14.3 Certified Design Material

This section provides the selection criteria and processes used to develop the AP600 Certified Design Material (CDM). This document provides the principal design bases and design characteristics that are certified by the 10 CFR Part 52 rulemaking process and included in the design certification rule.

The top-level design information in the Certified Design Material is extracted directly from the AP600 design information. Limiting the certified design contents to top-level information reflects the tiered approach to design certification endorsed by the U.S. Nuclear Regulatory Commission (see References 1 through 5).

The objective of this section is to define the bases and methods that were used to develop the Certified Design Material for the AP600. This section contains no new technical information regarding the AP600 design.

The AP600 Certified Design Material consists of the following:

- An introduction section which defines terms used in the Certified Design Material and lists general provisions that are applicable to all Certified Design Material entries. Also included is a list of acronyms and legends used in the Certified Design Material. (Because this material is self-explanatory, it is not discussed in this section.)

- Design descriptions for selected systems that are within the scope of the AP600 design certification, and the applicable portions of those selected systems that are only partially within the scope of the AP600 design certification. The Certified Design Material design descriptions delineate the principal design bases and principal design characteristics that are referenced in the design certification rule. The design descriptions are accompanied by the inspections, tests, analyses, and acceptance criteria (ITAAC) required by 10 CFR 52.47(a)(1)(vi) to be part of the design certification application. The ITAAC define verification activities that are to be performed for a facility with the objective of confirming that the plant is built and will operate in accordance with the design certification. Completion of these certified design ITAAC, together with the Combined License applicant's ITAAC for the site-specific portions of the plant, will be the basis for NRC authorization to load fuel per the provisions of 10 CFR Part 52.103.

- Design descriptions and their associated ITAAC for design and construction activities that are applicable to more than one system. Design-related processes have been included in the Certified Design Material for:

  - Aspects of the AP600 design likely to undergo rapid, beneficial technological developments in the lifetime of the design certification. Certifying the design processes associated with these areas of the design, rather than specific design details, permits future license applicants referencing the AP600 design certification to take advantage of the best technology available at the time of combined license application and facility construction.

   - Aspects of the design dependant upon characteristics of as-procured, as-installed systems, structures, and components. These characteristics are not available at the time of certification and, therefore, cannot be used to develop and certify design details.

- Interface requirements as defined by 10 CFR Part 52.47(a)(1)(vii). Interface requirements are defined as those which must be met by the site-specific portions of the complete nuclear power plant that are not within the scope of the certified design. These requirements define characteristics of the site-specific features that must be provided for the certified design to comply with certification commitments. AP600 has no interfaces meeting this definition. The Certified Design Material does not include ITAAC or a requirement for COL developed ITAAC for interface requirements.

- Site parameters used as the basis for AP600 design presented in the Tier 2 Material. These parameters represent a bounding envelope of site conditions for any license application referencing the AP600 design certification. No ITAAC are necessary for the site parameters entries because compliance with site parameters will be verified as part of issuance of a license for a plant that references the AP600 design certification.

The following is a description of the criteria and methods used to select specific technical entries for the Certified Design Material. The structure of the description is based on the Certified Design Material report structure.

The criteria and methods discussed in the following sections are guidelines only. For some matters, the contents of the Certified Design Material may not directly correspond to these guidelines because special considerations related to the matters may warrant a different approach. For such matters, a case-by-case determination is made regarding how or whether the matters should be addressed in the Certified Design Material. These determinations are based upon the principles inherent in Part 52.

14.3.1 CDM Section 1.0, Introduction

This section provides definitions, general provisions, a figure legend, and a list of acronyms used in the AP600 Certified Design Material.

Selection Criteria - Section 1.1 is used to define terms used throughout the Certified Design Material. Selection of entries is based on a judgment that a particular word/phrase merits definition - with particular emphasis on terms associated with implementation of the ITAAC. Section 1.2 contains a mixture of provisions that is selected on the basis that the provision is necessary to either define technical requirements applicable to multiple systems in the Certified Design Material or to provide clarification and guidance for future users of the Certified Design Material.

Selection Methodology - Entries in the Definition section are made on the basis of a self-evident need for a term to be defined. These terms are accumulated during the preparation and review of the Certified Design Material. Entries in the General Provisions section also are developed as part of the Certified Design Material selection and review process. Each entry has a unique background, but the overall intent is to state the broad guidelines and interpretations that are used to prepare Certified Design Material for the AP600.

14.3.2 CDM Section 2.0, System Based Design Descriptions and ITAAC

This section of the Certified Design Material has the design description and ITAAC material for the selected AP600 systems. The intent of this list of AP600 systems is to define at the Certified Design Material level the full scope of the certified design.

14.3.2.1 Design Descriptions

The certified design descriptions for selected AP600 systems address the top-level design features and performance standards that pertain to the safety of the plant and include descriptive text and supporting figures. The intent of the Certified Design Material design descriptions is to define the AP600 design characteristics referenced in the design certification rule as a result of the certification provisions of 10 CFR Part 52.

Selection Criteria - The following criteria are considered in determining the information included in the certified design descriptions:

- The information in the certified design descriptions is selected from the technical information presented in the Tier 2 Material. This reflects the approach that the Certified Design Material contains top-level design information and is based on the NRC directive in Reference 2 that there "be less detail in a certification than in an application for certification." In this context, the certification is the Certified Design Material and the application for certification includes the Tier 2 Material.

- The certified design descriptions contain only the information from the Tier 2 Material that is most significant to safety. The Tier 2 Material contains a wide spectrum of information on various aspects of the AP600 design. Not all of this information is included in the certified design descriptions. This selection criterion reflects the NRC directive in Reference 2 that the certified design should "encompass roughly the same design features that Section 50.59 prohibits changing without prior NRC approval." In determining those structures, systems, or components for which certified design descriptions and ITAAC must be prepared, the following questions are considered for each structure, system, or component:

  - Are there any features or functions classified as Class A, B, or C?

  - Are there any defense–in–depth features or functions provided?

  - For nonsafety–related systems, are there any features or functions credited for mitigation of design basis events?

- For nonsafety–related systems, are there any features or functions that have been identified in Reference 6 as candidates for additional regulatory oversight?

If the answer to the first question is yes, then a certified design description and ITAAC are prepared using the safety function stated in the Tier 2 Material and the parameters from the safety analysis.

If the answer to either of the next two questions is yes, then a certified design description and ITAAC are prepared using the functions stated in the Tier 2 Material and the parameters from the system design calculations.

If the answer to the last question is yes and the feature or function is not a programmatic requirement related to operations, maintenance or other programs, then a certified design description and ITAAC are prepared using the functions stated in the Tier 2 Material and the parameters from system design calculations.

In addition, the following questions were considered for each structure, system, or component not already selected for ITAAC using the above selection criteria:

- Are any features or functions necessary to satisfy the NRC's regulations in Parts 20, 50, 52, 73 and 100?

- Are there any features or functions that represent an important assumption for probabilistic risk assessment?

- Are any features or functions important in preventing or mitigating severe accidents?

- Are there any features or functions that have a significant impact on the safety and operation of the plant

- Are any features or functions the subject of a provision in the Technical Specifications.

If the answer to any of the above questions is yes, then a design description and ITAAC are prepared using the appropriate functions stated in the Tier 2 material and the parameters from the system design calculations.

A summary of the AP600 structures, systems, or components considered for selection is given in Table 14.3-1.

- In general, safety-related and defense-in-depth features and functions of structures, systems, and components are discussed in the certified design descriptions. Structures, systems, and components that are not classified as safety-related or defense-in-depth are discussed in the certified design descriptions to the extent that they have features or functions that mitigate a design basis event.

- The certified design descriptions for structures, systems, and components are limited to a discussion of design features and functions. The design bases of structures, systems, and components, and explanations of their importance to safety, are provided in the Tier 2 Material and are not included in the certified design descriptions. The Certified Design Material design descriptions define the certified design. Justification that the design meets regulatory requirements is presented in the Tier 2 Material.

- The certified design descriptions focus on the physical characteristics of the facility. The certified design descriptions do not contain programmatic requirements related to operating conditions or to operations, maintenance, or other programs. These matters are controlled by other means such as the technical specifications.

- The certified design descriptions in Section 2.0 of the Certified Design Material discuss the functional arrangement and performance characteristics that the structures, systems, and components should have after construction is completed. In general, the certified design descriptions do not address the processes that will be used for designing and constructing a plant that references the AP600 design certification. This is acceptable because the safety-function of a structure, system, or component is dependent upon its final as-built condition and not the processes used to achieve that condition. Exceptions to this criterion are the selected design and qualification processes defined in the instrumentation and control portions of Section 2 and the human factors portion of Section 3.

  The programmatic aspects of the design and construction processes (training, qualification of welders, and the like) are part of the licensee's programs and are subject to commitments made at the time of combined license issuance. Consequently, these issues are not addressed in the AP600 Certified Design Material.

- The certified design descriptions address fixed design features expected to be in place for the lifetime of the facility. Portable equipment and replaceable items are controlled through operational related programs.

- The certified AP600 design descriptions do not discuss component types (for example, valve and instrument types), component internals, or component manufacturers. This approach is based on the premise that the safety function of a particular design element can be performed by a variety of component types from different manufacturers.

- The certified design descriptions do not contain proprietary information.

- For the applicant or licensee of a plant that references the AP600 design certification to take advantage of improvements in technology, the certified design descriptions in general do not prescribe design features that are the subject of rapidly evolving technology.

- The Certified Design Material design description is intended to be self-contained and does not make direct reference to the Tier 2 Material, industrial standards, regulatory requirements, or other documents. (There are some exceptions involving the ASME Code and the Code of Federal Regulations.) If these sources contain technical information of sufficient safety significance to warrant Certified Design Material treatment, the information is extracted from the source and included directly in the appropriate system design description.

This approach is appropriate because it is unambiguous and it avoids potential questions regarding how much of a referenced document is encompassed in, and becomes part of, the Certified Design Material.

- Selection of the technical terminology to be used in the Certified Design Material is guided by the principle that the terminology should be as consistent as possible with that used in the Tier 2 Material and the body of regulatory requirements and industrial standards applicable to the nuclear industry. This approach is intended to minimize problems in interpreting Certified Design Material commitments.

A review of those sections of the AP600 Tier 2 Material that document plant safety evaluations was conducted. Specifically, reviews were conducted of the following chapters of the AP600 Tier 2 Material; the flooding analysis in Chapter 5, the analysis of overpressure protection in Chapter 5, containment analysis in Chapter 6, the core cooling analysis in Chapters 6 and 15, the analysis of fire protection in Chapter 9, the safety analysis of transients in Chapter 15, the analysis of anticipated transients without scram (ATWS) in Chapters 7 and 15, the radiological analysis in Chapter 15, the resolution of unresolved or generic safety issues and Three Mile Island issues in Chapter 1, and the PRA and severe accident information in Chapter 19. These reviews were important in identifying safety-related system design information warranting consideration in the design descriptions and the accompanying design commitments.

Selection Methodology - The Certified Design Material uses a system report structure. The certified design description entry for any system is based on review of the multiple sources having technical information related to that system. Using the selection criteria listed, design description material is developed for each system by reviewing the Tier 2 Material, safety analysis, test programs, and design documents relating to that system.

Application of the criteria listed results in a graded treatment of the systems. This leads to variation in the scope of the design description entries. The following lists the types of AP600 systems and is a summary of this graded treatment:

System Type	Scope of Design Description
Systems with safety-related functions that contribute to plant performance during design basis accidents	Major safety-related features and performance characteristics

System Type	Scope of Design Description
Systems with defense-in-depth functions that contribute to plant performance during design basis accidents	Major defense-in-depth features and performance characteristics
Nonsafety-related systems potentially impacting safety	Brief discussion of design features that prevent or mitigate the potential safety concern
Nonsafety-related systems with no relationship to safety	No discussion

For safety-related systems, application of this criteria results in design description entries that include the following information, as applicable:

- System name and scope
- System purpose
- Summary of the system's safety-significant components (usually shown by a figure)
- Equipment seismic and ASME classifications
- Piping ASME classification and Leak-Before-Break criteria
- Type of electrical power provided for the system
- System's important instruments, controls, and alarms to the extent located in the main control room or remote shutdown workstation
- Equipment to be qualified for harsh environments
- Motor-operated valves within the system that have an active safety-related function
- Other features or functions that are significant to safety

The certified design descriptions for nonsafety-related systems include the information listed to the extent that the information is relevant to the system and is significant to safety. Since much of this information is not relevant to nonsafety-related systems, the certified design descriptions for nonsafety-related systems are less extensive than the descriptions for safety-related systems.

14.3.2.2 Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC)

A table of ITAAC entries is provided for each system that has design description entries. The intent of these ITAAC is to define activities that will be undertaken to verify the as-built system conforms with the design features and characteristics defined in the design description. ITAAC are provided in tables with the following three-column format:

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria

Each design commitment in the left-hand column of the ITAAC tables has an associated inspections, tests, or analyses (ITA) requirement specified in the middle column. The acceptance criteria for the ITA are defined in the right-hand column.

Selection Criteria - The following are considered when determining what information is included in the Certified Design Material ITAAC entries:

- The scope and content of the ITAAC correspond to the scope and content of the certified design descriptions. There are no ITAAC for aspects of the design not addressed in the design description. This is appropriate because the objective of the ITAAC design certification entries is to verify that the as-built facility has the design features and performance characteristics defined in the Certified Design Material descriptions.

  Each AP600 system with a design description has an ITAAC table. This reflects the assessment that a design feature meriting a Certified Design Material description also merits an ITAAC entry to verify that the feature has been included in the as-built facility.

- One inspection, test, or analysis may verify one or more provisions in the certified design description. An ITAAC that specifies a system functional test or an inspection may verify a number of provisions in the design description. There is not necessarily a one-to-one correspondence between the ITAAC and the design descriptions.

- As required by 10 CFR 52.103, the inspections, tests, and analyses must be completed (and the acceptance criteria satisfied) prior to fuel loading. Therefore, the ITAAC do not include any inspections, tests, or analyses that are dependent upon conditions that exist only after fuel load.

- Because the design descriptions are limited to fixed design features expected to be in place for the lifetime of the facility, the ITAAC are limited to a verification of fixtures in the plant. There are no ITAAC for nuclear fuel, fuel channels, and control rods because they are changed by a licensee.

- The ITAAC verify the as-built configuration and performance characteristics of structures, systems, and components as identified in the Certified Design Material design descriptions.

Selection Methodology - Using the selection criteria, ITAAC table entries are developed for each selected system. This is achieved by evaluating the design features and performance characteristics defined in the Certified Design Material design description and preparing an ITAAC table entry for the design description criteria that satisfied the selection criteria. There is a close correlation between the left-hand column of the ITAAC table and the corresponding design description entries.

The ITAAC table is completed by selecting the method to be used for verification (either a test, an inspection, or an analysis [ITA]) and the acceptance criteria for the as-built feature.

The selection of the ITAs is guided by the following:

ITA Approach	Application
Inspection	To be used when verification can be accomplished by visual observations, physical examinations, review of records based on visual observations, or physical examinations that compare the as-built structure, system, or component condition to one or more design description commitments.
Test	To be used when verification can be accomplished by the actuation or operation, or establishment of specified conditions, to evaluate the performance or integrity of the as-built structures, systems, or components. The type of tests identified in the ITAAC tables includes activities such as factory testing, special test facility programs, and laboratory testing.
Analysis	To be used when verification can be accomplished by calculation, mathematical computation, or engineering or technical evaluations of the as-built structures, systems, or components.

The proposed verification activity is identified in the middle column of the ITAAC table. Where appropriate, the Tier 2 Material provides details regarding implementation of the verification activity. This Tier 2 Material is not referenced in the Certified Design Material and is not part of the Certified Design Material; Tier 2 Material is considered as providing one of potentially several acceptable methods for completing the ITA.

Selection of acceptance criteria is dependent upon the design characteristic being verified by the ITAAC table entry: in most cases, the appropriate acceptance criteria is self-evident and is based upon the Certified Design Material design description. For many of the AP600 ITAAC, the acceptance criteria is a statement that the as-built facility has the design feature or performance characteristic identified in the design description. A guiding principle for acceptance criteria preparation is the recognition that the criteria should be objective and unambiguous. The use of objective and unambiguous terms for the acceptance criteria will minimize opportunities for multiple, subjective (and potentially conflicting) interpretations as to whether an acceptance criteria has, or has not, been met. In some cases, the ITAAC acceptance criteria contain numerical parameters from the Tier 2 Material that are not specifically identified in the Certified Design Material design description or the design commitment column of the ITAAC table. This is acceptable because the design description defines the important design feature/performance that merits Certified Design Material treatment. The acceptance criterion defines a measurement standard for determining if the as-built facility is in compliance with the Certified Design Material design description commitment. Where appropriate, the Tier 2 Material identifies criteria applicable to the same design feature or function that is the subject of more general acceptance criteria in the ITAAC table.

For numerical acceptance criteria, ranges and/or tolerances are included. This is necessary and acceptable because of the following:

- Specification of a single-value acceptance criteria is impractical because trivial deviations will represent unnecessary noncompliances.

- Tolerances recognize that legitimate site variations can occur in complex construction projects.

- Minor variations in plant parameters within the tolerance bounds have no impact on plant safety.

14.3.3 CDM Section 3.0, Non-System Based Design Descriptions and ITAAC

Entries in this section of the Certified Design Material have the same structure as the system material discussed in Section 14.3.2; that is, design description text and figures and a table of ITAAC entries. The objective of this Certified Design Material is to address selected design and construction activities which are applicable to more than one system. There are six entries in Section 3.0 of the Certified Design Material: nuclear island buildings, initial test program, emergency response facilities, human factors engineering, Design Reliability Assurance Program, and radiation protection.

14.3.4 Certified Design Material Section, 4.0 Interface Requirements

AP600 is a plant design incorporating the nuclear island, the annex buildings and associated equipment, the diesel/generator building and associated equipment, the turbine/ generator building, the turbine/generator equipment, and the radwaste facilities. As a result, no interfaces need to be identified between or among these portions of the plant. There are no safety-related interfaces between the AP600 certified design and other portions of a facility with a combined license under 10 CFR Part 52.

The combined license applicant is responsible for initial testing of interfacing non-safety systems in portions of the plant outside the scope of design certification. Section 1.8,

Table 1.8-1, lists the interfacing systems and structures. Those systems that meet the requirements of 10 CFR 52.47(a)(1)(viii) are tabulated in subsection 14.4.5.

14.3.5 CDM Section 5.0, Site Parameters

This section of the Certified Design Material defines the site parameters used as a basis for the design defined in the AP600 certification application. These entries respond to the 10 CFR 52.47(a)(1)(iii) requirement that the design certification documentation include site parameter information. It is intended that applicants referencing the AP600 design certification demonstrate that these parameters for the selected site are within the certification envelope or provide additional analysis to show acceptability of deviations from the interface envelope.

Site-specific external events that relate to the acceptability of the design (and not to the acceptability of the site) are not considered site parameters and are addressed as interface requirements in the appropriate system entry in Section 4 of the Certified Design Material.

Section 5.0 of the Certified Design Material does not include any ITAAC and is limited to defining the AP600 site parameters. This is an appropriate approach because compliance of the site with these parameters is demonstrated by a license applicant prior to issuance of the license.

Selection Criteria - Section 2.0, Table 2.0-1, provides the envelope of site design parameters used for the AP600 design. The corresponding Certified Design Material Section 5.0 is based on using Table 2.0-1. Section 5.0 is limited to a tabular entry; no supporting text material is required.

14.3.6 Initial Test Program

The AP600 Initial Test Program defines testing activities that will be conducted following completion of construction and construction-related inspections and tests. The Initial Test Program extends through the start of commercial operation of the facility. This program is discussed in Chapter 14.

A summary of the Initial Test Program is included in Certified Design Material Section 3.4. This summary includes an overview of the Initial Test Program structure. This information is included in the Certified Design Material because of the importance of the Initial Test Program defining pre- and post-fuel load testing for the as-built facility. Key pre-fuel load Initial Test Program testing for individual systems is defined in the system ITAAC in Certified Design Material Sections 2 and 3.

No ITAAC entries have been included in the Certified Design Material for the Initial Test Program. This is acceptable because of the following:

- The Initial Test Program activities involve testing with the reactor at various power levels and thus cannot be completed prior to fuel load (Part 52 requires ITAAC to be completed prior to fuel load).
- Testing activities specified as part of the ITAAC in Certified Design Material Sections 2 and 3 must be performed prior to fuel load. Because these ITAAC testing activities address the design features and characteristics of safety significance, additional ITAAC for the Initial Test Program are not necessary to ensure that the as-built plant conforms with the certified design.

14.3.7 Elements of AP600 Design Material Incorporated into the Certified Design Material

Tables 14.3-2 through 14.3-8 summarize the design material that has been incorporated into the CDM in the areas of 1) Design Basis Accident Analysis, 2) Anticipated Transients Without Scram (ATWS), 3) Fire Protection, 4) Flood Protection, 5) Probabilistic Risk Assessment, 6) Radiological Analysis, and 7) Severe Accident Analysis. PRA assumptions incorporated into these tables encompass elements of the system design and assumptions that were expressly included in Tier 1 due to their importance. Both types of PRA assumptions were included for completeness, but are not distinguished in the tables. CDM falling outside of the seven subject areas are intentionally not incorporated in these tables. However, the referenced AP600 DCD sections may contain more information than encompassed by these seven subject areas. Each table may also include design information (certified or non-certified) that is not directly related to the particular subject area. Further, these tables are not intended to include all system-specific CDM information that is provided in the AP600 Tier 2 system descriptions.

14.3.8 Summary

An element of the design certification processes deriving from 10 CFR Part 52 is the selection and documentation of the technical information to be included in the design certification rule as the certified design. The certified design material is a subset of the design information presented in the Tier 2 Material. It includes the following:

- Key, important safety-significant aspects of the design described in the certification application
- Inspections, tests, analyses, and acceptance criteria (ITAAC) that will be used to verify that the as-built facility conforms with the certified design
- Interface requirements and site parameters

The information presented in the AP600 Certified Design Material is prepared using the selection criteria and methodology described in this section and is intended to satisfy the above Part 52 requirements for design certification. The ITAAC entries in Sections 2.0 and 3.0 confirm that key design performance characteristics and design features are implemented in the as-built facility.

14.3.9 References

1. SECY-90-377, "Requirements for Design Certification under 10 CFR Part 52," February 15, 1991.

2. 10 CFR, Part 52, "Statement of Considerations," (54 Federal Regulations 15372, 154377 [1989]).

3. SECY-90-241, "Level of Detail Required for Design Certification under Part 52," August 31, 1990.

4. SECY-90-377, "Requirements for Design Certification Under 10 CFR Part 52," November 8, 1990.

5. SECY-91-178, "Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for Design Certifications and Combined Licenses, "June 12, 1991.

6. WCAP-13856, "AP600 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process," Revision 1, January 1998.

Table 14.3-1 (Sheet 1 of 4)

ITAAC SCREENING SUMMARY

Structure/ System Acronym	Structure/ System Description	Selected for ITAAC
ADS	Automatic Depressurization System	X
ASS	Auxiliary Steam Supply System	X
BDS	Steam Generator Blowdown System	X
CAS	Compressed Air System	X
CCS	Component Cooling Water System	X
CDS	Condensate System	X
CES	Condenser Tube Cleaning System	X
CFS	Turbine Island Chemical Feed System	X
CMS	Condenser Air Removal System	X
CNS	Containment System	X
CPS	Condensate Polishing System	X
CVS	Chemical and Volume Control System	X
CWS	Circulating Water System	X
DAS	Diverse Actuation System	X
DDS	Data Display Processing System	X
DOS	Standby Diesel and Auxiliary Boiler Fuel Oil System	X
DRS	Storm Drain System	X
DTS	Demineralized Water Treatment System	X
DWS	Demineralized Water Transfer and Storage System	X
ECS	Main AC Power System	X
EDS	Non Class 1E DC and UPS System	X
EFS	Communication System	X
EGS	Grounding and Lightning Protection System	X

Table 14.3-1 (Sheet 2 of 4)

ITAAC SCREENING SUMMARY

Structure/ System Acronym	Structure/ System Description	Selected for ITAAC
EHS	Special Process Heat Tracing System	X
ELS	Plant Lighting System	X
EQS	Cathodic Protection System	X
FHS	Fuel Handling System	X
FPS	Fire Protection System	X
FWS	Main and Startup Feedwater System	X
GSS	Gland Seal System	X
HCS	Generator Hydrogen and CO_2 Systems	X
HDS	Heater Drain System	X
HSS	Hydrogen Seal Oil System	X
IDS	Class 1E DC and UPS System	X
IIS	Incore Instrumentation System	X
LOS	Main Turbine and Generator Lube Oil System	X
MES	Meteorological and Environmental Monitoring System	
MHS	Mechanical Handling System	X
MSS	Main Steam System	X
MTS	Main Turbine System	X
OCS	Operations and Control Centers	X
PCS	Passive Containment Cooling System	X
PGS	Plant Gas System	X
PLS	Plant Control System	X
PMS	Protection and Safety Monitoring System	X
PSS	Primary Sampling System	X

Table 14.3-1 (Sheet 3 of 4)

ITAAC SCREENING SUMMARY

Structure/ System Acronym	Structure/ System Description	Selected for ITAAC
PWS	Potable Water System	X
PXS	Passive Core Cooling System	X
RCS	Reactor Coolant System	X
RDS	Gravity and Roof Drain Collection System	X
RMS	Radiation Monitoring System	X
RNS	Normal Residual Heat Removal System	X
RWS	Raw Water System	X
RXS	Reactor System	X
SDS	Sanitary Drainage System	X
SES	Plant Security System	X
SFS	Spent Fuel Cooling System	X
SGS	Steam Generator System	X
SJS	Seismic Monitoring System	X
SMS	Special Monitoring System	X
SSS	Secondary Sampling System	X
SWS	Service Water System	X
TCS	Turbine Building Closed Cooling Water System	X
TDS	Turbine Island Vents, Drains and Relief Systems	X
TOS	Main Turbine Control and Diagnostics System	X
TVS	Closed Circuit TV System	X
VAS	Radiologically Controlled Area Ventilation System	X
VBS	Nuclear Island Nonradioactive Ventilation System	X
VCS	Containment Recirculation Cooling System	X
VES	Main Control Room Emergency Habitability System	X

Table 14.3-1 (Sheet 4 of 4)

ITAAC SCREENING SUMMARY

Structure/ System Acronym	Structure/ System Description	Selected for ITAAC
VFS	Containment Air Filtration System	X
VHS	Health Physics and Hot Machine Shop HVAC System	X
VLS	Containment Hydrogen Control System	X
VPS	Pump House Building Ventilation System	
VRS	Radwaste Building HVAC System	X̲
VTS	Turbine Island Building Ventilation System	X̲
VUS	Containment Leak Rate Test System	X̲
VWS	Central Chilled Water System	X
VXS	Annex/Auxiliary Nonradioactive Ventilation System	X
VYS	Hot Water Heating System	X̲
VZS	Diesel Generator Building Ventilation System	X
WGS	Gaseous Radwaste System	X
WLS	Liquid Radwaste System	X
WRS	Radioactive Waste Drain System	
WSS	Solid Radwaste System	X
WWS	Waste Water System	X̲
ZAS	Main Generator System	X̲
ZBS	Transmission Switchyard and Offsite Power System	
ZOS	Onsite Standby Power System	X
ZVS	Excitation and Voltage Regulation System	X̲

Legend: X = Selected for ITAAC
X̲ = Selected for ITAAC - title only, no entry for Design Certification
Blank = Not selected for ITAAC

Table 14.3-2 (Sheet 1 of 22)

DESIGN BASIS ACCIDENT ANALYSIS

Reference		Design Feature	Value
Section	5 1. 2	Safety valves are installed above and connected to the pressurizer to provide overpressure protection for the reactor coolant system.	
Section	5. 1. 2	The RCS has two hot legs and four cold legs.	
Section	5. 1. 2	The RCS has two steam generators and four reactor coolant pumps.	
Section	5. 1. 2	The RCS contains a pressurizer and a surge line connected to one hot leg.	
Section	5. 1. 3. 3	Rotating inertia needed for flow coast-down, is provided.	
Table	5. 1-3	Minimum measured flow rate with 10% tube plugging (gpm/loop)	96,600
Table	5. 1-3	Initial rated reactor core thermal power (MWt)	1933
Section	5. 2. 2	Reactor coolant system and steam system overpressure protection during power operation are provided by the pressurizer safety valves and the steam generator safety valves, in conjunction with the action of the PMS.	
Section	5. 2. 2. 1	Safety valve capacity exists to prevent exceeding 110 percent of system design pressure for the following events: - Loss of electrical load and/or turbine trip - Uncontrolled rod withdrawal at power - Loss of reactor coolant flow - Loss of normal feedwater - Loss of offsite power to the station auxiliaries	
Section	5. 2. 2. 1	Overpressure protection for the steam system is provided by steam generator safety valves	

Table 14.3-2 (Sheet 2 of 22)

DESIGN BASIS ACCIDENT ANALYSIS

Reference	Design Feature	Value
Section 5. 3. 2. 3	Non-destructive examination (NDE) of the reactor vessel and its appurtenances is conducted in accordance with ASME Code Section III requirements.	
Section 5. 3. 2. 5	The initial Charpy V-notch minimum upper shelf fracture energy levels for the reactor vessel beltline base metal traverse direction and welds are 75 foot-pounds, as required by Appendix G of 10 CFR 50.	
Section 5. 4. 1. 2. 1	Resistance temperature detectors (RTDs) monitor motor cooling circuit water temperature. These detectors provide indication of anomalous bearing or motor operation. They also provide a system for automatic shutdown in the event of a prolonged loss of component cooling water.	
Section 5. 4. 1. 3. 4	It is important to reactor protection that the reactor coolant continues to flow for a time after reactor trip and loss of electrical power. To provide this flow, each reactor coolant pump has a high-inertia rotor.	
Section 5. 4. 1. 3. 4	A safety-related pump trip occurs on high bearing water temperature.	
Section 5. 4. 5. 2. 3	Power to the pressurizer heaters is blocked when the core makeup tanks are actuated	
Section 5. 4. 6	Automatic depressurization system stage 1, 2 and 3 valves are connected to the pressurizer and discharge via the spargers to the in-containment refueling water storage tank.	
Section 5. 4. 6	Automatic depressurization system stage 4 valves are connected to each hot leg.	

Table 14.3-2 (Sheet 3 of 22)

DESIGN BASIS ACCIDENT ANALYSIS

Reference		Design Feature	Value
Section	5. 4. 9. 3	In the analysis of overpressure events, the pressurizer safety valves are assumed to actuate at 2500 psia. The safety valve flowrate assumed is based on full flow at 2575 psia, assuming 3 percent accumulation.	
Section	5. 4. 9. 3	The pressurizer safety valves prevent reactor coolant system pressure from exceeding 110% of system design pressure.	
Section	5. 4. 9. 3	In certain design basis events, the pressurizer safety valves are predicted to operate with very low flow rates.	
Section	5. 4. 12. 2	The reactor head vent valves can be operated from the main control room to provide an emergency letdown path.	
Table	5. 4-1	Minimum reactor coolant motor/pump moment of inertia ($lb\text{-}ft^2$).	≥ 5,000
Table	5. 4-11	Reactor Coolant System Design Pressure Settings: - Safety valves begin to open (psig)	 2485
Table	5. 4-17	Pressurizer Safety Valves - Design Parameters: - Number - Minimum required relieving capacity per valve (lbm/hr) - Set pressure (psig)	 2 ≥400,000 2485± 25
Section	6. 1. 2. 1. 1	The exterior of the containment vessel (above plant elevation 135'-3") and the interior of the containment vessel (above 7' above the operating deck) is coated with an inorganic zinc coating.	
Section	6. 1. 2. 1. 5	The nonsafety-related coatings used inside containment on walls, floors, ceilings, structural steel which is part of the building structure, and on the polar crane have a minimum dry film density (lb/ft^3).	≥100
Section	6. 2. 1. 1. 3	Internal containment structures, both metallic and concrete, act as passive internal heat sinks during a LOCA or a MSLB.	

Table 14.3-2 (Sheet 4 of 22)

DESIGN BASIS ACCIDENT ANALYSIS

Reference	Design Feature	Value
Figure 6. 2. 2-1	The passive containment cooling system consists of a water storage tank, cooling water flow discharge path to the containment shell, a water distribution system for the containment shell, and a cooling air flow path.	
Figure 6. 2. 2-1	The minimum duration the PCS cooling water flow is provided from the PCCWST (hours)	≥ 72
Table 6. 2. 2-1	The water coverage of the containment shell exceeds the amount used in the safety analysis.	
Table 6. 2. 2-1	The minimum drain flow rate capacity of the upper annulus drain (gpm).	≥ 450
Table 6. 2. 2-1	The minimum makeup flow rate capability from an external source to the PCS water storage tank (gpm).	≥ 62.7
Table 6. 2. 2-1	The minimum makeup flow rate capability from the PCS water storage tank to the spent fuel pit (gpm).	≥ 50
Table 6. 2. 2-1	The minimum PCS water storage tank volume for makeup to the spent fuel pit (non-coincident with PCS operation) (gallons).	≥ 400,000
Table 6. 2. 2-1	The minimum long term makeup capability from the PCCAWST to the PCCWST (days)	≥ 4
Table 6. 2. 2-1	The minimum long term makeup flow capability from the PCCAWST to the PCCWST (gpm)	≥ 62.7
Table 6. 2. 2-2	The first or top standpipe's elevation above the lowest or bottom standpipe (feet).	21.6 ± 0.25

Table 14.3-2 (Sheet 5 of 22)

DESIGN BASIS ACCIDENT ANALYSIS

Reference	Design Feature	Value
Table 6. 2. 2-2	The second standpipe's elevation above the lowest or bottom standpipe (feet).	14.0 ± 0.25
Table 6. 2. 2-2	The third standpipe's elevation above the lowest or bottom standpipe (feet).	6.1 ± 0.25
Figure 6. 2. 2-3	The minimum passive containment cooling water flow rate with water inventory at a height above the lowest standpipe of 13.05 ± 0.25 ft. (gpm)	≥ 72.5
Figure 6. 2. 2-3	The minimum passive containment cooling water flow rate with water inventory at a height above the lowest standpipe of 23.70 ± 0.25 ft. (gpm)	≥ 442
Figure 6. 2. 2-3	The minimum passive containment cooling water flow rate with water inventory at a height above the lowest standpipe of 20.65 ± 0.25 ft. (gpm)	≥ 123.5
Section 6. 3	The passive core cooling system provides core decay heat removal during design basis events.	
Section 6. 3	The passive core cooling system provides RCS makeup, boration, and safety injection during design basis events.	
Section 6.3	The passive core cooling system provides pH adjustment of water flooding the containment following design bases events.	
Section 6. 3. 1. 1	The passive core cooling system is designed to provide emergency core cooling during events involving increases and decreases in secondary side heat removal and decreases in reactor coolant system inventory.	

Table 14.3-2 (Sheet 6 of 22)

DESIGN BASIS ACCIDENT ANALYSIS

Reference	Design Feature	Value
Section 6. 3. 2. 1. 1	The heat exchanger consists of a bank of C-tubes, connected to a tubesheet and channel heat arrangement at the top (inlet) and bottom (outlet). The passive exchanger connects to the reactor coolant system through an inlet line from one reactor coolant system hot leg and an outlet line to the associated steam generator cold leg plenum (reactor coolant pump suction).	
Section 6. 3. 2. 1. 1	For the passive residual heat removal heat exchanger, the normal water temperature in the inlet line will be hotter than the discharge line.	
Section 6. 3. 2. 1. 2	The actuation of the core makeup tanks following a steam line break provides injection of borated water via water recirculation to mitigate the reactivity transient and provide the required shutdown margin.	
Section 6. 3. 2. 2. 1	The CMT inlet diffuser has a minimum flow area (in^3).	≥ 165
Section 6. 3. 2. 2. 6	The connection of the sparger branch arms to the sparge hub are submerged below the in-containment refueling water storage tank overflow level (ft).[3]	≤ 11.5
Section 6. 3. 2. 2. 3	The in-containment refueling water storage tank contains one passive residual heat removal heat exchanger.	
Section 6. 3. 2. 2. 6	Automatic depressurization system stage 1, 2 and 3 valves are connected to the pressurizer and discharge via the spargers to the in-containment refueling water storage tank.	
Section 6. 3. 2. 2. 7	The containment recirculation screens have plates located above them that are no more than 10 feet above the top of the screens and extend out at least 10 feet from the trash rack portion of the screen.	

Table 14.3-2 (Sheet 7 of 22)

DESIGN BASIS ACCIDENT ANALYSIS

Reference	Design Feature	Value
Section 6. 3. 2. 2. 7	The type of insulation used on ASME Class 1 lines inside containment and on the reactor vessel, reactor coolant pumps, pressurizer and steam generators is a metal reflective or suitable equivalent insulation.	
Section 6. 3. 2. 2. 7	The surface materials used in the vicinity of the containment recirculation screens are stainless steel or have coatings that are qualified to remain attached to those surfaces during design basis events. In the vicinity of the containment recirculation screens includes surfaces located above the bottom of the recirculation screens up to and including the bottom surface of the plate discussed in subsection 6. 3. 2. 2. 7, and the surfaces 10 feet in front and 10 feet to the sides of the screen face.	
Section 6. 3. 2. 2. 7	The bottom of the containment recirculation screens are located above the loop compartment floor (ft).	≥ 2
Section 6. 3. 3. 2. 1	For a loss of main feedwater event, the passive residual heat removal heat exchanger is actuated. If the core makeup tanks are not initially actuated, they actuate later when passive residual heat exchanger cooling sufficiently reduces pressurizer level.	
Section 6. 3. 3. 2. 2	For a feedwater system pipe failure event, the passive residual heat removal heat exchanger and the core makeup tanks are actuated.	
Section 6. 3. 3. 3. 1	For a steam generator tube rupture event, the nonsafety-related makeup pumps are automatically actuated when reactor coolant system inventory decreases and a reactor trip occurs, followed by actuation of the startup feedwater pumps. Makeup pumps automatically function to maintain the programmed pressurizer level. The core makeup tanks subsequently actuate on low pressurizer level, if they are not already actuated. Actuation of the core makeup tanks automatically actuates the passive residual heat removal system heat exchanger.	

Table 14.3-2 (Sheet 8 of 22)

DESIGN BASIS ACCIDENT ANALYSIS

Reference		Design Feature	Value
Section	6. 3. 6. 1	The piping resistances connecting the following PXS components and the RCS are bounded by the resistances assumed in the Chapter 15 safety analysis: - Core makeup tanks - Accumulators - In-containment refueling water storage tank injection - Containment recirculation - Automatic depressurization system valves	
Section	6. 3. 6. 1. 3	The bottom of the core makeup tanks are located above the reactor vessel direct vessel injection nozzle centerline (ft).	≥ 7. 5
Section	6. 3. 6. 1. 3	The bottom of the in-containment refueling water storage tank is located above the direct vessel injection nozzle centerline (ft).	≥ 3. 4
Section	6. 3. 6. 1. 3	The pH baskets are located below plant evaluation 107'-2".	
Figure	6. 3-1	The passive core cooling system has two direct vessel injection lines.	
Table	6. 3-4	The passive core cooling system has two core makeup tanks, each with a minimum required volume (ft^3)	≥ 2,000
Table	6. 3-4	The passive core cooling system has two accumulators, each with a minimum required volume (ft^3)	≥ 2,000
Table	6. 3-4	The passive core cooling system has an in-containment refueling water storage tank with a minimum required water volume (ft^3)	≥ 75,000
Table	6. 3-4	Each sparger has a minimum discharge flow area (in^3).	≥ 274

Table 14.3-2 (Sheet 9 of 22)

DESIGN BASIS ACCIDENT ANALYSIS

Reference	Design Feature	Value
Table 6. 3-4	The passive core cooling system has two pH adjustment baskets each with a minimum required volume (ft^3).	≥ 107
Table 6. 3-4	The passive residual heat removal heat exchanger minimum heat transfer rate (BTU/hr) - With 520°F Hot Leg and 120°F IRWST - With 420°F Hot Leg and 212°F IRWST	 ≥ 106,000,000 ≥ 43,400,000
Figure 6. 3.-1	The CMT level sensors (PXS-11A/B/C/D, -12A/B/C/D, -13A/B/C/D, and -14A/B/C/D) upper level tap centerlines are located below the centerline of the upper level tap connection to the CMTs(in).	1" ± 1"
Figure 6. 3-1	The CMT inlet lines (cold leg to high point) have no downward sloping sections.	
Figure 6. 3-1	The maximum elevation of the CMT injection lines between the connection to the CMT and the reactor vessel is the connection to the CMT's.	
Figure 6. 3-2	The PRHR inlet line (hot leg to high point) has no downward sloping sections.	
Figure 6. 3-2	The maximum elevation of the IRWST injection lines (from the connection to theIRWST to the reactor vessel) and the containment recirculation lines (from the containment to the IRWST injection lines) is less that the bottom inside surface of the IRWST.	
Figure 6. 3-2	The maximum elevation of the PRHR outlet line (from the PRHR to the SG) is less than the PRHR lower channel head top inside surface.	
Section 7. 1. 2. 11	Isolation devices are used to maintain the electrical independence of divisions and to see that no interaction occurs between nonsafety-related systems and the safety-related system. Isolation devices serve to prevent credible faults in circuit from propagating to another circuit.	

Table 14.3-2 (Sheet 10 of 22)

DESIGN BASIS ACCIDENT ANALYSIS

Reference	Design Feature	Value
Section 7.1.4.1.6	The protection and safety monitoring system equipment is seismically qualified to meet design basis earthquake levels.	Section
Section 7.1.4.1.6	The ability of the protection and safety monitoring system to initiate and accomplish protective functions is maintained despite degraded conditions caused by internal events such as fire, flooding, explosions, missiles, electrical faults and pipe whip.	
Section 7.1.4.1.6	The design of the protection and safety monitoring system equipment has margin to accommodate a loss of the normal HVAC.	
Section 7.1.4.2.6	The flexibility of the protection and safety monitoring system enables physical separation of redundant divisions.	
Section 7.2.2.2.1	The protection and safety monitoring system initiates a reactor trip whenever a condition monitored by the system reaches a preset level.	
Section 7.2.2.2.8	The reactor is tripped by actuating one of two manual reactor trip controls from the main control room.	
Section 7.3.1.2.14	The demineralized water system isolation valves close on a signal from the protection and safety monitoring system derived from either a reactor trip signal, a source range flux doubling signal, or low input voltage to the 1E dc and uninterruptible power supply battery chargers.	
Section 7.3.1.2.15	The chemical and volume control system makeup line isolation valves automatically close on a signal from the protection and monitoring system derived from either a high-2 pressurizer level, high-2 steam generator level signal, a safeguards signal coincident with high-1 pressurizer level, or high-2 containment radioactivity.	

Table 14.3-2 (Sheet 11 of 22)

DESIGN BASIS ACCIDENT ANALYSIS

Reference	Design Feature	Value
Section 7. 3. 1. 2. 2	The in-containment refueling water storage tank is aligned for injection upon actuation of the fourth stage automatic depressurization system via the protection and safety monitoring system.	
Section 7. 3. 1. 2. 3	The core makeup tanks are aligned for operation on a safeguards actuation signal or on a low-2 pressurizer level signal via the protection and safety monitoring system.	
Section 7. 3. 1. 2. 4	The fourth stage valves of the automatic depressurization system receive a signal to open upon the coincidence of a low-2 core makeup tank water level in either core makeup tank and low reactor coolant system pressure following a preset time delay after the third stage depressurization valves receive a signal to open via the protection and safety monitoring system.	
Section 7. 3. 1. 2. 4	The first stage valves of the automatic depressurization system open upon receipt of a signal generated from a core makeup tank injection alignment signal coincident with core makeup tank water level less than the Low-1 setpoint in either core makeup tank via the protection and safety monitoring system.	
Section 7. 3. 1. 2. 4	The second and third stage valves open on time delays following generation of the first stage actuation signal via the protection and safety monitoring system.	
Section 7. 3. 1. 2. 5	The reactor coolant pumps are tripped upon generation of a safeguards actuation signal or upon generation of a low-2 pressurizer water level signal.	
Section 7. 3. 1. 2. 7	The passive residual heat removal heat exchanger control valves are opened on low steam generator water level or on a CMT actuation signal via the protection and safety monitoring system.	

Table 14.3-2 (Sheet 12 of 22)

DESIGN BASIS ACCIDENT ANALYSIS

Reference	Design Feature	Value
Section 7. 3. 1. 2. 9	The containment recirculation isolation valves are opened on a safeguards actuation signal in coincidence with low-3 in-containment refueling water storage tank water level via the protection and safety monitoring system.	
Section 7. 3. 2. 2. 1	The protection and monitoring system automatically generate an actuation signal for an engineered safety feature whenever a monitored condition reaches a preset level.	
Section 7. 3. 2. 2. 9	Manual initiation at the system-level exists for the engineered safety features actuation.	
Section 7. 4. 3. 1	If temporary evacuation of the main control room is required because of some abnormal main control room condition, the operators can establish and maintain safe shutdown conditions for the plant from outside the main control room through the use of controls and monitoring located at the remote shutdown workstation.	
Section 7. 4. 3. 1. 1	The remote shutdown workstation equipment is similar to the operator workstations in the main control room and is designed to the same standards. One remote shutdown workstation is provided.	
Section 7. 4. 3. 1. 3	The remote shutdown workstation achieves and maintains safe shutdown conditions from full power conditions and maintains safe shutdown conditions thereafter.	
Section 7. 5. 4	The protection and safety monitoring system provides signal conditioning, communications, and display functions for Category 1 variables and for Category 2 variables that are energized from the Class 1E uninterruptible power supply system.	

Table 14.3-2 (Sheet 13 of 22)

DESIGN BASIS ACCIDENT ANALYSIS

Reference		Design Feature	Value
Section	7. 6. 1. 1	An interlock is provided for the normally closed motor-operated normal residual heat removal system inner and outer suction isolation valves. Each valve is interlocked so that it cannot be opened unless the reactor coolant system pressure is below a preset pressure.	
Section	8. 2. 2	Following a turbine trip during power operation, the reverse-power relay will be blocked for a minimum time period (sec).	$\geq$ 15
Section	8. 3. 2. 1. 2	The non-Class 1E dc and UPS system (EDS) consists of the electric power supply and distribution equipment that provides dc and uninterruptible ac power to nonsafety-related loads.	
Section	9. 1. 1. 2. 1	In the unlikely event of a dropping of an unirradiated fuel assembly, accidental deformation of the fuel rack will be determined and evaluated in the criticality analysis to demonstrate that it does not cause criticality criterion to be violated.	
Section	9. 1. 3. 5	The spent fuel pool is designed such that a water level is maintained above the spent fuel assemblies for at least 7 days following a loss of the spent fuel cooling system using only safety-related makeup water sources (See Table 9.1-4).	
Section	9. 1. 3. 5	The spent fuel pool cooling system includes safety-related connections to establish safety-related makeup to the spent fuel pool following a design basis event including a seismic event.	
Section	9. 1. 4. 1. 1	In the event of a safe shutdown earthquake (SSE), handling equipment cannot fail in such a manner as to prevent required function of seismic Category 1 equipment.	

Table 14.3-2 (Sheet 14 of 22)

DESIGN BASIS ACCIDENT ANALYSIS

Reference	Design Feature	Value
Section 9. 3. 6. 3. 7	The chemical and volume control system contains two redundant safety-related valves to isolate the demineralized water system from the makeup pump suction.	
Section 9. 3. 6. 3. 7	The chemical and volume control system contains two safety-related valves to isolate the makeup flow to the reactor coolant system.	
Section 9. 3. 6. 4. 5	The chemical and volume control system contains two safety-related valves to isolate the makeup flow to the reactor coolant system.	
Section 9. 3. 6. 4. 5. 1	The chemical and volume control system contains two redundant safety-related valves to isolate the demineralized water system from the makeup pump suction.	
Section 9. 3. 6. 7	The demineralized water system isolation valves close on a signal from the protection and safety monitoring system derived from either a reactor trip signal, a source range flux doubling signal, low input voltage to the 1E dc and uninterruptible power supply battery chargers, or a safety injection signal.	
Section 9. 3. 6. 7	The chemical and volume control system makeup line isolation valves automatically close on a signal from the protection and safety monitoring system derived from either a high-2 pressurizer level, high steam generator level signal, or a safeguards signal coincident with high-1 pressurizer level.	
Section 10. 1. 2	Safety valves are provided on both main steam lines.	

Table 14.3-2 (Sheet 15 of 22)

DESIGN BASIS ACCIDENT ANALYSIS

Reference	Design Feature	Value
Section 10. 2. 2. 4. 3	The flow of the main steam entering the high-pressure turbine is controlled by four stop valves and four governing control valves. The stop valves are closed by actuation of the emergency trip system devices.	
Section 10. 3. 1. 1	The main steam supply system is provided with a main steam isolation valve and associated MSIV bypass valve on each main steam line from its respective steam generator.	
Section 10. 3. 1. 1	Main steam isolation valve (MSIV) prevent the uncontrolled blowdown of more than one steam generator and isolate nonsafety-related portions of the system.	
Section 10. 3. 1. 2	Power-operated atmospheric relief valves are provided to allow controlled cooldown of the steam generator and the reactor coolant system when the condenser is not available.	
Section 10. 3. 2. 1	The main steam supply system includes: - One main steam isolation valve and one main steam isolation valve bypass valve per main steam line. - Main steam safety valves. - Power-operated atmospheric relief valves and upstream isolation valves.	
Section 10. 3. 2. 3. 2	In the event that a design basis accident occurs, which results in a large steam line break, the main steam isolation valves with associated main steam isolation bypass valves automatically close.	
Figure 10. 3. 2-1	The steam generator system consists of two main steam, two main feedwater, and two startup feedwater lines.	

Table 14.3-2 (Sheet 16 of 22)

DESIGN BASIS ACCIDENT ANALYSIS

Reference		Design Feature	Value
Table	10. 3. 2-1	Design data for main steam supply system valves: - Number per main steam line - Minimum relieving capacity per valve at 110% of design pressure (lb/hr)	 3 1,540,000
Table	10. 3. 2-2	The minimum flow capacity of the steam generator safety valves (lbm/hr)	≥ 4,600,000
Table	10. 3. 2-2	The maximum set pressure of the steam generator safety valves (psig)	≤ 1,195
Section	10. 3. 8. 3	The safety-related portions of the steam generator blowdown system are located in the containment and auxiliary buildings and are designed to remain functional after a safe shutdown earthquake.	
Section	10. 4. 7. 1. 1	Double valve main feedwater isolation is provided via the main feedwater control valve and main feedwater isolation valve. Both valves close automatically on main feedwater isolation signals, an appropriate engineered safety features isolation signal, within the time established with the Technical Specifications, Section 16.1. The startup feedwater control valve also serves as a containment isolation valve.	
Section	10. 4. 7. 1. 1	The condensate and feedwater system provides redundant isolation valves for the main feedwater lines routed into containment.	
Section	10. 4. 7. 1. 1	For a main feedwater or main steam line break (MSLB) inside the containment, the condensate and feedwater system is designed to limit high energy fluid to the broken loop.	

Table 14.3-2 (Sheet 17 of 22)

DESIGN BASIS ACCIDENT ANALYSIS

Reference	Design Feature	Value
Section 10. 4. 7. 1. 2	The booster/main feedwater pumps are tripped simultaneously with the feedwater isolation signal to close the main feedwater isolation valves.	
Section 10. 4. 7. 2. 1	The main feedwater pumps and booster pumps are tripped with the feedwater isolation signal that closes the main feedwater isolation valves. The same isolation signal closes the isolation valve in the cross connect line between the main feedwater pump discharge header and the startup feedwater pump discharge header.	
Section 10. 4. 7. 2. 2	One MFIV is installed in each of the two main feedwater lines outside the containment and downstream of the feedwater control valve. The MFIVs are installed to prevent uncontrolled blowdown from the steam generators in the event of a feedwater pipe rupture. The main feedwater check valve provides backup isolation. In the event of a secondary side pipe rupture inside the containment, the MFIVs limit the quantity of high energy fluid that enters the containment through the broken loop and limit cooldown. The MFCV provides backup isolation to limit cooldown and high energy fluid addition.	
Section 10. 4. 7. 2. 2	In the event of a secondary side pipe rupture inside the containment, the main feedwater control valves provide a redundant isolation to the MFIVs to limit the quantity of high energy fluid that enters the containment through the broken loop	
Section 10. 4. 7. 3	For a main feedwater line break inside the containment or a main steam line break, the MFIVs and the main feedwater control valves automatically close upon receipt of a feedwater isolation signal.	

Table 14.3-2 (Sheet 18 of 22)

DESIGN BASIS ACCIDENT ANALYSIS

Reference	Design Feature	Value
Section 10. 4. 7. 3	For a steam generator tube rupture event, positive and redundant isolation is provided for the main feedwater (MFIV and MFCV) with isolation signals generated by the protection and safety monitoring system (PMS).	
Section 10. 4. 8. 2. 2. 7	Blowdown system isolation is actuated on low steam generator water levels. The isolation of steam generator blowdown provides for a continued availability of the steam generator as a heat sink for decay heat removal in conjunction with operation of the passive residual heat removal system and the startup feedwater system.	
Section 10. 4. 8. 3	The safety-related portions of the steam generator blowdown system located in the containment and auxiliary buildings are designed to remain functional after a safe shutdown earthquake.	
Section 10. 4. 9. 1. 1	Double valve startup feedwater isolation is provided by the startup feedwater control valve and the startup feedwater isolation valve. Both valves close on a startup feedwater isolation signal, an appropriate engineered safeguards features signal, within the time established within the Technical Specifications, Section 16.1.	
Section 10. 4. 9. 1. 1	For a steam generator tube rupture event, positive and redundant isolation is provided for the startup feedwater system (startup feedwater isolation signal and startup feedwater control valve), with isolation signals generated by the protection and safety monitoring system.	
Section 10. 4. 9. 2. 2	In the event of a steam generator tube rupture, the startup feedwater isolation valve and startup feedwater control valve limit overfill of the steam generator by terminating startup feed flow.	

Table 14.3-2 (Sheet 19 of 22)

DESIGN BASIS ACCIDENT ANALYSIS

Reference	Design Feature	Value
Section 10. 4. 9. 2. 2	In the event of a secondary pipe rupture inside containment, the startup feedwater isolation valve and startup feedwater control valve provide isolation to limit the quantity of high energy fluid that enters the containment.	
Section 10. 4. 9. 2. 2	The startup feedwater isolation valve is provided to prevent the uncontrolled blowdown from more than one steam generator in the event of startup feedwater line rupture. The startup feedwater isolation valve provides backup isolation.	
Table 15. 0-1	Initial core thermal power (MWt)	1933
Table 15. 0-3	Nominal values of pertinent plant parameters used in accident analysis with 10% steam generator tube plugging - Reactor coolant flow per loop (gpm)	9.48 E+04
Section 15. 1. 2. 1	Continuous addition of excessive feedwater is prevented by the steam generator high-2 water level signal trip, which closes the feedwater isolation valves and feedwater control valves and trips the turbine, main feedwater pumps and reactor.	
Section 15. 1. 4. 1	For an inadvertent opening of a steam generator relief or safety valve, core makeup tank actuation occurs from one of four sources: - Two out of four low pressurizer pressure signals - Two out of four low pressurizer level signals - Two out of four low T_{cold} signals in any one loop - Two out of four low steam line pressure signals in any one loop	
Section 15. 1. 4. 1	After an inadvertent opening of a steam generator relief or safety valve, redundant isolation of the main feedwater lines closes the feedwater control valves and feedwater isolation valves, and trips the main feedwater pumps.	

Table 14.3-2 (Sheet 20 of 22)

DESIGN BASIS ACCIDENT ANALYSIS

Reference	Design Feature	Value
Section 15. 1. 5. 1	Following a steam line rupture, core makeup tank actuation occurs from one of five sources: - Two out of four low pressurizer pressure signals - Two out of four high-2 containment pressure signals - Two out of four low steam line pressure signals in any loop - Two out of four low T_{cold} signals in any one loop - Two out of four low pressurizer level signals	
Section 15. 1. 5. 1	After a steam line rupture, redundant isolation of the main feedwater lines closes the feedwater control valves and feedwater isolation valves, and trips the main feedwater pumps.	
Section 15. 1. 5. 2. 1	Core makeup tanks and the accumulators are the portions of the passive core cooling system used in mitigating a steam line rupture.	
Section 15. 1. 6. 1	The heat sink for the PRHR heat exchanger is provided by the IRWST, in which the PRHR heat exchanger is submerged.	
Section 15. 2. 6. 2. 1	Following a loss of ac power, the PRHR heat exchanger is actuated by the low steam generator water level (wide range).	
Section 15. 2. 8. 2. 1	Receipt of a low steam line pressure signal in at least one steam line initiates a steam line isolation signal that closes all main steam line and feed line isolation valves. This signal also gives a safeguards signal that initiates flow of cold borated water from the core makeup tanks to the reactor coolant system.	
Section 15. 3. 3. 2. 2	The pressurizer safety valves are fully open at 2575 psia. Their capacity for steam relief is described in Section 5.4.	

Table 14.3-2 (Sheet 21 of 22)

DESIGN BASIS ACCIDENT ANALYSIS

Reference	Design Feature	Value
Section 15. 4. 6. 2. 2	A safety signal from the protection and safety monitoring system automatically isolates the potentially unborated water from the demineralized water transfer and storage system and thereby terminates the dilution.	
Section 15. 5. 1. 1	Following inadvertent operation of the core makeup tanks during power operation, the high-3 pressurizer level signal actuates the PRHR heat exchanger and blocks the pressurizer heaters.	
Section 15. 5. 2. 1	The pressurizer heaters are blocked, and the main feedwater lines, steam lines, and chemical and volume control system are isolated.	
Table 15. 6. 5-11	ADS Valve Flow Areas (in^2)	
	- ADS Stage 1 Control Valve	≥ 4.6
	- ADS Stage 2 Control Valve	≥ 21
	- ADS Stage 3 Control Valve	≥ 21
	- ADS Stage 4A Valve	≥ 38
	- ADS Stage 4B Valve	≥ 38
Table 15. 6. 5-11	ADS Valve Opening Times (sec)	
	- ADS Stage 1 Control Valve	≤ 30
	- ADS Stage 1 Isolation Valve	≤ 20
	- ADS Stage 2 Control Valve	≤ 80
	- ADS Stage 2 Isolation Valve	≤ 30
	- ADS Stage 3 Control Valve	≤ 80
	- ADS Stage 3 Isolation Valve	≤ 30
	- ADS Stage 4A Valve	≤ 30
	- ADS Stage 4B Valve	≤ 30
Section 18. 8. 3. 2	The main control area includes two reactor operator workstations, the supervisor's workstation, the dedicated safety panel and the wall panel information system.	

Table 14.3-2 (Sheet 22 of 22)

DESIGN BASIS ACCIDENT ANALYSIS

Reference	Design Feature	Value
Section 18. 8. 3. 2	The human system interface resources available at each workstation are the plant information system displays, the control displays (soft controls), the alarm system support displays, procedure system, and the screen and component selector.	

Table 14.3-3

ANTICIPATED TRANSIENT WITHOUT SCRAM

Reference	Design Feature	Value
Section 7.7.1.11	The diverse actuation system is a nonsafety-related system that provides a diverse backup to the protection and safety monitoring system.	
Section 7.7.1.11	The diverse actuation system trips the reactor control rods and the turbine on low wide range steam generator water level and on low pressurizer water level.	
Section 7.7.1.11	The diverse actuation system initiates passive residual heat removal on low wide range steam generator water level or high hot leg temperature; actuates core makeup tanks and trips the reactor coolant pumps on low pressurizer water level; and isolates selected containment penetrations and starts passive containment cooling on high containment temperature.	
Section 7.7.1.11	The manual actuation function of the diverse actuation system is implemented by wiring the controls located in the main control room directly to the final loads in a way that bypasses the normal path through the control room multiplexers, the engineered safety features actuation cabinets, and the diverse actuation system logic	
Section 7.7.1.11	The diverse actuation system uses a microprocessor board different from those used in the protection and safety monitoring system.	
Section 7.7.1.11	The diverse actuation system hardware implementation is different from that of the protection and safety monitoring system.	
Section 7.7.1.11	The operating system and programming language of the diverse actuation system is different from that of the protection and safety monitoring system.	

Table 14.3-4 (Sheet 1 of 3)

FIRE PROTECTION

Reference		Design Feature	Value
Section	3. 4. 1. 1. 2	Separation is maintained between Class 1E divisions and between Class 1E divisions and non-Class 1E cables in accordance with the fire areas.	
Section	3. 4. 1. 1. 2	The AP600 arrangement provides physical separation of redundant safety-related components and systems from each other and from nonsafety-related components.	
Section	3. 8. 4. 1. 1	The conical roof supports the passive containment cooling system tank, which is constructed with a stainless steel liner on reinforced concrete walls.	
Section	7. 1. 4. 1. 6	The ability of the protection and safety monitoring system to initiate and accomplish protective functions is maintained despite degraded conditions caused by internal events such as fire and flooding.	
Section	7. 4. 3. 1	If temporary evacuation of the main control room is required because of some abnormal main control room condition, the operators can establish and maintain safe shutdown conditions for the plant from outside the main control room through the use of controls and monitoring located at the remote shutdown workstation.	
Section	7. 4. 3. 1. 1	The remote shutdown workstation equipment is similar to the operator workstations in the main control room and is designed to the same standards. One remote shutdown workstation is provided.	
Section	7. 4. 3. 1. 3	The remote shutdown workstation achieves and maintains safe shutdown conditions from full power conditions and maintains safe shutdown conditions thereafter.	

Table 14.3-4 (Sheet 2 of 3)

FIRE PROTECTION

Reference	Design Feature	Value
Section 8. 3. 2. 2	The four Class 1E battery chargers Class 1E voltage regulating transformers are independent, located in separate rooms, cannot be interconnected, and their circuits are routed in dedicated, physically separated raceways.	
Section 8. 3. 2. 3	Each safety-related circuit and raceway is given a unique identification number to distinguish between circuits and raceways of different voltage level or separation groups.	
Section 8. 3. 2. 4. 2	Cables of one separation group are run in separate raceway and physically separated from cables of other separation groups. Group N raceways are separated from safety-related groups A, B, C, and D. Non-class 1E circuits are electrically isolated by isolation devices, shielding and wiring techniques, physical separation, or an appropriate combination thereof.	
Section 9. 5. 1. 2. 1. 1	Separation is maintained between redundant safe shutdown components, including equipment, electrical cables, and instrumentation controls, in accordance with the fire areas.	
Section 9. 5. 1. 2. 1. 5	The standpipe system is supplied with water from the safety-related passive containment cooling system storage tank and normally operates independently of the rest of the fire protection system. The supply line draws water from a portion of the storage tank, using water allocated for fire protection.	
Section 9. 5. 1. 2. 1. 5	The standpipe system serving areas containing equipment required for safe shutdown following a safe shutdown earthquake is designed and supported so that it can withstand the effects of a safe shutdown earthquake and remain functional.	

Table 14.3-4 (Sheet 3 of 3)

FIRE PROTECTION

Reference	Design Feature	Value
Section 9. 5. 1. 2. 1. 5	The volume of the water in the PCS tank is sufficient to supply two hose streams, each with a flow of 75 gallons per minute, for two hours (gal).	≥ 18,000
Table 9. 5. 1-3	Each fire pump is rated: - Flow rate (gpm) - Total head (ft)	 ≥ 2000 ≥ 300
Section 18. 8. 3. 2	The human system interface resources available at each workstation are the plant information system displays, the control displays (soft controls), the alarm system support displays, procedure system, and the screen and component selector.	
Section 18. 8. 3. 4	The mission of the remote shutdown workstation is to provide the resources to bring the plant to a safe shutdown condition after an evacuation of the main control room.	
Section 18. 12. 3	The controls, displays, and alarms listed in Table 18.12.2-1 are retrievable from the remote shutdown workstation.	

Table 14.3-5 (Sheet 1 of 2)

FLOOD PROTECTION

Reference		Design Feature	Value
Section	Appendix 1-A RG 1.143 Section C.1.1.3 Clarification	The lowest level of the auxiliary building, elevation 66'-6", contains the components of the radwaste system within a common flood zone with watertight floors and walls. This volume of this enclosed flood zone is sufficient to contain the contents of the radwaste system.	
Table	2 -1	Plant elevation for maximum flood level (ft)	≤ 100
Section	3. 4. 1. 1. 1	The seismic category I structures below grade are protected against flooding by waterstops and a waterproofing system.	
Section	3. 4. 1. 1. 2	The boundaries between mechanical equipment rooms and the electrical and instrumentation and control equipment rooms of the auxiliary building are designed to prevent flooding of rooms that contain safe shutdown equipment up to the maximum flood level for each room.	
Section	3. 4. 1. 2. 2	The boundaries between mechanical equipment rooms inside containment and the electrical and instrumentation and control equipment rooms of the auxiliary building are designed to prevent flooding of rooms that contain safe shutdown equipment up to the maximum flood level for each room.	
Section	3. 4. 1. 2. 2	Boundaries exist to prevent flooding between the following rooms which contain safety-related equipment: PXS valve/accumulator room A, PXS valve/accumulator room B, and chemical and volume control room.	
Section	3. 4. 1. 2. 2	The AP600 arrangement provides physical separation of redundant safety-related components and systems from each other and from nonsafety-related components.	
Section	3. 4. 1. 2. 2	The safety-related components available for safety shutdown are located in the auxiliary building and inside containment. No credit is taken for operation of sump pumps to mitigate the consequences of flooding.	

Table 14.3-5 (Sheet 2 of 2)

FLOOD PROTECTION

Reference		Design Feature	Value
Section	3. 4. 1. 2. 2. 1	The PXS-A compartment, PXS-B compartment and the chemical and volume control system compartment are physically separated and isolated from each other by structural walls such that flooding in any one of these compartments is in the reactor coolant system compartment cannot cause flooding in any of the other compartments.	
Section	3. 6	In the event of a high- or moderate-energy pipe failure within the plant, adequate protection is provided so that essential structures, systems, or components are not impacted by the adverse effects of postulated pipe failure.	
Section	7. 1. 4. 1. 6	The ability of the protection and safety monitoring system to initiate and accomplish protective functions is maintained despite degraded conditions caused by internal events such as fire and flooding.	

Table 14.3-6 (Sheet 1 of 12)

PROBABILISTIC RISK ASSESSMENT

Reference		Design Feature	Value
Section	3. 2. 1. 3	The Nuclear Island structures include the containment and the shield and auxiliary buildings. These structures are seismic Category I.	
Table	3. 2-3	The components identified under Reactor Systems in Table 3. 2-3, as ASME Code Section III are designed and constructed in accordance with ASME Code Section III Requirements.	
Table	3. 2-3	The Nuclear Island structures include the containment and the Shield and Auxiliary Buildings. These structures are seismic Category I.	
Section	3. 4. 1. 1. 2	The boundaries between mechanical equipment rooms and the electrical and instrumentation and control equipment rooms of the auxiliary building are designed to prevent flooding of rooms that contain safe shutdown equipment up to the maximum flood level for each room.	
Section	3. 4. 1. 1. 2	The AP600 arrangement provides physical separation of redundant safety-related components and systems from each other and from nonsafety-related components.	
Section	3. 4. 1. 1. 2	Separation is maintained between Class 1E divisions and between Class 1E divisions and non-Class 1E cables in accordance with the fire areas.	
Section	3. 4. 1. 2. 2	Boundaries exist to prevent flooding between the following rooms which contain safety-related equipment: PXS valve/accumulator room A, PXS valve/accumulator room B, and chemical and volume control room.	

Table 14.3-6 (Sheet 2 of 12)

PROBABILISTIC RISK ASSESSMENT

Reference	Design Feature	Value
Section 3. 4. 1. 2. 2	The boundaries between mechanical equipment rooms inside containment and the electrical and instrumentation and control equipment rooms of the auxiliary building are designed to prevent flooding of rooms that contain safe shutdown equipment up to the maximum flood level for each room.	
Section 3. 4. 1. 2. 2	The safety-related components available for safety shutdown are located in the auxiliary building and inside containment. No credit is taken for operation of sump pumps to mitigate the consequences of flooding.	
Section 3. 4. 1. 2. 2. 1	The PXS-A compartment, PXS-B compartment and the chemical and volume control system compartment are physically separated and isolated from each other by structural walls such that flooding in any one of these compartments or in the reactor coolant system compartment cannot cause flooding in any of the other compartments.	
Section 3D. 6	RXS equipment in Appendix 3D is seismically qualified.	
Section 5. 1. 3	ADS has four stages. Each stage is arranged into two separate groups of valves and lines. - Stages 1, 2, and 3 discharge from the top of the pressurizer to the IRWST. - Each stage 4 discharges from a hot leg to the RCS loop compartment.	
Section 5. 3. 1. 1	The reactor vessel provides a high integrity pressure boundary to contain the reactor coolant, heat generating reactor core, and fuel fission products. The reactor vessel is the primary boundary for the reactor coolant and the secondary barrier against the release of radioactive fission products.	

Table 14.3-6 (Sheet 3 of 12)

PROBABILISTIC RISK ASSESSMENT

Reference	Design Feature	Value
Section 5. 4. 6	ADS has four stages. Each stage is arranged into two separate groups of valves and lines. - Stages 1, 2, and 3 discharge from the top of the pressurizer to the IRWST. - Each stage 4 discharges from a hot leg to the RCS loop compartment.	
Section 5. 4. 6. 2	Each ADS stage 1, 2, and 3 line contains two normally closed motor-operated valves (MOVs).	
Section 5. 4. 6. 2	Each ADS stage 4 line contains a normally open MOV valve and a normally closed squib valve.	
Section 5. 4. 7	The RNS removes heat from the core and reactor coolant system at reduced RCS pressure and temperature conditions after shutdown.	
Section 5. 4. 7	The normal residual heat removal system (RNS) provides a safety-related means of performing the following functions: - Containment isolation for the RNS lines that penetrate the containment - Long-term, post-accident makeup water to the RCS	
Section 5. 4. 7. 1. 1	The RNS containment isolation and pressure boundary valves are safety-related. The motor-operated valves are powered by Class 1E dc power.	
Section 5. 4. 7. 1. 2. 1	The component cooling water system (CCS) provides cooling to the RNS heat exchanger.	
Section 6. 2. 4	The containment hydrogen control system provides nonsafety-related hydrogen igniters for control of the containment hydrogen concentration for beyond design basis accidents.	

Table 14.3-6 (Sheet 4 of 12)

PROBABILISTIC RISK ASSESSMENT

Reference	Design Feature	Value
Section 6. 2. 4. 2. 3	At least 64 hydrogen igniters are provided.	
Table 6. 2. 4-2	The minimum full size passive autocatalytic recombiner depletion rate at 120° F and atmospheric pressure (scfm)	≥ 1
Section 6. 3	The automatic depressurization system provides a safety-related means of depressurizing the RCS.	
Section 6. 3	The in-containment refueling water storage tank subsystem provides a safety-related means of performing the following functions: - Low-pressure safety injection - Core decay heat sink during design basis events - Flooding of the lower containment, the reactor cavity and the loop compartment by draining the IRWST into the containment. - Borated water	
Section 6. 3. 1	The core makeup tanks provide safety-related means of safety injection of borated water to the RCS.	
Section 6. 3. 1	Passive residual heat removal (PRHR) provides a safety-related means of removing core decay heat during design basis events.	
Section 6. 3. 2	The ADS valves are powered from Class 1E dc power.	
Section 6. 3. 2	There are two CMTs, each with an injection line to the reactor vessel/DVI nozzle. - Each CMT has a pressure balance line from an RCS cold leg. - Each injection line is isolated with a parallel set of air-operated valves (AOVs). - These AOVs open on loss of air. - The injection line for each CMT also has two check valves in series.	

Table 14.3-6 (Sheet 5 of 12)

PROBABILISTIC RISK ASSESSMENT

Reference	Design Feature	Value
Section 6. 3. 2	The IRWST subsystem has the following flowpaths: - Two (redundant) injection lines from the IRWST to the reactor vessel/DVI nozzle. Each line is isolated with a parallel set of valves; each set with a check valve in series with a squib valve. - Two (redundant) recirculation lines from the containment to the IRWST injection line. Each recirculation line has two paths: one path contains a squib valve and an MOV, the other path contains a squib valve and a check valve. - The two MOV/squib valve lines also provide the capability to flood the reactor cavity.	
Section 6. 3. 2	There are screens for each IRWST injection line and recirculation line.	
Section 6. 3. 2	PRHR is actuated by opening redundant, parallel air-operated valves. These air-operated valves open on loss of air.	
Section 6. 3. 2. 2	The passive core cooling system (PXS) is composed of the following: - Accumulator subsystem - Core makeup tank (CMT) subsystem - In-containment refueling water storage tank (IRWST) subsystem - Passive residual heat removal (PRHR) subsystem. - The automatic depressurization system (ADS), which is a subsystem of the reactor coolant system (RCS), also supports passive core cooling functions.	
Section 6. 3. 2. 2. 2	There are two accumulators, each with an injection line to the reactor vessel/direct vessel injection (DVI) nozzle. Each injection line has two check valves in series.	
Section 6. 3. 2. 2. 2	The accumulators provide a safety-related means of safety injection of borated water to the RCS.	

Table 14.3-6 (Sheet 6 of 12)

PROBABILISTIC RISK ASSESSMENT

Reference		Design Feature	Value
Section	6. 3. 2. 2. 8. 7	The accumulator discharge check valves are of a different type than the CMT discharge check valves.	
Section	6. 3. 3	IRWST squib valves and MOVs are powered by Class 1E dc power.	
Section	6. 3. 3	The CMT AOVs are automatically and manually actuated from PMS and DAS.	
Section	6. 3. 3	The PRHR air-operated valves are automatically actuated and manually actuated from the control room by either PMS or DAS.	
Section	6. 3. 3	The squib valves and MOVs for injection and recirculation are automatically and manually actuated via PMS, and manually actuated via DAS.	
Section	6. 3. 3	The squib valves and MOVs for and reactor cavity flooding are manually actuated via PMS and DAS from the control room.	
Section	6. 3. 7	The positions of the containment recirculation isolation MOVs are indicated in the control room.	
Section	6. 3. 7	The position of the inlet PRHR valve is indicated in the control room.	
Section	6. 3. 7. 6. 1	The ADS first-, second-, and third -stage valve positions are indicated in the control room.	
Section	7. 1. 1	The diverse actuation system provides a nonsafety-related means of performing the following functions: - Initiates automatic and manual reactor trip - Automatic and manual actuation of selected engineered safety features - Main control room display of selected plant parameters.	

Table 14.3-6 (Sheet 7 of 12)

PROBABILISTIC RISK ASSESSMENT

Reference	Design Feature	Value
Section 7. 1. 1	The protection and safety monitoring system provides a safety-related means of performing the following functions: - Automatic and manual reactor trip - Automatic and manual actuation of engineered safety features (ESF).	
Section 7. 1. 1	PMS provides for the minimum inventory of fixed position controls and displays in the control room.	
Section 7. 1. 2	Each PMS division is powered from its respective Class 1E dc division.	
Section 7. 1. 2	PMS has four divisions of reactor trip and ESF actuation.	
Section 7. 1. 2. 10	PMS automatically produces a safety-related reactor trip or ESF initiation upon an attempt to bypass more than two channels of a function that uses 2-out-of-4 logic.	
Section 7. 1. 2. 15	The PMS hardware and software are developed using a planned design process which provides for specific design documentation and reviews during the design requirement, system definition, development, test and installation phases.	
Section 7. 1. 2. 6	PMS has two divisions of safety-related post-accident parameter display.	
Section 7. 1. 4. 1. 6	The ability of the protection and safety monitoring system to initiate and accomplish protective functions is maintained despite degraded conditions caused by internal events such as fire and flooding.	
Section 7. 1. 4. 1. 6	The design of the protection and safety monitoring system equipment has margin to accommodated a loss of the normal HVAC.	

Table 14.3-6 (Sheet 8 of 12)

PROBABILISTIC RISK ASSESSMENT

Reference		Design Feature	Value
Section	7. 1. 4. 2. 6	The flexibility of the protection and safety monitoring system enables physical separation of redundant divisions.	
Figure	7. 1-8	PMS has two divisions of safety-related post-accident parameter display.	
Section	7. 2. 2. 2. 1	The protection and safety monitoring system initiates a reactor trip whenever a condition monitored by the system reaches a preset level.	
Section	7. 3	The PMS allows for the transfer of control capability from the main control room to the remote shutdown workstation. The minimum inventory of displays and controls in the remote shutdown workstation is provided.	
Section	7. 3. 1	The ADS valves are powered from Class 1E dc power.	
Section	7. 3. 1	The ADS valves are automatically and manually actuated via the protection and safety monitoring system (PMS), and manually actuated via the diverse actuation system (DAS).	
Section	7. 3. 1	The CMT AOVs are automatically and manually actuated from PMS and DAS.	
Section	7. 3. 1	The squib valves and MOVs for injection and recirculation are automatically and manually actuated via PMS, and manually actuated via DAS.	
Section	7. 3. 1	The squib valves and MOVs for reactor cavity flooding are manually actuated via PMS and DAS from the control room.	
Section	7. 3. 1	The PRHR air-operated valves are automatically actuated and manually actuated from the control room by either PMS or DAS.	

Table 14.3-6 (Sheet 9 of 12)

PROBABILISTIC RISK ASSESSMENT

Reference	Design Feature	Value
Section 7. 3. 1	The RNS containment isolation MOVs are actuated via PMS.	
Section 7. 6. 1. 1	An interlock is provided for the normally closed motor-operated normal residual heat removal system inner and outer suction isolation valves. Each valve is interlocked so that it cannot be opened unless the reactor coolant system pressure is below a preset pressure.	
Section 7. 7. 1. 11	The diverse actuation system is a nonsafety-related system that provides a diverse backup to the protection and safety monitoring system.	
Section 7. 7. 1. 11	The diverse actuation system trips the reactor control rods and the turbine on low wide range steam generator water level and on low pressurizer water level.	
Section 7. 7. 1. 11	DAS manual initiation functions are implemented in a manner that bypasses the signal processing equipment of the DAS.	
Section 7. 7. 1. 11	The DAS automatic actuation signals are generated in a functionally diverse manner from the PMS signals. Diversity between DAS and PMS is achieved by the use of different architecture, different hardware implementations, and different software.	
Section 8. 3. 1. 1. 1	On loss of power to a 4160V diesel-backed bus, the associated diesel generator automatically starts and produces ac power. The source circuit breakers and bus load circuit breakers are opened, and the generator is connected to the bus. Each generator has an automatic load sequencer to enable controlled loading on the associated buses.	
Section 8. 3. 1. 1. 2. 1	Two onsite standby diesel generator units provide power to the selected nonsafety-related ac loads.	

Table 14.3-6 (Sheet 10 of 12)

PROBABILISTIC RISK ASSESSMENT

Reference	Design Feature	Value
Section 8. 3. 1. 1. 3	The main ac power system distributes non-Class 1E power from onsite sources to selected nonsafety-related loads.	
Section 8. 3. 2. 1	The Class 1E dc and uninterruptible power supply (UPS) system (IDS) provides dc and uninterruptible ac power for the safety-related equipment.	
Section 8. 3. 2. 1. 1. 1	There are four independent, Class 1E 125 Vdc divisions. Divisions A and D are each composed of one battery bank, one switchboard, and one battery charger. Divisions B and C are each composed of two battery banks, two switchboards, and two battery chargers. The first battery bank in the four divisions is designated as the 24-hour battery bank. The second battery bank in Divisions B and C is designated as the 72-hour battery bank.	
Section 8. 3. 2. 1. 1. 1	Battery chargers are connected to dc switchboard buses. The input ac power for the Class 1E dc battery chargers is supplied from onsite diesel-generator-backed low-voltage ac power supplies.	
Section 8. 3. 2. 1. 1. 1	The 24-hour battery banks provide power to the loads for a period of 24 hours without recharging. The 72-hour battery banks supply a dc switchboard bus load for a period of 72 hours without recharging.	
Section 8. 3. 2. 1. 2	The non-Class 1E dc and UPS system (EDS) consists of the electric power supply and distribution equipment that provides dc and uninterruptible ac power to nonsafety-related loads.	
Section 8. 3. 2. 1. 2	EDS load groups 1, 2, and 3 provide 125 Vdc power to the associated inverter units that supply the ac power to the non-Class 1E uninterruptible power supply ac system.	

Table 14.3-6 (Sheet 11 of 12)

PROBABILISTIC RISK ASSESSMENT

Reference	Design Feature	Value
Section 8. 3. 2. 1. 2	Battery chargers are connected to dc switchboard buses. The input ac power for the non-Class 1E dc battery chargers is supplied from onsite diesel-generator-backed low-voltage ac power supplies.	
Section 8. 3. 2. 1. 2	The onsite standby diesel-generator-backed low-voltage ac power supply provides the normal ac power to the battery chargers.	
Section 8. 3. 2. 1. 3	Separation is provided between Class 1E divisions, and between Class 1E divisions and non-Class 1E cables.	
Section 9. 2. 1	The service water system is a nonsafety-related system that transfers heat from the component cooling water heat exchangers to the atmosphere.	
Section 9. 2. 1. 2. 1	The SWS is arranged into two trains. Each train includes one pump and one cooling tower cell.	
Section 9. 2. 2	The component cooling water system is a nonsafety-related system that removes heat from various components and transfers the heat to the service water system (SWS).	
Section 9. 2. 2. 2	The CCS is arranged into two trains. Each train includes one pump and one heat exchanger.	
Section 9. 3. 6	The CVS provides a nonsafety-related means to perform the following functions: - Makeup water to the RCS during normal plant operation - Boration following a failure of reactor trip - Coolant to the pressurizer auxiliary spray line.	

Table 14.3-6 (Sheet 12 of 12)

PROBABILISTIC RISK ASSESSMENT

Reference	Design Feature	Value
Section 9. 3. 6. 1	The chemical and volume control system (CVS) provides a safety-related means to terminate inadvertent RCS boron dilution.	
Section 9. 4. 1	The main control room has its own ventilation system and is pressurized. The ventilation system for the remote shutdown room is independent of the ventilation system for the main control room.	
Section 9. 5. 1. 2. 1. 1	The PMS allows for the transfer of control capability from the main control room to the remote shutdown workstation. The minimum inventory of displays and controls in the remote shutdown room is provided.	
Section 9. 5. 1. 2. 1. 1	Class 1E divisional cables are routed in their respective divisional raceways.	
Section 9. 5. 1. 2. 1. 1	Separation is maintained between Class 1E divisions and between Class 1E divisions and non-Class 1E cables in accordance with the fire areas.	
Section 17. 4. 1	D-RAP provides reasonable assurance that the design of risk-significant SSCs is consistent with their PRA assumptions.	
Section 18. 8. 3. 2	The main control area includes two reactor operator workstations, the supervisor's workstation, the dedicated safety panel and the wall panel information system.	
Section 18. 12. 2	The minimum inventory of instrumentation includes those displays, controls, and alarms that are used to monitor the status of the critical safety functions and to manually actuate the safety-related systems that achieve the critical safety functions. The minimum inventory resulting from the implementation of the selection criteria is provided in Table 18.12.2-1.	

Table 14.3-7 (Sheet 1 of 4)

RADIOLOGICAL ANALYSIS

Reference	Design Feature	Value
Table 2 -1	Plant elevation for maximum flood level (ft)	≤ 100
Section 2. 3. 4	Atmospheric dispersion factors - X/Q (sec/m^3) - Site Boundary X/Q 0 - 2 hour time interval - Low Population Zone Boundary X/Q 0 - 8 hours 8 - 24 hours 24 - 96 hours 96 - 720 hours	 ≤ 1.0×10^{-3} ≤ 1.35×10^{-4} ≤ 1.0×10^{-4} ≤ 5.4×10^{-5} ≤ 2.2×10^{-5}
Table 6. 2. 3-1	Containment penetration isolation features are configured as in Table 6.2.3-1	
Table 6. 2. 3-1	Maximum closure time for remotely operated containment purge valves (seconds)	≤ 5
Table 6. 2. 3-1	Maximum closure time for all other remotely operated containment isolation valves (seconds)	≤ 60
Section 6. 4. 2. 3	The minimum storage capacity of each set of storage tanks in the VES (scf)	≤ 122,021
Section 6. 4. 3. 2	The maximum temperature rise in the main control room pressure boundary following a loss on the nuclear island nonradioactive ventilation system over a 72-hour period (° F)	≤ 15
Section 6. 4. 3. 2	The maximum temperature in the instrumentation and control rooms and dc equipment rooms following a loss of the nuclear island nonradioactive ventilation system remains over a 72-hour period (°F).	≤ 125
Section 6. 4. 4	The main control emergency habitability system nominally provides 65 scfm of ventilation air to the main control room from the compressed air storage tanks.	65 ± 2

Table 14.3-7 (Sheet 2 of 4)

RADIOLOGICAL ANALYSIS

Reference	Design Feature	Value
Section 6. 4. 4	Sixty-five ± five scfm of ventilation flow is sufficient to pressurize the control room to 1/8th inch water gauge differential pressure (WIC).	1/8th
Figure 6. 4-2	The main control room emergency habitability system consists of two sets of emergency air storage tanks and an air delivery system to the main control room.	
Section 6. 5. 3	The passive heat removal process and the limited leakage from the containment result in offsite doses less than the regulatory guideline limits.	
Section 8. 3. 1. 1. 6	Electrical penetrations through the containment can withstand the maximum short-circuit currents available either continuously without exceeding their thermal limit, or at least longer than the field cables of the circuits so that the fault or overload currents are interrupted by the protective devices prior to a potential failure of a penetration.	
Section 9. 4. 1. 1. 1	The VBS isolates the HVAC ductwork that penetrates the main control room boundary on high particulate or iodine concentrations in the main control room supply air or on extended loss of ac power to support operation of the main control room emergency habitability system.	

Table 14.3-7 (Sheet 3 of 4)

RADIOLOGICAL ANALYSIS

Reference	Design Feature	Value
Section 12. 3. 2. 2. 1	During reactor operation, the shield building protects personnel occupying adjacent plant structures and yard areas from radiation originating in the reactor vessel and primary loop components. The concrete shield building wall and the reactor vessel and steam generator compartment shield walls reduce radiation levels outside the shield building to less than 0.25 mrem/hr from sources inside containment. The shield building completely surrounds the reactor components.	
Section 12. 3. 2. 2. 2	The reactor vessel is shielded by the concrete primary shield and by the concrete secondary shield which also surrounds other primary loop components. The secondary shield is a structural module filled with concrete surrounding the reactor coolant system equipment, including piping, pumps and steam generators. Extensive shielding is provided for areas surrounding the refueling cavity and the fuel transfer canal to limit the radiation levels.	
Section 12. 3. 2. 2. 3	Shielding is provided for the liquid radwaste, gaseous radwaste and spent resin handling systems consistent with the maximum postulated activity. Corridors are generally shielded to allow Zone II access, and operator areas for valve modules are generally Zone II or III for access. Shielding is provided to attenuate radiation from normal residual heat removal equipment during shutdown cooling operations to levels consistent with radiation zoning requirements of adjacent areas.	

Table 14.3-7 (Sheet 4 of 4)

RADIOLOGICAL ANALYSIS

Reference	Design Feature	Value
Section 12. 3. 2. 2. 4	The concrete shield walls surrounding the spent fuel cask loading and decontamination areas, and the shield walls surrounding the fuel transfer and storage are sufficiently thick to limit radiation levels outside the shield walls in accessible areas to Zone II. The building walls are sufficient to shield external plant areas which are not controlled to Zone II.	
Section 12. 3. 2. 2. 5	Shielding is provided as necessary for the waste storage areas in the radwaste building to meet the radiation zone and access requirements.	
Section 12. 3. 2. 2. 7	Shielding combined with other engineered safety features is provided to permit access and occupancy of the control room following a postulated loss-of-coolant accident, so that radiation doses are limited to five rem whole body from contributing modes of exposure for the duration of the accident, in accordance with General Design Criteria 19.	
Section 12. 3. 2. 2. 9	The spent fuel transfer tube is shielded to within adjacent area radiation limits, is completely enclosed in concrete, and there is no unshielded portion of the spent fuel transfer tube during the refueling operation.	

Table 14.3-8

SEVERE ACCIDENT ANALYSIS

Reference	Design Features	Value
Section 1. 2	The discharge from the IRWST vents located in the roof of the IRWST next to the containment vessel are oriented away from the containment vessel.	
Section 5. 3. 1. 2	There are no penetrations in the reactor vessel below the core.	
Section 5. 3. 5	The reflective reactor vessel insulation provides an engineered flow path to allow the ingression of water and venting of steam for externally cooling the vessel.	
	- A flow path exists from the loop compartment to the reactor vessel cavity (ft^2).	≥ 6
	- A flow path area to vent steam exists between the vessel insulation and the reactor vessel (ft^2).	≥ 7.5
Section 6. 2. 4. 2. 2	The hydrogen control system consists of two passive autocatalytic recombiners installed inside the containment within the upper compartment between elevations 150'-0" and 175'-0".	
Section 6. 2. 4. 2. 3	The hydrogen ignition subsystem consists of 64 hydrogen igniters strategically distributed throughout the containment.	
Table 6. 2. 4-3	The minimum surface temperature of the hydrogen ignitors (°F).	≥ 1,700
Section 6. 3	The ADS provides a safety-related means of depressurizing the RCS.	
Section 6. 3	The PXS provides a safety-related means of flooding the reactor cavity by draining the IRWST into the containment.	
Section 7. 3. 1. 2. 9	Signals to align the IRWST containment recirculation isolation valves are generated by manual initiation.	
Section 7. 7. 1. 11	Initiation of containment recirculation is a diverse manual function.	