Enclosure 2

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"Blue Energy Center - Port of Victoria Nuclear Unit 1: Preliminary Safety Analysis Report", Revision 0

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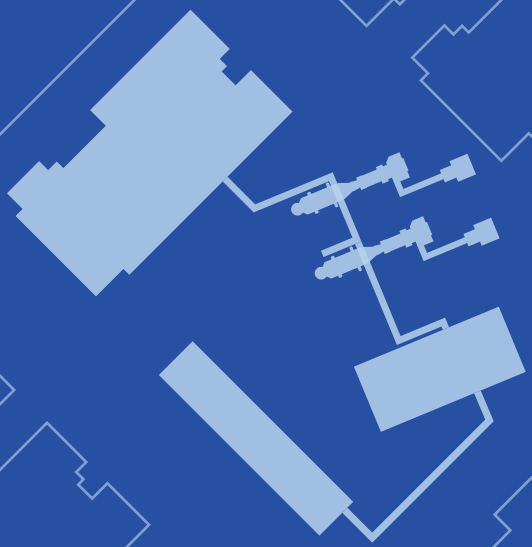
Construction Permit Application Part 1

Blue Energy Center - Port of Victoria Nuclear Unit 1: Preliminary Safety Analysis Report

Non-Proprietary

BE-POV-LCNRG-LIC-5009

Revision 0



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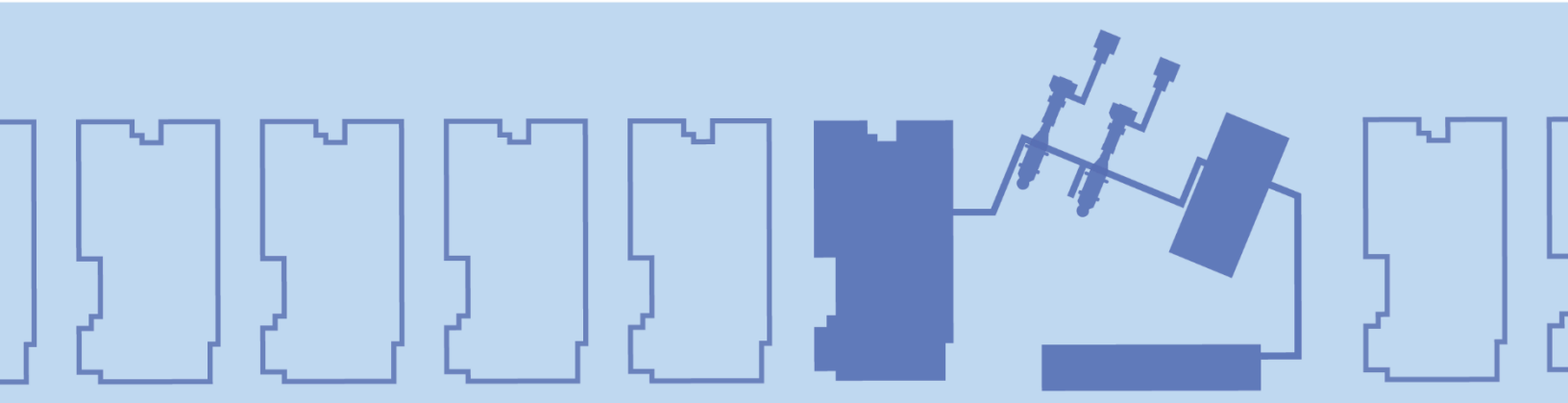
BE-POV-LCNRG-LIC-5009

Revision 0

September 2026

Non-Proprietary Information

Chapter 1
Introduction and General Plant Description



ACRONYM TABLE

Acronym	Explanation
ABWR	Advanced Boiling Water Reactor
AC	Alternating Current
ADS	Automatic Depressurization System
ALARA	As Low As Reasonably Achievable
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
ARI	Automatic Rod Insertion
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
BOP	Balance of Plant
BPVC	Boiler and Pressure Vessel Code
BTP	Branch Technical Position
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
CFS	Condensate and Feedwater Heating System
CHF	Critical Heat Flux
CIV	Containment Isolation Valve
CP	Construction Permit
CPR	Critical Power Ratio
CR	Control Room
CRD	Control Rod Drive
CWS	Circulating Water System
D-in-D	Defense-in-Depth
DBA	Design Basis Accident
DC	Direct Current
DEC	Design Extension Condition
DRG	Design Review Guide

Acronym	Explanation
EAB	Exclusion Area Boundary
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
ESBWR	Economic Simplified Boiling Water Reactor
ESF	Engineered Safety Feature
ESP	Early Site Permit
FMCRD	Fine Motion Control Rod Drive
FP	Fire Protection
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
GVH	GE Vernova Hitachi Nuclear Energy
GMRS	Ground Motion Response Spectra
GNF	Global Nuclear Fuel
GT	Gamma Thermometer
GWM	Gaseous Waste Management System
HPCS	High Pressure Core Spray
HVS	Heating, Ventilation, and Cooling System
I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency
IC	Isolation Condenser
I&C	Instrumentation and Control
ICS	Isolation Condenser System
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronics Engineers
INPO	Institute of Nuclear Power Operations
ISG	Interim Staff Guidance
ISI	Inservice Inspection

Acronym	Explanation
IST	Inservice Testing
LOCA	Loss-of-Coolant Accident
LOCV	Loss of Condenser Vacuum
LPZ	Low Population Zone
LTR	Licensing Topical Report
LWR	Light Water Reactor
MCPR	Minimum Critical Power Ratio
MCR	Main Control Room
MOV	Motor Operated Valve
MSIV	Main Steam Isolation Valve
N/A	Not Applicable
NBS	Nuclear Boiler System
NEI	Nuclear Energy Institute
NPP	Nuclear Power Plant
NSSS	Nuclear Steam Supply System
OL	Operating License
OPEX	Operating Experience
PAM	Post-Accident Monitoring
PCCS	Passive Containment Cooling System
PORV	Power Operated Relief Valve
PPS	Plant Pneumatics System
PRA	Probability Risk Assessment
PSAR	Preliminary Safety Analysis Report
PWR	Pressurized Water Reactor
QA	Quality Assurance
QAPD	Quality Assurance Program Description
RB	Reactor Building
RCIC	Reactor Core Isolation Cooling

Acronym	Explanation
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RG	Regulatory Guide
RIPB	Risk-Informed Performance-Based
RIV	Reactor Isolation Valve
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RTNSS	Regulatory Treatment of Non-Safety Systems
RWST	Refueling Water Storage Tank
SA	Severe Accident
SAFDL	Specified Acceptable Fuel Design Limits
SAR	Safety Analysis Report
SBO	Station Blackout
SC	Steal-Plate Composite
SC1	Safety Class 1
SCCV	Steel-Plate Composite Containment Vessel
SCN	Non-Safety Class
SCR	Secondary Control Room
SDC	Shutdown Cooling System
SDG	Standby Diesel Generator
SGI	Safeguards Information
SI	International System of Units
SLCS	Standby Liquid Control System
SMR	Small Modular Reactor
SRP	Standard Review Plan
SRV	Safety Relief Valve
SSCs	Structures, Systems, and Components
SSI	Soil-Structure Interaction

Acronym	Explanation
TAF	Top of Active Fuel
TBS	Turbine Bypass System
TMI	Three Mile Island
TS	Technical Specifications
TSC	Technical Support Center
UHS	Ultimate Heat Sink
UPS	Uninterruptible Power Supply
USCS	United States Customary System
USNRC	United States Nuclear Regulatory Commission

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1.0 INTRODUCTION AND GENERAL PLANT DESCRIPTION

1.1 Introduction

This Preliminary Safety Analysis Report (PSAR) supports the Blue Energy Global Inc. (Blue Energy) Construction Permit Application (CPA) for a single unit, Small Modular Reactor (SMR) plant at the Blue Energy Center - Port of Victoria (BECPV) site. The plant is designated as BECPV Nuclear Unit 1 (N01). The purpose of the PSAR is to describe the proposed facility and the safety analyses that have been performed to evaluate whether the plant can be constructed and operated without undue risk to the health and safety of the public. The PSAR content is consistent with regulatory requirements provided in Title 10 of the Code of Federal Regulations (CFR) Section 50.34.

Blue Energy is a privately held, domestically owned Delaware corporation and the ultimate parent of the Project entities, serving as the controlling applicant and operator of BECPV-N01. Sister 1 LLC, a wholly owned subsidiary of Blue Energy held through Blue Energy DevCo LLC and River 1 LLC, is the special purpose entity that will serve as the NRC-licensed owner of BECPV-N01 and will own or control all aspects of the nuclear Project prior to issuance of an operating license. The BECPV is a 245-acre site located in Victoria County, Texas, adjacent to the Victoria Barge Canal within the Port of Victoria industrial complex. The Applicant is pursuing a phased generation strategy at the BECPV, under which a combined-cycle gas turbine (CCGT) plant is developed and financed prior to, and separate from, the Project, to enable near-term power delivery and project financing. The Project is developed and licensed as its own phase of the overall BECPV. The balance-of-plant (BOP) and fire protection system (FPS) are initially built to support CCGT plant operations with a retained optionality to transition to support the BECPV-N01. This approach and financing model is jointly designated as the Applicant's Gas2Nuclear (or G2N) approach. This BOP resequencing and subsequent transition to nuclear plant option is assumed to be realized in this CPA.

This PSAR is based on the construction of a BWRX-300 SMR designed by GE Vernova Hitachi Nuclear Energy (GVH) with a nominal electrical output of 300 MWe.

Commercial operation of BECPV-N01 is targeted for 2032. If Blue Energy decides to build additional units at BECPV the request for additional units will be submitted as another application.

1.1.1 Format and Content

Regulatory Guide (RG) 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (NPPs) Light Water Reactor (LWR) Edition," provides guidance on the format and content of Safety Analysis Reports and is the basis of this PSAR. Because the PSAR supports a phased, two part, CPA, design information is preliminary and is sufficient to assess and demonstrate the safety of the design. The PSAR provided herein, in support of the CPA Part 1, has no site-specific information, which is reserved for Part 2. The PSAR will be updated in support of CPA Part 2 to include all information required for to support issuance of a construction permit. The PSAR will again be updated after issuance of a Construction Permit (CP) to a FSAR, which provides the design information needed to support an Operating License (OL). In situations where there is uncertainty as to whether material is to be provided in the PSAR or FSAR, a statement is provided to clarify information that will be presented in the FSAR.

Consistent with common licensing practice, most of the PSAR text is written in present tense, active voice, including discussions of facilities and programs not yet built or implemented.

Exceptions to this approach include discussion of completed studies and evaluations, which are written in the past tense. It should be understood, however, that statements regarding facilities (e.g., Structures, Systems, and Components (SSCs)) and pre-construction, preoperational, and operational activities (e.g.,

procedures and programs) typically address activities that have not yet been performed and will not be performed until it is reasonable and appropriate to do so.

1.1.1.1 Standard Review Plan

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," also referred to as the Standard Review Plan (SRP), provides regulatory guidance and acceptance criteria to support a United States Nuclear Regulatory Commission (USNRC) safety evaluation. A review of the facility against the SRP will be performed to support CPA Part 2.

Section numbering in this PSAR generally aligns with the SRP numbering with significant differences as follows:

- Chapter 7, Instrumentation and Controls is presented in terms of the "Design Review Guide (DRG): Instrumentation and Controls for Non-Light-Water Reactor (Non-LWR) Reviews," (Reference 1.1-1), in lieu of SRP Section 7
- Chapter 15, Transient and Accident Analysis is presented in terms of International Atomic Energy Agency (IAEA) SSG-61, "Format and Content of the Safety Analysis Report for Nuclear Power Plants," (Reference 1.1-2), in lieu of SRP Section 15
- Chapter 18, Human Factors Engineering, as defined by SRP Section 18, will be provided in the FSAR
- Chapter 19, Severe Accidents, as defined by SRP Section 19, will be provided in the FSAR. Section 15.6 provides risk insights available at the time of the PSAR submittal.

1.1.1.2 Tables, Figures, and Page Numbering

Tables, figures, and page numbering correspond to the PSAR section number at the "X.Y" level followed by a specific numerical identification, (e.g., Table 1.2-3 refers to the 3rd table in Section 1.2).

Summary information is presented in tables. Tables are located at the end of the applicable section.

Figures, graphs, and diagrams are provided with the level of detail needed to support the safety analysis. Thus, the PSAR does not provide detailed engineering drawings or figures drawn to scale. Figures are located at the end of the applicable section following the tables.

1.1.1.3 Numerical Information

Calculation output and data are provided with the number of significant figures that reflect the accuracy of the calculation and data. Similarly, values of design parameters such as power output or component dimensions are "nominal" rather than "as-built" values. Units are presented in United States Customary System (USCS), International System of Units (SI), or both.

1.1.1.4 Organization

The PSAR is organized into the following chapters:

<u>Chapter 1: Introduction and General Plant Description</u>

Chapter 1 summarizes the principal aspects of the design, conformance with regulatory requirements and guidance, and material that is incorporated by reference in the PSAR.

<u>Chapter 2: Site Characteristics</u>

Chapter 2 provides site characteristics and siting parameters for the BECPV.

<u>Chapter 3: Design of Structures, Systems, and Components</u>

Chapter 3 describes the classification of SSCs as well as compliance with General Design Criteria (GDC) and Principal Design Criteria (PDC).

Chapter 4: Reactor

Chapter 4 describes the design of the nuclear fuel, reactor, and reactor core, including the fuel rods, fuel assemblies, reactivity control system, nuclear design, and thermal-hydraulic parameters.

Chapter 5: Reactor Coolant System and Connected Systems

Chapter 5 describes the Reactor Coolant System (RCS) and connected systems that form the Reactor Coolant Pressure Boundary (RCPB).

Chapter 6: Engineered Safety Features

Chapter 6 describes Engineered Safety Features (ESFs) provided to mitigate the consequences of postulated accidents.

Chapter 7: Instrumentation and Controls

Chapter 7 describes Instrumentation and Control (I&C) systems used for normal, abnormal, and accident conditions.

Chapter 8: Electric Power

Chapter 8 describes the Alternating Current (AC) and Direct Current (DC) power systems and power requirements for normal, abnormal, and accident conditions.

Chapter 9: Auxiliary Systems

Chapter 9 describes systems used to support fuel storage and handling, normal cooling water, process auxiliaries (such as compressed air, sampling, water chemistry, containment inerting), Heating, Ventilation and Cooling System (HVS), fire protection, power auxiliaries, communications, and lighting.

Chapter 10: Steam and Power Conversion System

Chapter 10 describes the systems used for steam and power conversion with a representative heat balance presented.

Chapter 11: Radioactive Waste Management

Chapter 11 describes the capabilities of the plant to control, collect, handle, process, store, and dispose of liquid, gaseous, and solid wastes that may contain radioactive materials, and the instrumentation used to monitor and control the release of radioactive effluents and wastes.

Chapter 12: Radiation Protection

Chapter 12 describes policy, design and operational considerations that ensure occupational radiation exposure to personnel is kept As Low As Reasonably Achievable (ALARA).

Chapter 13: Conduct of Operations

Chapter 13 describes the Blue Energy organizational structure, programs and procedures developed to support safe plant operation during normal, abnormal, and accident conditions, and provisions for establishing and maintaining a qualified staff. Chapter 13 also describes emergency preparedness, physical security, and fitness for duty.

Chapter 14: Initial Test Program

Chapter 14 describes the initial test program for SSCs and design features to verify the capability to meet functional requirements.

Chapter 15: Safety Analyses

Chapter 15 describes the results of the safety analyses based on a Hazard Analysis, Deterministic Safety Analysis, and Probabilistic Safety Assessment.

Chapter 16: Technical Specifications

Chapter 16 provides the methodology for developing Technical Specifications (TS) and associated bases that ensure compliance with the safety analysis inputs, assumptions, and results.

Chapter 17: Quality Assurance

Chapter 17 describes the quality assurance program used during the design and construction of BECPV-N01 to ensure conformance with regulatory requirements and the design bases specified in the CPA.

1.1.2 References

1.1-1 “Design Review Guide (DRG): Instrumentation and Controls for Non-Light-Water Reactor (Non-LWR) Reviews,” U.S. Nuclear Regulatory Commission, February 2021.

1.1-2 IAEA SSG-61: “Format and Content of the Safety Analysis Report for Nuclear Power Plants,” International Atomic Energy Agency, 2021.

1.2 General Plant Description

A summary description of the Blue Energy Center – Port of Victoria and the Applicant follows.

1.2.1 Principal Site Characteristics

The BECPV will be defined in Chapter 2 which will be included in CPA Part 2.

1.2.2 Plant Description

The BECPV Nuclear Unit 1 plant, a BWRX-300 design, is a water-cooled SMR with core heat removal by natural circulation. The BOP design is based on standard components in use throughout the BWR fleet adjusted for gross power output. The steam and power conversion system for the BWRX-300 consists of the following systems: Turbine Generator (TG), Main Steam subsystem (MS) of the Nuclear Boiler system (NBS), Main Condenser and Auxiliaries (MCA), Moisture Separator Reheater System (MSR), Turbine Gland Seal System (TGSS), Turbine Bypass System (TBS), Circulating Water System (CWS), Condensate Filters and Demineralizers (CFD), and the Condensate and Feedwater Heating System (CFS).

Fuel storage for new and spent fuel is in the fuel pool located in the Reactor Building (RB). The fuel storage racks are designed to hold a reload of fresh fuel and at least 5 years of spent fuel with space remaining for a full core of off-loaded fuel.

Systems performing Safety Category 1 functions are simple and driven by natural passive mechanisms based on fundamental physical and thermodynamic principles. In the event of a Design Basis Accident (DBA), reactor shutdown, Reactor Pressure Vessel (RPV) heat removal, RPV isolation, containment heat removal, and containment isolation are achieved without reliance on offsite or standby diesel generator AC power. Key safety features of the design are summarized, as follows:

- Passive plant: Safety Category 1 functions are not dependent on external sources of power nor operator action to control reactivity, remove heat from the fuel, and confine the radioactive material for 72 hours after a DBA.
- RPV design: The RPV height is increased, relative to a forced circulation BWR, with a tall chimney region. The natural circulation flow resulting from the increased height provides adequate thermal margins during power operations and off-normal conditions. RPV nozzles are located above the Top of Active Fuel (TAF).
- Reactor Isolation Valves (RIVs): Integral RIVs are directly flanged to the RPV nozzles on the reactor vessel. The RIVs allow LOCAs to be quickly isolated, thereby preserving the fluid inventory in the RPV for core cooling.
- Isolation condensers: The passive Isolation Condenser System (ICS) provides residual and decay heat removal, overpressure protection, and is the Emergency Core Cooling System (ECCS) in conjunction with the RIVs. The IC pools are the ultimate heat sink for the ICS, thereby cooling the reactor core.
- RCPB Safety and relief valves: Safety and relief valves for the RCPB are not required in the design. The ICS, in conjunction with the reactor scram function, provides overpressure protection for the RCPB.
- Containment design:
  - The containment is comprised of a Steel-Plate Composite (SC) Containment Vessel (SCCV), consisting of cylindrical wall, mat foundation, and top slab; a steel containment closure head, and other ASME Boiler and Pressure Vessel Code (BPVC), Section III, Division 1, Class MC components. The containment is enclosed within an RB that is largely below grade.

  - The containment is nitrogen inerted and has limited subcompartments with large openings. Analyses demonstrate the subcompartments are sufficiently mixed and that gases generated by radiolysis do not create a deflagration concern.
  - The Passive Containment Cooling System (PCCS) is a passive system for transferring heat by natural circulation from the containment atmosphere to the reactor cavity pool and equipment pool to maintain containment pressure and temperature within the design limits during accident conditions, or loss of active containment cooling provided by the Containment Cooling System. The reactor cavity pool and equipment pool can provide containment heat removal for 72 hours without operator action following a DBA.

- Containment Isolation Valves (CIV): The containment isolation function assures containment integrity while allowing the passage of fluids across the boundary as needed to mitigate a postulated accident. To support isolation of fluid lines penetrating containment, CIVs are included in each process system line and instrumentation line. Containment penetrations are designed to maintain structural integrity during environmental conditions resulting from a postulated accident and protect against leakage from the containment.
- Emergency Power Systems: The emergency power systems consist of three independent divisions of battery backed DC power. Only two of the three corresponding Emergency Power System divisions are required to mitigate a DBA.

  No offsite or standby diesel generator AC power is required to support SC1 loads. Emergency Power System batteries provide 72-hours' worth of power to loads supporting Safety Category 1 functions and Post-Accident Monitoring (PAM).

Table 1.2-1 provides key design characteristics of the BWRX-300.

1.2.3 General Arrangements of Major Structures and Equipment

General arrangement drawings illustrating major structures are provided in Figure 1.2-3 through Figure 1.2-15.

Table 1.2-1: Key Design Characteristics

Characteristic	Value
Intended purpose	Commercial electric power generation with load following capability
Reference site design	Single unit
Type of plant	Boiling Water Reactor (BWR)
Core coolant, moderator	Light water, light water
Nuclear Steam Supply System layout	Direct-Cycle
Primary circulation	Natural
Gross electrical output (nominal)	300 MWe
Reactor thermal power level (estimated)	870 MWth
Type of containment structure	Dry
Containment construction	Steel-plate composite
Normal Heat Sink	Closed loop cooling system
Ultimate Heat Sink	IC pools, fuel pool, reactor cavity pool, and equipment pool
Fuel enrichment	3.84 to 4.68 w/o U235
Fuel cladding	Zircaloy-2
Number of fuel bundles	240
RPV diameter (ID)	~ 4 m
RPV height (Inside)	~ 27 m
Spent fuel storage capacity	At least 5 years of full-power operation plus a full core
Emergency Power Supply	72 hr coping period
Design life	60 years

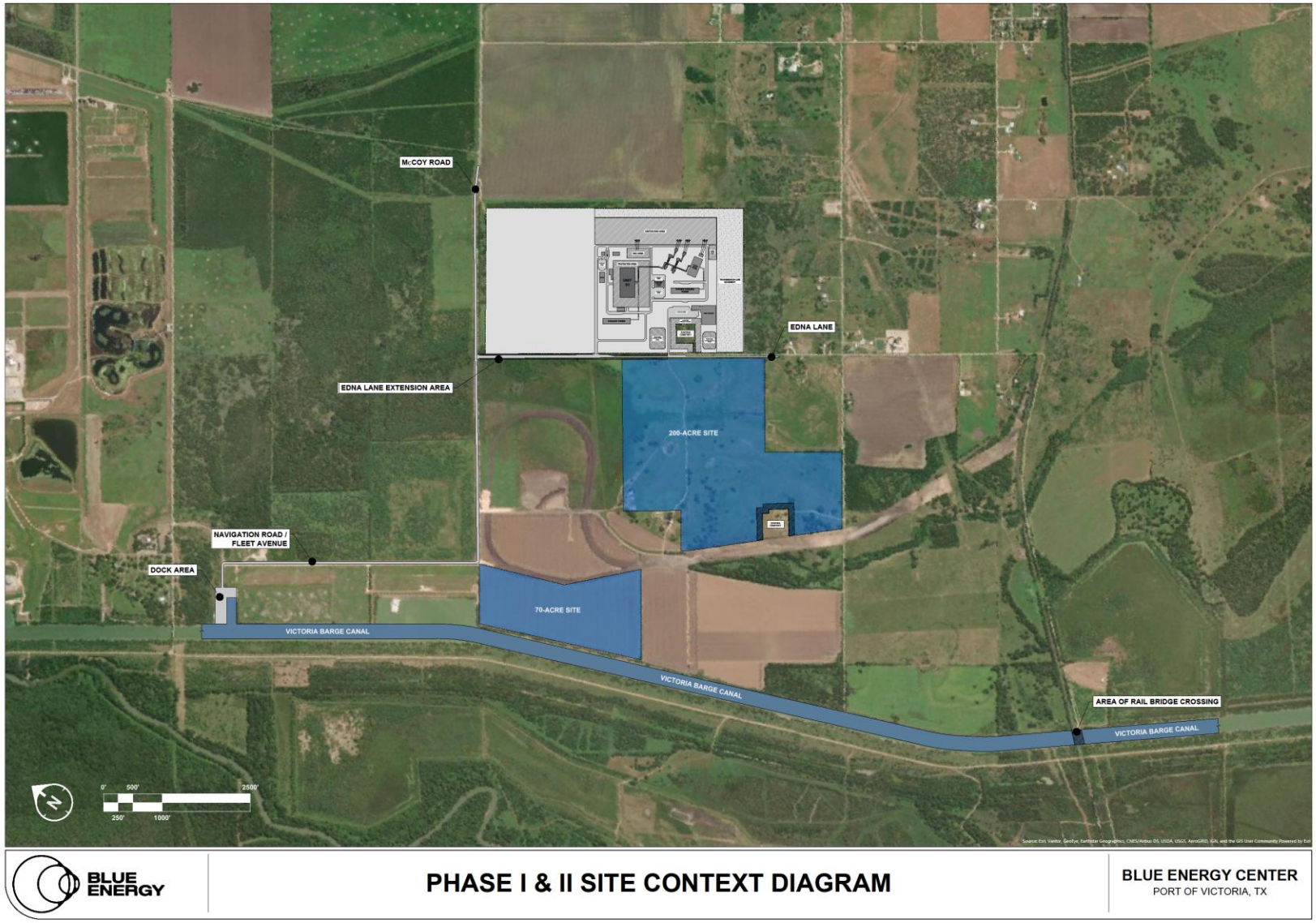


Figure 1.2-1a: Blue Energy Center – Port of Victoria Site

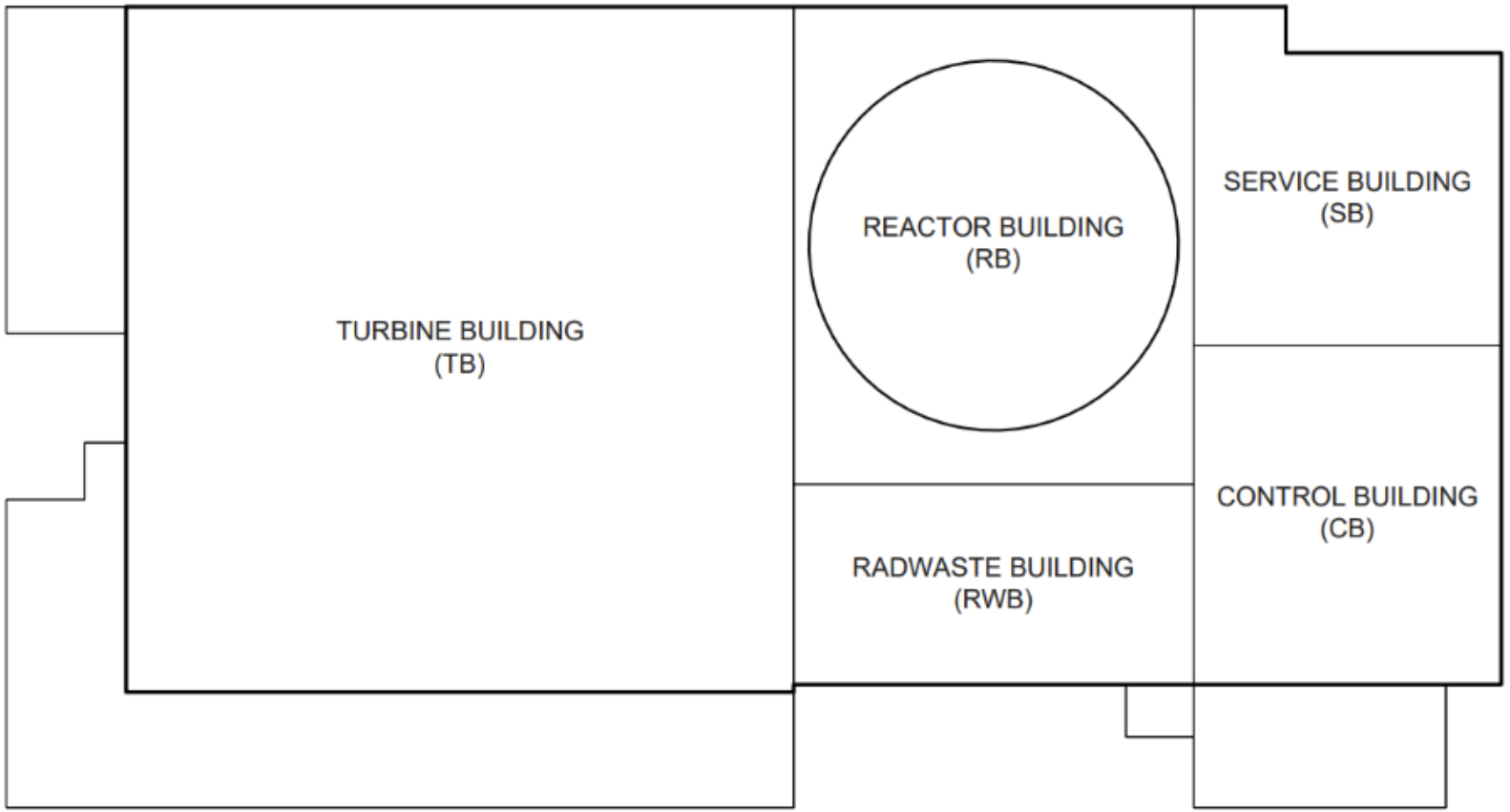


Figure 1.2-1b: BWRX-300 Power Block General Arrangement

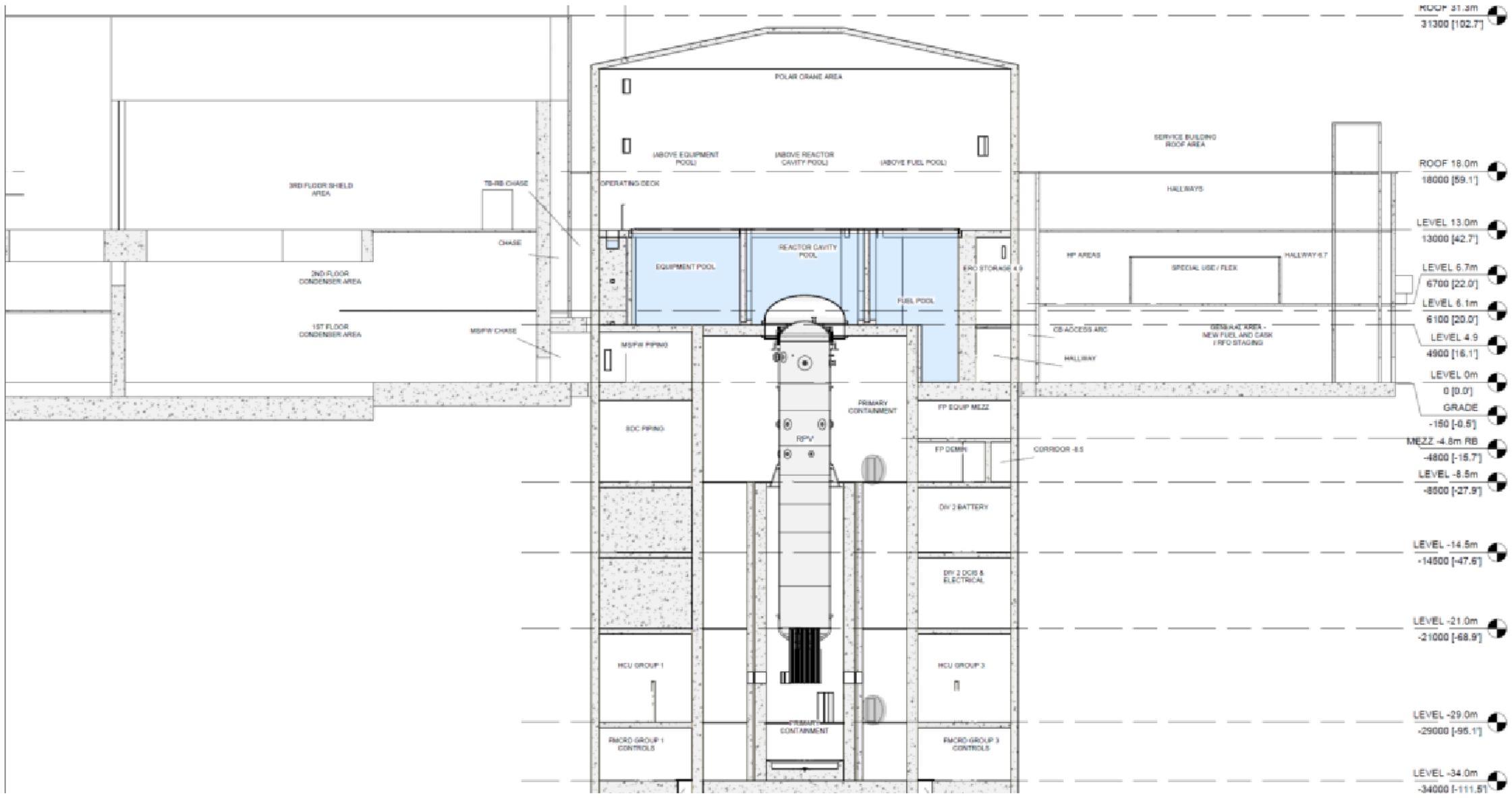


Figure 1.2-2: BWRX-300 Elevation View

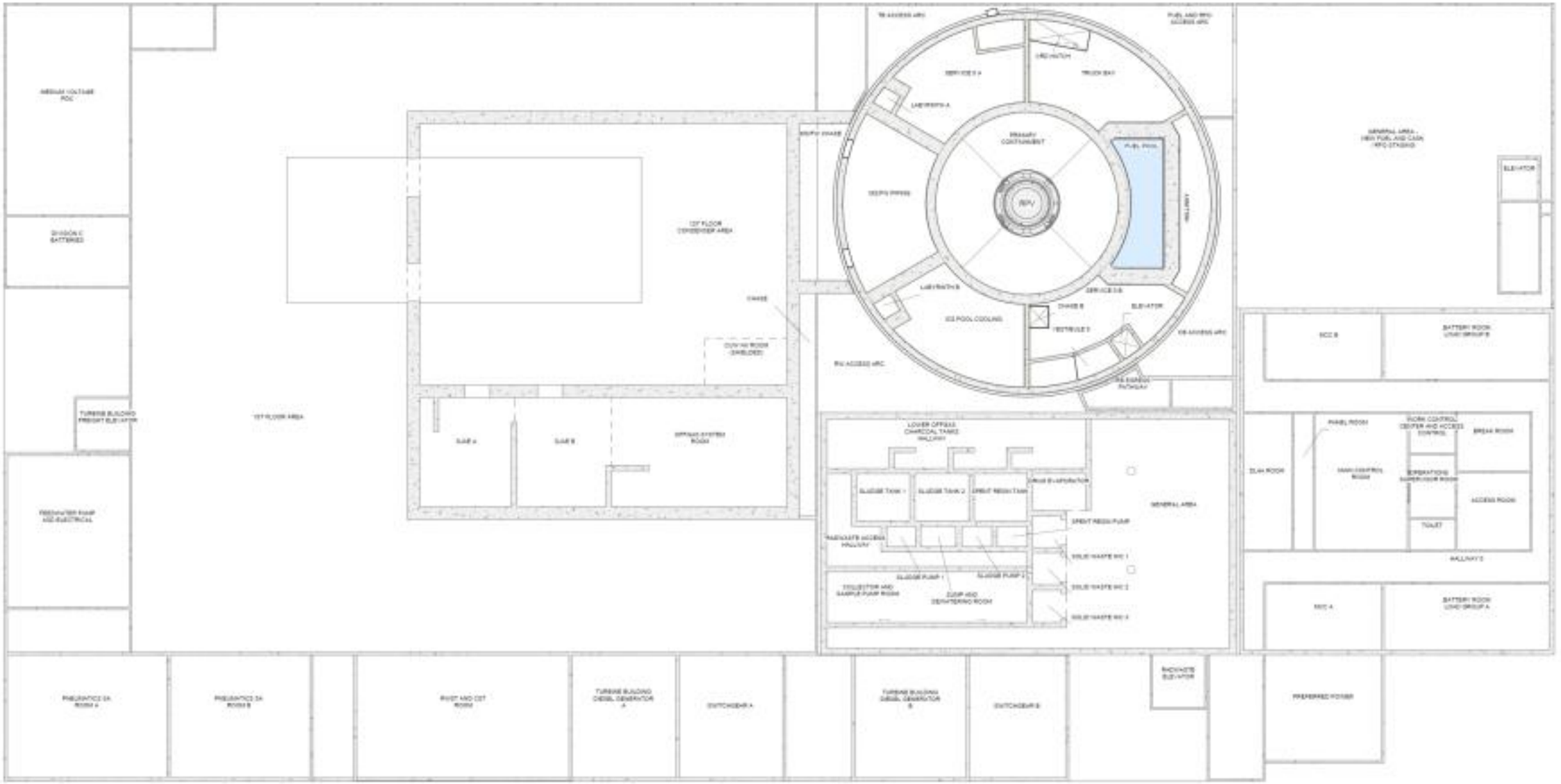


Figure 1.2-3: BWRX-300 Plan View Elevation 0m

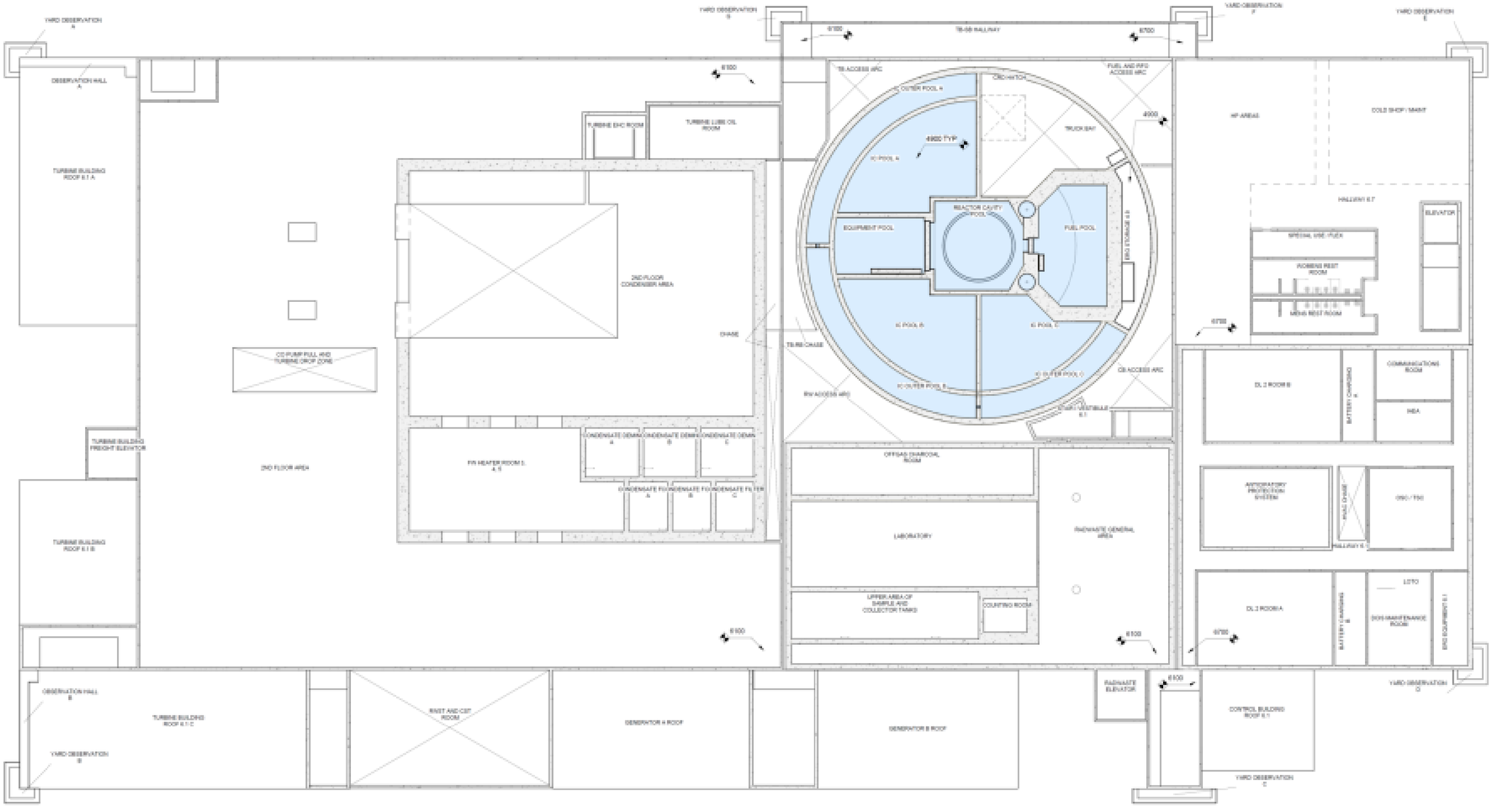


Figure 1.2-4: BWRX-300 Plan View Elevation 4.9m/6.1m/6.7m

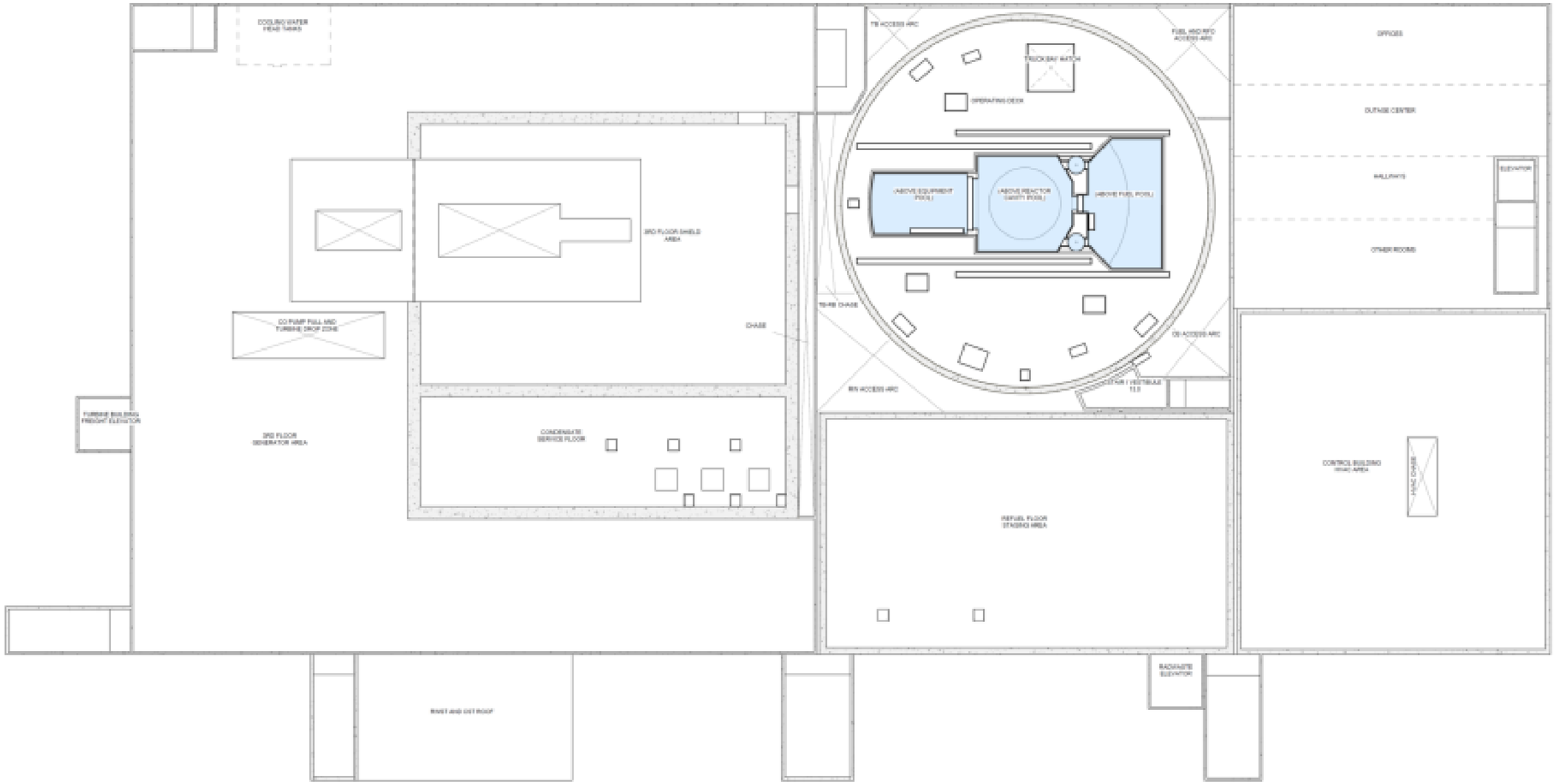


Figure 1.2-5: BWRX-300 Plan View Elevation 13.0m

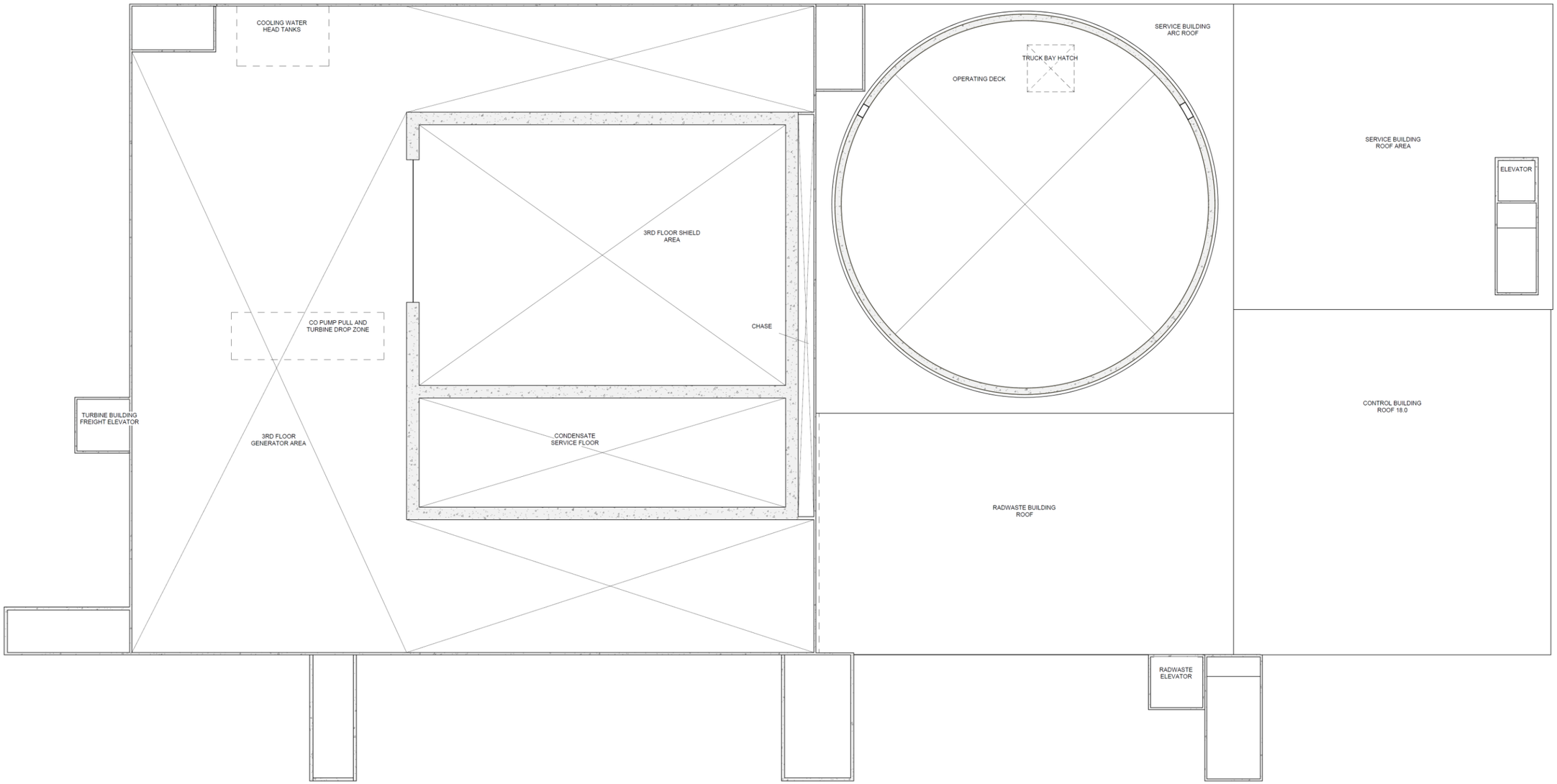


Figure 1.2-6: BWRX-300 Plan View Elevation 18.0m (Roof)

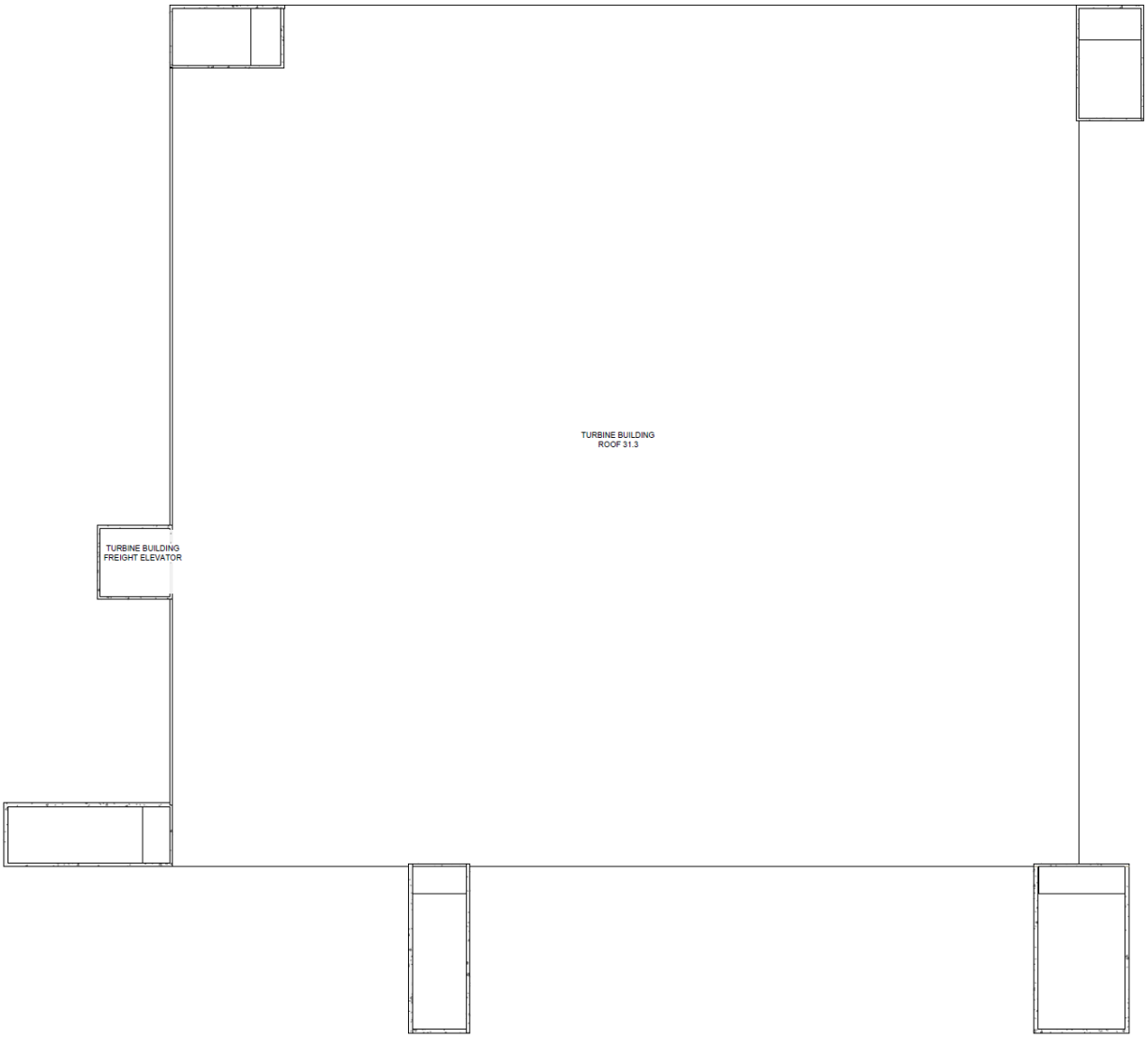


Figure 1.2-7: BWRX-300 Plan View Elevation 31.3 m (Roof)

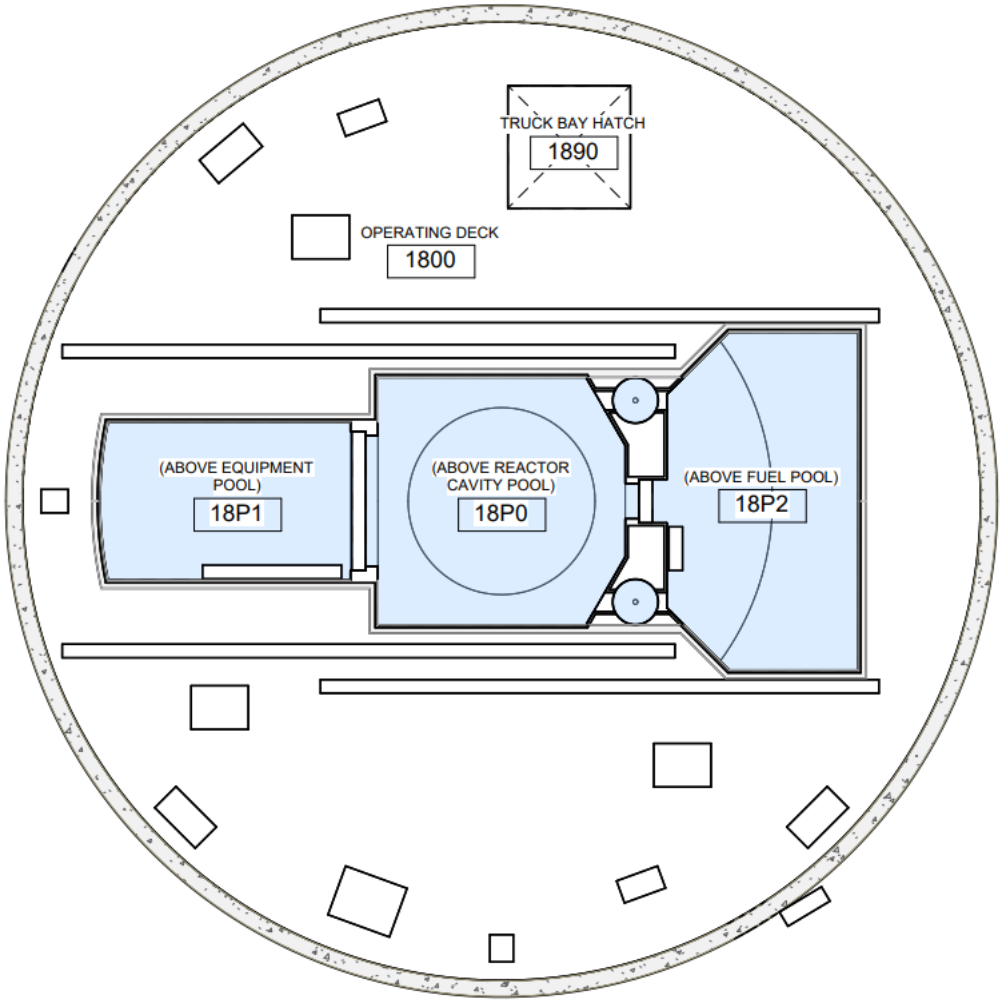


Figure 1.2-8: Reactor Building Plan View Elevation 13.0m

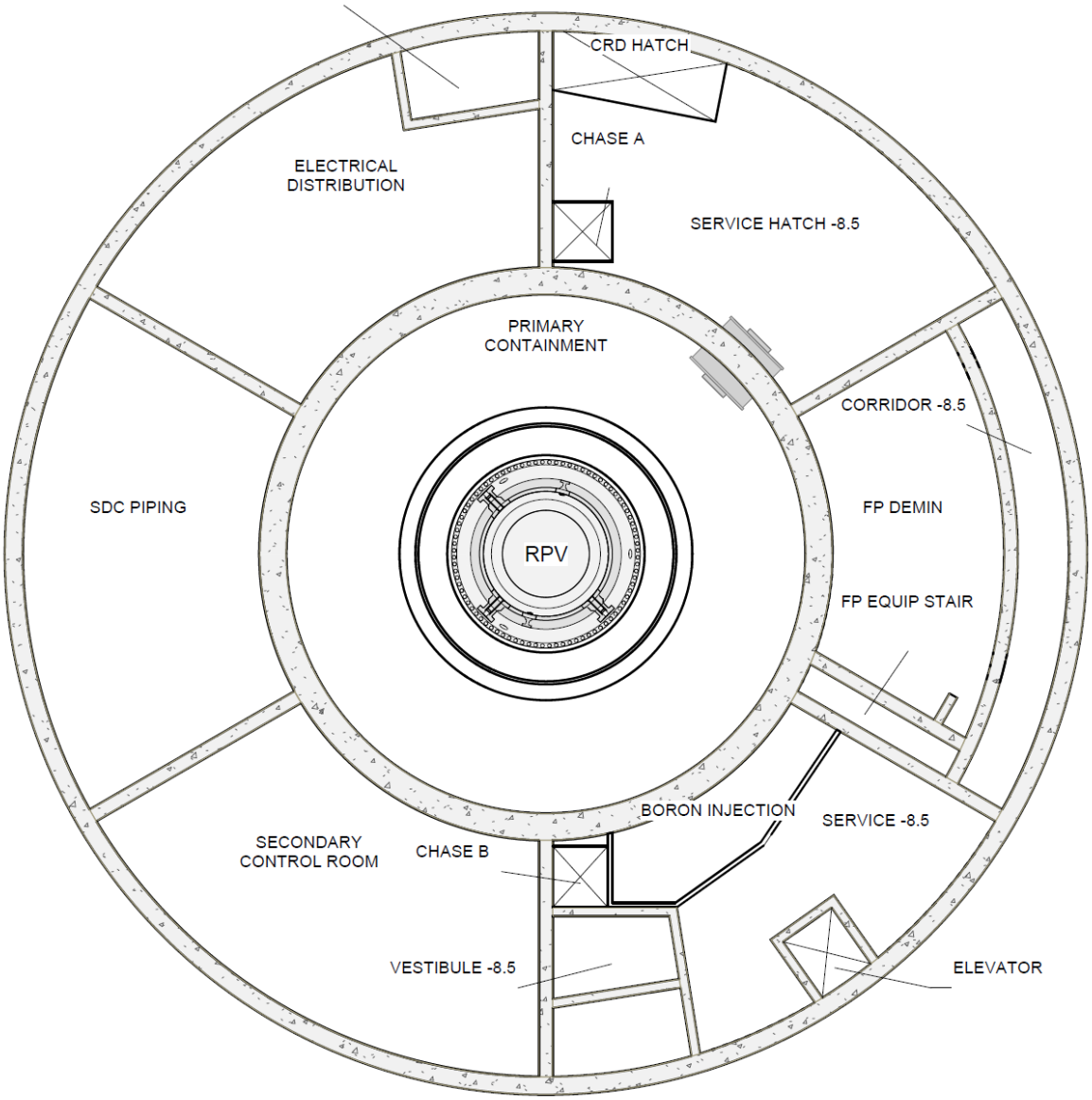


Figure 1.2-9: Reactor Building Plan View, Elevation -8.5m

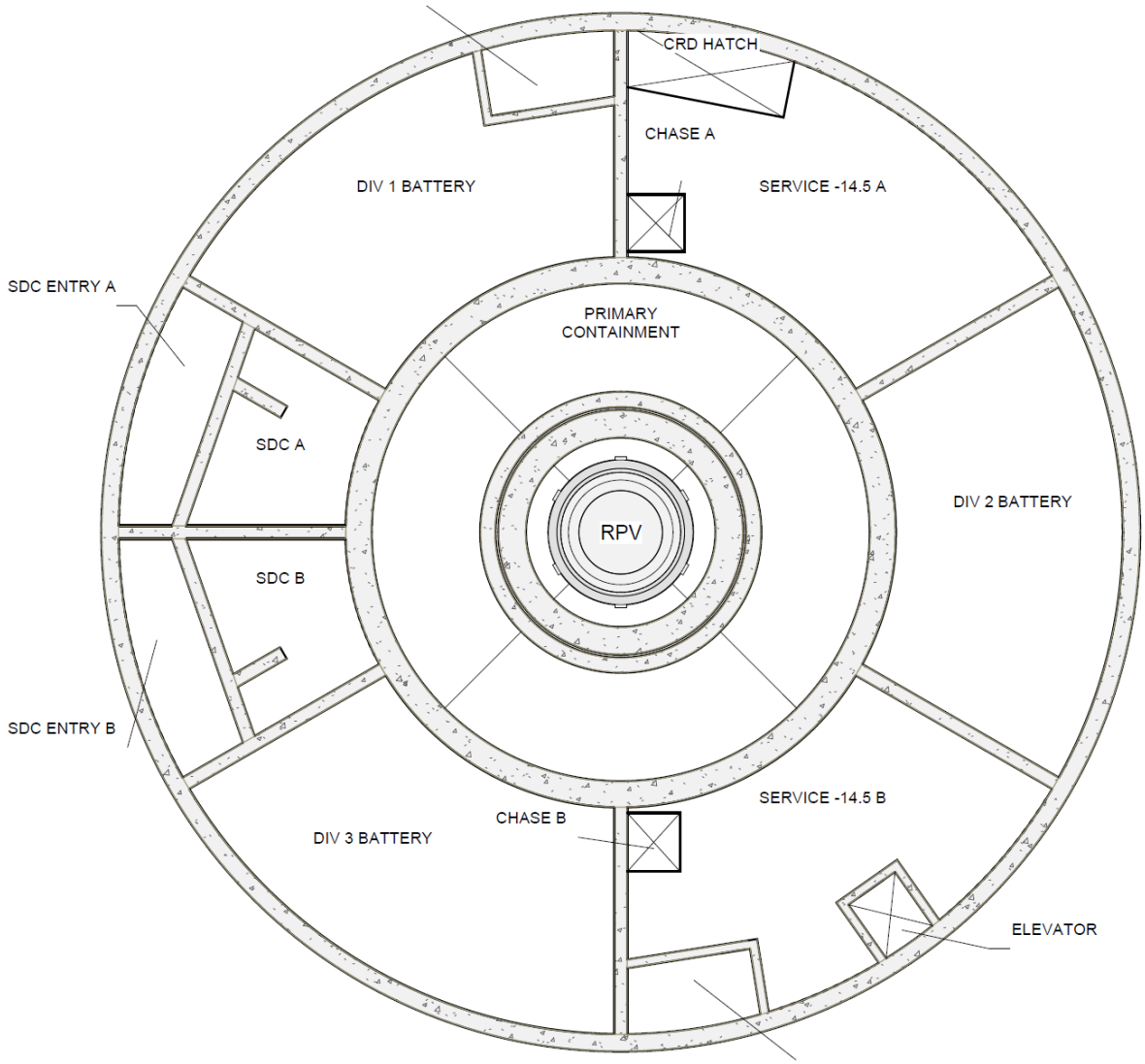


Figure 1.2-10: Reactor Building Plan View, Elevation -14.5m

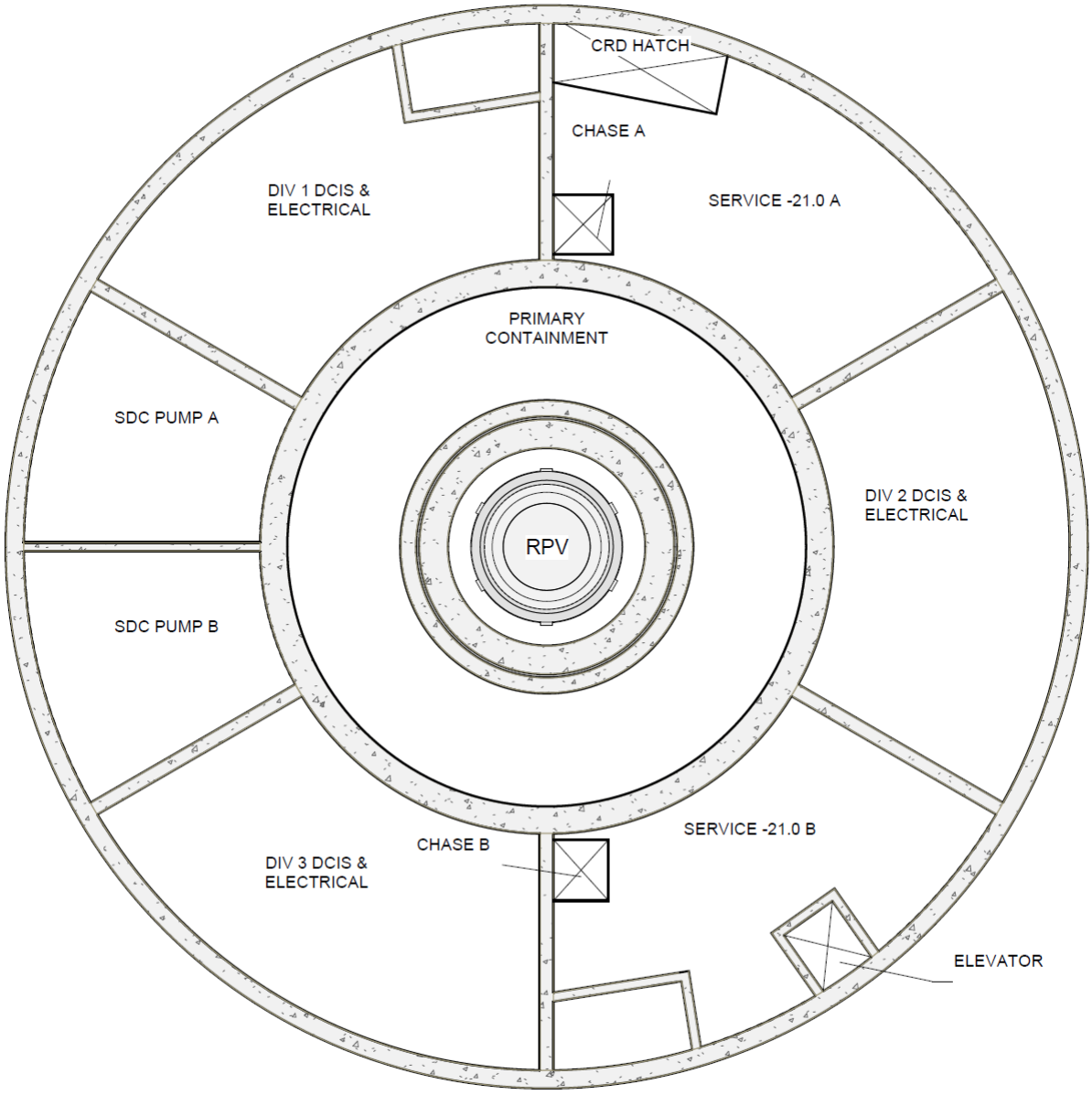


Figure 1.2-11: Reactor Building Plan View, Elevation -21.0m

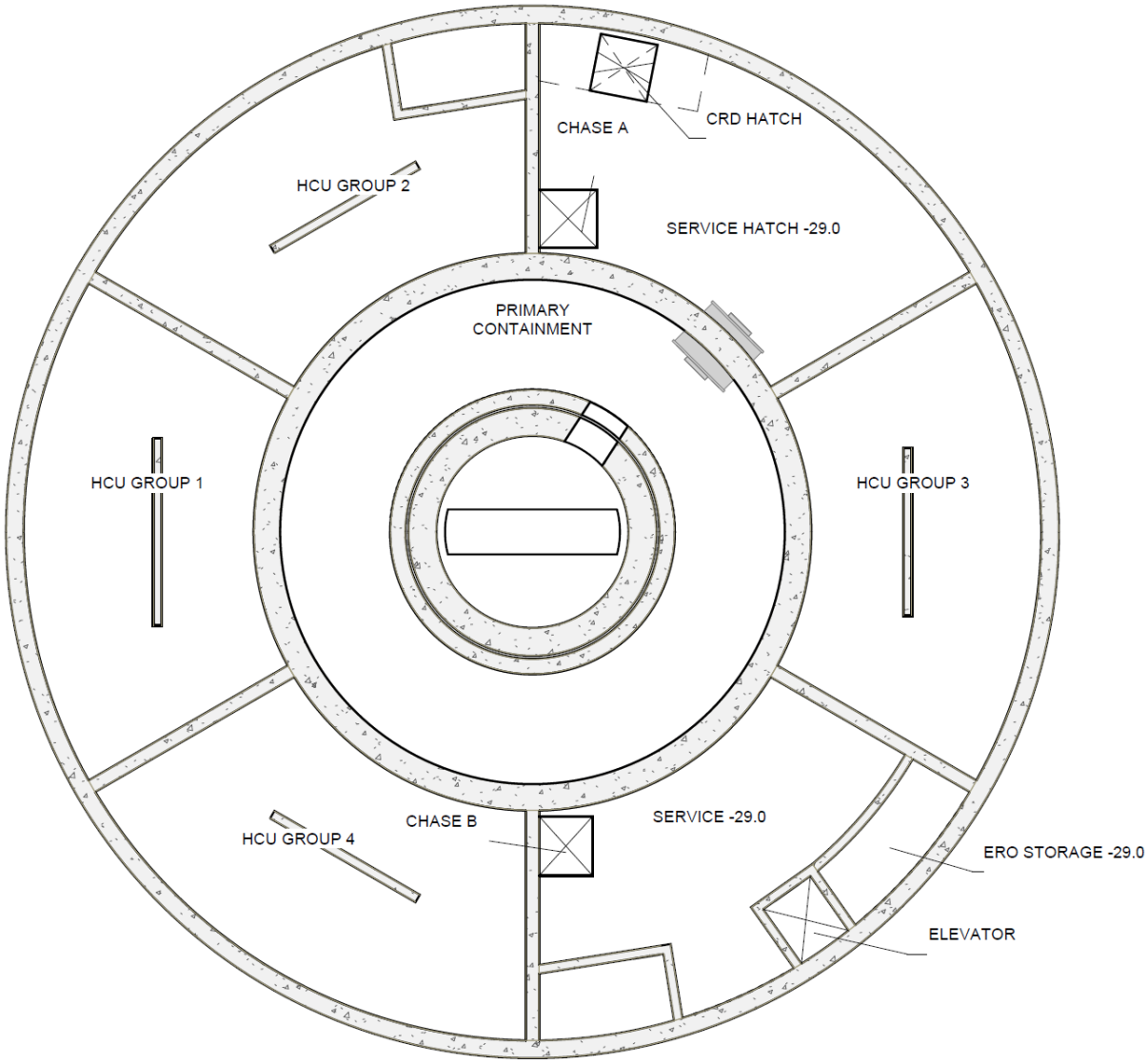


Figure 1.2-12: Reactor Building Plan View, Elevation -29.0m

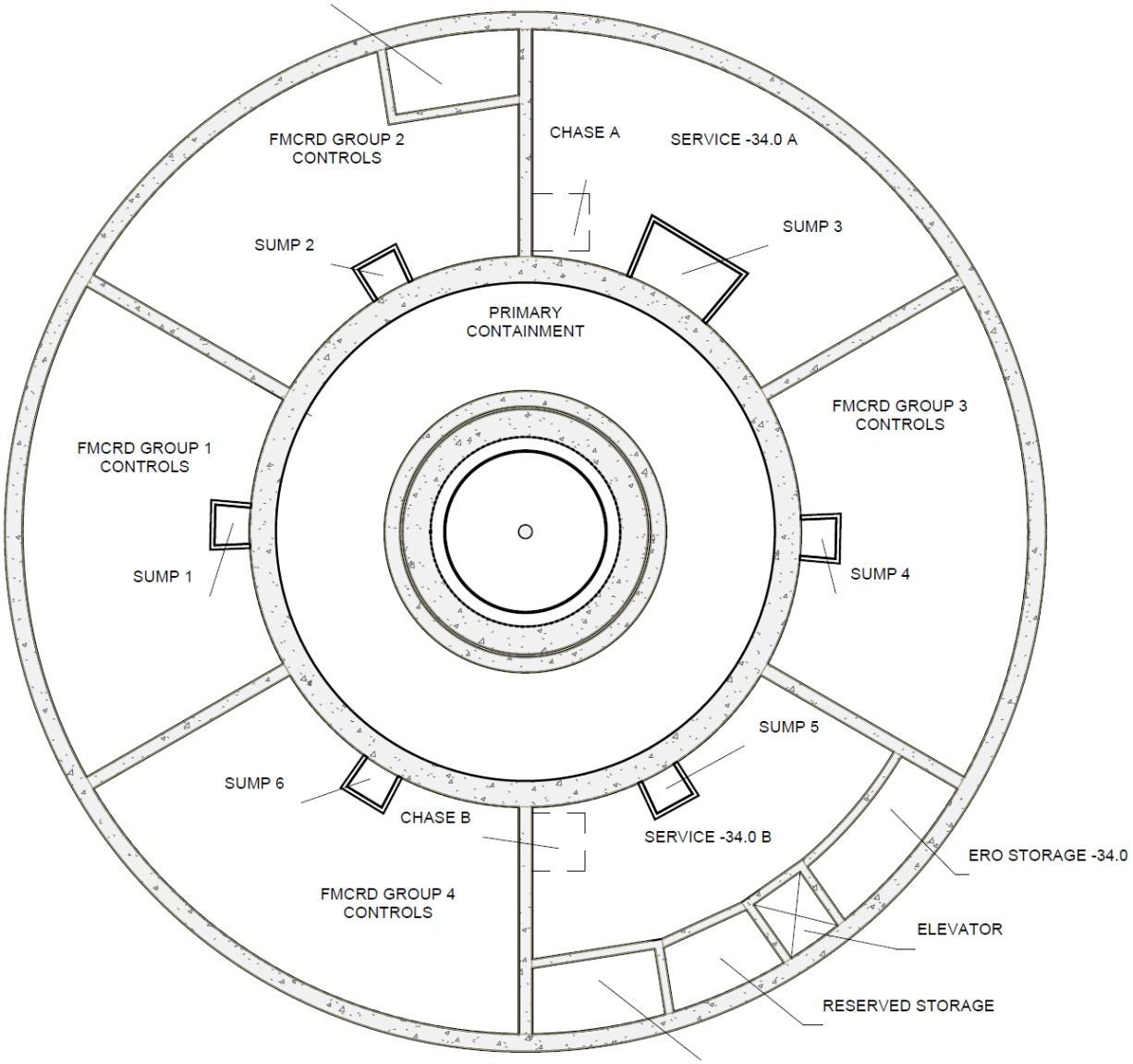


Figure 1.2-13: Reactor Building Plan View, Elevation -34.0m

1.3 Comparison with Similar Facility Designs

The BWRX-300 is the tenth generation of the BWR and is the simplest in the BWR product line. The BWRX-300 is the next evolution of the Economic Simplified Boiling Water Reactor (ESBWR) which received design certification (Appendix E to 10 CFR Part 52) and the Advanced Boiling Water Reactor (ABWR) which received design certification (Appendix A to 10 CFR Part 52).

The BWRX-300 incorporates the lessons learned in design, construction, operations, and maintenance from the many generations of BWRs that have been designed, built, and are currently operating or have been decommissioned. This includes the progress of the Darlington New Nuclear Project and the Clinch River Nuclear Site. The BWRX-300 is designed to enhance safety through simplification and reducing dependence on human intervention. This is achieved through increasing its reliance on natural circulation and natural phenomena-driven safety systems. These safety enhancements, in combination with its reduction in scale and complexity, enable reductions in operating staff, maintenance, and security requirements, as well as being easier to decommission.

Table 1.3-1 provides a comparison of facilities with similar designs.

Table 1.3-1: Comparison of Facilities with Similar Designs

Feature	ABWR	ESBWR	BWRX-300
Plant Type	Direct Cycle BWR	Direct Cycle BWR	Direct Cycle BWR
Plant Gross Electrical Output	~1,350 MWe	~1,600 MWe	~300 MWe
Reactor Coolant Recirculation	Reactor Internal Pumps	Natural Circulation	Natural Circulation
RPV Diameter (ID)	7.1 m	7.1 m	~4 m
RPV Height (Inside)	21.1 m	27.6 m	~27 m
Fuel Type	GE14	GNF14	GNF2
Number of Fuel Bundles	872	1,132	240
Fuel Rod Cladding Material	Zircaloy-2	Zircaloy-2	Zircaloy-2
Control Rod Blade Type	Cruciform B4C (optional Hf edges/tips)	Cruciform B4C (optional Hf edges/tips)	Cruciform B4C (optional Hf edges/tips)
Control Rod Drive (CRD) Type	Bottom Entry Fine Motion Control Rod Drive (FMCRD)	Bottom Entry FMCRD	Bottom Entry FMCRD
Number of Control Rods	205	269	57
Primary Containment Vessel Type	Reinforced Concrete Containment with steel liner	Reinforced Concrete Containment with steel liner	SCCV
Primary Containment Wetwell	Concrete Cylinder	Concrete Cylinder	None
ECCS	Active: 3-Divisional	Passive	Passive
Automatic Depressurization System (ADS)	Safety Relief Valves (SRVs)	SRVs and Depressurization Valves	RIVs and ICS in conjunction with reactor scram used for overpressure protection
Incore Monitor Calibration	Traversing Incore Probe System	Fixed, Incore Gamma Thermometers (GTs)	Fixed, Incore GTs

1.4 Identification of Agents and Contractors

1.4.1 Applicant/Program Manager

Blue Energy Global Inc. (Blue Energy) is the controlling applicant for the construction permit for BECPV-N01 at the Port of Victoria site in Victoria County, Texas and Sister 1 LLC (Sister 1) is the owner of this Project. Blue Energy is a privately held, domestically owned Delaware corporation led by a management team with substantial nuclear licensing, construction, and operations experience, including former senior NRC officials and executives previously with Exelon Corporation, Constellation Energy, Holtec International, TerraPower, and X-energy. Blue Energy is developing BECPV-N01 through a phased Gas2Nuclear strategy, under which non-safety-significant balance-of-plant infrastructure is constructed and initially energized as part of a combined-cycle gas turbine plant, with nuclear operations to follow in a subsequent phase.

1.4.2 Principal Contractors and Participants

Experience and qualifications of these agents and contractors, as well as the division of responsibility between Blue Energy and its agents and contractors is discussed below.

Blue Energy design and construction management teams provide oversight of design and construction activities prior to transitioning from the design/construction phase to the operational phase.

GVH is the reactor technology vendor and is responsible for design control of the Power Block. GVH provides engineering and technical services to the fleet of operating BWRs for the nuclear island, turbine island, and balance of plant equipment; plant life extension; power uprates; performance services; spare and renewal parts; outages; and inspections. GVH has proven experience, knowledge, and capability to design, manufacture, and furnish technical assistance and hardware for the construction, startup, and servicing of the BWRX-300. GVH will provide technical assistance for construction of BECPV-N01. Operating experience from design, construction, and operation of earlier GVH BWRs is applied in the design, construction, and operation of the BWRX-300.

1.4.2.1 GE Vernova Hitachi Nuclear Energy Americas, LLC

GVH is a business alliance of GE and Hitachi's respective nuclear businesses, established to serve the global nuclear industry. In addition, GE Nuclear Energy (predecessor to GVH) has designed, developed, constructed, and manufactured hardware for nuclear facilities since the 1950s, with the Dresden Unit 1 station being licensed in 1959 as the first United States commercial nuclear facility and the ABWR receiving the first design certification, followed by ESBWR design certification. GVH has designed and delivered more than 60 commercial nuclear power plants, both domestic and international. GVH is headquartered in Wilmington, North Carolina.

1.4.2.2 Construction Company

The contractor responsible for the construction of the project has not yet been decided and will be discussed in CPA Part 2.

1.4.2.3 Engineering and Design Company

The company responsible for the engineering and design of the project has not yet been decided and will be discussed in CPA Part 2.

1.5 Requirements for Further Technical Information

This section identifies, describes, and discusses those safety features or components for which further technical information is required in support of the issuance of a construction permit, but which have not been supplied with the CPA.

The descriptions of safety features or components in the PSAR do not require further technical information in support of the issuance of a construction permit. However, there are open items intended to be fulfilled by the CPA Part 2 submittal identified throughout the PSAR.

The Applicant recognizes that GVH is evaluating lessons learned from the ongoing construction of the Darlington BWRX-300 project in Ontario, Canada, with the objective of optimizing the design for improved constructability. The design reflected in the Applicant's PSAR is "frozen" and does not incorporate any of these optimizations. Accordingly, these potential optimizations do not give rise to a need for further technical information in support of the issuance of the construction permit. If the Applicant decides to pursue any design changes, the Applicant will first evaluate them to determine whether a formal licensing action is needed. If that evaluation determines that a formal licensing action is needed, the Applicant will either amend the application, include the change(s) in the CPA Part 2 submittal, incorporate the change(s) into the operating license application, or submit a license amendment request.

1.6 Material Referenced

Documents that are incorporated by reference into the PSAR are identified using the bases provided in Nuclear Energy Institute (NEI) 98-03, “Guidelines for Updating Final Safety Analysis Reports,” (Reference 1.6-1), which is endorsed by RG 1.181. Subsection 1.6.1 lists Topical Reports which provide BWRX-300 design information that supports the CPA.

1.6.1 Topical Reports

The following Topical Reports have been Incorporated By Reference:

- NEDC-33922P-A, “BWRX-300 Containment Evaluation Method,” (Reference 1.6-2), discusses the analysis method used for the thermal hydraulics performance to demonstrate that the containment design satisfies the acceptance criteria listed in Section 6.2
- BE-QAPD-001, “Quality Assurance Program Description” (Reference 1.6-3), establishes the Blue Energy QA program compliant with 10 CFR Part 50, Appendix B, 10 CFR Part 21, ASME NQA-1-2022, and RG 1.28, Revision 6, covering the full range of QA controls (organization, design control, procurement, inspection, test control, corrective action, records, and audits) for the current design and pre-construction phase, with a Phase 2 update planned to extend it through construction and operations.

Other topical or technical reports referenced in the PSAR provide design and analysis information that supports the CPA.

1.6.2 References

1.6-1 NEI 98-03, “Guidelines for Updating Final Safety Analysis Reports,” Nuclear Energy Institute, Rev 1, June 1999.

1.6-2 NEDC-33922P-A, “BWRX-300 Containment Evaluation Method,” Revision 3, GE Hitachi Nuclear Energy, Revision 3, June 2022.

1.6-3 Blue Energy Global Inc., Topical Report BE-QAPD-001-A, “Quality Assurance Program Description (QAPD),” Revision 2, December 2025 (ADAMS Accession No. ML25340A000).

1.7 Drawings and Other Detailed Information

The section summarizes the electrical diagrams and mechanical diagrams. Diagrams provided are simplifications of engineering drawings and provide the necessary information to support the safety analysis.

1.7.1 Electrical Diagrams

Table 1.7-1 lists simplified electrical diagrams provided in this PSAR. Figure 1.7-1 provides a common legend for these diagrams.

1.7.2 Mechanical Diagrams

Table 1.7-2 lists simplified mechanical diagrams provided in this PSAR. Figure 1.7-2 provides a common legend for these diagrams.

1.7.3 Other Detailed Information

Other detailed information is provided by reference in the applicable sections.

Table 1.7-1: Simplified Electrical Diagrams

Title	PSAR Figure
Main Electrical One-Line	Figure 8.1-1
Typical BWRX-300 Switchyard Interface	Figure 8.1-2
Preferred Power System Medium Voltage Bus A Train	Figure 8.3-1
Emergency Power System	Figure 8.3-2
Standby Power System Low Voltage Distribution Train A	Figure 8.3-3
Standby Power 250 VDC Subsystem	Figure 8.3-4
Preferred Power 250 VDC System Part One	Figure 8.3-5
Preferred Power 250 VDC System Part Two	Figure 8.3-6

Table 1.7-2: System Schematics and Simplified Diagrams

Title	PSAR Figure
Hydraulic Control Unit Simplified Diagram	Figure 4.6-8
Control Rod Drive Hydraulic Subsystem Simplified Diagram	Figure 4.6-9
BWRX-300 Reactor Coolant System Simplified Diagram	Figure 5.1-1
BWRX-300 Reactor Coolant Pressure Boundary Simplified Diagram	Figure 5.1-2
BWRX-300 Nuclear Boiler System Simplified Diagram	Figure 5.1-3
Shutdown Cooling System Simplified Diagram – Decay Heat Removal	Figure 5.4-2
Shutdown Cooling System Simplified Diagram – Overboarding	Figure 5.4-3
Reactor Water Cleanup System Simplified Diagram	Figure 5.4-4
Isolation Condenser System Simplified Diagram	Figure 6.3-1
Containment Monitoring Subsystem Simplified Diagram	Figure 6.8-1
Fuel Pool Cooling System Simplified Diagram	Figure 9.1-3
Service Water Subsystem Simplified Diagram	Figure 9.2-1
Plant Cooling Water System Simplified Diagram	Figure 9.2-2
Condensate Storage and Transfer Subsystem Simplified Diagram	Figure 9.2-4
Chilled Water Equipment Simplified Diagram	Figure 9.2-5
Isolation Condenser Pools Cooling and Cleanup System Simplified Diagram	Figure 9.2-6
Normal Heat Sink System Simplified Diagram	Figure 9.2-7
Refueling Water Storage and Cleanup Subsystem Simplified Diagram	Figure 9.2-8
Plant Pneumatics System Simplified Diagram	Figure 9.3-1
Normal Sump Simplified Diagram	Figure 9.3-2
Pressurized Sump Simplified Diagram	Figure 9.3-3
Oily Waste Sump Simplified Diagram	Figure 9.3-4
Containment Inerting System Simplified Diagram	Figure 9.3-5
Boron Injection System Simplified Diagram	Figure 9.3-6
Control Building Heating, Ventilation, and Cooling Simplified Diagram	Figure 9.4-1
Radwaste Building Heating, Ventilation, and Cooling Simplified Diagram	Figure 9.4-2
Turbine Building Heating, Ventilation, and Cooling and Plant Vent Stack Simplified Diagram	Figure 9.4-3
Reactor Building Heating, Ventilation, and Cooling Simplified Diagram	Figure 9.4-4
Secondary Control Room Heating, Ventilation, and Cooling Simplified Diagram	Figure 9.4-5
Containment Cooling System Simplified Diagram	Figure 9.4-6
BWRX-300 Fire Protection System Simplified Diagram	Figure 9.5-1

Title	PSAR Figure
Diesel Fuel Oil Storage and Transfer System Simplified Diagram	Figure 9.5-2
Simplified Flow Diagram with Representative Heat Balance of the Steam and Power Conversion System	Figure 10.1-1
Main Turbine Equipment System Simplified Diagram	Figure 10.2-1
Turbine Lube Oil System Simplified Diagram	Figure 10.2-2
Extraction Steam Subsystem Simplified Diagram	Figure 10.2-3
Electro-Hydraulic Control System Simplified Diagram	Figure 10.2-4
Main Steam Subsystem Simplified Diagram (RB & TB Portion)	Figure 10.3-1
Main Condenser and Auxiliaries System Simplified Diagram	Figure 10.4-1
Moisture Separator Reheater System Simplified Diagram	Figure 10.4-2
Turbine Gland Seal System Simplified Diagram	Figure 10.4-3
Turbine Bypass System Simplified Diagram	Figure 10.4-4
Circulating Water Supply Simplified Diagram	Figure 10.4-5
Condensate Filters and Demineralizers System Simplified Diagram	Figure 10.4-6
CFS Condensate Portion Simplified Diagram	Figure 10.4-7
CFS Feedwater Portion Simplified Diagram	Figure 10.4-8
CFS HP Heaters and Drains Simplified Diagram	Figure 10.4-9
CFS LP Heaters and Drains Simplified Diagram	Figure 10.4-10
Waste Collection and Filtering Subsystem Simplified Diagram	Figure 11.2-1
Waste Sampling Subsystem Simplified Diagram	Figure 11.2-2
Offgas System Simplified Diagram	Figure 11.3-1
Solid Waste Management System Simplified Diagram	Figure 11.4-1
Process Radiation Monitoring Subsystem Simplified Diagram	Figure 11.5-1

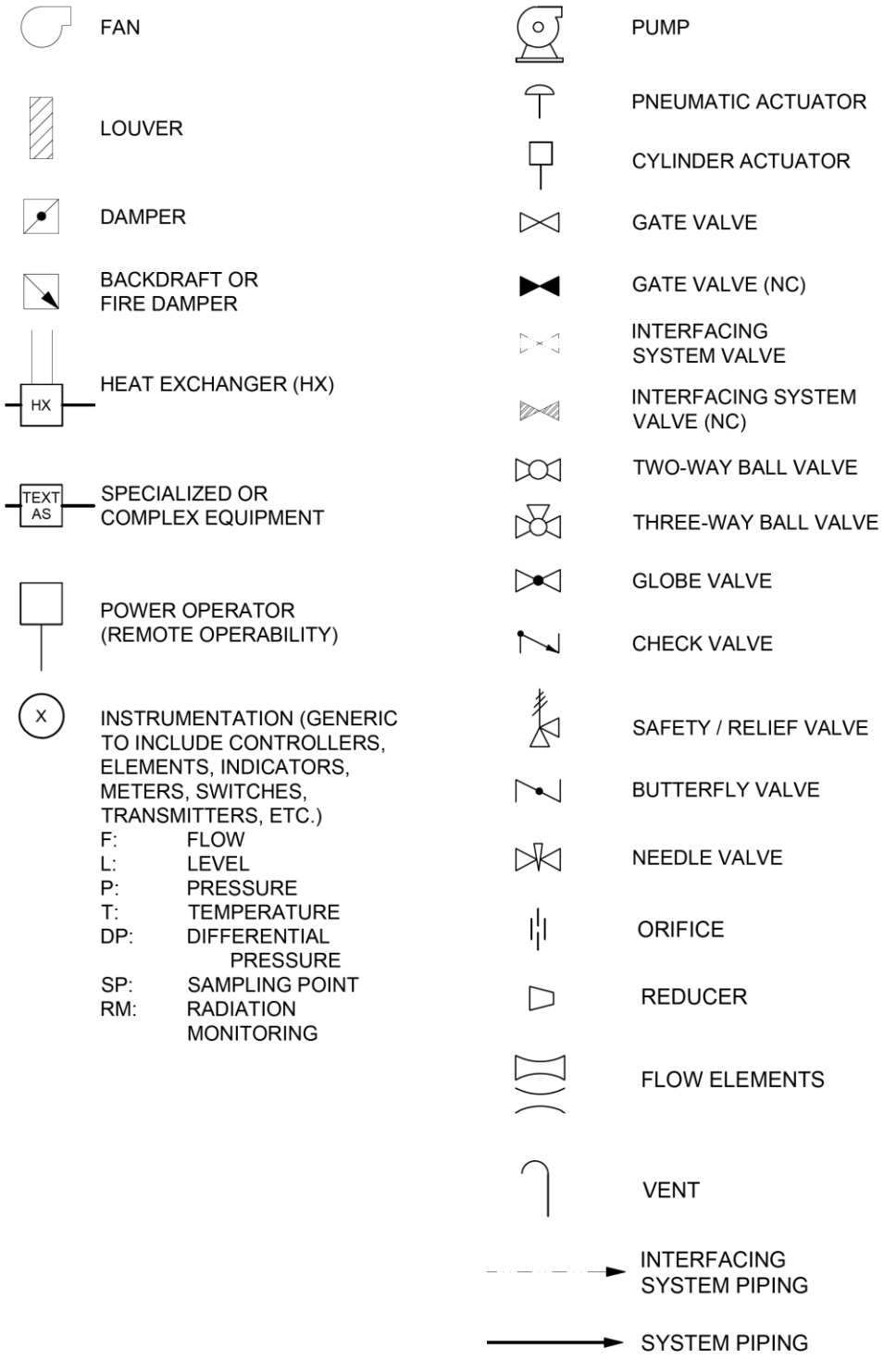


Figure 1.7-1: Simplified System Drawing Symbol Guide

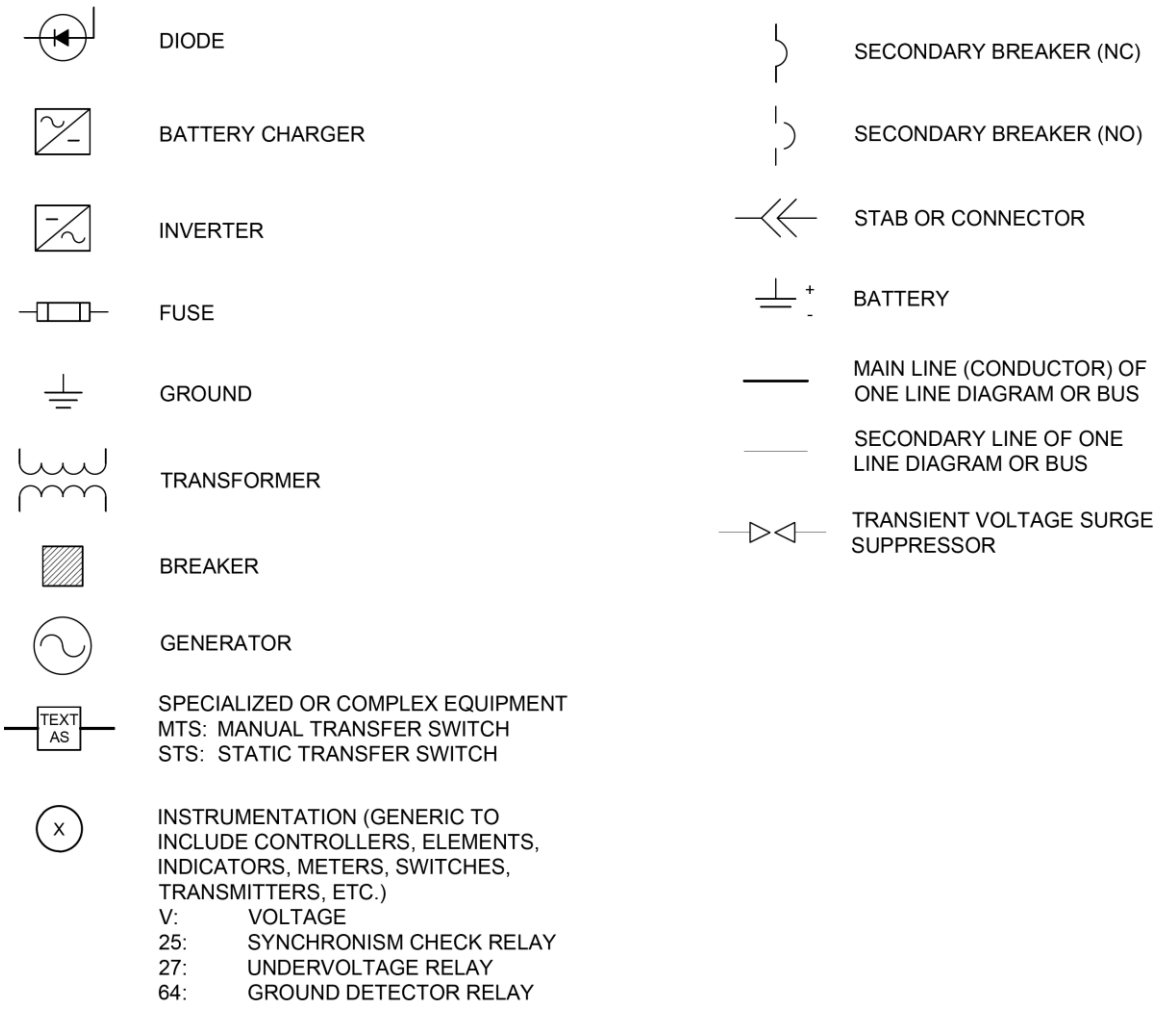


Figure 1.7-2: Simplified Electrical Mechanical Symbol Guide

1.8 Not Used

1.9 Regulatory Conformance

This section addresses compliance with regulations and conformance with regulatory guidance.

1.9.1 Compliance with General Design Criteria

Section 3.1 provides a summary of compliance to 10 CFR 50 Appendix A, General Design Criteria.

1.9.2 Conformance with Standard Review Plan

Tables 1.9-1 through 1.9-19 summarizing conformance with “SRP Acceptance Criteria” provided in NUREG-0800 will be provided in CPA Part 2.

The “Design Review Guide (DRG): Instrumentation and Controls for Non-Light-Water Reactor (Non-LWR) Reviews,” (Reference 1.9-1), is used in lieu of SRP Chapter 7. The DRG provides guidance for the USNRC to use in reviewing the I&C portions of applications for advanced non-LWRs within the bounds of existing regulations. Table 7.3-5, Table 7.3-6, Table 7.3-7, and Table 7.3-8 depict how the BWRX-300 design meets the DRG based on the equipment safety classification.

1.9.3 Conformance with Regulatory Guides

Table 1.9-20 summarizing conformance with Regulatory Guides will be provided in CPA Part 2.

1.9.4 Compliance with Three Mile Island Requirements

Table 1.9-21 summarizing conformance with TMI Requirements will be provided in CPA Part 2.

1.9.5 Conformance with Advanced and Evolutionary Light Water Reactor Design Issues

Conformance with Commission papers (SECY) and Staff Requirements Memoranda relating to advanced and evolutionary light water reactor design issues are addressed in the relevant sections.

1.9.6 References

1.9-1 “Design Review Guide (DRG): Instrumentation and Controls for Non-Light-Water Reactor (Non-LWR) Reviews,” U.S. Nuclear Regulatory Commission, February 2021.

Table 1.9-1: Conformance with NUREG-0800 (Chapter 1 Introduction and Interfaces)

To be provided in CPA Part 2.

Table 1.9-2: Conformance with NUREG-0800 (Chapter 2 Site Characteristics and Site Parameters)

To be provided in CPA Part 2.

Table 1.9-3: Conformance with NUREG-0800 (Chapter 3 Design of Structures, Components, Equipment, and Systems)

To be provided in CPA Part 2.

Table 1.9-4: Conformance with NUREG-0800 (Chapter 4 Reactor)

To be provided in CPA Part 2.

Table 1.9-5: Conformance with NUREG-0800 (Chapter 5 Reactor Coolant System and Connected Systems)

To be provided in CPA Part 2.

Table 1.9-6: Conformance with NUREG-0800 (Chapter 6 Engineered Safety Features)

To be provided in CPA Part 2.

Table 1.9-7: Conformance with Design Review Guide (Chapter 7 Instrumentation and Controls)

To be provided in CPA Part 2.

See Table 7.3-5, Table 7.3-6, Table 7.3-7, and Table 7.3-8 for how the BWRX-300 design meets the DRG based on the Safety Classification.

Table 1.9-8: Conformance with NUREG-0800 (Chapter 8 Electric Power)

To be provided in CPA Part 2.

Table 1.9-9: Conformance with NUREG-0800 (Chapter 9 Auxiliary Systems)

To be provided in CPA Part 2.

Table 1.9-10: Conformance with NUREG-0800 (Chapter 10 Steam and Power Conversion System)

To be provided in CPA Part 2.

Table 1.9-11: Conformance with NUREG-0800 (Chapter 11 Radioactive Waste Management)

To be provided in CPA Part 2.

Table 1.9-12: Conformance with NUREG-0800 (Chapter 12 Radiation Protection)

To be provided in CPA Part 2.

Table 1.9-13: Conformance with NUREG-0800 (Chapter 13 Conduct of Operations)

To be provided in CPA Part 2.

Table 1.9-14: Conformance with NUREG-0800 (Chapter 14 Initial Test Program and ITAAC-Design Certification)

To be provided in CPA Part 2

Table 1.9-15: Conformance with NUREG-0800 (Chapter 15 Transient and Accident Analysis)

To be provided in CPA Part 2.

Table 1.9-16: Conformance with NUREG-0800 (Chapter 16 Technical Specifications)

To be provided in CPA Part 2.

Table 1.9-17: Conformance with NUREG-0800 (Chapter 17 Quality Assurance)

To be provided in CPA Part 2.

Table 1.9-18: Conformance with NUREG-0800 (Chapter 18 Human Factors Engineering)

To be provided in CPA Part 2.

Table 1.9-19: Conformance with NUREG-0800 (Chapter 19 Severe Accidents)

To be provided in CPA Part 2.

Table 1.9-20: Conformance with Regulatory Guides

To be provided in CPA Part 2.

Table 1.9-21: Compliance with Three Mile Island Requirements (10 CFR 50.34(f))

To be provided in CPA Part 2.

1.10 Nuclear Power Plants to Be Operated on Multi-Unit Site

BECPV-N01 is not co-located at the BECPV Site with an existing nuclear plant or a nuclear plant under construction. Thus, there are no construction hazards associated with co-located units.

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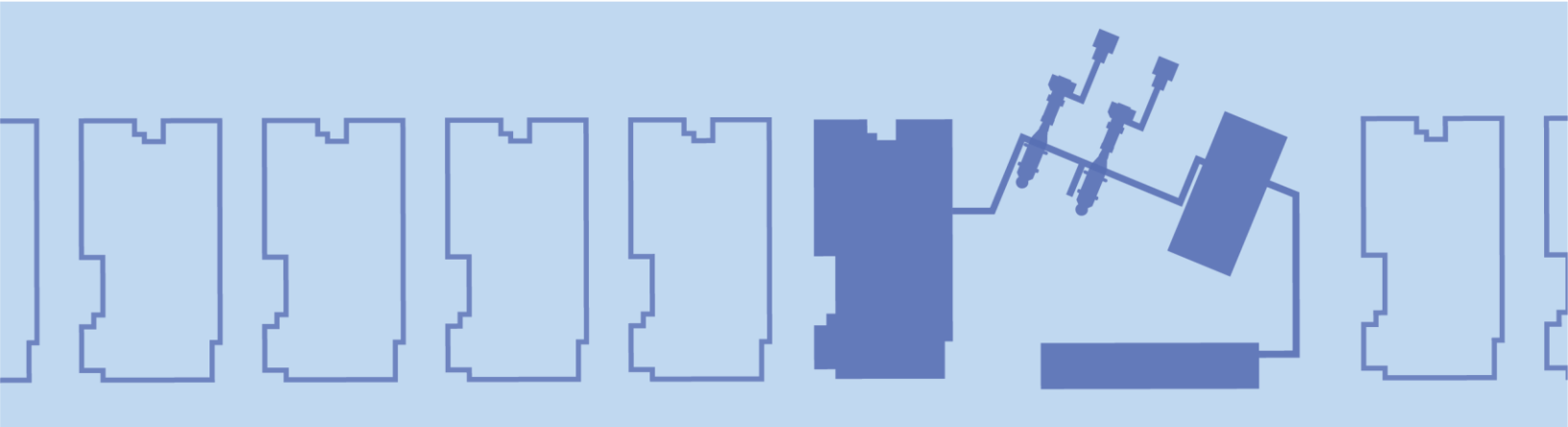
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Non-Proprietary Information

Chapter 2

Site Characteristics and Site Parameters



2.0 SITE CHARACTERISTICS AND SITE PARAMETERS

[Reserved] This section will be provided for CPA Part 2.

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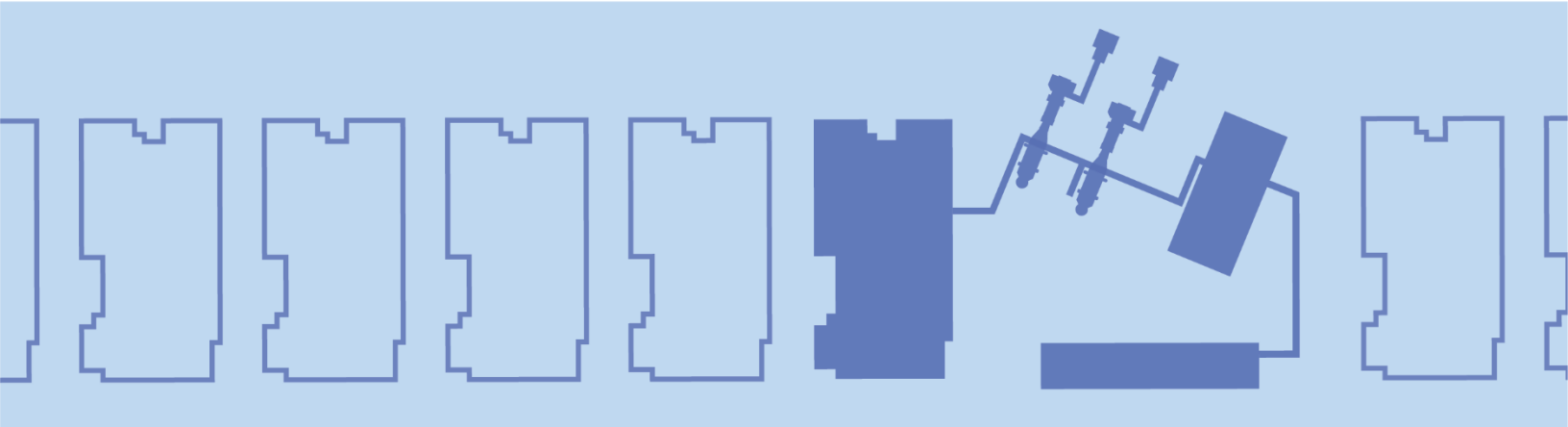
September 2026

Non-Proprietary Information

Chapter 3 Design of Structures, Components, Equipment, and Systems

Section 3.1 - Conformance with USNRC General Design Criteria

Section 3.2 - Classification of Structures, Systems and Components



ACRONYM LIST

Acronym	Explanation
AC	Alternating Current
AISC	American Institute of Steel Construction
ALARA	As Low As Reasonably Achievable
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
ARM	Area Radiation Monitoring
ASME	American Society of Mechanical Engineers
BDBA	Beyond Design Basis Accident
BIS	Boron Injection System
BL-DSA	Baseline Deterministic Safety Analysis
BPVC	Boiler and Pressure Vessel Code
BWR	Boiling Water Reactor
CCF	Common Cause Failure
CFR	Code of Federal Regulations
CFS	Condensate and Feedwater Heating System
CIS	Containment Inerting System
CIV	Containment Isolation Valve
CMon	Containment Monitoring Subsystem
CN-DSA	Conservative Deterministic Safety Analysis
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
CUW	Reactor Water Cleanup
CWE	Chilled Water Equipment
D-in-D	Defense-in-Depth
DBA	Design Basis Accident
DBE	Design Basis Event
DC	Direct Current
DCIS	Distributed Control and Information System
DEC	Design Extension Condition
DL	Defense Line
DL2	Defense Line 2

Acronym	Explanation
DL3	Defense Line 3
DL4a	Defense Line 4a
DL4b	Defense Line 4b
DSA	Deterministic Safety Analysis
ECCS	Emergency Core Cooling System
EFS	Equipment and Floor Drain System
FMCRD	Fine Motion Control Rod Drive
FPC	Fuel Pool Cooling and Cleanup System
FPS	Fire Protection System
FSF	Fundamental Safety Function
GDC	General Design Criteria
GEH	GE Hitachi Nuclear Energy
GVH	GE Vernova Hitachi Nuclear Energy
GNF	Global Nuclear Fuels
GOTHIC	Generation of Thermal-Hydraulic Information for Containments
HCU	Hydraulic Control Unit
HVS	Heating, Ventilation, and Cooling System
I&C	Instrumentation and Control
IC	Isolation Condenser
ICC	Isolation Condenser Pools Cooling and Cleanup System
ICS	Isolation Condenser System
ILRT	Integrated Leak Rate Test
ISI	Inservice Inspection
IST	Inservice Testing
LOCA	Loss-of-Coolant Accident
LTR	Licensing Topical Report
LWM	Liquid Waste Management System
MCR	Main Control Room
MS	Main Steam
NBS	Nuclear Boiler System
NDTT	Nil-Ductility Transition Temperature
NHS	Normal Heat Sink

Acronym	Explanation
NUREG	Nuclear Regulatory Report
OE	Operating Experience
OGS	Offgas System
OL	Operating License
OPEX	Operating Experience
PAM	Post-Accident Monitoring
PCCS	Passive Containment Cooling System
PCS	Primary Containment System
PDC	Principal Design Criterion
PIE	Postulated Initiating Event
PPS	Plant Pneumatics System
PSA	Probabilistic Safety Assessment
PSAR	Preliminary Safety Analysis Report
PVS	Plant Vent Stack
QA	Quality Assurance
RB	Reactor Building
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RG	Regulatory Guide
RIV	Reactor Isolation Valve
RPV	Reactor Pressure Vessel
RWB	Radwaste Building
SAFDL	Specified Acceptable Fuel Design Limits
SC	Steel-Plate Composite
SC1	Safety Class 1
SC2	Safety Class 2
SC3	Safety Class 3
SCCV	Steel-Plate Composite Containment Vessel
SCN	Non-Safety Class
SCR	Secondary Control Room
SDC	Shutdown Cooling System
SDG	Standby Diesel Generator

Acronym	Explanation
SRP	Standard Review Plan
SSCs	Structures, Systems, and Components
SWM	Solid Waste Management System
TAF	Top of Active Fuel
TB	Turbine Building
TEDE	Total Effective Dose Equivalent
TMR	Triple Modular Redundant
TRACG	Transient Reactor Analysis Code General Electric
TS	Technical Specifications
UHS	Ultimate Heat Sink
USNRC	United States Nuclear Regulatory Commission

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None.

3.0 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

This chapter identifies and describes the principal design and analysis of the BWRX-300 structures, systems, and components (SSCs). The appendices to this chapter provide supporting information for:

- SSC safety class, seismic category, and Quality Group classification, Appendix 3A
- Design and analysis of seismic structures, Appendices 3B through 3H
- Computer programs used for design and analysis of SSCs, Appendices 3I through 3M
- Aircraft impact assessment, Appendix 3N

3.1 Compliance with U.S. Nuclear Regulatory Commission General Design Criteria

This section contains an evaluation of the GE Vernova Hitachi Nuclear Energy (GVH) Boiling Water Reactor (BWR), BWRX-300 standard plant design, analyses, procurement, and construction requirements for compliance with 10 Code of Federal Regulations (CFR) 50, Appendix A, “General Design Criteria for Nuclear Power Plants.”

The BWRX-300 design incorporates passive safety features that are designed to safely shut down the reactor and maintain it in a safe shutdown condition without reliance on offsite or Standby Diesel Generator (SDG), Alternating Current (AC) power, external cooling water, or operator action for Anticipated Operational Occurrences (AOOs), Design Basis Accidents (DBAs), and Postulated Initiating Events (PIEs). The motive power for the passive safety systems is supplied by stored energy sources such as Direct Current (DC) power, compressed gas, springs, or gravity. In addition, the BWRX-300 design is provided with a complement of highly reliable systems that reduce the challenge to the passive safety systems and provide the capability for extending the safe shutdown period.

Because the General Design Criteria (GDC) are not technology-neutral and are predicated on legacy design, some GDC have been substituted with Principal Design Criteria (PDC) in accordance with 10 CFR 50.34(a)(3)(i) and (ii). Technology neutral guidance from Appendix A, “Advanced Reactor Design Criteria,” to Regulatory Guide (RG) 1.232, “Guidance for Developing Principal Design Criteria for Non-Light-Water-Reactors,” is used to inform the development of the BWRX-300 PDCs.

Table 3.1-1 provides references to the other Preliminary Safety Analysis Report (PSAR) sections where additional design and analysis information is discussed. GDC 1-5 are omitted from Table 3.1-1 because they have plant-wide applicability.

3.1.1 Group I – Overall Requirements

3.1.1.1 GDC 1 – Quality Standards and Records

GDC 1 Statement

Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

Evaluation Against GDC 1

Safety Class 1 (SC1) SSCs, and their Safety Category 1 functions, are subject to the GVH Quality Assurance (QA) program, which satisfies the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants" and American Society of Mechanical Engineers (ASME) NQA-1-2015, "Quality Assurance Requirements, for Nuclear Facility Applications," (Reference 3.1-6) with the exceptions and clarifications noted in RG 1.28, "Quality Assurance Program Criteria (Design and Construction)." Quality controls for non-SC1 SSCs that are risk-significant are consistent with "U. NONSAFETY-RELATED SSC QUALITY CONTROLS" of Nuclear Regulatory Report (NUREG)-0800, Standard Review Plan (SRP), Section 17.5. Subsection 17.4.5 describes Risk-Significant Criteria established for SSCs.

As discussed throughout PSAR sections, codes and standards are applied to SSCs relied on to satisfy GDCs or PDCs commensurate with the safety class of the SSC.

Documents are maintained to demonstrate that the requirements of the QA program are satisfied. This documentation shows that appropriate codes, standards, and regulatory requirements are identified, correct materials are specified, correct procedures are utilized, qualified personnel are provided, and the finished parts and components meet the applicable specifications. These records are maintained and retrievable for the life of the Operating License (OL).

Accordingly, the requirements of 10 CFR 50, Appendix A, GDC 1 are met.

3.1.1.2 GDC 2 – Design Bases for Protection Against Natural Phenomena

GDC 2 Statement

Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

Evaluation Against GDC 2

SSCs classified as SC1 are designed to retain capability to perform their Safety Category 1 functions during and after design basis natural phenomena events. SSCs whose failure during a seismic event could prevent a Seismic Category I SSC from performing its Safety Category 1 functions are designed to ensure such failure does not occur. For non-SC1 SSCs:

- Structures housing radiological waste materials are designed for external hazard protection in accordance with RG 1.143, "Design Guidance for Radioactive Waste Management Structures, Systems, and Components Installed in Light-Water-Cooled Nuclear Power Plants."
- The Main Control Room (MCR) and associated egress route between the MCR and Secondary Control Room (SCR) are protected such that human occupants of those areas are not caused incapacitating physical harm as a result of any design basis natural phenomena event

- Equipment that is qualified and located to remain functional during and following design basis external hazards is provided to support display of a suitable inventory of indications that allow confirmation of Fundamental Safety Function (FSF) performance and status of physical barriers to release
- Equipment determined to be risk-significant based on external hazard-specific Probabilistic Safety Assessment (PSA) modeling, such as Seismic PSA, is provided with additional protection and/or designated for qualification to survive the hazard conditions

Selection of design basis natural phenomena events reflects consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated. Selection of design basis natural phenomena events also reflect appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 2 are met.

3.1.1.3 GDC 3 – Fire Protection

GDC 3 Statement

Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat-resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

Evaluation Against GDC 3

The following design features minimize the probability of a fire event or assure adequate protection in the event of a fire event:

- Probability of fires occurring in the plant is minimized by the following measures:
  - Noncombustible and heat resistant materials such as metal cabinets, metal wireways, high melting point insulation, and flame-resistant markers for identification are used wherever practicable
  - Electrical cabling is suitably rated and protected, and cable tray loading is designed to avoid unacceptable internal heat buildup
- Redundant divisions of SC1 SSCs and fire safe shutdown SSCs are in separate fire areas or fire zones as needed to ensure one path to achieve fire safe shutdown is available (except the inerted containment)
- Electrical cabling supporting SC1 SSCs and fire safe shutdown SSCs is routed with separation to ensure one path to achieve fire safe shutdown is available (except the inerted containment)
- SC1 and fire safe shutdown SSCs are designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions

- The Fire Protection System (FPS) is provided with appropriate detection, capacity, and capability to minimize impact of fires on SC1 and fire safe shutdown SSCs
- Location and method of fire suppression are selected such that the impacts of rupture or inadvertent operation do not prevent the performance of a Safety Category 1 function

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 3 are met.

3.1.1.4 GDC 4 – Environmental and Dynamic Effects Design Bases

GDC 4 Statement

Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

Evaluation Against GDC 4

SC1 SSCs are designed to be compatible with environmental conditions associated with normal operation, maintenance, testing, and postulated pipe failure accidents including Loss-of-Coolant Accidents (LOCA). Design requirements specify the duration that SC1 SSCs must survive the environmental conditions following a LOCA. The Safety Class 2 (SC2) and Safety Class 3 (SC3) SSCs that mitigate PIEs are confirmed to be compatible with the environmental and dynamic effects present during the event scenarios where these functions are credited. Additionally, the failure of non-SC1 SSCs due to environmental and dynamic effects will not impair SC1 SSCs.

For piping in containment penetration areas, the probability of fluid system piping rupture is demonstrated to be extremely low under conditions consistent with the design basis for the piping, such that line breaks do not need to be postulated for the purpose of the evaluation of potential dynamic effects. Accordingly, SC1 SSCs do not require protection against dynamic effects, including the effects of pipe whipping, jet impingement, and blast wave and blowdown loads that may result from piping failure. The effects of missiles originating from inside or outside the plant are also considered.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 4 are met.

3.1.1.5 GDC 5 – Sharing of Structures, Systems, and Components

GDC 5 Statement

Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

Evaluation Against GDC 5

BWRX-300 is a single unit and does not share any SSCs with other nuclear power units. Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 5 are met.

3.1.2 Group II – Protection by Multiple Fission Product Barriers

3.1.2.1 GDC 10 – Reactor Design

GDC 10 Statement

The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Evaluation Against GDC 10

The “reactor core” includes both the mechanical and fuel design aspects of fuel assemblies and internal/supporting core structures. Design measures are applied to the reactor core design to assure margin to Specified Acceptable Fuel Design Limits (SAFDL) during normal operating conditions and AOO event sequences. Examples of such design measures include the following:

- The mechanical design is based on conservative application of stress limits, Operating Experience (OE), and experimental test results
- The reactor core is designed to maintain integrity considering a complete range of power levels, including normal operation and AOO event sequence conditions
- The reactor core is sized with sufficient heat transfer area and coolant flow to assure that SAFDLs are not exceeded
- Mechanical analyses confirm that Reactor Pressure Vessel (RPV) internals remain intact during normal operation, AOO event sequence conditions and beyond with appropriate margins (using ASME Boiler and Pressure Vessel Code (BPVC) Service Level approach). These structures are assumed to stay intact in the evaluations that ensure SAFDLs are not exceeded.
- Evaluations of normal operation and AOO event sequence conditions are performed for each fuel design and fuel cycle based on the design of the reactor core and associated coolant, control, and protection systems to confirm SAFDLs are not exceeded

“Associated coolant” systems include the passive Reactor Coolant Pressure Boundary (RCPB) equipment, RPV internal components including the core support structure, passive Isolation Condenser System (ICS) equipment, and connected process systems such as main steam and feedwater. In this context, passive means vessels, piping, heat exchanger tubing, and other equipment, and example design and analysis measures include the following:

- The mechanical design of RCPB components, reactor internals, and connected coolant systems is based on conservative application of stress limits, OE, and experimental test results
- These associated coolant systems are assumed to stay intact and perform as designed in the evaluations that ensure SAFDLs are not exceeded because they are designed in accordance with the appropriate ASME criteria for the applicable service level

“Associated control systems” include the control, electrical, mechanical and measurement elements that support control during normal operations, with example measures such as:

- Reactor power, pressure, and level control ensure normal operation is consistent with the normal operation conditions assumed in SAFDL evaluations. Level and pressure control are included in Deterministic Safety Analysis (DSA) of AOO event sequence conditions to demonstrate SAFDLs are not exceeded.

"Associated protection systems" include the control, electrical, mechanical and measurement elements that support mitigation during AOO event sequence conditions. The protective functions performed by these elements are Safety Category 2 or Safety Category 3 functions. Example measures include the following:

- Protective function setpoints are selected to ensure the FSFs are performed and SAFDLs are not exceeded during AOO event sequences
- Equipment performing protective functions is tested to confirm that its performance attributes are consistent with the performance modeled in evaluation of SAFDLs

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 10 are met.

Additional information discussing the compliance approach to GDC 10 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.2.2 GDC 11 – Reactor Inherent Protection

GDC 11 Statement

The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

Evaluation Against GDC 11

The New Fuel Licensing requirements established in NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel (GESTAR II)," (Reference 3.1-7), stipulate that the net prompt reactivity feedback due to prompt heating of the moderator and fuel shall be negative for a reactivity insertion event originating from any operating condition. GNF2 has been evaluated generically and affirmed to satisfy this requirement, and the results are summarized in NEDC-33270P, "GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II)," (Reference 3.1-8). The GNF2 application to the BWRX-300 with the N-lattice has separately been confirmed to satisfy this requirement.

Additionally, NEDE-24011-P-A establishes requirements governing other nuclear dynamic parameters pertaining to reactor safety.

Together, compliance with these requirements assures the inherent reactivity feedback mechanisms are consistent with reactor safety. GNF2 has been evaluated, generically, to satisfy all requirements governing nuclear dynamic parameters, including the prompt power coefficient.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 11 are met.

Additional information discussing the compliance approach to GDC 11 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.2.3 GDC 12 – Suppression of Reactor Power Oscillations

GDC 12 Statement

The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

Evaluation Against GDC 12

The BWRX-300 design addresses stability utilizing an RPV chimney that increases natural circulation core flow so that a margin to instability that could challenge SAFDLs is maintained for all modes of operation. In accordance with GDC 12, it is required that the power response to a sudden insertion of reactivity results in a coupled power and core flow response such that SAFDLs are not exceeded.

The BWRX-300 maintains a coupled power and flow response such that any initial perturbation that does not cause an immediate scram, decays quickly to steady state even at the limiting point of the cycle. Also, the relatively small core of the BWRX-300 is a design measure which causes it not to be susceptible to regional modes of oscillation. Therefore, a special stability detection and trip system (i.e., a “protection system”) is not required. The natural circulation driving head provided by the BWRX-300 chimney along with the core orifice design causes core flow and power perturbations to be naturally damped.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 12 are met.

Additional information discussing the compliance approach to GDC 12 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.2.4 GDC 13 – Instrumentation and Control

GDC 13 Statement

Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

Evaluation Against GDC 13

Instrumentation capable of monitoring important parameters across the anticipated ranges for normal operation, AOO event sequences, and accident conditions are provided by the mechanical systems that are controlled and monitored by the Distributed Control and Information System (DCIS). The instruments monitor system variables in the reactor core, RCPB, containment and supporting systems. Normal and off-normal operating ranges are defined to support process control during normal operation and the automatic initiation of safety functions and/or accident monitoring in the event of AOO event sequences or DBA event sequences.

Instruments are classified in accordance with the highest safety category function they support and qualified to operate under the conditions in which their functions are required to be performed.

Safety Category 1 and Safety Category 3 functions are performed by independent and diverse SC1 and SC3 Instrumentation and Control (I&C) systems, respectively, to maintain the specified variables within prescribed operating ranges through normal control or, when required, by automatic safety actuations to assure adequate safety. Diversity is provided to mitigate the risks of common cause failure, and redundancy is provided to meet reliability and availability requirements. The I&C systems also display the monitored parameters during normal operation and for accident monitoring.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 13 are met.

Additional information discussing the compliance approach to GDC 13 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.2.5 GDC 14 – Reactor Coolant Pressure Boundary

GDC 14 Statement

The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

Evaluation Against GDC 14

The piping and equipment within the RCPB (as defined in 10 CFR 50.2) are designed, fabricated, erected, and tested in accordance with the requirements in 10 CFR 50.55a, Codes and Standards, providing a high degree of integrity throughout the plant lifetime. Piping and equipment within the RCPB are classified as Quality Group A or B, as appropriate. The RCPB is protected from overpressure by ICS. The design requirements and codes and standards applied to Quality Group A and B components ensure high safety function integrity.

To minimize the possibility of brittle fracture within the RCPB, the fracture toughness properties and the operating temperature of ferritic materials are controlled to ensure adequate toughness. Materials are impact tested in accordance with ASME BPVC, Section III, where applicable. Where RCPB piping penetrates the containment, the fracture toughness temperature requirements of the RCPB materials apply.

Piping and equipment within the RCPB are assembled and erected by welding unless applicable codes permit flanged or threaded joints. Welding procedures are employed that produce complete fusion welds that are free of unacceptable defects. All welding procedures, welders, and welding machine operators used in producing pressure containing welds are qualified in accordance with the requirements of the ASME BPVC, Section IX, (Reference 3.1-5) for materials welded. Qualifications records, including the results of procedure and performance qualification tests and identification symbols assigned to each welder are maintained.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 14 are met.

Additional information discussing the compliance approach to GDC 14 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.2.6 GDC 15 – Reactor Coolant System Design

GDC 15 Statement

The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

Evaluation Against GDC 15

The Reactor Coolant System (RCS) includes the SC1 portion of the Nuclear Boiler System (NBS) and the SC1 portion of the Condensate and Feedwater Heating System (CFS). The RCS extends to and includes the outermost Containment Isolation Valve (CIV) in the Main Steam (MS) and Feedwater (FW) piping. The RCPB includes the pressure retaining components of the RCS and connected systems up to and including their respective outboard CIVs.

The RCS for a BWR includes the RPV, the Nuclear Boiler System (NBS), which includes the Main Steam subsystem to the outboard CIVs, and a portion of the Feedwater system to the outboard CIVs. The RCPB includes the RCS plus ICS and portions of the Control Rod Drive (CRD) and Reactor Water Cleanup (CUW) systems.

Safety Category 3 functions, implemented in at least SC3 equipment, are provided to automatically initiate hydraulic scram and pressure control functions (e.g., turbine bypass valves, or ICS pressure control) so that the design conditions of the RCPB are not exceeded during AOO event sequences. If monitored parameters exceed their predetermined setpoints, the Safety Category 3 functions automatically respond to maintain the parameters within their allowable design limits.

Overpressure protection for AOO event sequences is provided in accordance with ASME BPVC, Section III, NB-7120(b) and (c) by Safety Category 3 functions and ICS, respectively. These ensure that the acceptance criteria for each component in the protected system are met including 1) the overpressure does not exceed 1.1 times design pressure for the expected system pressure transient condition, and 2) the calculated stress intensity and other design limitations for Service Level C are not exceeded for the unexpected system excess pressure transient condition.

Design measures are also employed in the design of the RCPB, such that it accommodates pressure conditions expected for normal operations, AOO event sequences, DBA event sequences, and Design Extension Condition (DEC) event sequences. This means significant margin for normal operation and AOO event sequences. For example, the mechanical design of RCPB components is based on conservative application of ASME BPVC stress limits, Operating Experience (OPEX), and qualification test results.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 15 are met.

Additional information discussing the compliance approach to GDC 15 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.2.7 GDC 16 – Containment Design

GDC 16 Statement

Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

Evaluation Against GDC 16

An essentially leak-tight Steel-Plate Composite Containment Vessel (SCCV) encloses the RPV, including the majority of the RCPB and other branch connections for the NBS, and includes containment penetrations with essentially leak-tight isolation design features including CIVs, blind flanges, hatches, and electrical penetrations. A steel head encloses the opening in the top of the SCCV for servicing and refueling the RPV. The major piping systems (main steam, feedwater, ICS, and other systems) are located in the upper SCCV region. The lower SCCV region encloses the lower portion of the RPV and encloses the Fine Motion Control Rod Drives (FMCRD) and associated piping and other systems as well as providing maintenance space below the RPV. Temperature and pressure conditions inside the SCCV are controlled and maintained below acceptance criteria following an accident for at least 72 hours by RPV decay heat removal using the ICS and condensation on the SCCV walls and with containment heat removal using the Passive Containment Cooling System (PCCS).

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 16 are met.

Additional information discussing the compliance approach to GDC 16 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.2.8 PDC 17 – Electric Power Systems

PDC 17 Statement

During normal operations and design basis events that do not involve the loss of AC power, an offsite electric power system shall be provided to permit functioning of normally operating systems and components. The offsite electric power system shall not be required to perform Safety Category 1 functions. In the event of a loss of offsite AC power, onsite electric power system(s) shall be provided when required to permit functioning of systems and components for two layers of defense. The electric power system(s) shall have sufficient capacity and capability to ensure that (1) the design limits for the fission product barriers are not exceeded as a result of anticipated operational occurrences and (2) safety functions that rely on electric power are maintained in the event of DBAs. Each layer of defense shall have sufficient independence, redundancy, and testability to ensure the reliability of the functions being supported.

Evaluation Against PDC 17

The electric power provisions in PDC 17 apply to GDC 18, 34, 35, 38, 41, and 44.

The BWRX-300 design has a connection to the switchyard (i.e., source of offsite power) that provides AC power to normally operating systems and components such as the normal heat sink system and feedwater pumps. Offsite power is not required to perform any Safety Category 1 functions. The Preferred Power System provides the interconnecting electrical distribution system between the plant main generator, onsite power system, and the offsite power source.

The offsite power source is described in Section 8.2, and the Preferred Power System is described in Subsection 8.3.1.2.

In the event of a loss of offsite AC power, two onsite electric power systems are provided to support two layers of defense for systems and components that are relied upon to mitigate AOOs and DBAs, as follows:

- Electric Power System for AOOs:

  Safety Category 3 functions ensuring fission product barrier design limits are not exceeded as a result of AOOs are provided power by the SC3 Preferred Power 250 VDC Subsystem. The SC3 Preferred Power 250 VDC Subsystem has the capacity and capability to provide power to SC3 DC loads, such as DCIS. The SC3 Preferred Power 250 VDC Subsystem provides redundant, battery backed feeds such that no single power failure results in the loss of a Safety Category 3 function to mitigate AOOs. Similarly, a single power feed can be removed from service for testing or maintenance without impacting Safety Category 3 functions. The independence, redundancy, and testability of the SC3 Preferred Power 250 VDC Subsystem is sufficient to ensure that the Safety Category 3 functions it supports for AOO mitigation achieve their reliability target in accordance with NEDC-33934P, “BWRX-300 Safety Strategy,” (Reference 3.1-1).

- Electric Power System for DBAs:

  The BWRX-300 design assures that the FSFs are maintained for DBAs. The electric power system is the SC1 Emergency Power System for monitoring and actuating SC1 passive systems, such as emergency core cooling provided by the ICS. All passive SC1 systems for DBAs are actuated by SC1 DCIS control systems that are supplied power from the Emergency Power System 72-hour DC emergency batteries. The SC1 Emergency Power System provides three divisions of SC1 DC power such that no single failure (or removal of service for testing or maintenance) results in the loss of a Safety Category 1 function. The independence, redundancy, and testability of the SC1 Emergency Power System is sufficient to ensure that the Safety Category 1 functions it supports achieve their reliability target in accordance with NEDC-33934P, "BWRX-300 Safety Strategy," (Reference 3.1-1).

For AOOs that do not involve a loss of offsite AC power, deterministic safety analyses model the plant response without a loss of AC power (e.g., with feedwater pump running). To mitigate such AOOs, Safety Category 3 functions are initiated by SC3 DCIS, which is powered by SC3 Preferred Power 250 VDC Subsystem, and may rely on offsite power for normally operating systems and components (e.g., feedwater pumps, turbine bypass valves, turbine control valves). The reliability target of these Safety Category 3 functions is 1E-02 failures/demand, with consideration for all components needed to perform the function, including offsite electrical support.

For DBAs that do not involve a loss of offsite AC power, deterministic safety analyses model the feedwater pump running since AC power is available; however, only Safety Category 1 functions are credited to mitigate DBAs. Safety Category 1 functions are powered by the SC1 Emergency Power System. Offsite power is not required for any Safety Category 1 function.

Accordingly, the requirements of PDC 17 are met.

Additional information discussing the compliance approach to PDC 17 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.2.9 GDC 18 – Inspection and Testing of Electric Power Systems

GDC 18 Statement

Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

Evaluation Against GDC 18

All SC1 loads are normally supplied directly through DC to AC inverters. Capability is provided for testing each battery, rectifier, battery charger, and inverter without disrupting power to the SC1 loads.

Design of the SC1 power system provides testability in accordance with the requirements of GDC 18.

The SC3 Preferred Power 250 VDC subsystem supports functions used to ensure that the design limits for the fission product barriers are maintained. This subsystem is designed to permit appropriate periodic inspection and testing of each SC3 battery, rectifier, battery charger, and inverter in accordance with the requirements of GDC 18.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 18 are met.

Additional information discussing the compliance approach to GDC 18 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.2.10 PDC 19 – Control Room

PDC 19 Statement

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and anticipated operational occurrences, and to maintain it in a safe condition during design basis accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room during design basis accidents without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent as defined in 10 CFR 50.2.

Equipment at appropriate locations outside the control room shall be provided with a design capability for safe shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe shutdown condition.

Evaluation Against PDC 19

PDC 19 requires the capability to operate the plant under normal conditions, AOOs, and DBAs while providing adequate radiation protection, state-of-the-art human factors principles, and control room habitability such as heating, ventilation, cooling, and lighting. Additionally, PDC 19 clarifies the radiation protection requirements for control room inhabitants consistent with current industry practice.

BWRX-300 design demonstrates adequate radiation protection to permit access and occupancy of the MCR or SCR so that personnel do not receive radiation exposures in excess of 5 rem Total Effective Dose Equivalent (TEDE) for DBAs.

The BWRX-300 design includes means outside the MCR for operators to place and maintain the reactor in a safe condition in the event of a control room evacuation. For the BWRX-300 reactor design, the establishment of PDC 19 to require “safe shutdown” capability outside the MCR is supported and consistent with United States Nuclear Regulatory Commission (USNRC) guidance, such as SECY-94-084, “Policy and Technical issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs,” which applies to passive residual heat removal systems and RG 1.189, “Fire Protection for Nuclear Power Plants,” regarding fire in the MCR. In the event of an MCR evacuation, the reactor is tripped, and decay heat removal and containment isolation are initiated prior to operators evacuating the MCR, provided that the control capabilities remain operable. Operators can also place the reactor in safe shutdown from the Seismic Category I SCR within the Reactor Building (RB).

Accordingly, the BWRX-300 control room design provides a means from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition during DBAs.

Accordingly, the requirements of PDC 19 are met.

Additional information discussing the compliance with PDC 19 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.3 Group III – Protection and Reactivity Control Systems

3.1.3.1 GDC 20 – Protection System Functions

GDC 20 Statement

The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

Evaluation Against GDC 20

NEDC-33934P, "BWRX-300 Safety Strategy," (Reference 3.1-1), discusses the applicability of the "protection system" described in GDC 20 to the BWRX-300 design.

The protection functions are organized into layers of Defense Lines (DL). Analyses demonstrating the capability of the different DLs to protect the fission product barriers (i.e., fuel cladding, reactor coolant pressure boundary, and containment pressure boundary) to the applicable acceptance criteria are layered as well. The different DLs are implemented and analyzed for each AOO and DBA PIE.

The BWRX-300 layered safety analyses include a Baseline Deterministic Safety Analysis (BL-DSA) and a Conservative Deterministic Safety Analysis (CN-DSA). The BL-DSA is performed to demonstrate the capability of Safety Category 3 functions to mitigate AOO PIEs to ensure SAFDLs are not exceeded. The CN-DSA is performed to demonstrate that Safety Category 1 functions alone can mitigate the effects of the same AOO PIEs with the added assumption that Safety Category 3 functions fail to respond. The CN-DSA also demonstrates that Safety Category 1 functions can mitigate the effects of DBA PIEs. The CN-DSA demonstrates, for each AOO and DBA PIE, that the Safety Category 1 functions independently assure the integrity of the RCPB, the capability to shut down the reactor and maintain it in a safe shutdown condition, and the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 20 are met.

Additional information discussing the compliance approach to GDC 20 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.3.2 GDC 21 – Protection System Reliability and Testability

GDC 21 Statement

The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

Evaluation Against GDC 21

The systems implementing Safety Category 1 protection functions are designed with redundancy and independence between redundant divisions to fulfill the single-failure criterion. No single component failure, intentional bypass maintenance operation, calibration operation, or test to

verify operational availability, impairs the ability of the systems to perform their Safety Category 1 functions. Independence between redundant divisions prevents failures occurring in one division from impairing the ability of the other divisions to function correctly. With one division inoperable, the required minimum redundancy is temporarily lost. Acceptable reliability of operation is established by limiting the time such a condition can exist through Technical Specifications (TSs).

Design features, such as self-testing routines in digital equipment, that permit Inservice Testing (IST) are included in the design, and the Safety Category 1 functions can be tested during reactor operation.

The systems implementing Safety Category 3 protection functions are designed to ensure acceptable reliability of operation. Each Safety Category 3 DL function is designed with a target reliability of 1E-02 failures per demand or less. This is accomplished by using Triple Modular Redundant (TMR) controllers for the majority of Defense Line 2 (DL2) functions (control rod block functions are implemented in dual redundant hardware and are designed to fail-safe), as well as triple redundant input sensors. TMR controller inputs and outputs for Safety Category 3 functions use dedicated communication paths; execution of these Safety Category 3 functions does not rely on the Unit Data Highway. Triple modular redundancy is a fault tolerant technique where three controllers perform the same process simultaneously. The BWRX-300 design uses triplicated sensor inputs. The results from the controllers are then processed by a majority voting system producing a single output. This means that if one of the three sensors (or their associated communication paths) or one of the TMR controllers fails, the other two can still provide the correct output, ensuring reliability. SSCs performing Safety Category 3 protection functions are redundantly powered from battery-backed feeds supplied by the SC3 Preferred Power 250 VDC Subsystem. This redundancy improves reliability, ensuring that a single power feed failure does not result in the loss of a protection function. The criteria to ensure acceptable reliability of operation for Safety Category 3 protection functions are:

- No single sensor or controller failure results in the loss of a protection function
- Removal of a single sensor or controller from service does not result in the loss of a protection function
- No single power feed failure results in the loss of a protection function
- Removal of a single power feed from service does not result in the loss of a protection function

The SC3 hardware does not have the same types of independence between redundancies as in the SC1 systems; however, this is accounted for in calculating the reliability achieved for the DL2 functions. Acceptable reliability and availability of operation is established by monitoring DL2 functions in accordance with the Availability Controls Manual. Extensive hardware and software diagnostics are performed continuously during operation to detect and alarm individual failures that may occur. Cross-checks between redundant sensor measurements are performed continuously during operation to detect single sensor failures that may occur during operation.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 21 are met.

Additional information discussing the compliance approach to GDC 21 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.3.3 GDC 22 – Protection System Independence

GDC 22 Statement

The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

Evaluation Against GDC 22

Safety Category 1 functions are implemented with independence between redundant equipment such that the effects of natural phenomena, and of normal operation, maintenance, testing, and postulated accident conditions do not result in loss of the minimum redundancy required for successful performance of Safety Category 1 functions. Redundant SC1 sensors are electrically and physically separated, and only circuits of the same division are run in the same raceway.

Signals required to cross divisional boundaries to implement the 2 out of 3 voting logic use fiber optic communication pathways to maintain electrical isolation between divisions. SC1 equipment is also seismically and environmentally qualified to function in the expected environments resulting from natural phenomena events or accidents. The SC1 I&C systems are designed to permit maintenance and diagnostic work while the reactor is operating, without restricting the plant operation or hindering the output of safety functions. Operational system testing can be performed using independent input for each division. When a monitored variable supporting a Safety Category 1 function exceeds its scram trip point, it is sensed by three independent sensors, each located in a separate instrumentation channel.

DL2 (i.e., Safety Category 3) functions do not have the same level of independence that exists within Defense Line 3 (DL3) (i.e., Safety Category 1) functions; however, DL2 functions are implemented with independence adequate to meet their reliability requirement pursuant to an "other defined basis" described in GDC 22. TMR controllers are used for the majority of DL2 functions, as well as triple redundant input sensors, except control rod block functions are implemented in dual redundant hardware. SSCs performing a DL2 function are designed to operate in the environment expected during normal operations and in response to AOOs relying on the DL2 function for mitigation. The resulting reliability of DL2 functions is adequate to provide confidence that these functions will perform their functions when needed.

Additionally, the BWRX-300 systems making up the protection system have layers of protection, with both DL2 and DL3 functions being able to mitigate AOOs to within acceptance criteria described in Table 15.3-1 and Table 15.3-2, respectively. There is independence between these layers, as SC3 equipment is independent from SC1 equipment except in cases where sharing of incore components (e.g., Local Power Range Monitors data is utilized to initiate Safety Category 1 hydraulic scram functions and Safety Category 3 rod block functions) is necessary due to practical limitations. Additionally, there is diversity in hardware, as the SC1 I&C systems are diverse from other I&C systems (i.e., hardware platforms are diverse). The independence and diversity between layers of protective functions achieve high plant-level protection reliability.

The SC3 I&C systems are designed to ensure they cannot adversely affect SC1 I&C systems from fulfilling their Safety Category 1 protection functions. There is no communication from SC3 equipment to online SC1 equipment. Safety Category 3 functions may use signals from SC1 equipment sent via qualified isolation devices for some control purposes. Plant safety is not adversely affected by the loss or inaccuracy of these individual signals. The physical separation and signal isolation devices for the SC3 I&C systems meet the requirements of IEC 60709, "Nuclear Power Plants – Instrumentation, Control and Electrical Power Systems Important to Safety – Separation."

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 22 are met.

Additional information discussing the compliance approach to GDC 22 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.3.4 GDC 23 – Protection System Failure Modes

GDC 23 Statement

The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

Evaluation Against GDC 23

The systems implementing Safety Category 1 protection functions are designed to fail into a safe state (or a state demonstrated to be acceptable on another basis) if conditions such as disconnection of the system, loss of energy, or postulated adverse environments are experienced. Use of independent divisions allows the system to sustain any single divisional failure without preventing redundant divisions from initiating the Safety Category 1 protection functions. The environmental conditions in which the instrumentation and equipment of the SC1 systems must operate are considered in establishing the component specifications. Instrumentation specifications are based on the worst expected ambient conditions in which the instruments must operate.

The SC3 I&C systems have the required fail-as-is design features to perform intended functions and avoid spurious actuation. The fail-as-is, energize to actuate, design is used to prevent lesser classified systems from creating unnecessary challenges to plant safety (e.g., spurious actuations for expected failures) requiring action by Safety Category 1 functions. Where software is used for SC3 equipment diagnostics, the SC3 equipment includes watchdog timers, sensor range checks, and monitoring of power supplies, communications, and actuators. The SC3 equipment is designed to operate in the environment that is to be expected during both normal operations and anticipated off-normal conditions, and the equipment is qualified to perform its intended functions and meets IEC 61513, "Nuclear Power Plants – Instrumentation and Control Important to Safety – General Requirements for Systems," requirements and the equipment standards defined in Section 7 in IEC 61226, "Nuclear Power Plants – Instrumentation, Control and Electrical Power Systems Important to Safety – Categorization of Functions and Classification of Systems," requirements for Class 3 systems.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 23 are met.

Additional information discussing the compliance approach to GDC 23 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.3.5 GDC 24 – Separation of Protection and Control Systems

GDC 24 Statement

The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

Evaluation Against GDC 24

Safety Category 1 functions are implemented separately from Safety Category 3 functions to the extent that either of the following conditions leaves intact a group of equipment satisfying all reliability and independence requirements necessary to assure performance of the FSFs for design basis events:

- Failure of any single component performing Safety Category 1 functions
- Removal from service of any single component that supports both Safety Category 3 and Safety Category 1 functions

Interconnections between equipment performing Safety Category 3 functions and Safety Category 1 functions are limited such that one category of functions remains available in the presence of any credible single failure, or credible Common Cause Failure (CCF), postulated to affect mitigation capabilities of either category. Credible means those considered within the "Design Basis Event (DBE)" analysis set of BL-DSA and CN-DSA.

Electrical isolation, data communication isolation and functional independence are implemented to prevent faults in a lower classified system from propagating to a higher classified system.

Safety Category 3 protection, reactor control, and rod block functions are implemented in separate systems with dedicated controllers. Safety Category 3 control functions are allocated to separate control processors (referred to as separate control segments) than the Safety Category 3 protection functions. The control functions and protection functions may use common input sensors. These sensors are triple redundant such that single failure of any sensor does not prevent either the control or protection function from being performed.

Safety Category 3 protective functions are implemented separately from Safety Category 3 control functions to the extent that either of the following failures leaves intact a group of equipment satisfying all reliability and independence requirements for Safety Category 3 functions:

- Failure of any single sensor performing Safety Category 3 control and protective functions
- Failure of any single processor performing Safety Category 3 control or protective functions
- Removal from service of any single component that supports both Safety Category 3 control and protective functions

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 24 are met.

Additional information discussing the compliance approach to GDC 24 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.3.6 GDC 25 – Protection System Requirements for Reactivity Control Malfunctions

GDC 25 Statement

The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

Evaluation Against GDC 25

Safety Category 3 protection system functions assure that SAFDLs are not exceeded for reactivity control malfunctions. To minimize the likelihood of a rod withdrawal error, the rod control system incorporates redundancy to limit the effect of single failures. If there is a malfunction of the rod control system that results in a rod withdrawal error, a Safety Category 3 rod movement block function is initiated to terminate rod movement.

As a design measure, redundancy is employed throughout the reactivity control systems such that:

- No single equipment failure leads to inadvertent movement of more than one control rod.
- Vulnerabilities to single equipment failures which could cause inadvertent movement of a single control rod are minimized.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 25 are met.

Additional information discussing the compliance approach to GDC 25 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.3.7 PDC 26 – Reactivity Control Systems Capability

PDC 26 Statement

Two independent reactivity control systems of different design principles shall be provided.

One system shall use two diverse means of inserting control rods. This system shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded.

The second reactivity control system shall be capable of inserting negative reactivity into the core to assure a reactor shutdown from full power operating conditions if both diverse means of inserting control rods were to fail.

Both of these reactivity control systems shall be capable of holding the reactor core subcritical under cold conditions.

Evaluation Against PDC 26

Two independent reactivity control systems utilizing different design principles are provided. The normal method of reactivity control is the CRD system which employs control rod assemblies. A Boron Injection System (BIS) is also provided for diversity.

Reactivity control methods utilized during normal operation and AOOs employ control rod assemblies controlled by the CRD system. Diverse positive insertion means of controlling reactivity include: 1) the hydraulic scram control rod insertion function using the Hydraulic Control Units (HCUs) and control rods, and 2) the motor-driven control rod positioning function for normal power maneuvering which includes a separate motor-driven control rod run-in insertion function using the FMCRDs and control rods. Each FMCRD controls movement of a single control rod, whereas each HCU controls no more than a control rod pair during hydraulic scram insertion.

The CRD system is designed such that no single active failure of a CRD component, structure, or supporting system function will prevent a reactor scram. The CRD system provides positive means of inserting negative reactivity and is capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits (SAFDLs) are not exceeded. Hydraulic scram functions performed by the CRD system HCUs are controlled by redundant and diverse control systems. A separate motor-driven control rod run-in insertion function is performed by the FMCRDs following a hydraulic scram command as a diverse, alternate means of rod insertion.

The CRD system is capable of effecting the reactor core to become subcritical, with margin, in the most reactive condition throughout the operating cycle with the most reactive control rod fully withdrawn or rod pair associated with a single HCU in the full out position, with all other rods fully inserted. The CRD system is also capable of maintaining the reactor core subcritical under cold conditions, even if the highest worth rod pair associated with a single HCU is assumed to be in the fully withdrawn position. This shutdown capability of the CRD system is made possible by designing the core configuration to ensure the reactor core is held subcritical under cold conditions.

The means of normal reactivity control employs the FMCRDs to provide movement of the control rods; however, the hydraulic scram function is not precluded by the operation or malfunction of the motor-driven FMCRD control rod positioning or run-in design features. The FMCRDs motor-driven function is independent from the hydraulic-driven function such that FMCRD function does not introduce any additional challenges or failure mechanisms to the scram functions. The control rods are physically independent and located separately in the reactor core (i.e., each control rod is physically separated from each other by fuel assemblies). There are no identified mechanical interactions between control rods that would affect more than a control rod pair at a time.

The BIS is a diverse, Defense-in-Depth (D-in-D) system which is redundant to the CRD system with regards to its capability of shutting down the reactor and maintaining it in shutdown conditions. The BIS is a complementary design feature used to mitigate the consequences of DECs, which are Beyond Design Basis Accidents (BDBAs), in the event the controls rods fail to insert. It performs the Safety Category 3 function to assure reactor shutdown from full power operation to cold subcritical conditions without control rod movement by mixing a neutron absorber with the primary coolant. The BIS is capable of holding the reactor subcritical under cold conditions. The BIS and the CRD system have no shared components and operate on different design principles.

Accordingly, the requirements of PDC 26 are met.

Additional information discussing the compliance approach to PDC 26 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.3.8 GDC 27 – Combined Reactivity Control Systems Capability

GDC 27 Statement

The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

Evaluation Against GDC 27

The control rods are capable of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods, the capability to shut down the reactor (and cool the core) is maintained. Positive automatic insertion of the control rods (i.e., scram) is provided by the HCU accumulators in response to DBAs that require emergency core cooling system operation (i.e., initiation of reactor isolation and the ICS). Successful scram and initiation of reactor isolation and the ICS is shown to both achieve and maintain safe shutdown of the reactor without the possibility of a return to criticality, even with the unlikely occurrence of the most reactive control rod fully withdrawn or rod pair associated with a single HCU postulated stuck in the full out position. GDC 27 is satisfied for the BWRX-300 without the injection of soluble boron.

Accordingly, the requirements of 10 CFR 50, Appendix A, GDC 27 are met.

Additional information discussing the compliance approach to GDC 27 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.3.9 PDC 28 – Reactivity Limits

PDC 28 Statement

The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of steam line rupture, changes in reactor coolant temperature and pressure, and cold-water addition.

Evaluation Against PDC 28

The combined features of the CRD system and the rod control system incorporate appropriate limits on the potential amount and rate of reactivity increase. The fine motion capability of the FMCRD allows reactivity additions from rod withdrawal to be finely controlled by the rod control system. Normal rod movement and the rod withdrawal rate are limited by the rod control system and follow sequences assigned by the core designers to operate the core within thermal limits. The rod control system controls rod patterns and provides control rod blocks to limit the rate and amount of reactivity addition for control rod movement. The FMCRD includes an electro-mechanical brake that further prevents inadvertent rod withdrawal by engaging to prevent rod motion if the control system is not commanding rod motion. The BWRX-300 FMCRD and control rod guide tube designs prevent rod drop and rod ejection events through positive design means. Control rod drop is prevented by use of a bayonet style coupling between the FMCRD and the control rod, CRD mechanism latches that maintain the control rod inserted after a scram, and two redundant SC3 FMCRD separation switches that detect a control rod that has become separated from the FMCRD ball nut. Upon detection of separation from the ball nut, the rod control system prevents further outward rod motion of the ball nut by enforcing a withdrawal block. Similarly, control rod ejection is prevented by physical constraints (positive means) including a flange at the top of the control rod guide tube that is above the core plate to prevent guide tube ejection and the FMCRD connection to the control rod guide tube via a bayonet coupling. The FMCRD braking mechanism prevents rod ejection in the event of failure in the pressure retaining parts of the drive mechanism. The ball check valve in the drive flange scram insert port prevents rod ejection in the event of an insert pipe break. The ball check valve functions as SC1 component because it serves as a RCPB function and prevents reverse flow from the scram inlet port caused by a break of the scram line, thus preventing the loss of pressure from the underside of the hollow

piston and the generation of loads on the drive that would tend to cause a rod withdrawal and associated reactivity insertion.

The Limitation and Condition 5.2 of NEDC-33912P is satisfied by PDC 28 compliance and justification for categorizing the CRDA as a beyond-design-basis event. The CRDA will be analyzed as a DEC event in the FSAR.

The safety analyses evaluate the remaining postulated reactivity accidents, as well as abnormal operational transients, in detail. Analyses are included for steam line break, changes in reactor coolant temperature and pressure, and cold water addition. The initial conditions, assumptions, calculational models, sequences of events, and anticipated results of each postulated occurrence are covered in detail. The results of these analyses indicate that none of the postulated reactivity transients or accidents results in damage to the RPV internals, so that the capability to cool the core is not impaired.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, PDC 28 are met.

Additional information discussing the compliance approach to PDC 28 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.3.10 PDC 29 – Protection Against Anticipated Operational Occurrences

PDC 29 Statement

The protection and reactivity control systems shall be designed to assure: (1) an extremely high probability of accomplishing fundamental safety functions in the event of anticipated operational occurrences and (2) a high probability that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences.

Evaluation Against PDC 29

The BWRX-300 design provides two layers of defense for AOOs; Safety Category 3 functions ensure SAFDLs are not exceeded, and Safety Category 1 functions ensure fundamental safety functions are met. The combination of Safety Category 3 functions and independent Safety Category 1 functions, arranged in DLs providing layered protection against AOO PIEs, results in extremely high functional reliability for performance of the FSFs during and after design basis events, including AOOs. The reliability of Safety Category 1 functions imposed by the BWRX-300 Safety Strategy is 1E-04 failures per demand, assuring an extremely high probability of accomplishing their Safety Category 1 functions. The reliability of Safety Category 3 functions imposed by the BWRX-300 Safety Strategy is 1E-02 failures per demand, assuring a high probability of accomplishing their Safety Category 3 functions. The reliability of these functions is confirmed, as described in NEDC-33934P, “BWRX-300 Safety Strategy,” (Reference 3.1-1).

Accordingly, the requirements of 10 CFR Part 50, Appendix A, PDC 29 are met.

Additional information discussing the compliance approach to PDC 29 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.4 Group IV – Fluid Systems

3.1.4.1 GDC 30 – Quality of Reactor Coolant Pressure Boundary

GDC 30 Statement

Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

Evaluation Against GDC 30

The components of the RCPB, including the Reactor Isolation Valves (RIV), CIVs, ICS, and portions of the CRD and CUW systems, and the overpressure protection equipment (ICS and ICS actuation valves) are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed in accordance with generally recognized codes and standards, and under an approved QA program with approved control of records, as required by 10 CFR 50.55a and 10 CFR 50, Appendix A, GDC 1. In addition, means are provided to detect and identify the location of the source of reactor coolant leakage for components of the RCPB. The BWRX-300 design includes only unidentified leakage that is potentially from the RCS. As identified leakage is not defined, there is no leakage separation.

Leakage detection is provided to detect, annunciate, and, in some cases, isolate the RCPB from potentially hazardous leaks before predetermined limits are exceeded. Leaks/breaks may be detected by temperature and pressure changes, increased frequency of sump pump operation, increased airborne radioactivity, changes in flow rates in process lines, or changes in reactor water level.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 30 are met.

Additional information discussing the compliance approach to GDC 30 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.4.2 GDC 31 – Fracture Prevention of Reactor Coolant Pressure Boundary

GDC 31 Statement

The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady-state and transient stresses, and (4) size of flaws.

Evaluation Against GDC 31

Brittle fracture control of pressure-retaining ferritic materials is provided to ensure protection against non-ductile fracture. To minimize the possibility of brittle fracture failure of the RPV, the RPV is designed to meet the requirements of ASME BPVC, Section III.

The reactor assembly design provides an annular space from the outermost fuel assemblies to the inner surface of the RPV that serves to attenuate the fast neutron flux incident upon the RPV wall. This annular volume contains the core shroud and reactor coolant. Assuming plant operation at rated power and 100% availability over the plant life, the cumulative neutron fluence at the inner surface of the RPV causes a slight shift in the transition temperature. Expected shifts in transition temperature during design life as a result of environmental conditions, such as

neutron flux, are considered in the design. Operational limitations ensure that Nil-Ductility Transition Temperature (NDTT) shifts are accounted for in the reactor operation.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 31 are met.

Additional information discussing the compliance approach to GDC 31 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.4.3 GDC 32 – Inspection of Reactor Coolant Pressure Boundary

GDC 32 Statement

Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

Evaluation Against GDC 32

The RPV design includes provisions for ISI. Access to the RPV is provided by removable panels in the insulation. These openings provide access for examination of the RPV and its appurtenances. Removable insulation is provided on the NBS piping and valves extending out to and including the outermost containment isolation valve. Inspection of the RCPB, which includes the ICS, is in accordance with ASME BPVC, Section XI.

Material surveillance samples are located within the RPV.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 32 are met.

Additional information discussing the compliance approach to GDC 32 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.4.4 GDC 33 – Reactor Coolant Makeup

GDC 33 Statement

A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

Evaluation Against GDC 33

GDC 33 applies to small leaks in the RCPB that are smaller than the double-ended rupture of an instrument line (e.g., leakage from flanges or cracks in piping or other components), and which do not exceed the capability of the SC3 high-pressure CRD system used as normal reactor coolant makeup during power operations. The maximum allowed leakage rate for continuing power operation is stipulated in the plant TS. Automatic reactor scram and automatic actuation of the ICS are not anticipated for most of these small leaks because the normal means of makeup from the high-pressure CRD system is sufficient and feedwater maintains the level in the normal operating range. These small leaks, which do not exceed the capability of the high-pressure CRD system, are evaluated using SAFDLs rather than the BWRX-300 acceptance criteria in response to a LOCA.

The high-pressure CRD system pumps can operate using AC power from the SC3 Standby Power System which can be powered from onsite power (assuming offsite power is not available) or offsite power (assuming onsite power is not available).

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 33 are met.

Additional information discussing the compliance approach to GDC 33 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.4.5 GDC 34 – Residual Heat Removal

GDC 34 Statement

A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Evaluation Against GDC 34

The SC1 ICS provides the necessary residual heat removal capability for ensuring the SAFDLs are met and RCPB design conditions are not exceeded.

Residual heat removal is performed passively by the ICS system utilizing only the battery-backed SC1 Emergency Power System.

Three trains of ICS are provided, such that a single failure of any one train does not prevent successful performance of the residual heat removal function.

During refueling, when ICS is unavailable, SC3 active systems are employed to remove decay heat. In case the active systems are lost, the decay heat from the fuel in the RPV can be removed by the UHS.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 34 are met.

Additional information discussing the compliance approach to GDC 34 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.4.6 GDC 35 – Emergency Core Cooling

GDC 35 Statement

A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Evaluation Against GDC 35

The combined design features of the ICS and RIVs meet the definition of an Emergency Core Cooling System (ECCS) as described in 10 CFR 50.46(a)(1)(i) that has a calculated cooling performance following postulated LOCAs in compliance with the BWRX-300 acceptance criteria in response to a LOCA, which bound the acceptance criteria set forth in 10 CFR 50.46(b). In addition, the ICS and RIVs are effective as an ECCS for breaks in pipes in the RCPB up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the RCS in compliance with the definition of a LOCA in 10 CFR 50.46(c)(1). The ICS has the capability to provide more than sufficient emergency core cooling, which is assured for breaks in pipes in the RCPB up to and including a break equivalent in size to the double ended rupture of the largest pipe in the RCS through the use of the RIVs for LOCA break sizes greater than 19 millimeters (0.75 inch) diameter.

The BWRX-300 acceptance criteria in response to a LOCA is that reactor water level is maintained at or above the Top of Active Fuel (TAF) or fuel cladding temperature is maintained within normal operating temperature range, such that the performance of the ICS and RIVs is sufficient to ensure that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

The ICS design includes features that provide redundancy in components and functions. The worst-case single failure can only affect one ICS train, and their design function can be accomplished with the remaining ICS trains. The ICS is a passive system only requiring condensate return valves to open to place the system in operation. Only the battery-backed SC1 Emergency Power System is required for operation of the ICS condensate return valve actuators, controls, and actuation systems, and no power is required after the initial actuation of the ICS condensate return valves. The condensate return valves fail open on loss of all power. Diversity of the ICS automatic actuation valves and automatic actuation signals is provided to ensure that single active failures do not prevent an ICS train from being placed into service. The heat capacity of the Isolation Condenser (IC) heat exchangers and the opening times for the ICS condensate return valves are selected to assure that the BWRX-300 acceptance criteria in response to a LOCA, which bound the acceptance criteria of 10 CFR 50.46(b), are met.

The RIV design includes design features that provide redundancy in components and features, including isolation capability, because the worst-case single failure can only affect one RIV in each RPV connection, and their design function can be accomplished with the remaining RIV in each RPV connection. Other than the ICS RIVs, all RIVs fail to the closed position upon loss of power. If necessary upon a break of an ICS line, the battery-backed SC1 Emergency Power System is required for ICS RIV closure. ICS RIVs fail as-is upon loss of control power. Diversity of the RIV automatic actuation signals is provided. The isolation times of the RIVs are selected to assure that the BWRX-300 acceptance criteria in response to a LOCA, which bound the acceptance criteria of 10 CFR 50.46(b), are met.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 35 are met.

Additional information discussing the compliance approach to GDC 35 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.4.7 GDC 36 – Inspection of Emergency Core Cooling System

GDC 36 Statement

The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.

Evaluation Against GDC 36

The ICS and RIVs, which provide the ECCS function, are designed to permit appropriate periodic inspection of important components, such as piping and valves, to ensure integrity and capability of the ECCS.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 36 are met.

Additional information discussing the compliance approach to GDC 36 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.4.8 GDC 37 – Testing of Emergency Core Cooling System

GDC 37 Statement

The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Evaluation Against GDC 37

The ICS and RIVs, which provide the ECCS function, are designed for periodic pressure and functional testing to assure (1) the structural and leaktight integrity of these components, (2) the operability and performance of components which change state to place the ECCS function into operation, and (3) the operability of the systems as a whole.

The Safety Category 1 ICS initiation functions are implemented to allow for periodic testing of the operational sequences that bring the ICS cooling trains into operation, as required by plant TSs.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 37 are met.

Additional information discussing the compliance approach to GDC 37 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.4.9 GDC 38 – Containment Heat Removal

GDC 38 Statement

A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Evaluation Against GDC 38

Containment peak pressure and temperature are limited by condensation on containment walls and RPV heat removal by the ICS as well as containment heat removal by the PCCS by natural convection and condensation. The PCCS reduces containment peak pressure rapidly for a large break LOCA, which is the limiting BWRX-300 DBA event sequence. Heat is rejected to the

equipment pool and reactor cavity pool above containment by natural circulation using PCCS piping that penetrates the containment. Unisolated small breaks are not limiting for containment peak pressure or temperature. Small breaks less than or equal to 19 millimeters (0.75 inch) do not need to be isolated. These breaks are mitigated by the elevated placement of the small nozzles on the RPV assembly compared to TAF, containment size and the IC heat exchanger size, which helps to depressurize the RPV such that the RPV pressure and containment pressure equalize, and the inventory loss is limited. The PCCS does not require electrical power to operate, and the ICS operation only requires one-time actuation using the battery-backed SC1 Emergency Power System. Additionally, the ICS fails to an in-service state upon loss of all electrical power.

The PCCS does not have a high duty function for long-term containment cooling as most of the decay heat is removed by the ICS. Three independent redundant trains of PCCS are included in the design to provide a high degree of reliability. For RPV isolation events, containment pressure and temperature are limited by condensation on containment walls and containment heat removal by the PCCS and by RPV decay heat removal by the ICS.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 38 are met.

Additional information discussing the compliance approach to GDC 38 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.4.10 GDC 39 – Inspection of Containment Heat Removal System

GDC 39 Statement

The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system.

Evaluation Against GDC 39

The PCCS is designed, fabricated, erected, inspected, and tested in accordance with ASME BPVC, Section III, Subsections NCA and NCD (for Class 2 components) to permit preservice baseline and Inservice Inspections (ISIs) per ASME BPVC, Section XI, Subsection IWC, as applicable.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 39 are met.

Additional information discussing the compliance approach to GDC 39 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.4.11 GDC 40 – Testing of Containment Heat Removal System

GDC 40 Statement

The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and under conditions as close to the design as practical the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Evaluation Against GDC 40

The PCCS accomplishes the containment heat removal function during a large break LOCA. The PCCS forms a closed system inside containment. The PCCS is designed to be periodically pressure tested as part of the ISI Program required by ASME BPVC, Section XI, Subsection IWC

to demonstrate structural and leak-tight integrity. The PCCS piping can be pressure and leak tested during maintenance or ISI using various non-destructive methods.

Functional and operability testing of PCCS components is performed as required by the ASME OM Code and 10 CFR 50, Appendix J. System performance is established analytically for the range of in-containment environmental conditions following a LOCA.

The components of the PCCS are designed with sufficient margin to assure that these requirements for periodic pressure and functional testing during normal operations and emergency events are met. The operation of the PCCS does not require offsite electric power system operation.

The PCCS heat removal function is based on natural circulation and heat removal from condensation, which is analyzed as part of plant design. PCCS operability and performance are ensured by the physical arrangement of the PCCS piping and verified by periodic pressure testing and verification of an open flow path through the system.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 40 are met.

Additional information discussing the compliance approach to GDC 40 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.4.12 GDC 41 – Containment Atmosphere Cleanup

GDC 41 Statement

Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

Evaluation Against GDC 41

The BWRX-300 containment is maintained nitrogen-inerted by the Containment Inerting System (CIS). Fission products, hydrogen, oxygen, and other substances released from the reactor are contained within the low-leakage containment. Because the containment is an inert nitrogen environment, the containment integrity is maintained without employing any further combustible gas control system. Monitoring of oxygen and hydrogen is provided by the Containment Monitoring Subsystem (CMon). Containment leakage after an accident is limited such that the dose guidelines of 10 CFR 50.34(a)(1)(ii)(D) are not exceeded. The BWRX-300 containment is subterranean with underwater pools in the upper RB. As a result, containment leakage pathways to the environment are limited.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 41 are met.

Additional information discussing the compliance approach to GDC 41 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.4.13 GDC 42 – Inspection of Containment Atmosphere Cleanup Systems

GDC 42 Statement

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.

Evaluation Against GDC 42

The CIS inerts the containment atmosphere for plant operation and subsequently exhausts the containment atmosphere to the Heating, Ventilation, and Cooling System (HVS) when the containment is de-inerted. All CIS components are outside containment, accessible for visual inspection during normal plant operation at power, and available for ISI during all normal plant states to assure integrity and functionality of components.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 42 are met.

Additional information discussing the compliance approach to GDC 42 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.4.14 GDC 43 – Testing of Containment Atmosphere Cleanup Systems

GDC 43 Statement

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

Evaluation Against GDC 43

The CIS inerts the containment atmosphere for plant operation and subsequently exhausts the containment atmosphere to the HVS during refueling. The CIS is designed to accommodate preservice testing, pressure testing, and functional testing to ensure the system performs its intended functions.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 43 are met.

Additional information discussing the compliance approach to GDC 43 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.4.15 GDC 44 – Cooling Water

GDC 44 Statement

A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Evaluation Against GDC 44

During normal operation, the Normal Heat Sink (NHS) System transfers heat from plant equipment to the environment. If the NHS System is not available, heat removal is performed by the Ultimate Heat Sink (UHS). The UHS is a passive safety feature and includes the IC pools, reactor cavity pool, fuel pool, and equipment pool. Reactor residual heat is transferred through the ICS to the Isolation Condenser Pools during off-normal conditions, including AOO event sequences, DBA event sequences, and DEC event sequences.

Heat from the containment is transferred to the equipment pool and reactor cavity pool by the PCCS, as discussed for GDC 38. Decay heat from fuel in the fuel pool can be removed by its volume of water as discussed for GDC 61. During refueling, the decay heat for the fuel in the RPV can be removed by the ultimate heat sink.

The IC pool volumes are independent from the equipment pool, fuel pool, and reactor cavity pool.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 44 are met.

Additional information discussing the compliance approach to GDC 44 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.4.16 GDC 45 – Inspection of Cooling Water System

GDC 45 Statement

The cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.

Evaluation Against GDC 45

The ICS heat exchangers are located above containment in the RB and can be made accessible for periodic inspections. During outages, the IC pool compartments can be drained to permit inspection of the pools and condensers.

See GDC 39 for discussion of inspection of the PCCS.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 45 are met.

Additional information discussing the compliance approach to GDC 45 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.4.17 GDC 46 – Testing of Cooling Water System

GDC 46 Statement

The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

Evaluation Against GDC 46

The ICS is designed to allow the performance of surveillance testing in accordance with plant TSs. The ICS performance testing also confirms the proper functioning of the IC pools in support of heat removal. Each ICS train will be tested for heat transfer performance on a rotating basis.

Maintenance and testing activities are expected to be minimal as the only active mechanical components are the two per-train condensate return valves, the two per-train standby gas purge isolation valves and the two per-A and -B train Shutdown Cooling System (SDC) interface CIVs. The IC pools used as the heat sink perform their function without reliance on active component actuations.

See GDC 37 for discussion of testing of ICS. See GDC 40 for discussion of testing of the PCCS.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 46 are met.

Additional information discussing the compliance approach to GDC 46 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.5 Group V – Reactor Containment

3.1.5.1 GDC 50 – Containment Design Basis

GDC 50 Statement

The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by § 50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

Evaluation Against GDC 50

Containment design is based upon consideration of a full spectrum of postulated accidents that would result in the release of reactor coolant to the containment. These accidents are evaluated using Transient Reactor Analysis Code General Electric (TRACG) as boundary condition to Generation of Thermal-Hydraulic Information for Containment (GOTHIC) to calculate containment response. These accidents include liquid, steam, and partial (both steam and liquid) breaks. The evaluation of the containment design is based upon enveloping the results of this range of analyses, plus provision for appropriate margin. The most limiting short-term and long-term pressure and temperature responses are assessed to verify the integrity of the containment structure. The GOTHIC computer methodology for measuring containment response is provided in NEDC-33922P-A, “BWRX-300 Containment Evaluation Method,” (Reference 3.1-4). BWRX-300 does not have potential energy sources that are not accounted for in calculating the mass and energy release from RPV.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 50 are met.

Additional information discussing the compliance approach to GDC 50 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.5.2 GDC 51 – Fracture Prevention of Containment Pressure Boundary

GDC 51 Statement

The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.

The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady-state, and transient stresses, and (3) size of flaws.

Evaluation Against GDC 51

The BWRX-300 containment possesses ductility and energy absorbing capacity, which permits inelastic deformation without failure. The seismic design is qualified to meet the ductility detailing and design requirements for steel and steel composite structures of American National Standards Institute (ANSI) / American Institute of Steel Construction (AISC) N690-18, "Specification for Safety-Related Steel Structures for Nuclear Facilities," (Reference 3.1-9), with the supplementary guidance of RG 1.243, "Safety-Related Steel Structures and Steel-Plate Composite Walls for other than Reactor Vessels and Containments." Additionally, the ductility is confirmed by the ultimate capacity analysis of the BWRX-300 containment. The design material selection reflects consideration of service temperatures and other conditions of the pressure boundary during operation, maintenance, testing, and DBA conditions, and the uncertainties in determining: (a) material properties; (b) residual, steady-state, and transient stresses; and (c) the size of flaws. The steel composite design is capable of containing the loads resulting from design basis transients and accidents, including dynamic loads due to seismic and high-energy piping system events.

The preoperational test program and the QA program ensure the integrity of the containment and its ability to meet all normal operating and accident requirements.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 51 are met.

Additional information discussing the compliance approach to GDC 51 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.5.3 GDC 52 – Capability for Containment Leakage Rate Testing

GDC 52 Statement

The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

Evaluation Against GDC 52

The BWRX-300 containment and other equipment that may be subjected to containment test conditions are designed so that periodic integrated leakage rate testing can be conducted at containment design pressure at periodic intervals in accordance with 10 CFR 50, Appendix J.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 52 are met.

Additional information discussing the compliance approach to GDC 52 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.5.4 GDC 53 – Provisions for Containment Testing and Inspection

GDC 53 Statement

The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.

Evaluation Against GDC 53

The BWRX-300 containment and associated penetrations have provisions for conducting individual leakage rate tests on applicable penetrations. Penetrations are visually inspected and pressure tested for leaktightness at periodic intervals in accordance with 10 CFR 50, Appendix J.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 53 are met.

Additional information discussing the compliance approach to GDC 53 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.5.5 GDC 54 – Piping Systems Penetrating Containment

GDC 54 Statement

Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

Evaluation Against GDC 54

Piping systems that penetrate containment are designed to provide the required isolation and testing capabilities. These piping systems are provided with test connections to allow periodic leak detection as necessary to determine if valve leakage is within acceptable limits in accordance with 10 CFR 50, Appendix J and the IST Program.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 54 are met.

Additional information discussing the compliance approach to GDC 54 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.5.6 GDC 55 – Reactor Coolant Pressure Boundary Penetrating Containment

GDC 55 Statement

Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

1. One locked closed isolation valve inside and one locked closed isolation valve outside containment; or
2. One automatic isolation valve inside and one locked closed isolation valve outside containment; or
3. One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or
4. One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for ISI, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

Evaluation Against GDC 55

Each process line that penetrates the containment boundary and is part of the RCPB is provided with two in-series CIVs that are independently able to isolate the line with single failure proof automatic isolation. For systems with integral RIVs, either RIV can perform the inboard CIV function. The CIVs outside containment are located as close to the containment boundary as practicable. The integral RIV design includes design features that provide redundancy in components and features, including isolation capability, because the worst-case single failure can only affect one valve, and their design function can be accomplished with the remaining valve in each integral RIV body. Process line configurations are designed for containment leakage rate testing and IST programs that include functional testing, local leakage rate testing, and Integrated Leakage Rate Testing (ILRT). Automatic isolation valves are designed to take the position that provides greater safety upon loss of actuating power.

The NBS, CUW, and CFS have inboard integral RIVs and outboard CIVs. The supply lines for SDC Trains A and B are part of the ICS and comply with GDC 55 with an outboard CIV and the inboard ICS RIVs. The injection line for the BIS complies with GDC 55 with an outboard CIV and an inboard CIV.

Process lines that comply with GDC 55 on an “other defined basis” are:

- ICS, which can be isolated by the integral RIVs inside containment and is a closed loop design outside containment. The ICS steam supply and condensate return lines to the RPV each have an integral RIV inside containment. The ICS steam supply line to the isolation condensers is inside a guard pipe. This arrangement complies with GDC 55 based on an “other defined basis” because integral RIVs function as inboard CIVs for both steam supply and condensate return lines and the closed-loop piping outside containment can prevent radioactive releases outside containment.
- CRD system uses internal ball check valves that serve as the automatic isolation CIV inside containment to isolate the hydraulic lines to the FMCRDs. A manual isolation valve is provided outside containment to isolate the HCU from the hydraulic insertion line. This configuration complies with GDC 55 based on an “other defined basis” because keeping flow path open is necessary to perform the scram function and comparing risk between a degraded scram function and the negative radioactive fission product releases without an outboard automatic CIV, the scram function demonstrates higher risk significance; thus, automatic isolation is not required.

Instrument lines are provided with excess flow check valves located outside containment meeting RG 1.11, “Instrument Lines Penetrating the Primary Reactor Containment,” and comply with GDC 55 on an “other defined basis.”

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 55 are met.

Additional information discussing the compliance approach to GDC 55 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.5.7 GDC 56 – Primary Containment Isolation

GDC 56 Statement

Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

1. One locked closed isolation valve inside and one locked closed isolation valve outside containment; or
2. One automatic isolation valve inside and one locked closed isolation valve outside containment; or
3. One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or
4. One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Evaluation Against GDC 56

Each process line that connects directly to the containment atmosphere and penetrates the containment boundary is provided with in-series CIVs that are independently able to isolate the line with single failure proof automatic isolation. The CIVs outside containment are located as close to the containment boundary as practicable. Automatic isolation valves are designed to take the position that provides greater safety upon loss of actuating power.

The CMon, Chilled Water Equipment (CWE), Plant Pneumatics System (PPS), Demineralized Water Storage and Distribution Subsystem, and Liquid Waste Management System (LWM) process lines have inboard and outboard CIVs.

The Containment Inerting System (CIS) and the Equipment and Floor Drain System (EFS) process lines comply with GDC 56 on an “other defined basis.” The CIS and EFS process lines have two CIVs outside containment because locating a valve inside containment is not practicable.

Instrument lines are provided with excess flow check valves located outside containment meeting RG 1.11 and comply with GDC 56 on an “other defined basis.”

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 56 are met.

Additional information discussing the GDC 56 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.5.8 GDC 57 – Closed System Isolation Valves

GDC 57 Statement

Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

Evaluation Against GDC 57

The PCCS is a closed system inside containment that penetrates the containment boundary and is neither part of the RCPB nor connected directly to the containment atmosphere. The PCCS has three trains that each have one remote manually-operated CIV on the supply line and one remote manually-operated CIV on the return line. Both CIVs are outside containment and located as close to the containment wall as practical.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 57 are met.

Additional information discussing the compliance approach to GDC 57 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.6 Group VI – Fuel and Radioactivity Control

3.1.6.1 GDC 60 – Control of Releases of Radioactive Materials to the Environment

The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

Evaluation Against GDC 60

Gaseous releases come primarily from the Offgas System (OGS) and the ventilation systems. Noble gases and radioiodine activity that enters the OGS are held-up by activated carbon charcoal adsorber beds, allowing for the decay of these gases. Ventilation releases are through the Plant Vent Stack (PVS) located in the Turbine Building (TB). Automatic isolation functions are provided in the event of high radiation detection.

The LWM design has the capability to process liquid waste from normal operation as well as from occasional emergent issues, for reuse by other plant systems. In the unlikely event that the plant's overall water inventory will not allow for the water to be recycled, the filtered water can be discharged to the environment. Any environmental discharge is controlled to ensure concentrations of contaminants are maintained within permissible limits.

The Solid Waste Management System (SWM) is designed to handle solid waste generated by the plant during normal operations and AOOs. The SWM includes design features that prevent the release of liquid or gaseous effluents from the system to the environment.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 60 are met.

Additional information discussing the compliance approach to GDC 60 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.6.2 GDC 61 – Fuel Storage and Handling and Radioactivity Control

GDC 61 Statement

The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

Evaluation Against GDC 61

The fuel pool has adequate water shielding for stored spent fuel. Adequate shielding for transporting fuel is also provided between the RPV and fuel pool. Sensors are installed to detect low pool water level. The fuel pool is designed with no penetrations below the water level that is needed for adequate shielding at the operating floor. Anti-siphoning provisions protect against draining the fuel pool in the event of a line break.

Fuel storage racks are provided in the fuel pool adjacent to the reactor cavity. The fuel storage racks are top loading, allowing for inspection of the cells and identification of individual fuel assemblies from the poolside or refueling platform. The fuel storage rack design allows natural convection from the space underneath of the rack up through the fuel seating elevation to ensure adequate cooling water flow to all stored fuel assemblies removing decay heat under normal and abnormal conditions.

The Fuel Pool Cooling and Cleanup System (FPC) normally removes decay heat from the fuel pool. Without the active cooling trains of the FPC, sufficient pool water inventory is provided to permit boiling for up to 7-days without makeup. If required, makeup water is provided from onsite sources. FPC piping is used to transport makeup water to the fuel pool from various sources in accordance with the FLEX/EME strategy; this includes a connecting point in the yard area to portable water sources.

The refueling platform includes a load handling system designed to safely handle, store, and support inspection of fuel and core components. The refueling platform handles fuel within the reactor, lifts from the reactor to the fuel pool, and is utilized to load casks for long term storage of spent fuel. The load handling system design ensures that no single probable event or component failure results in dropping or damaging a fuel assembly. The refueling platform components allow access for inspection and surveillance and maintenance activities. Mechanical stops and interlocks are provided to prevent damage to fuel assemblies and conditions that pose a radiation hazard or unintentional exposure risk to personnel. Travel limits physically block hoisting of fuel and components above approved levels ensuring exposure is kept As Low As Reasonably Achievable (ALARA).

The fuel storage and handling systems are designed to ensure adequate safety under normal and postulated abnormal conditions.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 61 are met.

Additional information discussing the compliance approach to GDC 61 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.6.3 GDC 62 – Prevention of Criticality in Fuel Storage and Handling

GDC 62 Statement

Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

Evaluation Against GDC 62

Fuel handling and storage facilities are provided to preclude inadvertent criticality during handling and storage of new and spent fuel. Criticality of fuel stored in the fuel pool is prevented by racks made of fixed neutron absorbing material maintained in geometrically safe storage and subcritical arrays during normal and credible abnormal conditions. Criticality in the fuel storage racks is prevented by presence of fixed neutron absorbing material. The fuel storage racks are Seismic Category I components.

The new and spent fuel is stored under water in the fuel pool. A full array of loaded fuel storage racks with spent fuel is designed to be subcritical. Neutron absorbing material, as an integral part of the rack design, is employed to assure that the calculated K_{eff}, including biases and uncertainties, does not exceed 0.95 under all normal and abnormal conditions. The abnormal conditions accounted for are an earthquake, accidental dropping of equipment, or impact caused by the horizontal movement of the fuel handling equipment without first disengaging the fuel from the hoisting equipment.

Refueling interlocks include circuitry that senses conditions of the refueling equipment and the control rods. These interlocks reinforce the operational procedures that prohibit making the reactor critical. The fuel handling system is designed to provide a safe, effective means of transporting and handling and is designed to minimize the possibility of mishandling or maloperation.

Fuel handling equipment is designed to prevent inadvertent criticality and to maintain shielding and cooling of spent fuel as necessary to meet operating and off-site dose constraints.

The presence of fixed neutron absorbing material in the fuel storage racks and the design of fuel handling systems precludes accidental criticality.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 62 are met.

Additional information discussing the compliance approach to GDC 62 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.6.4 GDC 63 – Monitoring Fuel and Waste Storage

GDC 63 Statement

Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

Evaluation Against GDC 63

Fuel pool temperature and level are monitored as part of the FPC. High pool temperature would signal the need for providing additional cooling. Area radiation monitors are provided as part of the Area Radiation Monitoring (ARM) Subsystem, which monitors the operating/refueling floors for high radiation levels.

The LWM and SWM have no active decay heat removal functions because the decay heat from the activity in the inputs to these systems is not sufficient to warrant concern. Radwaste Building (RWB) area radiation monitors are provided to monitor radiation levels in the RWB.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 63 are met.

Additional information discussing the compliance approach to GDC 63 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.6.5 GDC 64 – Monitoring Radioactivity Releases

GDC 64 Statement

Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

Evaluation Against GDC 64

The BWRX-300 is provided with the following subsystems to monitor for radioactivity releases from normal operations, AOOs, and DBAs:

- A Process Radiation Monitoring Subsystem, which monitors various process and effluent streams for potentially hazardous contamination such as radioactivity, noble gases, air particulates, and halogens
- An Area Radiation Monitoring Subsystem, which monitors gamma radiation levels in strategic locations throughout the plant
- A CMon, which monitors containment pressure, containment temperature, containment water level, containment hydrogen concentration, containment oxygen concentration, containment fission products, and containment area radiation levels

Note that the BWRX-300 design does not include a “recirculation” mode for LOCAs in which post-accident fluids are recirculated.

Accordingly, the requirements of 10 CFR Part 50, Appendix A, GDC 64 are met.

Additional information discussing the compliance approach to GDC 64 is described in the applicable PSAR sections listed in Table 3.1-1.

3.1.7 References

3.1-1 NEDC-33934P, “BWRX-300 Safety Strategy,” GE Vernova Hitachi Nuclear Energy Americas, LLC, Revision 2, March 2026.

3.1-2 NEDC-33910P-A, “BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection,” GE-Hitachi Nuclear Energy Americas, LLC, Revision 2, June 2021.

3.1-3 NEDC-33912P-A, “BWRX-300 Reactivity Control,” GE-Hitachi Nuclear Energy Americas, LLC, Revision 2, February 2023.

3.1-4 NEDC-33922P-A, “BWRX-300 Containment Evaluation Method,” GE-Hitachi Nuclear Energy Americas, LLC, Revision 3, June 2022.

3.1-5 NEDC-33926P-A, “BWRX-300 Steel-Plate Composite (SC) Containment Vessel (SCCV) and Reactor Building Structural Design,” GE Vernova Hitachi Nuclear Energy, Americas, LLC, Revision 4, December 2025.

3.1-6 ASME NQA-1-2015, “Quality Assurance Requirements, for Nuclear Facility Applications,” American Society of Mechanical Engineers, 2015.

3.1-7 NEDE-24011-P-A-31, “General Electric Standard Application for Reactor Fuel (GESTAR II),” Global Nuclear Fuel – Americas, LLC, Revision 31, November 2020.

3.1-8 NEDC-33270P, “GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II),” Global Nuclear Fuel - Americas, LLC, Revision 11, August 2020.

3.1-9 ANSI/AISC N690-18, “Specification for Safety-Related Steel Structures for Nuclear Facilities,” American National Standards Institute/American Institute of Steel Construction, 2018.

Table 3.1-1: Compliance with General Design Criteria/Principal Design Criteria

GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
II	10	Reactor Design	3.9.5	Reactor Pressure Vessel Internals	Complies
II	10	Reactor Design	4.2	Fuel System Design	Complies
II	10	Reactor Design	4.3	Nuclear Design	Complies
II	10	Reactor Design	4.4	Thermal and Hydraulic Design	Complies
II	10	Reactor Design	7.1	I&C Systems	Complies
II	10	Reactor Design	15.3	Safety Objectives and Acceptance Criteria	Complies
II	10	Reactor Design	LTR 5.2.5(1)	GDC 10	Complies
II	11	Reactor Inherent Protection	4.3	Nuclear Design	Complies
II	12	Suppression of Reactor Power Oscillations	4.3	Nuclear Design	Complies
II	12	Suppression of Reactor Power Oscillations	4.4	Thermal and Hydraulic Design	Complies
II	12	Suppression of Reactor Power Oscillations	15.3	Safety Objectives and Acceptance Criteria	Complies
II	12	Suppression of Reactor Power Oscillations	LTR 5.2.6(1)	GDC 12	Complies
II	12	Suppression of Reactor Power Oscillations	LTR 4.1.2(4)	10 CFR 50 Appendix A, GDC 12	Complies
II	13	Instrumentation and Control	4.3	Nuclear Design	Complies
II	13	Instrumentation and Control	6.2.1	Primary Containment System	Complies
II	13	Instrumentation and Control	7.1	I&C Systems	Complies
II	13	Instrumentation and Control	9.3.2	Process Sampling System	Complies

GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
II	13	Instrumentation and Control	15.2.4.3	Reactivity and Power Distribution Anomalies Bounding Event Selection	FSAR
II	14	Reactor Coolant Pressure Boundary	3.9.1	Special Topics for Mechanical Components	Complies
II	14	Reactor Coolant Pressure Boundary	3.9.2	Dynamic Testing and Analysis of Systems, Components, and Equipment	Complies
II	14	Reactor Coolant Pressure Boundary	3.9.3	ASME BPVC Class 1, 2, and 3 Components, Component Supports, and Core Support Structures	Complies
II	14	Reactor Coolant Pressure Boundary	3.9.4	Control Rod Drive System	Complies
II	14	Reactor Coolant Pressure Boundary	3.9.6	Functional Design, Qualification, and IST Programs for Pumps, Valves, and Dynamic Restraints	Complies
II	14	Reactor Coolant Pressure Boundary	3.10	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	Complies
II	14	Reactor Coolant Pressure Boundary	3.12	ASME Code Class 1, 2 and 3 Piping Systems, Piping Components and Their Associated Supports	Complies
II	14	Reactor Coolant Pressure Boundary	3.13	Threaded Fasteners – ASME Code Class 1, 2, and 3	Complies
II	14	Reactor Coolant Pressure Boundary	4.5.1	Control Rod Drive System Structural Materials	Complies
II	14	Reactor Coolant Pressure Boundary	5.2.3	Reactor Coolant Pressure Boundary Materials	Complies

GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
II	14	Reactor Coolant Pressure Boundary	5.3.1	Reactor Vessel Materials	Complies
II	14	Reactor Coolant Pressure Boundary	5.3.2	Pressure-Temperature Limits	Complies
II	14	Reactor Coolant Pressure Boundary	5.3.3	Reactor Vessel Integrity	Complies
II	14	Reactor Coolant Pressure Boundary	5.4.8	Reactor Water Cleanup System	Complies
II	14	Reactor Coolant Pressure Boundary	5.4.13	Reactor Pressure Vessel Head Vent Subsystem	Complies
II	14	Reactor Coolant Pressure Boundary	6.1	Engineered Safety Feature Materials	Complies
II	14	Reactor Coolant Pressure Boundary	6.3	Emergency Core Cooling System (Isolation Condenser System)	Complies
II	14	Reactor Coolant Pressure Boundary	9.3.2	Process Sampling System	Complies
II	14	Reactor Coolant Pressure Boundary	10.4.6	Condensate Filters and Demineralizers System	Complies
II	14	Reactor Coolant Pressure Boundary	LTR 4.1.7(2)	10 CFR 50 Appendix A, GDC 14	Complies
II	15	Reactor Coolant System Design	3.9.1	Special Topics for Mechanical Components	Complies
II	15	Reactor Coolant System Design	3.9.2	Dynamic Testing and Analysis of Systems, Components, and Equipment	Complies
II	15	Reactor Coolant System Design	3.9.3	ASME BPVC Class 1, 2, and 3 Components, Component Supports, and Core Support Structures	Complies

GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
II	15	Reactor Coolant System Design	3.9.6	Functional Design, Qualification, and IST Programs for Pumps, Valves, and Dynamic Restraints	Complies
II	15	Reactor Coolant System Design	3.12	ASME Code Class 1, 2 and 3 Piping Systems, Piping Components and Their Associated Supports	Complies
II	15	Reactor Coolant System Design	6.3	Emergency Core Cooling System (Isolation Condenser System)	Complies
II	15	Reactor Coolant System Design	7.1	I&C Systems	Complies
II	15	Reactor Coolant System Design	15.3	Safety Objectives and Acceptance Criteria	Complies
II	15	Reactor Coolant System Design	LTR 5.2.7(1)	GDC 15	Complies
II	15	Reactor Coolant System Design	LTR 4.1.8(2)	10 CFR 50 Appendix A, GDC 15	Complies
II	16	Containment Design	3.8.1	Steel-Plate Composite Containment Vessel	Complies
II	16	Containment Design	3.8.2	Class MC Steel Components of the BWRX-300 Containment	Complies
II	16	Containment Design	6.2.1	Primary Containment System	Complies
II	16	Containment Design	6.2.4	Containment Isolation Function	Complies
II	16	Containment Design	6.2.7	Fracture Prevention of Containment Pressure Boundary	Complies
II	16	Containment Design	7.1	I&C Systems	Complies
II	16	Containment Design	LTR 2.1.1.9(5)	10 CFR 50 Appendix A, GDC 16	Complies
II	PDC 17	Electric Power Systems	5.4.13	Reactor Pressure Vessel Head Vent Subsystem	Complies
II	PDC 17	Electric Power Systems	6.2.2	Passive Containment Cooling System	Complies

GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
II	PDC 17	Electric Power Systems	6.3	Emergency Core Cooling System (Isolation Condenser System)	Complies
II	PDC 17	Electric Power Systems	8.1.3	Design Criteria, Regulatory Guides, and Standards	Complies
II	PDC 17	Electric Power Systems	8.3.1.8	Containment Building Electrical Penetrations	Complies
II	PDC 17	Electric Power Systems	9.5.4	Standby Diesel Generator Fuel Oil Storage and Transfer System	Complies
II	PDC 17	Electric Power Systems	15.3	Safety Objectives and Acceptance Criteria	Complies
II	18	Inspection and Testing of Electric Power Systems	8.1.3	Design Criteria, Regulatory Guides, and Standards	N/A GDC 18 does not apply to offsite power, SDG AC power, or SC2 DC backed power
II	PDC 19	Control Room	5.4.7	Shutdown Cooling System	N/A PDC 19 does not apply for SDC normal instrumentation
II	PDC 19	Control Room	5.4.13	Reactor Pressure Vessel Head Vent Subsystem	N/A PDC 19 does not apply for the RPV head vent subsystem normal instrumentation.
II	PDC 19	Control Room	6.4	Control Room Habitability	Complies
II	PDC 19	Control Room	7.1	I&C Systems	Complies
II	PDC 19	Control Room	9.4.1	Control Building Heating, Ventilation, and Cooling System	Complies
II	PDC 19	Control Room	9.4.6	Reactor Building Heating, Ventilation, and Cooling System	Complies

GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
II	PDC 19	Control Room	9.5.2	Non-Distributed Control and Information Communication System	N/A Subsystem does not contain any SC1 SSCs relied on for safe shutdown
II	PDC 19	Control Room	11.5	Radiation Monitoring	Complies
II	PDC 19	Control Room	15.3.1	Deterministic Safety Analysis Acceptance Criteria	Complies
II	PDC 19	Control Room	15.5.1.2.7	Description of the Computer Codes or Standards Used in the Safety Analyses	Complies
II	PDC 19	Control Room	15.5.8	Analysis of Fuel Handling Accident	Complies
II	PDC 19	Control Room	15.5.9.1	Analysis of Loss of Cooling Accidents Outside Containment	Complies
III	20	Protection System Functions	4.3	Nuclear Design	Complies
III	20	Protection System Functions	7.1	I&C Systems	Complies
III	20	Protection System Functions	15.3	Safety Objectives and Acceptance Criteria	Complies
III	20	Protection System Functions	LTR 5.2.9(1)	GDC 20	Complies
III	20	Protection System Functions	LTR 4.1.3(4)	10 CFR 50 Appendix A, GDC 20	Complies
III	21	Protection System Reliability and Testability	7.1	I&C Systems	Complies
III	21	Protection System Reliability and Testability	LTR 5.2.10(1)	GDC 21	Complies
III	21	Protection System Reliability and Testability	LTR 4.1.4(4)	10 CFR 50 Appendix A, GDC 21	Complies
III	22	Protection System Independence	7.1	I&C Systems	Complies
III	22	Protection System Independence	LTR 5.2.11(1)	GDC 22	Complies

GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
III	22	Protection System Independence	LTR 4.1.5(4)	10 CFR 50, Appendix A, GDC 22	Complies
III	23	Protection System Failure Modes	3.11	Environmental Qualification of Mechanical and Electrical Equipment	Complies
III	23	Protection System Failure Modes	4.6	Design of Reactivity Control Systems	Complies
III	23	Protection System Failure Modes	7.1	I&C Systems	Complies
III	23	Protection System Failure Modes	LTR 5.2.12(1)	GDC 23	Complies
III	23	Protection System Failure Modes	LTR 4.1.6(4)	10 CFR 50 Appendix A, GDC 23	Complies
III	24	Separation of Protection and Control Systems	7.1	I&C Systems	Complies
III	24	Separation of Protection and Control Systems	LTR 5.2.13(1)	GDC 24	Complies
III	24	Separation of Protection and Control Systems	LTR 4.1.7(4)	10 CFR 50 Appendix A, GDC 24	Complies
III	25	Protection System Requirements for Reactivity Control Malfunctions	4.3	Nuclear Design	Complies
III	25	Protection System Requirements for Reactivity Control Malfunctions	4.6	Design of Reactivity Control Systems	Complies
III	25	Protection System Requirements for Reactivity Control Malfunctions	7.1	I&C Systems	Complies
III	25	Protection System Requirements for Reactivity Control Malfunctions	15.3	Safety Objectives and Acceptance Criteria	Complies

GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
III	25	Protection System Requirements for Reactivity Control Malfunctions	LTR 5.2.14(1)	GDC 25	Complies
III	25	Protection System Requirements for Reactivity Control Malfunctions	LTR 4.1.8(4)	10 CFR 50 Appendix A, GDC 25	Complies
III	PDC 26	Reactivity Control System Redundancy and Capability	3.9.4	Control Rod Drive System	Complies
III	PDC 26	Reactivity Control System Redundancy and Capability	4.3	Nuclear Design	Complies
III	PDC 26	Reactivity Control System Redundancy and Capability	4.5.2	Regulatory Requirements	Complies
III	PDC 26	Reactivity Control System Redundancy and Capability	4.6	Design of Reactivity Control Systems	Complies
III	PDC 26	Reactivity Control System Redundancy and Capability	7.1	I&C Systems	Complies
III	PDC 26	Reactivity Control System Redundancy and Capability	9.3.2	Process Sampling System	N/A
III	PDC 26	Reactivity Control System Redundancy and Capability	15.3	Safety Objectives and Acceptance Criteria	Complies
III	PDC 26	Reactivity Control System Redundancy and Capability	9.3.10	Boron Injection System	Complies
III	27	Combined Reactivity Control Systems Capability	3.9.4	Control Rod Drive System	Complies
III	27	Combined Reactivity Control Systems Capability	4.2	Fuel System Design	Complies
III	27	Combined Reactivity Control Systems Capability	4.3	Nuclear Design	Complies

GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
III	27	Combined Reactivity Control Systems Capability	4.6	Design of Reactivity Control Systems	Complies
III	27	Combined Reactivity Control Systems Capability	6.3	Emergency Core Cooling System (Isolation Condenser System)	Complies
III	27	Combined Reactivity Control Systems Capability	7.1	I&C Systems	Complies
III	27	Combined Reactivity Control Systems Capability	9.3.10	Boron Injection System	N/A
III	PDC 28	Reactivity Limits	4.3	Nuclear Design	Complies
III	PDC 28	Reactivity Limits	4.6	Design of Reactivity Control System	Complies
III	PDC 28	Reactivity Limits	7.1	I&C Systems	Complies
III	PDC 28	Reactivity Limits	15.5.5.1	Control Rod Drop Accident	Complies
III	PDC 28	Reactivity Limits	LTR 4.1.11(4)	10 CFR 50 Appendix A, GDC 28	N/A – comply with PDC 28
III	PDC 29	Protection Against Anticipated Operational Occurrences	3.9.4	CRD System	Complies
III	PDC 29	Protection Against Anticipated Operational Occurrences	4.6	Design of Reactivity Control Systems	Complies
III	PDC 29	Protection Against Anticipated Operational Occurrences	7.1	I&C Systems	Complies
III	PDC 29	Protection Against Anticipated Operational Occurrences	LTR 5.2.15(1)	GDC 29	N/A - comply with PDC 29
III	PDC 29	Protection Against Anticipated Operational Occurrences	LTR 4.1.12(4)	10 CFR 50 Appendix A, GDC 29	N/A - comply with PDC 29
IV	30	Quality of Reactor Coolant Pressure Boundary	3.10	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	Complies

GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
IV	30	Quality of Reactor Coolant Pressure Boundary	3.13	Threaded Fasteners - ASME Code Class 1, 2, and 3	Complies
IV	30	Quality of Reactor Coolant Pressure Boundary	5.2.3	Reactor Coolant Pressure Boundary Materials	Complies
IV	30	Quality of Reactor Coolant Pressure Boundary	5.2.5	Reactor Coolant Pressure Boundary Leakage Detection	Complies
IV	30	Quality of Reactor Coolant Pressure Boundary	5.3.1	Reactor Vessel Materials	Complies
IV	30	Quality of Reactor Coolant Pressure Boundary	5.3.3	Reactor Vessel Integrity	Complies
IV	30	Quality of Reactor Coolant Pressure Boundary	5.4.13	Reactor Pressure Vessel Head Vent Subsystem	Complies
IV	30	Quality of Reactor Coolant Pressure Boundary	6.3	Emergency Core Cooling System (Isolation Condenser System)	Complies
IV	30	Quality of Reactor Coolant Pressure Boundary	LTR 4.1.9(2)	10 CFR Part 50 Appendix A, GDC 30	Complies
IV	31	Fracture Prevention of Reactor Coolant Pressure Boundary	3.13	Threaded Fasteners – ASME Code Class 1, 2, and 3	Complies
IV	31	Fracture Prevention of Reactor Coolant Pressure Boundary	5.2.3	Reactor Coolant Pressure Boundary Materials	Complies
IV	31	Fracture Prevention of Reactor Coolant Pressure Boundary	5.3.1	Reactor Vessel Materials	Complies
IV	31	Fracture Prevention of Reactor Coolant Pressure Boundary	5.3.2	Pressure-Temperature Limits	Complies
IV	31	Fracture Prevention of Reactor Coolant Pressure Boundary	5.3.3	Reactor Vessel Integrity	Complies
IV	31	Fracture Prevention of Reactor Coolant Pressure Boundary	6.1	Engineered Safety Features Materials	Complies
IV	31	Fracture Prevention of RCPB	LTR 4.1.10(2	10 CFR 50 Appendix A, GDC 31	Complies

GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
IV	32	Inspection of Reactor Coolant Pressure Boundary	5.2.4	Reactor Coolant Pressure Boundary Inservice Inspection and Testing	Complies
IV	32	Inspection of Reactor Coolant Pressure Boundary	5.3.1	Reactor Vessel Materials	Complies
IV	32	Inspection of Reactor Coolant Pressure Boundary	5.3.2	Pressure-Temperature Limits	Complies
IV	32	Inspection of Reactor Coolant Pressure Boundary	5.3.3	Reactor Vessel Integrity	Complies
IV	32	Inspection of Reactor Coolant Pressure Boundary	5.4.13	Reactor Pressure Vessel Head Vent Subsystem	Complies
IV	32	Inspection of Reactor Coolant Pressure Boundary	6.3	Emergency Core Cooling System (Isolation Condenser System)	Complies
IV	33	Reactor Coolant Makeup	4.6	Design of Reactivity Control Systems	Complies
IV	33	Reactor Coolant Makeup	5.4.6	Isolation Condenser System	Complies
IV	33	Reactor Coolant Makeup	6.3	Emergency Core Cooling System (Isolation Condenser System)	Complies
IV	33	Reactor Coolant Makeup	7.1	I&C Systems	Complies
IV	33	Reactor Coolant Makeup	8.1.3	Design Criteria, Regulatory Guides, and Standards	Complies
IV	33	Reactor Coolant Makeup	15.3	Safety Objectives and Acceptance Criteria	Complies
IV	33	Reactor Coolant Makeup	LTR 4.1.11[(2)]	10 CFR Part 50 Appendix A, GDC 33	Complies

GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
IV	34	Residual Heat Removal	5.4.7	Residual Heat Removal System	N/A SDC residual heat removal capabilities are not credited in the Chapter 15 safety analyses and ICS is relied on for compliance with GDC 34.
IV	34	Residual Heat Removal	6.3	Emergency Core Cooling System (Isolation Condenser System)	Complies
IV	34	Residual Heat Removal	7.1	I&C Systems	Complies
IV	34	Residual Heat Removal	8.1.3	Design Criteria, Regulatory Guides, and Standards	Complies
IV	34	Residual Heat Removal	10.4.4	Turbine Bypass System	Complies
IV	34	Residual Heat Removal	15.3	Safety Objectives and Acceptance Criteria	Complies
IV	35	Emergency Core Cooling	4.2	Fuel System Design	Complies
IV	35	Emergency Core Cooling	6.1	Engineered Safety Features Materials	Complies
IV	35	Emergency Core Cooling	6.3	Emergency Core Cooling System (Isolation Condenser System)	Complies
IV	35	Emergency Core Cooling	7.1	I&C Systems	Complies
IV	35	Emergency Core Cooling	8.1.3	Design Criteria, Regulatory Guides, and Standards	Complies
IV	35	Emergency Core Cooling	LTR 5.2.17(1)	GDC 35	Complies
IV	35	Emergency Core Cooling	LTR 4.1.12(2)	10 CFR 50 Appendix A, GDC 35	Complies
IV	36	Inspection of Emergency Core Cooling System	6.3	Emergency Core Cooling System (Isolation Condenser System)	Complies

GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
IV	36	Inspection of Emergency Core Cooling System	6.6	Preservice and ISI and Testing of Class 2 and 3 Components and Piping	Complies
IV	37	Testing of Emergency Core Cooling System	3.9.6	Functional Design, Qualification, and IST Programs for Pumps, Valves, and Dynamic Restraints	Complies
IV	37	Testing of Emergency Core Cooling System	6.3	Emergency Core Cooling System (Isolation Condenser System)	Complies
IV	37	Testing of Emergency Core Cooling System	6.6	Preservice and Inservice Inspection of Class 2 and 3 Components	Complies
IV	37	Testing of Emergency Core Cooling System	LTR 5.2.18(1)	GDC 37	Complies
IV	37	Testing of Emergency Core Cooling System	LTR 4.1.13(2)	10 CFR 50 Appendix A, GDC 37	Complies
IV	38	Containment Heat Removal	6.2.2	Passive Containment Cooling System	Complies
IV	38	Containment Heat Removal	7.1	I&C Systems	N/A I&C systems not required because of the passive safety features for reactor and containment cooling
IV	38	Containment Heat Removal	8.1.3	Design Criteria, Regulatory Guides, and Standards	N/A Electrical power systems not required because of the passive safety features for reactor and containment cooling
IV	39	Inspection of Containment Heat Removal System	6.2.2	Passive Containment Cooling System	Complies

GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
IV	39	Inspection of Containment Heat Removal System	6.6	Preservice and Inservice Inspection of Class 2 and 3 Components	Complies
IV	40	Testing of Containment Heat Removal System	3.9.6	Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	Complies
IV	40	Testing of Containment Heat Removal System	6.2.2	Passive Containment Cooling System	Complies
IV	40	Testing of Containment Heat Removal System	6.6	Preservice and Inservice Inspection of Class 2 and 3 Components	Complies
IV	41	Containment Atmosphere Cleanup	6.1	Engineered Safety Features Materials	N/A Containment atmosphere cleanup is not an ESF function
IV	41	Containment Atmosphere Cleanup	6.2.5	Combustible Gas Control in Containment	N/A Containment atmosphere cleanup is not an ESF function
IV	41	Containment Atmosphere Cleanup	6.5.1	Engineered Safety Feature Atmosphere Cleanup Systems	N/A Containment atmosphere cleanup is not an ESF function
IV	41	Containment Atmosphere Cleanup	7.1	I&C Systems	N/A I&C systems not required because of the passive safety features for reactor and containment cooling

GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
IV	41	Containment Atmosphere Cleanup	8.1.3	Design Criteria, Regulatory Guides, and Standards	N/A Containment atmosphere is maintained inerted such that cleanup systems are not required
IV	41	Containment Atmosphere Cleanup	9.3.6	Containment Inerting System	Complies
IV	42	Containment Atmosphere Cleanup	6.5.1	Fission Product Removal and Control System	N/A Containment atmosphere cleanup is not an ESF function
IV	42	Inspection of Containment Atmosphere Cleanup Systems	6.6	Preservice and Inservice Inspection of Class 2 and 3 Components	Complies
IV	42	Inspection of Containment Atmosphere Cleanup Systems	9.3.6	Containment Inerting System	Complies
IV	43	Testing of Containment Atmosphere Cleanup Systems	3.9.6	Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	Complies
IV	43	Testing of Containment Atmosphere Cleanup Systems	6.5.1	Fission Product Removal and Control System	N/A
IV	43	Testing of Containment Atmosphere Cleanup Systems	6.6	Preservice and Inservice Inspection of Class 2 and 3 Components	Complies
IV	43	Testing of Containment Atmosphere Cleanup Systems	9.3.6	Containment Inerting System	Complies
IV	44	Cooling Water	6.2.2	Passive Containment Cooling System	Complies

GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
IV	44	Cooling Water	6.3	Emergency Core Cooling System (Isolation Condenser System)	Complies
IV	44	Cooling Water	7.1	I&C Systems	GDC 44 is not applicable to I&C systems because of the passive safety features for reactor and containment cooling
IV	44	Cooling Water	8.1.3	Design Criteria, Regulatory Guides, and Standards	N/A Electrical power systems not required because of the passive safety features for reactor and containment cooling
IV	44	Cooling Water	9.2.1	Service Water Subsystem	N/A Subsystem not required to provide cooling to any SC1 components
IV	44	Cooling Water	9.2.2	Plant Cooling Water System	N/A Subsystem not credited for heat removal during DBA
IV	44	Cooling Water	9.2.5	Ultimate Heat Sink	Complies
IV	44	Cooling Water	9.2.6	Condensate Storage and Transfer System	N/A No cooling functions provided by Subsystem
IV	44	Cooling Water	9.2.7	Chilled Water Equipment	N/A Subsystem not required to mitigate a DBA or credited for cooling SC1 equipment

GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
IV	44	Cooling Water	9.2.8	Isolation Condenser Pool Cooling and Cleanup System	N/A Cooling function of ICC not assumed for accident conditions.
IV	44	Cooling Water	9.2.9	Normal Heat Sink	N/A NHS function not relied upon in safety analysis.
IV	45	Inspection of Cooling Water System	6.2.2	Passive Containment Cooling System	Complies
IV	45	Inspection of Cooling Water System	6.3	Emergency Core Cooling System (Isolation Condenser System)	Complies
IV	45	Inspection of Cooling Water System	6.6	Preservice and Inservice Inspection of Class 2 and 3 Components	Complies
IV	45	Inspection of Cooling Water System	9.2.1	Service Water Subsystem	N/A Subsystem not required to provide cooling to any SC1 components
IV	45	Inspection of Cooling Water System	9.2.2	Plant Cooling Water System	N/A Subsystem not credited for heat removal during DBA
IV	45	Inspection of Cooling Water System	9.2.5	Ultimate Heat Sink	Complies
IV	45	Inspection of Cooling Water System	9.2.6	Condensate Storage and Transfer System	N/A No cooling functions provided by this system.
IV	45	Inspection of Cooling Water System	9.2.7	CWE System	N/A No SC1 equipment cooled by CWE.

GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
IV	45	Inspection of Cooling Water System	9.2.8	Isolation Condenser Pool Cooling and Cleanup System	N/A Cooling function of ICC not assumed for accident conditions.
IV	45	Inspection of Cooling Water System	9.2.9	Normal Heat Sink	N/A NHS not relied on in safety analysis.
IV	46	Testing of Cooling Water System	3.9.6	Functional Design, Qualification, and IST Programs for Pumps, Valves, and Dynamic Restraints	Complies
IV	46	Testing of Cooling Water System	6.2.2	Passive Containment Cooling System	Complies
IV	46	Testing of Cooling Water System	6.3	Emergency Core Cooling System (Isolation Condenser System)	Complies
IV	46	Testing of Cooling Water System	6.6.12	Preservice and Inservice Inspection of Class 2 and 3 Components	Complies
IV	46	Testing of Cooling Water System	9.2.1	Service Water Subsystem	N/A Subsystem not required to provide cooling to any SC1 components
IV	46	Testing of Cooling Water System	9.2.2	Plant Cooling Water System	N/A Subsystem not credited for heat removal during DBA
IV	46	Testing of Cooling Water System	9.2.5	Ultimate Heat Sink	Complies
IV	46	Testing of Cooling Water System	9.2.6	Condensate Storage and Transfer System	N/A No cooling functions provided by Subsystem

GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
IV	46	Testing of Cooling Water System	9.2.7	Chilled Water Equipment	N/A Subsystem not required to mitigate a DBA or credited for cooling SC1 equipment
IV	46	Testing of Cooling Water System	9.2.8	Isolation Condenser Pool Cooling and Cleanup System	N/A Cooling function of ICC not assumed for accident conditions
IV	46	Testing of Cooling Water System	9.2.9	Normal Heat Sink	N/A NHS function not relied upon in safety analyses
V	50	Containment Design Basis	3.8.1	Steel-Plate Composite Containment	Complies
V	50	Containment Design Basis	3.8.2	Class MC Steel Components of the BWRX-300 Containment	Complies
V	50	Containment Design Basis	3.8.3	Containment Internal Structures	Complies
V	50	Containment Design Basis	6.2	Containment System	Complies
V	50	Containment Design Basis	8.1.3	Design Criteria, Regulatory Guides, and Standards	Complies
V	50	Containment Design Basis	LTR 2.1.1.10(5)	10 CFR 50 Appendix A, GDC 50	Complies
V	51	Fracture Prevention of Containment Pressure Boundary	3.8.1	Steel-Plate Composite Containment Vessel	Complies
V	51	Fracture Prevention of Containment Pressure Boundary	3.8.2	Class MC Steel Components of the BWRX-300 Containment	Complies

GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
V	51	Fracture Prevention of Containment Pressure Boundary	6.2.1	Primary Containment System	Complies
V	51	Fracture Prevention of Containment Pressure Boundary	6.2.7	10 CFR 50 Appendix A, GDC 51	Complies
V	51	Fracture Prevention of Containment Pressure Boundary	LTR 2.1.1.11(5)	Steel-Plate Composite Containment Vessel	Complies
V	52	Capability for Containment Leakage Rate Testing	3.8.1	Steel-Plate Composite Containment Vessel	Complies
V	52	Capability for Containment Leakage Rate Testing	3.8.2	Class MC Steel Components of the BWRX-300 Containment	Complies
V	52	Capability for Containment Leakage Rate Testing	6.2.1	Primary Containment System	Complies
V	52	Capability for Containment Leakage Rate Testing	LTR 2.1.1.12(5)	10 CFR 50 Appendix A, GDC 52	Complies
V	53	Provisions for Containment Testing and Inspection	3.8.1	Steel-Plate Composite Containment Vessel	Complies
V	53	Provisions for Containment Testing and Inspection	3.8.2	Class MC Steel Components of the BWRX-300 Containment	Complies
V	53	Provisions for Containment Testing and Inspection	6.2.1	Primary Containment System	Complies
V	53	Provisions for Containment Testing and Inspection	LTR 2.1.1.13(5)	10 CFR 50 Appendix A, GDC 53	Complies
V	54	Piping Systems Penetrating Containment	3.9.6	Functional Design, Qualification, and IST Programs for Pumps, Valves, and Dynamic Restraints	Complies
V	54	Piping Systems Penetrating Containment	5.4.5	Main Steam and Feedwater Isolation	Complies
V	54	Piping Systems Penetrating Containment	5.4.8	Reactor Water Cleanup System	Complies

GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
V	54	Piping Systems Penetrating Containment	6.2.4	Containment Isolation Function	Complies
V	55	Reactor Coolant Pressure Boundary Penetrating Containment	3.6.1.1.1	Break Exclusion Zone	Complies
V	55	Reactor Coolant Pressure Boundary Penetrating Containment	5.4.5	Main Steam and Feedwater Isolation	Complies
V	55	Reactor Coolant Pressure Boundary Penetrating Containment	5.4.8	Reactor Water Cleanup System	Complies
V	55	Reactor Coolant Pressure Boundary Penetrating Containment	6.2.4	Containment Isolation Function	Complies
V	55	Reactor Coolant Pressure Boundary Penetrating Containment	15.5.9.1.4	Large Isolation Condenser Line Break Outside Containment	Complies
V	56	Primary Containment Isolation	6.2.4	Containment Isolation Function	Complies
V	57	Closed System Isolation Valves	6.2.4	Containment Isolation Function	Complies
VI	60	Control of Releases of Radioactive Materials to the Environment	5.4.8	Reactor Water Cleanup System	Complies
VI	60	Control of Releases of Radioactive Materials to the Environment	9.2.1	Service Water Subsystem	Complies
VI	60	Control of Releases of Radioactive Materials to the Environment	9.2.4	Potable Water Distribution and Sanitary Sewer Handling Subsystems	Complies

GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
VI	60	Control of Releases of Radioactive Materials to the Environment	9.2.6	Condensate Storage and Transfer System	Complies
VI	60	Control of Releases of Radioactive Materials to the Environment	9.2.10	Refueling Water Storage and Cleanup Subsystem	Complies
VI	60	Control of Releases of Radioactive Materials to the Environment	9.3.2	Process Sampling System	Complies
VI	60	Control of Releases of Radioactive Materials to the Environment	9.3.3	Equipment and Floor Drain System	Complies
VI	60	Control of Releases of Radioactive Materials to the Environment	9.4.1	Control Building Heating, Ventilation, and Cooling System	Complies
VI	60	Control of Releases of Radioactive Materials to the Environment	9.4.3	Radwaste Building Heating, Ventilation, and Cooling System	Complies
VI	60	Control of Releases of Radioactive Materials to the Environment	9.4.4	Turbine Building Heating, Ventilation, and Cooling System	Complies
VI	60	Control of Releases of Radioactive Materials to the Environment	9.4.6	Reactor Building Heating, Ventilation, and Cooling System	Complies
VI	60	Control of Releases of Radioactive Materials to the Environment	10.4.1	Main Condenser and Auxiliaries System	Complies
VI	60	Control of Releases of Radioactive Materials to the Environment	10.4.3	Turbine Gland Sealing System	Complies

GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
VI	60	Control of Releases of Radioactive Materials to the Environment	11.2	Liquid Waste Management Systems	Complies
VI	60	Control of Releases of Radioactive Materials to the Environment	11.3	Gaseous Waste Management Systems	Complies
VI	60	Control of Releases of Radioactive Materials to the Environment	11.4	Solid Waste Management System	Complies
VI	60	Control of Releases of Radioactive Materials to the Environment	11.5	Process Radiation Monitoring	Complies
VI	61	Fuel Storage and Handling and Radioactivity Control	5.4.8	Reactor Water Cleanup System	Complies
VI	61	Fuel Storage and Handling and Radioactivity Control	9.1.2	New and Spent Fuel Storage	Complies
VI	61	Fuel Storage and Handling and Radioactivity Control	9.1.3	Fuel Pool Cooling and Cleanup System	Complies
VI	61	Fuel Storage and Handling and Radioactivity Control	9.1.4	Light Load Handling System	Complies
VI	61	Fuel Storage and Handling and Radioactivity Control	9.4.6	Reactor Building Heating, Ventilation, and Cooling System	Complies
VI	61	Fuel Storage and Handling and Radioactivity Control	11.2	Liquid Waste Management Systems	Complies
VI	61	Fuel Storage and Handling and Radioactivity Control	11.3	Gaseous Waste Management Systems	Complies
VI	61	Fuel Storage and Handling and Radioactivity Control	11.4	Solid Waste Management System	Complies
VI	62	Prevention of Criticality in Fuel Storage and Handling	9.1.4	Light Load Handling System	Complies

GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
VI	62	Prevention of Criticality in Fuel Storage and Handling	9A	Criticality Safety of New and Spent Fuel Storage	Complies
VI	63	Monitoring Fuel and Waste Storage	7.1	I&C Systems	Complies
VI	63	Monitoring Fuel and Waste Storage	9.1.2	New and Spent Fuel Storage	Complies
VI	63	Monitoring Fuel and Waste Storage	9.1.3	Fuel Pool Cooling and Cleanup System	Complies
VI	63	Monitoring Fuel and Waste Storage	9.3.2	Process Sampling System	Complies
VI	63	Monitoring Fuel and Waste Storage	11.4	Solid Waste Management System	Complies
VI	63	Monitoring Fuel and Waste Storage	11.5	Process Radiation Monitoring	Complies
VI	63	Monitoring Fuel and Waste Storage	12.3.4.1	Area Radiation Monitoring	Complies
VI	64	Monitoring Radioactivity Releases	6.2.1	Safety Evaluation (PCS)	Complies
VI	64	Monitoring Radioactivity Releases	7.1	I&C Systems	Complies
VI	64	Monitoring Radioactivity Releases	9.2.1	Service Water Subsystem	Complies
VI	64	Monitoring Radioactivity Releases	9.2.9	Normal Heat Sink	Complies

GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
VI	64	Monitoring Radioactivity Releases	9.3.2	Process Sampling System	N/A PS provides online monitoring and grab sampling. Monitoring radioactive releases is not applicable to PS Subsystem
VI	64	Monitoring Radioactivity Releases	11.5	Process Radiation Monitoring	Complies

NOTES:

1. NEDC-33934P, Rev. 2, BWRX-300 Safety Strategy (Reference 3.1-1)
2. NEDC-33910P-A, Rev. 2, BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection (Reference 3.1-2)
3. Not Used.
4. NEDC-33912P-A, Rev. 2, BWRX-300 Reactivity Control (Reference 3.1-3)
5. NEDC-33926P-A, Rev. 4, BWRX-300 Steel-Plate Composite (SC) SCCV and RB Structural Design (Reference 3.1-5)

3.2 Classification of Structures, Systems, and Components

Section 4.0 of NEDC-33934P, "BWRX-300 Safety Strategy," (Reference 3.2-1), describes the approach to classifying SSCs as identifying functions that affect nuclear safety, assigning a safety category to those functions based on their importance, and then assigning a safety class to the components that perform those functions.

The primary factors influencing the outcome of SSC classification are:

- Consequences of failure to perform the functions
- Expected frequency of an SSC being called on to perform its functions
- Time following a postulated initiating event (PIE) at which, or the period for which, the SSC performs its functions

Safety importance of plant functions (i.e., defense line (DL) functions) is determined based on how functions are credited to mitigate PIEs and event sequences in the DSA. The organization of PIEs and event sequences is based on event categories, which reflect the frequency of occurrence of the PIE or event sequence from the PSA.

The BWRX-300 Safety Strategy defines the FSF that protect physical barriers. For a given event sequence, if the functional DLs required to fulfill the FSFs are performed successfully, then the corresponding physical barriers remain effective.

The FSFs for the BWRX-300 are as follows:

- Control of reactivity
- Removal of heat from the fuel (in the reactor, during fuel storage and including long-term heat removal)
- Confinement of radioactive material, shielding against radiation and control of planned radioactive releases, as well as limitation of accidental radioactive releases

A systematic approach is taken to identify functions that are necessary to fulfill the FSFs and to identify the inherent features that contribute to fulfilling or affecting the FSFs. The results of applying the systematic approach are in the fault list, which provides traceability between DL functions with a direct role in fulfilling an FSF, plant states and event sequences in which each function performs that role.

Subsection 3.2.1 through Subsection 3.2.5 below summarize each area of the approach to classifying SSCs:

- Assignment of Safety Category to Functions
- Assignment of Safety Class to Components
- Seismic Categorization
- Quality Group

3.2.1 Assignment of Safety Category to Functions

Section 4.2 of NEDC-33934P describes the process for assigning safety categories to functions. This process is illustrated in NEDC-33934P, Table 4-1: Functional Safety Category Assignment, and identifies the SSC functions that apply to each safety category for the following groups of functions:

- DL2 functions—Actively control key plant parameters associated with FSFs and detect and mitigate AOO PIEs
- DL3 functions—Detect and mitigate DBA PIEs and event sequences comprising AOO PIEs and failure of DL2 functions
- Defense Line 4a (DL4a) functions—Detect and mitigate DEC, including event sequences associated with some DBA PIEs and failure of DL3 functions
- Defense Line 4b (DL4b) Functions—Detect and mitigate DECs to prevent core damage or mitigate the consequences of core damage events (severe accidents)
- Normal Functions—Functions typically operating during normal plant operation
- Post-Accident Monitoring (PAM) functions - support monitoring and display of PAM variables

NEDC-33934P assigns each of these functions to a safety category based on importance: Safety Category 1, Safety Category 2, Safety Category 3, or Safety Category N.

3.2.2 Assignment of Safety Class to Components

Section 4.3 of NEDC-33934P provides a description of the process for assigning safety class to components. Safety class is assigned to components based on the safety category of the functions they perform as follows:

- Components that are required to perform Safety Category 1 functions are assigned to SC1
- Components that are required to perform Safety Category 2 functions are assigned to at least SC2
- Components that are required to perform Safety Category 3 functions are assigned to at least SC3
- Non-Safety Class (SCN) is assigned to all other components.

In addition, the following classifications are made for components and structures that perform FSFs but may not be explicitly defined as part of a DL function:

- Components that perform a FSF, whose failure is considered “practically eliminated,” are assigned to SC1. An example is the reactor pressure vessel
- Components that make up the fission product barriers (i.e., fuel cladding, RCPB, and containment) are assigned to SC1
- Components that are part of the RCPB, as defined in 10 CFR 50.2, are assigned to SC1
- Structures (excluding fuel handling equipment) are assigned a safety classification based on the highest safety classification of the components they house or support, excluding components whose failure, due to failure of the structure, results in fail-safe performance of the component’s safety category function(s)

NEDC-33934P, Table 5-1: Safety Class and Safety Category Mapping to USNRC Regulatory Requirements and Guidance, provides an overview of the assignment of safety class to components. Table 3A-1 identifies the safety classification of components.

3.2.3 Seismic Categorization

Seismic categorization of structures and components is based on the importance of functions performed, as described in Section 4.5 of NEDC-33934P. SC1 components that perform Safety

Category 1 functions are Seismic Category I, except that SC1 components not in a Seismic Category I structure whose failure results in fail-safe performance of the component's Safety Category 1 function are not required to be Seismic Category I. Structures and components that are significant contributors to PSA risk for seismic events are categorized as Seismic Category I. Structures and components that are not required to remain functional during or after a seismic event but whose failure during a seismic event could adversely affect the ability of any Seismic Category I structure or component to accomplish its Safety Category 1 function are classified as Seismic Category II. Structures and components for management and storage of radiological material that meet the criteria for RW-IIa in RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," are classified as Seismic Category "RW-IIa." Structures and components that fit none of these categories are categorized as Seismic Category "Non-Seismic." Table 3A-1 identifies the seismic categorization of components.

The seismic categorization approach described in Section 5.7 of NEDC-33934P is an alternative approach to RG 1.29, "Seismic Design Classification for Nuclear Power Plants," Position C.1. RG 1.29, Position C.1, provides a discrete list of SSCs expected to be Seismic Category I, while the BWRX-300 seismic categorization approach is based on the importance of functions being performed within the BWRX-300 design. The BWRX-300 design conforms with RG 1.29, Positions C.2, C.3 and C.4. This approach is consistent with requirements of GDC 2, 10 CFR 100, Appendix A, and 10 CFR 50, Appendix S.

In addition, the BWRX-300 seismic categorization conforms with to the following RG:

- RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants."

The BWRX-300 seismic design is discussed in Section 3.7. The seismic categories of structures are provided in Table 3A-2.

3.2.4 Quality Group

The Quality Group designation per RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," is used for establishing the appropriate codes and standards based on the importance of the pressure-retaining function of BWRX-300 components. Components are classified as Quality Group A, B, C, D, or not applicable. Table 3A-1 identifies the Quality Group of components.

3.2.5 References

3.2-1 NEDC-33934P, "BWRX-300 Safety Strategy," GE Vernova Hitachi Nuclear Energy Americas, LLC, Revision 2, March 2026.

Blue Energy Global Inc.

BE-POV-LCNRG-LIC-5009

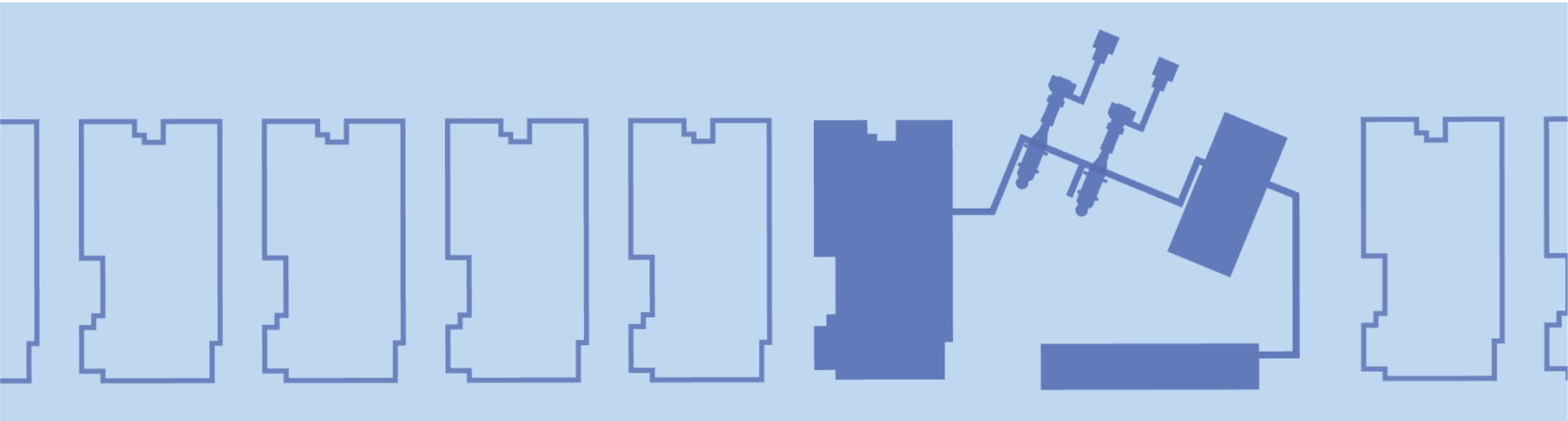
Revision 0

September 2026

Non-Proprietary Information

Chapter 3 Design of Structures, Components, Equipment, and Systems

Section 3.3 - Wind and Tornado Loadings



ACRONYM LIST

Acronym	Explanation
AISC	American Institute of Steel Construction
ANSI	American National Standard Institute
ASCE	American Society of Civil Engineers
CB	Control Building
GDC	General Design Criteria
GVH	GE Vernova Hitachi Nuclear Energy
MCR	Main Control Room
RB	Reactor Building
RG	Regulatory Guide
SC1	Safety Class 1
SCR	Secondary Control Room
SEI	Structural Engineering Institute
SRP	Standard Review Plan
SSCs	Structures, Systems and Components
USNRC	United States Nuclear Regulatory Commission

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None.

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None.

3.3 Wind and Tornado Loadings

The BWRX-300 Seismic Category I structures are designed to withstand the effects of extreme wind (tornado and hurricane) loadings in conformance with Title 10 of the Code of Federal Regulations (10 CFR) Part 50, Appendix A, General Design Criteria (GDC) 2, following the NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (also referred to as the Standard Review Plan or SRP) Subsections 3.3.1 and 3.3.2 (SRP 3.3.1 and 3.3.2). Overall compliance to GDC 2, for wind and tornado is provided in Subsection 3.1.1 and Table 3.1.1. The BWRX-300 design meets the GDC 2 requirements by following the applicable guidance provided in Regulatory Guide (RG) 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," RG 1.221, "Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants," and RG 1.76, "Design Basis Tornado and Tornado Missiles for Nuclear Power Plants," for the wind protection. Appropriate preliminary Safety Class 1 (SC1) Structures, Systems, and Components (SSCs) are listed in Table 3A-1.

3.3.1 Wind Loadings

As discussed in SRP 3.3.1, the design wind speed, its recurrence interval, the speed variation with height, and the applicable gust factors are used in defining the input parameters for the structural design criteria appropriate to account for wind loadings. The procedures that are utilized to transform the design wind speed into an effective pressure applied to structures take into consideration the geometrical configuration and physical characteristics of the structures and the distribution of wind pressure on the structures.

3.3.1.1 Design Wind Speed and Recurrence Interval

The design wind speed is defined as the (straight-line) 3-second gust wind speed based on appropriate standards, with suitable corrections for local conditions per SRP 2.3.1. Seismic Category I structures are designed for a basic wind speed value, that will be provided in CPA Part 2, based on American Society of Civil Engineers/Structural Engineering Institute (ASCE/SEI) 7-16, "Minimum Design Loads and Associated Criteria for Buildings and Other Structures," (Reference 3.3-1), Risk Category IV, basic wind speed maps (3-second gust), which correspond to a mean recurrence interval of 3,000 years.

The site-specific value remains below the standard plant design-basis wind speed for the Blue Energy Center – Port of Victoria Nuclear Unit 01 (BECPV-N01). This information will be provided in Construction Permit Application (CPA) Part 2. The basic wind speed used in the design of Seismic Category I structures, (to be reported in CPA Part 2) when adjusted for the ASCE/SEI 7-05 importance factor, bounds the most severe wind that has been historically reported for the site and surrounding area with sufficient margin for the limited accuracy, quantity, and period of time in which historical data has been accumulated.

3.3.1.2 Determination of Applied Forces

Design wind speeds for the Seismic Category I structures are translated into structural loading in accordance with the methodology specified in ASCE/SEI 7-16 (Reference 3.3-1). Appropriate load combinations (including wind) to be used in design are given in American National Standards Institute/American Institute of Steel Construction (ANSI/AISC) N690-18, "Specification for Safety-Related Steel Structures for Nuclear Facilities," (Reference 3.3-2). Procedures used are in conformance with Subsection II of SRP 3.3.1. However, SRP 3.3.1 refers to the ASCE/SEI 7-05 version. The BWRX-300 design uses the ASCE/SEI 7-16 (Reference 3.3-1) version. A comparison performed between these two ASCE/SEI versions concludes that the ASCE/SEI 7-16 (Reference 3.3-1) version bounds the ASCE/SEI 7-05 design parameters.

3.3.2 Extreme Wind (Tornado & Hurricane) Loadings

3.3.2.1 Applicable Design Parameters

The design input tornado and hurricane wind parameters applicable to the Seismic Category I structures will be provided in CPA Part 2 for the Blue Energy Center – Port of Victoria Nuclear Unit 01 (BECPV-N01). The tornado wind design parameters bound or are equivalent to the hurricane wind design parameters and will be confirmed to bound the wind that has been historically reported for the site and surrounding area with sufficient margin for the limited accuracy, quantity, and period of time in which historical data has been accumulated. Civil structures supporting and protecting SSCs for storage and processing of radioactive gas, liquid, and solid materials are classified in accordance with Regulatory Position 5 of RG 1.143. RW-IIa class SSCs are designed to withstand the tornado wind demand of Table 2 of RG 1.143. The Control Building (CB), including the Main Control Room (MCR) and egress route from the MCR to the Secondary Control Room (SCR) in the Reactor Building (RB), is designed to maintain its structural integrity under an extreme wind event to prevent incapacitating injury to MCR occupants or their egress to SCR in the RB, as discussed further in Subsection 3.8.4.4.

3.3.2.2 Determination of Forces on Structures

The procedures for transforming tornado wind speed into pressure-induced forces to apply to structures and the distribution across the structures are based on BC-TOP-3-A, “Tornado and Extreme Wind Design Criteria for Nuclear Power Plants,” (Reference 3.3-3) and follow the guidance of RG 1.76.

The procedure for transforming the tornado-generated missile impact into an effective or equivalent static load on structures is given in Subsection 3.5.3. The tornado missile spectrum is discussed in Subsection 3.5.1.4.

The individual tornado loading components and the load factors are in accordance with SRP 3.3.2/ BC-TOP-3-A and are selected to produce the most adverse total tornado effect on structures. The combined tornado effects are determined based on Equations 1 and 2 provided in SRP 3.3.2. The factor of 1.2 specified in BC-TOP-3-A, Sections 3.4 and 3.5.2 is applied to recognize potential non-conservative results for external walls when using the formulas specified in Appendix D of BC-TOP-3-A due to potential three-dimensional effects. As the formulas in Appendix D of BC-TOP-3-A are not used, the 1.2 factor is not applied for the BWRX-300 design.

SRP 3.3.2 invokes ASCE/SEI 7-05 for calculating maximum velocity pressure, q_z. After the values for coefficients, factors, and importance factor are substituted into the equation from SRP 3.3.2, the equation for maximum velocity pressure becomes $q_z = 0.00256V^2$, where V is the maximum tornado wind speed. This aligns with Equation 3-8 of BC-TOP-3-A. The size coefficient, C_s, from BC-TOP-3-A considers the load distribution length and the radius of maximum tornado wind. This size coefficient is not explicitly specified in ASCE/SEI 7-05. However, the values of external pressure coefficient, C_p, in Section 4.3.2 of BC-TOP-3-A, are similar to those provided for wall pressure coefficients in Figure 6-6 of ASCE/SEI 7-05. The approach in BC-TOP-3-A and ASCE 7-05 are similar. Therefore, BC-TOP-3-A is considered to be appropriate for use in the BWRX-300 design.

As discussed in Subsection 3.3.1.2, the BWRX-300 design uses the ASCE/SEI 7-16 version, which bounds the ASCE/SEI 7-05 design parameters.

Tornado loading considers tornado wind pressures, differential pressure loads due to rapid atmospheric pressure change, and tornado or hurricane-generated missile impact. For the purpose of structural analysis, the BWRX-300 RB is an enclosed (unvented) structure. The

exposed exterior roof and walls of the RB will be designed for the full pressure drop to be provided in Blue Energy Center - Port of Victoria Nuclear Unit 1 (BECPV-N01) CPA Part 2.

3.3.2.3 Effects of Failure of Structures or Components Not Designed for Tornado Loads

The Seismic Category II power block structures surrounding the RB are designed to maintain their structural integrity during an extreme wind event such that they do not collapse on the Seismic Category I RB.

Wind interaction evaluation follows the interaction requirements and guidance per the methodology described in Section 6.0 of NEDO-33914-A, “BWRX-300 Advanced Civil Construction and Design Approach,” (Reference 3.3-4).

3.3.3 References

3.3-1 ASCE/SEI 7-16, “Minimum Design Loads and Associated Criteria for Buildings and Other Structures,” American Society of Civil Engineers,” 2016.

3.3-2 ANSI/AISC N690-18, “Specification for Safety-Related Steel Structures for Nuclear Facilities,” American Institute of Steel Construction, 2018.

3.3-3 BC-TOP-3-A, “Tornado and Extreme Wind Design Criteria for Nuclear Power Plants,” Bechtel Power Corporation, Revision 3, August 1974.

3.3-4 NEDO-33914-A, “BWRX-300 Advanced Civil Construction and Design Approach,” Revision 2, GE Hitachi Nuclear Energy Americas, LLC, June 2022.

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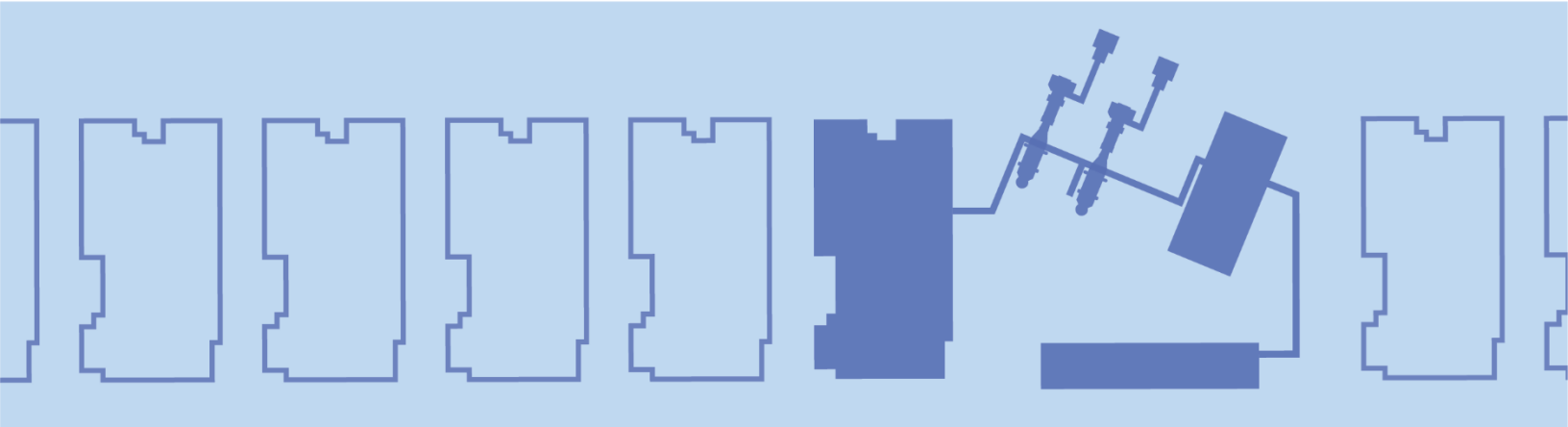
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Chapter 3 Design of Structures, Components, Equipment, and Systems

Section 3.4 - Water Level (Flood) Design



ACRONYM LIST

Acronym	Explanation
ANS	American Nuclear Society
ANSI	American National Standards Institute
BECPV-N01	Blue Energy Center Port of Victoria Nuclear Unit 1
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
PSA	Probabilistic Safety Assessment
RB	Reactor Building
SC1	Safety Class 1
SSCs	Structures, Systems, and Components

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3.4 Water Level (Flood) Design

External flooding hazards will be discussed in Chapter 2, which will be included in CPA Part 2.

Design of the plant flood protection includes all Structures, Systems, and Components (SSCs) whose failure could prevent safe shutdown of the plant or result in uncontrolled release of significant radioactivity. Overall compliance for BWRX-300 design to General Design Criteria (GDC) 2 and 4 for flood protection is provided in Subsection 3.1.1 and Table 3.1-1. A list of preliminary BWRX-300 primary component classifications, including safety class, location, seismic category, and quality group is provided in Table 3A-1. The building and site structure of the US seismic category and seismic design basis is provided in Table 3A-2. Appropriate preliminary Safety Class 1 (SC1) SSCs listed in Table 3A-1 are to be protected from internal flood. SSCs are also evaluated for risk effects in the internal flooding Probabilistic Safety Assessment (PSA) using the methods that will be described in Subsection 15.6.3.7.

Evaluation of external flooding from natural phenomena will be provided in Blue Energy Center Port of Victoria Nuclear Unit 1 (BECPV-N01) CPA Part 2. The analysis procedures that are used to transform the static and dynamic effects of the highest flood and groundwater levels into effective loads applied, are discussed in Subsection 3.4.2.

3.4.1 Internal Flood Protection for On-Site Equipment Failures

The internal flood design determines the effects of discharged fluid from leaks due to pipe cracks and breaks of fluid systems. The analysis method includes an outline of the approach to show the sources of flooding, and the potential target equipment due to potential flooding or spray. The review of building layout, elevations, and rooms determines the potential list of targets that require flood protection. In the buildings where there are large areas without dividing walls, the areas are subdivided to show any source of flooding and any target equipment in each of these areas. The design criteria for protection against the effects of compartment flooding follows guidelines provided in American National Standards Institute/American Nuclear Society, (ANSI/ANS) 56.11 “Design Criteria for Protection Against the Effects of Compartment Flooding in Light Water Reactor Plants,” (Reference 3.4-1) as applicable.

3.4.1.1 Internal Flooding Evaluation Methodology

The internal flood evaluation methodology considers the flow paths associated with potential internal floods, to improve the flood protection capability and to determine the additional loads on the structure (floor and walls) associated with a particular flood event. Fluid may be transferred from one area to another within the building through many pathways. Typical pathways that flood water could traverse include pipe conduits, drains, or openings in walls or floors, stairwells, vents, and elevators. Doors are also an important flood propagation pathway.

A structured approach is utilized to identify potential sources of flooding, affected equipment, model development, and evaluation criteria. Given the multitude of sources and equipment at risk, a series of evaluations are undertaken to identify the bounding scenarios. These limiting scenarios are then used to develop mathematical models, leading to a more comprehensive and detailed analysis.

Buildings that contain SC1 SSCs are evaluated for potential flooding or areas where spray can occur. Buildings that do not contain SC1 equipment, but house systems which could cause flooding that has the potential to reach SC1 equipment through connecting flow paths, are also evaluated. A review of each building layout, elevation, and rooms is completed. Identification of flood area consider hydraulic communication between rooms and buildings.

The piping failure criteria is provided in Section 3.6. The internal flood evaluation includes, but is not limited to the following examples of sources for flooding and spray:

- Pipe ruptures:
  - High-energy piping: breaks and cracks
  - Moderate energy piping: through-wall cracks in seismically supported piping
  - Moderate energy piping: breaks and through-wall cracks in non-seismically supported piping
- Tank ruptures
- Overfill of tanks
- Pump seals
- Fire suppression piping
- Fire sprinklers

The potential sources of water that can cause internal flooding are evaluated to determine possible local effects, such as water height and flow. All SC1 equipment are evaluated for possible flood impacts.

If the equipment is deemed to be inoperable, protective features are used to mitigate or eliminate the consequences of internal flooding. The mitigating strategy applied to the design is intended to ensure the equipment is compatible with the environmental conditions. If required, design features such as drainage paths, sumps, pumps, or any combinations are used to mitigate the effects of flooding. Possible mitigating strategies also include crediting operator actions, physical relocation, mitigating hardware, or environmental qualification (submergibility).

3.4.1.2 Internal Flooding Analysis

The bounding results from the internal flooding analysis, including the limiting flood sources, bounding flooding levels, location of SSCs subject to flooding, and required protective and mitigation features, are to be presented in the Final Safety Analysis Report (FSAR).

3.4.2 Analysis Procedures for Protection of Structures Against Flood from External Sources

The values for the maximum external flood and maximum groundwater level parameters will be provided in Blue Energy Center Port of Victoria Nuclear Unit 1 (BECPV-N01) CPA Part 2. CPA Part 2 will confirm that the maximum flood level and Local Intense Precipitation elevations do not exceed plant grade, so there are no external flooding protection features required in the BWRX-300 design, and no need for a permanent dewatering system, or CPA Part 2 will provide for the design of flood protection features and/or a permanent dewatering system, as needed. All exterior access openings and exterior penetrations for the Seismic Category I Reactor Building (RB) are above grade.

The RB, which is embedded partially below grade, is analyzed, and designed to withstand the effects of a maximum external flood defined at grade. Since the flood level is considered at the finished grade level, only hydrostatic effects are considered in the analysis and design of structures, while dynamic phenomena associated with a flooding event, such as currents, wind waves, and their hydrodynamic effects are not considered.

The BWRX-300 design considers the groundwater level at finished grade elevation. The hydrostatic pressure associated with the design flood level, or with the design groundwater level is considered as a structural load on the basemat and basement walls for structural design. Uplift or floating of structures is considered and the total buoyancy force is based on the hydrostatic pressure due to the design flood level, excluding wave action, or the design groundwater level.

The lateral, overturning and upward hydrostatic pressures acting on the side walls and on the foundation slab, respectively, are also considered in the structural design of these elements.

3.4.3 References

3.4-1 ANSI/ANS 56.11-1988, “Design Criteria for Protection Against the Effects of Compartment Flooding in Light Water Reactor Plants,” American National Standards Institute / American Nuclear Society, 1988.

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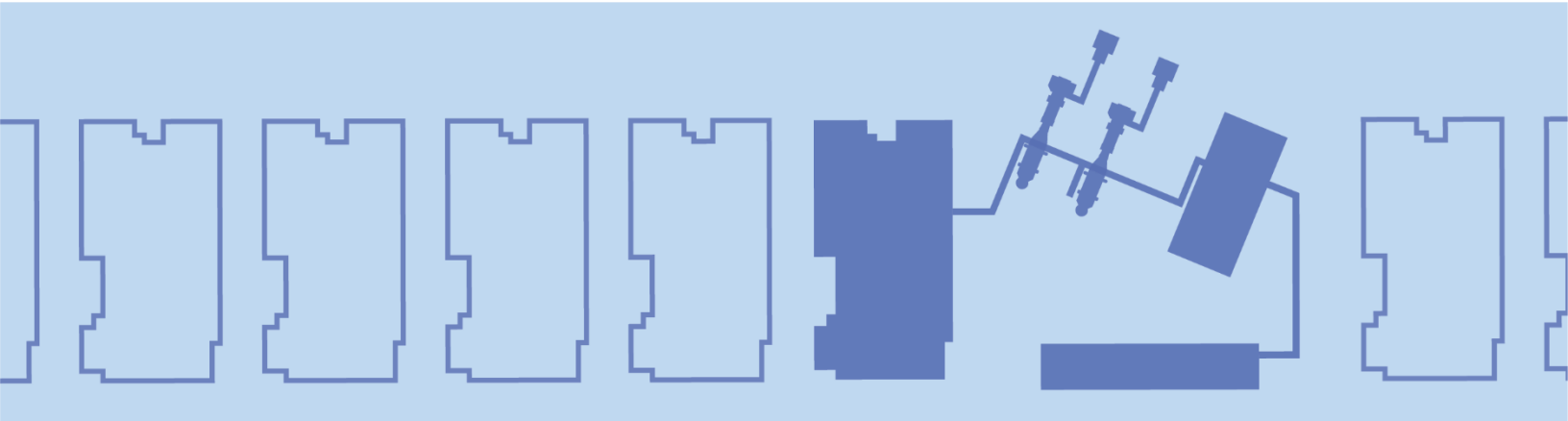
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Non-Proprietary Information

Chapter 3 Design of Structures, Components, Equipment, and Systems

Section 3.5 - Missile Protection



ACRONYM LIST

Acronym	Explanation
ACI	American Concrete Institute
AISC	American Institute of Steel Construction
ANS	American Nuclear Society
ANSI	American National Standards Institute
CB	Control Building
CFR	Code of Federal Regulations
DP-SC	Diaphragm Plate Steel-Plate Composite
GDC	General Design Criteria
MCR	Main Control Room
NS	Non-Seismic
RB	Reactor Building
RG	Regulatory Guide
RWB	Radwaste Building
SC1	Safety Class 1
SRP	Standard Review Plan
SSCs	Structures, Systems, and Components
TB	Turbine Building
USNRC	United States Nuclear Regulatory Commission

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3.5 Missile Protection

The missile protection design basis for Structures, Systems, and Components (SSCs) is described in this section. General arrangement drawings showing locations of the SSCs are presented in Subsection 1.2.3.

Missiles considered are those that could result from a plant-related failure or incident including failures within and outside of containment, environmental-generated missiles, and site proximity missiles. The structures, shields, and barriers that are designed to withstand missile effects, the possible missile loadings, and the procedures to which each barrier is designed to resist missile impacts are discussed in this section. Overall compliance to General Design Criteria (GDC) 2, 3, and 4 for the missile protection is provided in Subsection 3.1.1. The BWRX-300 design meets GDC 2, 3, and 4 requirements by following applicable guidance provided in Regulatory Guide (RG) 1.13, “Spent Fuel Storage Facility Design Basis,” RG 1.76, “Design Basis Tornado for Nuclear Power Plants,” RG 1.115, “Protection Against Low-Trajectory Turbine Missiles,” modified for a 3600 rpm turbine, RG 1.117, “Tornado Design Classification,” RG 1.143, “Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants,” and RG 1.221, “Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants,” for the missile protection.

Structures classified as Seismic Category I (e.g., Reactor Building (RB)), and structures classified as Seismic Category RW-IIa (e.g., Radwaste Building (RWB)) are designed to resist missiles, including missiles generated by natural phenomena. Seismic Category II and Non-Seismic (NS) structures are not required to be designed to resist missiles generated from tornado and extreme wind phenomena unless they have special tornado/hurricane storm hardening provisions. Appropriate preliminary Safety Class 1 SSCs listed in Table 3A-1 are designed to be protected from the missiles.

3.5.1 Missile Selection and Description

The missile protection criteria to which the plant is designed follows GDC 2 and 4 for external missiles and GDC 4 for internal missiles. SSCs are required to be protected from a wide spectrum of missiles. Missiles are classified as internal or external.

Internally generated missiles are those resulting from in-plant equipment component failures or other events within the nuclear island as defined in Nuclear Regulatory Report (NUREG)-0800, (also referred to as the Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition), Section 3.5.1.1, “Internally Generated Missiles (Outside Containment),” (SRP 3.5.1.1) and SRP 3.5.1.2, “Internally-Generated Missiles (Inside Containment).” Potential missile sources from rotating equipment, high-pressure system ruptures, and missiles caused by, or as a consequence of, gravitational effects (e.g., heavy load drop) are considered, when applicable.

Externally generated missiles are those that originate from an event of natural or human-induced origin that originates outside the site and whose effects on the reactor facility are considered as potentially hazardous. Examples include environmental-generated missiles and transportation accidents.

Potential missiles inside and outside containment and turbine missiles are identified, and their statistical significance is determined when applicable. A statistically significant missile is defined as a missile that could cause unacceptable plant consequences or exceedance of radiological release limits.

The examination of potential missiles and their consequences is done in the following manner to determine statistically significant missiles:

- If the probability of occurrence of the missile, P1, is determined to be less than 10^{-7} per year, the missile is dismissed from further consideration because at that likelihood it is considered not to be a statistically significant risk
- If P1 is found to be greater than 10^{-7} per year, it is examined for its probability of impacting a design target P2
- If the product of P1 and P2 is less than 10^{-7} per year, the missile is dismissed from further consideration
- If the product of P1 and P2 is greater than 10^{-7} per year, the missile is examined for its damage probability P3. If the combined probability (i.e., P1 x P2 x P3 = P4) is less than 10^{-7} per year, the missile is dismissed.
- Finally, measures are taken to design acceptable protection against missiles with P4 greater than 10^{-7} per year to reduce P1, P2, and/or P3, so that P4 is less than 10^{-7} per year

Preventative and mitigative measures considered in the BWRX-300 design against missiles include the following:

- Location of the system or component in an individual missile-proof structure
- Physical separation of redundant systems or components of the system from the missile trajectory path or calculated range
- Provision of localized protection shields or barriers for systems or components
- Design of the particular structure or component to withstand the impact of the most damaging missile
- Provision of design features on the potential missile source to prevent missile generation
- Orientation of the potential missile source to prevent unacceptable consequences caused by missile generation

The following criteria are considered as an acceptable design basis to withstand the statistically significant missiles:

- No loss of containment function as a result of missiles generated internal to containment
- Reasonable assurance that a safe plant shutdown condition can be achieved and maintained
- The failure of non-safety class equipment, components, or structures, whose failure could result in a missile, do not cause failure of nuclear safety class equipment
- No high energy lines are located near the off-gas charcoal bed adsorbers (located in the RWB)

In some cases, structural walls and slabs are credited for missile-consequence mitigation from missiles generated by rotating and pressurized components. These walls and slabs are designed to withstand internal missile effects in order to preclude perforation by internally generated missiles.

3.5.1.1 Internally Generated Missiles (Outside Containment)

SSCs (outside containment) are protected from internally generated missiles in accordance with the requirements of GDC 4.

Potential missile sources from rotating equipment or high-pressure system ruptures, and missiles caused by, or as a consequence of, gravitational effects (e.g., heavy load drop) are considered in the design, where applicable.

Equipment within the general categories of pumps, fans, blowers, diesel generators, compressors, and components in systems normally functioning during power reactor operation, are examined for possible sources of credible and significant missiles.

Potential missiles that result from the failure of pressurized components are categorized as contained fluid energy missiles or stored energy (elastic) missiles.

Examples of potential contained fluid energy missiles are valve bonnets, valve stems, and retaining bolts. Valve bonnets are considered jet-propelled missiles. Valve stems are analyzed as piston-type missiles, while retaining bolts are examples of stored strain energy missiles.

Pressurized gas bottles are to be placed with their operating orientation vertical and their ends facing concrete slabs. The tops of the bottles are protected with steel covers.

3.5.1.2 Internally Generated Missiles (Inside Containment)

SSCs (inside containment) are protected from internally generated missiles to ensure support to the reactor facility and conformance with the requirements of GDC 4. As discussed in Subsection 3.5.1.1, potential missile sources from both rotating equipment and pressurized components are considered, when applicable. Gravitational missiles inside the containment are also considered.

3.5.1.3 Turbine Missiles

Essential SSCs in RG 1.115 Appendix A required for mitigation of the effects of a turbine failure event for the BWRX-300 design are protected from the effects of turbine missiles to ensure:

1. The integrity of the reactor coolant pressure boundary.
2. The plant's capability to shut down the reactor and maintain it in a safe shutdown condition.
3. The plant's capability to prevent accidents which could result in potential offsite exposures comparable to those referred to in Title 10 of the Code of Federal Regulations (10 CFR) 50.34(a)(1), 50.67(b)(2), or 100.11, as applicable.

This protection is provided by favorable orientation of the turbine combined with quality assurance in design and fabrication, maintenance, inspection programs, and overspeed protection systems, as described in Chapter 10; physical protection via barriers; or analysis to show that the probability of failure of an essential SSC because of turbine missiles (P_4) is less than 10^{-7} per year.

General arrangement drawings illustrating major structures are provided in Figures 1.2-3 through 1.2-15. The turbine generator orientation is favorable per RG 1.115 based on the 25-degree strike zone, which renders the hazard rate for low trajectory missiles acceptably low and excludes them from further consideration. Any changes to the strike zone for a 3600-rpm turbine will be determined after selection of a turbine vendor, but there is sufficient margin as shown in the Chapter 1 general arrangement drawings to conclude that the turbine orientation is favorable.

High trajectory missiles are considered in accordance with Table 1 of RG 1.115. The turbine missile characteristics, trajectory paths, and probabilities are specific to each turbine vendor and

turbine type, and the P_1 through P_3 probabilities are subject to change for a 3600-rpm turbine. Therefore, detailed turbine missile evaluations are provided after selection of a turbine vendor.

3.5.1.4 Missiles Generated by Extreme Winds

The possible environmental hazards are due to missiles generated by the design basis tornado and hurricane.

The design for missiles generated by tornadoes and other extreme winds meets the requirements of GDC 2 and 4. The methodology used for designing BWRX-300 complies with RG 1.76 for tornado missiles, RG 1.221 for hurricane missiles, and RG 1.143 missile spectrum for the RWB with modifications to reconcile with more recent guidance in American National Standards Institute/American Nuclear Society (ANSI/ANS)-2.3, “Estimating Tornado, Hurricane, and Extreme Straight Line Wind Characteristics at Nuclear Facility Sites,” (Reference 3.5-1) consistent with RG 1.76.

Design for missile impact includes the bounding extreme winds (tornado or hurricane) and the corresponding wind-generated missiles as listed in Table 3.5-1. The design basis tornado and hurricane missile spectrum shown in Table 3.5-1 is the most limiting natural phenomena hazard in the design of missile barriers and structures. This spectrum is considered to envelope missiles generated by less intense natural phenomena.

RG 1.76 guidelines are followed for design basis missiles for nuclear power plants which includes at least:

1. A massive high-kinetic-energy missile that deforms on impact.
2. A rigid missile that tests penetration resistance.
3. A small rigid missile of a size sufficient to pass through any opening in protective barriers.

Non-storm-resistant building superstructures are constructed from materials such as reinforced concrete block, or structural steel with metal siding and roof deck. Potential missiles or debris from these materials, resulting from failure of the superstructure or from items blown off, when subjected to winds of tornado or hurricane intensity, are considered to be less severe than the missiles shown in Table 3.5-1.

The site-specific value for the hurricane wind speed will be provided in CPA Part 2 and will be confirmed to be bounded by the design wind speed. The design hurricane wind speed used for determination of hurricane missile velocities for the BWRX-300 is 89.4 m/s (200 mph).

3.5.1.5 Site Proximity Missiles (Except Aircraft)

CPA Part 2 will discuss the evaluations performed on potential accidents that could generate missiles (i.e., explosions) and confirm the hazards do not pose a threat to Blue Energy Center Port of Victoria Nuclear Unit 01 (BECPV-N01).

3.5.1.6 Aircraft Hazards

In accordance with the acceptance criteria in NUREG-0800 SRP 3.5.1.6, “Aircraft Hazards,” when the probability of aircraft accidents resulting in radiological consequences greater than the 10 CFR 100, “Reactor Site Criteria” exposure guidelines is greater than an order of magnitude of 10^{-7} per year, a detailed review of aircraft hazards is to be performed. The results of this review will be provided in CPA Part 2.

3.5.2 Structures, Systems, and Components to be Protected from Externally Generated Missiles

This subsection discusses the SSCs to be protected from externally generated missiles that are required to safely shut down the reactor and maintain it in a safe condition. SSCs that are required to be protected from externally generated missiles are located in a structure that is designed as storm (tornado and hurricane) resistant or other provisions are made to protect them. Seismic Category I and RW-IIa structures, and portions of the CB structure, defined by the different systems requirements, are designed to withstand the effects of externally generated missiles.

The design for missiles generated by extreme winds conforms with the Regulatory Positions C1 and C2 of RG 1.76, Regulatory Positions C1 and C2 of RG 1.221, Regulatory Positions C1, C2, and C3 of RG 1.117, and the Regulatory Position C2 of RG 1.13. SSCs to be protected from externally generated missiles are designed to meet the SRP 3.5.2, “Structures, Systems, and Components to be Protected from Externally-Generated Missiles,” acceptance criteria.

For the RWB, design for missiles is in accordance with RG 1.143.

Main Control Room (MCR) and MCR to secondary control room egress route are hardened to meet requirements of RG 1.117 and to provide a qualified route for egress.

3.5.3 Barrier Design Procedures

The procedures by which structures and barriers are designed to resist the missiles described in Subsection 3.5.1 are presented in this section. The following procedures are in accordance with NUREG-0800 SRP 3.5.3, “Barrier Design Procedures” and ensure that the design of structures, shields, and barriers that need to withstand the effects of environmental and natural phenomena meet the relevant requirements of GDC 2 and GDC 4.

3.5.3.1 Local Damage Prediction

The prediction of local damage in the impact area depends on missile characteristics, the basic material of construction of the structure or barrier (i.e., reinforced concrete, steel, or steel-plate composite) or Diaphragm Plate Steel-Plate Composite (DP-SC)), and the structural response. It is assumed that the missile strikes the target normal to the surface, and the axis of the missile is assumed parallel to the line of flight. These assumptions result in a conservative estimation of local damage.

Explicit dynamic inelastic analysis may be used, in lieu of the simplified formulas, to predict the local damage modes associated with the penetration of a missile into the barrier and the minimum required thickness.

3.5.3.1.1 Concrete Structures and Barriers

Local damage modes resulting from hard missile impact to concrete structures are presented in Figure 3.5-1 adopted from R.P. Kennedy, "A Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effect," (Reference 3.5-2) as explained below:

- Missiles with extremely low velocity can strike the target and bounce off without creating any local damage
- As the missile velocity increases, the concrete surface is spalled. The spall area extends larger than the missile cross-sectional area as shown in Figure 3.5-1 (a).
- As the velocity increases, the missile can penetrate the target as shown in Figure 3.5-1 (a)
- Further increase in the missile velocity can result in cracking the back surface which may result in scabbing (scabbing area is much wider than that of spalling but less deep) as shown in Figure 3.5-1 (b)
- Further increase in the missile velocity can result in perforation of the target as shown in Figure 3.5-1 (c)
- Finally, with increased velocity the missile can exit from the rear side of the target

To prevent perforation, criteria in Appendix F, Section F.7 of ACI 349 "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary," (Reference 3.5-3) is satisfied as listed below.

1. When perforation of concrete structural elements needs to be precluded, the concrete thickness is to be at least 20% greater than that required to prevent perforation.
2. Concrete structural elements protecting required systems or equipment which could be damaged by secondary missiles (fragments of scabbed concrete) are designed to prevent scabbing, or a properly designed scab shield is based on applicable formulas or pertinent test data. In the absence of scab shields, the concrete thickness is to be at least 20% greater than that required to prevent scabbing.
3. When it can be demonstrated by applicable formulas or pertinent test data that the concrete thickness is at least 20% greater than that required to prevent perforation and hence punching shear failure, design for punching shear in accordance with Section F.5 of ACI 349 (Reference 3.5-3) is not required.
4. For concrete slabs or walls subjected to missile impact effects where the concrete thickness is less than twice that required to prevent perforation, the minimum percentage of reinforcement is to be 0.2% each way, each face.

Empirical equations provided in, "A Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effects," are used to determine the required barrier thicknesses. Resulting thickness cannot be less than those listed in Table 3.5-2, which specifies the minimum thicknesses necessary to protect against tornado missiles.

3.5.3.1.2 Steel Structures and Barriers

Perforation is the local damage mode considered for a hard missile impacting steel plate. The minimum plate thickness to prevent perforation is obtained by multiplying 1.25 by the more conservative thickness of the formula described in C. R. Russel, "Reactor Safeguards,"

(Reference 3.5-4) and the Stanford equation in ORNL-NSIC-5, "U.S. Reactor Containment Technology," (Reference 3.5-5) as recommended by Bechtel Topical Report BC-TOP-9A, "Design of Structures for Missile Impact," (Reference 3.5-6).

3.5.3.1.3 Steel-Plate Composite Barriers

Steel-plate composite barriers when utilized in the BWRX-300 for missile protection are designed in accordance with Section 5.8.2 of NEDC-33926P-A (Reference 3.5-7) to ensure perforation does not occur. Local damage modes resulting from hard missile impact to steel-plate composite structures are presented in Figure 3.5-2. Scabbing is not a design limit state because it is prevented by the rear steel faceplates of the steel-plate composite structure as discussed in Section 5.8.2.2 of NEDC-33926P-A.

3.5.3.2 Overall Damage Prediction

The overall response of a structure or barrier to missile impact depends upon the location of impact (e.g., near mid-span or near a support), dynamic properties of the structure/barrier and missile, and on the kinetic energy of the missile.

The overall response of a structure subjected to impactive load is determined by one of the following methods:

1. Energy Balance Methods: Overall response can be assessed by using energy balance techniques to demonstrate that the calculated value of the required ductility ratio, μ_r, is less than or equal to the applicable value of the permissible ductility ratio, μ_p:

    a. Williamson and Alvy "Impact Effect of Fragments Striking Structural Elements," (Reference 3.5-8): An acceptable procedure for such an analysis, where the impact is assumed to be plastic, is presented in Reference 3.5-8. In this reference, the deformability of the missile is not considered.

    b. BC-TOP-9A (Reference 3.5-6): When a missile strikes a target, large forces develop at the missile target interface. If the interface forcing function is known (e.g., frontal impact of an automobile), the target structure can be modeled to predict the structural response. In this method, the deformability of the missile is accounted for in the forcing function. The overall structural response of a target is determined by equating the available target strain energy to the required strain energy to stop a missile using the analysis procedure provided in BC-TOP-9A.

2. Nonlinear Time-History Dynamic Analysis Method: The dynamic effects of impactive loads can be considered by performing a time-history dynamic analysis. The mass and inertial properties are included, as well as the nonlinear stiffnesses of the structural members under consideration.

3.5.3.3 Dynamic Increase Factors

Dynamic increase factors are based on the material strain rate effects. Increase factors are applied to static material strengths of steel, concrete, and steel-plate composite constituents for purposes of determining section strength but not to exceed those specified in Table 3.5-3.

The dynamic increase factors are limited to 1.0 for all materials where the dynamic load factor associated with the impactive loading is less than 1.2 (ACI 349, Clause F.2.1 (Reference 3.5-3) and American National Standards Institute/American Institute of Steel Construction (ANSI/AISC) N690-18, Clause Appendix N9.1.6a, "Specification for Safety-Related Steel Structures for Nuclear Facilities," (Reference 3.5-9).

3.5.3.4 Required Ductility Ratio

Required ductility is calculated by dividing the maximum target displacement resulting from the missile impact by the component yield displacement. Maximum target displacement is estimated from the applied impact load combined with other applicable loads using one of the energy balance methods described in Williamson and Alvy (Reference 3.5-8) and Energy Balance Method, BC-TOP-9A (Reference 3.5-6). The component adequacy for the resulting inelastic deformation is determined by comparing the calculated required ductility ratio (μ_r) with the applicable permissible ductility ratio (μ_p) provided in Table 3.5-4 for steel and reinforced concrete and Table 3.5-5a for steel plate composite. Table 3.5-5a and Table 3.5-5b provide structural acceptance criteria for load categories that include impactive and impulsive loads per NEDC-33926P-A (Reference 3.5-7), including the modifications required by Limitation and Condition 8.7.

3.5.4 References

3.5-1 ANSI/ANS-2.3-2011, “Estimating Tornado, Hurricane, and Extreme Straight Line Wind Characteristics at Nuclear Facility Sites,” American National Standards Institute/American Nuclear Society, 2011 (reaffirmed 2016).

3.5-2 R. P. Kennedy, “A Review of Procedures for the Analysis and Design of Concrete Structures to Resist Missile Impact Effects,” Nuclear Engineering and Design, Volume 37, Issue 2, May 1976.

3.5-3 ACI 349-13: “Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary,” American Concrete Institute, 2014.

3.5-4 C. R. Russel, “Reactor Safeguards,” MacMillan, New York, 1962.

3.5-5 ORNL-NSIC-5, “U.S. Reactor Containment Technology: A Compilation of Current Practice in Analysis, Design, Construction, Test, and Operation,” Volume 1, Oak Ridge National Laboratory, Chapter 6, 1965.

3.5-6 BC-TOP-9A, Topical Report, “Design of Structures for Missile Impact,” Bechtel Power Corporation, Revision 2, September 1974.

3.5-7 NEDC-33926P-A, “BWRX-300 Steel-Plate Composite (SC) Containment Vessel (SCCV) and Reactor Building Structural Design,” GE Vernova Hitachi Nuclear Energy Americas, LLC, Revision 4, December 2025.

3.5-8 Williamson, R.A. and Alvy, R.R, “Impact Effect of Fragments Striking Structural Elements,” Holmes and Narver, 1973.

3.5-9 ANSI/AISC N690-18: “Specification for Safety-Related Steel Structures for Nuclear Facilities,” American Institute of Steel Construction, 2018.

3.5-10 ASCE/SEI 59-11, “Blast Protection of Buildings,” American Society of Civil Engineers, 2011.

3.5-11 IAEA SR No. 87, “Safety Aspects of Nuclear Power Plants in Human Induced External Events: Assessment of Structures,” International Atomic Energy Agency, 2018.

Table 3.5-1: Design Extreme Wind-Generated Missile Spectrum

To be provided in CPA Part 2.

Table 3.5-2: Minimum Acceptable Barrier Thickness Requirements for Local Damage Protection Against Tornado Generated Missiles

Concrete Strength MPa (Psi)	Wall Thickness mm (in)	Roof Thickness mm (in)
20.7 (3000)	462 (18.2)	335 (13.2)
27.6 (4000)	429 (16.9)	312 (12.3)
34.5 (5000)	406 (16.0)	297 (11.7)

Table 3.5-3: Dynamic Increase Factor

Material	Dynamic Increase Factor	
	Yield Strength	Ultimate Strength
Reinforced Concrete		
Concrete Axial and Flexural Compressive Strength	–	1.25
Concrete Shear Strength	–	1.10
Diagonal tension and direct shear for reinforcing steel (stirrups)	–	1.0
Diagonal tension and direct shear (punchout) and bond for reinforced concrete	–	1.0
Reinforcing Steel Grade 280 MPa (40 ksi)	1.2	1.05
Reinforcing Steel Grade 350 MPa (50 ksi)	1.15	1.05
Reinforcing Steel Grade 420 to 550 MPa (60 – 80 ksi)	1.10	1.05
Structural Steel		
Carbon Steel Plate	1.29	1.10
Stainless Steel Plate	1.18	1.0

Table 3.5-4: Permissible Ductility Ratios-Steel and Reinforced Concrete

Barrier Material	Limit State	Permissible Ductility Ratio (μ_p)	Reference
Steel Structures Barriers	Tension(1)	≤ min (0.25 $\varepsilon_u/\varepsilon_y$, $0.1/\varepsilon_y$)	ANSI/AISC N690, "Specification for Safety-Related Steel Structures for Nuclear Facilities," Table NB3.1 (Reference 3.5-9)
	Flexure(1)(2)–Steel plates	≤ 20	
	Flexure(1)(2)–Open sections (W, S, WT)	≤ 10	
	Flexure(1)(2)–Closed sections (pipe, box section)	≤ 20	
	Members where shear(1)(2) governs the design	≤ 5	
	Compression, applicable when $F_e \geq 4.5F_y$	≤ min $(\frac{0.225}{F_y/F_e}, \frac{\varepsilon_{st}}{\varepsilon_y}, 10)$	
Reinforced Concrete Structures	Flexure(3)–Steel beams, walls, and slabs	min $(\frac{0.05}{(\rho-\dot{\rho})}, 10)$ Rotational capacity r_θ in radians of any yield hinge is limited to $(0.0065\ d/c)$ but not to exceed 0.07 radians	ACI 349 Clause F.3.3 and F3.4 (Reference 3.5-3)
	Shear-controlled beams, walls, and slabs: shear is resisted by concrete alone	≤ 1.0	ACI 349 Clause F.3.7 (Reference 3.5-3) as modified by RG 1.142
	Shear-controlled beams, walls, and slabs: shear is resisted by concrete and stirrups, headed bars, or bent bars	≤ 1.3	
	Shear-controlled beams, walls, and slabs: shear is resisted by stirrups alone	≤ 3.0	
	Compression and Flexural produced from impactive loads Case 1: When compression controls the design, as defined by a load-moment interaction diagram	≤ 1.0	ACI 349 Clause F.3.8 (Reference 3.5-3)

Barrier Material	Limit State	Permissible Ductility Ratio (μ_p)	Reference
	Compression and Flexural produced from impactive loads Case 2: for beams columns, walls, and slabs When the compression load does not exceed $0.1f_c'A_g$ or one-third of that which would produce balanced strain conditions, whichever is smaller	$\min(\frac{0.05}{(\rho - \dot{\rho})}, 10)$ Rotational capacity r_θ in radians of any yield hinge is limited to $(0.0065\, d/c)$ but not to exceed 0.07 radians.	
	The permissible ductility ratio varies linearly for conditions between those specified in Case 1 and Case 2.		
	Axial compressive load	≤ 1.3	ACI 349 Clause F.3.9 (Reference 3.5-3)

Notes:

(1) For net sections with ductile behavior, the plastic resistance is based on yielding of the net section. For net sections with either brittle or limited ductile behavior, the member's plastic resistance is based on yielding of the gross section provided that the net section's tensile rupture based available strength exceeds its gross section's yielding based available strength.

(2) Accompanying compression force, if any, is less than the smaller of $0.1F_eA_g$ and $0.1F_yA_g$.

(3) For flexure to control the design, the shear strength of a structural member is at least 20% greater than the flexural strength; otherwise, the ductility ratios given for shear-controlled limit state are to be used.

Where:

ε_u: strain corresponding to elongation at failure (rupture) using the value corresponding to an 8-in. (200 mm) long specimen.
ε_y: strain corresponding to nominal yield stress = F_y/E
F_y: specified minimum yield stress, (ksi)
E: Modulus of elasticity 29,000 ksi (200,000 MPa) for carbon steel
ε_{st}: strain corresponding to the onset of strain hardening
F_e: elastic buckling stress determined according to $F_e = \frac{\pi^2 E}{(L_c/r)^2}$, (ksi)
r: radius of gyration, (in)
L_c: Effective length of member for buckling about the minor axis, (in)
f_c': Concrete compressive strength (psi)
A_g: Gross cross-sectional Area (in^2)

Table 3.5-5a: Structural Acceptance Criteria for Flexure in Terms of Support Rotations and for Shear in Terms of Ductility[10][11]

Element Type	Controlling Behavior [1] [5]	Superficial Damage	Limited, Moderate, Severe Damage	Limited Damage	Moderate Damage	Severe Damage
			Ductility μ_d	Support Rotation r_θ (deg)[2] [6]	Support Rotation r_θ (deg)[2] [3] [7]	Support Rotation r_θ (deg)[3] [7]
SC Slabs and Walls (including DP-SC)	Out-of-Plane Flexure	Essentially Elastic Behavior [4]	10[6]	1	4	6
	Out-of-Plane Shear:					
	• Ties or Diaphragms spaced at no more than ½ t_{sc}		1.3	N/A	N/A	N/A
	• Ties or Diaphragms spaced at more than ½ t_{sc}		1.0	N/A	N/A	N/A
	Compression		1.0[8]	N/A	N/A	N/A

Element Type	Controlling Behavior (1) (5)	Superficial Damage	Limited, Moderate, Severe Damage	Limited Damage	Moderate Damage	Severe Damage
			Ductility μ_d	Support Rotation r_θ (deg)(2) (6)	Support Rotation r_θ (deg)(2) (3) (7)	Support Rotation r_θ (deg)(3) (7)
SC Shear Walls, and Diaphragms (including DP-SC) (9)	In-Plane Flexure (shear walls and diaphragms)	Essentially Elastic Behavior (4)	N/A	-	1.5	2
	In-Plane Shear (shear walls)		1.5	N/A	N/A	N/A
	In-Plane Shear (diaphragms)		1.5	N/A	N/A	N/A

Notes:

(1) SC components (including DP-SC) shall be classified as flexure-controlled if their available strength for the limit state of flexural yielding is less than their available strength for the limit state of out-of-plane shear failure by at least 25%.

(2) When flexure controls, the criteria in terms of support rotations, from Table 3.5-5a, and in terms of strains, from Table 3.5-5b, are fulfilled simultaneously in order to control damage.

(3) These rotation criteria (in degrees) are in general consistent with those in the ASCE/SEI 59-11 (Reference 3.5-10), which does not specify allowable inelastic deformation in terms of ductility ratio-criteria for flexure.

(4) Essentially elastic behavior means elastic structural analysis using design strain acceptance criteria of 1% for steel plates and 0.35% for concrete in compression, which corresponds to superficial damage of elements. The permissible ductility ratio μd is 1.0.

(5) When impact/impulse loading results in net tension, shear capacity of concrete is not considered.

(6) 1 degree support rotation is related to buildings with internal explosions producing internal blast pressures or chamber pressurization. This is a global structural response and is similar to a structural drift criterion that governs the entire structure's integrity. This is a higher damage level than the essentially elastic response threshold defined for superficial damage. The limit of ductility of 3 may be used in lieu of support rotations of 1 degree complying with Regulatory Position C11.1.6 of United States Nuclear Regulatory Commission (USNRC) RG 1.243.

(7) This is a semi-global response criterion (i.e., for a wall or slab or part of the structure). The collapse of this structural member does not lead to the collapse of the entire structure.

(8) For additional information refer to Regulatory Position C11.1.5 of USNRC RG 1.243.

(9) Per the acceptance criteria of IAEA SR No. 87 (Reference 3.5-11).

(10) The following limitations are placed on the acceptance criteria for DP-SC containment under DBAs: For normal and severe environmental load categories that include impactive/impulsive loads, the acceptance criteria is limited to superficial damage in Table 3.5-5a and Table 3.5-5b; and, for abnormal, extreme environmental, and abnormal and extreme environmental load categories that include impactive/impulsive loads, the allowable limits for ductility, support rotation, and strain shall not exceed those for limited damage in Table 3.5-5a and Table 3.5-5b.

(11) For DP-SC non-containment, under abnormal and extreme environmental load categories that include impactive and impulsive loads, the limits for moderate damage criteria in Table 3.5-5a and Table 3.5-5b are acceptable for DBAs as listed in the design load combinations given in RG 1.243. Those limits are per ANSI/AISC N690 (Reference 3.5-8) as endorsed and modified by Regulatory Position C11.1 of RG 1.243.

Table 3.5-5b: Structural Acceptance Criteria – Allowable Strains for Steel

Structural Acceptance Criteria - Allowable Strains for Steel[2][3]				
Material	**Strain Measure**	**Superficial Damage**	**Limited Damage**	**Moderate Damage**
Carbon Steel Plate	Membrane principal strain (tension)	0.010	0.025	0.050
	Local ductile tearing effective strain	N/A	0.070/*TF* [1]	0.140 / TF [1]
304 Stainless Steel Plate	Membrane principal strain (tension)	0.010	0.033	0.067
	Local ductile tearing effective strain	N/A	0.138/*TF* [1]	0.275 / TF [1]

Notes:

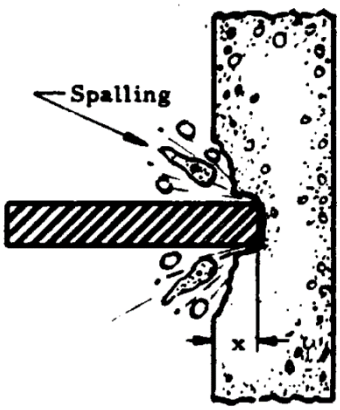
(1) The tri-axiality factor, TF, is defined as

$$TF = \frac{\sigma_1 + \sigma_2 + \sigma_3}{\sigma_e}$$

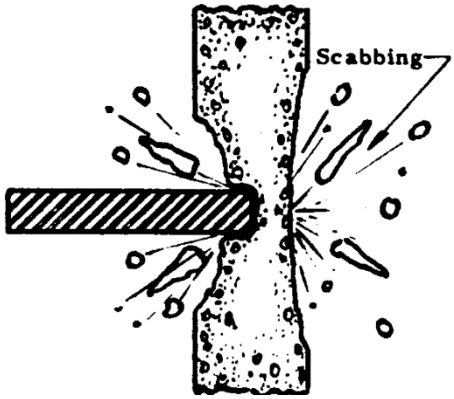
Where σ_1, σ_2 and σ_3 are principal stresses and σe is effective or equivalent stress. Conservatively, the tri-axiality factor, TF, value is taken as 2 for DP-SC modules. The values in Table 3.5-5a and Table 3.5-5b are maximum values under given loading condition. Note that the acceptance criteria presented in Table 3.5-5a and Table 3.5-5b are applicable to large structural components impacted by large impulsive loading. The strain values for severe damage are not provided as it is difficult to measure strains at that level of damage.

(2) The following limitations are placed on the acceptance criteria for DP-SC containment under DBAs: For normal and severe environmental load categories that include impactive/impulsive loads, the acceptance criteria is limited to superficial damage in Table 3.5-5a and Table 3.5-5b; and, for abnormal, extreme environmental, and abnormal and extreme environmental load categories that include impactive/impulsive loads, the allowable limits for ductility, support rotation, and strain shall not exceed those for limited damage in Table 3.5-5a and Table 3.5-5b.

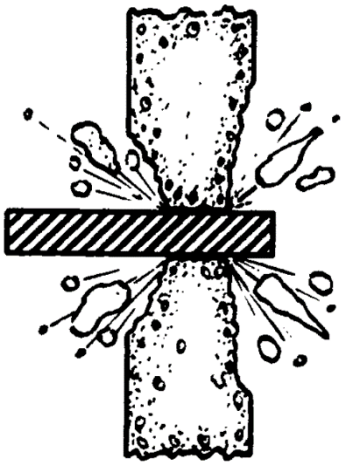
(3) For DP-SC non-containment, under abnormal and extreme environmental load categories that include impactive and impulsive loads, the limits for moderate damage criteria in Table 3.5-5a and Table 3.5-5b are acceptable for DBAs as listed in the design load combinations given in RG 1.243. Those limits are per ANSI/AISC N690 (Reference 3.5-8) as endorsed and modified by Regulatory Position C11.1 of RG 1.243.



(a) Missile Penetration and Spalling



(b) Target Scabbing



(c) Perforation

Figure 3.5-1: BWRX-300 Missile Impact Local Damage Modes for the Concrete Barrier

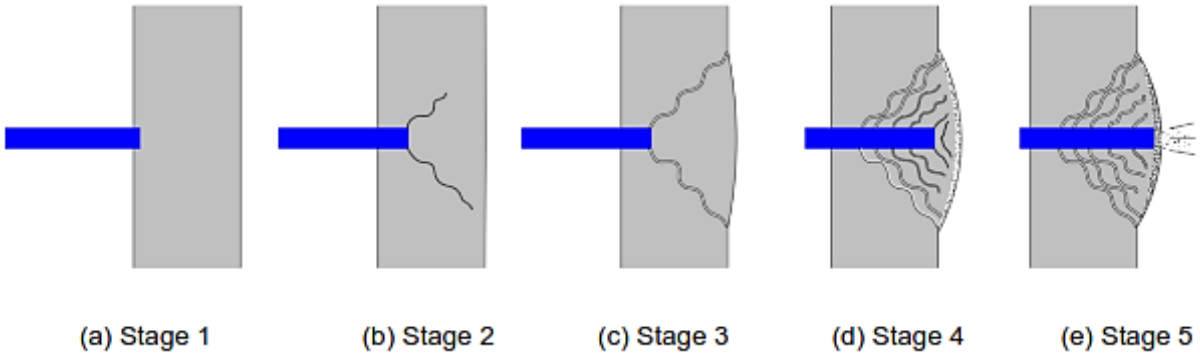


Figure 3.5-2: BWRX-300 Missile Impact Local Damage Modes for the Steel-Plate Composite Barrier

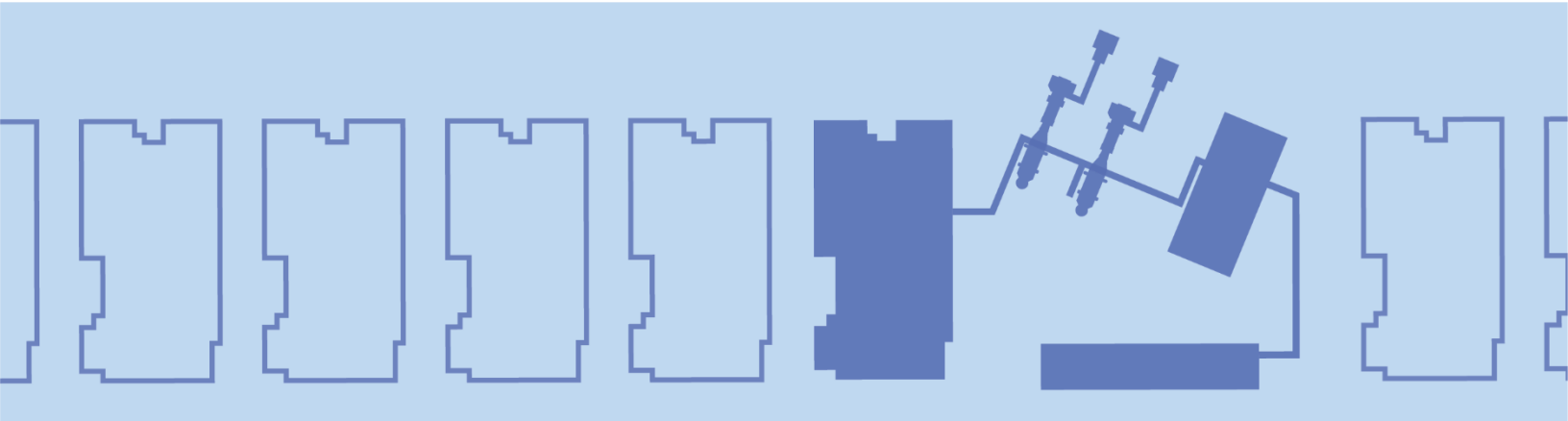
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Non-Proprietary Information

Chapter 3 Design of Structures, Components, Equipment, and Systems

Section 3.6 – Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping



ACRONYM LIST

Acronym	Explanation
AISC	American Institute of Steel Construction
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
BPVC	Boiler and Pressure Vessel Code
BTP	Branch Technical Position
CIV	Containment Isolation Valve
EPRI	Electric Power Research Institute
FMCRD	Fine Motion Control Rod Drive
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
HELB	High Energy Line Break
ICS	Isolation Condenser System
ISI	Inservice Inspection
NDE	Non-Destructive Examination
OM	Operations & Maintenance
RIV	Reactor Isolation Valve
RPV	Reactor Pressure Vessel
SRP	Standard Review Plan
SSCs	Structures, Systems, and Components
SSE	Safe Shutdown Earthquake
USNRC	United States Nuclear Regulatory Commission

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3.6.2 Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping 3-99

3.6.3 References 3-104

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None.

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None.

3.6 Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping

In accordance with NUREG-0800, Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition, Sections 3.6.1 and 3.6.2, the plant is designed for protection against piping failures inside and outside containment to assure that such failures do not cause the loss of needed functions of the systems and to ensure that the plant can be safely shut down in the event of such failures. The design includes consideration of high energy and moderate energy fluid system piping located inside and outside of containment. Design bases and measures used to protect these Structures, Systems, and Components (SSCs), referred to in the following subsections as essential SSCs, are discussed in Subsections 3.6.1 and Subsection 3.6.2. Essential SSCs are those that are required to shut down the reactor and mitigate the consequences of the postulated piping failure. In addition, any systems, or portions of systems, that are designed to mitigate the consequences of a postulated pipe failure are provided with design features that can assure the performance of their safety function. A list of preliminary BWRX-300 SSCs classification, their location, seismic category, and Quality Group is provided in Table 3A-1. Overall compliance for BWRX-300 design to General Design Criteria (GDC) 4 is provided in Subsection 3.1.1.4. An analysis of pipe break events will be performed in the Final Safety Analysis Report (FSAR) to identify essential SSCs to be protected from the postulated piping failures.

3.6.1 Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Inside and Outside Containment

3.6.1.1 Design Bases

The BWRX-300 pipe break event protection complies with GDC 4. The design bases and criteria for the analysis required to demonstrate that essential SSCs are protected from piping failure is provided in United States Nuclear Regulatory Commission (USNRC) Branch Technical Position (BTP) 3-3 “Protection Against Postulated Piping Failures in Fluid Systems Outside Containment,” and BTP 3-4, “Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment.” BTP 3-4 describes criteria for selecting the locations and orientations of postulated breaks and cracks in fluid systems piping.

3.6.1.1.1 Break Exclusion Zone

The break exclusion zone typically runs from the inboard Containment Isolation Valve (CIV) through the containment penetration to the outboard CIV as described in BTP 3-4 B.1(ii). Those sections of piping systems in containment penetration areas that meet the augmented requirements for a break exclusion zone in accordance with BTP 3-4, are exempted from the evaluation of the dynamic effects of postulated breaks in high-energy lines, and the installation of the corresponding whip restraints and jet shields. The dynamic effects that are eliminated are:

- Pipe whip
- Jet impingement
- Blast wave
- Blowdown (decompression water hammer)

In the BWRX-300 design, the Reactor Isolation Valve (RIV) is acting as the inboard CIV for all high energy systems subjected to BTP 3-4 B.1(ii) criteria. The boundaries of the break exclusion zone run from the piping weld at the flanged connection to RIVs (inboard CIVs) to the outboard CIVs and past the outboard CIVs to the seismic interface restraints The Isolation Condenser

System (ICS) does not have an outboard CIV as discussed in Subsection 6.2.4 and all of the ICS piping is within the break exclusion zone.

Section 5.1.7 of NEDC-33911P-A discusses the pipe break exclusion zone boundaries where breaks and cracks are postulated in those portions of piping from the RIVs that function as the inboard CIVs to the containment wall. NEDC-33910P-A, "BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection," (Reference 3.6-1) discusses that each RIV assembly is also connected to the outboard piping using bolted flange connections. Section 2.4.1 of NEDC-33910P-A discusses that for piping connected to the RIV assemblies, extending to the containment wall, the BWRX-300 design requirements include identifying postulated pipe rupture locations and configurations inside containment as specified in BTP 3-4, Part B.1(iii)(2), and identifying leakage cracks as specified in BTP 3-4, Part B, Item 1(v)(2). Additionally, NEDC-33910P-A states that a terminal end break would be postulated at the piping connection to the reactor pressure vessel outboard isolation valve as specified in BTP 3-4, B.1(iii)(2).

For the piping in containment penetration areas, breaks and cracks are not postulated in portions of piping from the piping weld to the RIV flange up to and including the nearest piping anchor outboard of the outside CIV as discussed above. This approach is acceptable as all requirements from BTP 3-4 B.1(ii) are met for the BWRX-300 design, which allows eliminating postulated breaks and cracks in those portions of piping within the containment penetration areas from the containment wall to the inboard or outboard CIVs.

The NRC safety evaluation report for NEDC-33911P-A (Reference 3.6-2) has a limitation and condition specified in Section 6 which states that the "USNRC staff must review and approve in future licensing activities the potential postulated breaks and cracks in the closed-loop piping being used for meeting GDC 55 on the "other defined basis." In addition, the USNRC staff reviews the consequences of pipe failures outside containment resulting from the ICS steam supply and condensate return piping and Fine Motion Control Rod Drive (FMCRD) hydraulic lines. The evaluation includes dynamic and environmental effects of pipe failures and fission product releases in the reactor building."

The FMCRD hydraulic lines evaluation includes dynamic and environmental effects of pipe failures. However, for the BWRX-300 design, the piping within the ICS component jurisdiction is considered as a break exclusion zone. This approach is acceptable as all requirements from BTP 3-4 B.1(ii) are met for the BWRX-300 design which allows eliminating postulated breaks and cracks in those portions of piping within the containment penetration areas from the containment wall to, the inboard or outboard CIVs.

As described in NEDC-33910P-A (Reference 3.6-1), GDC 4 dynamic effects for the area between the piping weld at the flanged connection to the RIVs and the Reactor Pressure Vessel (RPV) are eliminated by design provisions and Inservice Inspection (ISI) that ensure the probability of gross rupture at this location is extremely low.

3.6.1.1.2 Environmental Effects in Break Exclusion Zone

BTP 3-3 B.1 (a)(1) specifies: "Even though portions of the main steam and feedwater lines meet the break exclusion requirements of item 2.A(ii) of BTP 3-4, they should be separated from essential equipment. Designers are cautioned to avoid concentrating essential equipment in the break exclusion zone. Essential equipment must be protected from the environmental effects of an assumed non-mechanistic longitudinal break of the main steam and feedwater lines. Each assumed non-mechanistic longitudinal break should have a cross sectional area of at least one square foot and should be postulated to occur at a location that has the greatest effect on essential equipment."

Non-mechanistic longitudinal breaks are postulated in the main steam and main feedwater piping at the location having the most adverse effect on essential SSCs for evaluation for environmental, flooding, and over-pressure damage. A flow area of 1.0 ft^2 (0.0929 m^2) in BTP 3-3 B1(a)(1) is applicable to large, light-water reactors but is disproportionately large for a small modular reactor with small pipe sizes. Therefore, the flow area will be proportionally scaled to reflect the smaller main steam and feedwater pipe size in the BWRX-300 design and provided in the FSAR.

3.6.1.1.3 Augmented Inservice Inspections

In accordance with BTP 3-4 B.1(ii)(7), "A 100 percent volumetric in-service examination of all pipe welds should be conducted during each inspection interval as defined in American Society of Mechanical Engineers (ASME) Code, Section XI, IWA-2400." The welds and base metal in the break exclusion zone are pre-examined and ISI in accordance with ASME Boiler and Pressure Vessel Code (BPVC)-XI, Division 1– "Rules for Inservice Inspection of Nuclear Reactor Facility Components," (Reference 3.6-3). The ISI is performed on 100% of the welds during each inspection interval.

The break exclusion zone piping is subjected to augmented pre-operational testing for evidence of vibration, in accordance with ASME OM Code, "Operation and Maintenance of Nuclear Power Plants," (Reference 3.6-4).

Initial ISI programs use the latest edition and addenda of ASME BPVC, Section XI (Reference 3.6-3) and the ASME OM Code subject to the conditions identified and incorporated by reference into 10 CFR 50.55a, on the date no more than 18 months before the date of issuance of the Operating License and is addressed in the FSAR. The break exclusion zone piping is designed and laid out to provide access for ISI of every pipe segment, bend, and weld.

In addition to compliance with BTP 3-4 inspection criteria, the break exclusion zone piping is included in the aging management population assessed for the potential of damage mechanisms other than weld cracking. The specific damage mechanisms are identified, based on industry data, including using the latest revisions of:

- NUREG-1801 "Generic Aging Lessons Learned"
- NUREG-2191 "Generic Aging Lessons Learned for Subsequent License Renewal"

The piping is systematically assessed for all potentially applicable damage mechanisms using guidance from Generic Aging Lessons Learned (NUREG-1801 and NUREG-2191), reports published by Electric Power Research Institute (EPRI), and reports from the International Atomic Energy Agency (IAEA). Corresponding augmented inspection procedures are developed as part of the FSAR, and these augmented inspections provide supplementary assurance that piping in the break exclusion zone is not subjected to leakage or rupture. The ISI are implemented using Non-Destructive Examination (NDE) techniques that have been demonstrated and qualified to detect threshold values of metal loss, cracking, or embrittlement as part of the break exclusion zone ISI procedure.

3.6.1.1.4 Guard Pipe Assembly Design Criteria

In accordance with NUREG-0800, BTP 3-4, B.1.(ii)(3), where guard pipes are used, the enclosed portion of fluid system piping utilizes seamless construction without circumferential welds unless specific access provisions are made to permit in-service volumetric examination of the longitudinal and circumferential welds.

Guard pipes provided for those portions of piping in the containment penetration areas are designed in accordance with the criteria of the ASME BPVC, Section III, Subsection NE, Class MC, where the guard pipe is part of the containment boundary. In addition, the guard pipe assembly is designed to meet the following design criteria and tests specified in NUREG-0800, BTP 3-4, B.1.(ii)(6):

1. The design pressure and temperature are not less than the maximum operating pressure and temperature of the enclosed pipe under normal plant conditions.
2. The Level C stress limits in ASME BPVC, Section III, NE-3220 are not exceeded under the loading associated with containment design pressure and temperature in combination with the Safe Shutdown Earthquake (SSE).
3. Guard pipe assemblies are subjected to a single pressure test at a pressure not less than its design pressure.
4. Guard pipe assemblies do not prevent the access necessary to conduct the in-service examination specified in BTP 3-4, 2.A(ii)(7). Inspection ports, if used, are not located in that portion of the guard pipe through the annulus of dual barrier containment structures.

3.6.1.1.5 Protection of Essential Structures, Systems, and Components

The design of the plant provides adequate general protection against postulated high energy and moderate energy fluid system pipe ruptures, to the extent necessary to ensure the integrity and operability of essential SSCs required to safely shut down the plant and mitigate the consequences of the postulated piping failure in accordance with regulatory requirements. Both direct dynamic effects on essential SSCs inside the zone of influence and indirect dynamic effects are considered. Examples of an indirect effect would be the whipping pipe impacting a structural frame to which essential SSCs are attached, dislodged grating resulting in a secondary missile, or missiles from back face scabbing or front face spalling on reinforced concrete caused by pipe impact.

SRP Subsections 3.6.1 and 3.6.2 describe acceptable measures that could be taken for protection against the breaks and cracks and for restraint against pipe whips that result from breaks as discussed in Subsection 3.6.1.3.

If the consequence of dynamic effects on essential SSCs is unacceptable, the following mitigation options may be applied:

1. The essential SSCs are relocated.
2. The whipping pipe is restrained by a whip restraint, resulting in a smaller restrained zone of influence that does not encompass the essential target.
3. A protective barrier or jet shield is used to protect the essential SSCs from impact (barrier) or jet impingement (barrier or jet shield).
4. The essential SSCs are qualified to operate following the pipe rupture interaction.

3.6.1.2 Definition of High Energy and Moderate Energy Fluid Systems

High energy fluid systems are defined as those systems or portions of systems that, during normal plant conditions are either in operation or are maintained pressurized under conditions where either or both of the following are met:

- Maximum operating temperature exceeds 200°F
- Maximum operating pressure exceeds 275 psig

Systems or portions of systems pressurized above atmospheric pressure during normal plant conditions that are not classified as high energy are classified as Moderate Energy Fluid Systems.

Through-wall leakage cracks in lieu of breaks are postulated in the piping of those fluid systems that qualify as high energy fluid systems for only a short operational period but qualify as moderate energy fluid systems for the major operational period. An operational period is considered “short” if the fraction of time that the system operates within the pressure-temperature conditions specified for high energy fluid systems is less than 2% of the time that the system operates as a moderate energy fluid system.

For the BWRX-300, the following high energy piping systems are considered as potential candidates for a postulated pipe break during normal plant conditions and are analyzed for potential damage resulting from damage effects:

- Nuclear Boiler System
- Isolation Condenser System
- Control Rod Drive System
- Reactor Water Cleanup System
- Condensate and Feedwater Heating System
- Offgas System
- Condensate Filters and Demineralizers System
- Moisture Separator Reheater System
- Main Condenser and Auxiliaries

Moderate energy piping systems considered as potential candidates for a postulated pipe crack include the following:

- Boron Injection System
- Isolation Condenser Pools Cooling and Cleanup System
- Shutdown Cooling System
- Fuel Pool Cooling and Cleanup System
- Liquid Waste Management System
- Turbine Bypass System
- Circulating Water System
- Chilled Water Equipment
- Plant Cooling Water System
- Plant Pneumatics System
- Passive Containment Cooling System
- Containment Inerting System
- Containment Cooling System
- Heating, Ventilation, and Cooling System

- Fire Protection System
- Equipment and Floor Drain System
- Water, Gas, and Chemical Pads

3.6.1.3 Design (Safety) Evaluation

Essential SSCs are protected against postulated piping failures. The effects of postulated breaks and cracks include pipe whipping, jet impingement, fluid decompression waves within the ruptured pipes and environmental effects such as temperature, pressure, humidity, chemical exposure, radiation, spray wetting, flooding, and sub compartment pressurization.

An evaluation is performed to verify that the consequences of failures of high and moderate energy lines do not affect the ability to safely shut down the plant. Analysis of the postulated failure of piping is provided during the Operating licensing review process.

3.6.2 Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

Information concerning break and crack location criteria and methods of analysis for dynamic effects are discussed in this Subsection in accordance with SRP 3.6.2. This includes location criteria and methods of analysis needed to evaluate the dynamic effects associated with postulated breaks and cracks in high and moderate energy fluid system piping inside and outside of the primary containment. The information in this section provides the basis for the requirements for the protection of essential SSCs.

3.6.2.1 Criteria Used to Define Break and Crack Location and Configuration

The following subsections establish the criteria for the location and configuration of postulated breaks and cracks.

3.6.2.1.1 Locations of Postulated Pipe Breaks

Postulated pipe breaks are selected as follows:

Piping in the Containment Penetration Areas

The augmented requirements of BTP 3-4 are implemented, the dynamic effects resulting from the postulation of guillotine breaks in high energy lines in containment penetration areas are excluded from the design.

ASME BPVC Code Section III Class 1 Piping in Areas Other Than Containment Penetration

Pipe breaks are postulated at the following locations in each piping and branch run designed in accordance with the criteria provided in ASME “BPVC Section III, “Rules for Construction of Nuclear Facility Components – Division 1 – Subsection NB – Class 1 Components,” (Reference 3.6-5):

- At terminal ends
- At intermediate locations where the maximum stress range or fatigue usage values exceed the limits specified in BTP 3-4

ASME BPVC Code Section III Class 2 and 3 Piping in Areas Other Than Containment Penetration

Pipe breaks are postulated at the following locations in those portions of each piping and branch run designed in accordance with the design criteria provided in ASME, BPVC, Section III, “Rules

for Construction of Nuclear Facility Components – Division 1 – Subsection NCD - Class 2 and 3 Components," (Reference 3.6-6):

- At terminal ends
- At intermediate locations where the maximum stress values exceed the limits specified in BTP 3-4

Non-ASME BPVC Section III Class Piping

Breaks in seismically analyzed high energy non-ASME BPVC Section III piping systems are postulated according to the same criteria as for ASME BPVC Section III, Class 2 and 3 high energy piping systems.

Breaks in non-seismically analyzed high energy non-ASME BPVC Section III piping systems are postulated at terminal ends and at each intermediate location such as pipe fittings, valves, flanges, and welded attachments.

3.6.2.1.2 Locations of Postulated Pipe Cracks

Postulated pipe crack locations are selected as follows:

Piping in Containment Penetration Areas

Regions of high and moderate energy piping associated with reactor containment penetrations consider analytical concepts to eliminate the need to consider postulated cracks as described in BTP 3-4.

ASME BPVC Code Section III Class 1, 2, 3 and non-ASME High Energy Piping in Areas Other Than Containment Penetration Areas

Cracks in high-energy piping are postulated as follows:

1. For ASME Class 1 piping, at axial locations where the calculated stress range values exceed the limits specified in BTP 3-4.
2. For ASME Class 2 and 3 or non-ASME class piping, at axial locations where the calculated stress values exceed the limits specified in BTP 3-4.
3. For non-seismically analyzed non-ASME BPVC Section III piping which has not been evaluated to obtain stress information, through-wall cracks are postulated at axial locations that produce the most severe environmental effects.

ASME BPVC Code Section III Class 1, 2, 3 and non-ASME Moderate Energy Piping in Areas Other Than Containment Penetrations

Cracks in moderate energy piping adjacent to essential SSCs are postulated except where:

1. For ASME Class 1 piping, the calculated stress range values are less than the limits specified in BTP 3-4.
2. For ASME Class 2 or 3 and non-ASME class piping, the calculated stress values are less than the limits specified in BTP 3-4.

Through-wall cracks, unless the piping system is exempted above, are postulated at axial and circumferential locations that result in the most severe environmental effects.

3.6.2.1.3 Types of Breaks and Cracks to be Postulated

Pipe Breaks

Where not precluded by other means, the following criteria are used to postulate breaks in high energy fluid system piping at the identified locations:

Circumferential Pipe Breaks:

1. Circumferential breaks are postulated in piping having a diameter nominal greater than DN25 (1-inch nominal pipe size), except where the maximum stress range exceeds the limits specified in BTP 3-4 B.1(iii)(1) and B.1(iii)(2), and the circumferential stress range is at least 1.5 times the axial stress range.
2. Where break locations are selected without the benefit of stress calculations, breaks are postulated at the piping welds to each fitting, valve, or welded attachment.
3. Circumferential breaks are assumed to result in pipe severance and separation amounting to at least a one-diameter lateral displacement of the ruptured piping sections unless physically limited by piping restraints, structural members, or piping stiffness as may be demonstrated by inelastic limit analysis (e.g., a plastic hinge in the piping is not developed underloading).
4. The dynamic force of the jet discharge at the break location is based on the effective cross-sectional flow area of the pipe and on a calculated fluid pressure as modified by an analytically or experimentally determined thrust coefficient. Limited pipe displacement at the break location, line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs may be taken into account, as applicable, in the reduction of jet discharge.
5. Pipe whipping is assumed to occur in the plane defined by the piping geometry and configuration and to initiate pipe movement in the direction of the jet reaction.

Longitudinal Pipe Breaks:

1. Longitudinal breaks are postulated only in piping having a nominal diameter equal to or greater than DN100 (4-inch nominal pipe size), except where the maximum stress range exceeds the limits specified in BTP 3-4 B.1(iii)(1) and B.1(iii)(2), and the axial stress range is at least 1.5 times the circumferential stress range.
2. Longitudinal breaks are not postulated at terminal ends.
3. Longitudinal breaks are assumed to result in an axial split without pipe severance. Splits are oriented (but not concurrently) at two diametrically opposed points on the piping circumference such that the jet reactions cause out-of-plane bending of the piping configuration. Alternatively, a single split may be assumed at the section of highest tensile stress as determined by detailed stress analysis (e.g., finite element analysis).
4. The dynamic force of the fluid jet discharge is based on a circular or elliptical (2D x 1/2D) break area equal to the effective cross-sectional flow area of the pipe at the break location and on a calculated fluid pressure modified by an analytically or experimentally determined thrust coefficient as determined for a circumferential break at the same location. Line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs may be taken into account, as applicable, in the reduction of jet discharge.
5. Piping movement is assumed to occur in the direction of the jet reaction unless limited by structural members, piping restraints, or piping stiffness as demonstrated by inelastic limit analysis.

Pipe Cracks

Where not precluded by other means, the following criteria are used to postulate through-wall leakage cracks in high-or moderate energy fluid system piping at the identified locations:

1. Leakage cracks are postulated only in piping having a diameter nominal greater than DN25 (1-inch nominal pipe size).
2. Crack openings are assumed as a circular orifice of area equal to that of a rectangle having dimensions one-half pipe diameter in length and one-half pipe wall thickness in width.
3. The flow from the crack opening is assumed to result in an environment that wets all unprotected components within the compartment, with consequent flooding in the compartment and communicating compartments, based on a conservatively estimated time period to effect corrective actions.
4. For high energy fluid system piping, the leakage cracks are postulated to be in those circumferential locations that result in the most severe environmental consequences. For moderate energy fluid system piping, through-wall cracks are postulated at axial and circumferential locations that result in the most severe environmental consequences.

3.6.2.2 Analytical Methods to Define Forcing Functions and Response Models

3.6.2.2.1 Analytical Methods to Define Blowdown Forcing Functions

Thrust force and its magnitude is determined at the postulated circumferential and longitudinal break are developed by analytical solutions (closed-form equations) or by numerical simulation (computational fluid dynamics techniques). Analytical methods used to establish pipe rupture blowdown and jet thrust forcing functions are in accordance with American National Standards Institute/ American Nuclear Society (ANSI/ANS) 58.2, “Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture,” Appendix B, (Reference 3.6-7) modified by NUREG/CR-7275, “Jet Impingement in High-Energy Piping Systems,” to address non conservatisms identified in SRP 3.6.2, Appendix A.

3.6.2.2.2 High Energy Pipe Whip Analysis

The analysis of pipe whip follows analytical methods using dynamics equations or a numerical method using computational fluid dynamics and/or finite element analysis. The following are some of the considerations that apply to pipe whip analysis:

- Two ends of the broken pipe move clear of each other unless physically limited by piping restraints, structural members, or piping stiffness
- Determine the direction of the thrust force, axial to the broken pipe segment (perpendicular to the break area)
- The unrestrained zone of influence is determined in the direction of the jet reaction initially, with the total path controlled by the piping geometry
- If unrestrained, a whipping pipe having a constant energy source sufficient to form a plastic hinge is considered to form a plastic hinge and rotate about the plastic hinge or the nearest rigid pipe whip restraint, anchor, or wall penetration capable of resisting the pipe whip loads; or rotate about the calculated dynamic plastic hinge location

- Where the hinge moment has a torsional and a bending component, the hinge does not result in a planar unrestrained zone of influence. In this case, a half-sphere unrestrained zone of influence is assumed with the pipe sweeping a 180° arc from its initial position to its final resting position, unless a more detailed analysis is performed to justify a smaller unrestrained zone of influence.
- If the plane of motion of a whipping pipe is normal to a flat surface at impact, it is assumed that the pipe comes to rest against that surface
- The internal fluid energy level associated with the pipe break reaction considers any line restrictions (e.g., flow limiter) between the pressure source and break location, and the effects of either single-ended or double-ended flow conditions, as applicable. The energy level in a whipping pipe is considered insufficient to rupture an impacted pipe of equal or greater nominal pipe size and equal or heavier wall thickness.
- Credit is taken for the limited amount of energy in closed ended piping runs near dead ends or normally closed valves
- For those portions of high energy piping systems that are normally pressurized only during plant conditions other than 100 percent power, the thermodynamic state that produces the most severe fluid reaction forces is used

3.6.2.3 Dynamic Analysis Methods to Verify Integrity and Operability

3.6.2.3.1 Jet Impingement Analyses

Modeling of the jet geometry and determination of the jet impingement force acting on a target is calculated according to ANSI/ANS 58.2, with modifications applied as identified in NUREG/CR-7275 to address non conservatisms identified in SRP 3.6.2, Appendix A.

3.6.2.3.2 Design Codes and Load Combinations for Pipe Whip Restraint

Pipe whip restraints are Seismic Category II. They are seismically designed to prevent adverse interaction with Seismic Category I SSCs. Pipe whip restraints are designed to the following codes and standards:

- Pipe whip restraints that only act to restrain the effects of pipe breaks, and are not active for other loads, are designed to ANSI/American Institute of Steel Construction (AISC) N690-18 “Specification for Safety-Related Steel Structures for Nuclear Facilities,” (Reference 3.6-8). They are not ASME III NF restraints.
- Pipe whip restraints that also act as restraints for other loads (e.g., deadweight, seismic) are ASME BPVC-III-1-NF “Rules for Construction of Nuclear Facility Components - Division 1 – Subsection NF - Supports,” (Reference 3.6-9) restraints. However, the qualification for the pipe break impact load follows ANSI/AISC N690-18.
- The pipe whip restraints are designed to allow access for ISIs. The measures taken for the protection of essential SSCs do not preclude the conduct of in-service examinations of ASME Class 2 and 3 pressure-retaining components as required by the rules of ASME BPVC- Section XI.
- The pipe-restraint gap is sufficiently small to limit the kinetic energy of the whipping pipe and therefore the impact load
- The pipe-restraint gap is sufficiently large to prevent interference from thermal expansion of the line
- Pipe whip restraints are located to prevent the formation of a plastic hinge

The pipe whip restraints, jet shields, and barriers are designed for the following load and load combinations (Service Level D):

$$DW + TH + \sqrt{DBPB^2 + (|SSE| + |SSES| + |SWE|)^2}$$

Where:

DBPB = Design Basis Pipe Break (including jet impingement)

DW = Deadweight

SSE = Safe Shutdown Earthquake

SSES = SSE seismic anchor motion of the restraint structure

SWE = Seismic Self-Weight excitation of the restraint structure

TH = thermal loads due to the expansion of the whip restraint relative to the post-break hot building structure.

3.6.2.4 Analytical Methods to Define Blast Wave Interaction to Essential SSCs

Blast Wave Effects

- High energy blast wave analysis is performed assuming the break opening time as instantaneous, maximizing blast formation
- The formation and effects of a blast wave caused by a High Energy Line Break (HELB) is evaluated using three-dimensional computational fluid dynamics modeling technique that reflects the thermohydraulic parameters at the instant of the postulated pipe rupture, within 1 millisecond of the onset of the break
- These blast wave forces on target surfaces are impulsive loads that last a few milliseconds or less
- The blast analysis includes amplification caused by blast wave reflections against walls, and the effects of angled incidence
- Because of pressure-relieving clearing at the edges of a target surface, and the short duration of the pressure impulse, small structures are not exposed to significant loading. The analysis of blast wave effect is therefore limited to walls, floors, and large vessels.
- Blast force and impulse may be bounded by the jet thrust forces that subsequently develop

3.6.2.5 Pipe Break Analysis Results and Protection Methods

A summary of the dynamic analyses applicable to high energy piping systems follows the guidelines provided in SRP 3.6.2. This information is provided and completed in conjunction with the Operating License application process.

3.6.3 References

3.6-1 NEDC-33910P-A, Revision 2, “BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection,” GE Hitachi Nuclear Energy, June 2021.

3.6-2 Final Safety Evaluation For GE-Hitachi Licensing Topical Report NEDC-33911P, Revision 0, Supplement 1, BWRX-300 Containment Performance,” U.S. Nuclear Regulatory Commission, 3/12/2021.

3.6-3 ASME BPVC Section XI, Division 1, “Rules for Inservice Inspection of Nuclear Reactor Facility Components,” American Society of Mechanical Engineers.

3.6-4 ASME OM Code, “Operation and Maintenance of Nuclear Power Plants,” American Society of Mechanical Engineers.

3.6-5 ASME BPVC-III NB, BPVC Section III – “Rules for Construction of Nuclear Facility Components - Division 1 - Subsection NB – Class 1 Components,” American Society of Mechanical Engineers, 2021.

3.6-6 ASME BPVC-III NCD, BPVC Section III – “Rules for Construction of Nuclear Facility Components - Division 1 - Subsection NCD – Class 2 and Class 3 Components,” American Society of Mechanical Engineers, 2021.

3.6-7 ANSI/ANS 58.2, “Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture,” American National Standards Institute/American Nuclear Society, 1998.

3.6-8 ANSI/AISC N690-18, “Specification for Safety-Related Steel Structures for Nuclear Facilities,” American National Standards Institute/American Institute of Steel Construction, 2018.

3.6-9 ASME BPVC, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” Subsection NF, “Supports,” American Society of Mechanical Engineers, 2021.

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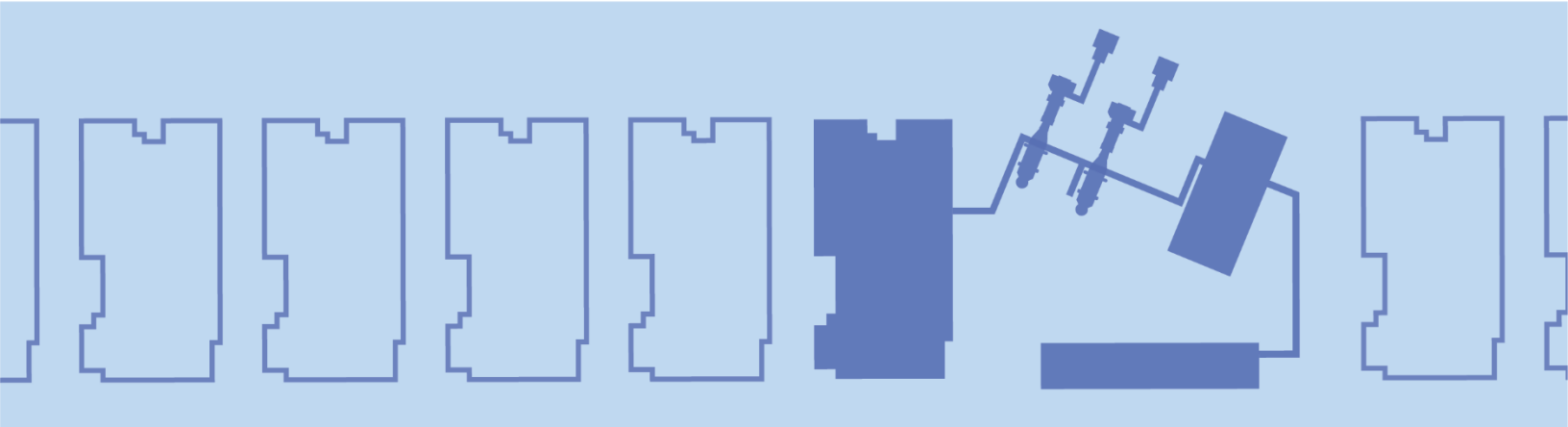
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Non-Proprietary Information

Chapter 3 Design of Structures, Components, Equipment, and Systems

Section 3.7 – Seismic Design



ACRONYM LIST

Acronym	Explanation
1D	One-Dimensional
2D	Two-Dimensional
3D	Three-Dimensional
ACI	American Concrete Institute
ANS	American Nuclear Society
ANSI	American National Standards Institute
ARS	Acceleration Response Spectra
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ATH	Acceleration Time History
BE	Best Estimate
BECPV	Blue Energy Center Port of Victoria
BPVC	Boiler and Pressure Vessel Code
CB	Control Building
CRD	Control Rod Drive
DP-SC	Diaphragm Plate Steel-Plate Composite
FE	Finite Element
FIRS	Foundation Input Response Spectra
GDC	General Design Criteria
IBC	International Building Code
ISRS	In-Structure Response Spectra
LB	Lower Bound
LMS	Lumped Mass Stick
NEI	Nuclear Energy Institute
OBE	Operating-Basis Earthquake
PBIRS	Performance Based Intermediate Response Spectra
PBSRS	Performance Based Surface Response Spectra
PSHA	Probabilistic Seismic Hazard Assessment
RB	Reactor Building
RG	Regulatory Guide
RPV	Reactor Pressure Vessel

Acronym	Explanation
RWB	Radwaste Building
SCCV	Steel-Plate Composite Containment Vessel
SEI	Structural Engineering Institute
SMR	Small Modular Reactor
SRA	Site Response Analysis
SRP	Standard Review Plan
SRSS	Square-Root-of-the-Sum-of-the-Squares
SSCs	Structures, Systems and Components
SSE	Safe Shutdown Earthquake
SSI	Soil-Structure Interaction
SSSI	Structure-Soil-Structure Interaction
TB	Turbine Building
UB	Upper Bound
USNRC	U.S. Nuclear Regulatory Commission
V/H	Vertical to Horizontal
ZPA	Zero Period Acceleration

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3.7 Seismic Design

The seismic design of the BWRX-300 Structures, Systems and Components (SSCs) complies with the General Design Criteria (GDC) 2, “Design Bases for Protection Against Natural Phenomena,” as stated in Section 3.1 and 10 CFR 50, Appendix S, “Earthquake Engineering Criteria for Nuclear Power Plants.”

The SSCs are assigned a seismic category that reflects the SSCs’ functional and performance requirements during or after a seismic event and affects their design.

Seismic Category I SSCs required to remain functional during and after a seismic event are seismically qualified to withstand the effects of a Safe Shutdown Earthquake (SSE). This SSE is developed following the regulatory guidance of U.S. Nuclear Regulatory Commission (USNRC) Regulatory Guide (RG) 1.208, “A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion,” and is based on the results of the site investigations discussed in Section 2.5. Subsection 3.7.1 provides the site-specific SSE ground motions used for the seismic design and qualification of the BWRX-300 plant, along with other design parameters developed for the seismic analysis.

Table 3A-1 provides the seismic categorization of the BWRX-300 systems and components. The seismic categorization of the BWRX-300 Power Block structures is presented in Table 3A-2.

As listed in Table 3A-2, the containment structure, the containment internal structures and the Reactor Building (RB) enveloping them are the only civil structures categorized as Seismic Category I.

The Radwaste Building (RWB) is categorized as Seismic Category RW-IIa per RG 1.143, “Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants,” and is designed for ½ SSE. The RWB is adjacent to and can interact with the RB during seismic or extreme wind events. The seismic interaction evaluation of the RWB is performed following the requirements and guidelines in Section 6.2 of NEDO-33914-A, “BWRX-300 Advanced Civil Construction and Design Approach,” (Reference 3.7-1) to ensure the structural integrity and safety functions of the RB Seismic Category I SSCs are not compromised.

The Control Building (CB), Turbine Building (TB), Service Building and Reactor Auxiliary Structures (see Section 1.2) support or protect SSCs that are not credited in the safety analysis to perform Safety Category 1 functions or that can perform their desired function regardless of structure performance. These structures also have the potential to interact with the RB during seismic or extreme wind events due to their proximity to the RB. As a result, the structures are assigned the BWRX-300 Seismic Category II described in Section 3.2 and are designed in accordance with the International Building Code (IBC) (Reference 3.7-2). The seismic interaction evaluation of the structures with the RB is performed following the requirements and guidelines in Section 6.2 of NEDO-33914-A.

The remaining Power Block structures consisting of detached pre-fabricated/modularized Equipment Structures are categorized as Non-Seismic and are designed per the IBC.

Table 3.7-1 summarizes the seismic design basis for the Power Block structures based on their seismic categories. As indicated in Table 3.7-1, the seismic design of Seismic Category I structures considers Limit State LS-D response defined in Table 1-1 of American Society of Civil Engineers (ASCE) / Structural Engineering Institute (SEI) 43,” Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities,” (Reference 3.7-3) as essentially elastic response. As explained in RG 1.208, this ensures a consistent level of safety from

earthquake-caused failures defined by level of response resulting in an onset of significant inelastic deformations with a probability of unacceptable performance:

- Less than 1% for an SSE ground motion level
- Less than 10% for ground motion with 1.5 times the SSE intensity

Analyses performed to qualify the Seismic Category I structures are discussed in Subsection 3.7.2. Analyses performed to qualify the Seismic Category I subsystems are discussed in Subsection 3.7.3. The interaction of Non-Seismic Category I SSCs with Seismic Category I SSCs is addressed in Subsections 3.7.2.8 and 3.7.3.8. Seismic instrumentation used to monitor earthquake events and their effects are discussed in Subsection 3.7.4. For more information on the design of the RWB, refer to Subsection 3.8.4.3.

In the following subsections, reference to the integrated RB is inclusive of the RB, containment, and containment internals, whereas RB is used to refer to the reactor building structure outside of containment.

3.7.1 Seismic Design Parameters

3.7.1.1 Design Ground Motion

The BWRX-300 design considers Operating-Basis Earthquake (OBE) and SSE design ground motions following the guidance of Standard Review Plan (SRP) 3.7.1, “Seismic Design Parameters.”

Subsections 3.7.1.1.1 and 3.7.1.1.2 will present the SSE design response spectra and time histories used for the design of the BWRX-300 Seismic Category I SSCs at Blue Energy Center – Port of Victoria (BECPV) in CPA Part 2.

The OBE at BECPV is set as 1/3 of the SSE. Per Appendix S to 10 CFR 50 and SRP 3.7.1, design load combinations that consider OBE loads are not required except for the design of containment metal components. OBE loads are considered for post-flooding condition and cyclic loading considerations for the containment metal components following the regulatory guidance of SRP 3.8.2, “Steel Containment.” The use of OBE for plant shutdown is discussed in Subsection 3.7.4.4. Fatigue evaluation criteria for seismic subsystems are addressed in Subsection 3.7.3.2 and Sections 3.9 and 3.12.

3.7.1.1.1 Design Response Spectra

The ground motion design spectra presented in this section define the amplitude and the frequency content of the design ground motion used for the seismic design of the BWRX-300 at BECPV. The ground motion design spectra are developed based on the results of the Probabilistic Seismic Hazard Assessment (PSHA) discussed in Section 2.5, considering as-built subgrade conditions. Chapter 2, including the PSHA results, will be provided in CPA Part 2. The SSE ground motion design spectra are developed in accordance with SRP 3.7.1 and RG 1.208, using the performance-based approach of ASCE/SEI 43 and the additional requirements provided in Section 5.2.2 of NEDO-33914-A.

In accordance with the requirements of Section 5.2.2 of NEDO-33914-A, the following horizontal and vertical spectra define the site-specific SSE ground motion:

1. Foundation Input Response Spectra (FIRS) defining the SSE ground motion at the bottom of the integrated RB foundation.
2. Performance Based Surface Response Spectra (PBSRS) defining the SSE ground motion at the finished plant grade elevation.
3. Performance Based Intermediate Response Spectra (PBIRS) defining the SSE ground motion at intermediate embedment depth elevation established, following the guidelines in Section 5.2.2 of NEDO-33914-A.

The purpose of PBIRS is to ensure the ground motions used as input for the Soil-Structure Interaction (SSI) analyses of deeply embedded integrated RB are adequate throughout the depth of the embedment.

The 5% damped horizontal FIRS, PBSRS and PBIRS are obtained from the probabilistic Site Response Analysis (SRA) results which will be provided in CPA Part 2.

The vertical FIRS, PBSRS and PBIRS for the BWRX-300 at BECPV are developed by applying Vertical to Horizontal (V/H) ratios to the horizontal spectra which will be provided in CPA Part 2.

The Nuclear Energy Institute (NEI) checks, to be provided in CPA Part 2, are performed following the procedure described in Section 5.3.4.1 of NEDO-33914-A to ensure the ground motion used as input for the deterministic SSI analyses of the deeply embedded integrated RB at its foundation bottom elevation meets the guidance of USNRC DC/COL-ISG-017, “Ensuring Hazard-Consistent Seismic Input for Site Response and Soil Structure Interaction Analyses,” to be hazard consistent with the results of the probabilistic SRA. The NEI check requires that the PBSRS be enveloped by the FIRS propagated upward through the three strain-compatible soil profiles used as input for the SSI analyses, presented in Subsection 3.7.1.3. When the enveloped Acceleration Response Spectra (ARS) of the propagated motion do not meet or exceed the PBSRS, the FIRS are augmented until the augmented input ground motion satisfies the NEI check.

3.7.1.1.2 Design Time Histories

Table 3.7-2, to be provided in CPA Part 2, provides details of the selected five sets of time history records used for the development of the design ATHs for the SSI analyses of the BWRX-300 integrated RB at BECPV.

Figure 3.7-6, Figure 3.7-7, and Figure 3.7-8, to be provided in CPA Part 2, present examples of comparisons of the 5 % damped target spectra and the corresponding scaled response spectra of the modified time histories. Figure 3.7-9, to be provided in CPA Part 2, shows for each set of time histories, a smooth power spectrum density calculated as the geometrical mean of the power spectrum density of the two horizontal matched time histories.

Table 3.7-3, to be provided in CPA Part 2, presents the ground motion parameters of the matched time histories for FIRS. The correlation coefficient between the two horizontal components for each set of the matched time histories for FIRS are presented in Table 3.7-4, to be provided in CPA Part 2.

Figure 3.7-10 through Figure 3.7-24, to be provided in CPA Part 2, show the matched time history for the selected records listed in Table 3.7-2 matched to the horizontal and vertical FIRS.

3.7.1.2 Percentage of Critical Damping Values

Table 3.7-5 lists damping values used in the seismic analysis of the integrated RB structures and components. OBE and SSE damping values assigned to the integrated RB structures and components are in accordance with RG 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," with steel-plate composite damping values being based on the values provided for steel-plate composite walls in Tables 1 and 2 of RG 1.61. Damping values listed in Table 3.7-5 are applicable to all structural elements constructed of the same material.

Damping properties assigned to soil materials in the SSI analysis model take into account the stress-strain properties corresponding to the level of seismic input per the guidance of SRP 3.7.1.

Following the regulatory guidance of RG 1.61, Section C.1.2, lower OBE damping ratios are assigned to structural elements for generating in-structure demands for seismic qualification of equipment and systems. The higher SSE damping values are used for development of seismic demands for structural design per ASCE/SEI 43, Section 3.3.3 and RG 1.61, Section C.1.2, respectively.

Damping values for seismic evaluations of subsystems including piping and equipment are obtained using the procedures described in Subsection 3.7.3.5.

3.7.1.3 Supporting Media for Seismic Category I Structures

The integrated RB structures are built on a common mat foundation deeply embedded in soil. For a description of the supporting media at BECPV, refer to Sections 2.4 and 2.5. For the foundation embedment depth and total structural height of the Seismic Category I structures, refer to Figure 1.2-2 and Figure 1.2-3. For the Foundation dimensions, refer to Figure 3.8-8.

To address the uncertainties related to the determination and variation of subgrade conditions, the SSI analysis, to be provided in CPA Part 2 in Appendix 3C, uses three sets of subgrade profiles representing LB, BE and UB estimates of the subgrade material properties. These profiles are representative of the as-built conditions at BECPV. The design uses an envelope of results from the SSI analysis of BE, LB and UB subgrade profiles to account for the variation and uncertainty in subgrade properties in accordance with Section 5.3 of NEDO-33914-A.

The development of dynamic subgrade profiles considers the soils located below the nominal groundwater table to be fully saturated. Figure 3.7-25 and Figure 3.7-26, to be provided in CPA Part 2, present the strain-compatible shear wave velocity profiles, compression wave velocity and damping ratio profiles used for the seismic analyses of BWRX-300 Seismic Category I structures discussed in Subsection 3.7.2.

3.7.2 Seismic System Analysis

In accordance with the requirements and guidelines of Section 5.0 of NEDO-33914-A, the seismic SSI analyses of the BWRX-300 Seismic Category I integrated RB consider the following effects on its structural integrity, seismic response, and distribution of stress demands:

- Interaction of the deeply embedded RB structure with the surrounding subgrade (SSI)
- Variation in the soil and structural parameters
- Hydrodynamic loads (mass and stiffness)
- Structure-Soil-Structure Interaction (SSSI) with the adjoining RWB, CB, TB, Service Building, and Reactor Auxiliary Structures

To adequately account for the SSI and SSSI effects per guidance of Section 5.1 of NEDO-33914-A, the one-step approach, as defined in Section 3.1.2 of ASCE/SEI 4, "Seismic

Design Safety-Related Nuclear Structures," (Reference 3.7-7), is implemented for the seismic design of the integrated RB structures. The surrounding subgrade is represented in the SSI analysis models by layered half-space continuum with equivalent linear elastic stiffness properties and complex damping.

The methodology used to develop the integrated RB 3D FE model used in the seismic SSI analyses is described in Subsection 3.7.2.3.

3.7.2.1 Seismic Analysis Method

The BWRX-300 one-step seismic SSI analysis approach provides demands for the seismic design and qualification of SSCs for all frequencies of interest and adequately addresses the effects of SSSI for the integrated RB with adjacent structures and foundations.

The BWRX-300 seismic analysis approach follows the guidance of Section 5.0 of NEDO-33914-A with a few modifications aimed to represent realistic site, structures and construction conditions described in Subsections 3.7.2.3 and 3.7.2.4. Section 5.0 of NEDO-33914-A addresses current limitations in SRP 3.7.2 when capturing the effects of seismic response and addressing the effects of seismic interaction of the deeply embedded RB structure with adjacent structures through the subgrade. These issues are identified in Section 1.5.11 of NUREG/CR-7193, "Evaluations of NRC Seismic-Structural Regulations and Regulatory Guidance, and Simulation-Evaluation Tools for Applicability to Small Modular Reactors (SMRs)," as an area of concern for deeply embedded SMRs such as the BWRX-300.

The seismic SSI analyses are performed using the sub-structuring method in Section 5.4 of ASCE/SEI 4, and the ACS SASSI (a system for analyses of SSI, see Appendix 3I) computer program to calculate the seismic response of the RB SSI system. The sub-structuring method allows the seismic response of the SSI system to be obtained by subdividing the problem into a series of simple subproblems that can be solved separately. Using the principle of superposition, the results of different sub-analyses are combined to obtain the final solution for the SSI problem. The solution for the seismic response of the integrated RB structures is obtained in the frequency domain for a selected set of frequencies and then interpolated for other frequency points.

The linear elastic SASSI analyses are performed on one-step structural models that accurately represent the geometry and dynamic properties of the integrated RB and its interaction with the subgrade. These structural models have a refined FE mesh that can transmit the entire frequency range of interest for the seismic design of the RB SSCs. These models assume isotropic elastic material properties of structural members and surrounding subgrade and neglect any non-linearity at the soil-structure contact interfaces.

The linear elastic assumption, based on the superposition principal, allows seismic demands obtained from the SASSI analyses to be combined with design demands obtained from other structural analyses in seismic design load combinations.

Far-field interaction nodes are established at the surface of each soil layer through the RB shaft embedment depth to calculate the horizontal and vertical components of the far-field motion in the SSI model. The responses calculated from these far-field interaction nodes are used to monitor the propagation of the input control motion through the RB embedment depth.

To account for the primary non-linearity of subgrade materials, a set of profiles of strain-compatible subgrade properties are used as input for the SSI analysis. These subgrade properties are compatible to the free-field strains generated by SSE level earthquake and are developed based on the results of equivalent linear probabilistic SRA which will be provided in CPA Part 2. The profiles of strain-compatible subgrade properties are presented Subsection

3.7.1.3. The uncertainties related to variation of subgrade properties will be addressed in CPA Part 2.

As described in Subsection 3.7.1.1.2, five sets of three input motion ATHs, corresponding to the three orthogonal directions of motion, are used as input for the seismic analyses to mitigate the uncertainty in the computed responses due to the phasing of the time history frequency components.

Input ground motion ATHs are applied to the SASSI model at the RB foundation bottom elevation as vertically propagating coherent:

- Shear waves for horizontal components of the input motion
- Compression waves for the vertical component of the input motion

The horizontal control motion is applied to the SASSI model in a manner that is consistent with the probabilistic One-Dimensional (1D) SRA approach which will be discussed in CPA Part 2. The requirement of USNRC DC/COL-ISG-017 for the ground motion used as input for the deterministic SSI analysis being consistent with the motions used for the probabilistic SRA is met as described in Subsection 3.7.2.4.1.

Uncertainties related to variations of the input SSI parameters and the wave propagation pattern through BECPV are addressed by results of sensitivity analyses performed as discussed in Subsection 3.7.2.9, following the guidance in Section 5.3 of NEDO-33914-A.

Design responses are obtained from the analysis of each of the three ground motion components for each of the five sets of ATHs using the methodology described in Subsections 3.7.2.5 and 3.7.2.6. The seismic design demands are calculated as the average of the results obtained from the analyses of the five sets of ATHs. Following the guidance of Section 5.3 of NEDO 33914-A, results of the sensitivity analyses discussed in Subsection 3.7.2.9 are included in the RB seismic design basis when significant exceedances are observed.

3.7.2.1.1 Frequencies of Analysis

The frequencies of analyses are determined following the guidelines of Section 5.3.2 of NEDO-33914-A.

Cutoff frequencies are established based on the largest value required by the following criteria:

1. Twice the highest dominant frequency of the coupled soil structure system, or
2. The highest structural frequency of interest, or
3. The frequency at which the Fourier amplitude of input motion has passed its peak value and has reached 10% of the peak value, and
4. 20 Hz

The value of cut-off frequency determined by the criteria described above is used for the analysis of the stiffest subgrade profile and models with UB structural stiffness properties. The analyses of softer subgrade profiles or reduced structural stiffness properties may use lower values for the cut-off frequency as per the guidance of Section 5.3.2 of NEDO-33914-A.

The frequencies of analysis are selected at sufficiently small frequency intervals. Transfer function amplitude results for responses at the key locations are inspected to detect any numerical anomalies in the interpolated transfer functions (e.g., sharp narrow spikes) that can potentially affect the accuracy of results. If present, the effects of these anomalies in the interpolated transfer function results are evaluated using additional frequencies of analysis to ensure the anomalies in the transfer function interpolations do not affect the accuracy of the calculated responses.

3.7.2.2 Natural Frequencies and Response Loads

Refer to Appendix 3C for the significant natural frequencies and SSE responses of the Seismic Category I structures at BECPV.

The seismic responses of the RB at BECPV are characterized using the results of the SSI analyses for responses at key structural locations and stress responses of key structural members selected per the guidance provided in Section 5.3.1 of NEDO-33914-A.

3.7.2.3 Procedures Used for Analytical Modeling

The 3D FE model used in the SSI analysis of the integrated RB meets the guidance of SRPs 3.7.1 and 3.7.2 and RG 1.61. The model also meets the structural modeling requirements of Section 3 of ASCE/SEI 4, and the one-step modeling requirements and guidelines of Section 5.1.1 of NEDO-33914-A.

Appendix 3B, to be provided in CPA Part 2, presents the details of the 3D FE model used in the seismic SSI analysis of the integrated RB.

The integrated RB structure is primarily constructed of Diaphragm Plate Steel-Plate Composite (DP-SC) modules. The DP-SC system has a configuration similar to the conventional steel-plate composite system except that the tie-rods in the conventional steel-plate composite system are replaced by diaphragm plates as described in Section 3.4 of NEDC-33926P-A, “BWRX-300 Steel-Plate Composite (SC) Containment Vessel (SCCV) and Reactor Building Structural Design,” (Reference 3.7-8).

3.7.2.3.1 Dynamic Finite Element Modeling of Integrated RB

The details of the integrated RB seismic SSI model, also known as the dynamic FE model, are provided in Appendix 3B.

In accordance with SRP 3.7.2, Section II.3.D and Section 3.4.2 of ASCE/SEI 4, the integrated RB structural FE model represents masses expected to be present at the time of the earthquake. These include 50% of the specified design live loads, 25% of the specified roof design snow loads and the inertia associated with the hydrodynamic effects of the fluids contained in various pools inside the RB.

The hydrodynamic effects that consist of the impulsive and convective (or sloshing) components are considered in accordance with Subsection II.1.A of SRP 3.7.2 and following the provisions of Section 3.6.3 of ASCE/SEI 4, and Chapter 5 of American Concrete Institute (ACI) 350.3, “Seismic Design of Liquid-Containing Concrete Structures and Commentary,” (Reference 3.7-9). The hydrodynamic mass is included in the model by:

- Distributing the horizontal impulsive fluid mass over the pool walls that are perpendicular to the direction of motion in accordance with the guidelines in ACI 350.3
- Lumping the entire vertical fluid mass on the pool slab bottom

The convective (sloshing) component of the hydrodynamic mass is not explicitly included in the global analysis model because its contribution is small and is associated with very low frequencies insignificant for the overall response. To account for the sloshing hydrodynamic effects, the design considers quasi-static sloshing pressure loads applied on the pool walls in accordance with Section 9.3 of ASCE/SEI 4.

In addition, the hydrodynamic effects of the breathing mode, associated with the liquid response due to vertical excitation, is considered by applying quasi-static pressure loads on the pool in accordance with Section 9.4 of ASCE/SEI 4.

Beam and shell elements are used to adequately represent the configuration of all main structural members in the integrated RB, with the steel-plate composite shear walls, slabs, and mat foundation being modeled using thick shell elements. The FE model includes gross discontinuities such as large openings and member eccentricity. Member eccentricity and offsets are accounted for by using rigid beam and shell elements or by appropriately adjusting the properties of the centerline modeled elements to account for the effect of large member offsets.

The integrated RB steel-plate composite structural elements are physically connected with welded connections. The FE model generally utilizes centerline shell element models of the steel-plate composite components where adjacent component elements share nodes at their intersecting lines. This modelling assumption effectively suggests rigid connection behavior.

Spring elements with stiffness properties appropriate to capture the interaction at the soil-structure interface connect the RB structural and subgrade FE models. The results obtained from the contact spring elements serve to:

- Calculate dynamic earth pressures on the below grade RB shaft exterior wall and mat foundation
- Determine whether separation between RB shaft wall and soils occurs under SSE loading as discussed in Subsection 3.7.2.9.4

The evaluation of effects of conditions at the contact interfaces with surrounding subgrade on the RB seismic response is discussed in Subsection 3.7.2.9.

The values of Young's modulus and Poisson's ratio representing the structural material stiffness properties are determined in accordance with the governing design codes in Section 3.8. Stiffness properties are assigned to the steel-plate composite structures in accordance with the guidance of SRP 3.7.2 and the requirements of Section 3.3.2 of ASCE/SEI 4.

The effective stiffness for analysis of the thick shell elements representing the steel-plate composite members is developed following the approach in Section 5.5 of NEDC-33926P-A. The stiffness calculations account for the expected state of stress and level of cracking for different loading conditions during normal operation and accident conditions.

The effects of variation of structural stiffness and damping properties is considered in the modeling of the integrated RB to ensure accuracy of the calculated seismic responses and seismic demands as discussed in Subsection 3.7.2.9.1.

The FE models used for seismic SSI analyses have a sufficiently refined mesh to be capable of transmitting the entire frequency range of interest for the seismic design of the RB SSCs. In accordance with the requirements of Section 5.3.4 of ASCE/SEI 4, the FE mesh is smaller than or equal to one-fifth of the smallest wavelength transmitted through the soil model, i.e., the maximum mesh size:

$$d_{max} \leq \frac{V_S}{5\, f_{cutoff}}$$

where: V_S is the shear wave velocity of the transmitting soil material; and

f_{cutoff} is the cutoff frequency of analysis determined as described in Subsection 3.7.2.1.1

The lower boundary of the SSI model is established at a distance that is deeper than at least two times the depth of the RB embedment and at least three times the largest foundation dimension from the bottom of the slab, meeting the guidance of SRP 3.7.2, Section II.4.

3.7.2.3.2 Dynamic Modeling of Subsystems, Components and Equipment

The dynamic properties of subsystems, components, and equipment are included in the integrated RB structural model based on the decoupling criteria of SRP 3.7.2, Section II.3.B as stated in NEDO-33914-A and Section 3.7 of ASCE/SEI 4. They depend on the ratios of the mass and first natural frequency of the subsystem, component, or equipment to those of the supporting structure.

To capture the dynamic coupling effects of the Reactor Pressure Vessel (RPV), the dynamic properties of the RPV and its components are represented by a Lumped Mass Stick (LMS) model capable of capturing the significant modes of the RPV seismic response. Procedures used to develop this LMS model are presented in Subsection 3.7.3.3.3. The RPV LMS model is connected to the RB structural model using local spring elements, representing the stiffness of the RPV support skirt and the horizontal stabilizers.

3.7.2.4 Soil-Structure Interaction

The SSI analyses of the Seismic Category I integrated RB are performed per the methodology presented in Subsection 3.7.2.1, for the range of subgrade conditions presented in Subsection 3.7.1.3. These subgrade conditions represent as-built conditions as stated in Subsection 3.7.1.1.1.

The foundation types utilized for all BECPB Power Block structures adjacent to the Reactor Building will be described in CPA Part 2.

To address the SSSI effects of the surrounding Power Block structures on the deeply embedded RB, SSI analyses are performed on a model combining the integrated RB FE model described in Subsection 3.7.2.3.1 with simplified models of the RWB, CB, TB, Service Building and Reactor Auxiliary structures representing the dynamic properties of these structures. The SSI analyses are performed for a range of as-built subgrade conditions. The combined FE model, input ground motions and subgrade properties used to evaluate the SSSI effects are provided in Appendix 3C. Note that the UB SSI analysis profiles are used in conjunction with the LB damping values, while the LB SSI analysis profiles are used in conjunction with the UB damping values.

Sensitivity studies performed to evaluate the effects of parameter variations on the integrated RB responses are discussed in Subsection 3.7.2.9. These sensitivity studies are performed using the SSI model with the adjacent Power Block structures.

The following are key requirements and approaches considered in the seismic SSI analyses to ensure the structural integrity and stability of the deeply embedded BWRX-300 RB throughout the life of the plant and to address specifics related to its design and construction.

3.7.2.4.1 Implementation of DC/COL-ISG-017 Guidance

BWRX-300 approaches for meeting USNRC DC/COL-ISG-017 guidance and addressing current limitations in USNRC DC/COL-ISG-017 related to the seismic analysis of deeply embedded structures, as identified in Section 1.5.8 of NUREG/CR-7193, are described in Section 5.3.4 of NEDO-33914-A.

The intent of USNRC DC/COL-ISG-017 is to ensure that the deterministic SSI analysis of the embedded RB structure uses ground motion inputs that are hazard consistent with the results of probabilistic SRA at the foundation bottom elevation and at ground surface.

The consistency between the free-field motion at the bottom of the RB foundation used as input for the deterministic SSI analysis and probabilistic SRA is checked as described in Subsection 3.7.1.1, using the procedure described in Section 5.3.4.1 of NEDO-33914-A.

The augmented and smoothed horizontal and vertical 5% damped spectra presented in Figure 3.7-3, to be provided in CPA Part 2, define the amplitude and frequency content of the SSI input control motion applied to the SSI model at the RB foundation bottom that is hazard consistent with the results of the probabilistic SRA.

3.7.2.4.2 Coupling of Soil and Structures

The seismic SSSI of the RB with the adjacent RWB, CB, TB, Service Building, and Reactor Auxiliary Structures is explicitly addressed in the seismic analysis and design as described in Subsection 3.7.2.4.

Coarse FE models representing the dynamic properties of the surrounding buildings and foundations are included in the combined FE model used for evaluating the SSSI effects of the adjacent Power Block structures on the RB. These models are sufficiently refined to capture the global modes of vibration of the RWB, CB, TB, Service Building and Reactor Auxiliary Structures with significant (> 20% per Section 5.3.7 of NEDO-33914-A) modal mass participations in the three orthogonal directions.

Subsection 3.7.2.8 presents the approach for addressing the requirements related to the seismic interaction of the RB with the surrounding RWB, CB, TB, Service Building and Reactor Auxiliary Structures and foundations.

3.7.2.5 Development of In-Structure Responses

In-Structure Response Spectra (ISRS) and ATHs are developed from the seismic analysis to serve as input for the seismic design and evaluation of subsystems, components, and equipment in accordance with Section II of SRP 3.7.3.

3.7.2.5.1 In-Structure Response Spectra

The development of the ISRS follows the guidance of Section II.5 of SRP 3.7.2, RG 1.122, “Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components,” and Section 6.2 of ASCE/SEI 4. Equipment structure interaction effects may be considered for the development of ISRS using the guidance in Section 5.3.6 of NEDO-33914-A.

ISRS are developed for required damping levels defining the amplitude and frequency content of in-structure design motion at different locations within the RB, in the two horizontal and the vertical directions.

If the equipment or component is supported by flexible slabs or attached to flexible walls, ISRS are developed considering additional nodal responses that capture the local effects of out-of-plane vibrations of the supporting slab or wall.

In accordance with the requirements of Section 6.2.1.1(a) and (b) of ASCE/SEI 4, the ISRS are developed from the calculated nodal in-structure responses by:

1. First combining in the time domain, the three co-direction responses due to the three orthogonal components of seismic input motion as an algebraic sum at each time step and then calculating the ARS of the combined ATHs, or
2. Combining the co-directional ARS results obtained from the analysis with the three orthogonal components of seismic input motion using the Square-Root-of-the-Sum-of-the-Squares (SRSS) method specified in Subsection 3.7.2.6.

The ISRS are developed at small frequency intervals to ensure they are sufficiently close to the peak response frequencies of the supporting structure. To meet the guidance of Section II.5 of SRP 3.7.2, the ISRS are calculated at 301 frequency points equally distributed on the logarithmic scale at the frequency range from 0.1 Hz to 100 Hz.

The ISRS are calculated as an envelope of the results from the seismic design basis SSI analysis of all subgrade profiles. Following the guidance of Section 5.3 of NEDO 33914-A, the enveloping ISRS are increased as described in Subsection 3.7.2.9 to address variations of important parameters. In accordance with RG 1.122 and Section 6.2.3 of ASCE/SEI 4, the peaks of the enveloping ARS are broadened by a minimum of +/-15% to address uncertainties related to the modeling of natural frequencies of the supporting structure and the SSI analysis methodology. The sharp valleys between peaks are filled to further account for the uncertainties in subgrade properties.

3.7.2.5.2 In-Structure Acceleration Time Histories

In accordance with the requirements of Section 6.3 of ASCE/SEI 4, time histories used in the analysis of subsystems are obtained either:

- Directly from the results of the SSI analysis as time histories of nodal responses at reference of subsystem support locations; or
- By generating synthetic time histories compatible to multi-damping ISRS developed as described in Subsection 3.7.2.5.1.

When obtained directly from the SSI analysis results:

- Time histories of the co-directional in-structure responses due to the three components of the SSI analysis input motion are combined in the time domain.
- Time histories are obtained from SSI analysis cases that are critical for the designed subsystem and include those obtained from BE soil case.
- Time shifting is applied to the time histories to address uncertainties related to the modeling of natural frequencies of supporting structure.

3.7.2.5.3 Relative Displacement

Relative displacement between different support points of subsystems with multiple or distributed supports are evaluated using the support displacement time histories.

The time history of the relative displacements corresponding to each SSI analysis is obtained by algebraic calculation of the difference in displacement time histories at the support locations. Directional combination of the support displacement time histories is carried out on a time-step-by-time-step basis or using the SRSS method as described in Subsection 3.7.2.6. Maximum design relative displacements are calculated as the envelope of the maximum relative displacements obtained for each SSI analysis case.

3.7.2.6 Three Components of Design Ground Motion

Earthquake motion is three dimensional and seismic design takes into account the effects of the three orthogonal components (two horizontal and one vertical) of the prescribed design earthquake.

The SSI analyses are performed separately for each of the three directional components of input ground motion using five sets of time histories as discussed in Subsection 3.7.1.1.2. For each set of time histories used as analysis input, the seismic response parameters obtained from the analysis of each of the three ground motion components are combined to get the total co-

directional response with either of the three methods permitted under Section 4.2.2 of ASCE/SEI 4, and RG 1.92, “Combining Modal Responses and Spatial Components in Seismic Response Analysis.”

1. The time histories of responses due to the three earthquake components are combined algebraically on the time-step-by-time-step approach.
2. The maximum co-directional responses can be combined using the 100-40-40 method.
3. The maximum responses due to the three earthquake components can be combined using SRSS method.

The absolute sum method used in time domain is implemented (e.g., for calculations of seismic demands for foundation bearing pressure and stability evaluations) as an alternative to performing the algebraic sum method for all possible combinations of the input motion directions.

3.7.2.7 Combination of Modal Responses

Modal combination is not utilized for the analysis of the Seismic Category I structures. These structures are evaluated using ACS SASSI FE models. ACS SASSI utilizes the frequency domain analysis method in which the equations of motion are solved for the selected range of frequencies defined in Subsection 3.7.2.1.1.

3.7.2.8 Interaction of Non-Seismic Category I Structures with Seismic Category I SSCs

To meet the interaction guidance of Section II.8 of SRP 3.7.2, evaluations are performed of the lateral load resisting system of the RWB, CB, TB, Service Building and Reactor Auxiliary Structures following the approach in Section 6.2 of NEDO-33914-A. These evaluations are based on seismic responses of RWB, CB, TB, Service Building and Reactor Auxiliary Structures obtained from the SSI analyses that incorporate the dynamic response of the RB and surrounding Power Block structures. The FE models of the RWB, CB, TB, Service Building and Reactor Auxiliary Structures included in the SSI combined model are refined sufficiently to provide accurate stress demands on the major lateral load resisting structural members and accurate seismic displacements in the direction of the adjacent RB.

The seismic interaction evaluations consider limited permanent deformations (Limit State LS-C) structural response to calculate SSE demands for the main lateral load resisting structural members following the guidance of Section 6.2 of NEDO-33914-A.

The gaps between the RB and adjacent structures are evaluated per guidance in Section 6.2 of NEDO-33914-A to ensure no physical interaction between the RB structure and surrounding structures. The gaps are evaluated along the entire height of the adjacent structures considering construction tolerances, inelastic deformations, and possible differential settlements.

3.7.2.9 Effects of Parameter Variations on Responses

Uncertainties in structural properties, damping values, subgrade properties, and variations of SSI parameters are accounted for in the analysis and design of BWRX-300 Seismic Category I structures in accordance with Section II.4 of SRP 3.7.2, Section 5.1 of ASCE/SEI 4, and following the guidance of Section 5.3 of NEDO-33914-A.

As stated in Subsection 3.7.2.4, the sensitivity analyses are performed using the SSI model with the adjacent Power Block structures. Evaluations of effects of parameter variations on floor responses are based on comparisons of key in-structure responses and stress demands at key locations selected as discussed in Subsection 3.7.2.2, obtained from the sensitivity analyses as described below.

For the results of the sensitivity analyses performed for the BWRX-300 at BECPV, refer to Appendix 3C.

3.7.2.9.1 Effects of Variation of Structural Stiffness and Damping Properties

Effective structural BE stiffness and lower OBE damping properties are assigned to the integrated RB FE model as discussed in Subsection 3.7.2.3. Effective stiffness assigned to steel-plate composite members considers the level of stress due to the most critical seismic load combinations following the recommendations in Section 5.3.5 of NEDO-33914-A.

The effects of structural stiffness variations are assessed by comparing key in-structure responses selected as described in Subsection 3.7.2.2 and stress demands obtained from the sensitivity analysis of the models with reduced stiffness properties with results of the design basis analysis performed on the model with effective stiffness properties. If the comparisons show significant exceedances (> 10%) in the RB seismic response due to the structural stiffness variations, the results of the sensitivity analysis are included in the RB seismic design basis.

3.7.2.9.2 Excavation Support, Backfill and Friction Effects

Excavation support and backfill effects are addressed following the guidelines of Section 5.3.8 of NEDO-33914-A.

The technique used for the RB shaft construction and corrosion allowance can also affect the friction at the interfaces between the RB exterior wall and the surrounding concrete fill material. To address the uncertainties related to the modeling of friction at the RB shaft interfaces, a sensitivity SSI analysis is performed by considering the no-friction conditions assuming no friction resistance between the RB exterior wall and the surrounding concrete fill following the guidance in Section 5.3.8 of NEDO-33914-A.

Results of these sensitivity analyses for key in-structure responses and stress demands selected as described in Subsection 3.7.2.2 are compared with the corresponding results of the design basis SSI analyses of the FE model that excludes the excavation support and assumes fully bonded conditions at the RB-subgrade interfaces. If the comparisons show significant exceedances (> 10%) in the RB seismic response due to the interaction with the excavation support or no-friction conditions, the results of these sensitivity analyses are included in the RB seismic design basis.

3.7.2.9.3 Groundwater Variation Effects

The potential effects of groundwater level variability on the seismic design of the RB are addressed following the guidelines of Section 5.3.10 of NEDO-33914-A.

The seismic design of the RB is based on the analysis of models that reflect fully saturated conditions for soil materials located below the nominal groundwater elevation. The potential effects of groundwater level variability on the seismic design are addressed by comparing the seismic responses obtained from the SSI analysis with fully saturated conditions with the responses obtained from the SSI analysis of a dry soil profile. The dry soil profile may use BE soil dynamic properties representative of the conditions of the LB groundwater level at BECPV. Alternatively, the groundwater may be conservatively considered to be located below the RB foundation bottom elevation.

If the comparisons show that the effects of groundwater variation exceed 10%, the design basis seismic demands are developed based on the fully saturated soil profile.

3.7.2.9.4 Soil Separation Effects

The SSI analysis of the RB addresses the uncertainties related to the inability of linear models used for the seismic analysis to explicitly represent the separation between the soil and the structure, following the guidance of Section 5.1.9(b) of ASCE/SEI 4, and Section 5.3.9 of NEDO-33914-A.

The extent of soil separation is assessed by comparing the maximum lateral earth pressure calculated from the seismic SSSI analysis of BE subgrade profile with a LB estimate of static earth pressures. The static lateral pressures calculated from static design SSI analysis with 1-g loading (see Subsection 3.8.4.1.4), are reduced by 10% to account for uncertainties in calculation of soil unit weights and surcharge loads. The regions where the static lateral pressure is lower than the seismic lateral pressure are considered separated in the model used for the sensitivity analysis.

The key in-structure responses selected as described in Subsection 3.7.2.2, and stress demands calculated from this sensitivity analysis are compared to the corresponding results of the SSI analysis of the model with BE properties representing fully bonded conditions. If the comparisons indicate that the seismic in-structure responses and stress demands from the model with soil separation exceed those obtained from the SSI analysis of fully bonded models by more than 10%, the results of the sensitivity analysis are included in the RB seismic design basis.

3.7.2.9.5 Effects of Non-Vertically Propagating Seismic Waves

The potential for non-vertically propagating seismic waves at BECPV is assessed following the guidelines in Section 5.3.3 of NEDO-33914-A, based on the geological and seismological conditions of the site. This investigation will be provided in CPA Part 2.

3.7.2.9.6 Non-Linear Sensitivity Evaluation

This evaluation will be provided in CPA Part 2.

3.7.2.9.7 Concrete Fill Strength

This evaluation will be provided in CPA Part 2.

3.7.2.10 Use of Equivalent Vertical Static Factors

Constant vertical static factors are not used in the design of the Seismic Category I structures. Vertical seismic loads are directly obtained from the results of SSI analyses performed using the one-step design analysis approach discussed in Subsection 3.7.2.1.

3.7.2.11 Methods Used to Account for Torsional Effects

Considerations are given in the modeling of the integrated RB structure to represent the actual locations of the center of masses and centers of rigidity of structural elements to account for torsional effects as discussed in Subsection 3.7.2.3.1.

In accordance with the requirements of Section 3.1 of ASCE/SEI 4, the seismic design of the RB structure also considers accidental torsion to account for:

- Non-vertically propagating seismic waves
- Rotational components of ground motion
- Distributions of structural mass and stiffness that differ from those represented in the 3D FE model used for the seismic response analysis

Following the guidance of Section II.11 of SRP 3.7.2, the accidental torsional moment demands are calculated at each floor level as the product of the story shear and 5% of the floor plan

dimension perpendicular to the story shear direction. Alternatively, the horizontal shear force demands on walls may conservatively be increased by 5% to account for the accidental torsion.

3.7.2.12 Comparison of Responses

Because only the time history method is used for the dynamic analysis of Seismic Category I structures, a comparison of responses with the response spectrum method is not applicable.

3.7.2.13 Analysis Procedure for Damping

Refer to Subsection 3.7.1.2 for damping values used in the SSI analysis of Seismic Category I SSCs and Subsection 3.7.2.1 for the seismic analysis method.

3.7.2.14 Determination of Seismic Overturning Moments and Sliding Forces, Structure to Soil Pressures and Frictional Forces for Seismic Category I Structures

Consistent with the guidance of Section II.14 of SRP 3.7.2, contact spring elements installed in the one-step SSI models at interfaces between the structure and the subgrade are used for calculation of dynamic bearing pressure demands, seismic driving forces and overturning moments on the Seismic Category I common mat foundation. Uncertainties related to friction at the RB shaft interfaces are addressed as described in Subsection 3.7.2.9.2.

As described in Subsection 3.7.2.6, time histories of the horizontal and vertical seismic forces in the three directions are calculated as the algebraic sum of the forces in the three directions at each step for all contact spring elements. Overturning moments about the two horizontal axes are calculated as the algebraic sum of the moments resulting from each spring force with respect to the foundation bottom centerline.

The seismic inertia forces and overturning moments for the foundation stability evaluations and seismic bearing pressure calculations may be obtained from SSI models with higher SSE structural damping values.

The RB seismic bearing pressure demands are also calculated in the time domain. Maximum bearing pressure values are calculated for the total duration of earthquake for each of the five sets of ATHs used as input for the SSI analysis discussed in Subsection 3.7.1.1.2. The dynamic bearing pressure demand under the RB foundation is defined as the average of the results obtained from the five sets of ATHs.

3.7.3 Seismic Subsystem Analysis

This section focuses on the seismic qualification by analysis of the BWRX-300 Seismic Category I subsystems which consists of:

- Distribution systems, including piping and supports, cable trays and supports, heating, ventilation and air conditioning ductwork and supports, conduits and supports,
- Equipment supports,
- Intervening structural elements between distribution systems and equipment supports and the RB structure, and
- RPV and internals.

Input motions for the qualification of these subsystems are usually in the form of floor response spectra or ATHs obtained from the integrated RB seismic SSI analysis discussed in Subsection 3.7.2. Input motions in terms of acceleration time histories are generally used. For other methods used in the seismic and dynamic qualifications of subsystems, refer to Section 3.10.

3.7.3.1 Seismic Analysis Methods

Seismic analysis of subsystems is performed using one of the following methods in accordance with SRP 3.7.2, Section II.1:

- Time history analysis
- Response spectrum analysis
- Static coefficient

3.7.3.1.1 Time History Analysis

Assuming velocity proportional damping, the dynamic equilibrium equations for a lumped mass, distributed stiffness system are expressed in matrix form as:

$$[M] \{\ddot{u}(t)\} + [C] \{\dot{u}(t)\} + [K] \{u(t)\} = \{P(t)\}$$

Where:

$\{u(t)\}$ = time dependent displacement of nonsupport points relative to the supports.

$\{\dot{u}(t)\}$ = time dependent velocity of nonsupport points relative to the supports.

$\{\ddot{u}(t)\}$ = time dependent acceleration of nonsupport points relative to the supports.

$[M]$ = mass matrix.

$[C]$ = damping matrix.

$[K]$ = stiffness matrix.

$\{P(t)\}$ = time dependent applied force column vector.

The above equation is solved by modal superposition or direct integration in the time domain.

Modal superposition involves two steps. First, the characteristic equation corresponding to undamped, free vibration of the model is solved to obtain the eigenvalues, eigenvectors, and generalized masses. The system coupled equations are then decoupled via the eigenvector transformation matrix which is simply the matrix of eigenvectors written as columns. The equations are decoupled in the generalized coordinate system because of the orthogonality of the matrix of eigenvectors with respect to the “weighted” mass and stiffness matrices. The decoupled modal equations are then solved independently to obtain the generalized coordinates. The physical solution is then given by the eigen transformation once the generalized coordinates are known.

The direct integration method involves the numerical integration of the simultaneous differential equations of equilibrium in their original form, without transformation to the generalized coordinates. For systems subjected to short duration, high frequency excitation, the direct integration method requires less computation and is recommended over the modal superposition method.

For the time domain solution, the numerical integration time step is sufficiently small to accurately define the dynamic excitation and to render stability and convergence of the solution up to the highest frequency (or shortest period) of significance. This condition is satisfied if Δt is selected to limit the amplitude decay per cycle of free vibration of the highest significant mode to less than 20%. This corresponds to approximately 3.5% numerical damping for that highest significant mode. The integration time step for both the direct numerical integration of the system coupled equations of motion and the numerical integration of the n decoupled equations (modal superposition) satisfies the following requirement:

$$\Delta t \leq T_m/10$$

where Δt is the numerical integration time step magnitude and T_m is the period of the highest significant mode considered in the analysis or the reciprocal of the cutoff frequency in Hz as defined in Subsection 3.7.3.4.

3.7.3.1.2 Response Spectrum Analysis

This method is used if only peak dynamic responses are required.

The response spectrum method is a modal superposition analysis in which only the peak values of the solution of the decoupled modal equations are obtained. The method is based on writing the solution of each decoupled modal equation in terms of the convolution integral. The major advantage of this form of solution is that for a given input motion the only variables under the integral are the damping factor and the frequency. Thus, for a specified damping factor, it is possible to construct a curve which gives the maximum value of the integral as a function of frequency. This curve is called a response spectrum for the particular input motion and the specified damping factor. The integral has units of velocity, consequently the maximum of the integral is called the spectral velocity.

For a subsystem analysis of a secondary system the input floor response spectra, obtained from a time history analysis of the primary system, is broadened ±15% to account for modeling uncertainties in both the primary and secondary systems in accordance with RG 1.122 and Section 6.2.3 of ASCE/SEI 4.

Using the calculated natural frequencies of vibration of the system, the maximum values of the modal responses are determined directly from the appropriate response spectrum. The modal maxima are then combined as discussed in Subsection 3.7.3.7.

3.7.3.1.3 Static Coefficient

The static coefficient method may be applied to certain equipment in lieu of the required dynamic analysis. Response loads are determined statically by multiplying the equipment mass by a static coefficient equal to 1.5 times the maximum spectral acceleration in accordance with SRP 3.7.2, Section II.1.B. This coefficient is intended to account for the effect of both multi-frequency excitation and multi-mode response. This method is applicable only to equipment corresponding to a simple column, beam, or frame type structure supported at a single point. Justification is required for applying this method or coefficient to equipment having configurations other than simple frame or beam type structures.

A factor of less than 1.5 may also be used if adequate justification is provided in accordance with SRP 3.7.2, Section II.1.B. For example, if the equipment is simple enough such that it behaves essentially as a single degree-of-freedom model and the fundamental frequency is greater than the seismic excitation frequency, the factor 1.0 can be used instead of 1.5.

If the fundamental frequency of the equipment is greater than the cutoff frequency but less than the Zero Period Acceleration (ZPA) frequency, the static coefficient can be taken as 1.5 times the peak spectral acceleration which occurs between the cutoff frequency and the ZPA frequency in the equipment input response spectra.

3.7.3.1.4 Seismic Analysis of Piping Systems

The time history and the response spectrum methods are utilized in the piping analysis as required. The procedure for multi-support excitation described in Subsection 3.7.3.9 is followed with both methods. When the multi-support response spectrum method is used to calculate the dynamic response of the piping system, all multi-support response spectra components are simultaneously applied to each piping model for each load case.

Secondary stresses due to anchor displacements may be calculated based on total anchor displacements or on a mode-by-mode basis. As a conservative approach, the maximum displacement of each support point can be computed by either time history analysis or response spectrum analysis for the supporting RB Structures. The computed individual maximum support displacements are then imposed on the supported piping and equipment systems in the most limiting, or severe, directional combination for the piping and equipment design verification consideration.

The dynamic loads on pipe mounted pumps and valves are determined by dynamic analysis of the piping system including the pumps and valves in the analytical model. If a pump or valve has a fundamental frequency greater than the cutoff frequency described in Subsection 3.7.3.4, it may be modeled as a rigid body. The pump or valve fundamental frequency is calculated for the equipment alone when it is rigidly supported at the nozzle where it interfaces with the piping system.

3.7.3.1.5 Seismic Analysis of Equipment

The time history and response spectrum methods are also utilized in the equipment analysis as required. When the equipment is supported at two or more points located at different elevations in the building, the response spectrum for the most severe single point of attachment is chosen as the design spectra. Alternatively, the multi-support excitation procedure described in Subsection 3.7.3.9 is used.

The relative displacement between supports is determined from the dynamic analysis of the structure. The relative support-point displacements are used for static analysis to determine the additional stresses due to support displacements. As discussed in Subsection 3.7.3.1.3, the static coefficient method may be used in lieu of the required dynamic analysis for rigid equipment and frame type structures supported at a single point.

Seismic qualification of equipment by testing is discussed in Section 3.10.

3.7.3.1.6 Seismic Analysis of the Reactor Pressure Vessel and Internals

To capture the interaction of the integrated RB structures with the RPV and its internals components, such as fuel, guide tubes, and Control Rod Drive (CRD) System housing, a LMS model representing the RPV dynamic properties is included in the integrated RB model as discussed in Subsections 3.7.2.3.2 and 3.7.3.3.3.

The dynamic responses required as input for the design verification of RPV and internals components are generated directly in the dynamic analysis of the RB. Seismic acceleration time histories for the design of the RPV and internals are applied to the RPV and internals model simultaneously in the three orthogonal directions. Alternatively, two analyses can be performed to calculate the horizontal and vertical responses of the RPV separately.

The evaluation of RPV internals components in the horizontal direction is performed using a two-step analysis approach, where seismic accelerations at RPV support locations are applied to more detailed FE models of the RPV and internals. The first step of the two-step analysis consists of obtaining ATHs developed as described in Subsection 3.7.2.5 at the RPV/RB interface locations from the RB SSI analyses discussed in Subsection 3.7.2. The second step is a multi-support excitation time history analysis of the RPV, and internals subjected to the ATHs generated in the first step. The procedure for multi-support excitation time history analysis, as described in Subsection 3.7.3.9, is followed in the second step analysis of the RPV and internals. In accordance with the requirements of Section 6.3.2 of ASCE/SEI 4, time step of the input time histories is varied to address uncertainties related to the modeling of natural frequencies of the supporting structure and the SSI analysis methodology as stated in Subsection 3.7.2.5.2.

3.7.3.2 Determination of Number of Earthquake Cycles

The determination of the number of earthquake cycles for subsystem analysis is in accordance with Section II.2 of SRP 3.7.3, “Seismic Subsystem Analysis Review Responsibilities.”

3.7.3.3 Procedures Used for Analytical Modeling

The mathematical models for Seismic Category I components are developed to realistically reflect the dynamic characteristics of the components.

To do so, each component is discretized into a series of interconnected beam elements or FEs. The node points are generally selected to coincide with the locations of large masses, such as at structure floors or at heavy equipment supports, and at points corresponding to any significant change in physical geometry.

The number of mass node points in the model is sufficient if additional node points (independent of number) do not result in more than 10% increase in the responses in the frequency range below the cutoff frequency specified in Subsection 3.7.3.4.

Following the guidance of SRP 3.7.2 and 3.7.3, the element mesh size is selected on the basis that further refinement only has a negligible effect on the solution results.

3.7.3.3.1 Modeling of Piping Systems

The continuous Seismic Category I piping systems are modeled as an assemblage of pipe elements (e.g., straight sections, elbows, and bends) supported by hangers and anchors, and restrained by pipe guides, struts, and snubbers. Pipe and fluid masses are lumped at the nodes and connected by the weightless elastic beam elements which reflect the physical properties of the corresponding piping segment. The mass node points are selected to coincide with the locations of large masses, such as valves, pumps, and motors, and with locations of significant geometry change. Concentrated weights on the piping system, such as the valves, pumps, and motors, are modeled as lumped mass rigid systems if their fundamental frequencies are greater than the cutoff frequency in Subsection 3.7.3.4. The torsional effects of valve operators and other equipment with off-set center of gravity with respect to the piping centerline are included in the analytical model. The pipe length between mass points is no greater than the length with a fundamental frequency equal to the cutoff frequency stipulated in Subsection 3.7.3.4 when calculated as a simply supported beam with uniformly distributed mass.

All pipe guides and snubbers are modeled to produce representative stiffness to reduce modeling uncertainties. Snubbers are modeled with an equivalent stiffness based on dynamic tests or on data provided from the vendor. The stiffness of the supporting structures is included in the analysis unless the supporting structure is shown to be rigid.

3.7.3.3.2 Modeling of Equipment

For dynamic analysis, Seismic Category I equipment is represented by lumped mass systems which consist of discrete masses connected by weightless beam elements and/or by any other appropriate FE representation. The criteria used to lump the masses are:

- The number of modes of dynamic system is controlled by the number of masses used. Therefore, the number of masses is chosen so that all significant modes are included. The modes are considered as significant if the corresponding natural frequencies are less than the cutoff frequency specified in Subsection 3.7.3.4.
- Mass is lumped at any point where a significant concentrated weight is located.
- For equipment with a free-end overhang span whose flexibility is significant compared to the center span, a mass is lumped at the overhang span.

- When a mass is lumped between two supports, it is located at a point where the maximum displacement is expected to occur. This tends to conservatively lower the natural frequencies of the equipment. Similarly, in the case of live loads and a variable support stiffness, the location of the load and the magnitude of support stiffness are chosen to yield the lowest frequency content for the system.

3.7.3.3.3 Modeling of Reactor Pressure Vessel and Internals

Because of the significant dynamic interaction between the RB and RPV and internals, the latter are integrated into the RB model as discussed in Subsection 3.7.2.3 and Appendix 3B.

The mathematical model of the RPV and internals consists of a LMS model connected by linear elastic members and 3D FE models. Using the elastic properties of the structural components, the stiffness properties of the model are determined. This includes the effects of both bending and shear.

To facilitate hydrodynamic mass calculations, mass points (e.g., representing the fuel, shroud, vessel) are selected at the same elevation. The various lengths of CRD housings are grouped into two representative lengths. These lengths represent the longest and shortest housings to adequately represent the full range of frequency response of the housings. To reduce the complexity of the dynamic model, the light components (such as incore guide tubes and housing, sparger, and their supply headers) are excluded from the RPV mathematical model. However, the dynamic response of selected components is determined from a subsystem analysis after the system response is found.

Dynamic effects of water enclosed by the RPV are accounted for by introduction of a hydrodynamic mass matrix, which serves to link the acceleration terms of the equations of motion of points at the same elevation in concentric cylinders with a fluid entrapped in the annulus. Although the dynamic coupling between the vertical hydrodynamic masses is not considered, the vertical hydrodynamic masses themselves are properly accounted for. Dynamic loads due to vertical motion are added to, or subtracted from, the static weight of component, whichever is more conservative.

The shroud support plate is modeled as a rigid link in the translational direction because it is loaded in its own plane during a horizontal dynamic event. The shroud support legs, and the local flexibilities of the vessel and shroud contribute to the rotational flexibilities and are modeled as an equivalent torsional spring.

Due to the small clearances in the horizontal directions, the fuel assembly is adequately modeled as a linear system for subsystem and system analysis. In the vertical direction, the fuel assembly has the potential to lift off from its seat and a non-linear representation is required if the vertical applied and reaction forces are sufficient to cause fuel lift. Furthermore, the interface between the fuel channel and lower tie plate is not rigid and a non-linear model to account for slippage may be appropriate.

The weight of asymmetric secondary components, such as attached equipment, is uniformly redistributed around the node point circle. Asymmetric equipment is modeled using FE or LMS methods.

3.7.3.4 Basis for Selection of Frequencies

The cutoff frequency selected in the time history and response spectrum analyses ensures that all significant modes are included in the superposition. Higher modes which cumulatively contribute less than 10% of the total system response are not considered in the superposition of the individual modal values.

The cutoff frequency for seismic and other dynamic loads follows Subsection 3.7.2.1.1. For seismic load, modes up to 100 Hz are included. For other dynamic analysis, the cutoff frequency is 100 Hz, as long as no more than 5% of the total strain energy of the system remains beyond this cutoff frequency.

Where practical, to avoid adverse resonance effects, equipment and components are designed/selected such that their fundamental frequencies are approximately less than half or more than twice the dominant frequencies of the support structure following the guidance of Subsection II.4 of SRP 3.7.3. Moreover, the equipment is analyzed or tested or both to demonstrate that it is adequately designed for the applicable loads considering both its fundamental frequency and the forcing frequency of the applicable support structure.

3.7.3.5 Analysis Procedure for Damping

3.7.3.5.1 RPV and Internals Damping

Damping values for seismic analysis of the RPV and internals using the modal superposition method are presented in Table 3.7-6. These damping values are in accordance with ASCE/SEI 43 and are consistent with RG 1.61, with the CRD damping values exceeding the guidance for a welded steel construction. Fuel damping values correspond to bolted steel with bearing connections from RG 1.61, with additional conservatism added for consistency with previous BWR designs, including the ESBWR approved in NUREG-1966.

The Rayleigh damping is used for FE analysis of the RPV and internals when using the direct integration method.

3.7.3.5.2 Damping for Piping, Equipment and Equipment Support

Damping coefficients used in the seismic analysis of Seismic Category I piping, equipment, equipment supports and intermediate structures between subsystems are presented in Table 3.7-7. These damping values are consistent with RG 1.61.

3.7.3.6 Three Components of Design Ground Motion

Applicable methods for spatial combination of responses due to each of the three input motion components are described in Subsection 3.7.2.6.

3.7.3.7 Combination of Modal Responses

Modal combination procedures are in accordance with Section II.7 of SRP 3.7.2 and RG 1.92.

3.7.3.8 Interaction of Non-Seismic Category I Subsystems with Seismic Category I SSCs

The methodology used for evaluation and design of Non-Seismic Category I subsystems interacting with Seismic Category I SSCs is consistent with Section II.8 of SRP 3.7.3.

3.7.3.9 Multiple-Supported Equipment and Components with Distinct Inputs

The BWRX-300 approach for analyzing equipment and components supported at several points by either a single structure or two separate structures conforms with the guidance of Section II.9 of SRP 3.7.3.

The time history direct integration, time history modal superposition and response spectrum modal superposition methods discussed in Subsection 3.7.3.1 can be used in multi-support excitation analysis. However, the mode superposition procedure described in Section 3.7.3.1 for an applied load vector is replaced with the corresponding mode superposition procedure for multi-support excitation analysis.

When using the time history or response spectrum methods, the following methods are acceptable:

1. The uniform support motion method, where uniform time histories corresponding to the envelopes of all individual ISRS at the various support locations are applied at each attachment point simultaneously.
2. The independent support motion method, where time histories corresponding to the envelopes of the ISRS for each attachment point in each of the three directions are applied at each corresponding attachment point simultaneously.

The above time history methods of analysis are performed such that inertial (primary) and static stresses (secondary) due to differential displacements are separated. The inertial forces are used for primary stress calculations. For closely spaced modes of the supporting structure, the response is calculated as discussed in Subsection 3.7.3.7. The total secondary stress for triaxial excitation is computed as the SRSS of the resultant secondary stresses for each excitation direction. The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) Section III, “Rules for Construction of Nuclear Facility Components,” (Reference 3.7-10) requires that the secondary stresses must be combined with the primary stress.

Using the response spectrum method, the support points response spectra are generated from support point acceleration time histories with ±15% peak broadening to account for the RB support structure modeling uncertainties in accordance with RG 1.122 and Section 6.2.3 of ASCE/SEI 4. For uncorrelated dynamic inputs, the use of the SRSS method to combine modal responses is appropriate because the modal excitation maximum responses do not occur simultaneously and therefore are statically uncorrelated. For situation of high modal density, i.e., when the significant modes are “closely spaced”, the modal responses could be highly correlated and the SRSS superposition may yield unconservative responses. In this case, the modal responses are initially combined algebraically, then combined with the uncorrelated responses using the SRSS method.

3.7.3.10 Use of Equivalent Vertical Static Factors

Equivalent vertical static factors are used when the requirements for the static coefficient method in Subsection 3.7.3.1.3 are satisfied.

3.7.3.11 Torsional Effects of Eccentric Masses

Torsional effects of eccentric masses are considered in the modeling of subsystems as discussed in Subsection 3.7.3.3.

3.7.3.12 Seismic Category I Buried Piping, Conduits and Tunnels

The BWRX-300 design does not include Seismic Category I buried pipes, conduits, or tunnels.

3.7.3.13 Methods for Seismic Analysis of Seismic Category I Concrete Dams

The design does not include nor require the presence of a dam.

3.7.3.14 Methods for Seismic Analysis of Above-Ground Tanks

The BWRX-300 design does not include Seismic Category I above-ground tanks.

3.7.3.15 Effect of Differential Building Movements

In most cases, subsystems are anchored and restrained to floors and walls of buildings that may have differential movements during a seismic event.

Differential endpoint or restraint deflections cause forces and moments to be induced in the system. As discussed in Subsection 3.7.3.9, the stress thus produced is a secondary stress. It is justifiable to place this stress, which results from restraint of free-end displacement of the system, in the secondary stress category because the stresses are self-limiting and, when the stresses exceed yield strength, minor distortions or deformations within the system satisfy the condition which caused the stress to occur.

Refer to Subsection 3.7.2.8 for the methodology used to obtain differential displacements used in the evaluation of subsystems.

3.7.4 Seismic Instrumentation

As required in 10 CFR 50, Appendix S, instrumentation is provided so that the seismic response of safety class SSCs can be evaluated promptly after an earthquake.

The BWRX-300 seismic instrumentation system meets the relevant requirements of GDC 2, 10 CFR 50.65, 10 CFR 50, Appendix S, and 10 CFR 50.55a “Codes and Standards and is consistent with SRP 3.7.4, “Seismic Instrumentation,” and RG 1.12, “Nuclear Power Plant Instrumentation for Earthquakes.” The instrumentation used for the measurements is capable of recording the effects produced by the most severe earthquakes that have been historically reported for the unique site considered and surrounding area, with sufficient margin for the limited accuracy, quantity and period of time in which historical data has been accumulated.

Provisions for inservice testing, remote alarm, recording capabilities, dynamic range and sampling rate with a low and adjustable actuating level or trigger, and data processing are in accordance with RG 1.12 and RG 1.166, “Pre-Earthquake Planning, Shutdown, and Restart of a Nuclear Power Plant Following an Earthquake.”

3.7.4.1 Comparison with Regulatory Guide 1.12

The seismic instrumentation program described in the following subsections is consistent with RG 1.12.

3.7.4.2 Location and Description of Instrumentation

3.7.4.2.1 Free-Field Instrumentation

The location of the free-field instrumentation follows the guidance of Section C.1.2 of RG 1.12, and include:

- At least two triaxial accelerometers installed outside of the structure-ground interaction influence of the Power Block, but as close as practicable to the RB following the guidance of Section C.1.3.1 of RG 1.12.
- At least one free-field downhole accelerometer installed at the depth of bottom of the RB foundation, below the free-field accelerometer at finished grade level, because the deeply embedded RB is founded at a depth more than 12 m (40 ft) below finished grade elevation.

3.7.4.2.2 Structure and Component Instrumentation

In accordance with the requirements of Section C.1.2 of RG 1.12, triaxial accelerometers are installed at several locations inside the RB:

1. At least one accelerometer at the top of the RB base slab
2. One accelerometer on the containment internal structure close to the reactor vessel
3. One accelerometer in the RB close to the top of the containment internal structure

4. One accelerometer close to the top of the containment structure
5. One accelerometer at the top of the operating floor elevation

Triaxial accelerometers are installed at three locations outside the RB deemed important in accordance with Section C.1.2 of RG 1.12.

The specific locations for instrumentation are determined to obtain the most pertinent information consistent with the selected key locations in the RB model to enable comparison between the measured and calculated in-structure responses in accordance with Section C.1.3.2 of RG 1.12.

Most of the instrumentation stations are configured to be accessible for maintenance during full-power operation in compliance with the guidance of Section C.4.3 of RG 1.12. For sensors installed in inaccessible areas, provisions for data recording, external remote alarm indicating actuation and remote inservice inspections are provided.

3.7.4.2.3 Recording and Playback Equipment

Recording and playback units are provided for multiple channel recording and playback of the triaxial accelerometer signals. Characteristics of the recording and playback equipment follow the guidelines in Section C.4 of RG 1.12.

3.7.4.3 Control Room Operator Notification

Activation of the seismic trigger causes an audible and visual annunciation in the two control rooms to alert the plant operator that a felt earthquake has occurred in accordance with Section C.7 of RG 1.12.

For other control room design requirements, refer to Section 6.4.

3.7.4.4 Comparison of Measured and Predicted Responses and Conformance with RG 1.166

The BWRX-300 pre-earthquake planning, and post-earthquake actions are in accordance with RG 1.166, including clarifications, and American National Standards Institute (ANSI)/American Nuclear Society (ANS)-2.23, “Nuclear Power Plant Response to an Earthquake,” (Reference 3.7-11) endorsed by RG 1.166.

Following a seismic event, operator actions and operator walkdown inspections are performed to assess the severity of the earthquake following the provisions of Section 6 of ANSI/ANS-2.23. The data from the seismic instrumentation, coupled with information obtained from a plant walkdown, is used to make the initial determination of whether the plant should be shut down, if it has not already been shut down by operational perturbations resulting from the seismic event. The plant is shut down if the walkdown inspections discover damage to equipment that would affect the safe operation of the plant, or if the OBE exceedance criteria determined in accordance with Section 6.4.1 of ANSI/ANS-2.23, are met.

Following plant shutdown, post-shutdown inspections and tests are performed in accordance with Sections 7.1, 7.2, 8.1 and 8.2 of ANSI/ANS-2.23 to determine the physical condition of the plant and its readiness to resume operation. Long term evaluations are performed following the requirements and guidelines of Section 9 of ANSI/ANS-2.23. After plant is restarted (or prior to restart if the earthquake caused significant damage to the plant), long-term evaluations are carried out for engineering assessments of plant structures and equipment using the actual event records to assure their long-term reliability. Results of post-shutdown inspections and tests are documented and are available for inspection prior to restart.

3.7.4.5 Design, Installation, and Inservice Surveillance

The seismic instrumentation is expected to operate during all modes of plant operation including periods of plant shutdown. The design, selection, and location of the BWRX-300 seismic instrumentation includes consideration for long service life, ease and low frequency of maintenance and calibration.

The instrumentation is designed and installed following the guidance of Sections C.4 and C.5 of RG 1.12.

Prior to installation, the operational reliability of the seismic monitoring instrumentation is demonstrated in accordance with Section C.4.7 of RG. 1.12.

Maintenance, testing and administrative procedures for seismic instrumentation defined in accordance with Section C.9 of RG 1.12 are documented before the first facility start-up, and updated as necessary following any modification to the system. Following the guidance of Section C.3 of RG 1.12, the seismic instrumentation system components are maintained and tested to ensure that a maximum number of instruments are kept in service during plant operation and shutdown. The maintenance program addresses technical, testing, and administrative procedures.

The guidance in Appendix A to RG 1.166 is followed if the instrumentation is discovered to have been out of service during an earthquake.

3.7.4.6 Comparison with USNRC RG 8.8

Acceleration sensors are installed such that occupational radiation exposures associated with their location, installation, and maintenance are maintained as Low as is reasonably achievable consistence with 10 CFR 20.1101(b) and RG 8.8, “Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable,” and as discussed in Section 12.3.

3.7.5 References

3.7-1 NEDO-33914-A, “BWRX-300 Advanced Civil Construction and Design Approach,” Revision 2, GE-Hitachi Nuclear Energy Americas, LLC, June 2022.

3.7-2 IBC, “International Building Code,” International Code Council, 2021.

3.7-3 ASCE/SEI 43-19, “Seismic Design Criteria or Structures, Systems, and Components in Nuclear Facilities,” American Society of Civil Engineers, 2019.

3.7-4 To be provided in CPA Part 2

3.7-5 To be provided in CPA Part 2

3.7-6 To be provided in CPA Part 2

3.7-7 ASCE/SEI 4-16, “Seismic Design of Safety-Related Nuclear Structures,” American Society of Civil Engineers, 2017.

3.7-8 NEDC-33926P-A, “BWRX-300 Steel-Plate Composite (SC) Containment Vessel (SCCV) and Reactor Building Structural Design,” GE Vernova Hitachi Energy Americas, LLC, Revision 4, December 2025.

3.7-9 ACI 350.3-06, “Seismic Design of Liquid-Containing Concrete Structures and Commentary,” American Concrete Institute, 2006.

3.7-10 ASME BPVC, Section III, “Rules for Construction of Nuclear Facility Components,” Edition 2021, American Society of Mechanical Engineers ASME BPVC Code Section III (2021 Edition).

3.7-11 ANSI/ANS-2.23, “Nuclear Power Plant Response to an Earthquake,” American National Standards Institute, Inc., American Nuclear Society, 2016.

3.7-12 ANSI/AISC N690-18, “Specification for Safety-Related Steel Structures for Nuclear Facilities,” American National Standards Institute/American Institute of Steel Construction, 2018.

3.7-13 ACI 349-13, “Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary,” American Concrete Institute, 2014.

Table 3.7-1: Seismic Categories and Design Basis of BWRX-300 Structures

Structure	Seismic Category/Evaluation	Design/Evaluation Basis	Design Basis Earthquake (1)	Limit State (2)
SCCV and Containment Steel Structures	Seismic Category I	ASCE/SEI 43 and ASCE/SEI 4 for analysis ASME BPVC and NEDC-33926P-A for design	SSE	LS-D
Containment Internal Structures	Seismic Category I	ASCE/SEI 43 and ASCE/SEI 4 for analysis ANSI/AISC N690 (Reference 3.7-12) and NEDC-33926P-A for design	SSE	LS-D
RB Steel-Plate Composite and Steel Structures	Seismic Category I			
RWB Structure	Seismic Category RW-IIa	RG 1.143, ASCE/SEI 43 and ASCE/SEI 4 for analysis ACI 349 (Reference 3.7-13) and ANSI/AISC N690 for design	½ SSE (3)	LS-D
	To meet II/I Seismic Interaction requirements (4)	ASCE/SEI 43 and ASCE/SEI 4	SSE	LS-C
CB Structure	Seismic Category II	IBC		
	To meet II/I Seismic Interaction requirements (4)	ASCE/SEI 43 and ASCE/SEI 4	SSE	LS-C

Structure	Seismic Category/Evaluation	Design/Evaluation Basis	Design Basis Earthquake (1)	Limit State (2)
TB Structure	Seismic Category II	IBC		
	To meet II/I Seismic Interaction requirements (4)	ASCE/SEI 43 and ASCE/SEI 4	SSE	LS-C
Service Building Structure	Seismic Category II	IBC		
	To meet II/I Seismic Interaction requirements (4)	ASCE/SEI 43 and ASCE/SEI 4	SSE	LS-C
Reactor Auxiliary Structures	Seismic Category II	IBC		
	To meet II/I Seismic Interaction requirements (4)	ASCE/SEI 43 and ASCE/SEI 4	SSE	LS-C
Equipment Structures	Non-Seismic	IBC		

Notes:

1. SSE is defined in Subsection 3.7.1.
2. Limit States per ASCE/SEI 43:
   - LS-D essentially elastic response
   - LS-C response with limited permanent deformations
3. The RWB is designed in accordance with the radioactive waste management requirements for Category RW-IIa from RG 1.143.
4. Proximity of this building to the Seismic Category I RB structure requires seismic interaction evaluation to SSE.

Table 3.7-2: Selected Seed Time History Records

To be provided in CPA Part 2.

Table 3.7-3: Ground Motion Parameters of the Matched Time Histories

To be provided in CPA Part 2.

Table 3.7-4: Correlation Coefficients Between Pairs of Modified Time Histories

To be provided in CPA Part 2.

Table 3.7-5: Seismic Damping Values for BWRX-300 Structures

Material	Critical Damping	
	OBE (%)	SSE (%)
Steel-plate composite structures	3	5
Welded and Friction-bolted steel structures	3	4
Bearing-bolted steel structures	5	7
Reinforced concrete structures	4	7

Table 3.7-6: Seismic Damping Values for Reactor Pressure Vessel and Internals

Component	Critical Damping	
	OBE (%)	SSE (%)
Reactor Vessel	2	4
Vessel Support Skirt	2	4
Shroud	2	4
Shroud Support Spring	2	-
Shroud Head & Separator	2	4
Fuel	4	6
CRD Guide Tubes	1	2
CRD Housing	1	2
CRD Restraint Springs	2 (vertical direction only)	-
Stabilizer and Bellows	2 (vertical direction only)	-

Table 3.7-7: Seismic Damping Values for Piping and Equipment

Structure or Component	OBE Damping	SSE Damping
Equipment and piping system	3	4

Figure 3.7-1: Scale Factors for Adjusting the Foundation Input Response Spectra (FIRS)

To be provided in CPA Part 2.

Figure 3.7-2: Nuclear Energy Institute (NEI) Check Results for the Ground Motion Response Spectra

To be provided in CPA Part 2.

Figure 3.7-3: Augmented Safe Shutdown Earthquake Ground Motion Input Response Spectra at Bottom of RB Foundation

To be provided in CPA Part 2.

Figure 3.7-4: Safe Shutdown Earthquake Ground Motion Input Response Spectra at [TBD]

To be provided in CPA Part 2.

Figure 3.7-5: Safe Shutdown Earthquake Ground Motion Input Response Spectra at Plant Finished Grade Elevation

To be provided in CPA Part 2.

Figure 3.7-6: Comparison of Scaled Time Histories with Target Spectra for First Horizontal Component

To be provided in CPA Part 2.

Figure 3.7-7: Comparison of Scaled Time Histories with Target Spectra for Second Horizontal Component

To be provided in CPA Part 2.

Figure 3.7-8: Comparison of Scaled Time Histories with Target Spectra for the Vertical Component

To be provided in CPA Part 2.

Figure 3.7-9: Smoothed Power Spectrum Densities of the Geometric Mean of Modified Time Histories

To be provided in CPA Part 2.

Figure 3.7-10: Acceleration, Velocity, Displacement and Normalized Arias Intensity of the Matched First Horizontal Time History for the [TBD] Earthquake

To be provided in CPA Part 2.

Figure 3.7-11: Acceleration, Velocity, Displacement and Normalized Arias Intensity of the Matched Second Horizontal Time History for the [TBD] Earthquake

To be provided in CPA Part 2.

Figure 3.7-12: Acceleration, Velocity, Displacement and Normalized Arias Intensity of the Matched Vertical Time History for the [TBD] Earthquake

To be provided in CPA Part 2.

Figure 3.7-13: Acceleration, Velocity, Displacement and Normalized Arias Intensity of the Matched First Horizontal Time History for the [TBD] Earthquake

To be provided in CPA Part 2.

Figure 3.7-14: Acceleration, Velocity, Displacement and Normalized Arias Intensity of the Matched Second Horizontal Time History for the [TBD] Earthquake

To be provided in CPA Part 2.

Figure 3.7-15: Acceleration, Velocity, Displacement and Normalized Arias Intensity of the Matched Vertical Time History for the [TBD] Earthquake

To be provided in CPA Part 2.

Figure 3.7-16: Acceleration, Velocity, Displacement and Normalized Arias Intensity of the Matched First Horizontal Time History for the [TBD] Earthquake

To be provided in CPA Part 2.

Figure 3.7-17: Acceleration, Velocity, Displacement and Normalized Arias Intensity of the Matched Second Horizontal Time History for the [TBD] Earthquake

To be provided in CPA Part 2.

Figure 3.7-18: Acceleration, Velocity, Displacement and Normalized Arias Intensity of the Matched Vertical Time History for the [TBD] Earthquake

To be provided in CPA Part 2.

Figure 3.7-19: Acceleration, Velocity, Displacement and Normalized Arias Intensity of the Matched First Horizontal Time History for the [TBD] Earthquake

To be provided in CPA Part 2.

Figure 3.7-20: Acceleration, Velocity, Displacement and Normalized Arias Intensity of the Matched Second Horizontal Time History for the [TBD] Earthquake

To be provided in CPA Part 2.

Figure 3.7-21: Acceleration, Velocity, Displacement and Normalized Arias Intensity of the Matched Vertical Time History for the [TBD] Earthquake

To be provided in CPA Part 2.

Figure 3.7-22: Acceleration, Velocity, Displacement and Normalized Arias Intensity of the Matched First Horizontal Time History for the [TBD] Earthquake

To be provided in CPA Part 2.

Figure 3.7-23: Acceleration, Velocity, Displacement and Normalized Arias Intensity of the Matched Second Horizontal Time History for the [TBD] Earthquake

To be provided in CPA Part 2.

Figure 3.7-24: Acceleration, Velocity, Displacement and Normalized Arias Intensity of the Matched Vertical Time History for the [TBD] Earthquake

To be provided in CPA Part 2.

Figure 3.7-25: Hazard Consistent Strain-Compatible Shear Wave Velocity (V_s) and Compression (P)-Wave Velocity (V_p) Profiles

To be provided in CPA Part 2.

Figure 3.7-26: Hazard Consistent Strain-Compatible Damping Ratios

To be provided in CPA Part 2.

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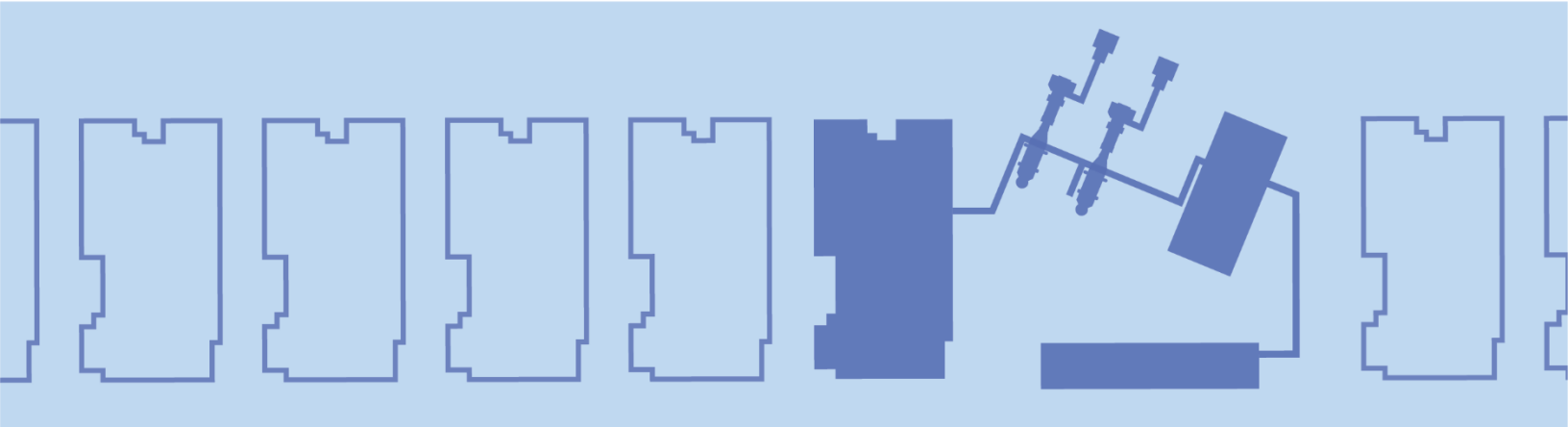
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Non-Proprietary Information

Chapter 3 Design of Structures, Components, Equipment, and Systems

Section 3.8 – Design of Seismic Category I Structures



ACRONYM LIST

Acronym	Explanation
3D	Three-Dimensional
ACI	American Concrete Institute
AISC	American Institute of Steel Construction
ALWR	Advanced Light-Water Reactor
ANSI	American National Standard Institute
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASTM	ASTM International
BECPV	Blue Energy Center – Port of Victoria
BPVC	Boiler and Pressure Vessel Code
CB	Control Building
CEPSS	Containment Equipment and Piping Support Structure
DBA	Design Basis Accident
DEC	Design Extension Condition
DP-SC	Diaphragm Plate Steel-Plate Composite
FE	Finite Element
FIA	Foundation Interface Analysis
GDC	General Design Criterion/Criteria
IBC	International Building Code
ILRT	Integrated Leak Rate Test
ISI	Inservice Inspection
LB	Lower Bound
LOCA	Loss-of-Coolant Accident
MCR	Main Control Room
NRIC	National Reactor Innovation Center
OBE	Operating-Basis Earthquake
RB	Reactor Building
RG	Regulatory Guide
RPV	Reactor Pressure Vessel
RWB	Radwaste Building
SC1	Safety Class 1

Acronym	Explanation
SCCV	Steel-Plate Composite Containment Vessel
SCR	Secondary Control Room
SEI	Structural Engineering Institute
SIT	Structural Integrity Test
SRP	Standard Review Plan
SSCs	Structures, Systems and Components
SSE	Safe Shutdown Earthquake
SSI	Soil-Structure Interaction
SSSI	Structure-Soil-Structure Interaction
TB	Turbine Building
UB	Upper Bound
UHS	Ultimate Heat Sink
NRC	U.S. Nuclear Regulatory Commission

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3.8 Design of Seismic Category I Structures

The BWRX-300 Seismic Category I structures consist of the Reactor Building (RB), the containment and containment internal structures, otherwise referred to as the integrated RB structures.

The BWRX-300 containment design is discussed in Subsections 3.8.1 and 3.8.2, with Subsection 3.8.1 focusing on the Steel-Plate Composite Containment Vessel (SCCV) portion of the containment and Subsection 3.8.2 focusing on the containment steel closure head and other Class MC components of the containment.

Internal structures housed in the containment are discussed in Subsection 3.8.3 and include the pedestal that supports the Reactor Pressure Vessel (RPV) and structures supporting piping and equipment within the containment.

The design of the RB structure outside of containment is discussed in Subsection 3.8.4, while the design of the common steel-plate composite mat foundation supporting the integrated RB is discussed in Subsection 3.8.5.

Subsections 3.8.4 and 3.8.5 also provide the design requirements for the adjacent Power Block structures and their foundations. For the location of these Power Block structures, refer to Section 1.2. General arrangement drawings of the RB and containment are also presented in Section 1.2.

In the following subsections, reference to the integrated RB is inclusive of the RB, containment, and containment internals, whereas RB is used to refer to the RB structure outside of containment.

3.8.1 Steel-Plate Composite Containment Vessel

The BWRX-300 uses a traditional containment system for the ultimate containment of radioactive materials for various postulated events. The configuration of the containment is shown in Figure 3.8-1. It comprises a SCCV, a steel containment closure head and other Class MC components. As shown in Figure 3.8-1, the BWRX-300 containment is completely enclosed within the RB.

The primary functions of the containment are:

1. To enclose and support the RPV and its connected piping systems.
2. To act as a leak-tight pressure boundary confining radioactive substances in different plant states, including normal operation, anticipated operational occurrences, Design Basis Accidents (DBAs) and Design Extension Conditions (DECs).
3. To provide radiation shielding to limit radiation dose within the applicable regulatory standards in different plant states.
4. To provide in conjunction with the surrounding RB, protection for plant safety equipment and systems against external hazards and human induced events.

The following subsections present the analysis and design requirements for the SCCV portion of the containment.

3.8.1.1 Description of the Steel-Plate Composite Containment Vessel

The BWRX-300 SCCV structure consists of a cylindrical wall, mat foundation, and top slab. The SCCV is constructed using steel-plate composite modules with diaphragm plate referred to as Diaphragm Plate Steel-Plate Composite (DP-SC) modules. The diaphragm plates of DP-SC modules have holes to allow the flow of concrete and are used, along with headed studs steel anchors, to provide the composite action between the steel faceplates and the concrete core.

The SCCV DP-SC modules, including the inner and outer faceplates, diaphragm plates, steel headed stud anchors and concrete infill are part of the containment pressure boundary, with the modules inner faceplate also serving as a leak-tight liner.

As shown in Figure 3.8-1, the SCCV structure includes two airlocks (see Subsection 3.8.2.1.2), penetrations (see Subsection 3.8.2.1.3), and other safety components, such as the corium shield installed on the lower portion of the SCCV and RPV pedestal. The corium shield protects the mat foundation and RPV pedestal from damage following core melt by preventing contact between the molten core and the DP-SC faceplates and concrete during severe accident conditions.

Besides the connection at the common mat foundation, the SCCV and the RB structures are also connected at the top slab and through the RB wing walls and floor slabs at intermediate floor levels as shown in Figure 3.8-1. The SCCV top slab directly supports the reactor cavity pool and portions of the Ultimate Heat Sink (UHS) in the RB, such as the fuel pool, equipment pool and isolation condenser inner pools.

A description of the steel containment closure head that covers the opening in the SCCV top slab over the RPV is provided in Subsection 3.8.2.1.1.

3.8.1.2 Applicable Codes, Standards, and Specifications

Codes, standards, specifications, and regulations applicable for the analysis, design, fabrication, construction, testing, and Inservice Inspection (ISI) of the SCCV are listed in NEDC-33926P-A, “BWRX-300 Steel-Plate Composite (SC) Containment Vessel and Reactor Building Structural Design,” (Reference 3.8-1). Conformance to U.S. Nuclear Regulatory Commission (NRC) regulatory requirements and guidance is as discussed in Chapter 1, Section 3.1 and Section 2.0 of NEDC-33926P-A.

As stated in Section 3.1, the design, fabrication, construction, testing, and ISI of the SCCV conforms to 10 CFR 50 and complies with the provisions of:

- General Design Criterion (GDC) 1 - A design, manufacturing, and operating quality program is established for the SCCV commensurate with the importance of its safety functions as discussed in Subsection 3.8.1.6.
- GDC 2 - The SCCV is designed to withstand the effects of a Safe Shutdown Earthquake (SSE) and tornadoes, hurricanes, floods, tsunami, and seiches, as applicable, without loss of capability to perform safety functions as discussed in Subsection 3.8.1.3.
- GDC 4 - The SCCV is designed to accommodate the effects of and be compatible with environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including Loss-of-Coolant Accidents (LOCAs) as discussed in Subsection 3.8.1.3.
- GDC 16 - The SCCV, including the containment closure head, access openings (airlocks) and penetrations, is designed to provide a leak-tight barrier and to contain the containment design pressure during design basis events as discussed in Subsections 3.8.1.2.1, 3.8.1.3, 3.8.1.4.5 through 3.8.1.4.7 and 3.8.1.7.
- GDC 50 - The SCCV, including the containment closure head, access openings and penetrations, is designed to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from LOCA as discussed in Subsections 3.8.1.3 and 3.8.1.4.5 through 3.8.1.4.7.

- GDC 51 - The SCCV, including the containment closure head, access openings and penetrations is designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design reflects consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady-state, and transient stresses, and (3) size of flaws, as applicable, as discussed in Subsections 3.8.1.4 and 3.8.1.7.
- GDC 52 - the SCCV, including the containment closure head, access openings and penetrations is designed with provisions to conduct periodic integrated leakage rate testing at containment design pressure as discussed in Subsections 3.8.1.7.2 and 6.2.
- GDC 53 - The containment is designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leak tightness of penetrations as discussed in Subsection 3.8.1.7.

Section 3.1 summarizes compliance with GDC.

The materials, analysis, design, fabrication, construction, inspection, examination and testing of the SCCV is in accordance with the provisions of Section 6.0 of NEDC-33926P-A. These provisions are adapted from American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section III, “Rules for Construction of Nuclear Facility Components,” Division 2, “Code for Concrete Containments,” Subsection CC, “Concrete Containments,” CC-1000 through CC-6000, including Division 2 Appendices (Reference 3.8-2) and changed based on analytical and engineering principles, including use of experimental results, to apply to a DP-SC containment vessel. Additional analysis and design guidance of NUREG-0800, Standard Review Plan (SRP) 3.8.1, “Concrete Containment,” and NRC Regulatory Guide (RG) 1.136, "Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments,” for concrete containments are also met, as applicable.

The applicable sections of the ASME BPVC, such as Section II, “Materials,” (Reference 3.8-3); Section III, Subsection NCA, “General Requirements for Division 1 and Division 2,” (Reference 3.8-4); Section V, “Non-Destructive Examination,” (Reference 3.8-5); and Section IX, “Qualification Standard for Welding, Brazing, and Fusing Procedures; Welders; Brazers; and Welding, Brazing, and Fusing Operators,” (Reference 3.8-6) are followed to the extent they apply to a DP-SC containment without reinforcing steel or tendons. ASME BPVC, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components," Division 1, “Rules for Inspection and Testing of Components of Light-Water-Cooled Plants,” incorporated by reference in 10 CFR 50.55a(g)(4) 18 months before the date of issuance of the operating license for the initial 120-month ISI interval as described in 10 CFR 50.55a(g)(4)(i), (Reference 3.8-7) is followed for the containment pre-service and periodic ISI and testing program.

3.8.1.2.1 Containment Jurisdictional Boundary

The BWRX-300 design codes jurisdictions are illustrated in Figure 3.8-2. As defined in Section 6.1 of NEDC-33926P-A, and shown in Figure 3.8-2, the SCCV boundary extends to the:

1. Outside diameter of the SCCV wall from mat foundation to containment top slab including the welds connecting the SCCV with the RB structural members.
2. Portion of the mat foundation under SCCV including the welds connecting the SCCV portion of the mat foundation with the remaining part of the RB mat foundation.

3. Containment top slab from containment closure head opening to the outside diameter of the SCCV including the welds connecting the slab with the RB structural members.

The SCCV along with the containment closure head, airlocks, and penetrations, provide the primary containment function as a leak-tight pressure boundary confining radioactive substances in different plant states. Although the internal RPV pedestal and other containment internal structures are completely within the SCCV, these internal structures are non-pressure retaining structures and are designed per the requirements of the American National Standard Institute (ANSI) / American Institute of Steel Construction (AISC) N690, "Specification for Safety-Related Steel Structures for Nuclear Facilities," (Reference 3.8-8). The design of welds connecting the containment internal structures to the containment pressure boundary is per Section 6.11 of NEDC-33926P-A and follows the quality assurance, welding procedures and inspection requirements of Sections 6.15 and 6.16 of NEDC-33926P-A. The connections of the RB walls and floors to the outside face of the SCCV wall are designed per Section 5.0 of NEDC-33926P-A, with the exception of attachment welds. Attachment welds connecting the RB walls and floors to the SCCV outside face are designed per Section 6.11 of NEDC-33926P-A and follow the quality assurance, welding procedures and inspection requirements of Sections 6.15 and 6.16 of NEDC-33926P-A.

3.8.1.3 Loads and Loading Combinations

The BWRX-300 containment is analyzed and designed for all credible conditions of loading, including normal loads, preoperational testing loads, loads during severe environmental conditions, loads during extreme environmental conditions, and loads during abnormal plant conditions. Load combinations considered in the design consider the probability of occurrence, loading time history and sequence of loads, as applicable.

Subsections 3.8.1.3.1 through 3.8.1.3.5 describe loads used in the design of the BWRX-300 containment. These loads are applicable to the SCCV, containment closure head and other Class MC components of the containment. Containment design loads are in accordance with the provisions of ASME BPVC, Section III, Division 1, Subsection NE, "Class MC Components," (Reference 3.8-9) and ASME BPVC Section III, Division 2, and are consistent with the guidance of NUREG-0800, SRP 3.8.1 and SRP 3.8.2, "Steel Containment." Loads resulting from the application of prestress and from relief valve or other high-energy device actuation are not considered in the design as these loads are not applicable to the BWRX-300 containment.

Load combinations used in the design of the SCCV are provided in Subsection 3.8.1.3.6. Load combinations used in the design of the containment closure head and other Class MC components are presented in Subsection 3.8.2.3.2.

SCCV design loads include RPV reactions due to seismic loads, LOCA loads, and acoustic loads, as well as containment penetrations, brackets, and attachment loads. The connections between the containment, RB, and containment internal structures in the integrated Finite Element (FE) model ensure a proper transfer of RPV reactions and other loads applied on the RB and containment internal structures to the SCCV structure.

3.8.1.3.1 Normal Loads

1. Dead load (D) which includes permanent dead weight of structural and shielding elements, and permanently located equipment.
2. Hydrostatic pressure of liquids in various pools (F).
3. Live loads (L, L_o) which include moveable equipment loads and other loads that vary in intensity and occurrence.

4. Indirect snow (S) and rain (R) Loads.
5. Thermal (T_o) effects and loads during normal operating, startup, or shutdown conditions.
6. Pressure (P_o) loads resulting from the pressure difference between the interior and exterior of the containment, considering both interior pressure changes due to heating or cooling and exterior atmospheric pressure variations. External pressure loads resulting from pressure variation either inside or outside the containment (P_v), as defined in ASME BPVC, Section III, Division 2, CC-3221.1, are considered under the P_o load case.
7. Pipe reactions (R_o) during normal operating or shutdown conditions based on the most critical transient or steady-state conditions.
8. Construction loads applied to the containment from start to completion of construction. The definitions for D, L and T_o given above are applicable, but are based on actual construction methods and/or conditions.
9. Indirect lateral soil and groundwater pressure loads (H).

3.8.1.3.2 Preoperational Testing Loads

1. Test thermal loads (T_t) due to thermal effects during the Structural Integrity Test (SIT) or Integrated Leak Rate Test (ILRT).
2. Test pressure loads (P_t) applied during the SIT or ILRT.

3.8.1.3.3 Severe Environmental Loads

Indirect design wind load (W) defined in Subsection 3.3.1.

3.8.1.3.4 Extreme Environmental Loads

1. Indirect extreme wind loads (W_t), including tornado and hurricane-generated missiles, defined in Subsections 3.3.2 and 3.5.
2. SSE seismic loads (E_s) determined for Blue Energy Center – Port of Victoria (BECPV) site-specific conditions considering Soil-Structure Interaction (SSI) effects, as discussed in Section 3.7, and include associated hydrodynamic loads and dynamic incremental soil pressures.

3.8.1.3.5 Abnormal Plant Loads

1. Accidental thermal effects (T_a) due to LOCA.
2. Accidental pressure loads (P_a) within the containment generated by a LOCA.
3. Accidental Pipe reaction loads (R_a) that consist of pipe reactions (including R_o) from thermal conditions generated by DBAs such as LOCA and SSE.
4. Local effects on containment due to LOCA (R_r) and blast loads (R_b) which includes:
    - R_{rr} load on the containment generated by the reaction of a ruptured high-energy pipe during the postulated event of the DBA
    - R_{rj} load on the containment generated by the jet impingement from a ruptured high-energy pipe during the postulated event of the DBA-
    - R_{rm} load on the containment resulting from the impact of a ruptured high-energy pipe during the DBA
    - Additional blast loads that may result from a postulated instantaneous break of a large pipe that could occur prior to jet loads

5. Internal flooding loads resulting from a DBA (H_a).
6. Other externally and internally generated missiles and design basis threats discussed in Section 3.5, as applicable.

3.8.1.3.6 Load Combinations

The SCCV is designed for the load combinations and load factors presented in Table 3.8-1. Load combinations used in the design of the SCCV are in accordance with ASME BPVC, Section III, Division 2, CC-3230, supplemented by NRC RG 1.136 and are consistent with the regulatory guidance of NUREG-0800, SRP 3.8.1.

3.8.1.4 Design and Analysis Procedures

3.8.1.4.1 Analytical Method

The RB, containment, containment internal structures, and their common mat foundation are analyzed as one integrated structure. Analysis procedures for the integrated structure, including a description of the Three-Dimensional (3D) model and computer codes used in the analysis, and assumptions made on boundary conditions, are discussed in Subsection 3.8.4.1.4. Details of the integrated RB FE model are provided in Appendix 3B.

As mentioned in Subsection 3.8.1.1, the SCCV and RB structures are connected at several locations. The connections between the SCCV and the RB members in the integrated RB FE model are modeled to reflect the appropriate load transfer for vertical, lateral, and thermal loads. The SCCV connections are modeled and designed as discussed in Sections 5.6 and 5.11 of NEDC-33926P-A, respectively. For more details on the SCCV connections, refer to Appendix 3B, and Appendix 3F.

Treatment of Axisymmetric and Non-axisymmetric Loads

ANSYS, the FE analysis software used to model and analyze the integrated RB structures, including the SCCV, (see Appendix 3I) evaluates the effects of the axisymmetric and non-axisymmetric loads applied on the 3D FE model by calculating the distribution of internal forces on the structural members.

The SCCV is shielded from the design wind, tornado, and hurricane by the RB, which completely encloses the structure. Forces from the design wind, tornado and hurricane are transmitted to the SCCV through the RB connections.

Indirect subgrade pressures are also transmitted to the SCCV through the RB connections. These pressures are obtained from the SSI analyses of the integrated RB discussed in Subsection 3.7.2. Methodology used to validate the subgrade pressure loads on the integrated RB is per Section 5.1.3 of NEDO-33914-A, “BWRX-300 Advanced Civil Construction and Design Approach,” (Reference 3.8-10).

Major Penetrations

The SCCV major penetrations are described in Subsection 3.8.2.1. These penetrations are included in the global integrated RB FE model used in the SSI analyses as stated in Subsections 3.7.2.3.1 and 3.8.4.1.4.

The state of stress and behavior of the SCCV around these openings is determined by use of acceptable analytical techniques, including but not limited to, the use of refined local FE models incorporating the imposed displacements calculated from the global integrated RB FE model discussed in Subsection 3.8.4.1.4.

Variation in Physical Material Properties

The effects of variation in assumptions and material properties on the integrated RB SSI analysis results are considered as discussed in Subsection 3.7.2.9.1.

Use of Vent Holes

Vent holes are used in the integrated RB steel-plate composite structures and the design of vent holes complies with the requirements of Sections 5.14 and 6.21 of NEDC-33926P-A. Detailed design implementation of vent holes is still progressing and Limitation and Condition (L&C) 8.16 of NEDC-33926P-A will be addressed prior to commencing nuclear construction using DP-SC modules.

3.8.1.4.2 Design Method

The SCCV structure design conforms to the requirements of Section 6.0 of NEDC-33926P-A and meets the acceptance criteria discussed in Subsection 3.8.1.5. As stated in NEDC-33926P-A, the design guidelines for the SCCV are adapted from ASME BPVC Section III, Division 2, for concrete containments using current research data for steel-plate composite structures.

Membrane forces, shear forces and bending moments used in the design of the SCCV structure are obtained from the linear elastic SSI analyses for the integrated RB FE model discussed in Subsection 3.8.4.1.4, for the loads and load combinations defined in Subsection 3.8.1.3. Appendix 3F presents design demands for critical SCCV sections identified per the criteria provided in Subsection 3.8.4.1.4.

3.8.1.4.3 Creep, Shrinkage, and Cracking of Concrete

The analysis of the SCCV considers the effects of creep, shrinkage, and cracking of concrete on the redistribution of forces and moments.

As stated in Subsection 3.7.2.3.1, stiffness calculations for structural modeling account for the expected state of stress and level of cracking for different loading conditions during normal operation and accident conditions, including accident thermal loading.

The effective stiffnesses, geometric and material properties, for operational and accidental conditions, of SCCV elements are computed using the equations in Section 5.5 of NEDC-33926P-A.

3.8.1.4.4 Tangential Shears/Stresses

Tangential (membrane) shears are captured in the 3D FE model of the SCCV. SCCV sections capacities are computed as discussed in Section 6.8 of NEDC-33926P-A.

3.8.1.4.5 Ultimate Capacity of Steel-Plate Composite Containment Vessel

The ultimate pressure capacity of the containment structure, including the SCCV, containment closure head, airlocks and penetrations, is determined to ensure its structural integrity and leak tightness under DEC internal pressure loads following the guidance of NUREG-0800, SRP 3.8.1 and NRC RG 1.216, “Containment Structural Integrity Evaluation for Internal Pressure Loadings Above Design Basis Pressure.”

The containment ultimate pressure capacity is obtained from the results of static FE analysis performed as described in Section 6.23.1 of NEDC-33926P-A, consistent with the guidelines of Regulatory Position 1 of NRC RG 1.216.

3.8.1.4.6 Steel-Plate Composite Containment Vessel Robustness Against Combustible Gas Pressure Loads

As stated- in Subsections 6.2.5 and 15.5.4.5, the containment design prevents combustible gases from creating a concern for deflagration in the containment during a DBA.

The containment structural robustness against DEC combustible gas pressure loads is evaluated using the methodology discussed in Section 6.23.2 of NEDC-33926P-A.

3.8.1.4.7 Steel-Plate Composite Containment Vessel Behavior Following a Severe Accident

In accordance with NRC SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” the BWRX-300 containment under the more likely severe accident conditions:

- Maintains its role as a reliable leak-tight barrier for approximately 24 hours following the onset of core damage.
- Continues to provide a barrier against the uncontrolled release of fission products following the initial 24-hour period.

The containment response to identified severe accidents is evaluated in the Final Safety Analysis Report using the methodology provided in Regulatory Position 3 of NRC RG 1.216 and as described in Section 6.23.3 of NEDC-33926P-A.

3.8.1.4.8 Corrosion Prevention

In accordance with Section 5.15 of NEDC-33926P-A, the corrosion protection of the SCCV DP-SC modules is met by one or a combination of the following approaches to meet the design life and decommissioning time of the plant:

- Corrosion tolerance by adding a sacrificial thickness
- Protective paint system suitable to the surrounding environment
- Membrane coating system
- Impressed Current Cathodic Protection

The additional sacrificial thickness for corrosion tolerance is not considered in the DP-SC modules strength or stiffness estimates.

NRC RG 1.54, "Service Level I, II, III, and In-Scope License Renewal Protection Coatings Applied to Nuclear Power Plants," and ASTM International (ASTM) D5144, “Standard Guide for Use of Protective Coating Standards in Nuclear Power Plants,” (Reference 3.8-11) are used for application, maintenance, and qualification of permanent corrosion protective coatings, as applicable.

Once the corrosion protection plan to be implemented for DP-SC modules is finalized, justification for the adequacy of the selected combination of measures will be provided prior to commencing nuclear construction using DP-SC modules per L&C 8.12 of NEDC-33926P-A.

3.8.1.5 Structural Acceptance Criteria

The acceptance criteria for the design of the SCCV are presented in Table 3.8-2a and Table 3.8-2b. Following the guidance of NUREG-0800, SRP 3.8.1, Subsection II.5 and as stated in Section 6.6 of NEDC-33926P-A, these allowables are based on criteria in Tables CC-3432-1 and CC-3432-2 of ASME BPVC, Section III, Division 2, Subsection CC modified to include the allowable stresses for the steel component of DP-SC modules. The stress/strain acceptance

criteria in Table 3.8-2a and Table 3.8-2b allow the SCCV to remain elastic under service load combinations and below the range of general yield under factored loads.

The seismic design criteria for the BWRX-300 containment, including the SCCV, containment closure head and other Class MC components are summarized in Table 3.7-1. The seismic design of the SCCV considers Limit State LS-D response in accordance with American Society of Civil Engineers (ASCE) / Structural Engineering Institute (SEI) 43, “Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities,” (Reference 3.8-12), ensuring an essentially elastic response when subjected to SSE.

The BWRX-300 SCCV is designed for impulsive and impactive loads per the guidance of Appendix A of NUREG-0800 SRP 3.8.1. ASME BPVC, Section III, Division 2 provisions are not fully applicable to the DP-SC containment. Design allowable stresses and deformation limits developed for the SCCV design are provided in Section 6.10 of NEDC-33926P-A.

3.8.1.6 Materials, Quality Control, and Special Construction Techniques

3.8.1.6.1 Materials

Materials used in the construction of the SCCV are in accordance with Section 6.2 of NEDC-33926P-A and NRC RG 1.136.

To ensure the acceptability of the DP-SC modules for the construction of the SCCV, the NEDC-33926P-A design requirements are supported by results from a test program performed under the National Reactor Innovation Center (NRIC) Advanced Construction Technology project in the United States. Mockups are used to demonstrate the efficiency of fabrication, installation, and construction processes for use of DP-SC modules for nuclear applications.

Properties of materials used in the construction of the SCCV are provided in the following subsections.

Concrete Infill

To enhance shielding, a high-density self-consolidating concrete designed in accordance with NRC RG 1.69, “Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants,” is used in the SCCV wall.

The design compressive strength for the high-density self-consolidating concrete used for the construction of the SCCV is determined as described in Section 5.2.1 of NEDC-33926P-A.

Concrete is batched and placed in accordance with Sections 6.2.1 and 6.15 of NEDC-33926P-A. Concrete and concrete constituents are examined and tested in accordance with Section 6.16 of NEDC-33926P-A. Supplementary concrete sampling requirements are used as described in Section 3.2.2.1 of NEDO-33914-A to address the effect of small volume of concrete placed.

Aggregates in concrete subjected to elevated temperatures for long periods of time do not contain pyretic materials.

Steel Materials

DP-SC steel components conform to the following:

- Steel faceplates/diaphragm plate: ASME SA-738, Grade B or SA-537, Class 1 or Class 2, or ASTM A709/A709M, "Standard Specification for Structural Steel for Bridges," (Reference 3.8-13), Grade HPS 70W, per ASME BPVC, Code Case N-763, "ASTM A 709-06, Grade HPS 70W (HPS 485W) Plate Material Without Postweld Heat Treatment as Containment Liner Material or Structural Attachments to the Containment Liner, Subsection CC, Section III, Division 2," (Reference 3.8-14)
- Studs: ASTM A108, "Standard Specification for Steel Bar, Carbon and Alloy, Cold-Finished," (Reference 3.8-15), Type B, as permitted by ASME BPVC Section III, Division 2, Subsection CC, MANDATORY APPENDIX D2-1, Table D2-1-2.2 and meeting the strength requirements of ASME BPVC, Section III, Division 2, Subsection CC, CC-2711, Table CC-2623.2-1

Structural Steel and Appurtenances

The structural steel and other appurtenances for the SCCV other than the primary DP-SC steel components covered above conform to the following:

- Structural steel embedded plate and attachments: ASTM A572, "Standard Specification for High-Strength Low-Alloy Columbium-Vanadium Structural Steel," (Reference 3.8-16), Grade 50, per ASME BPVC, Code Case N-632, "Use of ASTM A 572 Grades 50 and 65 for Structural Attachments to Class CC Containment Liners, Section III, Division 2," (Reference 3.8-17)
- Penetrations: ASME SA-516 Grade 60 or Grade 70, or SA-537 Class 2

Welding Materials

SCCV welding materials conform to the requirements of Section 6.0 of NEDC-33926P-A and ASME BPVC, Section IX.

Coatings

Corrosion protective coatings applied on the SCCV steel faceplates are discussed in Subsection 3.8.1.4.8.

3.8.1.6.2 Quality Control

Quality control procedures are established for the SCCV in the construction, fabrication and installation specifications and implemented during the fabrication, construction, installation, and inspection stages.

These specifications cover the fabrication, furnishing, and installation of each structural item and specifies the inspection and documentation requirements to ensure that the requirements of ASME BPVC, Section III, Division 2, CC-4000 and CC-5000, NRC RG 1.28, "Quality Assurance Program Criteria (Design and Construction)" and NRC RG 1.136 are met.

Procedures governing the design, fabrication, construction, installation, and inspection of the SCCV are developed in compliance with the GVH quality assurance program.

Welded Construction and Examinations

Refer to Subsection 3.8.1.2.1 for quality control procedures used for attachment welds connecting the SCCV to the RB walls and floors.

3.8.1.6.3 Special Construction Techniques

The SCCV is constructed using modular construction technique as discussed in Subsection 3.8.4.1.6.

The SCCV construction requirements are in accordance with Sections 6.15 and 6.16 of NEDC-33926P-A.

3.8.1.7 Testing and Inservice Surveillance Requirements

A pre-service and periodic ISI and testing program is established for the SCCV to meet the requirements of ASME BPVC, Section XI, Subsections IWE and IWL, as per 10 CFR 50.55a (b)(2)(viii) and (b)(2)(ix).

3.8.1.7.1 Structural Integrity Test

The SCCV pre-service SIT plan and instrumentation complies with Section 6.17 of NEDC-33926P-A, and with NRC RG 1.136 and NRC RG 1.216.

In accordance with Section 6.17 of NEDC-33926P-A, deformation and strain measurements are made to evaluate the behavior of the prototype SCCV, as defined per ASME BPVC, Section III, CC-6150, and confirm that the actual structural response is within the limits predicted by analysis.

3.8.1.7.2 Pre-Operational and Inservice Integrated Leak Rate Testing

The SCCV is designed such that periodic ILRT can be conducted as discussed in Subsection 6.2.6.

3.8.1.7.3 Containment Pre-Service and Inservice Inspection

The SCCV pre-service and periodic ISI plan is in accordance with Sections 6.17 and 6.22 of NEDC-33926P-A, respectively, and meets the requirements of ASME BPVC, Section XI.

3.8.2 Class MC Steel Components of the BWRX-300 Containment

The Class MC components of the BWRX-300 containment comprise the containment closure head, the two containment airlocks and the containment penetrations.

3.8.2.1 Description of the Class MC Steel Components

3.8.2.1.1 Containment Closure Head

The containment closure head is a removable steel head anchored to the SCCV top slab using a pair of mating flanges. The containment closure head is part of the containment pressure boundary and forms part of the reactor cavity pool. The containment closure head is designed for removal during reactor refueling and for replacement prior to reactor operation.

Provisions are made for testing the flange seals to ensure the containment closure head can perform its pressure retaining functions. Stainless-steel cladding is fixed to the outer surface of the containment closure head to protect it from the water in the reactor cavity pool. The cladding thickness is determined as discussed in Subsection 3.8.2.4.

A structural overview of the containment closure head is shown in Figure 3.8-1 and Figure 3.8-3.

3.8.2.1.2 Containment Airlocks

As shown in Figure 3.8-1, two airlocks are provided in the SCCV structure to provide access to the upper and lower portions of the containment. These airlocks serve the dual purpose of allowing personnel access to the containment and for movement of equipment. Figure 3.8-4 provides a structural overview of an airlock and its interface with the SCCV wall.

Each containment airlock has two pressure-seated doors interlocked to prevent simultaneous opening of both doors, and to ensure that one door is completely closed before the opposite door can be opened. The door operation is designed and constructed so either door may be operated from inside the containment, inside the lock, or from outside the containment. Also, each airlock has a barrel, which is an inner round portion that is slid into the fixed portion of the sleeve that is embedded in the SCCV wall. The barrel of the airlock can be removed if large equipment needs to be transported during outages.

3.8.2.1.3 Penetrations

Figure 3.8-5 provides a structural overview of a typical penetration and its interface with the SCCV.

The major piping penetrations through the SCCV are associated with main steam and feedwater lines. Sleeve length for hot penetrations is designed to meet the concrete temperature limitations specified in Section 6.5.1 of NEDC-33926P-A. SCCV electrical penetrations are sealed to the SCCV pressure boundary.

3.8.2.2 Applicable Codes, Standards, and Specifications

The containment closure head, airlocks and penetrations are part of the containment pressure boundary, and their design, fabrication, construction, testing, and ISI conform to 10 CFR 50 and comply with the provisions of GDC 1, 2, 4, 16, 50, 51, 52 and 53 as discussed in Section 3.1 and Subsection 3.8.1.2.

The containment closure head, airlocks and penetrations are analyzed, designed, fabricated, constructed, inspected, and tested following the provisions of ASME BPVC, Section III, Division 1, Subsection NE, Section XI, Subsection IWE, and 10 CFR 50 Appendix J in conformance with NUREG-0800, SRP 3.8.2, including applicable conditions listed in 10 CFR 50.55a.The code jurisdictional boundary for application of ASME BPVC, Section III, Division 1, Subsection NE, Class MC to the containment closure head, airlocks and penetrations is shown in Figure 3.8-3, Figure 3.8-4, and Figure 3.8-5, respectively.

Metal components backed by concrete are designed in accordance with Section 6.0 of NEDC-33926P-A.

3.8.2.3 Loads and Loading Combinations

3.8.2.3.1 Design Loads

Class MC components are designed for loading conditions applicable for the containment structure. Refer to Subsection 3.8.1.3 for the description of loads used in the design of the SCCV that are also applicable for the design of the containment Class MC components.

3.8.2.3.2 Design Load Combinations

Load combinations and associated load factors used in the design of the containment Class MC components are in conformance with NUREG-0800, SRP 3.8.2 and NRC RG 1.57, “Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components,” except for MC components backed by concrete are per the alternative design approach discussed in Sections 4.0 and 6.0 of the NEDC-33926P-A.

Table 3.8-3 and Table 3.8-4 summarize the load combinations and associated load factors used in the design of the containment closure head and other Class MC components, respectively.

As shown in Table 3.8-3 and Table 3.8-4, Class MC components combined loadings are categorized according to Level A, C and D service limits as defined in NE-3113. Because the Operating-Basis Earthquake (OBE) is set at one-third of the SSE for design purposes as

described in Subsection 3.7.1, OBE load combinations are not considered in the design. The OBE load combination involving post-flooding conditions is considered for the design of Class MC components, excluding containment closure head, in accordance with NUREG-0800, SRP 3.8.2. OBE loads are also considered for cyclic loading considerations following the guidance of NUREG-0800, SRP 3.8.2.

Metal components backed by concrete are designed for the load combinations and associated load factors as summarized in Table 3.8-1.

3.8.2.4 Design and Analysis Procedures

Analysis and design procedures for the containment closure head, airlocks, and penetrations, are presented in Subsections 3.8.2.4.1, 3.8.2.4.2 and 3.8.2.4.3, respectively.

The design of the Class MC steel components meets the acceptance criteria discussed in Subsection 3.8.2.5, including buckling and fatigue evaluations as required.

The design by analysis utilizes demands obtained from the linear elastic SSI analyses for the integrated RB FE model discussed in Subsection 3.8.4.1.4, for the loads and load combinations defined in Subsection 3.8.1.3. For the treatment of non-axisymmetric and localized loads on Class MC components, refer to Subsection 3.8.1.4.1.

Corrosion prevention methods considered for the containment closure head and Class MC components include the use of protective coatings and sacrificial thicknesses determined in accordance with NE-3121. The thickness of the cladding fixed to the wetted surface of the containment closure head is determined in accordance with NE-3122. The guidance of NRC RG 1.54 and ASTM D5144 are used as applicable for the application, maintenance, and qualification of corrosion protective coatings.

Effects of containment penetrations on the containment internal ultimate pressure capacity are considered as discussed in Subsection 3.8.1.4.5 and in Section 6.23.1 of NEDC-33926P-A.

3.8.2.4.1 Containment Closure Head

Design procedures for the containment closure head are shown in Figure 3.8-6.

The containment closure head is analyzed using ANSYS and ACS SASSI computer programs and manual calculation. The stresses, including discontinuity stresses induced by the combination of external pressure or internal pressure, dead load, live load, thermal effects, and seismic loads, are evaluated. Minimum thickness evaluations, including local strengthening of the containment closure head at locations of attachments and openings, are performed in accordance with NE-3100 and NE-3300. The required analyses and limits for the resulting stress intensities for the containment closure head are in accordance with ASME BPVC, Section III, Division 1, Subsection NE, NE-3200.

Following the guidance of Subsection II.4 of NUREG-0800, SRP 3.8.2, the evaluation of the containment pressure capacity considers buckling of the containment closure head under internal pressure as one of its limiting conditions. The compressive stress within the knuckle region caused by the internal pressure and the compression in other regions caused by other loads are limited to the allowable compressive stress values in accordance with NE-3222, or Code Case N-284-4, “Metal Containment Shell Buckling Design Methods, Class MC, TC, SC Construction Section III, Division 1 and Division 3,” (Reference 3.8-18).

The containment closure head is also evaluated for fatigue, with the total number of stress cycles considered in the fatigue evaluation following the requirements of NE-3221.5(d). Because the containment closure head is submerged in the reactor cavity pool and can be exposed to water temperature above 50°C during accident, the fatigue evaluation of the containment closure head

also follows the guidance of NRC RG 1.207, “Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors.”

3.8.2.4.2 Containment Airlocks

Design procedures for the containment airlocks are shown in Figure 3.8-7.

The containment airlocks are analyzed using ANSYS and/or manual calculation based on handbook formulas and tables. The discontinuity stresses induced by the combination of external, dead, and live loads, including the effects of earthquake loadings are evaluated. Minimum thickness evaluations for the containment airlocks are in accordance with NE-3100 and NE-3300. The airlocks required analyses and limits for the resulting stress intensities are in accordance with NE-3200, including buckling evaluation per Code Case N-284-4- and fatigue evaluation as required.

The design and detailing of airlock components backed by concrete also meet the requirements of Section 6.12 of NEDC-33926P-A.

3.8.2.4.3 Penetrations

Design procedures for the containment penetrations are shown in Figure 3.8-7.

Penetrations are subjected to various combinations of piping reactions, mechanical, thermal, and seismic loads transmitted through the SCCV structure. The resulting forces due to various load combinations are combined with the effects of external and internal pressures. Penetrations are analyzed using ANSYS and/or manual calculation based on handbook formulas and tables for applicable loads and load combinations. Minimum thickness evaluations for the containment penetrations are in accordance with NE-3100 and NE-3300. The penetrations required analysis and associated stress intensity limits are in accordance with NE-3200, including buckling evaluation per Code Case N-284-4 and fatigue evaluation as required.

The design and detailing of penetration components backed by concrete also meet the requirements of Section 6.12 of NEDC-33926P-A.

3.8.2.5 Structural Acceptance Criteria

The acceptance criteria for the design of the Class MC components of the containment are the allowable stress limits specified in NE-3220. Seismic design criteria for the containment Class MC components are the same as those for the SCCV described in Subsection 3.8.1.5.

The structural acceptance criteria for the post-flooding condition, which is only applicable for Class MC components other than the containment closure head, is in accordance with NUREG-0800, SRP 3.8.2. Table 3.8-5 and Table 3.8-6 summarize the acceptance criteria for testing, design, Service Level A, C and D, and Post-flooding conditions, as applicable, for the containment closure head and other Class MC components, respectively.

Stability against compression buckling is in accordance with NE-3222.1.

3.8.2.6 Materials, Quality Control, and Special Construction Techniques

3.8.2.6.1 Materials

Steel materials used in the fabrication of the containment closure head and other Class MC components of the containment are in accordance with NE-2000.

Structural steel and other appurtenances for the containment closure head and other Class MC components conform to the following:

- Structural steel: ASME SA-516 Grade 60 or 70, SA-537 Class 1 or Class 2 or SA-738 Grade B
- Non-pressure-retaining attachments: ASTM A572 Grade 50 as permitted by ASME BPVC, Section III, “Rules for Construction of Nuclear Facility Components,” Division 1, Subsection NF, “Supports,” (Reference 3.8-19) referenced by ASME BPVC, Section III, Division 1, Subsection NE
- Stainless-steel cladding: ASME SA-264 steel-clad plate alloy and SA-240, Type 304L alloy for general application in accordance with ASME BPVC, Section II, Part A
- Bolting: ASME SA-193 Grade B7 for Class MC components, and ASME SA-193 Grade B8 or SA-437 Grade B4B for containment closure head

3.8.2.6.2 Quality Control

Quality control procedures are established for the Class MC components of the containment, in the construction, fabrication, and installation specifications.

These procedures are in accordance with NUREG-0800, SRP 3.8.2, NRC RG 1.28, NE-2000, NE-4000, and NE-5000 of ASME BPVC, Section III, Division 1, Subsection NE, including applicable conditions listed in 10 CFR 50.55a(b)(1), NCA-4000 of ASME BPVC, Section III, Subsection NCA, and are developed in compliance with the GVH quality assurance program.

3.8.2.6.3 Special Construction Techniques

The Class MC components of the containment do not use special construction techniques.

3.8.2.7 Testing and Inservice Surveillance Requirements

SCCV testing and ISI requirements described in Subsection 3.8.1.7 apply to the Class MC components of the containment. Sufficient physical access is provided in the design to accommodate the ISI of the containment Class MC components.

3.8.3 Containment Internal Structures

The BWRX-300 containment internal structures include the RPV pedestal and the containment internal structural steel. The containment internal structural steel comprises the Containment Equipment and Piping Support Structure (CEPSS) and two support platforms as shown in Figure 3.8-1.

3.8.3.1 Description of Internal Structures

3.8.3.1.1 Reactor Pressure Vessel Pedestal

The RPV pedestal is a cylindrical-shaped structure that supports the RPV. The RPV pedestal is constructed using DP-SC modules and is integrated with the common mat foundation as shown in Figure 3.8-1.

The RPV pedestal is equipped with structural steel that lines the top of the RPV pedestal where the RPV skirt is anchored using anchor bolts. The RPV pedestal also provides structural support for the RPV bottom stabilizers, the CRD housing outer springs, and the CEPSS. The part of the RPV pedestal surrounding the core region of RPV and the non-structural short concrete wall with steel faceplates added on top of it help to attenuate the radiation emanating from the RPV. Openings are provided in the RPV pedestal to permit the routing of necessary piping to the RPV, to permit ISI of the RPV and piping, and to ensure personnel access into the under-vessel region.

3.8.3.1.2 Containment Equipment and Piping Support Structure

The CEPSS is supported vertically by the RPV pedestal and is laterally supported through tangential restraints to the SCCV as shown in Figure 3.8-1. The CEPSS framing connects directly to the RPV pedestal. Shear lug support connections with the SCCV are used to allow free thermal expansion of the CEPSS while resisting CEPSS lateral forces.

The CEPSS consists of various structural components, such as beams and columns, as shown in Figure 3.8-1. The CEPSS framing includes vertical bracing members around the CEPSS outer perimeter (near SCCV wall) and inner perimeter (above the pedestal), and horizontal bracing members to transfer lateral forces to the SCCV wall.

The CEPSS structure supports the RPV top stabilizers that provide horizontal bracing for the RPV. In addition to the RPV stabilizers, the CEPSS also supports piping, RIVs including actuator, and miscellaneous platforms needed for personnel access and equipment inspection and maintenance.

3.8.3.1.3 Support Platforms

The steel support platforms shown in Figure 3.8-1 provide support for miscellaneous components, including fans and coolers associated with the containment cooling system, and for apparatus needed for equipment inspection and maintenance.

The steel support platform beams are supported by the RPV pedestal and SCCV wall as shown in Figure 3.8-1.

3.8.3.2 Applicable Codes, Standards, and Specifications

The design, fabrication, construction, testing, and ISI of the containment internal structures conform to 10 CFR 50 and comply with the provisions of GDC 1, 2, 4, and 50 as discussed in Section 3.1 and explained below.

The analysis, design, fabrication and testing of the DP-SC RPV pedestal is in accordance with ANSI/AISC N690, endorsed and modified per NRC RG 1.243, “Safety-Related Steel Structures and Steel-Plate Composite Walls for Other Than Reactor Vessels and Containments,” including the modified design rules for the BWRX-300 non-containment steel-plate composite structures provided in Section 5.0 of NEDC-33926P-A (see Section 3.8.4.1.2 for more details). This approach meets the intent of the guidance of NUREG-0800, SRP 3.8.3, “Concrete and Steel Internal Structures of Steel or Concrete Containments,” and complies with the GDC requirements.

The analysis, design, fabrication and testing of the of the containment internal structural steel is in accordance with ANSI/AISC N690 consistent with NUREG-0800, SRP 3.8.3.

The design codes jurisdictional boundary for the RPV pedestal and internal structural steel are provided in Figure 3.8-2.

3.8.3.3 Loads and Loading Combinations

The containment internal structures are completely contained within and integrated with the RB and SCCV. As a result, the design of the containment internal structures considers both design loads applied directly to the containment internal structures and those applied indirectly through the RB and SCCV.

3.8.3.3.1 Design Loads

Refer to Subsections 3.8.1.3 and 3.8.4.1.3 for the description of design loads applicable for the SCCV and RB structures that are also generally applicable for the design of containment internal structures.

The design loads also include the reactions from the RPV at the support locations on the containment internal structures and other bracket and attachment loads applicable during different plant conditions. The RPV lumped mass beam model representing the mass and stiffness properties of the RPV is included in the integrated RB FE model as discussed in Subsection 3.7.2.3.2. Consequently, dead load and seismic load reactions from the RPV are obtained directly from the static and seismic analyses discussed in Subsection 3.8.4.1.4. Other normal and accidental plant operating loads are applied to the model as reaction force loads.

3.8.3.3.2 Load Combinations

Following the guidance of Section II.3 of NUREG-0800, SRP 3.8.3, load combinations and load factors for the design of the steel-plate composite structures and structural steel that form the containment internal structures are in accordance with ANSI/AISC N690, including the supplemental guidance of NRC RG 1.243.

They are covered by the comprehensive set of load combinations listed in Table 3.8-7 used for the design of the RB structure.

3.8.3.4 Design and Analysis Procedures

Analysis and design procedures for the containment internal structures are provided in the following subsections. These procedures meet the guidance provided in Section II.4 of NUREG-0800, SRP 3.8.3. Design demands for critical sections of the containment internal structures are provided in Appendix 3G.

3.8.3.4.1 Reactor Pressure Vessel Pedestal

The RPV pedestal is part of the integrated RB FE model and is analyzed and designed using the same procedures outlined in Subsection 3.8.4.1.4 for the RB.

The RPV Pedestal in the integrated RB FE model includes major openings and penetrations in accordance with Appendix N9 of ANSI/AISC N690. The state of stress and behavior of the RPV Pedestal around these openings is determined by use of analytical techniques.

The RPV pedestal supports and transfers vertical and lateral loads from the RPV, CEPSS and support platforms to the SCCV (inner) mat foundation as described in Subsection 3.8.4.1.4. The connections between the RPV pedestal, the RPV and other containment internal structures are appropriately modeled in the integrated RB FE model to reflect the load transfer for the vertical, thermal, reactions forces, and lateral loads discussed in Subsection 3.8.3.3.1.

3.8.3.4.2 Containment Equipment and Piping Support Structure

The CEPSS is part of the integrated RB FE model and is analyzed as discussed in Subsection 3.8.4.1.4.

The CEPSS supports and transfers vertical and lateral loads coming from the containment equipment and piping to the RPV pedestal and SCCV wall as described in Subsection 3.8.4.1.4. The connections between the CEPSS, the RPV pedestal and SCCV are appropriately modeled in the integrated RB FE model to reflect the load transfer for vertical and lateral loads.

The CEPSS is designed in accordance with ANSI/AISC N690 and is designed to carry piping dynamic loads without buckling and while remaining elastic. Pipe whip restraints attached to supporting CEPSS beams and columns are designed with the ability to sustain limited inelastic deformation without rupturing. SC1 SSCs affected by these inelastic deformations are evaluated to verify that their safety functions are not compromised.

3.8.3.4.3 Support Platforms

The support platforms shown in Figure 3.8-1 are part of the integrated RB FE model and are analyzed and designed using the same analytical and design procedures as those for the CEPSS discussed in Subsection 3.8.3.4.2.

The support platforms vertical and lateral loads are supported by the RPV pedestal and SCCV wall as described in Subsection 3.8.4.1.4.

3.8.3.5 Structural Acceptance Criteria

Acceptance criteria for the design of DP-SC RPV Pedestal, including welded and bolted connections, are in accordance with Section 5.0 of NEDC-33926P-A.

The acceptance criteria for the containment internal structural steel are in accordance with ANSI/AISC N690, including the guidance of NUREG-0800, SRP 3.8.3 and NRC RG 1.243.

3.8.3.6 Materials, Quality Control, and Special Construction Techniques

3.8.3.6.1 Materials

To enhance shielding, a high-density self-consolidating concrete designed in accordance with NRC RG 1.69 is used in the RPV pedestal DP-SC modules.

The design compressive strength for the pedestal high-density self-consolidating concrete infill is determined as described in Section 5.2.1 of NEDC-33926P-A. Aggregates used in areas subjected to elevated temperature for long durations are free of pyretic materials.

The RPV pedestal DP-SC steel components conform to the following:

- Steel faceplates/diaphragm plate: ASTM A572 Grade 50 or Grade 65
- Studs: ASTM A108 Type B

The RPV pedestal structural steel components conform to ASTM A572 Grade 50.

Materials used in construction of the containment internal structural steel are in accordance with ANSI/AISC N690.

The containment internal structural steel conforms to:

- Structural steel and connections: ASTM A572, ASTM A500, “Standard Specification for Cold-Formed Welded and Seamless Carbon Steel Structural Tubing in Rounds and Shapes,” (Reference 3.8-20) or A1085, “Standard Specification for Cold-Formed Welded Carbon Steel Hollow Structural Sections,” (Reference 3.8-21)
- High-strength structural steel plates: ASTM A572
- Structural steel attachments to SCCV: ASTM A572 Grade 50 per ASME Code Case N-632-

Corrosion protection coatings are applied on the exposed surface of the containment internal structures to protect the metal from corrosion.

3.8.3.6.2 Quality Control

The quality control procedures governing the design, fabrication, construction, installation, and inspection of the RPV pedestal are the same as those for the RB structure described in Subsection 3.8.4.1.6.

The quality control procedures governing the design, fabrication, construction, installation, and inspection of the containment internal structural steel are in accordance with NUREG-0800,

SRP 3.8.3, NRC RG 1.28 and Chapters NM and NN of ANSI/AISC N690. The procedures are developed in compliance with the GVH quality assurance program.

3.8.3.6.3 Special Construction Techniques

The RPV pedestal is constructed using modular construction technique as described in Subsection 3.8.4.1.6.

3.8.3.7 Testing and Inservice Surveillance Requirements

A formal program of testing and ISI is not required for the containment internal structures as these structures do not serve any pressure retaining function. However, due to their contribution to the containment structural function, the condition of the containment internal structures is monitored during the operating life of the plant in compliance with 10 CFR 50.65 and in accordance with NRC RG 1.160, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.” To accommodate the ISI of containment internal structures, the design provides sufficient physical access to the structures.

3.8.4 Other Seismic Category I Structures

The primary focus of this section is on the RB structure, the only Seismic Category I structure outside of the containment. Other structures discussed in this section include the Radwaste Building (RWB), which houses systems that collect, process and store radioactive waste, and the Control Building (CB) locally hardened by design to meet the control room habitability requirements. Level of content provided in the following subsections is commensurate with the safety classification of the structures. For information on other Power Block structures, refer to Section 3.7.

3.8.4.1 Reactor Building

3.8.4.1.1 Description of the Structure

The RB is a deeply embedded cylindrical-shaped building made of steel-plate composite floors and walls, with structural steel members supporting the roof structure. Two types of steel-plate composite modules are used for the construction of the RB structures which have the same overall performance characteristics but different steel ties configuration. DP-SC modules, the steel-plate composite type predominantly used for the construction of the RB structures, rely on diaphragm plates to attach the two faceplates and provide the composite action between the steel faceplates and the concrete core as described in Subsection 3.8.1.1. Conventional steel-plate composite modules use discrete tie bars instead of diaphragm plate to attach the faceplates as described in Section 3.4 of NEDC-33926P-A.

The below grade portion of the RB structure encloses the containment and protects the RPV, reactor support and SC1 SSCs, and most of vital and non-vital power supplies and equipment. While the majority of the RPV and SCCV is located below grade, the top elevation of the RPV and SCCV top slab are above grade. In addition, the above grade portion of RB structure houses the refueling floor, refueling and fuel handling systems, pools, and polar crane.

The RB fuel pool provides for storage of new and spent fuel, along with the in-core components removed from the RPV during refueling. New fuel is staged in fuel storage racks located in the fuel pool. An area in the deep pit is used for loading of a spent fuel cask. The fuel storage racks are Seismic Category I structures and are designed to withstand an SSE without failure of the basic structure or damage to the active region of irradiated fuel. For the performance characteristics, physical arrangement, operation, and design of the fuel storage racks, refer to Subsection 9.1.2.

The RB polar crane is composed of an overhead bridge of two deep girders supporting a trolley with a main and auxiliary hook. The top of circular rail is attached to the RB exterior wall and the bridge structure spans the full width of the refueling floor. The crane is classified as Seismic Category II to maintain structural integrity and is designed as a single failure proof system to mitigate the probability of a load drop event. For more information on the design of the RB polar crane, refer to Subsection 9.1.5.

Section 1.2 provides key dimensions of the RB while Figure 3.8-1 presents an elevation sectional view of the structure. Figure 3.8-2 depicts the separation of the RB structure boundary from the containment pressure boundary discussed in Subsection 3.8.1.2. The RB structural boundary includes the following areas:

1. Mat foundation portion between the SCCV wall and the RB exterior wall.
2. RB below grade (from the mat foundation) to the support for the polar crane, including the roof structure.
3. Isolation condenser pools, fuel pool, equipment pool, and the reactor cavity pool composed of steel-plate composite walls that are lined for corrosion protection and have a leakage detection system as applicable.
4. Rooms at several elevation levels outside the containment, structurally integral with the containment structure.

In addition to sharing the common mat foundation, the RB and SCCV structures are also integrated at the connections of wing walls and elevated slabs between the mat foundation and the reactor cavity pool top slab elevation of the structure as shown in Figure 3.8-1.

The primary functions of the RB are:

1. To house the reactor vessel, the containment structure, the reactor support structure of the reactor coolant system and fuel handling equipment, biological shielding, and associated equipment and structures.
2. To provide adequate space for the operation, maintenance, and removal of equipment housed within the containment structure during periodic maintenance.
3. To provide protection for SC1 SSCs defined in Section 3.2 from environmental and natural hazards phenomena, such as floods, winds, tornadoes, hurricanes, and earthquakes.
4. To provide protection for SC1 SSCs from external (e.g., explosions and missiles from nearby transportation/industry or aircraft impact) and internal (e.g., pipe breaks and heavy load drops) hazards.
5. To support the habitability functions for the Secondary Control Room (SCR) such as radiological shielding and toxic gas isolation and passive cooling for human occupancy in addition to environmental qualification for equipment.

3.8.4.1.2 Applicable Codes, Standards, and Specifications

Codes, standards, specifications, and regulations applicable for the analysis, design, fabrication, construction, testing, and ISI of the RB are listed in NEDC-33926P-A. Conformance to NRC regulatory requirements and guidance is discussed in Chapter 1, Section 3.1 and Section 2.0 of NEDC-33926P-A.

To comply with 10 CFR 50 and the provisions of GDC 1, 2 and 4, the materials, analysis, design, fabrication, construction, examination and testing of the RB structural steel and steel-plate

composite components are in accordance with ANSI/AISC N690, including the supplemental guidance of NRC RG 1.243 and NUREG-0800, SRP 3.8.4, “Other Seismic Category I Structures.”

The design of the RB DP-SC structural elements also meets the provisions of Section 5.0 of NEDC-33926P-A. NEDC-33926P-A, Section 5.0 provisions are adapted from ANSI/AISC N690 and adjusted to address the particularities of DP-SC construction. Sections 5.9 and 5.12 of NEDC-33926P-A justify why the provisions of Appendix N9 of ANSI/AISC N690 can be extended to the design of the RB DP-SC floors and curved walls. The design approach discussed in Section 5.0 of NEDC-33926P-A ensures a level of safety and performance for the RB DP-SC structures commensurate to that required by codes and standards listed in Section II.2 of NUREG-0800, SRP 3.8.4.

In addition to meeting the requirements of ANSI/AISC N690 and guidance of NRC RG 1.243, the RB polar crane rail clips, runway girders, and support corbels are also designed to meet the requirements of ASCE/SEI 7, “Minimum Design Loads and Associated Criteria for Buildings and Other Structures,” (Reference 3.8-22) and Sections 4160 and 4460 of ASME NOG-1, “Minimum Design Loads and Associated Criteria for Buildings and Other Structures,” (Reference 3.8-23). Crane loading is developed in accordance with Section 4.9 of ASCE/SEI 7.

3.8.4.1.3 Loads and Loading Combinations

In addition to the loads applicable directly to the RB, loads considered in the design of the RB include loads applied to the SCCV that are transferred to the RB structural components.

Loads and load combinations considered in the design of the RB meet the intent of Section II.3 of NUREG-0800, SRP 3.8.4.

Design Loads

Loads used in the design of the RB structure are in accordance with ANSI/AISC N690, following the guidance of NUREG-0800, SRP 3.8.4.

Loads, such as accident pressure and thermal transient loads due to a LOCA, internal to SCCV are also considered for the design of structural components of the RB that are integrated with the SCCV.

RB design loads consist of:

- Service category of loads that occur during construction, pre-operational testing, or normal operation. They include:
  - Dead loads (D) which consist of the weight of structures, weight of permanently attached major equipment, tanks, machinery, and cranes; and weight of piping, cable, cable trays and duct supports
  - Hydrostatic pressure (F) of liquids in various pools
  - Live loads (L, L_r) which consist of floor area loads, laydown loads, nuclear fuel, and equipment handling loads
  - Lateral Soil and groundwater pressure loads (H)
  - Snow/rain loads (S/R)
  - Normal plant operation and pre-operation pressure testing loads which consist of operation service pressure loads, pre-operation proof test pressure load, normal thermal conditions (T_o) and operation service pipe and equipment reaction loads (R_o)

  - Construction Loads
  - Settlement Loads
  - Crane Loads (C) developed as discussed in Subsection 3.8.4.1.2

- Abnormal and environmental category of loads that occur during postulated accidents and/or severe or extreme environmental events. They include:
  - Abnormal plant operation loads which include accident pressure (P_a) and thermal (T_a) loads, accident pipe reaction loads (R_a), high-energy pipe rupture loads (Y_r), missile generation, pipe whip (Y_m), jet impingement from large pipe breaks (Y_j), blast pressure and compartment pressurization
  - Wind and Tornado or Hurricane loads (W, W_t)
  - Seismic loads (E_s) discussed in Subsection 3.7.1, including hydrodynamic loads on the pool walls calculated based on the approach described in ASCE/SEI 4, "Seismic Analysis of Safety-Related Nuclear Structures," (Reference 3.8-24) and American Concrete Institute (ACI) 350.3, "Code Requirements for Seismic Analysis and Design of Liquid-Containing Concrete Structures (ACI 350.3-20) and Commentary," (Reference 3.8-25), and dynamic lateral pressures from the interaction of the deeply embedded structure with the surrounding rock and soil.

Load Combinations

Load combinations and load factors for the design of the RB steel-plate composite structures and structural steel are in accordance with the provisions of Section NB2.5 of ANSI/AISC N690 including the supplemental guidance of Regulatory Positions 2.1 and 2.2 of NRC RG 1.243.

Table 3.8-7 provides the load combinations and load factors used in the design of the RB.

3.8.4.1.4 Design and Analysis Procedures

[Reserved. To be addressed in CPA Part 2.]

3.8.4.1.5 Structural Acceptance Criteria

For a comparison of the calculated stresses and strains in key RB structural components with the allowable limits, refer to the analysis results in Appendix 3H.

Design Basis Acceptance Criteria

The RB steel-plate composite and steel structures, including welded and bolted connections, are designed to meet the acceptance criteria outlined in ANSI/AISC N690 and in Section 5.0 of NEDC-33926P-A.

The RB structure is evaluated for serviceability considerations including deflection, vibration, permanent deformation, cracking, and settlement. Serviceability evaluations meet the acceptance criteria in Chapter NL of ANSI/AISC N690.

Seismic Design Criteria

Per Table 3.7-1, The seismic design of the RB structure considers LS-D response in accordance with ASCE/SEI 43 to ensure an essentially elastic response without any significant permanent deformations when subjected to SSE and meets the deformation acceptance criteria of Section 5.2.3 of ASCE/SEI 43.

Interaction Evaluation Criteria

The interaction of the RB structure with the adjacent Power Block structures is discussed in Subsections 3.3.2.3 and 3.7.2.8.

The stability of foundations under SSE and extreme wind loads are checked following the criteria in Subsection 3.8.5.4.3.

Design for Impulsive and Impactive Loads

The RB structure is designed for impulsive and impactive loads per the guidance of NUREG-0800, SRP 3.8.4.

The RB is designed to withstand intrusion by regulatory defined threats by selecting design parameters for the steel-plate composite modules, such as steel thickness, steel ductility, and concrete thickness. The RB design for impulsive and impactive loads follows the provisions of ANSI/AISC N690 and the relevant guidance of NRC RG 1.243. Allowable stresses and limits for the RB design for impulsive and impactive loads are provided in Section 5.8 of NEDC-33926P-A.

3.8.4.1.6 Material, Quality Control and Special Construction Techniques

Material

Materials used in construction of the RB steel-plate composite structures outside of containment are in accordance with Section 5.2 of NEDC-33926P-A and Section NA3 of ANSI/AISC N690.

RB structural steel materials conform with the guidance of NUREG-0800, SRP 3.8.4.

The NRIC Demonstration Program Prototype test conclusions are used for demonstrating the structural performance of DP-SC modules and the adequacy of the proposed material requirements and design approaches discussed in Sections 5.0 of NEDC-33926P-A.

The following are the properties of materials used in the construction of the RB:

Concrete Infill

The design compressive strength for the self-consolidating concrete used for the construction of the RB is determined as described in Section 5.2.1 of NEDC-33926P-A.

Concrete constituents used in the RB steel-plate composite modules are tested, batched, and mixed in accordance with ACI 349, “Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary,” (Reference 3.8-26).

High-density concrete used for shielding purposes is designed in accordance with NRC RG 1.69. Aggregates used in high-density concrete for radiation shielding purposes conform to ASTM C637, “Standard Specification for Aggregates for Radiation-Shielding Concrete,” (Reference 3.8-27) per ACI 349.

Steel Materials

The steel components of the RB steel-plate composite structures conform to the following:

- Steel faceplates/diaphragm plate: ASTM A572 Grade 50 or Grade 65
- Studs: ASTM A108 Type B

The RB structural steel components conform to ASTM A572 Grade 50 or Grade 65.

Steel Connections

Bolts used in structural steel bolted connections are ASTM F3125, “Standard Specification for High-Strength Structural Bolts and Assemblies, Steel and Alloy-Steel, Heat Treated, Inch

Dimensions 120 ksi and 150 ksi Minimum Tensile Strength, and Metric Dimensions 830 MPA and 1040 MPa Minimum Tensile Strength," (Reference 3.8-28), Grade A325 Type 1.

Bolts of the following grades may be used otherwise, when deemed necessary:

- ASTM F3125, Grade A490 Type 1
- ASTM A354, "Standard Specification for Quenched and Tempered Alloy-Steel Bolts, Studs, and Other Externally Threaded Fasteners," (Reference 3.8-29), Grade BD-
- ASTM A449, "Standard Specification for Hex Cap Screws, Bolts and Studs, Steel, Heat Treated, 120/105/90 ksi Minimum Tensile Strength, General Use," (Reference 3.8-30), Type 1
- ASTM A193, "Standard Specification for Alloy-Steel and Stainless-Steel Bolting for High-Temperature or High-Pressure Service and Other Special Purpose Applications," (Reference 3.8-31) for high service temperature applications, grade is subject to specific design requirements-

Welding Materials

Welding materials for the RB structure comply with the requirements of ANSI/AISC N690.

Pool Liners and Appurtenances

Construction, fabrication, and installation of metallic liners, non-metallic liners, coatings, joint sealants, and waterstops for structures where leak tightness is critical meet the guidance of NUREG-0800, SRP 9.1.2, "New and Spent Fuel Storage."

Pool liners are selected to withstand the environmental conditions they are to encounter during the life of the facility. Pool liners, and their connections, are designed to sustain high-temperatures, resist radiation exposure, resist impactive and abrasive loads, maintain leak tightness and maintain the pool inventory at acceptable levels during and after normal operating and design basis events. Pool liner materials or coatings are also designed to resist flaking or chipping.

Pool liner plates may be of the following type and grade:

- Carbon steel with stainless clad: ASTM A264, "Standard Specification for Stainless Chromium-Nickel Steel-Clad Plate," (Reference 3.8-32), (A516, "Standard Specification for Pressure Vessel Plates, Carbon Steel, for Moderate- and Lower-Temperature Service," (Reference 3.8-33), Grade 70 plus A240, "Standard Specification for Chromium and Chromium-Nickel Stainless-Steel Plate, Sheet, and Strip for Pressure Vessels and for General Application," (Reference 3.8-34), Type 304L)
- Stainless-steel: ASTM A240 Type 304L

Coatings and Other Corrosion Prevention

Corrosion protection measures used for the RB steel faceplates and structural steel are the same as those for the SCCV discussed in Subsection 3.8.1.4.8.

A sacrificial thickness is added to the inner and outer faceplates of the DP-SC modules for the below grade portion of the RB structural elements to protect against corrosion. The calculation of this sacrificial thickness is based on a conservative corrosion rate and the selection of a concrete mix with chemistry control. As stated in Subsection 3.8.1.4.8, the additional sacrificial thickness for corrosion tolerance is not considered in the below grade DP-SC modules strength or stiffness estimates.

Quality Control

Quality control procedures are established and implemented during the construction and inspection phases of the RB. These procedures cover the fabrication, furnishing, and installation of each structural item and specify the inspection and documentation requirements in accordance with the requirements in Section NA5 and Chapters NM and NN of ANSI/AISC N690, NRC RG 1.28 as required by NRC RG 1.243, and as discussed in Sections 5.16 and 5.17 of NEDC-33926P-A.

The quality control procedures governing the design, fabrication, construction, installation, and inspection of the RB are developed in compliance with the GVH quality assurance program.

Special Construction Techniques

The integrated RB is built using a modular construction technique using predominantly DP-SC modules. The fabrication and construction of the RB follow the requirements of Section 5.16 of NEDC-33926P-A.

3.8.4.1.7 Testing and Inservice Surveillance Requirements

Periodic inspection, and inservice monitoring programs are implemented to ensure the RB structure continues to meet its functional and performance requirements in compliance with 10 CFR 50.65 and following the guidance of NRC RG 1.160.

The aging management, ISI and testing of the RB steel-plate composite structures, in accessible and inaccessible areas, are performed as discussed in Section 5.18 of NEDC-33926P-A. Inspections and monitoring during construction and commissioning are performed as described in Section 3.2 of NEDO-33914-A. The post-construction testing and inservice surveillance programs for the RB below grade wall and foundation include monitoring of groundwater chemistry, settlements and differential displacements as discussed in Sections 3.3 and 3.4 of NEDO-33914-A.

3.8.4.2 Intervening Structural Elements Between Distribution Systems and Structures

Cable trays, conduits and heating, ventilation, and air conditioning ducts are supported at intervals by appropriately designed supports. The supports are attached to the walls, floors, and ceilings of structures as required by the arrangement. The type of support and spacing considered is determined by systems allowable spans which are governed by rigidity and stress. Bracing is provided where required.

Intervening structural elements between the distribution systems and equipment supports and the building structural elements are accounted for in the design of the structure. The intervening structural elements are designed to withstand the effects of seismic and dynamic loads to prevent the collapse and collision of distribution systems with Seismic Category I SSCs in accordance with the requirements of GDC 2 and 4 and the guidance of NUREG-0800, SRP 3.8.4. The seismic qualification of the intervening structural elements and their interaction evaluations are performed as discussed in Subsection 3.7.3.

The design of embedded load bearing steel materials in concrete structures will conform to the guidance of NRC RG 1.199, “Anchoring Components and Structural Supports in Concrete.”

3.8.4.3 Radwaste Building

3.8.4.3.1 Description of the Structure

The RWB houses process systems such as the liquid waste management system, the solid waste management system and a portion of the Offgas System including the charcoal adsorbers.

For the RWB location and general dimensions, refer to Section 1.2.

The RWB structure consists of a reinforced concrete exterior wall system with reinforced concrete floor slabs and interior columns. The RWB roof structure consists of a reinforced concrete deck on open web steel joists supported by steel columns. The lateral force resisting system of the structure consists of reinforced concrete shear walls supporting the reinforced concrete diaphragms at each floor/roof level. Interior reinforced concrete walls are provided for radiological shielding of the liquid and solid management systems and the Offgas System charcoal adsorbers.

The foundation type utilized for all BECPV Power Block structures adjacent to the Reactor Building will be provided in CPA Part 2.

Because of its main function of housing systems that collect, process and store radioactive waste, the RWB is categorized as a RW-IIa structure per NRC RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants." The RWB is also evaluated for interaction with the integrated RB under seismic and extreme wind loadings due to its proximity to the RB as discussed in Sections 3.3 and 3.7. A structural gap is maintained between the RB and RWB to prevent physical contact between the buildings during a seismic or extreme wind event.

3.8.4.3.2 Applicable Codes, Standards, and Specifications

The design, fabrication, construction, testing, and ISI of the RWB conforms to 10 CFR 50 and complies with the provisions of GDC 1, 2, and 4.

The RWB is designed in accordance with NRC RG 1.143, which establishes the appropriate 10 CFR 50 requirements, following the guidance of NUREG-0800, SRP 3.8.4. Analysis procedures for the RWB are in accordance with ASCE/SEI 4 and ASCE/SEI 43. Seismic interaction evaluation of the RWB follows the requirements in Subsection 3.7.2.8.

3.8.4.3.3 Loads and Loading Combinations

In accordance with the guidance of NUREG-0800, SRP 3.8.4, loads and load combinations used in the design of the RWB are in accordance with RG 1.143, Tables 2, 3 and Table 4.

3.8.4.3.4 Design and Analysis Procedures

The RWB forms part of the combined model used to evaluate the Structure-Soil-Structure Interaction (SSSI) effects of the surrounding Power Block structures on the integrated RB as described in Subsection 3.7.2.4.

The design and seismic interaction evaluation of the RWB structure are performed following the linear elastic analysis methodology using a standalone FE model with a sufficient level of refinement to allow for the accurate estimation of design stresses and forces. The ASCE/SEI 4 two-step analysis approach is used for the seismic qualification of the RWB structure. Using this method, seismic demands in the form of RWB zero period accelerations or foundation acceleration response spectra are developed from the RB SSI analyses for the RWB simplified dynamic model discussed in Subsection 3.7.2.3. A fixed base quasi-static analysis using the maximum nodal accelerations or response spectrum analysis using the input foundation response spectrum enveloped with the RWB design earthquake (1/2 SSE) is then performed on the standalone refined FE model of the RWB. Resulting forces and stresses are used for the seismic design of the RWB. The demands for the RWB foundation design are taken directly from the SSI analysis of the integrated RB with the surrounding Power Block structures, with input corresponding to the RWB design earthquake.

The RWB interaction and structural integrity evaluations are performed for SSE loads using the results of the seismic SSI analysis discussed in Subsection 3.7.2.1 and for extreme wind loads.

Seismic interaction evaluation of the RWB follows the requirements in Subsection 3.7.2.8. As mentioned in Subsection 3.7.2.8, a gap is maintained between the RB and the RWB to prevent physical contact during a seismic or extreme wind event.

The RWB design follows the guidance of NUREG-0800, SRP 3.8.4 and NRC RG 1.143. Design and detailing of the RWB structural steel is in accordance with ANSI/AISC N690 as supplemented by NRC RG 1.243. Design and detailing of the RWB structural concrete is in accordance with ACI 349, as supplemented by NRC RG 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)."

3.8.4.3.5 Structural Acceptance Criteria

The RWB structure is designed to meet the acceptance criteria outlined in ANSI/AISC N690 as endorsed and modified by NRC RG 1.243 and ACI 349 as endorsed and modified by NRC RG 1.142, following the guidance of NUREG-0800, SRP 3.8.4.

3.8.4.3.6 Material, Quality Control and Special Construction Techniques

Materials used in the construction of the RWB are in accordance with ACI 349 and ANSI/AISC N690, following the guidance of NUREG-0800, SRP 3.8.4 and NRC RG 1.143.

The RWB quality control procedures meet the guidance of NRC RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," NRC RG 1.143, Position 7, and ANSI/ANS-55.6, "Liquid Radioactive Waste Processing System for Light-Water Reactor Plants," (Reference 3.8-35).

The quality control procedures governing the design, fabrication, construction, installation, and inspection of the RWB are developed in compliance with the GVH quality assurance program.

The RWB does not use special construction techniques.

3.8.4.3.7 Testing and Inservice Surveillance Requirements

Refer to Chapter 11 for the testing and inservice requirements for process systems inside the RWB.

3.8.4.4 Augmented Power Block Structures

3.8.4.4.1 Control Building

The CB houses the Main Control Room (MCR) and electrical, control and instrumentation equipment. The CB also houses the egress route between the MCR and SCR, which is located in the RB. For the CB location and general dimensions, refer to Section 1.2.

The CB structure consists of reinforced concrete walls with steel columns, beams/girders and composite floors made of concrete and steel decking.

The foundation type utilized for all BECPV Power Block structures adjacent to the Reactor Building will be provided in CPA Part 2.

As indicated in Table 3A-2 and Section 3.7, the CB is a Seismic Category II structure designed per the "International Building Code (IBC)," (Reference 3.8-36) and evaluated for seismic and extreme wind interaction with the integrated RB due to its close proximity to the RB structure. The CB interaction evaluation is performed following the approach in Section 6 of NEDO-33914-A, as discussed in Sections 3.3 and 3.7.

To meet the habitability and safety requirements for the MCR occupants, the MCR and MCR to SCR egress route are also assigned a seismic Category II and evaluated for a full SSE, considering limited permanent deformation (LS-C), and extreme wind loading per Section 6 of NEDO-33914-A.

Following the guidance of Section 6.0 of NEDO-33914-A and meeting the guidance in Section II.2 of NUREG-0800, SRP 3.7.2 and NRC RG 1.29, "Seismic Design Classification for Nuclear Power Plants," the CB, MCR and egress route between the MCR and SCR are designed to maintain their structural integrity under an SSE or extreme wind event to:

- Prevent adverse interaction with the RB Seismic Category I SSCs
- Prevent incapacitating injury to MCR occupants
- Ensure safe egress of MCR occupants to the SCR in the RB

In addition to withstanding the SSE and extreme wind conditions, the MCR and MCR to SCR egress route are hardened by design against perforations by extreme wind missiles as directed by NRC RG 1.117, "Protection Against Extreme Wind Events and Missiles for Nuclear Power Plants."

The MCR is also designed to meet the bullet resistant requirements of 10 CFR 73.55(e)(5). MCR doors are designed to remain functional and able to open during and after the design basis extreme environmental event.

Unobstructed access of the egress route during and after the events which can initiate an evacuation of the MCR is also ensured by the design to secure the egress of MCR occupants to the SCR.

3.8.5 Foundations

[Reserved. To be addressed in CPA Part 2.]

3.8.5.1 Description of the Foundations

3.8.5.1.1 RB and Containment Common Mat Foundation

[Reserved. To be addressed in CPA Part 2.]

3.8.5.1.2 Other Power Block Foundations

The foundation type utilized for all BECPV Power Block structures adjacent to the Reactor Building will be provided in CPA Part 2.

The foundations are separated from the RB exterior wall with a gap that can accommodate structural displacements due to seismic and wind loads.

The Power Block structures' foundations are assigned the same seismic category as the superstructures they support. The analysis and design of the Seismic Category RW-IIa RWB foundation is in accordance with ACI 349. Structural design and analysis of CB, TB, Service Building and Reactor Auxiliary structures' foundations are in accordance with the IBC and ACI 318, "Building Code Requirements for Structural Concrete and Commentary," (Reference 3.8-37).

The bearing capacity of the in-situ soil materials supporting the Power Block foundations will be provided in CPA Part 2. Stability evaluations of the Power Block structures' foundations against sliding and overturning due to SSE, and extreme wind, and against flotation due to flooding, are performed to ensure the stability requirements under normal operating and accidental conditions discussed in Subsection 3.8.5.4.3 are met. Stability evaluations are performed using the results of the seismic SSI analyses discussed in Subsections 3.7.2 and 3.8.4.1.4. The liquefaction potential of the Power Block structures' foundations will be assessed in CPA Part 2.

Groundwater effects on the Power Block structures' foundations will be discussed in CPA Part 2.

3.8.5.2 Applicable Codes, Standards, and Specifications

Applicable codes, standards and specifications for the containment and RB common mat foundation are the same as those for the superstructures discussed in Subsections 3.8.1.2 and 3.8.4.1.2.

The jurisdictional boundary for the application of the NEDC-33926P-A to the containment mat foundation is the portion within the outer perimeter of the SCCV referred to as the inner mat foundation (see Figure 3.8-2).

The jurisdictional boundary for application of the ANSI/AISC N690 to the non-pressure retaining portion of the mat foundation is the portion spanning from the exterior surface of the SCCV to the exterior surface of the RB referred to as the outer mat foundation.

3.8.5.3 Loads and Load Combinations

Design loads and load combinations for the containment and RB common mat foundation are those of the superstructures described in Subsections 3.8.1.3 and 3.8.4.1.3.

For foundation stability against flotation, the site-specific design basis flood is considered in combination with the dead load in accordance with NUREG-0800, SRP 3.8.5, “Foundations.”

3.8.5.4 Design and Analysis Procedures

[Reserved. TO be provided in CPA Part 2.]

3.8.5.4.1 Equivalent Linear Subgrade Static Properties

The structural design demands due to static earth pressures on the RB below grade exterior walls are obtained from the 1-g static analyses of the integrated RB FE model embedded in a layered half-space continuum model representing the surrounding soil and rock. To account for the interaction of the RB integrated structures with the surrounding subgrade, super-elements representing the stiffness properties of the layered subgrade materials are used in the static and thermal analyses, as described in Subsection 3.8.4.1.4.

The 1-g static SSI analyses, subgrade impedance analyses and thermal stress analyses use profiles of equivalent linear soil and rock properties developed using information from the BECPV site investigation and subsurface material testing programs summarized in Subsection 2.5.4. These properties are developed following Section 5.2.1 of NEDO-33914-A. The properties consist of:

- Effective unit weight that for soil materials below groundwater table are calculated as the total saturated unit weight of soil minus the unit weight of water
- Elastic and shear Modulus representing linearized stiffness properties of the soil and rock for long-term static loading conditions
- Soil and rock Poisson’s ratios representative of at-rest lateral pressure conditions

The site-specific geotechnical parameters and static subgrade properties used as inputs for the BECPV 1-g static Soil-Structure Interaction and subgrade impedance analyses will be discussed in CPA Part 2.

The groundwater level considered in the development of the equivalent linear subgrade static properties is discussed in Subsection 3.7.1.3.

Table 3.8-8 provides a summary of the linearized static properties for the 1-g SSI and impedance analyses for internal mechanical loads. Static properties for the thermal stress analysis are provided in Table 3.8-9.

In accordance with the guidance of Section 5.2.1.1 of NEDO-33914-A, the soil Poisson's ratios (v_{st}) are calculated as follows using the at-rest lateral (k_0) coefficient values provided in Table 3.8-8 and Table 3.8-9:

$$v_{st} = \frac{K_0}{1 + K_0}$$

The UB unit weight, LB static elastic modulus (E_{st}) and UB coefficient of at-rest pressure (K_0) are used for the 1-g SSI analyses resulting in larger deformation at soil-structure interfaces and conservative design stress demands.

3.8.5.4.2 Soil Bearing Stability of Integrated RB Mat Foundation

The static and dynamic stability of soil supporting the integrated RB mat foundation will be evaluated in CPA Part 2 per the guidance of NUREG-0800, SRP 2.5.4, "Stability of Subsurface Materials and Foundations."

The bearing capacity of the subgrade supporting the integrated RB mat foundation at BECPV will be provided in CPA Part 2.

The dynamic bearing pressure demands under SSE loads are calculated from the results of the seismic SSI analyses as described in Subsection 3.7.2.

3.8.5.4.3 Stability of Power Block Foundations

The integrated RB mat foundation stability is assessed against sliding and overturning due to SSE and extreme wind, and flotation following the guidance of Section II.4 of NUREG-800, SRP 3.8.5.

Explicit sliding and overturning stability evaluations are not performed for the deeply embedded RB because, in accordance with Section 7.2 of ASCE/SEI 43, its center of gravity is located deep below the grade elevation, and the structure is inherently stable against sliding and overturning.

Required safety factors against sliding and overturning for the Power Block foundations under normal operating and accidental conditions are presented in Table 3.8-10. Safety factors listed in Table 3.8-10 conform to the allowable values specified in Section II.5 of NUREG-0800, SRP 3.8.5.

3.8.5.5 Structural Acceptance Criteria

The structural acceptance criteria for the containment and RB common mat foundation are the same as those for their respective superstructures described in Subsections 3.8.1.5 and 3.8.4.1.5. For safety factors considered for soil bearing and foundations stability, refer to Subsections 3.8.5.4.2 and 3.8.5.4.3, respectively.

3.8.5.6 Materials, Quality Control, and Special Construction Techniques

3.8.5.6.1 Materials

[Reserved. To be addressed in CPA Part 2.]

3.8.5.6.2 Quality Control

Refer to Subsections 3.8.1.6.2 and 3.8.4.1.6 for discussion.

3.8.5.6.3 Special Construction Techniques

Refer to Subsection 3.8.4.1.6 for discussion.

3.8.5.7 Testing and Inservice Surveillance Programs

The inspection and testing of the integrated RB mat foundation follow the guidance of Sections 3.2.2 and 3.4 of NEDO-33914-A, and Sections 5.18 and 6.22 of NEDC-33926P-A. The foundation testing and surveillance program complies with the requirements of 10 CFR 50.65 and guidance of NRC RG 1.160 and includes the use of non-destructive testing and remote visual monitoring to monitor conditions in inaccessible areas of the foundation.

In accordance with Section 3.4 of NEDO-33914-A, field instrumentation is installed inside and outside of the RB to measure the distribution of pore pressures around and below the structure, total settlement and tilt of the RB including foundation, and measurement of the surrounding Power Block structures. The instrumentation provides recordings during excavation and continuing through the plant operation that can be benchmarked against design estimates using results of the FIA. Short-term and long-term settlement monitoring plans are also developed to detect:

- Vertical and horizontal movements in and around the integrated RB
- Differential distortion across the foundation footprint
- Differential settlements between the containment and RB portions of the mat foundation

3.8.6 References

3.8-1 NEDC-33926P-A, “BWRX-300 Steel-Plate Composite (SC) Containment Vessel (SCCV) and Reactor Building Structural Design,” Revision 4, GE Vernova Hitachi Nuclear Energy Americas, LLC, December 2025.

3.8-2 ASME BPVC, Section III, “Rules for Construction of Nuclear Facility Components,” Edition 2021, Division 2, “Code for Concrete Containments,” American Society of Mechanical Engineers.

3.8-3 ASME BPVC, Section II, “Materials,” Edition 2021, American Society of Mechanical Engineers.

3.8-4 ASME BPVC, Section III, “Rules for Construction of Nuclear Facility Components,” Edition 2021, Subsection NCA, “General Requirements for Division 1 and Division 2,” American Society of Mechanical Engineers.

3.8-5 ASME BPVC, Section V, “Non-Destructive Examination,” Edition 2021, American Society of Mechanical Engineers.

3.8-6 ASME BPVC, Section IX, “Qualification Standard for Welding, Brazing, and Fusing Procedures; Welders; Brazers; and Welding, Brazing, and Fusing Operators,” Edition 2021, American Society of Mechanical Engineers.

3.8-7 ASME BPVC, Section XI, Division 1, “Rules for Inspection and Testing of Components of Light-Water-Cooled Plants,” American Society of Mechanical Engineers.

3.8-8 ANSI/AISC N690-18, “Specification for Safety-Related Steel Structures for Nuclear Facilities,” American National Standards Institute/American Institute of Steel Construction, 2018.

3.8-9 ASME BPVC, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” Edition 2021, Subsection NE, “Class MC Components,” American Society of Mechanical Engineers.

3.8-10 NEDO-33914-A, “BWRX-300 Advanced Civil Construction and Design Approach,” Revision 2, GE-Hitachi Nuclear Energy Americas, LLC, June 2022.

3.8-11 ASTM D5144-08, “Standard Guide for Use of Protective Coating Standards in Nuclear Power Plants,” ASTM International, 2008.

3.8-12 ASCE/SEI 43-19, “Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities,” American Society of Civil Engineers, 2019.

3.8-13 ASTM A709/A709M-21, “Standard Specification for Structural Steel for Bridges,” ASTM International, 2021.

3.8-14 ASME BPVC, Code Case N-763, “ASTM A 709-06, Grade HPS 70W (HPS 485W) Plate Material Without Postweld Heat Treatment as Containment Liner Material or Structural Attachments to the Containment Liner, Subsection CC, Section III, Division 2,” American Society of Mechanical Engineers, Edition 2008.

3.8-15 ASTM A108-18, “Standard Specification for Steel Bar, Carbon and Alloy, Cold-Finished,” ASTM International, 2018.

3.8-16 ASTM A572/A572M-18, “Standard Specification for High-Strength Low-Alloy Columbium-Vanadium Structural Steel,” ASTM International, 2018.

3.8-17 ASME BPVC, Code Case N-632, “Use of ASTM A 572 Grades 50 and 65 for Structural Attachments to Class CC Containment Liners, Section III, Division 2” American Society of Mechanical Engineers, 1999.

3.8-18 ASME BPVC, Code Case N-284-4, “Metal Containment Shell Buckling Design Methods, Class MC, TC, SC Construction, Section III, Division 1 and Division 3,” American Society of Mechanical Engineers, 2012.

3.8-19 ASME BPVC, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” Edition 2021, Subsection NF, “Supports,” American Society of Mechanical Engineers.

3.8-20 ASTM A500/A500M-18, “Standard Specification for Cold-Formed Welded and Seamless Carbon Steel Structural Tubing in Rounds and Shapes,” ASTM International, 2018.

3.8-21 ASTM A1085/1085M-15, “Standard Specification for Cold-Formed Welded Carbon Steel Hollow Structural Sections,” ASTM International, 2015.

3.8-22 ASCE/SEI 7-16, “Minimum Design Loads and Associated Criteria for Buildings and Other Structures,” American Society of Civil Engineers, 2016.

3.8-23 ASME NOG-1-20, “Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder),” American Society of Mechanical Engineers, 2020.

3.8-24 ASCE/SEI 4-16, “Seismic Analysis of Safety-Related Nuclear Structures,” American Society of Civil Engineers, 2017.

3.8-25 ACI 350.3-06, “Seismic Design of Liquid-Containing Concrete Structures and Commentary,” American Concrete Institute, 2006.

3.8-26 ACI 349-13, “Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary,” American Concrete Institute, 2014.

3.8-27 ASTM C637-20, “Standard Specification for Aggregates for Radiation-Shielding Concrete,” ASTM International, 2020.

3.8-28 ASTM F3125-21, “Standard Specification for High-Strength Structural Bolts and Assemblies, Steel and Alloy-Steel, Heat Treated, Inch Dimension 120 ksi and 150 ksi Minimum Tensile Strength, and Metric Dimensions 830 MPA and 1040 MPA Minimum Tensile Strength,” ASTM International, 2021.

3.8-29 ASTM A354-17e2, “Standard Specification for Quenched and Tempered Alloy-Steel Bolts, Studs, and Other Externally Threaded Fasteners,” ASTM International, 2017.

3.8-30 ASTM A449-14(2020), “Standard Specification for Hex Cap Screws, Bolts and Studs, Steel, Heat Treated, 120/105/90 ksi Minimum Tensile Strength, General Use,” ASTM International, 2020.

3.8-31 ASTM A193/A193M-20, “Standard Specification for Alloy-Steel and Stainless-Steel Bolting for High-Temperature or High-Pressure Service and Other Special Purpose Applications,” ASTM International, 2020.

3.8-32 ASTM A264-12(2019), “Standard Specification for Stainless Chromium-Nickel Steel-Clad Plate,” ASTM International, 2019.

3.8-33 ASTM A516/A516M-17, “Standard Specification for Pressure Vessel Plates, Carbon Steel, for Moderate- and Lower-Temperature Service,” ASTM International, 2017.

3.8-34 ASTM A240/A240M-22a, “Standard Specification for Chromium and Chromium-Nickel Stainless-Steel Plate, Sheet, and Strip for Pressure Vessels and for General Application,” ASTM International, 2022.

3.8-35 ANSI/ANS-55.6, “Liquid Radioactive Waste Processing System for Light-Water Reactor Plants,” American Nuclear Society, 1993 (R2007).

3.8-36 IBC-2021, “International Building Code,” International Code Council, 2021.

3.8-38 ACI 318-19(R2022), “Building Code Requirements for Structural Concrete and Commentary,” American Concrete Institute, 2022.

Table 3.8-1: Load Combinations and Load Factors for Steel-Plate Composite Containment Vessel

Load Category	Combination #	Loads																		Load Condition
		D	F	L	H	P_t	T_t	P_v/ P_o	R_o	T_o	W	E_s	W_t	P_a	R_a	T_a	Y_j/ R_{rj}	Y_m/ R_{rm}	Y_r/ R_{rr}	
Test	SCCV-1	1.0	1.0	1.0	1.0	1.0	1.0													Service
Construction	SCCV-2	1.0	1.0	1.0	1.0					1.0	1.0									
Normal	SCCV-3	1.0	1.0	1.0	1.0			1.0	1.0	1.0										
Severe Environmental	SCCV-4	1.0	1.0	1.30	1.30			1.0	1.0	1.0	1.5									Factored
Extreme Environmental	SCCV-5	1.0	1.0	1.0	1.0			1.0	1.0	1.0		1.0								
	SCCV-6	1.0	1.0	1.0	1.0			1.0	1.0	1.0			1.0							
Abnormal	SCCV-7	1.0	1.0	1.0	1.0									1.5	1.0	1.0				
	SCCV-8	1.0	1.0	1.0	1.0									1.0	1.25	1.0				
	SCCV-9	1.0	1.0	1.0	1.0									1.25	1.0	1.0				
Abnormal / Extreme Environmental	SCCV-10	1.0	1.0	1.0	1.0						1.25			1.25	1.0	1.0				
	SCCV-11	1.0	1.0	1.0	1.0							1.0		1.0	1.0	1.0	1.0	1.0	1.0	

Notes:

(1) The load combination involving post-LOCA flooding condition with OBE (set at 1/3 of the SSE, see Subsection 3.7.1) is eliminated in accordance with NUREG-0800 SRP 3.8.1. This load combination is bounded by SCCV-11 because the maximum hydrostatic pressure produced by the post-LOCA flooding is less significant than the accident LOCA pressure.

(2) D = Dead Loads
F = Hydrostatic Pressure Loads
L = Live Loads
H = Soil and Underground Water Pressure Loads
P_t = Test Pressure Loads
T_t = Test Thermal Loads
P_v/P_o = Pressure Variant Load
R_o = Normal Operating Pipe Reaction Loads
T_o = Normal Operating Thermal Loads
W = Wind Loads
E_s = Seismic Loads (generated by SSE)
W_t = Tornado or Hurricane Loads
P_a = Accident Pressure Loads
R_a = Accident Pipe Reaction Loads
T_a = Accident Thermal Loads
Y_j/R_{rj} = Jet Impingement Loads
Y_m / R_{rm} = Missile Impact Loads
Y_r / R_{rr} = High-Energy Pipe Rupture

Table 3.8-2a: Acceptance Criteria for BWRX-300 Steel-Plate Composite Containment Vessel

Allowable Stress/Strain Limits for Factored Loads

Material	Force Classification	Type of Force Action	Criteria for Factored Loads	
			Stress Limit	Strain Limit, if any
Concrete	Primary	Membrane	$0.60f_c'$	-
		Membrane + Bending (2)	$0.75f_c'$	-
	Primary + Secondary (1)	Membrane	$0.75f_c'$	-
		Membrane + Bending (2)	$0.85f_c'$ (3)	-
Steel Plates	Primary	Membrane or Membrane + Bending (2)	$0.90F_y$	-
	Primary + Secondary (1)	Membrane or Membrane + Bending (2)	-	$2\varepsilon_y$ (4)

Notes:

(1) The primary portion of the calculated stress is not to exceed the allowable stress applicable when primary stress acts alone.

(2) The membrane portion of the calculated stress is not to exceed the allowable stress applicable when membrane stress acts alone.

(3) The maximum allowable primary-plus-secondary membrane and bending compressive stress of 0.85 f′c corresponds to a limiting strain of 0.002 in/in. The concrete allowable stresses for factored compression loads are reduced, if necessary, to maintain structural stability per CC-3421.1 of ASME BPVC, Section III, Division 2.

(4) Limit for mechanical (net) strain due to primary forces, which is calculated by subtracting strain induced by secondary force from the total strain as per CC-3422.1(e) of ASME BPVC, Section III, Division 2. For analysis purposes, ϵ_y is defined as the yield stress divided by Young's modulus.

Table 3.8-2b: Acceptance Criteria for BWRX-300 Steel-Plate Composite Containment Vessel

Allowable Stresses for Service Loads

Material	Force Classification	Type of Force Action	Criteria for Service Loads
			Stress Limit
Concrete	Primary	Membrane	$0.30f_c'$
		Membrane + Bending (2)	$0.45f_c'$
	Primary + Secondary (1)	Membrane	$0.45f_c'$
		Membrane + Bending (2)	$0.60f_c'$
Steel Plates	Primary	Membrane or Membrane + Bending (2)	$0.50F_y$ (3) (4)
	Primary + Secondary (1)	Membrane or Membrane + Bending (2)	$0.67F_y$ (3) (4)

Notes:

(1) The primary portion of the calculated stress is not to exceed the allowable stress applicable when primary stress acts alone.

(2) The membrane portion of the calculated stress is not to exceed the allowable stress applicable when membrane stress acts alone.

(3) Where the steel plates are under tension stress, the stress limit may be increased by 50%, with an upper limit of $0.9F_y$ when the temporary pressure loads during the test condition are combined with other loads in the load combination.

(4) Where the steel plates are under compression stress, the stress limit may be increased by 33%, with an upper limit of $0.9F_y$ when the temporary pressure loads during the test condition are combined with other loads in the load combination.

Table 3.8-3: Load Combinations and Load Factors for Containment Closure Head

Service Level	Load Combination									
	D	L	E_s	T_o	T_a	T_t	P_o	P_a	P_t	R_r
Test Condition	1	1				1			1	
Design Condition	1	1			1			1		
Level A	1	1		1			1			
	1	1			1			1		
Level C	1	1	1		1			1		
	1	1	1	1			1			
Level D	1	1	1		1			1		1

Notes:

(1) Based on NUREG-0800, SRP 3.8.2 and NRC RG 1.57

(2) D = Dead Load

L = Live Load

E_s = Seismic Loads (generated by SSE)

T_o = Normal Operating Thermal Loads

T_a = Accident Thermal Loads

T_t = Test Thermal Loads

P_o = Normal Operating Pressure Loads including Pressure Variant Loads (P_v)

P_a = Accident Pressure Loads

P_t = Test Pressure Loads

R_r = Local Effects on Containment due to LOCA (R_{rj} + R_{rm})

(3) In accordance with NRC RG 1.216, combustible gas load cases (Pg1, Pg2, and Pg3) and associated load combinations are associated with DECs and are currently not presented in this table. The structural integrity of the containment closure head is demonstrated by meeting the requirements in ASME BPVC, Section III, Subsection NE for the applicable load combinations comprising of dead load and internal pressure specified in NRC RG 1.57 with a minimum internal pressure of 310 kPa (45 psig).

Table 3.8-4: Load Combinations and Load Factors for Class MC Components

Service Level	Load Combination												
	D	L	E_s	T_o	T_a	T_t	P_o	P_a	P_t	R_r	R_o	R_a	H_a
Test Condition	1	1				1			1				
Design Condition	1	1			1			1				1	
Level A	1	1		1			1				1		
	1	1			1			1				1	
Level C	1	1	1		1			1				1	
	1	1	1	1			1				1		
Level D	1	1	1		1			1		1		1	
Post-flooding Condition	1	1	1[2]										1

Notes:

(1) Based on NUREG-0800, SRP 3.8.2 and NRC RG 1.57

(2) OBE consideration only for post-flooding condition and cyclic loading considerations, in accordance with NUREG-0800, SRP 3.8.2

(3) D = Dead Load

L = Live Load

E_s = Seismic Loads (generated by SSE)

T_o = Normal Operating Thermal Loads

T_a = Accident Thermal Loads

T_t = Test Thermal Loads

P_o = Normal Operating Pressure Loads including Pressure Variant Loads (Pv)

P_a = Accident Pressure Loads

P_t = Test Pressure Loads

R_r = Local Effects on Containment due to LOCA (Rrr + Rrj + Rrm)

R_o = Normal Operating Pipe Reaction Loads

R_a = Accident Pipe Reaction Loads

H_a = Internal Flooding Loads

(4) In accordance with NRC RG 1.216, Combustible gas load cases (Pg1, Pg2, and Pg3) and associated load combinations are associated with DECs and are currently not presented in this table. The structural integrity of the Class MC components is demonstrated by meeting the requirements in ASME BPVC, Section III, Subsection NE for the applicable load combinations comprising of dead load and internal pressure specified in NRC RG 1.57 with a minimum internal pressure of 310 kPa (45 psig).

Table 3.8-5: Acceptance Criteria for Containment Closure Head

Service Level	Acceptance Criteria[*1]			
	P_m	P_L	P_L+P_b [*2]	P_L+P_b+Q
Test Condition	0.8 S_y	1.15 S_y	1.15 S_y	N/A[*3]
Design Condition	1.0 S_{mc}	1.5 S_{mc}	1.5 S_{mc}	N/A[*3]
Level A	1.0 S_{mc}	1.5 S_{mc}	1.5 S_{mc}	3.0 S_m
Level C	1.2 S_{mc} or[*4] 1.0 S_y	1.8 S_{mc} or[*4] 1.5 S_y	1.8 S_{mc} or[*4] 1.5 S_y	N/A[*3]
Level D	S_f	1.5 S_f	1.5 S_f	N/A[*3]

Notes:

*1: Acceptance Criteria is defined by ASME BPVC, Subsection NE NE-3221.1 through 3221.4.

P_m = primary stress: general membrane.

P_L = primary stress: local membrane.

P_b = primary stress: bending.

Q = secondary stress: membrane plus bending.

S_y = material's yield strength at temperature as in ASME BPVC, Section II, Part D, Table Y-1.

S_m = allowable stress intensity S_m is the value given in ASME BPVC, Section II, Part D, Subpart 1, Tables 2A and 2B.

S_{mc} = allowable stress intensity S_{mc} is 1.1 times the S listed in ASME BPVC, Section II, Part D, Subpart 1, Tables 1A and 1B, except S_{mc} is not to exceed 90% of the material's yield strength at temperature shown in ASME BPVC, Section II, Part D, Subpart 1, Tables Y-1.

S_f = 85% of the general primary membrane allowable permitted in Mandatory Appendix XXVII of ASME BPVC, Section III. In the application of Appendix XXVII, S_m, if applicable, is as specified in NE-3112.4(a)(1).

*2: Values shown are for a rectangular section. See ASME BPVC, Subsection NE, NE-3221.3(d) for other than a solid rectangular section.

*3: N/A = Not applicable. No evaluation required.

*4: The larger of the two values listed is chosen as a limit load.

Table 3.8-6: Acceptance Criteria for Other Class MC Components

Service Level	Acceptance Criteria*1			
	P_m	P_L	P_L+P_b *2	P_L+P_b+Q
Test Condition	0.8 S_y	1.15 S_y	1.15 S_y	N/A*3
Design Condition	1.0 S_{mc}	1.5 S_{mc}	1.5 S_{mc}	N/A*3
Level A	1.0 S_{mc}	1.5 S_{mc}	1.5 S_{mc}	3.0 S_m
Level C	1.2 S_{mc} or*4 1.0 S_y	1.8 S_{mc} or*4 1.5S_y	1.8 S_{mc} or*4 1.5 S_y	N/A*3
Level D	S_f	1.5 S_f	1.5 S_f	N/A*3
Post-flooding Condition	1.2 S_{mc} or*4 1.0 S_y	1.8 S_{mc} or*4 1.5 S_y	1.8 S_{mc} or*4 1.5 S_y	3.0 S_m

Notes:

*1: Acceptance Criteria for other than Post-flooding Condition is defined by ASME BPVC, Subsection NE NE-3221.1 through 3221.4. For Post-flooding Condition, Service Level C limits apply to primary stress, and Service Level B limits apply to primary plus secondary stress, per item 5 of NUREG-0800 SRP 3.8.2.

P_m = primary stress: general membrane.

P_L = primary stress: local membrane.

P_b = primary stress: bending.

Q = secondary stress: membrane plus bending.

S_y = material's yield strength at temperature as in ASME BPVC, Section II, Part D, Table Y-1.

S_m = allowable stress intensity S_m is the value given in ASME BPVC, Section II, Part D, Subpart 1, Tables 2A and 2B.

S_{mc} = allowable stress intensity S_{mc} is 1.1 times the S listed in ASME BPVC, Section II, Part D, Subpart 1, Tables 1A and 1B, except S_{mc} is not to exceed 90% of the material's yield strength at temperature shown in ASME BPVC, Section II, Part D, Subpart 1, Tables Y-1.

S_f = 85% of the general primary membrane allowable permitted in Mandatory Appendix XXVII of ASME BPVC, Section III. In the application of Appendix XXVII, S_m, if applicable, is as specified in NE-3112.4(a)(1).

*2: Values shown are for a rectangular section. See ASME BPVC, Subsection NE, NE-3221.3(d) for other than a solid rectangular section.

*3: N/A = Not applicable. No evaluation required.

*4: The larger of the two values listed is chosen as a limit load.

Table 3.8-7: Load Combinations, Load Factors and Acceptance Criteria for Reactor Building Steel-Plate Composite Modules and Steel Structures

Load Category	Combination #	Loads																				
		D	F	L	H	L_r	R	S	C	R_o	P_o	T_o	W	E_s	W_t	S_x	P_a	R_a	T_a	Y_j	Y_m	Y_r
Normal	RB-1	1.4	1.4						1.0	1.4	1.0	1.0										
	RB-2.1	1.2	1.2	1.6	1.6	0.5			1.4	1.2	1.2	1.2										
	RB-2.2	1.2	1.2	1.6	1.6		0.5		1.4	1.2	1.2	1.2										
	RB-2.3	1.2	1.2	1.6	1.6			0.5	1.4	1.2	1.2	1.2										
	RB-3.1	1.2	1.2	0.8	0.8	1.6			1.4	1.2	1.2	1.2										
	RB-3.2	1.2	1.2	0.8	0.8		1.6		1.4	1.2	1.2	1.2										
	RB-3.3	1.2	1.2	0.8	0.8			1.6	1.4	1.2	1.2	1.2										
	RB-2/3	1.2	1.2	1.6	1.6			1.6	1.4	1.2	1.2	1.2										
Severe Environmental	RB-4.1	1.2	1.2	1.6	1.6	0.5			1.0	1.2	1.0	1.0	1.0									
	RB-4.2	1.2	1.2	1.6	1.6		0.5		1.0	1.2	1.0	1.0	1.0									
	RB-4.3	1.2	1.2	1.6	1.6			0.5	1.0	1.2	1.0	1.0	1.0									

Load Category	Combination #	Loads																				
		D	F	L	H	L_r	R	S	C	R_o	P_o	T_o	W	E_s	W_t	S_x	P_a	R_a	T_a	Y_j	Y_m	Y_r
Extreme Environmental	RB-5	1.0	1.0	0.8	1.0				1.0	1.0	1.0	1.0		1.0								
	RB-6	1.0	1.0	0.8	1.0					1.0	1.0	1.0			1.0							
	RB-7	1.0	1.0	0.8	1.0					1.0	1.0	1.0				1.0						
Abnormal	RB-8	1.0	1.0	0.8	1.0				1.0								1.4	1.0	1.0			
	RB-9	1.0	1.0	0.8	1.0									0.7			1.0	1.0	1.0	1.0	1.0	1.0

Notes:

(1) Load combinations are in accordance with ANSI/AISC N690 and NRC RG 1.243. Load combination RB2/3 is a conservative bounding load combination for RB-2.1, RB- 2.2, RB-2.3, RB-3.1, RB-3.2, and RB-3.3 required by ANSI/AISC N690.

D = Dead Loads, including settlements

F = Hydrostatic Pressure Loads

L = Live Loads

H = Soil and Underground Water Pressure Loads

L_r = Roof Live Loads

R = Rain Loads

S = Snow Loads

C = Crane Loads

R_o = Normal Operating Pipe Reactions

P_o = Normal Operating Pressure Loads

T_o = Normal Operating Thermal Loads

W = Wind Loads

E_s = Seismic Loads (Generated by SSE)

W_t = Tornado or Hurricane Wind Loads

S_x = Extreme Snow Loads

P_a = Accidental Pressure Loads

R_a = Accidental Piping Reactions

T_a = Accidental Thermal Loads

Y_j = Jet Impingement Loads from a ruptured high-energy pipe

Y_m = Missile Impact Loads from a ruptured high-energy pipe

Y_r = High-Energy Pipe Rupture

(2) When Pa and Ta are time-dependent loads, their effects are superimposed accordingly.

(3) Loads due to accidental explosions, accidental vehicle impacts, extreme snow load, and small aircraft impacts are evaluated using load combination RB-6 with those loads in lieu of W_t, following the guidance provided in Appendix N9 to ANSI/AISC N690.

Table 3.8-8: As-Built Subgrade Properties for Static 1-g Soil-Structure Interaction and Impedance Analyses

To be provided in CPA Part 2.

Table 3.8-9: As-Built Static Subgrade Properties for Thermal Analysis

To be provided in CPA Part 2.

Table 3.8-10: Stability Requirements for Power Block Foundations

Load Combination	Overturning	Sliding	Flotation
D + H + W	*1.5*	*1.5*	
D + H + E'	*1.1*	*1.1*	
D + H + W_t	*1.1*	*1.1*	
D + F'			*1.1*

Where:

D = Dead Load, W = Wind

H = Lateral soil pressure, E' = Safe Shutdown Earthquake

Wt = Tornado or Hurricane Wind

F' = Buoyant forces of design basis flood

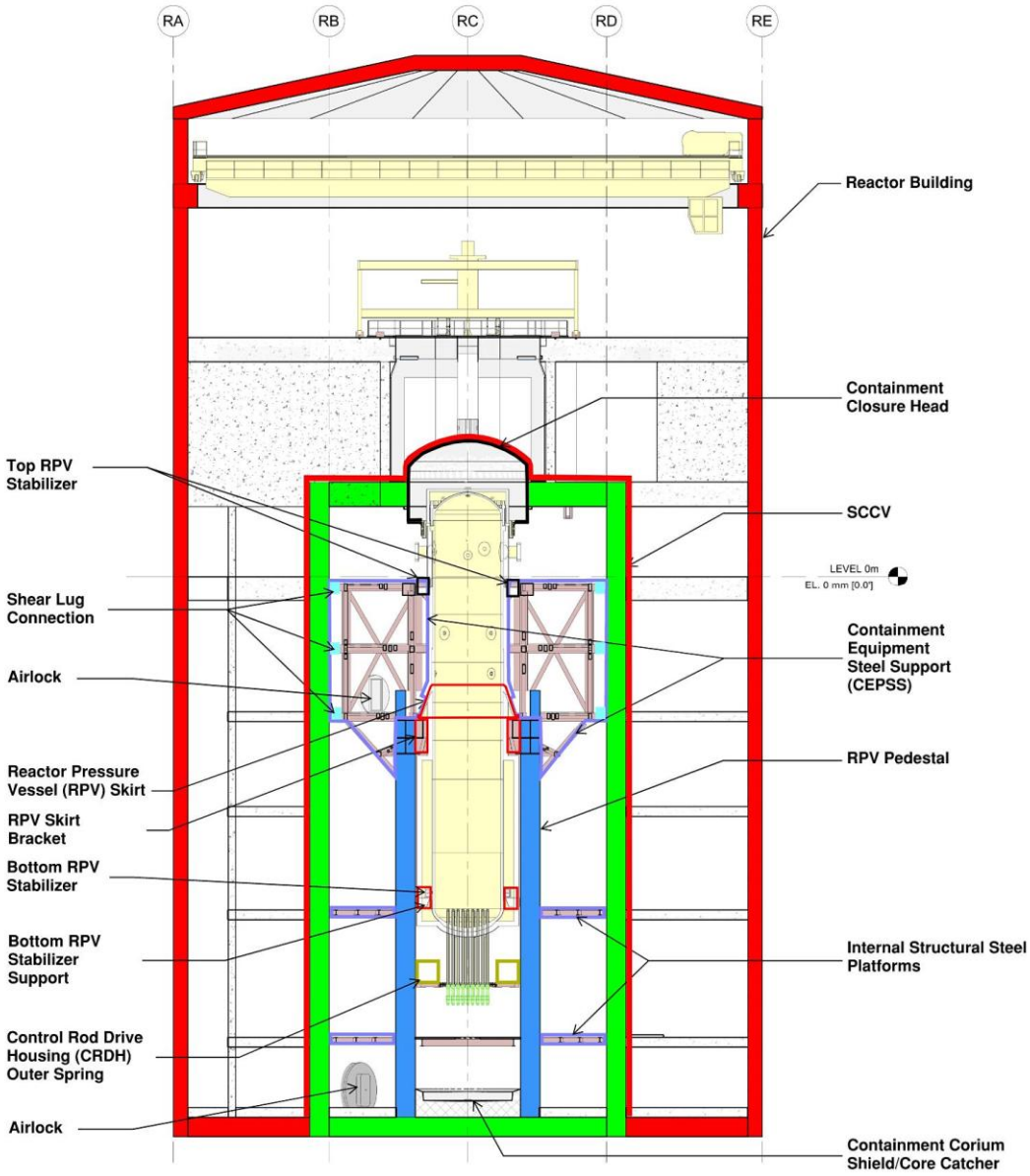
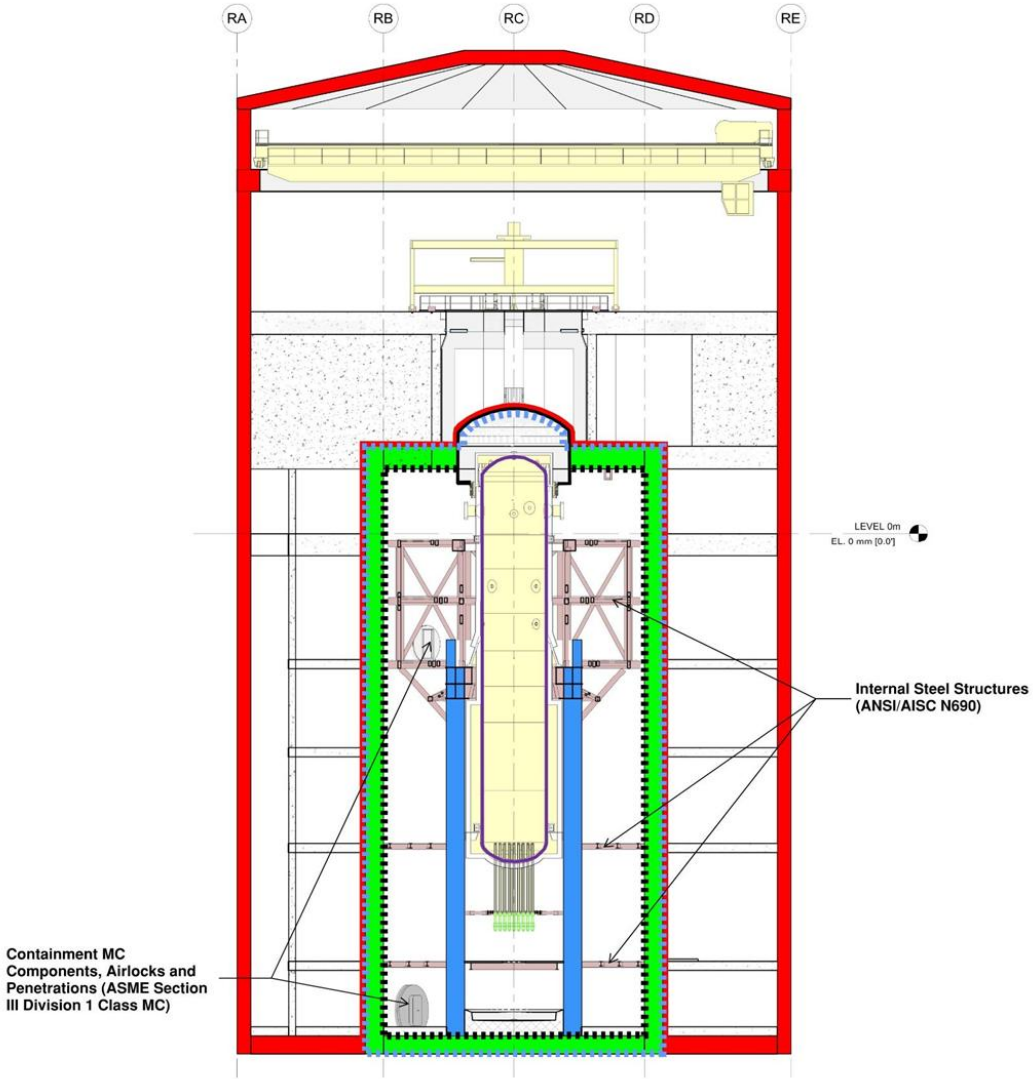


Figure 3.8-1: BWRX-300 Integrated Reactor Building Structures Including Mat Foundation

*For illustration purposes only. Integrated RB layout is based on current design and may change as the design progresses. Penetrations not included in this sectional view.



Legend	Component	Code Applicability
	Steel-Plate Composite Containment Vessel (SCCV)	NEDC-33926P, Section 6.0
	Reactor Building (RB)	NEDC-33926P, Section 5.0
	RPV Pedestal	NEDC-33926P, Section 5.0
	Reactor Pressure Vessel (RPV)	ASME Section III Division 1 Class 1
	SCCV Closure Head	ASME Section III Division 1 Class MC
	ASME Containment Leakage Boundary	
	Containment Pressure Boundary	

Figure 3.8-2: Structural Boundary of the BWRX-300 Containment, Containment Internal Structures and Reactor Building

*For illustration purposes only. Integrated RB layout is based on current design and may change as the design progresses

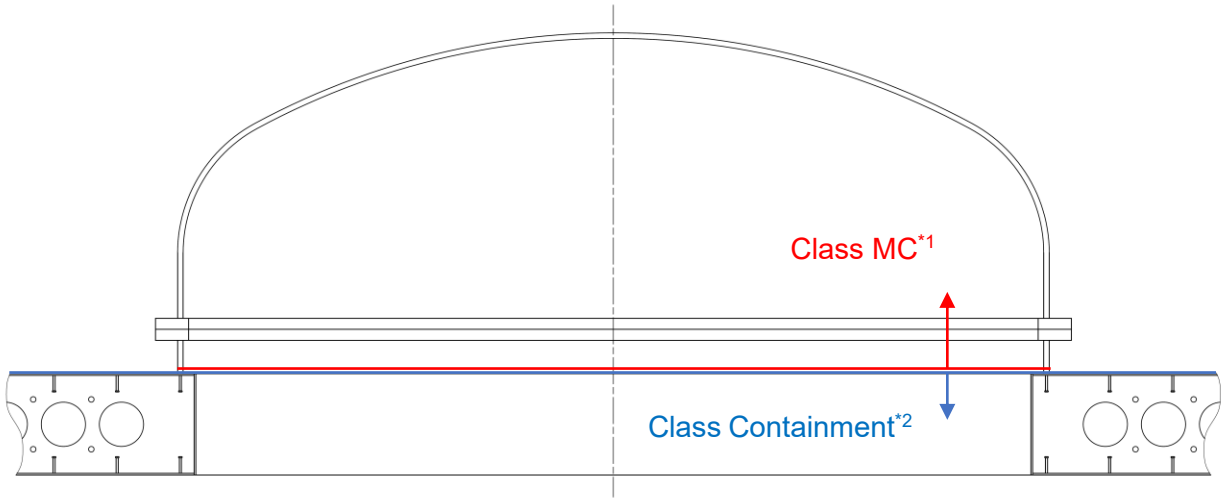


Figure 3.8-3: Containment Closure Head Structure Boundary

*1: Designed in accordance with ASME BPVC, Section III, Subsection NE (for Class MC)

*2: Designed in accordance with NEDC-33926P-A

*3: Structural overview is provided for illustration purposes only and will be updated as the design progresses

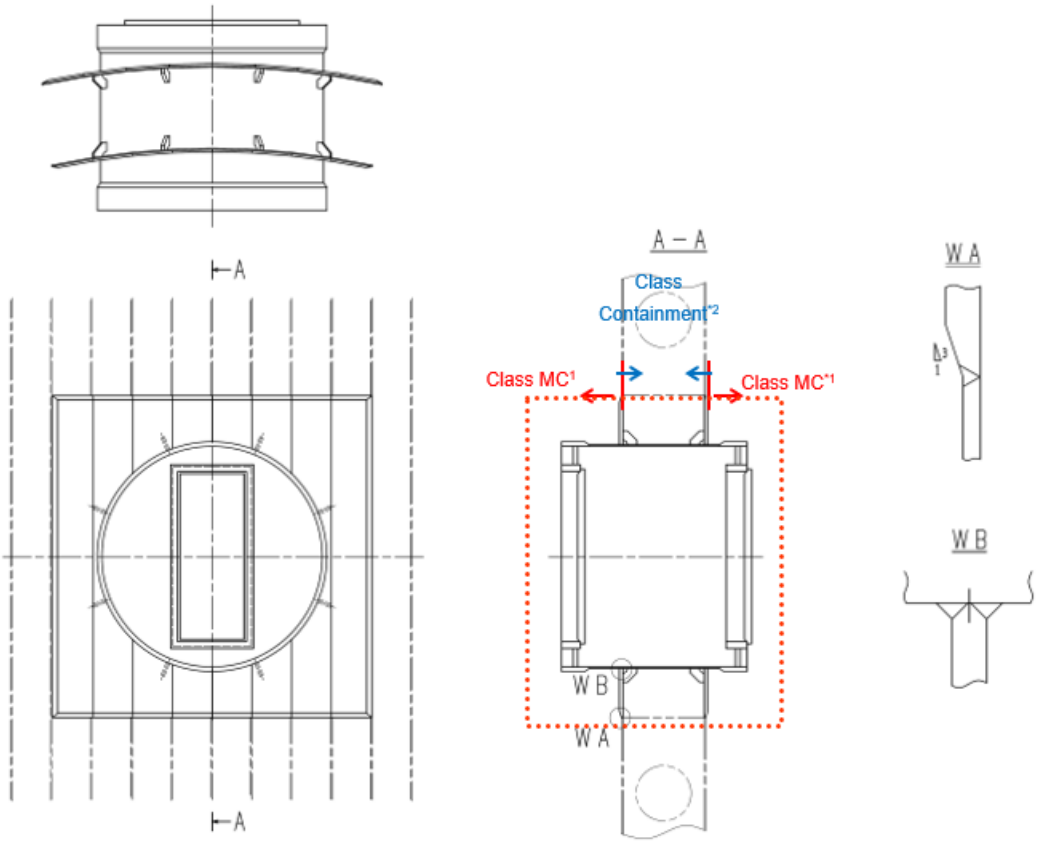


Figure 3.8-4: Containment Airlock Code Jurisdictional Boundary

*1: Designed in accordance with ASME BPVC, Section III, Subsection NE (for Class MC)

*2: Designed in accordance with NEDC-33926P-A

*3: Structural overview is provided for illustration purposes only and will be updated as the design progresses

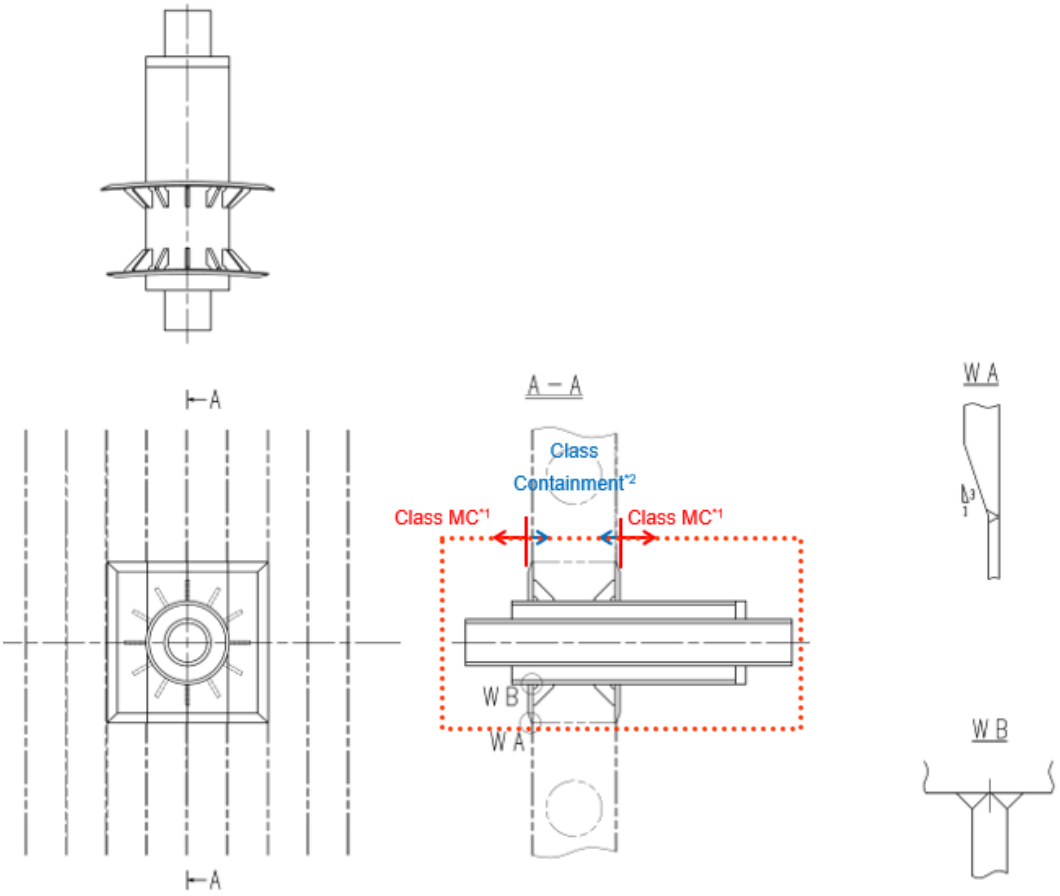


Figure 3.8-5: Penetrations Jurisdictional Boundary

*1: Designed in accordance with ASME BPVC, Section III, Subsection NE (for Class MC)

*2: Designed in accordance with NEDC-33926P-A

*3: Structural overview is provided for illustration purposes only and will be updated as the design progresses

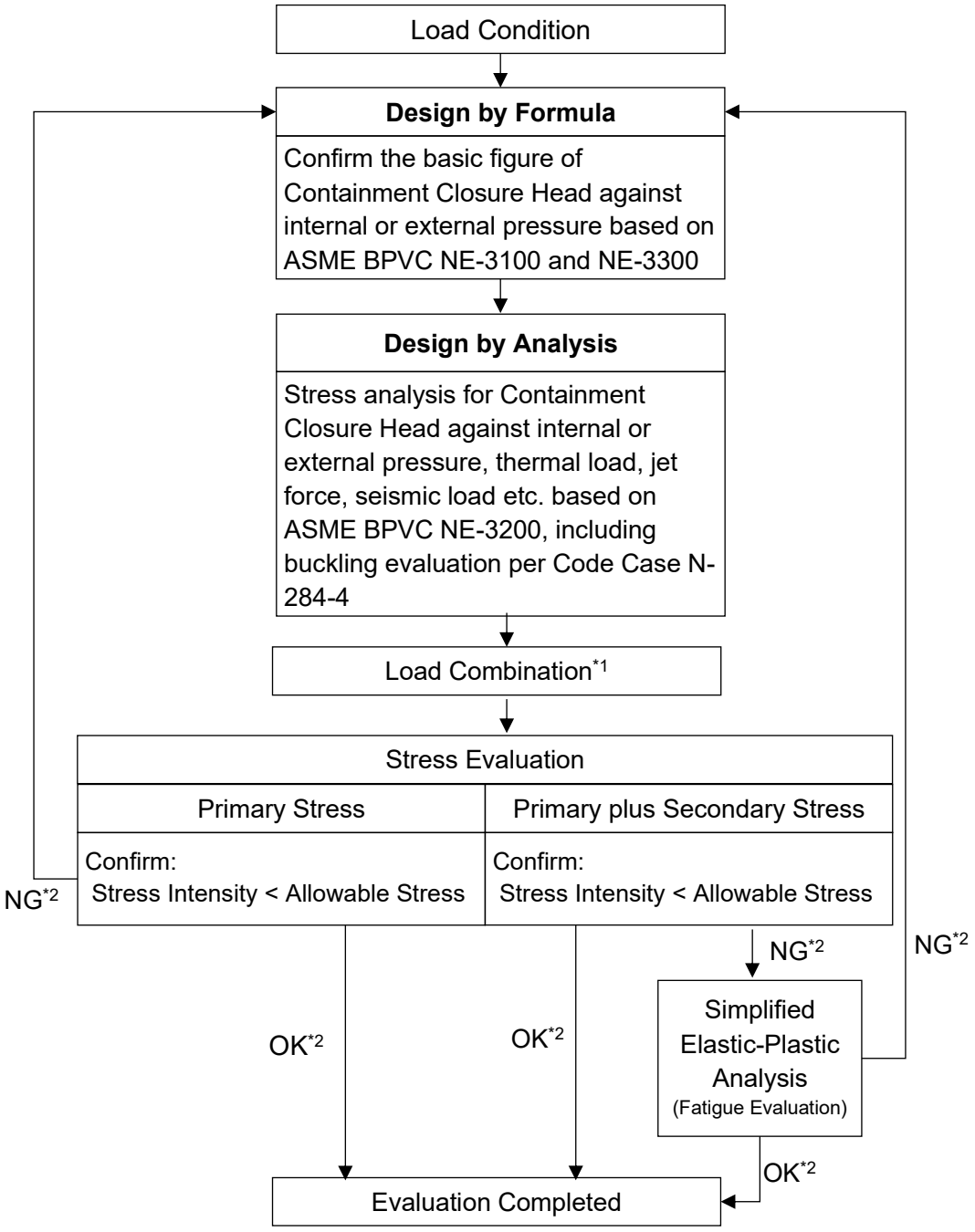


Figure 3.8-6: Design Procedures for the Containment Closure Head

*1: NUREG-0800, SRP 3.8.2 and NRC RG 1.57

*2: Ok refers to “Okay” and NG refers to “No Good” in the flowchart

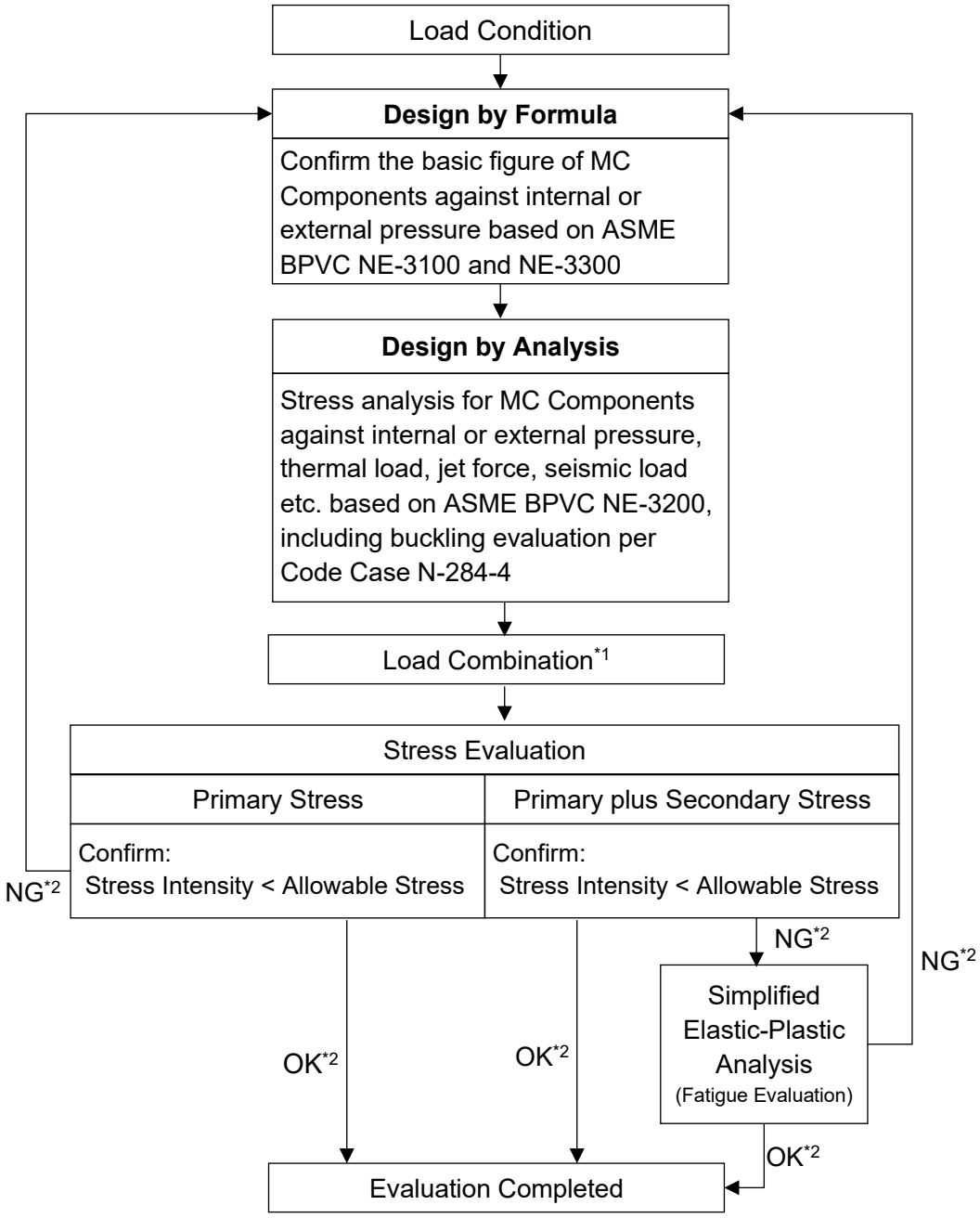


Figure 3.8-7: Design Procedures for Other Class MC Steel Components

*1: NUREG-0800, SRP 3.8.2 and NRC RG 1.57

For metal components backed by concrete, refer to NEDC-33926P-A

*2: OK refers to “Okay” and NG refers to “No Good” in the flowchart

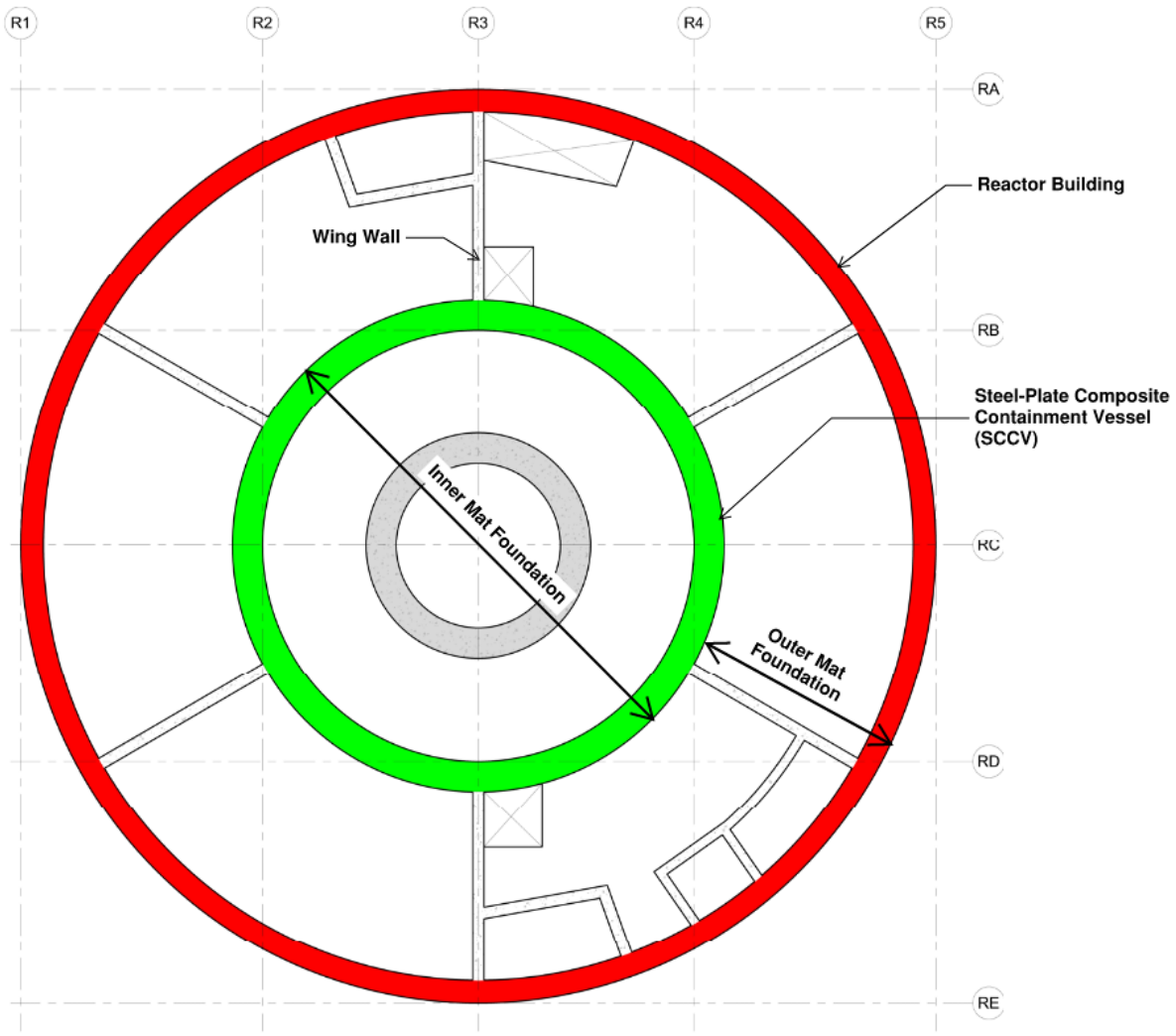


Figure 3.8-8: Plan View of BWRX-300 Mat Foundation

1. RB outer diameter is 37 meters.
2. SCCV outer diameter is 19.9 meters.
3. Foundation is 1.22 meters thick and is topped with a non-structural concrete cover to contain the sumps. Foundation embedment depth is provided in Figure 1.2-3.

*Plan view is provided for illustration purposes only and will be updated as the design progresses

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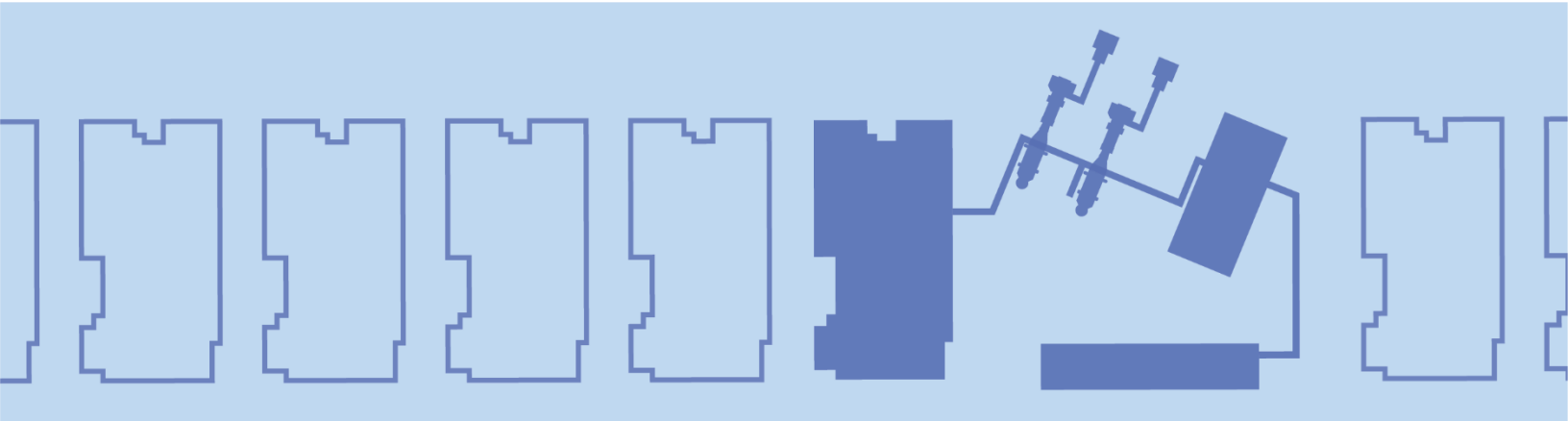
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Non-Proprietary Information

Chapter 3 Design of Structures, Components, Equipment, and Systems

Section 3.9 – Mechanical Systems and Components



ACRONYM LIST

Acronym	Explanation
ABWR	Advanced Boiling Water Reactor
AOO	Anticipated Operational Occurrences
ASME	American Society of Mechanical Engineers
BPVC	Boiler and Pressure Vessel Code
BWR	Boiling Water Reactor
CCF	Common Cause Failure
CFR	Code of Federal Regulations
CIV	Containment Isolation Valve
CRD	Control Rod Drive
CRDH	Control Rod Drive Housing
CVAP	Comprehensive Vibration Assessment Program
DBA	Design Basis Accident
DBE	Design Basis Earthquake
DEC	Design Extension Condition
ECCS	Emergency Core Cooling System
EMC	Electromagnetic Compatibility
FIV	Flow Induced Vibration
FMCRD	Fine Motor Control Rod Drive
GDC	General Design Criteria
GEH	GE Hitachi Nuclear Energy
GNF	Global Nuclear fuel
GVH	GE Vernova Hitachi Nuclear Energy
HCU	Hydraulic Control Unit
HVS	Heating, Ventilation, and Cooling System
ICS	Isolation Condenser System
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronic Engineers
ISI	Inservice Inspection
IST	Inservice Testing
ITP	Initial Test Program
LOCA	Loss-of-Coolant Accident

Acronym	Explanation
MS	Main Steam
MSCIV	Main Steam Containment Isolation Valve
MSRIV	Main Steam Reactor Isolation Valve
NPP	Nuclear Power Plant
NUREG	Nuclear Regulatory Report
PDC	Principal Design Criteria
RB	Reactor Building
RBV	Reactor Building Vibration
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RG	Regulatory Guide
RIV	Reactor Isolation Valve
RPV	Reactor Pressure Vessel
SAFDL	Specified Acceptable Fuel Design Limits
SC	Safety Class
SC1	Safety Class 1
SC2	Safety Class 2
SC3	Safety Class 3
SCN	Non-Safety Class
SDC	Shutdown Cooling System
SIR	Seismic Interface Restraint
SLCS	Standby Liquid Control System
SRP	Standard Review Plan
SSCs	Structures, Systems, and Components
SSE	Safe Shutdown Earthquake
TB	Turbine Building
TSV	Turbine Stop Valve
USNRC	United States Nuclear Regulatory Commission

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3.9 Mechanical Systems and Components

Section 3.9 provides the general design aspects used for Safety Class (SC) and Non-Safety Class (SCN) mechanical systems and components. It includes special considerations for mechanical components, dynamic testing and analysis of Structures, Systems, and Components (SSCs), required codes for American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section III, Division 1, Subsections NB and NCD for Class 1, 2, and 3 components, Subsection NF for component supports, and Subsection NG for core support structures. In addition, general design aspects for the Control Rod Drive (CRD) system and reactor vessel internals are presented. Further, this section discusses the functional design, qualification, and Inservice Testing (IST) program requirements for pumps, valves, and dynamic restraints. The design of Safety Class 2 (SC2) and Safety Class 3 (SC3) equipment follows ASME B31.1, “Power Piping,” (Reference 3.9-8) and ASME B31.3, “Process Piping,” (Reference 3.9-9).

Sections 3.1 and 3.2 are used as input to Section 3.9 and provide the general design principles, criteria, and classification used for design of mechanical systems and components. Among these principles are design for robustness, reliability, and fail-safe operation.

Section 3.7 develops the seismic input criteria and building spectra used as input to Section 3.9 and Section 3.10 for seismic qualification of Seismic Category I active mechanical components and system functionality. Additionally, Seismic Category I passive mechanical component supports, and equipment supports use the seismic spectra for qualification.

Equipment qualification requirements are provided in Section 3.10 for seismic and dynamic qualification of mechanical and electrical equipment, and Section 3.11 provides the equipment qualification requirements including environmental, functional qualification, and Electromagnetic Compatibility (EMC), which are used as input to safety classified mechanical systems and components.

Mechanical Equipment Separation for Safety Class 1 (SC1) Equipment

Simultaneous occurrence of identical failures in multiple, redundant components of the same technology are called Common Cause Failures (CCF). CCFs are an important consideration in the safety design of nuclear reactors and are considered in hazard analysis. Means of avoiding CCFs in SC1 SSCs include separation, diversity, redundancy, and independence. Separation of SSCs can be achieved using either distance, barriers, or both. Separation is also needed between redundant SSCs that perform Safety Category 1 functions to help ensure that these functions cannot be fully disabled by internal or external hazards.

Mechanical equipment separation measures for the BWRX-300 contribute to system reliability in the performance of any Safety Category 1 function including (but not necessarily limited to) interconnecting SC and SCN piping, valves, and associated mechanical controls, and instrumentation. Separation is also needed for plant features and functions where redundancy and independence are credited for ensuring acceptable safety analysis results.

Principles of physical separation include:

1. Separation by geometry (layout, distance, orientation, elevation, and, including separate structures).
2. Separation by barriers (e.g., walls, shields), both vertical and horizontal.
3. Separation by a combination of (1) and (2).

The choice of the means of separation depends on the hazards considered in the design basis, such as floods, missiles, fires.

Protection against hazards is accomplished by:

- Use of structures
- Spatial separation
- Three-hour rated fire barriers
- Isolation devices

The application of the single failure criterion is described in NEDC-33934P, “BWRX-300 Safety Strategy,” (Reference 3.9-17).

Separation of components may be by physical distance or by barriers. An example is the provision of principal fire barriers to delineate individual fire zones; such barriers may also serve as barriers to other hazards.

The following SC and SCN SSCs mechanical equipment items are considered for separation:

- Piping systems
- Valves
- Rotating equipment
- Vessels
- Ductwork systems
- Instrumentation

Piping Systems

Piping systems include piping to and from SC and SCN SSCs. These include their connected bellows, mechanical connections, support guides, and structural supports. They may include wall or floor sleeves and penetrations, pipe fittings including wells and branch connections, structural restraints (and appurtenances), and attached sampling. Piping systems also include vent/drain/test/flush/clean-out taps, including closures, instrument sensing line piping or tubing, and instrument racks. Finally, they also include pneumatic or hydraulic system tubing, manifolds, and controls appurtenances.

Valves

Valves include those that control fluid flow to and from SC and SCN SSCs. Valves include the valve body assembly, actuators, appurtenances, and all non-electrical connections.

Rotating Equipment

Rotating equipment includes SC and SCN pumps, fans, and compressors, gear sets or power coupling subsystems, and electric motors or other rotary-power driven subsystems. Their components include rotating casing, including base, frame, supports, and drive.

Vessels

Vessels include SC and SCN heat exchangers and tanks, including their supports, filter assemblies, and nozzles.

Ductwork Systems

SC and SCN ductwork systems include:

- Duct runs
- Active and pre-set dampers

- Fire dampers
- Screens
- Vents/reliefs/blow out panels
- Filters or air filtration assemblies/subsystems

Instrumentation

SC and SCN Instrumentation includes:

- Mechanically activated instruments used to monitor reactor and plant processes
- The associated non-electrical transmission
- Sensors
- Actuator systems
- In-line instruments with associated taps

Zone of Influence

The degree and type of separation required varies with the following potential hazards in a power plant zone:

1. Missiles – A missile is an unrestrained mass with sufficient kinetic energy to cause damage to SC1 components. Definition of missile and missile protection requirements are addressed in Section 3.5.
2. Pipe whip – Pipe whip is usually consequent to a pipe failure resulting in a complete segment separation break. The area in the vicinity of the postulated break of high-energy piping is defined as the pipe whip damage zone. Pipe whip protection requirements are addressed in Subsection 3.6.2.
3. Fluid jet – The fluid jet is usually consequent to a high-energy pipe break but may also be the result of intentional equipment action. Jet impingement protection requirements are addressed in Subsection 3.6.2.

Fire Area and Fire Zone

A fire area is an area sufficiently bounded to withstand the hazards associated with the fire area and to protect important equipment within the fire area from a fire outside the area. A fire zone, however, is a subdivision of fire area(s) for analysis purposes that is not necessarily bounded by fire-rated barriers.

Fire protection requirements are addressed in Subsection 9.5.1. Separation of vulnerable mechanical equipment from areas containing significant combustible materials is provided by fire barrier materials or housings, fire-rated walls, or doors (including consideration for ductwork isolations), barrier piping around processes containing flammable or combustible fluids to isolate the hazard, and in certain locations by atmospheric inerting (oxygen concentration suppression below combustible level or replacement with nitrogen, such as in containment).

Flood Zone

Internally generated flooding may occur by pipe or tank failure, fire suppression system operation, misaligned systems with openings in the affected zone, maintenance errors, or failure of a drainage system. Flood protection requirements are addressed in Section 3.4.

Separation by flood hazard containment walls, dikes, curbs, trenches or pits, watertight doors, elevated equipment mounting location (mezzanine or different floor), or pedestals, or placing vulnerable equipment in watertight housings may be used.

Design Load and Load Combination for Mechanical Systems and Components

Subsection 3.9.1.1 provides the operational transients, resulting loads, and load combinations. Design loads and load combinations (including design bases accidents) for fixed mechanical equipment are provided in Table 3.9-3. Refer to Tables 3.12-1 through 3.12-10 for additional load combination tables. These tables address faulted level transients. Fixed equipment includes the mechanical, electrical, and instrument components, and the component housings and structural supports that are anchored to civil structure(s) but are not a part of the civil structure itself, such as mechanical or electrical penetrations. Examples include the Reactor Pressure Vessel (RPV), RPV internals, RPV supports, instrumentation, piping, electrical equipment, and the component supports.

For SC1 containment structure penetrations, the anchor sleeve, where the sleeve itself acts as a containment boundary, is to meet the requirements of ASME BPVC, Section III, Subsection NE, “Class MC Components,” (Reference 3.9-5).

Design for System Duty of Mechanical Systems Based on Frequency of Occurrence and Plant State

Table 3.9-1 is used as a general event list for all hardware system duty design specifications. Service Levels are classified into service conditions as indicated below based on frequency of occurrence and plant state:

- Normal: Planned Operation
- Upset: AOO
- Emergency: Design Basis Accident (DBA) of low frequency
- Faulted: DBA of very low frequency

The BWRX-300 utilizes the four service levels used in ASME BPVC Section III, Division 1 and 2, Subsection NCA, “General Requirements for Division 1 and Division 2,” (Reference 3.9-4), Levels A, B, C and D, as well as testing conditions, in the design of fixed equipment. The design basis specifies the capabilities that are necessary for the plant in various operational states.

3.9.1 Special Topics for Mechanical Components

This subsection addresses information concerning methods of analysis for Seismic Category I components and supports, including those designated as ASME BPVC, Section III, Class 1, 2, 3, or Core Support Structures and those not covered by the ASME BPVC as discussed in SRP 3.9.1. Information is also presented concerning design transients for ASME BPVC Class 1 and Core Support components and supports.

The BWRX-300 design complies with the following General Design Criteria (GDC), which have plant-wide applicability, as discussed in Section 3.1:

GDC 1 – Quality standards and records

GDC 2 – Design bases for protection against natural phenomena

The following GDC have aspects specific to the BWRX-300 design and compliance is addressed as follows:

GDC 14 – Reactor coolant pressure boundary. The Reactor Coolant Pressure Boundary (RCPB) is designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal

leakage, of rapidly propagating failure, and of gross rupture. The design involves meeting the ASME BPVC, Section III acceptance criteria utilizing load combinations in Table 3.9-3 including transient plant duty cycles in Table 3.9-2 for the plant lifetime to assure minimal abnormal leakage.

GDC 15 – Reactor coolant system design. The mechanical components of the Reactor Coolant System (RCS) and associated auxiliary, control, and protection systems are designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOO and DBA event sequences. Subsection 3.9.1 through Subsection 3.9.6 provide design requirements for SC mechanical components that meet the margin and design condition requirements, therefore complying with GDC 15.

Section 3.1 summarizes compliance with GDC.

10 CFR 50, Appendix B, as it relates to design quality control, is discussed in Section 17.1.

10 CFR 50, Appendix S, as it relates to the suitability of the plant design bases for mechanical components established in consideration of site seismic characteristics, is discussed in Section 3.7.

Instrument sensing lines conform to the guidelines of Regulatory Guide (RG) 1.151, “Instrument Sensing Lines.”

3.9.1.1 Design Transients

The plant duty cycles represent transient events and their associated number of occurrences for the 60 year plant design life. These duty cycles are used for development of the BWRX-300 system and component design during normal operation, AOO and DBA. Requirements are evaluated for the system design and performance as they relate to complete reactor operation. The duty cycle is recorded as inputs to the system design for each specific primary and auxiliary hardware system. Duty is defined from a pressure and temperature perspective, mostly when variations in either variable are expected in important locations for the reactor including the RCPB and connecting piping and components.

The number of cycles associated with each event for the design of the RPV, RCPB, and other ASME pressure boundary components designed for fatigue are listed in Table 3.9-2. Table 3.9-1 breaks down the plant states by design condition. The plant operating conditions are identified as Normal, AOO, DBA, or testing as defined in Subsection 3.9.3.3. Appropriate service levels (A, B, C, D, or testing), as defined in the ASME BPVC, are designated for design limits. The design and analyses of ASME class piping and equipment using specific applicable thermal-hydraulic transients, which are derived from the system behavior during the events listed in Table 3.9-2 are documented in the design specifications and/or stress reports of the respective equipment.

Table 3.9-3 shows the load combinations and the standard acceptance criteria for ASME Section III components. Tables 3.12-1 through 3.12-10 provide the specific load combinations and acceptance criteria for ASME BPVC Section III Class 1, 2, and 3, B31.1, and B31.3 piping systems. Details about ASME B31.5, “Refrigeration Piping and Heat Transfer Components,” (Reference 3.9-18) piping is addressed in Section 3.12.

3.9.1.2 Computer Programs Used in Analyses

The computer programs described in Appendix 3J are used in the mechanical system and component analyses of the SC1 components, as listed in Table 3A-1.

The computer programs used in the analyses of Seismic Category I components are maintained

by GVH or by outside computer program developers.

The GVH software is controlled under NEDO-11209-A, “GE Hitachi Nuclear Energy Quality Assurance Program Description,” (Reference 3.9-11). The programs are verified for their

application by appropriate methods, such as hand calculations, or comparison with results from similar programs, experimental tests, or published literature.

3.9.1.3 Experimental Stress Analysis

Experimental stress analysis methods are used in compliance with the provisions of ASME BPVC, Section III, “Appendices,” (Reference 3.9-1).

3.9.1.4 Considerations for the Evaluation of Fault Conditions

Equipment designed to ASME BPVC, Section III, is evaluated for the faulted (Service Level D) loading conditions. In these cases, the calculated actual stresses are compared to the allowable ASME BPVC, Section III Service Level D limits. Analysis of ASME BPVC Class 1 components, component supports, core support structure components and reactor internals, and other non-code items, demonstrates that GDC 1, 14, and 15 requirements are met under faulted conditions when Service Level D limits are specified. The methods of analysis to calculate the stresses and deformations should conform to the methods outlined in the ASME BPVC, Section III, Mandatory Appendix XXVII, “Design by Analysis for Service Level D,” subject to the conditions addressed in Subsection II of SRP 3.9.1. ASME QME-1, “Qualification of Active Mechanical Equipment Used in Nuclear Facilities,” (Reference 3.9-14) is used for functional qualification of valves.

Deformations under faulted conditions are evaluated in critical areas and the necessary design deformation limits, such as clearance limits, are satisfied.

The following subsections address the evaluation methods and stress limits used for the equipment and identify the components evaluated for faulted conditions.

3.9.1.4.1 Fine Motion Control Rod Drive

The Fine Motion Control Rod Drive (FMCRD) components that are part of the RCPB are analyzed and evaluated for the ASME Service Level D faulted conditions in accordance with the ASME BPVC, Section III, Subsection NB, “Class 1 Components,” (Reference 3.9-2). Refer to Section 4.6 for FMCRD mechanism details.

3.9.1.4.2 CRD Hydraulic Control Unit

The Hydraulic Control Unit (HCU) is analyzed and tested for withstanding the faulted condition loads. Dynamic tests that are part of the seismic and dynamic qualification program establish the loads in the horizontal and vertical directions as the HCU capability for the frequency range that is likely experienced in the plant. These tests also ensure that the reactor trip function of the HCU can be performed under these loads. Dynamic analysis of the HCU with the mounting beams is performed to assure that the maximum faulted condition loads remain below the HCU capability.

Refer to Section 4.6 for HCU description details.

3.9.1.4.3 Reactor Pressure Vessel Assembly

The design of the RPV assembly as shown in Figure 4.1-1, out to and including the integral Reactor Isolation Valves (RIV), RPV top head, housings for FMCRD, and incore nuclear instrumentation complies with Subsections NB and NG of the ASME BPVC, Section III, as applicable. For faulted conditions, the reactor vessel is evaluated using elastic analysis.

Elastic analysis methods are defined in Subsection 3.7.3 and standard design rules, as defined in the ASME BPVC, are utilized in the analysis of the pressure boundary valves. The ASME BPVC, Section III allowable stress is applied to assure integrity under applicable loading conditions including faulted conditions. The functional qualification for the RIV includes analysis and/or testing for seismic and other dynamic conditions.

3.9.1.4.4 Core Support Structures and Other Safety Class Reactor Internal Components

The core support structures, the internal portion of nuclear instruments, and CRD housings and other SC1 reactor internal components are evaluated for faulted conditions. The allowable Service Level D limits for evaluation of these structures are per ASME BPVC, Section III Service Level D equations and Subsection NG, “Core Support Structures,” (Reference 3.9-7).

For the shroud support, an elastic analysis is performed, and buckling is evaluated for compressive load cases for certain locations in the assembly.

3.9.1.4.5 RPV Stabilizers, Reactor Skirt and FMCRD Housing and Nuclear Instrumentation Housing Restraints (Supports)

The calculated maximum stresses to meet the allowable stress limits are based on the ASME BPVC Section III, Subsection NF, “Supports,” (Reference 3.9-6), for the RPV Stabilizer, RPV Skirt, and supports for the FMCRD Housing, and nuclear instrumentation housing for faulted conditions. These supports restrain the components during earthquake, pipe rupture or other Reactor Building Vibration (RBV) events.

3.9.1.4.6 Reactor Isolation Valves, and Other ASME BPVC, Section III, Class 1 and 2 Valves

Elastic analysis methods and standard design rules, as defined in the ASME BPVC, are utilized in the analysis of the pressure boundary, Seismic Category I, ASME Class 1 and Class 2 valves. The ASME BPVC, Section III allowable stresses are applied to assure integrity under applicable loading conditions including faulted conditions. The functional qualification of the active valves, including RIVs; Containment Isolation Valves (CIV); Isolation Condenser System (ICS) valves (steam supply, standby gas purge, and condensate return valves) includes analysis and/or testing for seismic and/or other dynamic conditions.

3.9.1.4.7 Fuel Storage and Refueling Equipment

The Fuel Storage and Fuel Handling equipment is described in detail in Section 9.1. This includes the fuel pool structure, fuel storage racks, fuel pool cooling system, and fuel handling equipment.

A summary of the design considerations used to establish nuclear criticality safety under all operational and faulted (ASME Service Level D) conditions is described below.

All fuel storage racks are designed and qualified to operate within their performance requirements under the anticipated ranges of the normal, abnormal, or accident plant environments and are designed to withstand a Safe Shutdown Earthquake (SSE) without failure of the basic structure or damage to the active region of irradiated fuel.

3.9.1.4.8 Fuel Assembly (Including Channel)

The fuel assembly (including channel) is described in detail in Section 4.2.

The channel is subjected to mechanical tests to demonstrate the adequacy of the GNF2 channel for seismic/dynamic loads. The channel is tested to determine the allowable bending load that is sustained without buckling or collapsing the channel.

The fuel assemblies are designed for worst-case conditions that evaluate maximum stresses, fatigue, control rod insertion, fretting, corrosion/hydriding, and compatibility/dimensional changes. The results of the testing and analysis require that the SC1 components maintain the required functionality and structural capacity during ASME Level D service conditions.

3.9.1.4.9 ASME BPVC, Section III, Class 2 and 3 Vessels

The equivalent allowable stresses using elastic techniques are obtained from NCD-3300 and NCD-3200 of the ASME BPVC, Section III, Subsection NCD "Class 2 and Class 3 Components," (Reference 3.9-3).

3.9.1.4.10 ASME Class 2 and 3 Pumps

Elastic analysis methods defined in Subsection 3.7.3.1 are used for evaluating faulted loading conditions for Class 2 and Class 3 pumps. The equivalent allowable stresses for pumps using elastic techniques are obtained from NCD-3400. The preliminary SSC classification list in Table 3A-1 does not identify any pumps to be ASME Class 2 or Class 3 currently.

3.9.1.4.11 ASME Class 2 and 3 Valves

Elastic analysis methods defined in Subsection 3.7.3.1 and standard design rules are used for evaluating faulted loading conditions for Class 2 and 3 valves. The equivalent allowable stresses for valves using elastic techniques are obtained from NCD-3500.

3.9.1.4.12 ASME Class 1, 2 and 3 Piping

Elastic analysis methods defined in Subsection 3.7.3.1 are used for evaluating faulted loading conditions for Class 1, 2, and 3 piping. The equivalent allowable stresses using elastic techniques are obtained from NB-3600 (for Class 1 piping) and NCD-3600 (for Class 2 and 3 piping).

3.9.1.4.13 Inelastic Analysis Methods

The design criteria for pipe failure effects and mitigating features are provided in Section 3.6. Inelastic methods are not used in BWRX-300 piping design as described in Subsection 3.12.1.5.

3.9.2 Dynamic Testing and Analysis of Systems, Components, and Equipment

This subsection presents the criteria, testing procedures, and dynamic analyses employed to ensure the structural and functional integrity of piping systems, mechanical equipment, reactor internals, and their supports (including supports for conduit and cable trays, and ventilation ducts) under vibratory loadings, including those due to fluid flow and postulated seismic events discussed in SRP 3.9.2. Structural requirements for conduits and cable tray supports and Heating, Ventilation, and Cooling System (HVS) duct supports are specified in Subsection 3.9.2.2.

Dynamic testing and analysis of systems, components and equipment complies with the following GDC, which have plant-wide applicability, as discussed in Section 3.1:

GDC 1 – Quality standards and records

GDC 2 – Design bases for protection against natural phenomena GDC 4 – Environmental and dynamic design bases

The following GDC have aspects specific to Subsection 3.9.2 and compliance is addressed as follows:

GDC 14 – Reactor coolant pressure boundary. The RCPB is designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. The design involves meeting the ASME BPVC, Section III acceptance criteria utilizing load combinations in Table 3.9-3 including transient plant duty cycles in Table 3.9-2 for the 60 year life period to assure minimal abnormal leakage.

GDC 15 – Reactor coolant system design. The mechanical components of the RCS and associated auxiliary, control, and protection systems being designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOO and DBA event sequences. Subsection 3.9.1 through Subsection 3.9.6

provide design requirements for SC mechanical components that meet the margin and design condition requirements, therefore complying with GDC 15.

Additional Information on GDC compliance is found in Section 3.1.

10 CFR 50, Appendix B, as it relates to design quality control, is discussed in Section 17.1.

10 CFR 50, Appendix S, as it relates to the suitability of the plant design bases for mechanical components established in consideration of site seismic characteristics, is discussed in Section 3.7.

3.9.2.1 Flow Induced Vibration, Thermal Expansion, and Dynamic Effects

Piping vibration, thermal expansion, and dynamic effects testing is performed in accordance with the startup administrative manual as described in Chapter 14 and follows guidance from ASME OM Code, Division 2, Part 3, “Vibration Testing of Piping Systems” and Part 7, “Thermal Expansion of Nuclear Power Plant Piping Systems,” (Reference 3.9-10).

3.9.2.2 Seismic Qualification of SC1 Mechanical Equipment (Including Other Reactor Building Vibration Induced Loads)

This subsection discusses the testing and analytical qualification of the SC1 mechanical equipment, and other ASME BPVC, Section III, equipment including equipment supports.

3.9.2.2.1 Tests and Analysis Criteria and Methods

The testing and analytical criteria for the seismic qualification of SC1 mechanical equipment are discussed in Section 3.10.

3.9.2.3 Dynamic Response of Reactor Internals Under Operational Flow Transients and Steady-State Conditions

The reactor internal components within the vessel are subjected to extensive dynamic system analysis and testing to evaluate the dynamic response of the components during normal reactor operation (i.e., steady state conditions) and operational flow transients. The main concern for steady state dynamic response is to ensure that Flow Induced Vibration (FIV) experienced during normal operation will not cause structural failure or degradation. FIV analysis is performed as explained in the following paragraph. Component loads due to operational flow transients and non-normal operational transient conditions are performed by analysis, as described in Subsection 3.9.2.5.

As a FIV scoping activity, reactor internal components are evaluated by analytical calculations to identify any components that may require design changes. Vibration levels of each component are scaled based on legacy Boiling Water Reactor (BWR) plant measurements as detailed below. Analytical calculations are performed as stated below and follow the four main FIV mechanisms that occur in BWR’s:

Turbulence forcing function scaling was performed on existing (legacy) BWR component measured strains, flows and frequencies. The BWRX-300 component strain (and stress) is calculated from scaling the legacy plant conditions and component natural frequencies to the BWRX-300 conditions and component natural frequencies. Turbulence forcing function scaling uses the fundamental flow physics laws:

- Turbulence pressure is a function of the square of flow velocity
- Turbulence pressure decreases with increasing frequency. Turbulence loading is broadband and typically the largest response is at the component natural frequency.
- Vortex shedding (Strouhal) calculations on BWRX-300 components. This calculation also uses flow velocity past the structure compared to the structural natural frequency to ensure

no lock-in occurs. This analysis follows ASME BPVC, Nonmandatory Appendix N, "Dynamic Analysis Methods."

- There are no vane passing frequency FIV loads since there are no recirculation pumps
- There are no slip joint leakage FIV loads since there are no slip joints in the BWRX RPV

In addition, prior to initiation of the instrumented vibration measurement program, additional dynamic analyses of the reactor internals are performed. The results of these analyses are used to generate the allowable vibration levels during the vibration test.

3.9.2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

The FIV program description has been developed in accordance with RG 1.20, "Comprehensive Vibration Assessment Program (CVAP) for Reactor Internals During Preoperational and Startup Testing," to verify the adequacy of the reactor vessel internals with respect to FIV. The CVAP establishes the required testing for the reactor internals in accordance with RG 1.20 to ensure that flow-induced vibrations experienced during normal operation will not cause structural failure or degradation.

A "prototype" is a configuration of reactor internals, or a single component that, because of the arrangement, design, size, or operating conditions, represents a first-of-a-kind or unique design for which no previous valid prototype can be referenced. After the United States Nuclear Regulatory Commission (USNRC) staff accepts a prototype design and the CVAP is completed with no adverse vibration or excessive loading, it may be classified as a "valid prototype." A "non-prototype" is a configuration or a component that has substantially the same arrangement, design, size, and operating conditions as a valid prototype. A "limited prototype" is a configuration or a single component that is similar to a valid prototype but sufficiently different (in scale, nominal dimensions, boundary conditions, or operating parameters) that it cannot be considered a non-prototype.

Most of the BWRX-300 RPV internals SSCs are classified as "limited prototype" or "non-prototype" per RG 1.20 since the components are scaled versions of components in existing legacy BWR plants. However, the BWRX-300 reactor internals, as a whole, are considered a "prototype." The RG 1.20 classifications and applicable valid prototypes are listed in parentheses in the components list shown below:

- Steam Separator Assembly (limited prototype – Advanced Boiling Water Reactor (ABWR))
- Chimney (limited prototype - Dodewaard - Natural Circulation BWR)
- Shroud (limited prototype - ABWR)
- Feedwater Piping and Sparger (limited prototype - BWR/4-5)
- Core Plate (limited prototype - ABWR)
- Isolation Condenser Return Piping (non-prototype - BWR/5)
- Steam Dryer (limited prototype - BWR/4)

Although the steam dryer is a scaled version of the latest steam dryer technology, a full measurement program will be implemented because of the new piping/orifice arrangement and different steam dome acoustics.

New components included in the BWRX-300 design are the Reactor Water Cleanup piping and the Reactor Pressure Vessel Head Vent piping. These components are classified as "prototype" per RG 1.20.

The first operational BWRX-300 will implement a measurement program during initial startup for all prototype and limited prototype elements in accordance with RG 1.20. The primary purpose of this test series is to verify the anticipated effect of single- and two-phase flow on the vibration response of internals and to ensure these effects are within acceptable FIV criteria developed through the supporting dynamic analysis. Startup testing of the reactor internals is performed per the Initial Test Program (ITP) (Chapter 14).

For the initial prototype plant:

- Vibration measurements are recorded at incremental hold points up to and including up to 100% rated power during startup testing. The startup data will be recorded at plant steady state conditions at each hold point since FIV is a high cycle fatigue load. Measurements will occur during power ascension and will encompass the plant operational core-power map. It is anticipated that data will be recorded at 5% power increments, i.e., steady state hold points, during power ascension testing. The data recorded at each hold point is of sufficient duration to identify FIV mechanisms and to validate the analytical results:
  - There are no expected normal operation FIV transient loads since the BWRX-300 is a natural recirculation BWR, but if it is determined there is a need to measure transients between the hold points, then transient data will be recorded.
- Following completion of vibration measurement program, a final vibration testing report will be issued in accordance with RG 1.20
- Following the startup test program, reactor internal components will be inspected at the end of the first cycle to identify any indications of FIV per RG 1.20. Anomalies, such as excessive wear, will be noted and a plan for mitigation will be developed. A final inspection report will be issued in accordance with RG 1.20.

Following completion of the measurement program, the CVAP will be presented to the USNRC for review and acceptance as “valid” prototypes.

3.9.2.5 Dynamic System Analysis of Reactor Internals Under Faulted Conditions

Faulted Conditions are generally the bounding transients in BWR’s. Because these transients cannot be simulated in a working BWR, these events are analyzed and not measured. It should be noted that all transients, including anticipated operational occurrences, emergency and faulted conditions are analyzed to ASME BPVC, Section III and ASME Service Level definitions. Refer to Subsection 3.9.3.1 on Loads and Load Combinations. The dynamic analysis for faulted conditions events considers the following:

- Reactor Internal Pressures
- External Pressure and Forces on the Reactor Vessel
- Loss-of-Coolant Accident (LOCA) Loads
- Seismic Loads

These resulting loads to the reactor internals are included in the deterministic analysis performed for faulted conditions and described further in Subsection 3.9.3.

3.9.2.6 Correlations of Reactor Internals Vibration Tests with the Analytical Results

Prior to initiation of the instrumented vibration measurement program, extensive dynamic analyses of the reactor and internals are performed as described in Subsections 3.9.2.3 and 3.9.3.5. The results of these analyses are used to generate the allowable vibration levels during the vibration test. The vibration data obtained during the test are analyzed in detail.

The results of the data analyses, vibration amplitudes, natural frequencies, and mode shapes are then compared to those obtained from the theoretical analysis.

Such comparisons provide the analysts with added insight into the dynamic behavior of the reactor internals. In addition, knowledge gained from previous vibration tests is used in the generation of updated dynamic models for seismic and LOCA analyses of the plant. The models used for the BWRX-300 are similar to those used for the vibration analysis of earlier prototype BWR plants.

3.9.3 ASME BPVC Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

This subsection discusses the structural integrity and/or functional integrity requirements of pressure-retaining components, their supports, and core support structures that are designed in accordance with the requirements of ASME BPVC, Section III, Subsections NB, NCD, NE, NF, and NG and GDC 1, 2, 4, 14, and 15 and 10 CFR 50 Appendix S.

ASME Class 1, 2, and 3 Components, Component Supports, and Core Support Structures complies with the following GDC, which have plant-wide applicability, as discussed in Section 3.1:

GDC 1 – Quality standards and records

GDC 2 – Design bases for protection against natural phenomena

GDC 4 – Environmental and dynamic design bases

The following GDC have aspects specific to Subsection 3.9.3 and compliance is addressed as follows:

GDC 14 – Reactor coolant pressure boundary. The RCPB is designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. The design involves meeting the ASME BPVC, Section III acceptance criteria utilizing load combinations in Table 3.9-3 including transient plant duty cycles in Table 3.9-2 for the plant lifetime to have an extremely low probability of abnormal leakage.

GDC 15 – Reactor coolant system design. The mechanical components of the RCS and associated auxiliary, control, and protection systems being designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOO and DBA event sequences. Subsection 3.9.1 through Subsection 3.9.6 provide design requirements for SC mechanical components that meet the margin and design condition requirements, therefore complying with GDC 15.

Additional Information on GDC compliance is found in Section 3.1.

10 CFR 50, Appendix S, as it relates to the suitability of the plant design bases for mechanical components established in consideration of site seismic characteristics, is discussed in Section 3.7.

3.9.3.1 Loading Combinations, Design Transients, and Stress Limits

Subsection 3.9.3.3 delineates the criteria for selection and definition of design limits and loading combinations associated with Normal Operation, AOO, DBAs, and specified seismic and other RBV events for the design of ASME BPVC Section III components (except containment components which are discussed in Section 3.8).

This section discusses the ASME BPVC, Section III, Class 1, 2, and 3 equipment and associated pressure-retaining parts, and identifies the applicable loadings, calculation methods, calculated stresses, and allowable stresses. A discussion of equipment is included on a component-by-component basis to provide examples. Design transients and dynamic loading for

ASME BPVC, Section III, Class 1, 2 and 3 equipment are covered in Subsection 3.9.1.1, Subsection 3.9.3.6, and Subsection 3.9.3.7. Seismic-related loads and methodology for seismic qualification are discussed in Subsection 3.7.3. Table 3.9-3 presents the plant events to be considered for the design and analysis of all BWRX-300 ASME BPVC, Section III, Class 1, 2, and 3 components, component supports, and core support structures. Specific loading combinations considered for evaluation of specific equipment are derived from Table 3.9-3 and are contained in the design specifications and design reports for the respective equipment. For ASME Class 1 components where analysis for cyclic operation is evaluated in accordance with NB-3222.4, the fatigue usage evaluation includes the use of environmental fatigue curves in accordance with RG 1.207 and Nuclear Regulatory Report (NUREG)/CR-6909, “Effect of LWR Environments on the Fatigue Life of Reactor Materials.”

Specific load combinations and acceptance criteria for ASME Class 1, 2, 3, B31.1, and B31.3 piping is shown in Tables 3.12-1, 3.12-2, 3.12-3, and 3.12-4. Also, for ASME Class 1 piping, the operating temperatures above ambient or below ambient are included in the fatigue analysis.

The installation temperature state for the piping system is defined as a temperature of 21°C (70°F) for Class 1, 2, 3 or ASME B31.1 piping.

The design life for the BWRX-300 Standard Plant is 60 years. A 60 year design life is a requirement for all plant components. Additional life is added for components required during decommissioning. The design life requirement allows for replacement, refurbishment, and repair, as appropriate, to assure that the design life of the overall plant is achieved.

3.9.3.2 Design and Installation of Pressure Relief Devices

NEDC-33910P-A, “BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection,” (Reference 3.9-15) describes the methodology that is stated within ASME BPVC, Section III requirements for pressure relief. The BWRX-300 utilizes the ICS System for Pressure Relief as provided in Section 6.3.

3.9.3.3 Events Considered in Evaluating Effect of Loads on Fixed Equipment

Events that the BWRX-300 might credibly experience during a reactor-year are evaluated in Chapter 15, to establish the plant design basis, including plant fixed equipment. The associated loads and duty cycles associated with each event are considered in combination with additional events in load combinations as applicable. These event combinations are divided into the four plant conditions with associated frequency of occurrence and ASME BPVC, Section III service levels as shown in Table 3.9-1.

The following are the plant condition events and transients associated with the BWRX-300 design:

3.9.3.3.1 Normal Operation

See Table 3.9-3 for the Load Combinations and acceptance criteria which includes normal operation conditions.

3.9.3.3.2 Anticipated Operational Occurrences

See Table 3.9-3 for the Load Combinations and acceptance criteria which includes AOO conditions.

3.9.3.3.3 Design Basis Accident Events

See Table 3.9-3 for the Load Combinations and acceptance criteria which includes DBA events.

3.9.3.3.4 Seismic Events

Seismic design parameters and associated seismic events defined in Subsection 3.7.1 are used in qualification of mechanical system components which considers non-LOCA faulted and plant

testing events. Section 3.10 provides seismic qualification methodology to assure both component structural and/or functional capacity under seismic operational conditions are met.

All SSCs of the BWRX-300 design are designated by SC, Quality Group, and Seismic Category according to guidance in Section 3.2. Table 3A-1 provides the classification table for plant SSCs.

3.9.3.4 Establishment of Design, Service, and Test Loadings and Limits

Design, Service, and Test Loadings and Limits for fixed equipment components and supports are in accordance with ASME BPVC, Section III, Subsection NCA.

SC1 electrical equipment is evaluated with respect to the load combinations in Table 3.9-3 using guidance in International Electrotechnical Commission (IEC)/Institute of Electrical and Electronics Engineers (IEEE) 60780-323, “Nuclear Facilities – Electrical Equipment Important to Safety – Qualification,” (Reference 3.9-12) and IEC/IEEE 60980-344, “Nuclear Facilities – Equipment Important to Safety – Seismic Qualification,” (Reference 3.9-13).

For SC1, actuators and power operated valve assemblies are evaluated with respect to the load combinations in accordance with the provisions of ASME QME-1. In addition to meeting ASME QME-1, SC1 actuators should consider the qualification guidance provided in IEEE 382 as endorsed by RG 1.73, “Qualification of Safety-Related Actuators in Production and Utilization Facilities” for their qualification program.

3.9.3.5 Acceptance Criteria

Components and supports comply with the design rules established for design, service, and test loadings in the appropriate subsection of ASME BPVC, Section III.

Design documentation is completed in accordance with the requirements of the subsection of ASME BPVC applicable to the component or support.

3.9.3.6 Loading Criteria

3.9.3.6.1 Loading Conditions

As appropriate, ASME BPVC, Section III, NB-3111, NCD-3111, NE-3111, NF-3111, or NG-3111, is applied for a complete list of required load conditions to consider.

Consistent with the ASME BPVC, Section III, the stresses resulting from differential anchor movements during dynamic events are considered secondary stresses. Refer to Subsections 3.7.3.1.4 and 3.7.3.9 for guidance.

The combinations of events and acceptance criteria are shown in Table 3.9-3. The method of combination of the loads is in accordance with NUREG 0484, “Methodology for Combining Dynamic Responses.”

3.9.3.6.2 Design Loadings

The Design Loadings are established in accordance with NB-3112, NCD-3112, NE-3112, NF-3112, or NG-3112, as applicable.

Design Pressure – The specified internal and external Design Pressure is not to be less than the maximum difference in pressure between the inside and outside of the item, or between any two chambers of a combination unit, which exists under the most severe loadings for which the Level A Service Limits are applicable.

The Design Pressure includes allowances for pressure surges. Note this is referred to as Design Pressure Difference in NG-3112 for core support structures.

Design Temperature – Except as otherwise defined in NB-3112 for Class 1 components, the specified Design Temperature is not less than the expected maximum mean metal temperature through the thickness of the part considered for which Level A Limits are specified.

Design Mechanical Loads – The specified Design Mechanical Loads are in accordance with NCA-2142.4. The specified Design Mechanical Loads are selected so that when combined with the effects of Design Pressure, they represent the most severe coincident loadings for which the Level A Service Limits on primary stress are applicable.

3.9.3.6.3 Service Conditions

The Service Conditions are established in accordance with NB-3113, NCD-3113, NE-3113, NF-3113, or NG-3113, as applicable.

Each service condition to which the components may be subjected is classified in accordance with Service Limits designated in the component design specifications in such detail as to provide a complete basis for design, construction, and inspection. The loadings are taken into account in accordance with NB-3111, NCD-3111, NF-3111 or NG-3111, as applicable.

For ASME Class 1 components, the requirements of (1) and (2) below apply:

1. Level B Conditions – The estimated duration of service conditions for which Level B Limits are specified and included in the design specifications.
2. Level C Conditions – The total number of postulated occurrences for all specified service conditions for which Level C Limits are specified are limited to no more than twenty-five stress cycles having an Sa value greater than that for 10^6 cycles from the applicable fatigue design curves of ASME BPVC Section III, Mandatory Appendix I.

When the Component Design Specification requires computations to demonstrate compliance with specified Service Limits, the Component Design Specification provides information from which Service Loadings can be identified (pressure, temperature, mechanical loads, cycles, or transients).

3.9.3.6.4 Test Loadings

Test Pressure – The specified internal and external test pressures are as required by ASME BPVC, Section III.

Test Loads – Loads due to other types of required tests are included as required by ASME BPVC, Section III.

Test Temperature – Test temperature is defined to ensure that thermal effects are considered in test loads.

3.9.3.7 Loading Phenomena

Subsection 3.9.3.7 describes the types of load phenomena, which are considered for components, as applicable. Consists of pressure, dead weight, fluid reaction, prestress, fluid flow (including FIV when applicable) and other loads due to moving parts within a component or system.

“N” loads in Table 3.9-3 include Thermal Effects. Individual component specifications may elect to break out the thermal effects loads separately for the sake of clarity, if desired.

Thermal loads are included in this category for each service level, which are thermally induced loads caused by both steady state and transient operating conditions. The thermal anchor displacement loads are due to the anchor displacements (end motion) caused by thermal expansion, or pressure dilation, or other dynamic loads at anchor boundary locations.

3.9.3.7.1 Flow Induced Vibration

Flow of fluids past objects creates local pressure disturbances, which exert forces on the object. These forces can cause dynamic responses depending on the forcing function and dynamic characteristics of the object. Flow induced vibrations have been noted in nuclear power plant systems, which produce vortex shedding (e.g., heat exchangers), pump (reciprocating or centrifugal), and thermodynamic instability conditions. Mechanical systems and equipment are reviewed for potential flow induced vibration mechanisms, evaluating all modes of system operation including both normal and abnormal conditions.

Flow induced vibration loads may be associated with Service Level A for those structures (e.g., reactor internals) where the loads exist during normal operation. For flow induced vibration loads associated with transients that are not considered part of normal operation, the flow induced vibration loads are evaluated as part of the alternative service level.

Consistent with SIL No. 682, "Flow Induced Vibration Load," (Reference 3.9-19), the RPV component design considers the loads due to normal operation vibration (flow induced vibration) without combination of other primary loads.

Vortex Shedding

Vortex shedding occurs at certain fluid velocities when a fluid flows past objects. The vibration cannot be eliminated, but it can usually be controlled. It is important that these cases consider all potential modes of component operation. Vortex shedding hydrodynamic mass effects are considered. Other components susceptible to flow induced vibration are pressure, flow, and temperature sensors, which encroach upon the flow stream.

Pressure Fluctuations

Pressure fluctuations in a vapor or gas-state fluid (e.g., steam) occur due to flow past branch piping connections and branch connected components (e.g., safety valve "bell chamber" resonance), flow through short radius elbow fittings that induce flow separation effects, flow passing through valve chambers, flow past sharp-edged in-line pipe components (e.g., orifices, weld joint backing rings, valve seat rings), or two or more individual flows entering a common header or drum that generates an acoustic response. These various flow disturbances generate acoustic waves that can travel forward and backward in a piping system. If of sufficient strength and at a component's susceptible frequency, these acoustic resonances can cause cyclic fatigue and result in component failure.

Pumps create pressure fluctuations in a fluid system. In most system designs, these fluctuations are insignificant. However, the possibility exists that these fluctuations, coupled with unintentional but improper system or component structural characteristics, can cause resonant vibrational response in the system or component. Component structural characteristics are designed to assure a resonance value sufficiently high to avoid excitation by evaluated system fluid fluctuations. Pressure attenuation devices are used as applicable to significantly reduce the effects of this phenomenon.

Thermodynamic Instability

Under certain system design features and operating modes, fluid dynamic forces can be generated, which create large pressure variations. These have been noted in certain feedwater systems where a relatively cold fluid layer is in contact with a relatively hot steam region; under certain operating modes significant water-hammer-type phenomena have occurred causing a breach of the pressure-retaining boundary.

3.9.3.7.2 Rapid Valve Closure or Opening

Extremely rapid valve closure or opening in a fluid system can create large pressure waves which can propagate through a piping system and into connected components. This rapid motion could be caused by operating characteristics of the valve (e.g., stiffness of diaphragm in pneumatic operators), or the fluid flow forces acting on the valve parts during all modes of operations. Rapid valve closure or opening is defined as an AOO.

For example, Turbine Stop Valve (TSV) closure may be identified as being capable of generating large pressure waves which could cause significant dynamic response. Prior to TSV closure, saturated steam flows through Main System (MS) piping at nuclear boiler rated pressure and mass rate. Steam flow to the turbine comes to a stop at the instant the turbine-stop valve closes. The flow of steam travels in the MS line through the vessel nozzle and into the vessel. This results in a compressive acoustic load and a steam impingement load on the steam dryer outer hood. Additionally, repeated reflections of the compression wave in the MS line generate time-varying forces in the MS piping. SSCs in the Reactor Building (RB), Steam Tunnel and Turbine Building (TB) may be affected. TSV closure is conservatively evaluated as an AOO.

3.9.3.7.3 Isolation Condenser Operation

The thermal effects associated with operation of ICS and the loads such as pressure resulting from operation of ICS are considered. Loads associated with the breaks of ICS high pressure lines in the pool are considered. The major loads imposed on ICS result from:

- Sudden reactor isolation at power operating conditions
- During station blackout (i.e., unavailability of all alternate current power)
- Failure to SCRAM
- LOCA

3.9.3.7.4 Failures of High-Energy Fluid System Piping

The effects of postulated pipe breaks in high-energy fluid systems as well as measures used to protect SSCs are defined in Section 3.6.

3.9.3.7.5 Failures of Moderate-Energy Fluid System Piping

The effects of postulated pipe cracks in moderate-energy fluid systems as well as measures used to protect SSCs are defined in Section 3.6.

3.9.3.7.6 Fuel Lift Loads

Fuel lift is the postulated process under which a combination of vertical motion of the RPV support, scram uplift forces on the fuel assemblies and vertical hydraulic forces result in fuel assemblies lifting off from their seating surfaces on the fuel support. The reaction load of the fuel on the core support structures is considered. Fuel lift analysis is discussed in Subsection 4.2.3.4.

3.9.3.8 Safety Class Equipment and Safety Category Functional Criteria

For any normal or upset service level, the component or support must withstand these loadings without damage requiring repair.

For any emergency or faulted service level, these sets of limits promote large (Service Level C) or gross (Service Level D) general deformations with some consequent loss of dimensional stability and damage requiring repair that may require removal of the component or support from service.

3.9.3.9 Reactor Vessel Assembly

The reactor vessel assembly includes: The RPV pressure boundary out to and including the nozzles, the RIVs, and the housings for FMCRD and nuclear instrumentations. The RPV assembly is ASME Class 1.

The stress analysis is performed on the RPV for various plant operating conditions (including faulted conditions) by using elastic methods. Load combinations and acceptance criteria for reactor internals are provided in Table 3.9-3.

The RPV internals are listed in Table 3A-1.

3.9.3.10 Main Steam Piping

The MS piping trains extending from the outboard Main Steam Reactor Isolation Valve (MSRIV) up to the Seismic Interface Restraints (SIRs) that are outboard of the Main Steam Containment Isolation Valves (MSCIVs) are designed and constructed in accordance with the ASME BPVC, Section III, Subsection NCD requirements for Class 2 components. Stresses are calculated on an elastic basis for each service level and evaluated in accordance with NCD-3600. Table 3.12-2 shows the specific load combinations and acceptance criteria for Class 2 piping that apply to this piping.

The SIRs are designed to ASME BPVC, Section III, Subsection NF. The MS system piping extending from the outboard SIR to the TSV is constructed in accordance with the ASME B31.1 criteria.

3.9.3.11 Other Components

3.9.3.11.1 Isolation Condenser System

The ICS piping inside the containment is designed and constructed in accordance with the ASME BPVC, Section III, Class 1 requirements. The Isolation Condenser and piping outside containment are designed and constructed in accordance with ASME BPVC, Section III Class 1 requirements.

3.9.3.11.2 Reactor Water Cleanup System Heat Exchangers

The Reactor Water Cleanup System (CUW) heat exchanger does not perform any Safety Category 1 functions. The heat exchanger is Non-Seismic equipment. ASME BPVC, Section VIII, Division 1, “Rules for Construction of Pressure Vessels,” (Reference 3.9-16) requirements are used in the design and construction of the CUW heat exchanger.

3.9.3.11.3 Shutdown Cooling System Pump and Heat Exchangers

The Shutdown Cooling System (SDC) heat exchangers do not perform any Safety Category 1 functions. The pumps and heat exchangers are Non-Seismic equipment. ASME BPVC, Section VIII, Division 1 requirements are used in the design and construction of the SDC pump and heat exchanger.

3.9.3.11.4 ASME Class 2 and 3 Vessels

ASME Class 2 and 3 vessels are constructed in accordance with ASME BPVC, Section III.

3.9.3.11.5 ASME Class 1, 2, and 3 Valves

ASME Class 1, 2, and 3 valves are constructed in accordance with ASME BPVC, Section III. All valves and their extended structures are designed to withstand the accelerations due to seismic and other RBV loads. Refer to Subsection 3.9.3.12 for additional information on valve operational readiness.

3.9.3.11.6 ASME Class 1, 2, and 3 Piping

ASME Class 1, 2 and 3 piping is constructed in accordance with the ASME BPVC, Section III. For ASME Class 1 piping, stresses are calculated on an elastic basis and evaluated in accordance with NB-3600 and fatigue usage is determined. For ASME Class 2 and 3 piping, stresses are calculated on an elastic basis and evaluated in accordance with NCD-3600.

Tables 3.12-1 and 3.12-2 provide the specific load combinations and acceptance criteria for ASME Class 1, 2, and 3 piping systems.

3.9.3.12 Valve Operational Readiness

SC1 valves considers the qualification guidance provided in ASME QME-1 for their qualification program. Sections 3.10 and 3.11 provide details on the seismic and environmental qualification of valves.

The requirements in this section apply to SC1 valves and their associated Safety Category 1 functions. Functional qualification does not include independent verification and validation testing of software. Although functional qualification by testing or analysis is not required for SC2 and SC3 valves, SC2 and SC3 valves are designed to be capable of performing their required function(s) under the expected operating and environmental conditions for the required mission time.

3.9.3.13 Design of Pressure Relief Devices

The RCS does not utilize safety or relief valves for overpressure relief. During normal operation, the MS flow to the turbine is throttled to control system pressure. Section 6.3 describes the method of RPV overpressure relief that is based on NEDC-33910P-A (Reference 3.9-15).

3.9.3.14 Component Supports

The establishment of the design/service loadings and limits is in accordance with ASME BPVC, Section III, Subsection NF. These loadings and stress limits apply to the structural integrity of components and supports when subjected to combinations of loadings derived from plant and system operating conditions and postulated plant events.

ASME BPVC, Section III component supports are designed, manufactured, installed, and tested in accordance with all applicable codes and standards. The building structure component supports (connecting the NF support boundary component to the existing building structure) are designed as specified in Section 3.8.

The design of bolts for component supports is specified in ASME BPVC, Section III, Subsection NF. Stress limits for bolts are given in NF-3225. The rules and stress limits which must be satisfied are those given in NF-3324.6 multiplied by the appropriate stress limit factor for the particular service loading level and stress category specified in Table NF-3225.2-1.

3.9.3.15 Main Steam Piping

Supports and their attachments for ASME BPVC, Section III Class 1, 2, and 3 piping are designed in accordance with Subsection NF up to the interface of the building structure, with jurisdictional boundaries as defined by Subsection NF. Pipe supports include hangers, snubbers, struts, spring hangers, frames, energy absorbers and limit stops.

The design of supports for non-BPVC piping satisfies the requirements of ASME B31.1 and ASME B31.3.

3.9.3.16 Reactor Pressure Vessel Stabilizer

The RPV stabilizer is designed as an SC1 linear type component support in accordance with the requirements of ASME BPVC, Section III, Subsection NF. The main function of the stabilizers is

to provide horizontal support to limit vessel motion by transferring load from the vessel to the adjacent civil structure. Upper and lower vessel stabilizers function independently, depending on the direction of vessel motion due to an applied dynamic force, but horizontally and vertically parallel stabilizers act with combined resistance against vessel motion. The design loading conditions and stress criteria meet the ASME BPVC, Section III stresses in the critical support areas for various plant operating conditions.

3.9.4 Control Rod Drive System

The CRD system is described in Section 4.6. The CRD system is an SC1 system and portions of the CRD system are a part of the RCPB, the system is designed, fabricated, and tested to quality standards commensurate with the safety category functions that are performed.

The CRD system includes the FMCRD mechanisms, the HCU assemblies, and the CRD hydraulic subsystem. The system extends inside the RPV to the coupling interface with the control rod blades.

The BWRX-300 design complies with the relevant requirements of the following 10 CFR 50 Appendix A and 10 CFR 50.55a regulations.

The CRD system complies with the following GDC, which have plant-wide applicability, as discussed in Section 3.1:

GDC 1 – Quality standards and records

GDC 2 – Design bases for protection against natural phenomena

The following GDC and Principal Design Criteria (PDC) have aspects specific to the CRD system design and compliance is addressed as follows:

GDC 14 – Reactor coolant pressure boundary, as it relates to the CRD system, requires that the RCPB portion of the CRD system be designed, constructed, and tested for the extremely low probability of leakage or gross rupture. The design involves meeting the ASME BPVC, Section III, Subsection NB acceptance criteria utilizing load combinations in Table 3.9-3, including transient plant duty cycles in Table 3.9-2 for the 60 year life period to assure minimal abnormal leakage.

PDC 26 – Reactivity control system redundancy and capability, as it relates to the CRD system, requires that the control rods are used in the reactivity control system, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including AOOs, and with appropriate margin for malfunctions such as stuck rods, Specified Acceptable Fuel Design Limits (SAFDL) are not exceeded.

GDC 27 – Combined reactivity control systems capability, as it relates to the CRD system, requires that the reactivity control systems be designed to reliably control reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

PDC 29 – Protection against anticipated operational occurrences, as it relates to the CRD system requires that the CRD system, in conjunction with reactor protection systems, be designed to assure an extremely high probability of accomplishing its safety functions in the event of AOOs. The design of the CRD system to ASME BPVC, Section III Service Level A through D and test conditions as outlined in Table 3.9-3 assures accomplishment of safety functions in the event of an AOO, thus complying with PDC 29.

Additional Information on GDC and PDC compliance is found in Section 3.1.

3.9.4.1 Descriptive Information on Control Rod Drive System

Descriptive information on the FMCRDs as well as the entire CRD System is contained in Section 4.6.

3.9.4.2 Applicable Control Rod Drive System Design Specification

The CRD System, which is designed to meet the functional design criteria outlined in Section 4.6 consists of the following:

- Electro-hydraulic FMCRD mechanisms
- Hydraulic Control Unit
- CRD Hydraulic Subsystem

The subcomponents of the FMCRD forming part of the RCPB are designed according to ASME BPVC, Section III, Subsection NB requirements.

The quality group classification of the components of the CRD system is listed in Table 3A-1 and are designed to the codes and standards in accordance with their individual quality groups.

Pertinent aspects of the design and qualification of the CRD system components are discussed in the following locations: transients in Subsection 3.9.1.1, Subsection 3.9.3.6, and Subsection 3.9.3.7, faulted conditions in Subsection 3.9.1.4.1 and Subsection 3.9.1.4.2.

3.9.4.3 Design Loads, Stress Limits, and Allowable Deformations

The ASME BPVC, Section III, Subsection NB components of the CRD system are evaluated analytically, and the design loading conditions, and stress criteria are given in Table 3.9-3.

3.9.4.4 Control Rod Drive Performance Assurance Program

The assurance of the CRDM RCPB components to perform throughout the 60 year design life of the system is confirmed by the ASME Class 1 design report required by ASME BPVC, Section III, Subsection NB.

Refer to Section 4.6 for required tests to assure CRD operability.

3.9.5 Reactor Pressure Vessel Internals

Refer to Subsection 4.1.2 and Figure 4.1-1 for a detailed description, design, and function. In accordance with ASME BPVC, Section III, NG-1122 and NG-4110(b), the manufacturer (Certificate Holder) of core supports certifies that the construction of all internal structures is such as not to adversely affect the integrity of the core support structure.

The Reactor Pressure Vessel Internals comply with the following GDC, which have plant-wide applicability, as discussed in Section 3.1:

GDC 1 – Quality standards and records (SC1, SC2, and SC3)

GDC 2 – Design bases for protection against natural phenomena (SC1) GDC 4 – Environmental and dynamic design bases

The following GDC have aspects specific to the BWRX-300 design and compliance is addressed as follows:

GDC 10 – Reactor Design. This GDC requires that reactor internals be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs. The reactor internals consist of pressure boundary, non-pressure boundary components, and support structures that are

designed to maintain structural integrity per Subsection 3.9.1, Subsection 3.9.2, and Subsection 3.9.3, and maintain SAFDLs as discussed in Section 4.3.

Additional information on GDC compliance is found in Section 3.1.

3.9.6 Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints

Sections 3.10 and 3.11 provide the method for qualification of mechanical equipment. The qualification involves both determining component functionality while maintaining structural integrity under seismic, dynamic, and environmental conditions. Seismic testing of components is performed as well as use of analytical methods.

The Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints complies with the following GDC, which have plant-wide applicability, as discussed in Section 3.1:

GDC 1 – Quality standards and records

GDC 2 – Design bases for protection against natural phenomena

GDC 4 – Environmental and dynamic design bases

The BWRX-300 design complies with the relevant requirements of the following 10 CFR 50 Appendices A and 10CFR 50.55a regulations as stated below:

GDC 14 – Reactor coolant pressure boundary, as it relates to designing, fabricating, erecting, and testing pumps, valves, and dynamic restraints that form the RCPB so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture designed in accordance with Subsection 3.9.1, Subsection 3.9.2, Subsection 3.9.3, and Section 3.10, therefore complying with the requirements of GDC 2 for the 60 year life period to assure minimal abnormal leakage.

GDC 15 – Reactor coolant system design, as it relates to pumps, valves, and dynamic restraints that form the RCS and associated auxiliary, control, and protection systems being designed with sufficient margin to ensure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOO and DBA event sequences designed in accordance with Subsection 3.9.1, Subsection 3.9.2, Subsection 3.9.3, and Section 3.10, therefore complying with the requirements of GDC 15.

GDC 37 – Testing of emergency core cooling system, as it relates to designing the ECCS to permit periodic pressure and functional testing to ensure the leak tight integrity and operability and performance of its active components. The ECCS design incorporates adequate space and connections to allow for periodic pressure and functional testing. Design completion will occur during FSAR phase.

GDC 40 – Testing of containment heat removal system, as it relates to designing periodic pressure and functional testing of the passive containment cooling system to ensure the leak tight integrity and operability and performance of its active components. The passive containment cooling system design incorporates adequate space and connections to allow for periodic pressure and functional testing. Design completion will occur during FSAR phase.

GDC 43 – Testing of containment atmosphere cleanup systems, as it relates to designing the containment atmospheric cleanup systems to permit periodic pressure and functional testing to ensure the leak tight integrity and the operability and performance of the active components. The BWRX-300 containment atmosphere system is an inerted system design that incorporates adequate space and connections to allow for periodic pressure and functional testing as described in Subsection 9.3.6. Design completion will occur during FSAR phase.

GDC 46 – Testing of cooling water system, as it relates to designing the cooling water system to allow periodic pressure and functional testing to ensure the leak tight integrity and operability and performance of the active components. The cooling water system design incorporates adequate space and connections to allow for periodic pressure and functional testing. Design completion will occur during FSAR phase.

GDC 54 – Piping systems penetrating containment, as it relates to designing piping systems penetrating containment with the capability to test periodically the operability of the isolation valves and associated apparatus and determine valve leakage acceptability. The piping system containment penetration design incorporates adequate space and connections to allow for periodic pressure and functional testing. Design completion will occur during FSAR phase.

Additional Information on GDC compliance is found in Section 3.1.

Appendix B to 10 CFR 50, as it relates to quality assurance in the design, fabrication, construction, testing, and records control of SC1 pumps, valves, and dynamic restraints. Compliance with 10 CFR 50 Appendix B will occur during FSAR phase during design development.

IST Programs are developed for required operability and functional tests for components as described below:

- Necessary instrumentation, test connections, flow instruments, or other provisions that are required to fully comply with the requirements are included
- Valves required to be seat leakage tested have provisions for testing the leak-tightness of the valves in the direction in which the valves are required to be leak tight except as allowed in 10 CFR 50 Appendix J or ASME OM Code, “Operation and Maintenance of Nuclear Power Plants,” ISTC-3630(b) (Reference 3.9-10)
- ASME OM Code, Nonmandatory Appendix M provides design guidance of system and component testing

The IST Program includes periodic tests and inspections that demonstrate the operational readiness of SC1 components, pumps, valves, and snubbers (dynamic restraints) and their capability to perform their safety category functions. Table 3.9-4 provides a preliminary list of active valves. Active valves were identified from the perspective of their normal position and safety position.

The IST Program is based on the requirements of ASME OM Code as incorporated by reference in 10 CFR 50.55a as follows:

- The specific ASME OM Code requirements for functional testing of pumps are found in Subsection ISTF for plants designed after 2000
- Requirements for IST of valves are found in Subsection ISTC and Mandatory Appendices III (motor-operated valves) and IV (air-operated valves)
- Requirements for dynamic restraints (snubbers) are found in Subsection ISTD
- Requirements for IST of pressure relief devices are found in Mandatory Appendix I
- Requirements for Check Valve Condition Monitoring are found in Mandatory Appendix II
- General requirements for IST are found in Subsection ISTA
- Applicable edition

IST of Pumps

The BWRX-300 design does not require the use of pumps to perform a specific function in shutting down the reactor to the safe shutdown condition, maintaining the safe shutdown condition, or mitigating the consequences of an accident. Therefore, there are no pumps required to be included in the IST Program.

IST of Valves

Certain ASME BPVC Classes 1, 2, and 3 valves and pressure relief devices are subject to IST in accordance with ASME OM Code, Subsection ISTC or Appendix I, including the general requirements in Subsection ISTA. IST of valves assesses operational readiness, including actuating and position-indicating systems. The valves that are subject to IST include those valves that perform a specific function in shutting down the reactor to a safe shutdown condition, maintaining a safe shutdown condition, or mitigating the consequences of an accident. In addition, pressure relief devices used for protecting systems or portions of systems that perform a function in shutting down the reactor to a safe shutdown condition, maintaining a safe shutdown condition, or mitigating the consequences of an accident, are subject to IST.

ASME OM Code, Table ISTC-3500-1 requires the following four basic valve tests:

1. Exercise tests.
2. Seat leakage tests.
3. Remote position indicator tests.
4. Special tests (e.g., fail-safe tests, explosive valve tests, rupture disk tests).

Valve Categories

Non-exempt ASME BPVC Class 1, 2, and 3 valves are categorized in accordance with ASME OM Code, ISTC-1300(2) as follows:

1. Category A – valves for which seat leakage is limited to a specific maximum amount in the closed position for fulfillment of their required function(s).
2. Category B – valves for which seat leakage in the closed position is inconsequential for fulfillment of the required function(s).
3. Category C – valves that are self-actuating in response to some system characteristic, such as pressure (relief valve) or flow direction (check valve) for fulfillment of the required function(s).
4. Category D – valves that are actuated by an energy source capable of only one operation, such as rupture disks or explosively actuated valves.

When more than one distinguishing category characteristic is applicable, all requirements of each of the individual categories are applicable, although duplication or repetition of common testing requirements is not necessary.

Valve Exercise Tests

Active Category A valves, Category B valves, and Category C check valves are exercised periodically, except for self-actuated safety and relief valves. The ASME OM Code specifies a quarterly valve exercise frequency for all valves except power-operated safety and relief valves, which are required to be tested once per fuel cycle. Manual valve exercise tests are discussed in ISTC-3540. Where it is not practicable to exercise a valve during normal power operation, the valve exercise test is deferred to either cold shutdown or refueling outages. In some cases, quarterly stroke testing is deferred to refueling outages or cold shutdown. The bases for deferral

are consistent with ISTC-3500. Where practical, the BWRX-300 is designed to accommodate quarterly stroke testing.

During valve exercise tests, the necessary valve obturator movement is determined by exercising the valve while observing an appropriate direct indicator, such as indicating lights that signal the required changes of obturator position, or by observing other evidence or positive means, such as changes in system pressure, flow, level, or temperature that reflect change of obturator position.

Check valve exercise tests use direct observation or other positive means (ISTC-5221(a)) for verification of valve obturator position by performing both an open and a close test.

Active and passive Category A containment isolation valves are tested to verify seat leakage is within limits in accordance with 10 CFR 50 Appendix J. Frequencies of containment isolation valve seat leakage tests are in accordance with the 10 CFR 50 Appendix J requirements.

Other Category A valves are required to be seat leakage tested at least once every two years as specified by ISTC-3630.

Check valves that have seat leakage requirement are leak tested in accordance with ISTC-3600.

The USNRC allows licensees to apply the ASME OM Code Cases listed in RG 1.192, as incorporated by reference in 10 CFR 50.55a(a)(3)(iii), without prior USNRC approval, subject to the conditions in 10 CFR 50.55a(b)(6).

Relief from the testing requirements of the ASME OM Code is requested when compliance with requirements of the ASME OM Code is not practical. In such cases, specific information is provided which identifies the impractical code requirement, justification for the relief request, and the testing method to be used as an alternative. Demonstration of the impracticality of the testing required by the ASME OM Code, and justification for alternative testing proposed, are provided.

The BWRX-300 IST Program does not invoke the use of any ASME Code Cases for valve IST.

IST of Dynamic Restraints (Snubbers)

At this time, no snubbers are included in the BWRX-300 design. This could change as civil design matures. The following requirements are not all inclusive but provide IST requirements applicable to BWRX-300 if snubbers are incorporated at a future date.

- Preservice operability testing is to be performed on all snubbers. Tests verify the parameters described in ISTD-5131.
- Snubbers are tested at a load sufficient to verify the test parameters specified in ISTD-5131. Any snubbers that fail the preservice operational readiness test are evaluated for causes of failure. If a design deficiency is found, it is corrected, adjusted, repaired, or replaced. Failed snubbers are tested to meet the requirements in ISTD-5131.
- Snubber operational readiness checks verify the parameters described in ISTD-5210
- Snubbers are tested at a load sufficient to verify the test parameters specified in ISTD-5100, ISTD-5200, and ISTD-5300, or ISTD-5400 and ISTD-5500
- Snubbers are tested for operational readiness during each fuel cycle as defined in ISTD-5240. Test campaigns are required to be in accordance with a specified sampling plan as defined in ISTD-5260.

The snubbers IST sample is selected using one of the following two sample plans:

- 10% testing sample plan
- 37 testing sample plan (test 37 snubbers)

For more detailed requirements on testing sample plans, refer to ISTD-5300 and ISTD-5400:

- Snubbers preservice and IST, including their interval and percent of sample lot size requirements, are performed in accordance with ISTD-5131
- Snubbers testing documentation necessary to verify the results of the preservice and IST Program is in accordance with the requirements in ISTD-9000

3.9.7 Risk Informed Inservice Testing

This section will be addressed in the FSAR.

3.9.8 Risk Informed Inservice Inspection of Piping

This section will be addressed in the FSAR.

3.9.9 References

3.9-1 ASME BPVC, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” Appendices, American Society of Mechanical Engineers, 2021.

3.9-2 ASME BPVC, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” Subsection NB, “Class 1 Components,” American Society of Mechanical Engineers, 2021.

3.9-3 ASME BPVC, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” Subsection NCD, “Class 2 and Class 3 Components,” American Society of Mechanical Engineers, 2021.

3.9-4 ASME BPVC, Section III, “Rules for Construction of Nuclear Facility Components,” Subsection NCA, “General Requirements for Division 1 and Division 2,” American Society of Mechanical Engineers, 2021.

3.9-5 ASME BPVC, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” Subsection NE, “Class MC Components,” American Society of Mechanical Engineers, 2021.

3.9-6 ASME BPVC, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” Subsection NF, “Supports,” American Society of Mechanical Engineers, 2021.

3.9-7 ASME BPVC, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” Subsection NG, “Core Support Structures,” American Society of Mechanical Engineers, 2021.

3.9-8 ASME B31.1, “Power Piping,” American Society of Mechanical Engineers, 2022.

3.9-9 ASME B31.3, “Process Piping,” American Society of Mechanical Engineers, 2022.

3.9-10 ASME OM Code, “Operation and Maintenance of Nuclear Power Plants,” American Society of Mechanical Engineers.

3.9-11 NEDO-11209-A, “GE Hitachi Nuclear Energy Quality Assurance Program Description,” GE-Hitachi Nuclear Energy Americas, LLC, Revision 17, December 2022.

3.9-12 IEC/IEEE 60780-323, “Nuclear facilities – Electrical Equipment Important to Safety – Qualification,” International Electrotechnical Commission/Institute of Electrical and Electronics Engineers, 2016.

3.9-13 IEC/IEEE 60980-344, “Nuclear Facilities – Equipment Important to Safety – Seismic Qualification,” International Electrotechnical Commission/Institute of Electrical and Electronics Engineers, 2020.

3.9-14 ASME QME-1, “Qualification of Active Mechanical Equipment Used in Nuclear Facilities,” American Society of Mechanical Engineers, 2023.

3.9-15 NEDC-33910P-A, “BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection,” GE-Hitachi Nuclear Energy Americas, LLC, Revision 2, June 2021.

3.9-16 ASME BPVC, Section VIII, Division 1, “Rules for Construction of Pressure Vessels,” American Society of Mechanical Engineers, 2021.

3.9-17 NEDC-33934P, “BWRX-300 Safety Strategy,” GE Vernova Hitachi Nuclear Energy Americas, LLC, Revision 2, March 2026.

3.9-18 ASME B31.5, “Refrigeration, Piping, and Heat Transfer Component code,” American Society of Mechanical Engineers, 2019.

3.9-19 SIL No. 682, “Flow Induced Vibration Load,” GE-Hitachi Nuclear Energy Americas LLC, Revision 0, June 2019.

Table 3.9-1: Design Condition Definition

Service Condition	Typical ASME Service Level for Components	Plant State or Estimated Frequency Occurrence of PIE Used for Categorization
Normal	A	Planned Operation
Upset	B	AOO
Emergency	C	DBA of low frequency
Faulted	D	DBA of very low frequency
DEC	N/A; best-estimate methods may be applied	Event-specific basis (Note 1)

Note:

1. Events with a probability of occurrence in the DEC range (less than 1E-05 per reactor-year) are addressed on an event-specific basis.

Table 3.9-2: Cycles of Events

Event Number	Description	Design Basis Number of Cycles	Service Condition
1	Bolt-up	72	Normal
2A	Hydrostatic Testing	6	Normal
2B	System Leakage Test	144	Normal
3	Startup	200	Normal
4	Turbine Rolland Increase to Rated Power	200	Normal
5	Daily Load Reduction and Recovery	18,738	Normal
6	Rod Sequence/Pattern Change	60	Normal
7A	Inadvertent Isolation Condenser Initiation – One Train (Power Operation)	8	Upset
7B	Inadvertent Isolation Condenser Initiation – One Train (Hot Shutdown)	2	Upset
8	Feedwater Flow Increase	19	Upset
9	Turbine Valve Fail Open	25	Upset
10	Loss of Feedwater Heaters - Partial	59	Upset
11	Turbine Generator Trip, Other Scrams with Bypass Flow	67	Upset
12A	Scrams Without Bypass without Feedwater	37	Upset
12B	Scrams Without Bypass with Feedwater	30	Upset
13	Rated power Operation	N/A	Normal
14	Reduction to 0% Power	200	Normal
16	Hot Shutdown	200	Normal
16	Cold Shutdown	200	Normal
17	Vessel Flooding	72	Normal
18	Unbolt	72	Normal
19	Refuel	72	Normal
20	Design Basis Accident Pressurization With Feedwater	2	Emergency
21	Design Basis Accident Pressurization Without Feedwater	1	Emergency

Event Number	Description	Design Basis Number of Cycles	Service Condition
22	Design Basis Accident Without Pressurization	3	Emergency
23	Inadvertent Isolation Condenser Initiation – All Trains	5	Emergency
24	Small Pipe Rupture – Loss-of-Coolant Accident	1	Emergency
25	Bounding Transient Without Scram	N/A	N/A
26	Large Pipe Rupture – Loss-of-Coolant Accident	1	Faulted
27	Complex Sequence Pressure Increase	N/A	N/A

Table 3.9-3: Load Combinations and Acceptance Criteria

Plant Event / Event Combination	Service Loading Combination	ASME Service Level
Design	$P_D + T_D + R_D$	Design
Normal Operation	N	A
Plant / System AOO (Upset)	N + AOO	B
Seismic (Fatigue Only)	N + SSE	B (Note 1)
Emergency Level Transient, ATWS	N + ATWS	C
Emergency Level Transient, ELT	N + ELT	C
Small Break LOCA	N + SBLOCA	C
Small Break LOCA + SSE	N + SBLOCA + SSE	D
Large Break LOCA + SSE	N + LBLOCA + SSE	D
Faulted Level Transient, FLT	N + FLT	D
Test	$P_t + T_t + R_t$	Testing Limit

Notes:

1. Applies only to fatigue evaluation of ASME BPVC Class 1 components and core support structures.
2. Nomenclature:
   - AOO Loads for AOO event
   - D Dead Load
   - D_t Dead Load for Test Condition
   - ELT Emergency Level Transient
   - FLT Faulted Level Transient
   - LBLOCA Large Break Loss-of-Coolant Accident
   - N Normal Operation Loads
   - P_D Design Pressure
   - P_t Test Pressure
   - DBA Loads for DBA event
   - R_D Design Mechanical Loads
   - R_t Test Mechanical Loads
   - SBLOCA Small Break Loss-of-Coolant Accident
   - SSE Safe Shutdown Earthquake
   - T_D Design Temperature
   - T_t Test Temperature

Table 3.9-4: Preliminary Active Valve List

Quantity	Description	ASME Code Class
1	Reactor Pressure Vessel (RPV) Head Vent Inboard Isolation	1
1	RPV Head Vent Outboard Isolation	1
3	ICS Supply Inboard Isolation	1
3	ICS Supply Outboard Isolation	1
2	Main Steam Inboard Isolation	1
2	Main Steam Outboard Isolation	1
2	Reactor Water Cleanup Inboard Isolation	1
2	Reactor Water Cleanup Outboard Isolation	1
4	Feed Water Inboard Isolation	1
4	Feed Water Outboard Isolation	1
3	ICS Return Inboard Isolation	1
3	ICS Return Outboard Isolation	1
2	Main Steam Containment Isolation	2
1	RPV Leak Detection Excess Flow Check	2
6	Reference Leg Purge Water Check	2
3	RPV Water Level Reference Leg Excess Flow Check	2
3	RPV Water Level Variable Leg Excess Flow Check	2
1	RPV Head Vent Shutdown Water Level Excess Flow Check	2
6	A/B Main Steam Flow Venturi Low Pressure Tap Excess Flow Check	2
12	Containment Monitoring Isolation	2
7	Containment Monitoring Excess Flow Check	2
3	IC Purge Line Inboard Containment Isolation	1
3	IC Purge Line Outboard Containment Isolation	1
18	IC Steam Supply Leak Detection Excess Flow Check	2
18	IC Condensate Leak Detection Excess Flow Check	2
6	Condensate Return Isolation	1
2	Shutdown Cooling Inboard Isolation	1
2	Shutdown Cooling Outboard Isolation	1
1	Boron Injection Inboard Containment Check	1
1	Boron Injection Outboard Containment Isolation	1
29	HCU Charge Water Check	2
29	HCU SCRAM	2
29	HCU Purge Water Check	2

Quantity	Description	ASME Code Class
29	SCRAM Solenoid	2
57	FMCRD Ball Check Valve	1
4	Isolation Condenser (IC) Pool Check A check	2
4	IC Inner Pool B+C Check	2
2	IC Inner Pool Sparger Supply Isolation	2
1	IC Outer Pool Sparger Supply Isolation	2
2	IC Outer Pool B+C Suction Surge Tank Isolation	2
1	Filter/Storage Pool Sparger Supply Isolation	2
2	IC Outer Pool Drain Isolation	2
1	Reactor Water Cleanup Outboard Containment Isolation	2
2	Feedwater Containment Isolation	2
2	Feedwater System Isolation	2
2	Shutdown Cooling System Isolation	2
2	Containment Cooling Supply Outboard Containment Isolation	2
2	Containment Cooling Return Inboard Containment Isolation	2
2	Containment Cooling Supply Inboard Containment Check	2
2	Containment Cooling Return Outboard Containment Isolation	2
2	Containment Cooling Outboard Thermal Relief	2
1	Nitrogen Supply Outboard Containment Isolation	2
1	Nitrogen Supply Inboard Containment Check	2
6	Passive Containment Cooling System Isolation	2
3	Passive Containment Cooling Return Containment Isolation	2
3	Passive Containment Cooling Supply Containment Isolation	2
3	Passive Containment Cooling Air Supply Check	2
1	Containment Sump Inboard Containment Isolation	2
1	Containment Sump Outboard Containment Isolation	2

Note: Active valves identified from the perspective of their normal position and safety position.

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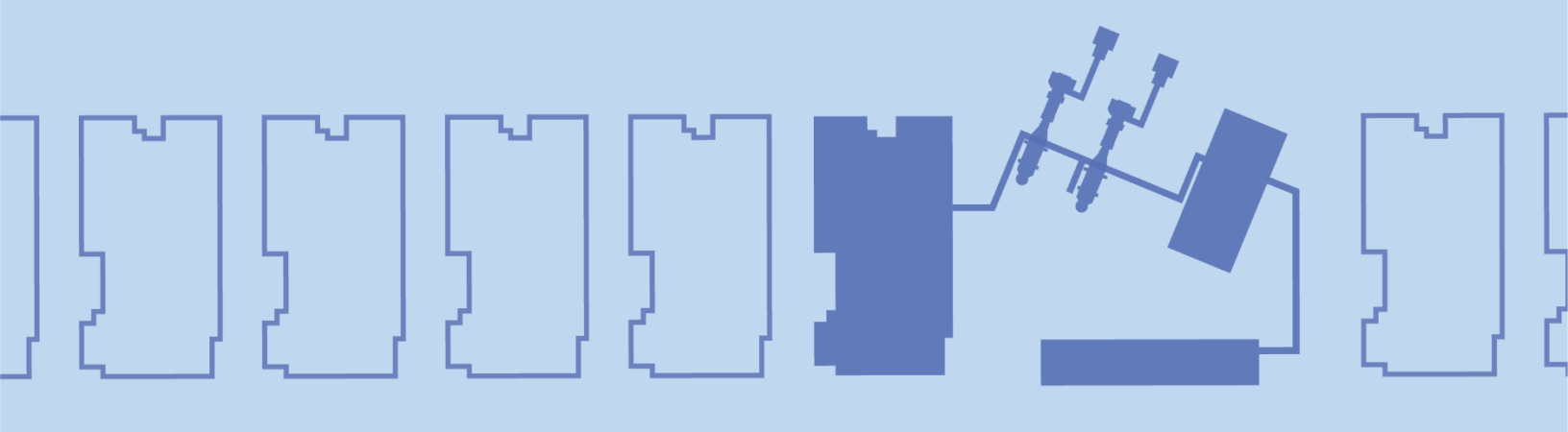
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Non-Proprietary Information

Chapter 3 Design of Structures, Components, Equipment, and Systems

Section 3.10 – Seismic and Dynamic Qualification of Mechanical and Electrical Equipment



ACRONYM LIST

Acronym	Explanation
ASME	American Society of Mechanical Engineers
BECPV-N01	Blue Energy Center Port of Victoria Nuclear Unit 1
BPVC	Boiler and Pressure Vessel Code
EQ	Environmental Qualification
GDC	General Design Criteria
I&C	Instrumentation and Control
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronics Engineers
ISRS	In-Structure Response Spectra
NPP	Nuclear Power Plant
RBV	Reactor Building Vibration
RG	Regulatory Guide
RCPB	Reactor Coolant Pressure Boundary
RRS	Required Response Spectra
SC1	Safety Class 1
SSCs	Structures, Systems, and Components
SSE	Safe Shutdown Earthquake
USNRC	United States Nuclear Regulatory Commission

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None.

3.10 Seismic and Dynamic Qualification of Mechanical and Electrical Equipment

The BWRX-300 Seismic Category I mechanical and electrical equipment (including Instrumentation and Control (I&C) components) are designed to withstand the effects of earthquakes (i.e., Seismic Category I requirements), and other accident-related dynamic loadings.

Mechanical equipment consists of pumps, valves, valve operators, vessels, and piping whose function is required to ensure safe operation or safe shutdown. Electrical equipment consists of all electrical power and I&C equipment, which includes all analog (non-digital) and digital I&C components. Computer-based I&C equipment is a subset of digital I&C components. Examples of electrical equipment are battery and battery racks, instrument and instrument racks, control consoles, electrical cabinets, electrical panels, valve operator motors, solenoid valves, pressure switches, relays, level transmitters, electrical penetrations, and pump and fan motors.

Structures, Systems, and Components (SSCs) that are credited to remain functional during or after a seismic event are Seismic Category I. SSCs whose failure during a seismic event could adversely affect the ability of Seismic Category I SSCs to accomplish their fundamental safety functions are classified as Seismic Category II. A list of preliminary BWRX-300 SSCs classification, their location, seismic category, and quality group is provided in Table 3A-1.

The BWRX-300 design discussed in Section 3.10 complies with the following General Design Criteria (GDC), which have plant-wide applicability, as discussed in Section 3.1:

GDC 1 – Quality standards and records

GDC 2 – Design bases for protection against natural phenomena

GDC 4 – Environmental and dynamic design bases

The following GDC have aspects specific to Section 3.10 and compliance is addressed as follows:

GDC 14 - Reactor Coolant Pressure Boundary (RCPB). Section 3.9.2 addresses dynamic testing of components of the RCPB to ensure it can withstand the applicable design-basis seismic and dynamic loads in combination with other environmental and natural phenomena loads without leakage, rapidly propagating failure, or gross rupture. Accordingly, the requirements of GDC 14 are met.

GDC-30 - Quality of reactor coolant pressure boundary. The methods of test and analysis employed to ensure the operability of mechanical and electrical equipment are based on joint standard International Electrotechnical Commission (IEC)/Institute of Electrical and Electronics Engineers (IEEE) 60980-344 “IEC/IEEE International Standard – Nuclear Facilities – Equipment Important to Safety – Seismic Qualification,” (Reference 3.10-1). Regulatory Guide (RG) 1.100 “Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants,” endorses IEEE-344-2013, “IEEE Standard for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating” and ASME QME-1-2017, “Qualification of Active Mechanical Equipment Used in Nuclear Power Plants.” The BWRX-300 design utilizes IEC/IEEE 60980-344-2020 (Reference 3.10-1) and ASME QME-1-2023 (Reference 3.10-2). Prior to issuance of the FSAR, the differences resulting from the revised reference documents will be reconciled along with the United States Nuclear Regulatory Commission (USNRC) exceptions and clarifications listed in RG 1.100. The additional guidance provided in RG 1.100 is used to identify individual components of the RCPB demonstrating through testing and analysis, or a combination of both, that a given component will not fail as a result of any combination of loadings for which it is qualified. This ensures that components are tested to the highest quality standards practical. Accordingly, the requirements of GDC 30 are met.

Additional Information on GDC compliance is found in Section 3.1.

Seismic design and design of Seismic Category I SSCs are addressed in Section 3.7 and Section 3.8. General Design aspects of mechanical systems and components are addressed in Section 3.9.

3.10.1 Seismic and Dynamic Qualification Criteria

A determination of the criteria for seismic and dynamic qualification is dependent on the type of equipment to qualify either mechanical, electrical, and/or I&C and the required seismic and dynamic inputs necessary to demonstrate structural and/or functional integrity. The criteria is provided in the following sections.

3.10.1.1 Qualification Standards

The guidance provided in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) Section III Division 1 “Rules for Construction of Nuclear Facility Components,” (Reference 3.10-3) is followed in the design of Safety Class 1 (SC1) mechanical equipment to achieve the structural integrity of pressure boundary components. SC1 valves consider the qualification guidance provided in ASME QME-1, “Qualification of Active Mechanical Equipment Used in Nuclear Facilities,” (Reference 3.10-2) for their qualification program.

Seismic and dynamic qualification of mechanical and electrical equipment and associated supports are considered for testing, analysis, or a combination of testing and analysis in accordance with IEC/IEEE 60980-344 in accordance with RG 1.100.

3.10.1.1.1 Qualification by Actual Seismic Experience

At this time in the design of the BWRX-300, there are no known experience-based qualifications that are considered appropriate for “qualification by actual seismic experience,” as permitted by RG 1.100. Therefore, this methodology is not utilized.

3.10.1.1.2 Qualification by Similarity

Qualification by similarity for Seismic Category I and II equipment is based on qualification of multiple similar pieces of equipment by testing and/or analyzing only one of the pieces of equipment. When extrapolation of data is made from similar equipment, a description of the differences between the equipment items involved is required. Justification that the differences do not degrade the environmental and/or seismic adequacy below acceptable limits and any additional supporting data is included.

Test results can be extrapolated for dynamic loading conditions in excess of, or different from, previous tests on a piece of equipment if the test results are in sufficient detail to allow an adequate dynamic model of the equipment to be generated. The model provides the capability of predicting failure under the increased or different dynamic load excitation. IEC/IEEE 60980-344 (Reference 3.10-1) defines the analytical method utilized in similarity qualification.

3.10.1.1.3 Functional Qualification of Active Mechanical Equipment

The seismic qualification of active mechanical equipment is performed considering the methods and requirements specified in ASME QME-1.

3.10.1.2 Qualification Program

The equipment qualification program follows the requirements provided in IEC/IEEE 60780-323, “IEC/IEEE International Standard – Nuclear Facilities – Electrical Equipment Important to Safety – Qualification,” (Reference 3.10-4) and is used to determine the overall equipment qualification test plan, including Environmental Qualification (EQ) provided in Section 3.11. The program

meets the qualification criteria contained in IEC/IEEE 60980-344 (Reference 3.10-1) that includes seismic and dynamic mechanical and electrical equipment qualification.

3.10.1.3 Seismic Qualification Report

The seismic qualification report follows the requirements that are defined in IEC/IEEE 60980-344 and is specific to the Seismic Category I electrical and mechanical equipment and associated supports to be qualified.

3.10.2 Methods and Procedures for Qualifying Mechanical and Electrical Equipment

3.10.2.1 Seismic Input Motion

Dynamic load conditions are simulated by testing using independent, random multi-frequency input or single frequency input motion (within equipment capability) over the frequency range of interest.

Acceptable justification for use of single frequency input includes, but is not limited to:

1. The characteristics of the required input motion are dominated by one frequency.
2. The anticipated response of the equipment is adequately represented by one mode.
3. The input has sufficient intensity and duration to excite all modes to the required magnitude so that the testing response spectra envelop the corresponding response spectra of the individual modes.
4. The time phasing of the inputs in the vertical or horizontal directions is such that a purely rectilinear resultant input is avoided.

The actual input motion used during testing, for both multi and single frequency, envelops the applicable input motion (floor, wall, response) at the location(s) of the equipment under test.

When the equipment is qualified by dynamic test, the In-Structure Response Spectra (ISRS) or time histories are used in determining required response spectra of input motion used for the test.

When both test and analysis are defined as acceptable methods, the deciding factors considered (as applicable) for choosing between tests or analysis includes:

- Magnitude of accelerations and frequency content of seismic and Reactor Building Vibration (RBV) dynamic loadings
- Environmental conditions associated with the dynamic loadings
- Nature of the function(s) required for a seismic event
- Size and complexity of the equipment
- Dynamic characteristics of expected failure modes (structural or functional)
- Partial test data upon which to base the analysis

Tests or analyses of assemblies are preferable to tests or analyses on separate components (e.g., a motor and a pump, including the coupling and other appurtenances, should be tested, or analyzed as an assembly). The replacement parts may be tested separately, if applicable.

Equipment that has been previously qualified by means of tests and analyses equivalent to those required for the current qualification program are used if proper documentation of such tests and analyses is available.

3.10.2.2 Qualification by Testing

Seismic qualification of mechanical and electrical equipment including I&C by testing is performed in accordance with the requirements of IEC/IEEE 60980-344.

3.10.2.2.1 Interface Requirements

Intervening structures or components (such as interconnecting cables, bus ducts, conduits) that serve as interfaces between the equipment to be qualified and that are supplied by others, are not qualified as part of the seismic equipment qualification program. When applicable, accelerations and frequency content at locations of interfaces with interconnecting cables, bus ducts, and conduits are determined and documented. This information is specified in the form of interface criteria.

3.10.2.2.2 Test Methods

The test methods presented in IEC/IEEE 60980-344 provide acceptable types of testing dependent on the type of motion selected based on the expected vibration environment and technical requirements of the specific application. Past testing demonstrates that BWRX-300 Seismic Category I electrical equipment has critical damping ratios equal to or less than 5%. Hence, Required Response Spectra (RRS) at 5% or less critical damping ratio are developed as input to the equipment base.

The preferred method for seismic testing is to use triaxial, multi-frequency testing. However, if justified, biaxial and single-axis testing is acceptable. If biaxial testing is justified to be used, then each test is performed in two steps, where the first step is to apply the input motion to both the vertical and horizontal axis simultaneously. For the second step, the test specimen is rotated 90 degrees in the horizontal plane, and a second test is performed with the input motion applied to the vertical and horizontal axis. Therefore, biaxial testing at a minimum requires twice the number of runs as triaxial testing. The preferred method for biaxial testing is independent, random tests.

For biaxial testing, when independent, random tests are not available, four tests are performed:

1. With the inputs in phase.
2. With one input 180 degrees out of phase.
3. With the equipment rotated 90 degrees horizontally and the inputs in phase.
4. With the same orientation as in the step (3), but with one input 180 degrees out of phase.

3.10.2.2.3 Selection of Test Specimen

Representative samples of equipment and supports are selected for use as test specimens. Variations in the configuration of the equipment are analyzed with supporting test data. Test specimen assemblies that represent multiple configurations are configured to represent the “worst case” configuration. For example, these variations may include mass distributions that differ from one cabinet to another. Therefore, it is analyzed and justified which mass distribution(s) results in the maximum stresses, such as response accelerations or frequency content, and this worst-case configuration(s) is used as the test specimen(s).

3.10.2.2.4 Mounting of Test Specimen

The test specimen is mounted to the test table so that the installed configuration, including interfaces, is adequately simulated and differences between the configuration are evaluated and resolved. If the test specimen is intended to be mounted to a panel or enclosure, the panel, enclosure, or a test fixture representative of the mounting conditions is included in the testing,

unless justified. If the test specimen cannot be mounted directly to the table due to mounting constraints, an interposing test fixture is designed and used as the mounting interface. However, the equipment-to-fixture mounting condition is to simulate its installed configuration and cause no dynamic coupling to the equipment. If the equipment being analyzed has no required orientation, the worst possible orientation is considered. The test specimen is considered to be in its operational configuration (i.e., filled with the appropriate fluid and/or solid). The investigation ensures that the point of maximum stress is considered. The test specimen mounting, and configuration includes hardware interface requirements. For interfaces that cannot be simulated on the test table, the accelerations and any resonances at such interface locations are recorded during the equipment test and documented in the test report.

3.10.2.2.5 Aging and Vibration Conditioning

The testing simulates the effects of aging. Equipment is reviewed in terms of design, function, materials, and environment for its specified application to identify potentially significant aging mechanisms.

If equipment is subjected to vibrational loads throughout its lifetime in its in-service mounted condition, then vibration aging to its end-of-life condition is performed prior to seismic qualification when required by the applicable qualification standard(s).

3.10.2.3 Qualification by Analysis

Qualification by analysis without testing may be acceptable on equipment that is only required to maintain its structural integrity to perform its safety function as described in IEC/IEEE 60980-344.

Dynamic analysis or an equivalent static analysis is employed to qualify the equipment when analysis is chosen as the method for qualification. The decision on using dynamic versus static analysis is typically defined based on whether the equipment is rigid or flexible.

If the fundamental frequency of the equipment is above the input excitation frequency (cutoff frequency of RRS), the equipment is considered rigid. The search for the natural frequency is done analytically, if the equipment shape is defined mathematically, or by prototype testing.

If the equipment is determined to be a rigid body (i.e., shown to have no resonance frequency within the expected frequency range), the static analysis method is able to be applied in place of dynamic analysis.

If the equipment is determined to be flexible (i.e., with the fundamental frequency of the equipment within frequency range of the input spectra) and not simple enough for equivalent static analysis, a dynamic analysis method is applied, unless justified otherwise.

If it is determined either dynamic or static analysis can be used, in general, the choice of the analysis is based on the expected design margin because the static coefficient method (the easiest to perform) is far more conservative than the dynamic analysis method.

For static analysis, the dynamic forces on each component can be obtained by concentrating the mass at the center of gravity and multiplying the mass by the appropriate floor acceleration. The dynamic stresses are then added to the operating stresses, and a determination is made of the adequacy of the strength of the equipment.

A static coefficient analysis may also be used for certain equipment in lieu of the dynamic analysis. No determination of natural frequencies is made in this case. The seismic loads are determined statically by multiplying the actual distributed weight of the equipment by a static coefficient equal to 1.5 times the peak value of the RRS at the equipment mounting location, at a conservative and justifiable value of damping.

Both types of analyses are to verify integrity of the equipment is maintained under low level earthquake loads, including appropriate RBV dynamic loads in combination with normal operating loads, and Safe Shutdown Earthquake (SSE) loads, including appropriate RBV dynamic loads, unless otherwise justified.

Subsection 3.9.3.1 defines acceptable load combinations and methods for combining dynamic responses for mechanical equipment. The same criteria are acceptable for electrical equipment.

3.10.2.4 Qualification by Combined Testing and Analysis

Qualification by combined testing and analysis is used as a method for qualification for complex or large equipment where it is not practical to test the entire assembly or it is too large to be tested at once, unless another method of qualification is justified.

One method of combined qualification is to use a representative prototype portion or scaled-down prototype of the assembly that is subjected to type testing. The data from the type testing is then used to develop and validate an analytical model of the prototype. The prototype analytical model is then extrapolated to represent the larger assembly and the results used to justify qualification of the equipment based on prototype testing.

A second method of combined qualification is to mount the full assembly to a rigid floor to simulate service mounting, and then a portable shaker test (or an impact or pull test if justified) is performed to excite the natural or resonance frequencies of the specimen. The amplification of resonance motion is used to determine the appropriate modal frequency and damping for a dynamic analysis of the equipment.

For equipment with multiple site configurations, the combined qualification method can be applied to reduce the number of configurations to be tested. In this case, an evaluation must be performed to determine the enveloping “worst-case” configuration(s), which is then tested. Analysis is then used to justify the various configurations based on the “worst-case” configuration(s).

The combination method is used for qualification of larger electrical equipment support assemblies containing SC1 equipment where it is not practical to test the entire assembly or it is too large to be tested at once, unless another method of qualification is justified. For this case, a test is run to determine if there are natural frequencies in the support equipment within the critical frequency range (any frequency below the cutoff frequency on the response spectrum). If the support is determined to be free of natural frequencies in the critical frequency range, then it is assumed to be rigid, and a static analysis is performed and calculations of transmissibility and responses to varying input accelerations are determined to see if SC1 equipment mounted in the assembly would operate without malfunctioning.

For digital I&C equipment qualification in a mild environment, analysis can be used in addition to testing if there is testing of an identical or similar item, or there is operating experience of equipment under identical or similar conditions with a supporting analysis to show that the equipment to be qualified is acceptable. Also, I&C equipment qualification can be performed using an analysis in combination with partial type test data that supports the analytical assumptions and conclusions.

3.10.3 Methods and Procedures of Analysis or Testing of Supports for Mechanical, Electrical Equipment and Instrumentation

Methods and procedures of analysis or testing of supports for mechanical and electrical equipment and instrumentation are in accordance with IEC/IEEE 60980-344 and ASME QME-1.

3.10.3.1 Supports for Battery Racks, Instrument Racks, Control Consoles, Cabinets, and Panels

SC1 control boards, panels, and racks should consider the qualification guidance provided in IEEE 420, “IEEE Standard for Design and Qualification of Class 1E Control Boards, Panels, and Racks Used in Nuclear Power Generating Stations,” (Reference 3.10-5) for their qualification program.

3.10.3.2 Cable Trays and Conduit Supports

SC1 cables consider the qualification guidance provided in IEEE 383, “IEEE Standard for Qualifying Electric Cables and Splices for Nuclear Facilities,” (Reference 3.10-6) and IEEE 384, “IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits,” (Reference 3.10-7), and test requirements in IEEE 1202, “IEEE Standard for Flame-Propagation Testing of Wire and Cable” (Reference 3.10-8).

Supports provided by the equipment supplier to be used for the equipment is to be qualified in accordance with this section by the equipment supplier.

Seismic Category I supports (hangers) that support trays or conduit that carry safety circuits are designed and analyzed to demonstrate qualification in accordance with IEEE 628, “IEEE Standard Criteria for the Design, Installation, and Qualification of Raceway Systems for Class 1E Circuits for Nuclear Power Generating Stations” (Reference 3.10-9).

SC1 connection assemblies consider the qualification guidance provided in IEEE 572, “IEEE Standard for Qualification of Class 1E Connection Assemblies for Nuclear Power Generating Stations and Other Nuclear Facilities,” (Reference 3.10-10) for the qualification program as endorsed by RG 1.156, “Qualification of Connection Assemblies for Production and Utilization Facilities.”

3.10.3.3 Line-Mounted Equipment

IEC/IEEE 60980-344 identifies special consideration is required for line-mounted (piping and duct system) equipment regarding seismic qualification as the most critical seismic loading condition can occur as a result of the piping or duct system.

Guidance and further clarification for special considerations for line-mounted equipment are provided in IEC/IEEE 60980-344 as well as IEEE 382, “IEEE Standard for Qualification of Safety-Related Actuators for Nuclear Power Generating Stations and Other Nuclear Facilities,” (Reference 3.10-11). Line-mounted equipment may also include active mechanical equipment subjected to ASME QME-1, including Non-mandatory Appendix QR-A.

3.10.4 References

3.10-1 IEC/IEEE 60980-344, “IEEE/IEC International Standard – Nuclear Facilities – Equipment Important to Safety – Seismic Qualification,” International Electrotechnical Commission/Institute of Electrical and Electronics Engineers, 2020.

3.10-2 ASME QME-1, “Qualification of Active Mechanical Equipment Used in Nuclear Facilities,” American Society of Mechanical Engineers, 2023.

3.10-3 ASME BPVC, Section III, “Rules for Construction of Nuclear Facility Components,” Subsections NB and NCD, American Society of Mechanical Engineers, 2021.

3.10-4 IEC/IEEE 60780-323, “IEC/IEEE International Standard – Nuclear Facilities – Electrical Equipment Important to Safety – Qualification,” International Electrotechnical Commission/Institute of Electrical and Electronics Engineers, 2016.

3.10-5 IEEE 420-2023, “IEEE Standard for Design and Qualification of Class 1E Control Boards, Panels, and Racks Used in Nuclear Power Generating Stations,” Institute of Electrical and Electronics Engineers, 2023.

3.10-6 IEEE 383-2023, “IEEE Standard for Qualifying Electric Cables and Splices for Nuclear Facilities,” Institute of Electrical and Electronics Engineers, 2023.

3.10-7 IEEE 384-2018, “IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits,” Institute of Electrical and Electronics Engineers, 2018.

3.10-8 IEEE 1202-2006, “IEEE Standard for Flame-Propagation Testing of Wire and Cable,” Institute of Electrical and Electronics Engineers, 2006.

3.10-9 IEEE 628-2020, “IEEE Standard Criteria for the Design, Installation, and Qualification of Raceway Systems for Class 1E Circuits for Nuclear Power Generating Stations,” Institute of Electrical and Electronics Engineers, 2020.

3.10-10 IEEE 572-2019, “IEEE Standard for Qualification of Class 1E Connection Assemblies for Nuclear Power Generating Stations and Other Nuclear Facilities,” Institute of Electrical and Electronics Engineers, 2019.

3.10-11 IEEE 382-2019, “IEEE Standard for Qualification of Safety-Related Actuators for Nuclear Generating Stations and Other Nuclear Facilities,” Institute of Electrical and Electronics Engineers, 2019.

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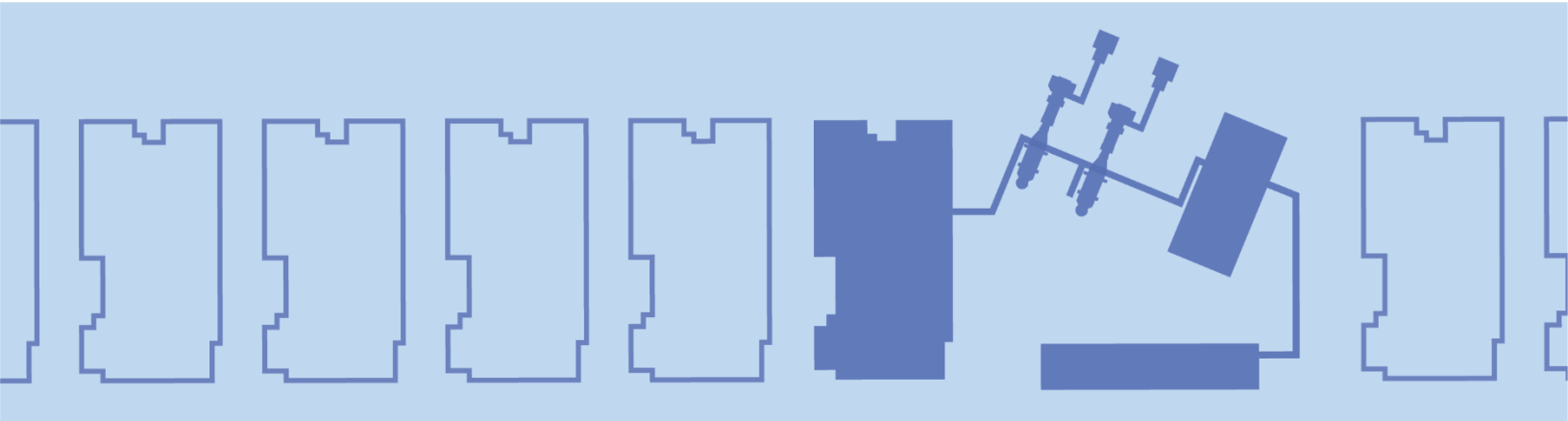
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Non-Proprietary Information

Chapter 3 Design of Structures, Components, Equipment, and Systems

Section 3.11 – Environmental Qualification of Mechanical and Electrical Equipment



ACRONYM LIST

Acronym	Explanation
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
BPVC	Boiler and Pressure Vessel Code
DBA	Design Basis Accident
EMC	Electromagnetic Compatibility
EMI/RFI	Electromagnetic Interference/Radio Frequency Interference
EPRI	Electric Power Research Institute
EQ	Environmental Qualification
GDC	General Design Criteria
I&C	Instrumentation and Control
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronics Engineers
RG	Regulatory Guide
SC	Safety Class
SC1	Safety Class 1
SC2	Safety Class 2
SC3	Safety Class 3
SCN	Non-Safety-Class
SSCs	Structures, Systems, and Components
SSE	Safe Shutdown Earthquake
USNRC	United States Nuclear Regulatory Commission
V&V	Verification and Validation

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3.11 Environmental Qualification of Mechanical and Electrical Equipment

Environmental Qualification (EQ) includes the generation and maintenance of evidence to ensure Structures, Systems, and Components (SSCs) can perform their intended design functions and remain fit for purpose in the conditions under which they are expected to perform. Section 3.10 provides the methodology and requirements used for the seismic and dynamic qualification of Seismic Category I Mechanical and Electrical equipment.

Mild Environment

An environment that would at no time be significantly more severe than the environment that would occur during normal plant operation, including Anticipated Operational Occurrences (AOOs), and does not give rise to significant aging mechanisms.

Harsh Environment

An environment that significantly changes from normal including design basis events and post-accident conditions as a result of a Design Basis Accident (DBA).

AOO Environment

AOO environmental conditions are the service conditions as a result of an operational deviation expected to occur during the operating plant lifetime that do not lead to accident conditions.

The methodology and requirements apply to the EQ of Safety Class 1 (SC1) mechanical and electrical equipment, including Instrumentation and Control (I&C), located in harsh and mild environments. International Electrotechnical Commission/ Institute of Electrical and Electronics Engineers (IEC/IEEE) 60780-323, “Nuclear Facilities – Electrical Equipment Important to Safety – Qualification,” (Reference 3.11-1), as endorsed by United States Nuclear Regulatory Commission (USNRC) Regulatory Guide (RG) 1.89, “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants,” defines the methodology and criteria used to qualify SC1 equipment for harsh and mild environments for the BWRX-300.

The BWRX-300 design discussed in Section 3.11 complies with the following GDC, which have plant-wide applicability, as discussed in Section 3.1:

GDC 1 – Quality standards and records

GDC 2 – Design bases for protection against natural phenomena

GDC 4 – Environmental and dynamic design bases

The following GDC aspects specific to Section 3.11 and compliance is addressed as follows:

GDC 23 - Protection system failure modes. The environmental conditions in which the instrumentation and equipment of the SC1 systems operate are considered in establishing the component specifications. Instrumentation specifications are based on the worst expected ambient environmental conditions in which the instruments operate. A list of preliminary BWRX300 SSCs classification, their location, seismic category, and quality category is provided in Table 3A-1. Accordingly, the requirements of GDC 23 are met.

Additional Information on GDC compliance is found in Section 3.1.

Qualification of mechanical equipment that performs a safety function is in accordance with American Society of Mechanical Engineers (ASME) QME-1, “Qualification of Active Mechanical Equipment Used in Nuclear Facilities,” (Reference 3.11-2).

3.11.1 Equipment Identification and Environmental Conditions

3.11.1.1 Equipment Identification

The equipment qualification program generates and maintains a list of SC1 equipment located in harsh and mild environments. The qualification plan includes the following parameters, at minimum and as applicable: the test and/or analysis sequence, environmental and/or seismic/dynamic or Electromagnetic Compatibility (EMC) requirements, test item functions, identification of industry codes and standards applicable to equipment, identification of the test equipment including description and calibration plan, and test item part numbers, quantity, mounting, and connection details.

3.11.1.2 Environmental Conditions

3.11.1.2.1 General Requirements

Environmental Design Bases

The environmental conditions consider normal, AOO, accident, and post-accident conditions, as applicable. Equipment located below the maximum flood level considers the effects of submergence and is qualified for flooding if it is required to function in this condition. Post-accident monitoring equipment considers the criteria for accident monitoring instrumentation and EQ guidance provided in IEEE 497, “IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations,” (Reference 3.11-3) as endorsed by RG 1.97, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants,” consistent with 10 CFR 50.49(b)(3).

The harsh environment qualification program verifies that the equipment is designed to be compatible and perform its safety functions during normal conditions, postulated environmental conditions, DBA, and post-accident conditions.

Equipment located within harsh environment conditions is exposed to environmental conditions including temperature, pressure, relative humidity, radiation, and chemical sprays.

Equipment determined to have a significant aging mechanism and located in a harsh environment account for the aging mechanism in the qualification program.

Aging mechanisms to be analyzed for equipment located in a harsh environment include time-temperature degradation (thermal), cycle aging (wear), and normal radiation exposure.

Analysis is performed to identify the environmental design bases for AOOs, normal, accident, and post-accident environments as applicable.

Equipment is qualified to the worst-case environmental conditions for the areas in which they are located for the duration that they are required to perform their Safety Category 1 function.

The Safety Category 1 functions are either functional performance requirements or fail-safe requirements. A fail-safe Safety Category 1 function consists of not failing in a manner detrimental to plant safety, accident mitigation, or prevention of a Safety Category 1 function. The basis for the Safety Category 1 function is included in the qualification documentation.

Although EQ by testing or analysis is not required for Safety Class 2 (SC2) and Safety Class 3 (SC3) components, these components are designed for their expected duty cycle and environmental conditions over the design life of the plant with due consideration for maintenance and aging management. Additionally, any non-SC1 electrical component whose failure under postulated environmental conditions could prevent satisfactory accomplishment of a Safety Category 1 function is qualified to the specified environmental conditions by testing or analysis in accordance with 10 CFR 50.49(b)(2).

The environments are considered for electrical and mechanical equipment in the EQ program such as temperature, pressure, humidity, chemical effects, radiation, and flooding effects. The EQ program uses the recommended environmental margins per IEC/IEEE 60780-323, Table 1.

Aging requirements apply to SC1 equipment. For equipment located in harsh and mild environments, the effect of aging is to be performed prior to DBA testing when a significant aging mechanism exists. Equipment is reviewed in terms of design, function, materials, and environment for its specified application to identify potentially significant aging mechanisms.

Equipment that could be exposed to radiation is environmentally qualified to a radiation dose that simulates the calculated radiation environment (normal and accident) that the equipment can withstand prior to completion of its required safety functions.

Radiation qualification considers that equipment damage is a function of total integrated dose and can be influenced by dose rate, energy spectrum, and particle type. The radiation qualification includes doses from all potential radiation sources at the equipment location. For equipment that is required to be functional post-accident, then the radiation dose is increased beyond the dose required for qualified life to envelop post-accident conditions as well, unless it is determined to cover post-accident conditions separately.

A mild radiation environment for electronic equipment is defined as a total integrated dose less than 10 gray (Gy) (1.0E03 rad), and a mild radiation environment for other equipment is less than 100 Gy (1.0E04 rad) as defined in RG 1.89.

Electronic and electrical equipment are tested with the equipment energized and performing its safety function if the required total integrated dose exceeds the mild environment level. This ensures equipment is qualified for the worst-case radiation with DBA margin per the requirements of IEC/IEEE 60780-323.

Electromagnetic Interference / Radio Frequency Interference and Voltage Surges

EMC requirements apply to all levels of safety class equipment, SC1, SC2, SC3, and Non-Safety Class (SCN) and provides qualification methods and implementation guidance. EMC qualifications for BWRX-300 design follow the requirements defined in (1) Electric Power Research Institute (EPRI) TR-102323, “Guidelines for Electromagnetic Compatibility Testing of Power Plant Equipment,” (Reference 3.11-4), or (2) Military Standards MIL-STD-461G, “Requirements for the Control of Electromagnetic Interference Characteristics of Subsystems and Equipment,” (Reference 3.11-5), or (3) IEC-62003, “Nuclear Power Plants – Instrumentation, Control, and Electrical Power Systems – Requirements for Electromagnetic Compatibility Testing,” (Reference 3.11-6). The qualification for Electromagnetic Interference/Radio Frequency Interference (EMI/RFI) and voltage surges for EQ equipment in harsh and mild environments is by test, consistent with USNRC RG 1.180 “Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems.” EMC Qualification and Acceptance Testing includes tests for susceptibility and emissions. Susceptibility and emissions requirements are applied to all Safety Class (SC) and SCN microprocessor-based I&C and electrical equipment.

3.11.2 Qualification Tests and Analyses

3.11.2.1 Qualification By Testing

Type testing is the preferred method for demonstrating that equipment is Environmentally Qualified. A type test subjects a representative sample of equipment, including interfaces, to a series of tests, and includes simulating the effects of significant aging mechanisms during normal operation. The sample is subsequently subjected to conditions that simulate DBA harsh conditions and thereby establishes the tested configuration for installed equipment service,

including mounting, orientation, interfaces, conduit sealing, and expected environments. A type test demonstrates that the equipment performs the intended Safety Category function(s) for the required operating time before, during, and/or following the DBA, as appropriate.

Tests are performed in accordance with applicable industry standards, such as IEC/IEEE 60780-323.

3.11.2.2 Qualification by Analysis

In general, analysis is used to supplement test data, and the analytical techniques and modeling assumptions are, when possible, based on a correlation of the analytical approach with testing or operating experience performed on similar equipment or structures.

Seismic and dynamic qualification by analysis is described in Subsection 3.10.2.3.

For qualification by analysis, a logical assessment, or a valid mathematical model of the equipment to be qualified is required, and the basis for the analysis includes physical laws of nature, results of test data, operating experience, and condition indicators, as applicable.

Analysis of data and tests for material properties, equipment rating, and environmental tolerance are acceptable methods to be used to demonstrate qualification.

Analysis alone is not used to demonstrate the initial qualification for electrical equipment in a harsh environment.

3.11.2.3 Qualification by Operating Experience

Qualification by use of operating experience requires documented data to be available confirming to the following conditions are met:

- The product providing the operating experience is identical or justifiably similar to the equipment to be qualified
- The product providing the operating experience has operated under service conditions which equal or exceed, in severity the service conditions and performance requirements for which the product is to be qualified are bounded by the product providing the operating experience
- The installed product in general is removed from service and subjected to partial type testing to include the DBA environments for which the product is to be qualified

3.11.2.4 Combined Qualification

Combination of test and analysis is used when it is deemed practical to use both methods to complete the qualification. The combined qualification method can be used for qualification for larger electrical equipment where it is not practical to test the entire assembly, or it is too large to be tested at once, unless another method of qualification is justified.

For digital I&C equipment qualification in a mild environment, analysis can be used in addition to testing if there is testing of an identical item of equipment under identical conditions or under similar conditions or operating experience with a supporting analysis to show that the equipment to be qualified is acceptable. Also, I&C equipment qualification can be performed using an analysis in combination with partial type test data that supports the analytical assumptions and conclusions.

3.11.2.5 Specific Equipment Requirements

3.11.2.5.1 Mechanical Equipment

SC1 mechanical equipment, which has the sole Safety Category 1 function of maintaining pressure integrity, and which is designed, fabricated, and qualified consistent with ASME Boiler and Pressure Vessel Code (BPVC), Section III, “Rules for Construction of Nuclear Facility Components,” (Reference 3.11-7), is considered qualified.

For mechanical equipment where the loading under normal service is more severe than loading under DBA, then the loading under normal service is considered in addition to the loading under DBA by test and/or analysis.

For mechanical equipment, the loading and capability under DBA conditions is analyzed in the qualification process to establish the suitability of materials, parts, and equipment needed for safety functions, and to verify that the design of such materials, parts, and equipment is adequate.

The qualification of mechanical equipment includes, as applicable, materials that are sensitive to environmental effects (e.g., seals, gaskets, lubricants, fluids for hydraulic systems, and diaphragms), required operating time, non-metallic subcomponents of such equipment, the environmental conditions and process parameters for which this equipment is qualified, non-metallic material capabilities, and the evaluation of environmental effects.

In addition, the qualification guidance provided in ASME QME-1 is considered for qualification of SC1 valves and SC1 mechanical pipe supports. The qualification of nonmetallic parts considers the qualification guidance provided in ASME QME-1, Nonmandatory Appendix QR-B.

3.11.2.5.2 Electrical Equipment

Additional qualification guidance is considered for specific electrical equipment, if applicable, as follows:

- RG 1.158 “Qualification of Safety-Related Vented Lead-Acid Storage Batteries for Nuclear Power Plants,” which endorses IEEE 535 “IEEE Standard for Qualification of Class 1E Vented Lead Acid Storage Batteries for Nuclear Power Generating Stations,” (Reference 3.11-8)
- RG 1.40 “Qualification of Continuous Duty Safety-Related Motors for Nuclear Power Plants,” which endorses IEEE 334, “IEEE Standard for Qualifying Continuous Duty Class 1E Motors for Nuclear Power Generating Stations,” (Reference 3.11-9) if considered applicable to BWRX-300 design
- RG 1.63 “Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants,” which endorses IEEE 317, “IEEE Standard for Electrical Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations,” (Reference 3.11-10)
- RG 1.73 “Qualification Tests for Safety-Related Actuators in Nuclear Power Plants,” which endorses IEEE 382, “IEEE Standard for Qualification of Safety-Related Actuators for Nuclear Power Generating Stations and Other Nuclear Facilities,” (Reference 3.11-11)
- RG 1.89 “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants” which endorses IEC/IEEE-60780-323 (Reference 3.11-1) that includes IEEE 638 “IEEE Standard for Qualification of Class 1E Transformers for Nuclear Power Generating Stations,” (Reference 3.11-12)
- RG 1.213 “Qualification of Safety-Related Motor Control Centers for Nuclear Power Plants,” considers conformance with the requirements of IEEE 649, “IEEE Standard for

Qualifying Class 1E Motor Control Centers for Nuclear Power Generating Stations," (Reference 3.11-13) if considered applicable to BWRX-300 design

- RG 1.210 "Qualification of Safety-Related Battery Chargers and Inverters for Nuclear Power Plants," which endorses IEEE 650 "IEEE Standard for Qualification of Class 1E Static Battery Chargers, Inverters, and Uninterruptible Power Supply Systems for Nuclear Power Generating Stations," (Reference 3.11-14)

3.11.2.5.3 Instrumentation & Control Equipment

Additional qualification guidance is considered for specific I&C equipment, if applicable, as follows:

- Control boards, panels, and racks classified as SC1 components – IEEE 420, "IEEE Standard for Design and Qualification of Class 1E Control Boards, Panels, and Racks Used in Nuclear Power Generating Stations," (Reference 3.11-15) for their qualification program

Qualification of computer based I&C systems is in accordance IEEE 7-4.3.2 "IEEE Standard Criteria for Programmable Digital Devices in Safety Systems of Nuclear Power Generating Stations," (Reference 3.11-16). The EMC requirements are specified in RG 1.180, IEEE 7-4.3.2 does not directly address RG 1.180 although the guidance in the RG is considered for I&C equipment.

When computer based I&C systems environmental type testing is performed:

- The system under test demonstrates that it functions and performs with safety software that has been validated and verified and is representative of the software to be installed in service
- The testing demonstrates performance of all safety function that affected by environmental factors under the environmental service conditions specified in the design specification. Software algorithms, which are tested during Verification and Validation (V&V) testing, are not tested unless their outputs exercise different hardware components which are affected impacted by environmental conditions
- The testing exercises all portions of the system that are necessary to accomplish the safety functions and those portions whose operation or failure could impair the safety functions
- The testing confirms the response of digital interfaces and verifies that the design accommodates the potential effect of environmental conditions on the overall response of the system

When computer based I&C systems environmental type testing is performed, the testing of a complete system is preferred. When testing of a complete system is not practical, confirmation of the dynamic response to the most limiting environmental and operational conditions is based on type testing of the individual modules and analysis of the cumulative effects of environmental and operational stress on the entire system to demonstrate required safety performance.

3.11.2.5.4 Cables, Raceways, Supports

For qualification of SC1 cables, the qualification guidance provided in IEEE 383 "IEEE Standard for Qualifying Electric Cables and Splices for Nuclear Facilities," (Reference 3.11-17) and IEEE 384, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits," (Reference 3.11-18) are considered. The test requirement guidance provided in IEEE 1202 "IEEE

Standard for Flame-Propagation Testing of Wire and Cable," (Reference 3.11-19) is used as a qualification program.

Seismic Category I supports (hangers) that support trays or conduit that carry SC1 circuits are designed and analyzed to demonstrate qualification in accordance with IEEE 628 "IEEE Standard Criteria for the Design, Installation, and Qualification of Raceway Systems for Class 1E Circuits for Nuclear Power Generating Stations," (Reference 3.11-20).

Seismic Category II supports used for SCN raceway (conduit and cable tray) in Seismic Category I and II structures are analyzed to withstand the effects of a Safe Shutdown Earthquake (SSE).

SC1 connection assemblies consider the qualification guidance provided in IEEE 572, "IEEE Standard for Qualification of Class 1E Connection Assemblies for Nuclear Power Generating Stations and Other Nuclear Facilities," (Reference 3.11-21) as endorsed by RG 1.156, "Qualification of Connection Assemblies for Production and Utilization Facilities."

3.11.2.5.5 Line-Mounted Equipment

Guidance in IEC/IEEE 60980-344, "IEEE/IEC International Standard-Nuclear Facilities Equipment Important to Safety-Seismic Qualification" (Reference 3.11-22) identifies that special consideration is required for line-mounted (pipe-supported) equipment regarding seismic qualification as the most critical seismic loading condition that occurs as a result of the piping or duct system. Guidance and further clarification for special considerations for line-mounted equipment is provided in IEEE 382, (Reference 3.11-11). Line mounted equipment also includes active mechanical equipment subjected to ASME QME-1 including the Non-mandatory Appendix QR-A.

3.11.3 References

3.11-1 IEC/IEEE 60780-323, IEC/IEEE International Standard "Nuclear Facilities – Electrical Equipment Important to Safety – Qualification," International Electrotechnical Commission/Institute of Electrical and Electronics Engineers, 2016.

3.11-2 ASME QME-1, "Qualification of Active Mechanical Equipment Used in Nuclear Facilities," American Society of Mechanical Engineers, 2023.

3.11-3 IEC 63147:2017/IEEE 497-2016 "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers, 2017.

3.11-4 EPRI 3002015757, TR-102323, "Guidelines for Electromagnetic Compatibility Testing of Power Plant Equipment," Electric Power Research Institute, Revision 5, 2019.

3.11-5 MIL-STD-461G, "Requirements for the Control of Electromagnetic Interference Characteristics of Subsystems and Equipment," US Department of Defense, 2015.

3.11-6 IEC-62003, "Nuclear Power Plants – Instrumentation, Control, and Electrical Power Systems – Requirements for Electromagnetic Compatibility Testing," International Electrotechnical Commission, 2020.

3.11-7 ASME BPVC-III, "Boiler and Pressure Vessel Code Section III – Rules for Construction of Nuclear Power Plant Components," American Society of Mechanical Engineers, 2021.

3.11-8 IEEE 535-2022, "IEEE Standard for Qualification of Class 1E Vented Lead Acid Storage Batteries for Nuclear Power Generating Stations," Institute of Electrical and Electronic Engineers, 2022.

3.11-9 IEEE 334-2006, “IEEE Standard for Qualifying Continuous Duty Class 1E Motors for Nuclear Power Generating Stations,” Institute of Electrical and Electronic Engineers, 2006.

3.11-10 IEEE 317-2013, “IEEE Standard for Electrical Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations,” Institute of Electrical and Electronic Engineers, 2013.

3.11-11 IEEE 382-2019, “IEEE Standard for Qualification of Safety-Related Actuators for Nuclear Power Generating Stations and Other Nuclear Facilities,” Institute of Electrical and Electronic Engineers, 2019.

3.11-12 IEEE 638-2013, “IEEE Standard for Qualification of Class 1E Transformers for Nuclear Power Generating Stations,” Institute of Electrical and Electronic Engineers, 2013.

3.11-13 IEEE 649-2006, “IEEE Standard for Qualifying Class 1E Motor Control Centers for Nuclear Power Generating Stations,” Institute of Electrical and Electronic Engineers, 2006.

3.11-14 IEEE 650-2006, “IEEE Standard for Qualification of Class 1E Static Battery Chargers and Inverters for Nuclear Power Generating Stations,” Institute of Electrical and Electronic Engineers, 2006.

3.11-15 IEEE 420-2023, “IEEE Standard for the Design and Qualification of Class 1E Control Boards, Panels, and Racks Used in Nuclear Power Generating Stations,” Institute of Electrical and Electronic Engineers, 2023.

3.11-16 IEEE 7-4.3.2-2016, “IEEE Standard Criteria for Programmable Digital Devices in Safety Systems of Nuclear Power Generating Stations,” Institute of Electrical and Electronic Engineers, 2016.

3.11-17 IEEE 383-2023, “IEEE Standard for Qualifying Electric Cables and Splices for Nuclear Facilities,” Institute of Electrical and Electronic Engineers, 2023.

3.11-18 IEEE 384-2018, “IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits,” Institute of Electrical and Electronic Engineers, 2018.

3.11-19 IEEE 1202-2023, “IEEE Standard for Flame-Propagation Testing of Wire and Cable,” Institute of Electrical and Electronic Engineers, 2023.

3.11-20 IEEE 628-2020, “IEEE Standard Criteria for the Design, Installation, and Qualification of Raceway Systems for Class 1E Circuits for Nuclear Power Generating Stations,” Institute of Electrical and Electronic Engineers, 2020.

3.11-21 IEEE 572-2019, “IEEE Standard for Qualification of Class 1E Connection Assemblies for Nuclear Power Generating Stations and Other Nuclear Facilities,” Institute of Electrical and Electronic Engineers, 2019.

3.11-22 IEC/IEEE 60980-344, “IEEE/IEC International Standard-Nuclear Facilities Equipment Important to Safety-Seismic Qualification,” 2020.

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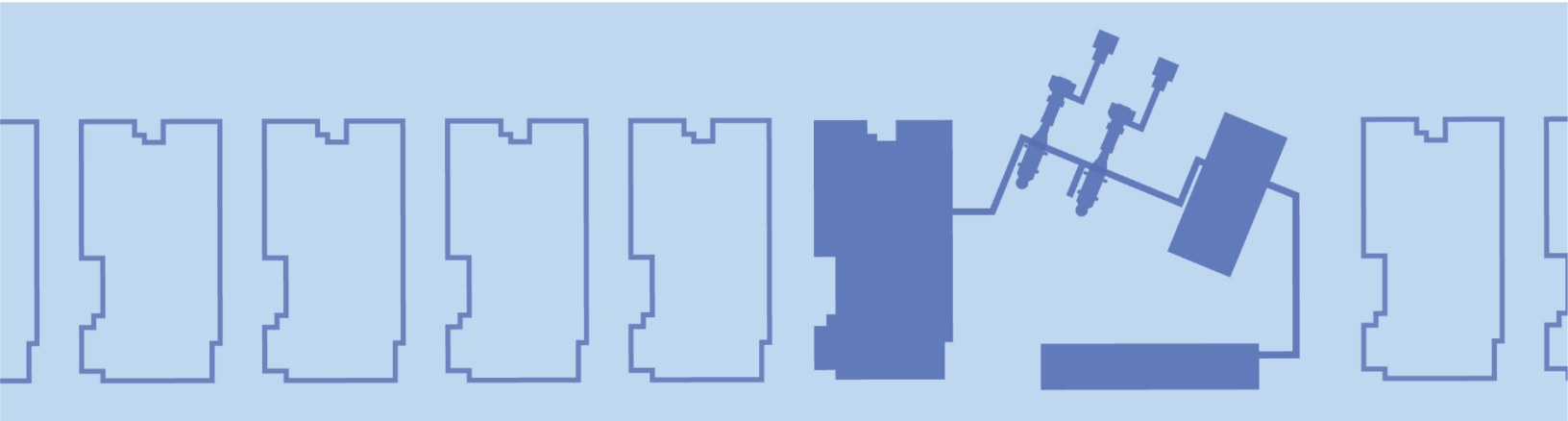
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Non-Proprietary Information

Chapter 3 Design of Structures, Components, Equipment, and Systems

Section 3.12 – ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and Their Associated Supports

Section 3.13 – Threaded Fasteners – ASME Code Class 1, 2, and 3



ACRONYM LIST

Acronym	Explanation
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
BPVC	Boiler and Pressure Vessel Code
CFR	Code of Federal Regulations
CMTR	Certified Material Test Report
CRD	Control Rod Drive
CUF	Cumulative Usage Factor
DBA	Design Basis Accident
DLF	Dynamic Load Factor
FMCRD	Fine Motion Control Rod Drive
GDC	General Design Criteria
GVH	GE Vernova Hitachi Nuclear Energy
ISI	Inservice Inspection
LOCA	Loss-of-Coolant Accident
NPP	Nuclear Power Plant
NPS	Nominal Pipe Size
OBE	Operating-Basis Earthquake
PIE	Postulated Initiating Event
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RG	Regulatory Guide
SBLOCA	Small Break Loss of Coolant Accident
SC	Safety Class
SC1	Safety Class 1
SCN	Non-Safety Class
SRSS	Square-Root-of-the-Sum-of-the-Square
SSCs	Structures, Systems, and Components
SSE	Safe Shutdown Earthquake
USNRC	United States Nuclear Regulatory Commission

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LIST OF FIGURES

None.

3.12 ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and Their Associated Supports

Section 3.12 provides the general design aspects used for Safety Class (SC) and Non-Safety Class (SCN) piping systems, piping components and pipe supports. The design of Safety Class 1 (SC1) piping systems, piping components and pipe supports is based on the rules established under the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section III, Division 1, "Rules for Construction of Nuclear Facility Components," for Class 1, Class 2, and Class 3 nuclear piping, components and supports. For non-SC1 components, ASME B31.1, "Power Piping," (Reference 3.12-5), ASME B31.3, "Process Piping" (Reference 3.12-6), and ASME B31.5, "Refrigeration Piping and Heat Transfer Components," (Reference 3.12-7) are used. Safety class, seismic categories, and quality groups for piping Structures, Systems, and Components (SSCs) are listed in Table 3A-1. The functional, operational, and safety requirements are unique to each system, and the required loading conditions are applied as specified in the specific ASME Code class sections.

Section 3.9 provides the general design principles, criteria, and classification used for design of mechanical systems and components.

The load combinations and acceptance criteria in Table 3.12-1 are applied to the analysis of Class 1 piping including its interfaces with components and penetrations.

The load combinations and acceptance criteria in Table 3.12-2 are applied to the analysis of Class 2 and 3 piping systems and containment penetrations.

The load combinations and acceptance criteria in Table 3.12-3 are applied to the analysis of non-nuclear (ASME B31.1) piping systems and components.

The load combinations and acceptance criteria in Table 3.12-4 are applied to the analysis of non-nuclear (ASME B31.3) piping systems and components.

The load combinations and acceptance criteria in Table 3.12-5 are applied to the analysis of ASME Class 1, 2, and 3 snubber type supports.

The load combinations and acceptance criteria in Table 3.12-6 are applied to the analysis of ASME Class 1, 2, and 3 rigid type supports.

The load combinations and acceptance criteria in Table 3.12-7 are applied to the analysis of ASME Class 1 piping connected to RPV nozzles.

The load combinations and acceptance criteria in Table 3.12-8 are applied to the analysis of ASME Class 2 piping connected to RPV nozzles.

The load combinations and acceptance criteria in Table 3.12-9 are applied to the analysis of ASME Class 1, 2, and 3 pipe-mounted valves.

The load combinations and acceptance criteria in Table 3.12-10 are applied to the analysis of ASME B31.1 and ASME B31.3 Linear Type Supports (Anchor, Guide, Struts).

The BWRX-300 design meets the relevant requirements of the following regulations:

The BWRX-300 design discussed in Section 3.12 complies with the following General Design Criteria (GDC), which have plant-wide applicability, as discussed in Section 3.1:

GDC 1 – Quality standards and records

GDC 2 – Design bases for protection against natural phenomena

GDC 4 – Environmental and dynamic design bases

The following GDC and 10 Code of Federal Regulations (CFR) 50 Appendix S have aspects specific to Section 3.12 and compliance is addressed as follows:

10 CFR Part 50, Appendix S – as it relates to design transients and resulting load combinations for SC1 piping and pipe supports necessary to withstand the effects of earthquakes as discussed in Subsection 3.7.3 combined with the effects of normal or accident conditions including a Loss-of-Coolant Accident (LOCA) and dynamic effects as defined in Table 3.12-1, Table 3.12-2, Table 3.12-3, and Table 3.12-4.

GDC 14 – Reactor coolant pressure boundary, as it relates to the Reactor Coolant Pressure Boundary (RCPB) of the piping systems being designed, fabricated, erected, and tested in accordance with the requirements in 10 CFR 50.55a, “Codes and Standards,” to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. The design involves meeting the acceptance criteria utilizing load combinations in Table 3.12-1 and Table 3.12-2, including transient plant duty cycles in Table 3.9-2 for the 60-year life period to assure that minimal abnormal leakage occurs.

GDC 15 – Reactor coolant system design, as it relates to the reactor coolant system and associated auxiliary, control, and protection systems being designed with sufficient margin to ensure that the design condition of the RCPB are not exceeded during any condition of normal operation, including Anticipated Operational Occurrences (AOOs). Subsections 3.9.1 through 3.9.6 provide design requirements for safety class and risk informed mechanical components and piping that meet the margin and design condition requirements, therefore complying with GDC 15.

Additional Information on GDC compliance is found in Section 3.1.

3.12.1 Piping Analysis Methods

ASME Class 1 piping design conforms to the requirements of ASME BPVC, Section III, Subsection NB (Reference 3.12-1) for both piping and piping components. The pipe supports attached to the ASME Class 1 piping meet the appropriate requirements of ASME BPVC, Section III, Subsection NF (Reference 3.12-2).

ASME Class 2 and 3 piping design conforms to the requirements of ASME BPVC, Section III, Subsection NCD (Reference 3.12-3) for both piping and piping components. The pipe supports attached to the ASME Class 2 and 3 piping meets the appropriate requirements of ASME BPVC, Section III, Subsection NF.

The containment penetration sleeve of ASME Class 2 piping is an anchor for the piping. The sleeve of the containment structure penetrations meets the requirements of ASME BPVC, Section III, Subsection NE (Reference 3.12-4). Section 3.8 and Figure 3.8-5 provide details of penetration jurisdictional boundaries.

Power piping is designed per ASME B31.1, “Power Piping,” (Reference 3.12-5) and process piping is designed per ASME B31.3, “Process Piping,” (Reference 3.12-6). Descriptions of systems that contain ASME B31.1 or ASME B31.3 piping and components including their functions are described in the system chapters.

Refrigerant piping meets the requirements of ASME B31.5, “Refrigeration Piping and Heat Transfer Components,” (Reference 3.12-7) as described in Subsection 9.2.7.

Table 3.12-1 and Table 3.12-2 provide the specific load combinations and acceptance criteria for ASME Class 1, 2, and 3 piping including its interfaces with components and penetrations.

Table 3.12-3 and Table 3.12-4 provide the specific load combinations and acceptance criteria for ASME B31.1 and ASME B31.3 piping.

Supports and their attachments for ASME Class 1, 2, and 3 piping are designed in accordance with Subsection NF up to the interface of the building structure, with jurisdictional boundaries as defined by Subsection NF. The building structure component supports (connecting the Subsection NF support boundary component to the existing building structure) are designed as specified in Subsection 3.8.4.2.

3.12.1.1 Experimental Stress Analysis Methods

Standard flanges used in SC1 piping systems are in accordance with ASME B16.5, “Pipe Flanges and Flanged Fittings: Nominal Pipe Size (NPS) 1/2 Through NPS 24, Metric/Inch Standard,” (Reference 3.12-8) with high-strength bolting per NCD-3658.3.

In cases where a fitting or joint design is used that is not covered by Table NB-3681(a)-1 or NCD-3673.2(b)-1, (or if it is desired to determine less conservative values), stress intensification factors and flexibility factors are established using experimental or analytical data.

Determination of non-standard stress intensification factors and flexibility factors is in accordance with the rules of NB-3672.7.

3.12.1.2 Modal Response Spectrum Method

The seismic response of the piping system is determined by performing a modal analysis by either the Response Spectrum Method or Time History Method. The procedure for multi-support excitation is followed with both methods.

The spectral peak shifting procedures of ASME BPVC, Section III, Nonmandatory Appendix N, (Reference 3.12-9) are used for design of systems with closely spaced structural modes and artificially broadened spectra input.

Nodal points are selected to coincide with the locations of concentrated masses, locations of significant geometry changes, or locations of supports. Node points need to be sufficiently close to ensure higher modes to be captured.

The support movement of multiple-supported piping and equipment systems subject to seismic and other dynamic load excitations is considered in design verification. As a conservative approach, the maximum displacement of each support point can be computed by either time history analysis, or response spectrum analysis. The computed individual maximum support displacements are then imposed on the supported piping and equipment systems in the most limiting, or severe, directional combination for the piping and equipment design verification consideration.

Refer to Subsection 3.7.3.1.4 and Section 3.10 for discussions on modal response methods of piping.

3.12.1.3 Response Spectra Method – Independent Support Motion Method

The Independent Support Motion or Multiple Response Spectra Method is used when piping systems are supported by multiple support structures or at multiple levels within a structure. In this method of analysis, supports are divided into support groups with different seismic excitation applied to each group. A single set of response spectra is applied to all supports of each group, but different sets of response spectra are applied to different groups. Typically, a support group is made up of supports attached to the same structure, floor, or portion of a floor.

In the Independent Support Motion Method, dynamic analysis is performed by applying the appropriate response spectra at one support group at a time, with no motion applied to the other supports. The process is repeated for each support group.

The combinations of modal responses and spatial components for systems analyzed using Independent Support Motion is performed consistent with NUREG-1061, “Evaluation of Other Loads and Load Combinations.” Inertial responses from multiple groups in the same direction are combined by the Absolute Summation Method. Modal and directional responses are combined by Square-Root-of-the-Sum-of-Squares (SRSS) method without considering closely spaced frequencies. The responses due to relative displacements at the support points are combined with the inertial responses by the SRSS method.

If the seismic inputs are known to be in-phase, the combination of responses is accomplished by the Absolute Summation Method.

See Subsection 3.7.3.1.4 for the Seismic Analysis of Piping Systems.

3.12.1.4 Time History Method

Subsection 3.7.2.6 describes the Three Components of Design Ground Motion analyses which conform to the acceptance criteria described in Regulatory Guide (RG) 1.92, “Combining Modal Responses and Spatial Components in Seismic Response Analysis.” The time history modal superposition, time history direct integration, and response spectrum modal superposition methods can all be used in Multi-Support Excitation analysis. The modal and spatial combinations in the time history method are described in RG 1.92.

3.12.1.5 Inelastic Analysis Method

Piping system stresses are calculated on an elastic basis for each service level and evaluated in accordance with NB-3600 (Class 1) and NCD-3600 (Class 2 and 3) as described in Subsection 3.9.1.4.12. Inelastic analysis methods are addressed in Subsection 3.9.1.4.13. Mathematical models for piping systems reflect the static and dynamic characteristics of the system. In general, a piping system consists of pipe elements such as straight sections, fittings, bends, and valves. The piping system is supported and restrained by hangers, anchors, pipe guides, struts, box frames, snubbers, and equipment nozzles.

Large equipment, such as vessels and pumps, is typically decoupled from piping analysis at the nozzle connection.

The mass of the pipe and inline equipment is lumped at corresponding nodal locations and connected by weightless elastic beam elements to reflect the physical properties of the corresponding piping segment.

Inelastic analysis method is not used in the BWRX-300 piping design and analysis except for piping whip restraints in Subsection 3.6.2. For Class 1 piping of which shakedown is not established by meeting Equation (10) of NB-3600, the simplified elastic-plastic discontinuity analysis is performed through Equation (12) to Equation (14) of NB-3600 to limit the accumulated plastic strains. The system piping analysis model includes the valves and nozzle connections to pumps and other large equipment. Level D loads acting on the components are determined through the system piping analysis.

3.12.1.6 Non-Seismic and Seismic Category II

Subsection 3.7.3.8 describes the design of the interaction of non-Seismic Category I subsystems with SSCs that are categorized as Seismic Category I.

In certain instances, Seismic Category II piping is connected to Seismic Category I piping at locations other than a piece of equipment which, for purposes of analysis, could be represented as an anchor. The transition points typically occur at Seismic Category I valves, which may or may not be physically anchored. Because a dynamic analysis is modeled from pipe anchor point to anchor point, two options exist:

- Specify and design a structural anchor at the Seismic Category I valve and analyze the Seismic Category I subsystem
- Analyze the subsystem from the anchor point in the Seismic Category I subsystem through the valve to either the first anchor point in the Seismic Category II subsystem, or for a distance such that there are at least two seismic restraints in each of the three orthogonal directions

For either option, seismic loads are applied on anchors and rigid supports in the model.

Where small Seismic Category II piping is directly attached to Seismic Category I piping, it may be decoupled from Seismic Category I piping following guidelines in Subsection 3.12.2.4.

3.12.1.7 Small Bore Piping Method

Small bore piping is defined as piping 50 mm and less NPS. Decoupling of small branch piping from main run piping follows decoupling criteria in Subsection 3.12.2.4. Seismic analysis of small-bore piping follows dynamic analysis methods per Subsection 3.12.1.2, or the equivalent static coefficient method if justified.

3.12.1.8 Seismic Category I Buried Piping

The BWRX-300 design does not include Seismic Category I buried piping.

3.12.2 Piping Modeling Techniques

3.12.2.1 Computer Codes

The computer programs used in the mechanical system and component analyses of the SC1 components are described in Appendix 3J and Subsection 3.9.1.2.

3.12.2.2 Dynamic Piping Model

Subsections 3.7.3.1.4 and 3.9.3 describe the analyses that are used to confirm that the Seismic Category I subsystems, including piping and supports, function during and after an earthquake.

3.12.2.3 Piping Benchmark Program

Refer to Subsection 3.12.2.1 and Appendix 3J for discussion of how the computer codes are benchmarked with the appropriate United States Nuclear Regulatory Commission (USNRC) benchmarks in accordance with NEDO-11209-A, “GE Hitachi Nuclear Energy Quality Assurance Program Description,” (Reference 3.12-10).

3.12.2.4 Decoupling Criteria

Decoupling is a method of dividing or separating analysis models based on pipe size/moment of inertia. Branch lines with a run to branch moment of inertia ratio of 25 to 1 or greater are decoupled. The decoupling criteria described in Subsection 3.7.2.3.2 includes Seismic Category I subsystems which includes piping and supports. Piping design specifications use an alternative approach stated in Welding Research Council Bulletin WRC-300 Part II, “Technical Position on Damping and on Industry Practice,” (Reference 3.12-11) to decouple the subsystem.

3.12.3 Piping Stress Analysis Criteria

Piping stress criteria for ASME Class 1, 2, and 3 piping are discussed in Section 3.9, Table 3.12-1, and Table 3.12-2.

3.12.3.1 Seismic Input

Seismic Input criteria for In-Structure Response Spectra for ASME Class 1, 2, and 3 piping are discussed in Subsection 3.7.2.5.1.

3.12.3.2 Design Transients

Design transients and dynamic loading for ASME Class 1, 2, and 3 piping are covered in Subsections 3.9.1.1 and 3.9.3.1.

3.12.3.3 Loadings and Load Combinations

Table 3.12-1, Table 3.12-2, Table 3.12-3, and Table 3.12-4 provide the specific load combinations and acceptance criteria for ASME Class 1, 2, and 3 and non-nuclear class piping systems, structures, and components.

3.12.3.4 Damping Ratios

Damping values for seismically qualified ASME piping are discussed in Subsections 3.7.1.2 and 3.7.3.5.2 and provided in Table 3.7-7 for seismic damping values for piping, equipment, and equipment supports.

Floor response spectra include 15% peak broadening to account for possible variations in structural frequencies.

Floor response spectra used for the qualification of Class 1, 2, and 3 piping for dynamic loads with Level B service limits use a 3% critical damping ratio for piping greater than or equal to 300 mm NPS, and a 2% critical damping ratio for piping less than 300 mm NPS.

Floor response spectra used for the qualification of Class 1, 2, and 3 piping for dynamic loads with Level C or Level D service limits use a 4% critical damping ratio for all pipe sizes.

3.12.3.5 Combination of Modal Responses

The modal and spatial combinations in the time history method are described in RG 1.92.

3.12.3.6 High-Frequency Modes

Cutoff frequency and effect of missing mass to be considered for dynamic analysis are determined in accordance with Subsection 3.7.3.4.

The effect of the missing mass associated with the cutoff frequency is properly considered in the design of the system and the support of the structures and components.

The additional constraint forces associated with the missing mass are obtained by applying an additional force at the center of mass of the equipment. This force is conservatively calculated as the product of the total missing mass and the maximum input acceleration spectrum values beyond the cutoff frequency. A less conservative method may be used when justified.

3.12.3.7 Fatigue Evaluation for ASME Class 1 Piping

The Fatigue Evaluation for ASME Class 1 piping conforms to ASME BPVC, Section III, Subsection NB requirements as described in Subsection 3.12.3.19 and Table 3.12-1, which includes the fatigue usage and stress limits and conforms to NB-3653. The Cumulative Usage Factor (CUF) is calculated in accordance with NB-3653.5 and RG 1.207, “Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors.”

The number of earthquake events of SSCs for seismic fatigue effects is based on the dynamic response resulting from the occurrence of only one Safe Shutdown Earthquake (SSE) during the operating lifetime of the plant.

3.12.3.8 Fatigue Evaluation of ASME Class 2 and 3 Piping

The Fatigue Evaluation for ASME Class 2 and 3 piping conforms to ASME BPVC, Section III, Subsection NCD requirements.

In cases where a fitting or joint design is used which is not covered by Table NCD-3673.2(b)-1 (or if it is desired to determine less conservative values), stress intensification factors and flexibility factors are established using experimental or analytical data.

Determination of non-standard stress intensification factors and flexibility factors is in accordance with the rules of NCD-3673.2.

Expansion and flexible hoses are designed in accordance with the requirements of NCD-3649. Expansion joints and flexible hoses are subject to limitations in displacement during operation. The component manufacturer provides these displacement limits. The displacement values obtained from the stress analysis are reviewed to ensure that the specified displacement limits have not been exceeded.

3.12.3.9 Thermal Oscillations in Piping Connected to the Reactor Coolant System

As described in USNRC Bulletin 88-08, thermal fatigue of unisolable piping connected to the Reactor Coolant System (RCS) can occur when the connected piping is isolated by a leaking block valve, the pressure upstream from the block valve is higher than RCS pressure, and the temperature upstream is significantly cooler than RCS temperature. An unrecognized phenomenon and possibly unanalyzed condition may exist for those reactors that are subjected to these conditions. Under these conditions, thermal fatigue of the unisolable piping can result in crack initiation. If these conditions are identified, a program meeting the criteria in Bulletin 88-08, which provides continued assurance that piping will not be subjected to unacceptable thermal stresses, will be developed.

3.12.3.10 Thermal Stratification

Transient loading due to hot and cold fluid not mixing at low flow conditions is considered in the design of the piping. Stresses due to the effects of thermal stratification are evaluated using ASME BPVC, Section III, Nonmandatory Appendix JJ. On run/branch connections where there is a closed valve and the resulting “dead leg” temperature tends toward ambient, the temperature distribution in the run/branch line is considered and properly included in the thermal expansion analysis.

The stresses and fatigue associated with thermal stratification and thermal striping are considered in the piping analyses that demonstrate compliance within ASME Class 2 and 3 code limits. The global effect of thermal stratification causes the bending of the piping system due to the temperature distribution in the element cross section. Consequently, this generates extra thermal stress in the load combination as defined in Table 3.12-2.

3.12.3.11 Safety Relief Valve Design, Installation, and Testing

No ASME Class 1, 2, or 3 safety relief valves are in the BWRX-300 design. Subsection 3.9.3.2 describes the installation of BWRX-300 overpressure protection pressure relief devices.

If ASME Class 1, 2, and 3 safety relief valves are required, design follows the criteria for pressure relief installations specified in ASME BPVC, Section III, Nonmandatory Appendix O. In addition, the following criteria are evaluated:

- Where more than one valve is installed on the same pipe run, the sequence of valve openings to be assumed in analyzing for the stress at any piping location is that sequence which is estimated to induce the maximum instantaneous value of stress at that location
- Stresses are evaluated, and applicable stress limits are satisfied for all components of the pipe run and connecting systems and the pressure relief valve station, including supports and all connecting welds between these components

- In meeting the stress limit requirements, the contribution from the reaction force and the moments resulting from that force include the effects of a Dynamic Load Factor (DLF) or use the maximum instantaneous values of forces and moments for that location as determined by the dynamic hydraulic/structural system analysis. This requirement is satisfied in demonstrating satisfaction of design limits at all locations of the pipe run and the pressure relief valve for Class 1, 2, and 3 piping. A DLF of 2.0 may be used in lieu of a dynamic analysis to determine the DLF.

In the case that a Class 1 piping system is designed without a safety relief valve, the overpressure protection design follows NB-7120(c).

In the case that a Class 2 or 3 piping system is designed without a safety relief valve, the overpressure protection design follows NCD-7120(c).

3.12.3.12 Functional Capability

Functional capability for ASME Service Level D load conditions is discussed in Subsection 3.9.1.4 and is applicable to SC1 SSCs.

If the requirements of NUREG-1367, “Functional Capability of Piping Systems,” are applicable, functional capability of ASME Class 1, 2, and 3 piping systems essential for the safe shutdown under postulated events is ensured provided the following requirements are met:

- The conditions of the equation in NB-3652, Equation (9) are met for Service Level C and D loads, (Class 1)
- The conditions of the equation in NCD-3653.1(a)(9a) are met for Service Level C and D loads, (Class 2 and 3)
- Dynamic moments are calculated using an elastic response spectrum analysis with +/- 15% peak broadening and with no more than 5% damping
- Steady state (weight) stress is less than or equal to 25% of yield stress
- The ratio of pipe diameter to wall thickness is less than or equal to 50
- External pressure is less than or equal to internal pressure

3.12.3.13 Combination of Inertial and Seismic Anchor Motion Effects

Combination of inertial and seismic anchor motion effects is discussed in Subsections 3.7.3.9 and 3.7.3.1.4.

3.12.3.14 Operating Basis Earthquake as a Design Load

The BWRX-300 design considers Operating Basis Earthquake (OBE) and site operating earthquake loads as 1/3 of the site SSE. In accordance with Appendix S to 10 CFR 50, design load combinations that consider OBE and site operating earthquake loads are not required, except for the design of metal containment components where the OBE loads are considered for post-flooding condition and cyclic loading considerations, as noted in Subsection 3.7.1.1.

3.12.3.15 Welded Attachments

The piping system and pipe supports are designed and located to minimize the necessity for welded attachments to the pipe. Where welded attachments exist, local stress evaluations are performed.

The code requirements are met with the local stress added to the regular piping stress.

The local stresses are evaluated per ASME BPVC, Section III, Nonmandatory Appendix Y (Reference 3.12-9).

Use of stiff pipe clamps may result in local pipe stresses. Stiff pipe clamps described in Issue 89 of NUREG-0933, "Resolution of Generic Safety Issues," are not used.

3.12.3.16 Modal Damping for Composite Structures

Modal Damping for Composite Structures is defined in Subsection 3.7.1.2.

Modal damping is addressed in Tables 3.7-5 and 3.7-6.

3.12.3.17 Temperature for Thermal Analyses

The analysis of thermal expansion includes all thermal operating modes, environmental conditions, cold water modes, and thermal attenuation.

Sufficient thermal expansion cases are established to account for various operating conditions to determine the maximum range of thermal expansion stresses.

The thermal stress-free state for the piping systems is defined as a temperature of 21°C (70°F) unless a basis is provided to use a higher temperature in accordance with ASME BPVC, Section II.

Applicable equipment nozzle movements are considered for their effect with respect to each operating mode. Radial and vertical movements of the RPV nozzles are considered in the thermal analysis.

For Class 1 piping systems, the moduli of elasticity are listed in ASME BPVC, Section II, "Materials," Part D, Subpart 2, Table TM (Reference 3.12-12).

Flexibility calculations of the forces and moments in a piping system due to thermal expansion and end motions are based on the hot modulus of elasticity E_h. However, the expansion stresses computed from the forces and moments are multiplied by the ratio of the cold modulus of elasticity E_c and the hot modulus, E_c/E_h in accordance with NB-3672.5.

For Class 2 and 3 piping systems, the moduli of elasticity are listed in the applicable tables of ASME BPVC, Section II.

For non-nuclear (ASME B31.1) piping systems, the moduli of elasticity are listed in ASME B31.1, Appendix C.

For non-nuclear (ASME B31.3) piping systems, the moduli of elasticity are listed in ASME B31.3, Appendix C.

Calculation of expansion stress is based on the modulus of elasticity at room temperature.

3.12.3.18 Intersystem Loss-of-Coolant Accident

The design includes consideration of high energy and moderate energy fluid system piping located inside and outside of containment. Design bases and measures used to protect these SSCs, referred to in the following subsections as essential SSCs, are discussed in Subsections 3.6.1 and 3.6.2.

To the extent practical, low-pressure systems (e.g., systems and subsystems connected to the RCS which extend outside the primary containment boundary and including valve stem seals, pump seals, HX tubes) are designed to withstand full RCS pressure. For those interfacing systems or subsystems which do not meet the full RCS Ultimate Rupture Strength requirement, the degree and quality of isolation or reduced severity of the potential pressure challenges are low enough to preclude an intersystem LOCA.

3.12.3.19 Effects of Environment on Fatigue Design

Refer to Subsection 3.9.3.1 which describes the design bases which includes the fatigue analysis.

Class 1 piping is evaluated for the effects of fatigue as a result of pressure and thermal transients and other cyclic events including earthquakes.

Without considering exemption for the seismic fatigue analysis, the seismic fatigue usage factor is determined from Equations 11 and 14 in NB-3653.2 and NB-3653.6, respectively. This fatigue factor is added to the fatigue usage factor associated with Levels A and B load ranges to obtain the CUF factor with the minimum number of cycles for the maximum peak response levels taken as 25 Cycles for floor-supported piping and 20 Cycles for ground-supported systems.

The CUF is calculated in accordance with NB-3653.5 combined with the environmental effects per RG 1.207.

3.12.4 Piping Support Design

Pipe support types include anchor, spring hanger, snubber, strut, and supporting steel frame.

Anchor type supports provide six-way restraints to a pipe. Spring hanger type supports are used to carry the dead weight of a pipe. Snubber type supports restrain a pipe in a seismic event but allow thermal displacement during normal operation. Strut type supports restrain pipe movements for all conditions. Supporting frames are usually used to support a standard support item and/or to restrain a pipe in a single or in multiple directions.

Pipe Whip Restraints are not in the jurisdictional boundary of ASME BPVC, Section III, Subsection NF, ASME B31.1, or ASME B31.3 and thus are not considered as pipe supports and are not within the scope of this section.

Pipe supports are designed and qualified to satisfy stiffness values used in the piping analysis. For struts and snubbers, the stiffness to consider is the combined stiffness of strut, snubber, pipe clamp, and piping support steel.

Steamline supports are designed for water-filled line loads under static loading conditions that may be encountered in plant operations. In addition, provisions are made for conveniently supporting the deadweight loads imposed during hydrostatic test of the Main Steam piping.

The stiffness of the building steel/structure (i.e., beyond the Subsection NF jurisdictional boundary) is not considered in pipe support overall stiffness. Response spectra input to the piping system include flexibility of the building structure. When attachment to a building structure is not possible, any intermediate structures are included in the analysis of the pipe support.

For multi-supported systems subjected to Independent Support Motion, the Independent Support Motion method of analysis can also be performed using the time history method.

3.12.4.1 Pipe Support Applicable Codes

The pipe supports attached to the ASME Class 1 piping meet the appropriate requirements of ASME BPVC, Section III, Subsection NF.

The pipe supports attached to the ASME Class 2 and 3 piping meet the appropriate requirements of ASME BPVC, Section III, Subsection NF.

The pipe supports attached to the non-nuclear (ASME B31.1) piping meet the requirements of Sections 120 and 121 of ASME B31.1.

The pipe supports attached to the non-nuclear (ASME B31.3) piping meet the requirements of Section 321 of ASME B31.3.

3.12.4.2 Pipe Support Jurisdictional Boundaries

Support design jurisdictional boundaries at interfaces with piping, structure, or intervening elements are defined in NF-1130. If piping supports transmit loads to surface-mounted baseplates as discussed in NF-1132(d), the baseplates are within the building structure jurisdiction. Refer to Subsection 3.9.3.15.

3.12.4.3 Pipe Support Loads and Load Combinations

The design loading is established considering plant and system operating conditions anticipated or postulated to occur during the intended service life of the component, system, or structure.

The criteria are described in Subsection 3.9.3.

Component and system piping loads and load combinations are provided in Table 3.12-1, Table 3.12-2, Table 3.12-3, Table 3.12-4, Table 3.12-5, and Table 3.12-6.

3.12.4.4 Pipe Support Baseplate and Anchor Bolt Design

Aspects of the anchor bolt design, including baseplate flexibility and factors of safety are used in the development of anchor bolt loads, as addressed in USNRC Bulletin 79-02, Rev. 2, “Pipe Support Base Plate Design Using Concrete Expansion Anchor Bolts,” and per NF-1132(d).

3.12.4.5 Use of Energy Absorbers and Limit Stops

ASME BPVC, Section III component supports are designed, manufactured, installed, and tested in accordance with applicable codes and standards. Supports include hangers, snubbers, struts, spring hangers, frames, energy absorbers and limit stops. Pipe whip restraints are not considered as pipe supports. If energy absorbers and limit stops are required for use in the piping system, the design methodology and acceptance criteria are defined and approved. The evaluation consists of iterative response spectra analyses of the piping and support system.

3.12.4.6 Use of Snubbers

Snubber type supports are used when restraint is needed for dynamic type loading such as seismic or relief valve discharge at a location in the pipe that has large thermal movement. Snubbers are only active during dynamic events. Snubbers allow free thermal movement during normal operation and therefore are not loaded during normal operation. Due to maintenance concerns, the use of snubbers is minimized or avoided.

Snubbers are modeled with an equivalent stiffness based on dynamic tests or on data provided from the vendor. The permissible loads that the piping system can place on snubbers is defined in ASME BPVC, Section III, Subsection NF.

The load combinations and acceptance criteria applied to the analysis of snubber type supports.

3.12.4.7 Pipe Support Stiffness

Rigid supports are sized and qualified for a specified deflection limit for all conditions in the restraint directions. Where the support deflection cannot meet the minimum criteria, the actual support stiffness is developed and either justified as sufficiently rigid or included in the piping analysis. In the event the actual stiffness is included, all supports within the stress case boundary are evaluated with actual stiffness values.

3.12.4.8 Seismic Self-Weight Excitation

For the component type supports (e.g., snubbers, struts, spring hangers), support component weights may be neglected if it is demonstrated that the support is dynamically rigid or that one half of the support mass is less than 10% of the mass of the straight pipe segment of the span at the support location, to preclude amplification. Otherwise, piping supports are evaluated to

include the effect of self-weight excitation on the support structure and anchorage, in detail, along with piping analyzed loads where this effect is significant.

Subsection 3.7.3.1 discusses the seismic analysis methodology used to analyze Seismic Category I components, piping, and their supports to function as needed during and after an earthquake.

3.12.4.9 Design of Supplementary Steel

The design of structural steel for use as frame type pipe supports are linear supports as defined in ASME Code, Section III, Subsection NF.

3.12.4.10 Consideration of Friction Forces

The friction force on the pipe during axial movement is considered in the design of the support.

3.12.4.11 Pipe Support Gaps and Clearances

Small gaps are provided for frame type supports allowing for radial thermal expansion of the pipe without imposing any thermal binding and to restrain the pipe.

3.12.4.12 Instrumentation Line Support Criteria

The instrumentation line supports attached to the ASME Class 1 piping meet the appropriate requirements of ASME BPVC, Section III, Subsection NF.

Subsection 3.9.1.4.5 discusses the instrumentation line support criteria for faulted conditions. Design loads and load combinations for fixed mechanical equipment are provided in Table 3.9-3 which also applies to Seismic Category I instrumentation and electrical equipment.

3.12.4.13 Pipe Deflection Limits

Rigid supports are sized and qualified for a specified deflection limit for all conditions in the restraint directions. Where the support deflection cannot meet the minimum criteria, the actual support stiffness is developed and either justified as sufficiently rigid or included in the piping analysis. In the event the actual stiffness is included, all supports within the stress case boundary are evaluated with actual stiffness values.

3.12.4.14 Clam-Induced Local Pipe Stress Evaluation

The piping system and pipe supports are designed and located to minimize the necessity for welded attachments to the pipe. Where welded attachments exist, local stress evaluations are performed. The code requirements are met with the local stress added to the regular piping stress.

The local stresses are evaluated per ASME BPVC, Section III, Nonmandatory Appendix Y. The use of stiff pipe clamps may result in local pipe stresses. Stiff pipe clamps described in Issue 89 of NUREG-0933 are not used.

3.12.5 References

3.12-1 ASME BPVC, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” Subsection NB, “Class 1 Components,” American Society of Mechanical Engineers, 2021.

3.12-2 ASME BPVC, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” Subsection NF, “Supports,” American Society of Mechanical Engineers, 2021.

3.12-3 ASME BPVC, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” Subsection NCD, “Class 2 and Class 3 Components,” American Society of Mechanical Engineers, 2021.

3.12-4 ASME BPVC, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” Subsection NE, “Class MC Components,” American Society of Mechanical Engineers, 2021.

3.12-5 ASME B31.1, “Power Piping,” American Society of Mechanical Engineers, 2022.

3.12-6 ASME B31.3, “Process Piping,” American Society of Mechanical Engineers, 2022.

3.12-7 ASME B31.5, “Refrigeration Piping and Heat Transfer Components,” American Society of Mechanical Engineers, 2019.

3.12-8 ASME B16.5, “Pipe Flanges and Flanged Fittings: NPS 1/2 Through NPS 24, Metric/Inch Standard,” American Society of Mechanical Engineers, 2020.

3.12-9 ASME BPVC, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” Appendices, American Society of Mechanical Engineers, 2021.

3.12-10 NEDO-11209-A, “GE Hitachi Nuclear Energy Quality Assurance Program Description,” GE-Hitachi Nuclear Energy Americas, LLC, Revision 17, December 2022.

3.12-11 WRC Bulletin 300, “Technical Position on Damping and on Industry Practice,” Welding Research Council, Inc. 1984.

3.12-12 ASME BPVC, Section II, “Materials,” American Society of Mechanical Engineers, 2021.

Table 3.12-1: Load Combinations and Acceptance Criteria for ASME Class 1 Piping

Condition	Load Combination for all Terms	Acceptance Criteria	ASME Code Reference
Design	Design Pressure + Weight + Sustained Mechanical Load (Note 1)	Eq 9 ≤ 1.5 S_m	NB-3652
Service Levels A and B (Note 2)	Peak Pressure + Weight + Sustained Mechanical Load + Thermal Expansion + AOO (Note 3) + SSE (Note 4)	Eq 10 ≤ 3.0 S_m or Eq 12 & 13 ≤ 3.0 S_m and Fatigue CUF < 1.0 and Range ΔT_1 ≤ Allowable (Note 5)	NB-3653.1 NB-3653.6 NB-3653.1 to NB-3653.6 NB-3653.7
Service Level B	Peak Pressure + Weight + AOO	Eq 9 ≤ 1.8 S_m, not greater than 1.5 S_y (Note 6) and Peak Pressure ≤ 1.1P_a	NB-3654.2(a) NB-3654.1
Service Level C	Peak Pressure + Weight + Design Basis Accident (DBA) Where DBA can be one of items as listed below: 1. Emergency Level Transient 2. Anticipated Transient Without Scram (ATWS) 3. Small Break Loss of Coolant Accident (SBLOCA)	Eq 9 ≤ 2.25 S_m, not greater than 1.8 S_y (Note 7) (Note 8) and Peak Pressure ≤ 1.5 P_a	NB-3655.2(a) NB-3655.1
Service Level D	Peak Pressure + Weight + DBA Where DBA can be one of items as listed below: 1. SBLOCA + SSE 2. Large Break LOCA + SSE 3. Faulted Level Transient	Eq 9 ≤ 3.0 S_m, not greater than 2.0 S_y (Note 9) (Note 10) and Peak Pressure ≤ 2.0 P_a (Note 10) and Stress caused by Seismic Anchor Movement due to SSE satisfies NB-3656(b)(4)	NB-3656(a)(2) NB-3656(a)(1) NB-3656(b)(4)

Notes:

(1) Effect of sustained mechanical load is considered together with pressure and weight. In normal operation load, fluid flow (including Flow-Induced Vibration when applicable), thermal and fluid reaction forces, and the like, are considered together with pressure and weight.

(2) Load combination is required to set up individual load set. Acceptance criteria for stress range including Equations 10, 12, and 13 are checked for each pair of load sets. One of the load sets included is that corresponding to zero pressure, zero moment, and room temperature.

(3) An event category assigned to a Postulated Initiating Event (PIE) or event sequence with an expected frequency of 1/yr to 1.0 E-02/yr that represents a deviation from normal operation but does not evolve into a postulated event..

(4) SSE is only used for piping fatigue analysis. For piping fatigue analysis, the full range of SSE maximum stress cycles are considered. Use 25 Cycles for floor-supported piping and 20 Cycles for ground-supported systems. SSE includes both Inertia Effect and Anchor Displacement Loads, which are combined using SRSS method.

(5) NB-3653.7 provides a variable limit of the linear thru-wall gradient, based on material properties, to prevent progressive deformation due to thermal stress ratcheting.

(6) This criterion is for Service Level B loadings that do not include reversing dynamic loads (NB-3622.2) or have reversing dynamic loads combined with non-reversing dynamic loads (NB-3622.4). For Level B loadings that include reversing dynamic loads that are not required to be combined with non-reversing dynamic loads, the above criterion for Levels A and B loadings applies.

(7) If the anchor motion due to reversing dynamic loads is excluded from NB-3654, it is assessed through the requirements of NB-3656(b)(4) using 70% of the allowable.

(8) As an alternative requirement for Level C loadings that include reversing dynamic loads that are not required to be combined with non-reversing dynamic loads, the requirements of NB-3656(b) are satisfied using the allowable stress in NB-3656(b)(2), 70% of the allowable stress in NB-3656(b)(3), and 70% of the allowable loads in NB-3656(b)(4). If the anchor motion due to reversing dynamic loads is excluded from Equation (9), it is assessed through the requirements of NB-3656(b)(4) using 70% of the allowable.

(9) If the anchor motion due to reversing dynamic loads is excluded from NB-3654, it is assessed through the requirements of NB-3656(b)(4).

(10) See alternative requirements in NB-3656(b).

(11) Nomenclature:

- AOO Anticipated Operational Occurrences
- DBA Design Basis Accident
- P_a Calculated maximum allowable internal pressure for a straight pipe that is at least equal the Design Pressure
- S_A Allowable stress range for expansion stresses = f (1.25 S_c+ 0.25 S_h), where f = stress range reduction factor for cyclic conditions
- S_c Basis material allowable stress at minimum (cold) metal temperature
- S_h Basis material allowable stress at maximum (hot) metal temperature
- S_m For Eq 9, allowable design stress intensity value at a temperature consistent with the loading under consideration; for Eq 10, 12 and 13, average of the allowable stress intensity value for the highest and the lowest temperature of the metal during the transient.
- S_y Yield Strength
- SRSS Square-root-of-the-sum-of-the-squares
- TE Thermal Expansion Range in Normal and Upset Conditions
- $\Delta T1$ Value of the range of the temperature difference between the temperature of the outside surface to and the temperature of the inside surface Ti of the piping product assuming moment generating equivalent linear temperature distribution.

Table 3.12-2: Load Combinations and Acceptance Criteria for ASME Class 2 and 3 Piping

Condition	Load Combination for all Terms	Acceptance Criteria	ASME Code Reference
Design	Design Pressure + Weight + Sustained Mechanical Load (Note 1)	Equation 8 ≤ 1.5 S_h	NCD-3652
Service Levels A and B	Thermal Expansion	Equation 10a ≤ S_A (Note 2)	NCD-3653.2
Service Levels A and B	Single Non-repeated Anchor Movement	Equation 10b ≤ 3.0 S_C	NCD-3653.2
Service Levels A and B	Design Pressure + Weight + Sustained Mechanical Load + Thermal Expansion (Note 1)	Equation 11 ≤ (S_h + S_A) (Note 2)	NCD-3653.2
Service Level B	Peak Pressure + Weight + AOO (Note 3) Where AOO can be the item as listed below: 1. AOO Plant/System Upset	Equation 9 ≤ 1.8 S_h, but not greater than 1.5 S_y (Note 4)	NCD-3653.1
Service Level C	Peak Pressure + Weight + DBA Where DBA can be one of items as listed below: 1. Emergency Level Transient 2. ATWS 3. SBLOCA	Equation 9 ≤ 2.25 S_h, but not greater than 1.8 S_y (Note 5) (Note 6)	NCD-3654.2
Service Level C	Peak Pressure	Peak Pressure ≤ 1.5 P_a	NCD-3654.1
Service Level D	Peak Pressure + Weight + DBA Where DBA can be one of items as listed below: 1. SBLOCA + SSE 2. Large Break LOCA + SSE 3. Faulted Level Transient	Equation 9 ≤ 3.0 S_h, but not greater than 2.0 S_y (Note 7) (Note 8)	NCD-3655
Service Level D	Seismic Anchor Movement due to SSE	Equation in NCD-3655(b)(4)	NCD-3655
Service Level D	Peak Pressure	Peak Pressure ≤ 2.0 P_a	NCD-3655

Notes:

(1) Effect of sustained mechanical load is considered together with pressure and weight. In normal operation load, fluid flow (including Flow-Induced Vibration when applicable), thermal and fluid reaction forces are considered together with pressure and weight.

(2) Meet either Equation 10a or 11.

(3) An event category assigned to a PIE or event sequence with an expected frequency of 1/year to 1.0 E-02/year that represents a deviation from normal operation but does not evolve into a postulated accident.

(4) If the anchor displacement due to dynamic loads is excluded from Equation 9, it is included in Equation 10a or 11.

(5) Service Loadings, which Level C Service Limits are designated for, includes reversing dynamic loads that are not required to be combined with non-reversing dynamic loads.

(6) If the anchor motion due to reversing dynamic loads is excluded from Equation 9, it is assessed through the requirements of NCD-3655(b)(4) using 70% of the allowable.

(7) Service Loadings, which Level D Service Limits are designated for, includes reversing dynamic loads that are not required to be combined with non-reversing dynamic loads.

(8) If the anchor motion due to reversing dynamic loads is excluded from Equation 9, it is assessed through the requirement of NCD-3655(b)(4).

(9) Nomenclature:

- AOO Anticipated Operational Occurrences
- DBA Design Basis Accident
- P_a Calculated maximum allowable internal pressure for a straight pipe that is at least equal the Design pressure
- PD Design Pressure
- PP Peak Pressure Operating Pressure associated with Transient
- TE Thermal Expansion Range in Normal and Upset Conditions
- SBLOCA Small Block LOCA
- SSE Safe Shutdown Earthquake. Includes both inertial and displacement components which are combined using SRSS method.
- S_A Allowable stress range for expansion stresses = f (1.25 S_c+0.25 S_h), where f = stress range reduction factor for cyclic conditions.
- S_c Basic material allowable stress at minimum (cold) metal temperature
- S_h Basic material allowable stress at maximum (hot) metal temperature
- S_y Yield Strength

Table 3.12-3: Load Combinations and Acceptance Criteria for ASME B31.1 Piping and Components

Description	Load Combination	Acceptance Criteria	ASME Code Reference
Sustained	Design Pressure + Weight + Other Mechanical Loads	Eq 15 ≤ 1.0 S_h	ASME B31.1 Paragraph 104.8.1
Occasional	Design Pressure + Weight + Other Mechanical Loads + Seismic Seismic event includes but is not limited to: • SSE	Eq 16 ≤ k S_h (Note 1)	ASME B31.1 Paragraph 104.8.2 (Reference 3.12-5)
Occasional	Design Pressure + Weight + Other Mechanical Loads + Occasional event other than Seismic Occasional event includes but is not limited to: • AOO Plant/System Upset (Note 2) • Emergency Level Transient • ATWS • SBLOCA • Large Break LOCA • Faulted Level Transient • Wind • Snow Load	Eq 16 ≤ k S_h (Note 1)	ASME B31.1 Paragraph 104.8.2
Thermal Expansion and other Cyclic Load	Displacement Stress Range	Eq 17 ≤ f (0.25 S_h + 1.25 S_C (Note 3) (Note 4)	ASME B31.1 Paragraphs 104.8.3 and 102.3.2(b)(1)
Noncyclic Displacement	Noncyclic Displacement Stress Range	Eq 17 ≤ 3.0 S_C (Note 5)	ASME B31.1 Paragraph 102.3.2(b)(2)
Test	Test Pressure	Circumferential (hoop) stress ≤ 0.9 * Yield Stress	ASME B31.1 Paragraph 102.3.3 (B)
Test	Test Pressure + Weight	Longitudinal stress ≤ 0.9 * Yield Stress	ASME B31.1 Paragraph 102.3.3 (B)

Notes:

(1) For conventional piping in accordance with ASME B31.1 rules, the k factor in the equation for stresses due to occasional loads including seismic loading is increased to 1.8. Alternatively, a conservative approach can be adopted in which the seismic stresses in the stress combination for occasional loads can be multiplied by factor 2/3 with the k factor equal to 1.2.

(2) An event category assigned to a PIE or event sequence with an expected frequency of 1/year to 1.0 E-02/year that represents a deviation from normal operation but does not evolve into a postulated accident.

(3) Stated Eq 17 of ASME B31.1 allowable stress is assuming a cycle stress range factor of 1 if total number of cycles is below 7,776. If total number cycles for displacement range exceeds 7,776 (ASME B31.1, Paragraph 102.3.2 (b)(1c)) then a factor less than 1.0 is calculated per Eq 1C in Section 102.3.2 and applied to the allowable stress.

(4) When S_h is greater than S_L, the difference between them may be added to the term 0.25 S_h, and the allowable stress range is calculated by $S_A = f\ (1.25S_C + 1.25\ S_h - S_L)$.

(5) Replace M_C with the moment range due to the noncyclic movement in Equation 17 to calculate the displacement stress range, S_E.

(6) Nomenclature:

- Sh Basis material allowable stress at maximum (hot) metal temperature
- Sc Basic material allowable stress at minimum (cold) metal temperature
- Sy Yield Strength

Table 3.12-4: Load Combinations and Acceptance Criteria for ASME B31.3 Piping and Components

Description	Load Combination	Acceptance Criteria	ASME Code Reference
Sustained	Design Pressure + Weight + Other Mechanical Loads	Eq 23a ≤ 1.0 S_h (Note 1)	ASME B31.3, Paragraph 320.2
Occasional	Design Pressure + Weight + Other Mechanical Loads + Seismic Seismic event includes but is not limited to: • SSE	Eq 23a ≤ k S_h (Note 1) (Note 2)	ASME B31.3, Paragraphs 302.3.6(a) and 320.2
Occasional	Design Pressure + Weight + Other Mechanical Loads + Occasional event other than Seismic Occasional event includes but is not limited to: • AOO Plant/System Upset (Note 3) • Emergency Level Transient • ATWS • SBLOCA • Large Break LOCA • Faulted Level Transient • Wind • Snow Load	Eq 23a ≤ k S_h (Note 2)	ASME B31.3, Paragraphs 302.3.6(a) and 320.2
Thermal and Settlement	Displacement Stress Range	Eq 17 ≤ f (0.25 S_h + 1.25 S_C) (Note 1) (Note 4) (Note 5) (Note 6) (Note 7)	ASME B31.3, Paragraphs 319.4.4 and 302.3.5(d)
Test	Test Pressure	Circumferential (hoop) stress ≤ Yield Strength or 1.5 x Component Rating	ASME B31.3, Paragraph 345.2.1(a)
Test	Test Pressure + Weight	Longitudinal stress ≤ Yield Strength or 1.5 x Component Rating	ASME B31.3, Paragraph 345.2.1 (a)

Notes:

(1) The cold and hot allowable stress is calculated as defined in ASME BPVC Section III Subsection NCD. For further guidance, refer to RG 1.143, Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants (NPPs).

(2) For conventional piping in accordance with ASME B31.3, the k factor in the equation for stresses due to occasional loads including seismic loading is increased to 1.8. Alternatively, a conservative approach can be adopted in which the seismic stresses in the stress combination for occasional loads can be multiplied by factor 2/3 with the k factor equal to 1.2.

(3) An event category assigned to a PIE or event sequence with an expected frequency of 1/year to 1.0 E-02/year that represents a deviation from normal operation but does not evolve into a postulated accident.

(4) Stated Eq 17 of ASME B31.3 allowable stress is assuming a cycle stress range factor of 1 if total number of cycles is below 7,776. If total number cycles for displacement range exceeds 7,776 (ASME B31.3-2022, Paragraph 302.3.5(d)(1c)) then a factor less than 1.0 is calculated per Equation 1c in Section 302.3.5 and applied to the allowable stress.

(5) The bending stress range used in Equation 17 for elbows, miter bends, and branch connections is calculated in accordance with Equation 18 in Section 319.4.4(b) of ASME B31.3.

(6) The displacement stress range due to earth settlement can be greater than that permitted by the allowable stress if due consideration is given to avoidance of excessive localized strain and end reactions, per ASME B31.3, Paragraph 319.2.1(c).

(7) When S_h is greater than S_L, the difference between them may be added to the term 0.25 S_h, and the allowable stress range is calculated by $S_A = f\,(1.25\,(S_C + S_h) - S_L)$.

(8) Nomenclature:

- S_h Basic material allowable stress at maximum (hot) metal temperature
- S_c Basic material allowable stress at minimum (cold) metal temperature
- S_y Yield Strength

The fatigue usage limit is reduced for piping locations exempt from pipe break consideration.

Table 3.12-5: Load Combinations and Acceptance Criteria for ASME Class 1, 2, and 3 Snubber Type Supports

Condition	Load Combination for all Terms	Acceptance Criteria
Service Level B	AOO (Note 1)	Vendor Load Capacity Datasheet or Vendor Design Report Summary
Service Level C	DBA Where DBA can be one of the items as listed below: 1. Emergency Level Transient 2. ATWS 3. SBLOCA	
Service Level D	DBA Where DBA can be one of the items as listed below: 1. SBLOCA + SSE 2. Large Break LOCA + SSE 3. Faulted Level Transient	

Notes:

* Notes under Table 3.12-1 and Table 3.12-2 are applied to Table 3.12-5.

(1) An event category assigned to a PIE or event sequence with an expected frequency of 1/year to 1.0 E-02/year that represents a deviation from normal operation but does not evolve into a postulated accident.

Table 3.12-6: Load Combinations and Acceptance Criteria for ASME Class 1, 2, and 3 Rigid Type Supports

Condition	Load Combination for all Terms	Acceptance Criteria
Service Level A	Weight + Thermal Expansion + Sustained Mechanical Load (Note 1)	Vendor Load Capacity Datasheet, Vendor Design Report Summary or NF-3623
Service Level B	Weight + Thermal Expansion + AOO (Note 2)	
Service Level C	Weight + Thermal Expansion + DBA Where DBA can be one of the items as listed below: 1. Emergency Level Transient 2. ATWS 3. SBLOCA	
Service Level D	Weight + Thermal Expansion + DBA Where DBA can be one of the items as listed below: 1. SBLOCA + SSE 2. Large Break LOCA + SSE 3. Faulted level Transient	

Notes:

* Notes under Table 3.12-1 and Table 3.12-2 are applied to Table 3.12-6.

(1) Effect of sustained mechanical load is considered together with pressure and weight. In normal operation load, fluid flow (including Flow-Induced Vibration when applicable), thermal and fluid reaction forces, etc. are considered together with pressure and weight.

(2) An event category assigned to a PIE or event sequence with an expected frequency of 1/year to 1.0 E-02/year that represents a deviation from normal operation but does not evolve into a postulated accident.

Table 3.12-7: Load Combinations and Acceptance Criteria for ASME Class 1 Piping Connected to Reactor Pressure Vessel Nozzles

Condition	Load Combination for all Terms	Acceptance Criteria
Design	Design Pressure + Weight + Sustained Mechanical Load	In accordance with RPV and Nozzle Qualification Documentation/Specification
Service Level B	Peak Pressure + Weight + Sustained Mechanical Load + Thermal Expansion + AOO	
Fatigue	Peak Pressure + Weight + AOO +SSE	
Service Level C	Peak Pressure + Weight + DBA Where DBA can be one of the items as listed below: 1. Emergency Level Transient 2. ATWS 3. SBLOCA	
Service Level D	Peak Pressure + Weight + DBA Where DBA can be one of the items as listed below: 1. SBLOCA + SSE 2. Large Break LOCA + SSE 3. Faulted level Transient	

Note:

* Notes under Table 3.12-1 apply to Table 3.12-7.

Table 3.12-8: Load Combinations and Acceptance Criteria for ASME Class 2 Piping Connected to Reactor Pressure Vessel Nozzles

Condition	Load Combination for all Terms	Acceptance Criteria
Design	Design Pressure + Weight + Sustained Mechanical Load (Note 1)	In accordance with RPV and Nozzle Qualification Documentation/Specification
Service Levels A and B	Peak Pressure + Weight + Sustained Mechanical Load + Thermal Expansion (Note 1)	
Service Level B	Peak Pressure + Weight + AOO (Note 2) Where AOO can be the item as listed below: 1. AOO Plant/System Upset	
Service Level C	Peak Pressure + Weight + DBA Where DBA can be one of the items as listed below: 1. Emergency Level Transient 2. ATWS 3. SBLOCA	
Service Level D	Peak Pressure + Weight + DBA Where DBA can be one of the items as listed below: 1. SBLOCA + SSE 2. Large Break LOCA + SSE 3. Faulted Level Transient	

Notes:

(1) Effect of sustained mechanical load is considered together with pressure and weight. In normal operation load, fluid flow (including Flow-Induced Vibration when applicable), thermal and fluid reaction forces are considered together with pressure and weight.

(2) An event category assigned to a PIE or event sequence with an expected frequency of 1/year to 1.0 E-02/year that represents a deviation from normal operation but does not evolve into a postulated accident.

Table 3.12-9: Load Combinations and Acceptance Criteria for ASME Class 1, 2, and 3 Pipe-Mounted Valve Acceleration

Condition	Load Combination for all Terms	Acceptance Criteria
Service Level B	AOO (Note 1)	In accordance with applicable valve specification
Service Level C	DBA Where DBA can be one of the following items listed below: 1. Emergency Level Transient 2. ATWS 3. SBLOCA	
Service Level D	DBA Where DBA can be one of the following items: 1. SBLOCA + SSE 2. Large Break LOCA + SSE 3. Faulted Level Transient	

Note:

* Notes under Table 3.12-1 and Table 3.12-2 are applied to Table 3.12-9.

(1) An event category assigned to a PIE or event sequence with an expected frequency of 1/year to 1.0 E-02/year that represents a deviation from normal operation but does not evolve into a postulated accident.

Table 3.12-10: Load Combinations and Acceptance Criteria for ASME B31.1 and ASME B31.3 Linear Type Supports (Anchor, Guide, Strut)

Description	Load Combination	Acceptance Criteria
Sustained	Design Pressure + Weight + Other Mechanical Loads	Vendor Load Capacity Datasheet, Vendor Design Report Summary, or in accordance with ASME B31.1 and ASME B31.3
Thermal and Sustained	Displacement Stress Range + Design Pressure + Weight + Other Mechanical Loads	
Occasional	Displacement Stress Range + Design Pressure + Weight + Other Mechanical Loads + Seismic Seismic event includes but is not limited to: SSE	
Occasional	Displacement Stress Range + Design Pressure + Weight + Other Mechanical Loads + Occasional event other than Seismic Occasional event includes but is not limited to: AOO Plant/System Upset (Note 1) Emergency Level Transient ATWS SBLOCA Large Break LOCA Faulted Level Transient Wind Snow Load	

Note:

(1) An event category assigned to a PIE or event sequence with an expected frequency of 1/year to 1.0 E-02/year that represents a deviation from normal operation but does not evolve into a postulated accident.

3.13 Threaded Fasteners – ASME Code Class 1, 2, and 3

RCPB fasteners are designed, fabricated, erected, and tested to the highest quality standards practical, in accordance with GDC 1 and 30, so as to have an extremely low probability of abnormal leakage/rapidly propagating failure/gross rupture, in accordance with GDC 14. In accordance with GDC 31, designs include sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions; (A) the boundary behaves in a nonbrittle manner, and (B), the probability of rapidly propagating fracture is minimized and reflect consideration of service temperatures, and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining; (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

10 CFR 50, Appendix G, “Fracture Toughness Requirements,” specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the RCPB and includes fasteners with specified minimum yield strengths not over 896 MPa (130,000 psi). The supplemental fracture toughness requirements specified in 10 CFR 50, Appendix G do not apply to ferritic fasteners in ASME Class 2 and 3 systems. In this case, refer to NCD-2300.

For ASME Class 1, 2, and 3 fasteners, additional requirements are invoked. SRP 3.13, “Threaded Fasteners – ASME Code Class 1, 2, and 3,” provides guidance for selection of materials, design, inspection and testing of threaded fasteners, refer to Table 3.13-1, “ASME BPVC Section III Criteria for Selection and Testing of Bolting Materials.”

The BWRX-300 design conforms to the relevant requirements of the following regulations:

GDC 14 – Reactor coolant pressure boundary, as it relates to the RCPB being designed, fabricated, erected, and tested in a manner that provides assurance of an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture by compliance with 10 CFR 50, Appendix G. Reactor vessel material requirements are discussed further in Subsection 5.3.1.

GDC 30 – Quality of reactor coolant pressure boundary, as it relates to components of the RCPB being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety category function to be performed by compliance with ASME BPVC, Section III.

GDC 31 – Fracture prevention of reactor coolant pressure boundary, as it relates to the requirement that the RCPB be designed with sufficient margin to ensure that when stressed under operating, maintenance, testing, and postulated accident conditions the boundary behaves in a nonbrittle manner and the probability of rapidly propagating fracture is minimized with compliance with ASME BPVC, Section III and 10 CFR 50, Appendix G.

Additional Information on GDC compliance is found in Section 3.1.

10 CFR Part 50, Appendix B, as it relates to controlling the cleaning of material and equipment to prevent damage or deterioration.

10 CFR Part 50, Appendix G, as it relates to materials testing and acceptance criteria for fracture toughness of RCPB components.

10 CFR 50.55a incorporates by reference the design criteria of ASME Code, Section III, for Class 1, 2, and 3 components. The selection of materials, design, testing, fabrication, installation and inspection of threaded fasteners and mechanical joints are acceptable if they meet the criteria of the ASME Code, Section III, for Class 1, 2, and 3 components. However, 10 CFR 50.55a(b)(4) permits use of code cases that have been adopted by the USNRC in RG 1.84, “Design,

Fabrication, and Materials Code Case Acceptability, ASME Section III," in lieu of applicable criteria of ASME Code, Section III, Class 1, 2, and 3 components.

3.13.1 Design Aspects

The design of threaded fasteners (e.g., bolts, studs) complies with NB-3000, NCD-3000, NE-3000, NF-3000, and NG-3000. Because NE-3000 does not explicitly state the loads and load combinations that should be considered in the design of metal containment components, the load combinations and associated load factors are in compliance with RG 1.57, "Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components." Fabrication of threaded fasteners complies with NB-4000, NCD-4000, NE-4000, NF-4000, and NG-4000. Inspection of threaded fasteners complies with NB-2500, NCD-2500, NE-5000, NF-2500 and NG-2500.

3.13.1.1 Materials Selection

Material used for threaded fasteners (e.g., bolts, studs) complies with the requirements of ASME BPVC, Section III, Subsection NCA, NB-2128 (Class 1), NCD-2128 (Class 2 and 3), NF-2128 (supports), or NG-2121 (core support structures). Material for nuts conforms to SA-194 or to the requirements of one of the specifications for nuts. Material for bolting conforms to ASME BPVC, Section II, Part D, Subpart 1, Table 4 (Class 1) or Table 3 (Class 2 and 3).

Note: NB-2122/NCD-2122 states that special requirements in NB/NCD-2000 apply in lieu of the requirements of the material specification (NCA-4256) when there is conflict between the two.

Fracture toughness testing is performed in accordance with NB-2300, NCD-2300, NF-2300, or NG-2300, (bolting, including studs with a nominal thickness of greater than 1 in). ASME BPVC, Section II, Part D, Subpart 1, provides the properties for various classes of bolting materials.

Refer to RG 1.65, Rev. 1, "Materials and Inspections for Reactor Vessel Closure Studs," for guidance on reactor vessel closure stud bolting.

3.13.1.1.1 ASME Class 1 Applications

Sections 4.2 and 4.6 describe the use of threaded fasteners for tie rods lower end plugs, upper tie plate to accept channel fastener bolt, and for the Fine Motion Control Rod Drive (FMCRD) middle flange attachments to Control Rod Drive (CRD) housing.

Subsection 5.2.3 discusses the specifications for pressure-retaining ferritic materials, nonferrous metals, and austenitic stainless steels, including bolting and weld materials, which are used for each component (vessels, piping, and valves) of the RCPB. The adequacy and suitability of the ferritic materials, stainless steels, and nonferrous metals specified for the above applications are discussed in Subsections 5.2.3.1 through 5.2.3.4. Tables 5.2-2a and 5.2-3 list the materials and ASME standards for the RCPB which include the pipe, body bolting and bolting nuts.

ASME BPVC, Section II, "Materials," Part D, provides the properties for the following Class 1 bolting materials:

- Class 1 Bolting: Table 4
- Class 1 Bolting for Core Support Structures: Tables 2A, 2B

3.13.1.1.2 ASME Class 2 and 3 Applications

ASME BPVC, Section II, Part D, provides the properties for the following Class 2 and 3 bolting materials:

- Class 2 Bolting: Table 3
- Class 3 Bolting: Table 3

3.13.1.2 Mechanical Testing, Special Process and Controls

The appropriate use and selection of lubricants or sealants for threaded fasteners conform to the recommended practices provided in NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants." Note that molybdenum sulfide lubricant is known to promote corrosion in low-alloy steel and is not used.

10 CFR Part 50, Appendix B, Criterion XIII, requires that measures be established to control the cleaning of material and equipment to prevent damage or deterioration. RG 1.28, "Quality Assurance Program Criteria (Design and Construction)," provides quality assurance criteria for cleaning fluid systems and associated components that ensure compliance with Appendix B. Application of the cleaning criteria in threaded fasteners provides assurance that contaminants to which they could be exposed will not damage or deteriorate the materials, alter their properties, accelerate effects associated with aging, or increase the susceptibility to failure mechanisms such as stress corrosion cracking. Application of these criteria reduces the likelihood that degradation of threaded fasteners could lead to loss of pressure boundary integrity.

For verification of conformance to the applicable ASME BPVC requirements, a chemical analysis is required for each heat of material and testing for mechanical properties is required on samples representing each heat of material and, where applicable, each heat treat lot.

3.13.1.3 Fracture Toughness Requirements for Ferritic Materials

ASME Class 2 and 3 pressure-retaining ferritic material, and material welded thereto, are impact tested in accordance with the requirements of ASME BPVC, Section III, Division 1, NCD-2300 and NCD-2400 to ensure adequate fracture toughness properties. Fracture toughness and the pressure-temperature limits of the RCPB and reactor vessel are discussed in Subsections 5.3.1.5 and 5.3.2.

The fracture toughness of ferritic bolts, studs, and nuts (i.e., made from either low-alloy steel or carbon steel materials) is acceptable if the ASME Code, Section III criteria are appropriately specified for ASME Class 1, 2, and 3 systems. Ferritic bolts, studs, and nuts used in RCPB applications also meet the fracture toughness requirements of 10 CFR 50, Appendix G, which specifies supplemental requirements for ASME Class 1 systems.

3.13.1.4 Fabrication Inspection

Fabrication of threaded fasteners complies with:

- ASME BPVC, Section III, Division 1, NB-4000 (Reference 3.13-3) for Class 1 components
- ASME BPVC, Section III, Division 1, NCD-4000 (Reference 3.13-2) for Class 2 and 3 components
- ASME BPVC, Section III, Division 1, NE-4000 (Reference 3.13-4)
- ASME BPVC, Section III, Division 1, NF-4000 (Reference 3.13-5)
- ASME BPVC, Section III, Division 1, NG-4000 (Reference 3.13-6)

Inspection of threaded fasteners complies with:

- ASME BPVC, Section III, Division 1, NB-2500 (Reference 3.13-3) for Class 1 components
- ASME BPVC, Section III, Division 1, NCD-2500 (Reference 3.13-2) for Class 2 and 3 components
- ASME BPVC, Section III, Division 1, NE-2500 (Reference 3.13-4)
- ASME BPVC, Section III, Division 1, NF-2500 (Reference 3.13-5)

- ASME BPVC, Section III, Division 1, NG-2500 (Reference 3.13-6)

The visual examinations are required to be performed in accordance with procedures qualified to:

- ASME BPVC, Section III, Division 1, NB-5100 (Reference 3.13-3) for Class 1 components
- ASME BPVC, Section III, Division 1, NCD-5100 (Reference 3.13-2) for Class 2 and 3 components
- ASME BPVC, Section III, Division 1, NE-5100 (Reference 3.13-4)
- ASME BPVC, Section III, Division 1, NF-5100 (Reference 3.13-5)
- ASME BPVC, Section III, Division 1, NG-5100 (Reference 3.13-6)

The visual examinations are performed by personnel qualified in accordance with:

- ASME BPVC, Section III, Division 1, NB-5500 (Reference 3.13-3) for Class 1 components
- ASME BPVC, Section III, Division 1, NCD-5500 (Reference 3.13-2) for Class 2 and 3 components
- ASME BPVC, Section III, Division 1, NE-5500 (Reference 3.13-4)
- ASME BPVC, Section III, Division 1, NF-5500 (Reference 3.13-5)
- ASME BPVC, Section III, Division 1, NG-5500 (Reference 3.13-6)

Bolts, studs, and nuts are visually examined for discontinuities including cracks, bursts, seams, folds, thread lap, voids, and tool marks.

3.13.1.5 Quality Records

For threaded fasteners, documentation related to fracture toughness (as applicable) and Certified Material Test Reports (CMTRs) are provided as part of the ASME BPVC, Section III records that are provided at the time the parts are shipped and are part of the required records that are maintained at the site in accordance with NCA-1224 and 10 CFR 50.70, “Inspections.”

3.13.2 Preservice and Inservice Inspection Requirements

Subsection 5.2.4 and Section 6.6 describe the Preservice Inspection program and Inservice Inspection (ISI) program of ASME components.

3.13.3 References

3.13-1 ASME BPVC, Section II, “Materials,” American Society of Mechanical Engineers, 2021.

3.13-2 ASME BPVC, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” Subsection NCD, “Class 2 and Class 3 Components,” American Society of Mechanical Engineers, 2021.

3.13-3 ASME BPVC, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” Subsection NB, “Class 1 Components,” American Society of Mechanical Engineers, 2021.

3.13-4 ASME BPVC, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” Subsection NE, “Class MC Components,” American Society of Mechanical Engineers, 2021.

3.13-5 ASME BPVC, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” Subsection NF, “Supports,” American Society of Mechanical Engineers, 2021.

3.13-6 ASME BPVC, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” Subsection NG, “Core Support Structures,” American Society of Mechanical Engineers, 2021.

3.13-7 ASME BPVC, Section III, “Rules for Construction of Nuclear Facility Components,” Subsection NCA, “General Requirements for Division 1 and Division 2,” American Society of Mechanical Engineers, 2021.

Table 3.13-1: ASME BPVC Section III Criteria for Selection and Testing of Bolting Materials

Code Category		ASME Class 1 Criteria	ASME Class 2 and 3 Criteria
Material Selection		NCA-1220 and NB-2128	NCD-2128
Material Test Coupons and Specimens for Ferritic Steel Material (Tensile Test Criteria)	Heat Treatment Criteria	NB-2210	NCD-2210
	Test Coupons requirements bolting and stud materials	NB-2224(b)	NCD-2224
Fracture Toughness Requirements	Materials to be impact tested	NB-2311 (a)(2)	NCD-2311
	Types of impact test	NB-2321	NCD-2321
	Location of Test Specimens Orientation of Impact Test Specimens	NB-2322.1 NB-2322.2(a)(3)	NCD-2322.1 NCD-2322.2
	Acceptance Standards	NB-2333	NCD-2332.3 NCD-2333.2
	Number of impact tests necessary	NB-2345	NCD-2345
	Retesting	NB-2350	NCD-2352
	Calibration of test equipment	NB-2360	NCD-2360
Examination criteria for bolts, studs, and nuts		NB-2541 NB-2580	NCD-2580
CMTR criteria		NB-2130 NCA-3860	NCD-2130 NCA-3860

Note:

* Section III paragraphs listed in this table represent those specified in the 2021 Edition of Section III.

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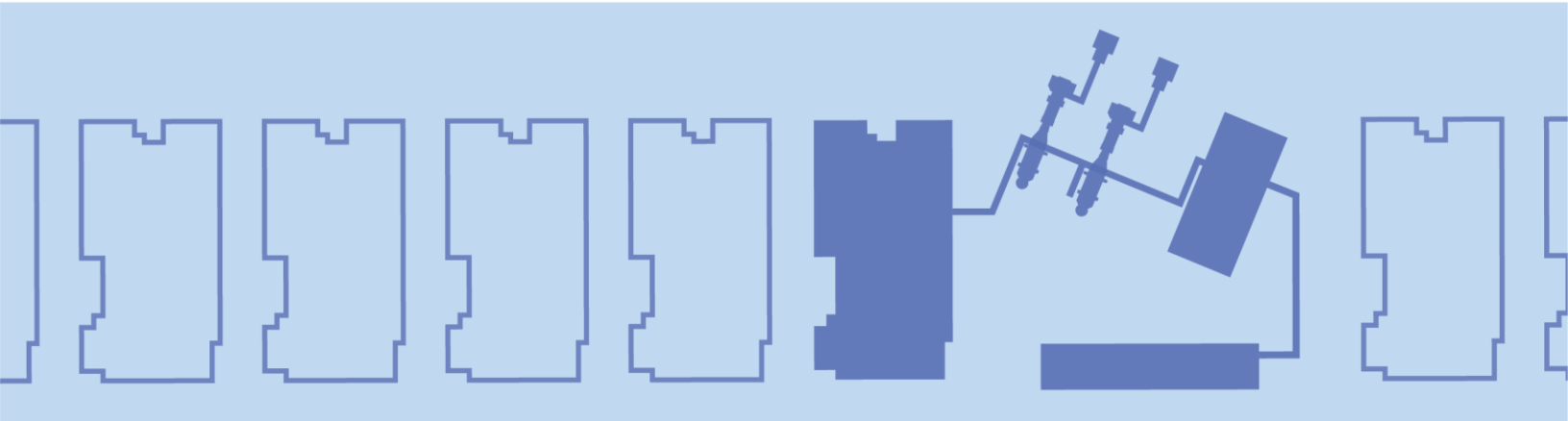
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Non-Proprietary Information

Chapter 3
Appendix 3A



ACRONYM LIST

Acronym	Explanation
1D	One-Dimensional
2D	Two-Dimensional
3D	Three-Dimensional
ALWR	Advanced Light Water Reactor
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrences
ARI	Alternate Rod Insertion
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
~~ATF~~	Acceleration Transfer Function
ATH	Acceleration Time History
AVR	Automatic Voltage Regulator
BIS	Boron Injection System
BWR	Boiling Water Reactor
CAD	Computer-Aided Design
CB	Control Building
CEPSS	Containment Equipment and Piping Support Structure
CIV	Containment Isolation Valve
CR	Control Room
CRD	Control Rod Drive
CST	Condensate Storage Tank
CUW	Reactor Water Cleanup System
CWS	Circulating Water System
DBA	Design Basis Accident
DCIS	Distributed Control and Information System
DEC	Design Extension Condition
DL2	Defense Line 2
DL3	Defense Line 3
DL4a	Defense Line 4a
DP-SC	Diaphragm Plate Steel-Plate Composite
EPRI	Electric Power Research Institute

Acronym	Explanation
FCU	Fan Coil Unit
FE	Finite Element
FIRS	Foundation Input Response Spectra
FMCRD	Fine Motion Control Rod Drive
FSAR	Final Safety Analysis Report
FW	Feedwater
GEH	GE Hitachi Nuclear Energy
GEZIP	General Electric Zinc Injection Passivation
GT	Gamma Thermometer
HCU	Hydraulic Control Unit
HRA	Human Reliability Analysis
HVS	Heating, Ventilation, and Cooling System
HWC	Hydrogen Water Chemistry
HX	Heat Exchanger
I&C	Instrumentation and Control
IC	Isolation Condenser
ICC	Isolation Condenser Pools Cooling and Cleanup System
ICS	Isolation Condenser System
ILRT	Integrated Leak Rate Test
ISRS	In-Structure Response Spectra
LB	Lower Bound
LMS	Lumped Mass Stick
LOCA	Loss-of-Coolant Accident
LV	Low Voltage
MAAP	Modular Accident Analysis Program
MCA	Main Condenser and Auxiliaries
MCC	Main Control Center
MCR	Main Control Room
MS	Main Steam
MSL	Main Steam Line
MSR	Moisture Separator Reheater System
MTE	Main Turbine Equipment

Acronym	Explanation
NBS	Nuclear Boiler System
NHS	Normal Heat Sink
NRV	Non-Return Valve
OLNC	On-Line NobleChem
PBIRS	Performance Based Intermediate Response Spectra
PCS	Primary Containment System
PBSRS	Performance Based Surface Response Spectra
PCV	Pressure Control Valve
PCW	Plant Cooling Water System
PIE	Postulated Initiating Event
PPS	Plant Pneumatics System
PRA	Probability Risk Assessment
QAPD	Quality Assurance Program Description
RB	Reactor Building
RCPB	Reactor Coolant Pressure Boundary
RG	Regulatory Guide
RIV	Reactor Isolation Valve
RPV	Reactor Pressure Vessel
RVT	Regulating Voltage Transformer
RWB	Radwaste Building
RWST	Refueling Water Storage Tank
SC	Safety Class
SC1	Safety Class 1
SC2	Safety Class 2
SC3	Safety Class 3
SCCV	Steel-Plate Composite Containment Vessel
SCR	Secondary Control Room
SDC	Shutdown Cooling System
SDG	Standby Diesel Generator
SEI	Structural Engineering Institute
SIR	Seismic Interface Restraint
SRSS	Square-Root-of-the-Sum-of-Squares

Acronym	Explanation
SSCs	Structures, Systems, and Components
SSE	Safe Shutdown Earthquake
SSI	Soil-Structure Interaction
SSSI	Structure-Soil-Structure Interaction
TB	Turbine Building
TBD	To Be Determined
TBV	Turbine Bypass Valve
TRACG	Transient Reactor Analysis Code General Electric
UB	Upper Bound
UPS	Uninterruptible Power Supply
USNRC	U.S. Nuclear Regulatory Commission
WRNM	Wide Range Neutron Monitor
ZPA	Zero Period Acceleration

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APPENDIX 3A – PRELIMINARY CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

3A.1 INTRODUCTION

The BWRX-300 approach to classifying Structures, Systems, and Components (SSCs) is based on classification principles described in Section 3.2 and NEDC-33934P, “BWRX-300 Safety Strategy,” (Reference 3A.1-1).

Classification of SSCs is conducted to identify the importance of the SSCs with respect to safety. As such, this approach is based primarily on deterministic methods, in that the classifications are not based on the calculated risk of each component but are assigned deterministically based on the functions performed by the SSCs and the importance of those functions.

The important factors in classifying SSCs are:

- Consequences of failure to perform the functions
- Expected frequency of an SSCs performing its functions
- Time following a Postulated Initiating Event (PIE) at which, or the period for which, the SSCs performs its functions

The methodology for classification of BWRX-300 SSCs is discussed in Section 3.2 in accordance with:

- Safety Class (SC)
- Quality Group
- Seismic Category

Table 3A-1 provides a preliminary list of the principal BWRX-300 components organized by system. Classification of structures is provided in Table 3A-2.

3A.2 References

3A.1-1 NEDC-33934P, “BWRX-300 Safety Strategy Licensing Topical Report,” GE-Hitachi Nuclear Energy Americas, LLC., Revision 2, March 2026.

Table 3A-1: Preliminary BWRX-300 Component Classification List

Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes
REACTOR SYSTEMS					
Core and Fuel					
Fuel Channel Assembly, Shell, Strip, Clip	SC1	PCS, RB	N/A	I	
Fuel Channel Fastener Assembly, Cap Screw, Guard (Machining and Casting)	SC1	PCS, RB	N/A	I	
Fuel Bundle Assembly	SC1	PCS, RB	N/A	I	
Core Support Structures: • Shroud • Core Support Components • Control Rod Guide Tubes • Non-Pressure Boundary Portion of Control Rod Drive (CRD) Housings	SC1	PCS	Note 10	I	
Internal Structures: • Chimney Barrel and Upper/Lower Flanges • Nuclear Instrumentation Incore Guide Tubes • Non-Pressure Boundary Portion of Incore Housings • ICS Return Internal Piping	SC1	PCS	Note 10	I	

Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes
Internal Structures: • Chimney Head and Steam Separator Assembly • Steam Dryer Assembly • FW Spargers • Head Vent Internal Piping • CUW Suction Piping • Nuclear Instrumentation Incore Guide Tube Stabilizers	SC3	PCS	Note 10	II	Note 8
Nuclear Instrumentation Dry Tube Wide Range Neutron Monitor (WRNM)	SC1	PCS	A	I	
WRNM	SC3	PCS	N/A	II	
Nuclear Instrumentation Housings, Flanges	SC1	PCS	A	I	
Local Power Range Monitors	SC1	PCS	N/A	I	
Gamma Thermometers	SC2	PCS	N/A	II	
CRD Housing	SC1	PCS	A	I	
Control Rods	SC1	PCS	N/A	I	
Control Rod Drive System					
Non-pressure retaining Fine Motion Control Rod Drive (FMCRD) Scram Subcomponents	SC1	PCS	N/A	I	
FMCRD Reactor Coolant Pressure Boundary (RCPB) Subcomponents	SC1	PCS	A	I	
Non-RCPB FMCRD Components that Prevent Rod Drop or Rod Ejection	SC1	PCS	N/A	I	
FMCRD Motor	SC2	PCS	N/A	Non-Seismic	
FMCRD Separation Switches	SC3	PCS	N/A	Non-Seismic	

Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes
FMCRD Position Indication Probe with Switches	SC3	PCS	N/A	Non-Seismic	
All other FMCRD Subcomponents not specified in rows above	SC3	PCS	N/A	Non-Seismic	
Hydraulic Control Unit (HCU) Nitrogen Tank	SC1	RB	B	I	
HCU Scram Valve	SC1	RB	B	I	
HCU Accumulator	SC1	RB	B	I	
HCU Scram Solenoid Pilot Valve Assembly	SC1	RB	N/A	I	
HCU Instrument Manifold Pressure Boundary Components	SC1	RB	B	I	
Alternate Rod Insertion (ARI) Valves	SC2	RB	D	Non-Seismic	
HCU Piping and Piping Between HCU and FMCRD	SC1	RB, PCS	B	I	
Charging Water Piping and Valves (except when directly above HCUs)	SC3	RB	D	Non-Seismic	
Charging Water Piping and Valves (directly above HCUs)	SC3	RB	D	II	
Purge Water Piping and Valves (except when directly above HCUs)	SC3	RB	D	Non-Seismic	
Purge Water Piping and Valves (directly above HCUs)	SC3	RB	D	II	
CRD Pumps	SC3	RB	D	Non-Seismic	
CRD Purge Pressure Control Valve	SC3	RB	D	Non-Seismic	

Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes
REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS					
Nuclear Boiler System					
Reactor Pressure Vessel (RPV)	SC1	PCS	A	I	
Reactor Isolation Valves (RIV) for Main Steam (MS), RPV Head Vent Subsystem, Isolation Condenser System (ICS), Feedwater (FW), and Reactor Water Cleanup System (CUW)	SC1	PCS	A	I	
RPV Stabilizers	SC1	PCS	Note 10	I	
RPV Support Skirt	SC1	PCS	Note 10	I	
Main Steam Line (MSL) piping from the RIV to the outboard MS Containment Isolation Valve (CIV)	SC1	PCS	B	I	
Outboard MSL CIVs	SC1	RB	B	I	
RPV Level Instrumentation Sensing Line including pressure-retaining parts of instrumentation located on these lines	SC1	RB	B	I	
MSL piping and components from the CIV to the Seismic Interface Restraint (SIR)	SC1	RB	B	I	
MSL SIR	SC1	RB	B	I	
MSL piping and components from the SIR to the Condensate and Feedwater System, Main Turbine Equipment, Moisture Separator Reheater System, Turbine Bypass System, Main Condenser and Auxiliaries, and Equipment and Floor Drain System	SC3	RB/TB	D	Non-Seismic	
MSL leak detection instrumentation in Reactor Building	SC1	RB	N/A	I	

Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes
MSL leak detection instrumentation in Turbine Building	SC1	TB	N/A	II	
Other Nuclear Boiler System (NBS) mechanical/instrumentation American Society of Mechanical Engineers (ASME) Section III pressure boundary components on the MSLs	SC1	RB	B	I	
Other NBS mechanical/instrumentation ASME B31.1 pressure boundary components on the MSLs	SC3	TB	D	Non-Seismic	
Shutdown Cooling System					
Pumps	SC3	RB	D	Non-Seismic	
Heat Exchangers (HX)	SC3	RB	D	Non-Seismic	
Leak Detection Equipment supporting Safety Category 1 functions	SC1	RB	C	I	
SDC Piping and Valves	SC3	RB	D	Non-Seismic	
Reactor Water Cleanup System					
HX	SC3	TB	D	Non-Seismic	
Leak Detection Equipment supporting Category 1 functions	SC1	RB	B	I	
Leak Detection Equipment supporting Safety Category 1 functions	SC1	TB	C	Non-Seismic	
Piping and Components from RIVs to SIR	SC1	RB	B	I	
Piping and Components Outboard of SIR	SC3	RB, TB	D	Non-Seismic	
Reactor Pressure Vessel Head Vent Subsystem					
Piping and Components from RIV to MSL	SC1	PCS	B	I	
Piping to Quench Tank Isolation Valve	SC1	PCS	B	I	

Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes
Quench Tank Isolation Valves	SC1	PCS	B	I	
Quench Tank	SC3	PCS	D	II	
Head Vent Quench Tank Vacuum Breaker	SC3	PCS	D	II	
ENGINEERED SAFETY FEATURES					
Primary Containment System					
Containment Closure Head	SC1	PCS	Note 10	I	
Personnel and Equipment Air Locks	SC1	PCS	Note 10	I	
Refueling Bellows Seal	SC1	PCS	Note 10	I	
All Containment Penetrations	SC1	PCS	B	I	
Integrated Leak Rate Test Piping and Valves forming part of the Containment Boundary	SC1	PCS	B	I	
Containment Sump	SC3	RB	N/A	I	
Passive Containment Cooling System					
Piping and Valves forming part of the Containment Boundary	SC1	PCS, RB	B	I	
Isolation Condenser System					
Steam Supply, Condensate Return, Standby Gas Purge Piping	SC1	PCS	A	I	
Shutdown Cooling System (SDC) Interface Piping to Redundant Outboard CIVs, Trains A and B	SC1	PCS, RB	A	I	
Boron Injection System (BIS) Interface Piping to BIS Interface Valve, Train C	SC1	PCS	A	I	
Isolation Condenser	SC1	PCS, RB	A	I	

Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes
Steam Supply and Condensate Return Valves	SC1	PCS	A	I	
Standby Gas Purge Valves	SC1	PCS	A	I	
INSTRUMENTATION AND CONTROL SYSTEMS					
SC1 Instrumentation and Control System					
Safety Class 1 (SC1) Instrumentation and Control (I&C) System	SC1	RB, CB	N/A	I	
SC2 and SC3 Instrumentation and Control System					
Equipment that supports Defense Line 2 (DL2) functions	SC3	RB, TB, and CB	N/A	Note 11	
Equipment that supports Defense Line 4a (DL4a) functions	SC2	RB, TB, and CB	N/A	Note 11	
Equipment that supports DL4b functions	SC3	FSAR	N/A	Non-Seismic	
Non-Safety I&C System	SCN	RB, TB, and CB	N/A	Non-Seismic	
ELECTRICAL POWER SYSTEMS					
Emergency Power System					
Uninterruptible Power Supply (UPS)	SC1	RB	N/A	I	
Battery Chargers	SC1	RB	N/A	I	
Battery Monitors	SC1	RB	N/A	I	
Battery Banks	SC1	RB	N/A	I	
Standby Power System					
Standby Diesel Generators	SC3	TB	N/A	Non-Seismic	
Low-Voltage (LV) Switchgear	SC3	CB, TB	N/A	Non-Seismic	
LV Motor Control Center (MCC)	SC3	RB, CB, TB	N/A	Non-Seismic	

Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes
Dry Type Transformers	SC3	CB, TB	N/A	Non-Seismic	
UPS Bypass Regulating Voltage Transformers	SC2	CB	N/A	Non-Seismic	
UPS	SC2	CB	N/A	Non-Seismic	
Battery Chargers	SC2	CB	N/A	Non-Seismic	
Battery Monitors	SC2	CB	N/A	Non-Seismic	
Battery Banks	SC2	CB	N/A	Non-Seismic	
FMCRD Packaged UPS	SC2	RB	N/A	Non-Seismic	
Preferred Power System					
Medium Voltage Switchgear	SC3	TB	N/A	Non-Seismic	
Protective Relaying	SC3	ALL	N/A	Non-Seismic	
DC power Systems that Support Protective Relays	SC3	ALL	N/A	Non-Seismic	
6900 V Breakers for Condensate Pumps and Feedwater Pumps	SC2	TB	N/A	Non-Seismic	
All other 6900 V Breakers	SC3	TB	N/A	Non-Seismic	
UAT	SC3	YARD	N/A	Non-Seismic	
RAT	SC3	YARD	N/A	Non-Seismic	
GSU	SC3	YARD	N/A	Non-Seismic	
LV Switchgear	SC3	CB, TB	N/A	Non-Seismic	
LV MCC	SC3	RB, CB, TB, RWB	N/A	Non-Seismic	
Dry Type Transformers	SC3	CB, TB	N/A	Non-Seismic	
UPS Bypass RVTs	SC3	TB	N/A	Non-Seismic	
UPS	SC3	TB	N/A	Non-Seismic	
Battery Chargers	SC3	TB	N/A	Non-Seismic	

Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes
Battery Monitors	SC3	TB	N/A	Non-Seismic	
Battery Banks	SC3	TB	N/A	Non-Seismic	
Offsite Power System					
Offsite Power System					Note 14
Cable Raceway System					
Reactor Building Raceway for DL3	SC1	RB	N/A	I	
Non-Reactor Building Raceway for DL3	SC1	CB, TB, RWB	N/A	Non-Seismic	
Reactor Building Raceway for DL4a	SC2	RB	N/A	Non-Seismic	
Non-Reactor Building Raceway for DL4a	SC2	CB, TB, RWB	N/A	Non-Seismic	
Reactor Building Raceway for DL4b	SC3	RB	N/A	Non-Seismic	
Non-Reactor Building Raceway for DL4b	SC3	CB, TB, RWB	N/A	Non-Seismic	
Reactor Building Raceway for DL2	SC3	RB	N/A	Non-Seismic	
Non-Reactor Building Raceway for DL2	SC3	CB, TB, RWB	N/A	Non-Seismic	
Other Reactor and Non-Reactor Building Raceway	SCN	RB, CB, TB, RWB	N/A	Non-Seismic	
Grounding Lightning Protection System					
All components	SCN	ALL	N/A	Non-Seismic	
AUXILIARY SYSTEMS					
FUEL STORAGE AND HANDLING					
New and Spent Fuel Storage					
Fuel Storage Rack	SC1	RB	N/A	I	
Fuel Pool Gate	SC1	RB	C	I	
Equipment Pool Gate	SC1	RB	C	I	

Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes
Control Rod and Fuel Support Grapple Tooling and Special Storage Racks	SCN	RB	N/A	FSAR	
Fuel Pool Cooling and Cleanup System					
FLEX/EME System Makeup	SC3	RB	D	I	
System equipment, piping, valves which penetrates the Fuel Pool	SC1	RB	B	I	
System equipment, piping, valves in or adjacent to the Fuel Pool (interaction)	SC3	RB	D	II	
General system equipment, piping, valves including normal makeup	SC3	RB	D	Non-Seismic	
Pumps	SC3	RB	D	Non-Seismic	
Heat Exchangers	SC3	RB	D	Non-Seismic	
Demineralizer	SC3	RB	D	Non-Seismic	
Filters	SC3	RB	D	Non-Seismic	
Light Load Handling System					
Refueling Platform	SCN	RB	N/A	FSAR	SSCs in vicinity of fuel storage racks are evaluated for seismic interaction
Fuel Prep Machine	SCN	RB	N/A	Non-Seismic	
Channel Handling Tool, Channel Handling boom, Channel Transfer Grapple	SCN	RB	N/A	Non-Seismic	
General refueling and servicing equipment to support outages	SCN	RB	N/A	Non-Seismic	
Overhead Heavy Load Handling System					
Polar Crane	SCN	RB	N/A	II	

Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes
WATER SYSTEMS					
Service Water Subsystem					
Pumps, HXs, piping associated with heat removal	SC3	TB, OO	D	Non-Seismic	
Piping, valves, equipment not associated with above	SCN	TB, OO	D	Non-Seismic	
Plant Cooling Water System					
Components associated with Makeup Water supply to the surge tanks	SCN	All	Note 10	Non-Seismic	
Components associated with ICC system cooling	SCN	All	D	Non-Seismic	
All other PCW System Equipment	SC3	All	D	Non-Seismic	
Demineralized Water Storage and Distribution Subsystem					
Demineralized water piping and valves forming part of the containment	SC1	PCS, RB	B	I	
All other Demineralized Water components	SCN	All	Note 10	Non-Seismic	
Potable Water Distribution and Sanitary Sewer Handling Subsystems					
All system components	SCN	All	Note 10	Non-Seismic	
Condensate Storage and Transfer Subsystem					
Condensate Storage Tank	SCN	TB	D	Non-Seismic	
Condensate Storage and Transfer Subsystem Piping and Valves	SCN	TB, RB	D	Non-Seismic	
Piping and Valves forming part of the Containment Boundary	SC1	PCS, RB	B	I	

Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes
Chilled Water Equipment					
Components such as Appurtenances supporting Heating, Ventilation, and Cooling System (HVS) after Shutdown I&C Equipment	SC3	RB, CB	D	II	
Piping and Valves Inside Containment that support Containment Cooling	SC3	PCS, RB	D	II	
Piping and Valves forming part of the Containment Boundary	SC1	PCS, RB	B	I	
Air-Cooled Chillers, Expansion Tanks, Chiller Pumps, and Air Separators	SC3	OO	D	Non-Seismic	
Components supporting HVS for Non-Safety Equipment	SCN	ALL	D	Non-Seismic	
Glycol Fill and Chemical Bypass	SCN	OO	D	Non-Seismic	
Isolation Condenser Pools Cooling and Cleanup System					
Suction Surge Tank Return Guard Tank	SC1	RB	B	I	
Piping enclosed by Return Guard Tank	SC1	RB	B	I	
Anti-siphon Device (including vent tubing connected to anti-siphon device)	SC1	RB	B	I	
Isolation Valves connected to RB Penetrations into Such Surge Tank and Return Guard Pipe	SC1	RB	B	I	
Unidirectional Pool Makeup Conduits	SC1	RB	B	I	
IC Pool Atmospheric Vents	SC1	RB	B	I	
IC Pool Drain Isolation Valves	SC1	RB	B	I	
Filter/Storage Pool Atmospheric Vent	SC3	RB	D	I	
Filter/Storage Pool Drain Isolation Valves	SC3	RB	D	I	

Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes
Drain Piping from IC Pool Atmospheric Vents and Filter/Storage Pool Atmospheric Vent	SC3	RB	D	Non-Seismic	
FLEX/EME Connection and Piping (including valves)	SC3	RB, YRD	D	I	
Other Isolation Condenser Pools Cooling and Cleanup System (ICC) Components and Piping, Pumps/Adjustable Speed Drives, Heat Exchangers, Demineralizer, Dosing Pot, ICC Piping and Distribution Spargers)	SCN	RB	D	Non-Seismic	
Normal Heat Sink					
Components associated with the Heat Sink for CWS SC3 Loads	SC3	YRD	Note 10	Non-Seismic	
All Remaining Components not associated with the functions above	SCN	YRD	Note 10	Non-Seismic	
Refueling Water Storage and Cleanup Subsystem					
Refueling Water Storage and Cleanup Subsystem equipment, piping and valves	SC3	TB, RB	D	Non-Seismic	SC3 for confinement of radioactive materials
Process Auxiliaries					
Plant Pneumatics System					
Piping and Valves forming part of the Containment Boundary	SC1	PCS, RB	B	I	
Service Air Distribution Piping (downstream of the service air isolation valve)	SCN	ALL	N/A	Non-Seismic	

Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes
Remaining Plant Pneumatics System (PPS) piping, valves, and equipment that do not form the containment boundary nor are downstream of the service air isolation valve (Note: includes Instrument Air Distribution and supply for Instrument Air)	SC3	ALL	N/A	Non-Seismic	
Sensors and Transmitters with a control function	SC3	TB	N/A	Non-Seismic	
Transmitters used for indication or alarms	SCN	TB	N/A	Non-Seismic	
Process Sampling Subsystem					
Non-pressure Boundary Sampling Equipment	SCN	RB, TB, RWB	N/A	Non-Seismic	
Pressure Boundary Sampling Equipment (non-contaminated)	SCN	RB, TB, RWB	Note 10	Non-Seismic	
Pressure Boundary Sampling Equipment (contaminated)	SC3	RB, TB, RWB	D	Non-Seismic	
Equipment and Floor Drain System					
Piping and Valves forming part of the Containment Boundary	SC1	PCS, RB	B	I	
Oily Waste Sump System and Other Non-Radioactive Subsystems	SCN	TB	Note 10	Non-Seismic	
General Equipment and Floor Drain System, Pressurized Sump, Drain Piping and Valves	SC3	ALL	D	Non-Seismic	
Containment Inerting System					
Piping and Valves forming part of the Containment Boundary	SC1	PCS, RB	B	I	
All other System Equipment and Piping	SC3	RWB, RB, YRD	D	Non-Seismic	

Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes
Hydrogen Water Chemistry System					
Hydrogen Water Chemistry (HWC) System	SCN	TB	N/A	Non-Seismic	
On-Line NobleChem System					
On-Line NobleChem™ (OLNC) System	SCN	TB	N/A	Non-Seismic	
Zinc Injection System					
Zinc Injection System	SCN	TB	N/A	Non-Seismic	
Boron Injection System					
Injection Pump	SC3	RB	D	Non-Seismic	
Storage Tank	SC3	RB	D	Non-Seismic	
Instrument Line Pressure Retaining Components	SC3	RB	D	Non-Seismic	
Instrumentation – Storage Tank Level, Discharge Pressure, Solution Temperature, Flow Measurement	SC3	RB	N/A	Non-Seismic	
Piping from Tanks to Pumps	SC3	RB	D	Non-Seismic	
Piping from Pumps to Outboard CIV	SC3	RB	D	Non-Seismic	
Piping and Valves forming part of the Containment Boundary	SC1	PCS, RB	A	I	
Piping and Valves forming part of the Containment Boundary	SC1	PCS, RB	A	I	
Piping, Valves, and Other Pressure Retaining Components with no Safety Function	SCN	RB	D	Non-Seismic	
Instrumentation with no Safety Function	SCN	RB	N/A	Non-Seismic	

Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes
HEATING, VENTILATION, AND COOLING SYSTEMS					
Control Building Heating, Ventilation, and Cooling System					
MCR Emergency HVS	SC3	CB	N/A	Non-Seismic	
SC2 and SC3 I&C DL2 Room FCUs	SC3	CB	N/A	Non-Seismic	
DL4a Room FCUs	SC2	CB	N/A	Non-Seismic	
All other CB HVS components	SCN	CB	N/A	Non-Seismic	
Radwaste Building Heating, Ventilation, and Cooling System					
RWB HVS Equipment	SCN	RWB	N/A	Non-Seismic	
Turbine Building Heating, Ventilation, and Cooling System					
TB HVS Equipment	SCN	TB	N/A	Non-Seismic	
Reactor Building Heating, Ventilation, and Cooling System					
SCR Emergency HVS	SC2	RB	N/A	Non-Seismic	
Refueling Floor Isolation Dampers and Battery Room Exhaust Fans	SC3	RB	N/A	Non-Seismic	
All other RB HVS Equipment	SCN	RB	N/A	Non-Seismic	
Containment Cooling System					
Air Handling Units, and Ductwork and related instrumentation	SC3	PCS	Note 10	II	
Drain Piping and related components	SCN	PCS	Note 10	II	
OTHER AUXILIARY SYSTEMS					
Fire Protection System					
Buried Piping and Valves	SCN	YRD, OL	Note 10	Non-Seismic	
Piping, Valves, and Sprinklers	SCN	RB, TB, CB, RWB, OL	Note 10	Non-Seismic	

Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes
Firewater storage tanks and firewater pumps	SCN	YRD, OL	Note 10	Non-Seismic	
Fire Protection Equipment (Smoke Detection/Local Control Panels/Manual Fire Alarm Stations, Extinguishers)	SCN	PCS, RB, CB, TB, RWB	Note 10	Non-Seismic	
Non-DCIS Communication System					
All Components excluding the Radio System UPS	SC3	ALL	N/A	Non-Seismic	
Radio System UPS	SCN	CB	N/A	Non-Seismic	
Lighting					
Emergency Lighting	SC3	ALL	N/A	I	
Normal Lighting	SCN	ALL	N/A	Non-Seismic	
Control Room Lighting	SC2	RB, CB	N/A	I	
Emergency Lighting Panels	SC3	ALL	N/A	I	
Normal Lighting Panels	SCN	ALL	N/A	Non-Seismic	
Standby Diesel Generator Fuel Oil Storage and Transfer System					
Diesel Fuel Oil Storage and Transfer System	SC3	YRD, TB	Note 10	Non-Seismic	
Cranes, Hoists, and Elevators					
Cranes, Hoists, and Elevators located within the RB	SCN	RB	N/A	II	
Cranes, Hoists, and Elevators located outside of the RB	SCN	CB, TB, RWB	N/A	Non-Seismic	

Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes
STEAM AND POWER CONVERSION SYSTEM					
Main Turbine Equipment					
Main Turbine Equipment (MTE) and Subsystem	SC3	TB	D	Non-Seismic	
Generator and Exciter					
Generator and Exciter System	SC3	TB	N/A	Non-Seismic	
Main Condenser and Auxiliaries					
MCA System (Main Condenser, Vacuum Pump, SJAE)	SC3	TB	D	Non-Seismic	
Piping and Valves	SC3	TB	D	Non-Seismic	
Wide Range Pressure Transmitters	SC2	TB	N/A	Non-Seismic	
Narrow Range Pressure Transmitters	SC3	TB	N/A	Non-Seismic	
All Remaining Components not associated with the functions above	SCN	TB	N/A	Non-Seismic	
Moisture Separator Reheater System					
Moisture Separator Reheater System	SC3	TB	D	Non-Seismic	
Turbine Gland Seal System					
Turbine Gland Seal System	SC3	TB	N/A	Non-Seismic	
Turbine Bypass System					
Turbine Bypass Valves and Components	SC3	TB	D	Non-Seismic	
All other System Equipment	SCN	TB	N/A	Non-Seismic	
Circulating Water Supply					
Pumps, piping, valves, equipment associated with SC3 heat removal	SC3	TB, OO	D	Non-Seismic	
Instrumentation			N/A		

Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes
Piping, valves, equipment not associated with above Instrumentation	SCN	TB, OO	D	Non-Seismic	
All Remaining Components not associated with the functions above	SCN	TB, OO	N/A	Non-Seismic	
Condensate Filters and Demineralizer System					
Condensate Filters and Demineralizers, Bypass Lines, and Related Components	SC3	TB	D	Non-Seismic	
All other System Equipment	SCN	TB	D	Non-Seismic	
Condensate and Feedwater Heating System					
Piping and Valves from the SIR near the RB wall to the FW RIVs	SC1	RB, PCS	B	I	
Differential Pressure Measurement for Feedwater Leak Detection	SC1	TB	Note 10	Non-Seismic	Note 13
Components supporting the detection of loss of FW flow	SC1	TB	Note 10	Non-Seismic	Note 13
System Components in the FW flow path from the Condenser interface to the Seismic Restraint near the RB wall	SC3	TB, RB	D	Non-Seismic	
System Components in the FW Heater drain path to the condenser	SC3	TB	D	Non-Seismic	
All other System Equipment	SCN	TB	N/A	Non-Seismic	
RADIOACTIVE WASTE MANAGEMENT SYSTEMS(7)					
Liquid Waste Management Systems					
Collection and Sample Tanks	SC3	RWB	D	RW-IIa	
Collection and Sample Pumps	SC3	RWB	D	RW-IIa	

Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes
LWM Piping and Valves (except Flexible Hoses and Hose Connections for Filtering Skids)	SC3	RWB	D	RW-IIa	
Waste Sampling Subsystem Piping and Valves	SCN	RWB	D	Non-Seismic	
Offgas System					
TB Piping and Valves	SC3	TB	D	Non-Seismic	
Offgas Recombiner	SC3	TB	D	Non-Seismic	
Cooler Condenser	SC3	TB	D	Non-Seismic	
Moisture Separator	SC3	TB	D	Non-Seismic	
Refrigeration Dryers	SC3	TB	D	Non-Seismic	
Gas Analyzers	SC3	TB	D	Non-Seismic	
RWB Piping and Valves	SC3	RWB	D	RW-IIa	
Offgas Reheater	SC3	RWB	D	RW-IIa	
Charcoal Vault, and Charcoal Adsorber Tanks	SC3	RWB	D	RW-IIa	
Offgas Post Filter	SC3	RWB	D	RW-IIa	
Solid Waste Management Systems					
Spent Resin Tank, Pump, Valves, and Piping	SC3	RWB	D	Non-Seismic	
Sludge Tanks, Pump, Valves, and Piping	SC3	RWB	D	Non-Seismic	
RADIATION MONITORING SYSTEMS					
Process Radiation Monitoring Subsystem					
Radiation Monitoring Equipment (non-pressure boundary)	SC3	RB, CB, TB, RWB	N/A	Non-Seismic	

Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes
Radiation Monitoring Equipment (pressure boundary)	SC3	RB, CB, TB, RWB	D	Non-Seismic	
Containment Monitoring Subsystem					
Piping and Valves forming part of the Containment Boundary	SC1	RB, PCS	B	I	
Containment Pressure Transmitters	SC1	RB	N/A	I	
Containment Temperature Transmitters	SC3	RB	N/A	I	
Containment Water Level Transmitters	SC3	RB	N/A	Non-Seismic	
Containment Hydrogen and Oxygen Monitoring Equipment (including process piping outboard of CIVs)	SC3	RB	D	I	
Containment Fission Product Monitoring Equipment (including process piping outboard of CIVs)	SC3	RB	D	Non-Seismic	
Containment Area Radiation Monitors	SC3	PCS	N/A	I	
Area Radiation Monitoring Subsystem					
Refueling Floor Area Radiation Monitors	SC3	RB	N/A	Non-Seismic	
General Area Radiation Monitors	SC3	RB, CB, TB, RWB	N/A	Non-Seismic	

Notes:

1. Safety Class determination and methodology is discussed in Subsections 3.2.1 and 3.2.2.
2. Location Codes:
   a. PCS: Primary Containment System
   b. RB: Reactor Building
   c. TB: Turbine Building
   d. CB: Control Building
   e. RWB: Radwaste Building
   f. OO: Outdoors On-Site
   g. OL: Any Other Location
   h. YRD: Yard

i. ALL: All Locations

3. Quality group classifications is discussed in Subsection 3.2.4.
4. Seismic categories are discussed in Subsection 3.2.3.
5. Structures, systems and components required to be designed in accordance with Radioactive Waste Management requirements from RG 1.143 for Category RW-IIa, shall meet the guidance of RG 1.143, as applied to radioactive waste management systems, with regard to quality, seismic, and Quality Group requirements.
6. FSAR: Classification information is to be provided later in the BWRX-300 design process and updated in the FINAL Safety Analysis Report (FSAR).
7. The Radioactive Waste Management System components conform to RG 1.143 Table 1 when RG 1.143 is applicable. For radwaste processing systems, RG 1.143 Table 1 modifies RG 1.26 Table 1 Quality Group D. This modification is acceptable per Standard Review Plan 3.2.2 Table A-1 Note (h).
8. These components are SC3 or SCN items and classified as Internal Structures as defined in ASME BPVC, Section III. Division 1, “Rules for Construction of Nuclear Facility Components,” Subsection NG, “Core Support Structures,” NG-1122. These components are not ASME BPVC components, but the design complies with the applicable requirements of ASME BPVC, Section III.
9. Components classified as SC1 may be assigned to a Seismic Category lower than I in accordance with NEDC-33934 (Reference 3A-1).
10. RG 1.26 Quality Group is not applicable because the component does not perform a SC1 pressure boundary function nor part of systems or portions of systems that contain or may contain radioactive material.
11. The component is qualified for seismic interaction, if required for mitigation of seismic-related events as described in Subsections 7.3.2 (SC2) and 7.3.3 (SC3).
12. SSCs that are not required to remain functional during or after a seismic event but whose failure during a seismic event could adversely affect the ability of any Seismic Category I SSC to accomplish its Safety Category 1 function are classified as Seismic Category II. Consequently, SSCs listed in Table 3A-1 as “Non-Seismic” are to be classified Seismic Category II as required for seismic interaction and mitigation of seismic-related events pending the physical layout in the final design (FSAR).
13. Design changes are being developed to relocate SC1 SSCs to the Seismic Category I RB and to change the SC1 SSC equipment seismic classification to Seismic Category I. Table 3A-1 will be updated in the FSAR.
14. The Offsite Power System is not part of the Clinch River Nuclear Site and its SSCs are not classified. See Section 8.2 for a description of offsite power.

Table 3A-2: Building and Site Structure and Seismic Design Basis

Structure	US Seismic Category	Seismic Demand	Interaction Requirement
RB	I	SSE	Not Applicable
SCCV			
Containment Internal Structures			
Radwaste Building (RWB)	RW-IIa	One-Half SSE	Potential Interaction with Seismic Category I RB
TB	II[(1)]	IBC Seismic Demand	Potential Interaction with Seismic Category I RB
CB			
Service Building			
Reactor Auxiliary Structures			

Note:

1. Seismic Category II applies to select Non-Seismic civil structures that cannot be ruled out for interaction with Seismic Category 1 SSCs during a Safe Shutdown Earthquake (SSE) or extreme wind event.

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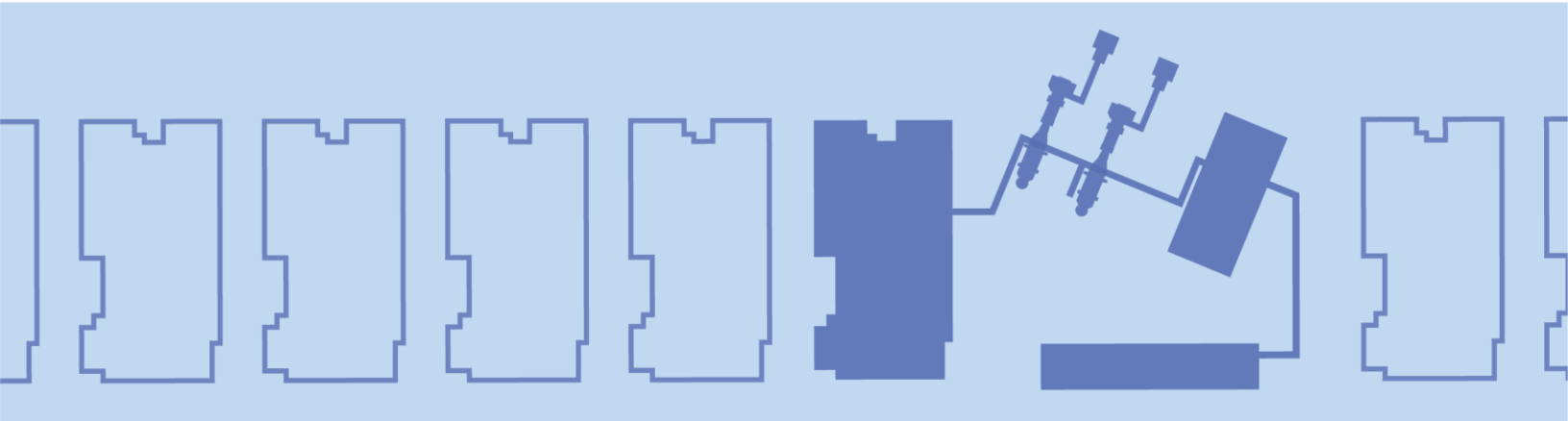
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Non-Proprietary Information

Chapter 3
Appendix 3B thru 3H



ACRONYM LIST

Acronym	Explanation
1D	One-Dimensional
2D	Two-Dimensional
3D	Three-Dimensional
ALWR	Advanced Light Water Reactor
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrences
ARI	Alternate Rod Insertion
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ATF	Acceleration Transfer Function
ATH	Acceleration Time History
AVR	Automatic Voltage Regulator
BE	Best Estimate
BECPV	Blue Energy Center – Port of Victoria
BIS	Boron Injection System
BWR	Boiling Water Reactor
CAD	Computer-Aided Design
CB	Control Building
CEPSS	Containment Equipment and Piping Support Structure
CIV	Containment Isolation Valve
CR	Control Room
CRD	Control Rod Drive
CST	Condensate Storage Tank
CUW	Reactor Water Cleanup System
CWS	Circulating Water System
DBA	Design Basis Accident
DCIS	Distributed Control and Information System
DEC	Design Extension Condition
DL2	Defense Line 2
DL3	Defense Line 3
DL4a	Defense Line 4a

Acronym	Explanation
DP-SC	Diaphragm Plate Steel-Plate Composite
EPRI	Electric Power Research Institute
FCU	Fan Coil Unit
FE	Finite Element
FIRS	Foundation Input Response Spectra
FMCRD	Fine Motion Control Rod Drive
FSAR	Final Safety Analysis Report
FW	Feedwater
GEH	GE Hitachi Nuclear Energy
GEZIP	General Electric Zinc Injection Passivation
GT	Gamma Thermometer
HCU	Hydraulic Control Unit
HRA	Human Reliability Analysis
HVS	Heating, Ventilation, and Cooling System
HWC	Hydrogen Water Chemistry
HX	Heat Exchanger
I&C	Instrumentation and Control
IC	Isolation Condenser
ICC	Isolation Condenser Pools Cooling and Cleanup System
ICS	Isolation Condenser System
ILRT	Integrated Leak Rate Test
ISRS	In-Structure Response Spectra
LB	Lower Bound
LMS	Lumped Mass Stick
LOCA	Loss-of-Coolant Accident
LV	Low Voltage
MAAP	Modular Accident Analysis Program
MCA	Main Condenser and Auxiliaries
MCC	Main Control Center
MCR	Main Control Room
MS	Main Steam
MSL	Main Steam Line

Acronym	Explanation
MSR	Moisture Separator Reheater System
MTE	Main Turbine Equipment
NBS	Nuclear Boiler System
NHS	Normal Heat Sink
NRV	Non-Return Valve
OLNC	On-Line NobleChem
PBIRS	Performance Based Intermediate Response Spectra
PBSRS	Performance Based Surface Response Spectra
PCV	Pressure Control Valve
PCW	Plant Cooling Water System
PIE	Postulated Initiating Event
PPS	Plant Pneumatics System
PRA	Probability Risk Assessment
QAPD	Quality Assurance Program Description
RB	Reactor Building
RCPB	Reactor Coolant Pressure Boundary
RG	Regulatory Guide
RIV	Reactor Isolation Valve
RPV	Reactor Pressure Vessel
RVT	Regulating Voltage Transformer
RWB	Radwaste Building
RWST	Refueling Water Storage Tank
SC	Safety Class
SC1	Safety Class 1
SC2	Safety Class 2
SC3	Safety Class 3
SCCV	Steel-Plate Composite Containment Vessel
SCR	Secondary Control Room
SDC	Shutdown Cooling System
SDG	Standby Diesel Generator
SEI	Structural Engineering Institute
SIR	Seismic Interface Restraint

Acronym	Explanation
SRSS	Square-Root-of-the-Sum-of-Squares
SSCs	Structures, Systems, and Components
SSE	Safe Shutdown Earthquake
SSI	Soil-Structure Interaction
SSSI	Structure-Soil-Structure Interaction
TB	Turbine Building
TBD	To Be Determined
TBV	Turbine Bypass Valve
TRACG	Transient Reactor Analysis Code General Electric
UB	Upper Bound
UPS	Uninterruptible Power Supply
USNRC	U.S. Nuclear Regulatory Commission
WRNM	Wide Range Neutron Monitor
ZPA	Zero Period Acceleration

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None.

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None.

APPENDIX 3B – INTEGRATED REACTOR BUILDING FINITE ELEMENT MODE

[Reserved. To be addressed in CPA Part 2.]

APPENDIX 3C – SEISMIC SOIL STRUCTURE INTERACTION ANALYSIS RESULTS

[Reserved. To be addressed in CPA Part 2.]

APPENDIX 3D – INTERACTION EVALUATIONS RESULTS

3D.1 INTRODUCTION

Interaction evaluations of the Seismic Category II and RW-IIa Power Block structures and foundations with the Seismic Category I structures and foundations under extreme loading will be provided in the FSAR.

This appendix provides interaction evaluations results of the Seismic Category II and RW-IIa Power Block structures and foundations with the Seismic Category I structures and foundations under extreme loading.

Implementation of the methodology for these evaluations discussed in Section 3.3 and Section 3.7 requires the design of the surrounding Seismic Category II and RW-IIa Power Block structures and foundations to be completed to provide quantifiable results, and, thus, the interaction evaluations will be provided to support the FSAR.

APPENDIX 3E – RESPONSE TO STATIC LOADS

[Reserved. To be addressed in CPA Part 2.]

APPENDIX 3F – DESIGN DETAILS AND EVALUATION RESULTS FOR THE CONTAINMENT

[Reserved. To be addressed in CPA Part 2.]

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[Reserved. To be addressed in CPA Part 2

APPENDIX 3H – DESIGN DETAILS AND EVALUATION RESULTS FOR THE REACTOR BUILDING STRUCTURE

[Reserved. To be addressed in CPA Part 2.]

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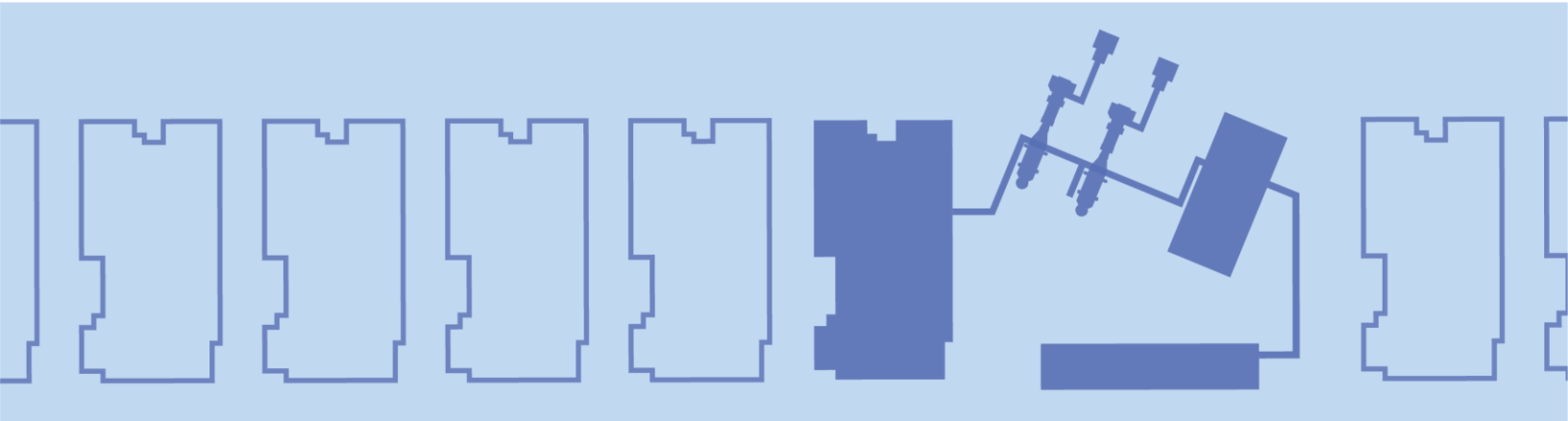
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Non-Proprietary Information

Chapter 3
Appendix 3I - 3M



ACRONYM LIST

Acronym	Explanation
1D	One-Dimensional
2D	Two-Dimensional
3D	Three-Dimensional
ALWR	Advanced Light Water Reactor
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrences
ARI	Alternate Rod Insertion
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ATF	Acceleration Transfer Function
ATH	Acceleration Time History
AVR	Automatic Voltage Regulator
BE	Best Estimate
BIS	Boron Injection System
BWR	Boiling Water Reactor
CAD	Computer-Aided Design
CB	Control Building
CEPSS	Containment Equipment and Piping Support Structure
CIV	Containment Isolation Valve
CR	Control Room
CRD	Control Rod Drive
CST	Condensate Storage Tank
CUW	Reactor Water Cleanup System
CWS	Circulating Water System
DBA	Design Basis Accident
DCIS	Distributed Control and Information System
DEC	Design Extension Condition
DL2	Defense Line 2
DL3	Defense Line 3
DL4a	Defense Line 4a
DP-SC	Diaphragm Plate Steel-Plate Composite

Acronym	Explanation
EPRI	Electric Power Research Institute
FCU	Fan Coil Unit
FE	Finite Element
FIRS	Foundation Input Response Spectra
FMCRD	Fine Motion Control Rod Drive
FSAR	Final Safety Analysis Report
FW	Feedwater
GEH	GE Hitachi Nuclear Energy
GEZIP	General Electric Zinc Injection Passivation
GT	Gamma Thermometer
HCU	Hydraulic Control Unit
HRA	Human Reliability Analysis
HVS	Heating, Ventilation, and Cooling System
HWC	Hydrogen Water Chemistry
HX	Heat Exchanger
I&C	Instrumentation and Control
IC	Isolation Condenser
ICC	Isolation Condenser Pools Cooling and Cleanup System
ICS	Isolation Condenser System
ILRT	Integrated Leak Rate Test
ISRS	In-Structure Response Spectra
LB	Lower Bound
LMS	Lumped Mass Stick
LOCA	Loss-of-Coolant Accident
LV	Low Voltage
MAAP	Modular Accident Analysis Program
MCA	Main Condenser and Auxiliaries
MCC	Main Control Center
MCR	Main Control Room
MS	Main Steam
MSL	Main Steam Line
MSR	Moisture Separator Reheater System

Acronym	Explanation
MTE	Main Turbine Equipment
NBS	Nuclear Boiler System
NHS	Normal Heat Sink
NRV	Non-Return Valve
OLNC	On-Line NobleChem
PBIRS	Performance Based Intermediate Response Spectra
PBSRS	Performance Based Surface Response Spectra
PCV	Pressure Control Valve
PCW	Plant Cooling Water System
PIE	Postulated Initiating Event
PPS	Plant Pneumatics System
PRA	Probability Risk Assessment
QAPD	Quality Assurance Program Description
RB	Reactor Building
RCPB	Reactor Coolant Pressure Boundary
RG	Regulatory Guide
RIV	Reactor Isolation Valve
RPV	Reactor Pressure Vessel
RVT	Regulating Voltage Transformer
RWB	Radwaste Building
RWST	Refueling Water Storage Tank
SC	Safety Class
SC1	Safety Class 1
SC2	Safety Class 2
SC3	Safety Class 3
SCCV	Steel-Plate Composite Containment Vessel
SCR	Secondary Control Room
SDC	Shutdown Cooling System
SDG	Standby Diesel Generator
SEI	Structural Engineering Institute
SIR	Seismic Interface Restraint
SRSS	Square-Root-of-the-Sum-of-Squares

Acronym	Explanation
SSCs	Structures, Systems, and Components
SSE	Safe Shutdown Earthquake
SSI	Soil-Structure Interaction
SSSI	Structure-Soil-Structure Interaction
TB	Turbine Building
TBD	To Be Determined
TBV	Turbine Bypass Valve
TRACG	Transient Reactor Analysis Code General Electric
UB	Upper Bound
UPS	Uninterruptible Power Supply
USNRC	U.S. Nuclear Regulatory Commission
WRNM	Wide Range Neutron Monitor
ZPA	Zero Period Acceleration

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APPENDIX 3I – COMPUTER PROGRAMS USED IN THE DESIGN AND ANALYSIS OF SEISMIC CATEGORY STRUCTURES

3I.1 INTRODUCTION

This appendix describes the major computer programs used in the design and analysis of the BWRX-300 seismic category structures. The programs are verified for their application by appropriate methods, such as hand calculations, or comparison with results from similar programs, experimental tests, or published literature, including analytical results or numerical results to the benchmark problems. The computer codes used for design and safety analysis are qualified in accordance with the GVH Quality Assurance Program.

3I.1.1 ACS SASSI v4

Description: ACS SASSI is a finite element computer code on the Microsoft Windows PC platforms for performing Three-Dimensional (3D) dynamic Soil-Structure Interaction (SSI) analysis to analyze the effect of seismic ground motions on structures. The analysis is performed in the frequency domain using linear or equivalent-linear material properties for the structure and soil.

Validation: The software is approved for production use per engineering software for design and analysis software procedure.

Summary: ACS SASSI is used to perform seismic and static SSI and Structure-Soil-Structure Interaction (SSSI) analyses, as applicable.

3I.1.2 ANSYS v17

Description: ANSYS is a Multiphysics computer program. ANSYS is a general-purpose large-scale finite element analysis computer program and has interactive capabilities. Finite element analysis is a numerical method for analyzing structure, thermal, fluid flow and other physical problems. The analysis method is based on displacement formulation of the finite element method. Typical applications include finding stress, deformation, thermal analysis, and modal analysis with user inputs of geometrical dimensions, element type, material properties, boundary conditions, and loadings.

Validation: The software is approved for production use per engineering software for design and analysis software procedure.

Summary: This program is used to model the structure and the hydrodynamics within the Boiling Water Reactor (BWR) and perform structural analysis for applicable loads.

3I.1.3 LS-DYNA R13.1.0

Description: LS-DYNA is an explicit simulation program capable of simulating the response of materials to short periods of severe loading. Its many elements, contact formulations, material models and other controls can be used to simulate complex models with control over all the details of the problem. LS-DYNA applications include explosion/penetration, impact analysis, and non-linear explicit structural analysis.

Validation: This software is not approved for production use per engineering software for design and analysis software procedure and requires output verification in accordance with the design process.

Summary: LS-DYNA analyzes structures for effects of blast loading, thermal profile, and aircraft impact.

3I.1.4 SSI_StressCoordTrans01P

Description: The SSI_StressCoordTrans01P program is an auxiliary program to post-process the ACS SASSI NQA V4 STRESS result binary database. The SSI_StressCoordTrans01P program includes an ensemble of STRESS database processing functionalities which were customized for application to the BWRX-300 seismic SSI analysis projects. The SSI_StressCoordTrans01P customized program is based on specific implementations incorporated in the ACS SASSI NQA V4 User Interface capabilities, such as the CTVEC and the CTCCV commands, and existing STRESS binary database verification tools used in-house during the development over years of the STRESS module.

Validation: The software is approved for production use per engineering software for design and analysis software procedure.

Summary: This SSI_StressCoordTrans01P program is used for post-processing the ACS SASSI STRESS binary databases for Integrated RB Walls and Floors in batch mode.

APPENDIX 3J – COMPUTER PROGRAMS USED IN THE DESIGN AND ANALYSIS OF COMPONENTS, SYSTEMS AND STRUCTURES – MECHANICAL SYSTEMS AND COMPONENTS

3J.1 INTRODUCTION

This appendix describes the major computer programs used in the design and analysis of the BWRX-300 safety class components, equipment, and structures. The programs are verified for their application by appropriate methods, such as hand calculations, or comparison with results from similar programs, experimental tests, or published literature, including analytical results or numerical results to the benchmark problems. The computer codes used for design and safety analysis are qualified in accordance with the GVH Quality Assurance Program.

3J.1.1 ANSYS v17

Description: ANSYS is a general-purpose large-scale finite element analysis computer program and has interactive capabilities. Finite element analysis is a numerical method for analyzing structure, thermal, fluid flow and other physical problems. The analysis method is based on displacement formulation of the finite element method. Typical applications include finding stress, deformation, thermal analysis, and modal analysis with user inputs of geometrical dimensions, element type, material properties, boundary conditions, and loadings.

Validation: The software is approved for production use per engineering software for design and analysis software procedure.

Summary: This program is used to model the structure and the hydrodynamics within the BWR and perform structural analysis for applicable loads.

3J.1.2 PBLE v1

Description: PBLE is used for performing the structural analysis of steam dryers.

Validation: The software is approved for production use per engineering software for design and analysis software procedure.

Summary: PBLE calculates the acoustic loads on a steam dryer based on measurements of pressure along the MSLs or pressures measured directly on the face of the steam dryer. The loads are then used in a finite element model to calculate the stresses in the dryer.

3J.1.3 SIMCENTER 3D Acoustics v2022

Description: SIMCENTER 3D is used for modeling dryer acoustic loads and instrumentation diagnostics. SIMCENTER 3D is a unified, scalable, open, and extensible environment for 3D computer-aided engineering with connections to design, One-Dimensional (1D) simulation, test, and data management. Fast and accurate solvers power structural, acoustics, flow, thermal, motion, and composites analyses, as well as optimization and multi-physics simulation.

Validation: The software is not qualified under the engineering software process, and the output of the software is verified with each use per the design process.

Summary: SIMCENTER Finite elements acoustic software models and calculates acoustic wave propagation in fluid mediums.

3J.1.4 MATHCAD v15

Description: MATHCAD is engineering math software that performs, analyzes, and shares calculations.

Validation: This software is not approved for production use per engineering software for design and analysis software procedure and requires output verification in accordance with the design process.

Summary: MATHCAD is used to run calculations on various mechanical components.

3J.1.5 MATLAB v2022

Description: MATLAB is a proprietary multi-paradigm programming language and numeric computer environment.

Validation: This software is not approved for production use per engineering software for design and analysis software procedure and requires output verification in accordance with the design process.

Summary: MATLAB is used for various calculations for mechanical components.

3J.1.6 OCTAVE v8

Description: Octave is a high-level programming language primarily intended for scientific computing and numerical computation.

Validation: This software is not approved for production use per engineering software for design and analysis software procedure and requires output verification in accordance with the design process.

Summary: Octave solves linear and non-linear problems/experiments.

3J.1.7 SEISM v5

Description: The SEISM program is used for the non-linear response prediction of structural system with spring, damper, friction and stop element, under dynamic loads. The program employs the component element method and can account for impact and friction forces effect. SEISM program performs calculations in double precision.

Validation: The software is approved for production use per engineering software for design and analysis software procedure.

Summary: SEISM is used for the non-linear time history response prediction of structural systems with spring, damper, friction and stop elements under dynamic loads. The program employs the component element method and can account for impact and friction force effects. When running SEISM, the user can select to run any of its four modules (CRTFI, SEPRE, SEISM, SEPST) individually or combined within a single session. Output of one module may be passed to and used as input to the next module.

3J.1.8 HyperMesh

Description: HyperMesh is a multi-disciplinary finite element pre-processor which manages the generation of the largest, most complex models, starting with the import of a Computer-Aided Design (CAD) geometry to exporting a ready-to-run solver file. With its advanced geometry and meshing capabilities, HyperMesh provides an environment for rapid model generation.

Validation: This software is not approved for production use per engineering software for design and analysis software procedure and requires output verification in accordance with the design process.

Summary: HyperMesh is a tool which is used to generate mechanical models for complicated mechanical components. This tool serves as a pre-processor to build mesh models, and no calculations are performed with Hypermesh.

3J.1.9 ERSIN v3

Description: ERSIN is a piping analysis software to generate secondary response spectra.

Validation: The software is approved for production use per engineering software for design and analysis software procedure.

Summary: ERSIN is used to generate secondary response spectra for pipe and floor mounted equipment. Example applications include control panels, equipment racks, Main Steam Isolation Valves (MSIVs), and Hydraulic Control Units. ERSIN software has three input options: 1) card decks, 2) SAP software decks, and 3) PISYS software decks. ERSIN can be used with SAP version 4G07P and PISYS version 08P structure/piping models.

3J.1.10 PYTHON

Description: Python is a high-level, general purpose programming language. Its design philosophy emphasizes code readability with the use of significant indentation.

Validation: This software is not approved for production use per engineering software for design and analysis software procedure and requires output verification in accordance with the design process.

Summary: Python supports multiple programming paradigms, including structures, object-oriented and functional programming.

APPENDIX 3K – COMPUTER PROGRAMS USED IN THE DESIGN AND ANALYSIS OF COMPONENTS, SYSTEMS, AND STRUCTURES – NUCLEAR FUELS

3K.1 INTRODUCTION

This appendix describes the major computer programs used in the design and analysis of the BWRX-300 safety class components, equipment, and structures. The programs are verified for their application by appropriate methods, such as hand calculations, or comparison with results from similar programs, experimental tests, or published literature, including analytical results or numerical results to the benchmark problems. The computer codes used for design and safety analysis are qualified in accordance with the GVH Quality Assurance Program.

3K.1.1 PANAC v11

Description: PANAC is the computer program used for the detailed nuclear calculations of the BWR Core. It is a steady-state, 3D, one and one-half energy group, diffusion theory computer program with coupled nuclear and thermal-hydraulic representation of the reactor core.

Validation: This software is approved for production use per engineering software for design and analysis software procedure.

Summary: The BWR Core Simulator (PANAC) is a steady-state, 3D coupled nuclear-thermal hydraulic computer program representing a BWR core. An automated plant heat balance option is used for modeling of the external flow loop. Provisions are made for fuel cycle and thermal limits calculations. The program is used for detailed 3D design and operational calculations of BWR neutron flux and power distributions and thermal performance as a function of control rod position, refueling pattern, coolant flow, reactor pressure, and other operational and design variables. A special power exposure iteration option is available for target exposure distribution and cycle length predictions. PANAC includes the effect of Doppler broadening as a function of moderator density, exposure, control, and moderator density history for a given fuel type. The nuclear model is based on coarse-mesh nodal, improved 1-1/2 group (quasi-two group), static diffusion theory. The diffusion equations are solved using the fast energy group. Resonance energy neutronic effects are included in the model by relating the resonance fluxes to the fast energy flux. The thermal flux is represented by an asymptotic expansion using a slowing down source from the epithermal region. A spectral history reactivity model and control blade history reactivity model are included. Control blade history local peaking effects are also incorporated in the nuclear model. A pin power reconstruction model is implemented to account for the effect of flux gradients across the nodes on the local peaking distribution. Neutronic parameters used by PANAC are obtained from the Two-Dimensional (2D) lattice physics code (TGBLA) and parametrically fitted as a function of moderator density, exposure, control, and moderator density history for a given fuel type.

3K.1.2 PRIME v3

Description: The PRIME computer program is used to calculate the thermal/mechanical response of nuclear fuel to time varying power histories.

Validation: The software is approved for production use per engineering software for design and analysis software procedure.

Summary: PRIME is used for steady-state and transient licensing analysis of UO_2 and (U, Gd)O_2 fuel with (and without) additive material. PRIME is used for steady-state and transient licensing analysis as well as qualification cases of recrystallized annealed zircaloy-2 cladding. Additionally,

PRIME may be used with stress-relieved annealed zircaloy-4 cladding of either 70 percent or 30 percent cold work for qualification cases.

3K.1.3 TGBLA v6

Description: Calculates lattice parameters for fuel bundles and the output is used by PANAC to model the behavior of the fuel in the core.

Validation: The software is approved for production use per engineering software for design and analysis software procedure.

Summary: TGBLA is a lattice design computer program for BWRs, which have the following lattices: 7x7, 8x8, 9x9, or 10x10. Water rods, including large central water rods and approximations for centered and offset water boxes, may be introduced into cells of the 2D mesh, which TGBLA solves. The 8x8 lattice can have up to four cells per water rod; the 9x9 lattice can have up to 3.5 cells per water rod; the 10x10 lattice can have up to four cells per water rod. Lattices with vanishing rods, thick-thin channels, or some water cross designs such as 8x8 and 10x10 water cross lattices, are qualified. TGBLA is qualified for water box designs where the water box is simulated by the use of nine water rods. Although TGBLA is capable of analyzing 11x11 and 12x12 lattices, MOX fuel and other design configurations, it has not been qualified for them. TGBLA solves 2D diffusion equations with diffusion parameters corrected by transport theory to provide the multiplication factor, the fission density distribution, the neutron balance, and the homogenized cross sections. Also, TGBLA performs burnup calculations for generating input to the BWR 3D simulator. In addition, TGBLA generates the rod-by-rod neutron cross sections, gamma smeared power distributions and flux discontinuity factors. The ring-by-ring gamma source distribution in gadolinium rods should not be used.

3K.1.4 TRACG v4

Description: Transient Reactor Analysis Code General Electric (TRACG) is a GVH version of the Transient Reactor Analysis Code representing a best-estimate code for the analysis of BWR transients. It is based on a multi-dimensional two-fluid model for the reactor thermal hydraulics and a 3D neutron kinetics model.

Validation: The software is approved for production use per engineering software for design and analysis software procedure.

Summary: TRACG is a computer program applicable for the calculation of thermal-hydraulic parameters and reactor power during BWR transients. TRACG is intended to be used as a 'best-estimate' computer code, with capabilities for 3D hydrodynamic calculations in the vessel components, and 1D calculations in the other components. A full two-fluid representation supplemented by air and boron models is employed for the characterization of two-phase flow, allowing application to transients where thermal non-equilibrium and counter-current flow between phases is significant. TRACG has point, 1D, and 3D neutron kinetics models for simulating the feedback effects of moderator density, fuel temperature, boron, and control blade movement on the core power. TRACG has a control system model capable of simulating the BWR feedback control system. TRACG is capable of modeling standard BWR fuels and advanced fuel designs including part length fuel rods and large water rods. In addition to modeling the BWR, TRACG is also applicable to experimental test facilities constructed from components representative of a BWR.

3K.1.5 SEISM v5

Description: The SEISM program is used for the non-linear response prediction of structural systems with spring, damper, friction, and stop elements, under dynamic loads. The program

employs the component elements method and can account for impact and friction forces effects. The SEISM program performs calculations in double precision.

Validation: This software is approved for production use per engineering software for design and analysis software procedure.

Summary: SEISM is used for the non-linear time history response prediction of structural systems with spring, damper, friction, and stop elements under dynamic loads. The program employs the component element method and can account for impact and friction force effects. When running SEISM, the user can select to run any of its four modules individually or combined within a single session. Output of one module may be passed to and used as input to the next module.

APPENDIX 3L – COMPUTER PROGRAMS USED IN THE DESIGN AND ANALYSIS OF COMPONENTS, SYSTEMS, AND STRUCTURES - ENVIRONMENTAL AND RADIOLOGICAL

3L.1 INTRODUCTION

This appendix describes the major computer programs used in the design and analysis of the BWRX-300 safety class components, equipment, and structures. The programs are verified for their application by appropriate methods, such as hand calculations, or comparison with results from similar programs, experimental tests, or published literature, including analytical results or numerical results to the benchmark problems. The computer codes used for design and safety analysis are qualified in accordance with the GVH Quality Assurance Program.

3L.1.1 GALE-BWR v3.2

Description: The Gaseous and Liquid Effluents (GALE-BWR) consists of codes that calculate the gaseous and liquid effluent releases from BWRs.

Validation: This software is not approved for production use per engineering software for design and analysis software procedure and requires output verification in accordance with the design process.

Summary: GALE-BWR uses a combination of input values and hardwired parameters to calculate the source term of radionuclides generated by a nuclear power plant during routine operation. Parameters that vary from plant to plant are treated as input values. GALE-BWR prompts the operator for input values on each run. Hardwired parameters are plant characteristics that are assumed to be the same for all reactors.

APPENDIX 3M – COMPUTER PROGRAMS USED IN THE DESIGN AND ANALYSIS OF COMPONENTS, SYSTEMS, AND STRUCTURES IN SAFETY ANALYSES (PROBABILITY RISK ASSESSMENT AND DETERMINISTIC)

3M.1 INTRODUCTION

This appendix describes the major computer programs used in the design and analysis of the BWRX-300 safety class components, equipment, and structures. The programs are verified for their application by appropriate methods, such as hand calculations, or comparison with results from similar programs, experimental tests, or published literature, including analytical results or numerical results to the benchmark problems. The computer codes used for design and safety analysis are qualified in accordance with the QAPD described in Chapter 17.

3M.1.1 ACUBE v11

Description: This is an advanced cut set upper bound estimator (ACUBE).

Validation: The software is not qualified under the engineering software process, and the output of the software is verified with each use per the design process.

Summary: The BWRX-300 project will use ACUBE to post-process result cut sets using a binary decision diagram method which will provide a more accurate point estimate of the results. ACUBE is a post-processing software that analyzes minimal cut sets and returns an estimate of the probability for a given top event using the binary decision diagram method. The binary decision diagram method is a more accurate estimation than the approximation calculations used in baseline results. The software can be used with manual inputs but typically is used with intermediate quantification software such as FRANX or PRAQuant.

3M.1.2 CAFTA v11

Description: Computer Aided Fault Tree Analysis (CAFTA) is an integrated tool to perform Probabilistic Risk Analysis, incorporating linking event tree/fault tree methodology.

Validation: The software is not qualified under the engineering software process, and the output of the software is verified with each use per the design process.

Summary: The CAFTA software will be qualified to complete all designed functions within the software. Note that the testing will not cover every possible variation or combination of use for the software, but it will validate that the software operates as intended for within the standard operating configuration of the software.

3M.1.3 FRANX v11

Description: FRANX is used in development of Probabilistic Risk Assessment (PRA) Hazards models (e.g., Fire, Flood, High Winds, Seismic).

Validation: The software is not qualified under the engineering software process, and the output of the software is verified with each use per the design process.

Summary: FRANX is used in the development of the Internal Fire, Internal Flood, Seismic, and High Winds hazard analyses. Specifically, FRANX will be used to build hazard specific scenarios and generate one-top models for later combination into an integrated hazard model. The FRANX software is a tool for analyzing external event risk. This tool is used to manage and develop the scenarios, calculate the probabilistic effect on core damage, and generate solution models.

3M.1.4 FTREX v2.0

Description: FTREX reads a fault tree that consists of Boolean equations for system failure and generates cut sets that are minimal combinations of component failures that cause system failure.

Validation: The software is not qualified under the engineering software process, and the output of the software is verified with each use per the design process.

Summary: This software will have all functionality qualified and be valid for use with the necessary interfacing software (e.g., FRANX, CAFTA, PRAQuant) or independently of that software. The software must be accessible from the interfacing software locations as well as have permission to read and write files to a temporary directory and a defined output file directory.

3M.1.5 HRA Calculator

Description: The Human Reliability Analysis (HRA) Calculator supports development of PRA human reliability analyses.

Validation: The software is not qualified under the engineering software process, and the output of the software is verified with each use per the design process.

Summary: The HRA Calculator is used to develop the HRA, calculate the human error probabilities, and develop a dependency analysis for the credited operator actions. The HRA Calculator provides a step-by-step process for developing the HRA applying one of the following methods: CBDTM, HCR/ORE, ASEP, SPAR-H, THERP.

3M.1.6 MAAP v5.06

Description: The Modular Accident Analysis Program (MAAP) Version 5.06, an Electric Power Research Institute (EPRI) owned and licensed computer software, is a fast-running computer code that simulates the response of light water and heavy water moderated nuclear power plants for both current and Advanced Light Water Reactor (ALWR) designs. It can simulate LOCA and non-LOCA transients for PRA applications as well as severe accident sequences, including actions taken as part of the Severe Accident Management Guidelines.

Validation: The software is approved for production use per engineering software process for design and analysis software procedure.

Summary: MAAP is used to analyze reactor thermal-hydraulic and containment response to transients and severe accident sequence progressions. MAAP is used to predict the timing of key events, evaluate the influence of mitigative systems, evaluate effectiveness of operator actions, predict magnitude and timing of fission product releases, and investigate uncertainties in severe accident phenomena.

3M.1.7 PRAQuant v11

Description: PRAQuant is used in accident sequence quantification. In performing a fault tree-based analysis, it is often necessary to solve the fault tree several times, using different subtrees, boundary conditions, truncations, or other assumptions about the model. These solutions can be performed manually in the CAFTA software, but it is often difficult to track and document the numerous results. PRAQuant is a general tool to configure several fault tree analysis solutions in advance, and to track the completion and results from each run.

Validation: The software is not qualified under the engineering software process, and the output of the software is verified with each use per the design process.

Summary: PRAQuant is used in the processing of the combined hazard model to generate a combined hazard cut set output. PRAQuant is a processing software used to configure several fault tree analysis solutions and track the completion and results from each run. The software is capable of defining specific criteria to be applied in each fault tree analysis solution (e.g., flag files, recovery rules, output file name, truncation) and processes the supplied inputs into a format that a quantification engine (e.g., FTREX) is capable of processing. Once the quantification

engine generates an output cut set file, the software can interface with QRecover to apply recovery rules before saving the final output to a defined directory.

3M.1.8 SysImp v11

Description: SysImp is a software tool used to calculate the importance of basic events, or collections of those events, in a risk model. SysImp is designed for risk models where components, equipment trains, and systems are represented by groups of basic events.

Validation: The software is approved for production use per engineering software for design and analysis software procedure.

Summary: SysImp is used to perform risk importance sensitivities, calculations, and grouping system importance. SysImp allows for deriving insights from risk importance rankings, estimating total plant risk given a specific change, and collective risk importance measures.

3M.1.9 UNCERT v11

Description: This is a PRA Uncertainty Propagation tool (UNCERT). UNCERT can read the cut set or sequence data created from CAFTA and calculate the uncertainty of the cut set result.

Validation: The software is approved for production use per engineering software for design and analysis software procedure.

Summary: UNCERT is used to perform the parametric uncertainty calculations on the output cut sets. The UNCERT software will take a defined input (e.g., cut set file and associated CAFTA database) and perform the uncertainty analysis utilizing either a Monte Carlo or Latin Hypercube sampling method. The output will calculate the metrics for the cut set using that defined method.

3M.1.10 TRACG v4

Description: TRACG is a GVH version of the Transient Reactor Analysis Code representing a best-estimate code for the analysis of BWR transients. It is based on a multi-dimensional two-fluid model for the reactor thermal hydraulics and a 3D neutron kinetics model.

Validation: The software is approved for production use per engineering software for design and analysis software procedure.

Summary: TRACG is a computer program applicable for the calculation of thermal-hydraulic parameters and reactor power during BWR transients. TRACG is intended to be used as a 'best-estimate' computer code, with capabilities for 3D hydrodynamic calculations in the vessel components, and 1D calculations in the other components. A full two-fluid representation supplemented by air and boron models is employed for the characterization of two-phase flow, allowing application to transients where thermal non-equilibrium and counter-current flow between phases is significant. TRACG has point, 1D, and 3D neutron kinetics models for simulating the feedback effects of moderator density, fuel temperature, boron, and control blade movement on the core power. TRACG has a control system model capable of simulating the BWR feedback control system. TRACG is capable of modeling standard BWR fuels and advanced fuel designs including part length fuel rods and large water rods. In addition to modeling the BWR, TRACG is also applicable to experimental test facilities constructed from components representative of a BWR.

3M.1.11 GOTHIC v8

Description: GOTHIC is used for design, licensing, safety and operating analysis of nuclear power plant containments, confinement buildings, and system components.

Validation: The software is approved for production use per engineering software for design and analysis software procedure.

Summary: GOTHIC is used to perform a sensitivity analysis for the passive containment cooling system while developing the design.

3M.1.12 RADTRAD

Description: RADTRAD uses a combination of tables and numerical models of source term reduction phenomena to determine the time-dependent dose at user-specified locations for a given accident scenario. It also provides the inventory, decay chain, and dose conversion factor tables needed for the dose calculation.

Validation: The software is approved for production use per engineering software for design and analysis software procedure.

Summary: The RADTRAD code is used for calculating accident doses, calculating transport of fission products inside the plant after an accident, and performing filter loading calculations for post-accident.

3M.1.13 ORIGEN2 v2.1

Description: ORIGEN is a one-group depletion and radioactive decay computer code. ORIGEN is used to calculate the radionuclide composition and other related properties of nuclear materials (irradiated fuel isotope inventory).

Validation: The software is approved for production use per engineering software for design and analysis software procedure.

Summary: ORIGEN is used for calculating core inventories of isotopes, and sometimes for performing activation analyses of various materials or components.

3M.1.14 PAVAN

Description: The PAVAN code calculations incorporate the results of several field tracer tests and diffusion experiments for elevated and ground-level releases from various locations on reactor facility buildings during stable atmospheric conditions with low wind speeds.

Validation: The software is approved for production use per engineering software for design and analysis software procedure.

Summary: PAVAN is an atmospheric-dispersion program for evaluating design-basis accidental releases of radioactive materials from nuclear power stations.

3M.1.15 ARCON

Description: ARCON is a computer code used to calculate atmospheric relative concentrations (X/Q) in support of control room habitability assessments required by 10 CFR Part 50, Appendix A, General Design Criterion 19.

Validation: The software is approved for production use per engineering software for design and analysis software procedure.

Summary: ARCON can account for both plume meander under low wind speed conditions and the plume dispersion due to building wakes.

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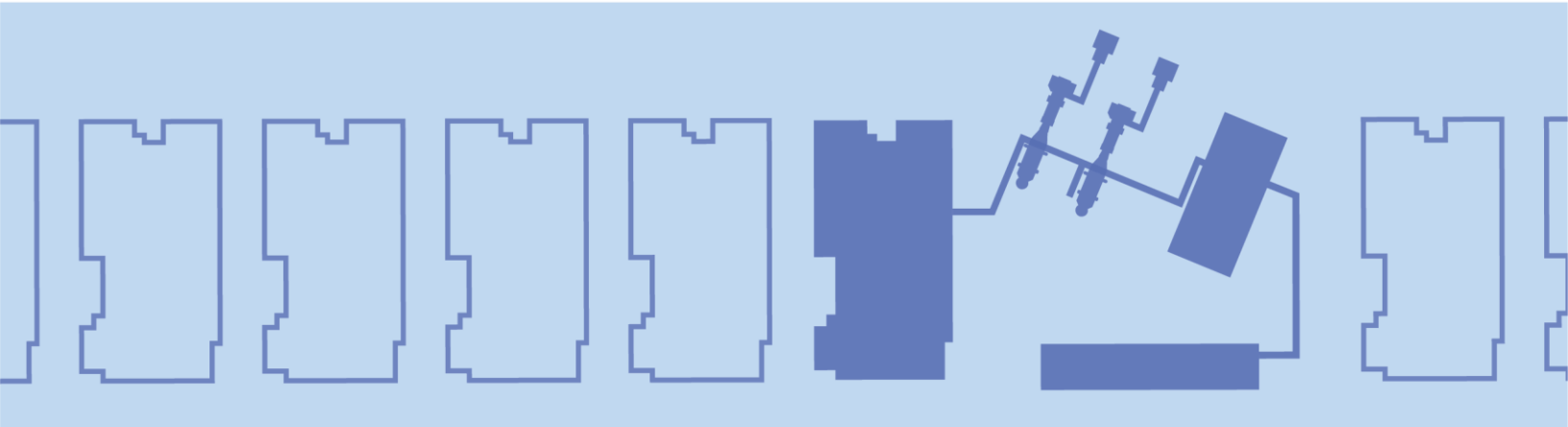
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Chapter 3
Appendix 3N



ACRONYM LIST

Acronym	Explanation
BE	Blue Energy
CFR	Code of Federal Regulation
CRD	Control Rod Drive
DEC	Design Extension Condition
GVH	GE Vernova Hitachi Nuclear Energy
ICS	Isolation Condenser System
PCCS	Passive Containment Cooling System
RB	Reactor Building
RG	Regulatory Guide
RIV	Reactor Isolation Valve
SCCV	Steel-Plate Composite Containment Vessel
SSCs	Structures, Systems, and Components

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APPENDIX 3N – AIRCRAFT IMPACT ASSESSMENT

3N.1 INTRODUCTION AND BACKGROUND

A design-specific assessment of the impact of a large commercial aircraft on the BWRX-300 is performed in accordance with 10 Code of Federal Regulations (CFR) 50.150(a) with the results provided herein in accordance with 10 CFR 50.150(b) and 10 CFR 50.34(a)(13). The aircraft impact assessment threat is a Design Extension Condition (DEC) that is performed using realistic analysis to demonstrate that, if a BWRX-300 is struck by a large commercial aircraft, design features and functional capabilities exist to ensure that necessary functions are maintained. This appendix describes the design features and functional capabilities, with reduced use of operator actions, of the BWRX-300 identified in the detailed assessment that assures these acceptance criteria are met:

- The reactor core remains cooled or the containment remains intact
- Spent fuel cooling or fuel pool integrity is maintained

The assessment is performed in accordance with the guidance for assessing effects for aircraft impact described in NEI 07-13, “Methodology for Performing Aircraft Impact Assessments for New Plant Designs,” (Reference 3N-1), as endorsed by Regulatory Guide (RG) 1.217.

3N.2 SCOPE OF THE ASSESSMENT

The evaluation of plant damage caused by the impact of a large, commercial aircraft is a complex problem involving phenomena associated with structural damage resulting from the initial impact, shock induced vibration, and the effects of aviation fuel fed fires. Analyses performed as part of this assessment include:

- Physical damage resulting from the impact of the aircraft fuselage and wing structure penetration of hardened aircraft components, such as engine rotor shafts and landing gear in accordance with Section 2 of NEI 07-13
- A shock-induced vibration analysis to evaluate functional effects on Structures, Systems, and Components (SSCs) required to support core cooling and fuel pool cooling based on the methodology in Section 3 of NEI 07-13
- An evaluation of damage from fuel fed fires using the guidance of Section 3 of NEI 07-13

The assessment considers numerous locations where an aircraft could potentially strike relevant BWRX-300 structures.

3N.3 ASSESSMENT METHODOLOGY

Structures of concern are those structures that contain SSCs necessary to ensure containment isolation and core cooling after reactor shutdown. For the BWRX-300 design, all of these SSCs are housed within the integrated Reactor Building (RB). Below grade portions of the integrated RB are not susceptible to a direct impact by an aircraft. There are no adjacent structures or buildings credited as intervening structures for the aircraft impact assessment.

A local damage analysis is conducted of the integrated RB structure based upon the methodology in Section 2 of NEI 07-13. If a missile does not perforate the exterior barrier of the above-grade RB structure and therefore has no residual velocity, then internal structures and components (e.g., the Steel-Plate Composite Vessel (SCCV), fuel pool, and isolation condensers) do not need to be assessed for potential local damage effects. For the local damage assessment, the thicknesses of the above-grade outer RB wall and RB roof are critical parameters and are evaluated using empirical equations. The BWRX-300 integrated RB is constructed of steel-plate composite modules as discussed in Subsection 3.8.4.1.1. Note that in the local damage evaluation of the

above-grade outer RB wall, an adjustment was made in the local damage analysis to Equation 2-3 in NEI 07-13 by applying a factor of 0.7 to adapt the guidance to account for the reduced thickness required for steel-plate composite modules compared to reinforced concrete for the same perforation performance.

A global analysis evaluates the global effects on the integrated RB structure, such as structural collapse, and demonstrates how the plant satisfies the acceptance criteria listed in Section 3N.1 based on the methodology in Section 2 of NEI 07-13.

Once damage from the initial impact is determined, the methods described in Section 3 of NEI 07-13 are followed to assess effects on SSCs credited in mitigating the event to assure that the required SSCs remain functional despite secondary physical effects of the aircraft impact on core cooling and fuel pool cooling capability of the existing design.

The shock damage distances are calculated in accordance with the guidance of NEI 07-13 to evaluate functional effects on SSCs required to maintain core cooling or fuel pool cooling. The evaluation also determines if sufficient systems remain functional despite vibration effects of the aircraft impact on core cooling capability of the existing design.

The BWRX-300 includes design features to prevent physical or fire damage internal to the RB due to the introduction of jet fuel through the RB openings such as ventilation and isolation condenser vents.

3N.4 DESIGN FEATURES AND FUNCTIONAL CAPABILITIES

The following key design features and functional capabilities ensure that the BWRX-300 design can meet the acceptance criteria listed in Section 3N.1, following an impact of a large commercial aircraft:

- RB, Containment, and Fuel Pool Structural Integrity
- Reactor Shutdown, Containment Isolation, and Core Cooling
- Containment, Core, and Fuel Pool Heat Removal Capability
- Fire Barriers and Fire Protection Features

3N.5 RESULTS OF ASSESSMENT

The following key design features and functional capabilities ensure that the BWRX-300 design can meet the acceptance criteria stated in Section 3N.1 following an impact of a large commercial aircraft.

3N.5.1 Reactor Building, Containment, and Fuel Pool Structural Integrity

Analyses demonstrate that the structural design of the RB and protection of penetrations are adequate to prevent perforation and structural collapse of the RB. The containment and fuel pool are in the RB and, thus, are protected from physical damage due to the impact of an aircraft. The design of the integrated RB is further discussed in Subsection 3.8.4.

The RB polar crane cannot be struck because there is no perforation of the integrated RB outer wall. The evaluations of the polar crane consider damage due to impact loads on the exterior wall to verify the crane will not fall and damage SSCs below. Refer to Subsection 9.1.5 for additional information on the RB polar crane.

The key design features identified are:

- The RB structural design as discussed in Subsection 3.8.4
- Individual RB penetration designs
- Design of the RB polar crane as discussed in Subsection 9.1.5

3N.5.2 Reactor Shutdown, Containment Isolation, and Core Cooling

If notified of an imminent threat of aircraft impact, the operators will manually scram the reactor and initiate isolation of the containment and cooling. The SSCs required to implement these actions are located within the RB and are thus protected from physical or fire damage. Once initiated, the reactor will remain shut down, the containment will remain isolated, and adequate core cooling will be provided for more than 72 hours.

The key design features that assure reactor shutdown, containment isolation, and core cooling are:

- Design of the control systems as described in Subsection 7.3.1 that permits operators to manually initiate reactor scram, containment isolation, and isolation condenser cooling, and
- SSCs credited to implement these actuations, including:
  - The Control Rod Drive (CRD) system.
  - Main Steam, Condensate, and Feedwater Containment Isolation Valves used to isolate the containment.
  - The Isolation Condenser System (ICS).
  - The Reactor Isolation Valves (RIVs) used to isolate the reactor.

Analyses demonstrate that secondary effects such as shock and vibration resulting from an aircraft impact do not adversely affect these SSCs within the RB.

Because these key design features ensure core cooling and the containment is protected from physical damage as discussed in 3N.5.1, the containment remains intact.

3N.5.3 Containment, Core, and Fuel Pool Heat Removal Capability

Containment heat removal is passive as described in Subsection 6.2.2 and is contained within the RB and therefore will remain functional.

Core decay heat removal is passive, utilizing the ICS described in Section 6.3, with actuation as described in Subsection 3N.5.2. The ICS is contained within the RB and therefore will remain functional.

Fuel pool heat removal is based on passively maintaining adequate pool inventory above the spent fuel as described in Subsection 9.1.2.2. The fuel pool is contained within the RB and therefore will remain functional.

Key design features that assure containment, core, and fuel pool heat removal are:

- Passive Containment Cooling System (PCCS) design and layout as discussed in Subsection 6.2.2
- ICS design and layout as discussed in Section 6.3

- Fuel pool design and depth of water maintained above spent fuel as discussed in Subsection 9.1.2.2

Refer to Subsection 9.2.5 for description of the ultimate heat sink.

Analyses demonstrate that secondary effects such as shock and vibration resulting from an aircraft impact do not adversely affect these SSCs within the RB.

3N.5.4 Fire Barriers and Fire Protection Features

As stated in Subsection 3N.5.1, the integrated RB above grade will not be perforated by an aircraft impact.

RB penetrations were considered in the evaluation of aircraft fuel fed fires. The design of the RB structure and penetrations are adequate to prevent the propagation of external fires into the RB.

Key design features to implement these fire protection measures are:

- Design and fire rating of walls and roof of the RB as determined by the Fire Hazards Analysis discussed in Appendix 9A
- Physical barriers utilized to ensure jet fuel does not enter the integrated RB penetrations

3N.6 CONCLUSIONS OF ASSESSMENT

This assessment concludes that in the event of an impact of a large commercial aircraft, as defined by RG 1.217, the design features and functional capabilities of the BWRX-300 assure that the RB is not perforated and does not experience structural collapse, SSCs in the RB required for continued core cooling and spent fuel cooling are not impacted by shock and vibration, and RB openings are protected from the intrusion of aircraft fuel fed fires. The assessment identifies key design features and functional capabilities, changes to which are required to be controlled in accordance with 10 CFR 50.150(c).

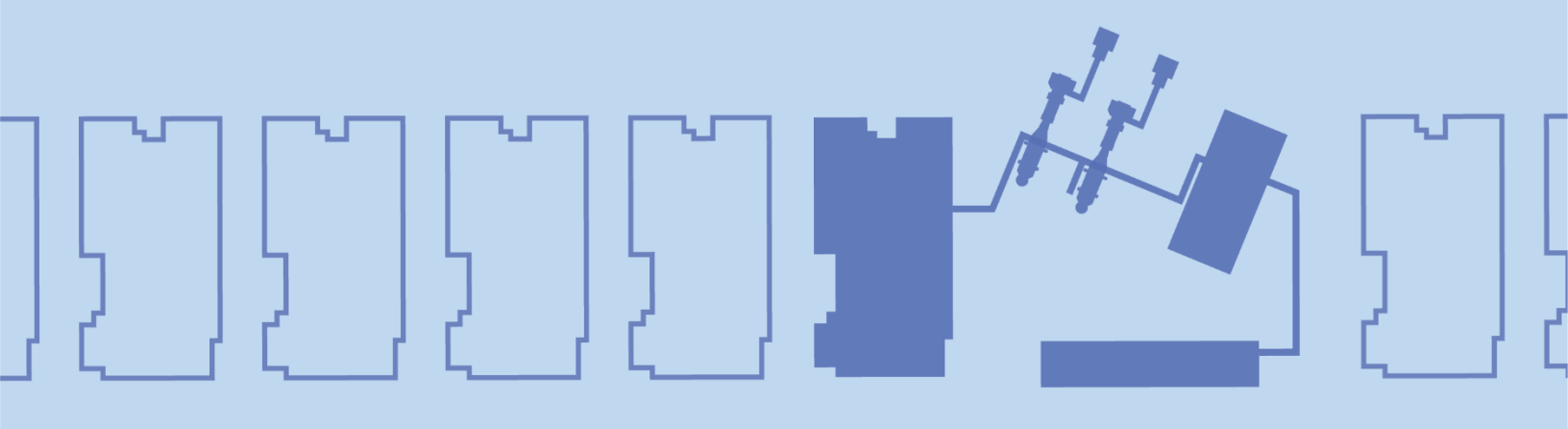
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3N-1 NEO 07-13, “Methodology for Performing Aircraft Impact Assessment for New Plant Design,” Nuclear Energy Institute, Revision 8, April 2011.

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Chapter 4
Reactor



ACRONYM LIST

Acronym	Explanation
1D	One-Dimensional
3D	Three-Dimensional
ABWR	Advanced Boiling Water Reactor
AOO	Anticipated Operational Occurrence
AOV	Air Operated Valve
ARI	Alternate Rod Insertion
ASME	American Society of Mechanical Engineers
ASTM	ASTM International
ATWS	Anticipated Transient Without Scram
AWS	American Welding Society
BPVC	Boiler and Pressure Vessel Code
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
CFS	Condensate and Feedwater Heating System
COLR	Core Operating Limits Report
CPR	Critical Power Ratio
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
CRDM	Control Rod Drive Mechanism
CST	Condensate Storage Tank
CUW	Reactor Water Cleanup System
DBA	Design Basis Accident
DSA	Deterministic Safety Analysis
ECP	Engineering Computer Program
ERIP	Emergency Rod Insertion Panel
ESBWR	Economic Simplified Boiling Water Reactor
FIV	Flow Induced Vibration
FMCRD	Fine Motion Control Rod Drive
FSAR	Final Safety Analysis Report
FW	Feedwater
GDC	General Design Criteria

Acronym	Explanation
GVH	GE Vernova Hitachi Nuclear Energy
GNF	Global Nuclear Fuel
GT	Gamma Thermometer
HCU	Hydraulic Control Unit
I&C	Instrumentation and Control
IASCC	Irradiation Assisted Stress Corrosion Cracking
ICS	Isolation Condenser System
ISI	Inservice Inspection
KKM	Kernkraftwerk Mühleberg (BWR/4 in Switzerland)
LFWH	Loss of Feedwater Heating
LHGR	Linear Heat Generation Rate
LOCA	Loss-of-Coolant Accident
LPRM	Local Power Range Monitor
LTR	Licensing Topical Report
MCPR	Minimum Critical Power Ratio
MCR	Main Control Room
NBS	Nuclear Boiler System
OLMCPR	Operating Limit Minimum Critical Power Ratio
PCV	Pressure Control Valve
PDC	Principal Design Criterion
PIE	Postulated Initiating Event
PRNM	Power Range Neutron Monitoring System
QA	Quality Assurance
RC&IS	Rod Control and Information System
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RG	Regulatory Guide
RPV	Reactor Pressure Vessel
SAFDL	Specified Acceptable Fuel Design Limit
SC1	Safety Class 1
SC2	Safety Class 2
SC3	Safety Class 3

Acronym	Explanation
SCC	Stress Corrosion Cracking
SCRRI	Selected Control Rod Run-In
SDC	Shutdown Cooling System
SDM	Shutdown Margin
SSCs	Structures, Systems, and Components
TAF	Top of Active Fuel
TRACG	Transient Reactor Analysis Code General Electric
USNRC	U.S. Nuclear Regulatory Commission
WRNM	Wide Range Neutron Monitor

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4.0 REACTOR

4.1 Summary Description

The BWRX-300 is a natural circulation, Boiling Water Reactor (BWR). This chapter describes the components for the BWRX-300, the nuclear fuel, reactor, and reactor core, including the fuel rods, fuel assemblies, reactivity control system (described in Subsection 3.9.4), nuclear design, and thermal-hydraulic design.

The reactor consists of the Reactor Pressure Vessel (RPV), pressure-retaining appurtenances, and reactor internal components, which are depicted in Figure 4.1-1. The reactor internal components are described in Subsection 4.1.2.

The fuel assembly and control rods are described in Section 4.2, including the design bases requirements, analytical methods, and evaluation results. Aspects pertaining to the nuclear design of the core, including the reference BWRX-300 design used for safety analysis, are described in Section 4.3. The thermal-hydraulic design bases requirements and important methodologies are described in Section 4.4. Reactor internals and the Control Rod Drive (CRD) system structural materials are discussed in Section 4.5. The functional design of the CRD system is discussed in Section 4.6. The thermal-hydraulic stability is discussed in Appendix 4A.

The Safety Classifications, Quality Groups, and Seismic Categories of the reactor components are provided in Table 3A-1.

4.1.1 Reactor Pressure Vessel

The BWRX-300 RPV assembly consists of the RPV with nozzles, and other appurtenances, a removable closure head, the reactor internals, and supports. The nuclear instrumentation that monitors the conditions within the reactor is designed to cover the full range of reactor power operation. The RPV, together with the internals, provides guidance and support for the Fine Motion Control Rod Drives (FMCRDs).

The RPV is a vertical, cylindrical pressure vessel fabricated with forged rings welded together with a removable top head by use of a head flange, seals, and bolting. Additional description of the RPV design is provided in Section 5.3.

The increased RPV height, relative to forced circulation BWRs, is achieved by a “chimney” in the space that extends from the top of the core (top guide) to the entrance to the chimney head and steam separator assembly. The natural circulation flow resulting from the tall RPV provides adequate thermal margins during power operation and off-normal conditions as described in Section 4.4.

4.1.2 Reactor Internal Components

Major reactor internal components are:

- Core components - control rods and nuclear instrumentation (Subsection 4.1.2.1)
- Core support structures (Subsection 4.1.2.2)
- Chimney (Subsection 4.1.2.3)
- Chimney head and steam separator assembly (Subsection 4.1.2.4)
- Steam dryer assembly (Subsection 4.1.2.5)
- Associated components and system interfaces (Subsection 4.1.2.6)

Except for the Zircaloy in the reactor core, reactor internals are stress corrosion-resistant stainless steels or other high alloy steels. The fuel assemblies (consisting of fuel bundles and channels),

control rods, chimney head and steam separator assembly, chimney, steam dryer assembly, and nuclear instrumentation are removable when the reactor vessel is opened for refueling or maintenance.

4.1.2.1 Core Components (Control Rods and Nuclear Instrumentation)

The major core components include control rods and nuclear instrumentation that are discussed in this chapter. The control rods (Figure 4.2-4) insert negative reactivity into the core to control reactor power and shutdown the reactor. Control rod distribution throughout the core is shown in Figure 4.2-3.

The control rods are discussed in Subsection 4.2.4. The nuclear instrumentation is discussed in Subsection 4.1.5.

4.1.2.2 Core-Support Structures

The core support structures include the shroud, core support ring, core plate, top guide, control rod guide tubes, CRD housing, and orificed/peripheral fuel supports.

The core support structures are designed, fabricated, and examined in accordance with the provisions of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) Section III, Division 1, “Rule for Construction of Nuclear Facility Components,” Subsection-NG, “Core Support Structures,” (Reference 4.1-1).

Shroud

The shroud provides a partition to separate the upward flow of coolant through the core from the downward recirculation flow. The shroud supports the weight of the chimney head and steam separator assembly, chimney, and top guide.

Core-Support Ring

The core support ring is a circular assembly with vertical stanchions (legs) that are welded to the RPV bottom head. The core support ring provides the vertical and lateral support for the shroud and other components like the top guide and core plate. It also supports the chimney, chimney head and steam separator assembly, and steam dryer assembly.

Core Plate

The core plate is at the bottom of the reactor core area. It provides the lateral support for the fuel assemblies and control rods through the orificed fuel supports. The core plate also provides vertical and lateral support for the nuclear instrumentation, the startup sources, and for the peripheral fuel supports and fuel assemblies.

Top Guide

The top guide is at the top of the reactor core. It provides the lateral support for the fuel assemblies, nuclear instrumentation, startup sources, and control rods. The top guide is attached between the top of the shroud and the chimney lower flange.

Control Rod Guide Tube

The control rod guide tubes fit into core plate holes and rest on their associated CRD housing that are welded to the reactor bottom head. The control rod guide tubes provide lateral support and channel water to the control rods as they move up and down within the core to change the reactor power levels. The control rod guide tubes also provide vertical support to the orificed fuel support and to four fuel assemblies landed on each orificed fuel support.

Control Rod Drive Housing

The CRD housings provide an extension of the RPV pressure boundary at the reactor vessel bottom head for installation of their respective CRD. They also support the weight of control rods, CRDs, control rod guide tube, thermal sleeve, orificed fuel supports, and fuel assemblies. Additionally, the CRD housing flanges also provide the ports for the attachment of CRD hydraulic supply lines.

Orificed and Peripheral Fuel Supports

The orificed fuel supports rest inside their respective control rod guide tubes with a flanged top end that sets down onto the control rod guide tube's top end that transfers vertical load. They provide a slot for the lower end of each fuel assembly to rest, and they provide cruciform slots that guide the control rod blade during movement. The orificed fuel supports slots keep the control rods properly aligned between the four fuel assemblies for rapid insertion into the core when fully withdrawn.

The peripheral fuel supports rest in the core plate around the outside of the orificed fuel supports and fuel assemblies. A single fuel assembly rests in each peripheral fuel support. The peripheral fuel support contains a flow restriction designed to ensure proper coolant flow to the peripheral fuel assembly. Otherwise, their function is the same as an orificed fuel support.

4.1.2.3 Chimney

The chimney barrel, the chimney lower and upper flanges constitute the chimney.

The relatively large RPV volume, along with the relatively tall chimney region, provides a substantial reservoir of water above the core. This provides assurance that the reactor water level is maintained at or above Top of Active Fuel (TAF) and fuel cladding temperature is maintained within normal operating temperature range following transients involving feedwater (FW) flow interruptions or Loss-of-Coolant Accidents (LOCAs).

The chimney barrel is an extended height cylindrical component welded to the chimney lower and upper flanges. The chimney barrel forms the annulus separating the subcooled recirculation flow returning downward from the upward steam-water mixture exiting the core. It provides the necessary additional flow space (volume) within the RPV that permits natural circulation driving forces to produce abundant core coolant flow.

The chimney lower flange is welded to the bottom of the chimney barrel and is mounted on the top guide and bolted to the top of the shroud. The chimney lower flange provides the vertical and horizontal support for the chimney barrel.

The chimney upper flange is welded to the top of the chimney barrel and supports the chimney head and steam separator assembly.

4.1.2.4 Chimney Head and Steam Separator Assembly

The chimney head and steam separator assembly is a perforated domed plate including a flange that is bolted to the chimney upper flange, which provides a path for the steam flowing from the core to the steam dryer assembly. The steam separator assembly is supported on and attached to the chimney head. The steam separator part of the assembly uses centrifugal force to remove most of the moisture in the steam before it enters the steam dryer assembly. The separated water drains to the downcomer annulus.

4.1.2.5 Steam Dryer Assembly

The steam dryer assembly completes the function of removing the moisture from the steam before it exits the reactor. The moisture content is lower than 0.1% weight percent at 100% reactor power when the steam leaves the reactor on its way to the high-pressure turbine.

The steam dryer assembly consists of multiple banks of dryer units mounted on a common structure that is removable from the RPV as an integral unit. The assembly includes the dryer banks, drain collecting troughs, drain ducts and a skirt that forms a water seal extending below the separator reference zero elevation. Upward and radial movements of the dryer assembly under the action of dynamic loads is limited by reactor vessel internal stops and brackets that are arranged to permit differential expansion growth of the dryer assembly with respect to the RPV. The steam dryer assembly is supported by a set of brackets equally spaced around the vessel shell internal wall. The dryer support ring rests on these brackets. Additional hold-down brackets are mounted inside the vessel head to restrain vertical movement of the dryer assembly under seismic or dynamic loads. These hold-down brackets are installed so that they are over the lifting lugs of the steam dryer assembly. Horizontal motion due to seismic or dynamic loads is limited by seismic blocks mounted to the dryer support ring.

4.1.2.6 Associated Components and System Interfaces

This section provides a description of the associated components and system interfaces within the reactor.

4.1.2.6.1 Feedwater Spargers

The FW spargers deliver makeup water to the reactor vessel during plant start up, power generation, and plant shutdown modes of operation. The FW spargers also provide a uniform distribution of FW flow within the downcomer annulus flow passage and delivers makeup and shutdown cooling water flow to the vessel.

4.1.2.6.2 Reactor Water Cleanup System

Two Reactor Water Cleanup System (CUW) internal drain lines are routed from flange connection inside the vessel to each respective nozzle, through an inverted loop, down the vessel shell internal wall, and to the RPV bottom head. The piping for each internal drain line is attached to the vessel shell internal wall by at least three brackets, spaced to provide support for anticipated loads and to inhibit pipe fatigue due to Flow Induced Vibration (FIV). The brackets are designed to the requirements of ASME BPVC, Section III, Division 1, Subsection NF, “Supports,” (Reference 4.1-2).

4.1.2.6.3 Reactor Pressure Vessel Head Vent

The RPV head vent design includes internal piping routed from the inside of the top of the RPV steam dome area to below the RPV shell flange. The RPV head vent subsystem is discussed in Subsection 5.4.13.

4.1.2.6.4 Incore Guide Tubes and Stabilizers

The incore guide tubes extend from the top of the incore housing to the top of the core plate. They provide the nuclear instrumentation with protection from flow of water in the bottom head plenum, and guidance for insertion and withdrawal from the core. The incore guide tube stabilizers provide lateral support and rigidity to the incore guide tubes.

4.1.2.6.5 Surveillance Sample Holders

The surveillance sample holder is a non-power generation internal device whose purpose is to enclose surveillance specimen capsules. The capsules are irradiated during the life of the plant.

The holder location permits capsule removal for examination incrementally over the life of the vessel. The surveillance sample holders are located next to the RPV inner wall around the periphery of the core at the core midplane. The surveillance sample holders are designed, fabricated, and analyzed to the requirements of ASME BPVC, Section III, Subsection NG (Reference 4.1-1).

4.1.3 Reactor Core

Each BWR reactor core is comprised of control cells. The reactor core is comprised of 240 fuel assemblies. Each control cell consists of a control rod and four fuel assemblies that immediately surround it. Each control cell is associated with a four–lobed fuel support. Around the outer edge of the core, twelve fuel assemblies are not immediately adjacent to a control rod and are supported by individual peripheral fuel supports. The four fuel assemblies are lowered into the control cell and, when seated, springs mounted at the tops of the channels force the channels into the corners of the cell such that the sides of the channels contact the grid beams. Core lattice designations are based upon relative water gap size between adjacent fuel assemblies and dimensional characteristics of the basic fuel assembly and channel.

4.1.4 Reactivity Control

The design of the fuel and control rods is described in Section 4.2. The CRD system is described in Section 4.6.

4.1.5 Nuclear Instrumentation

The performance of the core is monitored by fixed neutron detectors within the core. The nuclear instrumentation provides input to the automatic core protection control functions. The nuclear instrumentation consists of the Power Range Neutron Monitoring System (PRNM), Gamma Thermometer (GT), and Wide Range Neutron Monitor (WRNM).

The PRNM provides neutron monitoring power signals to the Instrumentation and Control (I&C), hydraulic scram logic. The PRNM also provides signals for post-accident monitoring purposes through isolated, one-way, optical data links to the core monitoring Three-Dimensional (3D) power distribution program and to the control rod blocking systems.

GTs are incore devices that convert local gamma flux to an electrical signal. Gamma flux and neutron flux are representative of core thermal power. It represents a diverse technology to the neutron detecting Local Power Range Monitors (LPRMs). The GTs are located in common instrument tubes with LPRM and WRNM detectors. There are a total of 184 GTs measuring dispersed radial and axial local power in the core. Each GT detector provides gamma sensitivity range up to 2.2 E+09 R/h. The LPRM strings are located between the corners of the fuel channels of the adjacent four fuel assemblies. The total number of 52 LPRMs are divided into the three divisions such that each division gets an approximately equal number of vertical and radial LPRM core locations. Each LPRM detector provides neutron monitoring sensitivity from approximately 10% core thermal power to >100% reactor thermal power and have a maximum perturbed thermal neutron flux capability of 2.8 E+14 neutrons/cm^2-sec.

The WRNM is performed by redundant processing units with a real time operating system. Each unit monitors the WRNM detectors. The WRNM detectors are distributed radially in the core at a fixed height. The detectors can be bypassed, but the bypass switches are arranged such that only one detector can be bypassed at a time in any quadrant. Each WRNM detector is sensitive to neutrons in the range of at least 11 decades of neutron flux from below criticality to >100% reactor thermal power.

Additional design information on the nuclear instrumentation is found in Section 7.3.

4.1.6 Analysis Techniques

The analytical techniques employed in core design are comprised of the computer codes summarized in Table 4.1-1 and engineering design practices. The computer codes in Table 4.1-1 are described in Appendix 3K.

Mechanical component qualification based on seismic and dynamic load combination analysis is discussed in Section 3.9. Fuel and control rod load conditions are described in Section 4.2.

4.1.6.1 Neutron Fluence

The GE methodology for neutron flux calculation was approved by the U.S. Nuclear Regulatory Commission (USNRC) in the Safety Evaluation Report for referencing in licensing actions, NEDC-32983P-A, “General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations, Licensing Topical Report (LTR),” (Reference 4.1-3). The GE methodology conforms to the guidance in Regulatory Guide (RG) 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” for neutron flux evaluation.

4.1.7 References

4.1-1 ASME BPVC, Section III, Division 1, Subsection NG, “Core Support Structures,” American Society of Mechanical Engineers, 2021.

4.1-2 ASME BPVC, Section III, Division 1, Subsection NF, “Supports,” American Society of Mechanical Engineers, 2021.

4.1-3 NEDC-32983P-A, “Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations, Revision 2, Licensing Topical Report,” GE-Hitachi Nuclear Energy Americas, LLC, January 2006.

Table 4.1-1: Analytical Techniques in Core Design

Analysis	Technique	Computer Code
Fuel rod design	Numerical solutions of 1D steady state and transient heat transfer and finite element mechanical analysis	PRIME03P
Fuel performance characteristics (temperature, internal pressure, and clad stress)		
Nuclear design Cross-sections and group constants	Lattice physics	TGBLA06
X-Y and X-Y-Z power distribution, reactivity coefficients	Steady state coupled nuclear thermal hydraulics Quasi 2 group diffusion theory	PANAC11
Axial power distributions, control rod worth		
Fuel rod power		
Thermal-hydraulic design steady state	Multi-dimensional, two-fluid model thermal hydraulics 3D reactor kinetics	TRACG04

Note:

1. Fuel load conditions and control load conditions are described in Section 4.2.

Legend:

1D: One-Dimensional

3D: Three-Dimensional

TRACG: Transient Reactor Analysis Code General Electric

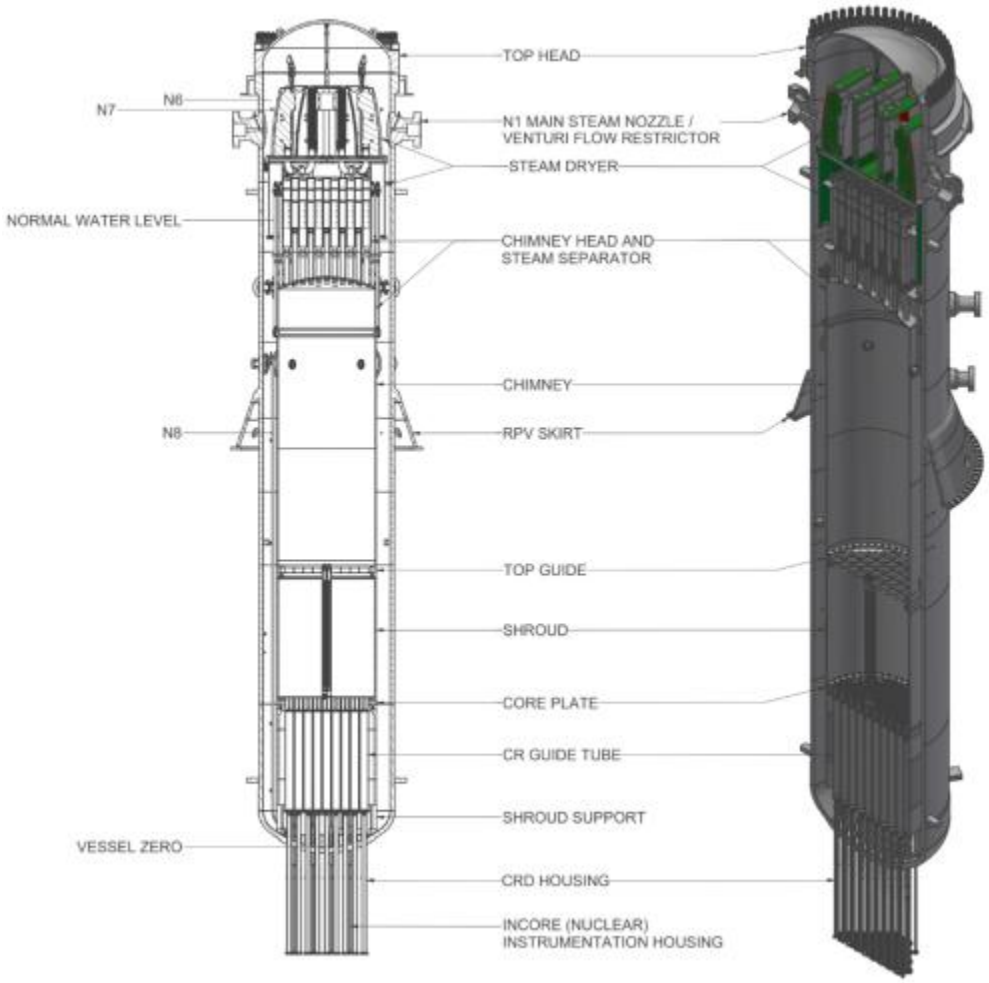


Figure 4.1-1: BWRX-300 Reactor Pressure Vessel and Internals Schematic

Note:

1. Schematic is not to scale. Some features are omitted or relocated for display purposes.

4.2 Fuel System Design

This section discusses both the fuel assembly and control rod design basis and system descriptions.

The reference fuel design for reactor licensing is GNF2 (Figure 4.2-1). GNF2 fuel conforms with the requirements in NEDE-24011-P-A-31-US GE, "General Electric Standard Application for Reactor Fuel (GESTAR II)," (Reference 4.2-1) as documented in NEDC-33270P, "GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II)," (Reference 4.2-2).

4.2.1 Fuel Assembly Design Bases

Operating limits are established to ensure that actual fuel operation is maintained within the fuel rod thermal–mechanical design and safety analysis bases. These operating limits define the maximum allowable fuel pellet operating power level as a function of fuel pellet exposure. The fuel assembly design basis is in accordance with NEDE-24011-P-A-31-US (GESTAR II) (Reference 4.2-1).

4.2.2 Fuel Assembly Description

GNF2 was designed for mechanical, nuclear, and thermal-hydraulic compatibility with the other Global Nuclear Fuel (GNF) fuel designs. The design includes many features of the GE10, GE11/13 and GE12/14 fuel including pellet-cladding interaction resistant barrier cladding, high performance spacers, partial length rods, thick corner/thin wall channel, and axial enrichment loading. New or improved features included in GNF2 are:

- Partial length rod configuration (both lattice position and axial extent) that improves efficiency and reactivity margins
- Alloy X-750 spacers with reduced pressure drop and improved resistance to boiling transition as compared to the Zircaloy ferrule spacer
- New fuel rod design with increased uranium content
- Defender™ Debris Filter Lower Tie Plate as standard equipment

The GNF2 design operates at a Linear Heat Generation Rate (LHGR) comparable to previous designs (GE11 and GE13), and higher than the GE14 design. The barrier fuel cladding is utilized to avoid capacity factor losses due to recommended operating restrictions associated with Pre-Conditioning Interim Operating Management Recommendations. The zirconium barrier liner that has been incorporated in GE6 through GE14 designs is standard in the GNF2 design. The combination and number of part length rods as well as the lengths of part length rods has been optimized. The GNF2 optimization maximizes the fuel assembly weight while maintaining pressure drop and stability characteristics compatible with existing BWR fuel designs.

The reference GNF2 fuel assembly components consist of a fuel bundle, a channel that surrounds the fuel bundle, and a channel fastener that attaches the channel to the bundle as shown on Figure 4.2-1. The fuel rods and water rods are spaced and supported by upper and lower tie plates and intermediate spacers. The upper tie plate has a handle for transferring the fuel bundle from one location to another.

The materials used in the GNF2 fuel assemblies are listed in Table 4.2-1.

Key attributes for the GNF2 fuel bundle are provided in Table 4.2-2. Additional details of the fuel assembly design are provided in NEDC-33270P (Reference 4.2-2).

4.2.2.1 Fuel Bundle

The GNF2 fuel bundle design consists of 92 fuel rods and two large central water rods contained in a 10x10 array. The two water rods encompass eight fuel rod positions. Three types of fuel rods are used in the GNF2 fuel assembly, as shown in Figure 4.2-2: tie rods, standard rods, and partial length rods. The upper and lower tie plates are connected by eight fueled tie rods. The upper tie plate has a handle for transferring the fuel bundle from one location to another. The fuel assemblies in the reactor are supported and positioned by the fuel support casting and core plate at their lower end and positioned horizontally by the top guide at their upper end. The fuel channel provides the structural lateral stiffness to the fuel assembly. The GNF2 lattice arrangement is shown on Figure 4.2-2.

4.2.2.1.1 Full-Length Fuel Rods

There are 78 full-length fuel rods in the GNF2 fuel bundle that reside in holes in the upper and lower tie plates. The end plugs of the standard fuel rods have shanks that fit into bosses in the tie plates. An expansion spring is located over the upper end plug shank of each fuel rod and tie rod in the assembly to maintain the fuel rods seated in the lower tie plate while accommodating irradiation growth. The fuel rod plenum also contains a compression spring that protects the column of fuel pellets during transportation while permitting the fuel column to expand axially during operation.

4.2.2.1.2 Tie Rods

The upper and lower tie plates are connected by eight fueled tie rods. The tie rods in each fuel bundle have lower end plugs that thread into the lower tie plate and threaded upper end plugs that extend through the upper tie plate. Nuts and locking tab washers are installed on the upper end plug to secure the upper tie plate. The tie rods support the weight of the bundle during fuel handling operations when the assembly is lifted by the handle. During operation, the fuel assembly is supported by the lower tie plate.

4.2.2.1.3 Gadolinia (Burnable Absorber) Fuel Rods

Gadolinia rods are full-length fuel rods with normal fuel pellets except sections of the fuel column contain pellets with Gd_2O_3 blended with UO_2. The resultant pellets function as a burnable neutron absorber for overall excess reactivity control and power shaping early in the fuel cycle. The Gd_2O_3 concentration is effectively depleted by the end of the first bundle operation cycle.

4.2.2.1.4 Partial Length Fuel Rods

Fourteen partial length rods are selectively located in the lattice as shown in Figure 4.2-2. GNF2 has eight long-partial length fuel rods and six short-partial length fuel rods. Partial length fuel rods are affixed to the lower tie plate and terminate above a specified spacer. The partial length fuel rod configuration improves efficiency and reactivity margins.

4.2.2.1.5 Water Rods

The GNF2 assembly is designed with two large circular water rods that are centrally located and occupy eight fuel rod lattice positions (i.e., each water rod occupies four fuel rod locations).

The water rods are Zircaloy-2 tubes with several holes around the circumference near each end. These holes allow water to be channeled from the bottom end of the bundle to the upper part of the bundle, without boiling, for increased neutron moderation. The water rod flow is controlled by the number and diameter of the inlet holes at the lower end.

Both water rods have a threaded lower end plug that is threaded into the lower tie plate. One is designed to position the spacers axially. Tabbed water rods that position the spacers are

designed with a D-shaped upper end plug and with spacer positioning tabs. These tabs are welded to the tube exterior above and below each spacer location. The tabbed water spacer-positioning rod is prevented from rotating by the engagement of its D-shaped upper end plug with a D-shaped hole in the upper tie plate. An expansion spring is located between the water rod shoulder and upper tie plate allowing for differential axial expansion.

4.2.2.1.6 Spacers

The GNF2 fuel design uses eight spacers made of Alloy X-750. The spacer spring force prevents fretting wear on the fuel rods due to fuel rod vibration. The top of the spacer features flow diversion devices to improve thermal hydraulic characteristics in the two-phase flow region.

4.2.2.1.7 Upper Tie Plate

Stainless steel upper and lower tie plates support the weight of the fuel assembly and positions the rod ends laterally during operation and handling. The upper tie plate has an internally threaded corner post fabricated to accept the channel fastener bolt. The upper tie plate is designed with locations for central water rods and partial length fuel rods to maximize flow area and minimize two–phase pressure drop.

4.2.2.1.8 Defender™ Debris Filter Lower Tie Plate

GNF2 features a Defender™ Fuel Debris Filter Lower Tie Plate. The lower tie plate, made of stainless steel, has cast-in flow holes. The fit-up with the fuel support casting, channel, and fuel rods is unchanged. One side of the lower tie plate has a cast-in opening to accept the Defender™ Debris Filter. The Defender™ Debris Filter is comprised of a series of formed vertical stainless steel plates welded together and captured in the cavity of the lower tie plate. The Defender™ Debris Filter is assembled into the lower tie plate through the opening on one side of the lower tie plate. A fully machined cover plate, which functionally replicates the lower tie plate side, is welded into place.

4.2.2.2 Channel and Channel Fastener

The GNF2 fuel channel (Figure 4.2-1) performs the following functions:

1. Forms the fuel bundle coolant flow path outer periphery.
2. Provides a surface for control rod guidance in the reactor core.
3. Provides structural lateral stiffness to the fuel bundle.
4. Controls, in conjunction with the lower tie plate, coolant bypass flow at the channel/lower tie plate interface.
5. Provides a stagnation envelope for in-core fuel sipping.

The channel is open at the bottom and makes a sliding seal fit over the lower tie plate. At the top of the channel, two opposite corners have welded tabs. These tabs support the weight of the channel on the upper tie plate posts. One of the tabs has a hole for attaching the channel fastener to the bundle. The channel design provides sufficient strength in the regions of highest stress while minimizing material that absorbs neutrons.

The overall stiffness of the fuel assembly is almost entirely a function of the channel stiffness. The channel is made of NSF material (GNF proprietary zirconium-based alloy channel material) as described in NEDE-33798P Supplement 1, “NSF Channel Annual Experience Summary Report,” (Reference 4.2-3).

The channel is supplied with a thin oxide film formed during thermal size annealing. The NSF alloy includes in its composition niobium, tin, and iron. NSF is considerably more resistant to

fluence gradient induced bow because it has a very low irradiation growth rate compared to Zircaloy. In addition, NSF exhibits good resistance to shadow corrosion induced bow.

The GNF2 channel fastener assembly consists of a stainless steel casting, a one-piece Alloy X-750 leaf spring with two active leaves at right-angles, a spring lock washer, and a fastener bolt attaching the fuel channel to the upper tie plate.

The channel fastener casting fits over the top of the channel and bolts through the channel tab into the upper tie plate. The channel fastener casting serves as a reaction support for the leaf springs, provides a captive housing and lead-in for the fastener spring, and protects the springs from being overstressed. The fastener bolt attaches the channel to the bundle and remains captive in the casting, even if the fastener bolt were to fail.

4.2.3 Fuel Assembly Design Evaluation

The GESTAR II licensing framework NEDE-24011-P-A-31-US (Reference 4.2-1) defines the generic requirements and approved methods for designing BWR fuel.

The principal fuel rod thermal-mechanical design analyses are performed using the PRIME methodology. The technical bases, qualification, and application of the PRIME steady-state methodology are described in NEDC-33256P-A, “The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance Part 1 – Technical Bases,” (Reference 4.2-4), NEDC-33257P-A, “The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance Part 2 – Qualification,” (Reference 4.2-5), and NEDC-33258P-A, “The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance Part 3 – Application Methodology,” (Reference 4.2-6).

The PRIME methodology is approved for application to Anticipated Operational Occurrence (AOO) transients in NEDC-33840P-A, “The PRIME Model for Transient Analysis of Fuel Rod Thermal-Mechanical Performance,” (Reference 4.2-7). PRIME analyses are performed for the following conditions:

1. For each analysis, fuel rod input parameters are based on either the most unfavorable manufacturing tolerances (i.e., ‘worst tolerance’ analyses) or statistical distributions of the input values. Calculations are performed providing either a ‘worst tolerance’ or statistically bounding tolerance limit for the resulting output parameter(s).
2. Operating conditions are postulated that cover the conditions anticipated during normal operating conditions and AOOs.

The fuel rod thermal-mechanical results are provided in NEDC-33270P (Reference 4.2-2), and NEDC-34042P “BWRX-300 GNF2 Fuel Assembly Thermal-Mechanical Design Report,” (Reference 4.2-8). The described GNF2 fuel design is within the limits of applicability for the fuel performance methodology as described in Table 2.1 of NEDC-33256P-A (Reference 4.2-4) and Table 3-1 of NEDC-33840P-A (Reference 4.2-7). Additionally, reactor operating parameters such as coolant temperature and pressure, water chemistry, and heat transfer coefficients used in the GNF2 design analysis are described in NEDC-33270P (Reference 4.2-2) and NEDC-34042P (Reference 4.2-8). The following subsections describe aspects of individual analyses and methods. GNF2 fuel bundle key attributes are provided in Table 4.2-2.

4.2.3.1 Design Analyses

GESTAR II established the basis for evaluating new fuel designs, developing the critical power correlation for these designs, and determining the applicability of generic analyses as discussed in NEDE-24011-P-A-31-US (Reference 4.2-1).

4.2.3.2 Worst Tolerance Analyses

The worst tolerance analysis methodology is described in NEDC-33258P-A (Reference 4.2-6).

4.2.3.3 Statistical Analyses

The statistical analysis methodology is described in NEDC-33258P-A (Reference 4.2-6).

4.2.3.4 Fuel Lift and Seismic and Dynamic Load Analysis

The analyses of fuel lift and dynamic (including seismic) loads evaluate whether these loads are sufficient to unseat the fuel assembly. Assurance that vertical uplift forces will not unseat the fuel assembly such that loss of lateral fuel bundle positioning could occur is provided by a nonlinear fuel lift analysis discussed in NEDE-24011-P-A-31-US (Reference 4.2-1), NEDC-33270P (Reference 4.2-2), and in NEDE-21175-3-P-A, "BWR Fuel Assembly Evaluation of Combined Safe Shutdown (SSE) and Loss-of-Coolant Accident (LOCA) Loadings (Amendment No. 3)," (Reference 4.2-9). LOCA and seismic loads conservatively bound the worst-case hydraulic loads possible during normal operation.

A seismic analysis will be performed using the Blue Energy site specific load combinations to determine that the maximum fuel accelerations are less than the fuel capacity. An evaluation demonstrating that these acceptance criteria are met will be provided in the FSAR.

4.2.3.5 Cladding Strain Analysis

The cladding strain analysis is performed using PRIME. The method and limits are described in NEDC-33258P-A (Reference 4.2-6).

4.2.3.6 Fuel Rod Internal Pressure Analysis

The fuel rod internal pressure analysis is performed using PRIME. The methodology and limits are described in NEDC-33258P-A (Reference 4.2-6).

4.2.3.7 Fuel Pellet Temperature

The maximum permissible thermal overpower is determined considering the difference between the fuel melting temperature and the fuel pellet centerline temperature at the overpower condition. A mean value and a standard deviation for the difference between the fuel melting temperature and the fuel pellet centerline temperature are determined. A specified percentile of this difference is required to be greater than or equal to zero to ensure that fuel pellet centerline melting will not occur in the event of a specified overpower during plant operation. Thermal and Mechanical Overpowers Analysis is discussed in Subsection 4.2.3.11.

4.2.3.8 Cladding Fatigue Analysis

The PRIME code is used to calculate cladding strains as part of GNF's statistical analysis methodology for evaluating cladding fatigue. Conservative fuel duty cycles inform the cladding fatigue analysis. The duty cycles represent conservative assumptions regarding power changes anticipated during normal reactor operation and AOOs, planned surveillance testing, normal control rod maneuvers, shutdowns and load following. Fatigue analyses are generically applicable based on fuel type and plant design. These analyses account for daily load following using the BWRX-300 plant duty cycles specification. The analyses of GNF2 fuel demonstrate that the cladding fatigue capability accommodates the expected fatigue duty for the projected life and operation of the fuel with sufficient margin. The fatigue usage must be consistent with the acceptance criteria of NEDE-24011-P-A-31-US GESTAR II (Reference 4.2-1).

4.2.3.9 Cladding Creep Collapse Analysis

Fuel pellets could undergo axial densification that could result in a large axial gap in the fuel column. As the cladding is unsupported by fuel pellets in the axial gap region, the ovality can become large enough to result in elastic instability and cladding collapse. The cladding creep collapse analysis consists of a detailed finite element mechanics analysis of the cladding. This evaluation is described in NEDC-33139P-A, “Cladding Creep Collapse,” (Reference 4.2-10).

4.2.3.10 Fuel Rod Stress Analysis

The fuel rod stress analysis is performed using the Monte Carlo statistical methodology. Results from PRIME analyses are used to generate inputs for the stress analysis. The cladding stress analysis is described in NEDC-33270P (Reference 4.2-2). Additionally, NEDC-33270P (Reference 4.2-2) discusses design criteria, compliance, and the safety implications of fuel rod bowing.

4.2.3.11 Thermal and Mechanical Overpowers Analysis

Thermal and mechanical overpowers are not licensing limits but are intended to prevent exceedance of licensing limits to be used during the nuclear design of a core or cycle. As part of the core design and AOO analysis, the calculated core mechanical and thermal overpowers defined in NEDC-33258P-A (Reference 4.2-6) and NEDC-33840P-A (Reference 4.2-7) are compared with the thermal overpower and mechanical overpower criteria and confirm conformance to these criteria. The steady-state thermal and mechanical overpower analyses are provided in NEDC-34042P (Reference 4.2-8).

4.2.3.12 Fretting Wear Testing

The GNF2 fuel assembly was tested to assure that the design features do not result in a significant increase in FIV response and thereby do not increase the potential for fretting. The method used to demonstrate the adequacy of the fuel assembly from a FIV perspective is included in NEDC-33270P (Reference 4.2-2).

4.2.3.13 Water Rod Analysis

Analyses are performed to determine component stresses at the bounding load conditions and compared to applicable criteria. The load conditions consider shipping and handling loads, and the pressure differential across the water rod. The design is also evaluated using finite element analysis to demonstrate that the design adequately sustains the design shipping and handling loads. Because the stresses are well below yield strength and there is no significant cyclic loading, the fatigue capability is not exceeded.

4.2.3.14 Tie Plate Analysis and Testing

Tie plate adequacy is demonstrated by detailed finite element analysis and mechanical testing as applicable, for bounding fuel handling and seismic load conditions. The upper and lower tie plates are not subjected to any significant cyclic loadings, and fatigue capability is therefore not exceeded.

4.2.3.15 Spacer Testing

Tests have been performed to demonstrate that the spacer design can withstand design bases loading without any significant deformation. Cyclic testing for seismic loading demonstrates that the spacer stresses and strains do not exceed failure values and that the fatigue capability is not exceeded. Additionally, based on the FIV test results, GNF2 fuel design meets the fretting design requirements summarized in NEDC-33270P (Reference 4.2-2).

4.2.3.16 Channel Analysis

Channel adequacy relative to applicable design criteria is confirmed by performing the following evaluations:

- Channel bending stress due to channel wall pressure gradient
- Channel fatigue and stress rupture fatigue with respect to the cyclic thermal stress gradients and pressures
- Channel/lower tie plate differential irradiation growth
- Channel lateral deformation due to bulge and bow
- Channel fastener bolt stress
- Channel fastener spring analysis
- Channel loads due to seismic events

4.2.4 Control Rod Design Bases

The Safety Classifications, Quality Groups, and Seismic Categories of the control rods are listed in Table 3A-1. The functions are described below:

Safety Category 1 Functions

- Provide variable negative nuclear reactivity worth to the reactor core
- Maintain insertion and withdraw capability of the control rod
- Provide positive coupling to the FMCRD

Safety Category 2 Functions

- None.

Safety Category 3 Functions

- None.

Safety Category N Functions

- None.

4.2.4.1 Control Rod Description

The control rods are cruciform shaped elements which occupy alternate spaces between fuel assemblies throughout the core. The control rods are connected to bottom vessel mounted drive mechanisms, which allow either axial positioning within the core or withdrawal into the guide tubes below the core. The outer fuel channel surfaces of the four fuel assemblies in a cell provide guidance for the insertion and withdrawal of the control rod. The control rods are cooled by the core leakage (bypass) flow and CRD purge flow to remove heat generated by neutron and gamma absorptions. A typical control rod consists of a handle, absorber section, and connector assembly as illustrated in Figure 4.2-4.

4.2.4.2 Control Rods Design Acceptance Criteria

The BWRX-300 control rods are designed to meet the control rod reactivity control requirements established by the safety analyses and to meet the following structural, materials and nuclear lifetime acceptance criteria. The control rod is based on the designs used in the operational BWR

fleet. An evaluation demonstrating that these acceptance criteria are met will be provided during future licensing activities.

Control Rod Structural Analysis

The control rod is analyzed in both the unirradiated condition and in the end-of-life irradiated condition and is able to withstand all applied loads without detrimental deformation or fatigue.

Control Rod Insertion

The control rod is designed and qualified to insert within limits under various conditions, including seismic events.

Control Rod Material

The control rod and its materials are designed to be compatible with the reactor coolant chemistry. Materials selected for control rod neutron absorber containment components have Irradiation Assisted Stress Corrosion Cracking (IASCC) resistance properties. Control rod un-irradiated and irradiated material properties are justified and documented.

Control rod mechanical and structural evaluations include allowances for corrosion and crud deposition as a function of time.

Control Rod Nuclear Reactivity

The nuclear lifetime of the control rod design is the neutron absorber depletion that corresponds to a 10 percent reduction in cold reactivity worth (Δk/k), relative to its zero-depletion cold reactivity worth.

Control rod characteristics in the PSAR equilibrium core analyses and the Chapter 15 safety analyses are representative of the BWRX-300 control rod.

4.2.5 Safety Evaluation

The GNF2 analyses utilize the following processes:

- Either worst tolerance assumptions are applied, or probabilistic analyses are performed to determine statistically bounding results
- Operating conditions are taken to bound the conditions anticipated during normal steady-state operation and AOOs

The integrity of the fuel rods is assured by designing to ensure the thermal-mechanical design criteria specified by NEDE-24011-P-A-31-US (Reference 4.2-1) are met.

The materials used in the GNF2 fuel assemblies are listed in Table 4.2-1.

Fuel mechanical and fuel rod thermal-mechanical evaluations, summarized in NEDC-33270P (Reference 4.2-2), demonstrate compliance with regulatory requirements. Detailed technical information pertaining to fuel assembly mechanical and fuel rod thermal-mechanical performance is provided in NEDC-34041P, “BWRX-300 GNF2 Fuel Assembly Mechanical Design Report,” (Reference 4.2-11) and NEDC-34042P (Reference 4.2-8).

Evaluation of the control rod design has not yet been completed. An evaluation demonstrating that the control rod design acceptance criteria in Subsection 4.2.4.2 are met will be addressed in the FSAR.

Fuel lift dynamic analyses require site-specific inputs. The fuel lift dynamic analyses described in Subsections 3.9.3.7.6 and 4.2.3.4 will be provided in the FSAR.

Regulatory Requirements

The following General Design Criteria (GDC) have aspects specific to the fuel system design:

GDC 10 – Reactor design: Compliance with this GDC is discussed in NEDC-33934P, "BWRX-300 Safety Strategy," (Reference 4.2-12) and the applicable PSAR sections are listed in Table 3.1-1.

GDC 27 – Combined reactivity control systems capability: The basis for GDC 27 compliance is discussed in Section 3.1.

In the Safety Evaluation to LTR NEDC-33912P-A, "BWRX-300 Reactivity Control," (Reference 4.2-13), Limitation and Condition 5.1 requires an applicant to demonstrate the probability of an Anticipated Transient Without Scram (ATWS) event is less than $1X10^{-5}$ per reactor year without the need for Standby Liquid Control System. An ATWS probability of less than $1X10^{-5}$ per reactor year is demonstrated and discussed in Subsection 15.6.1.4.5 that fulfills Limitation and Condition 5.1.

GDC 35 – Emergency core cooling: Compliance with GDC 35 relates to ICS function and is addressed in Section 6.3.

4.2.6 Testing, Inspection and Surveillance Plans

The GVH program for surveillance of both production and developmental fuel is documented in NEDE-24011-P-A-31, "General Electric Standard Application for Reactor Fuel," (Reference 4.2-14).

4.2.7 References

4.2-1 NEDE-24011-P-A-31-US, "General Electric Standard Application for Reactor Fuel," (GESTAR II) (Supplement for United States), Revision 31, Global Nuclear Fuel - Americas, LLC, November 2020.

4.2-2 NEDC-33270P, "GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II)," Revision 11, Global Nuclear Fuel - Americas, LLC, August 2020.

4.2-3 NEDE-33798P, Supplement 1, "NSF Channel Annual Experience Summary Report," Revision 7, Global Nuclear Fuel - Americas, LLC, January 2024.

4.2-4 NEDC-33256P-A, "The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance Part 1 – Technical Bases," Revision 2, Global Nuclear Fuel - Americas, LLC, October 2021.

4.2-5 NEDC-33257P-A, "The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance Part 2 – Qualification," Revision 2, Global Nuclear Fuel - Americas, LLC, October 2021.

4.2-6 NEDC-33258P-A, "The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance Part 3 – Application Methodology," Revision 2, Global Nuclear Fuel - Americas, LLC, October 2021.

4.2-7 NEDC-33840P-A, "The PRIME Model for Transient Analysis of Fuel Rod Thermal-Mechanical Performance," Revision 1, Global Nuclear Fuel - Americas, LLC, August 2017.

4.2-8 NEDC-34042P, "BWRX-300 GNF2 Fuel Assembly Thermal-Mechanical Design Report," Revision 0, GE-Hitachi Nuclear Energy Americas, LLC, April 2024.

4.2-9 NEDE-21175-3-P-A, "BWR Fuel Assembly Evaluation of Combined Safe Shutdown (SSE) and Loss-of-Coolant Accident (LOCA) Loadings (Amendment No. 3), Licensing Topical Report," Revision 0, General Electric Company, October 1984.

4.2-10 NEDC-33139P-A, “Cladding Creep Collapse,” Revision 0, Global Nuclear Fuel - Americas, LLC, July 2005.

4.2-11 NEDC-34041P, “BWRX-300 GNF2 Fuel Assembly Mechanical Design Report,” Revision 0, GE-Hitachi Nuclear Energy Americas, LLC, April 2024.

4.2-12 NEDC-33934P, “BWRX-300 Safety Strategy,” Revision 0, GE-Hitachi Nuclear Energy Americas, LLC, Rev. 2, March 2026.

4.2-13 NEDC-33912P-A, “BWRX-300 Reactivity Control,” Revision 2, GE-Hitachi Nuclear Energy Americas, LLC, February 2023.

4.2-14 NEDE-24011-P-A-31, “General Electric Standard Application for Reactor Fuel,” (GESTAR II), Revision 31, Global Nuclear Fuel – Americas, LLC, November 2020.

Table 4.2-1: GNF2 Fuel Assembly Material

Component	Material
Fuel Assembly	
Fuel Rod Spacers	Alloy X-750
Water Rods	Recrystallized Zircaloy-2
Tie Plates	Austenitic Stainless Steel
Fuel Rod	
Cladding	Recrystallized Zircaloy-2
Barrier	Recrystallized Zirconium
Fuel Pellet	UO_2 and UO_2 - Gd_2O_3
Channel and Channel Fastener	
Channel	Recrystallized Zircaloy-2 or NSF (GNF Proprietary Zirconium-Based Alloy)
Guard Material	Stainless Steel
Spring, cap screw and lock-washer material	Alloy X-750 XM-19 Stainless Steel

Note:

1. NEDC-34041P (Reference 4.2-11)

Table 4.2-2: GNF2 Fuel Bundle Key Attributes

	GNF2
Fuel Bundle	
Total number of fuel rods	92
Number of full length rods	78
Number of fueled tie rods	8
Number of partial length rods (long)	8
Number of partial length rods (short)	6
Lattice arrangement	See Figure 4.2-2
Fuel Rod	
Cladding material	See Table 4.2-1
Assembly active fuel length (mm)	3810
Fuel pellet material	See Table 4.2-1
Fuel rod fill gas	He
Water Rod	
Number of water rods	2
Water rod material	See Table 4.2-1
Spacer	
Spacer structure	Grid-type with flow diverters
Spacer material	Alloy X-750
Number of spacers	8

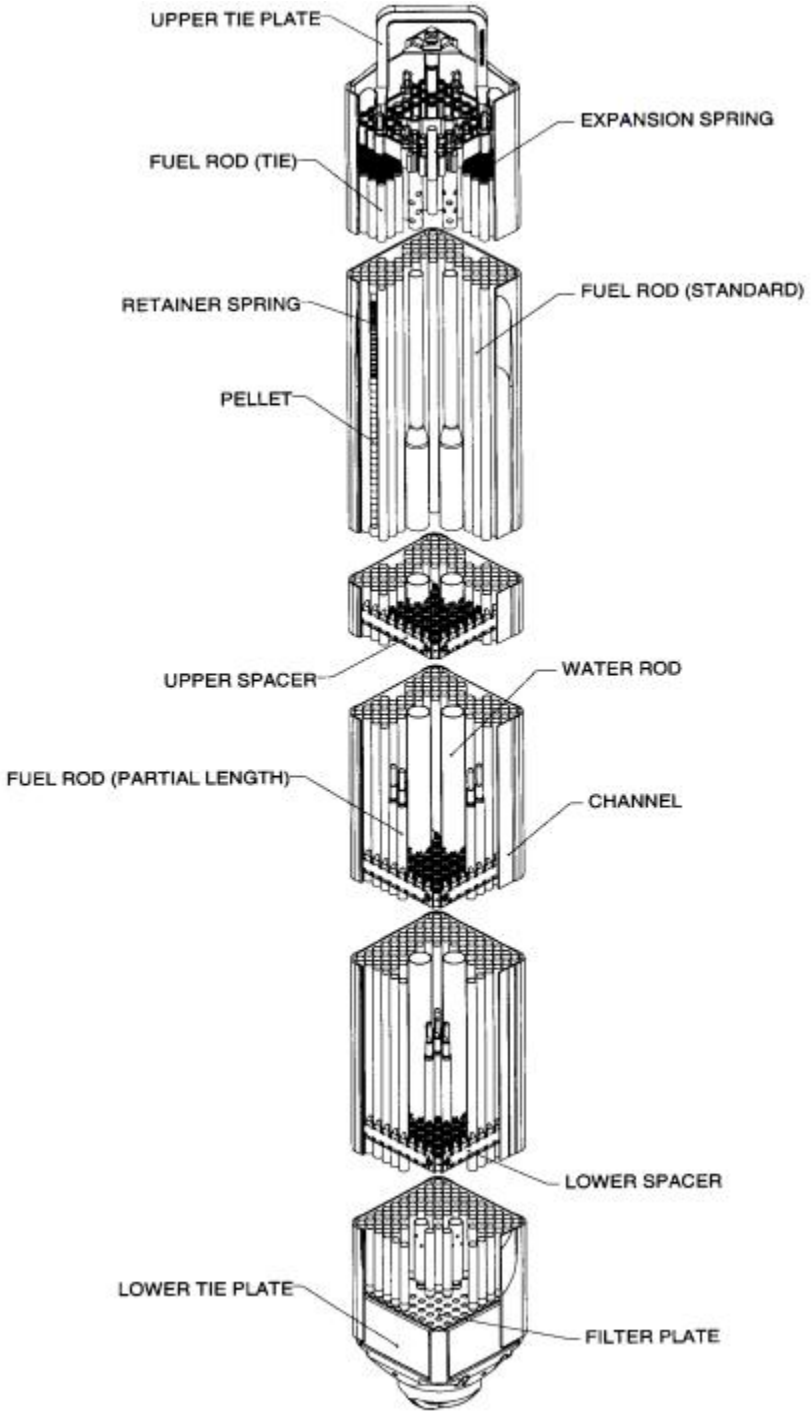


Figure 4.2-1: GNF2 Fuel Assembly

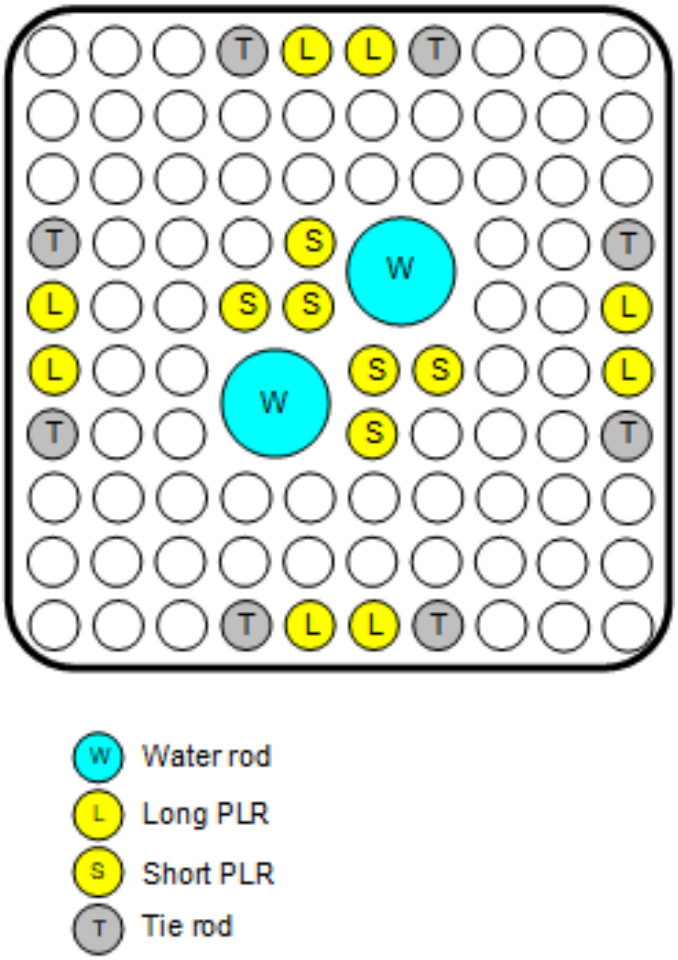


Figure 4.2-2: GNF2 Lattice Arrangement

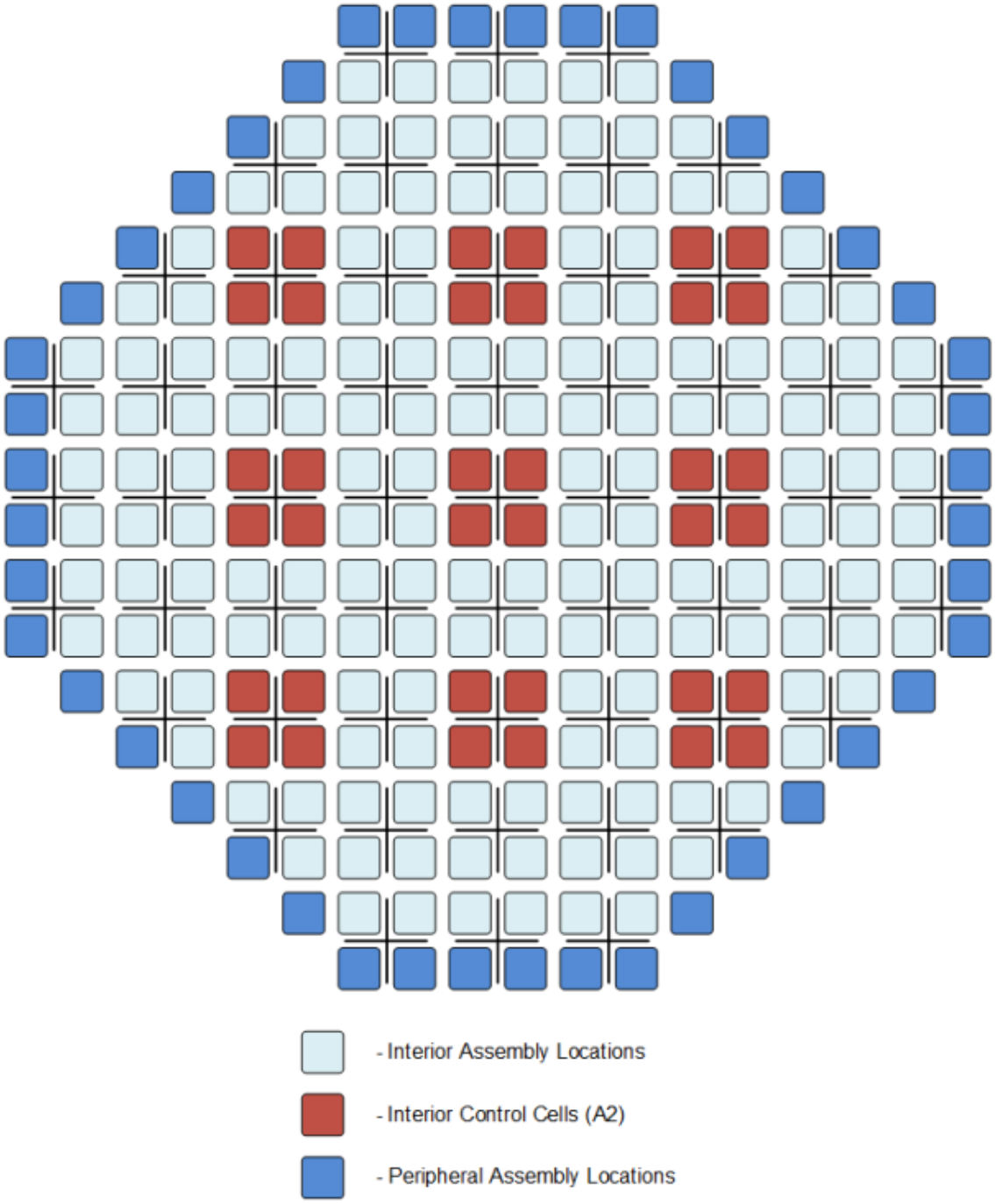


Figure 4.2-3: Control Rods Distributed Throughout Core

Note:

1. All available control cells are shown in red.

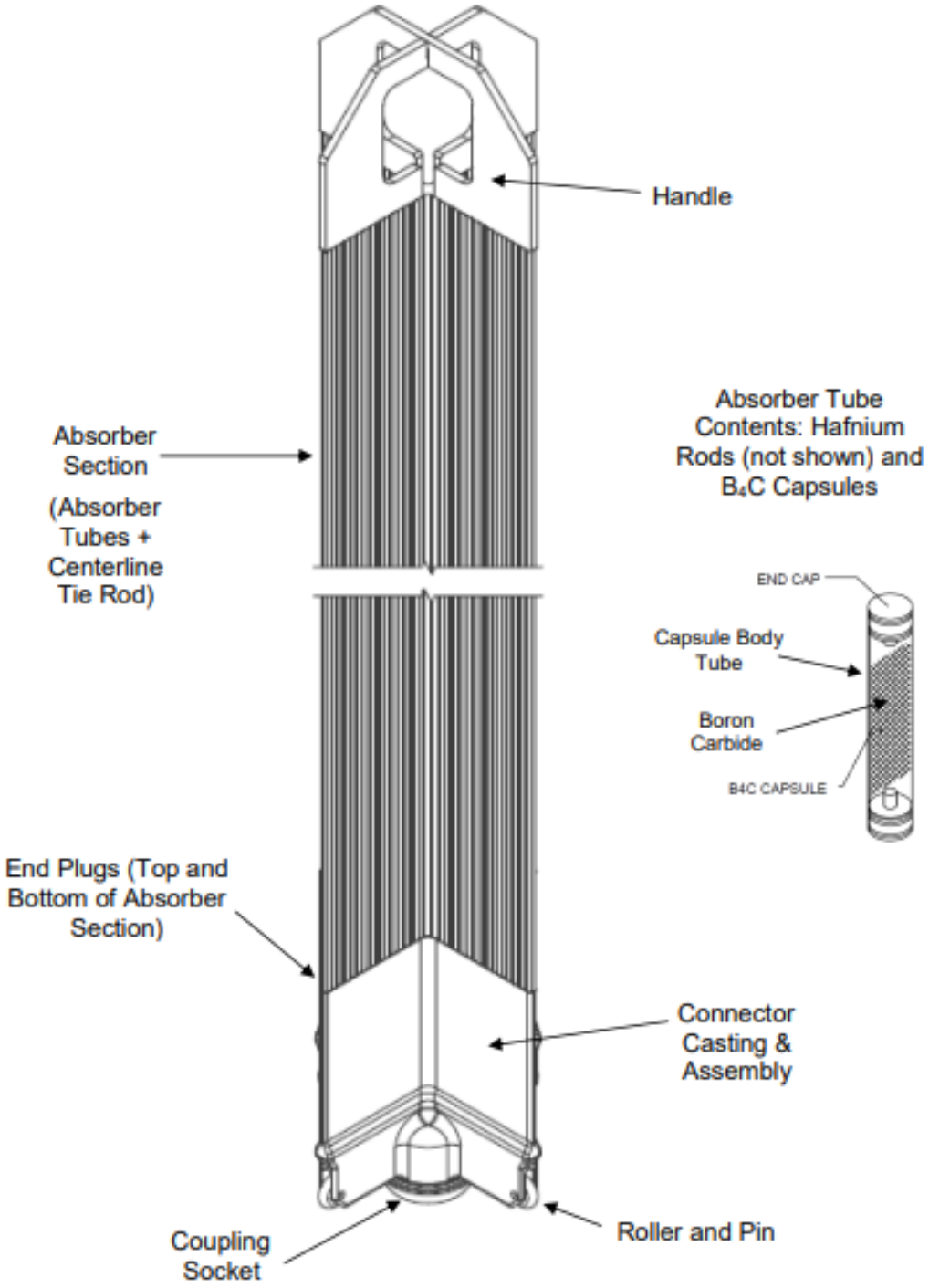


Figure 4.2-4: BWRX-300 Control Rod

4.3 Nuclear Design

This section describes the design bases and functional requirements used in the nuclear design of the fuel, core, and reactivity control system and relates these design bases.

4.3.1 Nuclear Design Bases

This section describes the design bases and functional requirements used in the nuclear design of the fuel, core, and reactivity control system.

4.3.1.1 Reactivity Feedback

Reactivity coefficients representing the differential changes in reactivity produced by differential changes in core conditions are useful for calculating stability parameters and evaluating the response of the core to external disturbances. The base initial condition of the system and the Postulated Initiating Event (PIE) determine which of the several defined coefficients are significant in evaluating the response of the reactor.

There are two primary reactivity coefficients that characterize the dynamic behavior of BWRs; these are the Doppler reactivity coefficient and the moderator void reactivity coefficient. Also associated with the BWR are a power reactivity coefficient and a temperature coefficient. The power coefficient is a combination of the Doppler and moderator void reactivity coefficients in the power operating range, and the temperature coefficient is merely a combination of the Doppler and moderator temperature coefficients. Power and temperature coefficients are not specifically calculated for reload cores.

The fuel reactivity acceptance criteria are established in NEDE-24011-P-A-31-US, “General Electric Standard Application for Reactor Fuel,” (Reference 4.3-1).

4.3.1.2 Control Requirements (Shutdown Margins)

The core must be capable of being made subcritical, with margin, in the most reactive condition throughout the operating cycle with the most reactive control rod fully withdrawn or rod pair associated with the coupled Hydraulic Control Unit (HCU) postulated stuck in the full out position, and all other rods fully inserted. The Shutdown Margin (SDM) is determined by using the 3D BWR core simulator code PANAC11 to calculate the core multiplication at selected exposure points with the most reactive control rod or rod pair associated with a single HCU fully withdrawn. The SDM is calculated based on the carryover of the minimum expected exposure at the end of the previous cycle. The core is assumed to be in the cold, xenon–free condition to produce conservative calculated values.

4.3.1.3 Control Requirements (Overpower Basis)

Technical Specification limits on Minimum Critical Power Ratio (MCPR) and the LHGR are determined such that the fuel will not exceed required licensing limits during abnormal operational occurrences or accidents. Explicit Maximum LHGR Limit and MCPR parameter definitions are provided as follows:

4.3.1.3.1 Linear Heat Generation Rate Operating Limit

The LHGR operating limit is the maximum allowable linear heat generation for each fuel rod in the bundle. The LHGR operating limit is fuel rod type-dependent and is a function of content and exposure. The LHGR is monitored, and operating limits are established to ensure that the fuel operation is maintained within the fuel rod thermal-mechanical design and safety analysis bases. Under AOO conditions, including the maximum overpower condition, the calculated overpower is confirmed to neither cause fuel melting nor exceed the cladding stress and strain limits as discussed in Subsection 4.2.3.

4.3.1.3.2 Minimum Critical Power Ratio

The Critical Power Ratio (CPR) is defined as the ratio of the critical power (bundle power at which some point within the assembly experiences onset of boiling transition) to the operating bundle power. The critical power is determined at the same mass flux, inlet temperature, and pressure that exists at the specified reactor condition. Thermal margin is stated in terms of the minimum value of the CPR, MCPR, which corresponds to the most limiting fuel assembly in the core. An Operating Limit Minimum Critical Power Ratio (OLMCPR) is established to provide adequate assurance that the fuel cladding integrity safety limit, specified in Subsection 4.4.4, is not exceeded for any moderate frequency AOO. This is further discussed in Section 15.5.

4.3.2 Core Nuclear Design Description

The BWRX-300 core nuclear design consists of a boiling light-water moderated reactor fueled with slightly enriched uranium dioxide fuel assemblies. The use of water as a moderator produces a neutron energy spectrum where fissions are caused principally by thermal neutrons. At normal operating conditions, the moderator boils, producing a spatially variable distribution of steam voids in the core. The BWR design provides a system for which reactivity is reduced by an increase in the steam void content in the moderator. The void reactivity feedback effect is an inherent safety feature of the BWR system. Any system change that increases reactor power, either locally or core-wide, produces additional steam voids that reduce reactivity and thereby reduces power.

A reference equilibrium core, as shown in Figure 4.3-2, has been developed for safety analysis to demonstrate that the BWRX-300 conforms to regulatory requirements with high confidence.

The core loading pattern, operating control rod patterns, and core performance results are described in NEDC-34044P, “BWRX-300 GNF2 Equilibrium 12 Month Cycle Nuclear Design Report,” (Reference 4.3-2), and the nuclear design of the fresh fuel assemblies is provided in NEDC-34045P, “BWRX-300 GNF2 Fuel Bundle Information Report for Equilibrium 12-Month Cycle,” (Reference 4.3-3). The BWRX-300 core configuration, including a representative nuclear instrumentation design and rod patterns, is depicted in Figure 4.3-1 and Figure 4.3-3 respectively. Select nuclear design information and core performance results are depicted in Figure 4.3-4a through Figure 4.3-8. Margin to the Specified Acceptable Fuel Design Limit (SAFDL) is maintained during normal steady state operation when the MCPR is greater than the required OLMCPR and the LHGR is maintained below the maximum LHGR limit(s). The steady state OLMCPR and maximum LHGR limits are established as discussed in Subsection 15.5.3.

4.3.3 Core Nuclear Analytical Methods

The analytical methods used in the design and analysis of the BWRX-300 core during steady state operation are summarized below and described in NEDC-34039P “BWRX-300 GNF2 Steady State Nuclear Methods: TGBLA06/PANAC11 Application Methodology,” (Reference 4.3-4).

4.3.3.1 Steady State

The principal Engineering Computer Programs (ECPs) used in the steady-state nuclear core analysis are the 2D lattice physics code TGBLA06 and the 3D BWR Core Simulator PANAC11. PANAC11 is a static coupled nuclear-thermal-hydraulic computer program representing the BWR core exclusive of the external flow loop.

TGBLA06/PANAC11 has hundreds of reactor operating years of successful experience evaluating BWR cores with similar configurations to the BWRX-300. This includes detailed 3D design and operational calculations of eigenvalue, neutron flux, power distributions and thermal performance as a function of control rod position, refueling pattern, coolant flow, reactor pressure, and other operational and design variables. The exposure accounting process provides the

opportunity for continuous comparison and validation of the nuclear methods against operating data. The purpose and general methodology of the BWR Core Simulator has not changed from the way it is used to evaluate the current BWR fleet to enable its application to BWRX-300. NEDC-34039P (Reference 4.3-4) provides a record that establishes the following for the combined method with respect to the BWRX-300 design:

- Adequate performance through operational qualification
- Appropriate uncertainties that characterize its capability
- Limitations associated with its use

4.3.3.2 Reactivity Coefficients

Reactivity coefficients, the differential changes in reactivity produced by differential changes in core conditions, are useful in calculating stability and evaluating the response of the core to external disturbances. The base initial condition of the system and the PIE determine which of the several defined coefficients are significant in evaluating the response of the reactor.

4.3.4 Core Nuclear Design Evaluation

This section provides an evaluation of reactivity based on GNF2 fuel.

4.3.4.1 Power Distribution

The core power distribution is a function of fuel bundle design, core loading, control rod pattern, core exposure distributions and core coolant flow rate. The thermal performance parameters, LHGR operating limit and MCPR, limit unacceptable core power distributions.

4.3.4.2 Reactivity Coefficients

The reactivity coefficients evaluation is performed as part of new fuel design development ensuring compliance with USNRC fuel licensing acceptance criteria. The GNF2 results are documented in NEDC-33270P, “GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II),” (Reference 4.3-5), which concludes that the criteria defined in GESTAR II have been met for the GNF2 fuel design.

4.3.4.3 Reactivity Variation

The excess reactivity designed into the core is controlled by the CRD system supplemented by gadolinia fuel rods. Control rods are used during the cycle partly to compensate for burnup and partly to control the power distribution. The hot excess reactivity and associated control rod patterns for the reference BWRX-300 equilibrium cycle are documented in NEDC-34044P (Reference 4.3-2).

4.3.4.3.1 Core Reactivity Effects

This section discusses the different effects associated with reactivity.

Control Reactivity

The core is capable of being made subcritical, with margin, in the most reactive condition throughout the operating cycle with the most reactive control rod fully withdrawn or rod pair associated with a single HCU postulated stuck in the full out position, with all other rods fully inserted.

The CRD system is designed to provide adequate control of the maximum excess reactivity anticipated during the plant operation. The shutdown capability is evaluated assuming a cold, xenon–free core.

The excess reactivity designed into the core is controlled by the CRD system supplemented by gadolinia fuel rods. Control rods are used during the cycle partly to compensate for burnup and partly to control the power distribution.

Control rods are the primary means of achieving shutdown in normal operations, AOOs, PIEs, and beyond design basis events and severe accident scenarios. During power operation, changes in core reactivity are controlled by movement and positioning of the control rods within the core, in fine increments, using FMCRD electric motors. Reactivity feedback mechanism is key for some PIE groups (see Table 15.2-1) in which rods are withdrawn in error.

Doppler Reactivity Coefficient

The Doppler reactivity coefficient is of high importance in reactor safety. The Doppler reactivity coefficient is a measure of the reactivity change associated with a change in the temperature of the fuel material. An increase in fuel temperature causes an increase in the absorption of resonance energy neutrons and a decrease in reactivity. The Doppler reactivity coefficient provides instantaneous negative reactivity feedback to any rise in fuel temperature, on either a gross or local basis. Although the reactivity change caused by the Doppler effect is small compared to other power–related reactivity changes during normal operation, it becomes very important during postulated rapid power excursions in which large fuel temperature changes occur. The GNF2 Doppler reactivity coefficient is negative for any operating conditions.

Moderator Void Coefficient

The moderator void coefficient of reactivity is associated with the change in moderating capability of the in-channel water. The moderator void coefficient should be large enough to prevent power oscillation due to spatial xenon changes yet small enough that pressurization transients do not unduly limit plant operation. In addition, the moderator void coefficient has the ability to flatten the radial power distribution and to provide ease of reactor control due to the void feedback mechanism. The overall void coefficient is always negative over the complete operating range because the design is under moderated. The analysis performed to calculate the moderator void coefficient used the lattice physics code TGBLA06 and the 3D core simulator PANAC11.

Table 15.2-1 focuses on events initiated from conditions off normal power operation. In the Deterministic Safety Analysis (DSA) TRACG is used for event analysis discussed in Subsection 15.5.1.

Moderator Temperature Coefficient

The moderator temperature coefficient is associated with a change in the moderating capability of the water. Once the reactor reaches the power producing range, nucleate boiling begins and the moderator temperature remains essentially constant. The moderator temperature coefficient is negative during power operation.

Boron Reactivity

For the BWRX-300, the Boron Injection System (BIS) acts as an emergency backup to the insertion of control rods to provide a diverse means of making the reactor subcritical and is discussed in Subsection 9.3.10.

Xenon Reactivity

Xenon reactivity feedback is not typically accounted for during events in DSA because BWRs do not have instability problems due to xenon.

Subsection 15.2.4.3 discusses bounding event selection for AOO and Design Basis Accident (DBA) reactivity and power distribution anomalies.

4.3.4.4 Shutdown Margins

The minimum cold SDM for the reference BWRX-300 equilibrium core is presented in Figure 4.3-8 for the most reactive rod pair associated with a single HCU and documented in NEDC-34044P (Reference 4.3-2).

Details on how the minimum cold SDM has been quantified can be found in NEDC-34039P (Reference 4.3-4).

The core must be capable of being made subcritical with margin in the most reactive condition throughout an operating cycle with the most reactive control rod fully withdrawn or rod pair associated with a single HCU postulated stuck in the full out position, with all other control rods fully inserted. This calculation is performed at cold temperatures, which are between 20°C and 286°C. The amount of reactivity in the core that is below the target eigenvalue is the cold SDM. The typical value of cold SDM required by plant Technical Specifications is 0.38% Δk/k. SDM is dependent upon the core loading. It is calculated for each plant cycle prior to the operation of that cycle.

The calculations that demonstrate compliance with this requirement are performed for every fuel reload. The results of the cycle specific calculations are documented in the reload license report for that cycle.

4.3.4.5 Criticality of Reactor During Refueling

The minimum allowable SDM will be provided in the Technical Specifications.

4.3.4.6 Thermal Limit

Figure 4.3-6 shows how the limiting (i.e., lowest) MCPR varies with cycle exposure. The shaded rectangle represents the range of expected operating limit MCPRs (OLMCPRs) that is based on various AOO analyses and analysis basis uncertainties used in the determination of the $MCPR_{99.9\%}$. Note that a definitive OLMCPR cannot be computed at this preliminary stage. As the steady-state MCPRs are well above the largest value of expected OLMCPR represented by the top of the shaded rectangle, the margin is illustrated.

The margin to thermal limits for the reference equilibrium core is depicted in Figure 4.3-7 and displays the maximum LHGR predicted during the cycle in relation to the maximum allowable steady-state LHGR. The Maximum Fraction of Limiting Power Density (MFLPD) shown in Figure 4.3-9 represents the thermal mechanical operating limit for GNF2 fuel rods as a function of exposure. The MFLPD is the maximum value of linear power density of the core to the steady state thermal mechanical limit on linear power density. MFLPD depicts the margin that includes the exposure dependency of the LHGR limit which is shown to be below 1.0.

4.3.4.7 Xenon Stability

BWRs are not susceptible to xenon instabilities. The xenon stability evaluation has been demonstrated by:

- No observed xenon instabilities in operating BWRs
- Special tests conducted on operating BWRs in an attempt to force the reactor into xenon instability
- Calculations

4.3.4.8 Thermal-Hydraulic Stability

The BWRX-300 core remains stable throughout the entire operating domain. Refer to Appendix 4A for a discussion of thermal-hydraulic stability.

4.3.4.9 Load Following Operation

Core power maneuvering is a normal occurrence for BWRs in response to several motivations (e.g., periodic surveillance testing, managing equipment performance degradation, control rod exchanges to reduce fuel duty, electricity demand). The load follow operation involves a rapid power increase or reduction. The thermal limits established for the fuel are developed to assure conformance to regulatory requirements during all modes of operation, including load following. Additionally, BWR Operating Guidelines have been developed to assist in mitigating low frequency Stress Corrosion Cracking (SCC) pellet clad interaction.

The fuel rod thermal-mechanical operating limits described in Section 4.2 have been established to be applicable to all modes of operation, including frequent power cycles associated with load following. The GNF2 fuel rod thermal-mechanical suite of analyses includes a fatigue evaluation assuming frequent power cycling and is discussed in NEDC-34042P, “BWRX-300 GNF2 Fuel Assembly Thermal-Mechanical Design Report,” (Reference 4.3-6).

The thermal-hydraulic limits established for every operating cycle (e.g., the OLMCPR) are developed to assure the Fuel Cladding Integrity Safety Limit specified in Subsection 4.4.4 is maintained during all conditions of power operation above approximately 25%.

During periods of frequent core power maneuvering, transient xenon affects the core reactivity and power distribution. The effects of transient xenon are predicted by the core monitoring function prior to performing the core power maneuver and control rod adjustments are applied to maintain the core power at specified levels, as described in Subsection 7.3.3.2.

4.3.5 Similarities to Previous Reactor Designs

The BWRX-300 reactor core operates via natural circulation flow and is considered a composite design comprised of the most desirable features developed and applied to the BWR fleet. The key features are:

1. There are 240 fuel assemblies arranged similarly to the Kernkraftwerk Mühleberg (BWR/4 in Switzerland) (KKM) reactor core.
2. The N-lattice type is that of the Advanced Boiling Water Reactor (ABWR). The N-lattice provides a larger dimension between fuel bundle centerlines than compared to earlier lattice types.
3. The core flow results from natural circulation. The driving force for natural circulation flow is the pressure head caused by the lower density water in the core and the higher density water entering below the core.
4. The CRDs are of the fine motion type that were similarly developed for the ABWR.
5. GTs are used in-lieu of the traversing incore probe system for neutron instrument calibration.

While the exact configuration of the BWRX-300 reactor core is enhanced, the configuration is similar to the BWR operating fleet, and the performance of all principal aspects has been proven in fleet application.

4.3.6 Safety Evaluation

This section discusses the regulatory requirements for the nuclear design for the BWRX-300.

Regulatory Requirements

The following GDC and PDC have aspects specific to the nuclear design and compliance is addressed as follows:

GDC 10 – Reactor design: The “reactor core” includes both the mechanical and fuel design aspects of fuel assemblies and internal/supporting core structures. Design measures are applied to the reactor core design to assure margin to Specified Acceptable Fuel Design Limits (SAFDL) during normal operating conditions and AOO event sequences.

GDC 11 – Reactor inherent protection: Requires that, in the power operating range, the prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity. GNF2 has been evaluated generically and affirmed to satisfy this requirement, and the results are summarized in NEDC-33270P (Reference 4.3-5). Additionally, NEDE-24011-P-A-31 establishes requirements governing other nuclear dynamic parameters pertaining to reactor safety. Together, compliance with these requirements assures the inherent reactivity feedback mechanisms are consistent with reactor safety.

GDC 12 – Suppression of reactor power oscillations: Requires that power oscillations that could result in conditions exceeding the SAFDLs are not possible or can be reliably and readily detected and suppressed. The BWRX-300 design addresses stability utilizing an RPV chimney that increases natural circulation core flow so that a margin to instability that could challenge the SAFDLs is maintained for all modes of operation. The power response to a sudden insertion of reactivity (typically a pressurization event) results in a coupled power and core flow response such that the SAFDLs are not exceeded. Also, the relatively small core of the BWRX-300 causes it to not be susceptible to regional modes of oscillation. Therefore, a special stability detection and trip system (i.e., a “protection system”) is not required. The natural circulation driving head provided by the BWRX-300 chimney along with the core orifice design causes core flow and power perturbations to be naturally damped.

GDC 13 – Instrumentation and control: Compliance with GDC 13 is discussed in Subsection 7.1.3.

GDC 20 – Protection system functions: Compliance with GDC 20 is discussed in Subsection 7.1.3.

GDC 25 – Protection system requirements for reactivity control malfunctions: Compliance with GDC 25 is discussed in Subsection 7.1.3.

PDC 26 – Reactivity control system redundancy and capability: Compliance with PDC 26 is discussed in Section 3.1 and Subsection 7.1.3.

GDC 27 – Combined reactivity control systems capability: The basis for GDC 27 compliance is discussed in Section 3.1.

PDC 28 – Reactivity limits: The combined features of the CRD system and the rod control system incorporate appropriate limits on the potential amount and rate of reactivity increase. The fine motion movement capability of the FMCRD allows reactivity additions from rod withdrawal to be limited. Control rod withdrawal sequences and patterns are selected to achieve optimum core performance and low individual rod worth. The BWRX-300 design prevents rod drop and rod ejection events through positive design means as discussed in Section 4.6 and Subsection 15.5.5.1. Section 4.6 also discusses design features that prevent inadvertent rod withdrawal and rapid rod withdrawal. Normal rod movement and the rod withdrawal rate are

limited by the FMCRD. The rod control system controls rod patterns and provides control rod blocks to limit the rate and amount of reactivity addition for control rod movement.

The safety analyses to demonstrate compliance with resulting transients or accidents including the steam line break and events that could challenge the reactor coolant temperature and pressure including cold water additions is provided in Section 15.5. Compliance with GDC 28 is also discussed in Subsection 7.1.3.

Section 3.1 summarizes compliance with GDC and PDC.

4.3.7 References

4.3-1 NEDE-24011-P-A-31-US, “General Electric Standard Application for Reactor Fuel,” (GESTAR II) (Supplement for United States), Global Nuclear Fuel - Americas, LLC, November 2020.

4.3-2 NEDC-34044P, “BWRX-300 GNF2 Equilibrium 12 Month Cycle Nuclear Design Report,” Revision 0, GE-Hitachi Nuclear Energy Americas, LLC, April 2024.

4.3-3 NEDC-34045P, “BWRX-300 GNF2 Fuel Bundle Information Report for Equilibrium 12-Month Cycle,” Revision 0, GE-Hitachi Nuclear Energy Americas, LLC, April 2024.

4.3-4 NEDC-34039P, “BWRX-300 GNF2 Steady State Nuclear Methods: TGBLA06/PANAC11 Application Methodology,” Revision 0, GE-Hitachi Nuclear Energy Americas, LLC, April 2024.

4.3-5 NEDC-33270P, “GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II),” Revision 11, Global Nuclear Fuel - Americas, LLC, August 2020.

4.3-6 NEDC-34042P, “BWRX-300 GNF2 Fuel Assembly Thermal-Mechanical Design Report,” Revision 0, GE-Hitachi Nuclear Energy Americas, LLC, April 2024.

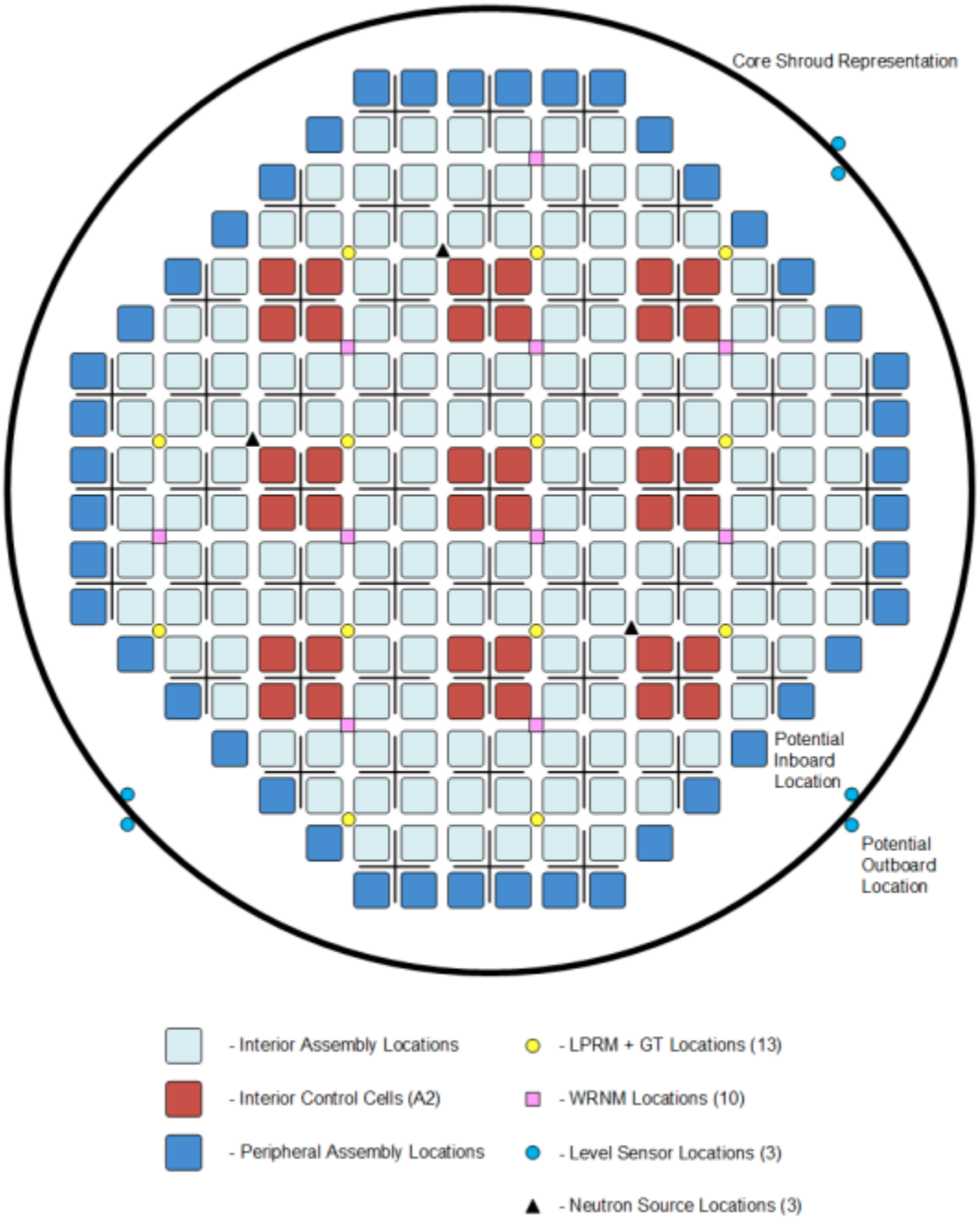


Figure 4.3-1: Core Map Instrumentation Layout

Note 1: Penetration locations are approximate.

Note 2: All available control cells are shown in red.

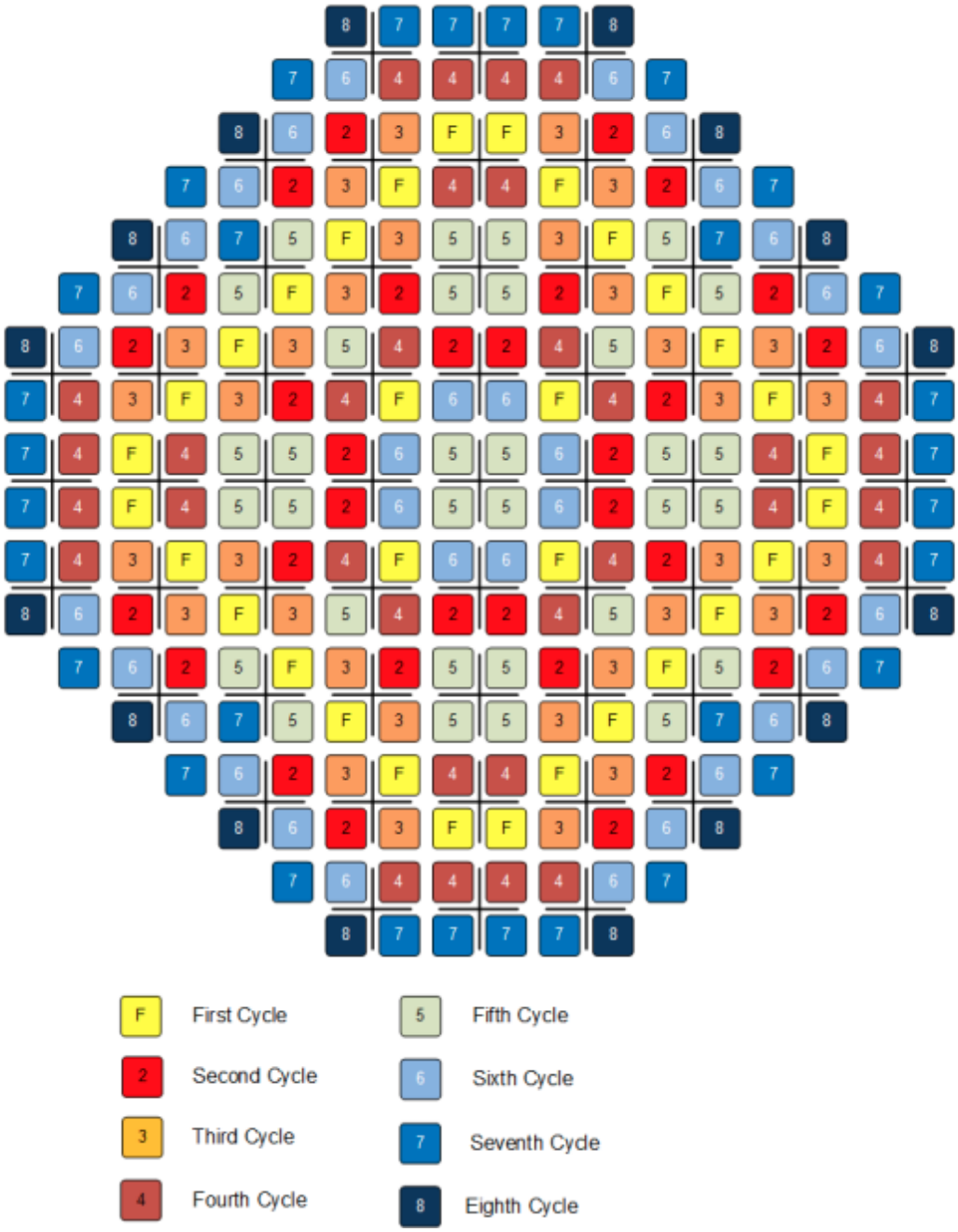


Figure 4.3-2: Core Loading Map – Reference Equilibrium Cycle

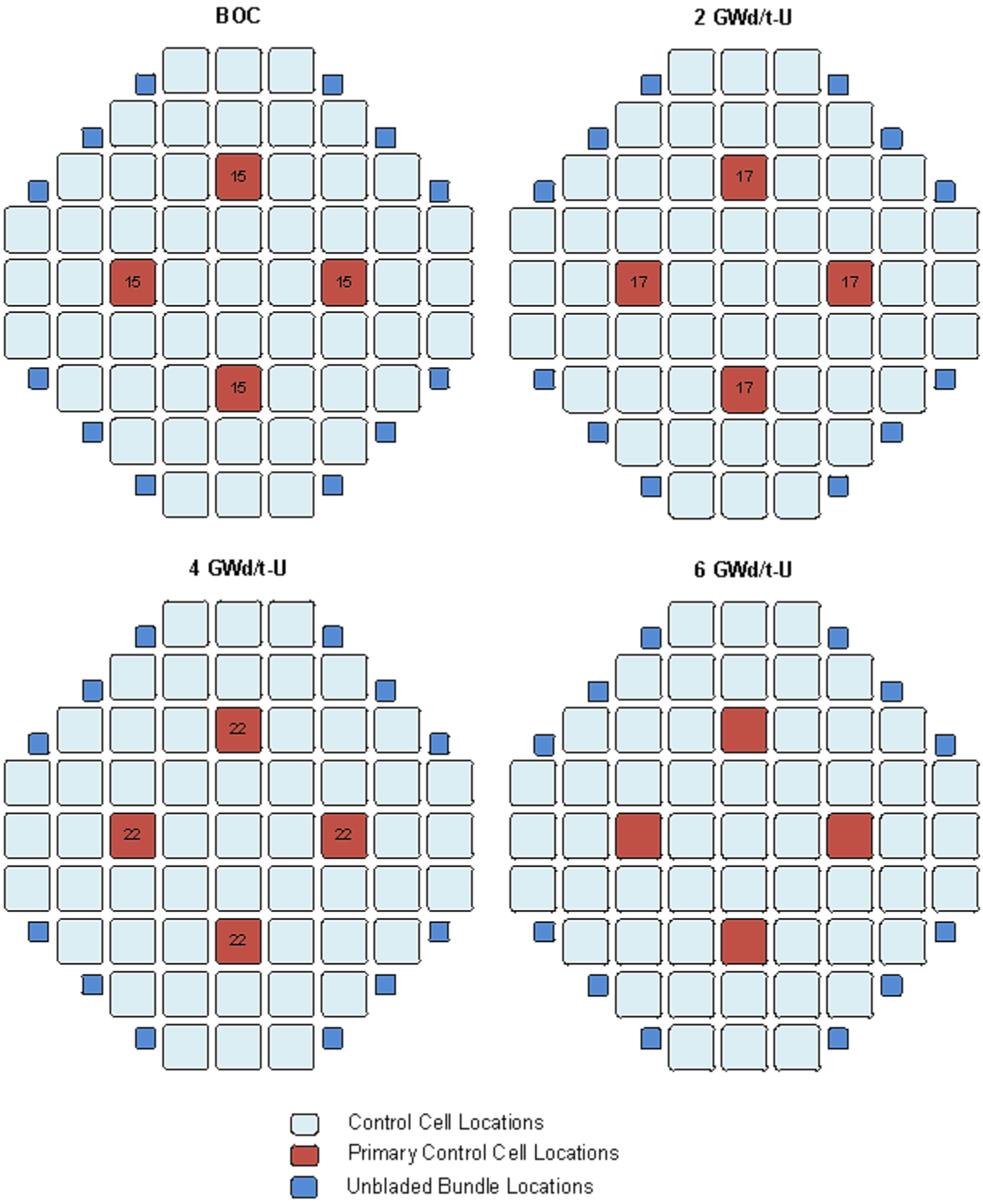


Figure 4.3-3: Representative Control Rod Patterns

Note: Control cells selected for use in the representative core depletion are shown in red.

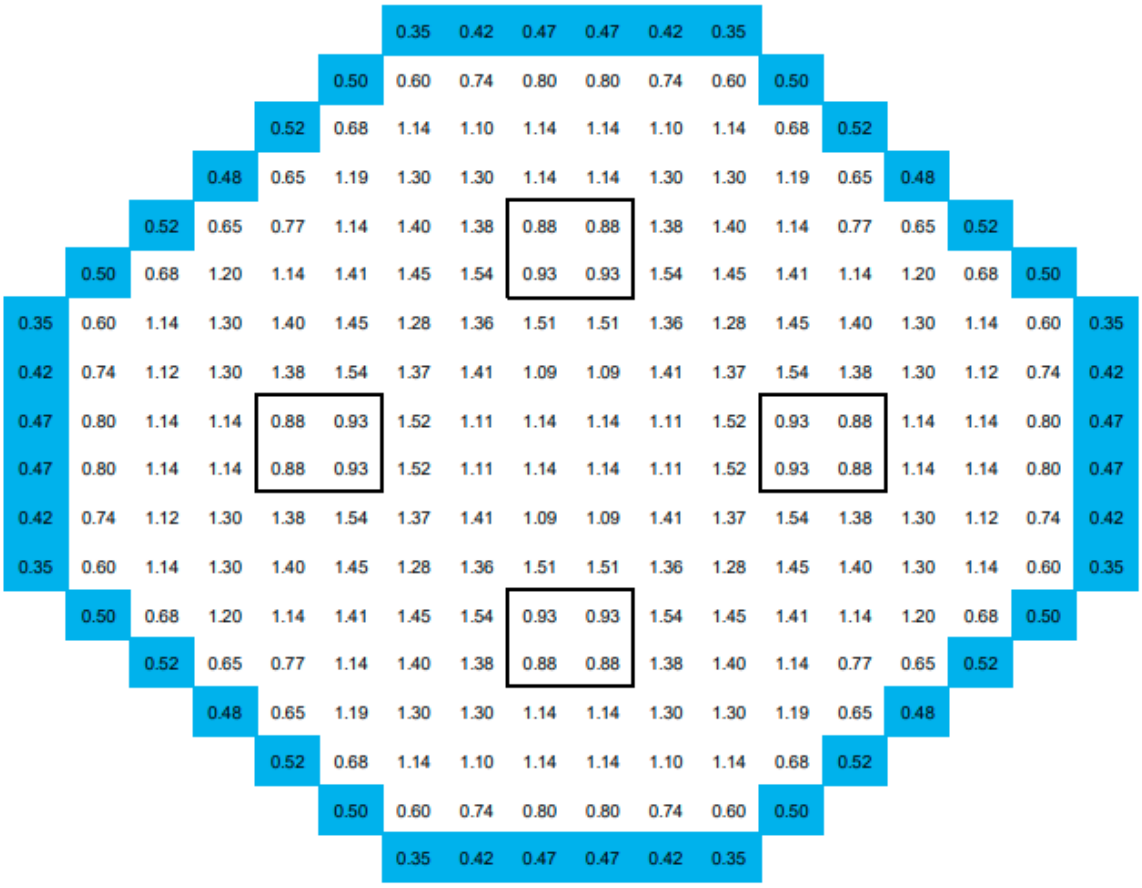


Figure 4.3-4a: Peak to Average Bundle Power Ratio at Beginning of Cycle (Average Bundle Power – 3.625 MW_{th})

Note: Control cells selected for use in the representative core depletion are outlined.

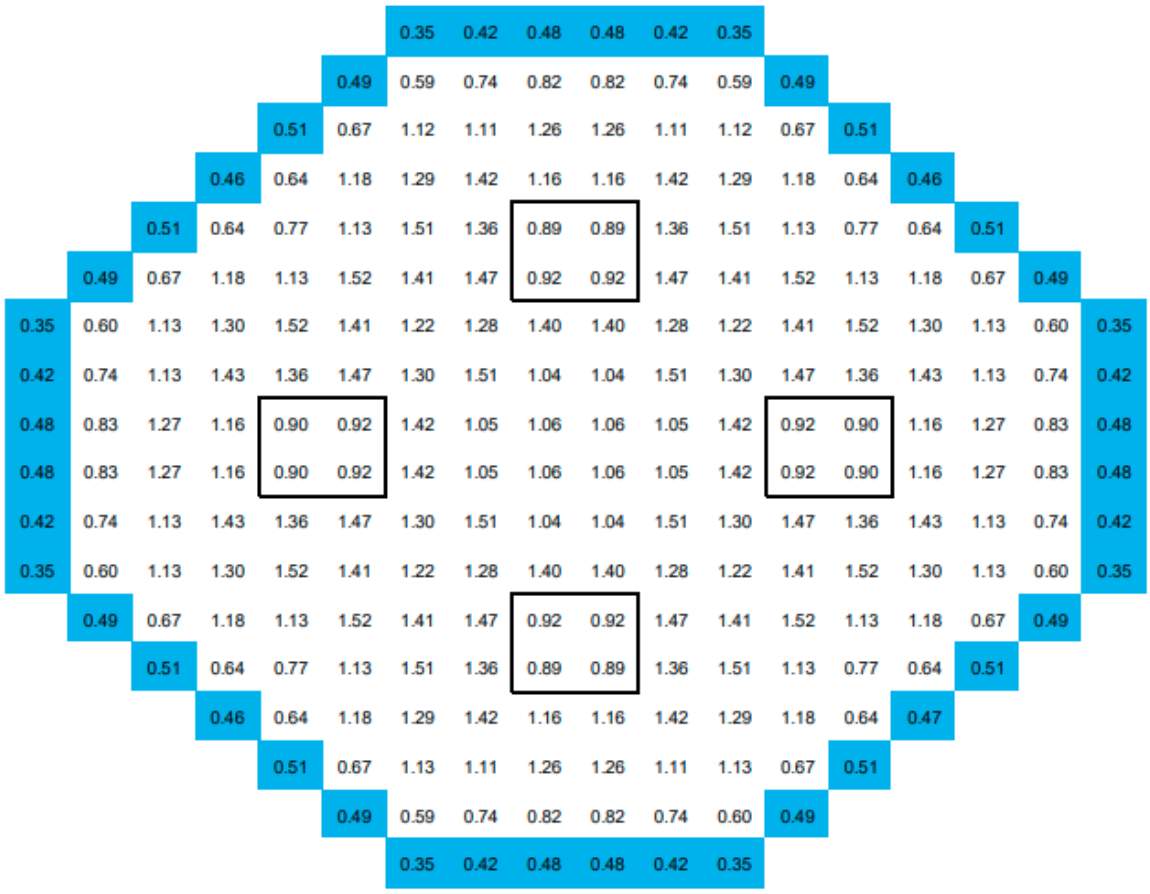


Figure 4.3-4b: Peak to Average Bundle Power Ratio at Middle of Cycle (Average Bundle Power – 3.625 MW_{th})

Note: Control cells selected for use in the representative core depletion are outlined.

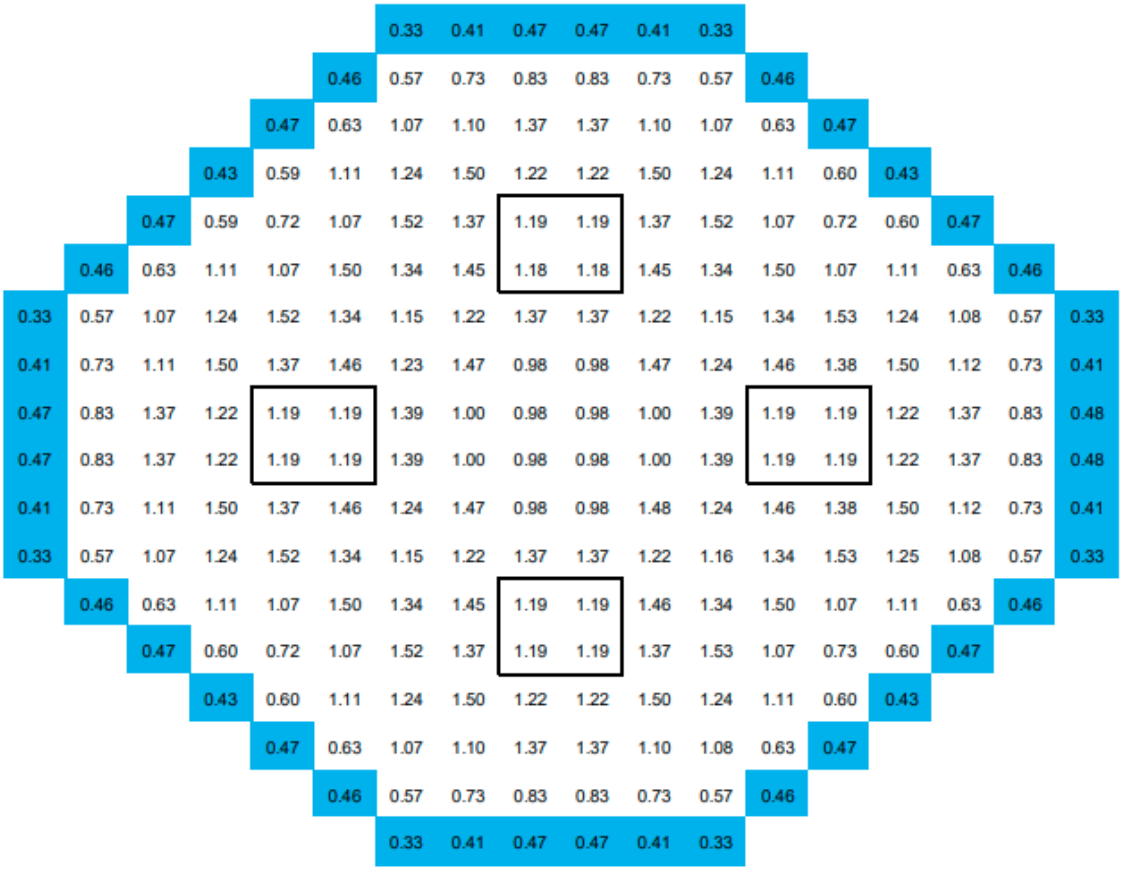


Figure 4.3-4c: Peak to Average Bundle Power Ratio at End of Cycle (Average Bundle Power – 3.625 MW_{th})

Note: Control cells selected for use in the representative core depletion are outlined.

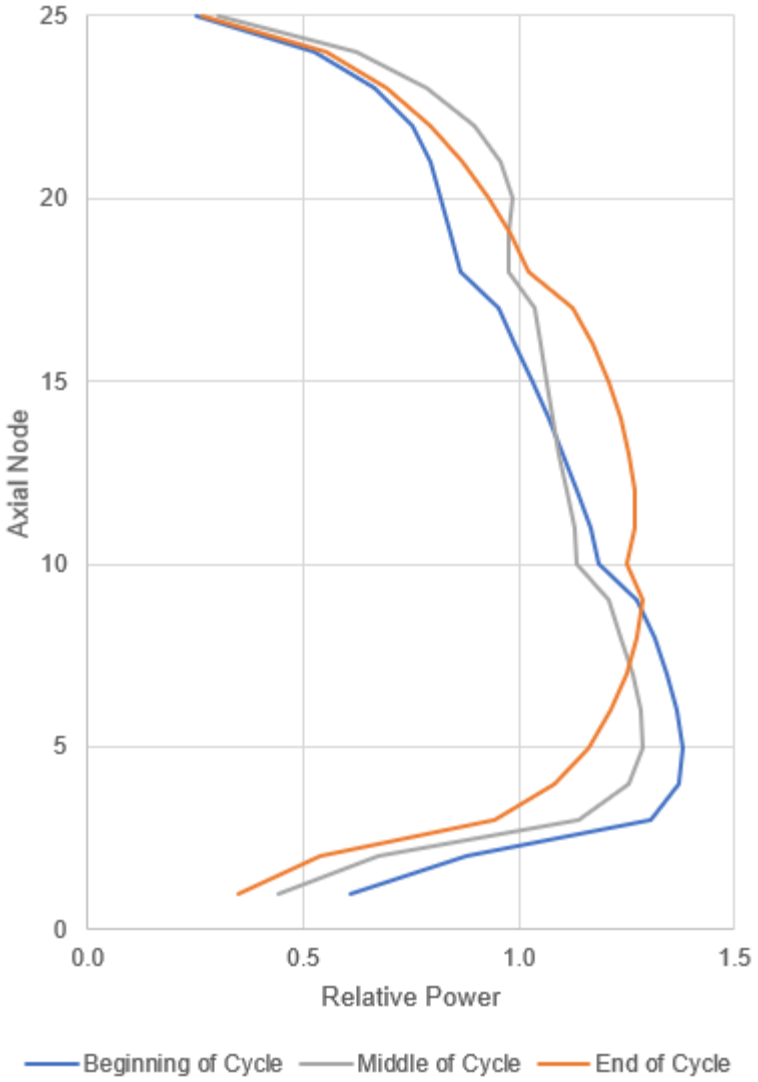


Figure 4.3-5: Axial Power Shapes

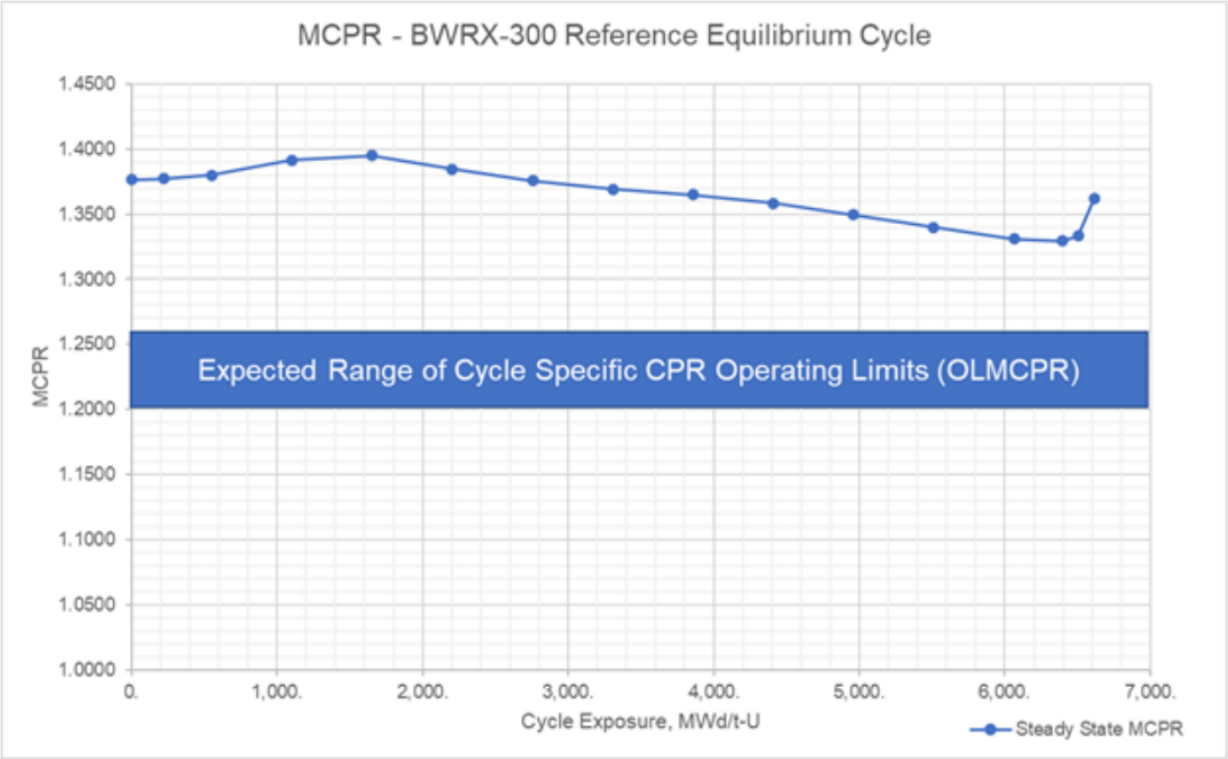


Figure 4.3-6: MCPR vs. Cycle Exposure for the Reference Equilibrium Cycle

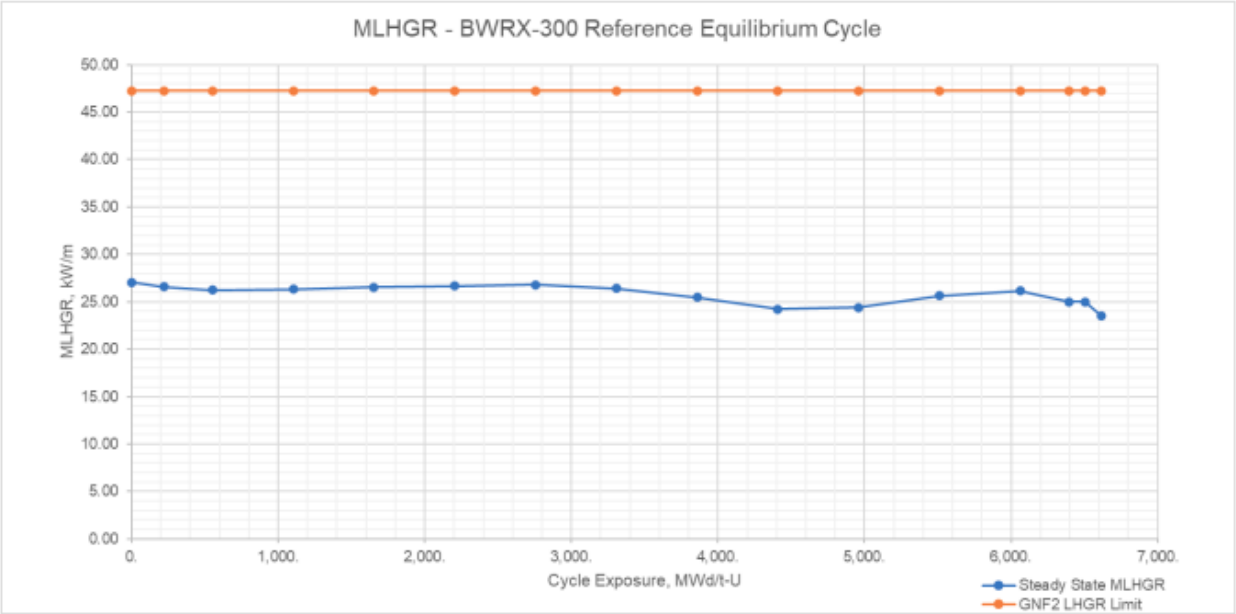


Figure 4.3-7: Reference Equilibrium Cycle Thermal Margin – Maximum Linear Heat Generation Rate

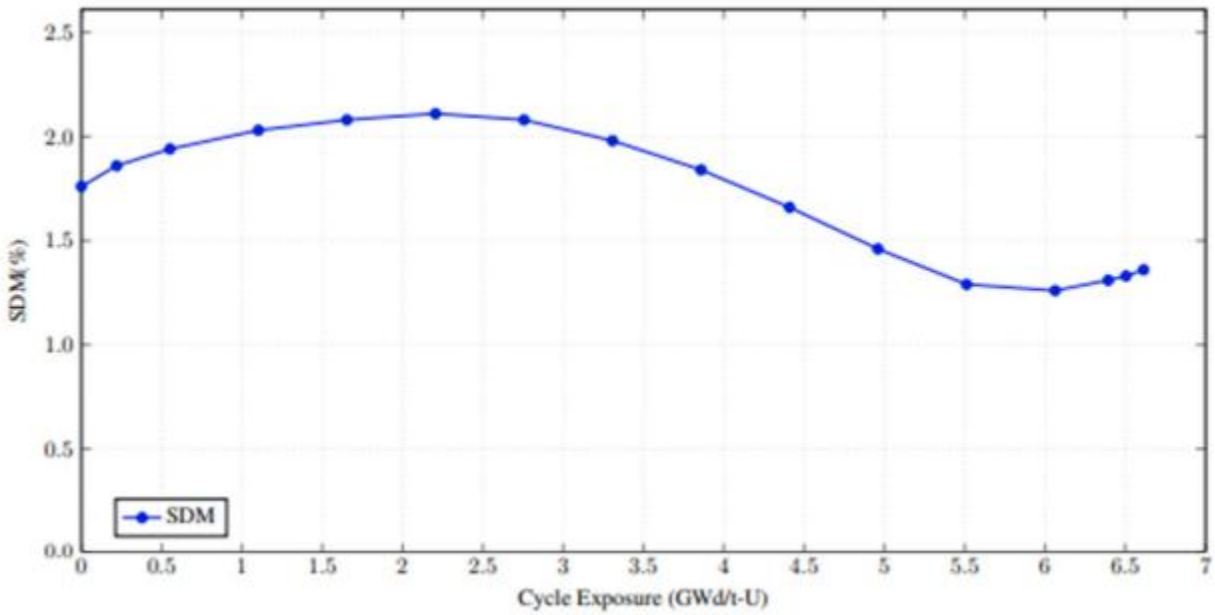


Figure 4.3-8: Shutdown Margin vs Cycle Exposure

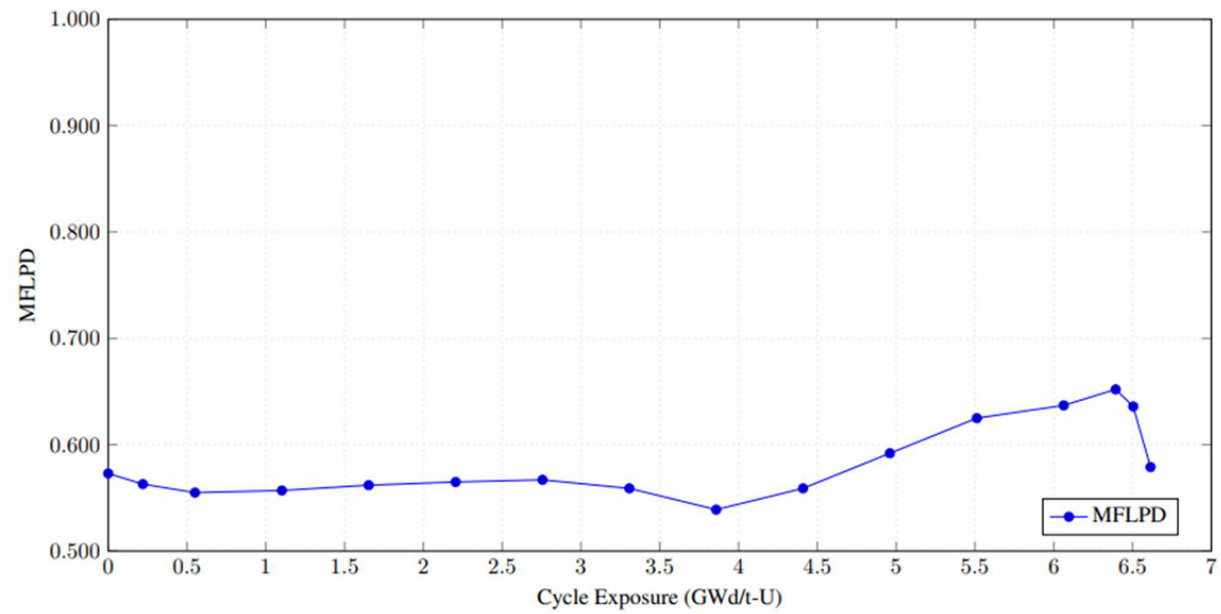


Figure 4.3-9: Maximum Fraction of Limiting Power Density (MFLPD) vs Cycle Exposure

4.4 Thermal and Hydraulic Design

The thermal and hydraulic design of the reactor core is established to provide adequate heat transfer from the fuel to the Nuclear Boiler System (NBS), Shutdown Cooling System (SDC), or ICS during normal operation and AOOs.

4.4.1 Thermal and Hydraulic Design Basis

Thermal–hydraulic design of the reactor core establishes the thermal-hydraulic safety limits for use in evaluating the safety margin relating the consequences of fuel cladding failure to public safety.

The steady state OLMCPR and thermal-mechanical operating limit are established for the most limiting AOO. The selection of bounding events is described in Subsection 15.2.4. Analysis of AOOs and the core and system performance are discussed in Subsection 15.5.3.

4.4.1.1 Critical Power Ratios

The CPR is the ratio of the predicted critical power (bundle power at which some point within the assembly experience onset of boiling transition) to the actual power of the particular fuel assembly, both evaluated at the same pressure, mass flux and inlet subcooling. The MCPR is defined as the minimum CPR for any fuel assembly within a core and is the figure of merit to represent the reactor thermal performance or margin.

The objective for normal operation and AOOs is maintaining nucleate boiling and precluding boiling transition.

CPR limits discussed in Subsection 4.3.1 are specified for maintaining adequate margin to the onset of the boiling transition during normal operation and the most limiting AOO. Adequate margin is defined to be a 95% probability at a 95% confidence level that no fuel rods are susceptible to boiling transition.

The evaluations of the critical power methods are discussed in Subsection 4.4.4. Analysis of AOOs and the core and system performance are discussed in Subsection 15.5.3.

4.4.1.1.1 Fuel Cladding Integrity Safety Limit

The generation of the MCPR limit requires a statistical analysis of each reload core near the limiting MCPR condition. The statistical analysis is used to determine the MCPR corresponding to the transient design requirement. The MCPR Fuel Cladding Integrity Safety Limit applies not only for core wide AOOs but is also applied to the localized rod withdrawal error AOO.

The Fuel Cladding Integrity Safety Limit is defined as the MCPR that ensures there is a 95% probability at a 95% confidence level that no fuel rods are susceptible to boiling transition. This limit is also referred to as Safety Limit MCPR or $MCPR_{95/95}$. This value is based only on the CPR correlation uncertainty and is therefore only dependent on fuel type. The maximum $MCPR_{95/95}$ value for any fuel product in the core establishes a lower limit on the $MCPR_{95/95}$ that can be applied in any cycle.

4.4.1.1.2 $MCPR_{99.9\%}$

The $MCPR_{99.9\%}$ is determined on a cycle-specific basis to support the determination of the OLMCPR. The $MCPR_{99.9\%}$ ensures that 99.9% of the fuel rods in the core are not susceptible to boiling transition. It is a function of the core design, plant system uncertainties, and power distribution related uncertainties.

4.4.1.1.3 Operating Limit Minimum Critical Power Ratio

The OLMCPR is established to provide adequate assurance that the cycle-specific fuel cladding integrity safety limit is not exceeded for any moderate frequency AOO. This operating requirement is obtained by addition of the maximum ΔCPR value for the most limiting AOO (including any imposed adjustment factors) from conditions postulated to occur at the plant to the cycle-specific fuel cladding integrity safety limit.

4.4.1.2 Linear Heat Generation Rate Operating Limit

The LHGR operating limit basis is described in Subsection 4.3.1. The LHGR operating limit is the maximum LHGR (expressed in kW/ft) in any fuel rod allowed by the technical specifications for a given fuel type. The LHGR operating limit is attained when the product of the local, radial, and axial peaking factors in an axial segment of a fuel bundle equals the total peaking factor for that fuel type. The adequacy of LHGR limits is evaluated for the most severe AOOs providing assurance that fuel rods are not expected to exhibit cladding failure.

4.4.1.3 Void Fraction Distribution Basis

The constitutive correlations for shear and heat transfer in TRACG use a relatively simple flow regime map, which consists basically of two distinct patterns: (a) liquid-continuous at low void fractions and (b) vapor-continuous at high void fractions with a transition zone in between. The liquid-continuous regime applies to the single-phase liquid flow, bubbly/churn and inverted annular flows. The vapor-continuous regime applies to the annular, dispersed droplet flow and single-phase vapor flow regimes. The transition regime involves churn to annular and churn to droplet; depending on the void fraction, flow rate and other variables.

4.4.1.4 Core Pressure Drop and Hydraulic Loads

The components of bundle pressure drop considered are friction, local, elevation and acceleration pressure drops. Pressure drop measurements made in operating reactors confirm that the total measured core pressure drop, and calculated core pressure drop are in good agreement.

The TRACG methods for core pressure drop modeling are described in NEDE-32176P, “TRACG Model Description,” (Reference 4.4-1). The models used in the core design analysis are described in NEDC-34039P, “BWRX-300 GNF2 Steady State Nuclear Methods: TGBLA06/PANAC11 Application Methodology,” (Reference 4.4-2). The fuel design specific loss coefficients and assembly pressure drop models are developed from and confirmed by data from full scale testing of prototypical assemblies spanning the range of hydraulic conditions where hydraulic models are applied. The adequacy of the pressure drop model applied to GNF2 fuel is summarized in GNF2 pressure drop characteristics and is inclusive of the BWRX-300 operating conditions in GNF2 pressure drop characteristics, NEDC-34040P “BWRX-300 GNF2 Fuel Assembly Pressure Drop Characteristics,” (Reference 4.4-3).

Hydraulic loads are determined based on the reactor internal pressure differences. The TRACG computer code is used to analyze the transient conditions within the reactor vessel following AOOs.

4.4.1.5 Core Coolant Flow Distribution

The flow distribution to the fuel assemblies and bypass flow paths is calculated on the assumption that the pressure drop across all fuel assemblies and bypass flow paths is the same. Pressure drop measurements made in operating reactors confirm that the total measured core pressure drop and calculated core pressure drop are in good agreement. There is reasonable assurance, therefore, that the calculated flow distribution throughout the core is in close agreement with the actual flow distribution of an operating reactor.

4.4.1.6 Fuel Heat Transfer

Fuel and cladding temperatures are calculated assuming steady-state, radial-only heat transfer from the pellet, across the pellet-cladding gap, through the cladding base metal, across the oxide and crud layers, and across the water film to the coolant.

4.4.2 Thermal and Hydraulic Analytical Methods

Reactor core thermal-hydraulic analytical methods are described below.

4.4.2.1 Fuel Bundle Critical Power Methods

The critical power is bundle power at which some point within the assembly experiences onset of boiling transition. Maintaining the bundle power below the critical power during steady-state operation and AOOs precludes the onset of boiling transition and satisfies the SAFDL pertaining to heat transfer from the fuel to the coolant. The methods applied in determining the bundle critical power and the associated operating limits are described below.

4.4.2.1.1 Fuel Bundle Critical Power Performance

The bundle critical power performance methodology was originally described in NEDE-10958P-A, "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, Licensing Report," (Reference 4.4-4). This original methodology evolved into the current form of the correlation, i.e., the GEXL correlation. The GEXL correlation is a critical quality and boiling length correlation used to predict the occurrence of boiling transition in BWR fuel. Each fuel bundle design has a specific set of correlation coefficients developed from full-scale test data. The specific GEXL correlation applied in analyzing GNF2 for all BWR types, including BWRX-300, is designated in NEDC-33292P, "GEXL17 Correlation for GNF2 Fuel," (Reference 4.4-5). The GEXL17 correlation application range established for fleetwide application envelopes the hydraulic conditions that the BWRX-300 experiences during normal operation and AOOs.

4.4.2.1.2 Fuel Cladding Integrity Safety Limit Method

The generation of the MCPR limit requires a statistical analysis of each reload core near the limiting MCPR condition. The statistical analysis is used to determine the MCPR corresponding to the transient design requirement given in NEDE-24011-P-A-31-US, "General Electric Standard Application for Reactor Fuel," (Reference 4.4-6). The MCPR Fuel Cladding Integrity Safety Limit applies not only for core wide AOOs but is also applied to the localized rod withdrawal error AOO.

The fuel cladding integrity safety limit, named the $MCPR_{95/95}$, ensures there is a 95% probability at a 95% confidence level that no fuel rods are susceptible to boiling transition using a limit that is derived from comparing the predicted critical power to the experimental data for a specific fuel bundle design. The Experimental CPR is defined as the ratio of the calculated critical power as determined by the GEXL correlation to the experimental critical power.

The determination of the $MCPR_{95/95}$ is further described in NEDE-24011-P-A-31, "General Electric Standard Application for Reactor Fuel," (Reference 4.4-7).

4.4.2.1.3 $MCPR_{99.9\%}$

The cycle specific $MCPR_{99.9\%}$ limit is established as described in Appendix C of NEDC-34039P (Reference 4.4-2).

4.4.2.1.4 Operating Limit Minimum Critical Power Ratio

BWRX-300 AOOs are analyzed using the TRACG model described in NEDE-32176P (Reference 4.4-1). The application of TRACG to AOO analyses is described in NEDC-34043P,

"BWRX-300 TRACG Application," (Reference 4.4-8) along with the process to calculate the ΔCPR/ICPR.

The operating limit MCPR is determined by adding the change in the CPR for the limiting analyzed AOO to the Safety Limit MCPR. The OLMCPR calculational procedure and descriptions of the limiting AOO events are documented, respectively, in NEDE-24011-P-A-31-US. These limiting events were established based on sensitivity studies of bundle and plant parameters. Because the operating limit MCPR is dependent upon the core loading pattern, this limit is cycle dependent for each plant and is calculated just prior to operation of the cycle.

Subsection 15.5.3 provides the analysis of limiting AOOs.

4.4.2.2 Linear Heat Generation Rate Operating Limits Method

The LHGR operating limit methods are described in Subsection 4.3.1. Margin to design limits for circumferential cladding strain and centerline fuel temperature is evaluated for AOOs in accordance with NEDC-34043P (Reference 4.4-8).

4.4.2.3 Void Fraction Distribution Method

The TRACG void fraction model is described in NEDE-32176P (Reference 4.4-1). The core simulator model is described in NEDC-34039P (Reference 4.4-2). This report shows that the methods for core modeling have undergone extensive efficacy demonstrations and uncertainty quantifications.

4.4.2.4 Core Pressure Drop and Hydraulic Loads Methods

The components of bundle pressure drop considered are friction, local, elevation and acceleration pressure drops. Pressure drop measurements made in operating reactors confirm that the total measured core pressure drop and calculated core pressure drop are in good agreement.

The TRACG methods for core pressure drop modeling are described in NEDE-32176P (Reference 4.4-1). The models used in the core design analysis are described in NEDC-34039P (Reference 4.4-2). The adequacy of the pressure drop model applied to GNF2 fuel is summarized in NEDC-34040P (Reference 4.4-3).

The TRACG computer code is used to analyze the transient conditions within the reactor vessel following AOOs.

4.4.2.5 Core Coolant Flow Distribution Methods

The core coolant flow distribution methods used in TRACG are described in NEDE-32176P (Reference 4.4-1). TRACG treats fuel channels as One-Dimensional (1D) components, but the vessel is modeled as a 3D component.

The flow distribution to the fuel assemblies and bypass flow paths is calculated on the assumption that the pressure drop across all fuel assemblies and bypass flow paths is the same. This assumption has been confirmed by measuring the flow distribution in BWRs. The components of bundle pressure drop considered are friction, local, elevation, and acceleration. Pressure drop measurements made in operating reactors confirm that the total measured core pressure drop and calculated core pressure drop are in good agreement. The core monitoring function is discussed in Subsection 7.3.3.2.

4.4.2.6 Fuel Heat Transfer Methods

The PRIME code performs coupled thermal and mechanical interaction analyses. The incremental finite element mechanics model performs an axisymmetric radial mechanical interaction analysis to determine pellet and cladding stresses and strains. The methodology for

fuel cladding, gap and pellet heat transfer is described in NEDC-33256P-A, “The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance Part 1 – Technical Bases,” (Reference 4.4-9).

4.4.3 Thermal and Hydraulic Design Description

Several metrics of interest for the BWRX-300 are compared against a typical BWR/6 reactor, ABWR, and Economic Simplified Boiling Water Reactor (ESBWR) in Table 1.3-1 and Table 4.4-1. Figure 4.4-1 is an illustrative power/flow map. A final power/flow map will be developed during the operating license application; however, it is noted that the operating region for BWRX-300 is a single natural circulation curve. Preliminary power, flow, and feedwater temperatures are presented in Table 4.4-2.

4.4.3.1 Reactor Coolant System

The Reactor Coolant System (RCS) maintains adequate core cooling conditions (i.e., pressure, temperature, and coolant flow rate) for the fuel in power operation. It includes the RPV, the steam lines and FW lines up to and including the Outboard Containment Isolation Valves.

The RCS is described in Chapter 5 and is shown in Figure 5.1-1. The BWRX-300 thermal hydraulic design is similar to operating BWRs except that it does not require recirculation pumps and their associated piping. Circulation of the reactor coolant through the BWRX-300 core is accomplished via natural circulation. The core flow varies according to the power level because fluid density changes with the power level. Therefore, a core power-flow map reduces to a single line and there is no active control of the core flow at any given power level, as shown in Figure 4.4-1.

4.4.3.2 Core Hydraulics

Accurate prediction of bundle flow and power distribution is important in the calculation of margin to the thermal limits of each fuel bundle. Flow pressure drop characteristics are included in plant cycle specific analyses for the calculation of the OLMCPR. Because of the channeled configuration of BWR fuel assemblies, there is no bundle-to-bundle cross flow inside the core. The only consideration pertaining to hydraulic compatibility of various bundle types in a core is the bundle inlet flow rate variation and its effect on OLMCPR. The coupled thermal–hydraulic–nuclear analyses are performed each cycle to determine fuel bundle flow and power distribution. The analyses use the various bundle pressure loss coefficients to determine the flow distribution required to maintain total core pressure drop boundary conditions applied to all fuel bundles. The margin to thermal limits of each fuel bundle is determined using this consistent set of calculated bundle flow and power.

The pressure drop characteristics of the GNF2 spacers are based on the pressure drop data from full-scale testing of the GNF2 fuel assembly. Production spacers were used in the full-scale test assembly with no modifications. The measured pressure drops include static head, wall friction, acceleration pressure drop, and form losses. When compared with the predicted pressure drop, it is concluded that the models and methods used for the determination of pressure drop in the GNF2 fuel assembly accurately predict the test data over a wide range of power and flow conditions.

4.4.3.3 Core Response in Transient and Accident Analysis

The thermal response of the core for accident and off-normal condition is provided in Subsection 15.2.4. The AOO and DBA analyses are presented in Subsections 15.5.3 and 15.5.4, respectively.

4.4.3.4 Physics and Reactivity Coefficients Including Effects of Power Coefficient of Reactivity

There are two primary reactivity coefficients that characterize the dynamic behavior of BWRs; these are the Doppler reactivity coefficient and the moderator void reactivity coefficient. Also associated with the BWR are a power reactivity coefficient and a temperature coefficient. The power coefficient is a combination of the Doppler and moderator void reactivity coefficients in the power operating range, and the temperature coefficient is merely a combination of the Doppler and moderator temperature coefficients. Power and temperature coefficients are not specifically calculated for reload cores.

Power control to ensure compliance with LHGR limits, including aspects of loss of reactivity control, is based on the use of cruciform shaped control rods (which are sometimes referred to as blades). Fast acting shutdown capability is provided using stored hydraulic energy HCUs as a diverse motive force relative to the FMCRDs, as described in Section 4.6. The core monitoring function in concert with reactor operators ensure compliance with requirements implemented by the thermal limits described in Subsection 4.3.1.

4.4.3.5 Operation Limits and Conditions for Core, Core Instrumentation and Control, and Nuclear Fuel

The core design for any operating cycle is based on management of thermal limit margins for fuel design.

A cycle specific OLMCPR is established by considering the limiting AOOs for each operating cycle. This is calculated as a function of exposure. Reports are prepared for each refueling cycle interval.

4.4.3.6 Load Following Characteristics

The BWRX-300 is a natural circulation BWR. The power-flow map depicted in Figure 4.4-1 illustrates that the core flow is fairly constant in the normal range of core thermal powers associated with load following. As the reactor power is the principal parameter that varies, and reactor coolant flow is fairly constant, compliance to thermal limits is readily maintained during periods of low power operation. See Subsection 4.3.4.9 for information on load following operation.

4.4.4 Thermal and Hydraulic Evaluation

This section provides evaluations for thermal hydraulics.

4.4.4.1 Critical Power Evaluations

Compliance to representative steady-state MCPR operating limits is demonstrated for a typical simulation of an equilibrium cycle described in Section 4.3. The typical OLMCPR evaluation process is outlined in the sections that follow.

4.4.4.1.1 Fuel Cladding Integrity Safety Limit Evaluation

The GNF2 Fuel Cladding Integrity Safety Limit value is 1.07.

4.4.4.1.2 MCPR99.9%

The $MCPR_{99.9\%}$ is evaluated for a specific core design with uncertainties documented in NEDC-34039P (Reference 4.4-2). The $MCPR_{99.\%9}$ limit is computed on a cycle-specific basis.

4.4.4.1.3 Minimum Critical Power Ratio Operating Limit Evaluation

The OLMCPR is computed on a cycle-specific basis and reported in a cycle specific Core Operating Limits Report (COLR). The expected range of cycle-specific OLMCPRs for the reference BWRX-300 equilibrium cycle is depicted in Figure 4.3-6.

4.4.4.2 Linear Heat Generation Rate Evaluations

Compliance to steady-state LHGR operating limits is demonstrated for the reference equilibrium cycle in Section 4.3 and NEDC-34044P, "BWRX-300 GNF2 Equilibrium 12 Month Cycle Nuclear Design Report," (Reference 4.4-10). The AOO analysis for the reference equilibrium cycle is documented in Subsection 15.5.3. Compliance to design limits during AOO events are confirmed on a cycle-specific basis.

4.4.4.3 Void Fraction Distribution Evaluation

The TRACG void fraction model is described in NEDE-32176P (Reference 4.4-1). The core simulator model is described in NEDC-34039P (Reference 4.4-2). This report shows that the methods for core modeling have undergone extensive efficacy demonstrations and uncertainty quantifications.

Results demonstrate excellent and consistent predictive performance for a wide range of BWR operating conditions. These results fully support the method application to BWRX-300.

Relevant uncertainties for use in safety evaluations are also described, quantified, and justified as appropriate for use with BWRX-300. The consistency of predictive performance shown in these results provides confidence that operation at higher power-to-flow ratios with associated higher exit void fraction conditions that will be observed in BWRX-300 compared to the current fleet should not be expected to increase power distribution uncertainties. Representative values for the core average axial void fraction are depicted in Figure 4.4-2.

4.4.4.4 Core Pressure Drop and Hydraulic Loads Evaluations

The expected operating pressure for the BWRX-300 is within the qualification basis of the pressure drop methods. The $MCPR_{99.9\%}$ calculation method includes provision for pressure drop uncertainty.

4.4.4.5 Core Coolant Flow Distribution Evaluation

The core coolant flow distribution (i.e., the inlet flow to each fuel assembly) is determined by the coupled nuclear and thermal hydraulic steady-state methods in NEDC-34039P (Reference 4.4-2) and in the design and analysis of the reference core nuclear design summarized in Section 4.3.

4.4.4.6 Fuel Heat Transfer Evaluations

Fuel heat transfer evaluations are dependent upon the reactor state. The calculation of fuel heat transfer is described in NEDC-34043P (Reference 4.4-8).

4.4.5 Evaluation of the Validity of Thermal and Hydraulic Design Techniques

The TRACG method described in Subsection 4.4.2 demonstrates accurate model system performance in BWRs. The thermal hydraulic design bases evaluated for the BWRX-300 described in Subsection 4.4.1 are applicable and adequate to those established for the operating fleet. The BWRX-300 reactor core was developed from applying qualified analytical methods and the results demonstrate compliance to the design bases that are used in BWRs.

4.4.6 Core Monitoring

The 3D Core Thermal Power Distribution Monitor is discussed in Section 7.3.

4.4.7 Safety Evaluation

This section discusses the regulatory requirements for the thermal and hydraulic design for the BWRX-300.

Regulatory Requirements

The following GDC have aspects specific to the thermal and hydraulic design and compliance is addressed as follows:

GDC 10 – Reactor design: GDC 10 requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs. As described in Subsection 4.4.2.1, maintaining the bundle power below the critical power during steady-state operation and AOOs precludes the onset of boiling transition and satisfies the SAFDL pertaining to heat transfer from the fuel to the coolant.

GDC 12 – Suppression of reactor power oscillations: GDC 12 requires that power oscillations that could result in conditions exceeding the SAFDLs are not possible or can be reliably and readily detected and suppressed. Thermal-hydraulic stability is discussed in Appendix 4A. Additionally, compliance with GDC 12 is discussed in Subsection 4.1.2 of NEDC-33912P-A, "BWRX-300 Reactivity Control," (Reference 4.4-11).

Section 3.1 summarizes compliance with GDC.

4.4.8 References

4.4-1 NEDE-32176P, "TRACG Model Description," Revision 4, GE-Hitachi Nuclear Energy Americas, LLC, January 2008.

4.4-2 NEDC-34039P, "BWRX-300 GNF2 Steady State Nuclear Methods: TGBLA06/PANAC11 Application Methodology," Revision 0, GE-Hitachi Nuclear Energy Americas, LLC, April 2024.

4.4-3 NEDC-34040P, "BWRX-300 GNF2 Fuel Assembly Pressure Drop Characteristics," Revision 0, GE-Hitachi Nuclear Energy Americas, LLC, April 2024.

4.4-4 NEDE-10958-PA, "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, Licensing Report," GE-Hitachi Nuclear Energy Americas, LLC, January 1977.

4.4-5 NEDC-33292P, "GEXL17 Correlation for GNF2 Fuel," Revision 3, GE-Hitachi Nuclear Energy Americas, LLC, June 2009.

4.4-6 NEDE-24011-P-A-31-US, "General Electric Standard Application for Reactor Fuel," (GESTAR II) (Supplement for United States), Global Nuclear Fuel - Americas, LLC, November 2020.

4.4-7 NEDE-24011-P-A-31, "General Electric Standard Application for Reactor Fuel," (GESTAR II), Revision 31, Global Nuclear Fuel – Americas, LLC, November 2020.

4.4-8 NEDC-34043P, "BWRX-300 TRACG Application," Revision 0, GE-Hitachi Nuclear Energy Americas, LLC, May 2024.

4.4-9 NEDC-33256P-A, "The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance Part 1 – Technical Bases," Revision 2, Global Nuclear Fuel - Americas, LLC, October 2021.

4.4-10 NEDC-34044P, "BWRX-300 GNF2 Equilibrium 12 Month Cycle Nuclear Design Report," Revision 0, GE-Hitachi Nuclear Energy Americas, LLC, April 2024.

4.4-11 NEDC-33912P-A, “BWRX-300 Reactivity Control,” Revision 2, GE-Hitachi Nuclear Energy Americas, LLC, February 2023.

Table 4.4-1: Typical Thermal–Hydraulic Design Characteristics of the Reactor Core

General Operating Conditions	BWR/6	ABWR	ESBWR	BWRX-300
Reference design thermal output (MW_{th})	3579	3926	4500	870
Power level for engineered safety features (MW_{th})	3730	4005	4590	887
Steam flow rate (kg/s)	1940	2122	2433	503
Core coolant flow rate (kg/s)	13104	14502	8763-10376	1890
Feedwater flow rate (kg/s)	1936	2118	2451	507
System pressure, nominal in steam dome (kPa)	7171	7171	7171	7171
Coolant saturation temperature at core design pressure (°C)	288	288	288	288
Average power density (kW/L)	54.1	50.6	54.3	39.6
Core inlet enthalpy (kJ/kg)	1227	1227	1180-1196	1218
Core inlet temperature (°C)	278	278	269-272	276
Core maximum exit voids within assemblies (%)	79.0	75.1	90.0	90.0
Core average void fraction, active coolant	0.414	0.408	0.530	0.510
Total core pressure drop (kPa)	182.0	168.2	71.8	71.98
Core support plate pressure drop (kPa)	151.7	137.9	43.1	49.6

Table 4.4-2: Expected Beginning of Cycle Total Core Flow and Nominal Feedwater Temperature versus Power

Power (%)	Core Flow (%)	Feedwater Temperature	
		°C	°F
15	69.2	136	277
25	82.7	160	319
50	98.0	222	431
75	101.8	234	454
90	102.5	242	467
100	103.1	242	467
103	102.4	242	467

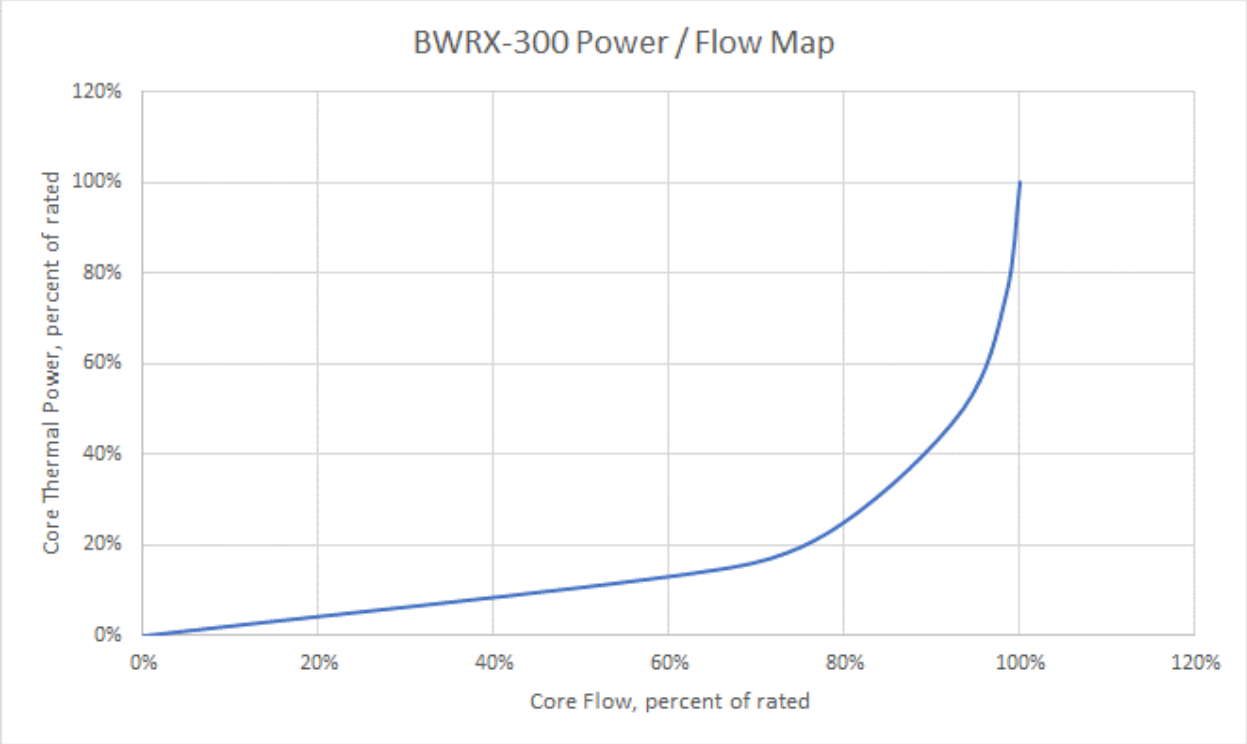


Figure 4.4-1: BWRX-300 Power / Flow Map

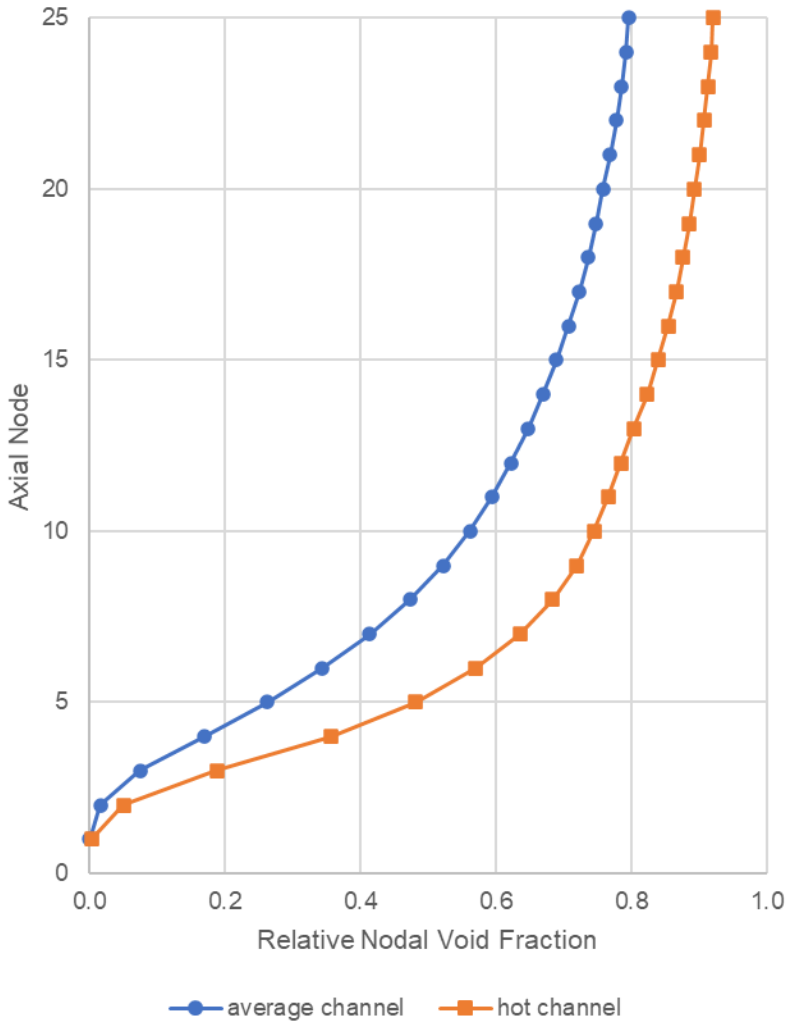


Figure 4.4-2: Relative Void Fraction for Analyzed Node (Hot Channel and Average Channel)

4.5 Reactor Material

This section describes the CRD System Structural Materials (Subsection 4.5.1) and the Reactor Internals Materials (Subsection 4.5.3).

4.5.1 Control Rod Drive System Structural Materials

This section discusses materials used in the Control Rod Drive Mechanism (CRDM) and extends to the coupling interface with control rod assemblies in the reactor vessel. The materials comprising the fuel and control rod assemblies are described in Section 4.2.

The RCPB components of the CRD, consist of FMCRD middle flange including the ball check valve, and the lower component housing, which enclose the lower part of the drive. These components are made with 300 series stainless steel materials compatible with the reactor coolant in accordance with ASME BPVC, Section III, “Rules for Constructions of Nuclear Facility Components,” (Reference 4.5-1). Compliance with RG 1.84, which addresses ASME BPVC, Section III Code Cases, is discussed in Subsection 5.2.1.2, and Code Cases, if used, are listed in Table 5.2-1. The installation bolts used to attach the middle flange and the lower component housing to the CRD housing are low alloy steel material as shown in Table 5.2-2a.

The metallic structural components of the CRD mechanism are made from four types of materials: 300 series stainless steel, Nickel-Chrome-Iron alloy X-750, XM-19 stainless steel and 17-4 PH stainless steel materials. The properties of the non-primary pressure retaining materials selected for the CRD mechanism are equivalent to those given in Parts A, B and D of Section II of ASME BPVC, Section II, “Materials,” (Reference 4.5-2) and Section III (Reference 4.5-1) or are included in RG 1.84, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III.” The requirements are intended to address the cold work concerns–- both cold work from forming operations and abrasive grinding. Surface cold work can be detrimental to SCC resistance of austenitic stainless steels. In conformance with RG 1.44, “Control of the Processing and Use of Stainless Steel,” cold worked 300 series austenitic stainless steels are not used except for minor forming and straightening which are controlled by limiting the material hardness, bend radius, or the amount of strain induced by a process. In these uses, the maximum yield strength to be used is 90 ksi. There are special materials also used for such components as the bayonet coupling (or coupling spud), the latch and latch spring which are made from XM-19 stainless steel, Alloy X-750 and 17-4 PH stainless steel. These materials have been used successfully in operating BWRs.

4.5.1.1 Materials Specifications

The CRD system’s material specification defines the material requirements for nuclear reactor FMCRD components and is intended to support design requirements. The FMCRD material specification is intended to be used with the design specification required by ASME BPVC, Section III, NCA-2021.

When applicable, such as for the RCPB ASME Class 1 components, ASME Code Section III, Division 1 Subsection NB-2300, Rules for Construction of Nuclear Facility Components, applies.

A list of materials and specifications is provided in Table 4.5-2. Table 5.2-2b lists material specifications for welds.

4.5.1.2 Austenitic Stainless Steel Components

The primary alloys to be used for the reactor internal components are austenitic structural alloys. These materials have a long history of use in all BWRs. Consequently, there is substantial experience with understanding and mitigating failures in these alloys. Specifically, the wrought austenitic stainless steel components are Type 304, 304L, 316, 316L. It should be noted that XM-19 stainless steel with its higher strength, developed for CRD components, is used in special

locations due to its excellent corrosion resistance. Maximum carbon content and heat treatment (i.e., solution annealing) of austenitic stainless steel materials are discussed in Subsection 5.2.3.4.1 and Table 5.2-2a . Welding of austenitic stainless steel, including control and measure of delta ferrite for austenitic stainless steel materials, is performed in compliance with RG 1.31, “Control of Ferrite Content in Stainless Steel Weld Metal,” and RG 1.44, “Control of the Processing and Use of Stainless Steel.”

4.5.1.3 Other Materials

The structural components of the FMCRD mechanism are made from four types of materials: 300 series stainless steel, Nickel-Chrome-Iron alloy X-750, XM-19 stainless steel, and 17-4 PH stainless steel materials. The RCPB components of the CRD, consist of FMCRD middle flange including the ball check valve, and the lower component housing, which enclose the lower part of the drive. The flange and lower housing are made with 300 series stainless steel materials in accordance with ASME BPVC, Section III (Reference 4.5-1).

Alloy 17-4 PH is a martensitic precipitation hardening stainless steel of high strength, and therefore, also potentially susceptible to SCC if the hardness is not limited.

4.5.1.4 Cleaning and Cleanliness Control

The degree of surface cleanliness meets the requirements of RG 1.28, “Quality Assurance Program Criteria (Design and Construction).” The Quality Assurance (QA) Program complies with 10 Code of Federal Regulations (CFR) 50, Appendix B and ASME NQA-1-2015, “Quality Assurance Requirements, for Nuclear Facility Applications,” (Reference 4.5-3). The GVH QA Program also complies with RG 1.28 and is organized to show its relationship to NQA-1-2015 (Reference 4.5-3) and the exceptions and clarifications noted in RG 1.28.

4.5.2 Regulatory Requirements

Compliance with 10 CFR 50, Appendix A

The CRD system structural material comply with the following GDC, which has plant-wide applicability, as discussed in Section 3.1:

GDC 1 – Quality standards and records.

The following GDCs and PDC have aspects specific to the CRD system structural materials and compliance is addressed as follows:

GDC 14 – Reactor coolant pressure boundary. The ICS forms part of the RCPB and, as such, compliance with GDC 14 with regard to RCPB materials is addressed in Chapter 5.

PDC 26 – Reactivity control system redundancy and capability. Compliance to PDC 26 is discussed in Section 3.1 and Subsection 7.1.3.

Section 3.1 summarizes compliance with GDC and PDC.

Compliance with 10 CFR 50.55a

The CRD system structural materials are designed to comply with the following regulation:

10 CFR 50.55a requires that reactor internals be designed, fabricated, erected, tested, and inspected to quality standards commensurate with the importance of the safety functions performed. Quality Groups of Structures, Systems, and Components (SSCs) are described in Table 3A-1. BWRX-300 conformance with RG 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” is described in Subsection 3.2.4.

4.5.3 Reactor Internals Materials

This section describes the material specification, welding control, nondestructive examinations, fabrication, and processing of austenitic stainless steel for the reactor internals and core support structure. Contamination protection and cleaning of austenitic stainless steel is described in Subsection 5.2.3. Other reactor materials used in this application are also discussed.

4.5.3.1 Materials Specifications

For the reactor internals, austenitic stainless steel is primarily used to minimize levels of radiation from corrosion products generated incore. Where higher strength requirements than those for standard austenitic stainless steels are needed, nickel base alloys, nitrogen strengthened austenitic stainless steel, or precipitation hardening stainless steels are used.

SCC requires three factors, susceptible materials, corrosive environment, and tensile stress. Materials selection with composition controls and defined fabrication processes is used to address material degradation issues in the reactor system; specifically, SCC. Design guidelines and controlled fabrication processes are used to manage component stresses, with an emphasis on surface stresses. The stainless steel structural materials used in the reactor core internals are primarily wrought materials along with their weld metals although cast austenitic stainless steels are used as well.

Cast austenitic stainless steels are not used for the RPV. However, cast austenitic stainless steels may be used in other pressure-retaining, cyclic, and/or dynamic load applications when supported by appropriate design analysis evaluations. As an example, fuel supports are subject to some cyclic and dynamic loads during reactor operation and operating events. Cast austenitic stainless steels are used for the fuel support due to the complexity of the part and challenges fabricating from wrought material.

Table 4.5-1 lists the materials used for the internal components including the shroud and core support ring, chimney, steam dryer, top guide, core support, CRD housings, and instrumentation housings, (Figure 4.1-1). All of these materials are procured to specifications from ASME BPVC Section II (Reference 4.5-2) as required by Section III (Reference 4.5-1). Table 4.5-1 lists the applicable material product forms as well as the ASME BPVC Section II (Reference 4.5-2) specifications. Table 5.2-2b lists material specifications for welds.

Cast Austenitic Stainless Steel

These cast versions of the wrought materials rely on carbon control and the presence of ferrite to assure SCC resistance which requires the ferrite level of 8% minimum for SCC. Additionally, the ferrite levels are also restricted to less than 20% to assure adequate toughness after long-term exposure to higher temperatures. The castings are given solution annealing to maintain SCC resistance.

Nickel Base Alloys

Similar composition controls and product form processing to stainless steels are used to assure that wrought Ni-Cr-Fe Alloy 600M and the associated Weld Metal Alloy 82 are free of any sensitization risks. Nickel base wrought materials are Alloy 600M with niobium modified chemistry.

RG 1.84 lists acceptable Code Cases. The primary one of concern is the use of Code Case N-580-2 which supports the use of Alloy 600M to address SCC concerns. Alloy 600 is not used in the BWRX-300.

High Strength Alloys

In special locations, the higher strength alloy, Alloy X-750, is used for bolting or fastener type of applications. The main approach used to prevent SCC susceptibility are accomplished by limiting the applied stresses in the design process. Additionally, Alloy 718 per Code Case N-60-6 may be used for high strength applications.

Alloy 17-4 PH is a martensitic precipitation hardening stainless steel of high strength, and therefore, also potentially susceptible to SCC if the hardness is not limited.

4.5.3.2 Controls on Welding

All welding, welding qualification, preheating, post weld heat treatment, and related nondestructive examinations are performed in accordance with ASME BPVC Section IX, "Qualification Standard for Welding, Brazing, and Fusing," (Reference 4.5-4), and when applicable, Section III (Reference 4.5-1). For non-ASME BPVC components or systems, alternate standards such as American Welding Society (AWS) may be applied if specified in the applicable design documents. For all ASME BPVC components, welder and welding procedures are qualified in accordance with the ASME BPVC, Section IX (Reference 4.5-4). General welding controls are described in Subsection 5.2.3.4.

Welding of austenitic stainless steel, including control and measure of delta ferrite for austenitic stainless steel materials, is performed in compliance with RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," and RG 1.44, "Control of the Processing and Use of Stainless Steel."

4.5.3.3 Nondestructive Examination

General Requirements for Examination

Weld repairs and heat affected zones on pressure retaining material shall be examined with the same techniques and criteria as are applied to welded joints on the same component.

Volumetric Examination

Radiographic examination shall be conducted in accordance with Section V, Article 2 as required and modified by ASME Code Section III. Ultrasonic examination shall be conducted in accordance with Section V, Article 4 and Article 5 as required and modified by ASME Code Section III.

Liquid Penetrant/Magnetic Particle Examinations

All liquid penetrant and magnetic particle examination shall be conducted in accordance with Section V, Article 6 and Article 7, respectively, as required by ASME Section III.

Visual Examination

Visual examinations shall be conducted in accordance with the ASME Code, Section V, Article 9.

4.5.3.4 Fabrication and Processing of Austenitic Stainless Steel Components

The reactor internal materials are selected to be resistant to corrosion and SCC in the BWR steam/water environment. Material specifications and fabrication techniques are established to reduce the probability that reactor internal components experience SCC. The austenitic stainless steel material for internals construction has a low carbon content and makes it less susceptible to sensitization, which in turn lowers the probability for SCC.

Specific steps, such as limiting alloy chemistry, controlling heat treatment, and eliminating cold work, are taken to mitigate SCC failures in the BWR environment. For L grade stainless steel materials, carbon is restricted to 0.02 weight percent to provide resistance to sensitization during welding while still being able to meet the specified strength levels at the use temperature. In low

carbon L grade materials, additional service temperature tensile tests are required to ensure that these materials still meet the strength requirements at service temperature per ASME BPVC, Sections II (Reference 4.5-2), III (Reference 4.5-1) and XI, “Rules for Inservice Inspection (ISI) of Nuclear Power Plant Components,” (Reference 4.5-5). These austenitic materials are given a solution heat treatment at a temperature range with a hold time designed to achieve a full solution annealed condition while minimizing grain growth. The heat treatment is then be followed by accelerated cooling to avoid any sensitization. Materials are tested for verification of non-sensitization in accordance with recommendations of RG 1.44, and Generic Letter 88-01. Typically, this includes an ASTM International (ASTM) A262, “Standard Practices for Detecting Susceptibility to Intergranular Attack in Austenitic Stainless Steels,” (Reference 4.5-6). It should be noted that XM-19 stainless steel, with its higher strength, developed for CRD components, is used in special locations due to its excellent SCC resistance.

4.5.4 Safety Evaluation

This section discusses the regulatory requirements for the CRD and the reactor internal structural materials for the BWRX-300.

Regulatory Requirements

The reactor internal structural materials are designed and fabricated to comply with the following regulations:

10 CFR 50.55a requires that reactor internals be designed, fabricated, erected, tested, and inspected to quality standards commensurate with the importance of the safety functions performed. Quality Groups of SSCs are described in Table 3A-1, BWRX-300 conformance with RG 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” is described in Subsection 3.2.4.

Regulatory Guidance

The reactor internal structural materials are designed and fabricated to conform to the following regulatory guidance:

RG 1.31 describes the acceptable method of implementing requirements with regard to the control of welding when fabricating and joining austenitic stainless steel components and systems. Written welding procedures that are approved by GVH are required for all RCPB welds performed for material fabrication and plant construction. These procedures conform with the requirements of Sections III (Reference 4.5-1) and IX (Reference 4.5-4) of the ASME BPVC and applicable USNRC RGs. This is described in Subsection 4.5.3.2.

RG 1.44 provides acceptance criteria for preventing intergranular corrosion of stainless steel components. Cleaning and protecting austenitic stainless steels from contamination during handling, storage, testing, and fabrication and for determining the degree of sensitization during welding. Processing controls of stainless steel including the removal of cold work. RG 1.44 also requires controls on contamination, controls to introduce cold work, the use of solution heat treatment to avoid sensitization and the use of testing to confirm absence of sensitization. The processes are provided in approved procedures by GVH. These procedures conform to the regulatory guidance of RG 1.44. This is described in Subsection 4.5.1.4.

RG 1.84, materials identified for each component or part used in the reactor internals and core support structures are compared with the materials identified as being acceptable in Sections II (Reference 4.5-2) and III (Reference 4.5-1) of the ASME Code and/or acceptable ASME Code Cases identified in RG 1.84. Conformance with the requirements of 10 CFR 50.55a, including

the use of ASME BPVC Section III Code Cases endorsed in RG 1.84 where necessary. This is described in Subsection 4.5.3.

4.5.5 References

4.5-1 ASME BPVC, Section III, “Rules for Constructions of Nuclear Facility Components,” American Society of Mechanical Engineers, 2021.

4.5-2 ASME BPVC, Section II, “Materials,” American Society of Mechanical Engineers, 2021.

4.5-3 ASME NQA-1-2015, “Quality Assurance Requirements, for Nuclear Facility Applications,” American Society of Mechanical Engineers, 2015.

4.5-4 ASME BPVC, Section IX, “Qualification Standard for Welding, Brazing, and Fusing,” American Society of Mechanical Engineers, 2021.

4.5-5 ASME BPVC, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” American Society of Mechanical Engineers, 2019.

4.5-6 ASTM A262-15(2021), “Standard Practices for Detecting Susceptibility to Intergranular Attack in Austenitic Stainless Steels,” 2021.

Table 4.5-1: Reactor Internal Components and Materials

Component(s)	Form	Material Type (Note 1)	ASME Specification
Steam Dryer	Plate, Bar or Forgings	Stainless Steel	SA-240/SA-240M, Gr 304/304L, 316/316L (Dual Grade) SA-182/SA-182M, Gr F304/F304L, F316/F316L (Dual Grade) SA-479/SA-479M, Type 304/304L, 316/316L (Dual Grade)
Shroud and Core Support Ring	Plate or Forgings	Stainless Steel	SA-240/SA-240M, Gr 316/316L (Dual Grade) SA-182/SA-182M, Gr F316/F316L (Dual Grade)
Core Support Structure	Plate or Forgings	Stainless Steel	SA-240/SA-240M, Gr 316/316L (Dual Grade) SA-182/SA-182M, Grade F316/F316L (Dual Grade)
Core Support Legs	Plate, Bar or Forgings	Ni-Cr-Fe	SB-168, SB-166, or SB-564 Nickel Alloy 600M modified per ASME Code Case N-580-2

Component(s)	Form	Material Type (Note 1)	ASME Specification
Top Guide	Plate or Forgings	Stainless Steel	SA-240/SA-240M, Gr 316/316L (Dual Grade) Gr 304/304L SA-182/SA-182M, Gr F316/F316L (Dual Grade) Gr 304/304L
Chimney	Plate or Forgings	Stainless Steel	SA-240/SA-240M, Gr 316/316L (Dual Grade) Gr 304/304L SA-182/SA-182M, Gr F316/F316L (Dual Grade) Gr 304/304L
Fuel Supports	Castings	Stainless Steel	SA-351/SA-351M, Gr CF3
Core Plate Bolts	Bar	Stainless Steel	SA-479/SA-479M, Type XM-19

Note:

1. Carbon content of RCPB wrought austenitic stainless steel (304/304L/316/316L) is 0.02% maximum.

Table 4.5-2: Control Rod Drive Components and Materials

Structural Component(s)	Form	Material Type (Note 2)	ASME Specification
Coupling Spud	Bar and Forgings	Nickel Alloy	SB-637 UNS N07750 Type 3, or UNS N07718 Type 2 (per ASME Code Case N-60-6)
Outer Tube	Seamless Pipe	Stainless Steel	SA-312/SA-312M, Gr TPXM-19
Outer Tube Head	Bar and Shapes	Stainless Steel	SA-479/SA-479M Type XM-19
Ball Screw	Bar and Shapes	Stainless Steel	SA-564/SA-564M Type 630 (Note 1)
Ball Nut	Bar and Shapes	Stainless Steel	SA-564/SA-564M, Type 630 (Note 1)
Lower Component Drive Shaft	Bar and Shapes	Stainless Steel	SA-479/SA-479M Type 316
Stop Piston	Seamless Pipe, Bar and Shapes	Stainless Steel	SA-312/SA-312M, Gr TP304, TP316 SA-376/SA-376M, Gr TP304, TP316 SA-479/SA-479M, Type 304/316
Guide Tube	Seamless Pipe	Stainless Steel	SA-312/SA-312M, Gr TP304 or TP316 SA-376/SA-376M, Gr TP304 or TP316

Notes:

1. To reduce risk of cracking, only the H1075, H1100, and H1150 conditions are allowed in 17-4 PH /Type 630 stainless steel.
2. Carbon content of RCPB wrought austenitic stainless steel (304/304L/316/316L) is 0.02% maximum

4.6 Design of Reactivity Control Systems

Reactivity controls consist of:

- Control rods (Section 4.2) and CRD systems
- Supplementary reactivity control in the form of gadolinia rods (Subsection 4.3.4.3)

This section describes the CRD system design, including the design bases, functions, system configuration and operation.

4.6.1 Control Rod Drive System Design Bases

The CRD system provides the primary means of reactivity control during normal, abnormal, and accident conditions. The system design bases include two diverse motive forces for control rod insertion, which are high pressure water for scram and fast motor insertion using the FMCRD motors. Incorporated into the design are positioning and protective features that prevent inadvertent withdrawal, drop, and ejection of the control rod due to a component failure or other malfunction in accordance with the requirements of GDC 25 and PDC 28.

Safety Category 1 Functions

- The CRD system provides a reliable and fail-safe means to rapidly insert all control rods based on signals from I&C Systems using stored hydraulic energy and maintain all control rods in the fully inserted position
- The CRD system provides features to prevent drop in or ejection of control rods
- The portions of the CRD system that interface with the NBS RPV provide the RCPB

Safety Category 2 Functions

- The CRD system provides a means by which to reposition control rods, either individually or in groups, based on signals from I&C Systems to enable power adjustments and to achieve and maintain subcriticality during normal and off-normal conditions
- The CRD system inserts all control rods using hydraulic pressure, by a diverse signal than that used for Safety Category 1 functions, when signaled to do so by I&C systems
- The CRD system inserts all control rods using hydraulic pressure, by an independent mechanism than that used Safety Category 1 functions, when signaled to do so by I&C systems
- The CRD system inserts control rods using FMCRDs when signaled to do so by I&C systems

Safety Category 3 Functions

- The CRD system provides rod status and position data to Rod Control and Information System (RC&IS)
- The CRD system stops control rod withdraw when signaled to do so by I&C systems
- The CRD system inserts all control rods using hydraulic pressure when signaled to do so by I&C systems
- The CRD system provides the capability to detect separation between a control rod and its associated ball nut or hollow piston
- The CRD system inserts control rods using FMCRDs when signaled to do so by I&C systems

- The CRD System aligns for maximum available flow when a signal is received from the RC&IS CRD hydraulic control systems
- The CRD System supplies purge water flow to the FMCRDs for cooling and foreign material exclusion
- The CRD System supplies water for sampling to the PRM
- The CRD System supplies a continuous flow of water to the NBS to keep the reactor water level reference leg instrument lines filled
- The CRD System supplies a makeup source for the RPV pressure test
- The CRD System supplies the required water to the SDC pumps for motor cooling and seal purge flow

Safety Category N Functions

None.

4.6.2 Control Rod Drive System Functions System Classification

The Safety Classification, Quality Groups, and Seismic Categories of the CRD SSCs are listed in Table 3A-1.

4.6.3 Control Rod Drive System Description

The CRD system is composed of three major elements:

- Electro-hydraulic FMCRD mechanisms
- HCU
- CRD Hydraulic Subsystem

The FMCRDs provide electric-motor driven positioning in normal plant operation for insertion and withdrawal of the control rods and hydraulic-powered rapid insertion (scram) of control rods during abnormal operating conditions. Verification of the adequacy of the FMCRDs to perform their mechanical functions and to maintain the RCPB is discussed in Subsection 3.9.4.

The hydraulic power required for scram is provided by high-pressure water stored in the individual HCUs. Each HCU contains a scram (nitrogen-water) accumulator, charged to high pressure, and the necessary valves and components to scram two FMCRDs, except that the center FMCRD has its own dedicated HCU. Additionally, during normal operation, the HCUs provide a flow path for purge water to the associated FMCRDs, which then flows into the reactor adding to reactor coolant inventory.

The CRD hydraulic subsystem provides clean, demineralized water that is regulated and distributed to provide charging of the HCU scram accumulators and purge water flow, via the HCUs, to the FMCRDs during normal operation. The CRD hydraulic subsystem is also the source of pressurized water for the SDC pump for motor cooling and seal purge flow and the reactor water level reference leg instrument lines.

The CRD hydraulic subsystem consists of the CRD pumps and valves that operate to pressurize the pump discharge header, which supplies the purge water header, charging water header and auxiliary system loads.

4.6.3.1 Fine Motion Control Rod Drive

The FMCRD used for positioning the control rod in the reactor core is an electro-hydraulic actuated mechanism as shown in Figure 4.6-1 and Figure 4.6-2. An electric motor-driven ball nut

and ball screw assembly positions the drive at both nominal increments and continuously over its entire range at a nominal speed. The FMCRDs also have the capability for motor-driven fast control rod insertion. Hydraulic pressure from the HCU is used for scrams. The FMCRD penetrates the bottom head of the RPV. The FMCRD does not interfere with refueling and is operative even when the RPV head is removed.

Fine Motion Control Rod Drive Components

A schematic of the FMCRD is provided in Figure 4.6-2. The basic elements of the FMCRD are as follows:

- Components required for electrical rod positioning or fine motion control (including the motor, brake release, ball screw, ball nut, and hollow piston)
- Components required for hydraulic scram (including hollow piston and buffer)
- Components required for pressure integrity (including the middle flange, installation bolts, and lower component housing)
- Rod position indication (position signal detectors)
- Reed position switches for scram surveillance and full-in indication
- Control rod separation detection switches (dual FMCRD separation switches)
- Bayonet coupling between the hollow piston and control rod
- Brake mechanism to prevent rod drop or ejection in the event of a break in the FMCRD primary pressure boundary
- Ball check valve to prevent rod drop or ejection in the event of a failure of the scram insert line
- Integral internal CRD blowout support (to prevent drive blowout)
- Magnetic coupling

These features and functions of the FMCRD are described below.

Components for Fine Motion Control

The fine motion capability is achieved with a ball nut and ball screw arrangement driven by an electric motor. The ball nut is keyed to the control rod guide tube (roller key) to prevent its rotation as it traverses axially as the ball screw rotates. A hollow piston rests on the ball nut and upward motion of the ball nut drives the control rod into the core. The weight of the control rod keeps the hollow piston and ball nut in contact during withdrawal.

The drive motor, located outside the pressure boundary, is magnetically coupled to the drive shaft located inside the pressure boundary. A splined coupling connects the drive shaft to the ball screw. The lower half of the splined coupling is keyed to the drive shaft and the upper half keyed to the ball screw. The tapered end of the drive shaft fits into the lower drive gear. The splined lower drive gear fits inside the splined upper drive gear keeping the drive shaft and the ball screw axially aligned. A drive shaft thrust bearing carries the entire weight of the control rod and drive internals.

The axially moving parts are centered and guided by radial rollers. The ball nut and bottom of the hollow piston includes radial rollers bearing against the control rod guide tube. Radially adjustable rollers at both ends of the labyrinth seal keep the hollow piston precisely centered in this region.

A stationary guide supports the top of the rotating ball screw against the inside of the hollow piston. A hardened bushing provides the circumferential bearing between the rotating ball screw and stationary guide. Rollers of the guide engage with axial grooves in the hollow piston to prevent the guide from rotating with the ball screw.

Components for Scram, Scram Follow, and Run-In

During hydraulic scram, high pressure water lifts the hollow piston off the ball nut and drives the control rod into the core. A spring washer buffer assembly stops the hollow piston at the end of its stroke. Departure from the ball nut releases spring-loaded latches in the hollow piston that engage slots in the control rod guide tube. These latches support the control rod and hollow piston in the inserted position. Simultaneous with the initiation of any hydraulic scram, a scram follow signal is generated such that each FMCRD motor drives the ball nut upward to a position just below the fully inserted and latched hollow piston. With the ball nut below the hollow piston, the piston remains latched to maintain the control rod in the fully inserted position.

The run-in of the ball nut using the electric motor drive following the hydraulic scram provides a diverse means of rod insertion as a backup to the accumulator scram. During motor run-in, the hollow piston remains in contact with the upper surface of the ball nut. This results in the weight of the hollow piston/FMCRD assembly being supported by the ball nut with the hollow piston latches remaining in the retracted position.

FMCRD Pressure Boundary

The RCPB components include the CRD housing (attached to the RPV), the FMCRD middle flange including the ball check valve, and the FMCRD lower component housing (which encloses the lower part of the drive). The middle flange is attached to the CRD housing by threaded bolts. The lower component housing is held to the middle flange and secured to the CRD housing by a separate set of main mounting bolts that become a part of the RCPB. This arrangement permits removing the lower component housing without disturbing the rest of the drive. Removing the lower component housing transfers the weight of the drive line from the drive shaft to a seat in the middle flange. Both the ball screw and drive shaft are locked to prevent rotation while the two are separated.

The part of the drive inserted into the CRD housing is contained within the outer tube. The outer tube is the FMCRD drive hydraulic scram pressure boundary, eliminating the need for designing the CRD housing for scram pressure. The bottom of the outer tube is welded to the middle flange at the bottom and is attached at the top with the CRD blowout support, which bears against the CRD housing as shown in Figure 4.6-3. The CRD blowout support and outer tube are attached by a slip-type connection that accounts for any slight variation in length between the drive and the CRD housing.

Purge water continually flows through the drive. The water enters through the ball check valve in the middle housing and flows around the hollow piston into the reactor. O-rings seal the lower component housing. A labyrinth seal near the top of the drive restricts the flow into the reactor.

Rod Position Indicator

Control rod position indication is provided by the FMCRDs to the control system by a position detection system that consists of position detectors and position signal converters.

Each FMCRD provides two position detectors, one for each control system channel, in the form of signal detectors directly coupled to the motor shaft. This configuration provides continuous detection of rod position during normal operation.

Scram Position Indicator

Scram position indication is provided by a series of magnetic reed switches to allow for measurement of adequate drive performance during scram. The magnetic switches are located at intermediate intervals of the drive stroke. They are mounted in a probe exterior to the CRD housing. A magnet in the hollow piston trips each reed switch in turn as it passes by.

As the bottom of the hollow piston contacts and enters the buffer, a magnet is lifted that operates a reed switch indicating scram completion. This continuous full-in indicating reed switch, shown on Figure 4.6-4, provides indication whenever the drive is at the full-in latched position. In the event of failure of the buffer mechanism a buffer contact reed switch is positioned above the full-in switch to provide failed buffer indication.

Control Rod Separation Detection

Two redundant switches are provided to detect the separation of the hollow piston from the ball nut or between the control rod and the hollow piston as shown in Figure 4.6-5. The separation switches function to detect a detached control rod and cause a rod withdrawal block, thereby preventing a rod drop accident. Actuation of either switch also initiates an alarm in the Main Control Room (MCR).

During normal operation, the weight of the control rod and hollow piston resting on the ball nut causes the ball screw assembly to compress a spring, on which the lower half of the splined coupling between the drive shaft and ball screw assembly rests (the lower half of the splined coupling is also known as the “weighing table”). When the hollow piston separates from the ball nut, or when the control rod separates from the hollow piston, the spring is unloaded and pushes the weighing table and ball screw assembly upward. This action causes a magnet in the weighing table to operate the reed switches located outside the lower component housing.

Bayonet Couplings

There are two bayonet couplings associated with the FMCRD. The first is the FMCRD/control rod guide tube/CRD housing interface as shown in Figure 4.6-3. This bayonet locks the FMCRD and the base of the control rod guide tube to the CRD housing and functions to retain the guide tube during normal operation and dynamic loading events. The bayonet also holds the FMCRD against ejection in the event of a hypothetical failure of the CRD housing weld. The locating pin on the core plate that engages the flange of the control rod guide tube and the bolt pattern on the FMCRD/CRD housing interface, assure proper orientation between the control rod guide tube and the FMCRD to confirm that the bayonet is properly engaged.

The second bayonet is located between the control rod and FMCRD as shown in Figure 4.6-6. The coupling spud at the top end of the hollow piston engages and locks into a mating socket at the base of the control rod. The coupling requires a 45° rotation for engaging or disengaging. Once locked, the drive and rod form an integral unit that must be manually unlocked before the components can be separated.

The FMCRD design allows the coupling integrity of this second bayonet to be checked by driving the ball nut down into an overtravel out position. After the weighing spring has raised the spindle assembly to the limit of its travel, further rotation of the spindle in the withdraw direction drives the ball nut down away from the piston (assuming the coupling is engaged). If the hollow piston is not properly coupled to the control rod, the hollow piston remains in contact with the ball nut and move with it to the overtravel position. A reed switch at the overtravel position detects this movement of the hollow piston and provide an alarm in the MCR.

FMCRD Brake and Ball Check Valve

The FMCRD design incorporates an electro-mechanical brake keyed to the motor shaft as shown in Figure 4.6-7. The brake is normally engaged by passive spring force when the FMCRD is stationary. It is disengaged for normal rod movements by signals from the RC&IS. Disengagement is caused by the energized magnetic force overcoming the spring load force. The braking torque and the magnetic coupling torque between the motor and the drive shaft are sufficient to prevent control rod drop or ejection in the event of failure in the pressure retaining parts of the drive mechanism. The brake is designed so that its failure does not prevent the control rod from rapid insertion (scram). However, if the brake fails to release when energized, the motor is unable to move the ball nut and control rod.

The electromechanical brake is located in the motor unit. The stationary spring-loaded disk and coil assembly is contained within the brake assembly. The rotating disk is keyed to the motor unit output shaft.

A ball check valve is located in the middle flange of the drive at the scram inlet port. The ball check valve actuates to close the scram inlet port under conditions of reverse flow caused by a break of the scram line. This prevents the loss of pressure to the underside of the hollow piston and the generation of loads on the drive that could cause a rod drop or ejection.

Integral Internal CRD Blowout Support

The CRD blowout support, as shown in Figure 4.6-3, is designed to prevent ejection of the FMCRD and the attached control rod in the unlikely event of:

- A failure of the weld between the CRD housing and the stub tube at the RPV bottom head, or a failure through the CRD housing along the fusion line of the weld
- A total failure of all flange bolts attaching the lower component housing flange and middle flange to the CRD housing flange

The internal CRD blowout support utilizes the outer tube integral with support internal to the RPV to provide the anti-ejection support. The outer tube, which is welded to the middle flange, attaches by a bayonet lock to the control rod guide tube base. The core plate in turn supports the control rod guide tube.

With a CRD housing failure, the weight plus pressure load acting on the drive and CRD housing would tend to eject the drive. In this event, the control rod guide tube base remains supported by the intact CRD housing extension inside the RPV and the FMCRD remains supported in turn by its positive lock to the control rod guide tube base. Coolant leakage is restricted to the small annular area between the CRD outer tube and the inside of the CRD housing. In the event of a total failure of the weld itself, with the entire CRD housing intact, the CRD housing would tend to be driven downward by the total weight plus RPV pressure. However, after the interconnected assembly of the CRD housing, FMCRD and control rod guide tube move down a short distance, the flange at the top of the control rod guide tube contacts the core plate and stops further movement of the assembly. As the FMCRD is positively locked to the control rod guide tube base, it cannot eject. In this case, the CRD housing that bears on top of the blowout support is also prevented from leaving the penetration. Coolant leakage for this condition is restricted to the small annular area between the outside of the CRD housing and the inside of the stub tube.

The FMCRD design provides an anti-rotation device that engages when the lower component is removed for maintenance as shown in Figure 4.6-10. This device prevents rotation of the ball screw and hence prevents control rod motion when the lower component is removed. The anti-rotation device consists of:

- The coupling piece on the bottom of the ball screw that engages with the lower component drive shaft
- The middle flange backseat

The coupling between the lower component drive shaft and ball screw is splined to permit removal of the lower component housing. The underside of the coupling piece on the ball screw has a circumferentially splined surface that engages with a mating surface on the middle flange backseat when the ball screw is lowered during lower component removal. When engaged, ball screw rotation is prevented. In the unlikely event of total failure of all the drive flange bolts attaching the lower component housing flange and middle flange to the CRD housing flange, the anti-rotation device engages when the lower component falls. The middle flange/outer tube/CRD blowout support is restrained by the control rod guide tube base bayonet coupling, thus preventing rod drop or ejection.

Magnetic Coupling

The magnetic coupling is located at the bottom of the lower component as shown in Figure 4.6-2. The magnetic coupling consists of an inner and an outer rotor. The inner rotor is located inside the lower component housing pressure boundary. The outer rotor is located outside the pressure boundary. Each rotor has permanent magnets mounted on it. As a result, the inner and outer rotors are locked together by the magnetic forces acting through the pressure boundary and work as a synchronous coupling. The outer rotor is coupled with the motor unit and driven by the motor such that the inner rotor follows the rotation of the outer rotor.

The magnetic coupling is designed so that its maximum coupling torque exceeds the maximum torque of the motor unit to prevent decoupling or slippage due to motor torque.

The magnetic coupling is designed to not have any undesirable effects on other magnetic sensitive subcomponents.

Materials of Construction

The materials of construction for the FMCRD components which form part of the RCPB are discussed in Section 4.5.

4.6.3.2 Hydraulic Control Units

Upon receipt of a scram signal, each HCU furnishes pressurized water for hydraulic scram to two FMCRD units (except for the FMCRD in the center of the core which has its own HCU). Additionally, each HCU provides the capability to adjust purge flow to the FMCRDs during normal plant operation. A test port is provided on the HCU for connection to a portable test station to allow for controlled venting of the scram insert line to test the ball check valve during plant shutdown. The check valves have an active safety function to close if charging header pressure drops below required HCU accumulator pressure for the hydraulic scram function. This ensures that the water stored in the HCU accumulator is delivered to the FMCRDs to accomplish the scram function. A simplified schematic of the HCU is provided in Figure 4.6-8. The major components of each HCU are provided below.

Scram Solenoid Pilot Valve

The scram solenoid pilot valve assembly is a three-way solenoid pilot operated valve. The solenoid valves are normally energized and closed. Upon loss of power to the solenoids, the valves open which vents air to open the scram valve. The assembly is designed so that the power must be removed from both solenoids before air pressure can be discharged from the scram valve operator. This prevents the inadvertent scram of the drives associated with a given HCU in the event of a failure of one of the valve solenoids.

Scram Valve

The scram valve opens to supply pressurized water to the bottom of the drive piston. This quick opening valve is operated by an internal spring and system pressure. It is closed by air pressure applied to the top of its diaphragm operator. A position indicator switch on this valve energizes a light in the MCR as soon as the valve starts to open.

Scram Accumulator

The scram accumulator stores sufficient energy to fully insert two control rods at any anticipated reactor pressure. The range of RPV pressures and the associated scram times are shown in Table 15.5-4. The accumulator is a hydraulic cylinder with a free-floating piston. The piston separates the water on top from the nitrogen below. A check valve in the accumulator charging line prevents loss of water pressure in case charging water header pressure is lost. During normal plant operation, the accumulator piston is seated at the bottom of its cylinder, and the nitrogen bottle maintains high pressure nitrogen under the accumulator piston. Pressure sensors provide local and MCR nitrogen pressure indication. Loss of nitrogen decreases the nitrogen pressure, which actuates a pressure switch and sounds an alarm in the MCR. The alarm would prompt operator action to repressurize the nitrogen bottle using an external supply of nitrogen gas. To ensure that the accumulator is always able to produce a scram, it is continuously monitored for water leakage. A level sensor actuates an alarm in the MCR if water leaks past the piston barrier and collects in the accumulator instrumentation block.

Purging Panel

The purging panel controls the purge water flow to the associated FMCRDs. Each panel has a throttling valve in the purge water line to control the purge water flow rate. This throttling valve maintains the flow at a constant value while the drives are stationary. A bypass line containing a solenoid-operated addition valve is provided around this normal purge flow path. The purge water addition valve is signaled to open and increase the purge water flow whenever either of the two associated FMCRDs is commanded to insert by the RC&IS. During FMCRD insertion cycles, the hollow piston moves upward, leaving an increased volume for water within the drive. Opening of the purge water addition valve increases the purge flow to offset this volumetric increase and precludes the backflow of reactor water into the drive, thereby preventing long-term drive contamination.

4.6.3.3 Control Rod Drive Hydraulic Subsystem

The CRD hydraulic subsystem supplies high pressure water to the following:

1. HCU accumulators for charging water.
2. FMCRDs for purge water.
3. SDC pumps for motor cooling and seal purge flow.
4. Reactor water level reference leg instrument lines.
5. Process Radiation and Environmental Monitoring System for sampling.

The subsystem consists of two sets of equipment, each providing the required functions with the pumps, valves, filters, piping, and instrumentation described below. A schematic line diagram for the CRD hydraulic system is provided in Figure 4.6-9.

4.6.3.3.1 Control Rod Drive Hydraulic Subsystem Description

The equipment and arrangement are comprised of the following:

1. CRD hydraulic subsystem pump suction piping from the Condensate and Feedwater Heating System (CFS) or the Condensate Storage Tank (CST).
2. Two 100% capacity sets of equipment, with major components arranged in a parallel manner, allowing for flexibility in operation by having the ability to align either set's suction/discharge filter with either pump. Each train consists of the following:
    a. One 100% capacity CRD pump with a suction and discharge filter to prevent foreign material and particulates from entering the pump, HCUs and FMCRDs. Each filter has a differential pressure sensor to monitor pressure drop across the filter and to provide a high differential pressure alarm in the MCR. The suction pressure of each pump is monitored with a pressure sensor which trips the pump on low suction pressure.
    b. Each CRD pump has an adjustable speed drive that can control discharge flow of the positive displacement pump.
    c. Each CRD pump has discharge pressure sensor that provides indication locally and in the MCR.
    d. The distribution piping, branching off the pump discharge line is downstream of the CRD pump discharge filter, and provides:
        i. Water flow to the SDC pumps for motor cooling and seal purge flow
        ii. Water flow to the reactor water level reference leg instrument lines
        iii. Sample water to the Process Radiation and Environmental Monitoring System
        iv. Water to the purge water header
        v. Water to the charging water header
    e. Downstream of the CRD discharge filters are pump discharge header pressure sensors used to start the standby CRD pump in the event of failure of the in-service pump and control the operating Pressure Control Valve (PCV).
3. The purge water header directs purge water flow to the individual HCUs, where it is then directed to the FMCRDs via the scram insert piping.
4. Piping connections are provided between the individual HCUs and their respective FMCRDs.
5. The charging water header includes an Air Operated Valve (AOV) that can be closed after a scram, a check valve, and a pressure transmitter on the charging header. A low-pressure indication would alarm in the MCR.
6. A scram air header, containing redundant PCVs, is connected to the scram valves in the individual HCUs. Pressure sensors are located on the scram air header to provide local and MCR pressure indication and to provide high and low-pressure alarms in the MCR.

7. A control air header, containing redundant PCVs, is used to control air flow to the purge water header PCV operators and the charging water header AOV.
8. Air filters are provided upstream of the PCVs for both the control air header and the scram air header and the charging water header AOV.
9. 9. Solenoid-operated Alternate Rod Insertion (ARI) valves are positioned in the scram air header to depressurize and isolate the scram air header on receipt of a scram signal.

4.6.3.3.2 Control Rod Drive Hydraulic Subsystem Operation

The operation mode of the CRD hydraulic subsystem is described in this subsection.

Normal Operation

Normal operation is defined as operating periods where the CRD hydraulic subsystem provides charging pressure to the HCUs and supplies purge water to the FMCRDs, the SDC pumps for motor cooling and seal purge flow, the reactor water level reference leg instrument lines, and the Process Radiation and Environmental Monitoring System.

The charging water header AOV is operated manually by the operator when desired or required for system maintenance. It does not receive any automatic signals.

During Power Operation, the CRD hydraulic subsystem is in normal operation as described below. During Power Operation, control rods may also be inserted and withdrawn, based on the power demands, as described below in “Control Rod Insertion and Withdrawal,” below.

During Startup, the CRD hydraulic subsystem is in normal operation, and the control rods are being withdrawn, as described in “Control Rod Insertion and Withdrawal” below, in order to achieve criticality and to start adding heat to the NBS.

During shutdown operations, the control rods are fully inserted, and the CRD hydraulic subsystem is in normal operation. While shutdown, withdrawal of control rods is blocked.

During refueling operations, the CRD hydraulic subsystem is in normal operation. Control rod testing (insertion and withdrawal) is enabled via an interlock which allows for movement of one control rod or pair of control rods.

Control Rod Insertion and Withdrawal

The FMCRD design provides the capability to move a control rod in fine steps. Normal control rod movement is under the control of the RC&IS. The RC&IS controls the input of actuation power to the FMCRD motor from the electrical power supply to complete a rod motion command. The FMCRD motor rotates a ball screw, which in turn causes the vertical translation of a ball nut on the ball screw. This motion is transferred to the control rod via a hollow piston that rests on the ball nut. Thus, the hollow piston with the control rod is raised (inserted) or lowered (withdrawn) depending on the direction of rotation of the FMCRD motor and ball screw. During a control rod insertion, the solenoid-operated purge water addition valve within the associated HCU is opened to increase the purge water flow to the FMCRD. The increased flow offsets the volumetric displacement within the FMCRD as the hollow piston is inserted into the core and precludes the backflow of reactor water into the drive.

Normal Reactor Shutdown

A planned shutdown for an outage can be initiated by either FMCRDs or a reactor scram.

Scram

In response to an automatic or a manual scram the power is interrupted to both scram solenoid valve assembly coils that vent the air from the scram valve operator in each HCU. Venting of the air causes the scram valves to open by spring action. When the scram valve opens in the associated HCU, hydraulic pressure provides an insertion force to the respective FMCRD. When the hydraulic force is applied, the FMCRD's hollow piston lifts off the ball nut and inserts the control rod rapidly. The water displaced from the FMCRD is discharged into the RPV. Indication that the scram has been successfully completed (i.e., all rods full-in position) is displayed in the MCR.

The CRD hydraulic subsystem is "fail-safe" in that loss of either electrical power to the scram solenoids, or loss of control air pressure to the scram valve operator, causes a scram.

Simultaneously with the hydraulic scram, each FMCRD motor is signaled to start from the RC&IS. These signals cause the FMCRD motor to drive the ball nut upward to a position just below the fully inserted and latched hollow piston. This action is known as the scram follow function.

At the end of the scram, the HCUs and pump discharge pressure is near RPV pressure with purge flow being primarily through the scram valves. In response to the scram signal, the CRD controller sends a fixed position signal to the in-service purge water header PCV to open to maintain flow to the auxiliaries. The purge flow is primarily through the scram valves until the scram logic is reset.

Reactor scram logic is described in Subsection 7.3.1.3.3.

Selected Control Rod Run-In

As described in Appendix 4A, the Selected Control Rod Run-In (SCRRI) function is credited in the loss of FW heating event. During SCRRI, a select number of control rods insert at the fast speed. Upon receipt of the SCRRI initiation signal, the CRD pump speeds up to provide increased purge water flow to the associated HCUs. At those HCUs, the solenoid-operated purge water addition valve opens to increase the purge water flow to the FMCRD to preclude the backflow of reactor water into the drives. In automatic control, the purge water header PCV automatically adjusts in the open direction to maintain pump discharge pressure which provides additional flow to the purge control valves.

Alternate Rod Insertion

The ARI function of the CRD system provides an alternate, or backup means for actuating hydraulic rod insertion that is diverse and independent from the Safety Class 1 (SC1) I&C System. The signals to initiate the ARI function are the same as those for automatic and manual scrams. Following receipt of any scram signal, the solenoid-operated ARI valves on the scram air header are energized to actuate the ARI valves and depressurize the scram air header which opens all HCU scram valves. The control rods are then inserted hydraulically in the same manner as an SC1 I&C System initiated scram. The same signals that initiate ARI simultaneously initiate an FMCRD scram follow function. To prevent spurious ARI actuations, each ARI vent path has two valves in series such that both valves are required to actuate to cause depressurization. Additionally, all ARI valves are energized to actuate in order to minimize the probability of a spurious scram due to ARI actuation.

ARI is not credited in either the deterministic or probabilistic safety analysis for BWRX-300. However, ARI is required for BWRs by 10 CFR 50.62(c)(3).

RPV Makeup Function

The CRD system can be used as a makeup source for the RPV. For this case, the CRD pumps' speed can be increased up to maximum and the purge water header PCVs will be fully opened. In this manner, the CRD pumps can be used to provide makeup up to and including full RPV design pressure. The RPV makeup function is not required to respond to any AOOs or DBAs.

4.6.4 Safety Evaluation

This section discusses the regulatory requirements for the reactivity control system for the BWRX-300.

Regulatory Requirements

The reactivity control system complies with the following GDC, which has plant-wide applicability, as discussed in Section 3.1:

- GDC 4 – Environmental and dynamic effects design bases

The following GDCs and PDCs have aspects specific to the reactivity control system and compliance is addressed as follows:

- GDC 23 – Protection system failure modes: Compliance to GDC 23 is described in Subsection 7.1.3
- GDC 25 – Protection system requirements for reactivity control malfunctions: Compliance to GDC 25 is described in Subsection 7.1.3
- PDC 26 – Reactivity control system redundancy and capability: Compliance with PDC 26 is discussed in Section 3.1 and Subsection 7.1.3
- GDC 27 – Combined reactivity control systems capability: The basis for GDC 27 compliance is included in the evaluation of PDC 26 as discussed in Section 3.1
- PDC 28 – Reactivity limits. Compliance to PDC 28 is described in Subsection 4.3.6 and Subsection 7.1.3
- PDC 29 – Protection against anticipated operational occurrences: Compliance to GDC 29 is described in Subsection 7.1.3
- GDC 33 – Reactor coolant makeup: The CRD system's normally operating Safety Category 3 function to supply purge water flow is sufficient to provide makeup for small leaks in the RCPB such that RPV level is maintained in the normal operating range, as described in Subsection 3.1.4.4

Section 3.1 summarizes compliance with GDC and PDC.

4.6.5 Testing and Inspection Requirements

Preoperational and startup testing is performed by the Initial Test Program (Chapter 14). Detailed testing procedures are to be provided in the Final Safety Analysis Report (FSAR).

4.6.6 Instrumentation and Control

The safety classes for the I&C system are described in this subsection.

4.6.6.1 Safety Class 1 Instrumentation and Control System

The SC1 I&C for the CRD system is described in Subsection 7.3.1.

4.6.6.2 Safety Class 2 and 3 Instrumentation and Control System

The Safety Class 2 (SC2) and Safety Class 3 (SC3) I&C for the CRD system are described in Subsection 7.3.2 and Subsection 7.3.3, respectively.

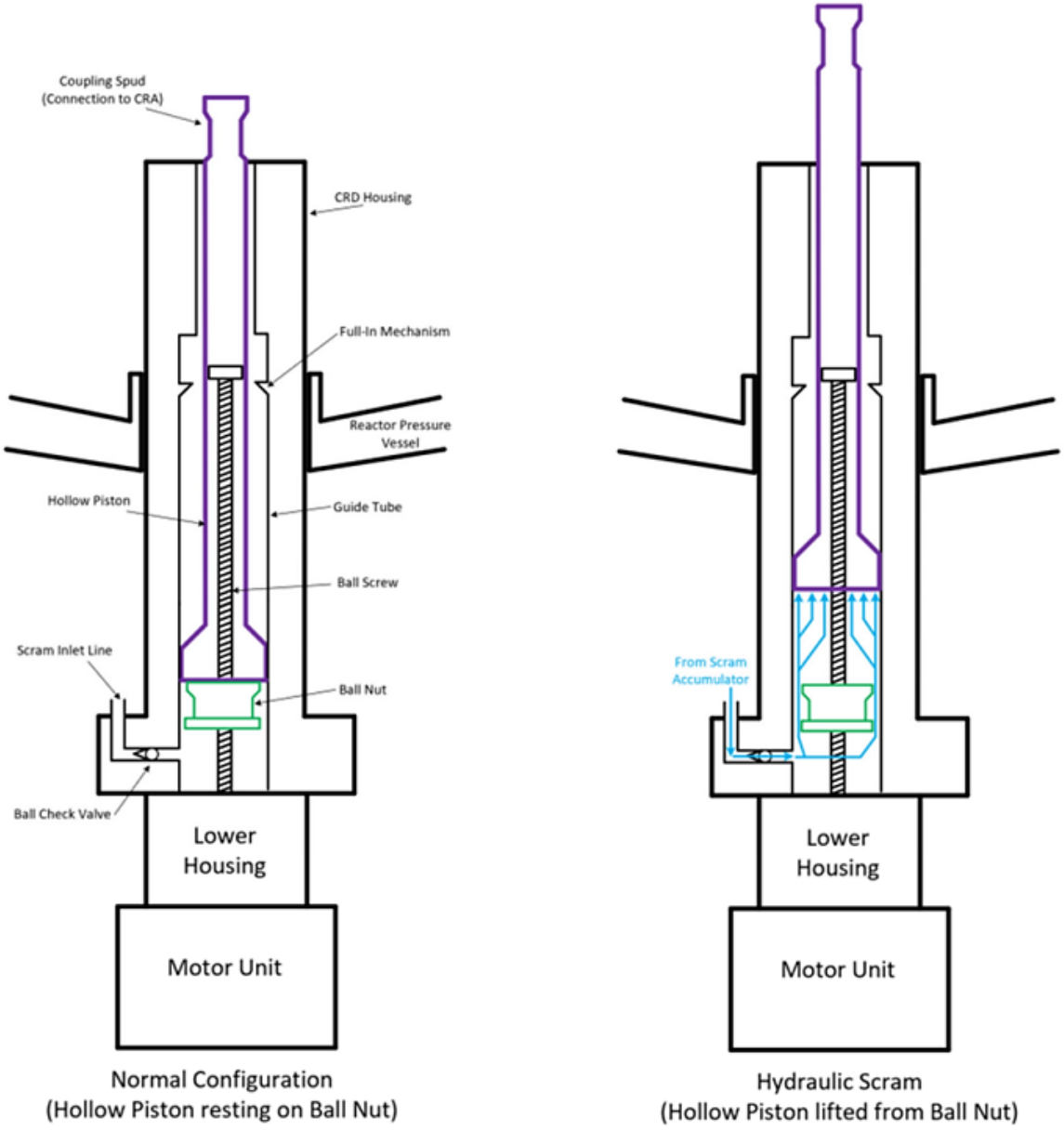


Figure 4.6-1: BWRX-300 Simplified View of FMCRD with Hydraulic Scram

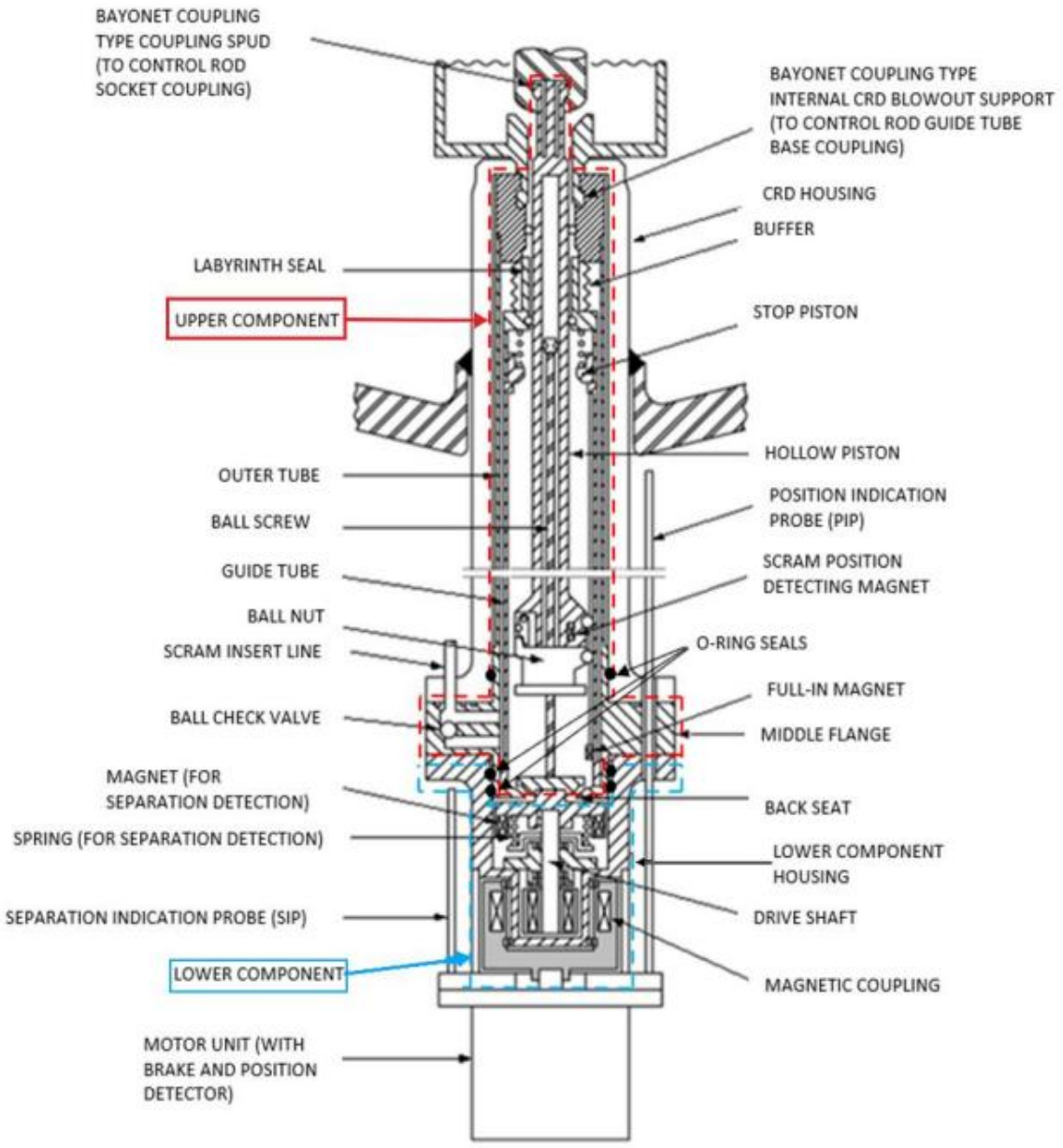


Figure 4.6-2: Schematic Representation of Magnet Coupling Fine Motion Control Rod Drive

Note:

1. The red and blue dashed lines indicate the upper and lower components that are bolted together at the middle flange.

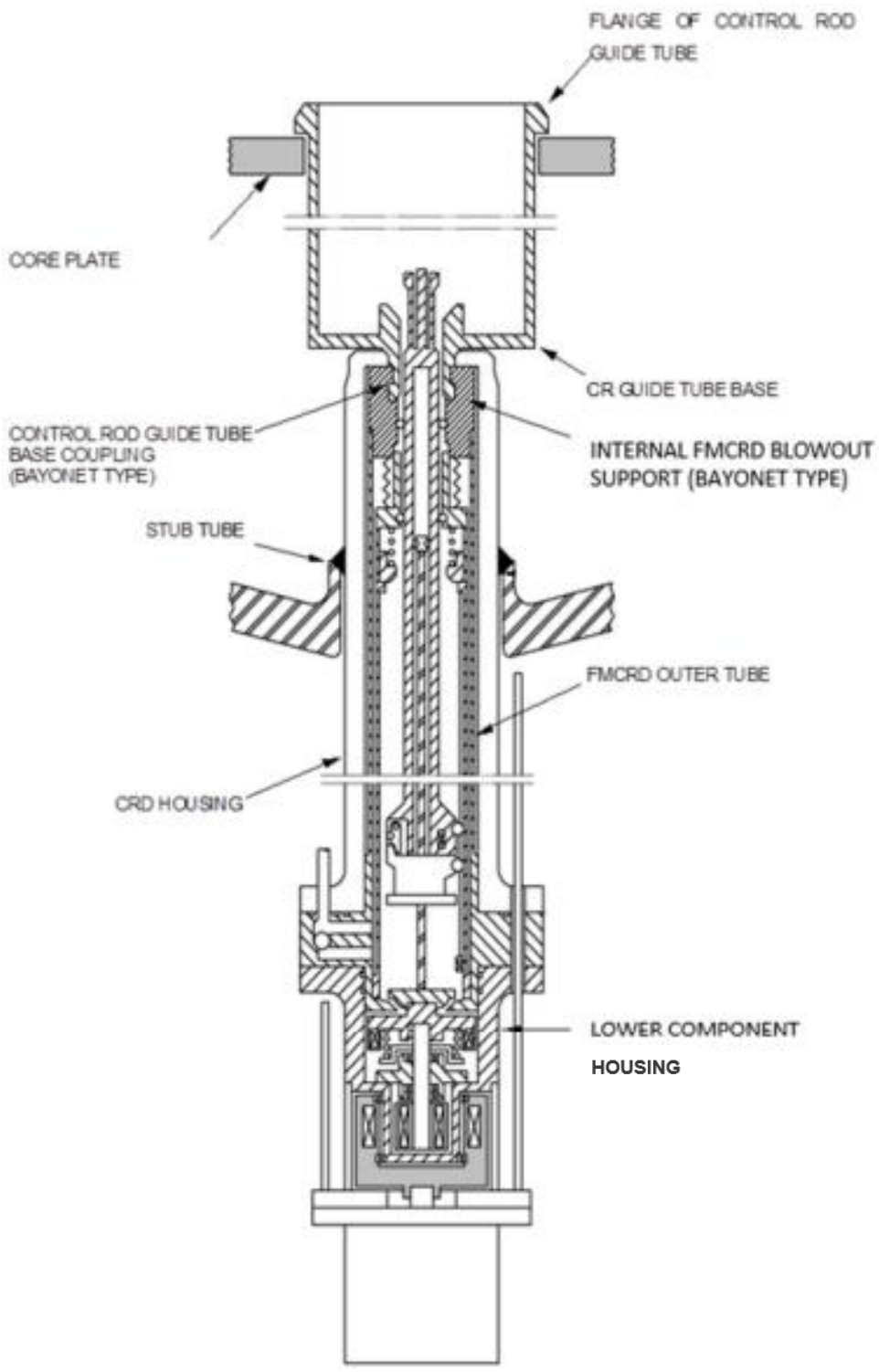


Figure 4.6-3: Representative Internal Control Rod Drive Blowout Support Schematic

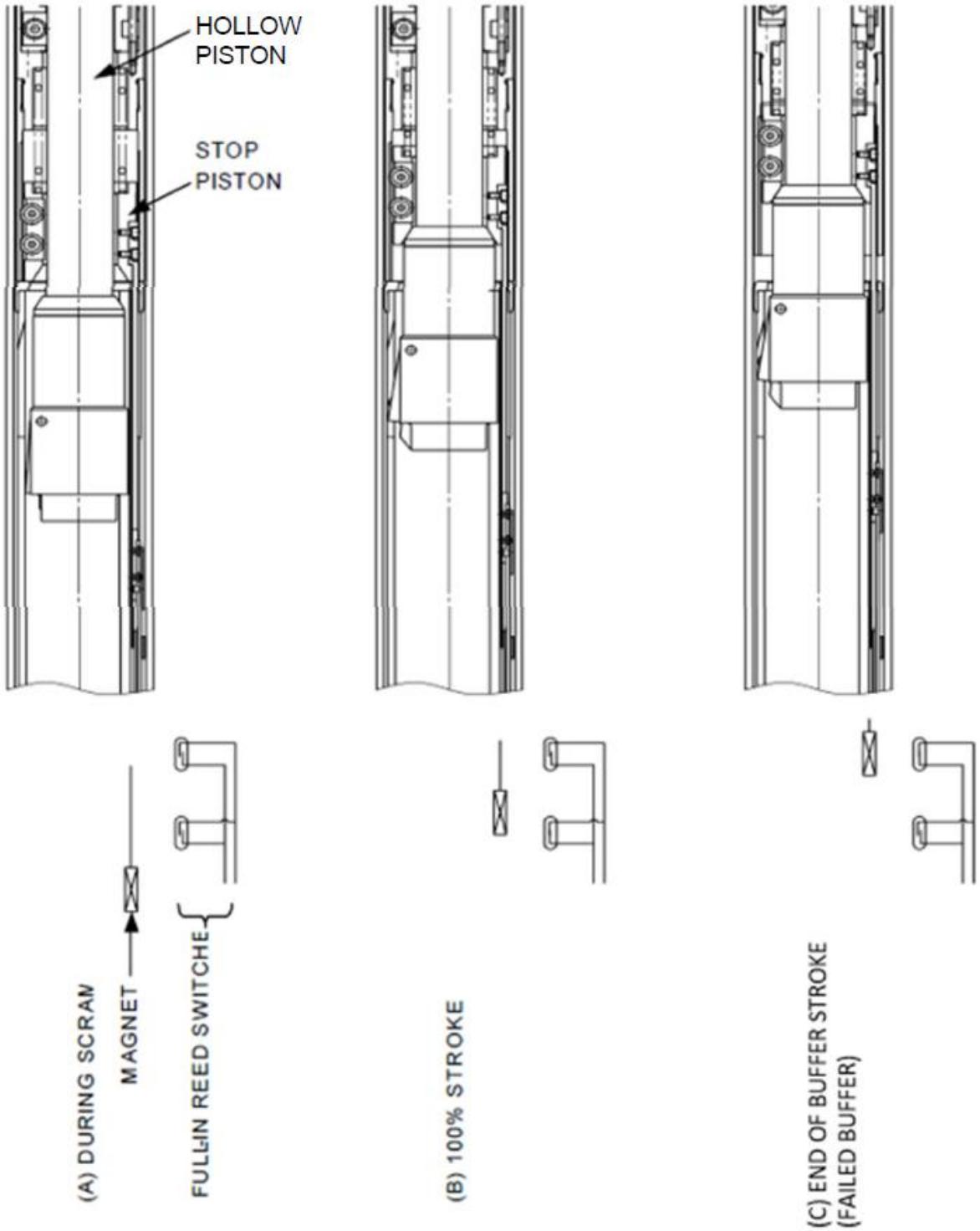


Figure 4.6-4: Representative Continuous Full-in Indicating Device

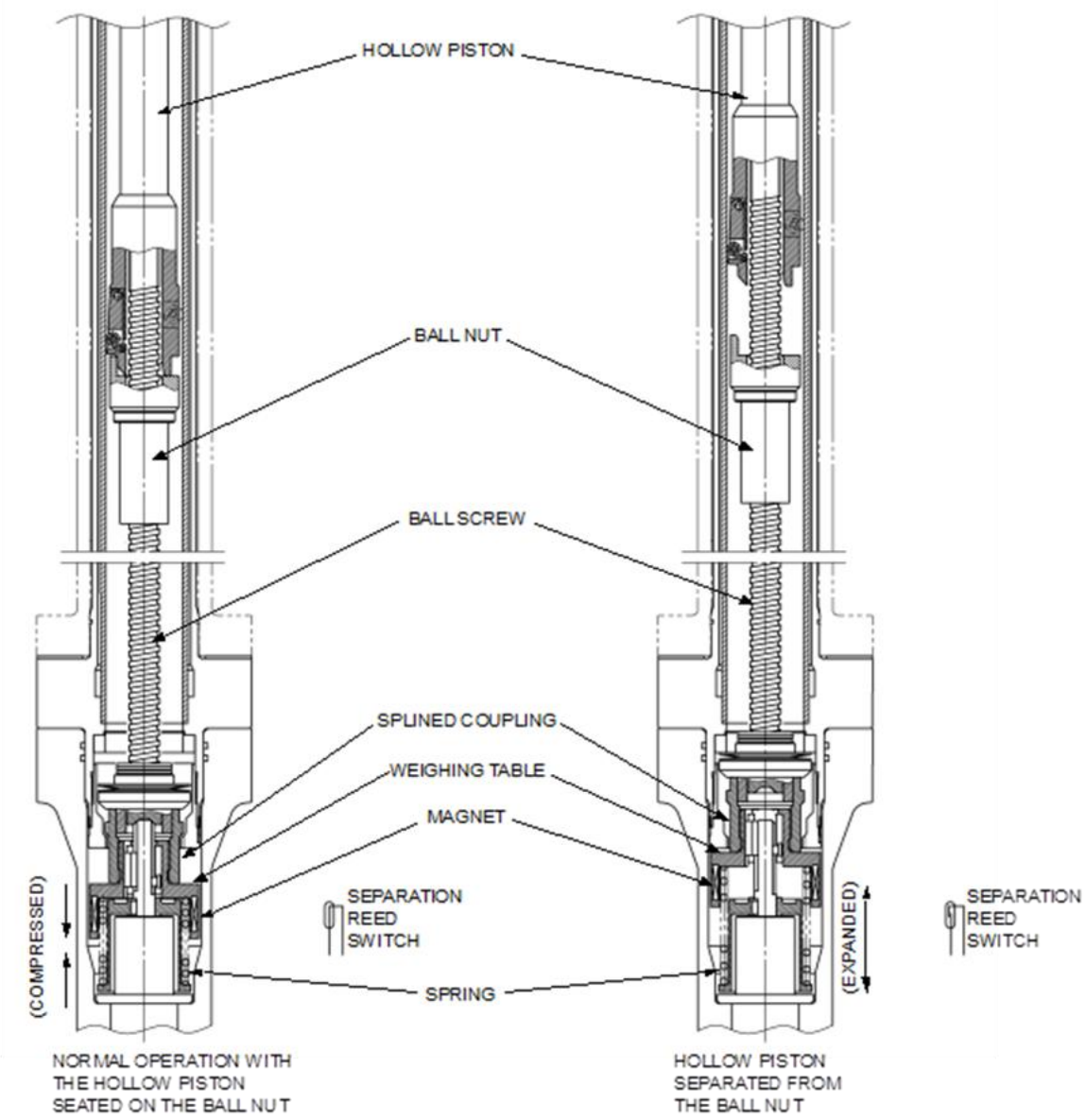


Figure 4.6-5: Representative Control Rod Separation Detection

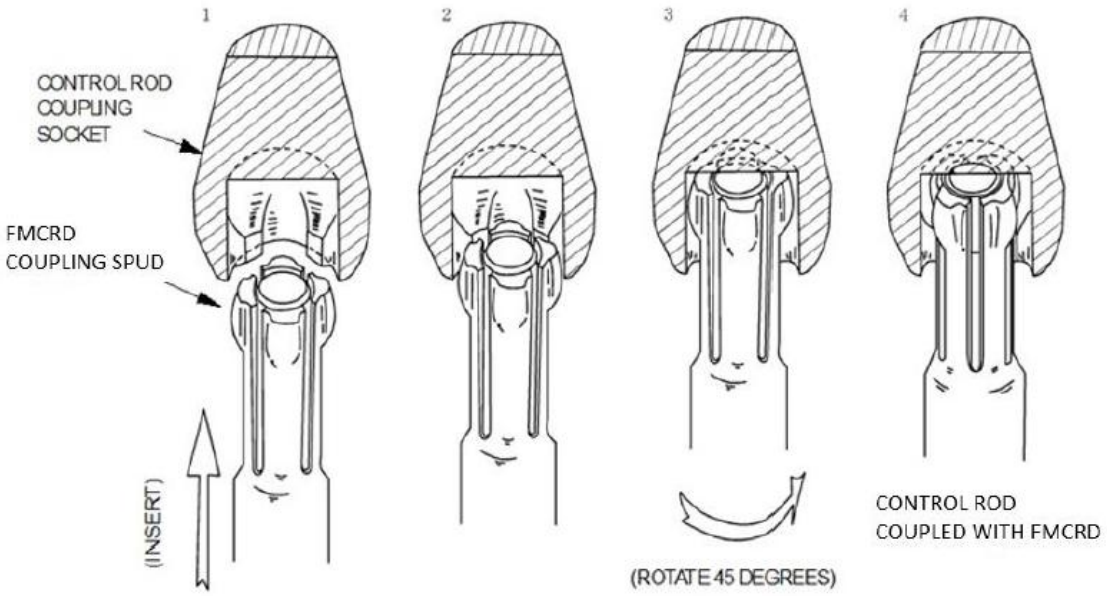


Figure 4.6-6: Representative Control Rod Coupling Socket to FMCRD Coupling Spud

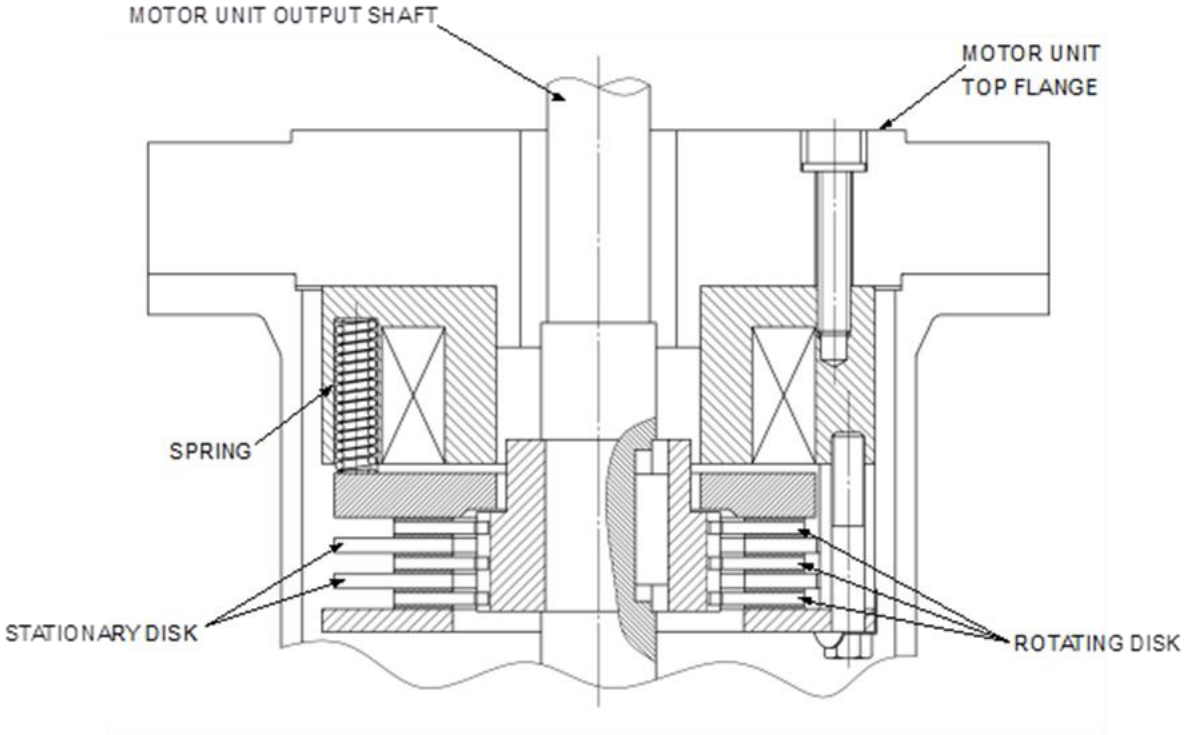


Figure 4.6-7: Representative Fine Motion Control Rod Drive Electro-Mechanical Brake

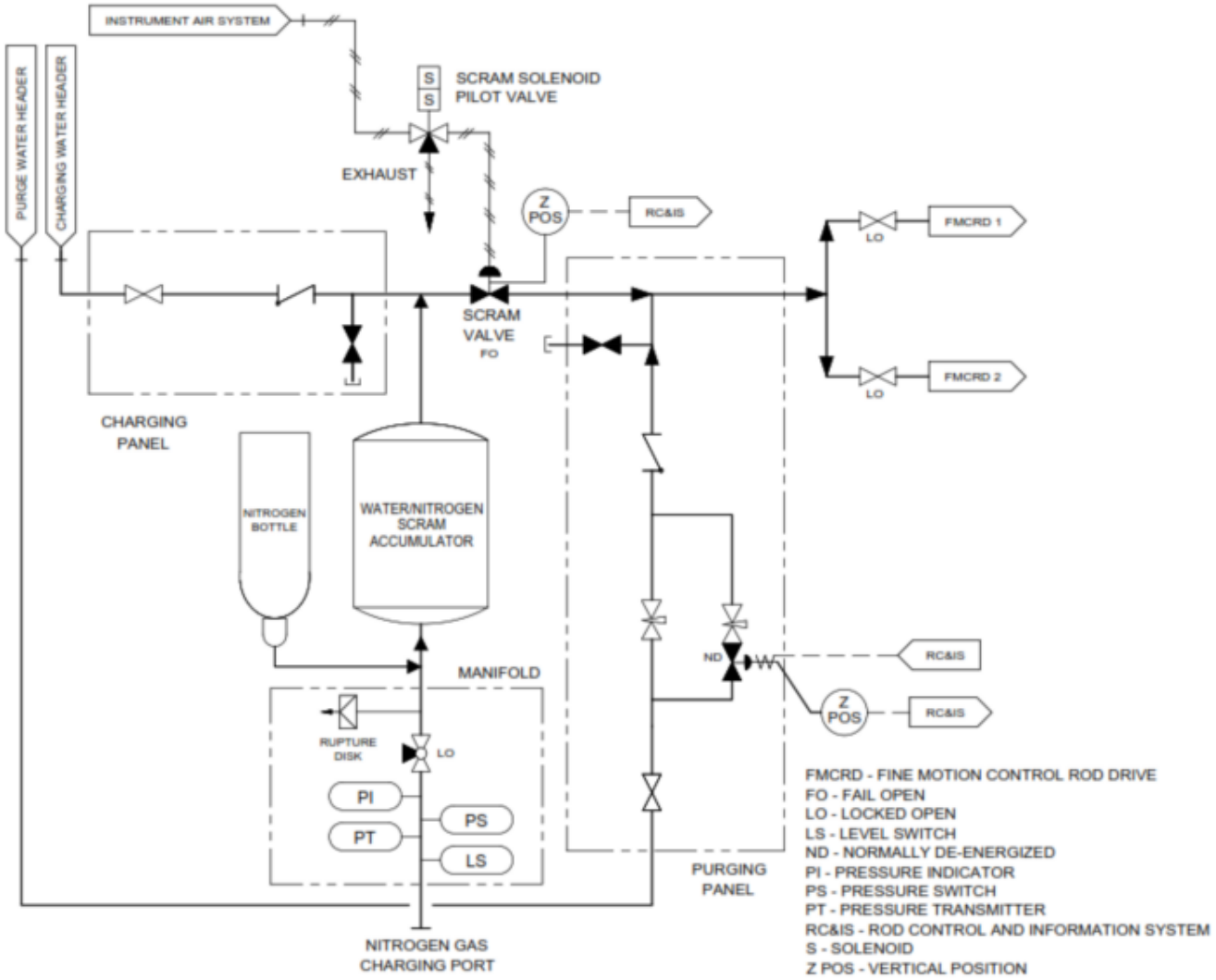


Figure 4.6-8: Hydraulic Control Unit Simplified Piping and Instrumentation Diagram

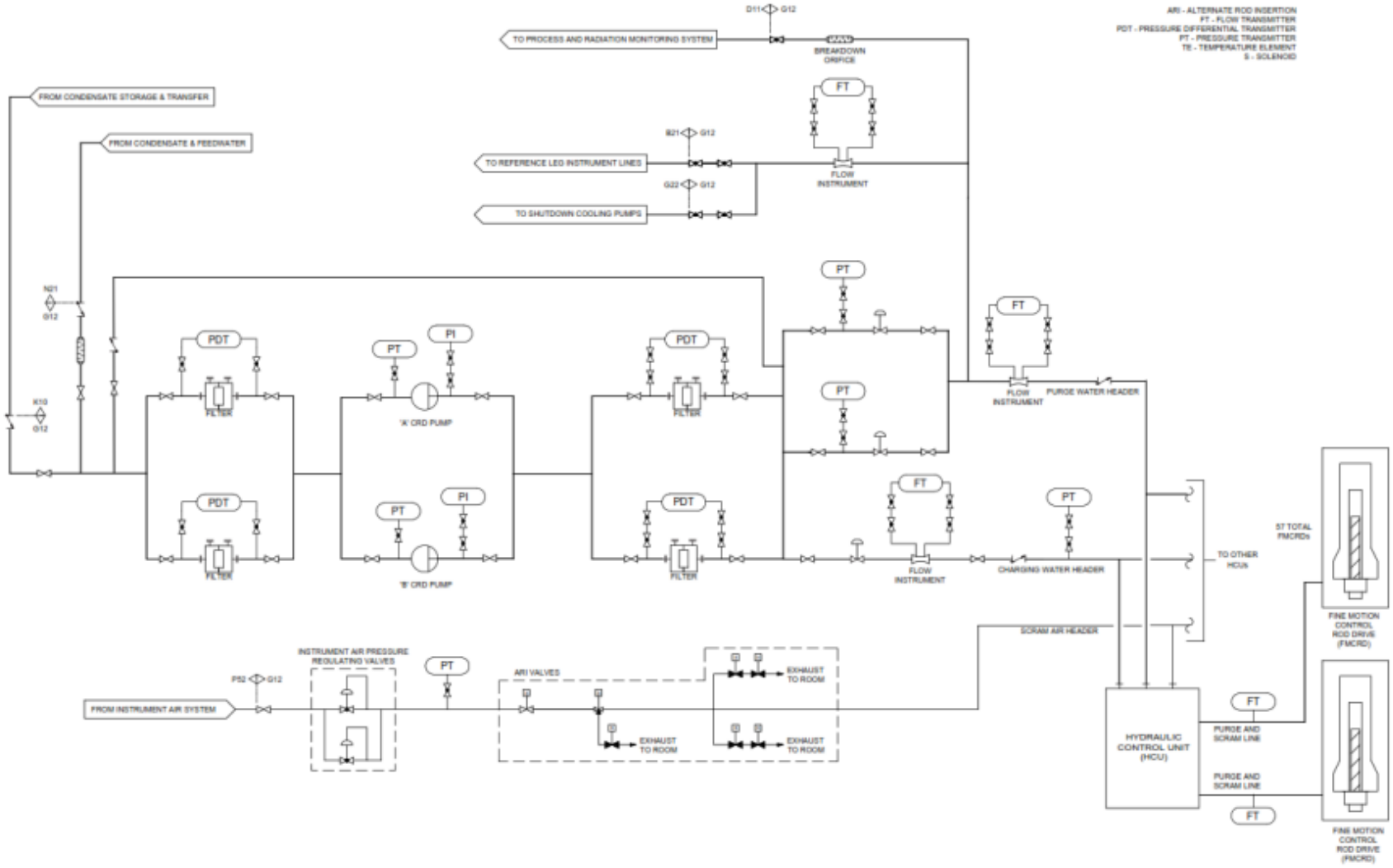


Figure 4.6-9: Control Rod Drive Hydraulic Subsystem Simplified Diagram

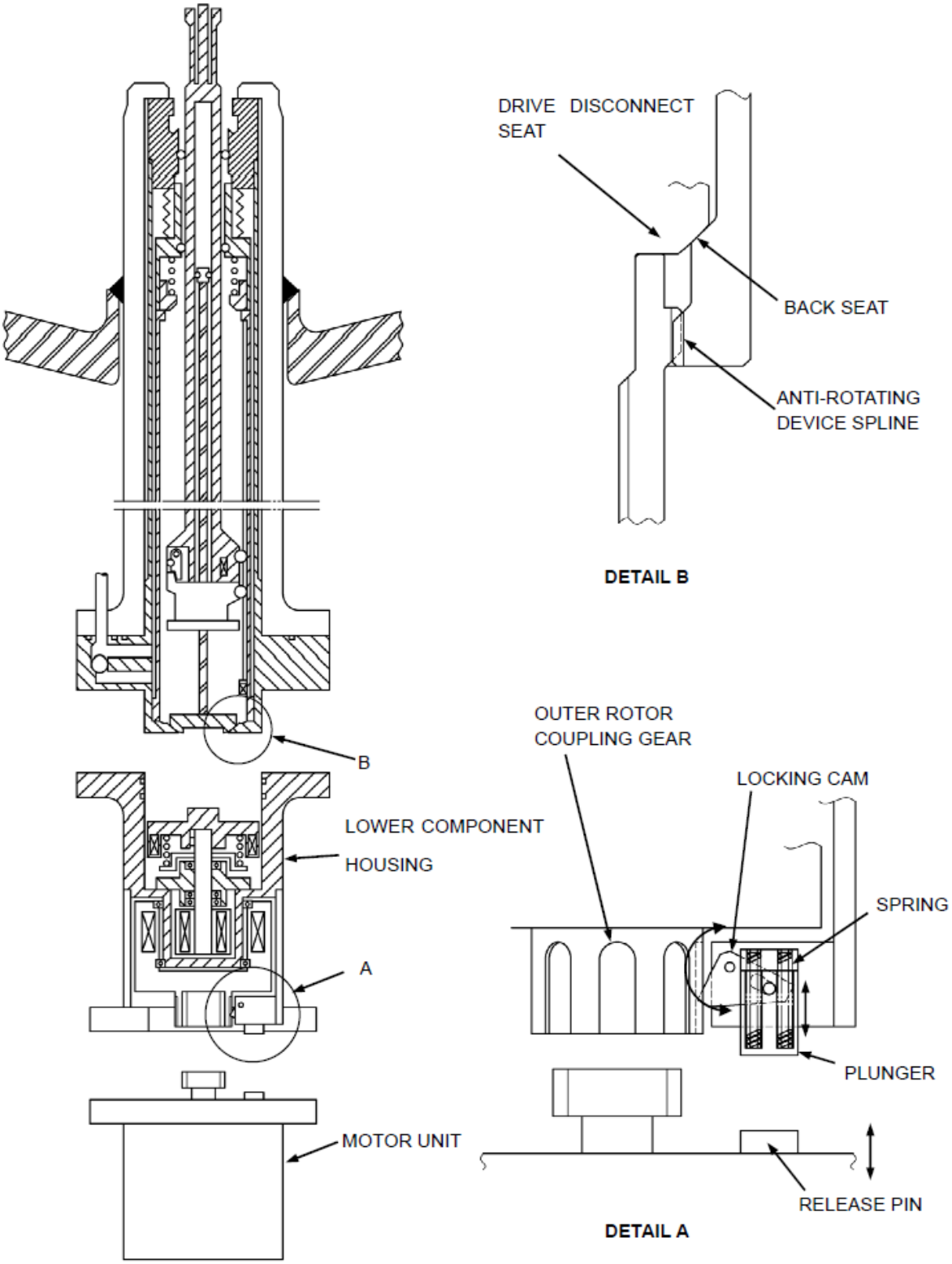


Figure 4.6-10: Representative Fine Motion Control Rod Drive Anti-Rotation Devices

4A THERMAL-HYDRAULIC STABILITY

4A.1 Background

Under certain conditions, BWRs can be susceptible to coupled neutronic/thermal-hydraulic oscillations (instabilities). These instabilities are characterized by periodic power and flow oscillations and are the result of density waves in the core. If the power and flow oscillations become large, and the density waves contain a sufficiently high void fraction, the Fuel Cladding Integrity Safety Limit could be challenged. Compliance to GDC12 was previously addressed in the USNRC-approved LTR NEDC-33912P-A, Revision 2, "BWRX-300 Reactivity Control," (Reference 4A-1). The BWRX-300 stability analysis is described in Subsection 15.5.2.

As documented in NEDC-33922-P-A, "BWRX-300 Containment Evaluation Method," (Reference 4A-2), the USNRC determined that TRACG is an acceptable analysis method for the BWRX-300. USNRC-approved TRACG methodology is utilized to support the BWRX-300 stability analysis. NEDC-34270P, "BWRX-300 Stability Analysis," (Reference 4A-3) requests USNRC approval to support BWRX-300 licensing applications and is specific to the BWRX-300 design. Use of an implicit integration scheme, BWRX-300 stability analysis, and a decay ratio acceptance criterion of ≤0.80 are the key purposes of the LTR. NEDC-34270 also supplements the stability analysis in Subsection 15.5.2.

4A.2 Design Bases

The reactor core and associated coolant, control, and protection systems are designed with appropriate margin to assure that power oscillations that can result in conditions exceeding the SAFDLs are not possible or can be reliably and readily detected and suppressed.

4A.3 Types of Boiling Water Reactor Oscillations

The potential for instability is a subset of normal operation and baseline AOO analyses. Reactor instabilities may be in the form of core-wide or regional oscillations.

Core-wide oscillations occur when all channels are in phase to create core-wide oscillations and instability.

Regional oscillations occur when some of the channels (~half) are out of phase with other channels. This is typically associated with larger cores where there is a possibility for less nuclear and thermal-hydraulic coupling. As the BWRX-300 is a very small core, regional instability is not expected. TRACG analyses are performed to confirm that the core decay ratio meets the design criteria to show that the BWRX-300 plant/core design will be stable through normal operations.

4A.4 Design Criteria

The most limiting stability condition, as defined by the highest decay ratio (defined below), in the BWRX-300 normal operating region is at full power conditions.

In the time domain analysis, decay ratio is defined as the ratio of the amplitude of two successive peaks. For the BWRX-300, the decay ratio is averaged over several cycles to reduce analytical uncertainty.

Conservative design criteria are imposed on the core-wide, regional, and single-channel decay ratios under all conditions of normal operation and anticipated transients. The limiting mode (i.e., highest decay ratio) for thermal-hydraulic oscillations is the core-wide mode.

Analyses are performed to confirm that the acceptance criterion for the core decay ratio of ≤ 0.80 is met, and to show that the BWRX-300 core will be stable through normal operations and AOOs considering uncertainties. Along with confirming stable operation, this also ensures that the Safety Limit MCPR is protected.

4A.5 Stability Solution Design

The BWRX-300 design has features that result in stable behavior in normal operation and minimize the effects of potential oscillations in off-normal conditions include:

- Small Core:

  The small core size of the BWRX-300 reduces the likelihood of regional mode oscillations. Core-wide oscillations are the dominant mode. Tighter neutronic coupling precludes regional mode oscillations. Mitigation of core-wide oscillations is described in Subsection 15.5.2.

- Natural circulation:
  - As there are no recirculation pumps, recirculation pump trip transients that can induce oscillations have been eliminated.
  - Loss of Feedwater Heating (LFWH) AOO effect on stability is mitigated by SCRRI operation that reduces the core thermal power. The SCRRI function is described in Subsection 7.3.3. The LFWH AOO analysis is described in Subsection 15.5.3.
- Tall chimney:
  - Increases volume of water
  - Increases driving head and natural circulation flow
  - Dampens oscillations
- Optimized inlet core orifice design

4A.6 Stability Analysis Methods

A detailed discussion of the methods used to analyze BWRX-300 thermal-hydraulic stability is presented in NEDC-34270P," BWRX-300 Stability Analysis," (Reference 4A-3). NEDC-34270P provides a methodology utilizing implicit numerical integration for channel components and a nominal core wide decay ratio acceptance criterion.

The methodology described in NEDC-34270P leverages NEDC-34043P, "BWRX-300 TRACG Application," (Reference 4A-4), documenting its applicability to stability analysis.

TRACG is a GVH proprietary version of the Transient Reactor Analysis Code. TRACG uses a multi-dimensional, two-fluid model for reactor thermal-hydraulics and a 3D reactor kinetics model. The models can be used to accurately simulate a large variety of test and reactor configurations. These features allow for realistic simulation of a wide range of BWR phenomena and are described in detail in NEDE-32176P, "TRACG Model Description," (Reference 4A-5).

TRACG has been extensively qualified against separate effects tests, component performance data, integral system effects tests, and operating BWR plant data. The details are presented in NEDE-32177P, "TRACG Qualification," (Reference 4A-6). TRACG has been used in a variety of applications for the operating BWRs as well as design and analysis for the ESBWR. Application for the ESBWR stability analysis is documented in NEDC-33083P-A, "TRACG Application for ESBWR," (Reference 4A-7).

The latest TRACG base deck and PANAC core simulator wrapups have been used to evaluate the BWRX-300 stability performance. This evaluation supports the BWRX-300 design to evaluate decay ratio performance for the 100% nominal power condition where the BWRX-300 would have the least margin for stability in normal operation.

This stability analysis performed in accordance with NEDC-34270P using the TRACG code demonstrates that the BWRX-300 maintains a coupled power-flow response such that any operational perturbation, maneuver, or AOO that does not cause an immediate scram is naturally damped, and decays quickly to steady state for all modes of operation and prevents the SAFDLs from being exceeded. The BWRX-300 is not susceptible to regional mode oscillations. Therefore, a special stability detection and trip system is not required as documented in NEDC-33912P-A (Reference 4A-1).

This analysis, including the decay ratio, is discussed in Subsection 15.5.2, and the acceptance criteria for the SAFDLs is provided in Section 15.3.

4A.7 References

4A-1 NEDC-33912P-A, “BWRX-300 Reactivity Control,” Revision 2, GE-Hitachi Nuclear Energy Americas, LLC, February 2023.

4A-2 NEDC-33922P-A, “BWRX-300 Containment Evaluation Method,” Revision 3, GE-Hitachi Nuclear Energy Americas, LLC, June 2022.

4A-3 NEDC-34270P, “BWRX-300 Stability Analysis,” GE-Hitachi Nuclear Energy Americas, LLC, Revision 0, March 2025.

4A-4 NEDC-34043P, “BWRX-300 TRACG Application,” Revision 0, GE-Hitachi Nuclear Energy Americas, LLC, May 2024.

4A-5 NEDE-32176P, “TRACG Model Description,” Revision 4, GE-Hitachi Nuclear Energy Americas, LLC, January 2008.

4A-6 NEDE-32177P, “TRACG Qualification,” Revision 3, GE-Hitachi Nuclear Energy Americas, LLC, August 2007.

4A-7 NEDC-33083P-A, “TRACG Application for ESBWR,” Revision 1, GE-Hitachi Nuclear Energy Americas, LLC, March 2005.

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BE-POV-LCNRG-LIC-5009

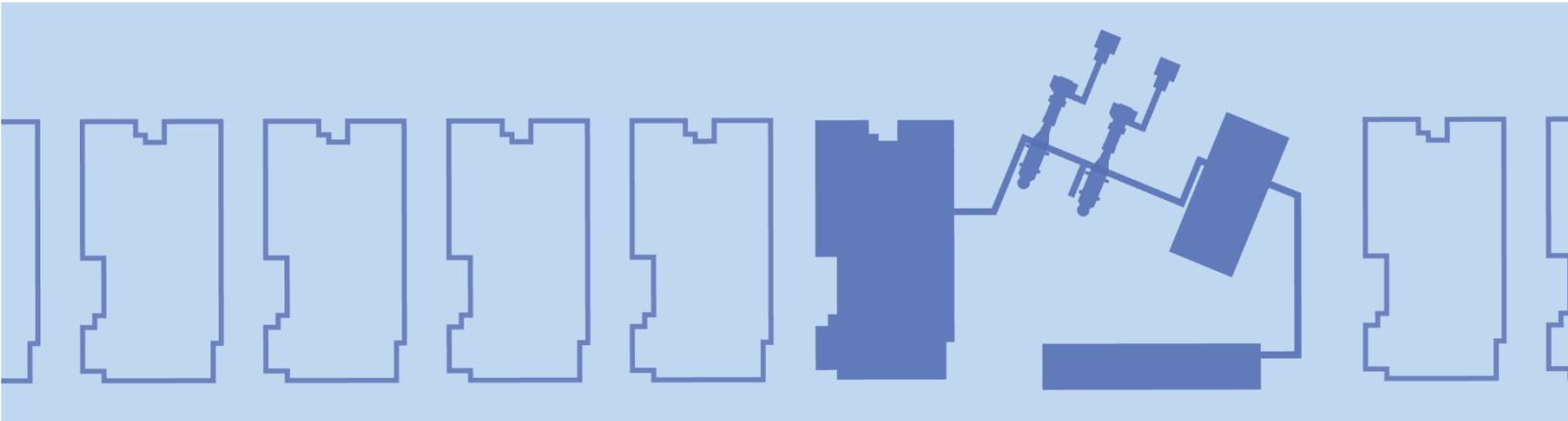
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Non-Proprietary Information

Chapter 5

Reactor Coolant System and Connected Systems



ACRONYM LIST

Acronym	Explanation
2oo3	Two Out of Three
ALARA	As Low As Reasonably Achievable
ALWR	Advanced Light Water Reactor
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
ASTM	ASTM International
AWS	American Welding Society
BIS	Boron Injection System
BPVC	Boiler and Pressure Vessel Code
BTP	Branch Technical Position
BWR	Boiling Water Reactor
CCS	Containment Cooling System
CFD	Condensate Filter and Demineralizers System
CFR	Code of Federal Regulations
CFS	Condensate and Feedwater Heating System
CIV	Containment Isolation Valve
CMon	Containment Monitoring Subsystem
CRD	Control Rod Drive
CUW	Reactor Water Cleanup System
ECCS	Emergency Core Cooling System
EPRI	Electric Power Research Institute
FAC	Flow Accelerated Corrosion
FE	Flow Element
FMCRD	Fine Motion Control Rod Drive
FPC	Fuel Pool Cooling and Cleanup System
FSAR	Final Safety Analysis Report
FT	Flow Transmitter
FW	Feedwater
FWCIV	Feedwater Containment Isolation Valve
FWH	Feedwater Heater
FWRIV	Feedwater Reactor Isolation Valve

Acronym	Explanation
GDC	General Design Criteria
GVH	GE Vernova Hitachi Nuclear Energy
HAZ	Heat Affected Zone
HVS	Heating, Ventilation, and Cooling System
HWC	Hydrogen Water Chemistry
I&C	Instrumentation and Control
IASCC	Irradiation Assisted Stress Corrosion Cracking
IC	Isolation Condenser
ICRRIV	Isolation Condenser Return Reactor Isolation Valve
ICS	Isolation Condenser System
ICSRIV	Isolation Condenser Supply Reactor Isolation Valve
IEEE	Institute of Electrical and Electronic Engineers
IGSCC	Intergranular Stress Corrosion Cracking
ISI	Inservice Inspection
IST	Inservice Testing
LOCA	Loss-of-Coolant Accident
LTR	Licensing Topical Report
LWM	Liquid Waste Management System
LWR	Light Water Reactor
MCA	Main Condenser and Auxiliaries
MCR	Main Control Room
MS	Main Steam
MSCIV	Main Steam Containment Isolation Valve
MSL	Main Steam Line
MSRIV	Main Steam Reactor Isolation Valve
NBS	Nuclear Boiler System
NDE	Non-Destructive Examination
NPSH	Net Positive Suction Head
NUREG	Nuclear Regulatory Report
OBE	Operating-Basis Earthquake
OLNC	On-Line NobleChem
OPEX	Operating Experience

Acronym	Explanation
PCS	Primary Containment System
PCW	Plant Cooling Water System
PDC	Principal Design Criterion
PREMS	Process Radiation and Environmental Monitoring System
PRM	Process Radiation Monitoring Subsystem
PT	Liquid Penetrant Testing
PWHT	Post-Weld Heat Treatment
RB	Reactor Building
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RG	Regulatory Guide
RHX	Regenerative Heat Exchanger
RIV	Reactor Isolation Valve
RPV	Reactor Pressure Vessel
SC	Safety Class
SC1	Safety Class 1
SC2	Safety Class 2
SC3	Safety Class 3
SCC	Stress Corrosion Cracking
SCN	Non-Safety Class
SDC	Shutdown Cooling System
SSCs	Structures, Systems, and Components
SSE	Safe Shutdown Earthquake
TAF	Top of Active Fuel
TB	Turbine Building
TS	Technical Specification
TSV	Turbine Stop Valve
USNRC	U.S. Nuclear Regulatory Commission
UT	Ultrasonic Testing

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5.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 Summary Description

The BWRX-300 is a natural circulation reactor and therefore does not rely on pumps to drive core coolant flow. The thermal-hydraulic design of the natural circulation reactor is described in Section 4.4.

The Reactor Coolant System (RCS) is comprised of the Safety Class 1 (SC1) portion of the BWRX-300 Nuclear Boiler System (NBS) and the SC1 portion of the Condensate and Feedwater Heating System (CFS). The RCS extends to and includes the outermost Containment Isolation Valve (CIV) in the Main Steam (MS) and Feedwater (FW) piping. The RCS is depicted in Figure 5.1-1.

The Reactor Coolant Pressure Boundary (RCPB) is comprised of the pressure retaining components of the RCS and connected systems up to and including their respective outboard CIVs. RCPB components are identified in Section 5.2 and depicted in Figure 5.1-2. The connected systems include the following:

- The MS subsystem which connects to the Reactor Pressure Vessel (RPV)
- The CFS which connects to the RPV
- The Isolation Condenser System (ICS) which connects to the RPV
- The Reactor Water Cleanup System (CUW) which connects with the RPV
- The Boron Injection System (BIS) which interfaces with ICS for the flow path to the RPV
- Reactor Pressure Vessel Head Vent Subsystem which connects to the RPV

The Isolation Condensers (IC), which are ICS components, are in the Reactor Building (RB) directly above the Primary Containment System (PCS). The ICS is described in Section 6.3. The Shutdown Cooling System (SDC), which interfaces with the ICS and CFS outside the PCS for the flow path with the RPV, is described in Subsection 5.4.7. The CUW is described in Subsection 5.4.8. The BIS is described in Subsection 9.3.10. System interfaces are depicted in Figure 5.1-2 and Figure 5.1-3.

The NBS includes the RPV and provides MS flow paths from the RPV to the steam turbine, and the CFS provides FW flow paths to the RPV. The NBS, up to and including the outermost Main Steam Containment Isolation Valves (MSCIVs), is illustrated on Figure 5.1-3. The MS subsystem and the CFS piping and components outboard of the CIVs are discussed in Chapter 10, Section 10.3 and Subsection 10.4.7, respectively. Additional descriptions of the MS and FW piping and components, within the scope of Chapter 5, are provided in Subsection 5.4.3, Subsection 5.4.4, Subsection 5.4.5, and Subsection 5.4.9.

5.1.1 Schematic Flow Diagrams

RPV volumes are shown in Figure 5.1-4. A schematic flow diagram of the RCS is shown in Figure 5.1-5 and depicts nominal pressures, temperatures, and flow rates for normal steady-state operating conditions at rated power.

5.1.2 Simplified Diagrams

Simplified Diagrams are provided for the following systems:

- RCS as a subsystem of NBS and CFS, Figure 5.1-1
- RCPB, Figure 5.1-2

- NBS up to and including MSCIVs, Figure 5.1-3
- SDC, Figure 5.4-2 and Figure 5.4-3
- CUW, Figure 5.4-4

5.1.3 Elevation Schematics

Figures in Section 1.2 provide plan views at various elevations showing the RPV in relation to the PCS and RB. The RPV height and diameter are provided in Table 1.2-1.

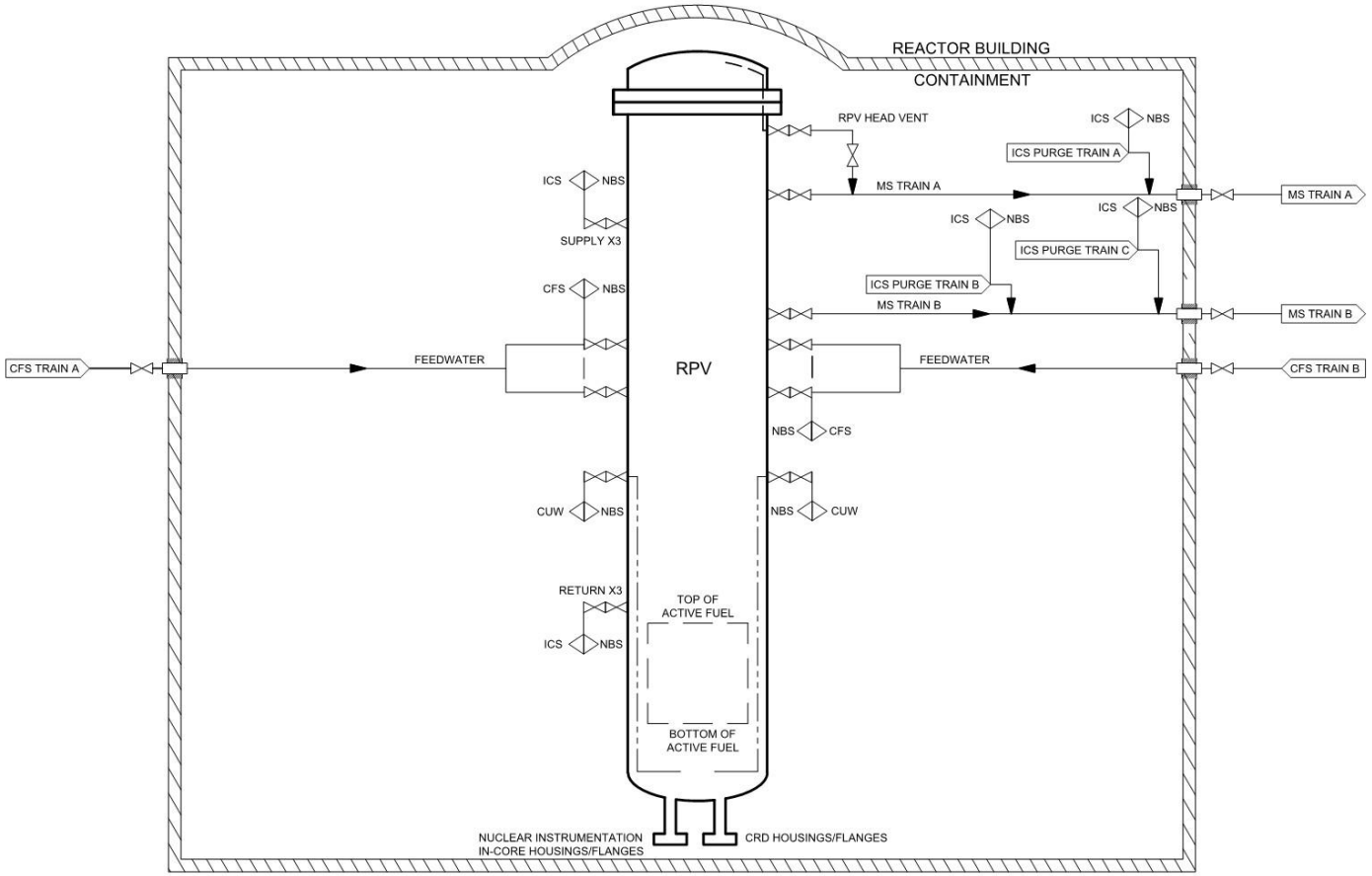


Figure 5.1-1: BWRX-300 Reactor Coolant System Simplified Diagram

Notes for this figure follow on the next page.

Notes for Figure 5.1-1:

1. Figure 1.7-2 provides a symbols legend. RPV head vent and CUW piping internal to the RPV is shown in phantom line.
2. Interfacing system breaks are provided to identify interfaces for the NBS, ICS, CFS, and CUW.
3. Reactor Isolation Valves (RIV) are two valves in a single body and directly bolted to RPV nozzles. RIVs are further discussed in Subsection 5.4.3.
4. All valves shown have power operators.
5. The RCS flow path starts at and includes the Feedwater Containment Isolation Valves (FWCIVs), passes through the FW piping, the RPV, the MS piping, and ends at the MSCIVs.
6. Safety Class/Quality Group/Seismic Category breaks are not shown.
7. Valve location on RPV is representative only.

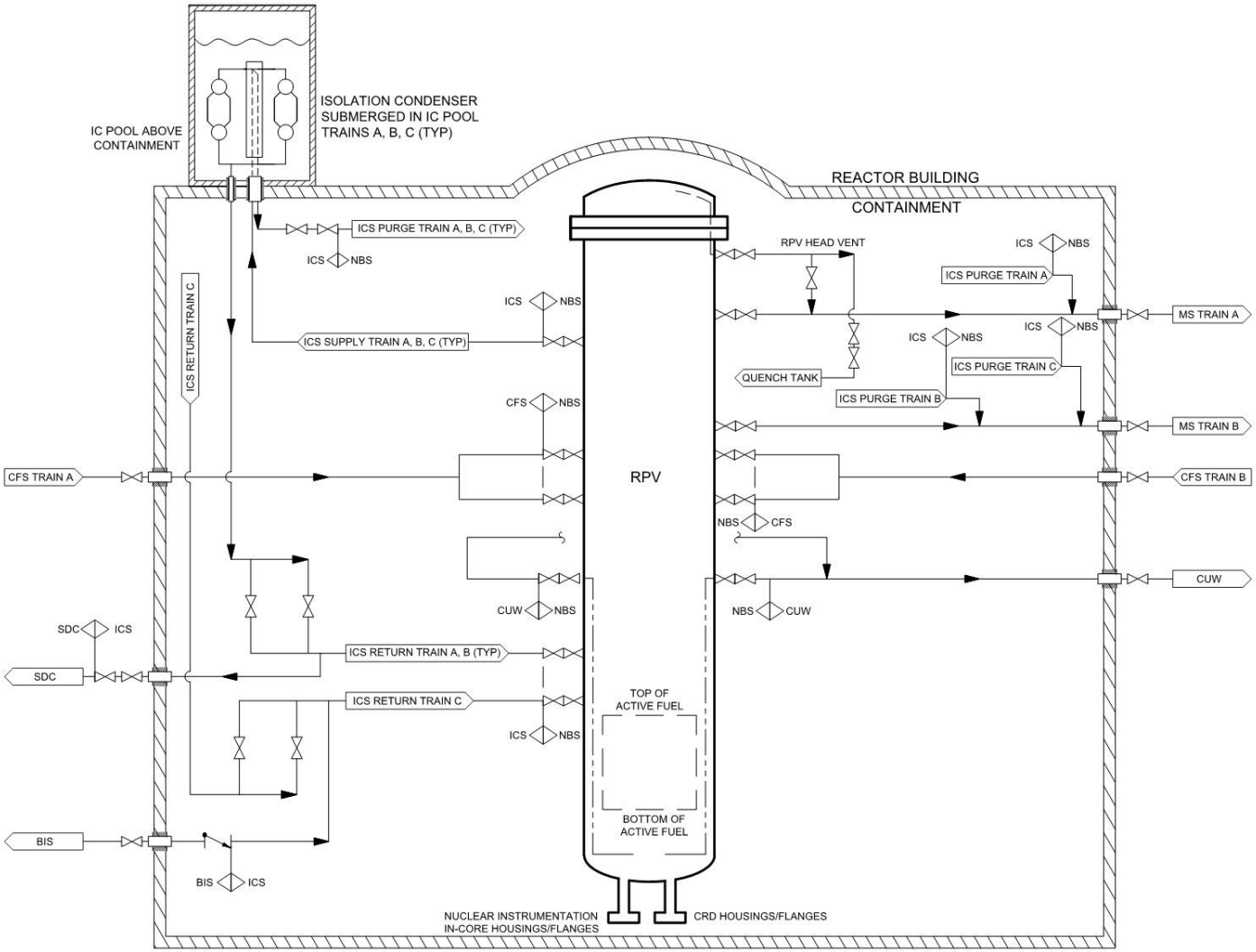


Figure 5.1-2: BWRX-300 Reactor Coolant Pressure Boundary

Notes for this figure follow on the next page.

Notes for Figure 5.1-2:

1. Figure 1.7-2 provides a symbols legend. RPV head vent and CUW piping internal to the RPV are shown in phantom line.
2. Interfacing system breaks are provided to identify interfaces for the NBS, ICS, CFS, CUW, BIS, and SDC.
3. RIVs are two valves in a single body and directly bolted to RPV nozzles. RIVs are further discussed in Subsection 5.4.3.
4. ICS consists of three independent trains (identified as Trains A, B, and C), each containing an IC.
5. All valves shown have power operators except for the BIS Inboard CIV, which is a check valve.
6. Safety Class/Quality Group/Seismic Category breaks are not shown.
7. Valve location on RPV is representative only.
8. TYP is an abbreviation for typical.

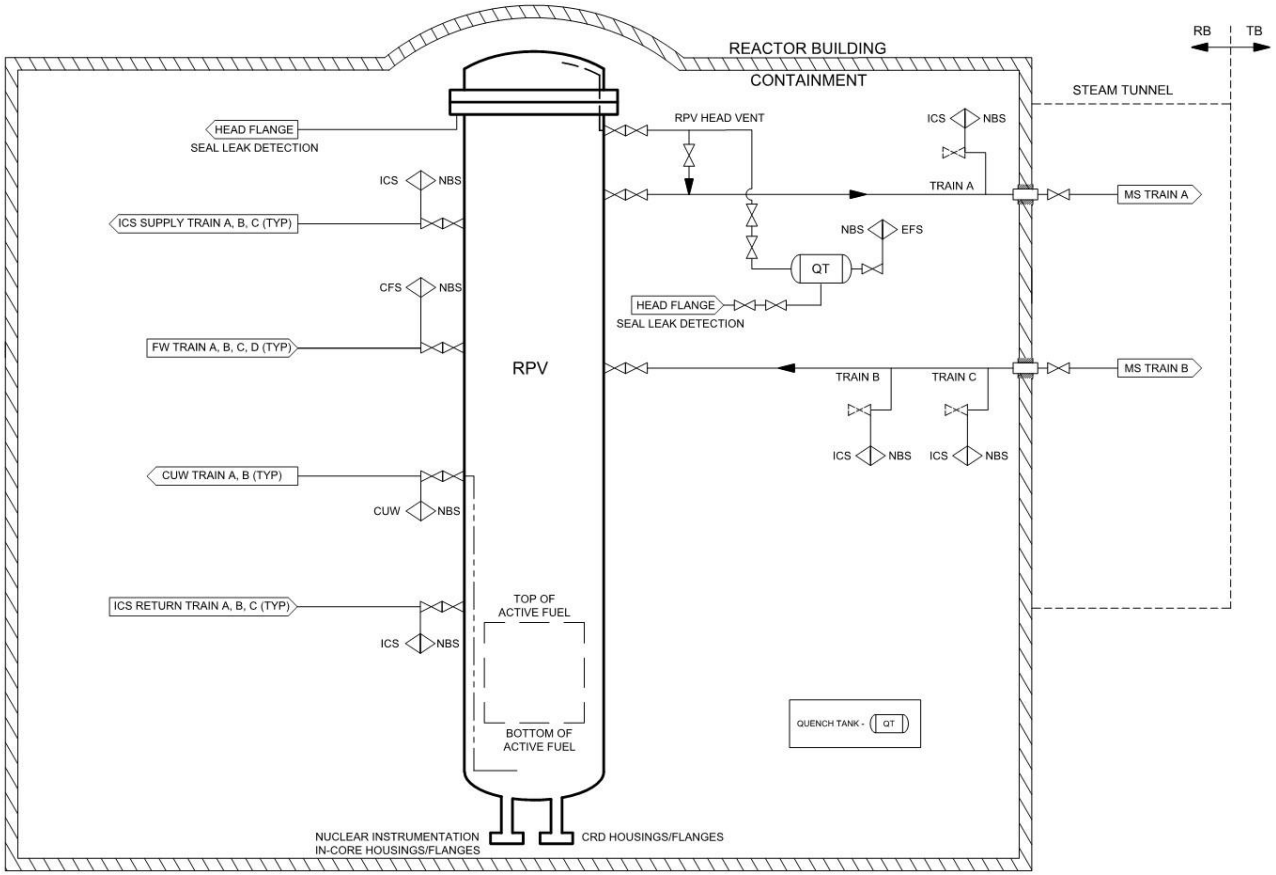


Figure 5.1-3: BWRX-300 Nuclear Boiler System Simplified Diagram

Notes for this figure follow on the next page.

Notes for Figure 5.1-3:

1. Figure 1.7-2 provides a symbols legend for symbols not identified in the figure. RPV head vent and CUW piping internal to the RPV are shown in phantom line.
2. Interfacing system breaks are provided to identify system interfaces for the NBS, ICS, CFS, and CUW.
3. RIVs are two valves in a single body and directly bolted to RPV nozzles. RIVs are further discussed in Subsection 5.4.3.
4. NBS piping downstream of the MSCIVs is illustrated on Figure 10.3-1.
5. MSCIVs are located within the MS Tunnel shown in phantom line.
6. All valves shown have power operators.
7. Safety Class/Quality Group/Seismic Category breaks are not shown.
8. Valve location on RPV is representative only.
9. TYP is an abbreviation for typical

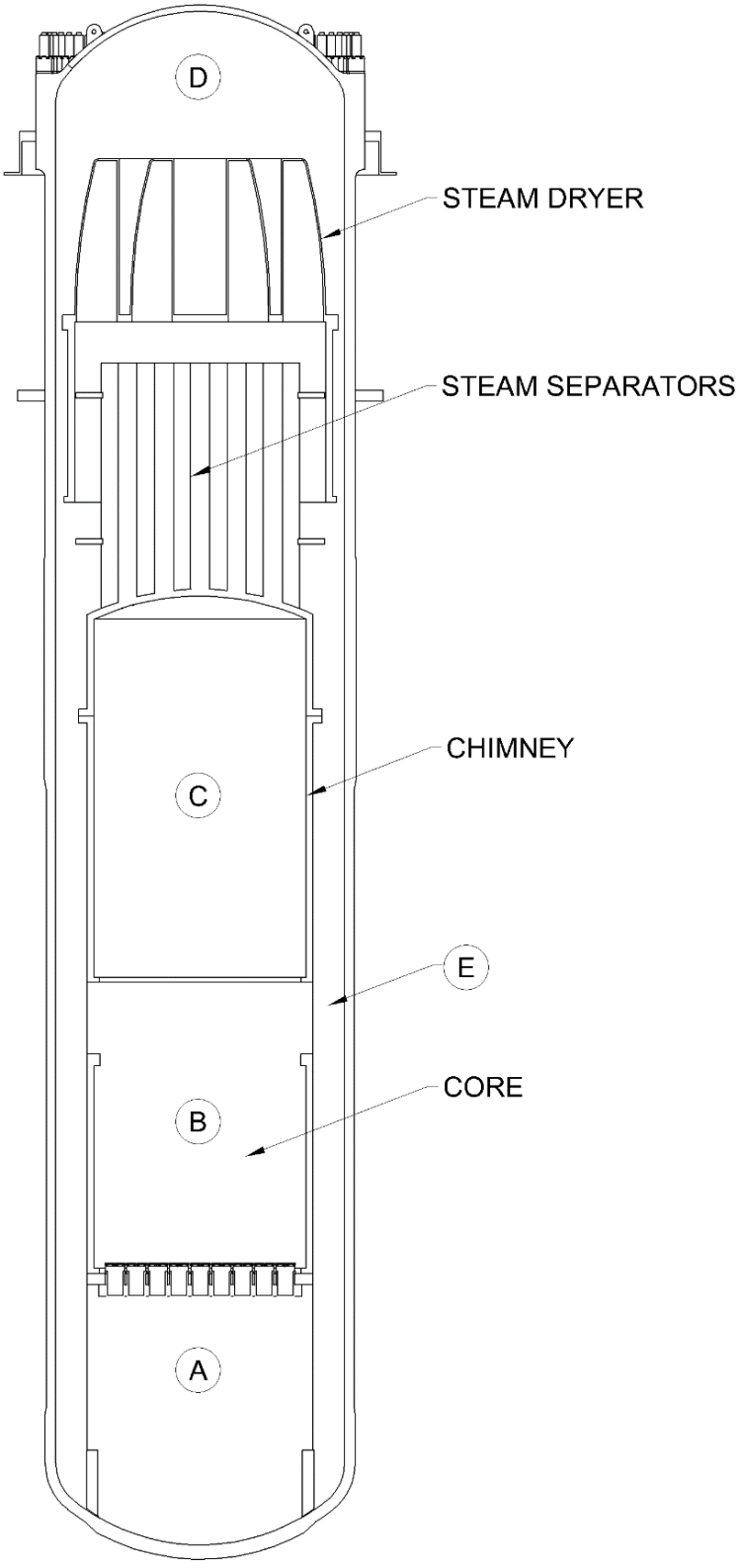


Figure 5.1-4: BWRX-300 Reactor Pressure Vessel Volumes at 100% Power

Note: Volume descriptions and their values at locations A, B, C, D, and E are provided in the table on the next page.

Volume Location	Volume Description	Fluid Volume m^3
A	Lower Plenum	32
B	Core	26
C	Upper Plenum (includes chimney and separator standpipe interior up to normal water level)	104
D	Dome (above normal water level)	75
E	Downcomer Region (below normal water level)	101

Figure 5.1-4: BWRX-300 RPV Volumes at 100% Power

Note: Volume locations are identified in the schematic on Figure 5.1-4 on the previous page.

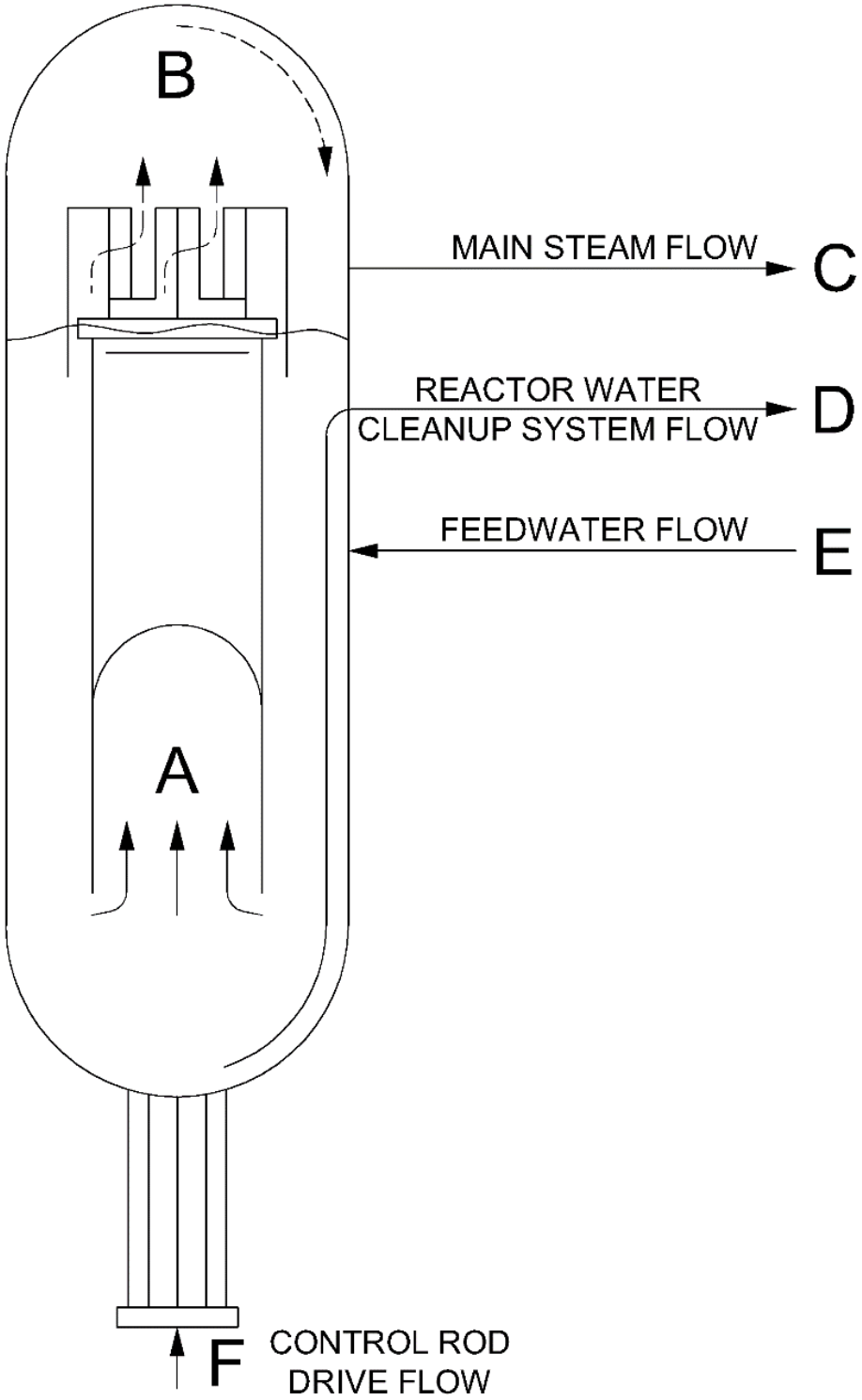


Figure 5.1-5: BWRX-300 Reactor Coolant System Schematic Flow Diagram at 100% Power

Note: Flow parameters and their values at locations A, B, C, D, E, and F are provided in the table on the next page.

Location (Note 1)	**Parameter** (Note 1)	**Value** (Approximate Nominal)	**Units**
A (Reactor Core)	Rated Thermal Power	870 MWth	-
	Core Flow	1890.0 (15)	kg/s (Mlbm/hr)
B (Reactor Dome)	Dome Pressure	7.17 (1040)	MPa (psia)
C (Main Steam) (Note 2)	MS Flow	503.2 (3.993)	kg/s (Mlbm/hr)
	MS Pressure	6.89 (1000)	MPa (psia)
	MS Temperature	284.8 (544.6) Saturation Temperature	°C (°F)
D (Reactor Water Cleanup System)	CUW Flow	5.1 (40240)	kg/s (lbm/hr)
	CUW Temperature	276.6 (529.9)	°C (°F)
E (Feedwater)	CFS Flow	507.0 (4.024)	kg/s (Mlbm/hr)
	CFS Temperature	241.8 (467.3)	°C (°F)
F (Control Rod Drive System)	Control Rod Drive (CRD) Flow	1.2 (9647.5)	kg/s (lbm/hr)
	CRD Temperature	43.4 (110.0)	°C (°F)

Figure 5.1-5: BWRX-300 RCS Schematic Flow Diagram at 100% Power

Notes:

1. Parameters are grouped by location, and locations are identified in the schematic portion of this figure.
2. Parameters for MS are upstream of the Turbine Stop Valve (TSV) except MS Temperature is at the turbine inlet.

5.2 Reactor Coolant Pressure Boundary Integrity

The BWRX-300 RCPB is designed in accordance with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) to maintain its integrity for the design life of the plant. The RCPB is consistent with the definition provided in 10 Code of Federal Regulations (CFR) 50.2 and includes the pressure retaining components of the RCS identified in Figure 5.1-1.

The RCPB, as depicted in Figure 5.1-2, includes:

- The RPV
- The RIVs
- The piping from the RIVs up to and including each respective outboard CIV (this includes the redundant isolation valve downstream of the CIV connected to the ICS to SDC interface piping)
- The ICS steam supply lines, ICs, ICS purge lines, and ICS condensate return lines
- The head vent piping to the Main Steam Line (MSL) and head vent interface piping to quench tank up to and including the isolation valves for the quench tank
- The reactor water level instrumentation lines
- The CRD flanges/housings
- The incore (nuclear) instrumentation housings

5.2.1 Compliance with Codes and Standards Rule, 10 CFR 50.55a

5.2.1.1 Compliance with 10 CFR 50.55a

The BWRX-300 meets the relevant requirements of the following regulations:

10 CFR 50, Appendix A, General Design Criterion 1, as it relates to the requirement that Structures, Systems, and Components (SSCs) are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. Regulatory Guide (RG) 1.26, “Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” is used to meet GDC 1. The quality group of BWRX-300 components including RCPB components are provided on Table 3A-1.

10 CFR 50.55a, as it relates to establishing minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of components within the RCPB and other fluid systems, by requiring conformance with appropriate editions of specified published industry codes and standards. Components of the RCPB and other fluid systems within the scope of 10 CFR 50.55a (Quality Groups A, B, and C) are identified in Table 3A-1. The BWRX-300 complies with the requirements of 10 CFR 50.55a(a) through 10 CFR 50.55a(e). The BWRX-300 is designed and fabricated in accordance with the 2021 Edition of ASME BPVC, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” (Reference 5.2-4). Limitations and modifications on ASME BPVC, Section III specified in 10 CFR 50.55a(b)(1) are met.

Pursuant to 10 CFR 50.55a(f)(4)(i) and 10 CFR 50.55a(g)(4)(i), the initial Inservice Testing (IST) and Inservice Inspection (ISI) programs, respectively, will use the edition and addenda of the ASME OM Code, “Operation and Maintenance of Nuclear Power Plants,” (Reference 5.2-8) and of the ASME BPVC, Section XI, Division 1, “Rules for Inspection and Testing of Components of Light-Water-Cooled Plants,” (Reference 5.2-6) that is incorporated by reference in 10 CFR 50.55a

18 months prior to the date of issuance of the operating license and the codes will be met subject to the conditions in 10 CFR 50.55a. Accordingly, the Code Edition for the ASME OM Code and ASME BPVC, Section XI will be addressed in the Final Safety Analysis Report (FSAR).

The RCPB is comprised of ASME Class 1 and ASME Class 2 components as detailed below.

The following RCPB components are Quality Group A, ASME Class 1 components designed to meet the requirements of ASME BPVC, Section III, Division 1, Subsection NB:

- The RPV
- The RIVs
- The ICS including steam supply lines, ICs, purge lines, purge valves, condensate return lines, condensate return valves, ICS to SDC interface piping (including CIV and redundant isolation valve), and ICS to BIS interface piping (including CIV)
- The CRD flanges/housings
- The incore (nuclear) instrumentation housings

The following RCPB components are Quality Group B, ASME Class 2 components designed to meet the requirements of ASME BPVC, Section III, Division 1, Subsection NCD:

- The piping from the RIVs up to and including each respective outboard CIV except for ICS components (this includes the redundant isolation valve downstream of the CIV connected to the ICS to SDC interface piping)
- The outboard CIVs for MS, CUW, and CFS
- The head vent piping to the MSL and head vent interface piping to quench tank up to and including the isolation valves for the quench tank
- The reactor water level instrumentation lines

The RCPB piping outboard of the RIVs is classified ASME Class 2, excluding ICS piping up to the ICs as allowed by 10 CFR 50.55a(c)(2)(ii). The RPV level instrumentation sensing lines are classified as ASME Class 2 per ASME BPVC, Section III, NB-3630(d).

In addition to identifying Quality Group A and Quality Group B components for the RCPB, Table 3A-1 also identifies Quality Group B and Quality Group C components for fluid systems that are not part of the RCPB. For these other fluid systems, Quality Group B components are ASME Class 2 components and Quality Group C components are ASME Class 3 components.

Subsection 5.2.4 describes compliance with the ISI and IST requirements of ASME BPVC, Section XI, for RCPB ASME Class 1 components. Section 6.6 describes compliance with the ISI and IST requirements of ASME BPVC, Section XI, for ASME Class 2 and ASME Class 3 components. Compliance with 10 CFR 50.55a(f), as it relates to IST programs for valves and dynamic restraints, is described in Subsection 3.9.6.

5.2.1.2 Applicable Code Cases

The BWRX-300 meets the relevant requirements of the following regulations:

10 CFR 50, Appendix A, GDC 1, as it relates to the requirement that structures, systems and components are designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. RG 1.84, RG 1.147, and RG 1.192 as discussed below are used to meet GDC 1.

10 CFR 50.55a, as it relates to establishing minimum quality standards for the design, fabrication, erection, construction, testing and inspection of components within the RCPB and other fluid systems, by requiring compliance with appropriate editions of specified published industry codes and standards. The ASME BPVC editions used to meet 10 CFR 50.55a are provided in Subsection 5.2.1.1. The editions of RG 1.84, RG 1.147, and RG 1.192 used to meet 10 CFR 50.55a are those incorporated by reference in 10 CFR 50.55a(a)(3) and clarified below.

To meet the requirements of GDC 1 and 10 CFR 50.55a, the following RGs are used pursuant to 10 CFR 50.55a(a)(3):

RG 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III," pursuant to 10 CFR 50.55a(a)(3)(i). This guide lists those ASME BPVC, Section III, Code Cases oriented to design and fabrication that are acceptable to the U.S. Nuclear Regulatory Commission (USNRC) staff, with conditions or restrictions for implementation in the licensing of nuclear power plants.

RG 1.147, "Inservice Inspection Code Case Acceptability, ASME BPVC, Section XI, Division 1," pursuant to 10 CFR 50.55a(a)(3)(ii). This guide lists those ASME BPVC, Section XI, Code Cases that are acceptable to the USNRC staff, with conditions or restrictions, for use in the ISI of light-water-cooled nuclear power plants. The revision of this RG will be addressed in the FSAR as discussed below.

RG 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code," pursuant to 10 CFR 50.55a(a)(3)(iii). This guide lists those ASME OM Code Cases that are acceptable to the USNRC staff, with conditions or restrictions, for implementation of a program for IST programs for valves and dynamic restraints in light-water cooled nuclear power plants. The revision of this RG will be addressed in the FSAR as discussed below.

ASME Code Cases provide alternatives to the ASME BPVC provisions and are developed and approved by ASME. The RGs pursuant to 10 CFR 50.55a(a)(3), identify ASME BPVC Code Cases determined to be acceptable alternatives to the code requirements of ASME BPVC, Section III, ASME BPVC, Section XI and the ASME OM Code. Code Cases listed in Table 1 of RG 1.84, RG 1.147, and RG 1.192 are acceptable for use without conditions while Code Cases listed in Table 2 of RG 1.84, RG 1.147, and RG 1.192 are subject to specified conditions.

Pursuant to 10 CFR 50.55a(f)(4)(i) and 10 CFR 50.55a(g)(4)(i), the initial IST and ISI programs, respectively, are to use the latest revisions of RG 1.147 and RG 1.192 subject to the conditions identified and incorporated by reference into 10 CFR 50.55a 18 months before the date of issuance of the operating license. Pursuant to 10 CFR 50.55a(f)(3)(iii), 10 CFR 50.55a(f)(3)(iv), 10 CFR 50.55a(g)(2)(ii), and 10 CFR 50.55a(g)(3)(ii), concerning design and accessibility requirements and preservice examination requirements, the revisions of RG 1.147 and RG 1.192 are to be the latest revisions subject to the conditions identified and incorporated by reference into 10 CFR 50.55a at the time the construction permit is issued. The revisions of these RGs will be addressed in the FSAR accordingly.

The Code Cases in RG 1.193, "ASME Code Cases Not Approved for Use," cannot be used unless authorized by the USNRC pursuant to the requirements of 10 CFR 50.55a(z).

Table 5.2-1 lists specific Code Cases that are applied to ASME BPVC, Section III, Division 1 components.

5.2.2 Overpressure Protection

Overpressure protection is discussed in Section 6.3. Materials evaluation of ICS RCPB components is discussed in Subsection 5.2.3. Chapter 15 discusses the acceptance criteria, calculational results, and safety margins for the analysis of Anticipated Operational Occurrences

(AOOs) and Design Basis Accident event sequences where the reactor coolant pressure boundary remains intact.

5.2.3 Reactor Coolant Pressure Boundary Materials

This subsection addresses materials of the RCPB which includes the RPV and supplements the information on RPV materials covered in Subsection 5.3.1.

The BWRX-300 meets the requirements of the following regulations:

10 CFR 50, Appendix A, GDC 1 and GDC 30, as they relate to quality standards for design, fabrication, erection, and testing, by compliance with ASME BPVC, Section III and conformance with RG 1.31, RG 1.50, RG 1.71, Generic Letter 88-01, and Nuclear Regulatory Report (NUREG)-0313, Revision 2. RG 1.34 does not apply as detailed in Subsection 5.2.3.3.2. RG 1.43 does not apply as detailed in Subsections 5.2.3.3.2 and 5.3.1.4.

10 CFR 50, Appendix A, GDC 4, as it relates to compatibility of components with environmental conditions, by conformance with RG 1.44, Generic Letter 88-01, and NUREG-0313, Revision 2

10 CFR 50, Appendix A, GDC 14 and GDC 31, as they relate to extremely low probability of rapidly propagating fracture and gross rupture of the RCPB, by conformance with RG 1.31, RG 1.36, RG 1.44, RG 1.71, Generic Letter 88-01, NUREG-0313, Revision 2, and compliance with ASME BPVC, Section III, Nonmandatory Appendix G.

10 CFR 50, Appendix B, Criterion XIII, as it relates to onsite material cleaning control, as detailed in Subsection 5.3.3.4.

10 CFR 50, Appendix G, as it relates to materials testing and acceptance criteria for fracture toughness of the RCPB as detailed in Subsection 5.2.3.3.1

10 CFR 50.55a, as it relates to quality standards applicable to the RCPB, by compliance with ASME BPVC, Section III and conformance with RG 1.31, RG 1.50, RG 1.71, Generic Letter 88-01, and NUREG-0313, Revision 2. RG 1.43 does not apply as detailed in Subsections 5.2.3.3.2 and 5.3.1.4.

Section 3.1 summarizes compliance with GDC.

5.2.3.1 Material Specifications

This subsection discusses the specifications for pressure-retaining ferritic materials, austenitic stainless steels, and nonferrous metals, including bolting and weld materials, that are used for each component (e.g., vessels, piping, pumps, and valves) of the RCPB. The adequacy and suitability of the ferritic materials, stainless steels, and nonferrous metals specified for the above application are discussed in Subsection 5.2.3.1 through Subsection 5.2.3.4.

Table 5.2-2a and Table 5.2-2b list the principal pressure-retaining materials and the appropriate material specifications for the RCPB components and welds. RCPB materials comply with ASME BPVC, Section III (Reference 5.2-4) (NB-2000 for ASME Class 1 components and NCD-2000 for ASME Class 2 components) and ASME BPVC, Section II, “Materials,” (Reference 5.2-3). RCPB ferritic steel materials comply with 10 CFR 50, Appendix G. Compliance with RG 1.84, which addresses ASME BPVC, Section III Code Cases, is discussed in Subsection 5.2.1.2, and Code Cases, if used, are listed Table 5.2-1. No exceptions are taken to the material specifications of the ASME BPVC.

BWR piping materials and materials processing conforms to Attachment A to Generic Letter 88-01, “NRC Position on IGSCC in BWR Austenitic Stainless-Steel Piping” and consider the recommendations in NUREG-0313, Revision 2, “Technical Report on Material Selection and Processing Guidelines for Boiling Water Reactor (BWR) Coolant Pressure

Boundary Piping" related to minimizing Stress Corrosion Cracking (SCC) in susceptible piping. Compliance with Attachment A to Generic Letter 88-01 includes carbon content and ferrite limits for stainless steel base and weld metals, and nickel base alloy material selection. Further discussion of the materials associated with the pressure retaining components of the RPV is provided in Subsection 5.3.1.

Cast austenitic stainless steels used in the RCPB have material specifications such as maximum ferrite levels, that ensure adequate fracture toughness over the design life to support use of the material under the expected environmental conditions (e.g., exposure to the reactor coolant operating temperatures).

Nickel-chromium-iron alloys such as Alloy 600M consistent with Code Case N-580-2 along with Alloy 82 weld metal are used in the RCPB. Alloy 600 and associated weld materials are not used in the RCPB, due to their susceptibility to SCC.

RPV and RPV Appurtenances

Table 5.2-2a also includes materials and specifications for the RPV and RPV appurtenances which form part of the RCPB. This includes RPV nozzles, stub tubes, CRD flanges/housings, and incore instrumentation housings and their associated RCPB components.

RPV Attachments

Table 5.2-3 lists materials for selected RPV attachments. RPV attachments are those components welded directly to the RPV but do not form parts of the RCPB. Many of these attachments are internal to the RPV and in contact with reactor coolant, while some are external to the RPV and not in contact with reactor coolant.

RPV attachments internal to the RPV and in contact with reactor coolant include the core support legs.

RPV attachments external to the RPV and not in contact with reactor coolant include the vessel stabilizer brackets and the vessel support skirt.

RPV Internals

Materials for RPV internals, such as steam dryer, chimney, and core support structure, are provided in Table 4.5-1. Discussion of RPV internals in this chapter is limited to their compatibility with reactor coolant and certain corrosion aspects or effects on water chemistry.

5.2.3.2 Compatibility with Reactor Coolant

General corrosion and SCC induced by impurities in the reactor coolant can cause failures of the RCPB. The chemistry of the reactor coolant and any additives whose function is to control corrosion are discussed in Subsection 5.2.3.2.1 and Subsections 9.3.7, 9.3.8, and 9.3.9. The compatibility of the materials of construction employed in the RCPB with the reactor coolant, contaminants, or radiolytic products to which the system is exposed has been considered by employing proven materials and processes. The extent of time-dependent general corrosion and other degradation mechanisms of materials of construction in contact with the reactor coolant has been considered. Similarly, consideration has been given to the control of the use of austenitic stainless steels in the sensitized condition.

5.2.3.2.1 Chemistry of Reactor Coolant

Proper material selection, fabrication and process controls, and control of water chemistry ensures the integrity of the RCPB. This section provides a brief review of the relationships between water chemistry variables and RCPB materials performance, fuel performance, and plant radiation fields. Electric Power Research Institute (EPRI) guidelines in EPRI 3002002623,

BWRVIP-190, “BWR Vessel and Internals Project, Volume 1: BWR Water Chemistry Guidelines – Mandatory, Needed and Good Practice Guidance and Volume 2: BWR Water Chemistry Guidelines – Technical Basis,” (Reference 5.2-1) are used as inputs to develop the basis for the BWRX-300 water chemistry program. The BWRX-300 water chemistry program may continue to be revised based on new EPRI guidelines, and as operational experience is gained.

The BWR uses high purity water as the fluid to transfer thermal energy. The quality of water to which the component is exposed, to a large extent, determines the reliability and lifetime of that component. The extent of materials corrosion from its environment is a function of the impurity and its concentration, temperature, and duration. Reliable maintenance of water quality is necessary to achieve the desired overall plant reliability and availability.

The major environment-related materials performance problem encountered to date in the RCPB of BWRs has been Intergranular Stress Corrosion Cracking (IGSCC) of sensitized austenitic stainless steel. Substantial research and development programs have been undertaken to understand the IGSCC phenomenon and develop remedial measures. As a result, many important features of the relationship between BWR water chemistry and IGSCC of sensitized stainless steels have been identified. Although operating within the BWR water chemistry guidelines for reactor water cannot prevent IGSCC, maintaining the lowest achievable impurity levels minimize the rate of progression. To aid in maintaining water quality, the BWRX-300 is designed to cascade Feedwater Heater (FWH) drains to the hotwell ensuring FW is processed through the Condensate Filter and Demineralizers System (CFD) while also minimizing insoluble metals such as iron. Furthermore, the BWRX-300 condensate demineralizer design of mixed bed demineralizers with upstream non-precoated filters, has a high efficiency for the removal of particulate and ionic impurities.

Materials of construction and their compatibility with reactor coolant is further addressed in Subsection 5.2.3.2.2.

Water Chemistry Regime

The chemistry regime at a nuclear power plant controls the reactor water chemistry in order to minimize SCC and radiation buildup in the plant’s austenitic stainless steel materials. BWRs were originally designed to operate with the chemistry controlled by the CFS and CUW without chemical additives. In the decades of Operating Experience (OPEX) since then, advancements have been made in understanding and improving water chemistry effects by controlling the chemistry via chemical additions. The planned BWRX-300 Chemistry Regime includes the following:

- Hydrogen Water Chemistry (HWC)
- On-Line NobleChem™ (OLNC)
- Zinc Injection using Depleted Zinc Oxide

Hydrogen Water Chemistry

Incore radiation from fuel bundles causes radiolysis, which breaks down the water molecules into (mainly) hydrogen, oxygen, and some peroxide. This creates an oxidizing environment which contributes to IGSCC crack initiation and crack propagation growth rates in austenitic stainless steel piping and internals. The addition of excess hydrogen to the reactor water promotes the recombination of hydrogen and oxygen, as well as peroxide suppression. Excess hydrogen is added to the reactor water in order for the recombination reaction to be effective and produce a more reducing environment that can reduce crack initiation and crack growth rates. The Hydrogen Water Chemistry Injection System is discussed in Subsection 9.3.7.

Noble Metal Technology

OLNC is the injection of noble metals developed by GVH as a process to add catalyst to reduce the amount of bulk water hydrogen needed to achieve mitigation of IGSCC. While mitigation can be achieved with hydrogen injection alone, the bulk water hydrogen concentration required for mitigation is high enough to cause MSL radiation monitor activity to increase sufficiently to warrant the injection of noble metals. Once the reactor internal surfaces have been coated with noble metals, the hydrogen/oxygen recombination reaction on internal surfaces is catalyzed, requiring a much lower bulk hydrogen concentration to maintain a similar level of IGSCC mitigation. Lower hydrogen concentrations therefore result in significantly reduced MSL radiation monitor activity. The On-Line NobleChem™ system is discussed in Subsection 9.3.8.

Zinc Injection

The largest contributor to radiation buildup on external piping and other out of core surfaces is from Co-60 in reactor water during power operation. Co-60 is responsible for >90% of the personnel occupational radiation exposure incurred during refueling outages. Zinc addition reduces shutdown dose rates by competing with cobalt for adsorption sites on the fuel and other surfaces, as well as stabilizing fuel oxide layers. A high concentration of soluble zinc can overwhelm soluble cobalt, giving cobalt less chance to incorporate into fuel crud where it is likely to be activated by exposure to neutron radiation and become Co-60. The presence of zinc in the fuel deposits as zinc ferrite promotes stability of the deposits and helps to retain Co-60 on the fuel, as opposed to its release to the water to later deposit on out of core surfaces. The Zinc Injection Passivation System is discussed in Subsection 9.3.9.

Water Chemistry Maintenance

A sampling and monitoring plan is a critical factor of maintaining water chemistry at optimal conditions and having the ability to respond to off-normal chemistry in a timely manner. The plant specific BWRX-300 water chemistry program is to be designed to analyze and monitor system chemistry for trending with alarm notification so actions can be taken to stay within operating specifications. This supports minimizing corrosion from chemical contaminants and monitoring chemical additives used to limit corrosion and radiation buildup.

BWR water chemistry control parameters, recommended operating limits, and recommended monitoring frequencies are to be developed to minimize the potential for IGSCC by controlling both ionic impurity and oxidizing radiolysis product concentrations in the reactor water. EPRI guidelines in Reference 5.2-1 are used as input to develop the basis for the BWRX-300 water chemistry program.

Sources of Impurities and Related Factors

The following sources of impurities and related factors are typically taken into consideration in BWR design and chemistry programs consistent with EPRI guidelines. Some of these factors may be adopted in the plant specific BWRX-300 water chemistry program and measured for control or diagnostic purposes to maintain proper water chemistry in the BWRX-300:

- Conductivity - general overall indictor of water purity
- Chloride - an aggressive contaminant
- Fluoride - a contaminant generally from welding
- Sulfate - an aggressive contaminant
- Silica (SiO_2) - more of a concern on fuel cladding
- Dissolved Oxygen - reduced with addition of hydrogen and noble metals

- Dissolved Hydrogen - indicates if sufficient hydrogen is added for mitigation of IGSCC
- Electrochemical Corrosion Potential - indicator for mitigation of IGSCC
- Organic Impurities - can mask some contaminants until they get in reactor water
- Corrosion Product Metals - from corrosion of piping and components
- Copper - concern for fuel and generally now eliminated from materials of construction
- Co-60 - largest source of gamma dose to workers
- Zinc - added to reduce buildup of Co-60
- Boron and Lithium - indicators of control blade leakage

Table 5.2-4 provides information on special purpose materials, that are not part of the RCPB but exposed to reactor coolant.

Typical Water Chemistry Control Parameters

Table 5.2-5 provides typical control parameters for BWRX-300 water chemistry under normal operation. The water chemistry program also implements diagnostic and good practice targets for water chemistry optimization.

Flow Accelerated Corrosion Considerations

General corrosion of plant components is managed in the design process based on the ASME BPVC which sets minimum corrosion allowances. However, under certain conditions of water chemistry, system flow, temperature, geometry, and component material, Flow Accelerated Corrosion (FAC) can degrade piping and components made of carbon steel (with low chromium composition) piping systems.

The most direct approach to assure mitigation against FAC is to employ stainless steel which has no risk of FAC. Alternatively, higher alloyed carbon steels (Cr-Mo steels with higher Cr) such as 2 ¼ Cr-1 Mo, can be used for piping lines that continuously meet the five (5) of the risk factors for FAC: 1) low temperature, 2) susceptible material composition, 3) turbulent flow, 4) pH <8, and 5) low oxygen content.

Components exposed to steam or condensate require materials resistant to FAC or erosion due to low oxygen content (<30 ppb) and lower temperatures (in the range of 100°C to 150°C). For the BWRX-300, stainless steel is used for the FW piping system to preclude any risk of FAC. Although not a part of the RCPB, stainless steel is to also be used for the condensate piping upstream of the FW piping. Other components, such as the MS piping, do not require higher chromium carbon steels to mitigate the risk of FAC because the temperature in the MS piping is very seldom in the range of 100°C to 150°C.

Irradiation Assisted Stress Corrosion Cracking (IASCC) Considerations

While SCC can occur in many locations, the SCC process is also facilitated by irradiation, identified as IASCC.

The mechanism of IASCC is associated with core components and typically involves annealed material that are subjected to irradiation. The exposure to irradiation leads to localized compositional changes at the material's grain boundaries. These changes can then lead to SCC at lower stress levels in the presence of high temperature oxygenated water with short lived oxidizing species, gamma, and neutron flux. IASCC does not require initial chromium depletion sensitization or high tensile stresses to produce failures.

IASCC is associated with time dependent degradation that occurs as austenitic stainless-steel material in the core region experiences material changes due to the effects of radiation. These effects include material hardening as well as radiation-induced segregation. It is accepted that the threshold for IASCC degradation occurs when components are subjected to a combination of cumulative fluence greater than $5x10^{20}$ n/cm^2 for E>1 MeV with applied stress levels in the component(s), and as fluence increases, the stress to initiate IASCC decreases.

In addition to material selection, fabrication, process controls and water chemistry controls, IASCC mitigation for the BWRX-300 includes the locating of welds away from the highest fluence regions.

5.2.3.2.2 Compatibility of Construction Materials with Reactor Coolant

It is the BWRX-300 design philosophy that the materials used maintain reliable operation of plant systems and equipment during the design life of each component.

Carbon Steel and Low-alloy Steel

In general, materials used for certain piping system components, the RPV (including the shell and top and bottom heads), and vessel attachments including nozzles and bolting are low-alloy steels and carbon steels.

The interior surfaces of the RPV are clad with stainless steel weld overlay, except for the closure head. The MS nozzles are clad with stainless steel weld overlay. The bottom head is clad with Ni-Cr-Fe alloy. The RPV closure sealing surfaces are clad with Ni-Cr-Fe alloy or stainless-steel overlay.

Stainless Steel and Nickel Based Alloys

In general, materials used for certain piping, reactor internals, and instrumentation and hardware are austenitic stainless steel and nickel-based alloys.

For the reactor internals, austenitic stainless steel is to be primarily used to minimize levels of radiation from corrosion products generated incore. Where higher strength requirements than those for austenitic stainless steels are imposed, nickel base alloys, nitrogen strengthened alloys, or precipitation hardening stainless steels are to be used.

The stainless-steel structural materials used in the BWRX-300 reactor core internals are primarily wrought materials along with their weld metals, although cast austenitic stainless steels are used as well.

In summary, construction materials of the RCPB and reactor internals exposed to the reactor coolant consist of the following:

- Solution-annealed austenitic stainless steels (both wrought and cast), Types 304, 304L, 316, 316L, CF3M, CF8M, and XM-19
- Solution-annealed tempered martensitic stainless steels, Type 410
- Nickel-based alloy (including niobium modified Alloy 600M with filler metal Alloy 82, Alloy 625 with filler metal ERNICMO-3, Alloy 718, and X-750)
- Precipitation Hardening stainless steels such as Grade 630 and Grade XM-13
- Carbon steel and low-alloy steel

Table 5.2-2a lists materials and material specifications for RCPB components. Table 4.5-1 lists material and material specifications for reactor internals.

Suitability of Materials of Construction Exposed to Reactor Coolant

Materials selection and processing controls address materials degradation issues in the RCS such as SCC, general corrosion, and FAC. The specific categories of materials that are to be used are wrought austenitic stainless steels, cast austenitic stainless steels, martensitic stainless steels, precipitation hardening stainless steels, and Ni-base alloys along with their weld metals used as part of construction. General corrosion for the plant components is managed in the design process based on the ASME BPVC, which sets minimum corrosion allowances.

Enhanced IGSCC resistance has been achieved through the use of IGSCC resistant materials such as Type 316L with controlled composition stainless steel, XM-19, and stabilized nickel-base niobium modified Alloy 600M and Alloy 82.

BWRX-300 materials provide adequate strength, fracture toughness, fabricability, and compatibility with the BWR environment. Their suitability has been demonstrated by long-term successful OPEX in reactor service.

Material Contamination and Transgranular Stress Corrosion Cracking

The requirements of GDC 4 relative to compatibility of components with environmental conditions are met by compliance with the applicable provisions of the ASME BPVC such as corrosion allowances and by conformance with the recommendations of RG 1.44, “Control of the Processing and Use of Stainless Steel.” Material contamination that can lead to transgranular stress corrosion cracking and process controls to minimize exposure to contaminants during fabrication and construction are discussed in Subsection 5.2.3.4.1.

Contaminants in the reactor coolant are controlled to very low limits and in conformance with RG 1.44. No detrimental effects occur on any of the materials from allowable contaminant levels in the high purity reactor coolant.

5.2.3.2.3 Compatibility of Construction Materials with External Insulation

RG 1.36, “Nonmetallic Thermal Insulation for Austenitic Stainless Steel.”

Nonmetallic insulation that is applied to austenitic stainless steel RCPB components conforms to RG 1.36 to minimize contamination that could promote stress-corrosion cracking.

5.2.3.3 Fabrication and Processing of Ferritic Materials

This subsection discusses fracture toughness properties of ferritic materials used for pressure retaining components of the RCPB, the control of welding in ferritic steels, and the requirements and methods for Non-Destructive Examination (NDE) of ferritic wrought seamless tubular products used for ASME Class 1 and ASME Class 2 components of nuclear power plants as specified in the ASME BPVC.

5.2.3.3.1 Fracture Toughness

ASME Class 1 and Class 2 RCPB carbon steel components are made from high toughness grade material. In addition, ferritic components comply with ASME BPVC, Section III requirements in accordance with the following:

- The ferritic materials used for piping and valves of the RCPB are usually 64 mm (2.5 in.) or less in thickness. Impact testing of ASME Class 1 components is performed in accordance with ASME BPVC, Section III, NB-2332 for thicknesses of 64 mm (2.5 in.) or less. Impact testing of ASME Class 1 components is performed in accordance with NB-2332 and NB-2331 for thickness greater than 64 mm (2.5 in.). Impact testing of ASME Class 2 components is performed in accordance with NCD-2330. In addition, ASME BPVC, Section III, Nonmandatory Appendix G, Paragraph G-3100 is met.

- Materials for ASME Class 1 and ASME Class 2 bolting with nominal diameters exceeding 25 mm (1 in.) are compliant with NB-2333 and NCD-2332.3, respectively, and are required to meet both the lateral expansion and the Charpy V values specified in Table 5.3-1).
- The reactor vessel complies with the requirements of NB-2331 and 10 CFR 50, Appendix G. The reference temperature is established for required pressure-retaining materials used in the construction of Class 1 vessels. This includes plates, forgings, and weld material. The reference temperature differs from the nil ductility temperature in that, in addition to passing the drop weight test, three Charpy V-Notch specimens (transverse) must exhibit 68 J (50 ft-lbf) absorbed upper-shelf energy and 0.89 mm (0.035 in.) lateral expansion at 33°C (91°F) above the reference temperature. The core beltline material must meet 102 J (75 ft-lbf) absorbed upper-shelf energy. Consideration has been given to the effects of irradiation on beltline fracture toughness by controlling the chemical composition residual elements of vessel beltline materials as specified in Table 5.3-1.
- Calibration of instruments and equipment meets the requirements of the ASME BPVC, Section III, NB/NCD-2360.

Fracture toughness tests will be performed for the BWRX-300 RCPB, and the test results are to be documented as required including description of the tests, locations of the test specimens and their orientation along with information on calibration of instruments and equipment.

5.2.3.3.2 Control of Welding

General Welding Controls

Welding of ferritic steels is conducted using procedures qualified according to the rules of Sections III and IX of the ASME BPVC. Control of welding variables during procedure qualification and production welding is performed according to ASME BPVC requirements.

Post-Weld Heat Treatment (PWHT) of low-alloy steel welds and their time/temperature are discussed in Subsection 5.3.1.2 and Table 5.3-1.

Avoidance of Hot Cracking

The following RGs have specific applicability to welding of ferritic materials:

RG 1.50, “Control of Preheat Temperature for Welding of Low-Alloy Steel.”

The BWRX-300 design conforms with RG 1.50. RG 1.50 delineates preheat and interpass temperature control requirements and welding procedure qualifications supplementing those in ASME BPVC, Sections III and IX.

For the BWRX-300, preheat and interpass temperatures employed for welding of low-alloy steel are to meet ASME BPVC, Section III (Subsection NB for Class 1 components and Subsection NCD for Class 2 components) and Non-Mandatory Appendix D, Article D-1000; and ASME BPVC, Section IX, “Qualification Standard for Welding, Brazing, and Fusing Procedures; Welders, Brazers; and Welding, Brazing, and Fusing Operators,” (Reference 5.2-5) as supplemented by RG 1.50.

RG 1.34, “Control of Electroslag Weld Properties”

Requirements of RG 1.34 are not applicable to the BWRX-300 RCPB or RPV because electroslag welding is not employed in structural welds of low-alloy steel. Electroslag welding is not used except for RPV cladding.

RG 1.43, “Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components”

RG 1.43 does not apply as discussed in Subsection 5.3.1.4. Stainless steel weld cladding is not used except for RPV cladding.

RG 1.71, “Welder Qualification for Areas of Limited Accessibility”

Welder qualification for areas of limited accessibility is discussed in Subsection 5.2.3.4.2.

Moisture Control for Low Hydrogen, Covered Arc Welding Electrodes

Suitable identification, storage, and handling of electrodes, flux, and other welding material are maintained. Precautions are taken to minimize absorption of moisture by electrodes and flux.

5.2.3.3.3 Nondestructive Examination of Tubular Products

Wrought tubular products that are used for pressure-retaining components of the RCPB are subject to the examination requirements of ASME BPVC, Section III, Subsection NB for Class 1 components and Subsection NCD for Class 2 components.

These RCPB components meet 10 CFR 50, Appendix B requirements (see process controls information in Subsection 5.2.3.4) and the ASME BPVC requirements, thus assuring adequate control of quality for the products.

5.2.3.4 Fabrication and Processing of Austenitic Stainless Steels

Austenitic stainless steels in a variety of product forms are used for construction of a limited number of pressure-retaining components in the RCPB. Process controls are exercised during various stages of component manufacturing and reactor construction to avoid severe sensitization of the material and to minimize exposure of the stainless steel to contaminants that could lead to SCC.

5.2.3.4.1 Avoidance of Stress Corrosion Cracking

Avoidance of Significant Sensitization

When austenitic stainless steels are subjected to the temperature range 427°C - 816°C (800°F - 1500°F), they are considered to become “sensitized” or susceptible to intergranular corrosion. The BWRX-300 design conforms with RG 1.44, “Control of the Processing and Use of Stainless Steel,” and with the guidelines of Generic Letter 88-01 and NUREG-0313, Revision 2, to avoid sensitization through the use of reduced carbon content and process controls.

All wrought austenitic stainless steels are supplied in the solution heat treated condition and special sensitization tests are applied when wetted by water above 93°C (200°F) to confirm proper heat treatment. For applications where stainless-steel surfaces are exposed to reactor water at temperatures above 93°C (200°F) in welded applications where solution heat treatment is not performed, L-grade materials with carbon content ≤ 0.02% are used.

During fabrication, heating operations (except welding) above 424°C (~800°F) are avoided, unless followed by solution heat treatment. During welding, heat input is controlled. The interpass temperature is also controlled. Where practical, shop welds are solution heat treated.

Process Controls to Minimize Exposure to Contaminants

Controls are implemented by limiting contaminant levels of elements (such as halogens, sulfur, lead) to as low as possible in miscellaneous materials used during fabrication and installation. These materials (such as tapes and penetrants) are completely removed and cleanliness is assured. Lubricant and gasket materials that remain in contact with the coolant during operation are evaluated on that basis.

While SCC generally proceeds through a material intergranularly, chloride contamination, surface cold work, or localized straining can lead to transgranular crack initiation which is designated as transgranular stress corrosion cracking.

IGSCC and transgranular stress corrosion cracking often occur in the same alloys depending on the environment or the microstructure. However, cases of transgranular stress corrosion cracking in small lines and other components have been associated with contaminants such as chlorides that can be present on the material's surface. These contaminants, related to salts or marking materials, are to be controlled during fabrication as well as operation.

Miscellaneous non-metallic materials, used for the fabrication and assembly of RCS components, are also controlled to minimize contamination with species known to have detrimental effects on alloys such as stainless steels and nickel alloys.

Contaminant materials, which come in contact with stainless steel or nickel alloys during these processes, either need to be low in halogens, sulfur, and lead or other low melting alloys or need to be removed in the final cleaning of the part or assembly. The known contaminants of concern include chlorides, fluorides, sulfur, lead, mercury, and other low melting point metals. In addition, when welding or solution heat treatment is involved, phosphates and carbonaceous materials are to be removed prior to welding or heat treatment.

Process controls are exercised during various stages of component manufacturing and reactor construction to avoid sensitization of the material, minimize fabrication-induced stresses and to minimize exposure of the stainless steel to contaminants that could lead to SCCs. There are controls of abrasive media to avoid cold work and contamination and review and control of pickling processes when employed. Pickling of fabricated stainless steels is avoided or performed only with appropriate controls. Pickling of sensitized austenitic stainless is not allowed.

Special care is exercised to ensure removal of surface contaminants prior to heating operations. Water quality for cleaning, rinsing, flushing, and testing is controlled and monitored. Suitable protective packaging is provided for components to maintain cleanliness during shipping and storage.

The degree of surface cleanliness obtained by these procedures meets the requirements of RG 1.44. Cleaning of RCPB components, including austenitic stainless steel components, complies with the GVH quality assurance program..

Cold-Worked Austenitic Stainless Steels

Cold work controls are applicable to the fabrication process of RCPB components made of austenitic stainless steel to minimize or avoid fabrication-induced surface stresses that could lead to SCCs. The cold work controls for the RCPB conform to RG 1.44. The processes that need to be controlled include bending or forming, final machining, grinding, and polishing.

5.2.3.4.2 Control of Welding

General Welding Controls

Welding of stainless steels is conducted using procedures qualified according to the rules of Sections III and IX of the ASME BPVC. Control of welding variables during procedure qualification and production welding is performed according to ASME BPVC requirements.

Control of welding of RCPB components conforms to the requirements of RG 1.44 to avoid severe sensitization that could lead to SCCs.

Avoidance of Hot Cracking

The following RGs have specific applicability to welding of austenitic stainless steels:

RG 1.31, “Control of Ferrite Content in Stainless Steel Weld Metal”

The BWRX-300 design conforms with RG 1.31. RG 1.31 describes the acceptable method of implementing requirements with regard to the control of welding when fabricating and joining austenitic stainless-steel components and systems.

Written welding procedures are required for RCPB welds performed for material fabrication and plant construction. These procedures are to comply with the requirements of Sections III and IX of the ASME BPVC and applicable USNRC RGs.

RCPB austenitic stainless steel weld filler materials are required by specification to have an average minimum delta ferrite content of 8 FN (ferrite number) and a maximum of 20 FN for 308L and 16 FN for 316L as determined on undiluted weld pads by magnetic measuring instruments calibrated in accordance with American National Standards Institute (ANSI)/American Welding Society (AWS) A4.2M/A4.2, “Standard Procedures for Calibrating Magnetic Instruments to Measure the Delta Ferrite Content of Austenitic and Duplex Ferritic-Austenitic Stainless Steel Weld Metal,” (Reference 5.2-7).

RG 1.34, “Control of Electroslag Weld Properties”

RG 1.34 is addressed in Subsection 5.2.3.3.2.

RG 1.71, “Welder Qualification for Areas of Limited Accessibility”

RG 1.71 requires that weld fabrication and weld repair comply with fabrication requirements of Sections III and IX of the ASME BPVC. It also requires additional performance qualification and requalification for welding in areas of limited access.

ASME BPVC, Section III welds are fabricated in accordance with the requirements of Sections III and IX of the ASME BPVC. The BWRX-300 is to meet the intent of RG 1.71 directly or by an alternate approach as described below:

Simulated access qualification testing of welders is required when access to a non-volumetrically examined production weld, (1) is less than 300 mm (12 inches) in any direction, and (2) allows welding from one access direction only. Such welds and weld repairs shall comply with the fabrication requirements specified in the ASME BPV Code, Section III, and the requirements of Section IX (invoked by Section III) supplemented by the following:

- The welder performance qualification test assembly shall be welded under simulated access conditions
- If bend tests are required, a specimen taken from the location in the qualification piece least favorable to the welder shall be included. If radiography is used to judge the qualification piece, it shall include the location least favorable to the welder.
- A welder shall be requalified where access to the production weld is less than the access for which the welder has been qualified

5.2.3.4.3 Nondestructive Examination of Tubular Products

For discussion of nondestructive examination of tubular products, refer to Subsection 5.2.3.3.3.

5.2.3.4.4 Thermal Aging of Cast Austenitic Stainless-Steel

The fabrication controls for cast austenitic stainless-steel components, in conjunction with acceptable base material and weld metal specifications, provide for sound weld joints and adequate fracture toughness to resist thermal aging for the design life.

The limitation placed upon the delta ferrite in austenitic stainless-steel castings is 8% minimum and a maximum value of 20%. The maximum limit is used for those castings designed for a 60-year life to limit the effects of thermal aging degradation.

The weld filler material used for austenitic stainless steel base metals is Type 308L/316L/309L with an average ferrite content not less than 8 FN (ferrite number). See Subsection 5.2.3.4.2 for information on control of welding.

5.2.4 Reactor Coolant Pressure Boundary Inservice Inspection and Testing

The BWRX-300 meets the relevant requirements of the following regulations:

10 CFR 50, Appendix A, GDC 32, as it relates to periodic inspection and testing of the RCPB. BWRX-300 is designed for the performance of preservice inspection and ISI of ASME BPVC Class 1 and ASME Class 2 components within the RCPB in accordance with ASME BPVC, Section XI pursuant to 10 CFR 50.55a(g). The edition of ASME BPVC, Section XI is provided in Subsection 5.2.1.1.

10 CFR 50.55a, as it relates to the requirements for inspecting and testing ASME BPVC Class 1 components of the RCPB as specified in ASME BPVC, Section XI. The BWRX-300 ISI program complies with the rules of 10 CFR 50.55a as indicated in Subsection 5.2.1.1 and Subsection 5.2.1.2.

The RCPB consists of ASME Class 1, and ASME Class 2 components as described in Subsection 5.2.1.1 and Figure 5.1-2. BWRX-300 RCPB components which require inspections and testing to satisfy ASME BPVC, Section XI, are examined by appropriate inspection and testing methods provided in ASME BPVC, Section XI. Examination categories, methods, and procedures for RCPB components are in accordance with ASME BPVC, Section XI, Subsection IWA, and Subsection IWB. Examination results are evaluated and accepted in accordance with ASME BPVC, Section XI, Subsection IWA, and Subsection IWB.

RCPB components are designed and provided with the access necessary to perform the required preservice and inservice inspections set forth in ASME BPVC, Section III and Section XI, including the examinations required by ASME BPVC, Section XI. The following are considered in the inspectability of the design:

- Preservice Inspection (PSI) for Class 1 RCPB are specified in the design specification and are in accordance with ASME BPVC, Section III, NB-5281 and NB-5282 and ASME BPVC, Section XI, Table IWB-2500-1, and will be implemented during construction
- Additional requirements such as NUREG-0313, Revision 2, and Generic Letter 88-01 that may include additional inspections
- Any augmented PSI and ISI for Class 1 RCPB welds, e.g., augmented inspections in Subsection 3.6.1.1.3
- The ability to perform system hydrostatic tests and system leakage tests.

RCPB components and their supports are designed and provided with the access necessary to perform the required preservice and inservice examinations set forth in ASME BPVC, Section III and Section XI, e.g., IWA-2200(b) for weld surface preparation for NDE, or the optional ASME

Code Cases listed in RG 1.147. Adequate clearance is provided in accordance with ASME BPVC, Section XI, IWA-1500 by the design and arrangement of system components.

RCPB components and their supports meet the preservice examination requirements set forth in ASME BPVC, Section III and Section XI or the optional ASME Code Cases listed in RG 1.147, applied to the construction of the particular component.

The implementation schedules in Table 13.4-1, are applicable to the Preservice Inspection Program and Inservice Inspection Program.

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection

RCPB leakage detection systems are designed to provide a means of detecting and, to the extent practical, identifying the source of the reactor coolant leakage. Although Licensing Topical Report (LTR) NEDC-33910P-A, "BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection," (Reference 5.2-2) describes conformance to RG 1.45, the BWRX-300 design includes only unidentified leakage that is potentially from the RCS. As identified leakage is not defined, there is no leakage separation. These are alternatives to RG 1.45 General Positions 1.1 and 1.2.

The RCPB leakage detection system detects leakage after an earthquake for an early indication of degradation so that corrective action can be taken before such degradation becomes severe enough to result in a leakage rate greater than the capability of the makeup system to replenish the coolant loss.

Historically, separation of identified leakage has been accomplished by routing leakage from known sources, such as the recirculation pump seal leakage, to an equipment drain system instrumented to quantify the total identified leak rate.

The BWRX-300 is a natural circulation design and has no pumps within the RCPB. Eliminating pump seal leakage significantly reduces anticipated leakage which could be identified. Additionally, the BWRX-300 does not make use of traditional safety or relief valves within the RCPB.

The design simplification achieved with the BWRX-300 has eliminated many historical sources of identified reactor coolant leakage such that total leakage is reduced. As a result, it is unnecessary to separate identifiable leakage to maintain a low threshold for unidentifiable leakage as was required with previous reactor designs. Instead, leakage within containment is routed to the containment sump and measured as unidentified.

To limit the potential for a gross failure, leakage monitoring systems are designed to detect RCPB degradation and to assist in locating the source of leakage. Typical leakage limits for BWRs are 5 gpm unidentified and 25 gpm total identified leakage. Studies of the failure behavior of reactor coolant systems contained in NUREG/CR-6861, "Barrier Integrity Research Program," show that for many piping systems, these limits provide significant margin against gross failure of reactor piping to sustained stress loads. The report concludes that these Technical Specification (TS) limits on unidentified leakage can be expected to provide significant margin against structural failure of piping systems.

The plant variables monitored for leakage/break detection are listed in Table 5.2-6.

Based on the discussion above, an acceptable alternative to RG 1.45 General Positions 1.1 and 1.2 is to monitor only unidentified leakage without leakage separation. The design of the RCPB leakage detection systems considers specific general design criteria and RGs. The integrated design of the RCPB leakage detection system is in accordance with the following criteria with identified alternatives:

GDC 2 as it relates to the capability of the systems to maintain and perform their safety functions following an earthquake, and meets the guidelines of RG 1.29, positions C.1 and C.2.

GDC 30 as it relates to the detection, identification and monitoring of the source of reactor coolant leakage, and meets the guidelines of RG 1.45, positions with alternatives to General Positions 1.1 and 1.2.

5.2.5.1 Leakage Detection Methods

The leakage detection methods employed inside, and outside containment are discussed and described separately in the following sections.

5.2.5.1.1 Detection of Leakage Within Containment

The primary methods used for monitoring small unidentified leakage are:

- Containment sump pump use
- Containment sump level and rate-of-rise
- Containment cooler condensate flow rate and increase
- Increase in radioactive fission product (noble gas and particulate) activity

These parameters are continuously monitored and/or recorded in the Main Control Room (MCR). Instrumentation for leakage rate monitoring is designed to detect changes in the containment leakage rate of 3.8 L/min increase within the previous one hour.

The containment sump collects unidentified leakage from the floor drain and condensate from containment atmosphere coolers. Excessive leakage is designed to be detected by the containment sump level changes, and frequency and duration of the pump runs.

The leak detection logic runs continuously to monitor leakage within the containment during normal plant operation, and during abnormal and accident conditions. The containment sump isolates on a containment isolation signal and is restored to service under operator control when the signal is clear. Leak detection measurements and trip actions are provided in the MCR for display, recording, and retrieval.

Leaks from line breaks in ICS piping upstream of the condensate return valves are detected by high flow rate in the ICS steam supply line, and downstream of the condensate return valves by high flow rate in the ICS condensate return line. ICS isolation is described in Section 6.3.

Gross unidentified leakage is detected by high containment pressure or temperature, low reactor pressure, low reactor water level, or high flow rate in steam piping (or breaks downstream of the MS Flow Elements (FEs).

To prevent leakage within containment, the reactor vessel head flange inner O-ring seal drain line pressure is monitored to detect failure of the inboard seal.

The plant variables monitored for leakage detection within containment are listed in Table 5.2-6.

5.2.5.1.2 Detection of Leakage External to Containment

The areas outside containment monitored for leakage are:

- The equipment areas in the RB
- The MS Tunnel area
- The Turbine Building (TB)
- The Radwaste Building (RWB)

Leaks from line breaks are detected by monitoring the differential mass flow rate in CUW and SDC. The differential flow rate for CUW is measured in the supply lines from the vessel and discharge or overboard lines. The differential flow rate in the SDC is measured in the supply line that is connected to the ICS and the return or overboard lines, such that a SDC leak/break can be detected and isolated as needed. The ICS condensate return flow element is located to optimize the ability to separately sense and isolate leaks in the SDC and the ICS. SDC and CUW isolation are described in Subsections 5.4.7 and 5.4.8, respectively.

The ambient temperatures in the MSL tunnel area, SDC area, CUW area, and FW heater area are monitored for indications of steam leakage. The steam tunnel is also monitored for area differential temperature. For room temperatures, the ambient temperature sensors are located near the ceiling, away from process piping and ventilation. Temperature elements used for differential temperature monitoring are located as near as possible to the area inlet and outlet ventilation flows.

5.2.5.2 Leakage Detection Instrumentation and Monitoring

The leaks requiring detection range from a double-ended shear of the largest diameter piping, typically the MSL piping, to a small crack in any of the process piping systems.

The instrumentation detects leakage in containment at or above the specified rate. Containment sump level and rate-of-change are the primary means of meeting this detection threshold. Containment airborne particulate activity provides a means of detection that is diverse and independent from containment sump monitoring:

- The instrumentation is designed to detect changes in the leakage rate outside containment, and the required sensitivity can vary depending upon the radiological release risk
- The instrumentation is designed to provide leak detection for lines within the containment and those that penetrate the containment and are a part of or are connected to the RCPB
- The instrumentation is designed to provide leak detection for lines that penetrate the containment and are neither a part of RCPB nor connected directly to the containment atmosphere. This instrumentation is an operational aid to help diagnose the source of potential leakage inside containment.

5.2.5.2.1 Leakage Detection Instrumentation and Monitoring Inside Containment

Reactor Pressure Vessel Water Level

The NBS includes three divisions for reactor water level detection.

Reactor Pressure Vessel Pressure

Low RPV pressure is an indicator of a leak, either inside or outside of containment. Pressure transmitters monitor RPV pressure.

Reactor Pressure Vessel Head Flange Inner O-Ring Seal

The reactor vessel closure head seal consists of two concentric O-rings placed between the reactor vessel shell and the vessel closure head. The O-rings are attached to the reactor vessel closure head in two concentric grooves. A line connected to the reactor vessel shell leads to the annulus between the two O-rings. Pressure buildup between the seals, caused by inner seal leakage, is sensed by a pressure transmitter, and annunciated in the MCR.

Reactor Pressure Vessel Head Vent Valve Discharge Temperature

A temperature element is included on the RPV head vent nozzle line to provide leak detection capability with the ability to trend the leak rate downstream of quench tank isolation valves. This line is normally isolated during normal operations and is therefore expected to be near ambient temperature. An increased temperature, in conjunction with containment sump in-leakage, is indicative of a leakage through the quench tank line isolation valves.

Containment Pressure

Pressure transmitters are provided for monitoring of containment pressure. Each pressure transmitter is located outside containment with instrument sensing lines routed through containment penetrations.

Pressure rise above the nominal level indicates a possible leak or loss of reactor coolant within containment. The Process Radiation and Environmental Monitoring System (PREMS) Containment Monitoring Subsystem (CMon) is responsible for monitoring containment pressure and providing the signal to the Distributed Control and Information System. Safety Category 1 and Safety Category 2 functions are supported by this instrumentation. The CMon is discussed in Section 6.8.

Containment Temperature

Temperature transmitters are installed throughout containment for monitoring air temperature. CMon containment air temperature and reactor coolant leakage detection functions are Safety Category 3.

Containment Fission Product Particulate and Noble Gas

CMon provides provides two independent particulate, iodine, and noble gas skids for monitoring of fission products in containment. The skids are located outside containment, with supply and return process piping routed through containment penetrations. Each process line is provided with an inboard/outboard pair of CIVs. Functions are credited for monitoring of fission products and indication of reactor coolant leakage within containment are Safety Category 3.

Containment Cooler Condensate Drain Flow

The Containment Cooling System (CCS) cooling coil condensate drain piping has flow rate measurement to detect the rates of leak inside containment during normal operations. The CCS is further described in Subsection 9.4.7.

Isolation Condenser Steam Supply Line and Condensate Return Line Flow

There are pressure differential-type transmitters that sense flow rates in the steam supply line to each IC and in each ICS return line to the RPV. ICS flow detection is train specific.

Containment Sump Level

The containment sump has triply redundant level transmitters to measure and trend containment leakage. Excessive containment boundary leakage is designed to be detected by the containment sump level changes, and frequency and duration of the pump runs. The Equipment and Floor Drain System is discussed in Subsection 9.3.3.

5.2.5.2.2 Leakage Detection Instrumentation and Monitoring Outside Containment

Main Steam Line Break

Differential pressure transmitter instruments from the MSL flow venturi taps measure steam flow and detect conditions that are indicative of a line break.

Instruments for sensing temperatures are provided in the RB steam tunnel and TB steam area.

Process Radiation Level

The Process Radiation Monitoring Subsystem (PRM) includes leak detection sensors for radiation levels in the following:

- Refuel Floor Area
- RB Heating, Ventilation, and Cooling System (HVS) Exhaust
- Refuel Area HVS Exhaust
- IC Pool Area

The PRM provides radiation monitoring of the lower RB HVS exhaust.

The PRM provides radiation monitoring of the Refuel Floor/Fuel Pool Area HVS exhaust.

The PRM monitors the IC cubicle pool area for radioactive materials that would indicate a failure of the IC heat exchanger pressure boundary.

The PRM is discussed in Section 11.5.

Real-time measurements are displayed in the MCR with alarms to notify personnel when measurements exceed preset limits. Local indication and alarms are provided where necessary to alert personnel of hazardous conditions.

Shutdown Cooling System Flow and Temperature

The SDC provides differential pressure transmitters that sense the flow rate in the supply line to each SDC train and in the associated return line to FW, as well as the overboard line to CUW (to perform a temperature-compensated mass flow balance). Upon detection of a significant drop in SDC mass flow rate, an off-normal condition is displayed in the MCR. A time delay allows for operator action to resolve the issue prior to an SDC pump trip and SDC loop isolation.

Area temperature sensors provided for the SDC rooms containing the pumps, heat exchangers, and piping. The system continuously monitors for leakage utilizing area temperature monitors. If a leak is detected via an increase in area temperature, an alarm is initiated in the MCR.

Reactor Water Cleanup

CUW provides differential pressure transmitters that sense the flow rate in the supply line to each CUW train and in the associated return line to the FW line and overboard lines to perform a temperature-compensated mass flow balance. CUW also provides ambient temperature in the areas of the TB containing CUW piping not already sensed by the NBS.

Condensate and Feedwater System Differential Pressure and Air Temperature

The CFS provides pressure transmitters that sense the differential pressure between the reactor vessel and the FW lines. As an operator aid, ambient temperature is monitored in the TB. The CFS is further described in Subsection 10.4.7.

5.2.5.3 Display and Indications in the Main Control Room

Although monitored leakage parameters are included in Table 5.2-6, plant leakage parameters monitored in the MCR will be provided in the FSAR.

5.2.5.4 Limits for Reactor Coolant Leakage Rates Within the Containment

Limits for reactor coolant leakage rates will be provided in the FSAR.

5.2.5.5 Criteria to Evaluate the Adequacy and Margin of Leak Detection System

Leak detection instrumentation detects leakage in containment at or above the specified rate of 0.19 L/min. Containment sump level and rate-of-change monitoring instrumentation provides the sensitivity necessary to meet the 0.19 L/min leak detection threshold. TS monitoring systems that respond to a leakage increase of 3.8 L/min in 1 hour or less are the containment sump level and airborne particulate activity.

5.2.5.6 Separation of Identified and Unidentified Leakages in the Containment

As discussed in Subsection 5.2.5.1, the BWRX-300 design includes only unidentified leakage that is potentially from the RCPB. As identified leakage is not defined, there is no leakage separation. These are alternatives from RG 1.45 General Positions 1.1 and 1.2.

5.2.5.7 Testing, Calibration, and Inspection Requirements

The requirements for testing, calibration, and inspection of the leak detection systems will be provided in the FSAR.

5.2.5.8 Regulatory Guide 1.45 Conformance

5.2.5.8.1 General Positions

Position 1.1

The BWRX-300 design considers only unidentified leakage; unidentified leakage is directed to the containment sump and is potentially from the RCPB. This is an alternative to RG 1.45 Position 1.1.

Position 1.2

The BWRX-300 design eliminates separation of identified and unidentified leakage. This is an alternative to RG 1.45 Position 1.2.

Position 1.3

RG 1.45 defines a critical component of the RCPB as a component which is risk-significant or potentially susceptible to material degradation. No critical components have been identified for leak detection monitoring. Position 1.3 is satisfied.

Position 1.4

The Plant Cooling Water System (PCW) provides cooling to the Fuel Pool Cooling and Cleanup System (FPC) and SDC heat exchangers. Intersystem leakage of radioactive material into the PCW from the FPC and SDC can be identified by use of sample points upstream of the PCW pumps and downstream of each FPC heat exchanger.

Leak detection provided for ICS and CUW is described in Subsection 5.2.5.2.2.

Leakage past closed RPV isolation valves following a reactor isolation is detected by RPV pressure and level changes. Position 1.4 is satisfied.

Position 1.5

Subsection 5.2.5.5 identifies the capabilities of TS monitoring systems. Although monitored leakage parameters are included in Table 5.2-6, associated capabilities will be provided in the FSAR.

5.2.5.8.2 Leakage-Monitoring-Related Positions

Position 2.1

Information included in plant procedures will be provided in the FSAR.

Position 2.2

Leakage detection systems with a response time of no greater than 1 hour for a leakage rate of 3.8 L/min are discussed in Subsection 5.2.5.5.

Position 2.3

Two independent and diverse instruments and/or methods that have the detailed monitoring capabilities identified in Position 2.2 are discussed in Subsection 5.2.5.5. Additional methods used to detect and monitor for leakage are identified in Table 5.2-6.

Position 2.4

Plant shutdown is required for any seismic event for which ground motion exceeds that of the Operating-Basis Earthquake (OBE). For BWRX-300, the OBE is set as one-third of the Safe Shutdown Earthquake (SSE) Ground Motion design response spectra. With the OBE set to this value, explicit response or design analysis is not required. Instrumentation associated with leak detection can be assumed to be functional for seismic events not requiring a shutdown. Position 2.4 is satisfied.

Position 2.5

Information related to calibration and plant testing during plant operation will be provided in the FSAR.

5.2.5.8.3 Operations Related Positions

Position 3.1

Operating plant activities will be described in the FSAR.

Position 3.2

Information included in plant procedures will be provided in the FSAR.

Position 3.3

Leakage monitoring system alarms, testing and calibration, and operator interface information will be included in the FSAR.

Position 3.4

Operating plant activities will be described in the FSAR.

5.2.5.8.4 Technical Specification Position

Position 4.1 TSs will be provided in the FSAR.

5.2.6 References

5.2-1 EPRI 3002025550, BWRVIP-190, Volume 1: “BWR Water Chemistry Guidelines – Mandatory, Needed and Good Practice Guidance” and Volume 2: “BWR Water Chemistry Guidelines – Technical Basis,” Electric Power Research Institute, Revision 2, 2024.

5.2-2 NEDC-33910P-A, “BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection,” GE-Hitachi Nuclear Energy Americas LLC, Revision 2, June 2021.

5.2-3 ASME BPVC, Section II, “Materials,” American Society of Mechanical Engineers, 2021.

5.2-4 ASME BPVC, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” American Society of Mechanical Engineers, 2021.

5.2-5 ASME BPVC, Section IX, “Qualification Standard for Welding, Brazing, and Fusing Procedures; Welders; Brazers; and Welding, Brazing, and Fusing Operators,” American Society of Mechanical Engineers, 2021.

5.2-6 ASME BPVC, Section XI, Division 1, “Rules for Inspection and Testing of Components of Light-Water-Cooled Plants,” American Society of Mechanical Engineers.

5.2-7 ANSI/AWS A4.2M/A4.2, “Standard Procedures for Calibrating Magnetic Instruments to Measure the Delta Ferrite Content of Austenitic and Duplex Ferritic-Austenitic Stainless Steel Weld Metal,” American Welding Society, 1997.

5.2-8 ASME OM Code, “Operation and Maintenance of Nuclear Power Plants,” American Society of Mechanical Engineers.

Table 5.2-1: Applicable Code Cases for ASME BPVC Components

Code Case Number (See Note 1)	Code Case Title
N-60-6	Material for Core Support Structures, Section III, Division 1
N-62-7	Internal and External Valve Items, Section III, Division 1, Classes 1, 2, and 3
N-249-17	Additional Materials for Subsection NF, Classes 1, 2, 3, and MC Supports Fabricated without Welding, Section III, Division 1
N-284-4	Metal Containment Shell Buckling Design Methods, Class MC, TC, and SC Construction, Section III, Divisions 1, and 3
N-548	Air Cooling of SA-182 Grades F304, F304L, F316, F316L Forgings Instead of Liquid Quenching After Solution Heat Treatment, Class 1, 2, and 3, Section III, Division 1
N-580-2	Use of Alloy 600 with Columbium Added, Section III, Division 1
N-632	Use of ASTM A 572 Grades 50 and 65 for Structural Attachments to Class CC Containment Liners
N-756	Alternative Rules for Acceptability for Class 1 Valves, NPS (DN 25) and Smaller with Nonwelded End connections Other Than Flanges, Section III, Division 1
N-779	Alternative Rules for Simplified Elastic-Plastic Analysis Class 1, Section III, Division 1
N-943 (Note 2)	Ni-Cr-Mo-Cb (UNS N06625) Forgings for Class 1 Construction, Section III, Division 1

Notes:

1. Unless otherwise noted, this table lists ASME BPVC, Section III Code Cases endorsed in RG 1.84 and applicable to the BWRX-300. Conditions imposed by RG 1.84 on Code Cases listed in this table are also applicable to the BWRX-300. See Subsection 5.2.1.2 for additional information.
2. Code Case N-943 has been approved by ASME. The code case is pending review by the NRC for incorporation in RG 1.84.

Table 5.2-2a: Reactor Coolant Pressure Boundary Materials

Component	Form	Material (Note 1)	ASME Specification
Reactor Pressure Vessel			
RPV Shells and Heads	Plate Forging	Low-Alloy Steel	SA-533/SA-533M Type B, Class 1 SA-508/SA-508M Gr 3, Class 1
RPV Shell and Head Flange	Forging	Low-Alloy Steel	SA-508/SA-508M Gr 3, Class 1
RPV Main Closure Bolting	Bolting	Low-Alloy Steel	SA-540/SA-540M Gr B23 or B24, Class 3
RPV Standard Flange Bolting	Bolting	Low-Alloy Steel	SA-540/SA-540M Gr B23 or B24, Class 3
RPV Nozzles	Forging	Low-Alloy steel	SA-508/SA-508M Gr 3, Class 1
RPV Level Instrumentation Nozzles	Forging	Stainless steel or Ni-Cr-Fe	SA-182/SA-182M Gr F316/F316L (Dual Grade), SA-965/SA-965M Gr F316/F316L (Dual Grade), SB-168, SB-166, or SB-564 Nickel Alloy 600M (Modified per ASME Code Case N-580-2)
Main Steam Reactor Isolation Valves (MSRIVs)			
MSRIV Body	Forging	Low-Alloy Steel	SA-350/SA-350M Gr LF2 Class1
MSRIV Cover	Forging	Low-Alloy Steel	SA-350/SA-350M Gr LF2 Class1
MSRIV stem	Rod or Bar	Stainless Steel	SA-564/SA-564M Type 630 (Notes 2,10), SA-479/SA-479M Type XM-19
MSRIV Cover Bolting	Bolting	Alloy steel	SA-193/SA-193M Gr B7
MSRIV Hex nuts	Bolting Nuts	Alloy steel	SA-194/SA-194M Gr 7
Main Steam Containment Isolation Valves			
MSCIV Body	Forging	Low-Alloy Steel	SA-350/350M Gr LF2 Class 1
MSCIV Cover	Forging	Low-Alloy Steel	SA-350/350M Gr LF2 Class 1
MSCIV Poppet	Forging	Low-Alloy Steel	SA-508/SA-508M Gr 3 Class 1

Component	Form	Material (Note 1)	ASME Specification
MSCIV stem	Rod or Bar	Stainless Steel	SA-564/SA-564M Type 630 (Notes 2,10), SA-479/SA-479M Type XM-19
MSCIV Body bolt	Bolting	Alloy steel	SA-193/SA-193M Gr B7
MSCIV Hex nuts	Bolting Nuts	Alloy steel	SA-194/SA-194M Gr 7
Main Steam Line RCPB Portion			
MS Pipe	Seamless Forged & Bored	Carbon steel Low-Alloy steel	SA-333/SA-333M Gr 6 SA-335/SA-335M Gr P22 SA-369/SA-369M Gr FP22
MS Contour nozzle	Forging	Low-Alloy steel	SA-508/SA-508M Gr 3 Class 1
MS 200 mm 1500 lb. large groove flange	Forging	Carbon steel	SA-350/SA-350M Gr LF2 Class 1
MS 50 mm special nozzle	Forging	Carbon steel	SA-350/SA-350M Gr LF2 Class 1
MS Elbow	Seamless Fitting	Carbon steel	SA-420/SA-420M Gr WPL-6 SA-508/SA-508M Gr 1
MS Head fitting/penetration piping	Forging	Carbon steel	SA-350/SA-350M Gr LF2 Class 1
MS Other Fitting	Forging	Low-Alloy steel	SA-234/SA-234M Gr WP22 SA-336/SA-336M Gr F22
Reactor Pressure Vessel Head Vent			
Head Vent Piping	Piping	Stainless Steel	SA-312/SA-312M Gr TP316/ TP316L (Dual Grade)
Head Vent to MSL Valves	Forging	Carbon Steel	SA-350/SA-350M Gr LF2 Class 1
Head Vent to Boundary Valves	Forging	Stainless Steel	SA-182/SA-182M Gr F316/F316L (Dual Grade)

Component	Form	Material (Note 1)	ASME Specification
RPV Appurtenances Control Rod Drive Flanges/Housings			
CRD Housings	Forging	Stainless steel	SA-182/SA-182M Gr F304/F304L, Gr F316/F316L (Dual Grade), SA-965/SA-965M Gr F304/F304L/F316/F316L (Dual Grade)
Lower Component Housing	Forgings	Stainless Steel	SA-965/SA-965M Gr F304
Middle Flange	Forgings	Stainless Steel	SA-965/SA-965M Gr F304
Upper Flange	Forgings	Stainless Steel	SA-965/SA-965M Gr F316
Ball Check Valve Retainer	Bar and Shapes	Stainless Steel	SA-479/SA-479M Type 316
CRD Spool piece (if used)	Forging	Stainless steel	SA-965/SA-965M Gr F304/F304L
CRD Mounting bolts	Bolting	Alloy steel	SA-193/SA-193M Gr B7
CRD Penetration Stub Tubes	Bar, Seamless Pipe, Forgings	Ni-Cr-Fe	SB-166, SB-167, SB-564 Nickel Alloy 600M (Modified per ASME Code Case N-580-2)
RPV Appurtenances Incore Instrumentation Flanges/Housings			
Incore Instrumentation Nozzles	Forging Bar, Seamless Pipe Forging	Stainless steel or Ni-Cr-Fe	SA-182/SA-182M Gr F304/F304L, Gr F316/F316L (Dual Grade), SA-965/SA-965M Gr F304/F304L, Gr F316/F316L (Dual Grade), SB-166, SB-168, or SB-564 Nickel Alloy 600M (Modified per ASME Code Case N-580-2)
Incore Instrumentation and CRD Penetration Stub Tubes	Bar, Seamless Pipe Forging	Ni-Cr-Fe	SB-166, SB-168, or SB-564 Nickel Alloy 600M (Modified per ASME Code Case N-580-2)

Component	Form	Material (Note 1)	ASME Specification
Incore Housings	Seamless Pipe, Forgings	Stainless Steel	SA-312/SA-312M Gr TP304/ TP304L, SA-182/SA-182M Gr F316/F316L (Dual Grade)
Isolation Condenser System Piping and Components			
IC Steam pipe	Pipe or tubing	Ni-Cr-Mo-Cb or Ni-Cr-Fe	SB-444 UNS N06625 or Nickel Alloy modified per ASME Code Case N-580-2
IC Standby Gas Purge Piping	Pipe or tubing	Ni-Cr-Mo-Cb or Ni-Cr-Fe	SB-444 UNS N06625 or Nickel Alloy modified per ASME Code Case N-580-2
IC Cover bolting (Note 3)	Forging	Stainless steel	SA-705/SA-705M Gr 630
IC Drum and Drum Covers	Forgings	Ni-Cr-Mo-Nb or Ni-Cr-Fe	SB-564 Gr 1 UNS N06625 or Nickel Alloy modified per ASME Code Case N-580-2
IC Condensate piping	Pipe or tubing	Ni-Cr-Mo-Cb or Ni-Cr-Fe	SB-444 UNS N06625 or Nickel Alloy modified per ASME Code Case N-580-2
ICS Recombiner vessel	Forgings	Stainless steel	SA-182/SA-182M Gr F316/F316L (Dual Grade), SA-965/SA-965M Gr F316/F316L (Dual Grade)
ICS Recombiner piping	Pipe or tubing	Ni-Cr-Mo-Cb or Ni-Cr-Fe	SB-444 UNS N06625 or Nickel Alloy modified per ASME Code Case N-580-2

Component	Form	Material (Note 1)	ASME Specification
Isolation Condenser System Isolation Condenser Supply Reactor Isolation Valve (ICSRIV)			
ICSRIV Body	Forging	Low-Alloy Steel	SA-350/SA-350M Gr LF2 Class 1
ICSRIV Disc	Forging	Low-Alloy Steel	SA-350/SA-350M Gr LF2 Class 1
ICSRIV Stem	Rod or Bar	Stainless Steel	SA-564/SA-564M Type 630 (Notes 2,10), SA-479/SA-479M Type XM-13, Type XM-19
Isolation Condenser System Isolation Condenser Return Reactor Isolation Valve (ICRRIV)			
ICRRIV Body	Forging	Low-Alloy Steel	SA-350/SA-350M Gr LF2 Class 1
ICRRIV Disc	Forging	Low-Alloy Steel	SA-350/SA-350M Gr LF2 Class 1
ICRRIV Stem	Rod or Bar	Stainless Steel	SA-564/SA-564M Type 630 (Notes 2,10), SA-479/SA-479M Type XM-13, Type XM-19
Isolation Condenser System Condensate Return Valves (CRV)			
ICS CRV Body	Forging	Stainless Steel	SA-182/SA-182M Gr F304/F304L, Gr F316/F316L (Dual Grade)
ICS CRV Disc	Forging	Stainless Steel	SA-182/SA-182M Gr F304/F304L, Gr F316/F316L (Dual Grade)
ICS CRV Stem	Rod or Bar	Stainless Steel	SA-564/SA-564M Type 630 (Notes 2,10), SA-479/SA-479M Type XM-19
Isolation Condenser System ICS to SDC Containment Isolation Valves			
ICS CIV	Forgings	Stainless Steel	SA-182/SA-182M Gr F304/F304L Gr F316/F316L (Dual Grade)

Component	Form	Material (Note 1)	ASME Specification
SDC Interface Redundant Isolation Valve (ICS Outboard CIV)	Forgings	Stainless Steel	SA-182/SA-182M Gr F304/F304L Gr F316/F316L (Dual Grade)
Isolation Condenser System Other			
ICS Purge Valves	Forging or Casting	Stainless Steel	SA-182/SA-182M Gr F316/F316L (Dual Grade), SA-351/SA-351M Gr CF3M
ICS Purge Piping	Seamless Piping	Stainless Steel	SA-312/SA-312M Gr TP316/ TP316L (Dual Grade)
ICS Purge Fittings	Forging	Stainless Steel	SA-182/SA-182M Gr F316/F316L (Dual Grade), SA-403/SA-403M Gr WP 316/ 316L (Dual Grade)
Condensate and Feedwater Heating System Feedwater Piping RCPB Portion			
FW Piping	Seamless	Stainless Steel	SA-312/SA-312M Gr TP316/ TP316L (Dual Grade)
FW Fittings	Forging or Fitting	Stainless Steel	SA-182/SA-182M Gr F316/316L (Dual Grade), SA-403/SA-403M Gr WP 316/ 316L (Dual Grade)
Condensate and Feedwater Heating System Feedwater Reactor Isolation Valves (FWRIVs)			
FWRIV Body	Forging	Low-Alloy Steel	SA-350/SA-350M Gr LF2
FWRIV Cover	Forging	Low-Alloy Steel	SA-350/SA-350M Gr LF2
FWRIV stem	Rod or Bar	Stainless Steel	SA-564/SA-564M Type 630 (Notes 2,10), SA-479/SA-479M Type XM-13, Type XM-19
FWRIV Body bolting	Bolting	Alloy steel	SA-193/SA-193M Gr B7
FWRIV Hex nuts	Bolting Nuts	Alloy steel	SA-194/SA-194M Gr 7

Component	Form	Material (Note 1)	ASME Specification
Condensate and Feedwater Heating System Feedwater Containment Isolation Valves			
FWCIV Body	Forging	Stainless Steel	SA-182/SA-182M Gr F304/F304L, Gr F316/F316L (Dual Grade)
FWCIV Cover	Forging	Stainless Steel	SA-182/SA-182M Gr F304/F304L, Gr F316/F316L (Dual Grade)
FWCIV stem	Rod or Bar	Stainless Steel	SA-564/SA-564M Type 630 (Notes 2,10), SA-479/SA-479M Type XM-13, Type XM-19
FWCIV Body bolting	Bolting	Alloy steel	SA-193/SA-193M Gr B7
FWCIV Hex nuts	Bolting Nuts	Alloy steel	SA-194/SA-194M Gr 7
Reactor Water Cleanup System (RCPB Portion)			
CUW Piping	Seamless	Stainless Steel	SA-312/SA-312M Gr TP316/ TP316L (Dual Grade)
CUW Fittings	Forging or Fitting	Stainless Steel	SA-182/SA-182M Gr F316/F316L (Dual Grade), SA-965/SA-965M Gr F316/F316L (Dual Grade), SA-403/SA-403M Gr WP 316/ 316L (Dual Grade)
CUW RIV Body (Note 4)	Forging	Low-Alloy Steel	SA-350/SA-350M Gr LF2
CUW RIV Stem	Rod or Bar	Stainless Steel	SA-564/SA-564M Type 630 (Notes 2,10), SA-479/SA-479M Type XM-13, Type XM-19
CUW CIV Body	Casting	Stainless Steel	SA-351/SA-351M Gr CF8M/CF3M
CUW CIV Stem	Rod or Bar	Stainless Steel	SA-564/SA-564M Type 630 (Notes 2,10), SA-479/SA-479M Type XM-13, Type XM-19

Component	Form	Material (Note 1)	ASME Specification
Boron Injection System (RCPB Portion)			
BIS CIV Body	Forging	Stainless Steel	SA-182/SA-182M Gr F316/F316L (Dual Grade)
BIS CIV Disc	Forging	Stainless Steel	SA-182/SA-182M Gr F316/F316L (Dual Grade)
BIS CIV Stem	Rod or Bar	Stainless Steel	SA-564/SA-564M Type 630 (Notes 2,10), SA-479/SA-479M Type XM-13, Type XM-19
BIS Check Valve Body	Forging	Stainless Steel	SA-182/SA-182M Gr F316/F316L (Dual Grade)
BIS Piping	Seamless	Stainless Steel	SA-312/SA-312M Gr TP316/TP316L (Dual Grade)
BIS Fittings	Forging or Fitting	Stainless Steel	SA-182/SA-182M Gr F316/F316L (Dual Grade), SA-403/SA-403M Gr WP 316/316L (Dual Grade)
Additional RCPB Bolting Material			
Flanges, Covers, and Bonnets	Stud or Bolting	Alloy & Carbon Steels	SA-354, SA-449

Notes:

1. Carbon content of RCPB wrought austenitic stainless steel (304/304L/316/316L) is 0.02% maximum.
2. To reduce risk of cracking in valve stems, the minimum nominal aging temperature in precipitation hardened stainless steels shall be 580°C (H1075) or higher.
3. SA-705/SA-705M Grade 630 bolting is limited to the H1100 condition per Table 4 of ASME BPVC, Section II, Part D.
4. Any wetted surfaces in carbon or low-alloy steel bodied valves in the CUW are clad with stainless steel.
5. Reference Table 5.3-1 and Subsection 5.3.1.5 for core beltline material composition limits.
6. Not used.
7. Not used.
8. Code Case N-943 for use of Alloy 625 (UNS N006625) is provided in Table 5.2-1.
9. Not used.
10. Items fabricated from these materials are wetted by reactor coolant but may not necessarily be part of the design ASME BPVC pressure boundary. Therefore, some of the specific material specifications or material specification grades identified under this category are not required to be listed in the ASME BPVC, Section II, Part D, Table 2A.

Table 5.2-2b: Reactor Coolant Pressure Boundary Welding Filler Metals

Base Material	Filler Metal Type	ASME Classification	AWS Classification
Carbon Steel Welds	Covered Electrode	SFA-5.1	E7018
	Filler metal	SFA-5.18	ER70S-2, ER70S-3, ER70S-6
Low Alloy Steel Welds (Note 1)	Covered Electrode	SFA-5.5	E9018-B3
	Filler metal	SFA-5.23	EF2, EF3 or other mutually agreed filler metal with appropriate chemistry selected for base metal composition and mechanical property requirements including toughness
		SFA-5.28	ER90S-XX with appropriate chemistry selected for base metal composition, shielding gas, and mechanical property requirements including toughness
Low Alloy Steel Piping Welds (2¼ Cr-Mo)	Covered Electrode	SFA-5.5	E9018-B3
	Filler metal	SFA-5.28	ER90S-XX with appropriate chemistry selected for base metal composition, shielding gas, and mechanical property requirements including toughness.
Stainless Steel Welds (Note 2)	Covered Electrode or Filler metal	SFA-5.4	E308L-XX, E309L-XX, E316LXX
	Filler Metal	SFA-5.9	ER308L, ER308LSi, ER309L, ER316L
Nickel Alloy Welds	Filler Wire	SFA-5.14	ERNiCr-3
Stainless Steel Welds	Consumable Inserts	SFA-5.30	IN308L, IN316L

Notes:

1. Dissimilar metal welds between carbon/low alloy steel and stainless steel are made with a layer (or layers) of 309L, followed by completion of the weld with either 308L, 316L, or 309L. Alternatively, the welds may be completed with Nickel Alloy 82 for the entire weld. Dissimilar metal welds between nickel alloy and carbon/low alloy steel or stainless steel are performed using Nickel Alloy 82. Welds between carbon and low alloy steel may be made with the filler metals listed for either base material, except that partial penetration welds may also be made with Nickel Alloy 82.
2. The maximum carbon content for stainless steel weld filler metal will be less than or equal to 0.035% in accordance with GL 88-01.

Table 5.2-3: Reactor Pressure Vessel Attachments

Component	Form	Material	ASME Specification
Reactor Pressure Vessel Internal Attachments			
Core Support Legs	Plate, Bar or Forgings	Ni-Cr-Fe	SB-168, SB-166, or SB-564 Nickel Alloy 600M (Modified per ASME Code Case N-580-2)
Reactor Pressure Vessel External Attachments			
Vessel Support Skirt	Forgings, Plates	Low-Alloy Steel	SA-508/SA-508M, Gr 3, Class 1; SA-533/SA-533M, Type B, Class 1
Vessel Stabilizer Brackets	Forgings, Plates	Low-Alloy Steel	SA-508/SA-508M, Gr 3, Class 1; SA-533/SA-533M, Type B, Class 1

Table 5.2-4: Special Purpose Materials

Component	Material	Location	Compatibility of Construction Materials with Reactor Coolant
Isolation Condenser System Catalyst Material			
ICS Recombiner Catalyst Material	Catalyst materials are typically noble metals such as palladium or platinum. Substrate materials are typically ceramics or stainless steels. The catalyst material covers the substrate material. (See Note)	Each IC is equipped with recombiner(s) containing catalyst material exposed to reactor coolant.	The catalyst material is not to introduce additional contaminants or particulates into the process fluid/gas that adversely affects reactor water chemistry.

Note:

The catalyst material must be specified to cover substrate surfaces if required by the substrate material selection to preclude degradation of the substrate.

Table 5.2-5: Typical BWRX-300 Water Chemistry Under Power Operation

System	Concentration (ppb)					Conductivity µS/cm at 25ºC (77°F)
	Iron	Copper	Chloride	Sulfate	Oxygen	
Feedwater	< 1	< 0.05	-	-	40-100	< 0.06
Reactor Water	-	-	< 1	< 2	Plant Specific	≤ 0.08 (See Note 2)
Control Rod Drive Cooling Water	-	-	-	-	40-100	< 0.06

Notes:

1. Typical control parameters are presented with best practice values that can be used for goal setting. Best practice values for the BWRX-300 are expected to evolve based on operational experience.
2. Conductivity is corrected for zinc and iron.

Table 5.2-6: Monitored Leakage Parameters

System	Instrument Type	Size	Process Measurement	Location (Note 1)
Nuclear Boiler	Differential pressure transmitter	Break	MS line flow	O
	Pressure transmitter	Break	Reactor pressure	I
	Level transmitter	Break	Reactor level	I
	Temperature element	Leak	MS tunnel ambient temperature	O
	Temperature element	Leak	Turbine building ambient temperature	O
	Temperature element	Leak	MS tunnel differential temperature	O
	Pressure transmitter	Leak	Reactor head O-ring pressure	I
	Temperature element	Leak	Reactor head vent valve seat leakage	I
Process Radiation System and Environmental Monitoring	Pressure transmitter	Leak/Break	Containment pressure	I
	Pressure transmitter	Leak/Break	Containment Pressure	I
	Radiation monitor	Leak	Refuel floor area radiation	O

System	Instrument Type	Size	Process Measurement	Location (Note 1)
Process Radiation System and Environmental Monitoring	Temperature element	Leak/Break	Containment temperature	I
	Radiation monitor	Leak/Break	Containment particulate	I
	Radiation monitor	Leak/Break	Containment noble gas	I
	Radiation monitor	Leak	RB HVAC exhaust	O
	Radiation monitor	Leak	Refuel floor area radiation	O
	Radiation monitor	Leak	ICS pool exhaust	O
Isolation Condenser	Differential pressure transmitter	Break	Differential Pressure	I
Shutdown Cooling	Temperature element	Break	Mass flow balance	O
	Differential pressure transmitter	Break		O
	Temperature element	Leak/Break	Ambient temperature	O
Reactor Water Cleanup	Temperature element	Break	Mass flow balance	O
	Differential pressure transmitter	Break		O
	Temperature element	Leak/Break	Ambient temperature	O

System	Instrument Type	Size	Process Measurement	Location (Note 1)
Condensate and Feedwater Heating	Differential pressure transmitter	Break	Reverse pressure from RPV to FW	O
	Temperature element	Leak/Break	Ambient temperature	O
Containment Cooling	Differential pressure transmitter	Leak/Break	Containment cooler drain flow	I
Equipment and Floor Drain	Level transmitter	Leak	Containment sump level and rate-of-change	I
	Level transmitter	Leak	Reactor building sump level and rate-of-change	O
	Level transmitter	Leak	Turbine building sump level and rate-of-change	O
	Level transmitter	Leak	Radwaste building sump level and rate-of-change	O

Note:

(1) I = Containment; O = Outside Containment

5.3 Reactor Vessel

The RPV is an SC1 component and is designed to meet the requirements of ASME BPVC, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” Subsection NB, Class 1 Components (Reference 5.3-1).

The RPV is used as a pressure-retaining barrier of primary coolant in a BWRX-300 reactor. It is supported horizontally by the vessel stabilizer brackets and vertically by the vessel support skirt. The RPV contains the light water coolant and moderator and forms the path for core recirculation flow. The vessel also contains nuclear fuel, fuel supporting structures, and internals required for operation of the reactor.

The vessel support skirt is a truncated conical support appurtenance of the RPV. The vessel support skirt attachment knuckle is integrally forged with the vessel cylinder located above the core beltline region. The bottom of the skirt transmits the vertical vessel load to the RPV pedestal. The connection is designed to allow for radial movement due to thermal expansion and contraction of the vessel shell during operational heat up and cool-down evolutions, respectively.

The RPV has two sets of horizontal motion stabilizer brackets with a set of six stabilizer brackets located above the RPV bottom head and a second set of six stabilizers located below the MS nozzles. The stabilizer bracket design is based on existing hardware design from previous operating BWRs. As discussed in Subsection 3.9.1.4.5, the stabilizer brackets are designed to ASME BPVC, Section III, Subsection NF. They are categorized as SC1 and Seismic Category I. The main function of the stabilizer brackets is to provide horizontal support to limit vessel motion by transferring load from the vessel to the adjacent civil structure. The twelve upper and lower vessel stabilizers function independently, depending on the direction of vessel motion due to an applied dynamic force. The stabilizer brackets provide resistance against vessel motion.

The RPV pedestal is discussed in Subsection 3.8.3.1.

Figure 4.1-1 provides an illustration of the RPV and reactor internals.

5.3.1 Reactor Vessel Materials

The BWRX-300 RPV meets the requirements of:

10 CFR 50, Appendix A, GDC 1 and GDC 30, as they relate to quality standards for design, fabrication, erection, and testing of SSCs, by compliance with Section III, Section IX, “Qualification Standard for Welding, Brazing, and Fusing Procedures; Welders; Brazers; and Welding, Brazing, and Fusing Operators,” (Reference 5.3-2), and Section XI, “Rules for Inspection and Testing of Components of Light-Water-Cooled Planted,” (Reference 5.3-3) of the ASME BPVC and by conformance with RG 1.31, RG 1.50, RG 1.65, Generic Letter 88-01 and NUREG-0313, Revision 2.

10 CFR 50, Appendix A, GDC 4, as it relates to compatibility of components with environmental conditions, by conformance with RG 1.44 and Generic Letter 88-01 and NUREG-0313, Revision 2.

10 CFR 50, Appendix A, GDC 14, as it relates to prevention of rapidly propagating fractures of the RCPB, by compliance with 10 CFR 50, Appendix G and conformance with RG 1.31 and Generic Letter 88-01 and NUREG-0313, Revision 2.

10 CFR 50, Appendix A, GDC 31, as it relates to material fracture toughness, by compliance with 10 CFR 50, Appendix G, and conformance with RG 1.65.

10 CFR 50, Appendix A, GDC 32, as it relates to the requirements for a materials surveillance program, by compliance with 10 CFR 50, Appendix H and ASTM International (ASTM) E185-21,

"Standard Practice for Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels," (Reference 5.3-6). See Subsection 5.3.1.6.1 for discussion on applicable Edition of ASTM E185.

10 CFR 50.55a as it relates to quality standards for design and determination and monitoring of fracture toughness, by compliance with Appendix G to 10 CFR 50, Section III, Section IX, and Section XI of the ASME BPVC, and conformance with RG 1.31, RG 1.50, RG 1.65, Generic Letter 88-01, and NUREG-0313, Revision 2. RG 1.34 does not apply as discussed in Subsection 5.2.3.3.2. RG 1.43 does not apply as discussed in Subsection 5.3.1.4.

10 CFR 50.60, as it relates to RCPB fracture toughness and material surveillance requirements of 10 CFR 50, Appendix G and Appendix H, by compliance with 10 CFR 50, Appendix G, 10 CFR 50, Appendix H, and ASTM E185. See Subsection 5.3.1.6.1 for discussion on applicable Edition of ASTM E185.

10 CFR 50, Appendix B, Criterion XIII, as it relates to onsite material cleaning control, as discussed in Subsection 5.3.3.4.

10 CFR 50, Appendix G, as it relates to materials testing and acceptance criteria for fracture toughness, by compliance with 10 CFR 50, Appendix H, (see Subsection 5.3.1.6.1 for discussion on applicable Edition of ASTM E185) and conformance with RG 1.65.

10 CFR 50, Appendix H, as it relates to the determination and monitoring of fracture toughness, by compliance with 10 CFR 50, Appendix H, and ASTM E185. See Subsection 5.3.1.6.1 for discussion on applicable Edition of ASTM E185.

Section 3.1 summarizes compliance with GDC.

5.3.1.1 Material Specifications

The materials used in the RCPB, RPV, appurtenances and selected attachments are shown in Table 5.2-2a and Table 5.2-3, with the applicable specifications.

The RPV materials comply with ASME BPVC, Section II, "Materials," (Reference 5.3-4) and 10 CFR 50, Appendix G. The RPV materials also meet the additional requirements discussed in the following subsections.

These materials provide adequate strength, fracture toughness, fabricability, and compatibility with the BWR environment. Their suitability has been demonstrated by long-term successful OPEX in reactor service.

5.3.1.2 Special Processes Used for Manufacturing and Fabrication

The RPV is constructed from low-alloy, high strength steel plate and forgings. Plates are ordered to ASME SA-533/SA-533M, Type B, Class 1, and forgings to ASME SA-508/SA-508M, Grade 3, Class 1. These materials are melted to fine grain practice and are supplied in the quenched and tempered condition. Further restrictions include a requirement for vacuum degassing to lower the hydrogen level and improve the cleanliness of the low-alloy steels. Specified limits for materials used in the core beltline region are presented in Table 5.3-1.

Materials for the RPV main closure bolting (studs, nuts, and washers) have special material controls for toughness and yield strength as presented in Table 5.3-1.

Plates, forgings, and bolting are ultrasonically tested, and surface examined by magnetic particle methods or liquid penetrant methods as required by ASME BPVC, Section III.

Fracture toughness properties of materials are also measured and controlled in accordance with ASME BPVC, Section III, Division 1 (Reference 5.3-1).

Fabrication of the RPV is performed in accordance with GVH approved drawings, fabrication procedures, and test procedures. The shells and vessel heads are made from formed plates or forgings, whereas flanges and nozzles are made from forgings. Welding performed to join these vessel components is in accordance with procedures qualified per ASME BPVC, Section III and IX requirements. Impact tests are performed to determine the properties of the base metal, Heat Affected Zone (HAZ), and weld metal.

Gas Tungsten Arc Welding, Gas Metal Arc Welding, Shielded Metal Arc Welding, and Submerged Arc Welding processes may be employed. Electroslag welding is not used except for cladding. Preheat and interpass temperatures will be controlled in accordance with ASME BPVC Section IX. The preheat and interpass temperatures for welding of low-alloy steel will meet or exceed the values given in ASME BPVC, Section III, Appendix D. PWHT of low-alloy welds is performed in accordance with ASME BPVC, Section III as specified in Table 5.3-1.

Volumetric examination and surface examination are performed on pressure-retaining welds as required by ASME BPVC, Section III, NB-5300. In addition, pressure-retaining welds are given a supplemental preservice ultrasonic examination in accordance with ASME BPVC, Section XI (Reference 5.3-3).

The materials, fabrication procedures, and testing methods used in the construction of the RPV meet or exceed requirements of ASME BPVC, Section III for Class 1 vessels.

5.3.1.3 Special methods for Nondestructive Examination

The materials and welds on the RPV are examined in accordance with methods and acceptance requirements specified by ASME BPVC, Section III. The pressure-retaining welds are volumetrically examined. In addition, the pressure-retaining welds are ultrasonically examined using acceptance standards that are equivalent or more restrictive than required by ASME BPVC, Section XI. The ultrasonic examination method, including calibration, instrumentation, scanning sensitivity, and coverage, is based on the requirements imposed by ASME BPVC, Section XI, Nonmandatory Appendix I.

5.3.1.4 Special Controls for Ferritic and Austenitic Stainless Steels

The following RGs have specific applicability to the design of the RPV:

RG 1.31, “Control of Ferrite Content in Stainless Steel Weld Metal.”

The BWRX-300 design conforms with RG 1.31. Controls on stainless steel welding are discussed in Subsection 5.2.3.4.2. Control of weld filler metal ferrite content is described in Subsection 5.2.3.4.2 and Subsection 5.2.3.4.4 consistent with Generic Letter 88-01 and NUREG-0313, Revision 2.

RG 1.34, “Control of Electroslag Weld Properties.”

Requirements of RG 1.34 are not applicable. See discussion in Subsection 5.2.3.3.2.

RG 1.43, “Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components.”

RG 1.43 is concerned with cracking of low-alloy steels underneath stainless steel weld deposited cladding. The requirements of this RG are not applicable to the BWRX-300 vessel because the RPV is constructed from SA-508/SA-508M, Grade 3, Class 1, or SA-533/SA-533M, Type B, Class 1 (low-alloy steel forgings or plates), which are produced to fine grain practice. The use of fine grain base metal prevents underclad cracking.

RG 1.44, “Control of the Processing and Use of Stainless Steel.”

The BWRX-300 design conforms with RG 1.44. Sensitization of stainless steel is controlled by the use of service proven low carbon materials and by use of appropriate design and processing steps, including solution heat treatment, control of welding heat input, control of heat treatment during fabrication, verification of non-sensitization per ASTM A262, “Standard Practices for Detecting Susceptibility to Intergranular Attack in Austenitic Stainless Steels,” (Reference 5.3-5) and control of stresses. These controls conform to the guidance of Generic Letter 88-01 and NUREG-0313, Revision 2 as described in Subsection 5.2.3.4.1.

RG 1.50, “Control of Preheat Temperature for Welding of Low-Alloy Steel.”

The BWRX-300 design conforms with RG 1.50. RG 1.50 delineates preheat and interpass temperature control requirements and welding procedure qualifications supplementing those in ASME BPVC, Sections III and IX. See Subsection 5.2.3.3.2 for additional information.

RG 1.71, “Welder Qualification for Areas of Limited Accessibility.”

Conformance with RG 1.71 and qualification for areas of limited accessibility is discussed in Subsection 5.2.3.4.2.

RG 1.99, “Radiation Embrittlement of Reactor Vessel Materials.”

The BWRX-300 design conforms with RG 1.99. Predictions for changes in reference temperature and upper-shelf energy for the BWRX-300 RPV are made in accordance with the requirements of RG 1.99. See Subsection 5.3.1.5 and 5.3.1.6 for additional information.

5.3.1.5 Fracture Toughness

Appendix G to 10 CFR 50 is applied to Class 1 RCPB components and is complied with as discussed in “Methods of Compliance” below and Subsection 5.3.2. The specific pressure-temperature limits for the operation of the BWRX-300 reactor are to be based on 10 CFR 50, Appendix G, Section IV.A.2. The FSAR will furnish bounding pressure-temperature limit curves either as part of the TSs or as part of a BWRX-300 Pressure and Temperature Limits Report.

Subsection 5.2.3.3.1 provides additional information on RPV fracture toughness compliance of ASME BPVC, Section III and 10 CFR 50, Appendix G.

Methods of Compliance

The following items provide details on methods of compliance with 10 CFR 50, Appendix G:

Material Test Coupons and Test Specimens (Appendix G, Section III.A)

Test coupons are removed from the location in each product form as specified in ASME BPVC, Section III, NB-2220. The heat treatment of the test coupons is performed in accordance with NB-2210.

Separate test coupons produced in accordance with NB-2223.3 may be used for forgings.

Location and Orientation of Test Specimens (Appendix G, Section III.A)

The test specimens are located and oriented in accordance with ASME BPVC, Section III, NB-2322. Charpy V-notch impact specimens for testing of plates are to be oriented so that the longitudinal axis is parallel to the surface and normal to the principal rolling direction (not in the thickness direction). For forged material, other than bolting and bars, the Charpy V-notch impact specimens are to be oriented so that the longitudinal axis is parallel to the surface and normal to the principal direction in which the material was worked. Axial specimens are used for bolting and bars.

In regard to 10 CFR 50, Appendix H, the surveillance test material is selected on the basis of the requirements of ASTM E185 “Standard Practice for Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels” and RG 1.99 to provide a conservative adjusted reference temperature for the beltline materials. See Subsection 5.3.1.6.1 for additional discussion on applicable Edition of ASTM E185.

Records and Procedures for Impact Testing (Appendix G, Section III.C)

Preparation of impact testing procedures, calibration of test equipment, and retention of the records of these functions and test data comply with the requirements of ASME BPVC, Section III. Personnel conducting impact testing are qualified by experience, training or qualification testing that demonstrates competence to perform tests in accordance with the testing procedure.

Charpy-V Curves for the RPV Beltline (Appendix G, Section III.A and Section IV.A.1)

In addition to the requirements of ASME BPVC, Section III, NB-2331, a full transverse Charpy-V curve is determined for heats of base material and weld metal used in the core beltline region with a minimum of three (3) specimens tested in the upper-shelf region in accordance with E185. The minimum initial upper-shelf energy level for base material and weld metal in the beltline region meets or exceeds 102 J (75 ft-lbf) and is predicted to maintain an upper-shelf energy of greater than 68 J (50 ft-lbf) throughout the life of the vessel, as required by 10 CFR 50, Appendix G, Section IV.A.1.

In regard to 10 CFR 50, Appendix G, Section III.A, separate, unirradiated baseline specimens per ASTM E185, Paragraph 5.4.1, are used to determine the reference temperature curve of the core beltline base material and weld metal.

Bolting Material

All bolting material exceeding 25 mm (1 in.) diameter has special material requirements as presented in Table 5.3-1.

Fracture Toughness Margins in the Control of Reactivity (Appendix G, Section IV.A)

ASME BPVC, Section XI, Nonmandatory Appendix G, is used in determining pressure-temperature limitations for phases of normal plant operation including AOOs.

5.3.1.6 Material Surveillance

The reactor vessel materials surveillance program monitors changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from exposure to neutron irradiation and thermal environment.

The objective of the BWRX-300 reactor vessel material surveillance program is to provide sufficient material data to (a) monitor irradiation embrittlement at the end of the period of extended operation and (b) determine the need for operating restrictions on the inlet temperature, neutron spectrum, and neutron flux for material exposed to a neutron fluence of 1.0×10^{17} n/cm^2 (E > 1 MeV) or greater for which impact tests are required. The extended core beltline region is the region that is exposed to neutron fluence at or above this neutron fluence threshold. The surveillance test program specimens are used to monitor the radiation-induced changes in the mechanical properties of the extended core beltline region materials of the RPV. Part of the specimens which are exposed to irradiation are to be installed in removable specimen capsules at the inside vessel wall opposite the active core. The remaining unirradiated baseline specimens are to be used to establish material reference data.

Material surveillance program data is to be used as described in 10 CFR 50, Appendix G, Section IV.

5.3.1.6.1 Compliance with Reactor Vessel Material Surveillance Program Requirements

Reactor vessel materials surveillance specimens are provided in accordance with requirements of ASTM E185 and 10 CFR 50, Appendix H "Reactor Vessel Material Surveillance Program Requirements." 10 CFR 50, Appendix H, Paragraph I, incorporates by reference ASTM E185-82 "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels" which is referenced in various subsections of Appendix H. Justification for use of the 2021 Edition of ASTM E185 will be provided in the FSAR.

Materials for the BWRX-300 materials surveillance program are selected to represent materials used in the extended beltline region. Specimens are manufactured from forgings used in the extended beltline region of the actual RPV and a weld typical of those in the beltline region and thus represent base metal and weld material. The assessment of the weld HAZ material is not to be included as it is considered optional in the 2021 Edition of ASTM E185 and 10 CFR 50, Appendix H. The base metal and weld are heat treated in a manner that simulates the actual heat treatment performed on the beltline region of the completed vessel. Four capsules provide the ability to monitor the RPV over its predicted 60-year Maximum Design Fluence. Each in-reactor surveillance capsule is to contain the minimum number of test specimens specified in ASTM E185-21 Subsection 5.4.2, including Charpy V notch, tensile, and fracture toughness specimens taken from base metal forgings that are located within the reactor beltline region and the weld material, as required. Each capsule is to also include fracture toughness specimens as detailed in ASTM E185-21. A set of out-of-reactor beltline Charpy V notch specimens, tensile specimens, and archive material are provided with the surveillance test specimens. Neutron dosimeters and temperature monitors are located within the capsules as required by ASTM E185-21.

The following proposed withdrawal schedule is based on Maximum Design Fluence from ASTM E185-21 to address the 60-year design life:

- First capsule: at 1/4 of Maximum Design Fluence
- Second capsule: at 1/2 of Maximum Design Fluence
- Third capsule: at 3/4 of Maximum Design Fluence
- Fourth capsule: at Maximum Design Fluence

Fracture toughness testing of irradiated capsule specimens is to be performed in accordance with the requirements of ASTM E185-21.

5.3.1.6.2 Neutron Flux and Fluence Calculations

An analysis of the neutron irradiation of the RPV has been performed and results of estimated peak fluence for the vessel base material and of the limiting weld are provided in Table 5.3-4. These peak fluence values are higher than the surveillance program threshold for peak fluence at the end of design life, but lower than the threshold value for causing IASCC as discussed in Subsection 5.2.3.2.1. These peak fluence values are also within the fluence range in which fully ductile fracture methods can be used for evaluating austenitic stainless steel. Discussion on predicting shift in reference temperature and drop in upper-shelf energy from these estimated peak fluence values are provided in Subsection 5.3.2.

A description of the methods of analysis and information on code validation is contained in Subsection 4.1.6.1.

5.3.1.6.3 Predicted Irradiation Effects on Beltline Materials

Reference temperature changes and changes in upper-shelf energy are calculated in accordance with the rules of RG 1.99. Reference temperatures are established in accordance with 10 CFR 50, Appendix G, and ASME BPVC, Section III, NB-2330.

Because weld material chemistry and fracture toughness data are not available at this time, the material residual limits in Table 5.3-1 were used to estimate worst case irradiation effects in accordance with RG 1.99.

These estimates for the adjusted reference temperature and upper-shelf energy at end of life for the beltline weld and forging are provided in Table 5.3-2.

5.3.1.6.4 Positioning of Surveillance Capsules and Methods of Attachment (Appendix H. Section III.B.2)

The BWRX-300 surveillance specimen capsules are to be located within the RPV around the periphery of the core near the inside wall of the beltline region and in accordance with ASTM E185-21. A description of the minimum and maximum planned lead factor will be provided in the FSAR. The sealed capsules are not attached to the RPV but secured in surveillance sample holders. The surveillance sample holders allow the removal and reinsertion of capsule sample holders. The surveillance sample holders are designed, fabricated, and analyzed to the requirements of ASME BPVC, Section III, Subsection NG.

5.3.1.7 Reactor Vessel Fasteners

The reactor vessel closure head (flange) is fastened to the reactor vessel shell flange by multiple sets of threaded studs and nuts. The lower end of each stud is installed in a threaded hole in the vessel flange. A nut and washer are installed on the upper end of each stud. The proper amount of preload can be applied to the studs by sequential tensioning.

Hardness tests are performed on RPV main closure bolting to demonstrate that heat treatment has been properly performed.

The BWRX-300 design conforms with RG 1.65. RG 1.65 defines acceptable materials and testing procedures for Light Water Reactor (LWR) vessel bolting, which includes stud, nuts, and washers. Materials and inspections for the RPV main closure bolting are in conformance with RG 1.65.

The design and analysis of RPV bolting materials are in compliance with ASME BPVC, Section III, Class 1, requirements. The RPV main closure bolting material(s) are specified in Table 5.2-2a. The bolting is to undergo NDE per ASME BPVC, Section III, NB-2580 before being placed into service. The NDE includes Visual Examination, Magnetic Particle or Liquid Penetrant Testing (PT), and Ultrasonic Testing (UT) examination. The maximum yield strength is specified in Table 5.3-1. The lateral expansion and the Charpy V values are specified in Table 5.3-1.

The RPV main closure bolting is not metal-plated. Bolting material may have a manganese phosphate (or other acceptable) surface treatment.

Lubricants are permissible provided that they are stable at operating temperatures and are compatible with the bolting and RPV materials and with the surrounding environment. Lubricants are to contain low levels of halogens, sulfur, lead, or other low melting point metals consistent with proven lubricants used in similar nuclear applications. Lubricants containing molybdenum sulfide (disulfide or polysulfide) should not be used for any Safety Class (SC) applications. Lubricants are to be approved by GVH prior to use.

During the venting and filling of the RPV and while the head is removed, the studs and stud holes in the RPV flange are adequately protected from corrosion and contamination.

5.3.2 Pressure-Temperature Limits

The regulations requiring the imposition of pressure-temperature limits on the RCPB are the following:

10 CFR 50.55(i) requires that SSCs subject to the codes and standards in 10 CFR 50.55a must be designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed. In addition, GDC 1, requires that the codes and standards used to ensure quality products in keeping with the safety function be identified and evaluated to determine their adequacy. .

10 CFR 50, Appendix A, GDC 14, requires that the RCPB be designed, fabricated, erected, and tested in order to have an extremely low probability of abnormal leakage, of rapid failure, and of gross rupture. Likewise, GDC 31, requires, in part, that the RCPB be designed with sufficient margin to assure that when stressed under operating, maintenance and testing, and postulated accident conditions, the boundary behaves in a non-brittle manner, and the probability of rapidly propagating fracture is minimized. In order to assess the structural integrity of the reactor vessel, GDC 32 requires, in part, an appropriate materials surveillance program for the reactor vessel beltline region.

The acceptability of the BWRX-300 RCPB pressure-temperature limits is demonstrated by meeting the relevant requirements of the following Commission regulations:

10 CFR 50, Appendix A, GDC 1, as it relates to quality standards for design, fabrication, erection, and testing, by compliance with ASME BPVC, Section XI, Nonmandatory Appendix G.

10 CFR 50, Appendix A, GDC 14, as it relates to assuring an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture of the RCPB, by compliance with 10 CFR 50, Appendix G.

10 CFR 50, Appendix A, GDC 31, as it relates to assuring that the RCPB behaves in a non-brittle manner and the probability of rapidly propagating fracture is minimized, by compliance with 10 CFR 50, Appendix G and RG 1.99.

10 CFR 50, Appendix A, GDC 32, as it relates to the reactor vessel materials surveillance program, by compliance with 10 CFR 50, Appendix H and ASTM E185 as discussed in Subsection 5.3.1.6.

10 CFR 50.55a, as it relates to quality standards for design, and determination and monitoring of material fracture toughness, by compliance with ASME BPVC, Section XI, Nonmandatory Appendix G.

10 CFR 50.60, as it relates to compliance with the requirements of 10 CFR 50, Appendices G and H. Based on the estimated irradiation effects on RPV beltline materials discussed in Subsection 5.3.2.1, it is expected that 10 CFR 50.60 is to be met with acceptable upper-shelf energy values throughout the life of the vessel.

10 CFR 50, Appendix G, as it relates to material testing and fracture toughness. Based on the estimated irradiation effects on RPV beltline materials discussed in Subsection 5.3.2.1, it is expected that 10 CFR 50, Appendix G is to be met with acceptable upper-shelf energy values throughout the life of the vessel.

5.3.2.1 Limit Curves

Pressure-temperature limit curves are based on the requirements of 10 CFR 50, Appendix G and RG 1.99. In addition to the FSAR furnishing bounding pressure-temperature limit curves as

discussed in Subsection 5.3.1.5, the FSAR will furnish the methodology used to determine these limits.

The ferritic materials in the RCPB, including pressure-temperature limits, will comply with the requirements of 10 CFR 50, Appendix G. 10 CFR 50, Appendix G specifically requires that pressure-temperature limits must be at least as conservative as limits obtained by following ASME BPVC, Section XI, Nonmandatory Appendix G during heatup, cooldown and test conditions.

The reactor vessel shells, RPV head and flange areas, FW nozzles, bottom head and the core beltline areas are evaluated using the material initial reference temperature data from the RPV specification. The operating pressure-temperature limit curves are based on the most limiting locations. The pressure-temperature limits are based on flaw sizes specified in Paragraph G-2120 of ASME BPVC, Section XI, Appendix G. The maximum through wall temperature gradient from continuous heating or cooling based on the average rate of change of reactor coolant temperature during heatup and cooldown is considered (for the BWRX-300, the rate of change value is specified in Table 5.3-1). The safety factors applied are as specified in ASME BPVC, Section XI, Appendix G.

To calculate the adjusted reference temperature (accounting for the effects of irradiation in the vessel beltline region), the copper and nickel specification limits are used in combination with the peak fluence values and the methodology of RG 1.99. This is considered conservative because the actual reference temperature values and chemical composition are normally much lower than the ones specified. Margins for the adjusted reference temperature calculation are as defined in RG 1.99.

The methodology for determining the stress (pressure and thermal) for the transient events for normal and upset conditions will be provided in the FSAR.

Temperature Limits for Inservice Inspection Hydrostatic and Leak Pressure Tests

Bounding pressure-temperature limit curves using BWRX-300 plant specific data such as materials, fluences and stresses will be developed and documented as appropriate in TS requirements.

Operating Limits During Heatup, Cooldown, and Core Operation

Bounding pressure-temperature limit curves using BWRX-300 plant specific data such as materials, fluences and stresses will be developed and documented as appropriate in TS requirements.

Reactor Vessel Annealing

In-place annealing of the reactor vessel, because of radiation embrittlement, is not necessary because the vessel is predicted to maintain an equivalent safety margin in accordance with the procedures of 10 CFR 50, Appendix G, Section IV.A.

Predicted Shift in Reference Temperature and Drop in Upper-Shelf Energy

For design purposes, the adjusted reference nil ductility temperature and drop in the upper-shelf energy for the BWRX-300 vessel is predicted in accordance with the requirements of RG 1.99. The estimated irradiation effects on RPV beltline materials are provided in Table 5.3-2.

The calculations are based on the limits on copper and nickel in the weld and forging material specified for the BWRX-300 in Table 5.3-1. The reference nil ductility temperature for select reactor components including the core beltline shell course is also provided in Table 5.3-1.

As required by 10 CFR 50, Appendix H, a surveillance program is to be conducted in accordance with the requirements of ASTM E185. The surveillance program is to include specimens of base metal and weld metal of the beltline forging. Subsection 5.3.1.6 provides additional detail on the surveillance program.

5.3.3 Reactor Vessel Integrity

This subsection provides a special summary discussion of factors relating to the integrity of the reactor vessel. The information in each area is discussed to ensure that the information is complete, and that no inconsistencies in information or requirements exist that would reduce the certainty of vessel integrity.

Design

Compatibility of the reactor vessel design with established quality standards for material properties and fabrication methods is described in Subsection 5.3.1. The basic design of the reactor vessel establishes compatibility with required inspections as described in Subsection 5.2.4. Selected RPV dimensions are provided in Table 5.3-3.

ASME BPVC, Section XI, Nonmandatory Appendix G is applied as a mandatory procedure for demonstrating protection against nonductile failure. The criteria of 10 CFR 50, Appendix G are interpreted as establishing the requirements of annealing. 10 CFR 50, Appendix G, Section IV.B requires the vessels to be designed for annealing of the beltline only where the existence of an adequate safety margin cannot be demonstrated in accordance with 10 CFR 50, Appendix G, Section IV.A. The BWRX-300 vessel is predicted in Table 5.3-2 to maintain an adequate safety margin throughout the life of the vessel; therefore, design for annealing is not required.

For further discussion of fracture toughness of the RPV, refer to Subsections 5.3.1.5 and 5.3.2.

Reactor vessel design data, including reactor vessel design pressure, design temperature and hydrostatic test pressure are presented in Table 5.3-1.

Materials of Construction

The materials of construction are described in Subsection 5.2.3 and in Subsection 5.3.1. Conformance with RG 1.99 is discussed in Subsections 5.3.1 and 5.3.2.

Fabrication Methods

The processes used to fabricate the reactor vessel, including forming, welding, cladding, and machining, are described in Subsection 5.3.1.

Inspection Requirements

The fabrication inspection and test methods and requirements are described in Subsection 5.3.1.

Shipment and Installation

Protective measures taken during shipment of the reactor vessel and its installation at the site verify that the as-built characteristics of the reactor vessel are not degraded by improper handling. See Subsection 5.3.3.4 for additional information.

Operating Conditions

Operating conditions as they relate to the integrity of the reactor vessel are considered in Subsection 5.3.2 as discussed in Subsection 5.3.3.5.

Inservice Surveillance

Plans and provisions for inservice surveillance of the reactor vessel are described in Subsection 5.3.1.6.

Operational Programs

The basic acceptance criteria for each review area listed above are covered by other subsections, so they are discussed here only in general terms. References are made to the subsections that include detailed criteria. The acceptance criteria in these subsections describe methods that meet the requirements of the following Commission regulations:

- 10 CFR 50, Appendix A (GDC 1, GDC 4, GDC 14, GDC 30, GDC 31, and GDC 32)
- 10 CFR 50, Appendix B
- 10 CFR 50.60 and associated Appendices G and H
- 10 CFR 50.55a

The Quality Assurance Program described in Chapter 17 ensures the following:

- Reactor vessel materials, equipment, and services associated with the reactor vessel and appurtenances conform to design document requirements and enforce measures to ensure conformance, including (1) provisions for source evaluation and selection, (2) objective evidence of quality furnished, (3) inspection at the vendor source and (4) examination of the completed reactor vessel.
- Inspection surveillance of the reactor vessel fabricator in-process manufacturing, fabrication, and testing operations is performed in accordance with written approved inspection procedures
- Adequate documentary evidence that the reactor vessel material, manufacture, testing, and inspection conforms to the specified quality assurance requirements contained in the design documents

5.3.3.1 Materials of Construction

Material used in the construction of the RPV comply with the requirements of ASME BPVC, Section II. The vessel heads, shells, flanges, and major nozzles are fabricated from low-alloy steel purchased in accordance with ASME Specifications SA-533/SA-533M Type B, Class 1 plates, and SA 508/SA-508M, Grade 3, Class 1 forgings. Interior surfaces of the vessel are clad with austenitic stainless steel or Ni-Cr-Fe weld overlay as discussed in Subsection 5.2.3.2.2. The core beltline limiting forging is made of SA 508, Grade 3, Class 1 material. The RPV head fasteners are described in Subsections 5.3.1.7.

These construction materials were selected because they provide adequate strength, fracture toughness, fabricability, and compatibility with the BWR environment. Their suitability has been demonstrated by long-term successful OPEX in reactor service.

5.3.3.2 Fabrication Methods

The RPV is a vertical cylindrical pressure vessel of welded construction with a bolted top head fabricated in accordance with ASME BPVC, Section III, Class 1 requirements. The shell, RPV head, flanges and major nozzles are fabricated from low-alloy steel forgings or plates. The shell forgings are joined by circumferential welds only. The length of the shell forgings is chosen to minimize the number of circumferential welds. Welding performed to join these vessel components is in accordance with procedures qualified to ASME BPVC, Section III and Section IX requirements. Weld test samples support each procedure for full penetration welds. Fabrication of the RPV is performed in accordance with GVH-approved drawings, fabrication procedures, and test procedures in accordance with the Quality Assurance Program described in Chapter 17.

Submerged arc, gas metal arc, gas tungsten arc and shielded metal arc welding processes may be employed. Electroslag welding is not used except for cladding. Preheat and interpass temperatures employed for welding of low-alloy steel is discussed in Subsection 5.2.3.3.2, Subsection 5.3.1.2, and Subsection 5.3.1.4. Post-weld heat treatment of low-alloy welds is presented in Table 5.3-1.

Other fabrication processes such as cutting, bending, and forming are performed in accordance with the vessel design documents.

5.3.3.3 Inspection Requirements

Plates, forgings, and bolting are ultrasonically tested, and surface examined by magnetic particle methods or liquid penetrant methods as required by ASME BPVC, Section III, Subsection NB. Welds on the RPV are examined in accordance with methods prescribed and meet the acceptance requirements specified by ASME BPVC, Section III, Subsection NB. In addition, the pressure-retaining welds are ultrasonically examined using acceptance standards that are equivalent to or more restrictive than required by ASME BPVC, Section XI.

5.3.3.4 Shipment and Installation

Handling, storage, shipping, and cleaning complies with the GVH quality assurance program. Installation is performed in accordance with the GVH quality assurance program .

5.3.3.5 Operating Conditions

Controls on plant operation are to be implemented to hold thermal stresses within acceptable ranges and to meet the bounding pressure-temperature limits. A limit on rate of change of reactor coolant temperature is imposed per Table 5.3-1, which assures that the vessel closure, closure studs, vessel support, CRD housing, and stub tube stresses and fatigue usage remain within acceptable limits.

These operational limits, when maintained, ensure that the stress limits within the reactor vessel and its components are within the thermal limits to which the vessel was designed for normal operating conditions.

5.3.4 References

5.3-1 ASME BPVC, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” American Society of Mechanical Engineers, 2021.

5.3-2 ASME BPVC, Section IX, “Welding, Brazing, and Fusing Qualifications,” American Society of Mechanical Engineers, 2021.

5.3-3 ASME BPVC, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” Division 1, “Rules for Inspection and Testing of Components of Light-Water-Cooled Plants,” American Society of Mechanical Engineers, 2019.

5.3-4 ASME BPVC, Section II, “Materials - Part C - Specifications for Welding Rods, Electrodes, and Filler Metals,” American Society of Mechanical Engineers, 2021.

5.3-5 ASTM A262, “Standard Practices for Detecting Susceptibility to Intergranular Attach in Austenitic Stainless Steels,” ASTM International, 2021.

5.3-6 ASTM E185-21, “Standard Practice for Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels,” ASTM International, 2021.

Table 5.3-1: Reactor Pressure Vessel Material Controls

Component	Control(s)
Specified limits for RPV materials used in the core beltline limiting forging. (See Table 5.2-2a and Table 5.2-2b Notes, Table 5.3-2, Subsection 5.2.3.3.1, Subsection 5.3.1.2, Subsection 5.3.1.5, and Subsection 5.3.2.1)	Base Material Residual Elements: 0.05% maximum copper, 0.006% maximum phosphorous, and 0.010% maximum sulfur Weld Material Residual Elements: 0.05% maximum copper, 1.0% maximum nickel, 0.008% maximum phosphorous, 0.010% maximum sulfur, and 0.05% maximum vanadium
Studs, nuts, and washers for the main closure flange (See Subsections 5.3.1.2 and 5.3.1.7)	Material specification per Table 5.2-2a having maximum yield strength level of 1034 MPa (150 ksi)
RPV post-weld heat treatment of low-alloy steel welds (See Subsections 5.2.3.3.2, 5.3.1.2, and 5.3.3.2)	593°C (1100°F) minimum and not exceeding 635°C (1175°F) is applied to low-alloy steel welds in accordance with NB-4622. Specific post-weld time and temperature requirements are provided in NB-4622
Toughness of bolting material exceeding 25 mm (1 in.) diameter (See Subsections 5.2.3.3.1, 5.3.1.5, and 5.3.1.7)	Minimum of 61 J (45 ft-lbf) Charpy upper-shelf energy and 0.64 mm (0.025 in.) lateral expansion at the minimum bolt preload temperature per NB-2333 for ASME Class 1 bolting and at or below lowest service temperature per NCD-2332.3 for ASME Class 2 bolting The 61 J (45 ft-lbf) requirement of the ASME BPVC that applies to bolts over 101.6 mm (4 in.) in diameter, is conservatively applied to nominal bolt diameters exceeding 25 mm (1 in.)
RPV Design Data (See Subsection 5.3.3)	The RPV design pressure is 10.342 MPa gauge (1500 psig) and the design temperature is 314.4°C (598°F). The preservice hydrostatic test pressure is 1.25 times the design pressure per NB-6221
Average rate of change of reactor coolant temperature during heatup and cooldown (See Subsection 5.3.2.1)	Not to exceed 111.1°C (200°F) during any one-hour period
Reference nil ductility temperature (See Subsection 5.3.2.1)	-20°C for core beltline shell course -25°C for nozzle forgings, except nozzles integral with shell forgings -12°C for upper shell courses and head

Table 5.3-2: Predicted Irradiation Effects on Beltline Materials

Parameter	Value
Adjusted reference temperature at end of life for the weld (at ¼ wall thickness)	< 21.0°C (69.8°F)
Adjusted reference temperature at end of life for the vessel beltline forging (at ¼ wall thickness)	< 15.5°C (60.0°F)
Calculated shift in reference temperature for welds	15.6°C (28.1°F)
Calculated shift in reference temperature for beltline forging	12.1°C (21.9°F)
Predicted drop in upper shelf energy for welds	11.2 J (8.3 ft-lbf)
Predicted drop in upper shelf energy for beltline forging	15.3 J (11.3 ft-lbf)
The end-of-life upper shelf energy	> 86.7 J (63.9 ft-lbf)

Table 5.3-3: Reactor Vessel Selected Dimensions

Dimension (See Notes 1 and 2)	Value (mm)
Minimum inner diameter	4204
Nominal wall thickness including cladding (shell just below the vessel support skirt)	135
Nominal inner height (to the inside top of the reactor closure head)	27297.8
Nominal height at top of top guide	9260
Nominal height at top of core plate	4858

Notes:

1. Refer to Figure 4.1-1 for a depiction of reactor components and internals.
2. Elevation zero is located at the inside bottom of the reactor bottom head. Height dimensions provided in this table are relative to elevation zero.

Table 5.3-4: Reactor Vessel Estimated Fluence Values

Parameter (See Notes 3 and 4)	Value for 60 Years for Energies >1 MeV (n/cm^2)
Estimated peak neutron fluence at inside surface of RPV beltline shell (Note 1)	4.63×10^{18}
Estimated peak neutron fluence at inside surface of limiting RPV beltline shell weld (Note 1)	1.33×10^{18}
Estimated peak neutron fluence at RPV beltline shell 1/4 wall thickness (Note 2)	3.43×10^{18}
Estimated peak neutron fluence at limiting RPV beltline shell weld 1/4 wall thickness (Note 2)	9.85×10^{17}

Notes:

1. The estimated fluence at the inside surface is obtained from direct fluence analysis results.
2. The estimated fluence at 1/4 wall thickness is obtained using the procedure in Regulatory Position 1.1 of RG 1.99 with the corresponding estimated fluence at the inside surface reported in this table.
3. Surveillance program neutron fluence threshold for peak fluence at the end of design life is provided in Subsection 5.3.1.6.
4. Neutron fluence threshold for IASCC is provided in Subsection 5.2.3.2.1.

5.4 Reactor Coolant System Component and Subsystem Design

This section pertains to various components and subsystems associated with the RCS. Design bases, descriptions, evaluations, and necessary tests and inspections for the components or subsystems, including radiological considerations, are discussed.

The BWRX-300 instrumentation consists of sensors to measure and monitor parameters including reactor water level, reactor coolant pressure, flow, and temperature, and incore (nuclear) instrumentation to measure neutron flux for reactor power. Additionally, there are sensors to measure and monitor pressures, flows, and temperatures for leakage detection purposes as discussed in Subsection 5.2.5. Reactor water level instrumentation, reactor coolant pressure, flow, and temperature are discussed in Subsection 5.2.5, Subsection 5.4.4, and Subsection 5.4.5. Nuclear instrumentation is discussed in Subsection 4.1.5.

The BWRX-300 also has control requirements for valves (e.g., RIVs, CIVs, drain valves and head vent valves) that are repositioned as the plant is operated. Sensor and detector signals are made available to the applicable Instrumentation and Control (I&C) systems. The I&C architecture and associated systems and components are discussed in Section 7.3.

5.4.1 Reactor Coolant Pumps or Circulation Pumps

The BWRX-300 design does not include reactor coolant pumps or circulation pumps. The BWRX-300 relies on buoyancy forces within the reactor vessel to cause recirculation of reactor coolant through the core. There is no forced reactor coolant recirculation system for the BWRX-300. The thermal and hydraulic performance of natural recirculation within the reactor core is discussed in Section 4.4. Pump flywheels are not applicable to the BWRX-300 as the BWRX-300 has no reactor coolant pumps or circulation pumps.

5.4.2 Steam Generators

The BWRX-300 design does not include a steam generator.

5.4.3 Reactor Coolant System Piping and Valves

The RCS is comprised of the SC1 portion of the BWRX-300 NBS and the SC1 portion of the CFS. The RCS extends to and includes the outermost CIV in the MS and FW piping. The NBS includes the RPV and provides MS flow paths from the RPV to the steam turbine, and the CFS provides FW flow paths to the RPV. The RPV, RIVs, and CIVs are major components of the RCS. The RCS is depicted in Figure 5.1-1. The RCS maintains adequate core cooling conditions (i.e., pressure, temperature, and coolant flow rate) for the fuel during power operation.

RIVs and CIVs are discussed with their respective connected systems in this chapter and other referenced chapters as applicable.

Reactor Isolation Valves

A feature of the BWRX-300 is the integral RIVs that are directly flanged to the RPV nozzles on the reactor vessel (see Figure 5.4-1). The design of the integral RIVs consists of two valves in a single body that are independently able to isolate the respective line. RIVs on each RPV nozzle are designed and qualified to meet required testing criteria for the RPV isolation functions.

The RIVs allow Loss-of-Coolant Accidents (LOCAs) to be quickly isolated, thereby, preserving the fluid inventory in the RPV for core cooling.

Each RPV nozzle with an internal bore greater than DN20 (3/4 inch nominal pipe size) is required to have an integral RIV flanged directly to the RPV nozzle. The RIVs are mechanically redundant and have diverse I&C systems. The RPV automatic isolation concept consists of RIVs with two valves in series that are single failure proof with each of the valves independently able to isolate

the line. RIVs are provided for MS lines, FW lines, ICS lines, CUW lines, and the RPV head vent line.

The RIVs operate via hydraulic actuators. An independent hydraulic system provides motive force for the RIV actuators. Except for the ICS RIVs, the RIVs fail closed and maintain a closed position by positive mechanical means upon loss of hydraulic power.

As demonstrated by the BWRX-300 design characteristics listed below, a break in the RPV nozzles, RIV bodies, the four associated flanges, or their flanged connections is highly unlikely. Thus, postulation of breaks for 10 CFR 50.46 and GDC 35, as well as consideration of GDC 4 dynamic effects, at these locations is excluded from the BWRX-300 design bases.

- These locations are robustly designed as follows:
  - RIVs and their flanges consist of a single integral forging with no welds and are designed in accordance with ASME BPVC, Section III, NB-3200, NB-3500; ASME B16.10 and ASME B16.34.
  - The flanges of the RIVs, RPV nozzles and valve-to-pipe transition flange are designed in accordance with ASME BPVC, Section III, NB-3200, NB-3300 and ASME B16.5.
  - Bolting is designed in accordance with ASME BPVC, Section III, NB-3200.
  - The materials of construction for the RIVs and their associated flanges are selected to resist degradation mechanisms and fabricated by forging methods to minimize defects and their propagation. The RPV nozzle and nozzle flange is a single integral forging with no welds. Bolting materials follow the guidance of RG 1.65, Rev. 1.
  - Stress and fatigue analyses are performed for the RIVs, their associated flanges and bolting in accordance with ASME BPVC, Section III, NB-3200.
  - All applicable loads are considered, and limits established, in accordance with ASME BPVC, Section III, NCA-2140.
  - All RIV bodies are surface and volumetrically inspected in accordance with ASME BPVC, Section III during fabrication.
- The BWRX-300 design includes provisions for prevention and detection of degradation as follows:
  - As described in Subsection 5.2.3.2.1, water chemistry control is established with input from EPRI BWRVIP-190, BWR Water Chemistry Guidelines.
  - As described in Subsection 5.2.5, detection of leakage from the RIVs and their associated flanges is provided by multiple methods for monitoring leakage inside containment (e.g., containment sump monitoring, particulate and noble gas radiation monitoring, temperature monitoring). The response time of the leakage detection instrumentation inside containment is consistent with RG 1.45.
  - Preservice and inservice inspections of the RIVs and their associated flanges including all flange bolting are conducted in accordance with the requirements of ASME BPVC, Section XI. RIV inboard and outboard flange bolting is inspected during each examination cycle and cannot be deferred until valve disassembly or the end of an inspection interval. All RIV flange bolting is volumetrically examined and no exemptions for size or sampling are used.

- A preliminary probabilistic risk assessment of rupture at the RIVs and their associated flanges has been performed and concludes that the frequency of a break at these locations (i.e., the sum of breaks at the flanged RPV/RIV connections and ruptures of the RIV bodies for all large break locations) is less than 1E-05 per reactor-year, with consideration of uncertainty.

RCS Containment Isolation Valves

The MS and FW CIVs operate via hydraulic actuators. The CIVs fail closed and maintain a closed position by positive mechanical means upon loss of hydraulic power.

Containment isolation function is discussed in Subsection 6.2.4.

RCS Piping and Valves Methods of Analysis and Structural Integrity Considerations

Subsections 3.9.1, 3.9.2, and 3.9.3, provide methods of analysis and structural integrity considerations for RCS piping and valves. Subsection 3.9.6 provides information on the functional design, qualification, and IST of RCS valves. Section 3.10 provides information on seismic and dynamic qualification of RCS valves, associated electrical equipment, and I&C systems. Section 3.12 provides information on the design and analysis, and testing, of RCS piping, and associated piping supports. Section 3.11 provides information on environmental qualification of mechanical and electrical equipment.

Subsection 5.2.3 provides discussion of the evaluation of RCPB piping and valve materials. Subsection 5.2.3 also discusses provisions taken during design, fabrication, and operation to control factors that contribute to stress corrosion cracking.

Subsection 5.2.4 describes performance of preservice inspection and ISI for RCS piping and valves with requirements consistent with ASME BPVC, Section XI.

Section 6.3 addresses the Emergency Core Cooling System (ECCS) function of the ICS. Section 6.6 discusses preservice inspection and ISI of ASME Class 2 and Class 3 components.

5.4.4 Main Steam Line Flow Restrictors

A MSL flow restrictor is installed in each of the two RPV MS nozzles. The flow restrictor limits steam flow in the event of a break in the MSL subsystem.

5.4.4.1 Design Bases

Safety Category 1 Functions

- Limit steam flow following an MSL break
- Along with MSRIVs and MSCIVs, prevent uncovering the core following an MSL break
- Limit the amount of radiological release outside the PCS prior to MS isolation via MSRIVs and MSCIVs

Safety Category 2 Functions

None.

Safety Category 3 Functions

None.

Safety Category N Functions

None.

5.4.4.2 System Description

The MS consists of two MSLs. Figures 5.1-2 and 5.4-5 provide simplified diagrams of the MS subsystem.

An MS flow restrictor is installed in each of the two RPV MS nozzles. The flow restrictor limits steam flow in the event of an MSL break. The flow restrictor, together with the fast-closing MSL RIVs (described in Subsection 5.4.5), prevent uncovering the core following an MSL break.

The MS nozzle has pressure taps connected to the throat of the flow restrictor for measuring the steam flow/differential pressure through the nozzle. Differential pressure instrumentation connected to the flow restrictor provides an isolation signal for MSL isolation valve closure.

The MS flow restrictor is designed in accordance with ASME-RCFM, “Fluid Meters Their Theory and Application Report of ASME Research Committee on Fluid Meters,” (Reference 5.4-1) and has no moving parts.

5.4.4.3 Safety Evaluation

Analysis of the MSL break accident is provided in Subsection 15.5.4.5. Additional information on accident analysis involving the MSL flow restrictors (referred to in Chapter 15 as flow limiters) is provided in Section 15.5.

The MS flow restrictors meet material and construction standards as described in Subsection 5.2.3.

5.4.4.4 Testing and Inspection Requirements

Because the flow restrictor forms a permanent part of the RPV steam outlet nozzle, and has no moving components, no testing program beyond RPV ISI is planned.

5.4.4.5 Instrumentation Requirements

The MS flow instruments are differential pressure Flow Transmitters (FTs) used to determine MS flow for the purpose of MS isolation and reactor water level control. The high-pressure input for the transmitters is the steam dome pressure, taken from the reference leg instrument piping. The low-pressure input is from the flow restrictor taps located in the MS nozzles. The MS nozzle has pressure taps connected to the throat of the flow restrictor for measuring the steam flow/differential pressure through the nozzle.

5.4.5 Main Steam and Feedwater Isolation

The MS isolation function is accomplished by the MSRIVs and MSCIVs using associated I&C systems as discussed in Subsection 6.2.4 and Section 7.3.

The FW isolation function is accomplished by the FWRIVs and FWCIVs using associated I&C systems as discussed in Subsection 6.2.4 and Section 7.3.

As noted in Section 6.7, MSL Isolation Valve Leakage Control System is not applicable to the BWRX-300.

5.4.5.1 Design Bases

Safety Category 1 Functions

- MSL and FW line RPV and Containment isolation

Safety Category 2 Functions

None.

Safety Category 3 Functions

- MSL and FW line isolation when required during plant shutdown

Safety Category N Functions

- These functions for MS and FW are described in Sections 10.3 and 10.4.7, respectively.

The Safety Classifications, Quality Groups, and Seismic Categories of the MS and FW Isolation SSCs are listed in Table 3A-1.

5.4.5.2 System Description

Reactor Isolation Valves

As described in Subsection 5.4.3, the RPV is equipped with RIVs that are attached directly to the RPV. The RPV RIVs include the MSRIVs and FWRIVs.

Main Steam and Feedwater Containment Isolation Valves

To ensure containment integrity at the PCS penetrations, the MS and FW containment piping and isolation valves are designed to ASME BPVC, Section III, Division 1, Subsection NCD, and Quality Group B requirements.

Containment isolation function and valves are discussed in Subsection 6.2.4.

Main Steam Subsystem

The MS subsystem consists of two MSLs. Each MSL is equipped with an MSRIV consisting of two independent isolation valves in a single body and an MSCIV. Figures 5.1-2 and 5.4-5 provide simplified diagrams of MS. Only the portion of the MSL from the MSRIV up to and including the MSCIV is addressed in Chapter 5, while the remainder of the MS subsystem is addressed in Section 10.3.

Main Steam Isolation Function

The MSRIVs and MSCIVs provide Safety Category 1 isolation of the PCS in the event of accidents or other conditions and mitigate the release of containment contents.

As described in Subsection 5.4.4, an MS flow restrictor is installed in each of the two RPV MS nozzles. The flow restrictor limits steam flow in the event of a break in an MSL. The flow restrictor, together with the fast-closing MSRIVs, prevent uncovering the core following a MSL break.

The MSRIVs and MSCIVs also provide isolation of the MSLs when required during plant shutdown condition.

Feedwater Subsystem

Downstream of the FW heaters, the FW line divides into two lines that penetrate the PCS. Each line then splits to provide four connections to the RPV. Each of the four RPV connections is equipped with a FWRIV. Both PCS penetrations are equipped with a FWCIV located outside the PCS. Figure 5.1-2 provides a simplified diagram of the NBS, and system interfaces located in the RB. The CFS and the FW subsystems are addressed in Subsection 10.4.7.

Feedwater Isolation Function

The FWRIVs and FWCIVs provide the Safety Category 1 isolation function of the PCS in the event of accidents or other conditions.

The CFS SC1 I&C System provides isolation signals, indication, and alarms for the FW containment isolation valves and FW system isolation valves for loss of FW flow and leak detection.

5.4.5.3 Safety Evaluation

The MS and FW isolation subsystems provide Safety Category 1 isolation functions for the RPV and PCS in the event of accidents or other conditions.

Regulatory Requirements

The MS and FW isolation subsystems comply with the following GDC, which have plant-wide applicability, as discussed in Section 3.1:

GDC 2 – Design bases for protection against natural phenomena

GDC 4 – Environmental and dynamic effects design bases

The following GDC have aspects specific to MS and FW isolation subsystems and compliance is addressed as follows:

GDC 54 – Piping systems penetrating containment, as it relates to piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities. MS and FW isolation subsystems are provided with Safety Category 1 leak detection, isolation, and containment functions.

GDC 55 – Reactor coolant pressure boundary penetrating containment, as it relates to each line that is part of the RCPB and that penetrates primary reactor containment shall be provided with containment isolation valves. MS and FW isolation subsystems are provided with SC1 CIVs.

Section 3.1 summarizes compliance with GDC.

Regulatory Guidance

The following guidance has aspects specific to MS and FW isolation subsystems and conformance is addressed as follows:

NUREG-0800 Branch Technical Position (BTP) 3-4, Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment. BTP 3-4 conformance is described in Section 3.6.

5.4.5.4 Testing and Inspection Requirements

Preoperational testing requirements are defined by the Initial Test Program (Chapter 14).

The RIVs and CIVs are verified operable during plant operation by periodic test cycling and inspection. RIVs and CIVs are equipped for remote manual functional testing from the MCR.

5.4.5.5 Instrumentation Requirements

The MS flow instruments are differential pressure FTs used to determine MS flow for the purpose of MS isolation. The MSRIVs and MSCIVs are equipped to provide manual control and position indication data in the MCR.

CFS provides instrumentation for measuring the necessary FW parameters required by the Reactor Isolation Function Logic to detect a FW line break. The FWRIVs and FWCIVs are equipped to provide manual control and position indication data in the MCR.

More information on the CIVs and associated instrumentation is discussed in Subsection 6.2.4 and Section 7.3.

5.4.6 Isolation Condenser System

Section 6.3 describes RCPB overpressure protection and ICS ECCS and decay heat removal functions.

GDC 33 compliance related to reactor coolant makeup for small breaks is also discussed in Section 4.6 and Section 15.3.

5.4.7 Shutdown Cooling System

The BWRX-300 is a passive plant and does not have a traditional residual heat removal system. Removal of residual and decay heat is provided by the main condenser (discussed in Subsection 10.4.1) and the ICS (described in Section 6.3) in conjunction with SDC.

ICS provides Safety Category 1 residual and decay heat removal function during off-normal conditions. The ICS cools the reactor when the RCPB becomes isolated following certain events during power operations. The ICS also provides cooling of the reactor when the PCS is isolated. The SDC does not provide any low-pressure emergency core cooling function. SDC provides Safety Category 3 residual and decay heat removal.

5.4.7.1 Design Bases

Safety Category 1 Functions

- Leak Detection and Isolation to prevent a reduction of reactor water inventory

Safety Category 2 Functions

None.

Safety Category 3 Functions

- RPV pressure and temperature reduction during cooldown operation
- Decay heat removal during startup operations, hot shutdown operations, cold shutdown operations, and refueling operations
- Overboarding during startup operations and hot shutdown operations
- Manual initiation of SDC on loss of FW flow during startup operations and hot shutdown operations
- Decay heat removal for small line break inside containment during hot shutdown operations

Safety Category N Functions

None.

The Safety Classifications, Quality Groups, and Seismic Categories of SDC SSCs are listed in Table 3A-1.

5.4.7.2 System Description

SDC is comprised of two independent trains. The major components of each train include a pump and heat exchanger, along with valves, piping, I&C, and power inputs.

SDC is capable of operating at high pressures and temperatures up to the reactor operating pressure and temperature, however SDC operation at these elevated conditions is not expected during normal operation.

SDC Layout and Location

The SDC SSCs are housed in the RB. The SDC interfaces with multiple systems to minimize the number of RPV nozzles as well as containment penetrations. The SDC provides process water to the Process Sampling Subsystem (PS) for online chemistry monitoring. SDC Trains A and B take suction from ICS A and B condensate return lines between the ICS condensate return valves and the ICS reactor isolation valves. The containment isolation valves on the SDC suction lines are classified as ICS valves.

After passing through the system isolation valve and an FE located outside the PCS, each SDC suction line is routed to an SDC pump in a room outside the PCS. Trains A and B are located in separate, adjoining rooms due to As Low As Reasonably Achievable (ALARA) requirements of separating potentially contaminated equipment for maintenance. The discharge piping for each pump is routed up to the respective SDC Train A or B heat exchanger. The heat exchangers are located in adjoining rooms above the SDC pump. A heat exchanger bypass line is installed to increase operational flexibility.

After exiting the SDC heat exchanger or heat exchanger bypass, flow is either directed back to the CFS Train A or B injection line upstream of the CFS CIVs line for return to the reactor for the decay heat removal function or routed to an overboarding line to CUW. Overboard flow can be directed to the Main Condenser and Auxiliaries (MCA) hotwell, or the radwaste reactor water storage tank by the CUW.

SDC Decay Heat Removal Subsystem

The SDC decay heat removal subsystem comprises the flow path from the ICS return line to the SDC pump, through the tube side of the SDC heat exchanger, and then returned to the RPV through the CFS as depicted in Figure 5.4-2. The decay heat removal subsystem is available during startup operations, hot shutdown operations, cold shutdown operations, and refueling operations, although the main condenser is used initially to cool the reactor after shutdown. The decay heat removal subsystem is initially operated at low flow to allow the SDC SSCs to be brought up to operating temperature without overstressing the components. Once the subsystem is up to temperature, the flow rate increases and is controlled to maintain PCW (Subsection 9.2.2) heat exchanger cooling water exit temperature within system interface requirements.

SDC is designed to reduce the reactor coolant temperature to: 60.0°C (140°F) in 24 hours, 54.4°C (130°F) in 40 hours, and 48.9°C (120°F) at the completion of reactor cavity flooding from water sources with temperatures <= 35°C (95°F) in 96 hours.

SDC Overboarding Subsystem

The SDC overboarding subsystem supports Reactor Level Control during startup operations, hot shutdown operations, cold shutdown operations, and refueling operations when the fuel pool gate is installed and CUW suction line from the RPV is not available for use. The overboarding subsystem comprises the flow path from the ICS condensate return line to the SDC pump, through the tube side of the SDC heat exchanger, and then through the overboard flow path to CUW. Figure 5.4-3 illustrates multiple overboard flow pathway connections through CUW. However, one overboard path connection from SDC to CUW using one or two pumps is implemented at a time by controlling the position of the valve on each flow path. The SDC trains share the overboard control valves and each SDC train has its own isolation valve to prevent inadvertent overboarding. Each SDC overboard flow path provides sufficient overboarding flow to support the fastest heat up rate of the reactor combined with the expected water input from the CRD.

The preferred overboard flow path is from the SDC through CUW to the LWM. However, in the event that high temperature is detected in the effluent, the flow path is manually switched from the LWM to the MCA.

SDC Leak Detection and Isolation Subsystem

When SDC is operational, the SDC leakage detection and isolation subsystem continuously monitors for leakage utilizing density-compensated differential flow measurements. If a leak is detected in one of the SDC trains, the affected SDC train is isolated to prevent a reduction of reactor water inventory.

The SDC leakage detection and isolation subsystem is comprised of FEs along with FTs to support leakage detection requirements. The readings are density-compensated accounting for temperature change across the SDC heat exchanger. This avoids spurious isolation of the SDC, which would reduce the SDC capacity, or if both trains are isolated, would require restart of ICS.

The I&C system compares the flow readings from the SDC supply FTs against flow readings from the SDC return and overboard FTs. If the I&C system detects a flow differential greater than the acceptable value, the I&C system actuates FW Isolation and SDC Isolation for the affected SDC train.

5.4.7.2.1 Component Description

Piping and Valves

The SDC piping material is stainless steel. The pipe wall thickness has been sized to ensure it can withstand the maximum pressure and temperature combination, and the diameter has been chosen to ensure the maximum flow velocities stay within the acceptable limits. The SDC piping and wetted parts of the valves are stainless steel to support ALARA goals, as this system is in contact with potentially radioactive water.

Valves in the system are used for isolation (on/off) except for the heat exchanger bypass valves, overboard control valves, and return control valves. The bypass valve is used to throttle the bypass around the heat exchanger and can be used in conjunction with the return control valve to control the exit temperature of the heat exchanger.

SDC decay heat removal piping and valves, and SDC overboard piping, and valves are designed, manufactured, and tested in accordance with ASME B31.1. Sections of the piping that include SC1 leak detection instrumentation are designed, manufactured, and tested in accordance with ASME BPVC, Section III, Subsection NCD.

Pumps

Each train of SDC contains one centrifugal pump designed to pump the maximum flow required by the system to meet SDC cooldown requirements. The SDC pumps are designed to pump the maximum flow required to meet SDC cooldown requirements for decay heat removal. The pump is installed below the RPV inlet nozzle to provide Net Positive Suction Head (NPSH) at maximum flow conditions. Each pump is paired with a return control valve at the outlet of the SDC heat exchanger to allow for control of the cooldown rate and the reactor temperature during shutdown.

Heat Exchangers

Each train of the system contains one shell and tube heat exchanger to remove decay heat. The heat is transferred from the SDC process fluid to the PCW.

SDC heat exchangers are designed, manufactured, and tested in accordance with ASME BPVC, Section VIII, Division 1. SDC heat exchanger tubes are constructed of stainless steel.

5.4.7.2.2 Normal Operation

The SDC has the following operational alignments:

Power

During power operations, SDC is not in operation.

Startup

During startup operations, the SDC is initially in service to provide forced circulation in the core and a reactor water overboarding flow path for reactor water level control until the CUW is in service for this function. The system pressure rating is such that it can be operated at any pressure up to full reactor design pressure for flexibility. The overboarding function of SDC is taken out of service when RPV pressure is high enough to establish CUW flow without the aid of the SDC pump. The circulation function of SDC is taken out of service once natural circulation has been established in the RPV and before the SDC has inadequate net positive suction head due to voiding in the chimney section of the RPV.

Hot Shutdown

During hot shutdown operations, the SDC is placed into service after SDC loop warming is completed and prior to securing heat removal via the Turbine Bypass System (Subsection 10.4.4). SDC is used to remove decay heat from the reactor core after condenser cooling is no longer effective and can be in service if cold shutdown mode transition is planned. SDC operation can be initiated after there is an adequate water level in the chimney to provide the necessary net positive suction head for the SDC pump. The heat is rejected to the PCW.

When the motive force from RPV pressure drops to a point that the normal reactor water overboard flow path via CUW is ineffective, a shift to overboarding via SDC to main condenser hotwell or LWM can be established. The SDC can be placed into service to overboard excess inventory to the main condenser or LWM through an interface connection with CUW.

Cold Shutdown

During cold shutdown operations, both trains of the SDC are normally available, but one train may be taken out of service at a time for maintenance if needed after decay heat has decreased to the point that one train is adequate to maintain the desired RPV water temperature.

With the RPV depressurized, SDC can be placed into service to overboard excess inventory to the main condenser or LWM through an interface connection with CUW.

Refueling

During refueling operations, SDC is in operation. SDC overboard flow path is coordinated with the water level in the RPV, position of the fuel pool gate, and availability of FPC. When the reactor cavity is fully flooded and connected to the fuel pool, FPC can maintain a constant level.

At any time when the fuel pool gate and equipment pool gate are installed and water level is being held constant, SDC will be overboarding to the LWM to offset the CRD purge flow.

5.4.7.2.3 System Shutdown

SDC is shut down as the plant heats up in Plant Mode 2. The cooling function is no longer required, as heat is being intentionally added to the RPV, and the overboarding function transitions from SDC to CUW when the RPV pressure is sufficient to drive the necessary flow through CUW to the overboarding destination.

Upon shutdown, the pump is turned off and the system isolation valves closed. SDC remains water solid while shutdown.

5.4.7.2.4 Off-Normal Operation

The SDC performs the following off-normal functions:

Leak Detection and Isolation

When the SDC is operational, it continuously monitors for leakage utilizing density compensated differential flow measurements and ambient temperature monitors. If a leak is detected at the initial setpoint, an alarm initiates in the MCR for operators to investigate. If a leak is detected at the secondary setpoint, the SDC is automatically isolated from the reactor using the ICS SDC CIV and the CFS SDC CIV to prevent the excessive reduction of reactor water inventory.

5.4.7.3 Safety Evaluation

SDC performs the Safety Category 3 functions of decay heat removal and overboarding. SDC leak detection and isolation function is Safety Category 1. The ICS performs the Safety Category 1 heat removal function. The ICS is described in Section 6.3.

Regulatory Requirements

The SDC complies with the following GDC, which have plant-wide applicability, as discussed in Section 3.1:

GDC 2 – Design bases for protection against natural phenomena

GDC 4 – Environmental and dynamic effects design bases

GDC 5 – Sharing of structures, systems, and components

The following GDC/Principal Design Criterion (PDC) have aspects specific to the SDC system and compliance is addressed as follows:

GDC 2 – Design bases for protection against natural phenomena, as it relates to the seismic design of SSCs whose failure could cause an unacceptable reduction in the capability of the residual heat removal system, specifically based on meeting Regulatory Position C.2 of RG 1.29 or its equivalent. SDC leak detection function is Safety Category 1 and meets RG 1.29, Position C.2.

GDC 4 – Environmental and dynamic effects design bases, as it relates to dynamic effects associated with flow instabilities and loads (e.g., water hammer). SDC is water solid when shutdown. If necessary, the system piping is filled and vented to minimize water hammer upon system startup.

PDC 19 – Control room, as it relates to control room requirements for normal operations and shutdown. SDC can be controlled from the MCR.

GDC 34 – Residual heat removal, as it relates to provision of a residual heat removal system. Residual heat removal capabilities of SDC are not credited in the Chapter 15 Safety Analyses. The ICS (described in Section 6.3) is the Safety Category 1 heat removal system.

Section 3.1 summarizes compliance with GDC and PDC.

10 CFR 50.34(f)(2)(xxvi)

When SDC is operational, the system continuously monitors for leakage and provides system isolation as required.

10 CFR 50.34(f)(2)(xxi)

ICS meets this requirement as described in Section 6.3.

Regulatory Guidance

The following guidance has aspects specific to the SDC:

SECY-94-084 – Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs. SDC performs a Safety Category 3 normal residual heat removal function and provides a Safety Category 3 defense in depth residual heat removal function to the ICS (described in Section 6.3). Additional information on conformance with SECY-94-084 is provided in NEDC-33934P, Licensing Topical Report, "BWRX-300 Safety Strategy," (Reference 5.4-2) and Subsection 15.6.1.4.2.

BTP 5-4, Revision 4 – Design Requirements of the Residual Heat Removal System. SDC conforms with the following criteria of BTP 5-4:

- 1.B: Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities to ensure that for onsite electrical power system operation (assuming offsite power is not available) and offsite electrical power system operation (assuming onsite power is not available) the system function can be accomplished assuming a single failure
- 1.C: Capable of being operated from the control room (including instrumentation for monitoring and control functions) with either only onsite or offsite power available
- 2.A.i: Isolation shall be provided by at least two power-operated valves in series. The valve positions shall be indicated in the control room.
- 2.B.i: 2B.ii, 2.B.iii, and 2.B.iv: Valves, position indicators, and interlocks
- 4: Design and operating procedures shall have provisions to prevent damage from overheating, cavitation, or loss of adequate pump suction fluid
- 6: Operational procedures for bringing the plant from normal operating power to cold shutdown shall be in conformance with RG 1.33

BTP 5-4, Criteria 2.A.ii, 2.A.iii, and 3, as they relate to residual heat removal pressure limitations, are not applicable to the BWRX-300, as SDC design pressure is the same as the RPV design pressure.

SECY-90-016 – Evolutionary LWR Certification Issues and Their Relationship to Current Regulatory Requirements and SECY-93-087 – Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs, as they relate to intersystem LOCA. Conformance to the regulatory position with respect to minimizing the potential for an intersystem LOCA and other SECY-90-016 topics are addressed in Subsection 15.6.1.4.2. Intersystem LOCA is also discussed in Subsection 3.12.3.18. In addition, SDC is not susceptible to intersystem LOCA as SDC design pressure is the same as the RPV design pressure.

5.4.7.4 Testing and Inspection Requirements

Preoperational testing requirements are defined by the Initial Test Program (Chapter 14).

5.4.7.5 Instrumentation Requirements

SDC is a manually initiated system but is designed for automatic control of functions such as SDC piping pre-warming, adjusting the system heat removal rate to stay within vessel and heat exchanger cooling water limitations, as well as allowing the operator to input a temperature rate setpoint that allows automatic vessel water temperature cooldown rate.

Additionally, during reactor vessel heatup, the SDC controllers receive flow demand signals from the Reactor Level Control to allow "swell" to be overboarded to maintain reactor level.

Flow, temperature, and pressure sensing elements, along with associated transmitters, are provided at appropriate locations throughout the SDC to allow for automatic functions and operator monitoring of SDC operation.

Refer to Section 7.3 for additional information pertaining to I&C systems.

5.4.8 Reactor Water Cleanup System

CUW provides blowdown-type cleanup flow for the RPV during reactor power operating mode. CUW does not clean or filter reactor water. Reactor water cleanup, filtration, and ion removal is performed by the CFD as described in Subsection 10.4.6. CUW provides an overboarding flow path to the condenser hotwell or liquid radwaste directly from the RPV lower region.

5.4.8.1 Design Bases

Safety Category 1 Functions

- Leak Detection and Isolation to prevent a reduction of reactor water inventory

Safety Category 2 Functions

None.

Safety Category 3 Functions

- Blowdown-Type Cleanup Flow
- Overboarding

Safety Category N Functions

None.

The Safety Classifications, Quality Groups, and Seismic Categories of the CUW SSCs are listed in Table 3A-1.

5.4.8.2 System Description

CUW components are located in the RB and TB. Figure 5.4-4 depicts a simplified flow diagram of the CUW and interfacing systems.

Reactor water cleanup, filtration, and ion removal are performed by CFD, which is described in Subsection 10.4.6. CFD purifies the condensate and CUW water, as required, to maintain reactor FW purity.

CUW does not utilize a pump and relies on the reactor or SDC for motive pressure to generate flow, with sufficient reactor pressure required to overcome system flow resistance for CUW flow to be achieved without SDC. System flow is maintained through throttling as reactor pressure increases.

CUW consists of one train fed by two inlet nozzles located in the RPV. The train inlet nozzles are independently connected to RPV penetrations located at about the mid-vessel height and take suction from inlet piping located near the RPV bottom head. The inlet piping connects to the two RPV CUW RIV valves and then combines inside the PCS to form one inlet line. This line penetrates the PCS and is provided with a containment isolation valve. The CUW RIVs and CIV receive a signal from the CUW leak detection subsystem and close upon a detected leak.

Downstream of the CUW containment isolation valve the flow continues through a Regenerative Heat Exchanger (RHX) and a pressure reduction station. Upstream of the RHX is the primary interface with SDC, which is used for overboarding as described below. The RHX and pressure

reduction station are designed to condition the water to acceptable temperatures and pressures for processing to CFS or overboarding. The RHX recovers heat back to the condensate.

Downstream of the RHX and upstream of the flow control valves there is an interface with SDC. Discharge piping is connected either to a condensate line for the normal CUW function or routed to the condenser hotwell or LWM for overboarding.

Information on CFD condensate demineralizers, condensate filters, and resin handling is provided in Subsection 10.4.6. Details regarding the LWM are provided in Section 11.2. MCA is described in Subsection 10.4.1.

Subsection 6.2.4 discusses the containment isolation function and CIVs.

CUW Leak Detection and Isolation Subsystem

CUW leak detection and isolation subsystem performs the Safety Category 1 function to identify leaks in CUW and initiates automatic response actions. The leak detection subsystem continuously monitors for leakage utilizing density-compensated differential flow measurements. If a leak is detected, the containment isolation function isolates the system from the RCPB through the closure of the CUW containment isolation valve and the CUW RIVs.

The leak detection and isolation Safety Category 1 function utilizes a single FE in the intake flow path and a single FE in each downstream flow path. Each FE serves independent, redundant sets of divisional FTs for the Safety Category 1 leak detection functions. The basis for this arrangement is that a leak near the end of a flow path may not be detectable by a single FE/FT feature in an upstream location of the system due to the physical distance between those locations. The design is intended to ensure that a leak or break can be identified by any of the FE/FTs at a detection point, or by differential flow between an upstream and downstream FE/FT detection point. Hence the use of leak detection features upstream and downstream locations allows for detection of differential flow between the locations, or detection of a leak or break within detection proximity of any single FE. When overboarding from SDC, the SDC FTs are compared to the CUW FTs.

The FT readings are density-compensated by independent, redundant temperature transmitters. This feature accounts for density changes in the water across the RHX between the upstream FE and the FEs located in each downstream flow path. Density-compensation is used to avoid spurious isolation of CUW and spurious generation of a signal that would initiate unwanted response actions by other systems.

5.4.8.2.1 Component Description

Piping and Valves

CUW piping is sized to ensure it can withstand the maximum pressure and temperature combination while keeping the maximum flow velocities within acceptable limits.

The piping and the wetted parts of CUW valves are stainless steel to help with ALARA concerns because this system is in contact with potentially radioactive water. CUW RIVs are part of the NBS.

All valves in the system are used for isolation except for a throttling valve in the pressure reduction station. This valve is used to throttle the system to achieve desired flow rates.

Regenerative Heat Exchanger

The CUW heat exchanger is a RHX cooled by FW. The heat exchanger is designed to reduce the temperature of CUW flow. Most of the heat of CUW water is transferred to the condensate returning to the reactor, thus losing almost no thermal value.

5.4.8.2.2 Operational Alignments

CUW has the following operational alignments:

Reactor Water Cleanup

During power operation, CUW is in service, taking suction from the bottom of the RPV through two suction lines. Water flows through piping internally mounted inside the RPV downcomer region and exits the vessel about four meters above the Top of Active Fuel. At the RPV nozzles, there are RIVs attached to the vessel, which close to prevent RPV inventory from exiting the CUW line in case of a pipe break. Except for the containment isolation and leakage detection related functions, CUW performs Safety Category 3 functions, and the other system functions defined below are Safety Category 3.

The two suction lines join into a single line outside the RPV where it exits the PCS, routes through piping in the MS Tunnel and into the TB. The system contains a RHX in which CUW water is cooled by CFS. Most of the heat of the CUW water is transferred to the FW returning to the reactor. A flow control valve is used to regulate flow and pressure before the water enters the CFS downstream of the condensate pumps, where it is mixed with the condensate and cleaned in the CFD.

During startup and shutdown operations, CUW can operate in the Reactor Water Cleanup alignment if FW is in service. However, due to relying on the reactor to provide motive pressure for flow through CUW, sufficient reactor pressure is required to overcome system flow resistance for CUW flow to be achieved, with flow being maintained by the FCVs.

CUW Overboard to MCA or LWM

When there is sufficient pressure in the RPV for flow in CUW, the CUW can overboard to the MCA hotwell or LWM. In this alignment CUW operates the same as Reactor Water Cleanup alignment, with the exception of flow being diverted to the MCA hotwell or LWM.

Overboarding from SDC through CUW bypassing RHX

When there is insufficient pressure in the RPV for flow in CUW, SDC can overboard through an interface with the CUW downstream of the CUW heat exchanger to the MCA hotwell or LWM.

5.4.8.2.3 Normal Operation

Power

During power operations, CUW is in service in Mode A, taking suction from the bottom of the RPV through the normal flow path and then entering the CFS downstream of the condensate pumps, where it is mixed with the condensate and cleaned in the CFD.

Startup

During startup operations, CUW operation is the same as during power operation. However, due to relying on reactor pressure to produce flow through CUW, sufficient reactor pressure is required to overcome system flow resistance for CUW flow to be achieved, with flow increasing as reactor pressure increases.

During startup or shutdown operations, CUW can function to letdown (overboard) the excess reactor inventory if there is enough pressure. If not, SDC can be used to let down the excess reactor inventory. The primary SDC overboard line interfaces with CUW upstream of the RHX. The secondary SDC overboard line interfaces with CUW downstream of the pressure reduction station. Overboard flow can be directed to the condenser hotwell or LWM.

Hot Shutdown

During hot shutdown operations, CUW is in service if the RPV is pressurized. Vessel overboarding can be performed as discussed under startup operation.

Cold Shutdown

During cold shutdown operations, CUW is isolated at the vessel. SDC can be used via CUW if overboarding is required.

Refueling

During refueling operations, CUW is isolated at the vessel. SDC can be used via CUW if overboarding is required.

5.4.8.2.4 Off-Normal Operation

No manual operator actions are required to initiate or assure the completion of fundamental CUW isolation functions. The RIVs and containment isolation valve fail in the closed position, with valve actuators designed to maintain the valves closed by positive mechanical means.

For CUW leaks outside of the PCS, leak detection subsystems identify the leak condition and provide signals to close the affected system isolation valves.

5.4.8.3 Safety Evaluation

CUW provides Safety Category 3 blowdown-type cleanup flow for the RPV during reactor power operating mode. Reactor water cleanup, filtration, and ion removal is performed by the CFD as described in Subsection 10.4.6. CUW provides Safety Category 1 leakage detection and system isolation function.

Regulatory Requirements

The CUW complies with the following GDC, which have plant-wide applicability, as discussed in Section 3.1:

GDC 1 – Quality standards and records

GDC 2 – Design bases for protection against natural phenomena

The following GDC have aspects specific to the CUW, and compliance is addressed as follows:

GDC 14 – Reactor coolant pressure boundary, as it relates to ensuring the RCPB integrity. Compliance with GDC 14 for the CFD is discussed in Subsection 10.4.6. Measures used to ensure integrity of RCPB portions of the CUW are described in Subsection 5.2.3.

GDC 54 – Piping systems penetrating containment and GDC 55 – Reactor coolant pressure boundary penetrating containment, as they relate to piping systems penetrating primary reactor containment are provided with leak detection, isolation, and containment capabilities. CUW is provided with leak detection, isolation, and containment functions.

GDC 60 – Control of releases of radioactive materials to the environment, as it relates to the capability of the CUW to control the release of radioactive effluents to the environment. CUW continuously monitors for leakage. If a leak is detected, the Safety Category 1 containment isolation function isolates the system from the RCPB through the closure of the CUW RIVs and CUW CIV, which limits release of radioactive material during accident conditions. The CUW does not perform an RCS water purification function and does not retain radioactive materials generated from RCS water purification. RCS cleanup, filtration, and ion removal is performed by the CFD as described in Subsection 10.4.6.

GDC 61 – Fuel storage and handling and radioactivity control, as it relates to designing the CUW with appropriate confinement. CUW RIVs and CIV are within the PCS and the RB.

Section 3.1 summarizes compliance with GDC.

Regulatory Guidance

The following regulatory guidance has aspects specific to the CUW:

RG 1.26, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants. The Safety Classifications, Quality Groups, and Seismic Categories of the CUW SSCs are listed in Table 3A-1.

5.4.8.4 Testing and Inspection Requirements

Pursuant to 10 CFR 50.55a(f)(4)(i) and 10 CFR 50.55a(g)(4)(i), the initial IST and ISI programs, respectively, will use the edition and addenda of the ASME OM Code and of ASME BPVC, Section XI, subject to the conditions identified and incorporated by reference into 10 CFR 50.55a, on the date no more than 18 months before the date of issuance of the Operating License. Accordingly, the Code Editions for the ASME OM Code and ASME BPVC, Section XI, Division 1 will be addressed in the FSAR.

Preoperational testing requirements are defined by the Initial Test Program (Chapter 14).

5.4.8.5 Instrumentation Requirements

During plant operation and after plant shutdown (with sufficient reactor pressure), CUW removes water from the bottom of the reactor and routes it to the CFS or overboards excess “swell,” when required, to the condenser hotwell or liquid radwaste system. These functions require both monitoring and control.

CUW can be controlled from the MCR. The CUW controller can receive commands from the Plant Automation System or be manually operated. The controller automates the operation of the system.

Flow, temperature, and pressure sensing elements, along with associated transmitters, are provided at appropriate locations throughout the CUW to allow for automatic functions and operator monitoring of CUW operation.

Refer to Section 7.3 for additional information pertaining to I&C systems.

5.4.9 Main Steam Line, Feedwater, and Auxiliary Feedwater Piping

MSL and FW lines from the RIVs to the CIVs, including isolation functions, are addressed in Subsections 5.4.3 and 5.4.5. MS and FW piping beyond the CIVs is discussed in Sections 10.3 and 10.4.7, respectively.

5.4.10 Pressurizer

The BWRX-300 design does not include a pressurizer.

5.4.11 Pressurizer Relief Discharge System

BWRX-300 design does not include a pressurizer relief discharge system.

5.4.12 Reactor Coolant System High Point Vents

RPV systems that are stagnant during normal plant operation have lines that are sloped downward from the RPV to prevent buildup of noncondensible gases and promote proper drainage. Where horizontal process lines cannot be continuously sloped, the lines are equipped with high point vents and accessible vent valves to ensure that the piping can be filled and vented.

ICS gas purge and catalytic recombiners are discussed in Section 6.3.

5.4.13 Reactor Pressure Vessel Head Vent Subsystem

The BWRX-300 is equipped with a RPV head vent subsystem that handles noncondensible gas buildup in the RPV head. The RPV head vent subsystem is designed to transport steam and noncondensible gases from underneath the RPV head to MSL Train A during startup and normal plant operation.

The RPV head vent subsystem includes piping internal to the RPV head, an RIV consisting of two independent isolation valves in a single body flange mounted on the reactor, and piping to the MSL Train A or quench tank, both located inside the PCS. This permits the gases to be released from the RPV so that the RPV can be filled with water for hydrostatic testing and provides the upper tap for RPV level measurement during reactor shutdown.

The RPV head vent piping penetrates the RPV through a nozzle on the side of the reactor. The internal piping is split into a portion mounted inside the RPV closure head and a portion mounted below the vessel flange with an intersection of the internal piping at a guide rod bracket where the two ends are joined. The piping in the RPV closure head is mounted with at least two brackets that hold the vent in place while allowing for thermal expansion and mitigating flow induced vibration effects. The brackets are designed to ASME BPVC, Section III, Subsection NF.

5.4.13.1 Design Bases

Safety Category 1 Functions

- Containment of radioactive materials via isolation of the RCPB
- Isolation of the RPV to minimize the loss of reactor coolant

Safety Category 2 Functions

None.

Safety Category 3 Functions

- Transport steam and noncondensible gases from underneath the RPV head to MSL Train A during startup and normal plant operation

Safety Category N Functions

None.

The Safety Classifications, Quality Groups, and Seismic Categories of the RPV head vent SSCs are listed in Table 3A-1.

5.4.13.2 System Description

The BWRX-300 RPV design includes internal piping routed from the inside of the top of the RPV steam dome area to below the RPV shell flange. The head vent piping penetrates the RPV through a nozzle on the side of the reactor with a flange mounted RIV consisting of two independent isolation valves in a single body. This RIV is SC1 and perform the RPV single failure proof automatic isolation function.

The RPV head vent subsystem is designed to transport steam and noncondensible gases from underneath the RPV head to MSL Train A during startup and normal plant operation. The noncondensible gases are extracted by differential pressure between the reactor dome and MSL Train A and then swept from the steam line to the condenser.

The RPV head vent connection on the side of the RPV permits air and noncondensible gases to be released from the RPV into the Quench Tank inside the PCS when the plant is shutdown.

Horizontal portions of the head vent line piping to either the Quench Tank or the MSL Train A have a minimum slope in the direction of normal flow to ensure that no condensate stays in the line, thereby minimizing the potential for hydrogen and other noncondensible gases collecting in the lines.

The RPV head vent line is cross connected to MSL Train A upstream of the PCS penetration. A single solenoid valve is provided on the cross-connection to the MS line, which can close the head vent line cross-connection for routine operation, maintenance, and testing.

RPV head vent internal piping inside the RPV closure head is a discharge line that is integrated into and through the bracket of one of the internal guide rods. The internal piping is, thus, split into a portion mounted inside the RPV closure head and a portion mounted below the vessel flange with an intersection of the internal piping at a guide rod bracket where the two ends are joined. The piping in the RPV closure head is mounted with at least two brackets that hold the vent in place while allowing for thermal expansion and mitigating flow induced vibration effects. The brackets are designed to ASME BPVC, Section III, Subsection NF, categorized as SC1, and Seismic Category I.

The RPV head vent RIV is designed to fail closed and maintain a closed position by positive mechanical means upon loss of hydraulic power.

Section 5.2 provides information related to applicable material standards for the RPV head vent subsystem.

5.4.13.3 Safety Evaluation

The RPV head vent connection on the side of the RPV permits air and noncondensible gases to be released from the RPV into the Quench Tank inside the containment when the plant is in Plant Modes 3, 4, and 5. RPV head vent piping is sloped where appropriate to minimize the potential for noncondensible gases collecting in the lines. RPV head vent piping is equipped with an RIV that is designed to fail closed and maintain a closed position by positive mechanical means upon loss of hydraulic power.

Regulatory Requirements

The following GDC/PDC have aspects specific to RPV head vent subsystem and compliance is addressed as follows:

GDC 14 – Reactor coolant pressure boundary, as it relates to the RCPB being designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. This requirement is met as described in Section 5.2.

PDC 17 – Electric power systems, as it relates to the provision of normal and emergency power for the RPV head vent subsystem components. The RPV head vent RIV automatic closure function uses SC1 battery-backed AC power.

PDC 19 – Control room, as it relates to the RPV head vent subsystem controls being operable from the control room. The RPV head vent RIV is equipped with open and close position switches to provide position indication data to the MCR.

GDC 32 – RCPB periodic inspection and testing, as it relates to the RPV head vent subsystem being designed to permit periodic inspection. Areas requiring ISI inspection are provided with

access spaces/routes and with removable and reusable insulation coverings. The RPV head vent RIV is equipped for remote manual functional testing from the MCR.

Section 3.1 summarizes compliance with GDC and PDC.

The RPV head vent subsystem is designed to comply with the following regulations:

10 CFR 50.46a**,** as it relates to the provision of, and requirements related to, high-point vents for the RCS, the reactor vessel head, and other systems required to maintain adequate core cooling if the accumulation of noncondensible gases would cause the loss of function of these systems. Horizontal process lines are sloped downward in the direction of flow to promote proper drainage. Where horizontal process lines cannot be continuously sloped, the lines are equipped with high point vents and accessible vent valves to ensure that the piping can be filled and vented.

10 CFR 50.49**,** as it relates to environmental qualification of electrical equipment necessary to operate the RPV head vent subsystem. The RIVs, which include the RPV head vent RIV, are designed to withstand the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs.

10 CFR 50.55a and GDC 1 – Quality standards and records and GDC 30 – Quality of reactor coolant pressure boundary, as they relate to the RPV head vent subsystem components that are part of the RCPB being designed, fabricated, erected, and tested and maintained to high quality standards. These requirements are met as described in Section 5.2.

Regulatory Guidance

The following regulatory guidance has aspects specific to the RPV head vent subsystem:

RG 1.100, “Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants,” and RG 1.92, “Combining Modal Responses and Spatial Components in Seismic Response Analysis,” as they relate to the RPV head vent subsystem (i.e., vent valves, block valves, position indication devices, cable terminations, and piping) should be seismically and environmentally qualified in accordance with IEC/IEEE 60980-344, “Nuclear Facilities – Equipment Important to Safety – Seismic Qualification,” (Reference 5.4-3) as supplemented by RG 1.100 and RG 1.92. The RIVs, which include the RPV head vent RIV, are designed to withstand the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs.

5.4.13.4 Testing and Inspection Requirements

Pursuant to 10 CFR 50.55a(f)(4)(i) and 10 CFR 50.55a(g)(4)(i), the initial IST and ISI programs, respectively, use the latest edition and addenda of the ASME OM Code and of ASME BPVC, Division 1, Section XI subject to the conditions identified and incorporated by reference into 10 CFR 50.55a, 18 months before the date of issuance of the operating license. Accordingly, the Code Edition for the ASME OM Code and ASME BPVC, Section XI, Division 1 will be addressed in the FSAR.

Preoperational testing requirements are defined by the Initial Test Program (Chapter 14).

The RPV head vent RIV is equipped for remote manual functional testing from the MCR.

5.4.13.5 Instrumentation Requirements

The RPV head vent RIV is equipped with open and close position switches to provide position indication data to the MCR.

A temperature element is included on the RPV head vent nozzle line, between the quench tank and the second isolation valve upstream of the quench tank. The purpose of this temperature element is to provide leak detection capability.

During reactor shutdown periods, reactor water level can be measured using the RPV head vent connection as the upper tap for the level instrumentation sensors.

5.4.14 Reactor Coolant System Pressure Relief Devices/Reactor Coolant Depressurization Systems

The BWRX-300 utilizes the ICS System for pressure relief of the RCPB. ICS pressure relief is discussed in Section 6.3.

5.4.15 References

5.4-1 ASME-RCFM, “Fluid Meters Their Theory and Application Report of ASME Research Committee on Fluid Meters,” American Society of Mechanical Engineers, 6th edition, 1971.

5.4-2 NEDC-33934P, “BWRX-300 Safety Strategy Licensing Topical Report,” GE Vernova Hitachi Nuclear Energy Americas, LLC, Revision 2, March 2026.

5.4-3 IEC/IEEE 60980-344, “Nuclear Facilities – Equipment Important to Safety – Seismic Qualification,” Electrotechnical Commission and the IEEE Standards Association.

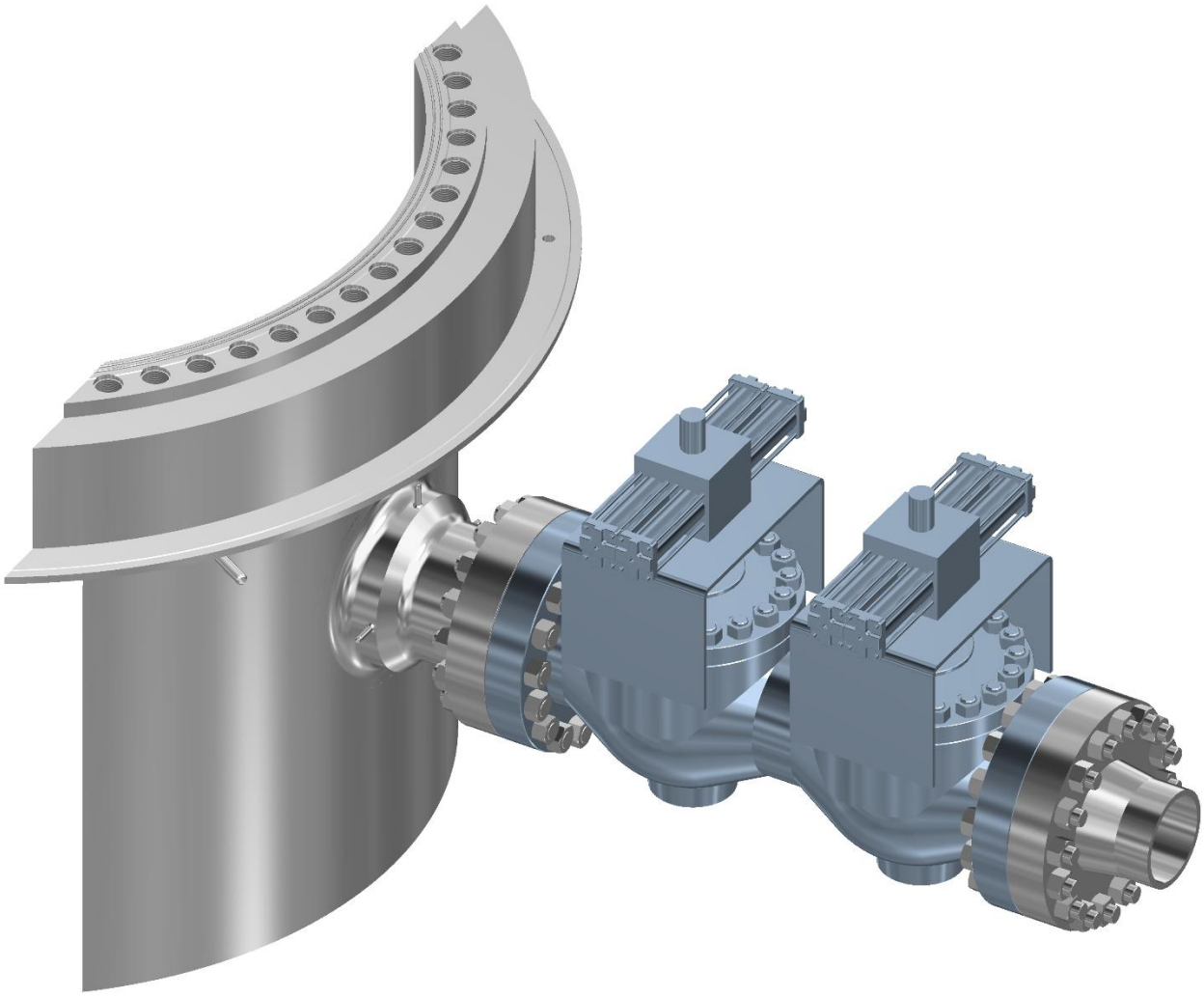


Figure 5.4-1: BWRX-300 Reactor Pressure Vessel Reactor Isolation Valve

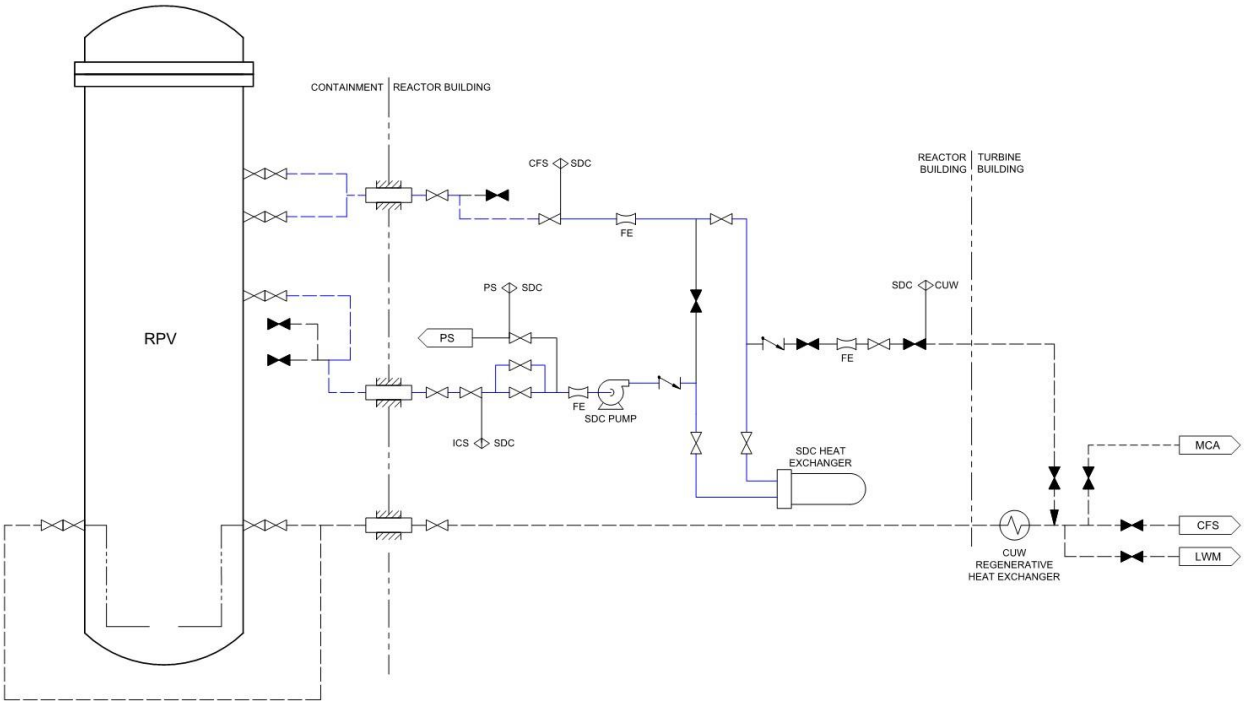


Figure 5.4-2: Shutdown Cooling System Simplified Diagram – Decay Heat Removal

Notes:

1. Valves shown, except check valves, have power operators.
2. Safety Class/Quality Group/Seismic Category breaks are not shown.
3. Blue line represents flow path.
4. Dotted lines are not part of the SDC.
5. Valve location on RPV is representative only.
6. Only one train of SDC is shown.

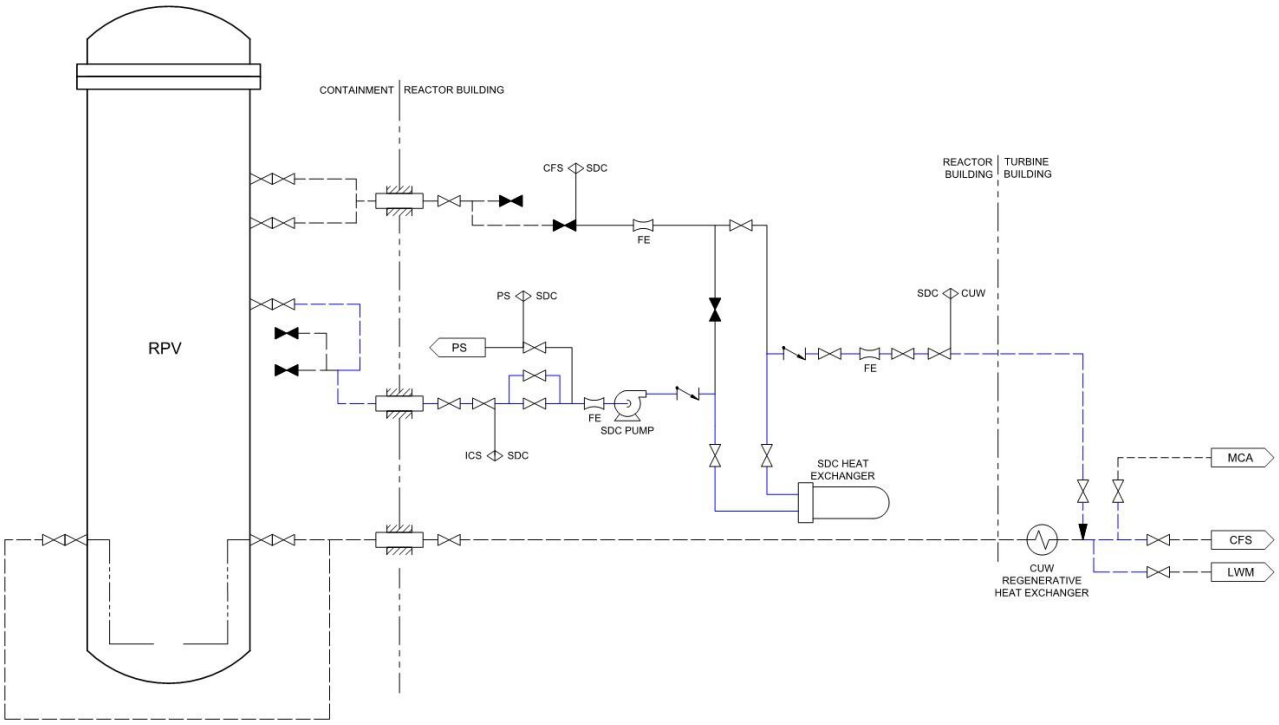


Figure 5.4-3: Shutdown Cooling System Simplified Diagram – Overboarding

Notes:

1. Valves shown, except check valves, have power operators.
2. Safety Class/Quality Group/Seismic Category breaks are not shown.
3. Blue line represents flow path.
4. Dotted lines are not part of the SDC.
5. Valve location on RPV is representative only.
6. Only one train of SDC is shown.

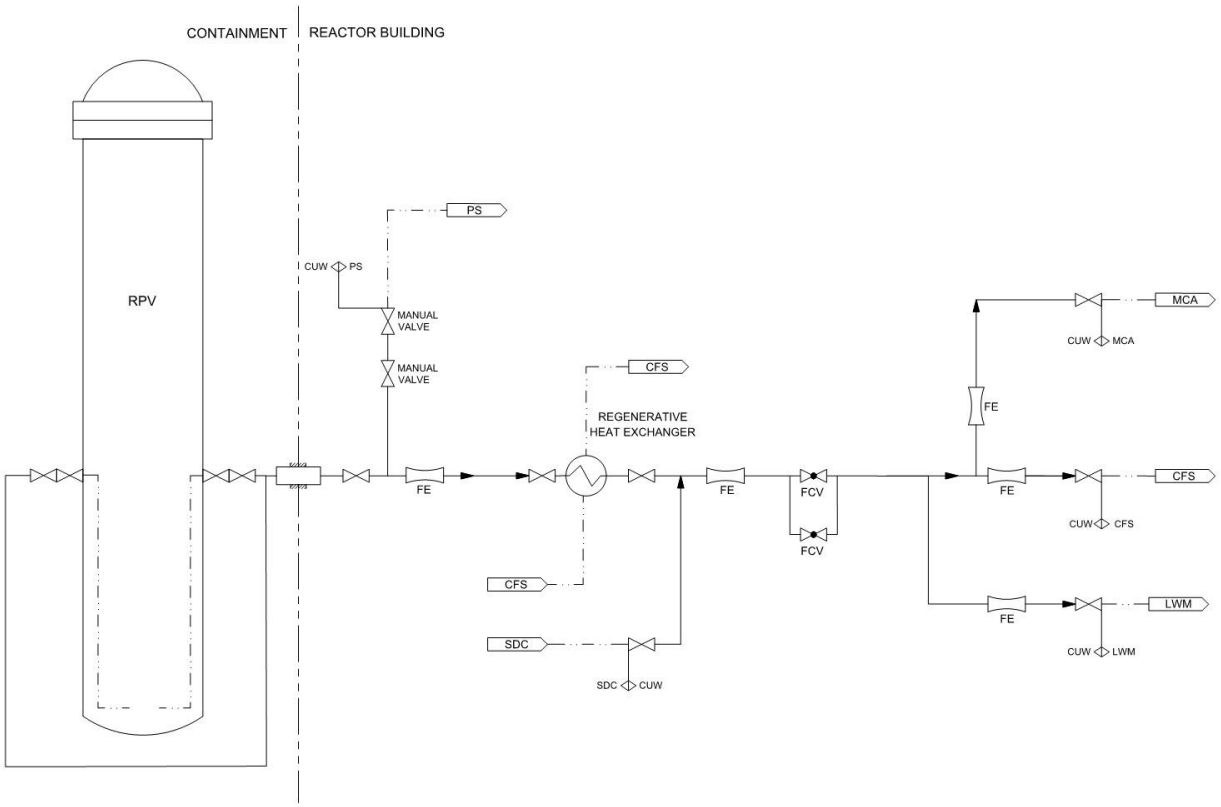


Figure 5.4-4: Reactor Water Cleanup System Simplified Diagram

Notes:

1. Valves shown have power operators.
2. Dotted lines are not part of the CUW.
3. Safety Class/Quality Group/Seismic Category breaks are not shown.
4. Valve location on RPV is representative only.

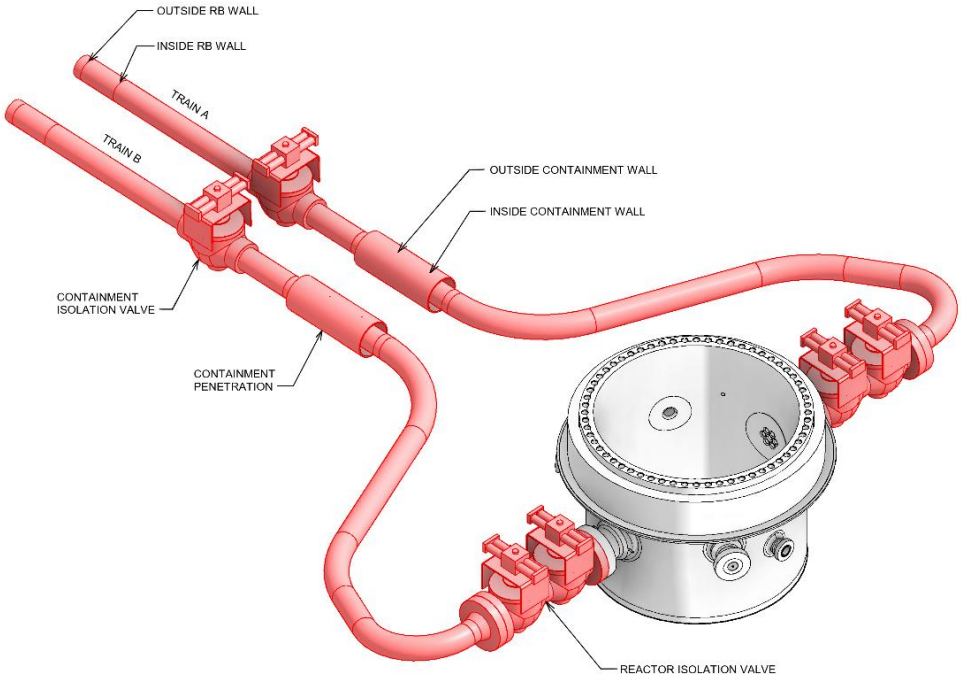


Figure 5.4-5: Main Steam System

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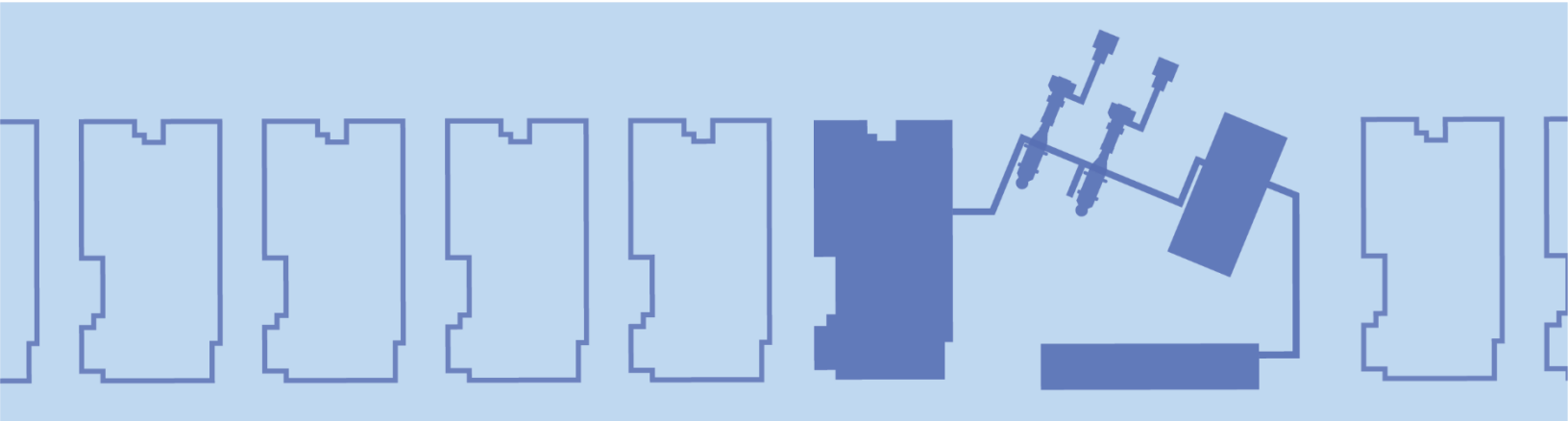
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September 2026

Non-Proprietary Information

Chapter 6
Engineered Safety Features



ACRONYM LIST

Acronym	Explanation
AC	Alternating Current
ALARA	As Low As Reasonably Achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
BIS	Boron Injection System
BPVC	Boiler and Pressure Vessel Code
CB	Control Building
CCS	Containment Cooling System
CFR	Code of Federal Regulations
CFS	Condensate and Feedwater Heating System
CIS	Containment Inerting System
CIV	Containment Isolation Valve
CMon	Containment Monitoring Subsystem
CRD	Control Rod Drive
CRE	Control Room Envelope
CRH	Control Room Habitability
CUW	Reactor Water Cleanup System
CWE	Chilled Water Equipment
DBA	Design Basis Accident
DC	Direct Current
DEC	Design Extension Condition
ECCS	Emergency Core Cooling System
EFS	Equipment and Floor Drain System
ESF	Engineered Safety Feature
FMCRD	Fine Motion Control Rod Drive
FPC	Fuel Pool Cooling and Cleanup System
FSAR	Final Safety Analysis Report

Acronym	Explanation
FSF	Fundamental Safety Function
GDC	General Design Criteria
I&C	Instrumentation and Control
IC	Isolation Condenser
ICC	Isolation Condenser Pools Cooling and Cleanup System
ICS	Isolation Condenser System
IGSCC	Intergranular Stress Corrosion Cracking
ILRT	Integrated Leakage Rate Test
ISI	Inservice Inspection
IST	Inservice Testing
LOCA	Loss-of-Coolant Accident
LWM	Liquid Waste Management System
MCR	Main Control Room
MSL	Main Steam Line
NBS	Nuclear Boiler System
PCCPA	Passive Containment Cooling Pipe Array
PCCS	Passive Containment Cooling System
PCS	Primary Containment System
PDC	Principal Design Criteria
PING	Particulate, Iodine and Nobel Gas
PPS	Plant Pneumatics System
PREMS	Process Radiation and Environmental Monitoring System
PWR	Pressurized Water Reactor
RB	Reactor Building
RCPB	Reactor Coolant Pressure Boundary
RG	Regulatory Guide
RIV	Reactor Isolation Valve
RPV	Reactor Pressure Vessel

Acronym	Explanation
SBO	Station Blackout
SC1	Safety Class 1
SCCV	Steel-Plate Composite Containment Vessel
SCR	Secondary Control Room
SDC	Shutdown Cooling System
SSCs	Structures, Systems, and Components
TMI	Three Mile Island
UHS	Ultimate Heat Sink
USNRC	U.S. Nuclear Regulatory Commission
WGC	Water, Gas, and Chemical Pads

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6.0 ENGINEERED SAFETY FEATURES

The Engineered Safety Features (ESF) for the BWRX-300 design are passive systems that are not dependent on external sources of power or operator action to fulfill the Fundamental Safety Functions (FSF) for at least 72 hours after a Design Basis Accident (DBA). The FSFs are:

1. Control of reactivity.
2. Removal of heat from the fuel.
3. Confinement of radioactive material.

The ESFs are the Primary Containment System (PCS), which includes the Passive Containment Cooling System (PCCS), and the Isolation Condenser System (ICS). These passive systems are automatically actuated, if needed, based on monitored parameters and automatically fulfill their ESF functions for at least 72 hours after a DBA.

The ESF complies with the following General Design Criteria (GDC), which have plant-wide applicability, as discussed in Section 3.1:

GDC 1 – Quality standards and records

GDC 2 – Design bases for protection against natural phenomena

GDC 4 – Environmental and dynamic effects design bases

GDC 5 – Sharing of structures, systems, and components

The technical descriptions of the BWRX-300 containment, PCCS, ICS, and Containment Isolation Valve (CIV) design features and design functions, acceptance criteria, regulatory bases provided in Section 2.0 of NEDC-33911P-A “BWRX-300 Containment Performance,” (Reference 6.2-3) are updated by the information in Chapter 6.

Chapter 6 sections are summarized as follows:

- The ESF materials for the PCS, which includes the PCCS, are described in Section 6.1. Because the ICS forms part of the Reactor Coolant Pressure Boundary (RCPB), its materials are described in Section 5.2.
- The PCS is described in Section 6.2. The PCCS, containment isolation function, combustible gas control, and containment leakage rate testing are addressed in Section 6.2.
- The ICS is described in Section 6.3
- Section 6.4 provides a summary description of Control Room Habitability (CRH) systems for the Main Control Room (MCR) and Secondary Control Room (SCR). CRH systems are not categorized as ESFs as discussed in Section 6.4.
- As discussed in Section 6.5, ESF fission product removal and control systems are not part of the design
- Preservice Inspection and Inservice Inspection (ISI) of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) Class 2 and 3 components are described in Section 6.6
- As discussed in Section 6.7, a Main Steam Line (MSL) isolation valve leakage control system is not part of the design
- The Containment Monitoring Subsystem (CMon) is described in Section 6.8

6.1 Engineered Safety Feature Materials

Engineered Safety Features for the BWRX-300 design are the ICS and PCS. Because the ICS forms part of the RCPB, its materials are addressed in Chapter 5. The PCS includes the Steel-Plate Composite Containment Vessel (SCCV), PCCS, Containment Isolation Valves (CIV), and other components as listed in Table 6.1-1. As discussed in Subsection 6.2.4.2.1, Reactor Isolation Valves (RIV) serve as CIVs inside containment.

Subsection 6.1.1 addresses ESF material selection and fabrication, compatibility with ESF fluids, and material controls for PCS components.

Subsection 6.1.2 addresses organic materials including protective coatings for the PCS components.

Subsection 6.1.3 and Subsection 6.1.4 summarize applicable regulatory requirements and guidance for PCS component materials.

6.1.1 Metallic Materials

6.1.1.1 Materials Selection and Fabrication

Quality requirements for the design, fabrication, installation, and inspection of PCS components are based on applicable codes and standards. Table 6.1-1 identifies the applicable sections of the ASME Boiler and Pressure Vessel Code (BPVC) and principal materials for the PCS.

Table 6.1-2 lists material specifications for welds. Material properties associated with both ASME BPVC and non-ASME BPVC requirements are selected to be consistent with the corresponding ASME BPVC or ASTM International specifications for the selected materials.

6.1.1.2 Compatibility of Construction Materials with Engineered Safety Feature Fluids

The PCS is a dry containment; its materials are compatible with the containment environment and are exposed only to high quality water during normal operation. The containment closure head forms part of the reactor cavity pool and separates the pool water from containment atmosphere during normal operations. The PCCS circulates water from the equipment pool to transfer heat from containment back to the equipment pool. The PCS components do not come into contact with core cooling water unless there is an event such as internal flooding in the PCS. A pipe break inside containment in the Nuclear Boiler System (NBS), ICS, Boron Injection System (BIS), Reactor Water Cleanup System (CUW), or Condensate and Feedwater Heating System piping could also result in PCS components coming into contact with core cooling water during a postulated accident.

Additional information on corrosion prevention is provided in Subsection 3.8.1.4.8.

6.1.1.3 Controls for Stainless Steel Materials

Controls for stainless steel materials are established during various stages of component manufacturing and construction in accordance with Regulatory Guide (RG) 1.31, “Control of Ferrite Content in Stainless Steel Weld Metal,” and RG 1.44, “Control of the Processing and Use of Stainless Steel,” for weld material and sensitized stainless steel.

Cold work controls are applicable to the fabrication process of ESF components made of austenitic stainless steel to minimize or avoid fabrication-induced surface stresses that could lead to stress corrosion cracking. The cold work controls for ESF components conform to RG 1.44. The processes that need to be controlled include bending or forming, final machining, grinding, and polishing.

6.1.2 Organic Materials

Corrosion prevention methods applied on SCCV steel faceplates and containment internal structural steel, such as protective paints or membrane coating systems, are described in Subsection 3.8.1.4.8. Corrosion prevention methods and protective coatings for the containment closure head and Class MC components are described in Subsection 3.8.2.4. As discussed in Section 6.3, the ICS circulates water from the Reactor Pressure Vessel (RPV). Thus, the design does not include an Emergency Core Cooling System (ECCS) that draws water from a source that is potentially affected by failure of protective coatings. Protective coatings and conformance to RG 1.54, "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants," are described in Subsections 3.8.1.4.8 and 3.8.2.4. These aspects of the design assure that ESF operation is not affected by potential coatings failure.

6.1.3 Regulatory Requirements

Regulatory requirements applicable to the PCS with respect to materials are Title 10 Code of Federal Regulations Part 50 (10 CFR 50), Appendix A, "General Design Criteria for Nuclear Power Plants," 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," and 10 CFR 50.55a, "Codes and Standards."

10 CFR 50, Appendix A

In accordance with the guidance of SRP 6.1, the PCS complies with the following General

Design Criteria (GDC), which have plant-wide applicability, as discussed in Section 3.1:

GDC-1 – Quality standards and records.

GDC 4 – Environmental and dynamic effects design bases.

The following GDC have aspects specific to the ESF materials and compliance is addressed as follows:

GDC 14 – Reactor coolant pressure boundary. The ICS forms part of the RCPB and, as such, compliance with GDC 14 with regard to RCPB materials is addressed in Chapter 5.

GDC 31 – Fracture prevention of reactor coolant pressure boundary. The ICS forms part of the RCPB and, as such, compliance with GDC 31 with regard to RCPB materials is addressed in Chapter 5.

GDC 35 – Emergency core cooling. Compliance with GDC 35 relates to ICS function and is addressed in Section 6.3.

GDC 41 – Containment atmosphere cleanup. Compliance with GDC 41 relates to the Containment Inerting System (CIS), which is not an ESF system. The CIS is discussed in Subsection 9.3.6.

Section 3.1 summarizes compliance with GDC.

10 CFR 50, Appendix B

10 CFR 50, Appendix B – Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants: Compliance with 10 CFR 50, Appendix B is addressed in Chapter 17.

10 CFR 50.55a

10 CFR 50.55a – Codes and Standards: Compliance with 10 CFR 50.55a with regard to the PCCS is addressed in Subsection 3.8.1.2 and Section 6.2. Chapter 5 addresses compliance with regard to the RCPB. Compliance with regard to ASME BPVC, Section III, Class 2 and 3 components is addressed in Section 6.6.

6.1.4 Regulatory Guidance

The following RGs and Generic Letter have specific applicability to the design of the PCS. Conformance to RGs related to instrumentation is addressed in Chapter 7.

RG 1.28 – Quality Assurance Program Criteria: The quality assurance program for the BWRX-300 is described in Chapter 17.

RG 1.31 – Control of Ferrite Content in Stainless Steel Weld Metal: The delta ferrite content of weld filler material is controlled in accordance with the guidance of RG 1.31.

RG 1.36 – Nonmetallic Thermal Insulation for Austenitic Stainless Steel: Nonmetallic thermal insulation used on fluid systems performing Safety Category 1 functions is selected to minimize contamination that could promote stress corrosion cracking of the stainless steel surface it contacts.

RG 1.44 – Control of the Processing and Use of Stainless Steel: The control of sensitization and the prevention of stress corrosion of stainless steels for the PCS components conforms to the guidance of RG 1.44.

RG 1.50 – Control of Preheat Temperature for Welding of Low-Alloy Steel: The control of preheat temperature for welding low-alloy steel for PCS components conforms to the guidance of RG 1.50.

RG 1.54 – Service Level I, II, III, and In-Scope License Renewal Protective Coatings Applied to Nuclear Power Plants: Protective coatings are qualified and capable of surviving a DBA without adversely affecting PCS components needed to mitigate the accident in accordance with the guidance of RG 1.54. Additional information on RG 1.54 conformance is provided in Subsections 3.8.1.4.8 and 3.8.2.4.

RG 1.84 – Design, Fabrication, and Materials Code Case Acceptability, ASME Section III: The ASME Code Case for the PCS mechanical design is N-284-4.

RG 1.136 – Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments: The ASME Code Case for the PCS mechanical design is N-632. Additional information about RG 1.136 conformance is provided in Subsection 3.8.1.

Generic Letter 83-02 – NUREG-0737 Technical Specifications contains a request for information for the current BWR licensees regarding NUREG-0737 items for which Technical Specifications are required, including guidance on the scope of a specification which the NRC Staff would find acceptable and sample technical specifications. Technical Specifications for the items related to containment and CIVs will be addressed in the Final Safety Analysis Report (FSAR).

Generic Letter 95-07 – Pressure Locking and Thermal Binding of Safety Related Power Operated Gate Valves contains a request to ensure that safety-related power operated gate valves that are susceptible to pressure locking, or thermal binding are evaluated to ensure they are capable of performing the safety functions described in the licensing basis. This guidance will be evaluated for applicability in the FSAR.

Generic Letter 88-01 – NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping: Conformance to Generic Letter 88-01 regarding IGSCC is addressed in Chapter 5.

Table 6.1-1: Containment Materials

Component	Applicable ASME Code Section III	Form	Material	ASME Code Section II Specification
Steel-Plate Composite Containment Vessel	See Subsection 3.8.1.6			
Containment Closure Head	See Subsection 3.8.2.6			
Penetrations	See Subsection 3.8.2.6			
Containment Airlocks	See Subsection 3.8.2.6			
Containment Internal Structural Steel	See Subsection 3.8.3.6			
PCCS Piping	Class 2, Subsection NCA & NCD	Pipe	Stainless Steel	Seamless SA-312/SA-312M Grade TP304/304L or TP-316/316L
PCCS Fittings	Class 2, Subsection NCA & NCD	Wrought	Stainless Steel	SA-403/SA-403M Grade WP304/304L or WP316/316L
Containment Isolation Valves	Class 1 & 2 Subsection NCA, NB & NCD	Forging	Stainless Steel	SA-182/SA-182M Grade F304/F304L or F316/F316L
Containment Isolation Valves	Class 2, Subsection NCA & NCD	Forging	Carbon Steel	SA-352 Grade LCB or SA-350 Grade LF2
Containment Isolation Valves	Class 2, Subsection NCA & NCD	Forging	Low-Allow Steel	SA-217 Grade WC6 or SA-217 Grade WC9
Reactor Isolation Valves	See Subsection 5.2.1.1 and Subsection 5.2.3.1			

Table 6.1-2: Containment Welding Filler Metals

Base Material	Filler Metal Type	ASME / AWS Specification	AWS Classification (Note 1)
Carbon Steel	Low-Alloy Steel (Covered) Electrodes	SFA-5.5 / AWS A5.5	E8018-X
	Low-Alloy Steel Electrodes and Rods (GMAW and GTAW)	SFA-5.28 / AWS A5.28	ER80S-XX
	Carbon Steel Electrodes and Fluxes (SAW)	SFA-5.17 / AWS A5.17	EM14K
	Carbon Steel Electrodes (FCAW)	SFA-5.20 / AWS A5.29	E7XT-12M-XX
	Low-Alloy Steel Electrodes (FCAW)	SFA-5.29 / AWS A5.29	E8XT1-K2M
Stainless Steel	Stainless Steel Electrodes and Rods (GMAW and GTAW)	SFA-5.9 / AWS A5.9	ER308L, ER316L
	Stainless Steel (Covered) Electrodes	SFA-5.4 / AWS A5.4	ER308L-XX, ER316L-XX

Note:

1. Alternative Specifications and/or Classifications meeting NEDC-33926P-A technical requirements are permitted with Purchaser approval.

6.2 Containment System

The BWRX-300 containment is termed the PCS. The PCS is an ESF that provides a physical barrier against radiological releases to the environment. The PCS is a dry, inerted containment type. Unlike many current generation BWRs, the design does not incorporate pressure suppression or a secondary containment. Containment heat removal during accident conditions is provided by the PCCS. Containment isolation valves, piping, and penetrations are part of the containment boundary.

The BWRX-300 containment performance acceptance criteria include the following:

- The containment pressure boundary and penetrations are designed for the design pressure and temperature to be established for DBAs in accordance with 10 CFR 50, Appendix A, GDC 2, GDC 4, GDC 16, GDC 38, GDC 41, GDC 50, and GDC 51
- In accordance with 10 CFR 50, Appendix A, GDC 4, GDC 16, GDC 38, GDC 41, GDC 50, and GDC 51, containment design pressure is evaluated to bound the peak accident containment pressure resulting from the most limiting large break Loss of Coolant Accident (LOCA) with margin, with no less than 10% margin in order to conform to SRP 6.2.1.1.A Acceptance Criteria.
- In accordance with 10 CFR 50, Appendix A, GDC 16, GDC 38, and GDC 50, the BWRX-300 containment design features establish an essentially leak-tight barrier, and is demonstrated to reduce containment pressure and temperature rapidly, and maintains them at acceptably low levels following a LOCA; and the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from a LOCA.

Subsection 6.2.1 describes the PCS. Subsection 6.2.2 describes the PCCS.

Subsection 6.2.3 establishes that secondary containment functional design is not applicable to the BWRX-300.

Subsection 6.2.4 describes containment isolation.

Subsection 6.2.5 describes control of combustible gases in containment. Subsection 6.2.6 describes containment leakage testing.

Subsection 6.2.7 describes fracture prevention of the containment boundary.

6.2.1 Primary Containment System

The following subsections address PCS design bases, system description, safety evaluation, tests and inspections, and instrumentation requirements.

6.2.1.1 Design Bases (Primary Containment System)

Safety Category 1 functions:

- Essentially leak-tight boundary against radioactivity release
- Structural support for Structures, Systems, and Components (SSCs) contained within the containment structure
- Capability to passively reject heat to the Ultimate Heat Sink (UHS)
- Containment isolation for closed system piping (i.e., PCCS)

Safety Category 2 functions:

None.

Safety Category 3 functions:

None.

Safety Category N functions:

- Personnel access to the containment structure during plant operations that do not require containment integrity
- Personnel access to the RPV head to allow the installation, removal, maintenance, and inspection of fuel and RPV internals during plant operations that do not require containment integrity

Subsection 6.2.1.3 discusses applicable regulatory requirements and guidance for the PCS. Codes and standards related to wind loadings, flooding, missile protection, and pipe rupture are addressed in Sections 3.3 through 3.6, respectively. Codes and standards related to seismic design are addressed in Section 3.7 and Section 3.8. The Safety Classifications, Quality Groups, and Seismic Categories of the PCS SSCs are listed in Table 3A-1.

6.2.1.2 System Description (Primary Containment System)

The PCS is a dry, essentially leak-tight containment which is inerted with nitrogen using the CIS, as discussed in Subsection 9.3.6. The PCS encloses the RPV and is largely below grade; it is housed within the Reactor Building (RB). Containment heat removal during normal operation is provided by the Containment Cooling System (CCS), as discussed in Subsection 9.4.7. During postulated accidents, containment heat is removed by the PCCS. The PCCS is a subsystem of the PCS and is described in Subsection 6.2.2. The PCS is designed to maintain its integrity for at least 72 hours without operator action.

Table 6.2-1 provides key PCS design parameters.

6.2.1.2.1 Major Primary Containment System Components

Primary Containment System components that form the containment boundary are the SCCV, containment closure head, containment penetrations, airlocks, and CIVs. Figure 3.8-1 illustrates the Integrated Reactor Building Structures. Containment isolation valves are discussed in Subsection 6.2.4. Other important aspects of the containment are the containment subcompartments, refueling bellows seal, PCCS, containment internal structures, containment sump, and corium shield.

Steel-Plate Composite Containment Vessel

The SCCV is a concrete and steel structure which functions as a containment boundary as well as a structural support. The SCCV top slab supports the reactor cavity pool and Isolation Condenser (IC) pools. The design of the SCCV is discussed in Subsection 3.8.1.

Containment Subcompartments

The potential for subcompartment differential pressures arising from a design basis event is addressed in NEDC-33922P-A, “BWRX-300 Containment Evaluation Method,” (Reference 6.2-2). Consistent with the evaluation provided in NEDC-33922P-A, large high energy piping is not routed through the containment subcompartments below the RPV, and the containment head area above the refueling bellows seal. Thus, the subcompartment boundaries are not subject to large transient loads resulting from postulated pipe breaks inside the subcompartments. Pipe breaks outside the subcompartments may result in transient loads on the subcompartment boundaries. However,

because the pipe breaks occur in a relatively large volume and subcompartments have large openings, these loads are not large in magnitude. As demonstrated in NEDC-33922P-A, the pressure differential across subcompartment boundaries for the most limiting case is found to be small. The design of the subcompartment includes the differential pressures resulting from pipe breaks outside the subcompartments.

Containment Closure Head

The containment closure head is a removable steel dome that covers the opening in the SCCV top slab over the RPV and is part of the containment pressure boundary. The head is designed for removal during reactor refueling and for replacement prior to reactor operation using the RB polar crane. The closure head is illustrated on Figure 3.8-3, and its design is discussed in Subsection 3.8.2.

Containment Penetrations

The design minimizes the number of containment penetrations while considering separation and redundancy requirements. Penetrations, as summarized below, form part of the containment boundary, and are designed to maintain containment integrity when containment isolation is required. Penetrations through the containment wall are shielded to reduce radiation streaming. Details of the penetration design are provided in Subsection 3.8.2. Isolation of the containment is discussed in Subsection 6.2.4.

Airlocks: There are two airlocks, each of which serves the dual purpose of personnel access to the containment and for movement of equipment. One airlock provides access to the upper portion of the containment and the other provides access to the lower portion of the containment. Each airlock has two pressure-seated doors interlocked to prevent simultaneous opening of both doors and to ensure that one door is completely closed before the opposite door can be opened. The door operation is designed and constructed so either door may be operated from inside the containment, inside the airlock, or from outside the containment.

Mechanical Penetrations: Mechanical penetrations allow transfer of the process medium across the containment boundary when CIVs are open.

Electrical Penetrations: Electrical penetrations allow the passage of continuous electrical cables across the boundary.

Instrument Penetrations: Instrument penetrations allow sensing of plant parameters. Instrument penetrations support ganged sensing lines.

Spare Penetrations: There are spare pipe penetrations with flanged connections for general use.

Refueling Bellows Seal

The refueling bellows seal is a mechanical appurtenance to the RPV that provides a structural barrier to retain the refueling cavity pool water above the PCS when the containment closure head is removed. The refueling bellows seal includes protection from puncture or damage from dropped items when performing RPV or containment closure head removal or installation activities. The design includes a drain to remove water from the bellows low point.

The refueling bellows seal allows a differential movement of the RPV with respect to the containment during normal plant operation caused by thermal expansion of the structures and the RPV. During seismic activity, additional movement is expected and the bellows acts as a buffer or damper so that excessive movement is not transmitted to the RPV from the containment. After operating as a flexible connection during normal plant operation, the refueling bellows provides a watertight seal between the RPV and containment to support a pool of water over the vessel flange of sufficient depth to allow fuel assembly transfer during refueling.

Containment Internal Structures

The major containment internal structures are the RPV pedestal, and the containment internal structural steel. The RPV pedestal is a cylindrical, steel-plate composite structure that structurally supports the RPV. The containment internal structural steel comprises the Containment Equipment and Piping Support Structure. Details of the containment internal structures design are provided in Subsection 3.8.3.

Maintenance structures (e.g., platforms, stairs, and ladders) are provided to support personnel entry and work areas to perform inspections, tests, and maintenance or modifications on equipment within containment.

Containment Sump

A drain line runs from containment basement to the pressurized sump outside of the PCS boundary to collect liquid leakage from the various systems, as described in Subsection 9.3.3.

Corium Shield

To mitigate the consequences of a potential severe accident, the lower portion of PCS and RPV pedestal is provided with a corium shield to prevent potential contact between molten core and containment components.

Passive Containment Cooling System

The PCCS is part of the PCS and is designed to provide containment heat removal in accident conditions. The PCCS is described in Subsection 6.2.2.

6.2.1.2.2 System Operation

The PCS has the following operational alignments:

Power

When the plant is in the power operation alignment, the PCS equipment and system interfaces are as follows:

- The containment closure head is installed
- The containment airlocks are closed
- The PCCS is aligned to allow flow of the equipment pool water through the closed loop within containment
- The containment isolation functionality is operational
- The bulkhead seal air passage openings are open
- Containment monitoring instrumentation used to identify leakage inside of containment is operational
- The CIS is aligned to supply nitrogen into containment and maintain an inerted atmosphere at a slight positive pressure as compared to external pressure
- The CCS is aligned to maintain the containment temperature within limits through use of fans and cooling coils
- The Containment Monitoring Subsystem (CMon) is aligned to monitor hydrogen, oxygen, temperature, pressure, and radiation within containment

Startup

When the plant is in the startup operation alignment, the PCS equipment and system interfaces are the same as the power operation alignment.

Hot Shutdown

When the plant is in the hot shutdown operation alignment, the PCS equipment and system interfaces are as follows:

- The containment closure head is installed
- The containment airlocks are closed
- The PCCS is aligned to allow flow of the equipment pool water through the closed loop within containment
- The containment isolation functionality is operational, including all associated measurement and control functions
- The bulkhead seal air passage openings are open
- Containment monitoring instrumentation used to identify leakage inside of the containment is operational
- The containment is not required to remain inerted and the CIS may be aligned to de-inert the containment atmosphere in order to exhaust the nitrogen and supply fresh breathable air into the containment
- The CCS is aligned to maintain the containment temperature within limits through use of fans and cooling coils
- The CMon is aligned to monitor hydrogen, oxygen, temperature, pressure, and radiation within containment

Cold Shutdown

When the plant is in the cold shutdown operation alignment, the PCS equipment and system interfaces are as follows:

- The containment closure head is removed in this mode
- Once the containment atmosphere is de-inerted and meets habitability requirements, the containment airlocks may be opened to allow access into the containment as needed
- The PCCS is not required to be in service
- The containment isolation functionality is not required
- The bulkhead seal air passage openings are closed and leak tested in preparation for flood-up of the reactor cavity pool
- Containment monitoring instrumentation used to identify leakage inside of the containment is not required to be in service
- The CIS is supplying breathable air from HVS to containment or in an idle state (i.e., not inerting or de-inerting)
- The CCS is aligned for personnel habitability if containment entry is desired, and temperature is maintained within limits through use of fans and cooling coils

- The CMon is aligned to monitor hydrogen, oxygen, temperature, pressure, and radiation within containment

Refueling

When the plant is in the refueling operation alignment, the PCS equipment and system interfaces are as follows:

- The containment closure head is removed in this mode
- The containment airlocks are disabled
- The PCCS is not required to be in service
- The containment isolation functionality is not required
- The bulkhead seal air passage openings are closed so the reactor cavity pool may be filled
- Containment monitoring instrumentation used to identify leakage inside of the containment is not required to be in service
- The CIS is supplying breathable air from HVS to containment or in an idle state (i.e., not inerting or de-inerting)
- The CCS is aligned for personnel habitability if containment entry is desired, and temperature is maintained within limits through use of fans and cooling coils
- The CMon is aligned to monitor hydrogen, oxygen, temperature, pressure, and radiation within containment

Off-normal Operation: If the CCS is not in operation, containment pressure and temperature are maintained within design limits by shutting down the reactor, depressurizing the RPV, and transferring containment heat to the equipment pool and reactor cavity pool by the PCCS and heat loss through the containment head.

During a postulated accident, the majority of the decay heat is removed from the RPV by the ICS, thereby reducing the heat added to the containment. Energy released to containment from a large line break is limited by the RIVs described in Subsection 5.4.3. Energy released to containment from unisolated small breaks is limited by the small break size. The ICS also depressurizes the RPV which reduces break flow to containment. The PCCS removes energy released from the break in containment to maintain pressure and temperature within design limits.

6.2.1.3 Safety Evaluation (Primary Containment System)

The PCS provides a barrier against radiological releases to the environment. The design of the PCS ensures that a potential radioactive release is As Low As Reasonably Achievable (ALARA), below the authorized limits on discharges during operation, and below accident limits. The design minimizes the number of containment penetrations, provides isolation valves in lines that penetrate containment, and provides a means for leak testing of mechanical penetrations and the PCS.

6.2.1.3.1 Regulatory Requirements

Regulatory requirements applicable to the PCS design are 10 CFR 50, Appendix A, 10 CFR 50, Appendix J, “Primary Containment Leakage Testing for Water-Cooled Power Reactors,” 10 CFR 50.34(f), “Additional Three Mile Island (TMI)-related requirements,” and 10 CFR 50.44, “Combustible Gas Control for Nuclear Power Plants.” Compliance with regulations related to the containment function is summarized in the following discussion. Compliance with requirements related to instrumentation is addressed in Chapter 7.

10 CFR 50, Appendix A

In accordance with the guidance of SRP 6.2.1, the PCS complies with the following GDC, which has plant-wide applicability, as discussed in Section 3.1:

GDC 4 – Environmental and dynamic effects design bases

The following GDC have aspects specific to the PCS and compliance is addressed as follows:

GDC 13 – Instrumentation and control. The design of the PCS complies with GDC 13. The PCS is provided with instrumentation to monitor variables important to the containment and its associated systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety. Section 6.8 identifies PCS instrumentation and monitored variables.

GDC 16 – Containment design. The design of the PCS complies with GDC 16. The PCS provides an essentially leak-tight barrier to the uncontrolled release of radioactivity following postulated accidents without operator action for at least 72 hours. The response of the PCS to postulated accidents is provided in Section 15.5.

GDC 50 – Containment design basis. The design of the PCS complies with GDC 50. The containment structure and heat removal system are designed to withstand postulated accident conditions without exceeding the design leakage rate. The methodology for evaluating the effect of potential accidents, including margin, is provided in Section 15.5. Results of the evaluation are summarized in Section 15.7.

GDC-51 – Fracture prevention of containment pressure boundary. The design of the containment boundary with regard to compliance with GDC 51 is addressed in Subsection 6.2.7.

GDC 52 – Capability for containment leakage rate testing. The design of the PCS complies with GDC 52. As described in Subsection 6.2.6, the containment is designed so that the periodic Integrated Leakage Rate Test (ILRT) can be conducted at design pressure.

GDC 53 – Provisions for containment testing and inspection. The design of the PCS complies with GDC 53. As described in Subsections 3.8.1, 6.2.1, 6.2.2, 6.2.4, and 6.3.4, the containment is designed to support periodic inspection of important areas. Subsection 6.2.6 describes containment periodic leakage rate testing.

GDC 64 – Monitoring radioactivity releases. The design of the PCS complies with GDC 64 with regard to monitoring the containment atmosphere. As described in Section 6.8, the containment is monitored for radioactivity during normal operations and off-normal events.

Section 3.1 summarizes compliance with GDC.

10 CFR 50, Appendix J

The design of the PCS complies with 10 CFR 50, Appendix J by providing the capability for leakage rate testing as described in Subsection 6.2.6.

10 CFR 50.34(f)

10 CFR 50.34(f)(2)(ix): The design of the PCS complies with 10 CFR 50.34(f)(2)(ix) by maintaining an inerted atmosphere in containment as discussed in Subsection 9.3.6.

10 CFR 50.34(f)(2)(xv): The design of the PCS complies with 10 CFR 50.34(f)(2)(xv) as the containment is monitored for fission products and area radiation levels. The PCS is provided with the capability for containment purge using the ILRT containment penetration that connects directly to the containment atmosphere and can be used to supply or vent air. Containment leakage rate testing is described in Subsection 6.2.6.

10 CFR 50.34(f)(2)(xvii): The design of the PCS complies with 10 CFR 50.34(f)(2)(xvii). As discussed in Section 6.8 and Section 7.3, the design includes MCR monitoring of instrumentation to measure containment pressure, water level, hydrogen and oxygen concentration, radiation level, and noble gas effluents. Section 7.3 addresses 10 CFR 50.34(f)(2)(xvii) as it relates to instrumentation capable of operating in the post-accident environment to monitor containment parameters.

10 CFR 50.34(f)(2)(xxvi): The design complies with 10 CFR 50.34(f)(2)(xxvi) by provision to isolate containment and test for leak tightness as discussed in Subsection 6.2.6. The program for controlling leakage from systems that could contain radioactive fluids outside containment will be described in the FSAR.

10 CFR 50.34(f)(3)(v)(A)(1): Evaluation of 100% fuel clad metal-water reaction will be addressed in the FSAR.

10 CFR 50.44

The design complies with 10 CFR 50.44 with regard to features designed to withstand the effects of DBAs, specifically:

10 CFR 50.44(c)(1): The design complies with 10 CFR 50.44(c)(1) with limited subcompartments that passively circulate containment atmosphere to prevent combustible gas concentration in a subcompartment.

10 CFR 50.44(c)(2): The design complies with 10 CFR 50.44(c)(2) by maintaining an inerted atmosphere in containment.

10 CFR 50.44(c)(3): The design complies with 10 CFR 50.44(c)(3) by maintaining an inerted atmosphere in containment.

10 CFR 50.44(c)(4): The design complies with 10 CFR 50.44(c)(4) by the use of oxygen and hydrogen analyzers for monitoring containment oxygen and hydrogen concentrations.

10 CFR 50.44(c)(5): The design complies with 10 CFR 50.44(c)(5) through analysis to demonstrate containment structural integrity.

10 CFR 50.55a

Pursuant to 10 CFR 50.55a(f)(4)(i) and 10 CFR 50.55a(g)(4)(i), the initial Inservice Testing (IST) and ISI programs, respectively, will use the edition and addenda of the ASME OM Code, “Operation and Maintenance of Nuclear Power Plants,” (Reference 6.2-5) and of ASME BPVC, Section XI, Division 1, “Rules for Inspection and Testing of Components of Light-Water-Cooled Plants,” (Reference 6.2-6), subject to the conditions identified and incorporated by reference into 10 CFR 50.55a, on the date no more than 18 months before the date of issuance of the Operating License. Accordingly, the Code Editions for the ASME OM Code and ASME BPVC, Section XI, Division 1 will be addressed in the FSAR.

6.2.1.3.2 Regulatory Guidance

The following RGs have specific applicability to the design of PCS. Conformance to RGs related to instrumentation is addressed in Chapter 7.

RG 1.7 – Control of Combustible Gas Concentrations in Containment: The design conforms to the guidance provided in RG 1.7 with regard to the use of an inerted containment. Discussion of combustible gas control is provided in Subsection 6.2.5.

RG 1.163 – Performance-Based Containment Leak-Test Program: RG 1.163 endorses American National Standards Institute (ANSI)/American Nuclear Society (ANS) 56.8-2020, “Containment

System Leakage Testing Requirements" (Reference 6.2-1), for acceptable industry standards on technical methods and techniques for performing Type A, B, and C tests. Discussion of containment leakage testing is provided in Subsection 6.2.6.

6.2.1.4 Mass and Energy Release Analysis for Secondary Steam Rupture (PWR)

The BWRX-300 design does not use a secondary system for feedwater or steam production.

6.2.1.5 Minimum Containment Pressure Analysis for Performance Capability Studies on Emergency Core Cooling System (PWR)

The BWRX-300 design does not rely on containment pressure to achieve emergency core cooling.

6.2.1.6 Tests and Inspections (Primary Containment System)

The PCS has provisions for personnel access during plant outages to perform maintenance, inspections, and tests required for assuring PCS integrity and the reliability of SSCs contained in the PCS boundary. The preoperational and startup test program is described in Chapter 14. The PCS supports periodic:

- ISI of non-isolable welds between penetrations and SCCV wall
- Local leakage rate testing of containment mechanical penetrations
- Integrated containment leakage rate testing
- Inspection of the refueling bellows seal
- Surveillance testing
- Inservice inspections

6.2.1.7 Instrumentation Requirements (Primary Containment System)

Containment parameters are monitored by the CMon. The CMon is discussed in Section 6.8. Refer to Section 7.3 for additional information pertaining to Instrumentation and Control (I&C) system.

6.2.2 Passive Containment Cooling System

The following subsections address PCCS design bases, system description, safety evaluation, tests and inspections, and instrumentation requirements.

6.2.2.1 Design Bases (Passive Containment Cooling System)

Safety Category 1 functions:

- Passive heat rejection to the equipment pool and reactor cavity pool which form the UHS for the PCCS

Safety Category 2 functions:

None.

Safety Category 3 functions:

None.

Safety Category N functions:

None.

Subsection 6.2.2.3 discusses applicable regulatory requirements and guidance for the PCCS. Codes and standards related to wind loadings, flooding, missile protection, and pipe rupture are

addressed in Sections 3.3 through 3.6, respectively. Codes and standards related to seismic design are addressed in Section 3.7 and Section 3.8. The Safety Classifications, Quality Groups, and Seismic Categories of the PCCS SSCs are listed in Table 3A-1.

6.2.2.2 System Description (Passive Containment Cooling System)

The PCCS is a passive system for transferring heat by natural circulation from the containment atmosphere to the reactor cavity pool and equipment pool. The PCCS consists of three independent trains each with a Passive Containment Cooling Pipe Array (PCCPA) which has piping connections to the equipment pool. When the plant is in normal operation, the equipment pool gate is removed, connecting the equipment pool and reactor cavity pool to provide a heat sink for the PCCS. The PCCS transfers containment heat to these pools after a DBA. In the event that normal containment cooling by the CCS becomes inoperable, containment pressure and temperature are maintained within the design limits by shutting down the reactor, depressurizing the RPV, and transferring the heat in containment to the equipment pool and reactor cavity pool by the PCCS and containment head.

The PCCS is always in-service unless portions are manually isolated, i.e., there are no active components, electric power, or actuation signals required to initiate or maintain function. For the most limiting line break inside containment, the PCCS heat removal capacity is sufficient to maintain the containment within design parameters for a period of at least 72 hours following an initiating event without reliance on operator action. The PCCS can achieve its function with a failure of a single train.

Figure 6.2-1 illustrates the configuration of the PCCS.

6.2.2.2.1 Major Passive Containment Cooling System Components

The major components of the PCCS are the PCCPAs, supply lines, and return lines.

Passive Containment Cooling Pipe Array

The PCCPA is a pipe array which contains vertical pipes connected by a top and bottom header. The PCCPAs are located in containment and are connected to the equipment pool by supply and return piping. These lines are sloped down from the equipment pool to the low point in the system on the supply side and sloped up to the outlet on the return side to allow system drainage. The PCCPAs are placed at locations that are not subject to jet loads or are protected from their effects.

PCCS Supply Piping

Supply piping from the equipment pool connects to each PCCPA. Each PCCS train has a CIV on the supply header outside containment as close to the containment boundary as practical. Additionally, each train has a system isolation valve on the supply header outside containment as close to the equipment pool boundary as practical. The function of the CIVs and system isolation valves is to isolate a train in the event of a PCCS pipe leak or to remove a train from service for maintenance or testing. The valves are normally open, fail as-is in the event of a containment isolation signal, and are remotely operated from the MCR. The valves are capable of manual operation.

The supply lines are equipped with strainers in the equipment pool to ensure that the piping remains free of debris. The supply line for each train has a drain valve upstream of the CIV to enable maintenance and testing.

PCCS Return Piping

Return piping from each PCCPA connects to the equipment pool. Each PCCS train has a CIV on the return header outside containment as close to the containment boundary as practical.

Additionally, each train has a system isolation valve on the return header outside containment as close to the equipment pool boundary as practical. The function of the CIVs and system isolation valves is to isolate a train in the event of a PCCS pipe leak or to remove a train from service for maintenance or testing. The valves are normally open, fail-as-is in the event of a containment isolation signal, and are remotely operated from the MCR. The valves are capable of manual operation.

The return lines are equipped with strainers in the equipment pool to ensure that the piping remains free of debris. The return line for each train has a drain valve downstream of the CIV to enable maintenance and testing.

6.2.2.2.2 System Operation

Because it requires no actuation or change in alignment to function, the PCCS configuration is the same for normal and off-normal operation.

Normal Operation

The PCCS is not required to maintain acceptable normal operating containment temperature conditions. The CCS maintains containment temperature during normal operations.

Off-Normal Operation

The PCCS heat removal becomes significant when steam is discharged into the containment following a pipe break. The steam discharged to containment raises containment temperature and increases the steam content and associated condensation on the outside of the PCCS piping. Heat transferred to the PCCS from the containment is removed by the natural circulation of water in single phase flow and rejected to the subcooled water in the equipment pool. Natural circulation is driven by an elevation difference between the supply and return line connections to the equipment pool.

Because the ICS has sufficient capacity to remove decay heat, the majority of the energy during an accident is removed directly from the RPV. PCCS must remove only the energy released from the break to the containment. The energy released from a break is limited for a large break (defined as a break having an area larger than a 0.75-inch (19 mm) equivalent diameter pipe) because such breaks are isolated rapidly by RIV closure. Energy released from unisolated small breaks is limited due to the small break size. Additionally, the ICS depressurizes the RPV making the break flow relatively small within a few hours. Depressurization of the RPV also reduces the RPV temperature, reducing the heat load from hot surfaces. The PCCS has sufficient capacity to reduce the pressure and temperature in the containment, minimizing RPV leakage following an accident and maintaining containment pressure and temperature below the design limits in isolation events.

Chapter 15 provides the plant response to design basis events. Approval of the containment evaluation model in NEDC-33922P-A included two Limitations and Conditions associated with PCCS performance. As stated in Subsection 15.5.4.5, the Loss-of-Coolant Accident (LOCA) Design Basis Analysis demonstrates that Limitation and Condition #3 and Limitation and Condition #4 of NEDC-33922P-A are met.

6.2.2.3 Safety Evaluation (Passive Containment Cooling System)

The PCCS passively provides containment heat rejection to the equipment pool and reactor cavity pool, which form the UHS for the PCCS, in response to a DBA. The UHS is described in Subsection 9.2.5.

Regulatory requirements applicable to the PCCS design are 10 CFR 50, Appendix A,10 CFR 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power

reactors," 10 CFR 50.55a, and 10 CFR 50.46, "Loss of all alternating current power." Compliance with regulations related to the passive containment cooling function is summarized in the following discussion:

10 CFR 50, Appendix A

The following GDC and Principal Design Criteria (PDC) have aspects specific to the PCCS, and compliance is addressed as follows:

PDC 17 – Electric power systems. The PCCS requires neither onsite nor offsite Alternating Current (AC) electric power to provide containment heat removal.

GDC 38 – Containment heat removal. The design of the PCCS complies with GDC 38. The PCCS provides the capability to rapidly remove containment heat after a LOCA consistent with the acceptance criteria defined in Subsection 15.3.1.2. The PCCS can achieve its function with a single failure. As described in Subsection 6.2.2, the PCCS does not require AC or Direct Current (DC) power to function and can meet heat removal requirements in the event of a single failure. Evaluation of the PCS response to postulated accidents is provided in Section 15.5.

GDC 39 – Inspection of containment heat removal system. The design of the PCCS complies with GDC 39. The PCCS is designed to permit periodic inspection of important components to assure system integrity and capability. As discussed in Subsection 6.2.1, the PCS has provisions for personnel access and for habitability during plant outages to perform maintenance, inspections, and tests.

GDC 40 – Testing of containment heat removal system. The design of the PCCS complies with GDC 40. The PCCS does not require active components or electric power to function. The system is designed so that each train can be isolated, and pressure tested during an outage.

GDC 44 – Cooling water. The PCCS transfers containment heat to the UHS during post-accident conditions. Compliance with GDC 44 is addressed in Subsection 9.2.5.

GDC 45 – Inspection of cooling water systems. Compliance with GDC 45 regarding inspection of UHS components is addressed in Subsection 9.2.5.

GDC 46 – Testing of cooling water systems. Compliance with GDC 46 regarding testing of UHS components is addressed in Subsection 9.2.5.

Section 3.1 summarizes compliance with GDC and PDC.

10 CFR 50.46(b)(5)

The PCCS is designed to passively provide long-term containment heat removal after a design basis event. Removal of containment heat does not require a containment spray, sump, external water source, or active components.

10 CFR 50.55a

Pursuant to 10 CFR 50.55a(f)(4)(i) and 10 CFR 50.55a(g)(4)(i), the initial IST and ISI programs, respectively, use the edition and addenda of the ASME OM Code and of ASME BPVC, Section XI, subject to the conditions identified and incorporated by reference into

10 CFR 50.55a, on the date no more than 18 months before the date of issuance of the Operating License. Accordingly, the Code Editions for the ASME OM Code and ASME BPVC, Section XI, Division 1 will be addressed in the FSAR.

10 CFR 50.63

The design of the PCCS supports the capability of the BWRX-300 to withstand a Station Blackout (SBO) as defined by 10 CFR 50.2. The PCCS for containment depressurization and heat removal does not require the onsite or offsite electric power system, or battery-backed DC power. Section 8.4 summarizes compliance with 10 CFR 50.63; Section 15.5 provides the results of the Loss-of-Preferred Power evaluation.

6.2.2.4 Tests and Inspections (Passive Containment Cooling System)

The PCCS is designed to support periodic testing and inspection. The PCCS has the capability of isolating a train from the equipment pool and draining for maintenance and inspection. The preoperational and startup test program is described in Chapter 14. The PCCS supports:

- Periodic pressure testing
- Periodic stroke testing of normally open valves during plant operation
- Local leakage rate testing of containment penetrations
- Surveillance testing
- Inservice Inspections and testing

6.2.2.5 Instrumentation Requirements (Passive Containment Cooling System)

No actuation signals for PCCS are required. PCCS availability is indicated by equipment pool level and temperature associated with the Fuel Pool Cooling and Cleanup System (FPC), which is described in Subsection 9.1.3. Containment isolation valves, which have MCR indication, are described in Subsection 6.2.4. There is also valving for PCCS system isolation with MCR indication. Section 7.3 addresses I&C functions.

6.2.3 Secondary Containment Functional Design

The BWRX-300 does not employ a secondary or dual containment.

6.2.4 Containment Isolation Function

The containment isolation function supports the capability of the containment to maintain an essentially leak-tight barrier to radionuclide release from postulated accidents while allowing the passage of fluids necessary to mitigate such events. The following subsections address the containment isolation function design bases, system description, safety evaluation, tests and inspections, and instrumentation requirements.

6.2.4.1 Design Bases (Containment Isolation)

The containment isolation function is achieved by systems interfacing with the RPV or containment. Systems are listed under multiple safety categories based on their actuation by multiple defense lines as discussed in Chapter 7.

The following systems perform a Safety Category 1 containment isolation function:

- Nuclear Boiler System (NBS)
- Containment Monitoring Subsystem (CMon)
- Isolation Condenser System (ICS)
- Boron Injection System (BIS)
- Control Rod Drive System (CRD)

- Reactor Water Cleanup System (CUW)
- Condensate and Feedwater Heating System (CFS)
- Chilled Water Equipment (CWE)
- Plant Pneumatics System (PPS)
- Containment Inerting System (CIS) with the exception of the containment overpressure vent path
- Equipment and Floor Drain System (EFS)
- Water, Gas, and Chemical Pads (WGC)
- Primary Containment System (PCS)
- Liquid Waste Management System (LWM)

The following systems perform a Safety Category 2 containment isolation function:

- Nuclear Boiler System (NBS)
- Containment Inerting System (CIS) with the exception of the containment overpressure vent path
- Reactor Water Cleanup System (CUW)
- Chilled Water Equipment (CWE)
- Equipment and Floor Drain System (EFS)
- Plant Pneumatics System (PPS)
- Condensate and Feedwater Heating System (CFS)
- Shutdown Cooling System (SDC) isolation (ICS interface)

The following systems perform a Safety Category 3 containment isolation function:

None.

The following systems perform a Safety Category N containment isolation function:

None.

6.2.4.2 System Description (Containment Isolation)

The containment isolation function assures containment integrity while allowing the passage of fluids across the containment boundary as needed to mitigate a postulated accident. To support isolation of fluid lines penetrating containment, CIVs are included in each process system line and instrumentation line. Containment penetrations are designed to maintain structural integrity during the extreme environmental conditions resulting from a postulated accident and protect against leakage from the containment. Design of containment penetrations is discussed in Subsection 3.8.2.

6.2.4.2.1 Major Containment Isolation Components

The major components supporting the containment isolation function are the CIVs. There are three categories of CIVs: CIVs associated with lines connecting to the RCPB, CIVs associated with lines connecting to the containment atmosphere, and CIVs associated with closed systems.

Mechanical systems with lines penetrating the containment boundary support containment integrity by being capable of isolating and limiting the radionuclide release from a postulated accident. The CIV configuration for containment isolation of process lines is indicated in Table 6.2-2. Table 6.2-3 summarizes isolation of instrument lines. Table 6.2-4 provides the configuration of CIVs during normal, hot shutdown, post-accident, and loss of motive power conditions and GDC compliance.

As indicated in Table 6.2-2, the two RIVs, which are in a single valve body, are considered a single CIV inside containment. The RIV design is discussed in Subsection 5.4.3. Isolation valves outside of containment are located as close to the containment boundary as practical.

6.2.4.2.2 Containment Isolation Operation

With the exception of passive mechanisms such as blank flanges, rupture disks, and locked closed valves, containment isolation occurs as a result of automatic actuation signals or remote manual actuation. Design of the control functions for automatic CIVs ensures that resetting the isolation signal does not result in the reopening of the CIV until the signal is removed and deliberate operator action is taken (e.g., by operation of a reset switch).

The CIVs, with exception of the CIS overpressure vent line isolation valve, close on receipt of a containment isolation signal from the Safety Class 1 (SC1) and Safety Class 2 (SC2) I&C systems. Also, CIVs fail closed on loss of air or control signal, except for the CIS overpressure vent line isolation valve, which fails open to allow for automatic venting of containment during a Design Extension Condition (DEC) overpressurization event sequence.

The CIS overpressure vent line, which consists of a rupture disk, rupture disk bypass valve, and downstream isolation valve meets the alternate acceptance criteria for GDC 56. The rupture disk is used in place of an automatic isolation valve and is considered a sealed-closed barrier. The downstream isolation valve is a normally open, fail open valve that does not automatically isolate. Maintaining this valve open ensures the availability of the containment vent flow path to maintain containment integrity under DEC overpressurization event sequence conditions and is considered the position of greatest safety. By preserving the Safety Category 3 function performed by the overpressure vent line, the CIS containment isolation alternative approach achieves at least an equivalent level of safety compared to the GDC.

Valve closure times are established to minimize radiological releases associated with a postulated accident. Minimizing potential effects of containment flooding due to a pipe break is also considered when determining closure times.

Normal Operation: When the plant is operating at power, CIVs associated with process lines supporting power operation are open. Table 6.2-4 summarizes the normal operation and hot shutdown CIV valve configurations. The table also summarizes compliance with GDC 55, 56, or 57, as applicable.

Off-Normal Operation: CIVs are designed with valve actuators having positive mechanical means (i.e., without power, motive force, or operator actions) to ensure that upon automatic actuation or a loss of signal or control power, the valves are maintained in the required post-accident valve position of greater safety. Upon receipt of a containment isolation signal, CIVs move to their post-accident configuration as indicated in Table 6.2-4. The table also indicates the CIV configuration on loss of motive power, which corresponds to the position of greater safety.

Chapter 15 describes the modeling of containment isolation in the plant response to design basis events.

6.2.4.3 Safety Evaluation (Containment Isolation)

The containment isolation function supports the PCS as an essentially leak-tight barrier to an uncontrolled radionuclide release. As such, the containment isolation function, including provisions for control, indication, and performance under loss/restoration of power conditions, is instrumental in maintaining containment barrier integrity and assuring essential flow paths for reactor core cooling.

6.2.4.3.1 Regulatory Requirements

Regulatory requirements applicable to containment isolation are 10 CFR 50, Appendix A, 10 CFR 50.34(f), 10 CFR 50.55a, and 10 CFR 50.63. Compliance with regulations related to the containment isolation function is summarized in the following discussion. Compliance with requirements related to instrumentation is addressed in Chapter 7.

10 CFR 50, Appendix A

In accordance with the guidance of SRP 6.2.4, the containment isolation function complies with the following GDC, which have plant-wide applicability, as discussed in Section 3.1:

GDC-1 – Quality standards and records

GDC 2 – Design bases for protection against natural phenomena

GDC 4 – Environmental and dynamic effects design bases

The following GDC have aspects specific to the containment isolation function and compliance is addressed as follows:

GDC 16 – Containment design: The design of the containment isolation function complies with GDC 16. The CIV design complies with the requirements of GDC 16 by preserving the capability of the containment boundary to prevent or limit the escape of fission products to the environment from postulated accidents. Section 15.5 demonstrates that containment pressures and temperatures are maintained below design values for 72 hours after postulated accidents.

GDC 54 – Piping systems penetrating containment: The CIV design complies with GDC 54. Piping systems penetrating the PCS are provided with isolation and containment capabilities consistent with their safety function. The isolation configuration is redundant where appropriate. As discussed in Subsection 6.2.4.4, the design includes the capability for leak detection and periodic testing.

GDC 55 – Reactor coolant pressure boundary penetrating containment: The CIV design complies with GDC 55. Lines that are part of the RCPB and that penetrate the PCS are provided with CIVs consistent with GDC 55 unless another basis is defined. Isolation valves outside containment are located as close to containment as practical and upon loss of actuating power, automatic isolation valves are designed to take the position that provides greater safety.

GDC 56 – Primary containment isolation: The CIV design complies with GDC 56. Lines that connect directly to the containment atmosphere and penetrate reactor containment are provided with CIVs consistent with GDC 56 unless another basis is defined. Isolation valves outside containment are located as close to containment as practical and upon loss of actuating power, automatic isolation valves are designed to take the position that provides greater safety.

GDC 57 – Closed system isolation valves: The CIV design complies with GDC 57. Each line that penetrates the containment and is not connected to the containment atmosphere and is not part of the RCPB has at least one isolation valve outside containment. Isolation valves outside containment are located as close to containment as practical.

Section 3.1 summarizes compliance with GDC.

10 CFR 50.34(f)

10 CFR 50.34(f)(2)(xiv): The containment isolation function provides automatic isolation and two isolation barriers in series for non-essential systems. Isolation valves do not reopen automatically upon resetting the isolation signal. Containment pressure setpoints as low as compatible with normal operation will be addressed in the FSAR. Automatic closing on high radiation will be addressed in the FSAR.

10 CFR 50.34(f)(2)(xv): The PCS is maintained dry and nitrogen-inerted above atmospheric pressure during operation. Thus, BTP 6-4, which relates to purging during normal operation, is not applicable to the design.

10 CFR 50.55a

10 CFR 50.55a: Pursuant to 10 CFR 50.55a(f)(4)(i) and 10 CFR 50.55a(g)(4)(i), the initial IST and ISI programs, respectively, use the edition and addenda of the ASME OM Code and of ASME BPVC, Section XI, subject to the conditions identified and incorporated by reference into 10 CFR 50.55a, on the date no more than 18 months before the date of issuance of the Operating License. Accordingly, the Code Editions for the ASME OM Code and ASME BPVC, Section XI, Division 1 will be addressed in the FSAR.

10 CFR 50.63

10 CFR 50.63(a)(2) - Loss of All Alternating Current Power: The containment isolation function complies with 10 CFR 50.63(a)(2) as the CIVs do not require offsite or standby diesel generator AC power to function and actuators are designed to hold their failure position for a minimum of 72 hours without offsite or standby diesel generator AC power.

6.2.4.3.2 Regulatory Guidance

The following RGs have specific applicability to the design of containment isolation. Conformance to RGs related to instrumentation is addressed in Chapter 7.

RG 1.11 – Instrument Lines Penetrating the Primary Reactor Containment: The CIV design conforms to the guidance in RG 1.11 regarding compliance with GDC 55 and GDC 56. Instrument lines penetrating containment are provided with an excess flow check valve or a valve that can be manually operated. The valves are located outside of containment.

RG 1.141 – Containment Isolation Provisions for Fluid Systems: The CIV design conforms to the guidance in RG 1.141. The containment isolation function complies with GDC 54, 55, 56, and 57. The design also implements the guidance provided in RG 1.141 with regard to redundancy, reset of actuation signals, and MCR indication.

RG 1.155 – Station Blackout: The CIV design conforms to the guidance in RG 1.155. No offsite or standby diesel generator AC power is required to place or maintain CIVs in their safety configuration. As discussed in Subsection 6.2.4.5, position indication in the MCR is provided for power-operated CIVs.

6.2.4.4 Tests and Inspections (Containment Isolation)

CIVs have individual leakage testing performed to support the overall requirement to verify containment leakage is within the allowable limits. Process system lines penetrating containment contain test connections to support leakage testing of the CIVs. Valve testing includes exercise tests, seat leakage tests, remote position indicator tests, and stroke testing as detailed in ASME OM Code, “Operation and Maintenance of Nuclear Power Plants,” (Reference 6.2-5). Closure of

CIVs for leakage test is accomplished by normal means and without preliminary exercising or adjustment. Details of containment leakage testing are provided in Subsection 6.2.6.

6.2.4.5 Instrumentation Requirements (Containment Isolation)

Power operated CIVs have position indication in the MCR. Section 15.5 provides the analytical modeling of containment isolation on high containment pressure. Refer to Section 7.3 for additional information pertaining to I&C systems.

6.2.5 Combustible Gas Control in Containment

As discussed in earlier sections, the BWRX-300 design employs a dry, inerted containment with limited subcompartments that have large openings. As discussed in NEDC-33922P-A, radiolytic gases mixed with the steam and liquid are released to containment in the event of a LOCA. The combustion potential from these gases is not a concern for large breaks because the rapid closure of the RIVs terminates the break flow and the release of radiolytic gases. Radiolytic gases may build up in the containment following unisolated small breaks over time although the rate of release is small. Analyses demonstrate the subcompartments are sufficiently mixed and that gases generated by radiolysis do not create a deflagration concern.

Evaluation of BWRX-300 containment structural integrity to demonstrate the survivability of the containment under structural loads generated by an accident in which 100% fuel clad-coolant reaction accompanied by hydrogen burning will be evaluated in the FSAR using the methodology described in NEDC-33926P-A, “BWRX-300 Steel-Plate Composite (SC) Containment Vessel (SCCV) and Reactor Building Structural Design,” (Reference 6.2-4).

Containment inerting is maintained by the CIS as described in Subsection 9.3.6, which addresses compliance with GDC 41. The containment design includes the capability to monitor containment hydrogen and oxygen concentration as discussed in Section 6.8.

6.2.6 Containment Leakage Testing

The containment is designed to be leak tested in accordance with 10 CFR 50, Appendix J. Test requirements for measuring the integrated containment leakage rate, “Type A” tests, are defined by American National Standards Institute (ANSI)/American Nuclear Society (ANS)-56.8-2020, “Containment System Leakage Testing Requirements,” (Reference 6.2-1). Requirements for local leakage rate testing of pressure-containing and leakage limiting boundaries, “Type B” tests, are defined by ANSI/ANS-56.8-2020. Requirements for local leakage rate testing of CIVs, “Type C” tests, are defined by ANSI/ANS-56.8-2020. Discussion on implementing Option B (performance-based) of 10 CFR 50 Appendix J in accordance with RG 1.163 guidance will be provided in the FSAR. The containment leakage test pressure shall be equal to, at minimum, the positive design pressure, which is provided in Table 6.2-1. Preoperational testing descriptions will be provided in Chapter 14 of the FSAR.

The tests assure that:

1. Leakage through the containment and systems and components penetrating containment does not exceed allowable leakage rate values as specified in the Technical Specifications; and
2. Periodic surveillance of containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during service life of the containment, and systems and components penetrating containment.

6.2.6.1 Integrated Leakage Rate Test (Type A)

After construction of the PCS, including installation of portions of mechanical, electrical, and instrumentation systems penetrating the PCS pressure boundary, and upon satisfactory completion of structural integrity testing described in Subsection 3.8.1.7.1, the initial (preoperational) ILRT is performed to verify that the containment leakage rate does not exceed the design limit. Closure of CIVs for Type A test is accomplished by normal means and without preliminary exercising or adjustment. Type B and Type C leakage rate tests are completed prior to the preoperational Type A test. To establish a point of reference for future in-service leak tests, the ILRT performed during commissioning is conducted at a pressure consistent with the pressure selected for the periodic in-service leak tests.

Following the preoperational test, ILRTs are conducted periodically according to 10 CFR 50, Appendix J to ensure containment leakage remains less than the maximum allowable leakage rate and to identify adverse trends since previous ILRT. The tests are performed at established intervals, after major repairs, and upon indication of excessive leakage. The periodic ILRTs follow the same method and acceptance criteria as the preoperational ILRT.

6.2.6.2 Containment Penetration Leakage Rate Test (Type B)

Preoperational and periodic Type B leakage rate tests are performed on containment penetrations in accordance with 10 CFR 50, Appendix J. Containment penetrations whose designs incorporate resilient seals, bellows, gaskets, or sealant compounds; airlocks and airlock door seals; equipment and access hatch seals; and electrical penetration canisters are subject to this test.

6.2.6.3 Containment Isolation Valve Leakage Rate Test (Type C)

Preoperational and periodic Type C tests are performed on CIVs and RIVs that are the Inside Containment CIV in accordance with 10 CFR 50, Appendix J. Type C tests are performed by local pressurization. The pressurization is applied in the same direction as when the valve is required to perform its safety function, unless it can be determined that the results from the tests for a pressure applied in a different direction provides equivalent or more conservative results. Each valve to be tested is closed by normal operation without preliminary exercising or adjustment.

6.2.6.4 Scheduling and Reporting of Periodic Tests

Containment is designed to permit appropriate periodic inspection of CIVs, penetrations, and pressure-retaining or leakage-limiting boundaries other than valves. Periodic Type A tests are performed in regular intervals following the completion of preoperational Type A test. Type B and Type C leakage rate tests are completed prior to the periodic Type A test.

Test results, analyses, and structural conditions of tested SSCs are documented in test reports.

6.2.6.5 Special Testing Requirements

This subsection is not applicable because a dual containment is not used in the BWRX-300 design.

6.2.7 Fracture Prevention of Containment Pressure Boundary

As discussed in Subsection 6.2.1.2, the containment is designed to provide an essentially leak-tight barrier to the release of radioactivity. As such, it is designed to assure that it is not subject to brittle fracture under loadings during operating, maintenance, testing and postulated accident conditions. The design of the containment boundary with regard to compliance with GDC 16 and GDC 51 is addressed in Section 3.8 and NEDC-33926P-A. The evaluation provided in Section 15.5 demonstrates that PCS temperature and pressure after a postulated accident are maintained within design parameters for at least 72 hours without operator action.

6.2.8 References

6.2.1 ANSI/ANS-56.8, “Containment System Leakage Testing Requirements,” American National Standards Institute/American Nuclear Society, 2020.

6.2.2 NEDC-33922P-A, “BWRX-300 Containment Evaluation Method,” Revision 3, GE-Hitachi Nuclear Energy Americas, LLC, June 2022.

6.2.3 NEDC-33911P-A, “BWRX-300 Containment Performance,” Revision 3, GE-Hitachi Nuclear Energy Americas, LLC, January 2022.

6.2.4 NEDC-33926P-A, “BWRX-300 Steel-Plate Composite (SC) Containment Vessel (SCCV) and Reactor Building Structural Design,” Revision 4, GE Vernova Hitachi Nuclear Energy Americas, LLC, December 2025.

6.2.5 ASME OM Code, “Operation and Maintenance of Nuclear Power Plants,” American Society of Mechanical Engineers.

6.2.6 ASME BPVC, Section XI, Division 1, “Rules for Inspection and Testing of Components of Light-Water-Cooled Plants,” American Society of Mechanical Engineers.

Table 6.2-1: Primary Containment System Key Design Parameters

Parameter	Value (Note 1)
Minimum Free Volume	245,790 ft^3(6960 m^3)
Containment Height, excluding the closure head	124.6 ft (38.0 m)
Containment Diameter (outer)	63.3 ft (19.3 m)
Containment Diameter (inner)	57.3 ft (17.4 m)
Containment Closure Head Diameter	22.3 ft (6.8 m)
Containment Atmosphere	Nitrogen with less than 4% Oxygen
Normal Pressure	0.25 - 1.30 psig (1.7 - 9.0 kPaG)
Design Internal Pressure	60.0 psig (413.7 kPaG)
Design Temperature	330 °F (165.5 °C)
Max. Negative Pressure	-2 psig (-13.8 kPaG)
Containment Design Leakage Rate (L_a) (Note 2)	0.35 wt%/day (Note 3)

Notes:

1. Values are based on preliminary design and are approximate.
2. L_a: maximum allowed mass leakage rate.
3. The containment design leakage rate is limited to less than 0.35% of the weight in the containment free volume per day at rated design pressure.

Table 6.2-2: Process Line Isolation

Line	Inside Containment CIV (Note 1)	Outside Containment CIV (Note 1)
Nuclear Boiler System		
Main Steam Line Train A, B	RIV, RIV	CIV
RPV Head Vent	RIV, RIV	CIV (Note 2)
Containment Monitoring Subsystem		
H2 & O2 Sample Panel Train A, B Return (Note 3)	CIV	CIV
H2 & O2 Sample Panel Train A, B Supply (Note 3)	CIV	CIV
PING Skid Return	CIV	CIV
PING Skid Supply	CIV	CIV
Isolation Condenser System		
Steam Supply Train A, B, C	RIV, RIV (Note 4)	N/A (Note 5)
Condensate Return Train A, B	RIV, RIV (Note 4)	N/A (Note 5)
Condensate Return Train C	RIV, RIV (Note 4)	N/A (Note 5)
Standby Gas Purge Line Train A, B, C	CIV, CIV	CIV (Note 2)
SDC Train A, B from Condensate Return Train A, B (Note 3)	RIV, RIV (Note 6)	CIV (Note 7)
Boron Injection System		
Boron Injection	CIV (Note 8)	CIV
Control Rod Drive		
Hydraulic Control Unit (each)	CIV (Note 9)	CIV (Note 10)
Reactor Water Cleanup System		
Reactor Water Cleanup Train A, B (Note 3)	RIV, RIV (Note 4)	CIV
Condensate and Feedwater Heating System		
Feedwater Train A (Nozzle A, B) (Note 11)	RIV, RIV (Note 4)	CIV
Feedwater Train B (Nozzle C, D) (Note 11)	RIV, RIV (Note 4)	CIV
Chilled Water Equipment		
Chilled Water Supply Train A, B (Note 3)	CIV (Note 8)	CIV
Chilled Water Return Train A, B (Note 3)	CIV	CIV
Plant Pneumatics System		
Service Air	CIV (Note 12)	CIV (Note 12)
Instrument Air (including Nitrogen Supply)	CIV (Note 8)	CIV
Containment Inerting System		
Inlet to Containment	N/A	CIV, CIV
Outlet from Containment to Plant Vent Stack	N/A	CIV, CIV

Line	Inside Containment CIV (Note 1)	Outside Containment CIV (Note 1)
Overpressure Vent (Note 13)	N/A	CIV, CIV
Equipment and Floor Drain System		
Containment Floor Drain to Pressurized Sump	N/A	CIV, CIV
Water, Gas, and Chemical Pads		
Demineralized Water	CIV (Note 12)	CIV (Note 12)
Passive Containment Cooling system		
Supply Cold Leg Train A, B, C	N/A	CIV
Return Hot Leg Train A, B, C	N/A	CIV
Liquid Waste Management System		
Utility Washdown Hose Connection	CIV (Note 12)	CIV (Note 12)
Primary Containment System		
Containment ILRT	N/A	YBV, FSFG
Spare Penetration	FSFG	FSFG
Temporary Outage Cables & Hoses	FSFG	FSFG
FMCRD Port	FSFG	FSFG

Notes:

1. CIVs have power operators unless noted otherwise.
2. Outside containment CIV is MSCIV
3. Each train has independent CIVs inside and outside containment.
4. RIVs are part of NBS.
5. Outside containment CIV to/from IC is not required to meet the intent of GDC 55.
6. Inside containment CIV is ICS Condensate Return RIVs.
7. Although there are additional system valves in the SDC line, the isolation valve closest to the containment is the outside containment CIV.
8. Check valve
9. Fine Motion Control Rod Drive (FMCRD) internal ball check valve.
10. Manual isolation valve
11. Each nozzle has independent RIVs
12. Locked closed manual valve
13. The inboard CIV is comprised of a FC valve in parallel with a rupture disk. The outboard CIV does not close on containment isolation signal.

Legend:

PING: Particulate, Iodine and Noble Gas

YBV: Manual Isolation Valve

CKV: Check Valve

CIV: Containment Isolation Valve

RIV: Reactor Isolation Valve

FSFG: Blind Flange

Table 6.2-3: Instrument Line Isolation

Description	Manual Isolation Valve	Excess Flow Check Valve
Nuclear Boiler System		
Head Flange Seal Leak Detection	YBV	CKV
Head Vent Shutdown Water Level Variable Leg	YBV	CKV
RPV Level Reference Leg	YBV	CKV
RPV Level Variable Leg	YBV	CKV
Containment Monitoring Subsystem		
Containment Pressure Monitoring	YBV	CKV
Containment Water Level Monitoring	YBV	CKV
Isolation Condenser System		
Differential Pressure Transmitter High Side	YBV	CKV
Differential Pressure Transmitter Low Side	YBV	CKV

Legend:

CKV: Excess Flow Check Valve
YBV: Manual Isolation Valve

Table 6.2-4: Process Line Isolation Configuration and GDC Compliance

System/GDC Compliance Basis	Description	CIV Position			
		Normal	Hot Shutdown	Post-Accident	Loss of Motive Power
NBS GDC 55	MSL Train A and B	Open	Open/Closed (Note 6)	Closed	Closed
	RPV Head Vent	Open	Open/Closed (Note 6)	Closed	Closed
CMon GDC 56	Hydrogen and Oxygen Sample Panel Train A, B Return	Open	Open	Open	As-Is
	Hydrogen and Oxygen Sample Panel Train A, B Supply	Open	Open	Open	As-Is
	PING Skid Return	Open	Open	Open	As-Is
	PING Skid Supply	Open	Open	Open	As-Is
ICS GDC 55 (Note 2)	Steam Supply Train A, B, and C	Open	Open	Open	As-Is
	Condensate Return Train A, B (Note 5)	Open	Open	Open	As-Is
	Condensate Return Isolation C (Note 5)	Open	Open	Open	As-Is
	Standby Gas Purge Lines	Open	Open	Closed	Closed
	SDC Supply Train A, B from Condensate Return Train A, B	Closed	Open/Closed (Note 6)	Closed	Closed
BIS GDC 55	Boron Injection	Closed	Closed	Closed	Closed
CRD GDC 55	Hydraulic Control Unit (Note 1)	N/A	N/A	N/A	N/A
CUW GDC 55	Reactor Water Cleanup Train A, B	Open	Open	Closed	Closed
CFS GDC 55	Feedwater Line Train A (Nozzle A, B)	Open	Open/Closed (Note 6)	Closed	Closed
	Feedwater Line Train B (Nozzle C, D)	Open	Open/Closed (Note 6)	Closed	Closed
CWE GDC 56	Chilled Water Supply Train A, B	Open	Open	Closed	Closed
	Chilled Water Return Train A, B	Open	Open	Closed	Closed
PPS GDC 56	Service Air	Locked Closed	Locked Closed	Locked Closed	N/A
	Instrument Air (includes Nitrogen Supply)	Open	Open	Closed	Closed

System/GDC Compliance Basis	Description	CIV Position			
		Normal	Hot Shutdown	Post-Accident	Loss of Motive Power
CIS GDC 56 (Note 3)	Inlet to Containment	Open/ Closed (Note 6)	Open/Closed (Note 6)	Closed	Closed
	Overpressure Vent Downstream	Open	Open	Closed	Open
	Overpressure Vent Disc Bypass	Closed	Closed	Closed	Closed
	Outlet from Containment to Plant Vent Stack	Open/ Closed (Note 6)	Open/Closed (Note 6)	Closed	Closed
EFS GDC 56 (Note 4)	Containment Floor Drain to Pressurized Sump	Open	Open	Closed	Closed
WGC GDC 56	Demineralized Water	Locked Closed	Locked Closed	Locked Closed	N/A
PCCS GDC 57	Supply Train A, B, C	Open	Open	As-Is	As-Is
	Return Train A, B, C	Open	Open	As-Is	As-Is
LWM GDC 56	Utility Washdown Hose Connection	Locked Closed	Locked Closed	Locked Closed	N/A
PCS GDC56	Containment Pressurization Test Penetration	Closed	Closed	Closed	N/A
	Spare Penetration	Closed	Closed	Closed	N/A
	Temporary Outage Cables, Hoses	Closed	Closed	Closed	N/A
	FMCRD Port	Closed	Closed	Closed	N/A

Notes:

1. Consistent with GDC 55 on “other defined basis,” keeping flow path open is necessary to perform scram function and comparing risk between a degraded scram function and the negative radioactive fission product releases without an outboard CIV, the scram function demonstrates higher risk significance; thus, automatic isolation is not required.
2. Consistent with GDC 55 on “other defined basis,” for a postulated ICS piping failure inside containment, the RIVs function as inboard CIVs for both steam supply and condensate return piping and the closed-loop piping outside containment can prevent radioactive releases outside containment.
3. Consistent with GDC 56 on “other defined basis,” locating a CIS valve inside containment when needed for accident mitigation is not practical.
4. Consistent with GDC 56 on “other defined basis,” locating a EFS valve inside containment above the floor elevation is not practical.
5. The containment isolation valves for the condensate return lines are the RIVs, which are separate from the ICS Condensate Return valves that are used to place the ICS in operation.
6. Operational considerations.

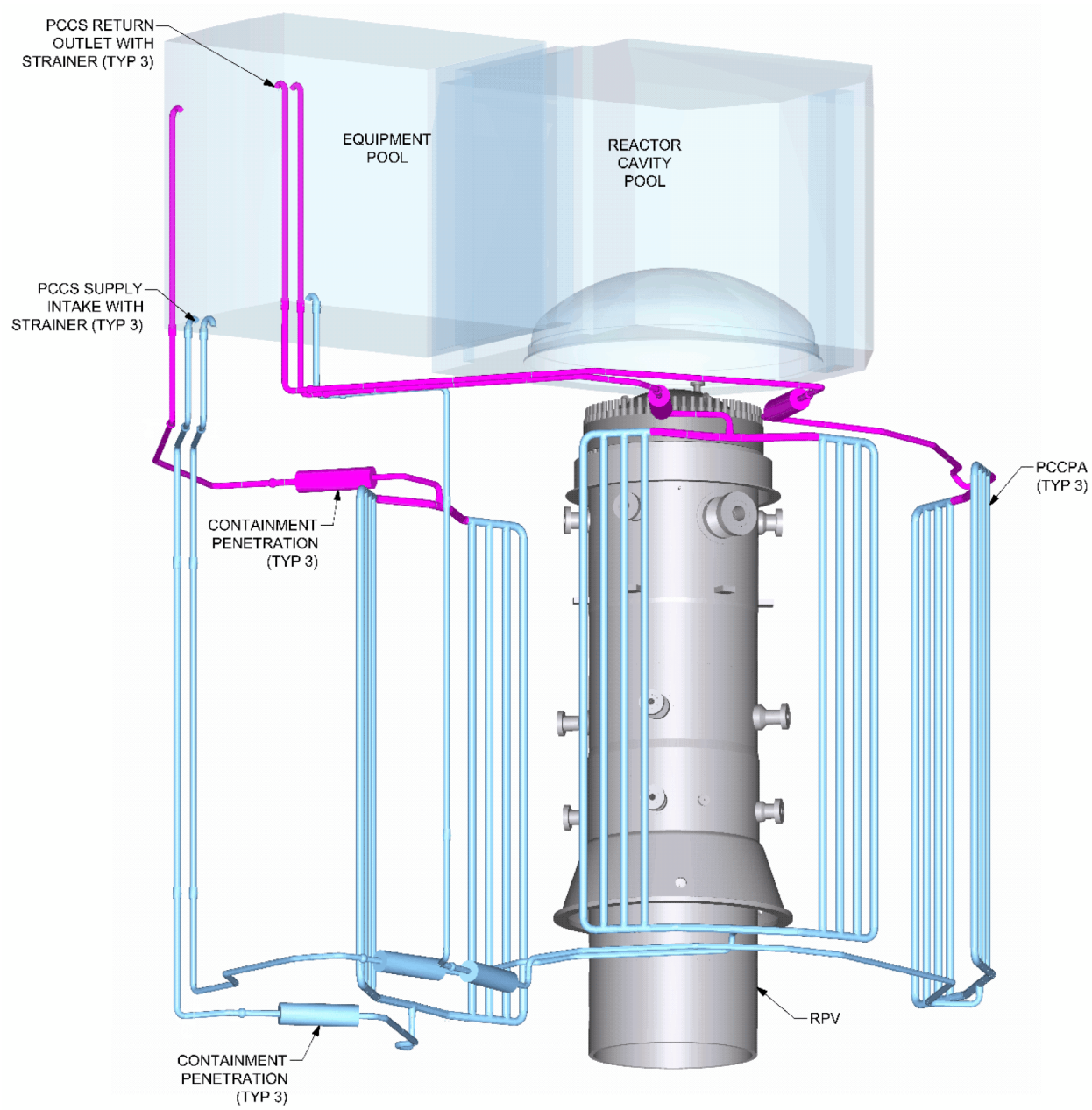


Figure 6.2-1: Passive Containment Cooling System Configuration

Note:

Valves are not shown.

6.3 Emergency Core Cooling System (Isolation Condenser System)

The RIVs together with the ICS comprise the ECCS. The concept for emergency core cooling is based on preserving RPV coolant inventory, removing decay heat using the ICS, and returning the reactor coolant to the RPV. Closure of the RIVs in response to a postulated pipe break limits the loss of RPV inventory to support successful ICS function. The RIVs are discussed in Subsection 5.4.3. The ICS, in conjunction with the reactor scram function, also provides overpressure protection for the RCPB.

Subsection 6.3.1 summarizes ICS functions and provides the design bases of the ICS. Subsection 6.3.2 describes the ICS design and operation.

Subsection 6.3.3 evaluates the ICS in terms of regulatory requirements and guidance. Subsection 6.3.4 summarizes ICS tests and inspections.

Subsection 6.3.5 summarizes instrumentation requirements associated with the ICS.

The following subsections address ICS design bases, system description, safety evaluation, tests and inspections, and instrumentation requirements.

6.3.1 Design Bases (Isolation Condenser System)

Some functions are listed under multiple safety categories based on their actuation by multiple defense lines as discussed in Chapter 7.

Safety Category 1 functions:

- Decay heat removal
- RCPB integrity
- ICS line break detection
- Closure of standby gas purge valves
- Isolation of SDC

Safety Category 2 functions:

- Decay heat removal
- ICS line break detection
- Closure of standby gas purge valves
- Isolation of SDC

Safety Category 3 functions:

- Closure of standby gas purge valves
- Prevention of non-condensable gas accumulation with ICS in standby

Safety Category N functions:

None.

The ICS is part of the RCPB and is the ASME BPVC-required overpressure protection system by functioning in conjunction with a reactor scram. The Safety Classifications, Quality Groups, and Seismic Categories of the ICS SSCs are listed in Table 3A-1.

6.3.2 System Description (Isolation Condenser System)

The ICS is a three-train system that provides the safety functions of emergency core cooling and overpressure protection for the RCPB. The simplicity and largely passive nature of the ICS design results in high reliability. The system is placed in operation by opening the condensate return valve(s) and closing standby gas purge valves. These valves are of a spring-to-safe position design, so that only de-energization of solenoid coils is required for the valves to re-position to their in-service position. The valves are redundant in both their mechanical design and the control systems which signal them into service.

Each ICS train is independent and contains a heat exchanger, i.e., an IC, that is submerged in a pool of water. Each IC is connected to the RPV by steam supply and condensate return piping. The ICS has connections to the SDC, which provides decay heat removal, as described in Subsection 5.4.7. The IC pools form the UHS for the ICS as described in Subsection 9.2.5. As described in Subsection 9.2.8, heat removal during normal operation from the IC pools is provided by the Isolation Condenser Pools Cooling and Cleanup System (ICC).

Table 6.3-1 provides key ICS design parameters.

Figure 6.3-1 illustrates the configuration of the ICS. The ICS forms part of the RCPB. An ICS system simplified diagram is provided as Figure 6.3-1.

6.3.2.1 Major Isolation Condenser System Components

The major components of the ICS are the ICs, steam supply lines, and condensate return lines.

6.3.2.1.1 Isolation Condensers

An IC is a tube-and-drum type heat exchanger submerged in an IC pool. The ICs are located outside of containment, and inside of the RB, at an elevation above the steam source so that steam and condensate flow are driven by passive forces. The ICS operates such that steam at high temperature, and pressure is delivered from the RPV to the upper header of an IC and continues down through the array of tubes that connect the upper drums (two per train) to the lower drums (two per train). The steam is condensed as it travels through the tubes, and condensate collects in the lower drums of an IC. Heat is transferred by conduction through the tube walls and is absorbed by the bulk fluid in the IC pool. Condensate drains from the lower drums, through the lower header, and back to the RPV via the condensate return piping.

When the ICS is in standby, condensate fills most of the volume in the upper region of the IC that is in direct communication with the steam supply piping from the RPV. As steam condenses in this region of the IC, radiolytic (non-condensable) gases can accumulate with the potential to displace the saturated steam. To prevent the radiolytic gases from accumulating, a continuous purge of the non-condensable gases is provided by a purge line that taps into the center region of the steam distributor and is routed to the MSL. These standby gas purge lines are isolated from the steam distributor by redundant valves designed to fail-closed on a loss of signal or power.

When an ICS train is placed in operation, radiolytic gases are removed from the ICS train by passive autocatalytic recombination. Figure 6.3-2 illustrates a preliminary configuration with a recombiner positioned on an IC.

Temperature indication is installed on the IC upper headers, lower headers, and steam distributors to provide indication that the IC is filled with water and in a ready state when the plant is operating at power.

Recombiner Validation Testing

A description of recombiner validation testing will be provided in CPA Part 2.

6.3.2.1.2 Steam Supply Lines

Each ICS train has one steam supply line routed from an RPV nozzle located in the upper region of the RPV to the steam distributor of the associated IC. Each supply line is routed from the RPV to the IC with a combination of vertical and upward sloping runs to ensure that noncondensible gases vent either to the IC and out through the standby gas purge lines during standby conditions or are processed by the catalytic recombiners when the ICS is in operation. Sloping the lines ensures that pockets of condensate are not present in the steam supply line.

Each steam supply line RIV is open when the ICS is in standby to provide a flow path to the IC. Each steam supply line is configured with instrumentation that can be used to detect flow, which is indicative of a significant pipe break. Temperature instrumentation on each line allows for periodic system performance testing and post-accident monitoring.

6.3.2.1.3 Condensate Return Lines

Each ICS train has one condensate return line routed from the lower IC drums to the RPV in the chimney region above the top of active fuel. Each condensate return line has two 100% flow capacity, mechanically diverse (valve type and actuator design) condensate return valves installed in parallel that are closed during normal plant operation, i.e., the ICS standby alignment. Thus, when the ICS is in standby, the condensate return line contains subcooled water. The full-open/full-close condensate return valves are pneumatically actuated valves and require pressurized nitrogen to maintain them in the closed position when the ICS is in standby. Each valve is designed to fail to the full-open position on a loss of control signal, control power, or motive power (pneumatic) and are held open by stored spring energy. Once the valves are fully open, they stay in that position indefinitely unless they are reclosed by the operator.

The piping that includes the condensate return valves is routed with a combination of vertical and downward sloping horizontal runs from the IC to the condensate return valves, after which a combination of vertical and upward sloping horizontal piping provides the connection with the RIVs. This configuration forms a loop seal to prevent steam bypass directly from the RPV chimney region to the IC by way of the condensate return line. This geometry also ensures that the valves have subcooled water on both sides of their sealing surfaces to prevent possible distortion and leakage while the ICS is in standby.

Trains A and B have piping connections for supplying the SDC from the RPV chimney region to support decay heat removal during plant shutdown. The interface connection is located in the loop seal region between the condensate return valves and the RIVs. The SDC lines are isolated from the ICS by redundant valves outside of containment. These valves are pneumatically actuated and require pressurized nitrogen to maintain them in the open position when the SDC is in operation; the valves are fail-closed and maintained closed by spring force.

Train C has a piping connection for injecting boron by the BIS which is described in Subsection 9.3.10. The interface connection is located in the loop seal region between the condensate return valves and the RIVs.

Each condensate return line is configured with instrumentation that can be used to detect flow, which is indicative of a significant pipe break. Temperature instrumentation on each line allows for periodic system performance testing and post-accident monitoring.

6.3.2.2 System Operation

Isolation Condenser System alignments are:

Standby: The ICS is in standby with its three trains aligned so that upon receiving an initiation signal they are placed in operation by opening the condensate return valves. In this alignment the standby gas purge valves are open and non-condensable gases are continuously purged to the MSLs.

In-Service: One or more ICS train is in operation to remove decay heat in response to an RPV pressurization event and thereby limit the pressure increase. The standby gas purge valves are closed to prevent loss of RPV inventory. Radiolytic hydrogen and oxygen are catalytically recombined within an ICS train.

Shutdown Cooling Operation, Train A and B: The ICS is in the Standby alignment with the exception that Trains A and B are aligned with SDC to provide the SDC with suction paths from the RPV chimney region. ICS Train A is aligned with SDC Train A and ICS Train B is aligned with SDC Train B.

Out-of-Service: If the pressure control and decay heat removal functions of ICS are not required, the ICS can be taken out-of-service (e.g., for maintenance, testing, or inspection). Trains A and B may be partially in operation to support the SDC suction path. When the plant is in refueling, Trains A and B are fully out-of-service when not supporting SDC suction path.

Boron Injection Operation, Train C: The ICS provides a direct flow path to the RPV chimney region for the injection of a boron solution from the BIS, in the event that the normal reactivity control system is not able to fully shutdown the reactor.

6.3.2.2.1 Normal Operation

When the plant is in power operations, the ICS is in standby and operable in accordance with the Technical Specifications. During a normal shutdown, a single train may be placed in operation for performance testing early in the normal shutdown sequence with the main condenser in operation providing the bulk of the decay heat removal.

When the plant is in hot shutdown operations, the ICS is in standby. During cold shutdown operations, the ICS provides a suction path for the SDC from the RPV. During cold shutdown operations, one train of the ICS must be operable as a backup to SDC in the event the SDC function is lost and the reactor repressurizes.

When the plant is in refueling operations, the ICS is out-of-service and can be maintained or inspected with the exception of a train (A or B) that may partially be in-service if supporting SDC operation.

6.3.2.2.2 Off-Normal Operation

The ICS provides emergency core cooling and overpressure protection. The ICS is placed in operation by opening the condensate return valves to the RPV and closing the standby gas purge valves. The subcooled condensate that is stored in the system when the ICS is in standby enters the RPV chimney region and provides additional subcooled inventory while quenching steam and lowering pressure. Simultaneously, steam from the RPV flows upward through the steam supply lines and enters the IC where it is condensed in the tubes and flows into the condensate return lines to complete the cycle. If the RPV conditions fall below the saturation point, the ICS becomes idle until decay heat drives conditions back to saturation, automatically and passively placing the ICS back into operation.

The number of ICS trains placed into service for the off-normal event depends upon the severity of the event. The plant response to design bases events described in Chapter 15 includes the number of ICS trains required to mitigate each design basis event. Each ICS train has two condensate valves in parallel that open when the RPV pressure exceeds the pressure setpoint for

the respective ICS train. The ICS initiation pressure setpoints are staggered across the three trains. The train that is assigned to initiate first is placed into service by the lowest high pressure setpoint which triggers opening of its two parallel condensate return valves. If RPV pressure continues to rise, a second then third train are placed into service in the staggered sequence. The standby gas purge isolation valves receive a signal to close simultaneously with the initiation of the first ICS train (i.e., there is no staggered closing sequence).

Because the ICS is part of the RCPB, a break in an ICS line necessitates that the RIVs for that train's steam supply and condensate return lines be closed to quickly terminate a loss of coolant. In such an event, the other two ICS trains remain intact to cool and depressurize the RPV. One ICS train is capable of removing decay heat for 72 hours to mitigate a LOCA. Instrumentation is provided to detect a break in an ICS line when a train is in standby or operation.

To ensure that a common cause failure of the control system cannot spuriously isolate the three ICS trains, the isolation functions for Trains A and C are allocated to one Safety Class 1 (SC1) Control System while the isolation functions for Train B are allocated to a second independent and diverse SC1 Control System. Each train is provided with its own leak detection instrumentation and isolation valves such that no single control failure results in isolation of all three trains.

6.3.3 Safety Evaluation (Isolation Condenser System)

The design concept for emergency core cooling is based on preserving vessel inventory by removing decay heat from the RPV using the ICS and returning reactor coolant to the RPV. Removing decay heat also provides overpressure protection of the RCPB. Because of the volume of water in the RPV, there is no need to provide additional water from an external source to maintain core cooling in response to a postulated LOCA. As discussed in Chapter 15, the RIVs are designed to close in a manner that preserves reactor coolant for breaks in piping that could cause an inventory loss large enough to prevent the ICS from cooling the core for 72 hours without operator action.

The ICS design is based on passive forces to provide overpressure protection and emergency core cooling. The only components required to place an ICS train in operation are the parallel condensate return valves, each with the capacity for full flow of an ICS train. The condensate return valves are fail-open and held open by positive means; the valves are redundant in each ICS train.

Chapter 15 provides the plant response to DBAs. Approval of the containment evaluation model in NEDC-33922P-A, "BWRX-300 Containment Evaluation Method," (Reference 6.3-2) included two Limitations and Conditions associated with ICS performance. As stated in Subsection 15.5.4.5, the LOCA Design Basis Analysis demonstrates that Limitation and Condition #1 and Limitation and Condition #2 of NEDC-33922P-A are met.

Subsection 6.3.3.1 and 6.3.3.2 summarize applicable regulatory requirements and guidance for the ICS.

6.3.3.1 Regulatory Requirements

Regulatory requirements applicable to the ICS design are 10 CFR 50, Appendix A, 10 CFR 50.34(f), 10 CFR 50.46, 10 CFR 50.55a, 10 CFR 50.63, and 10 CFR 50, Appendix K, "ECCS Evaluation Models." Compliance with regulations related to the ECCS and overpressure protection functions is summarized in the following discussion. Compliance with requirements related to instrumentation is addressed in Chapter 7.

10 CFR 50, Appendix A

In accordance with the guidance of SRP 6.3, the ICS complies with the following GDC, which have plant-wide applicability, as discussed in Section 3.1:

GDC 2 – Design bases for protection against natural phenomena

GDC 4 – Environmental and dynamic effects design bases

GDC 5 – Sharing of structures, systems, and components

The following GDC and PDC have aspects specific to the ICS and compliance is addressed in the following paragraphs. The applicable discussion is consistent with information provided in NEDC-33910P-A, “BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection,” (Reference 6.3-1).

GDC 14 – Reactor coolant pressure boundary. The ICS forms part of the RCPB and, as such, is designed to have an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture as discussed in Section 5.2.

GDC 15 – Reactor coolant system design. The ICS is designed with independent trains and control systems and with sufficient margin such that the RCPB design condition is not exceeded for normal operating conditions, including anticipated operational occurrences as demonstrated in Section 15.5.

PDC 17 – Electric power systems. The ICS requires neither offsite or standby diesel generator AC power to provide emergency core cooling and overpressure protection of the RCPB.

GDC 27 – Combined reactivity control systems capability. Compliance with GDC 27 is addressed in Section 3.1.

GDC 30 – Quality of reactor coolant pressure boundary. The ICS complies with GDC 30 with regard to inclusion of leak detection on steam supply and condensate return lines to detect and identify potential leakage location.

GDC 32 – Inspection of reactor coolant pressure boundary. The ICS complies with GDC 32 by including provisions for periodic inspection and testing to assess structural and leak-tight integrity. Inspection of the RCPB is discussed in Section 5.2.

GDC 33 – Reactor coolant makeup. Compliance with GDC 33 regarding small breaks is addressed in Section 4.6.

GDC 34 – Residual heat removal. The ICS supports decay heat removal such that specified acceptable fuel design limits are not exceeded during postulated accidents as demonstrated in Section 15.5. See Subsection 5.4.7 with regard to the SDC and Chapter 10 for normal methods of residual heat removal.

GDC 35 – Emergency core cooling. The ICS is designed to transfer heat from the reactor core in a postulated loss of coolant to prevent fuel damage. The system is redundant, has leak detection capability, and can be isolated. It can achieve its safety function assuming a single failure without reliance on offsite or standby diesel generator AC power.

GDC 36 – Inspection of emergency core cooling system. The ICS is designed to permit periodic inspection of important components to assure system integrity and capability as discussed in Subsection 6.3.4.

GDC 37 – Testing of emergency core cooling system. The ICS is designed to permit testing to assure structural and leak-tight integrity, performance of active components, and system operability as discussed in Subsection 6.3.4.

GDC 44 – Cooling water. The ICS transfers decay heat from the RPV to the UHS during post-accident conditions. Compliance with GDC 44 is addressed in Subsection 9.2.5.

GDC 45 – Inspection of cooling water systems. Compliance with GDC 45 regarding inspection of UHS components is addressed in Subsection 9.2.5.

GDC 46 – Testing of cooling water systems. Compliance with GDC 46 regarding testing of UHS components is addressed in Subsection 9.2.5.

Section 3.1 summarizes compliance with GDC and PDC.

10 CFR 50.34(f)

The ICS design complies with 10 CFR 50.34(f), as clarified in NUREG-0737. Many of the requirements identified in 10 CFR 50.34 are not technically relevant to the BWRX-300 design because of fundamental design differences, e.g., they specify a system that is not used in the BWRX-300. Compliance with these requirements is summarized in the following paragraphs, consistent with NEDC-33910P-A and design characteristics:

10 CFR 50.34(f)(1)(v): The requirement is not technically relevant. The intent of the requirement is met by the BWRX-300 design which is not subject to RPV stresses from cold water injection when actuating ECCS.

10 CFR 50.34(f)(1)(vi): The requirement is not technically relevant. The intent of the requirement is met by the ICS design which does not to use relief valves whose opening results in a loss of reactor coolant.

10 CFR 50.34(f)(1)(vii): The requirement is not technically relevant. The intent of the requirement is met by the ICS design which does not require manual actuation to assure adequate core cooling.

10 CFR 50.34(f)(1)(viii): The requirement is not technically relevant. The intent of the requirement is met by the ICS design which actuates by a one-time opening of valves with flow determined by natural circulation. The ICS is capable of maintaining core cooling without operator action for at least 72 hours.

10 CFR 50.34(f)(1)(ix): The requirement is not technically relevant. The intent of the requirement is met by the ICS design which does not require active pumps or space cooling to perform its function.

10 CFR 50.34(f)(1)(x): The requirement is not technically relevant. The intent of the requirement is met by the ICS design which does not require equipment other than SC1 components to perform its function. The only valves required to maintain core cooling are the condensate return valves which are held open by positive means (spring) upon actuation.

10 CFR 50.34(f)(1)(xi): The requirement is not technically relevant. The intent of this requirement is met by the ICS design which is based on removing decay heat using a passive method that does not challenge vessel integrity by rapid cooldown.

10 CFR 50.34(f)(1)(xii): The design complies with the requirement by the use of an inerted containment (See Subsection 6.2.5).

10 CFR 50.34(f)(2)(vi): The requirement is not technically relevant. The intent of this requirement is met by the configuration of the ICS piping and purge capability which control potential accumulation of non-condensable gases that could affect ECCS function. (See Subsection 5.4.13 for discussion of RPV venting).

10 CFR 50.34(f)(2)(x): The requirement is not technically relevant. The condensate return valves perform a similar function to relief valves in providing a path for overpressure protection and are

qualified by testing to perform this function for fluid conditions expected under operating conditions, transients, and accidents.

10 CFR 50.34(f)(2)(xi): The requirement is not technically relevant. The intent of the requirement is met by the ICS design. Pressure relief is provided by the ICS, which when actuated, does not result in inventory loss as is the situation with relief valves. Although valve position indication in the MCR is not needed to inform operators of a potential inventory loss, it is provided for RIVs, and ICS condensate return valves.

10 CFR 50.34(f)(2)(xviii): The requirement related to instrumentation for the detection of inadequate core cooling is addressed in Subsection 7.1.3.

10 CFR 50.34(f)(2)(xxi): The requirement is met by the design of the ICS which is initiated automatically and can be initiated manually as discussed in Section 7.3.

10 CFR 50.34(f)(2)(xxvi): The requirement related to leakage control outside of containment is addressed in Subsection 5.4.7 and Subsection 6.2.1.3.

10 CFR 50.46

The ICS, in conjunction with the RIVs, functions to assure core cooling in response to postulated LOCAs. The ICS circulates reactor coolant supplied as steam, condenses it through the ICs and returns cooled water to the RPV. The RIVs isolate pipe breaks that are large enough to exceed the capability of the ICS to cool the core. Section 5.4 establishes that breaks at the bolted flange connections between the RPV and RIVs are not postulated for the BWRX-300 design.

Chapter 15 discusses the analytical techniques used to model ECCS performance and provides the results of the modeling, as required by 10 CFR 50.46. The methods and assumptions of the deterministic safety analysis are consistent with those described in NEDC-33922P-A.

10 CFR 50.55a

Pursuant to 10 CFR 50.55a(f)(4)(i) and 10 CFR 50.55a(g)(4)(i), the initial IST and ISI programs, respectively, use the edition and addenda of the ASME OM Code and of ASME BPVC,

Section XI, subject to the conditions identified and incorporated by reference into

10 CFR 50.55a, on the date no more than 18 months before the date of issuance of the Operating License. Accordingly, the Code Editions for the ASME OM Code and ASME BPVC, Section XI, Division 1 will be addressed in the FSAR.

10 CFR 50.63

The ICS provides the capability to provide core cooling in response to a SBO as required by 10 CFR 50.63. RPV depressurization and decay heat removal is accomplished by the opening of the condensate return valve(s) and the closing of the standby gas purge valves. These valves are of a spring to safe position design, so that only the de-energization of solenoid coils is required for the valves to re-position to their ICS in-service position. SBO is addressed in Section 8.4.

10 CFR 50, Appendix K

As stated in NEDC-33910P-A, the ECCS evaluation model does not use the alternatives provided in 10 CFR 50, Appendix K.

6.3.3.2 Regulatory Guidance

The following RGs have specific applicability to the design of ICS. Conformance to RGs related to instrumentation is addressed in Chapter 7. Regulatory Guides related to ICS materials are addressed in Chapter 5.

RG 1.26 – Quality Group Classifications and Standards for Water, Steam, and Radioactive-Waste-Containing Components of Nuclear Power Plants: Quality Group classification of BWRX-300 components is discussed in Subsection 3.2.4.

RG 1.29 – Seismic Design Classification for Nuclear Power Plants: Seismic categorization of BWRX-300 components is discussed in Subsection 3.2.3.

RG 1.45 – Guidance on Monitoring and Responding to Reactor Coolant System Leakage: Conformance to RG 1.45 is discussed in Subsection 5.2.5.

RG 1.47 – Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems: Conformance to RG 1.47 is discussed in Subsection 7.1.3.3.

RG 1.53 – Application of the Single-Failure Criterion to Safety Systems: The ICS is designed such that no single active failure prevents a Safety Category 1 function from being performed.

RG 1.100 – Seismic Qualification of Electrical and Active mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants: The seismic and dynamic qualification of BWRX-300 components is discussed in Section 3.10.

RG 1.155 – Station Blackout: The ICS provides the capability to respond to a SBO as no offsite or standby diesel generator AC power is required to maintain adequate reactor coolant inventory and provide core cooling.

RG 1.157 – Best-Estimate Calculations of Emergency Core Cooling System Performance: As discussed in NEDC-33922P-A, only the RG 1.157 guidance relating to the general best practice elements of a rigorous method development are applicable. These best practice elements are also included in RG 1.203 which is discussed in Section 15.5.

6.3.4 Tests and Inspections (Isolation Condenser System)

The ICS is designed as ASME BPVC, Section III Class 1 and 2. As such there are mandatory surveillance tests including IST of components and ISI of pressure boundary components. The design and construction of the ICS integrates features that permit these tests and inspections to be performed. The preoperational and startup test program is described in Chapter 14. The ICS design supports:

- Periodic stroke testing of normally closed condensate return valves during plant operation
- Periodic stroke testing of normally open standby gas purge valves during plant operation
- Surveillance testing
- ISI and IST
- Heat transfer performance testing

6.3.5 Instrumentation Requirements (Isolation Condenser System)

Instrumentation associated with the ICS is:

- Flow transmitters on steam supply and condensate return lines
- Temperature instrumentation on steam supply line and condensate return lines
- Temperature instrumentation on the IC upper headers, lower headers, and steam distributors
- Pressure instrumentation for SDC interface CIVs and isolation valve nitrogen supply accumulators

- Pressure instrumentation for condensate return valve nitrogen supply accumulators

Refer to Section 7.3 for additional information pertaining to I&C systems.

6.3.6 References

6.3-1 NEDC-33910P-A, “BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection,” Revision 2, GE Hitachi Nuclear Energy Americas LLC, June 2021.

6.3-2 NEDC-33922P-A, “BWRX-300 Containment Evaluation Method,” Revision 3, GE Hitachi Nuclear Energy Americas LLC, June 2022.

6.3-3 To be provided in CPA Part 2

Table 6.3-1: Isolation Condenser System Parameters

Parameter	Value
Number of ICS trains	3
Number of isolation condensers (IC) per train	1
Number of condenser modules per IC	2
Heat removal capacity per IC	3.8% reactor thermal power (33.75 MWt)
IC design pressure	1500 psig (10.34 MPaG)
IC design temperature	598°F (314.4°C)

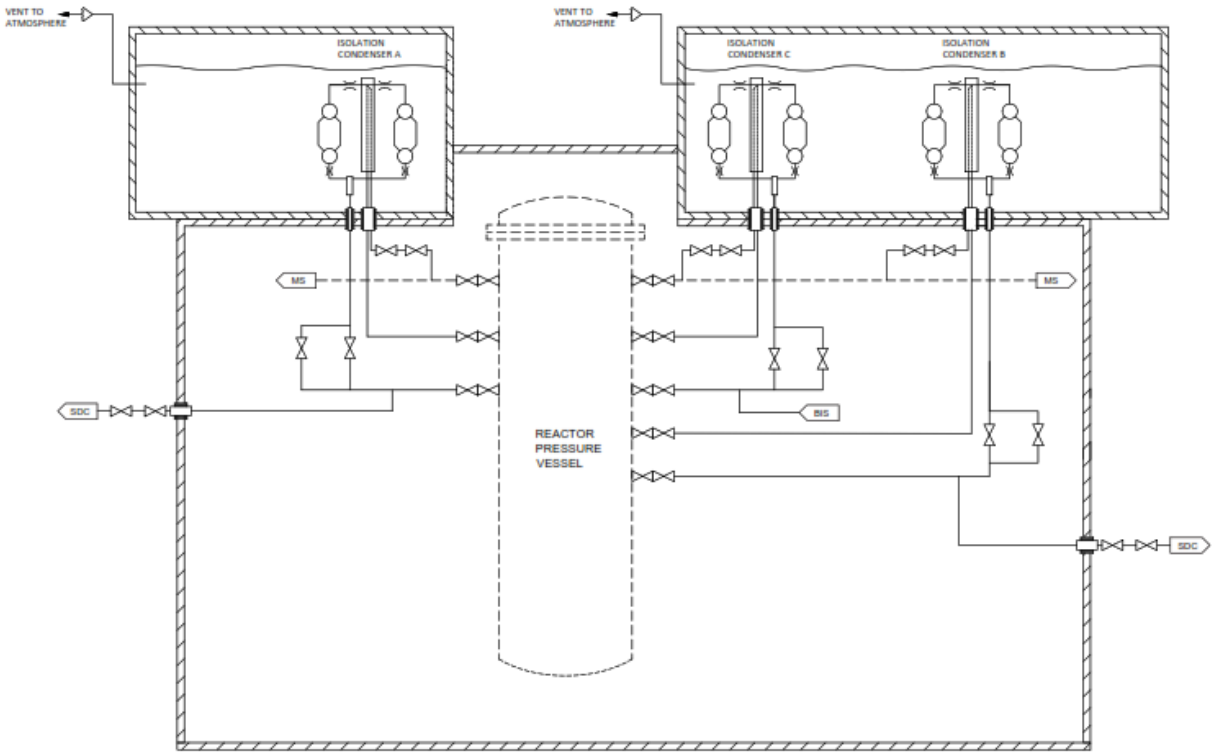
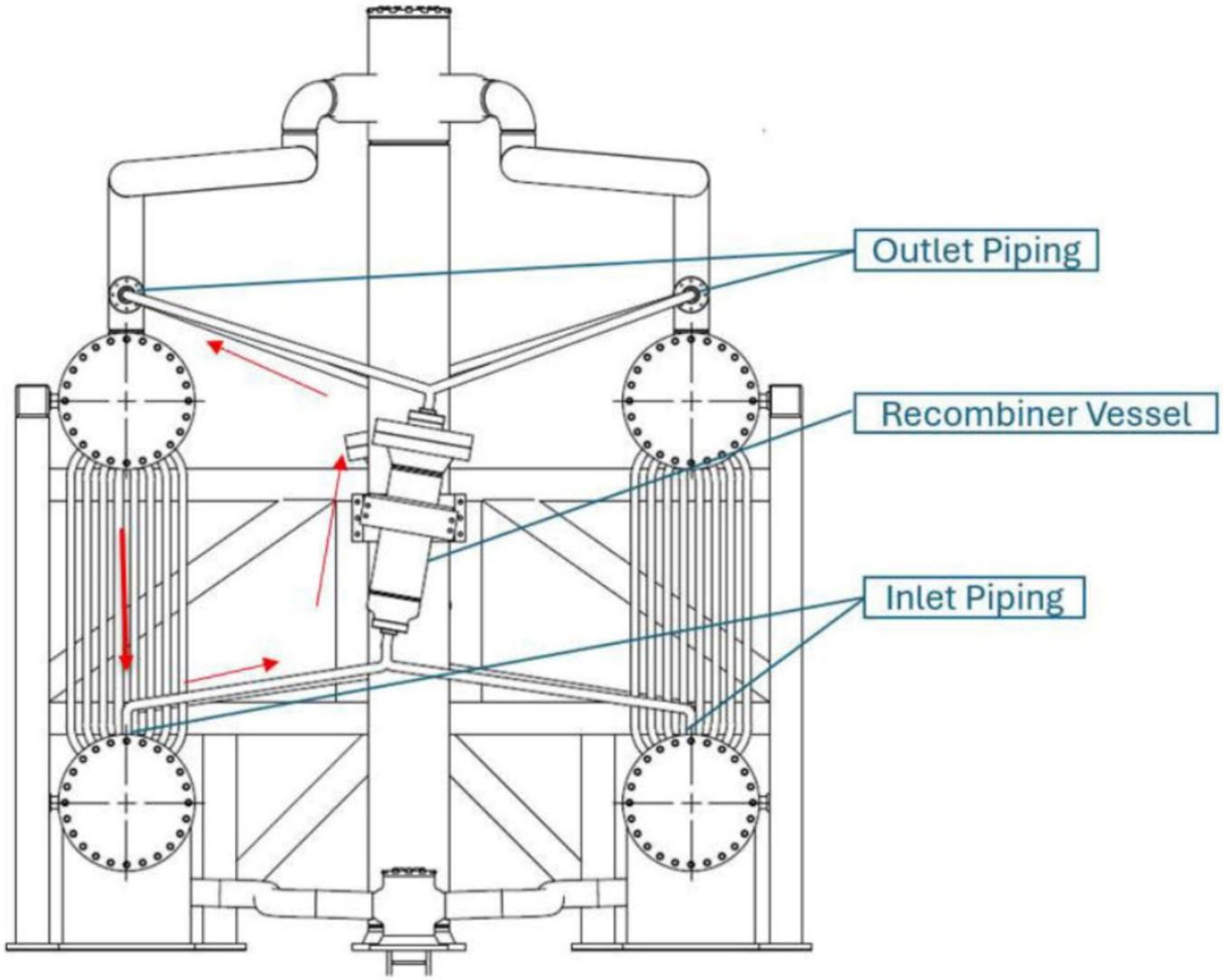


Figure 6.3-1: Isolation Condenser System Simplified Diagram



Note: Red arrows show flow direction.

Figure 6.3-2: Preliminary Hydrogen Recombiner Configuration

6.4 Control Room Habitability

The plant design includes two distinct and independent control rooms: an MCR and an SCR. The MCR and SCR are physically and electrically separated facilities such that at least one facility remains accessible and functional during and following a postulated initiating event. The MCR is located in the Control Building (CB) and has the capability to operate the plant during normal conditions as well as maintain the plant in a safe state and monitor critical parameters during an off-normal event. The SCR is located in the RB and provides the capability to initiate shutdown, maintain the plant in a safe shutdown condition, and monitor critical parameters during an off-normal event.

Control room habitability systems are those systems that support operator actions to operate the plant under normal and accident conditions. MCR and SCR habitability systems are not categorized as ESF systems because they are not associated with Safety Category 1 functions. The control room habitability systems ensure operator protection and support other Safety Category functions, as discussed in the following sections:

- Heating, ventilation, and cooling systems: Section 9.4 discusses CB and RB ventilation including protection against toxic gases, radioactive gases, and smoke. Section 9.4 defines the Control Room Envelope (CRE) for both the MCR and the SCR.
- Control room lighting: Subsection 9.5.3 provides lighting requirements for the MCR and SCR under normal and accident conditions.
- Control room instrumentation: Chapter 7 describes MCR and SCR instrumentation.
- Fire protection: Subsection 9.5.1 discusses fire protection capabilities for the MCR and SCR.
- Radiation shielding: Chapter 12 discusses radiation shielding and potential exposures in the MCR and SCR. An evaluation of potential DBA doses to control room personnel is described in Section 15.5.
- Protection against natural phenomena: Chapter 3 discusses the capability of the CB and RB with regard to protection against natural phenomena.

6.4.1 Design Bases

The safety categorization of systems supporting CRH is addressed in the relevant system sections. Functions associated with the control rooms are:

- The MCR provides:
  - Operators with the capability to monitor the status of FSFs during postulated initiating events and associated event sequences
  - Instrumentation and equipment needed to perform operational sequences in which continuous monitoring or timing of control actions is defined by task analysis as being critical
  - Instrumentation and equipment needed to detect abnormal conditions
  - Instrumentation and equipment needed to bring the plant to a safe shutdown condition
  - Capability to manually initiate safety functions that are initiated by automatic control logic

  - Indications of plant conditions and processes that have deviated from normal operation and that could affect safety
  - Display of information needed to monitor the effects of the automatic actions of control, safety, and safety support systems
- The SCR provides the capability to:
  - Initiate shutdown of the reactor, if necessary, and maintain the plant in a safe shutdown condition
  - Monitor FSFs

The CB is a Seismic Category II structure. The RB is a Seismic Category I structure. Codes and standards related to wind loadings, flooding, missile protection, and pipe rupture are addressed in Sections 3.3 through 3.6, respectively. Codes and standards related to seismic design are addressed in Section 3.7 and Section 3.8.

6.4.2 System Operation

Operators relocate from the MCR and transfer controls to the SCR if:

- The ability to operate the plant and maintain the plant from the MCR is lost (i.e., MCR functionality is unacceptably impaired)
- The MCR becomes uninhabitable
- Procedurally driven criteria associated with a specific event (e.g., anticipated loss of ability to operate from the MCR)

An egress route exists from the MCR to the SCR for events that necessitate evacuation of the MCR. The SCR becomes the assured shutdown location if controls are transferred from the MCR. For example, the SCR is the assured shutdown location for seismic events having a magnitude greater than the Safe Shutdown Earthquake as defined in Section 3.7. Either control room can be the assured location depending on the location of a fire. The Control Transfer Function is discussed in Subsection 7.6.1.

6.4.3 Safety Evaluation

The plant design includes two distinct and independent control facilities: an MCR and an SCR which assures that operator response can be accomplished for postulated initiating events. CRH systems do not support Safety Category 1 functions.

6.4.3.1 Regulatory Requirements

Regulatory requirements applicable to the CRH design are 10 CFR 50, Appendix A, and 10 CFR 50.34(f). Compliance with regulations related to the CRH functions is summarized in the following discussion. Compliance with requirements related to instrumentation is addressed in Chapter 7.

10 CFR 50, Appendix A

In accordance with the guidance of SRP 6.4, the CRH systems comply with the following GDC, which have plant-wide applicability, as discussed in Section 3.1:

GDC 4 – Environmental and dynamic effects design bases

GDC 5 – Sharing of structures, systems, and components

The following PDC has aspects specific to the CRH systems and compliance is addressed as follows:

PDC 19 – Control room. The BWRX-300 design complies with the requirement of PDC 19 as it relates to maintaining the plant in a safe condition under DBA conditions without requiring operator action from MCR or SCR, as described in Subsection 15.4.1. Dose acceptance criteria are met for DBAs, without credit for the heating, ventilation, and cooling system supporting the MCR and SCR. As such, CRH systems provide adequate radiation protection under DBA conditions and comply with PDC 19.

Section 3.1 summarizes compliance with PDC.

10 CFR 50.34(f)

10 CFR 50.34(f)(2)(xxviii): Compliance with 10 CFR 50.34(f)(2)(xxviii) is demonstrated with design commitments that address criteria that would preclude CRH. Specifically, for any operator actions claimed, including required monitoring of plant parameters, the design ensures that a habitable location is available and accessible during the associated event conditions for DBAs.

6.4.3.2 Regulatory Guidance

The following RGs have specific applicability to the design of CRH systems. Conformance to RGs related to instrumentation is addressed in Chapter 7.

RG 1.183 – Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors: Potential control room dose is evaluated in Section 15.5 and Appendix 15C.

RG 1.196 – Control Room Habitability at Light-Water Nuclear Power Reactors: The design of the MCR and SCR conforms to RG 1.196 by assuring that at least one control room is available to support FSFs for a DBA and not requiring operator action from the MCR or SCR during DBAs. Dose acceptance criteria are met during DBAs without credit for the heating, ventilation, and cooling system. CRH system maintenance and testing requirements, as applicable, are described in system descriptions as referenced earlier in this chapter.

RG 1.197 – Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors: Applicable regulatory guidance and industry standards for demonstrating CRE integrity are discussed in Section 9.4. The test program for CRE integrity will be established in the FSAR.

6.5 Fission Product Removal and Control Systems

6.5.1 Engineered Safety Feature Atmosphere Cleanup Systems

The BWRX-300 design does not require an ESF atmospheric cleanup system. The BWRX-300 low-leakage containment is constructed in the subterranean which is expected to further limit potential fission product leakage.

The CIS, which is not an ESF system, is discussed in Subsection 9.3.6. Compliance with GDC 41, GDC 42, and GDC 43 is addressed in Section 9.3.

6.5.2 Containment Spray Systems

This section is not applicable to the BWRX-300 design.

6.5.3 Fission Product Control Systems

The BWRX-300 design does not use a dual or secondary containment. Containment isolation and leakage rate testing is addressed in Section 6.2.

6.5.4 Ice Condenser as a Fission Product Cleanup System

This section is not applicable to the BWRX-300 design.

6.5.5 Pressure Suppression Pool as a Fission Product Cleanup System

This section is not applicable to the BWRX-300 design.

6.6 Preservice and Inservice Inspection of Class 2 and Class 3 Components

This section describes the Preservice Inspection, ISI, and system pressure test programs for Quality Groups B and C components, that is, ASME BPVC, Section III, Class 2 and 3 components, respectively. This description includes those programs implementing the requirements of ASME BPVC, Section XI, Division 1, "Rules for Inspection and Testing of Components of Light-Water-Cooled Plants," Subsections IWC and IWD.

Pursuant to 10 CFR 50.55a(g)(4)(i), the initial ISI program uses the edition and addenda of ASME BPVC, Section XI, subject to the conditions identified and incorporated by reference into 10 CFR 50.55a, on the date no more than 18 months before the date of issuance of the Operating License. Accordingly, the Code Edition for the ASME BPVC, Section XI, Division 1 will be addressed in the FSAR.

Preservice Inspection and ISI programs for Class 2 and 3 components are based on the ASME BPVC, Section XI, edition and addenda specified in accordance with 10 CFR 50.55a subject to limitations and modifications found therein. Additionally, 10 CFR 50.55a provides an allowance to request alternatives to, or relief from, ASME BPVC, Section XI requirements. The editions of RG 1.84, RG 1.147, and RG 1.192 used to meet 10 CFR 50.55a are those incorporated by reference in 10 CFR 50.55a(a)(3).

6.6.1 Components Subject to Examination

Quality Group B and C components that are subject to examination are classified in accordance with RG 1.26 and are listed in Table 3A-1.

6.6.2 Accessibility

Components within the ASME BPVC, Section III, Class 2 and 3 boundaries are designed to provide access for the examinations required by ASME BPVC, Section XI, IWA-1500.

6.6.3 Examination Categories and Methods

6.6.3.1 Examination Categories

The examination category of each component is consistent with ASME BPVC, Section XI, IWC-2000 and IWD-2000, and is listed in the Preservice Inspection and ISI Programs. The Preservice Inspection and ISI Programs state the method of examination for each component.

6.6.3.2 Examination Methods

Non-destructive examination methods are in accordance with ASME BPVC, Section V, "Nondestructive Examination," and ASME BPVC, Section XI.

Visual Examination

Visual examinations are used to indicate the presence of surface discontinuities and imperfections, leakage from pressure-retaining components, and the general mechanical and structural condition of components. General visual examination and visual examination methods VT-1, VT-2, and VT-3 are used in accordance with ASME BPVC, Section XI.

Surface Examination

Surface examinations are used to indicate the presence of surface discontinuities. Such examinations are performed by using Eddy Current Testing, Liquid Penetrant Testing, Magnetic Particle Testing, and Ultrasonic Testing methods. Surface examination is performed in accordance with ASME BPVC, Section XI.

Volumetric Examination

Volumetric examinations are used to indicate the presence of discontinuities throughout the volume of the material. Such examinations are performed by using Acoustic Emission Testing, Eddy Current Testing, Radiographic Testing, or Ultrasonic Testing methods. Volumetric examination is performed in accordance with ASME BPVC, Section XI.

Alternative Examination Techniques

As provided by ASME BPVC, Section XI, IWA-2240, "Alternate Examinations," a combination of methods, or newly developed techniques, may be substituted for the methods specified for a given component provided they are demonstrated to be equivalent or superior to the specified method. This provision allows for the use of newly developed examination methods, techniques which may result in improvements in examination reliability and reductions in personnel exposure. Newly developed techniques are approved by the U.S. Nuclear Regulatory Commission (USNRC) and if necessary, conditions provided by the USNRC staff are incorporated for use.

Qualification of Personnel and Examination Systems

Non-destructive examination personnel who perform Acoustic Emission Testing, Eddy Current Testing, Magnetic Particle Testing, Liquid Penetrant Testing, Radiographic Testing, Ultrasonic Testing, Leak Testing, and Visual Testing examinations, including applicable procedures, are qualified in accordance with ASME BPVC, Section V, Articles 2, 4, 5, 6, 7, 8, 9, 10, 12, and 13.

NDE personnel, procedures and NDE equipment (systems) will be qualified in accordance with ASME BPVC, Section XI, including Mandatory Appendices I, III, IV, VI, VII and VIII specified above since the PSI and ISI program is performed in accordance with ASME Code, Section XI.

6.6.4 Inspection Intervals

6.6.4.1 Class 2 Systems

The Inspection Intervals for Class 2 systems are consistent with ASME BPVC, Section XI, IWC-2400. Inspection Intervals have a nominal length of 10 years with allowance for up to a year variation to coincide with refueling outages.

6.6.4.2 Class 3 Systems

The Inspection Intervals for Class 3 systems are consistent with ASME BPVC, Section XI, IWD-2400. Inspection Intervals have a nominal length of 10 years with allowance for up to a year variation to coincide with refueling outages.

6.6.5 Examination Categories and Requirements

The ISI categories and requirements for Class 2 components comply with ASME BPVC, Section XI, IWC-2000. The ISI categories and requirements for Class 3 components comply with ASME BPVC, Section XI, IWD-2000.

6.6.6 Evaluation of Examination Results

Examination results are evaluated in accordance with ASME BPVC, Section XI, IWC-3100 for Class 2 components and repairs are performed in accordance with ASME BPVC, Section XI, IWA-4000. Examination results are evaluated in accordance with ASME BPVC, Section XI, IWD-3100 for Class 3 components and repairs are performed in accordance with ASME BPVC, Section XI, IWA-4000.

6.6.7 Code Exemptions

As provided in ASME BPVC, Section XI, IWC-1220 and IWD-1220, certain portions of ASME BPVC, Section III, Class 2 and 3 systems are exempt from the volumetric, surface, and visual examination requirements of IWC-2500 and IWD-2500.

6.6.8 System Pressure Tests

System leakage and hydrostatic testing is performed in accordance with ASME BPVC, Section XI. These tests meet the applicable requirements of ASME BPVC, Section XI, IWA-5000 and IWC-5000 for Class 2 components; and IWA-5000 and IWD-5000 for Class 3 components.

6.6.8.1 System Leakage Test

A system leakage test is performed in accordance with IWC-5000 on Class 2 systems and IWD-5000 on Class 3 systems.

6.6.8.2 Hydrostatic Pressure Tests

A system hydrostatic test may be performed in lieu of a system leakage test, and when required for repairs, replacements, and modifications in accordance with ASME BPVC, Section XI, IWA-4540. The hydrostatic test includes Class 2 or 3 pressure retaining components within the boundaries defined by ASME BPVC, Section XI, IWC-5000 and IWD-5000 or the boundary of a repair or replacement as applicable.

6.6.9 Augmented Inservice Inspection to Protect Against Postulated Piping Failures

High Energy Piping

High energy piping, as defined in Section 3.6, is subject to augmented inspection of the break exclusion zone, as discussed in Subsection 3.6.1.1.

Erosion-Corrosion

ASME BPVC, Section III, Class 1, 2, 3, and non-safety class piping and components, as described in NUREG-1344 and Generic Letter 89-08 determined to be susceptible to erosion-corrosion, are subject to a program of nondestructive examinations to verify system structural integrity. Examination methods and schedule are determined in accordance with the guidelines established within EPRI 3001000563, “Recommendations for an Effective Flow-Accelerated Corrosion Program,” (Reference 6.6-1).

6.6.10 Augmented Inspection of Boiling Water Reactor Piping Susceptible to IGSCC

Augmented inspections for IGSCC are performed in accordance with NUREG-0313, Generic Letter 88-01, and Supplement 1 to Generic Letter 88-01. Details of augmented inspections for intergranular stress corrosion cracking will be provided as part of the Inservice Inspection Program description in the FSAR.

6.6.11 Operational Programs

Preservice Inspection and ISI Implementation milestones are identified in Section 13.4.

6.6.12 Regulatory Compliance

Regulatory requirements applicable to ISI and Preservice Inspection of ASME BPVC, Section III, Class 2 and 3 components are defined in 10 CFR 50, Appendix A. The BWRX-300 design meets the requirements for periodic inspection and testing of Class 2 and 3 components in GDC 36, 37, 39, 40, 42, 43, 45, and 46, as specified in part in 10 CFR 50.55a, and as detailed in ASME BPVC, Section XI.

Subsection 6.3.3.1 provides the evaluation of compliance with GDC 36 and GDC 37 as pertains to periodic inspection and testing of the ICS. Subsection 6.2.2.3 provides the evaluation of compliance with GDC 39 and GDC 40 as pertains to periodic inspection and testing of the PCCS. Subsection 9.2.5.3 addresses GDC 45 and GDC 46 as pertains to periodic inspection and testing of the IC pools, equipment pool, and reactor cavity pool. Subsection 9.3.6.3 addresses GDC 42 and GDC 43 as pertains to periodic inspection and testing of the CIS.

Compliance with the preservice and inservice examinations of 10 CFR 50.55a, as detailed in ASME BPVC, Section XI, satisfies in part the requirements of GDC 36, 37, 39, 40, 42, 43, 45, and 46.

6.6.13 References

6.6-1 EPRI 3001000563, “Recommendations for an Effective Flow-Accelerated Corrosion Program (NSAC-202L-R4),” Electric Power Research Institute, 2013.

6.7 Main Steam Isolation Valve Leakage Control System

This section is not applicable to the BWRX-300 because the design does not include a main steam isolation valve leakage control system.

6.8 Containment Monitoring Subsystem

The Containment Monitoring Subsystem (CMon) is a subsystem of the Process Radiation and Environmental Monitoring System (PREMS). Descriptions of the other PREMS subsystems are included in other PSAR sections as indicated below. CMon process sample lines have Containment Isolation Valves (CIVs), which are described in Subsection 6.2.4.

The PREMS includes the following other subsystems:

- Process Sampling Subsystem (PS), Subsection 9.3.2
- Process Radiation Monitoring Subsystem (PRM), Section 11.5
- Area Radiation Monitoring Subsystem (ARM), Subsection 12.3.4

The CMon provides instrumentation to monitor:

- Containment pressure
- Containment temperature
- Containment water level
- Containment hydrogen concentration
- Containment oxygen concentration
- Containment fission products
- Containment area radiation levels

Real-time measurements are displayed in the MCR with alarms to notify personnel when measurements exceed preset limits. Local indication and alarms are provided where necessary to notify personnel of hazardous conditions.

6.8.1 Design Bases (CMon)

Safety Category 1 Functions

- Monitor containment pressure

Safety Category 2 Functions

- Monitor containment pressure

Safety Category 3 Functions

- Monitor process and effluent streams for potentially hazardous materials such as radioactive noble gases, particulates, and halogens
- Monitor containment environmental parameters that are indicative of RCPB leakage
- Monitor hydrogen and oxygen concentrations inside containment
- Monitor area radiation levels in containment

Safety Category N Functions

None.

The Safety Classification, Quality Groups, and Seismic Categories of the CMon SSCs are listed in Table 3A-1.

6.8.2 System Description (CMon)

A simplified diagram of the CMon is shown on Figure 6.8-1.

6.8.2.1 Containment Pressure Monitoring

Pressure transmitters are provided for monitoring of containment pressure. Each pressure transmitter is located outside containment with instrument sensing lines routed through containment penetrations.

These instruments provide:

- Indication of containment pressure for hydraulic scram
- Indication of containment pressure for ICS initiation
- Indication of containment pressure for RPV/Containment isolation
- Indication of containment pressure for CIS control
- Monitoring of bulk containment pressure

6.8.2.2 Containment Temperature Monitoring

Temperature transmitters are installed throughout containment for monitoring of containment air temperature. These instruments provide:

- Indication of reactor coolant leakage within containment
- Indication of containment temperature for CCS control
- Monitoring of bulk containment temperature

6.8.2.3 Containment Water Level Monitoring

Water level transmitters are installed at varying levels in containment. These instruments do not support any Safety Category functions.

6.8.2.4 Containment Hydrogen and Oxygen Concentration Monitoring

Sample skids are provided for monitoring of hydrogen and oxygen content in containment. The skids are located outside containment, with supply and return process piping routed through containment penetrations.

Samples are physically extracted from the containment atmosphere, analyzed, and returned to containment. Each skid includes independent oxygen and hydrogen analyzers, flow extraction/control equipment, grab sampling taps, a local control panel, and audible/visual alarms.

These instruments provide:

- Monitoring of containment hydrogen concentration
- Monitoring of containment oxygen concentration
- Indication of containment oxygen concentration for CIS control

6.8.2.5 Containment Fission Product Monitoring

A Particulate, Iodine and Nobel Gas (PING) skid is provided for monitoring of fission products in containment. The skid is located outside containment, with supply and return process piping routed through containment penetrations.

Samples are physically extracted from the containment atmosphere, analyzed, and returned to containment. The skid includes three independent particulate, iodine, and noble gas analyzers,

flow extraction/control equipment, grab sampling taps, a local control panel, and audible/visual alarms. Particulate filters, iodine filters (charcoal cartridges), and grab samples are periodically collected from the skid for isotopic analysis.

These instruments provide:

- Monitoring of containment for fission products
- Indication of reactor coolant leakage within containment

6.8.2.6 Containment Area Radiation Monitoring

Area radiation monitors are provided for monitoring gamma radiation levels in containment. These instruments provide:

- Monitoring of containment area radiation levels

6.8.3 Safety Evaluation (CMon)

The CMon complies/conforms with the following regulatory requirements and guidance.

Regulatory Requirements

The CMon is designed to comply with the following regulations:

- 10 CFR 20.1101(b), as it relates to the use of engineering controls in maintaining doses ALARA
- 10 CFR Part 20, Subpart F, as it relates to radiological monitoring

These regulations are met, as the CMon is designed to conform with RG 8.2 and RG 8.8, as it relates to information relevant to ensuring that occupational radiation exposures are ALARA.

10 CFR 50 Appendix A

The following GDC has aspects specific to the CMon, and compliance is addressed as follows: GDC 13 – Instrumentation and control. The design of the CMon complies with GDC 13. The CMon is provided with instrumentation to monitor variables important to the containment over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety.

GDC 41 – Containment atmosphere cleanup. The design of the CMon complies with GDC 41. The CMon is provided with instrumentation to monitor variables important to the containment over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety.

GDC 64 – Monitoring radioactivity releases: The design of the CMon complies with GDC 64 with regard to monitoring the containment atmosphere for radioactivity during normal operations and off-normal events.

Section 3.1 summarizes compliance with GDC.

10 CFR 50.34(f)

10 CFR 50.34(f)(2)(xvii) Accident monitoring instrumentation and control room display requirements: The design of the CMon complies with 10 CFR 50.34(f)(2)(xvii).

10 CFR 50.34(f)(2)(xix) Post-core damage accident plant condition monitoring requirements: The design of the CMon complies with 10 CFR 50.34(f)(2)(xix).

Regulatory Guidance

RG 1.7 – Control of Combustible Gas Concentrations in Containment: The design conforms to the guidance provided in RG 1.7 with regard to monitoring hydrogen and oxygen concentrations inside containment.

RG 1.21 – Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste: The design conforms to the guidance provided in RG 1.21 with regard to monitoring for potentially hazardous contamination such as radioactivity, noble gases, air particulates, and halogens.

RG 1.45 – Guidance on Monitoring and Responding to Reactor Coolant System Leakage: The design conforms to the guidance provided in RG 1.45 with regard to monitoring parameters in support of RCPB leakage detection.

RG 1.97 – Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants: The design conforms to the guidance provided in RG 1.97 with regard to monitoring parameters in support of post-accident monitoring. A preliminary list of accident monitoring variables is provided in Table 7.3-2.

6.8.4 Tests and Inspections (CMon)

The initial test program is described in Chapter 14. Preoperational testing descriptions will be provided in the FSAR.

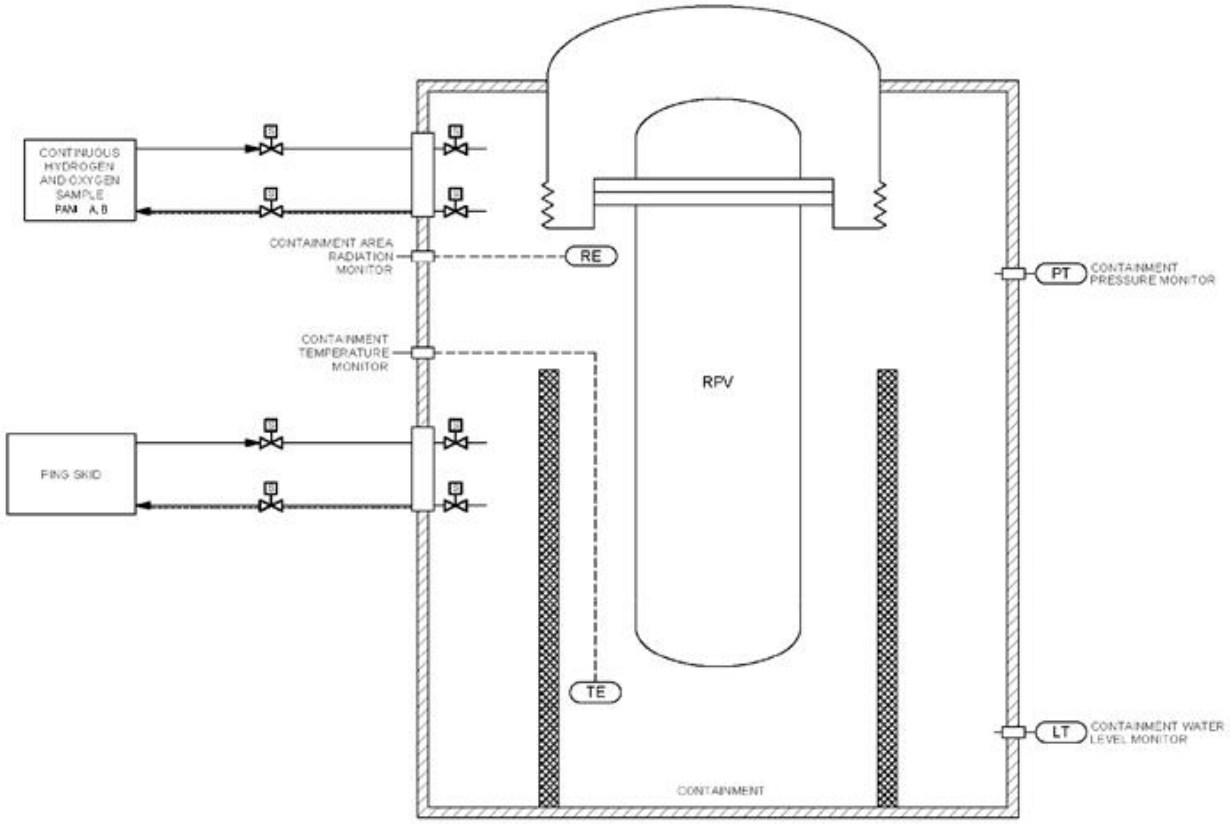


Figure 6.8-1: Containment Monitoring Subsystem Simplified Diagram

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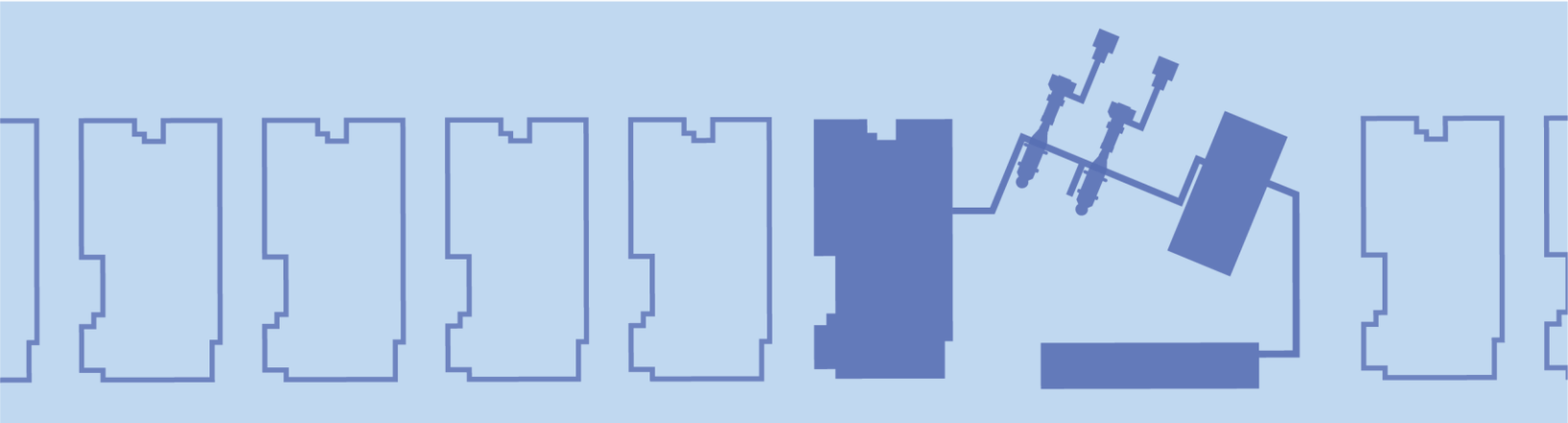
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Non-Proprietary Information

Chapter 7
Instrumentation and Controls



ACRONYM LIST

Acronym	Explanation
2oo2	Two-out-of-Two
2oo3	Two-out-of-Three
2oo4	Two-out-of-Four
3D	Three-Dimensional
AC	Alternating Current
AOF	Allocation of Functions
AOO	Anticipated Operational Occurrence
APR	Automatic Power Regulator
APRM	Average Power Range Monitor
ARI	Alternate Rod Insertion
ARM	Area Radiation Monitoring Subsystem
ATLM	Automatic Thermal Limit Monitor
BIS	Boron Injection System
BOP	Balance of Plant
CB	Control Building
CCF	Common Cause Failure
CFR	Code of Federal Regulations
CRD	Control Rod Drive
CUW	Reactor Water Cleanup System
D-in-D	Defense-in-Depth
DBA	Design Basis Accident
DC	Direct Current
DCIS	Distributed Control and Information System
DEC	Design Extension Condition
DL	Defense Line
DL1	Defense Line 1
DL2	Defense Line 2
DL3	Defense Line 3
DL4a	Defense Line 4a
DL4b	Defense Line 4b
DL5	Defense Line 5

Acronym	Explanation
DPS	Diverse Protection System
DRG	Design Review Guide
EMC	Electromagnetic Compatibility
EOP	Emergency Operating Procedure
ERICP	Emergency Rod Insertion Control Panel
ERIP	Emergency Rod Insertion Panel
FMCRD	Fine Motion Control Rod Drive
FRA	Functional Requirements Analysis
FSAR	Final Safety Analysis Report
FSF	Fundamental Safety Function
FW	Feedwater
FWCIV	Feedwater Containment Isolation Valve
FWRIV	Feedwater Reactor Isolation Valve
GDC	General Design Criteria
GVH	GE Vernova Hitachi Nuclear Energy
GT	Gamma Thermometer
HA	Human Action
HCU	Hydraulic Control Unit
HFE	Human Factors Engineering
HFEPP	Human Factors Engineering Program Plan
HPM	Human Performance Monitoring
HSI	Human-System Interface
HVS	Heating, Ventilation, and Cooling System
I&C	Instrumentation and Control
IC	Isolation Condenser
ICS	Isolation Condenser System
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronics Engineers
ISV	Integrated System Validation
LAN	Local Area Network
LPRM	Local Power Range Monitor
LWR	Light Water Reactor

Acronym	Explanation
MCR	Main Control Room
MRBM	Multi-Channel Rod Block Monitor
MSCIV	Main Steam Containment Isolation Valve
MSL	Main Steam Line
MSRIV	Main Steam Reactor Isolation Valve
NPP	Nuclear Power Plant
OE	Operating Experience
PAS	Plant Automation System
PDC	Principal Design Criteria
PDH	Plant Data Highway
PIE	Postulated Initiating Event
PRNM	Power Range Neutron Monitoring System
PSAR	Preliminary Safety Analysis Report
QA	Quality Assurance
RB	Reactor Building
RC&IS	Rod Control and Information System
RCPB	Reactor Coolant Pressure Boundary
RG	Regulatory Guide
RIV	Reactor Isolation Valve
RLC	Reactor Level Control
RPC	Reactor Pressure Control
RPV	Reactor Pressure Vessel
SAFDL	Specified Acceptable Fuel Design Limits
SAMG	Severe Accident Management Guideline
SC	Safety Class
SC1	Safety Class 1
SC2	Safety Class 2
SC3	Safety Class 3
SCN	Non-Safety Class
SCR	Secondary Control Room
SCRRI	Selected Control Rod Run-In
SDC	Shutdown Cooling System

Acronym	Explanation
SPDS	Safety Parameter Display System
SSCs	Structures, Systems, and Components
STP	Simulated Thermal Power
TMR	Triple Modular Redundant
TA	Task Analysis
TMI	Three Mile Island
TMR	Triple Modular Redundant
TS	Technical Specifications
UDH	Unit Data Highway
UPS	Uninterruptible Power Supply
USNRC	United States Nuclear Regulatory Commission
V&V	Verification and Validation
VDU	Visual Display Unit
WRNM	Wide Range Neutron Monitor

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7.0 INSTRUMENTATION AND CONTROLS

The information in Chapter 7 is organized to systematically present the design bases of the Instrumentation and Control (I&C) systems in the necessary context to support an understanding of the individual I&C system designs and safety category functions and their interconnection into a Distributed Control and Information System (DCIS). Information on the relationship of the I&C systems to plant level requirements is presented in Section 7.1. Next, information on the architecture of the I&C system of systems is presented in Section 7.2. Then information on the individual I&C systems, including accident monitoring instrumentation, is presented in Section 7.3. The development processes for the digital I&C systems are presented in Section 7.4. The I&C systems in the Main Control Room (MCR) are described in Section 7.5. The I&C systems in the Secondary Control Room (SCR) are described in Section 7.6. The I&C systems in the emergency response facilities are described in Section 7.7. Hazards analyses for the BWRX-300 I&C systems are described in Section 7.8.

7.1 Instrumentation and Control Introduction and Overview

The BWRX-300 systems are designed to support the plant safety strategy described in Chapter 15. The BWRX-300 Small Modular Reactor utilizes an integrated digital-based I&C design. The I&C architecture is arranged to support a plant level Defense-in-Depth (D-in-D) framework. The BWRX-300 D-in-D framework is supported by a safety analysis framework that provides a consistent analytical basis for the Defense Lines (DLs). The BWRX-300 D-in-D framework links to a classification scheme based on the importance of the individual DLs. The BWRX-300 I&C architecture and associated systems and components are designed in accordance with the relevant international standards and via proven engineering design practices and processes, which represent state-of-the-art methods.

The BWRX-300 passive safety features present a simpler and more flexible design with large safety margins. These safety improvements also reduce plant complexity and lead to design optimizations that can lower overall cost. The passive safety design requires fewer automatic actuation functions that also eliminate automatic control capabilities and eliminates required operator actions for Design Basis Accidents (DBAs) for 72-hours. The design does not require active monitoring of critical plant safety functions to support near-term operator actions or emergency planning decisions for design basis events. The simplicity of the BWRX-300 design has limited requirements for safety support systems for the highest classified I&C equipment performing safety functions to mitigate design basis events. The short times required for the actuation of safety category I&C functions allow for optimization of the BWRX-300 support systems and structures.

7.1.1 Relationship Between Instrumentation and Control Functions and Plant Level Defense Lines

Defense Line 1 (DL1) includes the quality measures and design features employed to minimize potential for failures and initiating events to occur in the first place and to minimize potential for failures to occur in subsequent lines of defense. These quality measures cover the design, construction, operation, and maintenance of the plant. DL1 also includes the use of appropriate conservatism in design and analyses. Accident monitoring instrumentation supports more than one DL and receives DL1 treatment by application of applicable industry standards for accident monitoring instrumentation, as shown in Table 7.1-2 and 7.1-3.

Defense Line 2 (DL2) contains Safety Category 3 plant functions (e.g., anticipatory reactor trips) designed to detect and mitigate Postulated Initiating Events (PIEs) in the Anticipated Operational Occurrences (AOOs) frequency range category and prevent plant conditions from escalating to accident conditions. Safety Category 3 functions that normally operate to actively control reactor parameters are also part of DL2 (e.g., reactor level and pressure control and control rod positioning).

Defense Line 3 (DL3) contains Safety Category 1 functions that act to mitigate PIEs consisting of AOOs, DBAs by preventing core damage. These Safety Category 1 functions provide a level of assurance for maintaining integrity of the physical barriers that prevent radiological release and place the plant in a safe state. DL3 also includes Safety Category 1 functions that maintain the plant in a safe condition following mitigation of PIEs until normal operations are resumed. Safety Category 1 functions typically include reactor scram and actuation of engineered safety features. Safety Category 1 functions are needed when Safety Category 3 functions are not effective at intercepting a PIE or when a PIE is beyond the capabilities of Safety Category 3 functions. Accordingly, DL3 provides D-in-D in that it provides an additional layer of mitigation for AOO PIEs. Primary Safety Category 1 functions include actuations of reactor scram, Reactor Pressure Vessel (RPV) and containment isolations, and Isolation Condenser System (ICS) initiation to mitigate the consequences of AOOs or DBAs.

Safety Category 1 functions are credited to mitigate PIEs independent of Safety Category 2 and 3 functions and are therefore required to be independent from Safety Category 2 and 3 functions to the extent needed to meet plant safety goals. Exceptions related to the I&C systems are cases where probabilistic safety analyses indicate safety is improved by the sharing of components (e.g., Local Power Range Monitors (LPRMs) for Safety Category 1 hydraulic scram and Safety Category 3 rod block functions).

Defense Line 4 comprises two subsets, designated as Defense Line 4a (DL4a) and Defense Line 4b (DL4b). DL4a includes Safety Category 2 functions required to place and maintain the plant in a safe state in case of PIEs and event sequences with failure of the Safety Category 1 functions (e.g., Design Extension Conditions (DEC) without core damage). DL4a provides D-in-D in that it provides an additional layer of mitigation for DBA PIEs. Safety Category 2 functions detect and mitigate DECs. The need for Safety Category 2 functions arises when specific, postulated Common Cause Failures (CCFs) occur in DL3. Safety Category 2 functions actuate for any condition that satisfies the actuation criteria logic.

Safety Category 2 functions are used as a backup to DL3 Safety Category 1 functions (e.g., CCFs postulated to occur in DL3 coincident with DBA PIEs). Safety Category 2 functions are designed to work in tandem with Safety Category 3 functions to ensure AOOs and DBAs resulting from a single failure are mitigated by two DLs among DL2, DL3, and DL4a. Accordingly, DL4a is not required to be independent and diverse from DL2. Safety Category 2 functions can be used, along with unaffected Safety Category 3 functions, to mitigate a PIE as part of the same event sequence (i.e., to act as a single DL and not as two independent DLs in an DEC analysis). All AOOs and DBAs resulting from a single failure are required to be mitigated by Safety Category 1 functions and separately by Safety Category 3 functions, Safety Category 2 functions, or a combination of Safety Category 2 and 3 functions. Safety Category 2 and 3 functions are not credited to mitigate the same PIEs independently of each other and are therefore not required to be independent from each other.

DL4b contains Safety Category 3 functions that are explicitly provided to prevent or mitigate a severe accident while keeping radioactive releases to acceptable levels. The Safety Category 3 functions intended for mitigating DECs have functional and design requirements derived from the supporting safety analyses.

The DL independence requirements are consistent with the D-in-D strategy (i.e., the crediting of DLs in the fault evaluation and deterministic safety analyses). Although not specifically required by NEDC-33934P, "BWRX-300 Safety Strategy," (Reference 7.1-26), there is independence between Safety Category 2 and Safety Category 3 functions to a level that does meet this definition of practicable; it is not justifiable or cost-effective to require full independence between Safety Category 2 and Safety Category 3 functions (i.e., the type of independence required for Safety Category 1 functions). The only aspects of independence required between equipment performing Safety Category 2 and Safety Category 3 functions are required electrical isolation and physical separation for cable raceways and I&C cabinets. Equipment performing DL4b functions are independent of any equipment postulated to have failed in the event sequence those functions are mitigating. These interface requirements stem from the different safety classifications of the equipment.

Defense Line 5 (DL5) includes emergency preparedness measures to cope with potential unacceptable releases in case the first four DLs are not effective. These are off-site measures taken to protect the public in a scenario involving substantial release of radiation. DL5 is supported by accident monitoring instrumentation.

7.1.2 Instrumentation and Control System Classification

The BWRX-300 DCIS is an integrated control and monitoring system for the plant. The DCIS is arranged in three safety classified DCIS segments and a Non-Safety Class (SCN) segment with appropriate levels of hardware and software quality corresponding to the system functions they control and their equipment locations, as described in Section 3.2. The BWRX-300 DCIS provides control, monitoring, alarm, and recording functions. The various bus segments of the integrated DCIS are designed to operate autonomously.

The Safety Category 1 functions are allocated to DL3 and implemented in Safety Class 1 (SC1) equipment, with the following exceptions:

1. Structures, Systems, and Components (SSCs) only needed after the first 72-hours of the event are classified as Safety Class 2 (SC2).
2. SSCs only needed after the first 7 days of the event are classified as Safety Class 3 (SC3).

The Safety Category 2 functions allocated to DL4a are implemented in at least SC2 equipment, except for SSCs that are only needed after the first 7 days of the event, which are classified as SC3. Safety Category 2 functions complete the safety objective via an independent and diverse means of logic and actuation if the Safety Category 1 function is not completed due to an SC1 equipment failure.

The Safety Category 3 functions are allocated to DL2 and implemented in at least SC3 equipment. The Safety Category 3 functions need to be performed independently from diverse Safety Category 1 functions providing protection for the same event.

The Safety Category 3 functions allocated to DL4b are implemented in at least SC3 equipment unless other requirements are justified.

Accident monitoring is a DL1 provision and Safety Category 3 is assigned to functions that support monitoring and display of PAM Type B, C, D, E, and F accident monitoring variables.

The Safety Category 1 DCIS functions are allocated to the SC1 systems which require minimal support equipment (e.g., no active systems such as Heating, Ventilation, and Cooling System (HVS) or cooling water for at least 72-hours). Safety Category 1 DCIS functions are implemented on SC1 I&C system hardware and software platforms located in three separate divisional fire barrier rooms in the Reactor Building (RB). The Safety Class 1 I&C Reactor Mode Switch is also

located in the Control Building (CB). The Safety Category 2 and Safety Category 3 actuation functions are implemented in different systems. The Safety Category 2 DCIS functions are implemented with SC2 equipment. The majority of the SC2 equipment is located in its own fire barrier room in the CB with the remainder located in compartmentalized fire barrier rooms in the RB. The Safety Category 3 functions are implemented with SC3 equipment. The SC3 equipment is located in two separate fire barrier rooms in the CB.

SSCs that are not required to be SC1, SC2, or SC3 are classified as SCN. The SCN controller equipment is designed to prevent random I&C component failures from initiating plant transients. The non-safety category functions can be implemented on vendor packages (or skids) located in a separate fire barrier room in the CB. Vendor supplied SCN equipment is integrated into the SC3 DCIS network through SC3 gateways, as necessary.

The system designations of Primary Protection System as SC1, Diverse Protection System (DPS) as SC2, Nuclear Controllers as SC3, and Balance of Plant (BOP) Controllers as SCN represent an initial decomposition of the overall I&C systems based on safety classification. Further refinement of the system decomposition, based on functional grouping within a Safety Class (SC) and equipment selection, is performed as the I&C architecture design process progresses. This process is described in Subsection 7.4.2.2.

7.1.3 Regulatory Requirements, Regulatory Guidance, and Industry Standards

The discussion of regulatory requirements, regulatory guidance, and industry standards applicable to the BWRX-300 I&C systems is organized into common topics. Subsection 7.1.3.1 discusses the regulations related to Preliminary Safety Analysis Report (PSAR) content. Subsection 7.1.3.2 discusses regulations directly related to I&C systems. Subsection 7.1.3.3 discusses regulatory guidance and industry standards applicable to I&C systems.

7.1.3.1 Regulations Related to Preliminary Safety Analysis Report Content

Title 10 of the Code of Federal Regulations (CFR) 50.34, “Contents of Applications; Technical Information,” specifies requirements regarding the content of an application.

The content of Chapter 7 is aligned with the review guidance in the U.S. Nuclear Regulatory Commission (USNRC) “Design Review Guide (DRG): Instrumentation and Controls for Non-Light-Water Reactor (Non-LWR) Reviews,” (Reference 7.1-1). The DRG was developed to address the immediate needs associated with the Non-LWR community. Because the DRG is technology inclusive, the DRG may be used for the review of Light Water Reactor (LWR) plant designs and other reactor technologies. Blue Energy requests that the DRG be used as an alternative approach to satisfying 10 CFR 50.34(h), “Conformance with the Standard Review Plan,” for the I&C system review.

7.1.3.2 Regulations Directly Related to Instrumentation and Control Systems

The following General Design Criteria (GDC) specified in 10 CFR Part 50, Appendix A, are considered applicable to the BWRX-300 I&C systems:

- GDC 1 – Quality standards and records
- GDC 2 – Design bases for protection against natural phenomena
- GDC 4 – Environmental and dynamic effects design bases
- GDC 10 – Reactor design
- GDC 13 – Instrumentation and control
- GDC 15 – Reactor Coolant System

- GDC 16 – Containment design
- GDC 20 – Protection system functions
- GDC 21 – Protection system reliability and testability
- GDC 22 – Protection system independence
- GDC 23 – Protection system failure modes
- GDC 24 – Separation of protection and control systems
- GDC 25 – Protection systems requirements for reactivity control functions
- GDC 27 – Combined reactivity control systems capability
- GDC 33 – Reactor coolant makeup
- GDC 34 – Residual heat removal
- GDC 35 – Emergency core cooling
- GDC 63 – Monitoring fuel and waste storage
- GDC 64 – Monitoring radioactivity releases

As described in Section 3.1, the following GDC have replaced by Principal Design Criteria (PDC):

- PDC 19 – Control room
- PDC 26 – Reactivity control system redundancy and capability
- PDC 28 – Reactivity limits
- PDC 29 – Protection against anticipated operational occurrences

Alignment of the GDC and PDC with the passive small modular reactor BWRX-300 design that incorporates the use of a multi-layered safety strategy using DLs to control and mitigate PIEs is described in Section 3.1.

Three GDC typically applied to I&C systems are not considered applicable because of the BWRX-300 passive safety features for reactor and containment cooling:

- GDC 38 – Containment heat removal
- GDC 41 – Containment atmosphere cleanup
- GDC 44 – Cooling water

Additional USNRC regulations that contain detailed design requirements related to the I&C design applicable to the BWRX-300:

- 10 CFR 50.34(f)(2)(iv) related to safety parameter display console with respect to displays and monitoring requirements
- 10 CFR 50.34(f)(2)(v) related to bypass and inoperable status indication with respect to operating bypass, maintenance bypass, and displays and monitoring requirements
- 10 CFR 50.34(f)(2)(xiv) related to containment isolation systems with respect to actuation system requirements
- 10 CFR 50.34(f)(2)(xvii) related to accident monitoring instrumentation

- 10 CFR 50.34(f)(2)(xviii) related to instrumentation for the detection of inadequate core cooling
- 10 CFR 50.34(f)(2)(xix) related to instruments for monitoring plant conditions following core damage
- 10 CFR 50.34(f)(2)(xxiv) related to central reactor vessel water level recording
- 10 CFR 50.44(c)(4) related to combustible gas control monitoring
- 10 CFR 50.55a(h)(3), Protection and Safety Systems, requires that for applications filed on or after May 13, 1999, safety systems need to meet the requirements in Institute of Electrical and Electronics Engineers (IEEE) Std 603-1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," (Reference 7.1-25)
- 10 CFR 50.62(c)(3) related to requirements for reduction of risk from Anticipated Transients Without Scram events

Four regulations are not considered applicable to the BWRX-300 design:

- 10 CFR 50.34(f)(2)(xi) related to direct indication of relief and safety valve position; the BWRX-300 design does not have traditional relief valves. The need for the direct indication is based on monitoring open relief valves for reactor coolant loss. This concern is not applicable to the closed system ICS used for overpressure protection.
- 10 CFR 50.34(f)(2)(xii) related to pressurizer heater power supply and associated motive and control power interfaces (applicable to pressurized water reactors only)
- 10 CFR 50.34(f)(2)(xx) related to power supplies for pressurizer relief valves, block valves, and level indicators (applicable to pressurized water reactors only)
- 10 CFR 50.34(f)(2)(xxii) related to failure modes and effects analysis of the integrated control system to include consideration of failures and effects of input and output signals to the integrated control system (applicable to Babcock and Wilcox designed plants only)

10 CFR 50.55a(z), Alternatives to codes and standards requirements, allows use of alternatives to the requirements of 10 CFR 50.55a(h), when authorized by the USNRC. A proposed alternative is submitted and authorized prior to implementation. The alternative needs to demonstrate that: (a) the proposed alternative would provide an acceptable level of quality and safety; or (b) compliance with the specified requirements in 10 CFR 50.55a, results in hardship or unusual difficulty without a compensating increase in the level of quality and safety. The BWRX-300 design uses IEEE Std 603-2018, "Standard Criteria for Safety Systems for Nuclear Power Generating Stations," (Reference 7.1-2) as an alternative to the IEEE Std 603-1991 (Reference 7.1-25) version to support a 3-division SC1 architecture. Subsection 7.3.1.3.2 describes the approach for addressing IEEE Std 603-2018 Clause 6.7.

The alignment of the regulations to the applicable BWRX-300 I&C system classes is shown in Table 7.1-1. SC1 systems are discussed in Subsection 7.3.1. SC2 systems are discussed in Subsection 7.3.2. SC3 systems are discussed in Subsection 7.3.3. SCN systems are discussed in Subsection 7.3.4.

7.1.3.3 Regulatory Guidance and Industry Standards Applicable to Instrumentation and Control Systems

The BWRX-300 I&C systems are designed using International Electrotechnical Commission (IEC) standards because they are proven engineering design practices and processes that represent state-of-the-art methods. The use of these standards supports harmonized licensing

internationally. These standards are tailored for nuclear sector requirements and needs. They represent a complete set of standards that are integrated and organized in a logical framework. The use of these international standards is allowed by the DRG.

The BWRX-300 I&C systems comply with the IEC 61513, “Nuclear Power Plants - Instrumentation and Control Important to Safety – General Requirements for Systems,” (Reference 7.1-3). The BWRX-300 uses Section 7 of IEC 61226, “Nuclear Power Plants – Instrumentation, Control and Electrical Power Systems Important to Safety – Categorization of Functions and Classification of Systems,” (Reference 7.1-4) to identify applicable I&C equipment standards. A summary of the key system and equipment standards applied to the I&C systems based on IEC 61226, (Reference 7.1-4) is shown in Table 7.1-2. The overall I&C regulatory conformance plan is documented in a BWRX-300 plant I&C systems nuclear regulations and standards compliance plan.

Each IEC digital I&C standard is complete with respect to associated V-model (i.e., system, hardware, software development V-models). These standards address process, documentation, and technical requirements that align with the BWRX-300 SCs. They use graded requirements based on the importance of system functionality. The standards are up to date and are proven to support industry needs. The IEC standards provide a more organized approach to digital I&C design standards than the approach currently endorsed by USNRC.

Section 7.3 provides a summary of the alignment for each I&C system to the regulatory guidance in the DRG and the key industry I&C standards IEC 61513 (Reference 7.1-3) and IEEE Std 603-2018 (Reference 7.1-2). Section 7.4 provides a summary of the alignment for the I&C system development process to the regulatory guidance in the DRG and the key industry I&C standards IEC 61513 (Reference 7.1-3) and IEEE Std 603-2018 (Reference 7.1-2).

The IEC standards align with the technical topics in the USNRC RGs, as shown in Table 7.1-3. The use of the IEC standards as alternatives to the IEEE standards endorsed in the relevant RG satisfy the underlying regulatory criteria (i.e., regulations and GDC) defined in the RGs.

7.1.3.4 Compliance with 10 CFR 50, Appendix A

The BWRX-300 I&C systems comply with the following GDC, which have plant-wide applicability, as discussed in Section 3.1:

- GDC 1 – Quality standards and records
- GDC 2 – Design bases for protection against natural phenomena
- GDC 4 – Environmental and dynamic effects design bases

The following GDC and PDC have aspects specific to I&C systems and compliance is addressed as follows:

GDC 10 – Reactor design - The SC3 I&C systems include the control, electrical, mechanical, and measurement elements that support DL2 functions during normal operations and AOOs. These functions are designed to ensure that Specified Acceptable Fuel Design Limits (SAFDLs) are protected with margin appropriate for AOO PIEs. Subsection 7.3.3.2 describes the functions associated with the SC3 systems. In support of DL2 functions, SC1 I&C systems provide isolated LPRM signals to the SC3 equipment. Subsection 7.3.1.2 describes the functions associated with the SC1 systems.

GDC 13 – Instrumentation and control - DL3 functions are performed by independent and diverse SC1 I&C systems to maintain the specified variables by automatic safety actuations to assure adequate safety. The lower classified information displays associated with the SC1 I&C systems provide the monitored parameters during normal operation and for accident monitoring by the

operators. Subsection 7.3.1.2 describes the functions associated with the SC1 systems. The SC3 I&C systems perform DL2 functions to monitor plant variables in the reactor core, Reactor Coolant Pressure Boundary (RCPB), containment and supporting systems. Normal and off-normal operating ranges are defined to support process control during normal operation and the automatic initiation of control functions to maintain the specified variables within prescribed operating ranges through normal control or, when required, by automatic safety actuations to assure adequate safety and accident monitoring in the event of AOOs or DBAs. Subsection 7.3.3.2 describes the functions associated with the SC3 systems.

GDC 15 – Reactor coolant system design - The SC3 I&C systems perform DL2 functions to automatically initiate hydraulic scram and pressure control functions so that the reactor coolant pressure design limits are not exceeded during AOOs. Subsection 7.3.3.2 describes the functions associated with the SC3 systems.

GDC 16 – Containment design - SC1 I&C systems perform the DL3 functions for reactor, containment, and system isolation to limit line break effects both inside and outside containment. Subsection 7.3.1.2 describes the containment isolation functions associated with the SC1 systems. SC2 I&C systems perform the DL4a functions for reactor, containment, and system isolation to limit line break effects both inside and outside containment. These functions are diverse from the SC1 DL3 functions. Subsection 7.3.2.2 describes the containment isolation functions associated with the SC2 systems.

PDC 19 – Control room - The SC1 I&C systems provide lower classified operator interfaces in the MCR and SCR for display of Type B and C accident monitoring parameters and provide the ability to place and maintain the reactor in safe shutdown in the event of a MCR evacuation event. Prior to evacuating the MCR, operators trip the reactor and initiate the ICS and containment isolation. The BWRX-300 design does not rely on operator action, instrumentation, or controls outside of the MCR to maintain safe shutdown. The design includes Type B and C accident monitoring parameters displayed in the SCR to allow operators to monitor the reactor for safe shutdown condition. The MCR and SCR design and functions related to I&C are described in Sections 7.5 and 7.6. The PDC 19 evaluation provided in Subsection 3.1.2.10 describes the basis of how the BWRX-300 design complies with the requirement of PDC 19 as it relates to maintaining the plant in a safe shutdown condition under DBA conditions without operator action from the MCR.

GDC 20 – Protection system functions - The SC1 I&C systems are designed to provide timely protection in sensing accident conditions and initiating the required DL3 functions. Subsection 7.3.1.2 describes the functions associated with the SC1 systems. The SC3 I&C systems automatically initiate the DL2 functions to assure that AOO acceptance criteria, including SAFDL, are not exceeded, if monitored variables exceed pre-established limits. Additionally, the SC3 I&C systems sense DBA conditions and automatically initiate DL2 functions to assure that DBA acceptance criteria are met. Subsection 7.3.3.2 describes the functions associated with the SC3 systems.

GDC 21 – Protection system reliability and testability - The SC1 I&C system functions provide assurance that, through redundancy and independent divisions of equipment, the SC1 I&C system has sufficient reliability to fulfill the single-failure criterion. No single component failure, intentional bypass maintenance operation, calibration operation, or test to verify operational availability, impairs the ability of the system to perform its intended DL3 functions. The SC1 I&C system reliability, single failure tolerance, and testability, including surveillance testing, is described in Subsection 7.3.1.3.2. The SC3 nuclear segment process controllers are designed with Triple Modular Redundant (TMR) controllers to prevent random I&C component failures from causing plant transients and ensure the system is not affected by a single failure. The Technical

Specifications (TS) Monitor provides surveillance tests and support, including support for the three SC1 divisions available through isolated, unidirectional boundary devices and gateways. The TS monitors are designed to support online and automatic surveillance testing to meet TS requirements. The systems implementing Safety Category 3 protection functions are designed to ensure acceptable reliability of operation, as such DL2 functions are designed with a target reliability of 1E-02 failures per demand or less. This is accomplished by using TMR controllers for the majority of DL2 functions (control rod block functions are implemented in dual redundant hardware and are designed to fail-safe), as well as triple redundant input sensors. TMR controller inputs and outputs for Safety Category 3 functions use dedicated communication paths; execution of these Safety Category 3 functions does not rely on the Unit Data Highway. Additionally, the Safety Category 3 protection functions are designed to ensure acceptable reliability of operation such that: (1) no single sensor or controller failure results in the loss of a protection function, (2) removal of a single sensor or controller from service does not result in the loss of a protection function, (3) no single power feed failure results in the loss of a protection function, and (4) removal of a single power feed from service does not result in the loss of a protection function. Acceptable reliability and availability of operation is established by monitoring DL2 functions in accordance with the Availability Controls Manual, as described in Subsection 16.4.2. The SC3 I&C system reliability, single failure tolerance, and testability, including surveillance testing, are described in Subsections 7.3.3.2 and 7.3.3.3.2.

GDC 22 – Protection system independence - The SC1 equipment that performs DL3 functions is implemented with independence between redundant equipment such that the effects of natural phenomena, and of normal operation, maintenance, testing, and postulated accident conditions do not result in loss of the minimum redundancy required for successful performance of DL3 functions. The SC1 I&C system independence is described in Subsection 7.3.1.3.3. The SC3 equipment implements DL2 functions with independence from the SC1 I&C systems. While DL2 functions do not have the same level of independence that exists within DL3 functions, DL2 functions are implemented with independence adequate to meet their reliability requirement pursuant to an “other defined basis” described in GDC 22. SSCs performing a DL2 function are designed to operate in the environment expected during normal operations and in response to AOOs relying on the DL2 function for mitigation. The resulting reliability of DL2 functions is adequate to provide confidence that these functions will perform their functions when needed. The BWRX-300 systems making up the protection system have layers of protection, with both DL2 and DL3 functions being able to mitigate AOOs to within acceptance criteria described in Table 15.3-1 and Table 15.3-2, respectively. There is independence between these layers, as SC3 equipment is independent from SC1 equipment except in cases where sharing of in-core components (e.g., Local Power Range Monitors (LPRMs) data is utilized to initiate Safety Category 1 hydraulic scram functions and Safety Category 3 rod block functions) is necessary due to practical limitations. Additionally, there is diversity in hardware, as the SC1 I&C systems are diverse from other I&C systems (i.e., hardware platforms are diverse). The independence and diversity between layers of protective functions achieve high plant-level protection reliability.
GDC 23 – Protection system failure modes - The SC1 I&C systems implementing DL3 functions are designed to fail into a safe state if conditions such as disconnection of the system, loss of energy, or postulated adverse environments are experienced. The SC1 I&C system fail safe design is described in Subsection 7.3.1.3.3. The SC3 nuclear segment equipment is designed to fail-as-is to avoid spurious actuation. The fail-as-is, energize to actuate, design is used to prevent the lesser classified SC3 I&C systems (e.g., Anticipatory Trip System) from creating unnecessary challenges to plant safety requiring action by DL3 functions. The SC3 I&C system fail-as-is design is described in Subsection 7.3.3.3.3.

GDC 24 – Separation of protection and control systems - The SC1 DL3 functions are implemented separately from the SC3 DL2 functions to the extent that any failure leaves intact a group of

equipment satisfying all reliability and independence requirements necessary to assure performance of the DL3 functions for design basis events. Interconnections between equipment performing DL2 functions and DL3 functions is limited such that either the DL3 or DL2 functions remain available in the presence of any credible single failure, or credible CCF, postulated to affect mitigation capabilities of either SC. The SC1 signals are electrically isolated and physically separated from SC3 I&C systems. Sensor signal outputs may be shared, but each signal is optically isolated before entering a redundant or lower classified channel interface. The independence features prevent faults in a lower classified system from propagating to a higher classified system. The SC1 I&C system independence features are described in Subsection 7.3.1.3.3. The SC3 system functions are described in Subsection 7.3.3.2.

GDC 25 – Protection systems requirements for reactivity control functions - SC3 DL2 I&C system functions, along with other mitigating design features, assure that SAFDL are not exceeded for reactivity control malfunctions. To prevent a rod withdrawal error, the rod control system has redundancy to limit the effect of single failures. Additionally, the rod patterns are enforced for rod withdrawals. If there is a malfunction of the rod control system that results in a rod withdrawal error during startup, a DL2 rod block is initiated based upon a Wide Range Neutron Monitor (WRNM) Short Reactor Period signal. If the rod withdrawal error were to result in a further decrease to the WRNM Short Reactor Period based setpoint, a reactor scram would occur. If a rod withdrawal error were to occur at higher power, the Average Power Range Monitor (APRM) scram would terminate the event if it were to continue to its setpoint. Subsection 7.3.3.2 describes the functions associated with the SC3 systems. In support of DL2 functions, SC1 I&C systems provide isolated LPRM signals. Subsection 7.3.1.2 describes the functions associated with the SC1 systems.

PDC 26 – Reactivity control system redundancy and capability. The SC1 I&C system, diverse SC2 I&C system, and the SC3 Anticipatory Trip System separately initiate DL2, DL3, and DL4a Hydraulic Scram functions. Each of these Hydraulic Scram functions also initiates a scram follow signal to the SC2 Fine Motion Control Rod Drive (FMCRD) control system which inserts the control rods with the motors operating at normal speed. A DL4a Fast Motor Run-In function is initiated if the SC2 I&C system does not obtain positive confirmation of successful Hydraulic Scram. The SC1, SC2, and SC3 system functions are described in Subsections 7.3.1.2, 7.3.2.2, and 7.3.3.2, respectively. A SC3 Boron Injection System (BIS) is also provided for diversity and is described in Subsection 9.3.10. The Boron Injection System (BIS) is a diverse, defense-in-depth system which is redundant to the CRD system with regards to its capability of shutting down the reactor and maintaining it in shutdown conditions. It performs Safety Category 3 DL4b functions to assure reactor shutdown from full power operation to cold subcritical conditions without control rod movement by mixing a neutron absorber with the primary coolant. The BIS and the CRD system have no shared components and operate on different design principles. The PDC 26 evaluation provided in Subsection 3.1.3.7 describes the basis of how the BWRX-300 design complies with the requirement of PDC 26 as it relates to the diversity, redundancy, and independence of the Hydraulic Control Units (HCUs) and the FMCRD motors for reactivity control during normal operation, planned power changes, AOOs, and postulated accidents

GDC 27 - The SC1 I&C system, diverse SC2 I&C system, and the SC3 Anticipatory Trip System separately initiate DL2, DL3, and DL4a Hydraulic Scram functions. Positive automatic insertion of the control rods (i.e., scram) is provided by the HCU accumulators in response to design basis accidents (DBAs) that require emergency core cooling system operation (i.e., initiation of reactor isolation and the isolation condenser system (ICS)). Successful scram and, when required, initiation of reactor isolation and the ICS is shown to both achieve and maintain safe shutdown of the reactor without the possibility of a return to criticality, even with the unlikely occurrence of the most reactive control rod fully withdrawn or rod pair associated with a single HCU postulated stuck

in the full out position. The SC1, SC2, and SC3 system functions are described in Subsections 7.3.1.2, 7.3.2.2, and 7.3.3.2, respectively. GDC 27 is satisfied for the BWRX-300 without the injection of soluble boron.

PDC 28 – Reactivity limits. The SC1 I&C system implements DL3 functions to provide timely protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and RCPB. The SC3 DL2 functions provide the control logic and positions the control rods using the SC2 FMCRD control system. The SC1, SC2, and SC3 system functions are described in Subsections 7.3.1.2, 7.3.2.2, and 7.3.3.2, respectively.

PDC 29 – Protection against anticipated operational occurrences. For events that result in AOOs, the DL2 functions ensure SAFDLs are not exceeded by providing a reactor scram to shut down the reactor. If the event reaches the DL3 function setpoints, a scram is enforced by the SC1 I&C system DL3 functions. The reliability of Safety Category 1 functions imposed by the BWRX-300 Safety Strategy is 1E-04 failures per demand, assuring an extremely high probability of accomplishing their Safety Category 1 functions. The reliability of Safety Category 3 functions imposed by the BWRX-300 Safety Strategy is 1E-02 failures per demand, assuring a high probability of accomplishing their Safety Category 3 functions. The combination of DL2 and independent DL3 functions provide layered protection against AOO PIEs, and results in extremely high functional reliability for performance of the Fundamental Safety Functions (FSFs) during and after design basis events, including AOOs. The SC1 and SC3 system functions are described in Subsections 7.3.1.2 and 7.3.3.2, respectively.

GDC 33 – Reactor coolant makeup - For normal operational leakage, the SC3 I&C system performs the DL2 functions to maintain Reactor Coolant System level. Subsection 7.3.3.2 describes the functions associated with the SC3 systems. For breaks, the SC1 I&C system performs the DL3 function to actuate Reactor Isolation Valves (RIVs). Subsection 7.3.1.2 describes the functions associated with the SC1 systems.

GDC 34 – Residual heat removal - The SC1 I&C system actuates DL3 ICS functions to provide the necessary residual heat removal capability for ensuring SAFDL are met, and RCPB design conditions are not exceeded. Subsection 7.3.1.2 describes the functions associated with the SC1 systems.

GDC 35 – Emergency Core Cooling - SC1 I&C system performs the DL3 functions to initiate ICS and reactor, containment, and system isolation functions. The ICS design includes features that provide redundancy in components, because the worst-case single failure can only affect one ICS train or RIV. Subsection 7.3.1.2 describes the functions associated with the SC1 systems.

GDC 63 – Monitoring Fuel and Waste Storage - The SC3 I&C systems provide monitoring and control for the spent fuel pool redundant heat exchangers and pumps and their associated valves to detect conditions that may result in loss of residual heat removal capability and implement appropriate DL2 functions. Subsection 7.3.3.2 describes the functions associated with the SC3 systems.

GDC 64 – Monitoring radioactivity releases - The BWRX-300 I&C systems provide process radiation monitoring and Area Radiation Monitoring Subsystems (ARMs) to monitor for radioactivity releases from normal operations, including AOOs and from DBAs. The SC3 DCIS provides displays in both the MCR and SCR as well as a Safety Parameter Display System (SPDS) function that provides a big picture overview display of plant parameters during transients and accidents. The SC3 DCIS displays and the SPDS function are described in Subsections 7.3.3.4 and 7.3.3.2, respectively.

Additional Information on GDC and PDC compliance is found in Section 3.1.

7.1.4 References

7.1-1 “Design Review Guide (DRG): Instrumentation and Controls for Non-Light-Water Reactor (Non-LWR) Reviews,” U.S. Nuclear Regulatory Commission, February 2021.

7.1-2 IEEE 603, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” Institute of Electrical and Electronics Engineers, 2018.

7.1-3 IEC 61513, “Nuclear Power Plants – Instrumentation and Control Important to Safety-General Requirements for Systems,” International Electrotechnical Commission, 2011.

7.1-4 IEC 61226, “Nuclear Power Plants – Instrumentation, Control and Electrical Power Systems Important to Safety – Categorization of Functions and Classification of Systems,” International Electrotechnical Commission, 2020.

7.1-5 IEC 60709, “Nuclear Power Plants – Instrumentation, Control and Electrical Power Systems Important to Safety – Separation,” International Electrotechnical Commission, 2018.

7.1-6 IEC 63147/IEEE Std 497, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations,” International Electrotechnical Commission/Institute of Electrical and Electronics Engineers, 2017/2016.

7.1-7 IEC 60812, “Failure Modes and Effects Analysis (FMEA and FMECA),” International Electrotechnical Commission, 2018.

7.1-8 IEC 61000-4 Series, “Electromagnetic Compatibility Package,” International Electrotechnical Commission.

7.1-9 IEC 61000-6-2, “Electromagnetic Compatibility (EMC) – Part 6-2: Generic Standards-Immunity Standard for Industrial Environments,” International Electrotechnical Commission, 2016.

7.1-10 IEC 60880, “Nuclear Power Plants – Instrumentation and Control Systems Important to Safety – Software Aspects for Computer-Based Systems Performing Category A Functions,” International Electrotechnical Commission, 2006.

7.1-11 IEC 60987, “Nuclear Power Plants – Instrumentation and Control Important to Safety-Hardware Requirements,” International Electrotechnical Commission, 2021.

7.1-12 IEC/IEEE 60780-323, “Nuclear Facilities – Electrical Equipment Important to Safety-Qualification,” International Electrotechnical Commission/Institute of Electrical and Electronics Engineers, 2016.

7.1-13 IEC/IEEE 60980-344, “Nuclear Facilities – Equipment Important to Safety – Seismic Qualification,” International Electrotechnical Commission/Institute of Electrical and Electronics Engineers, 2020.

7.1-14 IEC 61500, “Nuclear Power Plants – Instrumentation and Control Systems Important to Safety – Data Communication in Systems Performing Category A Functions,” International Electrotechnical Commission, 2018.

7.1-15 IEC 62138, “Nuclear Power Plants - Instrumentation and Control Systems Important to Safety - Software Aspects for Computer-Based Systems Performing Category B or C Functions,” International Electrotechnical Commission, 2018.

7.1-16 IEC 61508 CMV, “Functional Safety of Electrical/Electronic/Programmable Electronic Safety-Related Systems – Parts 1 to 7,” Commented Version, International Electrotechnical Commission, 2010.

7.1-17 IEC 60671, “Nuclear Power Plants – Instrumentation and Control Systems Important to Safety – Surveillance Testing,” International Electrotechnical Commission, 2007.

7.1-18 IEC 62385, “Nuclear Power Plants – Instrumentation and Control Important to Safety-Methods for Assessing the Performance of Safety System Instrument Channels,” International Electrotechnical Commission, 2007.

7.1-19 IEC 62808 CSV, “Nuclear Power Plants - Instrumentation and Control Systems Important to Safety - Design and Qualification of Isolation Devices, for Isolation Qualification,” Consolidated Version, International Electrotechnical Commission, 2018.

7.1-20 IEC 61888, “Nuclear Power Plants – Instrumentation Important to Safety – Determination and Maintenance of Trip Setpoints,” International Electrotechnical Commission, 2002.

7.1-21 IEC 62566, “Nuclear Power Plants – Instrumentation and Control Important to Safety-Development of HDL-Programmed Integrated Circuits for Systems Performing Category A Functions,” International Electrotechnical Commission, 2012.

7.1-22 IEC 62645, “Nuclear Power Plants - Instrumentation, Control and Electrical Power Systems - Cybersecurity Requirements,” International Electrotechnical Commission, 2019.

7.1-23 IEC 62859, “Nuclear Power Plants - Instrumentation and Control Systems-Requirements for Coordinating Safety and Cybersecurity,” International Electrotechnical Commission, 2016.

7.1-24 NEI 08-09, “Cyber Security Plan for Nuclear Power Reactors,” Nuclear Energy Institute, Rev. 6, 2010.

7.1-25 IEEE Std 603, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” Institute of Electrical and Electronics Engineers, 1991.

7.1-26 NEDC-33934P, “BWRX-300 Safety Strategy,” GE Vernova Hitachi Nuclear Energy Americas, LLC, Revision 2, March 2026.

Table 7.1-1: Alignment of Regulations to BWRX-300 Instrumentation and Control Safety Classes

Applicable GDC, PDC, and Regulations	BWRX-300 Safety Classes			
	SC1	SC2	SC3	SCN
10 CFR 50.34(f)(2)(iv)	-	-	X	-
10 CFR 50.34(f)(2)(v)	X	-	X	-
10 CFR 50.34(f)(2)(xi)	Not Applicable (Note 1)	-	-	-
10 CFR 50.34(f)(2)(xiv)	X	X	-	-
10 CFR 50.34(f)(2)(xvii)	X	-	X	-
10 CFR 50.34(f)(2)(xviii)	X	-	X	-
10 CFR 50.34(f)(2)(xix)	X	-	X	-
10 CFR 50.34(f)(2)(xxiv)	X	-	X	-
10 CFR 50.44(c)(4)	X	-	X	-
10 CFR 50.55a(h)(3)	X	-	-	-
10 CFR 50.62(c)(3)	-	X	-	-

Note:

(1) Not applicable because the BWRX-300 does not have relief valves and overpressure protection is provided by the ICS, as described in Section 6.3.

Table 7.1-2: Instrumentation and Control System and Equipment Standards

Safety Class	Description	Systems	Equipment
SC1, SC2, and SC3	I&C equipment supporting Safety Category 1, 2, and 3 functions	• IEC 61513, "Nuclear Power Plants – Instrumentation and Control Important to Safety – General Requirements for Systems," (Reference 7.1-3) (I&C architecture and general system requirements) • IEC 60709, "Nuclear Power Plants – Instrumentation, Control and Electrical Power Systems Important to Safety – Separation," (Reference 7.1-5) (separation) • IEC 63147/IEEE Std 497 "Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations," (Reference 7.1-6) (accident monitoring) • IEC 60812, "Failure Modes and Effects Analysis (FMEA and FMECA)" (Reference 7.1-7) (FMEA)	• IEC 61000-4 Series, "Electromagnetic Compatibility Package," (Reference 7.1-8) • IEC 61000-6-2, "Electromagnetic Compatibility (EMC) – Part 6-2: Generic Standards – Immunity Standard for Industrial Environments," (Reference 7.1-9) (electromagnetic compatibility)
SC1	I&C equipment supporting Safety Category 1 functions	• IEC 60880 "Nuclear Power Plants – Instrumentation and Control Systems Important to Safety – Software Aspects for Computer-Based Systems Performing Category A Functions," (Reference 7.1-10) (Category A software) • IEC 60987 "Nuclear Power Plants – Instrumentation and Control Important to Safety – Hardware Requirements," (Reference 7.1-11) (hardware) • IEEE Std 603 (Reference 7.1-2) (system)	• IEC/IEEE 60780-323, "Nuclear Facilities – Electrical Equipment Important to Safety – Qualification" (Reference 7.1-12) (environmental) • IEC/IEEE 60980-344, "Nuclear Facilities – Equipment Important to Safety – Seismic Qualification" (Reference 7.1-13) (seismic) • IEC 61500, "Nuclear Power Plants – Instrumentation and Control Systems Important to Safety – Data Communication in Systems Performing Category A Functions" (Reference 7.1-14) (Network communication)

Safety Class	Description	Systems	Equipment
SC2	I&C equipment supporting Safety Category 2 functions	• IEC 60987, (Reference 7.1-11) (hardware) • IEC 62138, "Nuclear Power Plants - Instrumentation and Control Systems Important to Safety - Software Aspects for Computer-Based Systems Performing Category B or C Functions" (Reference 7.1-15) (Category B software)	• IEC/IEEE 60780-323, (Reference 7.1-12) (environmental) • IEC 6150, "Functional Safety of Electrical/Electronic/Programmable Electronic Safety-Related Systems – Parts 1 to 7," (Reference 7.1-16) (can be considered for use on pre-developed equipment for SC2 systems)
SC3	I&C equipment supporting Safety Category 3 functions	• IEC 62138 (Reference 7.1-15) (Category C software)	• IEC/IEEE 60980-344 (Reference 7.3-13) (seismic) - Note 1 • IEC 61508, (Reference 7.1-16) (can be considered for use on pre-developed equipment for SC3 systems)

Note:

If required for mitigation of seismic-related events.

Table 7.1-3: Alignment of Regulatory Guides to BWRX-300 Alternatives

Regulatory Guides	BWRX-300 Alternative
RG 1.22, “Periodic Testing of Protection System Actuation Functions”	• IEC 60671, “Nuclear Power Plants – Instrumentation and Control Systems Important to Safety – Surveillance Testing,” (Reference 7.1-17) • IEC 62385, “Nuclear Power Plants – Instrumentation and Control Important to Safety – Methods for Assessing the Performance of Safety System Instrument Channels,” (Reference 7.1-18)
RG 1.47, “Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems”	• IEEE Std 603 (Reference 7.1-2)
RG 1.53, “Application of the Single-Failure Criterion to Safety Systems”	
RG 1.62, “Manual Initiation of Protective Actions”	
RG 1.75, “Physical Independence of Electric Systems”	• IEC 60709 (Reference 7.1-5) • IEC 62808, “Nuclear Power Plants - Instrumentation and Control Systems Important to Safety - Design and Qualification of Isolation Devices, for Isolation Qualification,” (Reference 7.1-19)
RG 1.89, “Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants”	• IEC/IEEE 60780-323 (Reference 7.1-12)
RG 1.97, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants”	• IEC 63147/IEEE Std 497 (Reference 7.1-6)
RG 1.100, “Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants”	• IEC/IEEE 60980-344 (Reference 7.1-13)
RG 1.105, “Setpoints for Safety-Related Instrumentation”	• IEC 61888, “Nuclear Power Plants – Instrumentation Important to Safety – Determination and Maintenance of Trip Setpoints,” (Reference 7.1-20)
RG 1.118, “Periodic Testing of Electric Power, and Protection Systems”	• IEC 60671 (Reference 7.1-17) • IEC 62385 (Reference 7.1-18)
RG 1.152, “Criteria for Use of Computers in Safety Systems of Nuclear Power Plants”	• IEC 61513 (Reference 7.1-3) • IEC 60880 (Reference 7.1-10) • IEC 62566, “Nuclear Power Plants – Instrumentation and Control Important to Safety – Development of HDL-Programmed Integrated Circuits for Systems Performing
RG 1.168, “Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants”	

Regulatory Guides	BWRX-300 Alternative
RG 1.169, Revision 1, “Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants”	Category A Functions” 2012 (Reference 7.1-21) • IEC 62138 (Reference 7.1-15) • IEC 62645, “Nuclear Power Plants - Instrumentation, Control and Electrical Power Systems - Cybersecurity Requirements,” (Reference 7.1-22) • IEC 62859, “Nuclear Power Plants - Instrumentation and Control Systems - Requirements for Coordinating Safety and Cybersecurity,”(Reference 7.1-23) (for secure development and operating environment) • IEC 61500 (Reference 7.1-14)
RG 1.170, Revision 1, “Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants”	
RG 1.171, Revision 1, “Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants”	
RG 1.172, Revision 1, “Software Requirement Specifications for Digital Computer Software and Complex Electronics Used in Safety Systems of Nuclear Power Plants”	
RG 1.173, Revision 1, “Developing Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants”	
RG 1.180, Revision 2, “Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems”	• IEC 61000-4 Series (Reference 7.1-8)
RG 1.209, Revision 0, “Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants”	• IEC/IEEE 60780-323 (Reference 7.1-12)
RG 1.227, Revision 0, “Wide-Range Spent Fuel Pool Level Instrumentation”	• IEC 63147/IEEE Std 497 (Reference 7.1-6)
RG 5.71, Revision 1, “Cyber Security Programs for Nuclear Facilities”	• NEI 08-09, “Cyber Security Plan for Nuclear Power Reactors,” (Reference 7.1-24) • IEC 62645 (Reference 7.1-22) • IEC 62859 (Reference 7.1-23) (for secure development and operating environment)

7.2 Instrumentation and Control Systems of Systems

The DCIS systems (i.e., Primary Protection System, DPS, Nuclear Controllers, and SCN Controllers) are developed and integrated into an architecture that implements the BWRX-300 D-in-D strategy, as described in Subsection 7.4.2. The DCIS systems are networked using a managed network switch scheme that allows the parts to operate independently and prevent faults from propagating while appearing seamless to the plant operator. The BWRX-300 I&C segments are interconnected (networked) to support common services like alarming, Visual Display Units (VDU), recording, and sending data to the emergency response facilities while maintaining appropriate separation between the segments and providing for the necessary cyber security.

The BWRX-300 I&C systems architecture incorporates defensive cyber security features designed to meet regulatory requirements, as described in Subsection 7.4.4. Managed network switches are standard industrial components with a proven record for reliable and safe operation in similar network applications. The BWRX-300 cyber security architecture provides cyber security defensive levels. Data flow and communications between defense levels are managed by boundary devices that implement unidirectional flow where required to prevent a lower level from compromising a higher level. Isolation devices that are provided at interfaces between systems with different classification levels are classified to the higher classification. Signals used from lower SC equipment are not used to perform higher safety category functions in the higher SC equipment.

7.2.1 Distributed Control and Information System

The various DCIS systems are implemented on different hardware and software platforms appropriate to the safety classification of the functions they are performing, as indicated by the following colors in Figure 7.2-1:

- Red – Safety Category 1 functions and SC1 equipment
- Green – Safety Category 2 functions and SC2 equipment
- Blue – Safety Category 3 functions and SC3 equipment
- Black – No Safety Category functions and SCN equipment
- Yellow – Enterprise network

The SC1 DCIS is comprised of three divisions that operate completely independently from each other (except for voting). It is completely isolated from the rest of the BWRX-300 DCIS except for the optically isolated data links provided through unidirectional boundary devices to SC3. Isolated information only flows through permanent connections in one direction through boundary devices to the SC3 networks so that SC1 system information may be alarmed, monitored, and recorded (but not controlled) with SC2 and SC3 equipment (e.g., VDUs, printers, alarm system, and historians). They can also be monitored by the VDUs associated with the SC1 systems. The SC1 DCIS divisions are separately powered by the three divisions of the SC1 electrical system.

DL4a performs Safety Category 2 DPS functions using SC2 equipment. Safety Category 2 DPS actuation logic operates on an I&C platform that is diverse from the SC1 equipment. It communicates status and diagnostic information to the SC3 portion which is based on digital I&C technology.

DL2 performs Safety Category 3 Nuclear Controller functions using SC3 equipment. The SC3 DCIS is redundant and segmented into two parts. The two network segments are the nuclear segment and the BOP segment. The two bus segments are classified as SC3. The nuclear segment uses controllers classified as SC3. The SC3 BOP segment interfaces with the SCN

controllers to acquire information to support Human-System Interface (HSI) integration in the SC3 DCIS. Both segments have their own redundant Unit Data Highway (UDH), and each independently provides the control and monitoring capability of nuclear and BOP I&C for their segment. Each segment has their own redundant network managed switches classified as SC3. Each segment can operate independently of the other (although it appears seamless to the operator). Each segment has associated VDUs, historians, and an alarm system. The SC3 segments use TMR and redundant controller architectures for various functions to support reliability goals or prevent adverse actuations for anticipated hardware component failures.

The nuclear segment supports the SC3 controllers (which include the major reactor control systems) by acquiring SC1 data from the SC1 controllers via unidirectional boundary devices.

The SC3 BOP segment provides for control of the plant equipment with no Safety Category function and controllers used for power generation. It also provides SC3 gateways to interface non-native controllers to the SC3 BOP segment. Although lower classified equipment, these functions use TMR controller architectures (referred to hereafter as TMR controllers) to prevent random I&C component failures from causing plant transients.

7.2.2 Unit Data Highway Network

The SC3 and SCN DCIS components are connected to the two UDH networks shown in Figure 7.2-1 using the general switch arrangement shown in Figure 7.2-2. The components are “dual ported,” which means that each is connected to separate but redundant network switches. The switches are discussed in the next section, but the important reliability point is that either connection to the network for the component and either network switch provides full functionality; failovers are seamless. The controllers are either dual redundant or TMR and the connections and switches are dual redundant. The design is fault tolerant, and components and connections are monitored and alarmed. The various components can be replaced online without affecting either safety or power generation. The UDH is characterized as nondeterministic and is not designed to be deterministic. For closed-loop control functions, logic and associated signals are confined to a single controller and communications with other controllers is designed to be reliable but it does not require guaranteed message delivery at exact specified response rates.

The UDH provides the control and monitoring capability of nuclear and BOP traditional I&C. Because of the network managed switches, the network segmentation appears transparent to the operator in normal operation (i.e., any SC3 VDU can monitor or control any SC3 or SCN equipment). However, in the unlikely event either of the individual redundant network segments becomes unavailable, the remaining segment remains operational.

7.2.3 Network Managed Switches

The blue lines in Figure 7.2-1 indicate that the redundant SC3 networks appear as busses; in fact, they are a rapid spanning tree network of managed Ethernet switches as shown on Figure 7.2-3.

Each managed network switch provides standard Ethernet switch capability but also provides additional cyber security advantages because they have security features. These include identification of authorized equipment addresses (locking out unauthorized equipment even if it is connected to the network), the capability to ignore or not uplink (to other segments), the capability to control which nodes are allowed to communicate and the capability to alarm abnormal network traffic. Each switch is programmed to allow intercommunication only between defined nodes and to refuse communication to or from a node whose identification has not been previously defined (i.e., a “stranger” plugging into the network). Only when a switch determines that an information data packet is destined for a node on another switch (and that communication is allowed) is the information put on an uplink to another switch. The network switches learn and maintain their own forwarding tables containing a list of the nodes and hosts on their respective network

segment. When a network switch receives a data communication packet, it forwards only that data packet to the segment to which the receiving host is connected. These protocols prevent data traffic between devices on the network from affecting devices on other segments of the network.

The uplink ports on the switches are connected radially and in a data communication ring because multiple interconnections increase reliability. Specifically, the switches use a “rapid spanning tree protocol” to automatically enable and disable ports so there is only one path from the nodes of one switch to another. Should a path become disabled, the switches automatically reconfigure to establish another path through the remaining switches and fiber optic cable paths. Reconfiguration requires no operator input and is usually accomplished in much less than a second.

As described in the previous section, each switch “node” (workstation, display, and controller) is connected to redundant switches of the segment. These connections support normal plant operation. Each switch has redundant power feeds and can work from either power source. The switches and connected controllers support extensive component and data self-diagnostics, and failures are indicated. Finally, the switches are connected to a cyber security monitoring system that provides alarms for intrusion detection (or more likely, intrusion attempts) even if the switches and the network remain unaffected. Additionally, each switch monitors, controls, rejects, and reports unexpected and excessive (“data storm”) traffic on its respective network segment.

The SC3 nuclear and BOP segments of the DCIS work together as a single network, but independently of one another if failures occur. The networks are redundant and segmented to support the I&C systems with high reliability. The loss of one switch per segment has no effect on plant operation or data. Such failures are indicated and are repairable online. In the highly unlikely event of both switches of a SC3 nuclear or BOP segment fail simultaneously, that particular segment is lost; however, the remaining segments are unaffected and individual nodes connected to the failed switches can continue to function. The remaining switches then automatically reconfigure their uplink ports such that the remaining segments automatically find available data paths between each other. No switch or network failure can adversely affect Safety Category 1 functions because the SC1 equipment is isolated from these networks by unidirectional boundary devices. The BWRX-300 I&C networks are highly fault tolerant such that no single network failure can adversely affect plant operation.

7.2.4 Plant Data Highway Network

Figure 7.2-1 shows the Plant Data Highway (PDH). The communications on the SC3 nuclear and BOP segment UDHs are rigorously controlled and monitored. The managed switches control the information sent to the PDH. The managed switches are the PDH. The PDH is used for important, but non-essential and non-control services like printers, plotters, long-term data storage (the current requirement is to keep plant data for at least two years). Information is sent outside the plant network through the unidirectional boundary devices at a nominal data rate of once per second through the unidirectional boundary device to the server. Various lower cyber security levels of users are connected to the server, including the on-site or off-site emergency response facilities and utility business or engineering networks that may have their own cyber security restrictions. Only the server can respond to requests for information from outside the plant network, and the only information it has is the data sent to it through the unidirectional boundary devices. No plant DCIS component receives or is able to respond to a data request from outside the plant network. No component outside the plant network can “reach” the PDH. The managed switches do not allow any such communication to reach the PDH, SC3, or SCN UDHs. The PDH is characterized as nondeterministic and is not designed to be deterministic; instead, the PDH is

designed to function as a communication channel that is reliable and responsive for displaying process information.

Access to plant cyber essential assets is appropriately protected from external or business networks by unidirectional boundary devices.

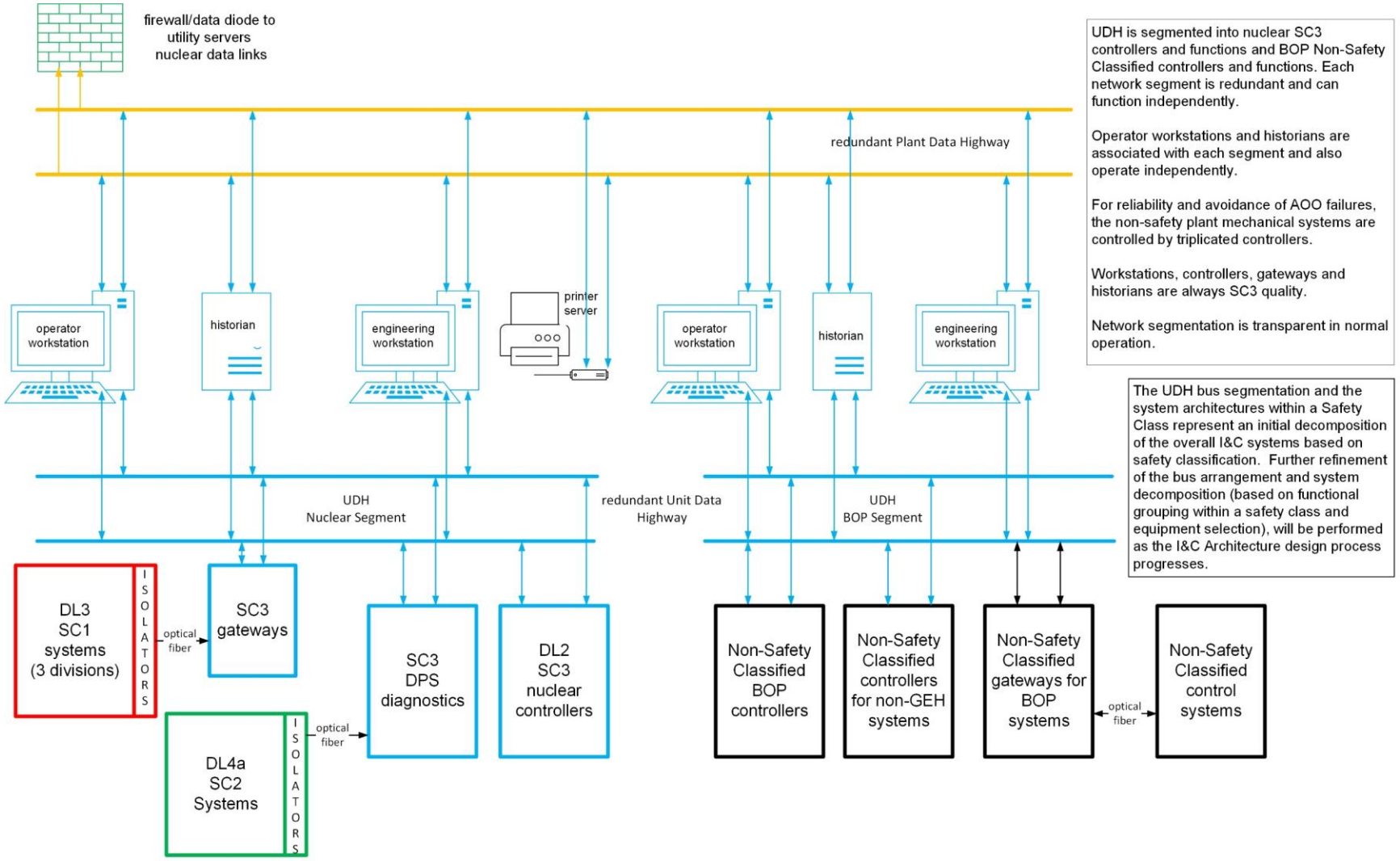


Figure 7.2-1: BWRX-300 Distributed Control and Information System Network Architecture

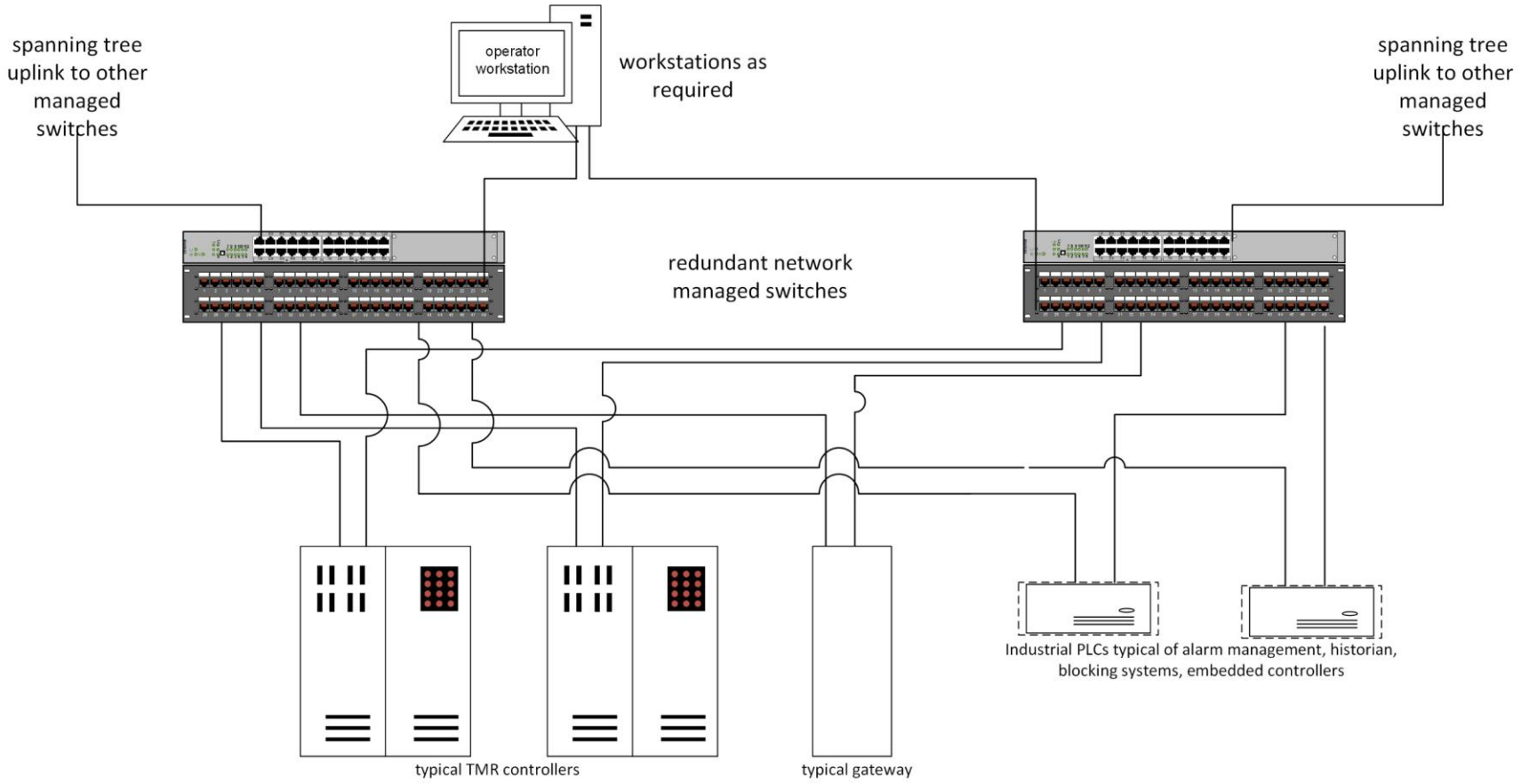


Figure 7.2-2: Typical Instrumentation and Control Controller and Component Network Connections

Redundant network UDH nuclear and BOP segments and PDH are actually pairs of network managed switches.

Most equipment and all controllers, operator workstations and historians are dual ported, one port to each switch.

Any redundant switch can be lost without affecting plant operation.

Any segment can be lost without affecting the other segments.

Managed switch uplinks (inter-switch communication) normally allow any segment controller, historian or workstation to send/receive data to any other.

Dark black links indicate normal switch communication, the lighter black links represent possible connections should ports fail or be shut down for self diagnostic monitoring reasons.

Managed switches control which nodes are allowed to communicate and can prevent any unauthorized internal or external communication.

Managed switches have dual power supplies.

Conceptual Cyber Security monitoring connections shown. Other cyber security configurations are possible.

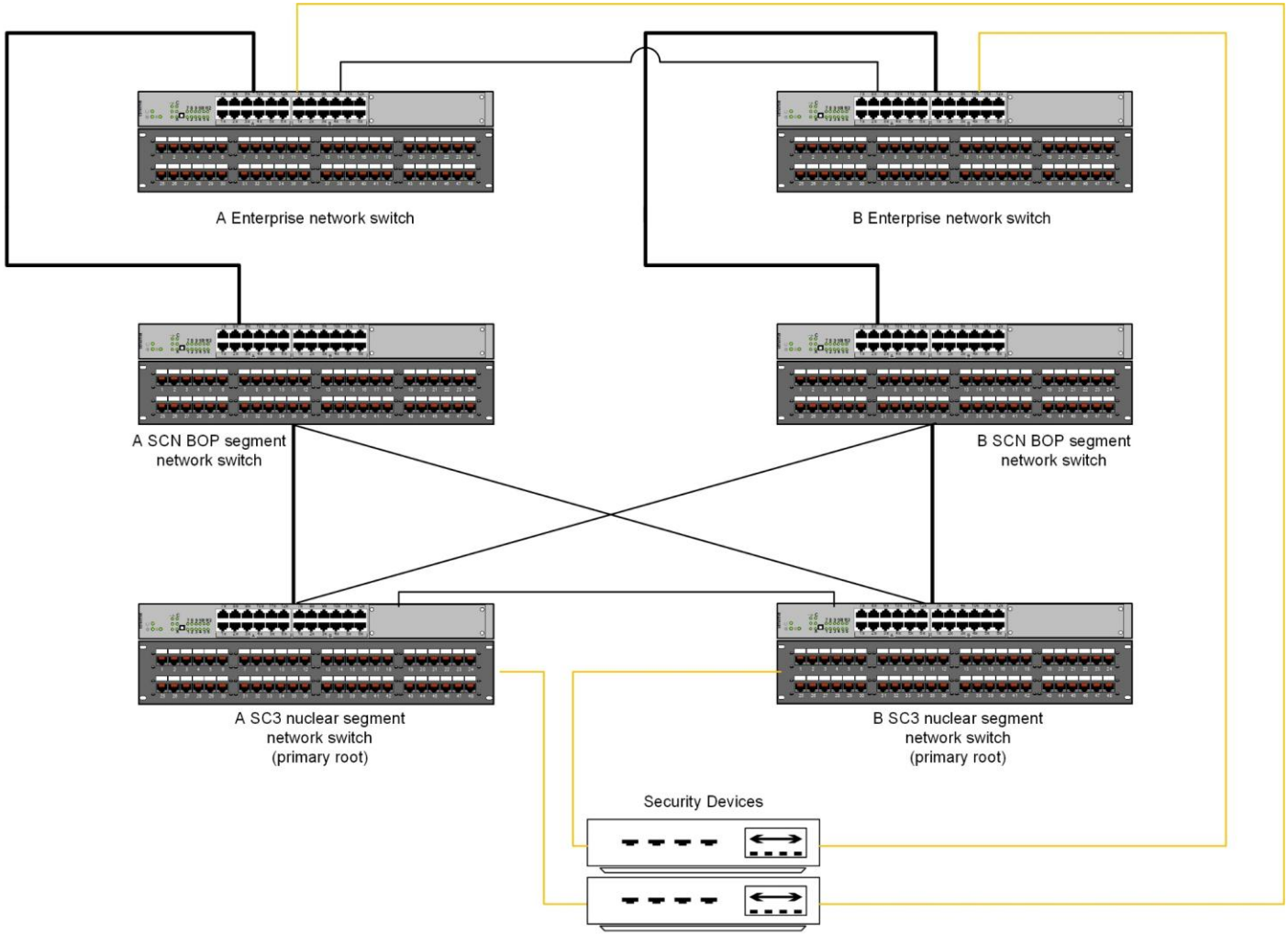


Figure 7.2-3: Rapid Spanning Tree Network Managed Switches

7.3 Distributed Control and Information System Functions

The Reactor Mode Switch is used as an input to the DCIS to insert or remove operating bypasses for the actuation of Safety Category 1, 2, and 3 functions. The Reactor Mode Switch is classified as a SC1 component, consistent with the highest Safety Category of the functions it serves. The Reactor Mode Switch is a four-position switch with the following positions:

- RUN
- STARTUP
- SHUTDOWN
- REFUEL

The Reactor Mode Switch is manually positioned by the reactor operator as the reactor is transitioned from cold (refueling outage) to rated power. The switch position determines and provides information to the various controllers needing the Reactor Mode Switch position information to provide the correct operating bypasses for system operation. As an example, the Safety Category 1 functional logic requires the Reactor Mode Switch position to be in STARTUP to enable a scram at ~15% power and in the RUN position to enable the high-power flux scram. Safety Category 3 functional logic requires the Reactor Mode Switch to be in STARTUP to enable the WRNM short period anticipatory scram and to disable the short period anticipatory scram in the RUN mode. The Reactor Mode Switch is located in the MCR to support normal plant operation. The Reactor Mode Switch is not needed to support any of the functions performed in the SCR.

The Reactor Mode Switch provides the necessary electrical isolation between each segment to maintain divisional independence or isolation between different SCs. The Reactor Mode Switch position information is sent to the following functions:

- SC1 Protection System (Three Divisions)
- SC2 DPS
- SC3 Nuclear Controllers (including the Anticipatory Trip System)
- SC3 Automatic Power Regulator (APR)
- SC3 Rod Control and Information System (RC&IS)

A block diagram of the Reactor Mode Switch is shown in Figure 7.3-1, which illustrates signals sent from the switch and the physical separation of the rotary switch segments.

Each of the three SC1 divisions performing Safety Category 1 functions receive the Reactor Mode Switch information, and each of the systems performing Safety Category 2 and Safety Category 3 functions receive the Reactor Mode Switch information three times to allow for validation. A contact is closed in the selected switch position to provide “continuity” to or from the receiver. The other three (not selected) received switch positions have no continuity. From the point of view of the receiver there are always three positions with no continuity and one position with continuity. The receiver logic can determine a switch position, determine if more than one position is selected (e.g., broken switch), or if no positions are selected (e.g., broken switch/loss of power).

The receiving system logic determines the resolution of “broken” switch positions. For example, the Safety Category 1 fail-safe safety functions trip for multiple switch positions or no switch positions. Fail-as-is systems alarm and stop their processes. The APR takes the plant out of automation and the RC&IS blocks control rod motion. An alarm is provided for inconsistent switch positions.

7.3.1 Primary Protection System – Defense Line 3/Safety Class 1

The SC1 system acquires the plant information from the sensors specified for systems performing Safety Category 1 functions. The specific list of parameters to be measured, the physical locations of the sensors, and environmental qualification envelope for the equipment are specified as part of the detailed design process. The plant information acquired is displayed on the associated VDUs and provided through qualified isolation devices to the SC3 integrated plant displays developed through the Human Factors Engineering (HFE) process as described in Appendix 7A.

7.3.1.1 Systems Architecture

The SC1 I&C systems have three independent divisions that are powered by three separate divisions of the SC1 electrical system. Each of the three divisions uses two Uninterruptible Power Supply (UPS) that are backed by a 72-hour battery. The UPS and the battery chargers can be powered by off-site power or either of the standby diesel generators. The SC1 I&C system automatic actuation functions are designed to be fail-safe, (except for ICS isolation). Fail-safe, as defined for the BWRX-300 design, is the ability of a system or component to move to a predetermined state when a failure occurs that ensures no harm comes to equipment, people, or the environment. If power to the SC1 signal acquisition, trip determination, or actuation devices is lost, sensor failure and sensor communication is lost or voting communication is lost the reactor scrams, and the ICS is automatically initiated to remove decay heat. The automatic and manual controls for the ICS RIVs are fail-as-is.

Each SC1 set of DCIS cabinets, each SC1 UPS, battery chargers, and each SC1 battery are located in a separate, fire barrier rooms in the RB. The instrument racks that mount the signal transmitters that provide the SC1 measured trip, isolation, and initiation signals are located in separate fire barrier rooms in the RB.

The main functions of Safety Category 1 functional logic are the initiation of the scram, reactor and containment isolation, and ICS initiation functions. The necessary Safety Category 1 function parameters are input to the three independent divisions of the SC1 I&C system. The SC1 I&C system (per division) provides signal conditioning and uninterruptible power for SC1 sensors. The SC1 signals are hardwired per division to the SC1 cabinets, and the cabinet outputs are hardwired to their actuators. Remote data acquisition multiplexers are not required because the SC1 I&C system and SC1 signals are mostly located in the RB. The functional architecture for the SC1 I&C system is shown in Figure 7.3-2.

The sensor trip logic is similar for each function:

1. Each division measures a parameter using its own sensors.
2. Each division compares the measured parameter to a setpoint.
3. Each division determines if the measured parameter exceeds the setpoint.

The three divisions communicate (i.e., vote) among themselves to determine if there have been sensor trips in Two-out-of-Three (2oo3) divisions. Each division independently makes the 2oo3 trip decision. The three-division sensor and logic designs are shown in Figure 7.3-3.

The data acquisition functions are implemented on SC1 I&C hardware and software platforms in the SC1 cabinets with a common backbone to the SC1 communications function. The backbone ensures that SC1 I&C platforms in a division have access to the SC1 signals in that division and the communication function ensures that divisional data are isolated and sent through a unidirectional boundary device to the SC3 gateways.

As an example, a reactor pressure signal may be used for both the Safety Category 1 hydraulic scram and ICS safety function initiation. Any SC1 I&C actuation signal within a division is available to any trip logic. Any signal within a division is available for display and alarm on the associated VDUs. Any SC1 I&C signal is available via an isolated link through unidirectional boundary device to the SC3 systems.

The voting communication between divisions contains message authentication and divisional or parameter trip status, and if there is a loss of communication from a division, the receiving division assumes a trip.

The divisions have additional logic to support monitoring and can include alarms for predetermined parameters on the associated VDUs. This design supports accident monitoring using signals acquired by SC1 equipment and the corresponding divisional support systems like SC1 electrical supply status. The SC1 divisions export plant process, accident monitoring, and diagnostic data via dedicated, unidirectional boundary devices to the SC3 systems.

Reactor scram, reactor and containment isolation, and ICS initiation can also be initiated without software via hard switches in either plant control room.

7.3.1.2 System Design Bases and Associated Safety Functions

The design bases of the SC1 I&C systems are to mitigate the effects of a PIE (i.e., AOO or DBA and most DECs) assuming no credit for Safety Category 2 or Safety Category 3 functions (e.g., CCF). The main functions of the SC1 I&C logic are the initiation of the following Safety Category 1 functions for reactor scram, reactor and containment isolation, and ICS initiation functions. The SC1 I&C system design includes provision of instrumentation to monitor plant variables and systems over the respective ranges for operational states and PIE in order to ensure that adequate information can be obtained on plant status. Safety Category 1 functions operating alone ensure that no AOO or DBA causes a radiation release greater than regulatory requirements. Safety Category 1 functions operate independently of Safety Category 3 functions using diverse equipment to perform required Safety Category 1 functions. The supporting safety analyses are described in Chapter 15.

SC1 I&C system is designed to be available during all modes of plant operation. Specific safety functions actions are mode dependent, as determined by the Fault Evaluation process described in Chapter 15 and specified by the Mode Switch position shown for each function in Table 7.3-1.

The DL3 functions categorized as Safety Category 1 and implemented in SC1 I&C equipment are:

1. Hydraulic scram – SC1 I&C initiates a trip when a measured parameter exceeds a predefined setpoint and initiates a hydraulic scram with a 2oo3 voting logic using the same parameter. The trip logic is used to scram the reactor by hydraulically inserting the control rod blades. HCUs are pressurized accumulators used to forcibly drive the control rod blade into the core. Scram initiation is enabled by de-energizing two scram solenoid valves to vent air holding the HCU scram valve closed against a spring force. When the HCU scram valve is opened, two control rod blades are inserted into the core except for one HCU that inserts a single control rod blade. The Safety Category 1 hydraulic scram functions and initiation signals are listed in Table 7.3-1.

   The HCUs are arranged in four spatial groups (three with seven HCUs and one with eight HCUs). 29 HCUs are used to drive the 57 control blades. The SC1 hydraulic scram uses divisional load drivers to de-energize the HCU scram solenoid valves.

   Simultaneous with the initiation of the Safety Category 1 hydraulic scram, a scram follow signal is sent to SC2 equipment to have each FMCRD motor drive its ball nut upward to a

position just below the fully inserted and latched hollow piston that is coupled to the control rod blade. This signal is sent whenever a hydraulic scram is demanded by SC1 I&C regardless of whether the hydraulic scram actuation has been successful.

More information on the Control Rod Drive (CRD) System is found in Section 4.6.

2. Power Range Neutron Monitoring (PRNM) – SC1 I&C acquires and processes the PRNM neutron monitoring power signals to support the Safety Category 1 hydraulic scram function. The PRNM acquires signals from 13 vertical "strings" of LPRMs, and Gamma Thermometers (GTs) arranged radially throughout the BWRX-300 core. Each string has four LPRM detectors and eight GTs distributed vertically over the length of the wet tube rod assembly. The LPRMs signals are averaged per division to produce an APRM signal. The averaged signal is normalized to thermal power in percent and used for high-power APRM flux trips.

   PRNM also provides the high Simulated Thermal Power (STP) signal. The APRM signal is put through a (typical) six second time constant to relate it to the heat flux (as opposed to neutron flux) through the fuel to produce the STP signal (also in percent thermal power). Because there are transients where neutron flux can spike but thermal flux does not, the STP trip setpoints for Safety Category 1 hydraulic scram function are set lower than the APRM (neutron) flux signal.

   The PRNM also provides signals for accident monitoring purposes to the Three-Dimensional (3D) Core Thermal Power Distribution Monitor, and to the SC3 control rod blocking systems. These additional functions are not required to support any Safety Category 1 function; however, SC1 I&C is used to acquire the sensor inputs.

   More information on core monitoring is found in Subsection 4.4.6.

3. Reactor, Containment, and System Isolation – SC1 I&C performs the Safety Category 1 functions for RPV and containment isolation by closing the Main Steam Reactor Isolation Valves (MSRIVs), Main Steam Containment Isolation Valves (MSCIVs), Feedwater Reactor Isolation Valves (FWRIVs), Feedwater Containment Isolation Valves (FWCIVs), and other valves identified in Subsection 6.2.4 to limit line break effects both inside and outside containment. The isolation functions and initiation signals are listed in Table 7.3-1. The different isolations are initiated when 2oo3 divisions agree on an isolation demand.

   More information on the different isolation valves is found in Section 5.4, and Subsection 6.2.4

4. ICS Actuation – SC1 I&C performs the Safety Category 1 function to initiate ICS, which is comprised of three Isolation Condensers (ICs). The three BWRX-300 ICs are each a simple loop including the vessel steam supply, piping to the heat exchangers submerged in the IC pools, and a line to return the condensed steam to the vessel. The ICS is initiated by opening a normally closed condensate return. The SC1 I&C logic activates each IC separately using 2oo3 voting logic. When initiated by high reactor pressure, a different initiation setpoint is used for each IC. An IC is initiated by opening a normally closed condensate return valve per IC. When the 2oo3 voting logic is satisfied, a pair of load drivers de-energize the solenoids that are being used to hold the fail-open valve closed. The ICS functions and initiation signals are listed in Table 7.3-1.

   More information on the ICS is found in Section 6.3.

5. Diverse ICS Isolation – The Safety Category 1 function is implemented in a separate and diverse SC1 platform for one train of ICS Isolation. The diverse platform utilizes redundant logic such that a single failure does not cause the isolation of the IC it controls.

More information on the ICS Isolation is found in Section 6.3.

6. Accident Monitoring – SC1 I&C acquires accident monitoring information associated with Type B and C accident monitoring variables (and Type A variables if later determined to be needed), as defined in IEC 63147/IEEE Std 497, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations,” (Reference 7.3-20). Type A variables provide the primary information required to permit the operating staff to take specific required manual actions for which no automatic control is provided. The BWRX-300 does not have any Type A variables, as noted in Table 7.3-2. Type B variables provide the primary information to the accident management personnel to assess the plant safety functions. These safety functions include accomplishing or maintaining reactivity control, core cooling, Reactor Coolant System integrity, and containment integrity (including radioactive effluent control). Type C variables provide the primary information to the accident management personnel to indicate the potential for breach or the actual breach of fission product barriers (e.g., fuel cladding, Reactor Coolant System pressure boundary, and containment pressure boundary). Type B and C variables are categorized as Safety Category 3.
7. Communications – Used to acquire the process, diagnostic, and monitoring signals within the divisional SC1 I&C components and organize them into a predefined message protocol. The divisional communication messaging is then isolated using SC1 hardware and converted into two optical fiber message streams per division. The three redundant pairs of divisional optical fiber message streams are sent through unidirectional boundary devices to two SC3 gateways and then to the SC3 networks. These gateways are redundant such that if one fails, the three divisions of SC1 I&C signals are still available for alarming, recording, and monitoring in the lesser classified I&C systems.

The setpoints for SC1 I&C actuation logic are determined with the setpoint methodology defined in IEC 61888, “Nuclear Power Plants – Instrumentation Important to Safety – Determination and Maintenance of Trip Setpoints,” (Reference 7.3-21) using the final analytical limits from the plant safety analyses and the measurement uncertainties associated with the SC1 I&C equipment (e.g., sensors and processing units). The design bases of Safety Category 1 functions are to act when Safety Category 3 functions are not effective at intercepting a transient or when an event is beyond the capabilities of the Safety Category 3 functions. If Safety Category 1 functions completely fail, Safety Category 2 functions are designed to prevent core damage. As such, for common physical parameters measured by the DLs, the Safety Category 3 function setpoint needs to be designed to act first, because the SC3 design bases are to mitigate/prevent SC1 from needing to respond. Safety Category 1 and 2 functions have common analytical limits to simplify transient and accident analyses and application of the formal setpoint methodology. The use of a common analytical basis has no adverse effect on plant operations or safety.

The BWRX-300 does not use interlocks to prevent overpressurization of low-pressure systems, interlocks to prevent overpressurization of the Reactor Coolant System during low temperature conditions, interlocks to isolate SC systems from SCN systems, and interlocks to preclude inadvertent interconnections between redundant or diverse safety systems for the purposes of testing or maintenance. No interlocks are required because the BWRX-300 design minimizes piping systems, major system components (e.g., pumps and valves), and subsystems connected to the RCPB and connected systems are designed to full reactor coolant pressure, as described in Section 5.4.

No Safety Category 1 I&C function for control room habitability has been identified at this stage of the design progression. More information on the Control Room Habitability System is found in Section 6.4, and Subsections 9.4.1 and 9.4.6.

7.3.1.3 Fundamental Design Properties in the System Design

The SC1 I&C system design includes DL1 properties that represent the quality measures implemented to minimize the potential for failures to occur by the use of conservatism in design and analyses. These DL1 properties of qualification, reliability, robustness, security, diversity, and other D-in-D features are discussed in the following five subsections.

7.3.1.3.1 Equipment Qualification

The SC1 I&C equipment is designed to operate in the environment that is to be expected during both normal operations and anticipated off-normal conditions. The SC1 I&C equipment is qualified to perform its intended functions. Qualification addresses both hardware and software aspects of the SC1 I&C system. The SC1 I&C design and manufacturing processes are of sufficient quality to ensure I&C systems can reliably perform their credited protection functions. The SC1 I&C system equipment is designed, developed, fabricated, and tested to quality standards commensurate with the safety significance of the functions to be performed. The SC1 I&C qualification measures confirm that the I&C systems and equipment are capable of reliably performing the design bases functions for which they are credited over the range of environmental conditions postulated for the area in which they are located.

The SC1 I&C hardware platform is developed in accordance with IEC 60987, “Nuclear Power Plants – Instrumentation and Control Important to Safety – Hardware Requirements,” (Reference 7.3-22) and meets IEC 61513, “Nuclear Power Plants – Instrumentation and Control Important to Safety – General Requirements for Systems,” (Reference 7.3-23) requirements and the equipment standards defined in Section 7 in IEC 61226, “Nuclear Power Plants – Instrumentation, Control and Electrical Power Systems Important to Safety – Categorization of Functions and Classification of Systems,” (Reference 7.3-24) for Class 1 systems.

SC1 software used to perform Safety Category 1 functions is developed in accordance with IEC 60880, Nuclear Power Plants – Instrumentation and Control Systems Important to Safety – Software Aspects for Computer-Based Systems Performing Category A Functions,” (Reference 7.3-25). SC1 hardware used to perform its Safety Category 1 functions is qualified for environmental conditions in accordance with IEC/IEEE 60780-323, “Nuclear Facilities – Electrical Equipment Important to Safety – Qualification,” (Reference 7.3-26) and seismic interaction in accordance with IEC/IEE 60980-344, “Nuclear Facilities – Equipment Important to Safety – Seismic Qualification,” (Reference 7.3-27).

The SC1 I&C equipment is qualified for electromagnetic compatibility in accordance with IEC 61000-4 Series “Electromagnetic Compatibility,” (References 7.3-1 through 7.3-17) and IEC 61000-6-2, “Electromagnetic Compatibility (EMC) – Part 6-2: Generic Standards – Immunity Standard for Industrial Environments,” (Reference 7.3-29).

7.3.1.3.2 Reliability

The SC1 I&C systems have the required reliability to perform its intended functions. It has an initial quantitative reliability target of less than 1E-4 probability of failure on demand. These reliability targets are achievable based on GVHs experience with similar Nuclear Power Plant (NPP) protection systems. The reliability of the SC1 I&C system is demonstrated as part of the final design.

The reliability analysis of the SC1 I&C systems uses qualitative and quantitative performance measures or criteria, as appropriate, to demonstrate reliability goals. The reliability assessment is used to optimize goals such as minimizing out-of-service time for repair and reducing the frequency of surveillance testing.

The SC1 I&C systems use digital technology and different Safety Category 1 I&C functions may be combined on a common platform. It uses three divisions with 2oo3 voting logic for trips and initiations. Spurious actuations due to single hardware failures are prevented.

The SC1 I&C design allows one division of sensors to be bypassed or one division of logic to be bypassed to support maintenance or testing. The sensor and logic bypass switches are used to support these activities. It is physically possible to bypass only one division of sensors or one division of logic at a time, which retains the capability to provide required safety actuations. When a bypass is used, the 2oo3 logic behaves as the equivalent of Two-out-of-Two (2oo2) to trip, isolate, or initiate. It should be noted that the design of the self-diagnostics puts a channel in trip when a critical fault is detected, which preserves the trip capability for the function. The three-division redundancy scheme is acceptable based on high reliability of the SC1 system, administrative controls that limit the time allowed for removal from service for maintenance bypass, and the availability of SC2 DPS as a backstop to ensure that the overall plant safety goals are satisfied.

For the BWRX-300 design, no manual online testing of SC1 equipment is required because on-line testing and self-diagnostics features continuously check the system during operation. The need for online maintenance is expected to be infrequent because of the high reliability requirements for the SC1 system. The time allowance for online maintenance is controlled by plant TS. During this period, the SC2 DPS provides an additional layer of protection for any additional problems affecting the SC1 system during the maintenance activity.

The use of measuring and test equipment that can impair SC1 I&C equipment requires deliberate manual intervention via hardware interlock features at the system interfaces.

The individual LPRMs can be bypassed to allow the updated sensor calibrated values to be periodically input to the logic as required by plant TS. APRM signals are periodically calibrated to current core thermal power. The LPRM averaging logic that is used to calculate the APRM value monitors the number of bypassed LPRMs going into the average. If the number drops below a predefined amount, the average is declared invalid, and the downstream logic assumes an APRM trip for that division.

Correct functioning of the SC1 digital I&C platforms is continuously monitored by self-testing features rather than by manual surveillance testing. Critical faults detected by the self-testing features trip the outputs of the platform and other faults are alarmed. The SC1 system is designed to support required surveillance testing without affecting plant operation. It provides signal and division bypass capabilities to support maintenance and testing. Bypass conditions are alarmed in the MCR. The SC1 I&C is designed to support periodic testing of the entire division of instrumentation logic, from sensing device to actuating device.

The SC1 I&C platforms and diagnostics and their communication to the SC3 nuclear segment are designed to support online and automatic surveillance testing that meets TS requirements. Testing and performance assessment requirements are developed using the guidance of IEC 60671, “Nuclear Power Plants – Instrumentation and Control Systems Important to Safety – Surveillance Testing,” (Reference 7.3-28) and IEC 62385, “Nuclear Power Plants – Instrumentation and Control Important to Safety – Methods for Assessing the Performance of Safety System Instrument Channels,” (Reference 7.3-29). Redundant divisions are monitored for deviations by an automated function that is performed continuously in the TS Monitor described in Subsection 7.3.3. Deviations greater than a specified value are alarmed.

For each pair of HCU scram solenoids, one scram solenoid uses a load driver from Division 1 and the other load driver from Division 2. Division 1 and Division 2 load drivers open on a 2oo3 vote from the three divisions communicated via optical fiber. Both scram solenoids in the HCU are

de-energized to initiate a scram. The HCU scram design allows surveillance testing of each division through to the solenoid without causing a scram in normal operation. Testing provisions that are permanently connected to safety systems are classified the same as the safety system. The arrangement of the scram solenoid control and power is shown as Figure 7.3-4.

Although Figure 7.3-4 indicates a single load driver in Division 1 and in Division 2, in fact there are multiple load drivers assigned to the scram solenoids in HCUs located in four separate rooms in the RB. In addition, the reactor can also be shut down with the complete failure of Safety Category 1 hydraulic scram function with the diverse backup of Safety Category 2 functions. The cables from the load drivers to the four HCU rooms are in individual grounded conduits to preclude the possibility of a hot short keeping the scram solenoids incorrectly energized. Specifically, the circuit shown in Figure 7.3-4 is replicated per Division 1 and Division 2 based on the assignment to HCU scram solenoids.

The reactor and containment isolation valves are closed by de-energizing two isolation valve solenoids mounted on the valve actuators. One of the isolation solenoids uses a load driver and uninterruptible power from Division 1 and the other solenoid is powered from Division 2. Division 1 and Division 2 load drivers open on a 2oo3 vote from the three divisions communicated via optical fiber. Both isolation solenoids per isolation valve are de-energized to initiate a valve closure. This valve design allows surveillance testing of each division through to the solenoid without causing an isolation in normal operation. The arrangement of the isolation solenoid control and power is shown as Figure 7.3-5.

The ICS actuation circuits have two normally energized solenoids on the IC open or close valve that keep the valve normally closed. One of the valve solenoids is in Division 1 and the other valve solenoid is in Division 2. Each solenoid has a normally closed load driver that opens on a 2oo3 vote from the three divisions communicated via optical fiber. Both solenoids per actuation valve are de-energized to initiate a valve opening. This valve solenoid allows surveillance testing of each division through to the solenoid without causing ICS initiation in normal operation.

Each IC isolation valve is designed with three divisional solenoids, and 2oo3 solenoids need to change state for the valve to open or close. Reliable isolation and prevention of inadvertent isolation is achieved by the assignment of the SC1 I&C divisions to solenoid valves that control each IC isolation valve and the mechanical arrangement of the solenoid valves to ensure that a single failure does not result in the unwanted isolation of an IC or prevent a required isolation of an IC. Manual actuation of the IC isolation function can be performed on a train-by-train basis. This manual isolation is a Safety Category 2 function, however it is implemented via SC1 equipment, because no lower classified equipment can have the ability to isolate an IC train (which potentially prevents SC1 actuation of that train).

The SC1 I&C system uses dual power supplies and dual power feeds per chassis to increase system reliability and availability.

7.3.1.3.3 Robustness

Robustness of the SC1 I&C design is the degree to which it can function correctly in the presence of invalid inputs or stressful environmental conditions. The robust SC1 I&C design reflects the use of design methods and adherence to engineering best practices to ensure its functions are achieved for operational states and accident conditions. The SC1 I&C design requirements address the full range of operating environments associated with normal operation, transient, and accident conditions, as well as foreseeable internal and external hazards.

The SC1 I&C systems have the required separation and independence to perform their intended functions.

The SC1 equipment is located in three separate divisional fire barrier rooms in the RB. The equipment is separated from the SC3 equipment that is located in a separate fire barrier room in the CB.

The SC1 I&C system uses sensors and actuators independent of the SC2 DPS.

The SC1 I&C system provides trip communication between divisions for voting using optical fiber. A unidirectional boundary device provides for isolated optical communications to SC3 systems through the SC3 gateways to prevent communication from lower SC systems from affecting SC1 I&C systems. The SC1 communication protocols meet the data communication requirements for Category A functions in IEC 61500, “Nuclear Power Plants – Instrumentation and Control Systems Important to Safety – Data Communication in Systems Performing Category A Functions,” (Reference 7.3-30). The isolation devices are classified as SC1.

The logic used to rapidly scram the reactor by hydraulically inserting the control rod blades is separate and independent of the normal FMCRD positioning controls using the RC&IS.

Each LPRM/GT string has four LPRMs (which are miniature fission chambers) equally spaced vertically along the approximate 12-foot height of the core. The LPRM string is located between the corners of the fuel channels of the adjacent four fuel bundles. The locations are determined such that the 3D Core Thermal Power Distribution Monitor can use mirror and rotational symmetry to virtually rotate and reflect the LPRM readings for unmonitored locations with similar surrounding fuel to gain complete core coverage. Figure 7.3-6 indicates the LPRM/GT locations in the BWRX-300 core.

Each LPRM is individually powered and signal conditioned by its associated division. The signals from active LPRMs are averaged to generate APRM and STP values, which are compared to setpoints to determine if a scram is required.

The SC1 I&C physical separation and signal isolation devices meet the requirements of IEC 60709, “Nuclear Power Plants – Instrumentation, Control and Electrical Power Systems Important to Safety – Separation,” (Reference 7.3-31).

The SC1 I&C systems are designed with three divisions with 2oo3 voting logic for trips, isolations, and initiations. This redundancy ensures that the protection functions can be actuated even with an assumed single failure. It is only physically possible to bypass one division of sensors or one division of logic at a time, which retains the capability to provide required safety actuations. When a bypass is used, the 2oo3 logic behaves as the equivalent of 2oo2 to trip, isolate, or initiate. Bypass conditions are alarmed in the MCR. The three-fold redundancy scheme is acceptable based on high reliability of the SC1 equipment performing Safety Category 1 functions, administrative controls limit the time allowed for removal from service for maintenance bypass, and the availability of Safety Category 2 functions as diverse D-in-D to ensure that the overall plant safety goals are satisfied.

The SC1 I&C system has the required fail-safe design features to perform its intended functions. Deliberate operator action is required to return the SC1 I&C system to normal after actuation.

The SC1 hydraulic scram, PRNM, and reactor and containment isolation functions use only fail-safe logic (including loss of sensor and communications data). The SC1 hydraulic scram logic is also equipped with individual manual scram switches in Division 1 and Division 2. These switches interrupt the power to the scram solenoids independently of the SC1 hydraulic scram logic and do not use any software. The switches are included in the SCR, and both switches are required to be operated to cause a scram. A loss of uninterruptible power or logic power causes an SC1 hydraulic scram, ICS actuation, and RPV and containment isolation but does not close the IC RIVs.

The Safety Category 1 PRNM logic provides flux related trip signals to initiate a SC1 hydraulic scram. It also provides power data to the SC3 DCIS for use to determine if a rod block is required (see Subsection 7.3.3) and to computer functions that determine the 3D power distribution of the core. A block diagram of the PRNM is shown as Figure 7.3-7.

The LPRM averaging logic that is used to calculate the APRM value monitors the number of available LPRMs going into the average to ensure it is representative. If the number drops below a predefined amount, the average is declared invalid, and the downstream logic assumes an APRM trip for that division.

The APRM logic receives inputs from the Reactor Mode Switch position. Certain trips based on the APRM signals are enabled with the Reactor Mode Switch in RUN and different Safety Category 1 trip functions are enabled with the Reactor Mode Switch position in STARTUP.

The SC1 ICS actuation logic initiates the three ICs separately (using 2oo3 voting logic) for both reliability reasons and to allow different initiation setpoints for each IC. A loss of power initiates all three trains of the ICS. The condensate return valves are open/close only and are operated by Safety Category 1 functional logic. The logic is similar to the Safety Category 1 hydraulic scram function logic.

The ICS isolations do not use a fail-safe design. Instead, the IC steam supply and IC condensate return isolation valves are designed to fail-as-is to minimize the potential for inadvertent isolation (i.e., the valves require active power to change position). Additional design features are provided to ensure that the ICS provides the required safety functions. The SC1 I&C logic actively energizes the IC isolation valve solenoids to isolate an IC, and each IC is isolated independently and only upon indication of a line break for that IC.

The IC isolation logic works by each division acquiring the isolation parameters. Bypass conditions are alarmed in the MCR. The signals are compared to their setpoints and vote to trip (per division) if the sensed signal exceeds the setpoint. The three divisions communicate their trip decisions, and each division determines if at least two divisions have tripped to initiate an isolation. Unlike fail-safe logic, which treats an invalid input or loss of communication as a vote to trip, fail-as-is logic ignores invalid inputs such that the logic always requires two valid trip votes prior to actuation. If at least two divisions have tripped, an isolation is initiated. If logic or valve solenoid power is lost or the logic fails self-diagnostic monitoring, the isolation valves remain at their last position. From this point the fail-as-is logic differs from the fail-safe logic used for SC1 I&C actuation features.

The divisional logic outputs change state (open or close the isolation valves) only with active power. Logic and sensor diagnostics are alarmed but do not initiate isolation. A loss of power to the division or to the solenoids does not initiate isolation. The isolation is active and latching to fail-as-is. Three solenoids are provided on each isolation valve and energizing any 2oo3 solenoids cause the valve to change position and latch. The three solenoids provide single failure tolerance (e.g., loss of one division of power) because the fail-as-is design requires power from at least two divisions. Reliable isolation, when required, is achieved by requiring 2oo3 solenoids and 2oo3 divisional logics to initiate an isolation. The 2oo3 logic minimizes the potential for inadvertent isolation for random I&C component failures.

The IC isolation controls are placed into two groups to prevent the remote possibility of a CCF in SC1 equipment performing Safety Category 1 functions from causing three ICs to be spuriously isolated. One group has the isolation controls for two ICs, and the second group has the isolation controls for the remaining IC. For the first IC group, the main SC1 hardware and software platform performs the IC isolation function. For the second group (one IC) separate and diverse hardware and software perform the isolation function. The diverse hardware and software are internally

redundant such that a single failure does not cause the isolation of the controlled IC The ICS design ensures that even with a postulated CCF of the primary SC1 I&C platform, at least one IC is maintained operational, and one IC is sufficient to mitigate AOOs.

A critical self-diagnostic hardware or software failure initiates a divisional trip for fail-safe logic circuits. Diagnostic failures may be classified as critical (such as the watchdog timer or loss of communication) and cause a trip in that division. Other diagnostics are non-critical (e.g., one of the redundant power supplies fails) and cause an alarm in that division.

No SC1 I&C function is dependent on whether the optical fiber unidirectional boundary devices are connected or disconnected, whether the fibers are severed, the gateways are powered, or the SC3 networks have received the SC1 signals.

A software-free interface is used in SC1 I&C systems to provide hardwired manual scram and isolation capabilities.

7.3.1.3.4 Security

The SC1 I&C systems have features that adequately address access control to limit cyber security vulnerabilities and ensure the system can perform its intended functions. The SC1 I&C design incorporates access control features to support the establishment of a secure operational environment. The SC1 I&C software is produced in a secure development environment that prevents the insertion of undocumented codes. The SC1 I&C systems are installed and maintained in accordance with the station administrative procedures and control of access programs.

The SC1 I&C network is isolated from the rest of the BWRX-300 DCIS. The three SC1 DCIS divisions operate completely independently from the SC2 and SC3 DCIS. Information only flows through unidirectional boundary devices from SC1 I&C to the SC3 networks when the system is in operation. Two-way data flow can be used with appropriate access control features to support LPRM calibration. The SC3 nuclear segment includes the SC2 gateways that receive isolated SC1 signals through a unidirectional boundary device and provide them to the SC3 nuclear segment. The SC2 gateways are discussed in Subsection 7.3.3. Measured parameters, setpoints, logic and trip status, and diagnostic information are continuously sent through these gateways to the SC3 nuclear segment TS Monitor, which can perform online surveillance testing, alarm on discrepancies, and additionally alarm on loss of these expected communications.

7.3.1.3.5 Diversity and Defense-In-Depth

The SC1 I&C system provides the main line of protection for AOOs and DBAs. The SC3 nuclear segment provides anticipatory trip functions to support the BWRX-300 D-in-D strategy. Safety Category 3 functions are designed to prevent or mitigate AOOs before either Safety Category 1 or Safety Category 2 functions are required. The design bases of Safety Category 2 functions provide diversity and D-in-D for the complete failure of the Safety Category 1 function (i.e., CCF). Safety Category 2 functions independently provide comparable Safety Category 1 functions using diverse platforms from those used in SC1 equipment.

Diversity measures are incorporated within the SC1 I&C system design as an additional level of protection for potential systematic faults caused by design and implementation defects within redundant divisions of the system. The IC isolation controls are designed to prevent the remote possibility of a CCF of a Safety Category 1 function causing three ICs to be spuriously isolated. The SC1 I&C systems are designed to be diverse from each other and are internally redundant such that a single failure does not cause the isolation of the controlled IC ICs. These diversity measures ensure that at least one IC remains operational for any CCF affecting the IC isolation features and one IC is enough to mitigate AOOs.

The hydraulic scram is equipped with individual manual scram switches in Division 1 and Division 2 (as shown on Figure 7.3-4). These switches use a software-free method to interrupt the power to the scram solenoids independently of the automatic actuation logic. The switches are included in the SCR, and both switches are required to be operated to cause a scram.

The SC1 ICS initiation logic is equipped with individual switches in Division 1 and Division 2 to open the ICS condensate return line valves. These switches use a software-free method to open the valves. The switches are included in the SCR, and both switches are required to be operated to cause an ICS Initiation.

The SC1 I&C reactor, containment, and system isolation logic is equipped with individual switches in Division 1 and Division 2 to close the MSRIV, MSCIV, FWRIV, and FWCIV. These switches use a software free method to close the valves. The switches are included in the SCR, and both switches are required to be operated to close these valves.

SC1 manual switches are provided in the SCR to open and close ICS RIVs.

7.3.1.4 Operator Interface and Accident Monitoring

Data from the SC1 I&C systems is available on appropriate displays and these displays are subject to an HFE evaluation to maximize operator responsiveness. Redundant VDUs are located in both the MCR and SCR and locally in the SC1 I&C equipment rooms. The displays also include an alarm system to provide operator awareness and to prompt to the displays containing further information relating to the alarm. SC1 I&C system bypass conditions are alarmed in the MCR. The displays are menu-driven, and it is possible to switch between related displays without going through the main menu.

A reactor scram, ICS initiation and reactor, containment, and system isolation can be manually initiated by switches in the MCR and switches in the SCR. Manual actuation of the IC isolation function can be performed on a train-by-train basis.

Some signals monitored by the SC1 I&C systems provide necessary information to achieve or verify correct plant response to transients and accidents. The necessary signals are determined through HFE analysis and applicable regulatory requirements and are referred to as accident monitoring signals.

The signals monitored by the SC1 I&C systems are recorded, alarmed, and displayed to the operator on an appropriate display (e.g., range check problems or alarm setpoints exceeded). The SC1 I&C signals and the various SC1 I&C components are alarmed for self-diagnostics.

The monitored parameters required to be available for 72-hours are supported by the SC1 UPS and batteries with no off-site power available.

Data from the SC1 I&C system is also transmitted through a unidirectional boundary device to SC3 DCIS and SC3 equipment qualified for Safety Category 3 accident monitoring functions, where the signals are recorded, alarmed, and displayed. The associated monitoring also includes signals needed to monitor divisional support equipment like the UPS and battery chargers, as well as the SC1 equipment room temperatures. The operator can view the information for the SC1 I&C system on the associated VDUs even if the SC3 DCIS is not operational.

The SC1 I&C system acquires and displays the accident monitoring information associated with Type B and C accident monitoring variables.

The BWRX-300 plant level safety diversity and D-in-D design has a direct bearing on the design of the accident monitoring instrumentation. The BWRX-300 passive reactor design automatically actuates Safety Category 1 functions that rely solely on natural passive mechanisms based on fundamental physical and thermodynamic principles to mitigate a design basis event (i.e., AOOs

or postulated accidents). The BWRX-300 does not require manual actions to mitigate design basis events. Automatic controls perform mitigation functions. As such, no Type A accident monitoring instruments are required. The simplicity of the BWRX-300 safety features to mitigate design basis events and safely shut the plant down, when coupled with the BWRX-300 D-in-D framework, ensures the capability to mitigate the effects of AOOs and DBAs even with a coincident loss of a complete DL.

The simplicity, diversity, and D-in-D framework change the role of the Type B and C accident monitoring instruments for the BWRX-300 accident management strategy. For the BWRX-300, these instruments are only used for immediate verification that DL safety category functions have actuated. They are not needed for subsequent operator actions to realign or control more complicated engineered safety features after the initial actuation phase. The multiple DLs and their capability to accommodate the loss of a complete DL eliminate the need for more immediate operator actions to respond to equipment failures for successful accident management. The specific accident monitoring variables that the operator should monitor to ensure safety during an accident and the subsequent long-term stable shutdown phase are determined through the normal BWRX-300 design process using the HFE process. The Type B and C accident monitoring information is displayed in both the MCR located in the CB and the SCR located in the RB. The accident monitoring information acquired by the SC1 I&C system is sent through qualified isolation devices to the SC3 DCIS network.

A preliminary list of the Type B and C accident monitoring variables is provided in Table 7.3-2.

The operator interface HSI for the SC1 protection system is described in Sections 7.5 and 7.6 for the MCR and SCR, respectively.

7.3.1.5 Compliance Alignment

Table 7.3-5 shows how the SC1 I&C system aligns to meet the regulatory guidance in USNRC DRG: “Instrumentation and Controls for Non-Light-Water Reactor Non-LWR Reviews,” (Reference 7.3-19) and the key industry I&C standards IEC 61513 (Reference 7.3-23) and IEEE Std 603-2018, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” (Reference 7.3-32).

7.3.2 Diverse Protection System – Defense Line 4a/Safety Class 2

The SC2 system acquires the plant information from the sensors specified for systems performing Safety Category 2 functions. The specific list of parameters to be measured, the physical locations of the sensors, and environmental qualification envelope for the equipment are specified as part of the detailed design process. The plant information acquired is displayed on the SC3 integrated plant displays developed through the HFE process.

7.3.2.1 Systems Architecture

The SC2 DPS is arranged into a segmented architecture. Three channels of input signals are acquired and compared to setpoints to produce trip or initiating signals when required. These channel trip and initiating signals are then voted using a 2oo3 logic to produce a DPS initiating or trip actuation signal. The 2oo3 logic is used to avoid inadvertent actuations. The functional architecture for the SC2 system is shown in Figure 7.3-8. The splitter arrangement used for shared sensors is shown in Figure 7.3-9. A representative DPS block diagram for a reactor scram is shown in Figure 7.3-10 and should be viewed with Figure 7.3-4, which shows the scram solenoid power configuration.

Protection function signals within the SC2 equipment do not use the plant networks. The RC&IS communicates with the FMCRD motor controllers (which perform a Safety Category 2 function) on a dedicated redundant network. The FMCRD network is not part of the plant DCIS network

and does not go through the network managed switches. This communication is not required for any Safety Category 2 function, nor could it interfere with any Safety Category 2 function.

7.3.2.2 System Design Bases and Associated Safety Functions

The design bases for Safety Category 2 functions assume the complete failure of Safety Category 1 functions (i.e., CCF). Safety Category 2 functions are implemented in SC2 equipment that is independent and diverse from the SC1 equipment. The SC2 equipment design for systems implementing Safety Category 2 functions include provisions for instrumentation to monitor plant variables and systems over the respective ranges for operational states and PIEs in order to ensure adequate information can be obtained on plant status. The supporting safety analyses are described in Chapter 15.

SC2 equipment is designed to be in service during all modes of plant operation. Specific safety functions are mode dependent, as determined by the Fault Evaluation process in Chapter 15 and the initiating signals are enabled based on the Reactor Mode Switch position shown for each function in Table 7.3-3.

The system functions performed by SC2 equipment are:

1. DPS – Provides the logic for the Safety Category 2 functions. The DPS is implemented on a technology diverse from the SC1 equipment and segmented into three channels. Input signals are measured, sent to comparison logic, and if the signals exceed their setpoints, produce trip or initiation signals per channel. The channel trip and initiation signals are then voted 2oo3 to produce a DPS initiation or trip and to be fault tolerant to avoid inadvertent trips. The DPS functions include:

    a. Hydraulic scram independent of Safety Category 1 functions

    b. RPV, containment and system isolations independent of Safety Category 1 functions

    c. ICS initiation independent of Safety Category 1 functions

    d. FMCRD Fast Motor Run-in

    e. Alternate Rod Insertion (ARI) pilot valve actuation

    f. Feedwater (FW) and Condensate Pumps Trip

    g. Reactor Cavity Pool Makeup from IC Pools

    The hydraulic scram and ARI signals are initiated whenever a Safety Category 2 scram function is demanded regardless of whether it has been successful. The Safety Category 2 fast motor run-in function occurs on any hydraulic scram signal followed by a reactor power level indicating that the hydraulic scram was not successful.

    The DPS functions and initiation signals are listed in Table 7.3-3. The DPS provides the capability for manual initiations using switches located in the MCR. These are separate from the switches associated with SC1 I&C which are located in the SCR.

    The setpoints for SC2 DPS actuation logic are determined with the setpoint methodology defined in IEC 61888 (Reference 7.3-21) using the final analytical limits from the plant safety analyses and the measurement uncertainties associated with the DPS equipment (e.g., sensors and processing units). Safety Category 2 functions are designed to prevent core damage for PIE assuming the complete failure of a Safety Category 1 function. As such, common physical parameters measured by Safety Category 1 and Safety Category 2 functions have common analytical limits to simplify transient and accident

analyses, as well as the application of the formal setpoint methodology. The use of a common analytical basis has no adverse effect on plant operations or safety.

2. Gamma Thermometers – Provide electrical signals from the GTs based on local gamma flux which is representative of core thermal power. The GT represents a completely diverse technology to the neutron detecting LPRMs used by the SC1 system. The DPS acquires signals from 13 vertical LPRM/GT "strings" arranged radially throughout the BWRX-300 core. Each string has four LPRM detectors and eight GTs distributed vertically over the length of the wet tube rod assembly.

   The GT signals are apportioned to four cabinets such that each cabinet gets a similar allocation of radial and axial GT signals.

3. FMCRD Motor Control – There are 57 FMCRDs in the BWRX-300 and each one has an individual motor and an individual motor controller. The motor controllers are allocated to SC2 equipment and interface with DPS, Emergency Rod Insertion Panels (ERIPs), and RC&IS. Each motor controller uses software to supply the appropriate commands to the FMCRD motors using individual position controllers (i.e., position demand and position feedback). Control rod position indication is provided to RC&IS. Normally the FMCRDs are individually (or in gangs) positioned by the RC&IS to control reactor power in response to either operator or automation system insert and withdraw commands. The FMCRD motor controller software is required for the fast motor run-in, but the DPS actuated hydraulic scram is independent from this function and does not rely on this software.

4. Emergency Rod Insertion Panels and Emergency Rod Insertion Control Panels (ERICPs) – There are four SC2 ERIPs. The ERIPs have two functions. The first is to multiply the scram demand signals from the SC3 and SC1 systems, as well as fast motor run-in demand signals from the SC2 DPS to each of the 57 FMCRD motor controllers. The second is to multiply the Multi-Channel Rod Block Monitor (MRBM) rod block demand signals to each of the 57 FMCRD motor controllers. The MRBM rod block stops the rods from being able to withdraw and the motor run-in has priority. The circuit design ensures control rod insertion is always available. The scram follow motor run-in is automatically initiated upon receipt of a valid hydraulic scram demand from any of the interfacing systems.

   These panels are contact multipliers that each receive manual scram signals, three rod motor withdraw power block signals, and three motor run-in signals from the ERICP. The three motor run-in signals and three motor withdraw power block signals are decoded using 2oo3 logic to pick up additional actuation devices, 29 in one panel and 28 in the other panel. The design ensures that a failure of an actuation device can only affect one FMCRD. The motor run-in actuation devices are sealed in once energized long enough to ensure the motors have time to drive the rods full in.

   The ERICP sends the manual scram and motor run-in and motor power block signals to the ERIP. The scram follow motor run-in is automatically initiated whenever Anticipatory Trip System, DPS, or Safety Category 1 hydraulic scram function initiates a hydraulic scram.

   The first two (SC1 hydraulic scram and SC3 anticipatory) hydraulic scram initiation signals are shared with SC2 equipment to generate an FMCRD scram follow motor run-in. Simultaneous with the initiation of the Safety Category 1 and Safety Category 3 anticipatory hydraulic scram functions, the scram follow function drive each FMCRD motor upward, stopping just short of re-engaging the control rod blade. This, in addition to the SC1 internal control rod blade full in rod latches, prevents the control rod blade from

dropping out of the reactor post scram. This also ensures that any rod scram failure (due to a failed HCU component, for instance) is inserted into the reactor within minutes of the scram signal. The Safety Category 2 DPS fast motor run-in is completely independent of the scrams initiated by Safety Category 1 and Safety Category 3 functions and serves as the backup to the complete failure of the hydraulic scram in case of complete failure of the mechanical/hydraulic portions of the CRD system. This function is performed by the ERICP sending signals to DPS to indicate that a hydraulic scram has been initiated from any source. After receiving such signals, the DPS logic expects its GT inputs to show a decreasing power due to a hydraulic scram. If the GT measurements do not decrease sufficiently within a short time, the DPS generates outputs to the ERICP to order a fast motor run-in. The FMCRD motors should have already started in their scram follow capacity and the DPS order to the ERICP causes the motors to increase insertion speed.

A motor power withdraw block signal comes from the MRBM, which stops the rods from being able to withdraw. The circuit design ensures that control rod insertion is always available.

5. FMCRD Motor UPS – These UPS are dedicated to the FMCRD motor controllers and motors and are not part of the SC1 or SC2/SC3 normal DCIS UPS. The FMCRD motors and controllers have four dedicated UPS that can supply the FMCRD motors in the absence of off-site power or the absence of standby diesel generator power.

   The FMCRDs are divided into four groups distributed throughout the core. Each of the four motor and motor controller groups has its own UPS that has a capacity that is at least twice as long as the duration of a motor run-in (i.e., rods driven full in by their motors) even if both off-site power and standby diesel generator power are lost. The four motor groups of FMCRDs are indicated by color code on the upper right of the core map in Figure 7.3-6.

   See Subsection 8.3.2 for additional detail.

7.3.2.3 Fundamental Design Properties in the System Design

The SC2 system includes DL1 properties that represent the quality measures implemented to minimize potential for failures to occur by the use of conservatism in design and analyses. These DL1 properties of qualification, reliability, robustness, security, diversity, and other D-in-D features are discussed in the following five subsections.

7.3.2.3.1 Equipment Qualification

The SC2 equipment is designed to operate in the environment that is to be expected during both normal operations and anticipated off-normal conditions. The SC2 equipment is qualified to perform its intended functions. Qualification addresses both hardware and software aspects of the SC2 equipment. The design and manufacturing processes are of sufficient quality to ensure I&C systems can reliably perform their credited protection functions. The SC2 equipment is designed, developed, fabricated, and tested to quality standards commensurate with the safety significance of the functions to be performed. The SC2 equipment qualification measures confirm the I&C systems and equipment are capable of reliably performing the design bases functions for which they are credited over the range of environmental conditions postulated for the area in which they are located.

The main SC2 hardware platform meets IEC 61513 (Reference 7.3-23)requirements and the SC2 equipment standards defined in Section 7 of IEC 61226 (Reference 7.3-24) requirements. SC2 software is used in the FMCRD motor controllers and is developed in accordance with the Category B requirements in IEC 62138, “Nuclear Power Plants – Instrumentation and Control Systems Important to Safety – Software Aspects for Computer -Based Systems Performing

Category B or C Functions," (Reference 7.3-33). SC2 hardware is developed in accordance with IEC 60987 (Reference 7.3-22). SC2 hardware is qualified for environmental conditions in accordance with IEC/IEEE 60780-323 (Reference 7.3-26) and seismic interaction in accordance with IEC/IEEE 60980-344 (Reference 7.3-27). The SC2 equipment is qualified for electromagnetic compatibility in accordance with the IEC 61000-4 Series (References 7.3-1 through 7.3-17) and IEC 61000-6-2 (Reference 7.3-29).

7.3.2.3.2 Reliability

The SC2 equipment has the required reliability to perform its intended functions. The SC2 equipment has an initial quantitative reliability target of less than 1E-3 probability of failure on demand. The reliability analysis of the SC2 equipment demonstrates it meets its reliability goals using qualitative and quantitative performance measures or criteria, as appropriate. The reliability assessment is used to optimize goals such as minimizing out-of-service time for repair and reducing the frequency of surveillance.

The DPS provides self-diagnostics to SC3 DCIS for equipment status monitoring through qualified isolation features classified as SC2. FMCRD motor controller self-diagnostics and rod separation signals are also used by RC&IS. The SC2 DPS is designed to support required surveillance testing without affecting plant operation. It provides the capability for signal and channel bypass to support maintenance and testing. Bypass conditions are alarmed in the MCR. The SC2 design provides for periodic testing of the entire channel of instrumentation logic, from sensing device to actuating device. Testing and performance assessment requirements for DPS are developed using the guidance of IEC 60671 (Reference 7.3-28) and IEC 62385 (Reference 7.3-29). The use of measuring and test equipment that can impair a Safety Category 2 function requires deliberate manual intervention via hardware interlock features at the system interfaces. Testing provisions permanently connected to safety systems are classified the same as the safety system.

The GT system supports averaging the four groups of GT signals and normalizing the average to core thermal power. The GT system provides the capability for bypassing individual GTs. Bypass conditions are alarmed in the MCR. The GT system provides for the calibration of the GTs. The GT system is designed to support required surveillance testing without affecting plant operation. It also provides the GT system self-diagnostics to the SC3 DCIS for display.

The SC2 DPS, GTs, and ERIPs are redundantly and uninterruptedly powered by the SC2/SC3 electrical system load groups. The FMCRD motor controllers are normally powered from the SC2/SC3 electrical system and have dedicated SC2 UPS. The FMCRD UPS have a capacity that is at least twice as long as the duration of a motor run-in (i.e., rods driven full in by their motors). The DPS, GTs, and ERIP UPS are backed by two load groups of batteries. The UPS and the battery chargers can be powered by off-site power or either of the standby diesel generators.

7.3.2.3.3 Robustness

Robustness of SC2 equipment is the degree to which it can function correctly in the presence of invalid inputs or stressful environmental conditions. The robust SC2 DPS design reflects the use of design methods and adherence to engineering best practices to ensure that the protection functions are achieved for the specified conditions.

The SC2 systems have the required separation and independence to perform their intended functions as a diverse backup to SC1 equipment.

The majority of the SC2 equipment performing Safety Category 2 functions is located in a separate fire barrier room in the CB with the remainder located in compartmentalized fire barrier rooms in the RB. It is separated from SC1 equipment performing Safety Category 1 functions

that is located in three separate divisional fire barrier rooms in the RB. The SC2 equipment is powered by separate power supplies than the SC1 equipment. The four ERIPs are located in the RB near the cabinets housing the FMCRD motor controllers.

The SC2 DPS uses sensors and actuators that are independent of SC1 equipment. Isolated communication is allowed from SC1 equipment to SC2 equipment but not vice versa. This communication cannot prevent either from fulfilling its protection functions. Safety Category 2 functions do not share any sensors with Safety Category 1 functions.

SC2 equipment is independent from SC3 equipment to an extent that is practicable in that shared sensors are not credited with mitigation for the same PIE. Some signals shared between SC2 and SC3 equipment, as identified in Figure 7.3-8, are used by both SC2 and SC3 TMR controllers allocated to each parameter performing Safety Category 2 and Safety Category 3 functions. The splitters for redundant signals are located in separate cabinets. The shared signals are developed using TMR architectures. The shared signals are analog signals. Signal sharing is accomplished using non-software, analog splitters that are powered with redundant uninterruptable power supplies. The splitter technology supplies isolated outputs that allow short circuits or common mode voltages up to 300 volts (Alternating Current (AC) or Direct Current (DC)) applied to one output without affecting the other output. The splitter design is shown in Figure 7.3-9. The splitter design ensures a splitter output is unable to adversely affect the other outputs. These isolation features are classified as SC2.

The three channels of the SC2 DPS receive signals from separate SC2 splitters to be used by the 2oo3 voting logic for trips and initiations.

The FMCRD motor controllers receive normal control signals from RC&IS; however, the motor controller SC2 software instructs the controller to ignore the SC3 RC&IS network commands and run the rods in when instructed by the ERIPs and DPS.

The control rods and motors are located inside containment in the RB. The FMCRD motor controllers are organized into four groups located in four separate rooms in the RB outside containment.

The SC2 FMCRD UPS provides the necessary independent power to operate the motors and monitor system operation. The FMCRD motors and controllers have four dedicated UPS that can supply the FMCRD motors in the absence of off-site power or the absence of standby diesel generator power. The FMCRD UPS can operate from and be charged by those sources. The power sources are also independent of the normal SC2 UPS that power the SC2 and SC3 DCIS. The ERIPs, FMCRD motor controllers, and the UPS to power them are arranged in four independent and separately located groups.

The SC2 physical separation and signal isolation devices meet the requirements of IEC 60709 (Reference 7.3-31).

Where software is used, SC2 equipment diagnostics include watchdog timers, sensor range checks, power supply monitoring, communications, and actuator monitoring. The DPS is monitored digitally through unidirectional isolated signals. The digital monitoring for self-diagnostics cannot adversely affect Safety Category 2 functions. The digital monitoring checks for inconsistencies between redundant signals, performs sensor range checks, and monitors actuator, communication, and power supply status. The ERIP logic is designed to ensure that an actuation device failure can only affect one FMCRD. Motor run-in is initiated when a manual scram is initiated from either control room. The motor run-in actuation devices are sealed in for at least four minutes once energized to ensure the motors have time to drive the rods full in.

The SC2 DPS has the required fail-as-is design features to perform its intended functions and avoid spurious actuations. The fail-as-is, energize to actuate, design is used to prevent lesser classified DPS from creating unnecessary challenges to plant safety (e.g., spurious actuations for expected failures) requiring action by Safety Category 1 functions.

Deliberate operator action is required to return the Safety Category 2 functions to normal after actuation.

SC2 equipment operating by itself (or in combination with Safety Category 3 functions unaffected by the PIE) and assuming a complete failure of a Safety Category 1 function, ensures that no AOO or DBA causes a radiation release greater than regulatory requirements. No SC1 equipment can prevent a scram, reactor isolation, containment isolation, or ICS initiation from SC2 equipment or vice versa.

The SC2 equipment performing Safety Category 2 functions provides diversity to support the BWRX-300 D-in-D strategy.

DPS initiates hydraulic scrams, isolations, and ICS using the same solenoids as Safety Category 1 functions but using actuation devices instead of load drivers. Independence between these safety category functions is discussed below.

For diversity, the SC2 equipment is able to independently shutdown the plant using the FMCRD motors. The SC2 ERIPs accept isolated hydraulic scram demand signals from Anticipatory Trip System Safety Category 1 hydraulic scram and Safety Category 3 anticipatory hydraulic scram functions to also initiate reactor shutdown using the FMCRD motors. The motor run-in inserts rods even if the hydraulic scram does not work or the entire SC1 system fails.

GTs provide a diverse technology to the neutron detecting LPRMs used in the SC1 system. The GT system segment supplies two types of outputs: average power signals and digital signals of individual sensor measurements. The GT signals are sent to four GT data acquisition cabinets located in four separate rooms of the RB. The signals are apportioned to the four cabinets such that each cabinet gets a similar allocation of radial and axial GT signals. The digital GT signals are sent to SC3 equipment via the SC3 DCIS network.

7.3.2.3.4 Security

The SC2 equipment performing Safety Category 2 functions adequately address access control to limit cyber security vulnerabilities and ensure the system can perform its intended functions.

The SC2 DPS and FMCRD designs incorporate features to support establishment of a secure operational environment. SC2 equipment diagnostics and internal data are available to the SC3 networks and controllers. The SC2 DPS and FMCRD software is produced in a secure development environment that prevents the insertion of undocumented codes and precludes their use. The SC2 digital equipment is installed and maintained in accordance with the station administrative procedures and control of access programs.

7.3.2.3.5 Diversity and Defense-In-Depth

The SC2 systems provide diverse protection functions to place and maintain the plant in a safe state in the event of PIE concurrent with failure of Safety Category 1 functions to support the D-in-D strategy.

The design bases of the SC2 systems provide diversity and D-in-D for the complete failure of a Safety Category 1 function (i.e., CCF). Safety Category 2 functions independently provide safety functions comparable to the Safety Category 1 functions using diverse platforms from the SC1 equipment. The SC2 sensors are selected to be as diverse as practical from SC1 sensors. The

ERICP provides diversity and D-in-D by sending the manual scram, SC1 hydraulic scram and SC3 anticipatory hydraulic scram signals to the ERIPs to drive in rods by the FMCRDs.

Diversity measures are incorporated within the SC2 system designs as an additional level of protection for potential systematic faults caused by design and implementation defects within redundant divisions of the SC1 I&C system. The SC2 method of actuation is diverse from the method of actuation implemented the by Safety Category 1 hydraulic scram function. The SC2 actuation relays shown in Figure 7.3-4 and Figure 7.3-5 are configured as energize to actuate whereas SC1 I&C load drivers are configured as de-energize to actuate which provides functional diversity in these actuation circuits. The ARI pilot valves described in Section 4.6 are energized to actuate and provide an alternate path to vent control air and open scram valves resulting in hydraulic insertion of control rods.

The FMCRD motor controllers are required to be diverse from the SC1 I&C platform equipment and credited as a required mitigating function for a failure of Safety Category 1 hydraulic scram function due to CCF.

A motor run-in results from scrams manually initiated via the MCR or the SCR to the ERIPs, with no involvement of RC&IS software.

The FMCRD UPS provide additional D-in-D if normal power is lost. See Subsection 8.3.2 for additional detail.

7.3.2.4 Operator Interface and Accident Monitoring

Data obtained by the SC2 equipment is sent to the SC3 DCIS for display, recording, monitoring, and alarming. SC2 system bypass conditions are alarmed in the MCR. A reactor scram can be manually initiated by switches in the MCR or the SCR.

The operator interface is described in Section 7.5 and Section 7.6 for the MCR and SCR, respectively. SC2 sensors signals shared with SC3 equipment provide measurements that are diverse from SC1 sensors for selected parameters.

7.3.2.5 Compliance Alignment

Table 7.3-6 shows how the SC2 systems align to meet the regulatory guidance in USNRC I&C DRG: Instrumentation and Controls for Non-Light-Water Reactor (Non-LWR) Reviews," (Reference 7.3-19) and the key industry I&C standards IEC 61513 (Reference 7.3-23) and IEEE Std 603-2018 (Reference 7.3-32).

7.3.3 Nuclear Controllers including Anticipatory Trip System – Defense Line 2/Safety Class 3/Nuclear Segment

This section addresses the SC3 nuclear segment, which complements the SC1 and SC2 I&C systems to provide a D-in-D for the plant protection functions. It also provides detail for the SC3 nuclear segment which contains the SC3 Anticipatory Trip System, major reactor control, and the control rod blocking safety functions.

The SC3 system acquires the plant information from the sensors specified for systems performing Safety Category 3 functions or from isolated signals shared from SC1 and SC3 systems. The specific list of parameters to be measured, the physical locations of the sensors, and environmental qualification envelope for the equipment are specified as part of the detailed design process. The plant information acquired is displayed on the SC3 integrated plant displays developed through the HFE process.

7.3.3.1 System Architecture

The functional architecture for the SC3 nuclear segment is shown in Figure 7.3-11. Depending on final DCIS design and processor loading, multiple functions may be combined and implemented on a common controller. A representative block diagram for the Safety Category 3 anticipatory scram functions is shown in Figure 7.3-12. This figure should also be viewed with Figure 7.3-4, which shows the scram solenoid power configuration.

The SC3 nuclear segment employs TMR and redundant controller architectures for processing units to improve reliability or eliminate spurious control actions caused by a single hardware failure, as shown in Figure 7.3-11.

Protection function and control signals needed between controllers for their functions are point-to-point wire or optical fiber.

The design provides both control and monitoring functions and supports both local and remote data acquisition.

7.3.3.2 System Design Bases and Associated Safety Functions

The design bases of the Safety Category 3 functions are to prevent or mitigate AOOs before either Safety Category 1 or Safety Category 2 functions are required. Safety Category 3 action precludes the need for other DL intervention. Safety Category 3 functions are implemented in SC3 equipment. These SC3 hardware and software are diverse from SC1 equipment performing Safety Category 1 functions. The SC3 system design includes provision of instrumentation to monitor plant variables and systems over the respective ranges for operational states and during a PIE in order to ensure that plant status information can be obtained. The supporting safety analyses are described in Chapter 15.

SC3 equipment is designed to be in service during all modes of plant operation. Specific safety functions are mode dependent, as determined by the Fault Evaluation process in Chapter 15 and specified by the Mode Switch position shown for each function in Table 7.3-4.

The main SC3 nuclear segment functions are:

1. Anticipatory Trip System – The Anticipatory Trip System is designed to perform the Safety Category 3 functions which are mitigating transients and providing investment protection for expected transients by tripping the plant in advance of any required Safety Category 1 or Safety Category 2 response. There is no failure of Anticipatory Trip System that can adversely affect the Safety Category 1 or Safety Category 2 functions. The Anticipatory Trip System uses logic similar to the Safety Category 1 hydraulic scram function and DPS, in that parameters are measured, compared to a setpoint, voted and an actuator initiated. In the unlikely event the SC3 Anticipatory Trip System fails to scram, the other independent and diverse SC1 system still scrams the reactor on flux or pressure to maintain the plant safety functions (to prevent unacceptable radiation release to the public). An Anticipatory Trip System malfunction cannot stop a SC1 hydraulic scram, nor can a SC1 hydraulic scram stop a Safety Category 3 ATS scram function. The Anticipatory Trip System scram initiation and rod block signals are listed in Table 7.3-4. Simultaneous with the initiation of the Anticipatory Trip System hydraulic scram demand, a scram follow signal is generated such that each FMCRD motor drives its ball nut upward to a position just below the fully inserted and latched hollow piston that is coupled to the control rod blade. This signal is sent whenever a hydraulic scram is demanded by Anticipatory Trip System regardless of whether the hydraulic scram actuation has been successful.

The Anticipatory Trip System provides Safety Category 3 non-scram functions, including:

a. Turbine Trip on Low Main Condenser Vacuum
b. Turbine Bypass Valves Closure on Low Main Condenser Vacuum
c. Turbine Bypass Valves Fast Open on Fast Closure of Turbine Control Valve/Turbine Stop Valve Demand
d. MSRIV/MSCIV Isolation on Low Main Condenser Vacuum
e. Standby Diesel Generator Start on Low Electrical Bus Voltage
f. FW and Condensate Pumps Trip on High RPV Level
g. FW Isolation on High-High RPV Level
h. ICS Pressure Control on High Reactor Pressure

The setpoints for SC3 Anticipatory Trip System actuation logic are determined with the setpoint methodology defined in IEC 61888 (Reference 7.3-21) using the final analytical limits from the plant safety analyses and the measurement uncertainties associated with the Anticipatory Trip System equipment (e.g., sensors and processing units). Safety Category 3 functions are designed to prevent or mitigate anticipated operational transients before either Safety Category 1 or Safety Category 2 action is required. As such, for common physical parameters, the Safety Category 3 function setpoint is designed to act first, because its design bases are to mitigate/prevent Safety Category 1 functions from needing to respond.

2. Plant Automation System (PAS) – The PAS coordinates automatic reactor power control as part of a larger automation scheme, including coordination of SC3 control systems (for control of nuclear equipment) and coordination of SCN control systems (for control of BOP equipment). The SC3 and SCN control systems need to individually be put into automatic mode by the operator for this automation to be functional; without this operator action, any commands originating from the PAS are ignored.

   The PAS provides batch control of the steps needed to automate the plant systems used to bring the reactor from cold conditions to rated power. One of these controllers is APR which interfaces with the nuclear control systems. Examples of these APR functions include:

   a. Bring reactor to critical
   b. Heatup and pressurization
   c. Increase power to 15%
   d. Roll turbine to rated speed
   e. Synchronize turbine-generator
   f. Increase electrical load to ~ 20%
   g. APR switch to power control

   The SC3 PAS does not communicate with SC1 or SC2 I&C systems, nor does it control SC1 plant equipment; instead, the operator is prompted to operate the Reactor Mode Switch to provide a necessary permissive before an automated function can occur.

   After the minimum 20% power is reached by predefined automation steps, the APR accepts reactor power demands from approximately 20% to 100%. The power demand

can be percent thermal power or desired generator MWe as selected by the operator. The desired generator MWe can come from a remote load dispatcher. In this case, the operator sets an allowed maximum and minimum reactor power and the allowed rate of change in reactor power. The remote load dispatcher modulates the desired generator MWe. The operator maintains control of reactivity by supervising and limiting the load dispatcher inputs.

3. Reactor Level Control (RLC) – The RLC provides reactor level monitoring and control from cold conditions through pressurization and heatup to power operation. This system controls reactor level at greater than 10 – 15% power by adjusting the frequency of the adjustable speed drives supplying power to the FW pump motors. At lower powers, the system uses a low flow control valve supplied with either a condensate pump or feed pump operating at fixed speeds, depending upon reactor pressure. The RLC also operates level control valves in the Shutdown Cooling System (SDC) and the Reactor Water Cleanup System (CUW) when water is rejected from the reactor vessel during cold and heatup operations.
4. Feedwater Temperature Control – The Feedwater Temperature Control controls the feedwater temperature delivered to the reactor by the sixth stage FW heater by measuring the heater exit temperature and modulating a steam heating valve as required to meet the demanded temperature setpoint. The FW temperature is programmed as a function of reactor power, and the temperature is held constant between 90 and 100% thermal power and then decreased as power is further reduced. Appropriate function logic features are provided to keep the FW nozzle temperatures from changing too rapidly. The controller also provides the sixth stage FW heater level control necessary to keep the FW heater level at the optimum setpoint for best heat transfer.
5. Reactor Pressure Control (RPC) – RPC provides control from a cold vessel through pressurization and heat up to power operation. During plant operation this system normally controls reactor pressure by sending a flow demand signal to the turbine controller. During plant startup or at other times when the turbine is off-line, RPC adjusts reactor pressure by sending a flow demand signal to the turbine bypass valves.
6. IC Pressure Control - To meet their safety requirements, the three BWRX-300 ICs have a very large capacity for reactor pressure reduction. While desirable in safety systems, this capacity can result in exceeding the normal allowed cooldown rate for the reactor vessel. The ICs are each initiated with two condensate return valves. Both valves are “open or close” and can be actuated by the SC1 protection system and the SC2 DPS. One valve is also controlled by a SC3 I&C system.
7. Turbine-Generator Control - The normal function of turbine-generator control is to position the turbine control valves and bypass valves. The turbine-generator controller provides the turbine protection functions (e.g., turbine overspeed protection). The turbine-generator controller also provides trip and load rejection signals to the Anticipatory Trip System and similar signals to the RPC to quickly open the bypass valves. The controller receives both high reactor level and vacuum trip signals from the SC2 DPS (to ensure these trips occur in the right sequence with other control systems). The turbine-generator controller is involved in plant automation to allow the turbine to be rolled, synchronized, and initially loaded. The operator deliberately puts the system into automated control. Several adverse conditions and diagnostics take the system out of automation.

8. RC&IS – The RC&IS provides the control logic and positions control rods using the fine motion drive motors.

   RC&IS stores at least two predefined control rod sequences from all control rods full in (shutdown) to all control rods full out (i.e., end of cycle power operation). The sequences are designed to avoid exceeding thermal limits and to meet fuel operating guidelines concerns under expected conditions of load following and transient xenon. Once a sequence is selected, RC&IS is constrained to operate in that sequence, but as described for the blocking systems, RC&IS is supervised by blocking functions to ensure that the sequences are followed, and no thermal limits or fuel operating guidelines are exceeded. The rod block functions are listed in Table 7.3-4.

   In the automatic mode, RC&IS accepts insert and withdraw commands from APR. These commands are issued as required to achieve the desired reactor power. Rod motion is constrained to the RC&IS selected sequence. The Automatic Thermal Limit Monitor (ATLM) and MRBM are always functioning to supervise and block control rod motion. Only the operator can put the RC&IS into the automatic mode, but certain plant events or diagnostics can take the system out of automation mode.

   In the semi-automatic mode, the operator controls the insert or withdraw from the MCR. Once momentarily depressed, the RC&IS inserts or withdraws the rods until the next predefined rod position breakpoint and then stops. Rod motion is constrained to the selected sequence and the ATLM and MRBM are actively enforcing rod motion blocks if unacceptable conditions are detected.

   In the manual mode, the operator can move rods out of sequence (until blocked by the ATLM or MRBM), but unlike the other modes, only single (as opposed to ganged) rod motion is allowed. At lower thermal powers, manual movement commands deviating from the selected sequence are blocked. At higher thermal power levels, manual movement commands deviating from the sequence are alarmed.

   RC&IS moves up to eight control rods to a predefined rod pattern (by insertion only) when a Selected Control Rod Run-In (SCRRI) demand is issued. The SCRRI rod pattern is chosen to achieve a specific reactor power.

   The RC&IS provides position demands to the control rod motor controllers and drives which are in SC2 equipment. The SC2 design features allow the FMCRDs to be commanded to full in or to be blocked by disabling rod withdrawal regardless of RC&IS status or RC&IS positioning demand signals. The design allows a Safety Category 3 function implemented in SC3 equipment system to give commands to an SC2 system because the SC3 system is supervised by the blocking systems.

9. Containment Inerting/Nuclear Boiler Control - The containment inerting function provides monitoring and control for the generation and storage of nitrogen gas to do the initial containment fill and thereafter to make up for the small long-term losses. It also supplies nitrogen to any inside containment pneumatic valves. The I&C Nuclear Boiler function acquires signals from the nuclear boiler system into the SC3 DCIS to provide monitoring and control functions for nuclear boiler system SC3 equipment.

10. Shutdown Cooling Control – The shutdown cooling function provide separate controllers for monitoring and control for the SDC system to remove decay heat from the reactor. There are two separate SDC controllers that respectively operate the redundant SDC. The SDC controllers can receive commands from the Plant Automatic System or can be manually operated.

11. Containment Cooling Control – The containment cooling function provides redundant monitoring and control of the redundant containment air handling units with chilled water cooling during normal operation or during an outage.

12. Isolation Condenser and Fuel Pool Cooling and Cleanup Control – The isolation condenser and fuel pool cooling, and cleanup function provides monitoring and control for the IC pool redundant heat exchangers and pumps and their associated valves. This system provides monitoring and control for the Fuel Pool Cooling and Cleanup System. The controller interfaces with redundant pumps and heat exchangers and filter demineralizers. The controller also includes logic to start a standby pump if an operating pump trips.

13. Service Water, Plant Cooling, and Chilled Water Equipment Control – The service water, plant cooling, and chilled water equipment function provides redundant monitoring and control for the Service Water, Plant Cooling Water, and Chilled Water Equipment System pumps and valves.

14. SC2/SC3 Electric System Control – The SC2/SC3 electric system feeder breakers, load center breakers, motor control center breakers, and diesel generator breakers interface with the two SC2/SC3 electric system controllers for control and monitoring, but these breakers perform their protection functions independently of I&C control. The electric system control is usually by operator manual action except for the diesel generator breakers which close automatically after a loss of associated bus power after the diesel has been started and reached rated speed and voltage. The SC2/SC3 electric system controllers also control the sequencing of the standby diesel generator loads to minimize operator involvement and prevent diesel overload and under or over voltage and frequency. The controller provides the ability to synchronize the standby diesels to an energized bus to allow periodic load testing of the diesel generator. The controllers also provide extensive monitoring of the electric system parameters and for the SC2 UPS, battery chargers, and batteries.

15. SC2/SC3 Protective Relaying Monitoring – The SC2/SC3 protective relaying function is to interface with the relays through gateways as necessary, so data is available to I&C to support operator displays and alarms of electrical system status and for advanced condition monitoring.

16. RB and CB HVS Control – The RB and CB HVS provides redundant monitoring and control for automatic operation of the normal RB and CB HVS requirements by controlling the redundant cooling of the various RB and CB air handling units, building intake and exhaust fans, and, if required, recirculating fans. The room and building temperatures are controlled by damper and chilled water control valve operation.

17. Process Radiation Monitoring – The process radiation monitoring provides radiation data to other systems for trips and also provides data for operator monitoring, alarming, and recording. The SC3 controller provides for the acquisition of the various process radiation measurements, alarms, and diagnostics, and transmission of these data to the SC3 nuclear segment networks with appropriate isolation or using appropriate safety class equipment to generate and transmit the data.

18. SC3 Gateways – The SC3 gateways are used to bring data-linked signals to the SC3 nuclear segment networks. The signals from SC1 systems are isolated with SC1 isolators at the source. The two gateways perform three Safety Category 3 functions and are implemented in SC3 equipment. The first function provides another layer of isolation. This isolation function also prevents communication back to the SC1 sources. The second

function is to provide protocol conversion of the signals from SC1 systems to be compatible with the plant networks and package the data into messages for specific SC3 controllers. The third function is to accept data links from SC2, SC3, or SCN sources that are input to the SC3 nuclear segment.

19. ATLM – The ATLM function provides redundant protection against critical power ratio (a fuel heat flux to flow consideration) and linear heat generation rate (maximum heat flux through the fuel clad) design limits. This online monitoring of thermal limits and fuel operating guidelines uses LPRM and APRM data sent via qualified isolation devices. The fuel operating guidelines keep the fuel duty per node below previously conditioned power levels to prevent fuel failures. Operation above the preconditioned envelope per node is constrained to slow power changes. The ATLM rod blocks are automatically operative at higher reactor powers. The ATLM receives online LPRM data and uses it with previously stored and periodically updated 3D power distribution and RC&IS data for current rod positions and the next selected rods to be moved. This data is used to predict the resulting thermal limits and nodal powers for the fuel operating guidelines. If the prediction anticipates a thermal limits or soft duty guideline violation, the control rod motion is blocked.
20. MRBM – The MRBM provides redundant protection against critical power ratio and linear heat generation rate design limits. The MRBM rod blocks are automatically operative at higher reactor powers. The redundant MRBMs use LPRM and APRM data received from qualified isolation devices and control rod status from RC&IS. The MRBMs operate after receiving a signal from RC&IS that rods have been selected for motion. The MRBMs, using the real time LPRM information received from the gateways, identify the LPRMs around the selected rods and normalize their average to a nominal value. As the selected rods are being withdrawn, the LPRM average around the selected rods increases in value. If the ratio of the withdrawing average to the initial average increases by too great an amount, it indicates that the local power has increased by an amount that would violate fuel thermal limits. Under those circumstances, the MRBMs issue a rod block.
21. WRNM – The WRNM function provides redundant monitoring of the fixed neutron monitoring detectors in the BWRX-300 core. It acquires signals from ten fixed neutron monitoring detectors (i.e., fission chambers). The detectors are distributed radially in the core at fixed heights with at least two detectors in each quadrant. The detectors can be bypassed but only one detector can be bypassed at a time in any quadrant. Each detector is sensitive to neutrons from below criticality to greater than 100% thermal power. Their absolute value is indicative of reactor power. The WRNM detectors reactor power indication is not accurate given how few detectors are in the core, but they can indicate that the core has been successfully shutdown post rod insertion and for controlling reactor flux rate of change during a reactor startup. The average of the detectors can be related to thermal power by comparison to APRM data. This relationship is continuously adjusted to ensure that there is always an overlap in the 1 – 10% power region. Although the very localized WRNM absolute values are not required to be accurate, the main use of the detector data is to determine reactor period per detector. These period data are representative of the whole core and are used for monitoring and by APR. Both the operator and APR use the data when starting the reactor. Either WRNM can cause a control rod withdrawal block (unless bypassed) for a short period and either can cause a hydraulic scram (when reactor mode switch is in STARTUP) via the Anticipatory Trip System for a shorter period.
22. Safety Parameter Display System – The SPDS provides a big picture overview display of plant parameters during transients and accidents. The SPDS terminology is used for

consistency with regulatory documents. The BWRX-300 SPDS is not a separate system; instead, it is functionality that is included with the SC3 equipment. The BWRX-300 SPDS presents sufficient information on safety-critical parameters and includes display and trending of the parameters associated with the emergency procedure guidelines for the plant. The specific displays are developed in accordance with the BWRX-300 HFE process. The displays cover the full range expected for the safety-critical parameters, track data trends, indicate when process or safety limits are being approached or exceeded, and display the status of safety systems. The same data can be transmitted to the emergency response facilities described in Section 13.3.

23. TS Monitor – The TS Monitor provides surveillance tests and support. It monitors signals and alarm deviations from plant TS requirements for those requirements where appropriate plant signals are available. The redundant TS Monitors have no signals of their own but instead have access to the signals on the plant networks including signals from three SC1 divisions available to the SC3 nuclear segment through isolated, unidirectional boundary devices and gateways. The TS Monitors are designed to support online and automatic surveillance testing to meet TS requirements. Redundant input signals from the SC1 divisions used for safety actuation are continuously compared by the TS Monitors during operation and alarmed for deviations outside of established tolerance. Similarly, the correct functioning of the SC1 digital I&C platforms is continuously monitored by self-testing features rather than by manual surveillance testing. Critical faults detected by the self-testing features trip the outputs of the platform and other faults are alarmed. The TS Monitors are a redundant pair of processing units with a real time operating system that each monitor available TS data. The TS Monitors communicate among themselves to indicate they are working and agree on their determinations. The TS monitors have their own self-diagnostics, and the loss of a TS Monitor is alarmed.

24. Core Thermal Power/Flow Monitor – The core thermal power/flow monitor provides reactor thermal power and core flow information using a redundant pair of processing units with a real time operating system that calculates reactor thermal power and core flow. The thermal power is calculated once per second and is made available plant wide over the plant networks. The lowest power levels are estimated from the WRNM system and the intermediate power levels from the low flow FW system and bypass valve position. The program seamlessly transitions between the calculation methodologies. In the normal power range, approximately 10 to 100% thermal power, the calculation is done using a heat balance. There are no assumed inputs or constants. Heat balance parameters are measured and, in most cases, redundant. The most important measurements, FW flow, FW temperature and reactor pressure (used to determine the FW and steam enthalpies that account for greater than 99% of the heat balance accuracy) are triply redundant. Natural circulation core flow is calculated by a heat balance around the core inlet. Alarms are provided if sensed parameters are missing or inconsistent.

    The core thermal power/flow monitors have their own self-diagnostics. The loss of a monitor results in an alarm.

25. 3D Core Thermal Power Distribution Monitor – The redundant 3D core thermal power distribution monitor computers use the thermal power calculation, core flow calculation, isolated (from SC1 and SC2 equipment) LPRM and GT data, and control rod position data to determine the core 3D power distribution. The program runs both periodically (typically every hour) and on demand from the operator or ATLM (discussed above). Thermal limits and other diagnostics are alarmed, and data are recorded in the plant historians. The 3D Core Thermal Power Distribution Monitors use a redundant pair of processing units with a real time operating system that each provide core monitoring, fuel operating guidelines,

and thermal limit information. The core monitors have their own self-diagnostics, and the loss of a monitor results in an alarm.

26. Plant Alarm System – The plant alarm system provides aggregated process alarms and self-diagnostic data from the plant control and monitoring systems over the plant networks. The alarms and data are prioritized and filtered by plant mode and events. The alarm system also provides annunciator displays integrated into the MCR video displays. The purpose of the annunciator display is to attract operator attention to specific display formats associated with the alarm. The operator can display currently active alarms, acknowledged alarms, and "first in" alarms. The alarms are available to the operator and recorded in the plant historians. Alarm information is available in both control rooms and to perform the technical support center function. The alarm systems are redundant pairs of processing units that each monitor plant data. The alarms are available even with a single alarm system failure.
27. Plant Historians and Monitoring – The plant historians and monitoring computer provides archiving of the plant control, alarm, and monitoring system information. The data are recorded at least once per second, depending on the parameter, (e.g., flux and pressure). Some data are recorded at higher rates, and some are recorded at lower rates. The recording capacity is designed to at least collect data from a fuel cycle. The historians allow archival storage and playback while the plant is online and support operator-initiated data trending on control room displays or report printouts. The historians are redundant pairs of processing units that each monitor plant data. The stored data is still available even with a single historian failure.
28. Fiber Optic Relay Panels – The fiber optic relay panels provide the capability to send data across SC boundaries when the use of data links and networks is not indicated. These non-software panels containing optical fiber transmitters and receivers can transmit either analog or discrete contact signals.

7.3.3.3 Fundamental Design Properties in the System Design

The SC3 system includes DL1 properties that represent the quality measures implemented to minimize potential for failures and initiating events to occur in the first place, by the use of conservatism in design and analyses. These DL1 properties of qualification, reliability, robustness, security, diversity, and other D-in-D features are discussed in the following five subsections.

7.3.3.3.1 Equipment Qualification

The SC3 equipment is designed to operate in the environment that is to be expected during both normal operations and anticipated off-normal conditions, and the equipment is qualified to perform its intended functions and meets IEC 61513 (Reference 7.3-23) requirements and the equipment standards defined in Section 7 in IEC 61226 (Reference 7.3-24) requirements for Class 3 systems. The SC3 nuclear segment software is developed to meet IEC 62138 (Reference 7.3-33) Category C requirements. Equipment qualification addresses both hardware and software aspects of the SC3 nuclear segment. The SC3 nuclear segment design and manufacturing processes are of sufficient quality to ensure I&C systems can reliably perform their credited protection functions. The SC3 nuclear segment equipment is designed, developed, fabricated, and tested to quality standards commensurate with the safety significance of the functions to be performed. The SC3 nuclear segment equipment qualification measures demonstrate that the I&C systems and equipment are capable of reliably performing the design bases functions for which they are credited over the range of environmental conditions postulated for the area in which they are located.

The SC3 equipment is qualified for seismic interaction in accordance with IEC/IEEE 60980-344 (Reference 7.3-27) if required for mitigation of seismic-related events. The SC3 equipment is qualified for electromagnetic compatibility in accordance with IEC 61000-4 Series (References 7.3-1 through 7.3-17) and IEC 61000-6-2 (Reference 7.3-29).

7.3.3.3.2 Reliability

The SC3 nuclear segment has the required reliability to perform its intended functions. The main SC3 hardware platform uses TMR controllers for the major plant control systems. It has an initial quantitative reliability target of less than 1E-2 probability of failure on demand for standby functions. The reliability analysis of the SC3 nuclear segment demonstrates it meets its reliability goals using qualitative and quantitative performance measures or criteria, as appropriate.

The SC3 nuclear segment process controllers are designed with TMR controllers to prevent random I&C component failures from causing plant transients. The TMR controller outputs are dual ported to the plant SC3 nuclear segment network. Where a mechanical system has safety functions and is redundant (i.e., “A” and “B”), two controllers are furnished so the redundant components may be operated separately, and the system is not affected by a single failure.

For the functions with TMR controllers, single controller failures (and some double failures) have no adverse effect on system or plant operation, and failed controller parts may be replaced online. The controllers perform Safety Category 3 functions and are implemented in SC3 equipment designed with a specific requirement that there be no failures more often than once per 100 years. The intent is that no random controller failure initiates an AOO. The reliability of the controller depends on final design and the formal analyses to determine failure rates are performed as part of the detailed plant design.

SC3 nuclear segment rod block functions are implemented on redundant processors for reliability. Normally both redundancies are online, and their data continuously compared. Any one redundancy can be bypassed at a time but, if bypassed, it takes the plant out of automation mode.

Design reliability assurance program and reliability, availability, and maintainability plan documents are used to quantify the required failure rates of the DCIS to assure plant safety and plant availability goals. Testing and performance assessment requirements for Anticipatory Trip System are developed using the guidance of IEC 60671 (Reference 7.3-28) and IEC 62385 (Reference 7.3-29).

Extensive hardware and software diagnostics are provided for the TMR controllers to provide operator monitoring and alarms.

The Safety Category 3 functions for ATLM, MRBM, WRNM, SPDS, Technical Specification Monitor, Core Thermal Power/Flow Monitor, 3D Core Thermal Power Distribution Monitor, Plant Alarm System, and Plant Historians and Monitoring are powered by the SC2/SC3 electrical system which provides redundant UPS and power feeds.

7.3.3.3.3 Robustness

Robustness of the SC3 nuclear segment is the degree to which it can function correctly in the presence of invalid inputs or stressful environmental conditions. The robust SC3 nuclear segment design reflects the use of design methods and adherence to engineering best practices to ensure that its functions are achieved for the specified conditions.

The SC3 nuclear segment has the required separation and independence to perform its intended functions. SC3 equipment is independent from SC1 equipment. Safety Category 3 functions are also independent from Safety Category 2 functions to an extent that is practicable in that shared sensors are not credited with mitigation of the same PIE. Safety Category 2 and Safety

Category 3 functions share support equipment. Safety Category 3 functions are designed to ensure that they cannot adversely affect Safety Category 1 or Safety Category 2 functions from fulfilling their protection functions. Safety Category 3 functions share some signals with Safety Category 2 functions sent via qualified isolation devices (as described in Subsection 7.3.2.3.3). Safety Category 3 functions may use signals from SC1 equipment sent via qualified isolation devices (as described in Subsection 7.3.1.3.3) for some control purposes. Plant operation is not adversely affected by the loss or inaccuracy of these individual signals.

The SC3 equipment implementing Safety Category 3 functions is located in two separate fire barrier rooms in the CB. The SC1 equipment implementing Safety Category 1 functions is located in three separate divisional fire barrier rooms in the RB. The SC2 equipment implementing Safety Category 2 functions is located in a separate fire barrier room in the CB.

There is no communication from SC3 equipment to online SC1 equipment and only rod block or motor run-in initiation (dry contact) communication to the SC2 ERIPs. There is no direct connection between SC1 equipment and the SC3 APR. The APR does use LPRM signals from SC1 equipment for power feedback, but it also uses GT signals and the core thermal power computers for similar feedback. These signals are continuously compared, and any discrepancy takes the plant out of automation mode to prevent any power signal from misleading the control system. Other than the APRM signals, Safety Category 1 functions and the SC3 APR controller have no sensors or actuators in common. The Safety Category 3 physical separation and signal isolation devices meet the requirements of IEC 60709 (Reference 7.3-31).

The SC3 nuclear segment has the required fail-as-is design features to perform its intended functions and avoid spurious actuation. The fail-as-is, energize to actuate, design is used to prevent lesser classified systems (e.g., Anticipatory Trip System) from creating unnecessary challenges to plant safety (e.g., spurious actuations for expected failures) requiring action by Safety Category 1 functions. Where software is used for SC3 equipment diagnostics, the SC3 equipment includes watchdog timers; sensor range checks, and monitoring of power supplies, communications, and actuators. Each output is produced separately by a 2oo3 discrete, or median select analog vote from each of the three controllers. The scheme is used to provide reliable trips and avoid inadvertent trips.

The Safety Category 3 blocking system functions (i.e., ATLM, MRBM, and WRNM) are redundant and are used to supervise the control systems and plant conditions. Either of the redundant blocking pair can block rod motion by removing power to the FMCRD motors. One of the redundant pair can be bypassed for maintenance activities. The remaining component of the blocking pair can still block rod motion. If both blocking pairs are not available, semi-automatic and manual control rod motion is halted until at least one is available. The blocking pairs alarm and block if they lose communication with the SC3 gateways or RC&IS, cannot receive an update from the core 3D Core Thermal Power Distribution Monitoring program, or if their internal data do not agree. As described, the system is fail-safe to block when both blocking pairs are not available, on a loss of communication between systems, or on a loss of input data.

Signals from the redundant WRNM controllers are used to initiate a rod withdrawal block for a short reactor period and initiate a Safety Category 3 anticipatory hydraulic scram for a shorter reactor period. The scram is only available when the Reactor Mode Switch is not in RUN (typically below 15% thermal power as measured by the APRMs). The WRNM detectors are distributed radially in the core at fixed heights with at least two detectors in each quadrant with two additional detectors (ten total) at the boundaries of two adjacent quadrants such that each quadrant has a backup in the event of a single WRNM failure. If no WRNM detector is available in any core quadrant, rod motion is prohibited except in manual mode. If no WRNM detector is available in any core quadrant during refueling activities, then fuel movement is halted.

The SC3 nuclear segment is predictable when performing its intended functions. Where Anticipatory Trip System or a reactor control system TMR controller uses three signals from SC2 splitters, and a fourth SC3 signal. Typically, these parameters include containment pressure, reactor level, reactor pressure, and condenser vacuum scrams. Each Anticipatory Trip System controller and reactor control system validates the analog signal with a range and consistency check, compares each of the four signals to a common setpoint, and then indicates the analog value and trip status. For example, the reactor level and pressure controller analog validation algorithms use the three splitter signals from SC2 equipment, and the fourth signal dedicated to Safety Category 3 functions to make a fault tolerant control signal that functions even if the splitter signals are lost:

- If four signals are valid, use average of two median signals
- If three signals are valid, use median signal
- If two signals are valid, use average of two signals
- If only one signal is valid, use signal

For trip signals derived from analog inputs the Safety Category 3 validation algorithms functions:

- If four signals are valid, use Two-out-of-Four (2oo4) logic
- If three signals are valid, use 2oo3 logic
- If two signals are valid, use 2oo2 logic
- If only one signal is valid, do not trip

The scheme is used to provide reliable trips and avoid inadvertent trips.

7.3.3.3.4 Security

The SC3 nuclear segment adequately addresses security to limit cyber security vulnerabilities and ensure the system can perform its intended functions.

The SC3 nuclear segment design incorporates features to support establishment of a secure operational environment. It is installed and maintained in accordance with the station administrative procedures and control of access programs.

The BWRX-300 network switches interconnect the SC3 DCIS segments. The network switches have the security features described in Subsection 7.2.3. The communications on the nuclear segment UDHs are rigorously controlled and monitored and information sent outside the plant network is through unidirectional boundary devices, as described in Subsection 7.2.4.

7.3.3.3.5 Diversity and Defense-In-Depth

The SC3 nuclear segment provides anticipatory trip functions to support the D-in-D strategy. An example of anticipatory trips is an Anticipatory Trip System scram on loss of off-site power, which is initiated before the Safety Category 1 and Safety Category 2 functions scram the plant on reactor low level on loss of the feed pumps or a high flux resulting from a reactor isolation on loss of vacuum. Another example is the Anticipatory Trip System scram on a turbine trip or load rejection before the Safety Category 1 and Safety Category 2 functions scram the plant on reactor high flux or pressure; this also mitigates the adverse thermal limit response because the reactor is scrammed before the pressurization wave impacts the reactor. The Safety Category 3 functions are implemented in SC3 hardware and software that are diverse from SC1 equipment. The SC3 Anticipatory Trip System actuation relays shown in Figure 7.3-4 are configured as energize to actuate and the SC1 I&C load drivers are configured as de-energize to actuate, which provides functional diversity in these actuation circuits.

Diversity measures are incorporated within the SC3 nuclear segment design as an additional level of protection for potential systematic faults caused by design and implementation defects when equipment is credited to as backup in the safety analyses.

The use of a fourth SC3 signal provides immunity against loss of control signals due to the postulated CCF of the SC2 signal splitters.

Safety Category 3 functions can initiate hydraulic scrams, automatically and manually initiate isolations, and operate the ICS. No SC3 equipment can prevent a scram, isolation, or ICS initiation actuated from SC1 or SC2 systems. Similarly, Safety Category 1 and Safety Category 2 functions cannot adversely affect Safety Category 3 functions. If Safety Category 3 functions do not send out their anticipatory trip commands, plant safety is ensured by Safety Category 1 and Safety Category 2 functions.

Although the SC2 equipment, the SC3 nuclear and BOP segments, and the SCN BOP controllers are powered by UPS with battery backups, the power supply is redundant, and a single failure does not affect either Safety Category function or segment. The UPS and the battery chargers can be powered by off-site power or either of the standby diesel generators. Double breakers or fuses are used to separate the different Safety Class equipment and to separate the SC3 nuclear and BOP segments such that a fault does not propagate. No power feeds to equipment performing Safety Category 2 or Safety Category 3 functions are shared with SC1 equipment performing Safety Category 1 functions.

The SC3 nuclear segment provides diversity to support the D-in-D strategy. The SC3 hardware and software performing Safety Category 3 functions are diverse from SC1 equipment performing Safety Category 1 functions. The SC3 software and logic is independent of the SC2 software and logic. The processing units used for the control rod blocking functions are implemented in equipment diverse from the SC1 equipment if it is credited as a backup in the safety analyses. The turbine overspeed protection control logic uses two diverse and independent systems to provide sufficient reliability to eliminate the risk of turbine missile generation and to avoid the need for a mechanical overspeed trip. These turbine overspeed protection systems use TMR control architecture to eliminate random hardware vulnerabilities that could result in spurious turbine trips.

7.3.3.4 Operator Interface and Accident Monitoring

The SC3 data are available on their appropriate displays. The layout and content of the displays are designed as part of the HFE program of activities integrated into the BWRX-300 design. The SC3 DCIS has its own displays in both the MCR and SCR and locally in the SC3 equipment rooms. The SC3 displays also include an alarm system to provide operator awareness and to prompt to the displays containing further information relating to the alarm. SC3 system bypass conditions are alarmed in the MCR.

Some SC3 monitoring signals provide necessary information to achieve or verify correct plant response to transients and accidents. The necessary signals are determined through HFE analysis and applicable regulatory requirements and are referred to as accident monitoring signals.

The SC3 signals are alarmed to prompt the operator to an appropriate display (e.g., range check problems or alarm setpoints exceeded). The SC3 signals and the various SC3 components are alarmed for self-diagnostics.

The SC3 system acquires accident monitoring information associated with Type B and C accident monitoring variables from the SC1 I&C through isolated interfaces. It also acquires accident monitoring information associated with Type D, E, and F accident monitoring variables from SC3 and SCN equipment. The aggregated SC3 displays of accident monitoring information are

provided to the MCR located in the CB and the SCR located in the RB. Type D, E, and F are defined in IEC 63147/IEEE Std 497 (Reference 7.3-20). Type D variables provide information to accident management personnel to verify safety system status and that indicate the performance of safety systems, required auxiliary support features, and other systems necessary to achieve and maintain a safe shutdown condition. Type E variables provide information to accident management personnel to monitor radioactive release identified pathways and environmental conditions used to determine the effect of releases in the plant environs and radioactivity levels in the control rooms and selected plant areas where access may be required for plant recovery. Type F variables provide information that indicates fuel damage and the effects of fuel damage to support accident management response to severe accidents. Type D, E, and F variables are categorized as Safety Category 3. A preliminary list of the Type D, E, and F accident monitoring variables is provided in Table 7.3-2.

The SC3 and SCN DCIS rooms, which include SC3 I&C and associated electrical support equipment performing Safety Category 3 functions and functions with no safety category are located in the CB. The MCR is used by plant operators unless conditions require relocation to the SCR. The CB provides the necessary seismic ruggedness and habitability capabilities to support the expected post-accident conditions with a SC3 level of quality and reliability for required protection functions.

The operator interface HSI for the SC3 nuclear segment is described in Sections 7.5 and 7.6 for the MCR and SCR, respectively.

7.3.3.5 Compliance Alignment

Table 7.3-7 shows how the SC3 systems align to meet the regulatory guidance in the USNRC I&C DRG (Reference 7.3-19) and the key industry I&C standards IEC 61513 (Reference 7.3-23) and IEEE Std 603-2018 (Reference 7.3-34).

7.3.4 Balance of Plant Controllers – Non-Safety Class/Balance of Plant Segment

This section addresses the SCN BOP controllers that feed the SC3 BOP segment, which contains the control and monitoring systems associated with power generation and support systems.

7.3.4.1 Systems Architecture

The functional architecture for the SCN BOP controllers is shown in Figure 7.3-13. The functions shown using TMR controllers are dual ported to the plant SC3 BOP segment network. The other functions are “packaged systems” (typically embedded controllers) provided by an equipment vendor. Depending on final DCIS design and processor loading, functions may be combined onto common or shared controllers. Similarly, the use of TMR may be changed as the design progresses based on assessment of controller failure effects on the plant.

Control signals needed between controllers for their functions are point-to-point wire or optical fiber. The signals needed by a controller for its functions are radially connected to the controller and do not use plant networks.

7.3.4.2 System Design Bases and Associated Safety Functions

The design bases of the SCN BOP controllers are to provide for the control and monitoring of the SCN power generation and support equipment. The SCN I&C system design includes provision of instrumentation to monitor plant variables and systems over the respective ranges for operational states in order to ensure that adequate information can be obtained on plant status. This equipment is not credited for protection functions although they may be available for beyond design basis events and severe accident scenarios.

The main functions of the SCN I&C system are:

1. Condensate and FW – The condensate and FW function provide monitoring and control for the condensate and FW systems as the plant is brought from cold to rated power conditions.
2. FW Heater Drains and Extraction Steam – The FW heater drains, and extraction steam function provides monitoring and active heater level control functions to prevent heater isolation, turbine trip, and to provide optimum heat transfer.
3. Condenser – The condenser function provides monitoring and “batch” process functions as the plant is brought from cold to rated power. The condenser also requires extensive monitoring for both condition monitoring and investment protection purposes.
4. Turbine Auxiliaries – The turbine auxiliaries function provides control of several auxiliary systems that are controlled for normal plant operation. The turbine auxiliaries also require extensive monitoring for both condition monitoring and investment protection purposes.
5. Generator Auxiliaries – The generator auxiliaries function provides control for the stator cooling system and may also provide the interface to the generator exciter to provide generator voltage regulation and reactive power (dispatch as requested by the load dispatcher). The generator auxiliaries also require extensive monitoring for both condition monitoring and investment protection purposes.
6. Moisture Separator Reheater – The moisture separator reheater function provides reheater stage and the moisture separator level control reheater exit steam temperature control. The Moisture Separator Reheater System also requires extensive monitoring for both condition monitoring and investment protection purposes.
7. Turbine Bypass Valves – The turbine bypass valves function provides position feedback using linear variable differential transformers. These servo valves are part of the mechanical turbine bypass valve system.
8. CUW – The CUW function provides monitoring and control for the CUW flow in normal operation to maintain required system flow rates and protect the cooling water temperatures of the heat exchanger.
9. Circulating Water – The circulating water function provides monitoring and control of the cooling water flow to the main condenser.
10. Intake Structure – The intake structure function provides monitoring and control for racks and travelling screens to prevent debris from entering the water systems and associated HVS to maintain the intake building environment.
11. Turbine and Radwaste Building HVS – The turbine and radwaste building HVS function provides monitoring and control for automatic operation of the plant SCN chillers.
12. Service Air and Instrument Air and Plant Gas Systems – The service air, instrument air, and plant gas systems function provide monitoring and control for the Service Air and Instrument Air systems.
13. Condensate Polishing System – The condensate polishing system function provides control and monitoring of the condensate polishing system, which consists of multiple vessels for filtration and demineralization of the condensate water before injection into the reactor. The system is designed to be highly reliable for maintaining reactor chemistry control and to prevent inadvertent operation that could disrupt the condensate/ FW flow to the reactor vessel. The polishing system vessels are automatically sequenced and

regenerated/backflushed by the controller as chemistry dictates. Additionally, the controller provides vessel flow control to provide equal flow such that the resins and filters are not simultaneously depleted.

14. SCN Electric System – The SCN electric system function provides extensive monitoring of the electric system parameters and for the SCN UPS, battery chargers, and batteries.
15. SCN Protective Relaying – The SCN protective relaying function uses digital protective relays that appropriately protect the SCN medium voltage and low voltage busses and loads, feeder circuits, main and unit auxiliary transformers, and the main generator. Where a single failure can cause a plant scram, the main generator, main transformer, and unit transformer are protected with a 2oo3 relaying scheme. The relays are also in the control system architecture because they are networked to the SC3 BOP segment and the source of most of the electrical system parameter measurements. The required I&C function of these systems is to interface the relays through SC3 gateways as necessary, so the various data are available to I&C to support operator displays and alarms of electrical system status and for advanced condition monitoring.
16. Condensate Storage and Transfer – The condensate storage and transfer function provide monitoring and control for the condensate storage and transfer system to store condensate quality water for power generation and transfer the water for various uses in the power plant.
17. Equipment and Floor Drains System – The equipment and floor drains function provides monitoring and control for the plant sumps and drains to collect and release or process depending on the radioactivity and chemistry of the contained water.
18. Potable Water and Sewage – The potable water and sewage function provides monitoring and control for the capability to ensure that there is at least a one-week supply of stored potable water available in each control room.
19. Hydrogen Water Chemistry and Noble Metal Injection – The hydrogen water chemistry and noble metal injection function provides a dedicated controller for skid-mounted mechanical components that mechanically inject chemicals into the FW flow to the reactor. Hydrogen injection is continuous at power and proportional to FW flow. Noble metal is injected at commissioning and then periodically over the plant lifetime.
20. Liquid and Solid Waste Management and Offgas – The liquid and solid waste management and offgas function provides monitoring and control for the offgas system and the liquid and solid waste systems. The offgas controller involves both motor-operated and modulating valves to operate the steam air ejectors and steam dilution systems. The controller provides the systems' monitoring and control capability, specifically including radiation monitoring before radioactive material or gas is released. The liquid and solid waste systems provide batch controller functions.
21. BOP Gateways – The BWRX-300 uses many packaged systems bought from vendors instead of being designed and implemented by GVH. These vendor systems typically include embedded control systems. These vendor systems are integrated into the plant DCIS for both monitoring and control purposes. Each packaged system is connected to the DCIS networks through a SC3 gateway to minimize cyber security vulnerabilities. The SC3 gateways are designed to provide any needed protocol conversion and cyber security data flow features that limit what the embedded controller can do.
22. Fire Protection (Packaged) – Packaged system that meets local fire safety standards. The system is self-contained with its own operator interfaces (see Subsection 9.5.1). A SC3

gateway is provided to interface to the plant DCIS so that the operator can monitor the system, obtain immediate fire alarm location and actuation information, and dispatch an appropriate response before the fire protection system console can be staffed.

23. Water Treatment (Packaged) – Packaged plant that inputs raw water and makes demineralized water (see Subsection 9.2.3). The packaged plant embedded controller controls its pumps and demineralizers and backwashing/regeneration and operates in either batch or continuous mode.

24. Clock (Packaged) – The DCIS includes a set of redundant global positioning system satellite synchronized clocks. The main use of the clocks is to distribute network time through SC3 gateways to the plant control systems, historians, and electrical system digital protective relays so that signals from different sources are synchronized when recorded and for playback.

25. Equipment Condition Monitoring (Packaged, if provided) – The equipment condition monitoring equipment consists of a central console that receives data from special remote multiplexing units located near power generation rotating machinery including the main turbine, motors, pumps, and fans.

26. Seismic Monitoring System (Packaged) – The seismic monitoring system is a packaged system (see Subsection 3.7.4). It has its own internal UPS, an operator interface, and recording capability. There are two and three axis accelerometers located in various plant locations, including the yard. These are directly wired to the seismic monitoring system cabinet. Whenever an accelerometer signal exceeds a predefined value, the seismic monitoring system begins recording accelerometer data. The seismic monitoring system provides an alarm signal into the plant DCIS through a SC3 gateway to the DCIS networks. The alarm alerts plant staff to go to the seismic monitoring system operator interface to determine if there is plant damage. The intent of the system is, post-earthquake, to allow the operator or plant engineering staff to determine if the earthquake was above or close to the operating basis earthquake. This informs the decision of whether to shut down the plant or how much inspection of plant equipment is required. The seismic monitoring system is connected to the plant networks only for alarming purposes.

27. Area Radiation Monitoring System (Packaged) – The ARM is a packaged system with detectors at many plant locations accessible to personnel. The areas may exhibit potentially high radiation. The ARM cabinet has an operator interface and a data link to the plant DCIS. There are local radiation detectors mounted in various plant locations with the ability to provide both radiation data and predefined high radiation alarms (see Subsection 12.3.4).

28. Meteorological Monitoring System (Packaged) – The meteorological monitoring system consists of a site tower containing wind speed, wind direction, temperature, and barometric pressure instruments. The data are used with environmental monitoring radiation release data to predict radiation plume magnitude, direction, and dispersion (see Subsection 2.3.3).

29. Environmental Monitoring System (Packaged) – The environment monitoring system consists of sensors located at or near site boundaries to monitor plant releases. Radiation releases are typically measured but the system may be used to monitor other parameters required by environmental permits (see Section 11.5).

7.3.4.3 Fundamental Design Properties in the System Design

The SCN system properties are based in the quality measures implemented to minimize the potential for failures and initiating events to occur. The SCN systems are conservatively designed and analyzed. These properties of qualification, reliability, robustness, security, diversity, and other D-in-D features are discussed in the following five subsections.

7.3.4.3.1 Equipment Qualification

The SCN BOP controllers are designed to operate in the environment that is to be expected during normal operations. The SCN BOP controller design and manufacturing processes are of sufficient quality to ensure that I&C systems can reliably perform their design bases functions. The SCN BOP controller equipment is designed, developed, fabricated, and tested to quality standards commensurate with the safety significance of the functions to be performed. The SCN BOP controller equipment qualification measures confirm that the I&C systems and equipment are capable of reliably performing the design functions for which they are credited over the range of environmental conditions postulated for the area in which they are located.

The SCN BOP controller equipment is assessed as a potential hazard if located near safety classified equipment and, if necessary, qualified for electromagnetic compatibility in accordance with IEC 6100-4 Series (References 7.3-1 through 7.3-17) and IEC 71000-6-2 (Reference 7.3-29) for emissions only.

7.3.4.3.2 Reliability

The SCN BOP controllers have the required reliability to perform their intended functions. The reliability analysis of the SC3 BOP segment demonstrates it meets its reliability goals using qualitative and quantitative performance measures or criteria, as appropriate. Separately, the SCN BOP controllers have the required reliability to perform their intended functions.

SCN BOP controllers are implemented with redundancy as appropriate to ensure that single controller failures have no adverse effect on system or plant operation. The controllers have a specific reliability requirement goal that there be no failures more often than once per 100 years. The intent is that no random controller failure initiates an AOO. Controller reliability will be formally analyzed to determine failure rates, as part of the detailed plant design.

Extensive hardware and software diagnostics are provided for the TMR controllers to provide operator monitoring and alarms. Failed TMR controller parts may be replaced online.

The main SCN BOP controller platform's control and monitor design support both local and remote data acquisition, the latter with optical fiber or wire. The intent is that the failure of a SCN controller does not initiate an AOO.

7.3.4.3.3 Robustness

Robustness of the SCN controllers is the degree to which it can function correctly in the presence of invalid inputs or stressful environmental conditions. The robust SCN BOP controller design reflects the use of design methods and adherence to engineering best practices to ensure that its functions are achieved for the specified conditions.

The SCN BOP controllers have no direct connections (i.e., commands or communication) with the SC1 systems.

The SCN BOP controller equipment is in different rooms than the SC3 nuclear segment rooms and physically separated from SC1 and SC2 equipment rooms. The SCN BOP controller physical separation and signal isolation devices meet the requirements of IEC 60709 (Reference 7.3-31). Vendor supplied SCN hardware is connected through a SC3 gateway.

7.3.4.3.4 Security

The SCN BOP controller design incorporates features to support establishment of a secure operational environment. This equipment is installed and maintained in accordance with the station administrative procedures and control of access programs.

The BWRX-300 network switches that interconnect the SC3 BOP segment have security features, as described in Subsection 7.2.3. The communications on the SC3 BOP segment UDHs are rigorously controlled and monitored, and information sent outside the plant network is through unidirectional boundary devices, as described in Subsection 7.2.4. The vendor packaged SCN BOP controllers are interfaced to the SC3 BOP segment through SC3 gateways to provide communication protocol conversion and eliminate cyber security vulnerabilities.

7.3.4.3.5 Diversity and Defense-In-Depth

The SCN BOP controller equipment has no specific diversity requirements because it performs no credited protection functions.

7.3.4.4 Operator Interface

The operator interface HSI for the SC3 BOP segment is described in Section 7.5 for the MCR.

7.3.4.5 Compliance Alignment

Table 7.3-8 shows how the SCN BOP controllers align to meet the regulatory guidance in USNRC I&C DRG (Reference 7.3-19) and in the key industry I&C standards IEC 61513 (Reference 7.3-23) and IEEE Std 603-2018 (Reference 7.3-32).

7.3.5 References

7.3-1 IEC 61000-4-2, “Electromagnetic Compatibility – Part 4-2: Testing and Measurement Techniques – Electrostatic Discharge Immunity Test,” International Electrotechnical Commission, 2008.

7.3-2 IEC 61000-4-3, “Electromagnetic Compatibility – Part 4-3: Testing and Measurement Techniques – Radiated, Radio-Frequency Electromagnetic Field Immunity Test,” International Electrotechnical Commission, 2020.

7.3-3 IEC 61000-4-4, “Electromagnetic Compatibility – Part 4-4: Testing and Measurement Techniques – Electrical Fast Transient/burst Immunity Test,” International Electrotechnical Commission, 2012.

7.3-4 IEC 61000-4-5, “Electromagnetic Compatibility – Part 4-5: Testing and Measurement Techniques – Surge Immunity Test,” International Electrotechnical Commission, 2014.

7.3-5 IEC 61000-4-6, “Electromagnetic Compatibility – Part 4-6: Testing and Measurement Techniques – Immunity to Conducted Disturbances, Induced by Radio-Frequency Fields,” International Electrotechnical Commission, 2013.

7.3-6 IEC 61000-4-8, “Electromagnetic Compatibility – Part 4-8: Testing and Measurement Techniques – Power Frequency Magnetic Field Immunity Test,” International Electrotechnical Commission, 2009.

7.3-7 IEC 61000-4-9, “Electromagnetic Compatibility – Part 4-9: Testing and Measurement Techniques – Impulse Magnetic Field Immunity Test,” International Electrotechnical Commission, 2016.

7.3-8 IEC 61000-4-11, “Electromagnetic Compatibility - Part 4-11: Testing and Measurement Techniques - Voltage Dips, Short Interruptions and Voltage Variations Immunity Tests,” International Electrotechnical Commission, 2020.

7.3-9 IEC 61000-4-12, “Electromagnetic Compatibility - Part 4-12: Testing and Measurement Techniques - Ring Wave Immunity Test,” International Electrotechnical Commission, 2017.

7.3-10 IEC 61000-4-13, “Electromagnetic Compatibility - Part 4-13: Testing and Measurement Techniques - Harmonics and Interharmonics Including Mains Signaling at A.C. Power Port, Low Frequency Immunity Tests,” International Electrotechnical Commission, 2002.

7.3-11 IEC 61000-4-14, “Electromagnetic Compatibility - Part 4-14: Testing and Measurement techniques - Voltage fluctuation immunity test for equipment with input current not exceeding 16 A per phase,” International Electrotechnical Commission, 2009.

7.3-12 IEC 61000-4-16, “Electromagnetic Compatibility - Part 4-16: Testing and Measurement Techniques - Test for Immunity to Conducted, Common Mode Disturbances in the Frequency Range 0 Hz to 150 kHz,” International Electrotechnical Commission, 2015.

7.3-13 IEC 61000-4-18, “Electromagnetic Compatibility – Part 4-18: Testing and Measurement Techniques – Damped Oscillatory Wave Immunity Test,” International Electrotechnical Commission, 2019.

7.3-14 IEC 61000-4-19, “Electromagnetic Compatibility – Part 4-19: Testing and Measurement Techniques – Test for Immunity to Conducted, Differential Mode Disturbances and Signaling in the Frequency Range 2 kHz to 150 kHz at A.C. Power Ports,” International Electrotechnical Commission, 2014.

7.3-15 IEC 61000-4-28, “Electromagnetic Compatibility – Part 4-28: Testing and Measurement Techniques - Variation of Power Frequency, Immunity Test for Equipment with Input Current Not Exceeding 16 A Per Phase,” International Electrotechnical Commission, 2002.

7.3-16 IEC 61000-4-31, “Electromagnetic Compatibility – Part 4-31: Testing and Measurement Techniques – AC Mains Ports Broadband Conducted Disturbance Immunity Test,” International Electrotechnical Commission, 2016.

7.3-17 IEC 61000-4-34, “Electromagnetic Compatibility – Part 4-34: Testing and Measurement Techniques – Voltage Dips, Short Interruptions and Voltage Variations Immunity Tests for Equipment with Mains Current More Than 16 A Per Phase,” International Electrotechnical Commission, 2005.

7.3-18 NEDO-33911-A, “BWRX-300 Containment Performance,” Revision 3, January 2022.

7.3-19 “Design Review Guide (DRG): Instrumentation and Controls for Non-Light-Water Reactor (Non-LWR) Reviews,” U.S. Nuclear Regulatory Commission, February 2021.

7.3-20 IEC 63147/IEEE 497, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations,” International Electrotechnical Commission/Institute of Electrical and Electronics Engineers, 2017/2016.

7.3-21 IEC 61888, “Nuclear Power Plants – Instrumentation Important to Safety – Determination and Maintenance of Trip Setpoints,” International Electrotechnical Commission, 2002.

7.3-22 IEC 60987, “Nuclear Power Plants – Instrumentation and Control Important to Safety – Hardware Requirements,” International Electrotechnical Commission, 2021.

7.3-23 IEC 61513, “Nuclear Power Plants – Instrumentation and Control Important to Safety – General Requirements for Systems,” International Electrotechnical Commission, 2011.

7.3-24 IEC 61226, “Nuclear Power Plants – Instrumentation, Control and Electrical Power Systems Important to Safety – Categorization of Functions and Classification of Systems,” International Electrotechnical Commission, 2020.

7.3-25 IEC 60880, “Nuclear Power Plants – Instrumentation and Control Systems Important to Safety – Software Aspects for Computer-Based Systems Performing Category A Functions,” International Electrotechnical Commission, 2006.

7.3-26 IEC/IEEE 60780-323, “Nuclear Facilities – Electrical Equipment Important to Safety – Qualification,” International Electrotechnical Commission/Institute of Electrical and Electronics Engineers, 2016.

7.3-27 IEC/IEEE 60980-344, “Nuclear Facilities – Equipment Important to Safety – Seismic Qualification,” International Electrotechnical Commission/Institute of Electrical and Electronics Engineers, 2020.

7.3-28 IEC 60671, “Nuclear Power Plants – Instrumentation and Control Systems Important to Safety – Surveillance Testing,” International Electrotechnical Commission, 2007.

7.3-29 IEC 62385, “Nuclear Power Plants – Instrumentation and Control Important to Safety – Methods for Assessing the Performance of Safety System Instrument Channels,” International Electrotechnical Commission, 2007.

7.3-30 IEC 61500, “Nuclear Power Plants – Instrumentation and Control Systems Important to Safety – Data Communication in Systems Performing Category A Functions,” International Electrotechnical Commission, 2018.

7.3-31 IEC 60709, “Nuclear Power Plants – Instrumentation, Control and Electrical Power Systems Important to Safety – Separation,” International Electrotechnical Commission, 2018.

7.3-32 IEEE Std 603, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” Institute of Electrical and Electronics Engineers, 2018.

7.3-33 IEC 62138, “Nuclear Power Plants - Instrumentation and Control Systems Important to Safety - Software Aspects for Computer-Based Systems Performing Category B or C Functions,” International Electrotechnical Commission, 2018.

Table 7.3-1: Safety Category 1 Instrumentation and Control Functions and Initiation Signals

Initiating Signal	Reactor Mode Switch Position
Hydraulic Scram	
HIGH RPV pressure [HP1]	RUN or STARTUP
LOW RPV pressure [LP1]	RUN
LOW RPV water level [L3]	RUN or STARTUP
HIGH APRM neutron flux [RUN setpoint]	RUN
HIGH STP	RUN
HIGH containment pressure	RUN or STARTUP
HIGH APRM neutron flux [STARTUP setpoint]	STARTUP
Indication of a line break (Main Steam Line (MSL) or FW)	RUN, STARTUP, or SHUTDOWN
Isolation Condenser Actuation	
HIGH RPV pressure [HP2] (ICS Train 1 actuation)	RUN, STARTUP, or SHUTDOWN
HIGH RPV pressure [HP3] (ICS Train 2 actuation)	RUN, STARTUP, or SHUTDOWN
HIGH RPV pressure [HP4] (ICS Train 3 actuation)	RUN, STARTUP, or SHUTDOWN
LOW RPV water level [L2]	RUN, STARTUP, or SHUTDOWN
HIGH containment pressure	RUN, STARTUP, or SHUTDOWN
Indication of a line break indication (MSL or FW)	RUN, STARTUP, or SHUTDOWN
RPV and Containment Isolation	
LOW RPV water level [L2]	RUN, STARTUP, or SHUTDOWN
HIGH containment pressure	RUN, STARTUP, or SHUTDOWN
MSRIV and MSCIV Isolation	
LOW RPV pressure [LP1]	RUN
LOW RPV water level [L2]	RUN, STARTUP, or SHUTDOWN
Indication of a MSL break	RUN, STARTUP, or SHUTDOWN
Indication of a FW line break	RUN, STARTUP, or SHUTDOWN
Indication of a SDC line break	RUN, STARTUP, or SHUTDOWN
FWRIV and FWCIV Isolation	
HIGH RPV water level [L9]	RUN, STARTUP, or SHUTDOWN
LOW RPV water level [L2]	RUN, STARTUP, or SHUTDOWN

Initiating Signal	Reactor Mode Switch Position
Loss of Normal FW Flow	RUN, STARTUP, or SHUTDOWN
Indication of a FW line break	RUN, STARTUP, or SHUTDOWN
Indication of a SDC line break	RUN, STARTUP, or SHUTDOWN
SDC Isolation	
Indication of a FW line break	RUN, STARTUP, or SHUTDOWN
Indication of a SDC line break	RUN, STARTUP, or SHUTDOWN
CUW Isolation	
Indication of a CUW line break	RUN, STARTUP, or SHUTDOWN
ICS Train Isolation (Train 1)	
Indication of an ICS train 1 line break	RUN, STARTUP, or SHUTDOWN
ICS Train Isolation (Train 2)	
Indication of an ICS train 2 line break	RUN, STARTUP, or SHUTDOWN
Diverse ICS Train Isolation (Train 3)	
Indication of an ICS train 3 line break	RUN, STARTUP, or SHUTDOWN

Table 7.3-2: BWRX-300 Candidate Accident Monitoring Variables

Variable	Type	Selection Reason	Remarks
None	A	BWRX-300 has no specific planned manually controlled actions for safety systems to perform their safety functions	NA
Containment Sump Water Level	B, C	10 CFR 50.34(f)(2)(xvii)	Designated for monitoring FSFs during the implementation of Emergency Operating Procedures (EOPs) and Severe Accident Management Guidelines (SAMGs) and needs to comply with the requirements for safety classifications.
Neutron Flux	B	BWR legacy knowledge	NA
Reactor Pressure	B, C, D	10 CFR 50.34(f)(2)(xvii)	Designated for monitoring FSFs during the implementation of EOPs and SAMGs and needs to comply with the requirements for safety classifications.
RPV Dome Pressure	B	BWR legacy knowledge	Designated for monitoring fundamental safety functions during the implementation of EOPs and SAMGs and needs to comply with the requirements for safety classifications.
Reactor Isolation Valve Positions	B	BWR legacy knowledge	Designated for monitoring FSFs during the implementation of EOPs and SAMGs and needs to comply with the requirements for safety classifications.
RPV Water Level	B	10 CFR 50.34(f)(2)(xviii) 10 CFR 50.34(f)(2)(xxiv)	Designated for monitoring FSFs during the implementation of EOPs and SAMGs and needs to comply with the requirements for safety classifications.
Containment Area Radiation Level	C, E	10 CFR 50.34(f)(2)(xvii) 10 CFR 50.34(f)(2)(xix)	N/A
Containment Hydrogen Concentration	C, F	10 CFR 50.34(f)(2)(xvii) 10 CFR 50.34(f)(2)(xix) 10 CFR 50.44(c)(4) NEDO-33911-A Revision 3, “Licensing Topical Report BWRX-300 Containment Performance,” (Reference 7.3-18)	Provided for severe accident monitoring

Variable	Type	Selection Reason	Remarks
Containment Oxygen Concentration	C, F	10 CFR 50.44(c)(4) NEDO-33911-A Revision 3, (Reference 7.3-18)	Provided for severe accident monitoring
Containment Pressure	C, D	10 CFR 50.34(f)(2)(xvii) 10 CFR 50.34(f)(2)(xix)	Designated for monitoring fundamental safety functions during the implementation of EOPs and SAMGs and needs to comply with the requirements for safety classifications.
Isolation Condenser Condensate Line Flowrate	C, D	10 CFR 50.34(f)(2)(xix)	N/A
Isolation Condenser Steam Line Flowrate	C, D	10 CFR 50.34(f)(2)(xix)	N/A
Off Gas Activity	C	10 CFR 50.34(f)(2)(xix)	Designated for monitoring FSFs during the implementation of EOPs and SAMGs and needs to comply with the requirements for safety classifications.
Boron Injection System Indications	D	BWR legacy knowledge	Not associated with any events requiring environmental or seismic qualification.
Containment Isolation Valve Positions	D	BWR legacy knowledge	Primary accident monitoring information for the containment is provided by reactor pressure, reactor water level, and containment pressure.
Containment Temperature	D	BWR legacy knowledge	N/A
Control Rod Position	D	BWR legacy knowledge	The rod position indication is a normal operating system that is not required to be seismically designed. Its function is completed before experiencing a harsh environment. Also, the proper functioning of the hydraulic scram and CRDs can be inferred from other parameters.
Electrical Power Status	D	BWR legacy knowledge	N/A
FW Flow	D	BWR legacy knowledge	Not associated with any events requiring environmental or seismic qualification
Isolation Condenser Valve Positions	D	BWR legacy knowledge	N/A
ICS Pool Water Level	D	BWR legacy knowledge	N/A

Variable	Type	Selection Reason	Remarks
Area Radiation Level	E	10 CFR 50.34(f)(2)(xix)	N/A
Control Building Air Intake HVS Radiation Level	E	10 CFR 50.34(f)(2)(xix)	N/A
Effluent Radioactivity - Noble Gases	E	10 CFR 50.34(f)(2)(xvii) 10 CFR 50.34(f)(2)(xix)	N/A
Meteorological Data (Wind Speed, Wind Direction, and Atmospheric Stability)	E	10 CFR 50.34(f)(2)(xix)	N/A
Plant Environment Radiation/Radioactivity Levels	E	10 CFR 50.34(f)(2)(xix)	N/A
Spent Fuel Pool Level	F	IEC 63147/IEEE Std 497 (Reference 7.3-20)	Provided for severe accident monitoring

Table 7.3-3: Safety Category 2 Diverse Protection System Functions and Initiating Signals

Initiating Signal	Reactor Mode Switch Position
Diverse Hydraulic Scram	
HIGH RPV pressure	RUN or STARTUP
LOW reactor water level [L3]	RUN or STARTUP
HIGH containment pressure	RUN or STARTUP
HIGH main condenser pressure	RUN
Indication of a line break (MSL or FW)	RUN, STARTUP, or SHUTDOWN
Manual initiation	RUN or STARTUP
CRD Fast Motor Run-In	
High Flux After Scram	RUN or STARTUP
Alternate Rod Insertion	
Hydraulic scram demand	RUN or STARTUP
Isolation Condenser Actuation	
HIGH containment pressure	RUN, STARTUP, or SHUTDOWN
LOW RPV level [L2]	RUN, STARTUP, or SHUTDOWN
Indication of an IC line break (unaffected ICS trains initiated)	RUN, STARTUP, or SHUTDOWN
Indication of a FW line break	RUN, STARTUP, or SHUTDOWN
Indication of a MSL line break	RUN, STARTUP, or SHUTDOWN
High Flux After Scram Signal	RUN or STARTUP
High RPV Pressure	RUN or STARTUP
Manual initiation	RUN, STARTUP, or SHUTDOWN
RPV and Containment Isolation	
HIGH containment pressure	RUN, STARTUP, or SHUTDOWN
LOW RPV level [L2]	RUN, STARTUP, or SHUTDOWN
MSRIV and MSCIV Isolation	
Sustained LOW FW flow	RUN, STARTUP, or SHUTDOWN
Indication of a MSL break	RUN, STARTUP, or SHUTDOWN
Indication of a FW line break	RUN, STARTUP, or SHUTDOWN
Indication of a SDC line break	RUN, STARTUP, or SHUTDOWN
FWRIV and FWCIV Isolation	
HIGH RPV level [L9]	RUN
Loss of normal FW flow	RUN, STARTUP, or SHUTDOWN

Initiating Signal	Reactor Mode Switch Position
Feedwater and Condensate Pump Trip	
Detection of FW line break	RUN, STARTUP, or SHUTDOWN
High Flux after Scram	RUN or STARTUP
Feedwater and SDC Isolation	
Indication of SDC line break	RUN, STARTUP, or SHUTDOWN
Indication of a FW line break	RUN, STARTUP, or SHUTDOWN
CUW Isolation	
Indication of a CUW line break	RUN, STARTUP, or SHUTDOWN
Cavity Pool Makeup from ICS Pools	
RPV level reduction	REFUEL
ICS Train Isolation	
Manual (Per Train)	RUN, STARTUP, or SHUTDOWN

Table 7.3-4: Safety Category 3 Anticipatory Trip and Block Functions and Initiating Signals

Initiating Signal	Reactor Mode Switch Position
Anticipatory Hydraulic Scram	
Turbine Trip Demand	RUN
Closure of any MSRIV or MSCIV (<90% Open)	RUN
Short reactor period (at low power) from WRNM	STARTUP
LOW bus voltage (from SC2/SC3 electrical system)	RUN or STARTUP
HIGH main condenser pressure	RUN
High RPV level	RUN or STARTUP
Sustained LOW FW flow	RUN
Manual initiation	RUN or STARTUP
Control Rod Withdrawal Block	
ATLM	RUN
MRBM	RUN
WRNM Short Reactor Period	STARTUP
Control Rod Separation	RUN, STARTUP, or SHUTDOWN

Table 7.3-5: Safety Class 1 Instrumentation and Control Compliance Alignment

I&C Topic	USNRC I&C Design Review Guide (Reference 7.3-19)
Qualified I&C	X.2.2.2, Qualification Measures X.2.2.2.1, Quality X.2.2.2.2, Equipment Qualification
Reliable I&C	X.1.2, Architecture Assessment X.2.1, Reliability
Robust I&C	X.1.2, Architecture Assessment X.2.2, Robustness X.2.2.1.1, Independence X.2.2.1.2, Redundancy X.2.2.1.3, Diversity in Support of D-in-D to Address CCFs X.2.2.1.4, Predictable and Repeatable Behavior
Secure I&C	X.1.2, Architecture Assessment A.6, Control of Access, Identification, and Repair
Defense-In-Depth	X.1.2, Architecture Assessment X.2.2.1, Defense-in-Depth Measures
Operator Interface	X.1.2, Architecture Assessment X.2.2.1.3, Diversity in Support of Defense-in-Depth to Address CCFs
I&C Performance Objectives	X.1.1, Systematic Assessment Review Criteria X.1.2, Architecture Assessment
I&C Topic	**IEC 61513 Section (Reference 7.3-23)**
Qualified I&C	Sections 5.5.2 and 6.2.2.7
Reliable I&C	Sections 5.4.2.2, 5.4.4.3, 5.5.6, 6.2.2.3.4, 6.2.2.3.5, 6.2.2.3.6, 6.2.3.5, and 6.2.4.2.2
Robust I&C	Sections 5.4.2.6, 5.4.4.2, 6.2.2.3.3, 6.2.3.3.3, and 6.2.3.3.4
Secure I&C	Section 5.5.3
Defense-In-Depth	Section 5.2.2
Operator Interface	Section 5.4.2.3
I&C Performance Objectives	Sections 5.2.2, 5.2.4, and 5.4
I&C Topic	**IEEE Std 603 Clause (Reference 7.3-32)**
Qualified I&C	Clauses 5.3 and 5.4
Reliable I&C	Clauses 5.10, 5.11, 5.12, 5.14, 5.15, 6.5, 6.7, 7.5, and 8.3
Robust I&C	Clauses 5.1, 5.2, 5.5, 5.6, 5.7, 5.8, 5.13, 6.1, 6.3, 6.6, 7.1, 7.3, and 7.4
Secure I&C	Clause 5.9
Defense-In-Depth	Clauses 5.16, 6.2 and 7.2
Operator Interface	Clauses 5.8, 5.14, 6.2 and 7.2
I&C Performance Objectives	Clauses 4, 6.4, and 6.8

Table 7.3-6: Safety Class 2 Instrumentation and Control Compliance Alignment

I&C Topic	USNRC I&C Design Review Guide (Reference 7.3-19)
Qualified I&C	X.2.2.2, Qualification Measures X.2.2.2.1, Quality X.2.2.2.2, Equipment Qualification
Reliable I&C	X.1.2, Architecture Assessment X.2.1, Reliability
Robust I&C	X.1.2, Architecture Assessment X.2.2, Robustness X.2.2.1.1, Independence X.2.2.1.2, Redundancy X.2.2.1.3, Diversity in Support of Din-D to Address CCFs X.2.2.1.4, Predictable and Repeatable Behavior
Secure I&C	X.1.2, Architecture Assessment A.6, Control of Access, Identification, and Repair
Defense-In-Depth	X.1.2, Architecture Assessment X.2.2.1, Defense-in-Depth Measures
Operator Interface	X.1.2, Architecture Assessment X.2.2.1.3, Diversity in Support of Defense-in-Depth to Address CCFs
I&C Performance Objectives	X.1.1, Systematic Assessment Review Criteria X.1.2, Architecture Assessment
I&C Topic	**IEC 61513 Section (Reference 7.3-23)**
Qualified I&C	Sections 5.5.2 and 6.2.2.7
Reliable I&C	Sections 5.4.2.2, 5.4.4.3, 5.5.6, 6.2.2.3.4, 6.2.2.3.5, 6.2.2.3.6, 6.2.3.5, and 6.2.4.2.2
Robust I&C	Sections 5.4.2.6, 5.4.4.2, 6.2.2.3.3, 6.2.3.3.3, and 6.2.3.3.4
Secure I&C	Section 5.5.3
Defense-In-Depth	Section 5.2.2
Operator Interface	Section 5.4.2.3
I&C Performance Objectives	Sections 5.2.2, 5.2.4, and 5.4
I&C Topic	**IEEE Std 603 Clause (Reference 7.3-32)**
Robust I&C	Clauses 5.6 and 6.3
Defense-In-Depth	Clauses 5.16, 6.2 and 7.2
I&C Performance Objectives	Clauses 4, 6.4, and 6.8

Table 7.3-7: Safety Class 3 Instrumentation and Control Compliance Alignment

I&C Topic	USNRC I&C Design Review Guide (Reference 7.3-19)
Qualified I&C	X.2.2.2, Qualification Measures X.2.2.2.1, Quality X.2.2.2.2, Equipment Qualification
Reliable I&C	X.1.2, Architecture Assessment X.2.1, Reliability
Robust I&C	X.1.2, Architecture Assessment X.2.2, Robustness X.2.2.1.1, Independence X.2.2.1.2, Redundancy X.2.2.1.3, Diversity in Support of D-in-D to Address CCFs X.2.2.1.4, Predictable and Repeatable Behavior
Secure I&C	X.1.2, Architecture Assessment A.6, Control of Access, Identification, and Repair
Defense-In-Depth	X.1.2, Architecture Assessment X.2.2.1, Defense-in-Depth Measures
Operator Interface	X.1.2, Architecture Assessment X.2.2.1.3, Diversity in Support of Defense-in-Depth to Address CCFs
I&C Performance Objectives	X.1.1, Systematic Assessment Review Criteria X.1.2, Architecture Assessment
I&C Topic	**IEC 61513 Section (Reference 7.3-23)**
Qualified I&C	Sections 5.5.2 and 6.2.2.7
Reliable I&C	Sections 5.4.2.2, 5.4.4.3, 5.5.6, 6.2.2.3.4, 6.2.2.3.5, 6.2.2.3.6, 6.2.3.5, and 6.2.4.2.2
Robust I&C	Sections 5.4.2.6, 5.4.4.2, 6.2.2.3.3, 6.2.3.3.3, and 6.2.3.3.4
Secure I&C	Section 5.5.3
Defense-In-Depth	Section 5.2.2
Operator Interface	Section 5.4.2.3
I&C Performance Objectives	Sections 5.2.2, 5.2.4, and 5.4
I&C Topic	**IEEE Std 603 Clause (Reference 7.3-32)**
Robust I&C	Clauses 5.6 and 6.3
I&C Performance Objectives	Clauses 4, 6.4, and 6.8

Table 7.3-8: Non-Safety Class Instrumentation and Control Compliance Alignment

I&C Topic	USNRC I&C Design Review Guide (Reference 7.3-19)
Qualified I&C	X.2.2.2, Qualification Measures X.2.2.2.1, Quality X.2.2.2.2, Equipment Qualification
Reliable I&C	X.1.2, Architecture Assessment X.2.1, Reliability
Robust I&C	X.1.2, Architecture Assessment X.2.2.1.1, Independence
Secure I&C	X.1.2, Architecture Assessment A.6, Control of Access, Identification, and Repair
Defense-In-Depth	X.1.2, Architecture Assessment X.2.2.1, Defense-in-Depth Measures
Operator Interface	X.1.2, Architecture Assessment
I&C Performance Objectives	X.1.1, Systematic Assessment Review Criteria X.1.2, Architecture Assessment

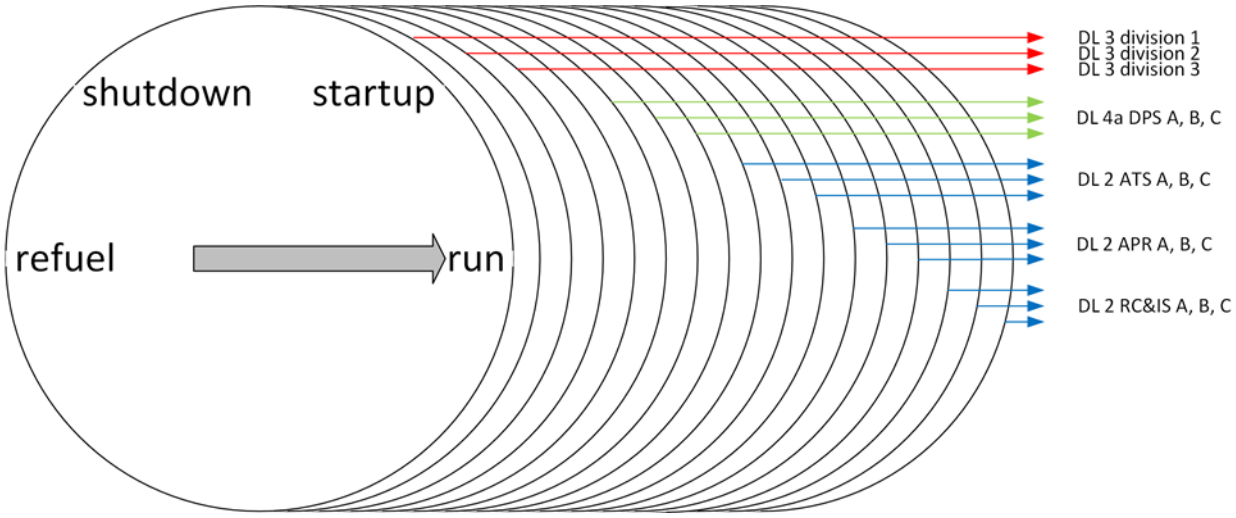


Figure 7.3-1: Reactor Mode Switch

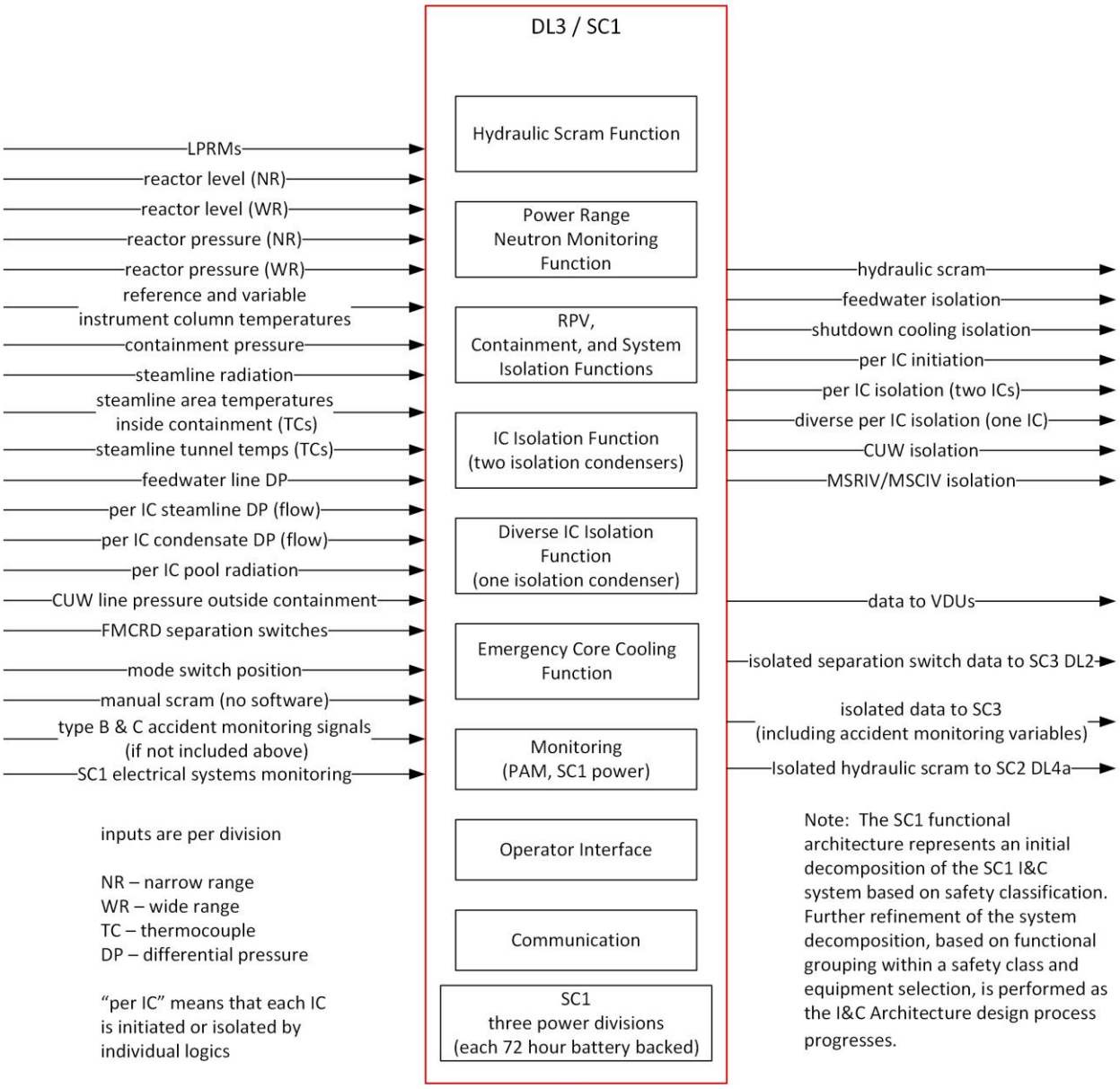


Figure 7.3-2: Safety Class 1 Functional Architecture

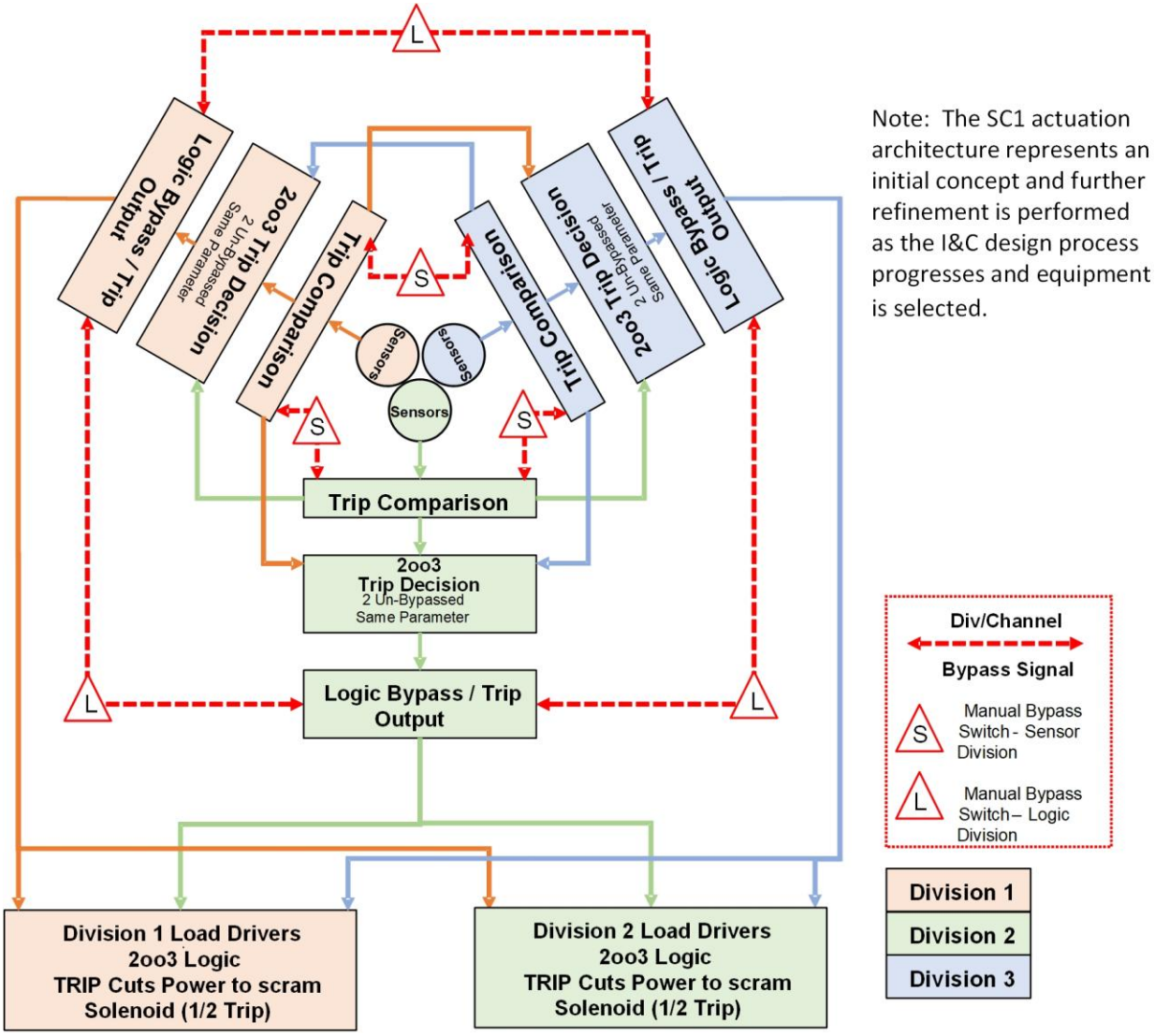


Figure 7.3-3: BWRX-300 Safety Category 1 Actuation Logic

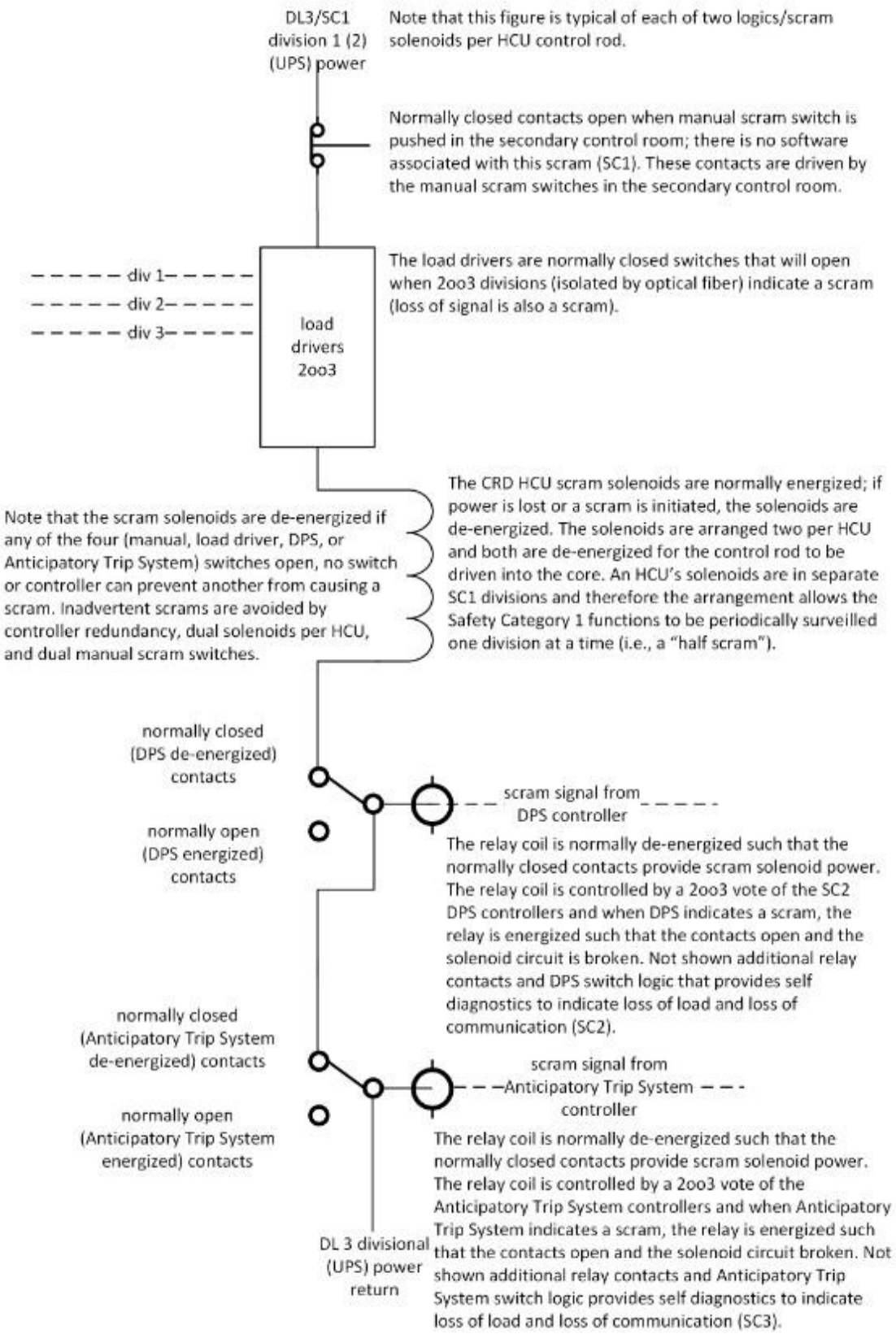


Figure 7.3-4: BWRX-300 Hydraulic Scram Solenoid Control and Power

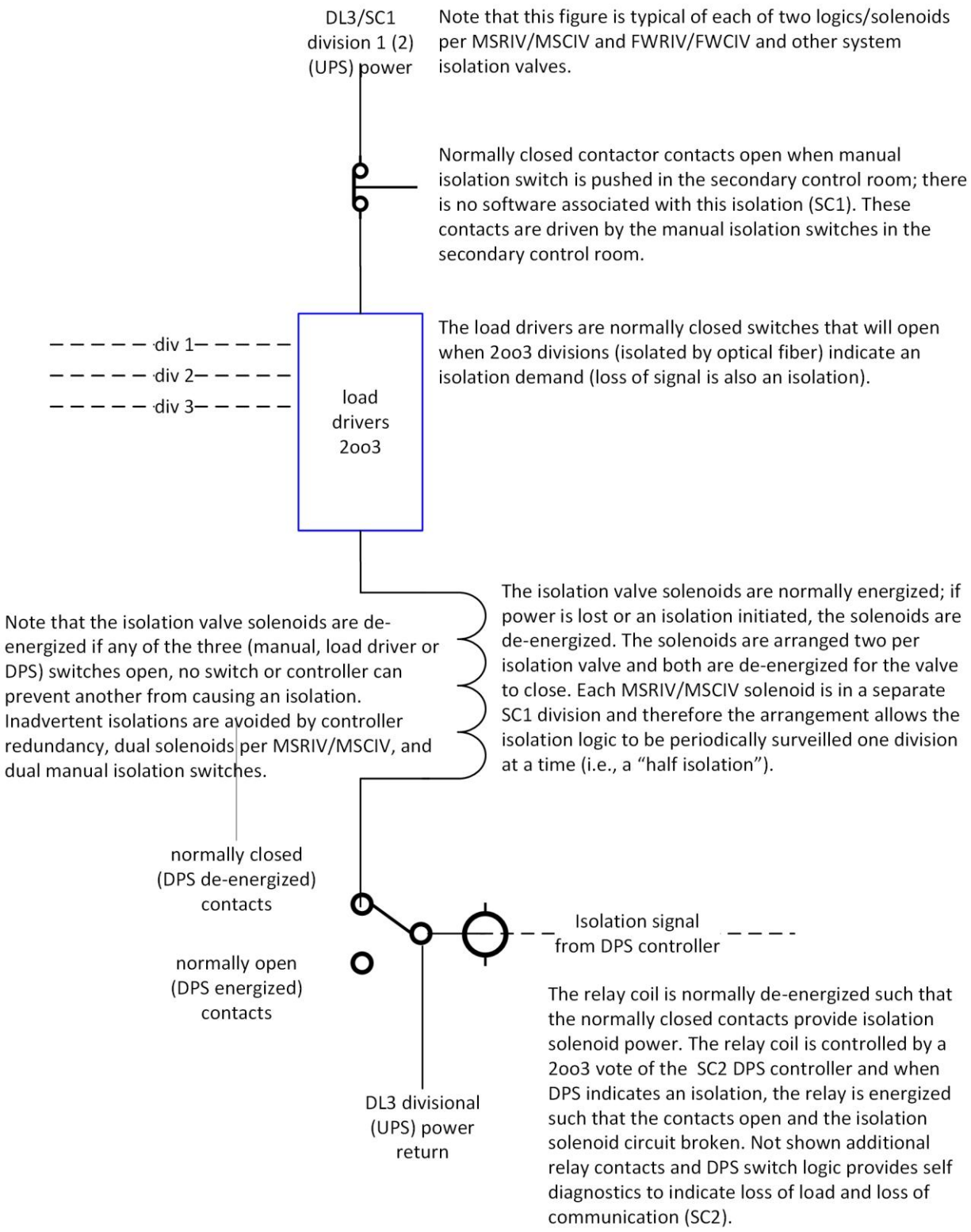


Figure 7.3-5: BWRX-300 Isolation Valve Solenoid Control and Power

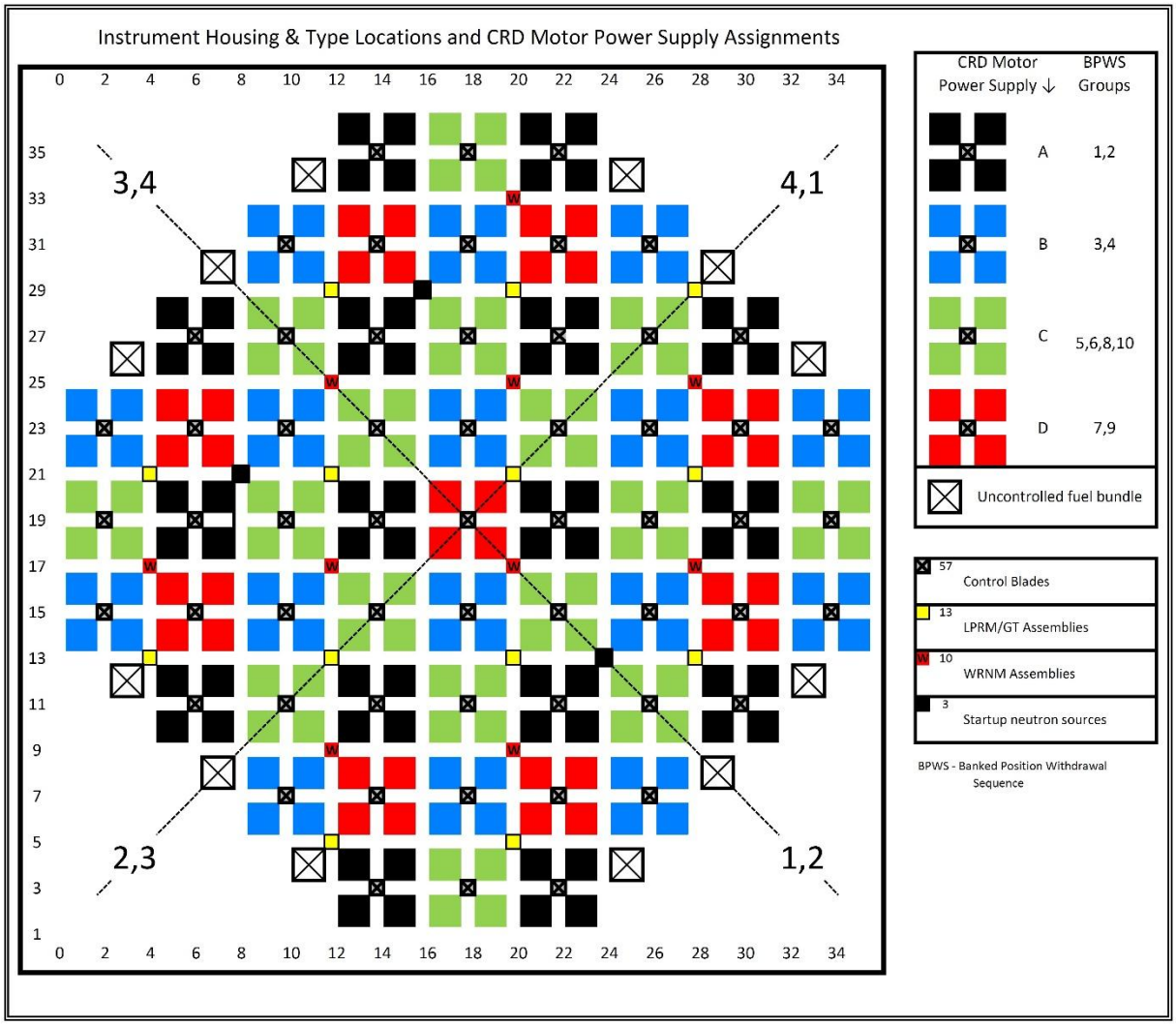


Figure 7.3-6: BWRX-300 Core Map with Monitoring Locations and Fine Motion Control Rod Drive Power Assignments

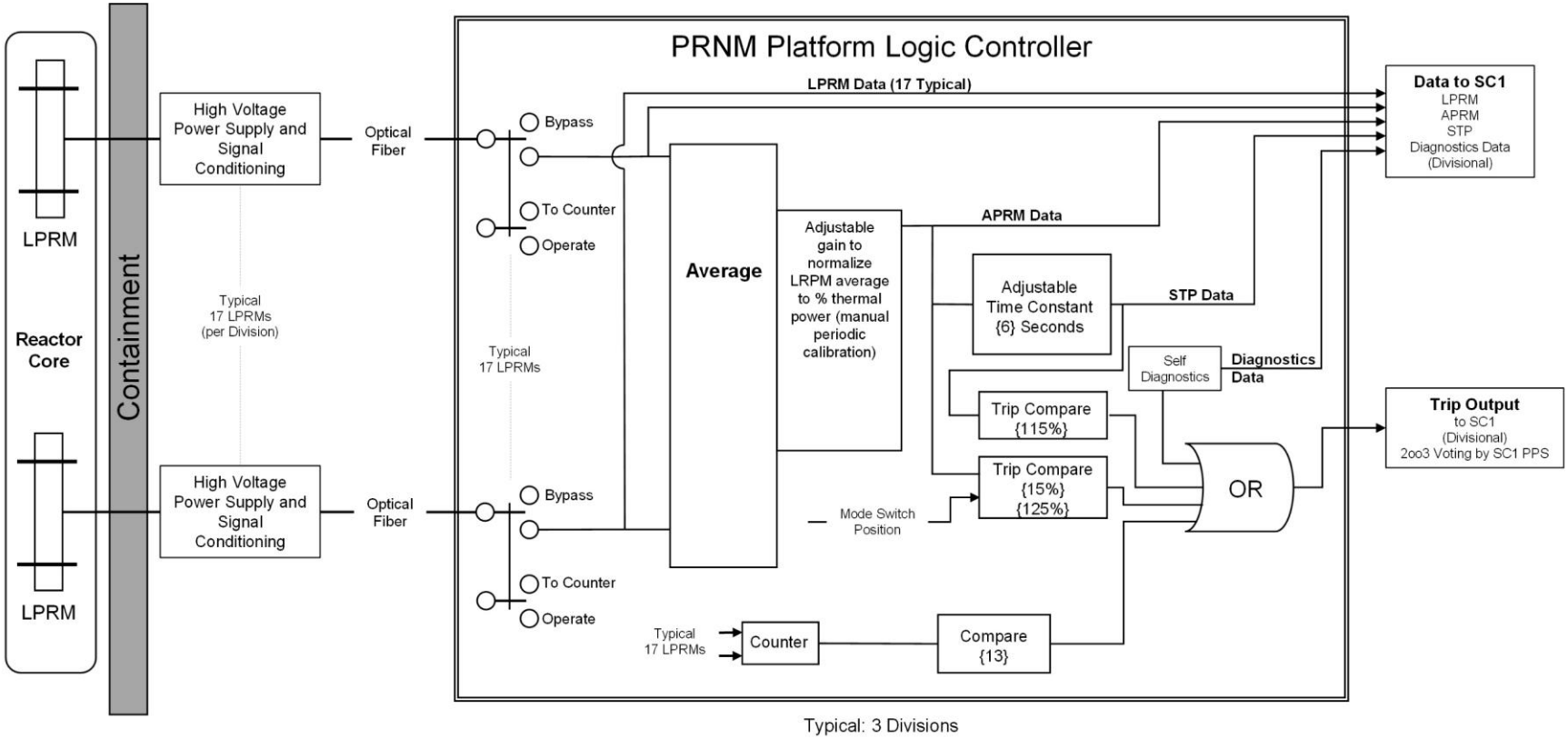


Figure 7.3-7: Representative Safety Class 1 Power Range Neutron Monitoring Block Diagram

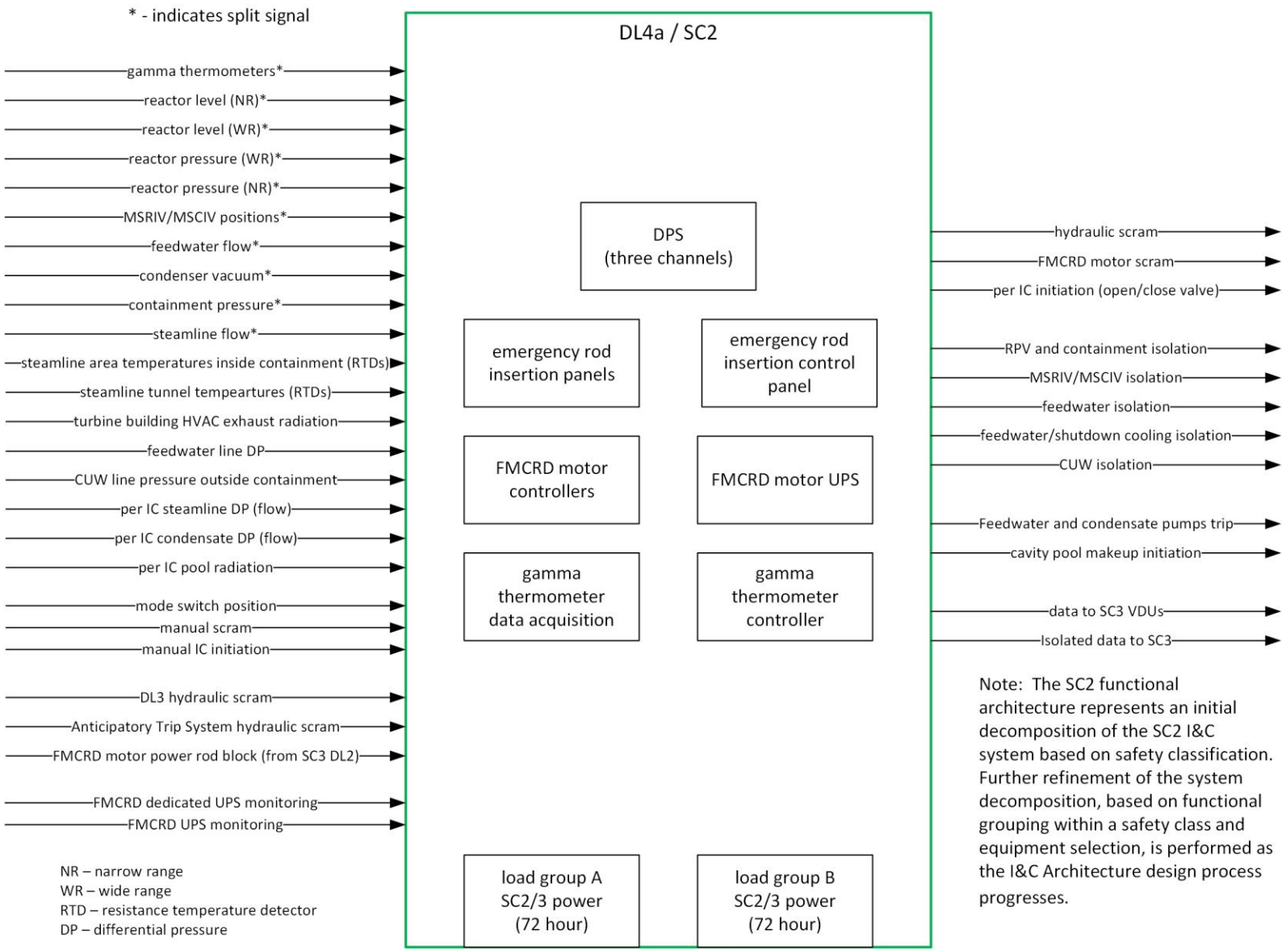


Figure 7.3-8: Safety Class 2 Functional Architecture

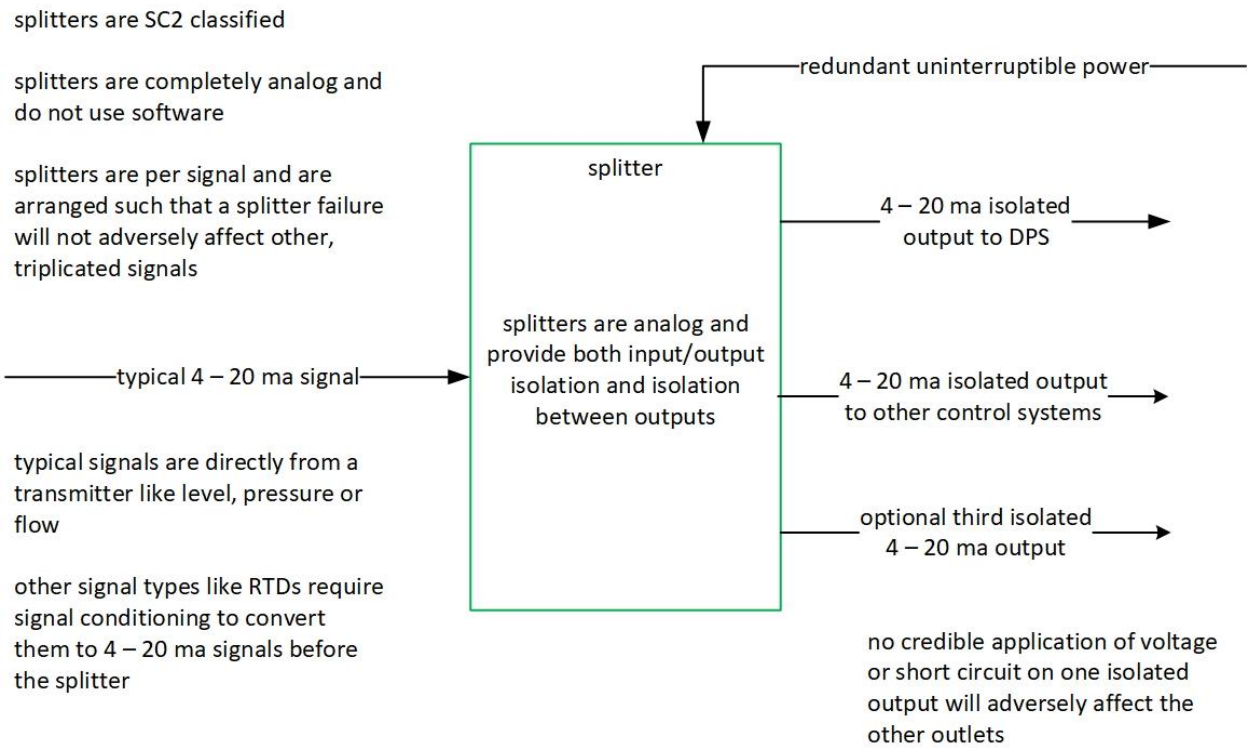


Figure 7.3-9: Safety Class 2 Analog Signal Splitters

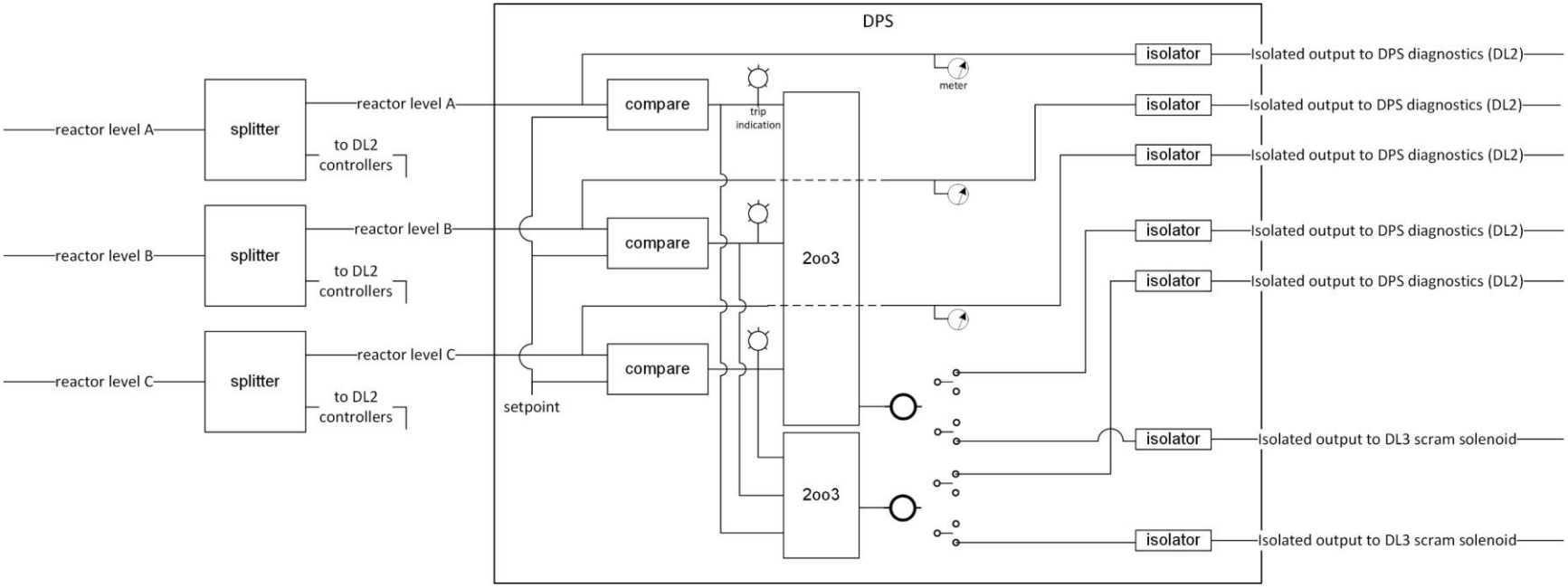


Figure 7.3-10: Representative Diverse Protection System Scram Block Diagram

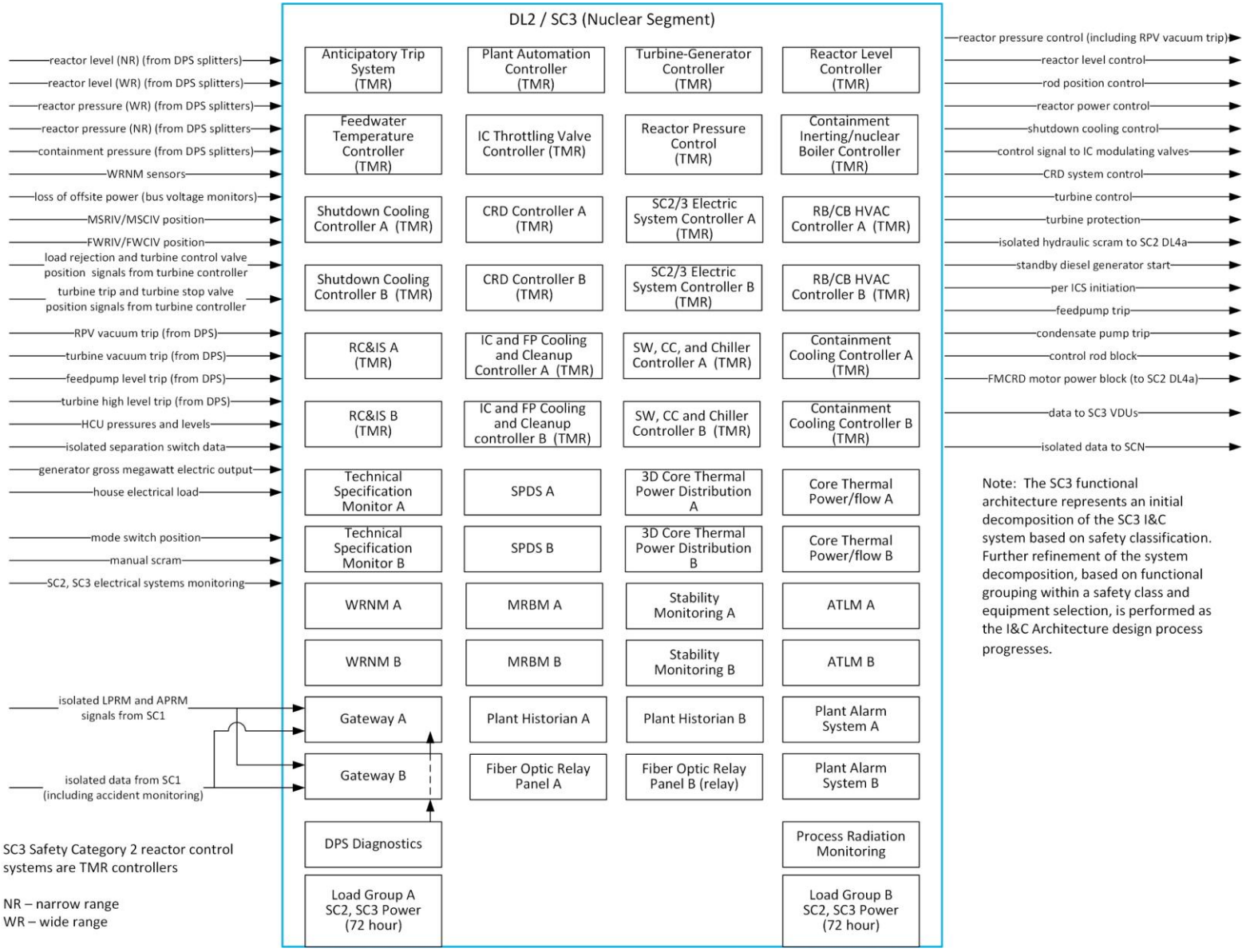


Figure 7.3-11: Safety Class 3 Nuclear Segment Functional Architecture

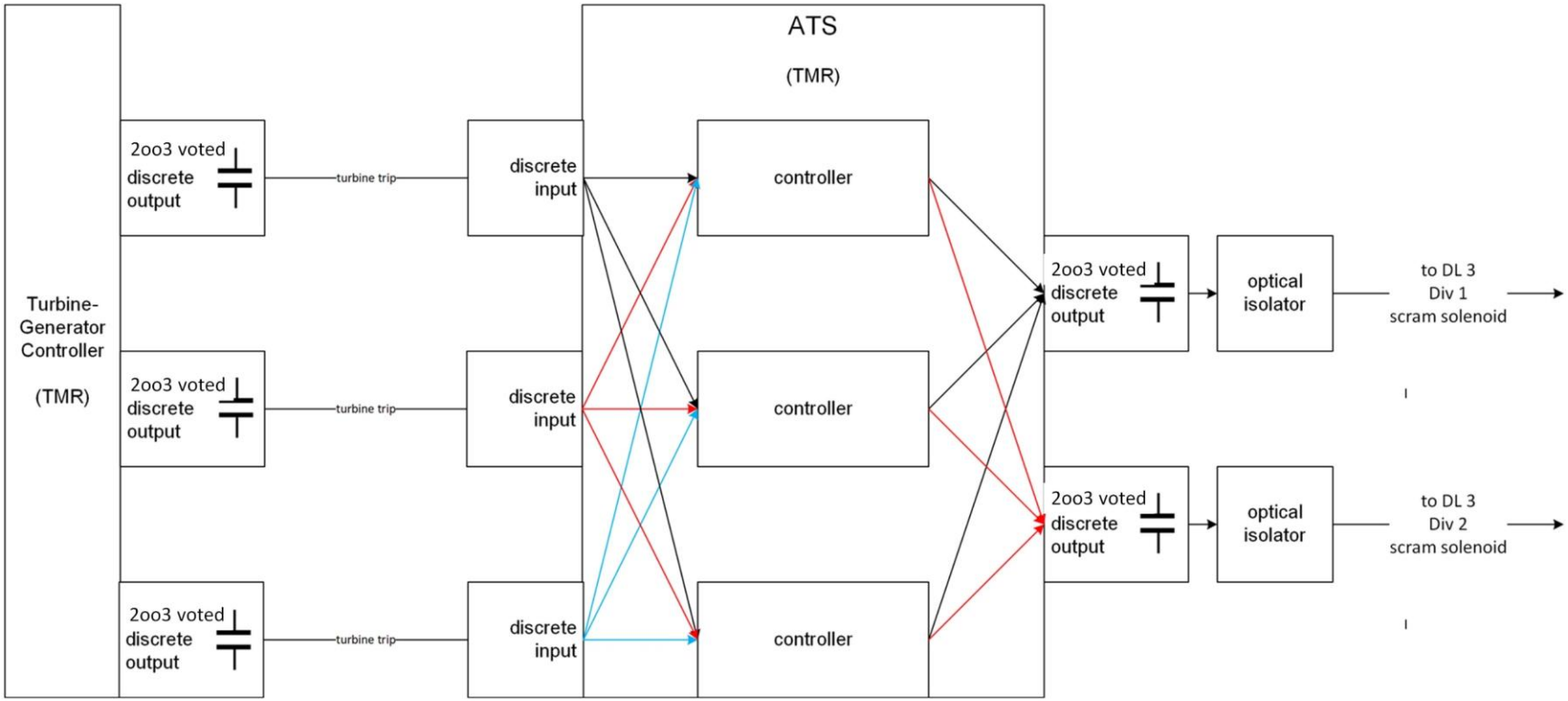


Figure 7.3-12: Representative Anticipatory Trip System Scram Block Diagram

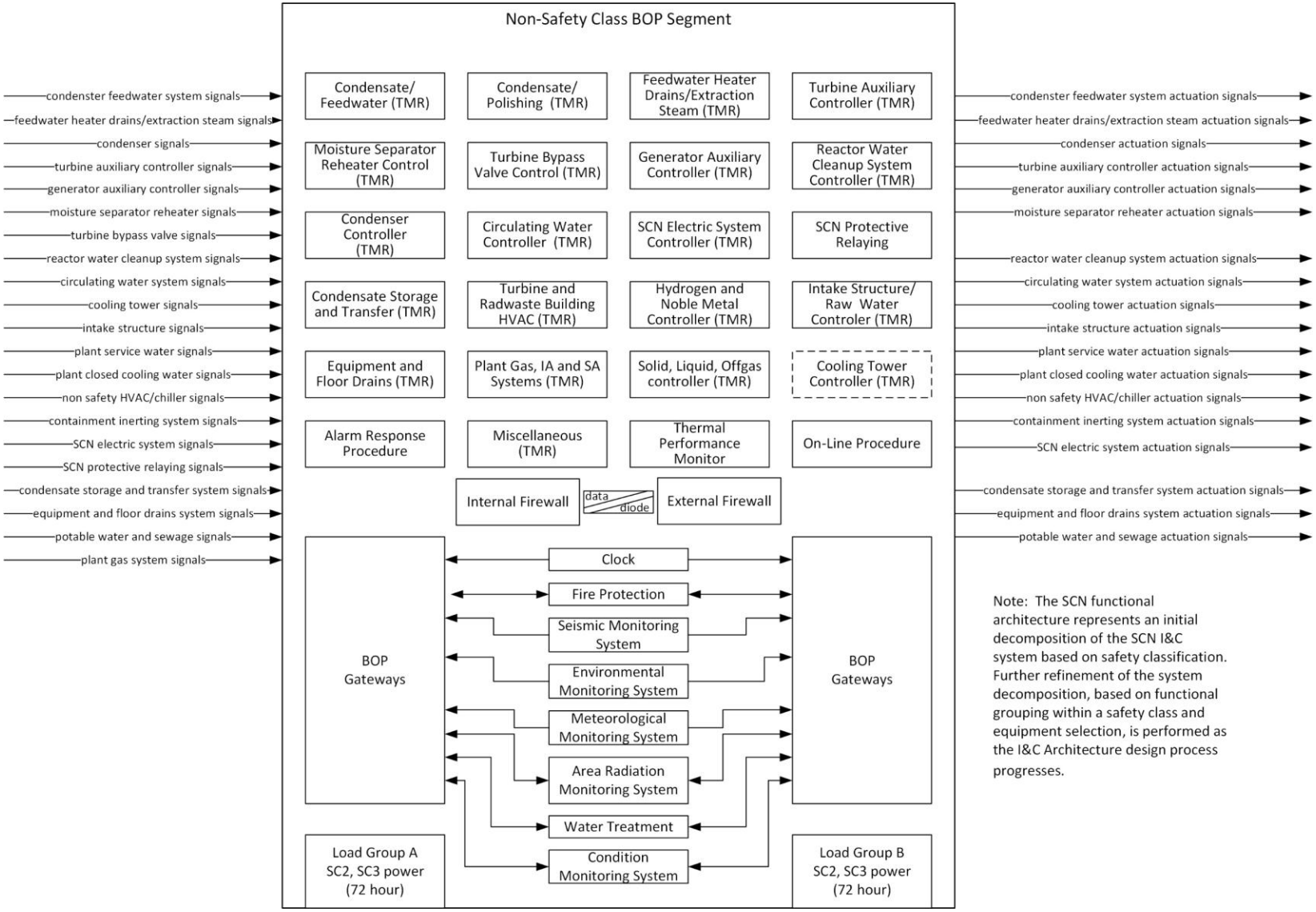


Figure 7.3-13: Non-Safety Class Balance of Plant Segment Functional Architecture

7.4 Digital Instrumentation and Control System Development Processes

The BWRX-300 I&C systems design is the result of a past and continuing top – down systematic design process. This process makes risk-informed decisions about the scope and level of design detail required at various stages of the BWRX-300 design process. The BWRX-300 I&C design process is derived from IEC 61513 (Reference 7.4-1). The BWRX-300 I&C design process is informed by first-hand experience and lessons learned from already licensed and delivered BWR I&C designs and proposed new plant I&C designs (e.g., United States, Finland, Taiwan, and United Kingdom).

7.4.1 Design Control

The BWRX-300 I&C design process is used to ensure that the design and its associated design documentation meet applicable technical requirements, regulatory requirements, codes and standards, and contractual requirements. It is an iterative and recursive design process to allow sequencing and development of requirements. The design process has four design phases (i.e., Requirements, Conceptual, Preliminary, Detailed). The design phases are defined by the maturity of each of the requirement levels, and deliverables are defined for each phase.

1. Baseline 0 Design Phase: The Baseline 0 Design Phase focuses on an abbreviated concept design cycle to more closely examine the design and technology selections suited for demonstration on the desired timeline. This phase focuses on rapid study and design of requirements, systems, and component design alternatives to support timely regulatory submittals, and refinement of the selections into an optimized safety and licensing package. Based on these studies, top-level design requirements are established for the standard plant design based on stakeholder and product requirements, including expected regulatory requirements for desired markets. I&C system requirements are defined and allocated based on high-level conceptual mechanical systems design, and high-level planning documents are prepared. An initial plant level architecture document is prepared.
2. Baseline 1 Design Phase: The Baseline 1 Design Phase focuses on developing the specific design definition of primary SSC. This includes detailed system and, where appropriate, component analysis, specifications, and drawing development, and detailed manufacturing and fabrication analyses for long lead procurement activities. Technical reviews are conducted to confirm the adequacy of design requirements and objectives. System interfaces are established and integrated models, baseline safety analyses, and probabilistic safety analyses to support construct permit application submittals to regulators are developed. System design descriptions are developed for primary systems based on known requirements allocated from identified plant level requirements, interfacing systems, and the overall plant operational concept. At the plant level, the I&C system of systems architecture is developed, and diverse hardware- and software-based implementing control and monitoring platforms are provisionally selected. Various plant level architecture and design documents and diagrams are prepared defining the relationships associated with topics including I&C control and monitoring functions from the reactor and power plant process systems, I&C functions to I&C systems or groups, I&C systems to I&C equipment or platforms, and interconnectivity “from-to” cable and wiring route tables between major I&C equipment or platform cabinets as well as their physical characteristics (e.g., space, weight, electric power demand, heat generation load, and location in the buildings, floors, and rooms).
3. Baseline 2 Design Phase: The Baseline 2 Design Phase focuses on the effort to complete the standard plant design and prepare for construction planning, detailed component design, and support for equipment procurement/fabrication required at this stage of design. Key system and component design documents and calculations of record is to be issued to support

licensing and construction, and equipment specifications are finalized for procurement. Design verification is performed to support submittal of the operating license application.

4. Baseline 3 Design Phase: The Baseline 3 Design Phase focuses on finalization of all remaining system and component design in preparation for construction activities. The remaining equipment specifications are finalized and executed for procurement, and all remaining system and component design documents and calculations of record are issued. This phase includes the detailed efforts necessary to apply the standard plant design to a specific project (e.g., satisfaction of local codes and engineering requirements for stamping of drawings and documents, plant/customer-specific equipment identification for turnover to customer operating systems, and finalization of commodity purchases and construction details). Updates may be needed to component specifications and datasheets based on feedback from selected vendors, and these are evaluated for application to the standard plant design.

7.4.2 Defense-In-Depth and Architecture Design

A structured, formal approach to defining the overall I&C architecture is a prerequisite for successful licensing of a new nuclear plant design. The development process output includes a definition of each I&C system in terms of its assigned functions, its safety classification, and its relationships to other systems. The design of the I&C architecture provides a top-level definition of the I&C systems of the BWRX-300 and the communication between BWRX-300 systems. The design process and supporting tools ensure a consistent interface between these systems. The BWRX-300 I&C architecture design process is shown in Figure 7.4-1.

7.4.2.1 Baseline 0 Phase and Activities

The requirements which define the necessary functions to control, operate, or monitor are specified. An I&C functional requirement is defined in such a way that it:

- Gives a complete representation of a functional objective
- Can be categorized according to its degree of importance to safety

Inputs to the I&C functional requirements include:

- Process and electrical system design descriptions
- Transient and accident analyses
- Human factors analyses

Requirements that define the necessary independence and diversity for I&C systems are specified. An I&C architectural requirement is defined in such a way that it specifies a safety classification and level of redundancy for each I&C system.

Inputs to the architectural requirements include:

- Transient and accident analyses
- Severe accident analyses
- D-in-D concept for the plant
- Applicable regulations, guidance, and endorsed standards
- Human factors analyses

These are the constraints placed on the BWRX-300 I&C architecture from external influences. Examples of external influences include:

- Plant building and room layouts
- Process system interfaces
- Support system designs
- Environmental conditions
- Sources of internal and external hazards
- Concept for online maintenance
- Human factors constraints
- Cyber security constraints

Requirements are defined and allocated based on high-level conceptual mechanical systems design, and high-level planning documents are prepared. An initial plant level architecture document is prepared.

7.4.2.2 Baseline 1 Phase and Activities

The Allocation of Functions (AOFs) activity involves the definition of a set of criteria to govern the process of allocating I&C functions to I&C systems and the act of allocating the I&C functions following the criteria. This can be done initially at the level of control and monitoring functions but eventually needs to be done for each I&C function. I&C sub-functions are created during this process.

The definition of architecture activity is the top-level definition of I&C systems in terms of the functional scope of each I&C system. At a minimum, the definition of an I&C system comprises its safety classification, its design bases functions, and its position in the lines of defense. This work can also include top-level assignment of I&C systems to physical locations in the plant.

Both the AOF and definition of architecture are iterative, both to each other and as the BWRX-300 plant design progresses.

The definition of interfaces activity represents the organization and documentation of the interfaces created by the architecture definition and function allocation processes. The key elements to document are:

1. Basis (need) for the interface.
2. Top-level nature of the interface (e.g., hardware, data communication, point-to-point, and switched network).
3. Primary requirements on the interface (e.g., independence, diversity, and failure modes).

Interfaces also include support systems for the I&C system.

HSIs are integrated in the architecture and design structure for each system control and monitoring area including the MCR, SCR, local control panels, and any emergency control location. The design of the I&C architecture implements the principles for plant operation established in the BWRX-300 plant design including:

1. Requirements between automatic signals and manually initiated control signals.
2. Requirements between the different HSI systems during normal, accident, and post-accident operation.

3. Requirements between normal and backup HSI systems (including any switchover rules).

7.4.3 Instrumentation and Control System Life Cycle

The I&C system life cycle applied to each I&C system follows the overall lifecycle of IEC 61513 (Reference 7.4-1) and the system engineering "V" model to show top-down design and bottom-up integration for Verification and Validation (V&V) testing, as shown in Figure 7.4-2. This plan includes the project timeline on the X axis and vertical level project progression with V&V arrows included. Configuration management requirements for the project are defined in the BWRX-300 configuration management plan. The requirements management plan for the project provides direction for managing requirements appearing in project documents developed during execution of the design process.

Each plant level I&C architecture lifecycle phase within the lifecycle contains activities and output documentation aimed at fulfilling the purpose of the phase. The phases are not strictly sequential in their execution; rather, they are iterative. For example, decisions and discoveries made in a 'subsequent' phase can necessitate a return to 'previous' phases to modify requirements, design decisions, or output documentation. In some cases, portions of different phases can be executed in parallel. In general, a 'subsequent' phase can be started prior to completion of the 'previous' phase.

The software-related activities in I&C system life cycle are shown in Figure 7.4-3 from IEC 60880, Figure 2 (Reference 7.4-5).

7.4.4 Cyber Security Life Cycle

The BWRX-300 cyber security program plan is designed to meet regulatory requirements, relevant industry standards and best practices for the design, operation, and protection of the BWRX-300 I&C systems. The BWRX-300 cyber security program plan applies advanced security principles throughout the product development and deployment lifecycle of the BWRX-300 I&C systems. The objective of the program plan is to achieve a high assurance that unauthorized access to the protection, control, and adjustment systems of the BWRX-300 is prevented. The program is based on three key principles:

1. Regulatory compliance.
2. Integration of cyber security controls into the BWRX-300 I&C systems development process.
3. Integration of cyber security features into the BWRX-300 I&C systems.

The BWRX-300 cyber security program plan recognizes that protecting cyber essential assets is an ongoing program throughout the I&C system development and operations stages. It is designed to:

- Implement security controls to protect the necessary digital assets from cyber attacks
- Apply and maintain D-in-D protective strategies to ensure the capability to detect, respond to, and recover from cyber attacks
- Mitigate the adverse effects of cyber attacks
- Ensure functions of protected assets are not adversely affected due to cyber attacks

The cyber security life cycle is shown in Figure 7.4-4. The cyber security architecture security boundaries are shown in Figure 7.4-5. The BWRX-300 cyber security program plan classifies the DCIS systems SC1, SC2, and SC3 based on importance to nuclear safety. These systems are protected from lower classified systems by the unidirectional communication flow between Level 3

and Level 2, as shown in Figure 7.4-5 and consistent with zone requirements in RG 5.71, “Cyber Security Programs for Nuclear Facilities,” (Reference 7.4-2). The BWRX-300 cyber security program plan adds an additional layer of protection between SC1 and SC2 I&C and the SC3 DCIS with unidirectional communication flow between Level 4 and Level 3. Data flow and communications between defense levels are managed by boundary devices that implement unidirectional flow where required to prevent a lower level from compromising a higher level.

7.4.5 Compliance Alignment

This section describes how the BWRX-300 I&C development life cycle aligns to meet the regulatory guidance in USNRC, “DRG: Instrumentation and Controls for Non-Light-Water Reactor (Non-LWR) Reviews,” (Reference 7.4-3) and the key industry I&C standards IEC 61513 (Reference 7.4-1) and IEEE Std 603-2018 (Reference 7.4-4), as shown in Table 7.4-1.

7.4.6 References

7.4-1 IEC 61513, “Nuclear Power Plants – Instrumentation and Control Important to Safety – General Requirements for Systems,” International Electrotechnical Commission, 2011.

7.4-2 Regulatory Guide 5.71, “Cyber Security Programs for Nuclear Facilities,” Revision 1, U.S. Nuclear Regulatory Commission, 2023.

7.4-3 “Design Review Guide (DRG): Instrumentation and Controls for Non-Light-Water Reactor (Non-LWR) Reviews,” U.S. Nuclear Regulatory Commission, February 2021.

7.4-4 IEEE Std 603, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” Institute of Electrical and Electronics Engineers, 2018.

7.4-5 IEC 60880, “Nuclear Power Plants – Instrumentation and Control Systems Important to Safety – Software Aspects for Computer-Based Systems Performing Category A Functions,” International Electrotechnical Commission, 2006.

7.4-6 IEC 62138, “Nuclear Power Plants - Instrumentation and Control Systems Important to Safety - Software Aspects for Computer-Based Systems Performing Category B or C Functions,” International Electrotechnical Commission, 2018.

7.4-7 IEEE Std 7-4.3.2, “IEEE Standard Criteria for Programmable Digital Devices in Safety Systems of Nuclear Power Generating Stations,” Institute of Electrical and Electronics Engineers, 2003.

Table 7.4-1: Digital Instrumentation and Control System Development Process Compliance Alignment

I&C Topic	USNRC I&C Design Review Guide
Qualified I&C	X.2.2.2.1, Quality
I&C Topic	**IEC 61513 Section (Reference 7.4-1)**
Qualified I&C	Sections 5.5.2. and 6.2.2.7 as supplemented by IEC 60880 (Reference 7.4-5) and IEC 62138 (Reference 7.4-6)
I&C Topic	**IEEE Std 603 Clause (Reference 7.4-4)**
Qualified I&C	Clauses 5.3 and 5.4 as supplemented by IEEE Std 7-4.3.2 (Reference 7.4-7)

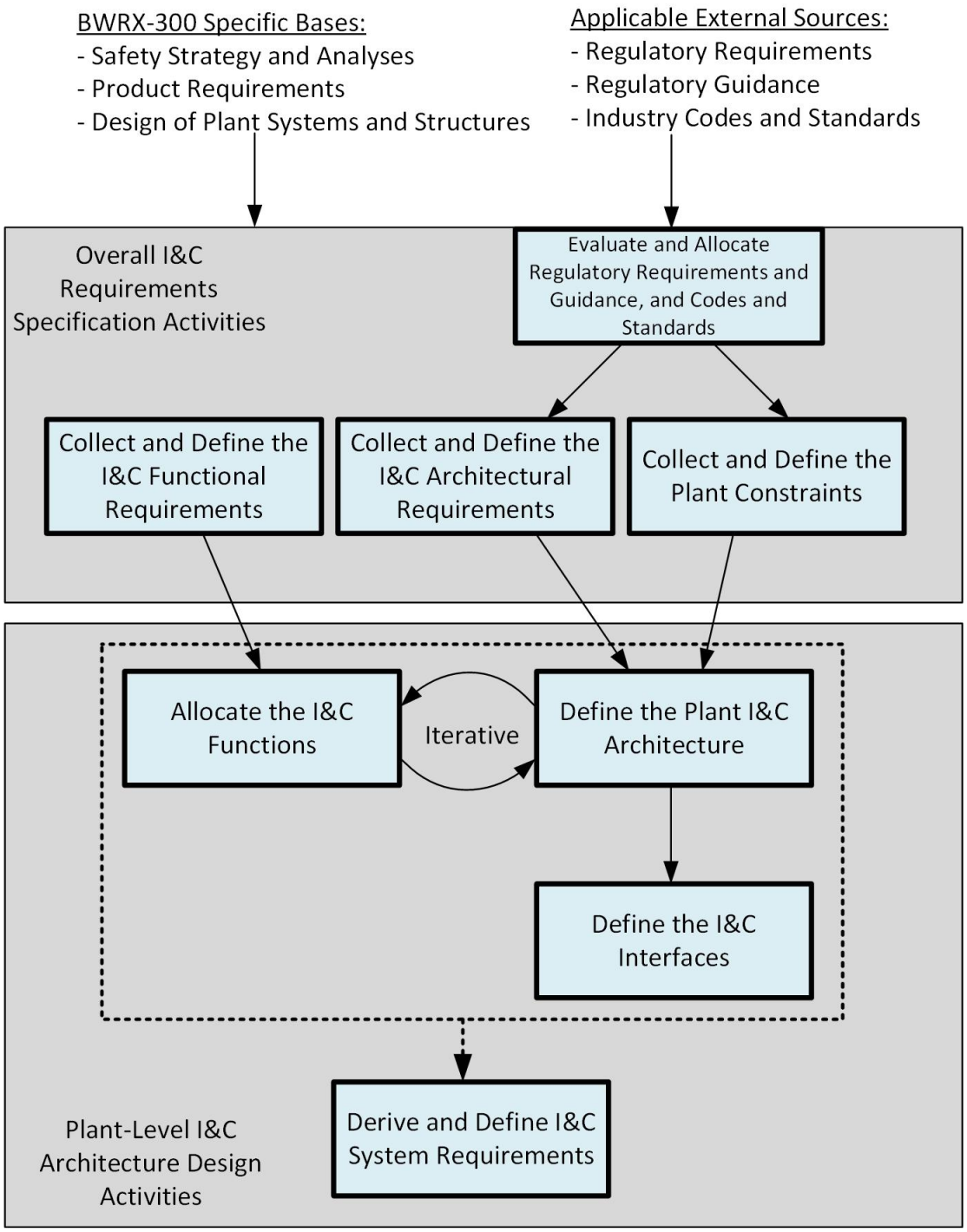


Figure 7.4-1: BWRX-300 Instrumentation and Control Architecture Design Process

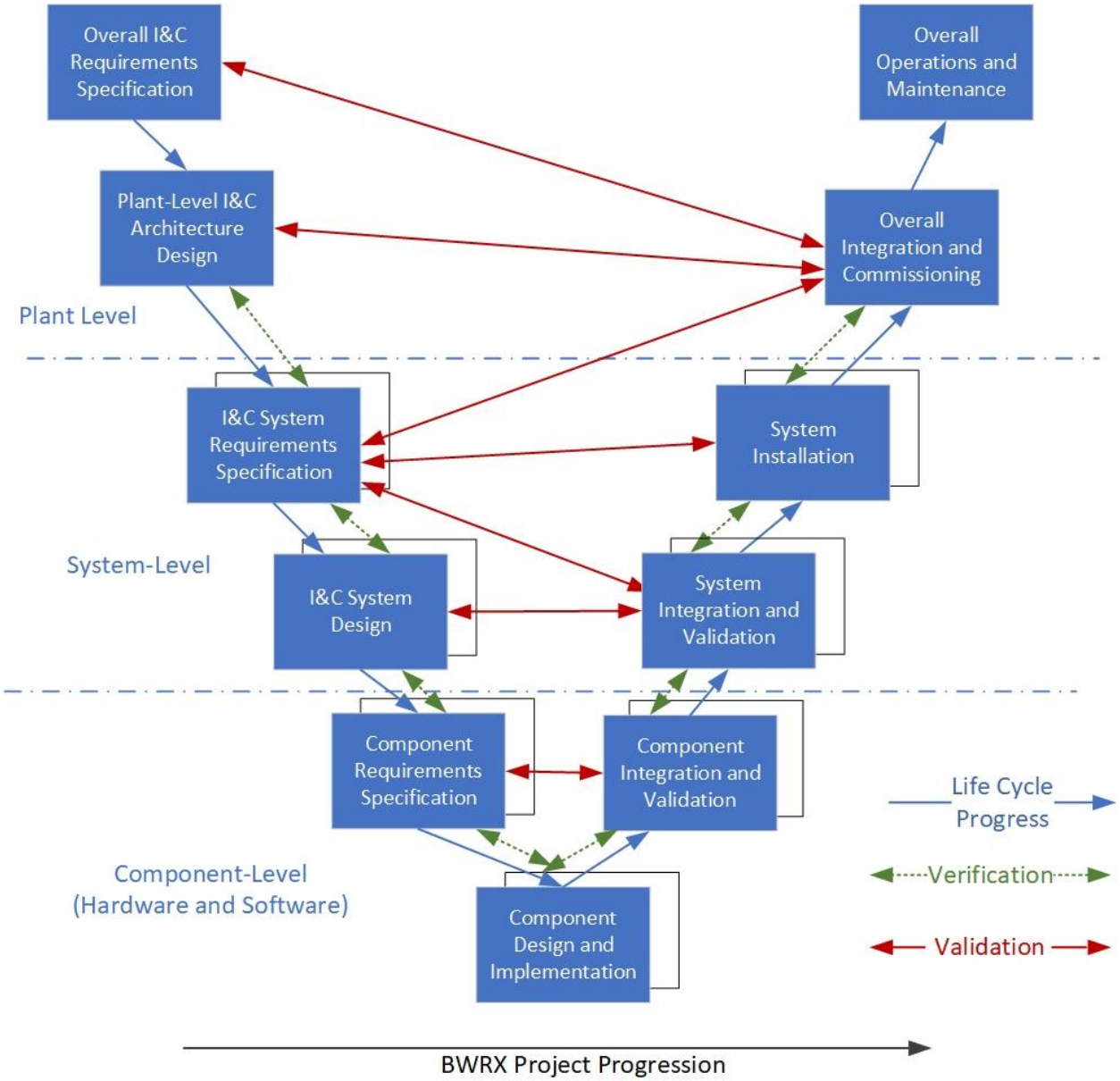


Figure 7.4-2: Overall BWRX-300 Instrumentation and Control System Life Cycle

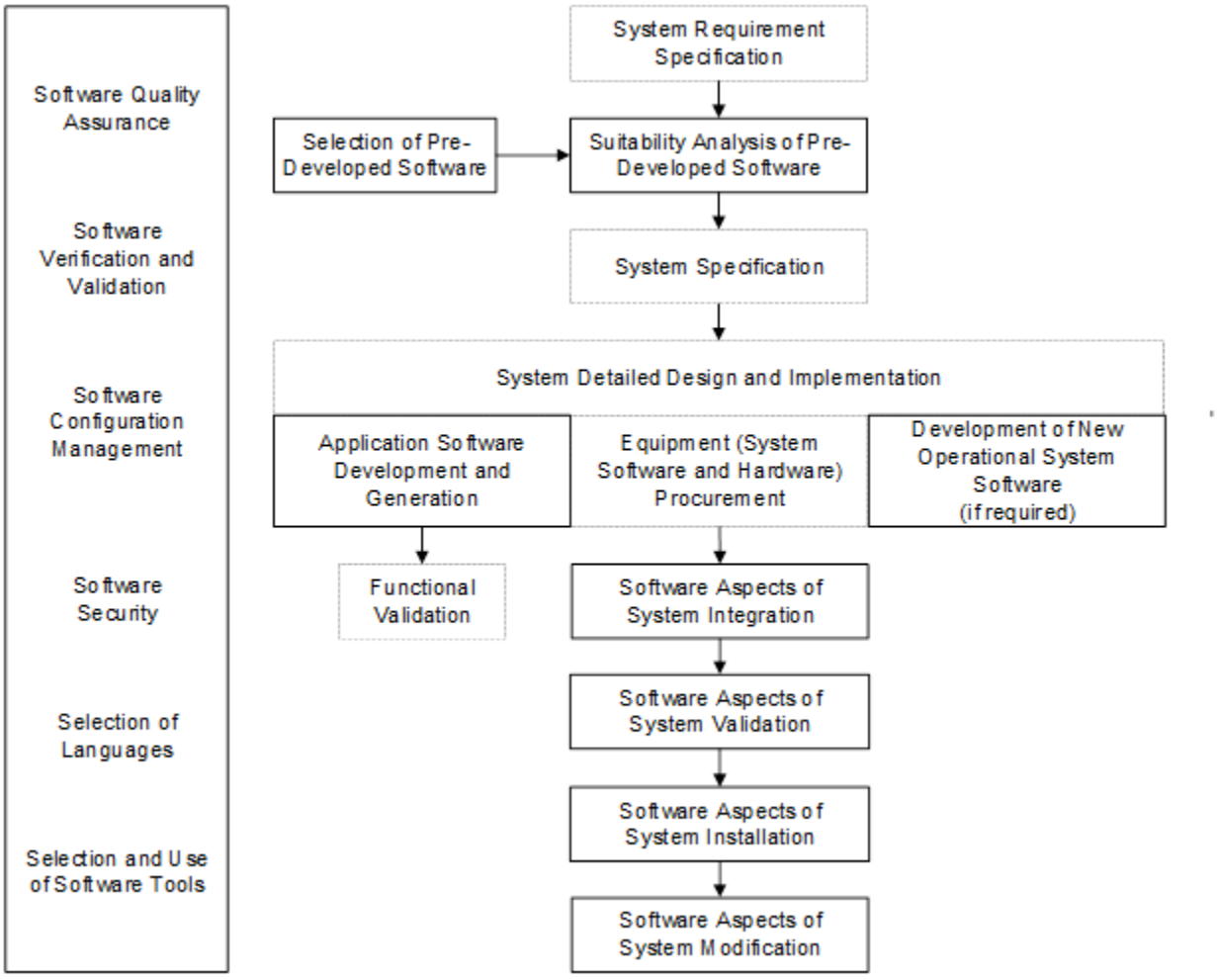


Figure 7.4-3: Software Related Activities in Instrumentation and Control System Life Cycle

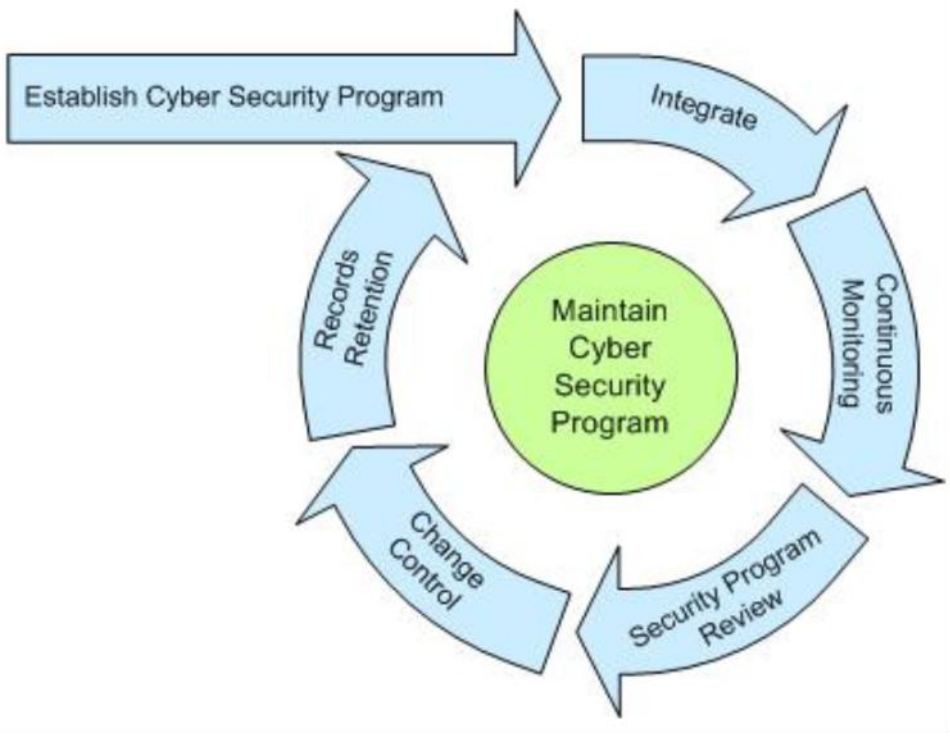


Figure 7.4-4: Cyber Security Life Cycle

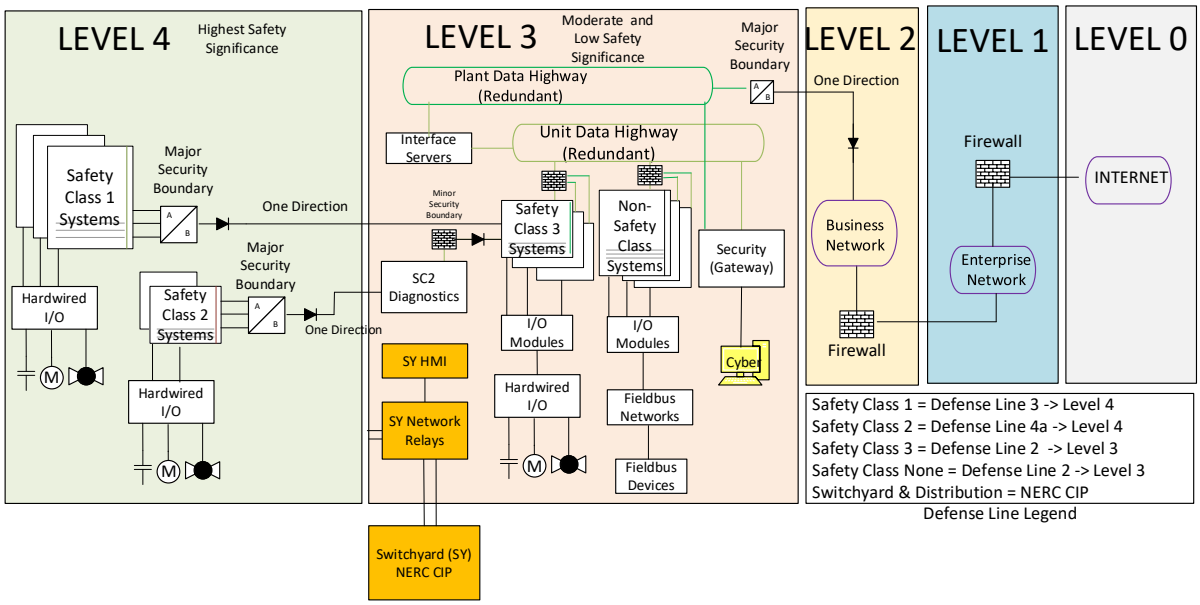


Figure 7.4-5: Cyber Security Architecture Security Boundaries

7.5 Instrumentation and Control in Main Control Room

The MCR is located in the CB. The MCR is the primary location for plant monitoring and control during normal, abnormal, and emergency conditions. The MCR includes controls, indications and alarms that enable operators to perform the defined set of functions during normal operation modes and PIE conditions.

The MCR equipment is designed in accordance with the HFE process as described in Appendix 7A.

7.5.1 Main Control Room Use

The MCR is designed with HSIs to support the following tasks:

1. Assessing the overall status and performance of the plant in any condition and providing necessary information to support operator actions.
2. Monitoring and controlling FSFs.
3. Monitoring the status and trends of key plant parameters (such as reactor power and rates of power change).
4. Operating the plant safely during operational states when the MCR is available, automatically, or manually.
5. Taking measures to maintain the plant in a safe state or to bring it back into a safe state after design basis events and DECs.
6. Maintaining the plant within the specified limits and conditions for the parameters associated with plant systems and equipment.
7. Monitoring for failure of critical instrumentation and equipment.
8. Confirming safety actions for the actuation of safety systems are automatically initiated when needed and that the relevant systems perform as intended.
9. Determining the need and the time for manual initiation or intervention of specified safety actions.
10. Implementing EOPs, emergency mitigating equipment guidelines, and SAMGs.

The MCR is designed to be optimal for nominal shift complement of the control room operator role-holders. The MCR at concept stage is sized and designed to accommodate the expected maximum number of people on a continuous basis. This maximum number is to be based on the MCR nominal staffing plus six other people. The basis for six additional people is to allow sufficient space for the supervisory role and five others (e.g., trainees, observers). The capability for six additional people also provides the necessary facilities to support plant conditions and evolutions that require more than the normal operations complement, for example during preoperational and startup testing. The size and layout of the MCR are expected to be adjusted as the design progresses as a result of the staffing analysis being conducted as part of the BWRX-300 HFE process. The required MCR personnel have workstations designed to support their specific control, information, communication, and work coordination needs. Each MCR workstation also contains space for a role trainee. The MCR is designed in accordance with current international best practice codes and standards for control room design, integrating results from HFE analyses and specified HFE design requirements.

7.5.2 Main Control Room Layout

Figure 7.5-1 represents the concept for the MCR. This concept is expected to evolve with the program of safety analyses and HFE activities being performed as an integral part of the

BWRX-300 design. The analyses are used to develop an MCR design that optimally supports required personnel roles for plant conditions.

Key to the MCR layout design is that tasks take place in positions that do not result in blocking views of the group-view display. The arrangement of the workstations in the MCR also facilitates communication between MCR personnel, including direct conversation and visual contact. MCR design also fosters the oversight and command and control responsibilities of the required supervisory roles. The layout of workstations provides sufficient clearance for foot traffic and maintenance between consoles. The layout of workstations provides enough space for the proper storage, placement, and use of tools and procedures in the MCR. The workstations are typically designed for seated operations; however, it is expected that the BWRX-300 HFE process identifies the correct configuration and height required to ensure optimal visibility of individual and shared workspaces within the MCR.

The MCR group-view display is an array of large VDUs displaying a consolidated overview of information about plant systems. The purpose of the group-view display is to support cohesive situational awareness and crew coordination through shared monitoring of key parameters and alarm information relating to overall plant status, including EOP entry conditions.

The group-view display VDUs are located and designed to support optimal line-of-sight and viewing angles by the necessary subset of MCR users. Group-view display and their content are designed and sized to support monitoring and readability of on-screen component labels and indications to the required MCR users. Group-view display content can be managed at the supervisory workstation via mouse and keyboard interface, including access to closed circuit televisions. Group-view display VDUs do not have plant control functions, as these are designed for monitoring purposes only. Group-view display alarms from SC1, SC2, SC3, and SCN systems are also included on the group-view display VDUs. Additional information about alarms is accessible at the workstation. The group-view display connected to the DCIS is also available at the workstation VDUs. The purpose of the group-view display is to be a supplemental, not a sole source, of information.

The MCR operator workstation is made up of three sections: two SC3 workstation sections and one SC1 workstation section. The MCR operator workstations contain the controls and indications needed to perform the assigned MCR tasks. VDU displays are designed using the BWRX-300 HFE and provide easy and simple navigation to available parameters and controls.

In the MCR concept, the SC1 workstation contains the necessary dedicated VDUs for Primary Protection System displays, for a total of six VDUs. The VDUs are divisionally separated so that information from one division does not populate VDUs of another division. Some VDUs are used primarily for monitoring parameters and overall system status indications and for backing a nonfunctional VDU. Other VDUs are used primarily for controls and confirmatory indications that the control has been actuated, a control signal has been sent, and the control action is having the intended effect on the system. SC1 VDUs are controlled by a touchscreen interface. Hardware and software-based controls are positioned in compliance with specified HFE design requirements and to preserve sufficient laydown areas for procedures.

The SC3 workstations contain the necessary VDUs to support monitoring, control, and alarm management of other plant functions. These VDUs are controlled by an appropriate number of wired keyboards and mouse interfaces. SC3 workstation VDUs are capable of displaying any provided plant and system information, as supported by the I&C architecture. These VDUs only allow for control of equipment with Safety Category 3 functions and equipment with no safety category functions. These VDUs do not allow more than one operator to control the same equipment at the same time.

MCR SC3 workstation VDUs are sized and positioned in compliance with specified HFE design requirements for viewing angles for frequently used indications and alarms, to support readability, and to preserve line-of-sight to group-view display. The arrangement of the SC3 workstations in the MCR facilitates verbal communication.

The MCR SC3 workstations also contain a VDU (each with a mouse and keyboard input device) dedicated for business Local Area Network (LAN) and one of the SC3 workstations contains a VDU (with a mouse and keyboard input device) for the fire protection system. The business LAN is separated from the plant I&C networks through a firewall and data diode. The SC3 workstation also contains space for communications equipment, procedure laydown, and administrative tasks.

The MCR supervisory role workstation is implemented with SC3 equipment and designed with a work area large enough to accommodate three seated people. The MCR supervisory workstation is located behind the SC3 workstation; it is suitably positioned to maintain a clear line-of-sight to the SC3 workstations and support verbal communication with the other workstations in the room. The supervisory workstation is also positioned to maintain a clear line-of-sight to the group-view display. The MCR supervisory workstation contains at least two VDUs with indications (no control) to monitor operations as they are being performed. The MCR supervisory workstation VDUs are controlled by a wired keyboard and mouse interface. The MCR supervisory workstation also contains two VDUs dedicated to the business LAN, as well as space for communications equipment, procedure and drawing laydown, and administrative tasks.

The MCR also contains an emergency communications workspace. The intent of the workspace is to allow designated personnel as required during events to communicate externally to the MCR, in such a way as to not disrupt the personnel focused on monitoring and controlling the plant in response to the event. The emergency communications workspace is located and designed with suitable provisions to minimize disturbance to the operating crew. The emergency communications workstation contains at least one VDU with indications (i.e., no control) from SC1, SC2, SC3, and SCN systems for monitoring plant conditions. The emergency communications workstation also contains at least two VDUs to access the business LAN. The emergency communications workstation VDUs are controlled by a wired keyboard and mouse interface. VDUs are sized and positioned in compliance with specified HFE design requirements. The emergency communications workspace also contains space for required communications equipment and administrative tasks.

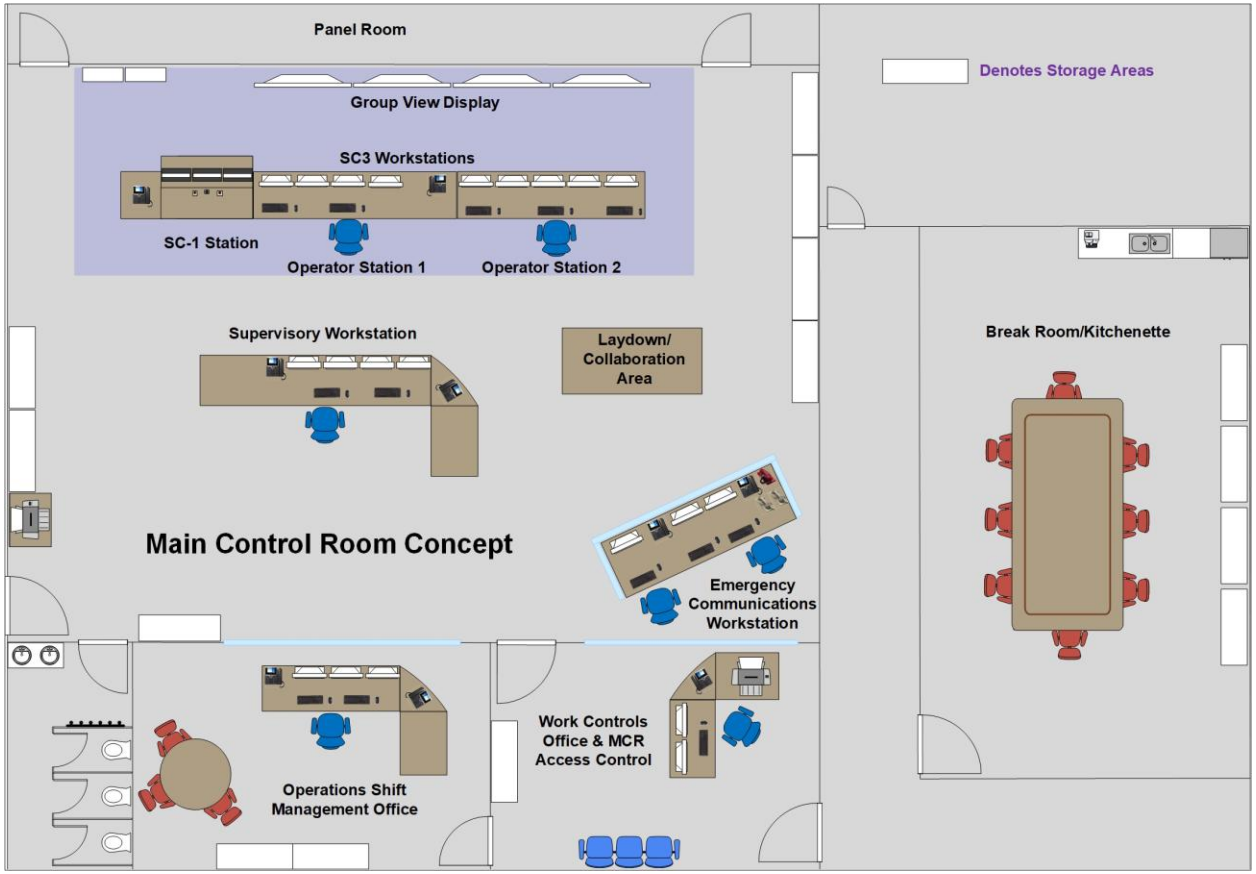


Figure 7.5-1: Main Control Room and Surrounding Areas (Top View)

7.6 Instrumentation and Control in Secondary Control Room

The SCR is located in the RB. The routes from the MCR to the SCR are described in Section 6.4. The SCR includes the required HSI that enables operators to perform the defined set of functions required for responding to the identified plant events and conditions for which the MCR cannot be used.

The SCR equipment is designed in accordance with the HFE process as described in Appendix 7A.

7.6.1 Control Transfer Function

Access to the SCR is under strict administrative controls and indicated by alarms in the MCR to ensure unauthorized access is detected. The SCR includes HSI inventory required to maintain the plant in a safe state for scenarios requiring MCR evacuation. Suitable provisions are provided inside the SCR for transferring control to the SCR whenever the MCR is abandoned, as well as to transfer control back to the MCR. Any design features, if needed to effect I&C control transfer, are located in a suitable location determined through HFE analysis, either in the SCR or accessible via the qualified access path.

Design features that prevent or mitigate spurious actuations due to fire are described in Subsection 9.5.1.

7.6.2 Secondary Control Room Use

The SCR is utilized to perform the functions required to keep the plant in a safe state when the MCR is unavailable. The required functions are derived as a result of safety and HFE analyses.

The SCR includes suitable facilities for habitability and well as workspace for tasks to support required usage. The SCR contains a suitable supply of food and water. The SCR also contains adequate space and provisions for sleeping as required by the postulated scenarios in which it is used.

The SCR is designed to accommodate the expected staffing based on BWRX-300 HFE staffing analysis for the expected usage conditions. A suitable number of workstations are provided to support the specific task, communication, and work coordination needs for expected personnel.

7.6.3 Secondary Control Room Layout

The layout of the workstations and HSI in the SCR provides the personnel with adequate information to assess the plant state and perform actions to maintain the plant in a safe state, if required for the expected usage scenarios.

To reduce the likelihood of human errors and time needed to resume monitoring and control tasks within the SCR, the layout and HSI of the SCR are designed to be consistent with the MCR to the extent possible, with differences driven by different purpose and user task needs.

The SCR operator workstation can accommodate the required staffing. The SC1 workstation contains the required HSI to perform the functions expected to be performed in the SCR.

The SCR SC3 workstation section contains two VDUs to support monitoring, control, and alarm management. The SC3 workstation VDUs can display any provided plant and system information, as supported by the I&C architecture, but only allows for control of equipment performing Safety Category 3 functions and equipment performing no safety category functions. SC3 VDUs allow for flexible use and are not dedicated to specific functions. These VDUs are controlled by a wired keyboard and mouse.

SCR SC3 workstation VDUs are sized and positioned in compliance with specified HFE design requirements for viewing angles for frequently used HSI components. The SCR SC3 workstation is suitably positioned to support effective verbal communication with the other workstations in the room. The SCR SC3 workstation also contains space for communications equipment, procedure laydown, and administrative tasks.

The SCR supervisory workstation is implemented with SC3 equipment and large enough to accommodate the required staffing. The SCR supervisory workstation is suitably positioned to maintain a clear line-of-sight to the SC1 and SC3 workstations and support effective verbal communication throughout the room. The supervisory workstation contains two SCN VDUs to provide SPDS indications (i.e., no control) to monitor plant conditions, and a VDU or tablet is provided for business LAN connection. The business LAN is separated from the plant I&C networks through a firewall and data diode. The SCR supervisory workstation also contains space for communications equipment, procedure and drawing laydown, and administrative tasks.

VDUs are controlled by a wired keyboard and mouse interface except where applicable or dictated by safety requirements. The SC1 VDUs are a touchscreen type interface. If supplied, tablets are controlled by a touchscreen interface.

The SCR has an emergency communications workspace located and designed with suitable provisions to minimize disturbance to the operating crew. The emergency communications workstation contains VDUs or Wi-Fi-enabled tablets for connection to business LAN with at least one VDU with displaying SPDS indications (i.e., no control) to monitor and report externally on plant conditions. The emergency communications workspace also contains space for communications equipment, procedures, and drawings in support of performing administrative tasks.

7.7 Instrumentation and Control in the Emergency Response Facilities

The BWRX-300 MCR is the primary location for plant monitoring and control during normal, abnormal, and most emergency conditions. The I&C capabilities in the MCR are described in Section 7.5. The SCR is utilized to perform the functions required to keep the plant in a safe state when the MCR is unavailable. The I&C capabilities in the SCR are described in Section 7.6.

The Emergency Preparedness and Response capabilities and associated facilities are described in Section 13.3.

The plant information necessary to support overall emergency response from the MCR, SCR, Technical Support Center, Operational Support Center, Central Emergency Control Center, and Local Recovery Center are provided either directly via the UDH or the PDH unidirectional boundary device interface shown in Figure 7.2-1, depending on cyber security requirements and I&C requirements required for to each location.

The MCR emergency communications workspace contains space for communications equipment and administrative tasks. The plant communication systems are described in Subsection 9.5.2.

7.8 Hazard Analysis for Instrumentation and Control Systems

Hazards for the BWRX-300 I&C systems are addressed in five ways. First, specific system requirements are developed based on plant safety analysis (i.e., plant level hazards), as described in Chapter 15. These are the Safety Category 1, 2, and 3 actuation functions. Specific safety functions actions are mode dependent, as specified by the Mode Switch position shown for each function in Tables 7.3-1, 7.3-3, and 7.3-4. Second, specific hardware and software technology hazards for the digital I&C equipment (i.e., digital technology hazards) are addressed by system level failure modes and effect analyses performed using the methods described in IEC 60812, “Failure Modes and Effects Analysis (FMEA and FMECA),” (Reference 7.8-1). Third, a rigorous development process, as described in Section 7.4, is used to minimize human errors during system development and operation (i.e., systematic hazards). Fourth, the I&C equipment is designed to provide protection against external and internal hazards, as described in Sections 3.3 through 3.7. And fifth, the I&C equipment is qualified, as described in Sections 3.10, 3.11, and 7.3.

7.8.1 References

7.8-1 IEC 60812, “Failure Modes and Effects Analysis (FMEA and FMECA),” International Electrotechnical Commission, 2018.

7.9 Appendix 7A – Human Factors Engineering Introduction

In accordance with NUREG-0800 "Standard Review Plan" Chapter 18, Revision 3 and NUREG-0711 Revision 3, "Human Factors Engineering Program Review Model," a HFE program is established. The BWRX-300 HFE plans describe the HFE activities and analyses informing the design of the BWRX-300 based on clear definition of the full plant set of users and a clearly defined scope of application across the full plant design, operational modes, and lifecycle stages, with focus on important Human Actions (HAs).

The high-level goal of the HFE program is to identify and assess the risks and consequences that arise from human interactions with the plant at the phases of its lifecycle and to reduce those risks through the design as far as is reasonably achievable. The Human Factors Engineering Program Plan (HFEPP) specifies a proportionate, integrated, and effective HFE program for the design of the BWRX-300 standard plant design as well as the first site-specific ("project-specific") design stage. It also forms the basis for any future site- and customer-specific BWRX-300 design HFE programs, carried out in other "project-specific" design stages. The HFEPP supports BWRX-300 licensing and workplans that fall within the timeframe of the standard plant and project-specific design stages.

The HFEPP describes and defines the following, to a level of detail appropriate for an overarching program plan:

- Technical considerations for HFE work during design of the plant, including descriptions and methods, intended tools, and technical guides
- Processes and procedures that establish the general steps taken for effective implementation and the measures used to ensure consistency across the work performed for each HFE activity or "Technical Element." The HFE technical elements are described in Appendix 7B.
- The role of HFE within the project, including the interaction among HFE and other engineering disciplines, the flow of work and information within and among each of the HFE technical elements, and the infrastructure to communicate requirements and issues across disciplines within the project
- The scope of the SSCs to which HFE technical elements are applied and the approach used to determine the level of HFE application in these areas
- The HFE outputs created for the BWRX-300 project, a summary of the processes implemented to produce these outputs, and the relative timeline under which the activities occur

The requisite technical elements are informed by various disciplines and engineering outputs. For example, the probabilistic risk assessment, DLs, human reliability analysis, and other studies, evaluations, tests, and validations outcomes inform the HFE technical elements. The technical elements inform one another and as the design progresses, information is fed iteratively into the related elements for modification and increasing precision and unequivocal detail. As each element is developed and outcomes of tests, measurements, validations, and analyses are completed, HSIs are established and modified. The data and HFE guidance inform the engineers, procedure writers, trainers, and other disciplines to ensure consistency and standardization in design and plant construction, and operations in the various plant modes. The overarching goals are to eliminate human errors, optimize production, and ensure safety. If in the unlikely event, a human error occurs, optimized human factors design input mitigates the effects of the error.

The HFE program applies to the plant conditions in the design basis, including normal, outage, abnormal, and emergency. The HFE program in design also applies across the scope of users

and activities that support BWRX-300 operation, testing, inspection, and maintenance, including functions such as fuel handling, chemistry, radioactive waste processing, and radiation protection. However, the application of HFE support and activities to the scope of each phase and task location is graded (or proportionate) to apply a higher level of emphasis and rigor for important human interactions that are safety-critical or hazardous.

HFE requirements management is performed in accordance with a requirements management process that is standardized and controlled across the entire plant design. Human factors best practices and compliance to regulatory guidelines reflect state-of-the-art human factors guidelines, principles, and methods.

Process Requirements are those related to how the HFE program is conducted and the interfaces among HFE and other disciplines. HFE process requirements are allocated to the HFEPP and are fulfilled via the construction and implementation of the architecture and processes defined within this document. As such, the HFE process requirements are allocated to an end with the HFEPP.

Product Requirements are for the design of, or provision for plant workspace and environmental attributes, SSCs, and HSIs. Product requirements are either derived from HFE design standards, codes, and guidance, or generated from HFE analyses as required to support successful task performance in the specific context of plant conditions. These requirements are implemented through design requirements specifications or design records communicated to and implemented by the relevant design teams. Requirement traceability and HFE support level is proportionately applied based on the assessed risk-based grading.

7.10 Appendix 7B – Human Factors Engineering Technical Elements

The technical elements for the BWRX-300 HFE programmatic approach are described briefly below. The full description of these elements, the objectives and scope, methods, and results, and how they constitute a comprehensive and robust program of HFE integration across the plant design is included in the Final Safety Analysis Report (FSAR) accompanied by subordinate documents, plans, strategies, processes, and analysis results. Among those supporting documents is the HFEPP, a concept of operations, and input to accommodate the regulatory bodies.

1. Operating Experience (OE) Review – Identification, review, and incorporation of any recommendations and learning (positive and negative) from past events and user feedback related to HFE in design. Particularly important is operator input.

   The criteria used to identify the OE applicable to the design:

   - The OE reviewed, including recognized industry HFE issues (e.g., Three Mile Island (TMI) issues, OE reports – NUREG-1275 series)
   - The process used in conducting the review (e.g., documentation reviews, interviews)
   - The results of the OE reviews, including how the design is addressing the human factors issues identified by the OE reviews.

   Methodology

   The OE review process methodology establishes the process and procedures for evaluating operating, design, and construction experience, thus ensuring that the applicable important industry experiences are provided in a timely manner to those designing and constructing the plant, as required by 10 CFR 50.34 (f)(3)(i).

   - The method for identifying the OE includes identifying:
     - OE for the selected HFE technology components from relevant predecessor plants and systems
     - Risk-important HAs, recognized industry issues
     - Issues identified by plant personnel
   - The methods for analysis and evaluation of OE include:
     - Use of summarized issues from industry sources
     - Development of insights from event reviews
     - Development of design solutions to reduce human error
   - The method for keeping track of the process includes the use of the HFE Issue Tracking System, which permits tracking, and review of the issues identified and addressed in the design.

2. Functional Requirements Analysis (FRA) – Determination of functions required to achieve plant goals and objectives. The FRA establishes:

   - Methods for conducting the FRA consistent with accepted HFE practices and principles
   - Strategies to define the system functions required that determine HSI requirements
   - The process to identify critical safety functions (e.g., those functions required to control radionuclide release)

Methodology

The function analysis includes:

- Methods to identify system and subsystem level functions based on the BWRX-300 mission and goals
- Logical function sequence arrangements that can be traced from BWRX-300 plant level goals to specific tasks
- The method to develop detailed function descriptions, which encompasses:
  - Identification of observable parameters that indicate system status
  - Control of processes and data required to accomplish the function
  - Determination of the way proper discharge of function is determined
- Analysis methods that define the integration of closely related sub-functions so that they are treated as a unit
- Analysis methods that divide identified sub-functions into two groups according to whether:
  - Common achievement of the sub-function is an essential condition for the accomplishment of higher-level function
  - The sub-function is an alternative supporting function to a higher-level function, or the sub-function accomplishment is not necessarily pre-required for a higher-level function

3. Allocation of Functions (AOF) – Assigning the identified functions to system (technology), human, or shared between them based on respective capabilities and limitations of each or those tasks that are accomplished by both the system and the human. The goal is to maintain situational awareness, full cognizance of plant state, and effects of actions taken by the operator or the system.

Methodology

The AOF process for BWRX-300 is based on the relevant best practice methodology presented in IAEA-TECDOC-668, “The Role of Automation and Humans in Nuclear Power Plants,” (Reference 7.10-1).

The methodology identifies functions that should not be assigned to humans due to criteria such as:

- Physical demands (forces, posture)
- Cognitive demands (multitasking, stress, situational awareness, and vigilance)
- Combination of physical and cognitive demands (accuracy, response time)
- Environmental conditions (temperature, radiation)

The AOF process also uses criteria from USNRC NUREG/CR-2623, “The Allocation of Functions in Man-Machine Systems: A Perspective and Literature Review,” that limit or preclude human participation in a function or, conversely, that make human participation mandatory. These combined criteria form the top-level, overriding criteria in the AOF process.

The FRA process provides the list of functions input to the AOF. The AOF process is composed of two stages: the first stage establishes an initial (hypothesized) allocation; the second stage

evaluates the hypothesized AOF to determine its adequacy and validate that it is optimized within the larger integrated task performance and work environment.

The initial AOF is determined following a formal decision flow with key allocation criteria informing each decision point. The decisions are based on expert judgement formed by a panel that includes a Human Factors Engineer/HFE Specialist, an HFE Operations/Maintenance representative, a plant integration engineering representative, and a system engineer for the respective system. The expert panel accounts for the mandatory criteria when hypothesizing the initial allocation. The panel also makes use of OE to determine how functions were allocated in previous or similar applications and evaluate how they have performed.

The second stage of the AOF process is the AOF evaluation. This stage is performed later in the design and HFE Testing and Evaluation (T&E) process, following completion of system level and integrated Task Analysis (TA). The AOF evaluation is a structured examination of function and task groupings that is used to assess allocations in a collective manner within an integrated work environment, instead of on a single function basis, where overload issues are less likely to be revealed. Functions and tasks allocated to humans are considered in combination using scenario development. The scenarios are then evaluated to determine acceptability based on expected concurrent task performance, workload (physical and cognitive), vigilance, and situation awareness.

As part of the AOF activity, functions are decomposed to tasks, and multiple tasks may be necessary to support each function. For example, to fulfill a core protection function, a task to perform a system readiness surveillance test supports the eventual task of safety system initiation. The function tasks require a “task allocation” that uses the same criteria applied at the overall AOF level. The safety analyses also provide input to the AOF, specifying when HAs are required to backup automatic (i.e., system) actions.

4. TA – Identification of the tasks required to achieve the allocated functions, and decomposition into task steps to allow the identification and characterization of HSIs, personnel, locations, and support equipment (e.g., communications, lighting, personnel protection) required to perform each task successfully. This includes workspaces, workplaces, hardware, software, and requisite materials or job performance aids to ensure error-free and timely task performance success.

Methodology

The task inputs provided by the AOF, and task grading process form the starting point for TA. These tasks are divided into levels of effort as defined through the task grading portion of the AOF process. Minimum HFE application-level applied to an HA is determined by the overall risk level. The minimum HFE application-level dictates the minimum degree of application when considering each HFE technical element.

HFE tasks regardless of application-level receive a TA. However, the level of detail within the TA varies based on HFE application-level. For example, those at the lowest risk level may be performed by the responsible system engineer and reviewed by HFE for acceptability.

5. Staffing Analysis – Determination of the numbers and roles of personal required to support optimal task performance for plant conditions.

Methodology

The process starts with an assumed initial staff complement taken from predecessor and similar plants, and representative OE from the operating fleet. Using this information, the initial staffing is optimized, considering the modern design features and new systems of the BWRX-300.

The optimized initial staffing level is subject to an expert panel staffing assessment that evaluates the minimum staffing to cope with selected credible events. The goal is to perform early and iterative assessments as the design progresses such that the risk of less than adequate staffing is reduced in part with each evaluation.

The evaluations are performed as a desktop (talk-through) exercise led by the expert panel. The expert panel is made up of personnel from the HFE team, supported by personnel from plant integration engineering and risk and reliability engineering. The next steps take place in conjunction with the TA and HSI design process. During TA, task steps are defined, and personnel assignments are made. The TA forms the basis for job design and qualifications for each role. With the TA, timeline analysis is conducted for the most resource-intensive credible events. These events are selected by the expert panel, and the expert panel also performs a review of the timeline analysis and input TA data to evaluate whether the minimum staff complement is adequate.

6. Treatment of important HAs – Activities supporting and providing input to the BWRX-300 safety analyses to ensure clear identification of important HAs, ensure claimed actions are achievable, and identify HSI requiring the highest level of HFE focus and effort. Consideration and integration of HFE within the safety analyses, and consideration of the results and assumptions of the safety analyses within the other HFE activities both comprise the technical element of treatment of important HAs. The set of activities supporting this HFE technical element was developed based on requirements for HFE expertise in safety analyses per IAEA SSG-51, “Human Factors Engineering in the Design of Nuclear Power Plants,” (Reference 7.10-2).

Methodology

HFE safety analysis activities use the various safety analyses, available system design information, (particularly that related to HSI)s, and available and applicable TA results as inputs. The activities that form part of this technical element include:

- Perform a human operation hazard evaluation
- Review the design safety analysis for explicit and implied HAs
- Review the probabilistic safety assessment and human reliability analysis for:
  - Ensuring event sequences introduce a human performance limiting value
  - Perform qualitative human error analysis to substantiate or refine the human error probabilities used for all HAs claimed in the human reliability analysis
  - Provide HFE qualitative basis and substantiation for any dependency assumptions and analysis
  - Provide HFE qualitative basis and substantiation for any timeline assumptions and analysis
  - Review the Severe Accident Analysis, including Level 2 and Level 3 probabilistic safety assessment, to identify explicit and implied HAs, determine task achievability in required timescales, and capture input for procedure development and emergency planning and response
  - Compile a database of all important HAs claimed in the safety analyses, including the source of the claim, their key characteristics, related assumptions and any associated HSIs

7. HSI Design – Identification and management of the set of HFE design requirements from standards, codes, and best practice guidance, and implementation of those requirements plus results from HFE analyses into the design of HSIs, including integration of HFE team design support; also includes HFE test and evaluation activities.

Specific Discussion of Human-System Interface: Main Control Room Design

The MCR design features are based upon proven technologies and are demonstrated through broad scope control room dynamic simulation tests and evaluations, to satisfy the BWRX-300 HSI design goals and design bases. Validation of the implemented MCR design includes evaluation of the design features performed as part of the HFE V&V process as defined by the performance measures specified for each validation activity.

The HSI design implementation activities include the development of dynamic models for evaluating the overall plant response as well as individual control systems, including operator actions. These dynamic models are used to:

- Analyze both steady-state and transient behaviors
- Confirm the design of the advanced alarm system concepts
- Confirm the adequacy of control schemes
- Confirm the allocation of control to a system or an operator
- Develop and validate plant operating procedures

Using part-task simulation, an initial set of systems is identified through modeling, including the development of the graphical user interfaces. The part-task simulator is used in preliminary BWRX-300 design and includes design features unique to the BWRX-300.

As the BWRX-300 design progresses, the part-task simulator proceeds through a series of iterative evaluations resulting in the development of a complete control room full scope simulator. Simulators are the focal point for operator evaluations and feedback checkpoints throughout the HSI design process.

Safety margins are used in the design safety analysis is to account for uncertainty and provide an added margin for adequate assurance that the various important limits or criteria are not encroached upon. Suitable margin is also added to the HFE analysis (TA and human error analysis) of HAs by ensuring suitable conservatism is included in things like the timeline analysis or the generation of human error probabilities.

Design goals and design bases for the design of HSIs in the MCR and in other applicable facilities are based on the HFEPP.

Methodology

For the HSIs, the HFE team provides HFE design requirements and task support requirements. Depending on the HFE application-level and the nature of the HSI, the HFE team:

A. Provides design requirement-compliant, application-specific wireframes, and templates for HSI displays, panel layouts, and HSI elements, housed within the HSI element library.

B. Works with the system engineer to implement the HFE requirements in the requirements management database (as compiled in the HFE Design Requirements Document) that apply to their discipline and system, allowing them to design or specify and select compliant SSCs or HSIs and to develop compliant system and equipment layouts.

C. Reviews, tests, and verifies HSI design work to ensure acceptable requirements are implemented and compliance is documented or (for lower HFE application levels) performs proportionate design work audits using the design and task support evaluation checklists to ensure acceptable requirements are implemented and compliance is documented.

The process remains the same, however the primary and secondary activity performers of the step may vary based on the HSI type and the HFE application level.

For hardware HSIs, including plant layout and physical environment, direct physical SSC interfaces and HSIs that form part of procured commercial off-the-shelf or bespoke design systems and equipment, the discipline engineer includes the applicable task and HFE design requirements as part of their system and component level design requirements. They flow down the HFE requirements into “lower” level system and component requirements specifications, and procurement specifications to ensure consistency and application throughout the BWRX-300 plant design. This is managed using the design requirements management tool, the standard content for system design specifications, and the integrated HFE design support activities and issues management process outlined in HFEPP. The HFE program HSI design element includes roll-out of the HFE requirements and support to their implementation for other disciplines.

8. Procedures – Process and activities for the development of usable and validated operational documentation, for plant task types. There are procedures for the conduct of human factors activities as well as human factors input to procedures for operation and maintenance of the plant.

Methodology

The procedures development methodology establishes the process for developing technical procedures that are complete, accurate, consistent, and easy to understand and follow.

For each procedure type, a procedure writer’s guide is established. The procedure writer’s guide establishes objective criteria so that the procedures developed with it are consistent in organization, style, and content. The procedure writer’s guide provides instructions for procedure content and format, writing of steps, and specifying lists of terms used.

Procedures are then written after the associated TAs are finished. TA is an iterative process due to the amount of information that is created at any point in the design process. TA is conducted in a prioritized manner and done on a per task, per system basis. System design descriptions are used along with the TA output to give a complete understanding of the system and its operation. These inputs provide understanding of each system, its associated tasks, and its interrelationship to other systems, which allows the development of system level and then integrated operational procedures.

As the TA progresses beyond operations, the system designers define test, inspection, and maintenance requirements. As the TA processes and the iterative safety analyses mature, procedures are developed for other than normal operations (e.g., test, maintenance, surveillance, alarm response, outages and any other conditions that are not included in the scope of normal plant operations). Procedures that support important HAs for the higher HFE application levels are developed by the HFE team based on the detailed TA and qualitative human error analysis. Procedures to support HAs for the lowest HFE application-level are developed by the system designer or vendor specification based on the Basic TA.

Initial procedures are tested and evaluated for their usability and efficacy using physical mock-ups, simulation, and plant 3D models early in the design, through the HFE T&E activities. The HSI design and procedures are evaluated together to make a more cohesive model of the

future operational plant. This provides contextual feedback for both the HSI design and the procedure, allowing optimization of the design of both simultaneously.

The developed procedures are verified and validated as part of the HFE V&V program, culminating in Integrated System Validation (ISV). Procedures that support important HAs for the higher HFE application levels are validated by individuals that are independent of the design. Individuals that review these levels of procedures include representatives of the end users. Final procedure validation is done on the installed physical plant hardware as part of the HFE design implementation once the plant has been built. Procedures to support HAs for the lowest HFE application-level are validated by the designer. The final procedure validations are done as specified by the plant preoperational testing and startup testing programs.

Following ISV, the procedures are provided to the plant operator to use as the bases for preoperational testing, startup testing, and operation of the plant. Once a procedure is validated and declared complete, procedure maintenance and control of updates falls under the engineering change management process to ensure that changes to individual procedures are reflected throughout the full suite of related procedures, and that changes are confirmed as accurate and supported by the design and TA.

9. Training and Qualifications – Process and activities for the development of relevant and validated training content, optimized for the plant design, operational documentation, and baseline personnel qualifications and attributes.

Methodology

The training and qualification program development follows the fundamentals of the systematic approach to training method. The analysis stage provides the identification of training needs, tasks, or competencies required for training and the knowledge, skills, and abilities required to perform a specified job position based on assigned tasks. Tasks that support plant functions are identified as part of detailed and basic TA. Tasks are selected for training based on difficulty, importance, and frequency analysis. Depending on the difficulty, importance, and frequency ranking, a decision is made to determine if initial training and periodic retraining is needed. This evaluation of training tasks is the training equivalent to grading HAs. The results of the TA, including identification of critical steps, inform the difficulty, importance, and frequency analysis and resulting rankings.

Depending on the difficulty, importance, and frequency ranking, a determination is made if initial and periodic retraining is required. Tasks that are selected for training are then analyzed to determine the required knowledge, skills, and attributes. The knowledge, skills, and abilities necessary for each job position, including entry-level education, training, and experience, is established to support training design. Any changes to the iterative HFE or system design inputs to the analysis phase are required to be assessed for impact on the training analysis.

10. Human Factors V&V – Detailed, staged set of activities to provide assurance of the correct and sufficient implementation of HFE requirements in the design, and the appropriate design to support required tasks. There are both system V&V and integrated systems V&V and typically both part-task and full scope simulation is required.

Methodology

The purpose of HFE Verification is two-fold:

A. Task Support Verification (TSV) verifies that the HSIs, as defined and baselined in the HSI inventory and characterization, include the necessary features (e.g., controls, information displays, and alarms) required to support tasks and that there are no unnecessary features.

B. HFE design verification verifies that the HSIs and plant SSCs, are compliant with the applicable HFE design requirements and design-to-analysis requirements input resulting from HFE analysis activities. Verification activities include identifying changes to the design that affect HSIs and other features due to competing design constraints, and checking for due consideration of OE items, user stakeholder input, and HFE T&E results.

HFE Validation is conducted through staged activities:

A. Early and partial system validation activities are performed in advance of the full-scope simulator and fully constructed plant and are generally performed only on partial systems. Although they require a sufficient maturity of the design, HFE participants from the V&V team, and end users with enough level of familiarity with the system, they do not require the full integrated system. The purpose of these validation activities is to identify and solve HFE issues in advance of a fixed design.

B. ISV is the performance-based evaluation of the fully integrated system design. Simulations are used to validate the ability of personnel, trained using the training and qualification program material, to use the integrated HSIs and finalized procedures in accordance with the task and scenario performance requirements. ISV is intended to evaluate those integrated aspects that were verified and validated singly through earlier, partial means.

HFE validation ensures that the design, particularly the HFE-specified aspects, accomplishes its intended goals for usability and reducing the risk of human error to As Low As Reasonably Achievable through design. Validation is an integrated, dynamic, performance-based test activity where participants are subjected to a set of simulated scenarios that represent a realistic, challenging, and generalizable set of conditions to ensure that the integrated HSI supports safe operation of the plant.

The scope of the HFE V&V activities applies to operations, maintenance, testing, and inspection activities and to the plant HSIs. As with other HFE program activities, the application of HFE V&V is graded to focus on the HSIs, tasks, and plant conditions that involve important HAs, are complex or novel, or inherently hazardous. The same risk-based approach is applied to the HFE V&V activities to determine scope, robustness, and level of detail used in the approach.

11. Design Implementation – Support and monitoring of the design from “on paper” to a realized constructed plant, including integration with configuration control to ensure no loss of integrity of the HFE program and process goals throughout fabrication and construction. The design implementation addresses the final “As-Built” implementation of the HFE design requirements into plant design. By its definition, this technical element sits outside the timeframe of this design stage HFEPP. However, it is included as it is an important “close-out” confirmation that the design as implemented meets the HFE design intent and supports user task performance in the optimized manner expected.

<u>Methodology</u>

The following activities are conducted during design implementation, and the detailed methodology is developed in the HFE design implementation methodology plan:

- Confirm that the final HSIs, procedures, and training (“As-Built”) conform to the design resulting from the HFE design process and V&V activities
- Evaluate any deviation from the design during implementation and identify their effect on the HFE aspects of the design
- Perform final procedure validation on the physical plant hardware

- Verify aspects of the design that may not have been evaluated previously in the V&V process:
  - This includes any hardware/software, new or modified displays that were absent from the simulator-based integrated V&V process, and any physical or environmental (e.g., noise, lighting) differences between those present at the V&V process and the "As-Built" control areas
- Verify resolution of remaining human engineering discrepancies and open items from the HFE Issue Tracking System
- Verify HFE application levels are based on the final set of inputs

Design implementation activities are performed by the HFE team, with support from commissioning, vendor, and pre-operations testing teams. All HFE deficiencies identified during commissioning are to be addressed prior to declaring the system available for service.

Unlike the other HFE technical elements, design implementation is performed after the design is complete, immediately prior to commencement of commercial operations. Despite the plant being built and undergoing startup testing and commissioning, the HFE team still performs the design implementation activities. The process follows the HFE V&V process. In this case, the sampling process includes identifying any changes to the standardized plant design (documented through engineering configuration control process) and assessing effects on HFE aspects of the design. The sampling process also identifies aspects of the design that were not able to be previously verified and validated. Additional items not previously identified for V&V but determined to require it are also added to the scope for design implementation HFE V&V.

The list of methods used are like those described for the HFE program activities, particularly HFE V&V, staffing analysis, procedure development, training, and qualification program development, and HFE issues tracking.

12. Human Performance Monitoring (HPM) – Continuous monitoring of user task performance throughout the lifetime of the plant to ensure optimum plant and organizational design and identify early trends and issues that require HFE design improvements. Human performance activities are typically the responsibility of the owner/operator; however, HFE gives input. The HPM program is not "monitor operator behaviors" but rather to check and confirm the goodness of the implemented design and the need for any improvements or modifications.

Methodology

The purposes of HPM are:

- To ensure that the high safety standards established by the HFE program during the design of the plant are maintained even when changes are made to the plant
- To detect any deterioration of task performance that may be attributable to latent or slow-developing HFE design issues
- To provide adequate assurance that the safety bases remain valid during the operational phase of the plant

There is no intent for the HSI designer or the applicant to periodically repeat a full set of ISV activities. The strategy is to provide a monitoring plan, building upon the HFE activities during the design that can be carried forward into the operational phase, using industry accepted methods. HPM incorporates the monitoring strategy into the problem identification and corrective action program, which identifies and classifies human errors, provide for evaluation of the root cause, and supports effectiveness verification and documentation of the corrective action.

7.10.1 References

7.10-1 IAEA-TECDOC-668, “The Role of Automation and Humans in Nuclear Power Plants,” International Atomic Energy Agency, 1992.

7.10-2 IAEA SSG-51, “Human Factors Engineering in the Design of Nuclear Power Plants,” International Atomic Energy Agency, 2019.

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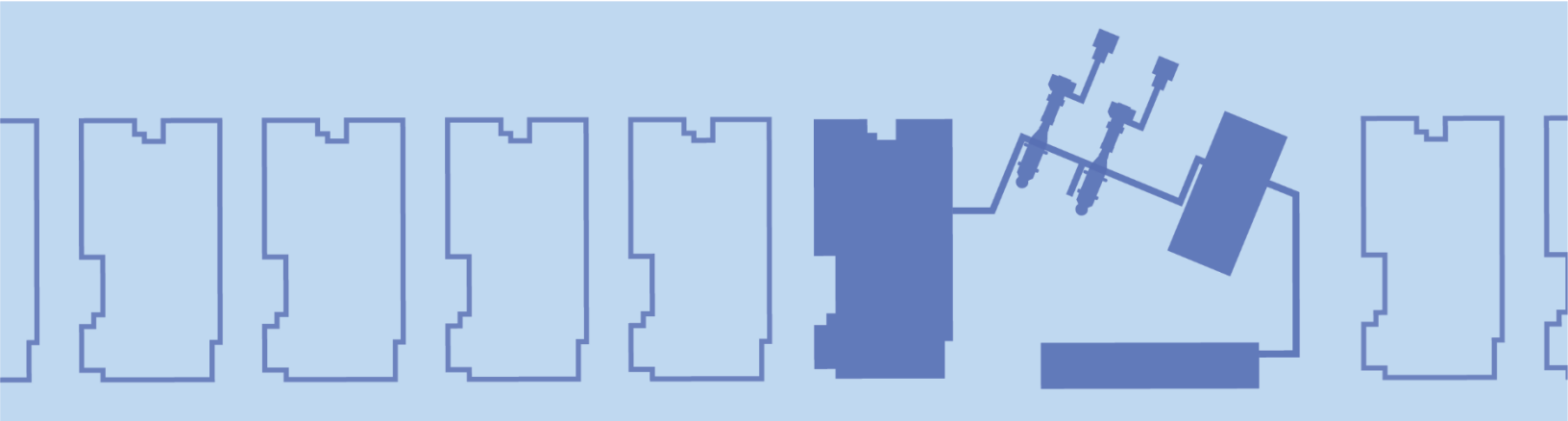
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Non-Proprietary Information

Chapter 8
Electric Power



ACRONYM LIST

Acronym	Explanation
AC	Alternating Current
AOO	Anticipated Operational Occurrence
BOP	Balance of Plant
BTP	Branch Technical Position
D-in-D	Defense-in-Depth
DBA	Design Basis Accident
DC	Direct Current
DCIS	Distributed Control and Information System
ECCS	Emergency Core Cooling System
EDS	Electrical Distribution System
EMC	Electromagnetic Compatibility
EMI	Electromagnetic Interference
FLEX	Diverse and Flexible Coping Strategy
FMCRD	Fine Motion Control Rod Drive
GDC	General Design Criteria
GSU	Generator Step Up Transformer
HV	High Voltage
I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency
IC	Isolation Condenser
ICS	Isolation Condenser System
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronics Engineers
LG	Load Group
LOCA	Loss-of-Coolant Accident
LOOP	Loss-of-Offsite Power
LOPP	Loss-of-Preferred Power
LV	Low Voltage
MCC	Motor Control Center
MCR	Main Control Room
MTS	Manual Transfer Switch

Acronym	Explanation
MV	Medium Voltage
NERC	North American Electric Reliability Corporation
NFPA	National Fire Protection Association
PAM	Post-Accident Monitoring
PCCS	Passive Containment Cooling System
PDC	Principal Design Criterion
PWR	Pressurized Water Reactor
RAM	Reliability, Availability, and Maintainability
RAT	Reserve Auxiliary Transformer
RB	Reactor Building
RFI	Radio Frequency Interference
RG	Regulatory Guide
RVT	Regulating Voltage Transformer
SBO	Station Blackout
SC1	Safety Class 1
SC2	Safety Class 2
SC3	Safety Class 3
SCN	Non-Safety Class
SCR	Secondary Control Room
SDG	Standby Diesel Generator
SSCs	Structures, Systems, and Components
STS	Static Transfer Switch
TB	Turbine Building
TMI	Three Mile Island
TVSS	Transient Voltage Surge Suppressor
UAT	Unit Auxiliary Transformer
UPS	Uninterruptable Power Supply
USNRC	United States Nuclear Regulatory Commission
VAC	Volts Alternating Current
VDC	Volts Direct Current

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8.0 ELECTRIC POWER

This chapter describes the BWRX-300 plant electrical system and provides its design bases.

8.1 Introduction

The Electrical Distribution System (EDS) architecture is a configuration of generators, buses, transformers, and load centers that supply power to the design loads. The design described is for a 60 Hz Alternating Current (AC) power system, with 6.9 kV for the Medium Voltage (MV) level and 600 Volts Alternating Current (VAC) for the Low Voltage (LV) level.

The EDS architecture is made up of the following systems:

- Generator and Exciter
- Preferred Power System:
  - Balance of Plant (BOP) Distribution Subsystem
  - Preferred Power 250 VDC Subsystem
- Standby Power System:
  - Standby Diesel Generators
  - Standby Power 250 VDC Subsystem
- Emergency Power System:
  - SC1 Direct Current (DC) Power Subsystem
  - Uninterruptable Power Supply (UPS) Subsystem
- Lighting and Service Power System
- Cable and Raceway System
- Grounding and Lightning Protection System
- Non-Distributed Control and Information System (DCIS) Communication System
- Freeze and Cathodic Protection System

In addition to describing the electrical systems and associated requirements, this chapter includes plant figures that depict the switchyard, switchyard interface, and the MV and LV AC distribution systems, including the DC battery and UPS systems. The Lighting and Service Power System and the Non-DCIS Communication System are discussed in Section 9.5.

The BWRX-300 design minimizes the reliance on electrical power to support Safety Category 1 functions. The passive design of the plant is not dependent upon offsite or Standby Diesel Generator (SDG) AC power, to mitigate a Design Basis Accident (DBA). Safety Class 1 (SC1) power is supplied from battery-backed DC power, which has a coping period of 72 hours for all DBAs.

The Defense-in-Depth (D-in-D) concept is achieved by defined safety classes that are further established in Section 3.1. The electrical systems provide an integral support function for Structures, Systems, and Components (SSCs) across the range of safety classes. A summary of the safety categories and description of how the electrical systems provide support is outlined below.

Safety Category 1 Functions

Safety category functions are designed to minimize reliance on electrical power. Safety Category 1 functions requiring power and control are controlled by the SC1 DCIS. Safety Category 1 functions are automatically initiated on loss of power except for Isolation Condenser System (ICS) isolation. The functions that fail to the required position upon loss of power do not require electrical power as an integral support function. Therefore, the only energize-to-actuate Safety Category 1 function supported by electrical power is isolation of a faulted ICS line.

In addition to the Emergency Power System, elements of the Cable Raceway System provide a passive Safety Category 1 function as portions of this system provide a Safety Category 1 support function.

Safety Category 2 Functions

The electrical systems provide a range of Safety Category 2 functions that are powered by Standby Power System DC backed power supplies. These power supplies are UPS Load Groups (LGs) A and B, which provide power to the Safety Class 2 (SC2) DCIS and four sets of dedicated Fine Motion Control Rod Drive (FMCRD) motor controllers.

Safety Category 3 Functions

The electrical systems provide AC or DC power to equipment supporting Safety Category 3 functions. Normal functions performing a fundamental safety function during normal plant operation or maintaining key reactor parameters (e.g., reactor pressure, reactor water level, and temperature) within normal ranges, and their integral support functions, are assigned to at least Safety Category 3.

Power for most systems with defense line functions is provided by sources backed by the Standby Diesel Generators (SDGs). Examples of AC loads serving these Safety Category 3 functions include shutdown cooling, plant cooling water, and fuel pool cooling system loads. Control power associated with these loads is derived from DC backed sources. There are a few instances where the AC power is provided by non-SDG backed sources (feedwater pump power and condensate pump power are derived from Safety Class 3 (SC3) MV buses). In such cases, the Safety Category 2 function supported (trip the pump breaker) is not required for a Loss-of-Preferred Power (LOPP) scenario and is provided by independent means for certain Design Extension Condition events.

Safety Category 3 functions also include those functions that mitigate severe accidents including RPV vent, containment vent and emergency boron injection. These functions are powered by an electrical bus that is backed by the SDGs, and the EDS infrastructure is designed to support Diverse and Flexible Coping Strategy (FLEX) connections for an alternate power source.

Safety Category N Functions

Safety Category N (SCN) components are those that do not perform Safety Category 1, 2, or 3 functions. For the EDS this includes:

- Grounding and Lightning Protection System
- Freeze and Cathodic Protection System

Figure 8.1-1 depicts the main BWRX-300 plant electrical one-line diagram for the Generator and Exciter, Standby Power System and Preferred Power System.

For further explanation on the assignment of safety classes, refer to Chapter 3, Subsection 3.2.2.

8.1.1 Offsite Power System

The BWRX-300 has two connections to the switchyard. The Preferred Power System provides the interconnecting EDS elements between the plant main generator, on-site power system, and the offsite power source. The Preferred Power System interfaces for the plant are at the two switchyard connections. Note that for the BWRX-300 design, a LOPP or a Loss-of-Offsite Power (LOOP) are the same and can be used interchangeably to represent a loss of the switchyard as a power source to the plant.

The BWRX-300 switchyard design allows significant flexibility. Because of the passive safety design of the BWRX-300 plant, only a single Preferred Power System connection between the plant and switchyard is required. However, two sources of offsite power are provided to improve reliability and operational flexibility.

8.1.1.1 Offsite Power System Design Basis

Safety Category Function

Offsite power has no Safety Category 1 function due to the passive design of the BWRX-300. Offsite electrical power is provided to normally operating systems and components in accordance with Principal Design Criterion (PDC) 17, “Electric Power Systems.” The offsite power system is not a plant system and has no classification. Further discussion is contained in Section 3.1.

8.1.2 Onsite Power Systems

The on-site AC power system starts at the Generator Step Up Transformer (GSU) and Reserve Auxiliary Transformer (RAT), and transitions from there to include the main generator, the main generator circuit breaker, the Unit Auxiliary Transformer (UAT), and the rest of the plant AC distribution as depicted on Figure 8.1-1. The preliminary switchyard interface for BECPV-N01 is depicted on Figure 8.1-2.

The Preferred Power System source is from the switchyard breakers through the High Voltage (HV) side of the GSU, through the UAT, and to the MV buses. A Preferred Power System source from the switchyard breakers to the RAT primary side is provided for maintenance and resiliency purposes. The UAT and RAT secondaries are connected to the plant MV buses.

The MV bus configuration and load allocation are such that buses can be shut down for maintenance during outages and can be shut down one at a time in normal operation while minimizing the effect to plant power generation.

The MV buses are normally aligned to the UAT. The MV buses can be aligned to the RAT while offline or during power generation, either manually or via automatic transfer. The RAT is normally energized and ready to accept the plant load. The MV buses have protective relays and are designed to automatically bus transfer (break before make) between the UAT and the RAT. The bus transfer is automatically initiated on loss of the UAT source to the MV buses or a faulted circuit in the normal Preferred Power System supply. The intent of this automatic action is to minimize the potential for a loss of power to the plant EDS. The transfer logic occurs if power is available from the RAT, preventing a transfer to a dead source. To avoid the possibility of cycling between the preferred sources, the automatic bus transfer is one-directional (i.e., there is no possibility of an automatic transfer from the RAT to the UAT). Transfer from the RAT to the UAT for the Preferred Power System source is performed manually with sufficient permissives to avoid transferring into a faulted or out of sync source.

8.1.2.1 Onsite Power System Design Basis

Safety Category Functions

The BWRX-300 design includes passive safety features that do not require offsite or SDG AC power to support core cooling or containment integrity. Safety functions are automatically initiated on loss of power except for isolation of a faulted ICS line. Therefore, the only Safety Category 1 function that requires electrical power is ICS Isolation.

Emergency Power System batteries provide 72 hours' worth of power to loads supporting Safety Category 1 functions and Post-Accident Monitoring (PAM). A loss of AC voltage, degraded voltage, and a loss of phase condition on the electrical power systems does not adversely affect the performance of plant Safety Category 1 functions.

The Standby Power System includes the SDGs and the interconnecting AC system that is derived from it. Additionally, the Standby Power System includes the SC2 Standby Power 250 VDC Subsystem. The primary purpose of this subsystem is to provide backup power in the event of a LOPP for loads which have a function required during a LOPP (or loads which require power for plant investment protection).

8.1.3 Design Criteria, Regulatory Guides, and Standards

Regulatory Requirements

The Offsite Power Systems and Onsite Power Systems comply with the following GDC, which has plant-wide applicability, as discussed in Section 3.1:

GDC 5 – Sharing of structures, systems, and components

The following GDCs and PDC have aspects specific to the Offsite Power Systems and Onsite Power Systems and compliance is addressed as follows:

GDC 2 – Design bases for protection against natural phenomena – This GDC was considered in the design of the Offsite Power Systems and Onsite Power Systems. The Emergency Power System is SC1 and is designed to retain the capability to perform its Safety Category 1 functions during and after design basis natural phenomena events. Therefore, the Emergency Power System meets the requirements of this GDC.

No Safety Category 1 functions rely on AC power or DC power from the Standby Power System or Preferred Power System. Therefore, these systems are not required to meet the requirements of this GDC.

GDC 4 – Environmental and dynamic effects design bases – This GDC was considered in the design of the Offsite Power Systems and Onsite Power Systems. The Emergency Power System is SC1 and is designed to be compatible with environmental conditions associated with normal operation, maintenance, testing, and postulated pipe failure accidents including Loss-of-Coolant Accidents (LOCAs). Design requirements specify the duration that Safety Class 1 SSCs must survive the environmental conditions following a LOCA. Emergency Power System components are protected against dynamic effects including the effects of missiles, pipe whipping, jet impingement, and applicable hydrodynamic loads that may result from equipment failure. The effects of missiles originating outside the plant are also considered. Therefore, the Emergency Power System meets GDC 4.

The SC2 and SC3 SSCs in the Preferred Power System and Standby Power System that support normal operations or mitigate Anticipated Operational Occurrences (AOOs) are confirmed to be compatible with the environmental and dynamic effects present during the event scenarios where these functions are credited. Therefore, these SSCs meet the requirements of GDC 4.

Additionally, the failure of non-SC1 equipment due to environmental and dynamic effects will not impair SC1 equipment. No Safety Category 1 functions rely on offsite AC power; in the event of a loss of offsite power, the electrical design provides two layers of protection via onsite power systems, as discussed in PDC 17. Therefore, the Offsite Power System is not required to meet the requirements of this GDC.

PDC 17, Electric Power Systems: The BWRX-300 plant design supports PDC 17 compliance by providing an offsite AC power source for normally operating systems and components and two onsite power sources in the event of a loss of offsite AC power. Offsite power is not needed to support Safety Category 1 functions. No offsite or SDG AC power is required for 72 hours after a DBA.

The reliability of the power source(s) supporting functions that mitigate AOOs and DBAs is sufficient to ensure that the function's reliability target is met in accordance with NEDC-33934P, "BWRX-300 Safety Strategy" (Reference 8.1-1). Additional information about compliance with PDC 17 is provided in Section 3.1.

GDC 18, “Inspection and Testing of Electric Power Systems” – SC1 DC power sources are provided to support passive core cooling and containment Safety Category 1 functions. The SC1 DC Power Subsystem has the capability for testing each battery, rectifier, battery charger and inverter without disrupting power to SC1 loads. Therefore, the design of the SC1 DC Power Subsystem meets the requirements of this GDC.

The Preferred Power 250 VDC Subsystem supports functions used to ensure that the design limits for the fission product barriers are maintained. This subsystem is designed to permit appropriate periodic inspection and testing of each SC3 battery, rectifier, battery charger, and inverter in accordance with the requirements of GDC 18.

No offsite or SDG AC power is required for 72 hours after a DBA. Two layers of defense utilizing onsite systems are available following a LOPP as discussed in PDC 17. Therefore, GDC 18 does not apply to offsite power, SDG AC power, or SC2 DC backed power.

GDC 33, Reactor coolant makeup – The high-pressure CRD system pumps can provide reactor coolant makeup using power from the SC3 Standby Power System which can be powered from onsite electric power (assuming offsite power is not available) or offsite electric power (assuming onsite power is not available). Control Power for the CRD Pumps is supplied by the SC3 Preferred Power 250 VDC Subsystem. Leaks larger than the CRD system pump capacity or if CRD pumps are not available are mitigated by ICS, which requires no offsite or SDG AC power. ICS control power is supplied by the Emergency Power System (SC1 DC Power Subsystem and UPS Subsystem). Therefore, the electrical power requirements of GDC 33 are met.

GDC 34, Residual heat removal – The residual heat removal function for DBAs is performed passively by the ICS system. The RPV isolation valve and ICS control power is supplied from the Emergency Power System (SC1 DC Power Subsystem and UPS Subsystem). Neither offsite nor SDG AC power supplies are utilized in performance of the function.

GDC 35, Emergency core cooling – The emergency core cooling function is performed passively by the ICS system. The RPV isolation valve and ICS control power is supplied from the Emergency Power System (SC1 DC Power Subsystem and UPS Subsystem). Neither offsite nor SDG AC power supplies are utilized in performance of the function.

GDC 38, Containment heat removal – The containment heat removal function is performed passively by the Passive Containment Cooling System (PCCS). Neither offsite nor onsite power supplies are utilized in performance of the function. Therefore, there are no electrical power requirements for this GDC.

GDC 41, Containment atmosphere cleanup – The containment atmosphere is maintained inerted such that cleanup systems, and therefore electrical power systems, are not required. Therefore, there are no electrical power requirements for this GDC.

GDC 44, Cooling water – The ICS and PCCS systems transfer heat to their associated pools passively, without the need for electrical power systems. Therefore, there are no electrical power requirements for this GDC.

GDC 50, Containment design basis – The containment electrical penetrations are designed to minimize leakage such that the containment does not exceed leakage limits. Therefore, the containment electrical penetrations meet the requirements of this GDC.

Refer to Section 3.1 for additional information on GDC and PDC compliance.

10 CFR 50.34(f)(2)(v), Three Mile Island (TMI)-Related Requirements Action Item I.D.3, “Safety System Status Monitoring” – met by conformance with RG 1.47 and BTP 8-5, and compliance with IEEE 603-2018.

10 CFR 50.34(f)(2)(xiii), Three Mile Island (TMI) Action Item II.E.3.1, “Emergency Power Supply for Pressurizer Heater” – This criterion is applicable only to PWRs and does not apply to the BWRX-300.

10 CFR 50.34(f)(2)(xx), TMI Action Item II.G.1, “Emergency Power for Pressurizer Equipment” – This criterion is applicable only to PWRs and does not apply to the BWRX-300.

10 CFR 50.55a, “Codes and Standards” – met for The SC1 Emergency Power System by compliance with IEEE 603-2018 or IEEE 603-1991 when specified by regulation or regulatory guidance.

10 CFR 50.63, “Loss of All Alternating Current Power” – The BWRX-300 design does not rely upon offsite power, SDG AC power, or operator action for the first 72 hours to achieve and maintain safe shutdown. As such, the passive design meets the requirements of this regulation.

10 CFR 50.65(a)(4) “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants” – Implementation of the programs required to meet this regulation will be provided with the operating license application.

Regulatory Guidance

United States Nuclear Regulatory Commission (USNRC) endorsed standards are used to show compliance to applicable USNRC Regulatory Guides (RGs). Section 1.9 includes Table 1.9-20 that lists conformance with RGs or describes an alternative approach including any applicable International Electrotechnical Commission (IEC) or Institute of Electrical Electronics Engineers (IEEE) standards.

RG 1.6, Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems – The BWRX-300 does not need or have SC1 standby AC power sources. However, portions pertaining to the Emergency Power System meet IEEE 603-2018 and are addressed within Subsection 8.3.2.

RG 1.9, Application and Testing of Safety-Related Diesel Generators in Nuclear Power Plants – The BWRX-300 SDG units are not SC1, nor is offsite or SDG AC power needed for the BWRX-300 to mitigate DBAs. Therefore, this RG is not applicable.

RG 1.32, Criteria for Power Systems for Nuclear Power Plants - SC1 DC power sources are provided to support passive core cooling and containment integrity safety functions. No offsite or standby diesel generator AC power is required for the initial 72 hours after a DBA. The

Emergency Power System meets RG 1.32 by conformance to the guidance in IEEE 308-2020, “IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations.”

RG.1.47, “Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems.” The design of the Emergency Power System conforms to the applicable requirements given in IEEE 603-2018, as supplemented by the requirements included in RG 1.47.

RG 1.53, Application of the Single-Failure Criterion to Safety Systems: The Emergency Power System conforms to this guidance and meets IEEE 379-2014, “IEEE Standard for Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems,” and IEEE 603-2018. See Subsection 8.3.2.1.

RG 1.63, “Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants.” Electrical penetrations conform with this guidance and meet IEEE 317-2013, “IEEE Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations.”

RG 1.75, Criteria for Independence of Electrical Safety Systems: Equipment and cable separation, and electrical isolation are in accordance with the requirements of IEEE 384-2018, “Standard Criteria for Independence of Class 1E Equipment and Circuits,” and IEC 60709 “Nuclear Power Plants – Instrumentation, Control and Electrical Power Systems Important to Safety - Separation.”

RG 1.81, “Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants” – The BWRX-300 Standard Plant is designed as a single-unit plant. Therefore, RG 1.81 is not applicable.

RG 1.106, “Thermal Overload Protection for Electric Motors on Motor-Operated Valves.” The BWRX-300 design does not include motor operated valves that perform a Safety Category 1 function. Therefore, this RG is not applicable.

RG 1.118, “Periodic Testing of Electric Power and Protection Systems.” The design of the Emergency Power System is designed to accommodate required surveillance testing in accordance with the requirements of IEEE 338-2022, “IEEE Standard for Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems,” as supplemented by the guidance in RG 1.118.

RG 1.128, “Installation Design and Installation of Large Lead-Acid Storage Batteries for Nuclear Power Plants.” The Emergency Power System batteries shall be designed with consideration of the recommended practices in IEEE 484-2019, “IEEE Recommended Practice for Installation Design and Installation of Vented Lead-Acid Batteries for Stationary Applications,” as supplemented by the guidance of RG 1.128.

RG 1.129, “Maintenance, Testing, and Replacement of Large Lead-Acid Storage Batteries for Nuclear Power Plants.” The SC1 batteries meet the requirements of IEEE 450-2020, “IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications.”

RG 1.153, “Criteria for Safety Systems.” The Emergency Power System is single failure tolerant in accordance with the guidance in IEEE 603-2018. See also conformance with BTP 8-5.

RG 1.155, “Station Blackout” – The BWRX-300 meets Station Blackout by having a 72 hour coping capability. Therefore, the intent of RG 1.155 is met, however, much of the guidance is not applicable due to the passive nature of the BWRX-300 safety systems. See Section 8.4 for regulatory compliance with 10 CFR 50.63.

RG 1.160, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.” Implementation of this guidance will be considered as part of the implementation of the maintenance rule (10 CFR 50.65) in the operating license application.

RG 1.204, “Guidelines for Lightning Protection of Nuclear Power Plants.” Refer to Subsection 8.3.1.10 for conformance and applicable standards.

RG 1.211, “Qualifications of Safety-Related Cables and Field Splices for Nuclear Power Plants.” Conformance with guidance is met by compliance with IEEE 383-2023, “IEEE Standard for Qualifying Electric Cables and Splices for Nuclear Facilities.”

RG 1.212, “Sizing of Large Lead–Acid Storage Batteries.” See Subsection 8.3.2.1 and Subsection 8.3.2.2 for conformance and applicable standards.

NUREG/CR 0660, “Enhancement of Onsite Diesel Generator Reliability” – The BWRX-300 SDGs are not SC1, nor is offsite or SDG AC power needed to achieve safe shutdown. Therefore, the NUREG is not directly applicable. However, D-in-D principles such as redundancy and diversity are incorporated in the design and integration of BWRX-300 systems.

Branch Technical Positions

BTP 8-1 Requirements on Motor-Operated Valves in the ECCS Accumulator Lines – BWRX-300 does not use accumulators. Therefore, BTP 8-1 is not applicable.

BTP 8-2, Use of Diesel Generator Sets for Peaking – SDGs are only tied to Offsite Power for load testing and are not used for peaking. Therefore, BTP 8-2 is not applicable.

BTP 8-3, Stability of Offsite Power Systems: IEEE 242-2001, “IEEE Recommended Practice for Protection and Coordination of Industrial and Commercial Power Systems (IEEE Buff Book)” and IEEE 399-1997, “IEEE Recommended Practice for Industrial and Commercial Power System Analysis (Brown Book),” (Reference 8.1-34) are used to provide technical information and guidance regarding the protection and performance of the offsite power system (see Subsection 8.2.4).

BTP 8-4 Application of the Single Failure Criterion to Manually Controlled Electrically Operated Valves – The BWRX-300 is a passive design that does not require any manual valve operations for mitigation of a DBA. Therefore, the guidance in BTP 8-4 is not applicable.

BTP 8-5 Supplemental Guidance for Bypass and Inoperable Status Indication for Engineered Safety Features – The Emergency Power System design includes provisions for bypassing an inoperable status indication. The Emergency Power System meets RG 1.47, RG 1.153, and IEEE 603-2018.

BTP 8-6 Adequacy of Station Electric Distribution System Voltages – The Offsite Power System does not supply power to any Class 1E loads and does not support Safety Category 1 functions. Therefore, BTP 8-6 is not applicable.

BTP 8-7 Criteria for Alarms and Indications Associated with Diesel-Generator Unit Bypassed and Inoperable Status – BWRX-300 design does not include SC1 diesels. Therefore BTP 8-7 is not applicable.

BTP 8-8 Onsite (Emergency Diesel Generators) and Offsite Power Sources Allowed Outage Time Extensions – The BWRX-300 design does not include SC1 diesel generators. Therefore, BTP 8-8 is not applicable.

BTP 8-9 Open Phase Conditions in Electric Power System – The BWRX-300 complies with BTP 8-9 Section B.1 and Section B.3 for new reactors that incorporate passive safety features. See Subsection 8.2.2

8.1.4 References

8.1-1 NEDC-33934P, “BWRX-300 Safety Strategy,” GE-Hitachi Nuclear Energy Americas, LLC, Revision 2, March 2026.

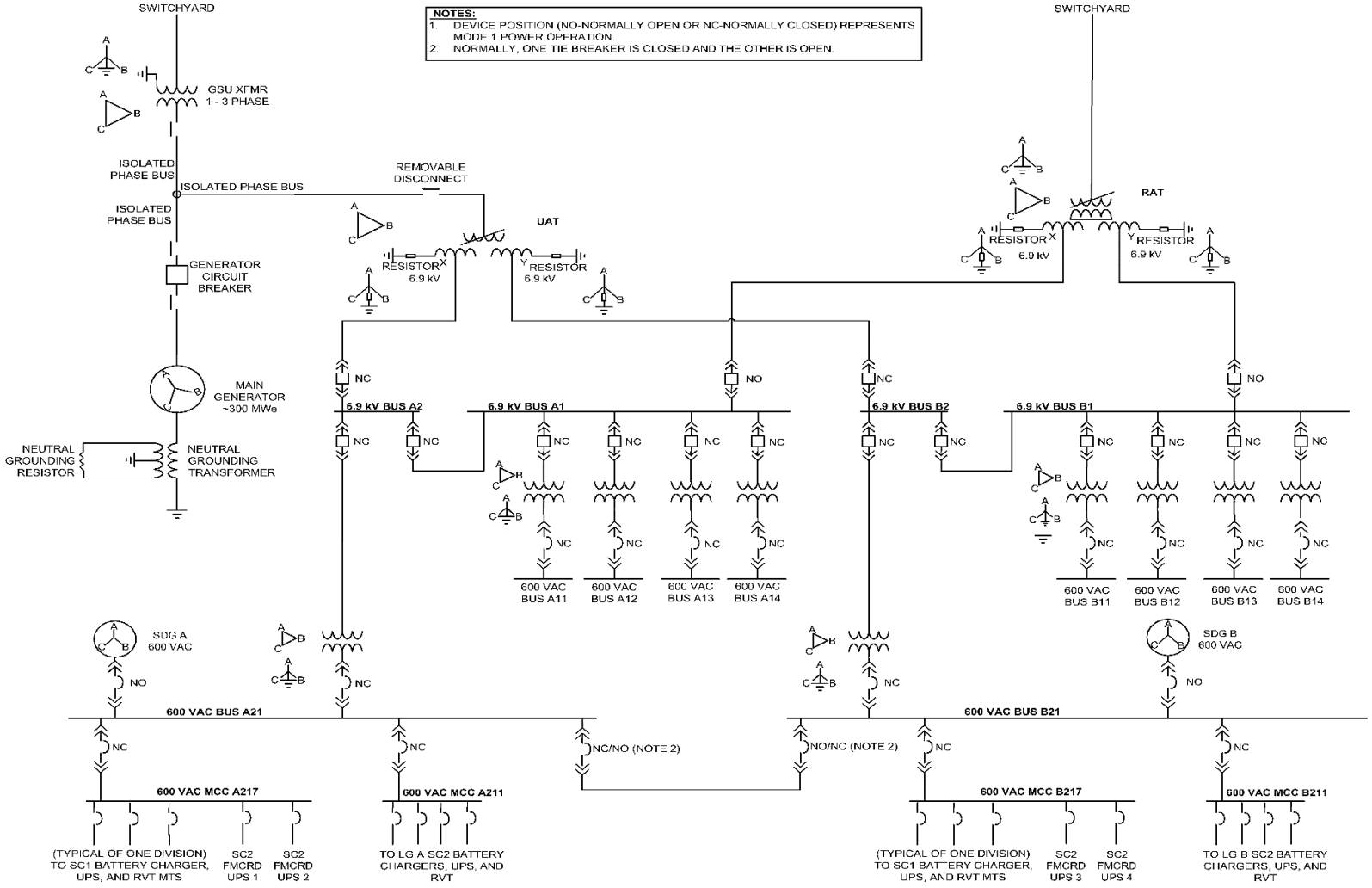


Figure 8.1-1: Main Electrical One-Line

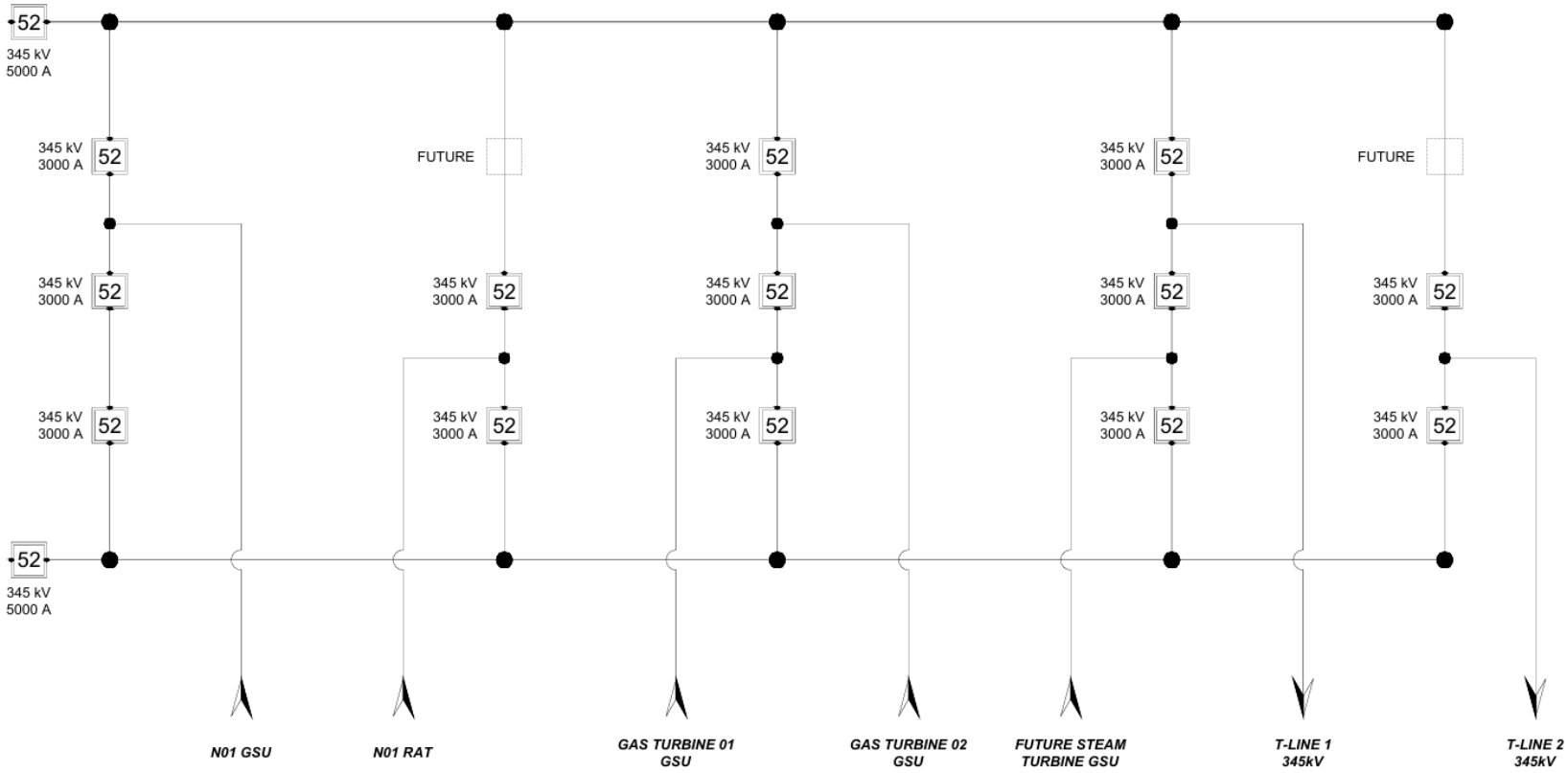


Figure 8.1-2: Preliminary BECPV-N01 Switchyard Interface

8.2 Offsite Power Systems

Power is supplied to the plant from the switchyard connected to the transmission grid offsite power sources. The BWRX-300 has two connections to the switchyard. One connection is to the HV side of the GSU. The LV side of the GSU is connected to the main generator breaker via isolated phase bus duct and includes a branch to the UAT. The other connection is to the HV side of the RAT. The UAT and RAT secondaries are connected to the plant MV buses. The configuration is shown in Figure 8.1-1.

The GSU provides transmission of the main generator output to the grid when the plant is online and can provide power to the plant when the main generator is offline. When the generator breaker is closed and the plant is synchronized with the grid, the generator gross power output is approximately 300 MWe. When the generator breaker is open, the switchyard can “backfeed” power to the plant through the GSU and the UAT. The generator breaker is designed to perform its intended function during steady-state operation, power system transients, and major faults. The ratings and required capabilities of the generator circuit breaker are the designated limits of operating characteristics based on definite conditions as defined in IEEE 62271-37-013, IEEE/IEC “International Standard for High-Voltage Switchgear and Control gear – Part 37-013: Alternating Current Generator Circuit-Breakers,” (Reference 8.2-1).

The normally energized RAT interface provides a supplemental power path from the grid to the plant power distribution system. Like the UAT, the RAT is sized for 100% of the plant loading in both normal and abnormal operating conditions. The purpose of the preferred power source from the RAT is to provide operational Reliability, Availability, and Maintainability (RAM) to the system.

8.2.1 Transmission

The transmission system provides electric power from the utility grid to the offsite power system.

8.2.2 Transmission Switchyard

The switchyard is planned to be a Breaker-and-a-Half design unless interconnection studies warrant a more robust configuration (Figure 8.1-2).

The preliminary switchyard design includes a total of two transmission lines from the 345 kV switchyard to the electrical grid. Connection to the regional transmission grid is provided by connecting to the existing 345 kV 80286 DULCE7A to 8676 STATIC7A transmission line, as well as the Black Bayou to Vistron 138 kV transmission line. These transmission lines will provide independent paths for off-site power to connect to the switchyard. Each line can be individually isolated without removing the other from service. Both transmission lines will connect with the grid lines at the point of interconnection.

Operational experience has shown that an open phase condition on the high side of the large power transformers connecting the transmission network to the onsite power system is difficult to detect and has the potential to adversely affect downstream equipment. The BWRX-300 complies with the requirements provided within BTP 8-9 for new reactors by including open phase detection on the high side of the GSU and RAT. The open phase detection detects an open phase during all operational electrical system configurations and plant load conditions. Following detection of an open phase an alarm is generated in the MCR.

8.2.3 Transformer Area

The transformer area contains the GSU, the UAT, and the RAT. One feeder connects the transformer area with the switchyard to supply power to/from the GSU for the unit. A separate switchyard feeder supplies power to the RAT. The arrangement is shown in Figure 8.1-2.

Fire protection for the transformers is in accordance with National Fire Protection Association (NFPA) 804-20, “Standard for Fire Protection for Advanced Light Water Reactor Electric Generating Plants,” (Reference 8.2-2) guidance.

8.2.4 Grid Stability

The BWRX-300 has passive systems for core cooling and containment integrity, and therefore, does not depend on the electric power grid for mitigating DBAs. This feature significantly reduces the importance of the grid connection and the importance of grid stability for anything other than for power production. An Interconnection study will be performed prior to plant operation to ensure that the transmission interface is adequate for Federal Energy Regulatory Commission and North American Electricity Reliability Corporation regulatory requirements.

8.2.5 Conformance and Guidance

The passive plant water supplies in the Isolation Condenser (IC) and fuel pools provide a cooling function for at least seven days without any active intervention. IC isolation valves fail as-is, and battery backed UPSs are needed to close isolation valves if an IC leak occurs. After seven days, refilling of cooling pools and the reactor can be accomplished using the onsite diesel with SC3 systems, FLEX connections, portable pumps, and other available sources. Offsite power is not required for mitigation of Anticipated Operational Occurrences (AOOs) and DBAs. The BWRX-300 does not require offsite or SDG AC power sources for mitigating DBAs (see PDC 17 discussion in Subsection 8.1.3); Section 15.5 describes the design bases assumptions utilized for analysis.

The AC power system is designed such that plant auxiliaries can be powered from the grid under all operating conditions. During LOOP, the AC power is supplied to the Standby Power System by the onsite SDGs or portable generators via FLEX connections. Preassigned loads and equipment are automatically loaded on the SDGs in a predetermined sequence. Additional loads can be manually added as required.

8.2.6 Design Criteria, Regulatory Guides, and Standards

See Subsection 8.1.3.

8.2.7 References

8.2-1 IEEE/IEC 62271-37-013-2021, “International Standard for High-Voltage Switchgear and Control Gear – Part 37-013: Alternating Current Generator Circuit-Breakers,” Institute of Electrical and Electronics Engineers, 2021.

8.2-2 NFPA 804-20, “Standard for Fire Protection for Advanced Light Water Reactor Electric Generating Plants,” National Fire Protection Association, 2020.

8.3 Onsite Power Systems

Onsite power systems provide power to the plant loads during all operating conditions. The onsite power systems include AC power systems and DC power systems.

The onsite AC power systems are described in Subsection 8.3.1. The DC power systems are described in Subsection 8.3.2. SSCs classification methodology is provided in Section 3.2.

8.3.1 Alternating Current Power Systems (Onsite)

The onsite AC power system consists of a 60 Hz standby onsite AC power supply system and electrical distribution equipment. Figure 8.1-1 shows the plant main one line diagram. The onsite power distribution system has multiple nominal bus voltage ratings. Equipment utilization voltages are designated as 6.9 kV, 600 VAC, 240 VAC, and 208/120 VAC.

The onsite AC power system starts at the GSU and RAT high side transformer bushings and transitions from there to include the GSU, the main generator, the main generator circuit breaker, the UAT, the RAT, and the rest of the plant AC distribution as depicted on Figure 8.1-1.

See Subsection 8.1.3 for GDC compliance.

8.3.1.1 Electric Circuit Protection

Load grouping facilitates the use of conventional protective relaying practices for isolation of faults. Protective relay schemes and trip devices are provided throughout the onsite power system to:

- Isolate faulted equipment and/or circuits from the power system
- Prevent damage to equipment
- Protect personnel
- Minimize system disturbances
- Maintain continuity of the power supply

8.3.1.2 Preferred Power System

The Preferred Power System provides the interconnecting EDS elements between the plant main generator, on-site power system, and the offsite power source. In addition, the preferred power system includes the entirety of the BOP Distribution Subsystem required for power generation. The preferred power system boundaries start at the high side bushings of the GSU and RAT and continue to the terminals of the Main Generator and downstream to the terminals of the BOP equipment. The system also includes protective relaying associated with the GSU, RAT, UAT and Main Generator, and a 250 Volts Direct Current (VDC) and UPS Subsystem.

The GSU provides transmission of the main generator output to the grid when the plant is online and can provide power to the plant when the main generator is offline. When the generator breaker is closed and the plant is synchronized with the grid, the generator gross power output is approximately 300 MWe. When the generator breaker is open, the switchyard can “backfeed” power to the plant through the GSU and the UAT. The UAT is sized to support 100% of the plant loading in both normal and abnormal operating conditions. Protection is provided for fault conditions and includes both transient and steady-state overvoltage protection. If a UAT fault occurs, the switchyard breaker, generator breaker, and secondary side breakers open, isolating the faulted transformer and causing automatic transfer to the RAT Preferred Power System path.

The normally energized RAT interface provides a supplemental power path from the grid to the plant power distribution system. Like the UAT, the RAT is sized for 100% of the plant loading in

both normal and abnormal operating conditions. The purpose of this Preferred Power System source is to provide operational RAM to the system.

The GSU, UAT, and RAT are monitored by a condition monitoring system.

As shown on Figure 8.1-1 the on-site power system includes four 6.9 kV MV buses. The normal Power System supply is connected through the UAT to the MV buses. The alternate supply to the buses is connected through the RAT to the MV buses and serves as backup to the UAT.

The MV bus configuration and load allocation is such that buses can be removed from service for maintenance during outages and can be removed from service one at a time in normal operation while minimizing the effect to plant power generation.

With the plant online providing power to the switchyard, the MV buses are normally aligned to the UAT. The MV buses can also be aligned to the RAT during power generation, either manually or via automatic transfer. The RAT is normally energized and ready to accept the plant load. With the plant offline and the generator breaker open, power is back fed through the GSU to the UAT supplying power to the MV buses. The MV bus A Train is shown in Figure 8.3-1.

8.3.1.3 Standby Power System

The Standby Power System includes the SDGs and the interconnecting AC system that is derived from it. Additionally, the Standby Power System includes the Standby Power System 250 VDC Subsystem. The primary purpose of this subsystem is to provide backup power in the event of a LOPP for loads which have a function required during a LOPP or loads which require power for plant investment protection.

Safety Category 2 functions supported by power from the Standby Power System have battery backed UPS power sources. These include functions associated with the SC2 FMCRDs and the SC2 DCIS. The non-battery backed portion of the system supports equipment that relies upon an SDG-backed power source. These include functions such as shutdown cooling, plant cooling water, boron injection and CRD Purge Water injection.

SDGs are the backup source of power to the Emergency Power System. When the associated bus loses power, the SDG automatically starts after a short delay. Specific loads on the LV buses (and the derivative Motor Control Center (MCC)) are automatically sequenced after the SDG successfully starts. Load sequencing is based on relative importance and coordinates with the dropout of battery chargers and UPS. The SDGs can be manually started/controlled from the MCR and locally.

The SDG auxiliary systems are further described in Section 9.5.

The Standby Power System provides redundant LV power feeds to each SC1 EDS division.

Each train of the Standby Power System is designed to supply the full load of all three SC1 divisions, inclusive of a single charger per division operating at rated load.

The Standby Power System provides two SDGs for 600 VAC backup power to the LV buses for plant mechanical systems, battery chargers and UPSs, and investment protection.

The SDGs meet the following requirements:

- Sized to supply 100% of the required SC1 EDS divisional loads in all three SC1 EDS divisions while also supplying power to the required EDS loads during a LOOP event
- Can be controlled from the MCR and locally
- Provide voltage and frequency regulation within the limits required by ANSI/NEMA MG1-2021, “Motors and Generators,” (Reference 8.3-1)

- Capable of closing into a dead bus
- Once started, operate without requiring any plant services other than those required to store and transfer fuel, with the capability of continuous operation for at least 7 days at rated power without any offsite resources

8.3.1.4 Electrical Equipment Layout

The Preferred Power System GSU, UAT, and RAT are located outside the Turbine Building (TB) at the generator end of the building.

The Emergency Power System divisions are in the Reactor Building (RB), which is a Seismic Category I structure. Each of the three Emergency Power System divisions have an independent and physically separate power feed from each of the Standby Power System LV SDG backed LV buses. Each of the three Emergency Power System divisions are independent and physically separate from other Emergency Power System divisions and from the non-SC1 electrical systems of the plant. The Standby Power System is located in the Turbine Building, Control Building, and the Reactor Building.

8.3.1.5 Analysis

Emergency Power System includes SC1 equipment including battery chargers, batteries, and UPSs. The UPSs supply AC power to the SC1 loads required for 72 hours after a DBA or loss of onsite and offsite AC power. The Preferred Power System and Standby Power System do not contain SC1 components.

The configuration and system designs for the BWRX-300 electrical systems must provide acceptable power and voltage to station loads for all configurations and plant operating conditions. The system is designed to meet the following criteria:

- The minimum and maximum allowable steady state voltages for connected equipment
- The minimum and maximum allowable transient voltages for connected equipment
- The maximum allowable voltage drop across connecting equipment
- The maximum allowable short circuit current for major electrical equipment
- Allowable configurations for the electrical systems
- The required operating conditions for the electrical systems
- Established margins consistent with IEEE standards for equipment sizing and operation
- Frequency and voltage tolerance for systems and equipment

8.3.1.6 Environmental Considerations

The EDS equipment is capable of performing under worst case conditions when required to perform its safety function. SC1 equipment that must operate in a harsh environment during and subsequent to an accident is provided in Section 3.11.

8.3.1.7 Freeze and Cathodic Protection System

The freeze protection subsystem provides environmental protection to plant equipment to prevent freezing. The electric heat tracing system provides freeze protection where required for outdoor service components and warming of process fluids if required, either indoors or outdoors.

The cathodic protection subsystem uses an electrical process that reverses the basic corrosion reaction by applying a DC current to counteract the corrosion current and thus prevent or decrease the rate of corrosion of a metal in an electrolyte.

The Freeze and Cathodic Protection System are considered SCN.

8.3.1.8 Containment Building Electrical Penetrations

Containment Electrical Penetrations meet the requirements of IEEE 317-2013 (Reference 8.3-2) to support compliance with RG 1.63, RG 1.75, and PDC 17.

8.3.1.9 Grounding Subsystem

Grounding is important for both personal safety of plant personnel and electrical system protection, and it is critical to the operability and performance of the BWRX-300 DCIS. The overall Grounding Subsystem consists of a plant ground grid and grounding system conductors with a low resistance path to the plant ground grid. The BWRX-300 grounding grid consists of a buried building grounding grid and local switchyard grounding grid, both with grounding electrodes where appropriate. Ground grids are tied together to ensure equal potential and tied to the remote switchyard grounding grids when required. Grounding conductors for equipment and instrument grounds are installed within buildings to provide a low resistance path to the grounding grid.

The Grounding Subsystem includes the following types of grounding networks:

- Building grounds (connections of the building foundations and external structures to the grounding grid)
- Equipment grounds (connection of equipment to the internal grounding system of the buildings)
- Instrument grounds (separate grounding connection to the grounding grid for analog and digital instrumentation systems)
- Lightning Protection Subsystem (to protect equipment against lightning induced surges)

The Grounding System is classified as SCN.

8.3.1.10 Lightning Protection Subsystem

Lightning protection is installed on structures and select electrical equipment to protect against lightning strikes. Structures, including buildings, above-ground tanks, stacks, antennas, construction cranes, and meteorological towers, are protected by a Lightning Protection Subsystem. Lightning protection devices are tied to the ground grid through low impedance paths. The Lightning Protection Subsystem is classified as SCN.

Analysis

Lightning protection is provided in accordance with RG 1.204, “Guidelines for Lightning Protection of Nuclear Power Plants.” Both the Lightning Protection Subsystem and Grounding Subsystem are designed and installed to the applicable sections of the following guides and standards:

1. IEEE 80-2013, “Guide for Safety in AC Substation Grounding,” (Reference 8.3-3).
2. IEEE 81-2012, “Guide for Measuring Earth Resistivity, Ground Impedance, and Earth Surface Potentials of a Ground System,” (Reference 8.3-4).
3. IEEE 665-1995, “Guide for Generation Station Grounding,” (Reference 8.3-5).
4. IEEE 1050-2004, “Guide for Instrumentation and Control (I&C) Equipment Grounding in Generating Stations,” (Reference 8.3-6).
5. NFPA 780-2022, “Standard for the Installation of Lightning Protections Systems,” (Reference 8.3-7).

6. IEEE C62.23-2017, “Application Guide for Surge Protection of Electric Generating Plants,” (Reference 8.3-8).
7. IEEE 666-2007, “Design Guide for Electric Power Service Systems for Generating Stations,” (Reference 8.3-9).
8. IEEE 142-2007, “Recommended Practice for Grounding of Industrial and Commercial Power Systems,” (Reference 8.3-23).

8.3.1.11 Motor Operated Valve and Thermal Overload Bypass

There are no SC1 motor operated valves in the BWRX-300 design.

8.3.1.12 Cables Subject to Submergence

The underground duct banks used in the plant design have sufficient drainage to minimize any potential for long term submergence.

8.3.1.13 Bypass and Inoperable Status Indication for Engineered Safety Features Systems

The SC1 EDS design includes provisions for bypass and inoperable status indications in accordance with NUREG-0800 BTP 8-5.

8.3.1.14 Cable Raceway System

The cable and raceway system includes raceway and supports necessary to route cable in and between buildings and equipment using cable trays, conduits, duct banks, or tunnels.

The cabling and raceways are designed to meet the following design requirements:

1. Equipment, cabling separation, and electrical isolation are based on IEEE 384-2018, (Reference 8.3-10) and IEC 60709:2018 (Reference 8.3-11). This supports conformance with RG 1.75.
2. Fire protection and fire safe shutdown program requirements are supported by cable separation, fire barriers, limiting cable tray fill, limiting cable ampacity to levels that prevent overheating and insulation failures (and resultant possibility of fire), and by use of fire resistant and non-propagating cable insulation. Copper cables within the power block are qualified to IEEE 1202-2006, “Standard for Flame-Propagation Testing of Wire and Cable,” (Reference 8.3-12) and IEEE 383-2023 (Reference 8.3-13). Fiber optic cables within the Power Block are qualified to IEEE 1202-2006 (Reference 8.3-12).

8.3.1.14.1 Cable Separation

Cables and raceways are separated on the basis of defense lines, division, and voltage class. Equipment and cable separation, and electrical isolation is in accordance with the requirements of IEEE 384-2018 and IEC 60709.

8.3.1.14.2 Cable Derating and Cable Tray Fill

Cable tray fill is limited to a percentage of cross-sectional area.

8.3.1.14.3 Cable Tray and Cable Routing

Cables are routed in horizontal and vertical runs of steel trays or through steel conduits. Trays or conduits are supported at intervals by appropriately designed supports. The supports are attached to walls, floors, and ceilings of structures as required by the arrangement. The type of support and spacing is determined by allowable tray or conduit spans which are governed by rigidity and stress. Underground raceways have limited application for the BWRX-300. When

underground raceways are required, they are appropriately sloped and drained to minimize the potential for long term cable submergence.

8.3.1.14.4 Inspection and Testing

Normally the cabling and raceways are designed to not require any inspections once they have passed installation and commissioning testing. If special inspection considerations are required, they are performed under plant inspection programs.

Cabling and raceways testing is based on preventive and predictive maintenance programs. Site testing requirements are used for post installation testing and inspection.

8.3.1.15 Standby Diesel Generator Testing and Inspection

The SC3 SDGs are automatically started and loaded upon a LOPP. The SDGs are designed and sized to carry the loads for plant recovery from DBA and LOPP events. SDG inspection and testing is based on preventive and predictive maintenance programs.

8.3.1.15.1 Standby Diesel Generator Capacity Testing

The SDGs are periodically tested by starting and loading to verify their functional capabilities. The SDG supply breakers are interlocked with the normal supply breakers to prevent the undesired paralleling of sources, while allowing paralleling of sources to allow periodic SDG testing by synchronizing and loading to the normal power feed. The testing is to verify that the function of the SDGs can operate under the anticipated loads.

8.3.2 Direct Current Power Systems

The on-site DC power system is divided into three subsystems with three safety classes, SC1, SC2, and SC3. See Subsection 8.1.3 for GDC compliance. The system and equipment functions are described below:

- Emergency Power System:
  - The SC1 DC Power Subsystem powers the SC1 DCIS equipment responsible for automatic shutdown and decay heat removal. The Safety Category 1 DCIS safety functions are executed in a fail-safe manner through removal of power with exception of isolation of a faulted ICS line.
  - As a seismically qualified DC power source, the SC1 DC Power Subsystem also supplies power to non-SC1 loads required during or following a seismic event.
- Standby Power 250 VDC Subsystem:
  - The loads assigned to the SC2 UPSs and batteries are those required for the BWRX-300 to achieve safe shutdown and include the equipment that provides a diverse, active alternative to Safety Category 1 functions.
- Preferred Power 250 VDC Subsystem:
  - The SC3 DC Power Subsystem is designed to provide power to SC3 DC loads, such as DCIS, protective relaying, and breaker control power.

8.3.2.1 Emergency Power System (SC1 DC Power Subsystem and Uninterruptable Power Supply Subsystem)

The on-site DC Power system description for each of the SC1 DC Power Subsystem and SC1 UPS Subsystem is provided below.

Like the three-division SC1 DCIS, the Emergency Power System consists of three separate and independent divisions of power. With the design of the SC1 DCIS, only two of the three divisions are required when a DBA occurs, and the plant safety functions are still achieved. Similarly, only two of the three corresponding Emergency Power System divisions are required to mitigate a DBA. Within each Emergency Power System division, a UPS supplies AC power for the DCIS cabinets, Scram and Isolation Valves, and various other equipment. The DCIS cabinets have an additional DC supply directly from a DC distribution panel fed from the battery bus to ensure that they remain in operation if a UPS is lost.

Figure 8.3-2 represents a typical design of one of three divisions of the Emergency Power System. Each division contains two battery chargers, each with the capacity to power loads while providing the maximum battery charge current. The 250 VDC SC1 batteries are sized to provide 72 hours of emergency power to ensure all functions requiring power are available and sufficient PAM is provided over the 72 hour period. The 72 hour requirement is consistent with the overall plant design basis and regulatory requirements including NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Plants: LWR Edition,” Section 8.4 for Station Blackout (SBO).

As shown in the Main Electrical One-Line in Figure 8.1-1 and Figure 8.3-2, the Standby Power System provides separate feeds from an SC3 LV bus to the three divisions of Emergency Power System. Each Emergency Power System division can be continuously energized from multiple sources: offsite power (normal supply), SDG power (backup supply), portable generator power (external building connections), or individual division battery power (emergency supply). Each Standby Power System supply train (diesel generator and associated equipment) is sized to supply all three Emergency Power System divisions. By division, SDG sizing considers one charger operating at rated capacity. Independent of offsite or Standby Power each division can be supplied by the Emergency Power System batteries (emergency supply). In the unlikely event that AC power provided by the LV buses is of insufficient quality, the battery chargers and UPS automatically isolate from the station EDS, and the Emergency Power System loads are supplied by the SC1 batteries. Additionally, external connections are provided for connection of FLEX as needed.

Loading on each of three divisions is similar with a few exceptions such as Secondary Control Room (SCR) lighting, which is distributed on only two of three divisions. Each division is designed to the maximum load and mission time of the three divisions.

The batteries, battery chargers, UPSs, and associated components are in the RB, which is a Seismic Category I structure. Divisional components are in separated rooms that are fire- and flood-barriered from each other by division. The SC1 electrical components and equipment in these areas are qualified to environmental and seismic conditions.

The following design requirements are applicable to the Emergency Power System:

The design of the Emergency Power System conforms to the applicable requirements given in IEEE 308-2020 (Reference 8.3-14).

The design of the Emergency Power System conforms to the applicable requirements given in IEEE 741-2022 (Reference 8.3-15).

Note that IEEE 741-2022 Annex C and Annex D are not applicable to the BWRX-300. Annex C applies to automatic bus transfer. IEEE 741-2022 applies to the SC1 Emergency Power System, which does not have an automatic bus transfer design feature. This guidance may be applied to the automatic bus transfer from the UAT to the RAT; however, this automatic bus transfer is not part of the SC1 Emergency Power System and is not required for compliance with IEEE 741-2022.

Annex D applies to high-speed magnetic circuit breakers used for motor protection. The SC1 Emergency Power System does not have these components.

The design of the Emergency Power System conforms to the applicable requirements given in IEEE 603-2018 (Reference 8.3-16) and conforms to RG 1.47 and RG 1.153.

The design of the Emergency Power System is designed to accommodate required surveillance testing in accordance with the requirements of IEEE 338-2022 (Reference 8.3-17) and conforms to RG 1.118.

The Emergency Power System batteries are designed with consideration of the recommended practices in IEEE 484-2019 (Reference 8.3-18) and conforms to RG 1.128.

Emergency Power System battery banks are sized such that the sum of the loads required in support of Category 1 functions and PAM does not exceed the battery ampere hour rating over the rated 72 hour discharge period or warranted capacity at the end of the installed life with 100% design demand.

The Emergency Power System is single failure tolerant in accordance with the requirements of IEEE 379-2014 (Reference 8.3-19).

The Emergency Power System design supports the recommended practices of IEEE 450-2020 (Reference 8.3-20) and conforms to RG 1.129.

The Emergency Power System design includes provisions for bypassing an inoperable status indication in accordance with USNRC BTP 8-5.

The Emergency Power System, the Standby Power System, and the Preferred Power System battery chargers and UPSs disconnect from the AC input whenever that input varies outside the range of voltage from nominal for a set period of time.

The Emergency Power System, the Standby Power System, and the Preferred Power System battery chargers are able to function as battery eliminators.

The Emergency Power System, the Standby Power System, and the Preferred Power System battery chargers produce a filtered DC output that meets the requirements of IEC 61000-4:2022, “Electromagnetic Compatibility (EMC) – Part 4: Testing and Measurement Techniques,” (Reference 8.3-21).

The Emergency Power System, the Standby Power System, and the Preferred Power System battery chargers include overvoltage protection on the output that shuts down the charger upon actuation. This meets the recommendations of IEEE 946-2020, “Recommended Practice for the Design of DC Power Systems for Stationary Applications,” (Reference 8.3-22).

The Emergency Power System, the Standby Power System, and the Preferred Power System battery chargers are of a current limiting design to provide additional overload protection for DC loads, including the batteries, and prevents the AC source from becoming a load on the batteries. This requirement meets the recommendations of IEEE 946-2020 (Reference 8.3-22).

The plant DC systems consist of an ungrounded system with ground detection. This requirement is based the recommendations of IEEE 142-2007, “Recommended Practice for Grounding of Industrial and Commercial Power Systems,” (Reference 8.3-23) for station battery systems and on sound engineering practice for DC systems to prevent interruption of DC power when a single line-to-ground fault occurs.

All UPSs maintain rated output voltage within an acceptable tolerance from zero to rated load. This value meets the range of service voltages provided in ANSI C84.1-2020, “American National

Standard for Electric Power Systems and Equipment – Voltage Ratings (60 Hertz)," (Reference 8.3-24).

All branch circuit protective devices supplied by the UPS are evaluated to ensure coordination with the UPS short circuit shutoff magnitude and time. This requirement prevents the complete loss of the circuits supplied by the UPS in the case of a faulted branch circuit.

The UPS bypass Regulating Voltage Transformers (RVTs) maintain a nominal output voltage over a varied input voltage for a given range. This requirement establishes that a well-conditioned and regulated output voltage is provided even when the UPSs are bypassed.

Transient Voltage Surge Suppressors (TVSSs) are used on the input circuit to battery chargers, UPSs, and associated bypass circuits to protect against voltage surges. This requirement supports compliance with RG 1.204.

In accordance with IEEE 1050-2004 (Reference 8.3-6), adequate filtering is provided by the battery charger, UPS, and regulating transformers to ensure that propagation of Electromagnetic Interference/Radio Frequency Interference (EMI/RFI) of the levels allowed by IEC 61000-3:2024, "Electromagnetic Compatibility – Part 3: Limit – All Parts," (Reference 8.3-25).

The Emergency Power System design meets the recommended practices in IEEE 946-2020 (Reference 8.3-22) for design of DC power systems for stationary applications.

A single failure in any individual Emergency Power System division does not adversely affect the other Emergency Power System divisions. This requirement is based on IEEE 603-2018 (Reference 8.3-16).

A single failure in the DC portion of the Emergency Power System does not result in conditions that prevent safe shutdown of the plant coincident with a separate Emergency Power System division being out of service. This requirement is based on IEEE 603-2018 (Reference 8.3-16).

Emergency Power System Divisions are physically and electrically separated and isolated from each other without electrical crosstie to other divisions or battery systems. This requirement supports compliance with RG 1.6.

The Emergency Power System batteries meet the qualification requirements of IEC/IEEE 60780-323:2016, "Nuclear Facilities – Electrical Equipment Important to Safety – Qualification," (Reference 8.3-26), IEC/IEEE 60980-344-2020, "International Standard - Nuclear Facilities - Equipment Important to Safety - Seismic Qualification," (Reference 8.3-27), and IEEE 535-2022, "Standard for Qualification of Class 1E Vented Lead Acid Storage Batteries for Nuclear Power Generating Stations," (Reference 8.3-28).

The Emergency Power System divisional batteries are sized with an expected service life of 20 years using the methodology given in IEEE 485-2020, "Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications," (Reference 8.3-29) and conforms to RG 1.212.

The Emergency Power System batteries have enough stored capacity to independently supply loads continuously for at least 72 hours with a minimum final discharge voltage sufficient to support the associated loads. The 72 hour mission time is consistent with the guidance in SECY-94-084 for passive plants.

Each of the Emergency Power System UPSs can supply 100% of the divisional load. This requirement is based on IEEE 603-2018 (Reference 8.3-16).

The Emergency Power System UPS outputs provide a single-phase nominal AC voltage of 240 VAC, 60 Hz.

The Emergency Power System battery banks are designed to supply DC power at 250 VDC nominal.

8.3.2.1.1 Emergency Power Uninterruptible Power Supply

The UPS in each division provides a single-phase 240 VAC output. All Emergency Power System UPSs have both AC and DC source inputs (the AC input is rectified internal to the UPS to match the DC input), allowing battery testing without losing the divisional outputs (albeit the 72 hour mission time is unavailable from the battery in test during that time). In addition, the UPS has a Static Transfer Switch (STS) which provides a bumpless transfer to an RVT upon loss of the inverter. In normal alignment the STS is inverter seeking. The UPS also contains a Manual Transfer Switch (MTS) as a means of transferring the loads to the RVT. In this configuration, maintenance or testing of a UPS can be done online without disturbing UPS loads. An MTS is included on the input side of the UPS and RVT to allow alignment to one of two redundant sources from the Standby Power System.

The power panels ultimately feed the individual loads. Where required, the UPSs are protected at the individual load level with current limiting fuses such that the resulting fault currents are within the transient rated current capability of the inverter. Individual faults have no effect on other loads on the same power panel. The DCIS cabinets are redundantly powered from the Emergency Power System divisional UPS and a DC supply directly from a DC distribution panel fed from the battery bus. Loads with the ability to accept only a single supply, such as scram and isolation valve solenoids, SCR emergency lighting, and SCR ventilation are supplied by the UPS. These loads are designed with sufficient redundancy so that a loss of a single UPS will not result in a plant transient. Therefore, failure of one divisional UPS has no adverse effect on plant operations or safety.

8.3.2.1.2 Physical Identification of Safety Class 1 Equipment

Electrical and control equipment, panels and racks, and cables and raceways grouped into separate divisions are identified so that their electrical divisional assignment is apparent, and so that an observer can visually differentiate between SC1 equipment and wiring of different divisions, and between SC1 and non-SC1 equipment and wiring. The BWRX-300 equipment, cabling separation and electrical isolation are based on function and performed in accordance with the requirements of IEEE 384-2018 (Reference 8.3-10), IEC 60709:2018 (Reference 8.3-11), and RG 1.75.

8.3.2.1.3 Independence of Redundant Systems

Each of the Emergency Power System divisional rooms in the RB are physically separate from and individually fire-barriered and flood-barriered from the other Emergency Power System divisional rooms.

Raceway and Cable Routing

Equipment, cabling separation, and electrical isolation are based on IEEE 384-2018 (Reference 8.3-10) and IEC 60709 (Reference 8.3-11). Separation and isolation by functions supports conformance with RG 1.75.

Separation Techniques

Where component and circuit independence are required by general safety standards such as International Atomic Energy Agency (IAEA) safety guides, IEC 61513, “Nuclear Power Plants – Instrumentation and Control Important to Safety – General Requirements for Systems,” (Reference 8.3-30), IEC 63046, “Nuclear Power Plants – Electrical Power System – General Requirements,” (Reference 8.3-31), and other project constraints, two ways of achieving this

independence are physical separation and electrical isolation between the components and circuits on the basis of defense lines.

It is noted that traditional separation and isolation are performed in part based on safety category function, but NEDC-33934P, “BWRX-300 Safety Strategy,” (Reference 8.3-34), is predicated on redundancy and diversity, which do not correlate to safety category function in every case. As such, electrical and Instrumentation and Control (I&C) cable and component separation and isolation is based on defense lines.

8.3.2.1.4 Safety Class 1 to Non-Safety Class 1 Isolation

Anticipated electrical faults are selectively isolated by protective devices. Such isolation is accomplished by ensuring acceptable coordination between protective devices in accordance with the requirements and recommendations of IEEE 308-2020, (Reference 8.3-14), IEEE 242-2001 (Reference 8.3-32), and IEEE 741-2022 (Reference 8.3-15).

Bus arrangement, load allocation, and equipment and cable separation prevent any single fault from compromising the performance of the safety category function.

8.3.2.2 Non-Safety Class 1 DC and UPS System

The on-site DC power system description for each of the two subsystems is provided below.

Standby Power 250 VDC Subsystem

The most important loads on the Standby Power System 600 VAC buses are the SC1 Emergency Power UPS, battery chargers, and RVTs. Next in line as a matter of importance are the SC2 Standby Power System LGs for the UPSs, battery chargers, and RVTs. Refer to Figure 8.3-3.

The loads assigned to the SC2 Standby Power System UPSs and batteries are those required for the BWRX-300 to achieve shutdown and include the equipment that provides a diverse, active means for reactor vessel injection and to remove heat from the reactor vessel and containment. The Standby Power 250 VDC Subsystem, depicted in Figure 8.3-4, is designed to carry the loads via dedicated batteries, independent of offsite or diesel AC power. The SC2 Standby Power System LV LGs include surge protection via a TVSS to prevent HV spikes propagating to the battery chargers, UPSs, and RVTs. The Standby Power System includes two LGs. Each LG has an individual UPS in support of the respective SC2 DCIS division, along with other loads. The LG UPSs provide three-phase 208/120 VAC outputs. Redundant circuits are provided from separate UPSs to support redundant power sources to the DCIS.

The Standby Power 250 VDC Subsystem includes the following design features:

1. Any SC2 load requiring power for completion of a Safety Category 2 function is powered from the SC2 UPSs, batteries or RVTs. The Standby Power 250 VDC Subsystem is designed with redundancy such that the failure of a single electrical bus or MCC has no adverse effect on the redundant counterpart.
2. Battery banks are sized such that the sum of the required loads does not exceed the battery ampere hour rating over the rated discharge period at the end of the installed life with 100% design demand.
3. Batteries have enough stored capacity without chargers to independently supply loads continuously with a minimum final discharge voltage sufficient to support the associated loads.
4. Redundant SC2 LG battery chargers support equalization and battery testing without effects to downstream loads.

5. LG UPS outputs provide a three-phase nominal AC voltage of 208/120 VAC, 60 Hz.
6. LG Battery banks are designed to supply DC power at 250 VDC nominal.
7. DC and UPS systems are powered from diesel-backed sources.

Preferred Power 250 VDC Power Subsystem

Preferred Power 250 VDC Subsystem is classified as SC3 and is designed to provide power to SC3 and SCN DC loads, such as SC3 DCIS, protective relaying, and breaker control power. As shown in Figure 8.3-5, two independent 250 VDC batteries are provided, supplying power to two independent 250 VDC buses. The redundant configuration is compliant with North American Electric Reliability Corporation (NERC) and Northeast Power Coordinating Council requirements for separate backup power for Bulk Power Systems. A battery mission time of eight hours is selected to align with typical designs for Bulk Power Systems substations. Each DC bus and each battery is normally supplied power and charging current from its associated charger. A standby charger capable of being aligned to either bus or battery is provided in case any given charger is out of service. Each charger is designed to supply 100% of the required load of a single bus while charging the battery. The two primary chargers derive LV AC power from separate SDG backed buses. The standby charger is supplied through a transfer switch, which can derive power from either SDG backed LV buses. The standby charger connection to the SDG buses provides operational flexibility in the event of an extended bus outage.

As shown in Figure 8.3-6, two independent UPSs are provided that supply a three-phase 208/120 VAC output to independent buses. The UPSs derive power from their associated 250 VDC bus. The UPSs are equipped with both static and manual transfer switches aligned to an RVT. The RVTs are normally energized and derive power from a separate AC source than that which supplies the associated battery charger. The static transfer switch operates automatically as needed for UPS output faults.

The following design features are applicable to the Preferred Power 250 VDC subsystem:

1. The Preferred Power 250 VDC Subsystem UPS outputs provide a three-phase nominal AC voltage of 208/120 VAC, 60 Hz.
2. The Preferred Power 250 VDC Subsystem battery banks are designed to supply DC power at 250 VDC nominal.
3. The Preferred Power 250 VDC Subsystem batteries have enough stored capacity without chargers to independently supply loads continuously for at least eight hours, with a minimum final discharge voltage sufficient to support the associated loads.
4. Preferred Power 250 VDC Subsystem batteries are physically separated and electrically independent in accordance with the applicable requirements of the grid connection authority having jurisdiction.
5. The Preferred Power 250 VDC Subsystem batteries are valve-regulated lead-acid (VRLA) batteries and conform to RG 1.212 and endorsed IEEE Standard 485. IEEE 485 is used for battery sizing and related design parameters. Standards IEEE 1187 and IEEE 1188, cited by IEEE 485 for VRLA batteries, are also used for design, installation, maintenance, and testing of the batteries.

8.3.2.3 Analysis

The DC subsystems are designed to perform their associated function without operator interaction. Upon loss of AC power, the systems automatically transition to being supplied by the associated battery.

Input AC power for the DC subsystems is identified in the figures included in this chapter.

The on-site DC power subsystems are expected to perform their design functions in normal and abnormal conditions. Electrical power is supplied to the I&C systems. Refer to Chapter 7 for further discussion.

The BWRX-300 electrical systems are monitored and controlled by DCIS equipment. Generally, the DCIS control systems are expected to operate and monitor the plant transformers and feeder breakers, including load breakers for pumps, adjustable speed drives, or other plant equipment.

MV breakers are controlled by protective relays to protect the electrical system components. For example, the condensate pump motor breaker includes protective relays that monitor overcurrent, balanced phases, undervoltage, underfrequency, and ground and phase faults. The protective relay trips the breaker, and a trip command has precedence over a close command. The plant DCIS can command a breaker to open or close, but a DCIS close command can be overridden by the protective relay. Either a DCIS trip or protective relay trip command opens the breaker. The protective relays are interfaced to the plant DCIS and can provide multiple signals to the attached plant controller. Modern microprocessor based protective relays are addressed as digital assets.

The design of the plant electrical systems accommodates online testing and inspection to the extent practical. Condition monitoring equipment is included where appropriate. For the Emergency Power System, the UPSs are in redundant pairs for each division, and testing can be done online without losing the complete divisional UPS output.

Required electrical system testing is governed by the plant technical specification and industry standards. The electrical systems are designed to allow required tests to be performed with minimal disturbance to the balance of the system. Specific steps and alignments necessary for testing are controlled by plant procedures.

Maintenance can be done during outages. However, given the redundancy in the electrical system, most maintenance can also be done with the plant online.

While the expected service life of the plant battery is less than the 60-year service life of the plant, the battery is optimized to the greatest extent possible by providing optimal environmental conditions (e.g., temperature and humidity) and limiting the discharge occasions to periodic surveillance and preventative maintenance testing.

Depending on the type, a battery may periodically need a maintenance equalizing charge that raises battery voltage to levels that can cause stress on battery loads. With the redundant battery charger design configuration, one charger can supply the equalizing charge to the battery and be isolated from the DC loads while the other charger carries the DC loads at normal voltages. It is noted that DC loads are designed to withstand the equalization voltage.

8.3.2.4 Cable Raceway System

The cable and raceway system includes raceway and supports necessary to route cable in and between buildings and equipment using cable trays, conduits, duct banks, or tunnels. The cable and raceway system also includes the primary containment electrical penetration assemblies.

The cabling and raceways are designed to meet the following SC design requirements:

1. Equipment, cabling separation, and electrical isolation are based on IEEE 384-2018 (Reference 8.3-10) and IEC 60709:2018 (Reference 8.3-11). This supports compliance with RG 1.75.

2. Fire protection and fire safe shutdown program requirements are supported by cable separation, fire barriers, limiting cable tray fill, limiting cable ampacity to levels that prevent overheating and insulation failures (and resultant possibility of fire), and by use of fire resistant and non-propagating cable insulation. Copper cables within the power block are qualified to IEEE 1202-2006 (Reference 8.3-12) and IEEE 383-2023 (Reference 8.3-13). Fiber optic cables within the power block are qualified to IEEE 1202-2006 (Reference 8.3-12).
3. IEEE 317-2013 (Reference 8.3-2) for containment electrical penetrations.

8.3.2.4.1 Cable Separation

The cabling for the BWRX-300 provides electrical power to designated loads.

Equipment, cabling separation, and electrical isolation are based on defense line function and performed in accordance with the requirements of IEEE 384-2018 (Reference 8.3-10) and IEC 60709:2018 (Reference 8.3-11). Separation for SCN components and systems are derived from the application of IEC 60709:2018 (Reference 8.3-11) as appropriate.

The cables and raceways are designed commensurate with the safety category functions being provided.

8.3.2.4.2 Raceway and Cable Routing

Cables and raceways are separated on the basis of defense lines, division, and voltage class. The raceways are physically arranged top to bottom based on the function and the voltage class of the cables. Cables are routed in horizontal and vertical runs of steel trays or through steel conduits. Trays or conduits are supported at intervals by appropriately designed supports. The supports are attached to walls, floors, and ceilings of structures as required by the arrangement. The type of support and spacing is determined by allowable tray or conduit spans which are governed by rigidity and stress. Underground raceways have limited application for the BWRX-300. When underground raceways are required, they are appropriately sloped and drained to minimize the potential for cable submergence.

8.3.2.4.3 Control of Compliance with Separation Criteria During Design and Installation

Normally the cabling and raceways are designed to not require any inspections once they have passed installation and commissioning testing. However, cabling and raceways are inspected based on operations surveillance programs, maintenance, and testing programs during normal operations and outages as appropriate.

Cabling and raceways testing is based on preventive and predictive maintenance programs. Site testing requirements are also used for post installation testing and inspection.

Electrical and control equipment, panels and racks, and cables and raceways grouped into separate divisions are identified so that their electrical divisional assignment is apparent, and so that an observer can visually differentiate between SC1 equipment and wiring of different divisions, and between SC1 and non SC1 equipment and wiring. The identification method is color-coding. Markers within a division have the same color. The BWRX-300 equipment, cabling separation, and electrical isolation are based on defense line function and performed in accordance with the requirements of IEEE 384-2018 (Reference 8.3-10), IEC 60709:2018 (Reference 8.3-11), and RG 1.75. Divisional separation requirements of individual pieces of hardware are shown in the system elementary diagrams. Identification of raceways and cables is compatible with the identification of the SC1 equipment with which it interfaces. Location of the

identification is such that points of change of circuit classification (at isolation devices) are readily apparent.

Cabling and raceways are passive components that provide power from electrical supplies to components and/or loads.

While electrical cables transmit I&C signals throughout the plant, there are no instrumention and control features specific to the electrical cabling and raceway systems.

The cables and raceways are designed to perform required functions under all anticipated normal and abnormal conditions.

Installation of the cabling and raceways is per approved procedures and testing requirements to ensure they can perform their intended functions.

8.3.2.4.4 Hazard Protection

Where redundant SC1 and non SC1 raceway systems traverse each other, separation in accordance with RG 1.75 and IEEE 384-2018 (Reference 8.3-10) is maintained.

Where hazards to SC1 raceways are identified, a predetermined minimum separation is maintained between the break and/or missile source and any SC1 raceway, or a barrier designed to withstand the effects of the hazard is placed to prevent damage to raceway of redundant systems. For details of missile protection and high-energy line break protection, see Sections 3.5 and 3.6, respectively.

Where redundant circuits, devices, or equipment (different separation groups) are exposed to the same external hazard(s), predetermined spatial separation is provided. Where the spatial separation cannot be met; qualified barriers are installed. For details on fire protection, see Section 9.5.

8.3.2.5 Inspection and Testing

The Emergency Power System is designed to accommodate required surveillance testing in accordance with the requirements of IEEE 338-2022, (Reference 8.3-17) and complies with RG 1.118.

Condition monitoring equipment is included where appropriate to support inspection activities and ensure equipment performance and health is as desired. Required electrical system testing is governed by the plant Technical Specifications and industry standards. The electrical systems are designed to allow required testing with minimal disturbance to the balance of the system. Specific steps and alignments necessary for testing are controlled by plant procedures.

8.3.2.5.1 Safety Class 1 Battery Capacity Testing

Maintenance, testing, and replacement activities for the SC1 EDS batteries are based on recommended practices in IEEE 450-2020 (Reference 8.3-20).

Each Emergency Power System division has its own battery and two redundant UPSs. The two UPSs in each division are primarily for the SC1 DCIS system to provide redundant divisional power feeds. Loss of any individual divisional UPS still allows the DCIS of that division to function. Therefore, reliability is enhanced with the redundant UPSs that provide power to each division of the DCIS. In addition, all UPSs normally operate directly from AC power, allowing battery testing without losing the divisional outputs (albeit the 72 hour coping time is unavailable from the battery in test during that time).

The plant electrical system battery banks are designed to permit replacement of individual cells.

The design of the plant electrical systems accommodates online testing and inspection to the extent practical. Condition monitoring equipment is included where appropriate. For the Emergency Power System, the UPSs are in pairs per division, and testing can be done online without losing the complete divisional UPS output.

The Emergency Power System batteries require periodic testing under load to verify capacity. During battery testing, the associated DCIS loads remain powered because the UPSs have AC power available.

The battery may periodically need a maintenance equalizing charge that raises battery voltage to levels that can cause stress on battery loads. While all DC loads are designed to withstand the equalization voltage, the redundant battery charger design configuration allows one charger to supply the equalizing charge to the battery and be isolated from the DC loads while the other charger carries the DC loads at normal voltages.

Required electrical system testing is governed by the Technical Specifications and industry standards. The SC1 batteries are designed to allow required testing with minimal disturbance to the balance of the system.

8.3.2.5.2 Safety Class 1 Inverter Capacity Testing

Each Emergency Power System UPS can be manually bypassed allowing the loads to be supplied directly from a RVT, allowing testing of the inverter and STS.

Required electrical system testing is governed by the Technical Specifications and industry standards. The SC1 inverters are designed to allow testing with minimal disturbance to the balance of the system.

8.3.2.5.3 Safety Class 1 Charger Capacity Testing

There are two battery chargers per division. Each battery charger has the capacity to power all loads for the division while also charging the battery. The redundant battery chargers and bus configuration are arranged to provide operational flexibility to the system.

Required electrical system testing is governed by the Technical Specifications and industry standards. The SC1 chargers are designed to allow testing with minimal disturbance to the balance of the system.

8.3.2.5.4 Safety Class 1 Auxiliary Equipment Capacity Testing

Required electrical system testing is governed by the Technical Specification and industry standards. The SC1 Auxiliary Equipment is designed to allow testing with minimal disturbance to the balance of the system.

8.3.2.5.5 Non-Safety Class 1 Electrical System Capacity Testing

Testing is conducted in accordance with applicable industry standards and vendor recommendations.

8.3.3 References

8.3-1 ANSI/NEMA MG1-2021, “Motors and Generators,” National Electrical Manufacturers Association, 2021.

8.3-2 IEEE 317-2013, “Standard for Electric Penetration Assemblies in Containment Structures or Nuclear Power Generating Stations,” Institute of Electrical and Electronics Engineers, 2013.

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8.3-4 IEEE 81-2012, “Guide for Measuring Earth Resistivity, Ground Impedance, and Earth Surface Potentials of a Grounding System,” Institute of Electrical and Electronics Engineers, 2012.

8.3-5 IEEE 665-1995, “Guide for Generating Station Grounding,” Institute of Electrical and Electronics Engineers, 1995.

8.3-6 IEEE 1050-2004, “Guide for Instrumentation and Control Equipment Grounding in Generating Stations,” Institute of Electrical and Electronics Engineers, 2005.

8.3-7 NFPA 780-2022, “Standard for the Installation of Lightning Protection Systems,” 2022.

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8.3-9 IEEE 666-2007, “Design Guide for Electrical Power Service Systems for Generating Stations,” Institute of Electrical and Electronics Engineers, 2007.

8.3-10 IEEE 384-2018, “Standard Criteria for Independence of Class 1E Equipment and Circuits,” Institute of Electrical and Electronics Engineers, 2018.

8.3-11 IEC 60709, “Instrumentation, Control and Electrical Power Systems Important to Safety – Separation,” International Electrotechnical Commission, 2018.

8.3-12 IEEE 1202-2006, “Standard for Flame-Propagation Testing of Wire and Cable,” Institute of Electrical and Electronic Engineers, 2006.

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8.3-14 IEEE 308-2020, “Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations,” Institute of Electrical and Electronics Engineers, 2020.

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8.3-19 IEEE 379-2014, “Standard for Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems,” Institute of Electrical and Electronics Engineers, 2014.

8.3-20 IEEE 450-2020, “Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications,” Institute of Electrical and Electronics Engineers, 2020.

8.3-21 IEC 61000-4 SER, “Electromagnetic Compatibility – Part 4: Testing and Measurement Techniques,” International Electrotechnical Commission, 2022.

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8.3-25 IEC 61000-3 Series, SER, “Electromagnetic Compatibility – Part 3: Limit – All Parts,” International Electrotechnical Commission, 2024.

8.3-26 IEC/IEEE 60780-323, “Nuclear Facilities – Electrical Equipment Important to Safety – Qualification,” Institute of Electrical and Electronics Engineers, 2016.

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8.3-33 IEEE 399-1997, “IEEE Recommended Practice for Industrial and Commercial Power System Analysis (Brown Book),” Institute of Electrical and Electronics Engineers, 1997.

8.3-34 NEDC-33934P, “BWRX-300 Safety Strategy,” GE Vernova Hitachi Nuclear Energy Americas, LLC, Revision 2, March 2026.

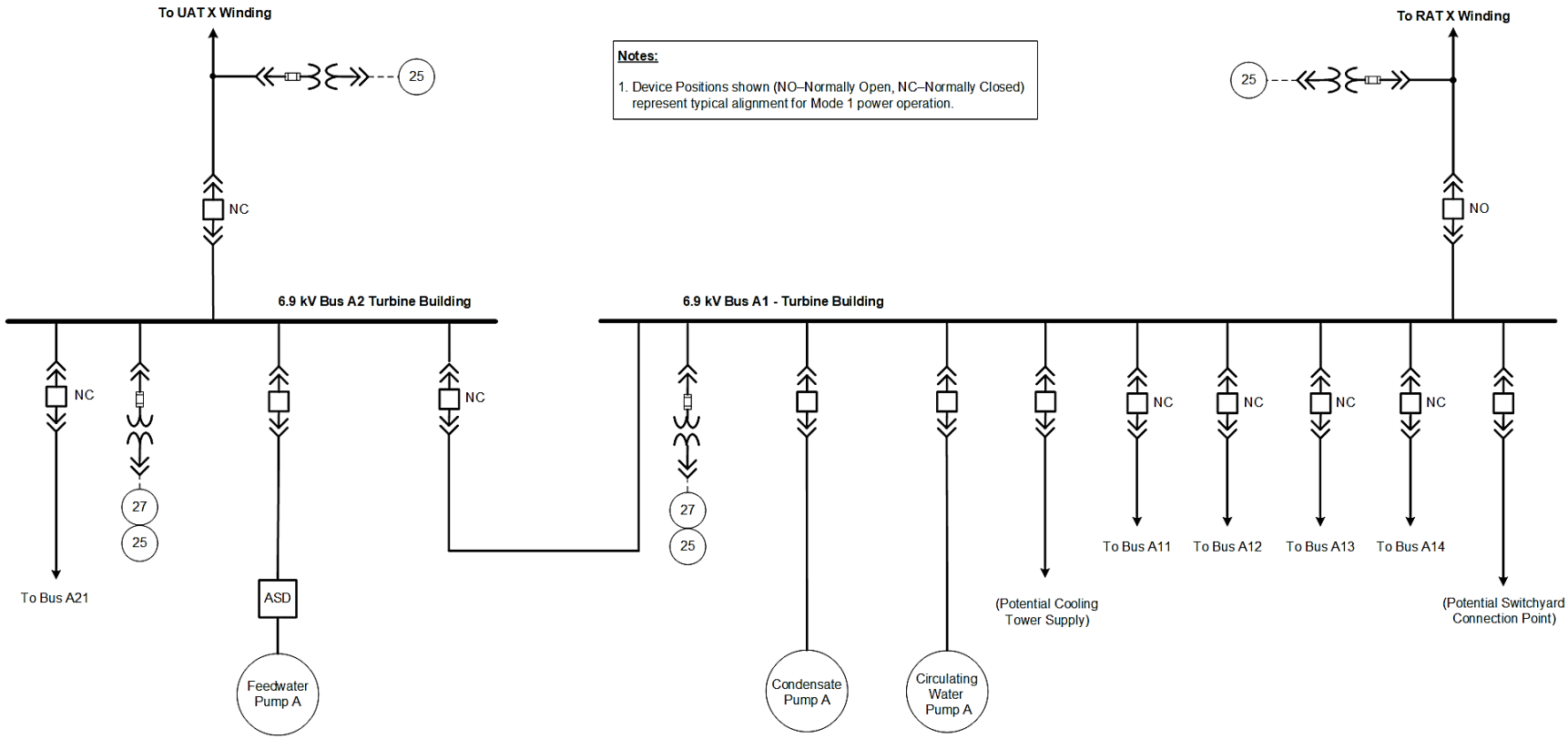


Figure 8.3-1: Preferred Power Medium Voltage Bus A Train

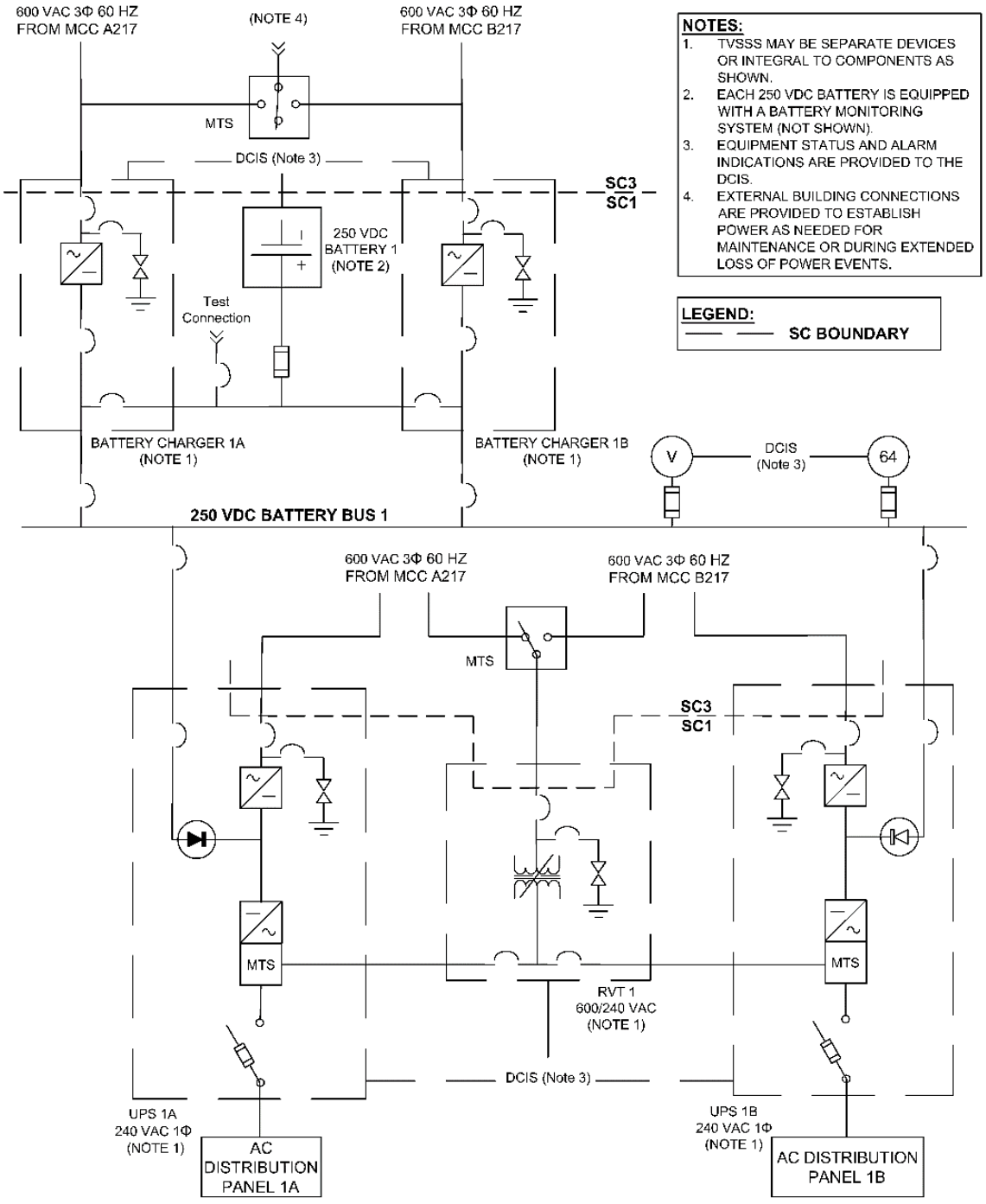


Figure 8.3-2: Emergency Power System

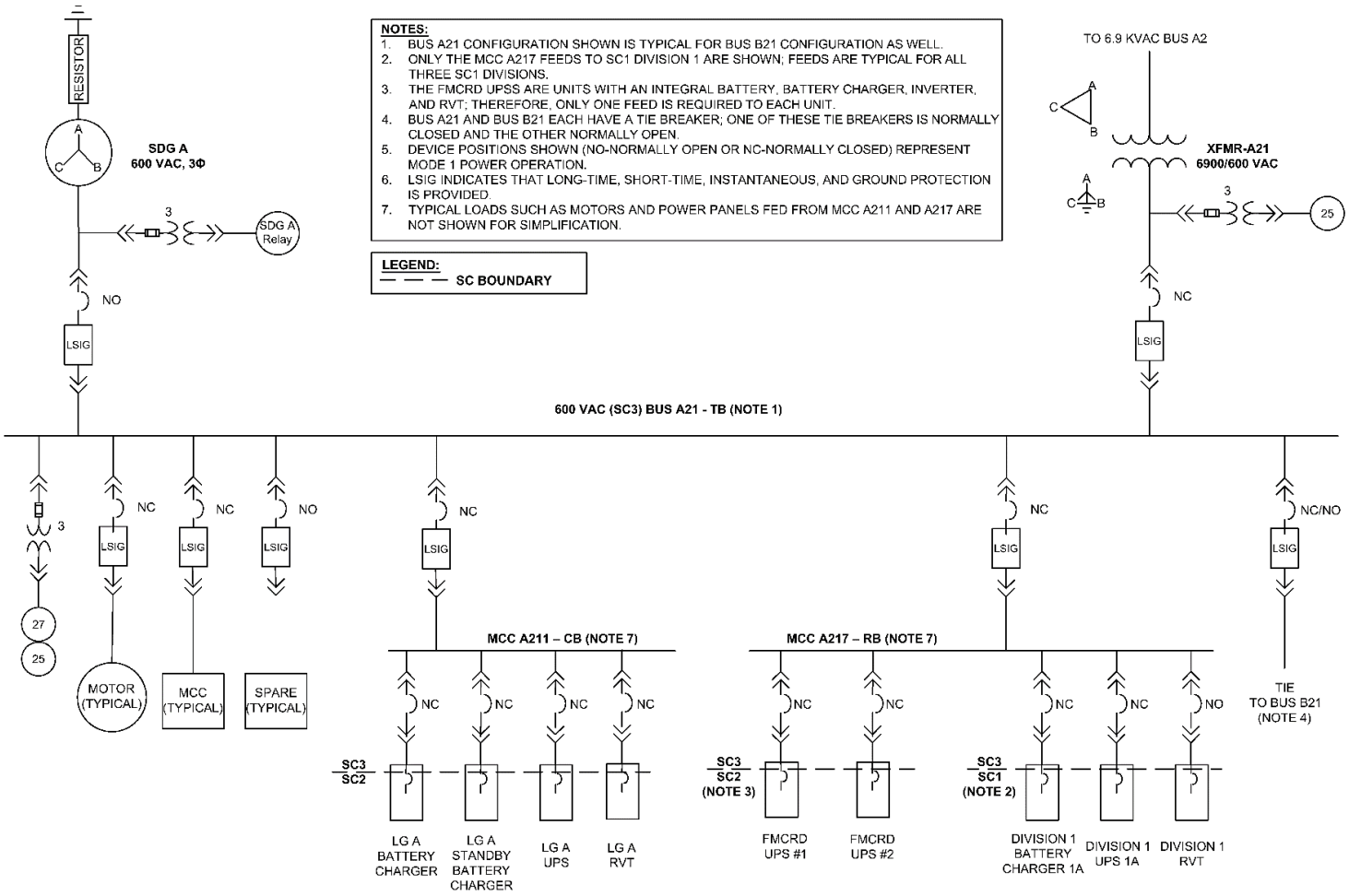


Figure 8.3-3: Standby Power System Low Voltage Distribution Train A

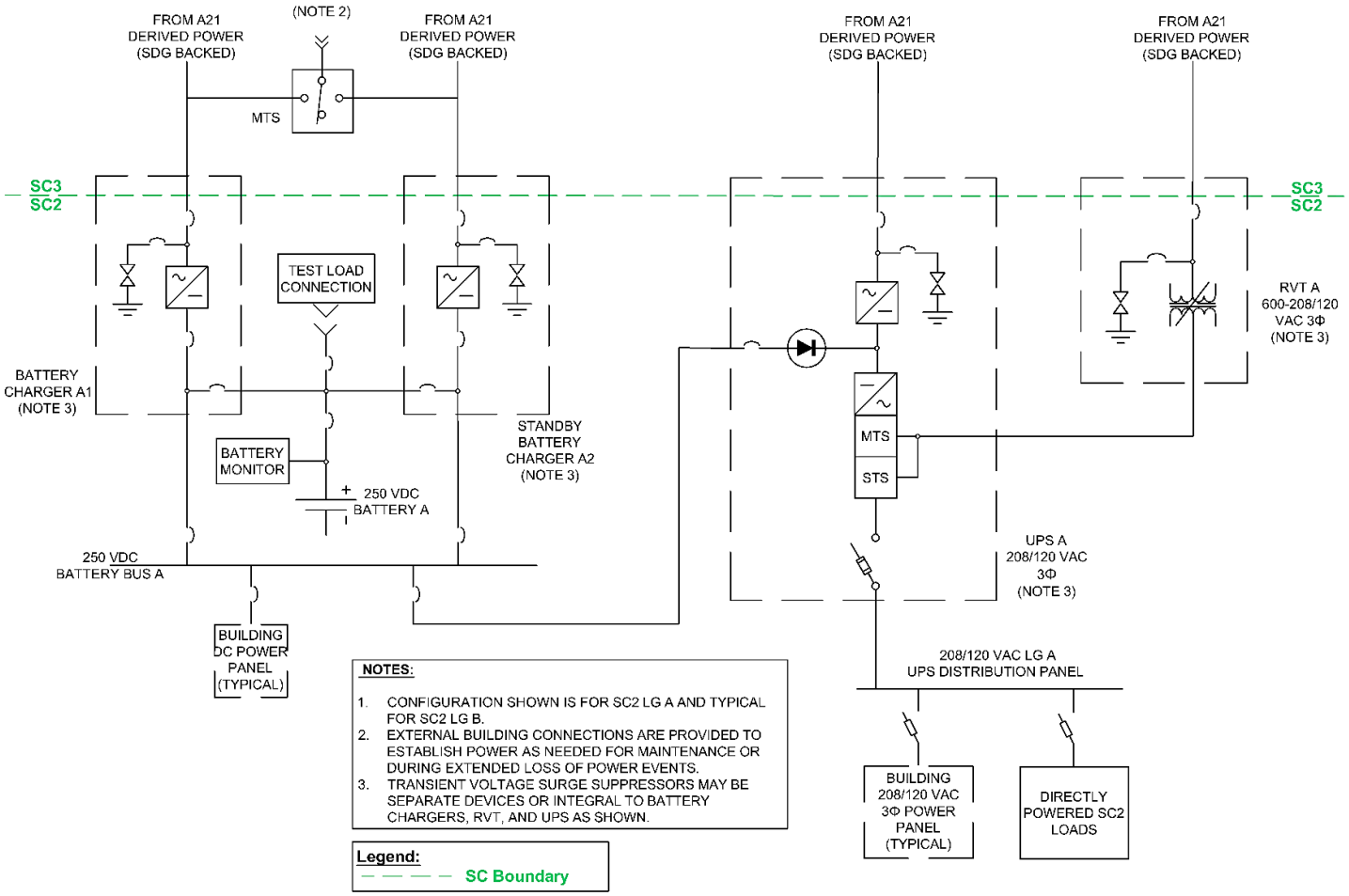


Figure 8.3-4: Standby Power System – 250 VDC System

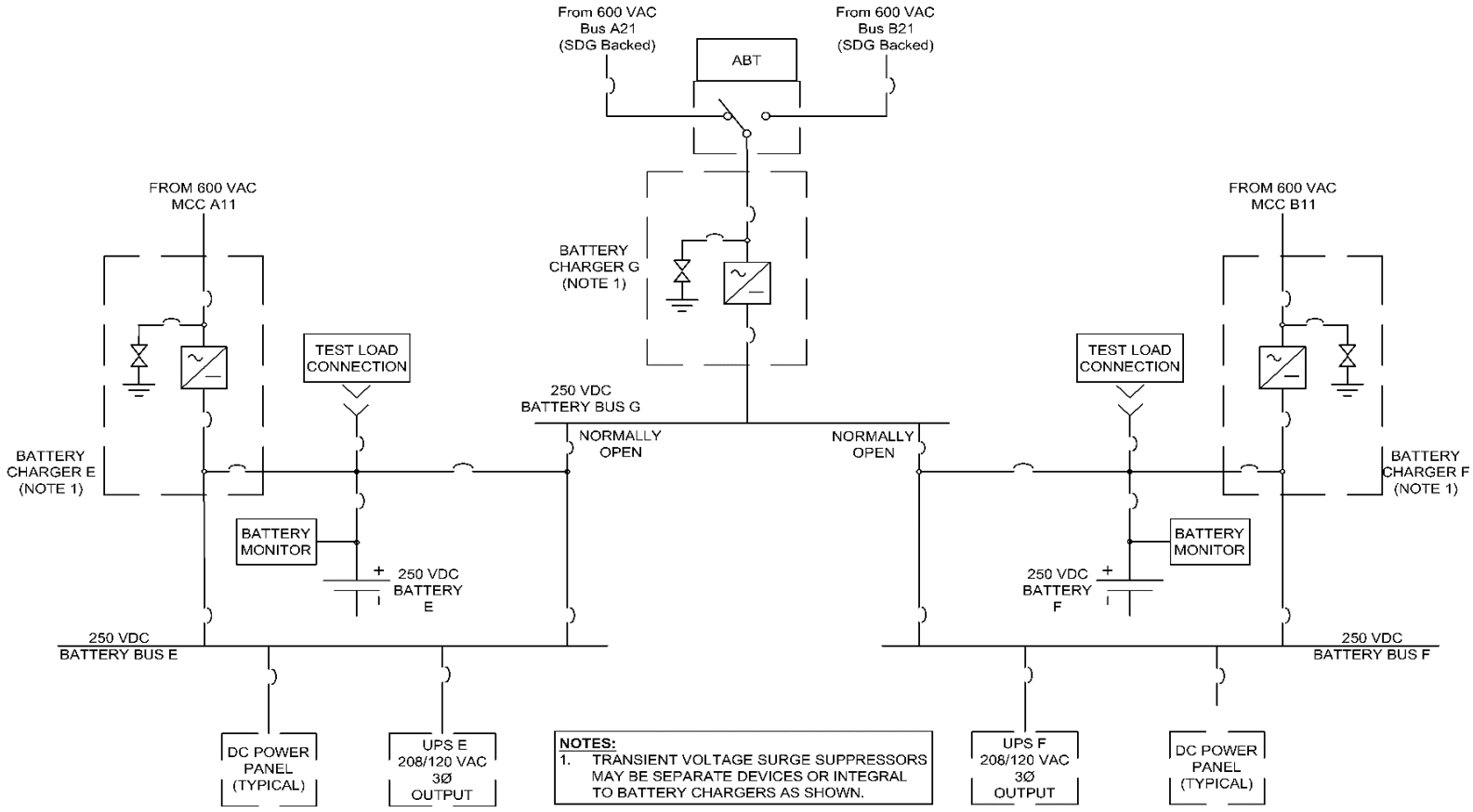


Figure 8.3-5: Preferred Power 250 VDC Subsystem Part One

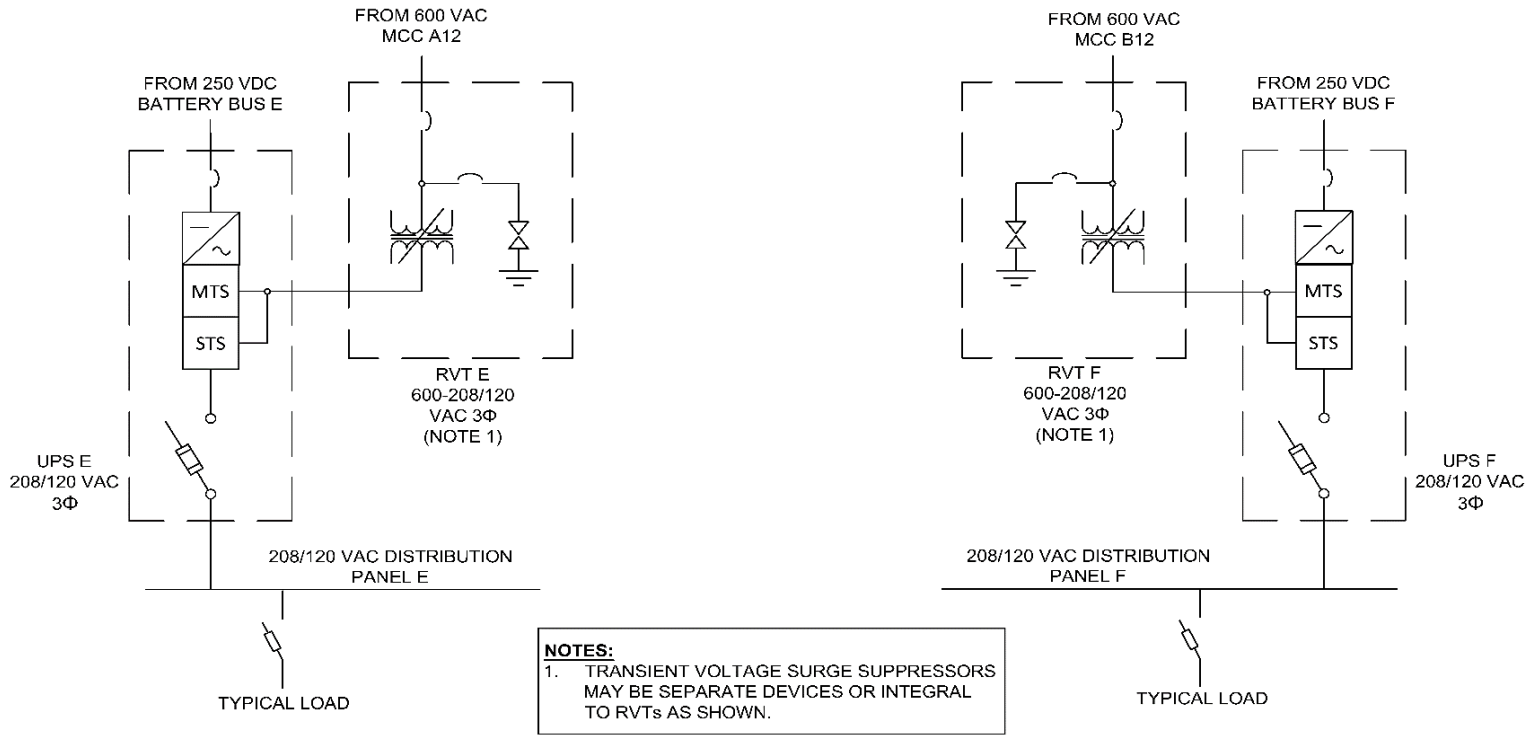


Figure 8.3-6: Preferred Power 250 VDC Subsystem Part Two

8.4 Station Blackout

The BWRX-300 design minimizes the reliance on electrical power to support safety class functions. The passive design of the plant is not dependent upon off site AC power sources or diesel generators to mitigate a DBA. The evaluation of a LOPP, with no reliance on the SDGs, is presented in Subsection 15.5.5.2.4. SC1 power is supplied from battery backed DC power which has a coping period of 72 hours for all DBAs. SC1 DC power sources are provided to support passive core cooling and containment Safety Category functions. This complies with the requirements of 10 CFR 50.63.

See Subsection 8.1.3 for compliance with GDCs.

Blue Energy Global Inc.

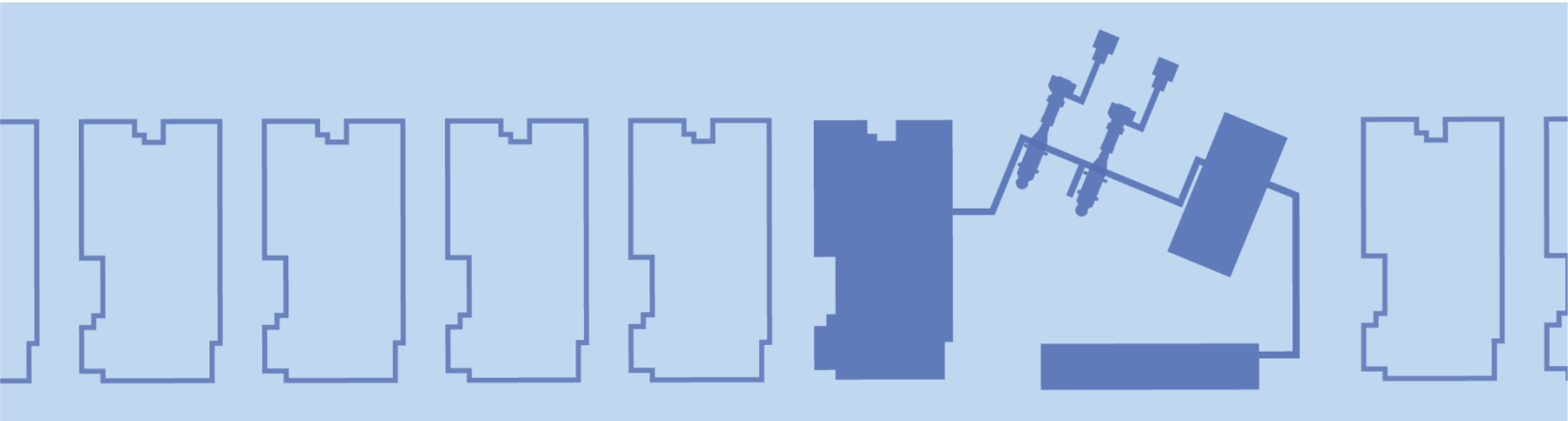
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September 2026

Non-Proprietary Information

Chapter 9
Section 9.1 – Fuel Storage and Handling



ACRONYM LIST

Acronym	Explanation
AHU	Air Handling Unit
ALARA	As Low As Reasonably Achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASME	American Society of mechanical Engineers
BPVC	Boiler and Pressure Vessel Code
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
CST	Condensate Storage Tank
DCIS	Distributed Control and Information System
EME	Emergency Mitigating Equipment
FPC	Fuel Pool Cooling and Cleanup System
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
GNF	Global Nuclear Fuel
I&C	Instrumentation and Controls
ISI	Inservice Inspection
ITP	Initial Test Program
LOCA	Loss-of-Coolant Accident
LWM	Liquid Waste Management System
MOX	Mixed Uranium-Plutonium Oxide
NEI	Nuclear Energy Institute
OHLHS	Overhead Heavy Load Handling System
PCCS	Passive Containment Cooling System
PCW	Plant Cooling Water System
RB	Reactor Building
RG	Regulatory Guide
RPV	Reactor Pressure Vessel
SC	Steel-Plate Composite
SCCV	Steel-Plate Composite Containment Vessel
SRP	Standard Review Plan

Acronym	Explanation
SSCs	Structures, Systems, and Components
SSE	Safe Shutdown Earthquake
UHS	Ultimate Heat Sink
USNRC	U.S. Nuclear Regulatory Commission

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9.0 AUXILIARY SYSTEMS

9.1 Fuel Storage and Handling

9.1.1 New Fuel Storage

The BWRX-300 design does not include separate new fuel storage facilities such as new fuel storage vaults, new fuel storage pool, and new fuel storage racks. Following receipt and inspection, new fuel is placed in storage racks located in the fuel pool. The storage racks used for new fuel storage are the same as those used for spent fuel storage. Because new fuel is stored with spent fuel in the same fuel pool, the Design Basis, Facilities Description and Safety Evaluation associated with spent fuel, as presented in Subsection 9.1.2, also addresses storage of new fuel in the fuel pool. The criticality safety analysis is discussed in Section 9.1.A.

9.1.2 New and Spent Fuel Storage

The new fuel assemblies and spent fuel assemblies are stored in fuel storage racks which are in the fuel pool. The fuel pool is in the Reactor Building (RB). The above grade-portion of the RB structure houses the refuel floor, refueling and fuel handling systems, fuel pool, water needed for the BWRX-300 passive cooling systems, and polar crane. The structures that comprise the fuel pool include the fuel storage racks, the fuel pool liner, and the walls/floor of the RB integral with the fuel pool.

9.1.2.1 Design Bases

Safety Category 1 Functions

- Maintain and protect water in the fuel pool. Refer to Subsection 9.2.5 regarding the Ultimate Heat Sink (UHS) function of the water in the pool.
- Provide safe, effective, and traceable storage of new and spent fuel

Safety Category 2 Functions

None.

Safety Category 3 Functions

None.

Safety Category N Functions

None.

The Safety Classification, Quality Groups, and Seismic Categories of the fuel pool and associated Structures, Systems, and Components (SSCs) are listed in Table 3A-1.

9.1.2.2 Fuel Facilities Description

Fuel Pool

The fuel pool is used to store spent fuel after removal from the reactor and new fuel after delivery to the site and before core loading. An area in the fuel pool is allocated for spent fuel cask loading operations. The fuel storage racks interface with the refueling platform fuel mast grapple, allowing for unobstructed installation and removal of fuel assemblies.

The deep pit in the fuel pool provides storage for spent fuel removed from the core during refueling activities. New fuel is staged in fuel storage racks located in the fuel pool. An area in the deep pit is used for loading of a spent fuel cask.

The fuel pool is adjacent to the reactor cavity pool, with the capability of being isolated by means of a removable gate. The equipment pool, opposite the other end of the cavity, holds the dryer and separator during the refueling outage. The equipment pool is capable of being isolated by a removable gate.

The fuel pool base mat is located at approximately the power block grade elevation. The fuel pool extends up to the refuel floor elevation.

The fuel pool size is determined by the volume of cooling water, other equipment required to be stored in the pool, including various refueling outage equipment and lay-down areas. Refueling outage equipment such as lights, test weights, and control rod blades are also stored in the fuel pool. Used reactor core instruments may also be stored in the fuel pool, until such time as they are packaged.

SSCs in the vicinity of the fuel storage racks are evaluated as either Seismic Category I or Seismic Category II to confirm that during or following a Safe Shutdown Earthquake (SSE) their interaction with spent fuel could not result in an increase of k_{eff} beyond the maximum allowable k_{eff}. Refer to Section 9.1.A for information pertaining to criticality analysis associated with the fuel pool and fuel storage racks.

The pool is constructed of a steel and concrete composite as described in Subsection 3.8.4. Figure 9.1-1 presents the general arrangement of the fuel pool, reactor cavity pool, and equipment pool.

The liner is a stainless steel or equivalent corrosion resistant material. Even though liner corrosion is not anticipated, a leak detection system is provided to detect and monitor fuel pool liner leaks.

For normal conditions of operation, the fuel pool is cooled by the Fuel Pool Cooling and Cleanup System (FPC) (Subsection 9.1.3). The fuel pool is designed with a gate and weir that prevents drainage of coolant inventory below an adequate shielding depth, and the ability to provide adequate coolant flow to the fuel racks.

In the event of a loss of the fuel pool cooling and cleanup functions, the primary method of maintaining fuel pool temperature is boiling of water in the fuel pool. Water level and temperature in the fuel pool is monitored to ensure adequate water level and temperature. Level instrumentation is maintained in accordance with Regulatory Guide (RG) 1.227. The design of the fuel pool liner is capable of withstanding the high temperatures associated with a boiling pool.

The refueling platform spans the reactor vessel cavity and fuel pool and is employed to handle fuel and perform other ancillary tasks in the RB. It is equipped with a traversing trolley on which is mounted a telescoping mast and a fuel grapple. The refueling platform is a rigid structure built to ensure accurate and repeatable positioning during the refueling process. Additional information pertaining to the refueling platform is provided in Subsection 9.1.4.

Shielding is provided by maintaining a minimum water level above the top of stored fuel assemblies.

As discussed in Subsection 9.1.5, the RB polar crane meets single-failure-proof criteria which provides protection from damage due to drops of heavy loads in the fuel storage area.

Fuel Storage and Rack Design

The fuel assemblies are stored in the fuel storage racks that are arranged in a grid pattern in the fuel pool as shown in Figure 9.1-2. The fuel storage racks provide space for storage of new fuel as well as space for longer term storage of the spent fuel assemblies for cooling. The fuel racks hold a single batch of new fuel and at least five years of spent fuel and have space remaining to accept a full core of off-loaded fuel. The fuel storage racks prevent inadvertent criticality. The

racks are top loading, with fuel bail extended above the rack. The fuel pool racks maintain a subcriticality of at least 0.05 Δk under flooded conditions. The rack arrangement is designed to prevent accidental insertion of fuel assemblies between adjacent racks and allows flow to prevent nucleate boiling for all fuel assemblies. This analysis assumes the BWRX-300 core design utilizes a GNF2 fuel assembly or an equivalent fuel configuration.

The racks are designed to withstand the effect of a stuck fuel assembly. The uplift force is evaluated for an event that the fuel or grappling device should hang up during removal of fuel. The racks are designed to allow natural convection through the rack and fuel for the removal of decay heat under normal and abnormal operating conditions. This is consistent with the guidance of RG 1.13, “Spent Fuel Storage Facility Design Basis,” Position C.11. For the case of loss of both FPC trains (Subsection 9.1.3), boiling is allowed to occur in the racks. The structural acceptance criteria for the fuel storage racks are that the storage rack design does not exceed the allowable stress levels given in the American Society of Mechanical Engineers, (ASME) Boiler and Pressure Vessel Code (BPVC), Section III “Rules for Construction of Nuclear Facility Components,” (Reference 9.1-1), Division I, Subsection NF, “Class 1, 2, 3, and MC Supports.” Boiling at the bottom of the rack is assumed to occur with a full head of pool water (i.e., water at normal pool level).

The fuel storage racks are used for the safe, effective, and traceable storage of new and spent fuel. After receipt and inspection of new fuel, the new fuel is placed directly into the fuel storage racks prior to insertion into the Reactor Pressure Vessel (RPV). Safe handling of fuel includes design considerations for maintaining occupational radiation exposures “As Low As Reasonably Achievable” (ALARA).

The fuel storage racks are designed to accommodate the bulge, twist, and bow associated with irradiated fuel assemblies.

The fuel storage racks are designed to accommodate inspection for the presence of a neutron poison at the time of installation and during the life of the equipment. A program for monitoring the effectiveness of the neutron poison present in the neutron absorbing panels is to be maintained.

The fuel storage rack components are designed and qualified to operate within their performance requirements under the anticipated ranges of the normal plant environment. The fuel storage racks are also designed and qualified to operate within their performance requirements under the anticipated ranges of the abnormal or accident plant environment and are designed to withstand an SSE without failure of the basic structure or damage to the active region of irradiated fuel.

The fuel storage racks are designed to ensure an acceptable margin of subcriticality less than 0.05 when combined with other multiplication factors as specified in American National Standards Institute (ANSI)/American Nuclear Society (ANS) 57.2 “Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants,” (Reference 9.1-2).

The fuel storage racks are designed to limit the effect of a dropped fuel assembly such that fuel in the storage racks remains in a subcritical array. A postulated worst case fuel handling accident is evaluated in Subsection 15.5.8.

The neutron absorbing material used to construct the fuel racks is permanently marked with identification traceable to the material certifications. The rack design ensures the fuel assembly handles are the highest point to allow for clear pick up and set down activities.

9.1.2.3 Safety Evaluation

The fuel pool complies with the following General Design Criteria (GDC), which have plant-wide applicability, as discussed in Section 3.1:

GDC 2 – Design bases for protection against natural phenomena

GDC 4 – Environmental and dynamic effects design bases

GDC 5 – Sharing of structures, systems, and components

The following GDC have aspects specific to the fuel pool liner and fuel racks, and compliance is addressed as follows:

GDC 61 – as it relates to the design for fuel storage and radioactivity control

GDC 63 – as it relates to monitoring for detection of radiological conditions and initiating appropriate safety actions

Additional information on GDC compliance is found in Section 3.1.

The fuel pool, liner and fuel storage racks are designed to comply with the following regulations:

Compliance with 10 CFR 20.1101(b) as to radiation doses kept as low as reasonably achievable.

Compliance with 10 CFR 50.68 as to design to preclude criticality accidents.

The design of the fuel pool liner and fuel racks complies with GDC 2 and GDC 4 and conforms to RG 1.13 “Spent Fuel Storage Facility Design Basis,” positions C.1 and C.2. The liner and fuel storage racks are designed to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including Loss-of-Coolant Accident (LOCA) and the dynamic effects of equipment failure. They also are designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, and floods without loss of capability to perform their safety functions. Guidance from RG 1.29 “Seismic Design Classification for Nuclear Power Plants,” and RG 1.117 “Protection Against Extreme Wind Events and Missiles for Nuclear Power Plants” are also considered. The liner and fuel storage racks meet Seismic Category I requirements and are protected from interaction with non-Seismic Category I SSCs to ensure that the SSE does not cause a loss of capability to perform their safety functions. The fuel pool liner and storage racks are protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluid, which may result from equipment failures and events and conditions outside the nuclear power unit.

Fuel storage and handling facilities are designed to comply with GDC 61 as described below

GDC 61 Sub Criterion 1 - Periodic Inspection and Testing of Important to Safety Components

Fuel storage racks are provided in the fuel pool adjacent to the reactor cavity. The fuel storage racks are designed to accommodate inspection for the presence of a neutron poison at the time of installation and during the life of the equipment. The fuel racks are top loading, allowing for identification of individual fuel assemblies from the poolside or refuel platform.

The fuel pool itself is part of the RB structure. As identified in Section 5.18 of NEDC-33926P-A, “BWRX-300 Steel-Plate Composite Containment Vessel and Reactor Building Structural Design,” Revision 2, (Reference 9.1-3), an Inservice Inspection (ISI) and testing program is established for the RB to ensure the structure can fulfill its intended functions throughout its design service life. Leak detection is provided to monitor the integrity of the fuel pool liner.

Fuel pool gate seals are designed to be inspected, tested, and properly maintained as needed.

GDC 61 Sub Criterion 2 - Shielding for Radiation Protection

The fuel pool provides shielding by maintaining a minimum water level above the top of stored fuel assemblies. Adequate shielding is also provided for transporting fuel between the vessel and fuel pool. Nominal water levels provided by the weirs maintain at least 23 feet of water above the top of the active fuel in the storage racks during refueling. This water depth is sufficient for scrubbing of fission gases released from damaged fuel rods during a postulated fuel handling accident per Subsection 15.5.8. Water depth also maintains adequate shielding for the lifted fuel assembly being transported between the fuel storage racks and the RPV. Liquid level sensors are installed to detect low pool water level. Refer to Sub Criterion 5 for more on Drainage Prevention.

GDC 61 Sub Criterion 3 - Containment, Confinement, and Filtering

The fuel pool liner, fuel pool, and the walls of the RB integral with the fuel pool provide containment of the fuel assemblies and the water cooling the assemblies. Subsection 9.1.3 describes the FPC which removes radionuclides from the fuel pool water by collecting them on a filter or in a demineralizer. The FPC has no drains, piping, or other systems that would allow fuel pool water to drain below the level needed for adequate shielding of the fuel assemblies. Refer to Sub Criterion 5 for more on Drainage Prevention.

GDC 61 Sub Criterion 4 - Residual Heat Removal

The FPC described in Subsection 9.1.3 normally removes decay heat from the fuel pool and other interconnected pools. The active heat removal portion of the FPC is not designed to withstand accident conditions and, therefore, the fuel pool structure, including liners are designed to withstand coolant boiling. Sufficient pool water inventory is provided to allow seven days of spent fuel decay heat to be absorbed without any additional inventory addition, while maintaining sufficient coverage of 10 feet over fuel. If required, makeup water is provided from onsite sources as described in Subsection 9.1.3.

The fuel storage racks are designed in a manner that allows for adequate coolant flow to stored fuel assemblies. The fuel rack design allows natural convection to ensure adequate cooling water flow to all stored fuel assemblies removing decay heat under normal and abnormal conditions.

Subsection 9.2.5 describes the water in the fuel pool is the UHS for fuel in the pool at all times and contributes towards UHS during refueling.

GDC 61 Sub Criterion 5 - Drainage Prevention

During power operations, the fuel pool is configured such that the fuel pool gate is installed, and the adjacent reactor cavity pool is filled. There are no drains, piping, or other systems installed in the fuel pool that would allow coolant levels to drain below adequate shielding depths of above the top of the fuel assemblies.

Weirs are designed to prevent the coolant inventory from being drained below the top of the fuel seated in the rack, while maintaining sufficient coverage of 10 ft over fuel even after 7 days of decay heat removal. The surge tank(s) receive inventory from the reactor cavity, equipment pool, and fuel pool as it overflows weirs located near the top of the reactor cavity and spent fuel pools. Surge tank(s) are utilized to provide natural protection from a breach in the FPC from draining the pool volume (versus direct suction) and provide a damping effect for volume changes in the pools due to outage or dry cask evolutions.

During plant outages, there are alternate configurations of the fuel pool. Conformance to RG 1.13 for these configurations are discussed in Section 1.5. The adjacent reactor cavity pool may be drained with the fuel pool gate installed to allow access to the Steel-Plate Composite Containment

Vessel (SCCV) and RPV heads for either installation or removal. Additionally, the fuel pool gate may be removed to join the fuel pool, reactor cavity pool, and the RPV to support underwater fuel transfer between the fuel pool and RPV.

The removable gate that separates the fuel pool from other pools is designed to support the full height of water remaining on the fuel pool side with the reactor cavity pool side completely drained. The fuel pool gate has two radiation-resistant seals in series, one non-inflatable static seal on the reactor cavity pool side and one pneumatic perimeter seal that is pressurized. The pneumatic seal is controlled from a control station in the RB. Supply air is normally provided from the Plant Pneumatics System (PPS), described in Subsection 9.3.1. The pneumatic seal control station has a backup air supply. The fuel pool gate is designed as Seismic Category I and designed in accordance with ANSI/ANS 57.2.

The reactor cavity pool permits maintaining water level at a height above the core required for shielding during fuel transfer. Appropriate seals are provided. The refueling bellows seal design includes protection from puncture or damage from dropped items during refueling activities. Controls to prevent inadvertent draining of the reactor cavity during refueling activities via the drain valves associated with the refueling bellows seal are to be evaluated in the Final Safety Analysis Report (FSAR). Due to the fuel racks located in the deep pit of the fuel pool, coolant inventory would not be reduced to a level below the top of fuel assemblies, even during a failure of the fuel pool gate or refueling bellows seal during refueling activities.

In compliance with GDC 63, RG 1.13 position C.7, and paragraph 5.4 of ANSI/ANS 57.2 (Reference 9.1-2), the fuel pool is provided with instrumentation to detect conditions that may cause loss of residual heat removal capability and excessive radiation levels, and to initiate appropriate actions. Instrumentation is provided by the FPC (Subsection 9.1.3) for fuel pool water level, pool temperature, and radiation monitoring. Instrumentation of the FPC generally falls into two categories. Instrumentation used to provide data which may be used for a control function, or as an input used to determine train alignment, is a triplex signal to provide redundancy and indication of a fault. Instrumentation intended only for local indication or alarm is single-channel, consistent with BWRX-300 instrumentation development.

In addition, a fuel handling accident with the potential to cause the greatest damage to stored fuel is analyzed to conform with U.S. Nuclear Regulatory Commission (USNRC) limits for radioactivity releases and avoidance of excessive personnel radiation exposure.

Consistent with the requirements of 10 CFR 20.1101(b), to the extent practicable, procedures and engineering controls based on sound radiation protection principles are employed to achieve ALARA occupational doses and doses to the public. The FPC design conforms with ANSI/ANS 57.2, and the FPC maintains the water level of the fuel pool as discussed in Subsection 9.1.3. The guidance of ANSI/ANS 57.2 paragraph 5.1 is followed for maintaining safe radiation levels for personnel during anticipated operating and accident conditions. The FPC also maintains the water quality of the fuel pool, equipment pool, and reactor cavity pool through filtration and demineralization. Refer to Section 11.5 (Process Radiation Monitoring) and Subsection 12.3.4 (Area Radiation and Airborne Radioactivity Monitoring Instrumentation) for radiation monitoring and alarm descriptions. Information pertaining to design of SSCs relative to ALARA is provided in Section 12.3.

Refer to Section 9.1.A regarding the fuel pool criticality analysis and 10 CFR 50.68 requirements.

9.1.2.4 Inspection and Testing

Fuel pool level instrumentation permits routine testing and calibration in-situ. Inspection and testing are to verify that there is no corrosion of the fuel pool liner or fuel storage racks, no buildup of crud or debris that may obstruct coolant flow in the fuel pool, as well as no degradation of any

strong fixed neutron absorbers. The design of the fuel pool liner and fuel storage racks facilitates inspections and testing. Most surfaces of these structures are accessible for underwater inspection.

Fuel pool gate seals are designed to allow for periodic inspection and testing for leakage as required.

9.1.2.5 Instrumentation and Controls

Instrumentation for the fuel pool is provided by the FPC. Refer to Subsection 9.1.3.7 for FPC Instrumentation and Controls (I&C).

9.1.3 Fuel Pool Cooling and Cleanup System

The functions of the FPC are to provide cooling of the water volume in the fuel pool to remove decay heat from spent fuel, and to provide replacement coolant inventory from a variety of sources, and to ensure spent fuel is kept cool and submerged until relocated for permanent storage. In addition, the FPC includes demineralization and particulate filtration to maintain coolant quality and to reduce general area dose. The FPC can be aligned to provide cooling and cleanup to the reactor cavity and equipment pools, as necessary.

9.1.3.1 Design Bases

Safety Category 1 Functions

- Refer to Subsection 9.2.5 regarding the UHS functions of the water in the fuel pool, and also of water in the equipment pool and reactor cavity pool for Passive Containment Cooling System (PCCS)
- Refer to Subsection 9.1.2 regarding the fuel pool structure functions to protect the water required for cooling, shielding, and subcriticality

Safety Category 2 Functions

None.

Safety Category 3 Functions

- Cooling of the fuel pool, reactor cavity pool, and equipment pool
- Makeup water capabilities for the fuel pool, reactor cavity pool, and equipment pool, delivered directly to the surge tanks and circulated to the pools
- Maintain the water level of the fuel pool for shielding and cooling
- Direct makeup water capabilities for the fuel pool during accident conditions, independent of the forced cooling portion of the system
- Maintain the water level of the fuel pool for shielding and cooling during accident conditions

Safety Category N Functions

- Maintain the water quality of the fuel pool, reactor cavity pool, and equipment pool through filtration and demineralization
- Restore the fuel pool temperature to normal operating limits from elevated temperatures, caused by a system outage or accident conditions, upon restoration of the forced cooling components of the system

The Safety Classifications, Quality Groups, and Seismic Categories of the FPC SSCs are listed in Table 3A-1.

9.1.3.2 System Description

Refer to Figure 9.1-3 which depicts the FPC.

The FPC is intended to maintain temperatures during plant operation and refueling outages and, as such, the operation of the system is primarily driven by bulk pool temperatures.

The FPC consists of two trains of equipment, each with a pump, a filter, and two heat exchangers. There is one demineralizer which can be aligned to either train. The capability to bypass the demineralizer while providing active cooling of the pool temperature from conditions exceeding the demineralizer operational temperature limits is provided. Each set of components are placed in parallel to provide single train operation and cross connecting of trains should a component fail. A single train is sufficient to prevent bulk boiling in the fuel pool under maximum expected fuel pool heat loads. If both trains are rendered inoperable, the fuel pool is sized such that it can retain sufficient coverage of the fuel for seven days, and the FPC can provide makeup water provisions independently of the forced cooling trains.

9.1.3.3 Modes of FPC Operations

The FPC operates in the following system modes:

9.1.3.3.1 Normal Operation

The FPC is designed to maintain maximum normal bulk temperatures in the fuel pool below 60°C (140°F). Heat removal capabilities during normal operation of the FPC are also sufficient to manage a full core offload during refueling while maintaining bulk temperatures in the fuel pool below 60°C (140°F).

Mode A – Normal Heat Load

The most frequently used mode of system operation of the FPC is “normal heat load” operation, intended to be used during all plant modes. Mode A includes a single train of the FPC operating to remove heat generated by spent fuel stored in the fuel pool. Cleanup consisting of particulate filtration and demineralization of the water passing through the system is performed. During Mode A, coolant inventory can be added from the Condensate Storage Tank (CST) as required to combat evaporation or to replace inventory following the removal of equipment. Excess inventory may be over-boarded to the CST after cleanup and cooling to create additional pool volume for submersion of spent fuel dry cask equipment.

Mode B – High Heat Load

Mode B utilizes both trains of FPC equipment running simultaneously to increase the volume of water cooled and cleaned. This mode still retains full filtration and demineralization of a single train and is intended to be used predominately during refueling outages, when high-activity spent fuel is transferred to the fuel pool, and water clarity and dose are of particular concern. As with Mode A, coolant inventory can be added or over-boarded to support outage operations and maintain water level.

9.1.3.3.2 Off-Normal Operation

The FPC is designed to respond to events that are the result of potential system failure, degraded equipment, and external events. The FPC may be placed in system Modes C and D. In all plant operational modes, it is expected that these events result in the plant transitioning to a shutdown state. The FPC may operate as such:

Mode C – High Heat Load – Bypass

This mode utilizes both trains of FPC equipment identical to high heat load operation, except the demineralizer is isolated from the system and a bypass is utilized.. Mixed bed resin demineralizers typically have operating limits to remain below 140°F, which when exceeded can cause resin excursions contaminating plant equipment and pools. To retain the high heat removal capability but prevent damage to the resin beds during high pool temperature events, bypass may be utilized until the water temperature is restored to a threshold allowing high heat load operation. Bypass may also be used to restore active cooling after an extended loss of cooling event.

Mode D – Active Cooling Inoperative

This mode is a condition where the active cooling and cleanup portion of the FPC is unable to be operated and consists of multiple sources for coolant addition directly to the fuel pool, bypassing the remaining system or any potential breeches that may have occurred. A makeup line utilizes multiple sources to maintain pool level in the event of a failure of the active cooling portion of the FPC. Sources include the CST (Subsection 9.2.6) and FLEX/Emergency Mitigating Equipment (EME). The fuel pool volume is intended to contain sufficient water inventory to allow seven days of spent fuel decay heat to be absorbed, without any additional inventory addition, while maintaining sufficient coverage of at least 10 ft over fuel.

The FLEX/EME connection is in an accessible location outside the RB. It is Seismic Category I and routed through the RB to the fuel pool to allow for external water sources to replenish the fuel pool within seven days if normal makeup from the CST is unavailable. As stated in Section 13.3, the FSAR will include Diverse and Flexible Coping Strategies that provide sufficient equipment and supporting staff to enable fuel pool cooling functions to be maintained.

9.1.3.4 Component Description

The following is a description of major equipment and components of the FPC.

Surge Tank

The surge tanks receive inventory from the equipment pool, reactor cavity pool, and fuel pool as it overflows weirs located near the top of the pools. Surge tanks are utilized to provide natural protection from a breach in the system draining either pool volume (versus direct suction) and provide a damping effect for volume changes in the pools due to outage or dry cask evolutions.

Pumps

Each train of the FPC begins with a pump. Pumps can be run individually for normal heat removal and cleanup, or in combination for high heat load operations.

Heat Exchangers

Each train includes two heat exchangers (four in total) used to reject waste heat to the Plant Cooling Water System (PCW) (Subsection 9.2.2). Each heat exchanger is sized to fit within the RB hatch to allow removal or replacement during the operating life of the plant.

Demineralizers/Filters

Each train of the FPC contains a particulate filter which can support 100% of the train's flow. The filter may be backwashed to the Liquid Waste Management System (LWM) to remove accumulation and reduce dose. A single deep bed mixed resin demineralizer can be aligned to either FPC train and can support 100% of a single train's flow. A mesh to trap resins or particles is downstream of the deep bed demineralizer. The demineralizer operates in both normal (A) and high heat (B) modes but can also be bypassed if the temperature of the system coolant exceeds the operational limits of the demineralizer (Mode C).

9.1.3.5 Safety Evaluation

The FPC complies with the following GDC, which have plant-wide applicability, as discussed in Section 3.1:

GDC 2 – Design bases for protection against natural phenomena

GDC 4 – Environmental and dynamic effects design bases

GDC 5 – Sharing of structures, systems, and components

The following GDC have aspects specific to the FPC, and compliance is addressed as follows:

GDC 61 – Fuel storage and handling and radioactivity control, as it relates to handling of radioactive materials

GDC 63 – Monitoring fuel and waste storage, as it relates to detecting conditions that may result in loss of residual heat removal capability and excessive radiation levels

Additional information on GDC compliance is found in Section 3.1.

The fuel pool is designed to comply with the following regulation:

Compliance with 10 CFR 20.1101(b) as to radiation doses kept ALARA.

Consistent with GDC 61, the FPC provides containment of radioactivity by providing a system for detecting and containing fuel pool liner leaks. The fuel pool is located inside of the RB, and confinement is afforded by virtue of the RB design to limit the potential release of radioactive materials. The design of the FPC has no drains, piping, or other systems that would allow fuel pool water to drain below the minimum level needed to support plant safety analyses, which is above the level needed for adequate shielding of the fuel assemblies. In the event of a fuel handling accident on the refuel floor in the Refuel Platform area, a radiation monitor alarm initiates closure of the RB isolation dampers and secures the RB upper-level supply Air Handling Units (AHUs).

The function of the FPC is to provide cooling of the water volume in the fuel pool to remove decay heat from spent fuel, and to provide replacement coolant inventory from a variety of sources, to ensure spent fuel is kept cool and submerged until relocated for permanent storage. The FPC is able to restore the fuel pool temperature to normal operating limits from elevated temperatures, caused by a system outage or an accident event, upon restoration of the forced cooling components of the system. The FPC provides the capability and provisions for removing radioactive materials, corrosion products, and impurities from the fuel pool water, thus meeting

the requirements of GDC 61 as it relates to filtering systems associated with fuel cooling and storage.

The FPC design meets the requirements of GDC 63. Reliable and frequently tested monitoring equipment are provided to alarm both locally and in a continuously manned location if the water level in the fuel storage pool falls below a predetermined level, if the water temperature exceeds a predetermined level, or if high local radiation levels are experienced. Radiation monitoring and alarms are addressed in Section 11.5. The above alarms provide notification of abnormal conditions for operators to initiate timely actions to ensure the safety functions are satisfied.

Information pertaining to design of SSCs relative to ALARA is provided in Section 12.3.

9.1.3.6 Inspection and Testing

FPC components are designed to facilitate maintenance and inspection during operation. Because portions of the FPC operate continuously, operation of key components is monitored via instrumentation.

Startup testing of the FPC is performed per the Initial Test Program (ITP) (Chapter 14).

9.1.3.7 Instrumentation and Controls

Refer to Section 7.3 for Distributed Control and Information System (DCIS) functions, and to Section 11.5 for radiation monitoring and alarms. Monitoring is provided for fuel pool water level and temperature.

The design of the instrumentation provided for monitoring fuel pool water level, fuel pool water temperature, and local area radiation, detects the loss of heat removal function. Remote monitoring of wide-range fuel pool levels to support implementation of event mitigation and recovery actions is provided in accordance with RG 1.227, “Wide-Range Spent Fuel Pool Level Instrumentation.”

Instrumentation of the FPC generally falls into two categories. Instrumentation used to provide data which may be used for a control function, or as an input used to determine train alignment, is a triplex signal to provide redundancy and indication of a fault. Instrumentation intended only for local indication or alarm is single-channel, consistent with BWRX-300 instrumentation development. During normal operation, pumped capacity is varied through use of one or both system trains, with output balanced to the equipment, reactor cavity, and fuel pool to distribute cooled water in a manner best matching the sources of thermal energy. Addition or reduction of inventory, both as a natural function of environmental conditions, as well as supporting maintenance and outage evolutions, can be performed via administrative procedures.

During off-normal events, with the forced cooling portion of the system assumed to be inoperable, makeup capacity can be initiated from various sources as redundant fuel pool level indication requires.

Major components (pumps, demineralizers, heat exchangers) have performance indication through the I&C System, and component protection such as bypassing of the demineralizer if system temperature exceeds limits and stopping of pumps on low flow and low suction pressure signals.

9.1.4 Light Load Handling System

The Light Load Handling System consists of the SSCs for handling new and spent fuel during its lifecycle (receipt, refueling, and loading into shipping or storage casks). The Light Load Handling System uses equipment during fuel movement and refueling operations designed to lift loads weighing up to and including a single fuel assembly and its handling device. Refer to

Subsection 9.1.5 for overhead heavy load handling equipment, such as the polar crane, designed to handle greater loads than a single fuel assembly and its handling device.

9.1.4.1 Design Bases

Safety Category 1 Functions

None.

Safety Category 2 Functions

None.

Safety Category 3 Functions

None.

Safety Category N Functions

- Support the periodic refueling of the reactor as well as movement of control rods and other radioactive components within the reactor core, equipment pool, reactor cavity pool, and fuel pool.

The Safety Classifications, Quality Groups, and Seismic Categories of the refueling equipment associated with the Light Load Handling System are discussed in Table 3A-1.

9.1.4.2 System Description

The Light Load Handling System consists of the equipment and structures needed for fuel storage and the refueling operation. The equipment is used to transfer new and irradiated fuel between the RPV, and fuel storage racks located in the fuel pool. The Light Load Handling System is also used for core maintenance activities, such as fuel shuffling and replacement of components in the reactor core. The structures associated with the fuel handling equipment are the equipment pool, reactor cavity pool, fuel pool, equipment pool gate, and fuel pool gate. The fuel handling equipment is designed to convey a fuel assembly underwater from the time it is lowered into the fuel pool, until such time as the spent fuel assembly is placed in a spent fuel transfer cask. Underwater transfer of spent fuel assemblies provides radiation shielding, as well as a medium for removal of decay heat.

Equipment design and layout consider personnel safety, radiation exposure, and human factors engineering. Figure 9.1-4 depicts the location of equipment on the refuel floor. The refueling bellows seal is described in Subsection 6.2.1.2.1 and provides a watertight seal between the RPV and containment to support a pool of water over the vessel of sufficient depth to allow fuel bundle transfer during refueling. The design includes protection from puncture or damage from dropped items during refueling activities. Drainage prevention is discussed in Subsection 9.1.2 with regards to GDC 61 compliance, as well as fuel pool design and shielding for radiation protection of operators.

9.1.4.3 Refueling Operations

Fuel handling activities are outlined below and illustrated in Figures 9.1-5 through 9.1-9. These illustrated activities include:

- Containment Closure Head Removal
- Reactor Pressure Vessel Head Removal
- Steam Dryer Removal
- Chimney Head and Separator Assembly Removal

- Fuel Handling

Overhead heavy load lifts of the closure head, RPV head, steam dryer, and assembly removals are through the use of the polar crane and is described in Subsection 9.1.5. Fuel handling is the only activity specified above that is applicable to this section.

The RB structure provides adequate space for operation, maintenance, and removal of equipment.

The equipment pool holds the dryer and steam separator assemblies. The equipment pool is capable of being isolated by a removable gate. The fuel pool, adjacent to the reactor cavity, has the capability of being isolated by means of an additional removable gate.

As required, the fuel pool gate and equipment pool gate are unsealed and removed by the RB polar crane for fuel handling. Water clarity is established in the reactor cavity pool and fuel pool. The FPC (Subsection 9.1.3) maintains pool level, chemistry, and temperature.

Fuel Handling (Refueling Outage)

The RB is supplied with a refueling platform for fuel movement and servicing. The refueling platform moves between the RPV and the fuel pool via the refueling platform in a safe and efficient manner.

A position indicating system and travel limit computer are provided to locate the grapple over the vessel core and prevent collision with pool obstacles.

The refueling platform fuel grapple is used for lifting and transporting fuel assemblies. It is designed with a telescoping grapple that extends from normal up to its full extension to reach the top guide. The fuel pool water level requirements and component clearance over the travel path establishes the shielded water depth for irradiated fuel transfers.

Storage racks are provided in the fuel pool for the temporary and long-term storage of new and spent fuel assemblies and associated equipment.

Fuel handling encompasses activities associated with the transfer of fuel between the reactor vessel and the fuel pool.

The refueling platform computer provides memory of all spent fuel rack positions and reactor core fuel positions. It also controls where a loaded fuel mast can travel, its loading, and speed for all areas of travel.

As one spent fuel assembly is removed from the core, a new fuel assembly is added to the core, or another fuel assembly is shuffled to its position using the refueling platform.

Figure 9.1-9 depicts fuel handling using the refueling platform.

The BWRX-300 design has integrated shielding for the fuel transfer zone from the core to fuel pool, to adequately shield the upper elevations of containment to protect personnel during the refueling outage.

9.1.4.4 Light Load Handling System Tools and Equipment

The following describes the use of major tools and equipment associated with the Light Load Handling System.

Fuel Prep Machine

Two fuel prep machines are provided for handling fuel assemblies while removing or installing channels on the fuel bundles. Each machine consists of a fuel bundle carriage, which rides on a frame that is mounted on the edge of the fuel pool and extends down to depth along the pool wall,

and a work platform. The work platform includes handrails and extends out over the pool to facilitate viewing of the channeling operations. The fuel inspection rotator fixture in conjunction with the fuel prep machine permits rotation of the fuel assembly in the carriage.

Channel Handling Tool

The channel handling tool is used in conjunction with the fuel preparation machine to remove, install, and handle fuel channels in the fuel pool. The bail hangs from a load balancer on the channel handling boom mounted adjacent to the fuel prep machines.

Channel Transfer Grapple

The channel transfer grapple is an air-actuated device consisting of a frame, air cylinder, and two jaws. It is used with the refueling platform auxiliary hoist to transport individual irradiated fuel channels between working and storage facilities in the fuel pool.

Channel Handling Boom

The channel handling boom is located in a socket adjacent to the fuel preparation machines adjacent to the fuel pool. The boom supports the channel handling tool and load balancer over the fuel preparation machine during channel removal and installation operations.

Refueling Platform

The refueling platform consists of a bridge which spans the width of the fuel pool and reactor cavity. A trolley rides on the bridge, traversing the width of the bridge to the maximum width. A full-length sized working platform where refueling, servicing, and inspection personnel perform their work tasks spans the length of the bridge. The refueling platform rides on two rails set into the refuel floor.

The refueling platform is equipped with a traversing trolley, an operator control console, a main hoist with a telescoping mast and fuel grapple, an auxiliary and a monorail hoist. An air compressor on the refueling platform provides compressed air to the pneumatic system for the fuel grapple and reactor service tooling. The auxiliary and monorail hoists support handling the smaller core components and tooling.

The refueling platform is a rigid structure built to ensure accurate and repeatable positioning during the refueling process. The telescoping mast and grapple are suspended from a trolley system and are used to lift and orient fuel bundles for placement in the core or storage racks. The refueling platform includes a control console providing the operator precise control of the motions of bridge, trolley, and hoists. Bridge and mast position readouts are located in a convenient position for the operators to monitor as they position the refueling platform. The control console includes a light indication to monitor hoist functions and refueling interlocks.

9.1.4.5 Safety Evaluation

The Light Load Handling System complies with the following GDC, which have plant-wide applicability, as discussed in Section 3.1:

GDC 5 – Sharing of structures, systems, and components

The following GDC have aspects specific to the FPC, and compliance is addressed as follows:

GDC 2 – Design bases for protection against natural phenomena, as it relates to the design of the Light Load Handling System against natural phenomena

GDC 61 – Fuel storage and handling and radioactivity control, as it relates to handling of radioactive materials

GDC 62 – Prevention of criticality in fuel storage and handling, as it relates to prevention of criticality in the fuel storage and handling system

Additional information on GDC compliance is found in Section 3.1.

The design of the Light Load Handling System complies with GDC 2 as it relates to protection of safety class equipment and spent fuel from the effects of earthquakes. The refueling platform is designed in accordance with RG 1.29, Positions C.1.d bullet 4, and C.2. The refueling platform is designed such that no single probable event or component failure results in the dropping or damaging of a fuel assembly on the fuel hoist. Interlocks on the refueling platform: (1) prevent hoisting a fuel bundle over the vessel, unless an all-control-rod-in permissive is present; (2) limit vertical travel of the fuel grapple to provide shielding over the grappled fuel during transit; and (3) prevent lifting of fuel without grapple hook engagement when the hoist is loaded.

The design of the Light Load Handling System complies with GDC 61 as it relates to handling of radioactive materials. Compliance with GDC 61 is achieved via application of the requirements of ANSI/ANS 57.1, “Design Requirements for Light Water Reactor Fuel Handling Systems,” (Reference 9.1-4) which provides guidance for prevention of unacceptable radioactivity releases and unacceptable radiation exposure due to dropping or mishandling of new or spent fuel.

During all modes of operation, the Cranes, Hoists and Elevators System (Subsection 9.5.9) provides the means to lift and lower equipment and materials, including heavy loads, and moves them horizontally along safe load paths. Administrative controls (e.g., lift procedures) are used to determine the load paths and equipment used for anticipated heavy load lifts to ensure safe load handling of new and spent fuel.

The design of the Light Load Handling System precludes system malfunctions or failures that could cause a release of radioactivity, or excessive personnel radiation exposures. The following attributes contribute to the prevention of release of radioactivity, or excessive personnel radiation exposures:

- The Light Load Handling System is designed such that probability of dropping a fuel assembly is minimized
- The Light Load Handling System is designed with the capability to permit periodic inspection and testing of components
- The Light Load Handling System operators are provided radiological shielding by water being maintained over the active fuel during transit

The design of the Light Load Handling System complies with GDC 62 as it relates to prevention of criticality in the fuel handling, and storage system, by physical systems or processes, and geometrically safe configurations. Compliance with GDC 62 is provided via application of ANSI/ANS 57.1 (Reference 9.1-4).

The design of the Light Load Handling System precludes system malfunctions or failures that could cause a criticality event.

The following attributes contribute to the prevention of criticality events:

- Fuel handling equipment is designed to prevent inadvertent criticality and to maintain shielding and cooling of spent fuel as necessary to meet operating and off-site dose constraints
- The refueling platform system provides a series of interlocks that restrict movement of the refueling platform and control rods in order to prevent an inadvertent criticality during refueling

9.1.4.6 Inspection and Testing Requirements

Inspection

The refueling platform is designed to allow for components to be inspected. Equipment is configured with sufficient internal access, so as to perform required inspections. The hoists, mast and grapple shall be designed to perform adequate inspections.

The fuel handling equipment also permits the inspection of fuel assemblies as follows:

The fuel inspection rotator fixture is used in conjunction with the fuel prep machine to permit remote inspection of fuel elements. The fuel inspection rotator fixture, when installed, permits rotation of the fuel assembly in the carriage, and in conjunction with the vertical movement of the carriage provides complete access for fuel assembly inspection.

The new fuel inspection stand is stored on the refuel floor of the RB. It consists of a single u-shaped platform with guardrails, a column which supports two fuel assemblies in a vertical position, and a personnel lift which raises or lowers a work platform along the fuel assemblies for inspection purposes. The lower base of the fuel assembly sits in the fuel seat at the bottom of the stand. The fuel assemblies are mounted on rotatable bearing surfaces which allow for rotation to desired quadrants for inspection.

Testing

Fuel handling and vessel servicing equipment preoperational tests are performed in accordance with the ITP (Chapter 14).

9.1.4.7 Instrumentation Requirements

An integrated control system for the refueling platform is designed considering ANSI/ANS 57.1 (Reference 9.1-4), such that the fuel move sequence can be performed in the manual, semi-automatic, and automatic modes.

Refueling platform interlocks restrict the operators to ensure safe operation of the equipment.

9.1.5 Overhead Heavy Load Handling System

The Overhead Heavy Load Handling System (OHLHS) is equipment which lifts loads, whose weight is greater than the combined weight of a single fuel assembly and its handling device. The regulatory emphasis of the OHLHS is to prevent inadvertent operations or equipment malfunctions, separately, or in combination that could cause a release of radioactivity, a criticality accident, inability to cool fuel within the reactor vessel, or fuel pool, or could prevent safe shutdown of the reactor.

9.1.5.1 Design Bases

Safety Category 1 Functions

None.

Safety Category 2 Functions

None.

Safety Category 3 Functions

None.

Safety Category N Functions

- Critical load handling where inadvertent operations or equipment malfunctions, separately or in combination, could cause a release of radioactivity, a criticality accident, inability to cool fuel within the reactor vessel, or spent fuel pool, or could prevent safe shutdown of the reactor

The Safety Classifications, Quality Groups, and Seismic Categories of the OHLHS SSCs are listed in Table 3A-1.

9.1.5.2 Codes and Standards

The design of cranes and lifting devices incorporate large margins, appropriate interlocks, and other safety features to accommodate the lifting of large loads. BWRX-300 follows the guidance of Standard Review Plan (SRP) 9.1.5 for overhead heavy load handling systems compliance with GDC 1, 2, 4 and 5.

Lifting and handling operations using overhead handling systems are performed consistent with ASME Standard NML-1-2019, “Rules for the Movement of Loads Using Overhead Handling Equipment in Nuclear Facilities,” (Reference 9.1-5) with clarifications as presented in RG 1.244, “Control of Heavy Loads at Nuclear Facilities.” Performance and design criteria applied to specific components of the heavy load handling system conform to the consensus standards identified in RG 1.244 and provide for reliable fuel handling.

As endorsed by RG 1.244, the guidelines of ASME Standard NML-1-2019 use a risk-informed approach by qualitatively considering the probability and consequences of a load handling event during a planned lift to accommodate variations in significance, and the associated controls applied to manage the risk of the lift. The probability of a load handling event is established by evaluating the handling system design, rigging configuration, and load characteristics. The potential consequences of a load handling event are classified considering the properties of the lifted load, and the SSCs in the vicinity of the load path. The classifications are “standard lift,” “special lift,” and “critical lift,” with “nuclear safety critical lift” a subset of critical lift. Nuclear safety critical lifts encompass lifts that could challenge the continued performance of fundamental safety functions.

The guidelines of ASME Standard NML-1 (Reference 9.1-5) are applied to address the program for conducting normal handling system operations, including maintenance, surveillance, and periodic testing of handling system SSCs. The heavy load lift program for normal operations includes development of lift procedures, crane operator and rigger qualifications, crane inspection, testing, and maintenance, and selection and use of lifting devices not specially designed. The guidelines of ASME Standard NML-1 are also applied to establish criteria for protecting safety functions during nuclear safety critical lifts. These criteria include control of load motion by design and interlocks, enhanced safety handling systems, and engineering controls that include analysis of a postulated load drop. The enhanced safety handling system includes a crane designed as single-failure-proof, such as an overhead crane designed to meet the Type I criteria of ASME Standard NOG-1, “Rules for Construction of Overhead and Gantry Cranes,” (Reference 9.1-6) as endorsed by RG 1.244, and lifting devices, which may include special lifting devices designed to ASME Standard BTH-1, “Design of Below-the-Hook Lifting Devices,” (Reference 9.1-7) for recurrent lifts as endorsed by RG 1.244.

The guidance of ASME Standard NML-1 is also utilized to establish controlled ranges of motion to ensure that safety functions are accomplished with appropriate consideration of the effects of natural phenomena, or credible equipment failures.

9.1.5.3 System Description

The design of the OHLHS equipment, in conjunction with procedures and safe load paths, ensures the safe movement of loads. Safe load paths are defined for the movement of heavy loads to minimize the potential for a load drop on irradiated fuel in a reactor vessel or fuel pool, and safe shutdown equipment.

A heavy load handling program is developed consistent with ASME Standard NML-1 (Reference 9.1-5). Implementation of the heavy load handling program ensures that safe heavy load paths are referenced in procedures and on equipment layout drawings. This minimizes the potential to affect irradiated fuel in the reactor vessel, and in the fuel pool, as well as safe shutdown equipment from movement of heavy loads.

The RB polar crane is designed according to ASME Standard NOG-1 (Reference 9.1-6) for a Type I single-failure proof crane. Redundancy and other features ensure that the failure of a single component in the load path does not result in the loss of capability to stop and hold the critical load.

The RB polar crane is composed of an overhead bridge of two deep girders supporting a trolley with a main and auxiliary hook. This rail supports the two girders allowing the crane bridge 360 degrees of rotation. The bridge structure spans the full width of the refuel floor. The crane is classified as Seismic Category II. Refer to Section 3.2 for additional information pertaining to designation of seismic categories. Design for both the main and auxiliary hoists is in accordance with ASME NOG-1, which provides a high degree of reliability and safety. Redundancy and other features ensure that the failure of a single component in the load path does not result in the loss of capability to stop and hold a critical load. The principal heavy loads handled by this crane consist of the components associated with reactor vessel refueling, and spent fuel transfer, including the reactor head, reactor vessel internals including dryer/separators, fuel assemblies (associated with new fuel receipt), and various containment support components. During spent fuel handling activities, the maximum load of the crane is the combination of the spent fuel cask and canister, and lifting yoke, which weighs less than the cranes overall capacity. The crane has access to a truck bay for fuel transfer to spent fuel shipping and/or transfer casks, new fuel assemblies, and replacement components.

Additional OHLHS equipment and lifting devices that do not handle fuel are addressed in Subsection 9.5.9.

9.1.5.4 Safety Evaluation

The OHLHS complies with the following GDC, which have plant-wide applicability, as discussed in Section 3.1:

GDC 1 – Quality standards and records

GDC 5 – Sharing of structures, systems, and components

The following GDC have aspects specific to the OHLHS, and compliance is addressed as follows:

GDC 2 – Design bases for protection against natural phenomena, as it relates to the design of the OHLHS against natural phenomena

GDC 4 – Environmental and dynamic effects design bases, as it relates environmental and dynamic effects on the design of OHLHS

Additional information on GDC compliance is found in Section 3.1.

The design of the OHLHS is consistent with GDC 2 for protection against natural phenomena. The RB polar crane is designed to ensure that the system retains its load throughout an SSE.

The RB polar crane is designed as a Seismic Category II structure in accordance with RG 1.29, Position C.2.

The design of the OHLHS includes features to minimize the potential for a load drop and for the safe handling of heavy loads. The design includes single-failure-proof cranes, mechanical stops, electrical interlocks, well-defined safe load paths, established load handling procedures, and a plant configuration that provides redundancy and duality in certain components, so the probability of load drops is minimized. The RB polar crane is designed to support a “standard lift,” “special lift,” and “critical lift,” during and after an SSE, as required by the specific lifting device. The RB polar crane design conforms to the single-failure-proof guidelines; therefore, the failure of a single component does not result in the loss of capability to stop and hold a load. The use of this single-failure-proof crane precludes the need to perform load drop evaluations, and as a result, accident analysis is not required to assess radiological consequences of a spent fuel cask drop accident.

The design of the OHLHS is consistent with GDC 4. The design of the polar crane provides protection against the effects of internally generated missiles (i.e., dropped loads). The guidance of position C.5 of RG 1.13, “Spent Fuel Storage Facility Design Basis,” is utilized to meet this protection in the fuel pool storage areas. Dynamic effects associated with missile impact are provided in Section 3.5.

In addition to being designed to meet single-failure-proof criteria, the polar crane is designed with a system of interlocks that prevent movement outside safe load paths, which is intended to prevent deleterious impacts with SSCs. Physical limits and administrative controls are included to ensure safe handling of loads. The design of the Overhead Heavy Loads Handling System in conjunction with the designation of safe load paths, which represent travel paths associated with the movement of heavy loads, and areas where heavy loads are not allowed to transit, ensures that the movement of heavy loads in the plant can be performed without affecting plant operations, including safe shutdown and cooldown.

9.1.5.5 Inspection and Testing

Quality assurance requirements for the RB polar crane are defined in ASME NOG-1 Section 2000. The supplier shall have a quality assurance program that complies with ASME NQA-1, “Quality Assurance Requirements, for Nuclear Facility Applications,” (Reference 9.1-8), or otherwise complies with the requirements of 10 CFR Part 50 Appendix B. As a Type I crane, the RB polar crane complies with both paragraph 6170 and Table 7210-1 of ASME NOG-1 which specify inspection, testing, and enhanced quality assurance requirements for various electrical, mechanical, and structural subcomponents.

Surveillances of a load handling system are performed. Inspections and tests of the load handling system are performed in accordance with Operations and Maintenance Manual guidance, as well as load handling procedures.

Maintenance activities for OHLHS SSCs, which are either preventive or the result of adverse conditions determined through an inspection program, are corrected by adjustment, repair, or replacement of components before continuing the use of the SSC. Operational testing is performed in accordance with plant procedures.

OHLHS SSCs which are altered, repaired, and modified, are required to be operationally tested for the functions affected by the alteration, repair, or modification. In addition, load testing of altered, repaired, and modified load handling systems, may be limited to the functions affected by the alteration, repair, or modification.

9.1.5.6 Instrumentation and Control Requirements

The RB polar crane control circuits are arranged, so that tripping of an overload relay or limit switch defeats the permissive to crane controllers. Overload relay or limit switch operation does not interrupt lighting circuits.

The prescribed path for RB polar crane hook travel is enforced by limit switches for hook travel in both vertical and horizontal directions.

The RB polar crane is controlled by a local master control panel on the crane bridge. Primary control of crane motions is directed from a cab located on the crane bridge, and remote control with backup pendant control. The cab, and remote and pendant controls provide for bridge, trolley, main hoist, hook rotation (cab and radio control only), and auxiliary hoist motions. Pendant controls are back up for cab/remote and will be isolated during normal operation by a transfer switch on the pendant.

Redundancy and other features ensure that the failure of a single component in the load path does not result in the loss of capability to stop and hold the critical load.

The RB polar crane control system uses a programmable logic controller, housed in a bridge control panel. The programmable logic controller is integral to a restricted zone scheme for the crane, which is activated during critical lift evolutions such as movement of the spent fuel cask.

9.1.6 References

9.1-1 ASME BPVC Section III, “Rules for Construction of Nuclear Facility Components,” American Society of Mechanical Engineers, 2021.

9.1-2 ANSI/ANS 57.2, “Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants,” American National Standards Institute/American Nuclear Society, 1983.

9.1-3 NEDC-33926P-A, “BWRX-300 Steel-Plate Composite (SC) Containment Vessel (SCCV) and Reactor Building Structural Design,” GE Vernova Hitachi Nuclear Energy, Revision 4, December 2025.

9.1-4 ANSI/ANS-57.1, “Design Requirements for Light Water Reactor Fuel Handling Systems,” American National Standards Institute/American Nuclear Society, R2024.

9.1-5 ASME NML-1-2019, “Rules for the Movement of Loads Using Overhead Handling Equipment in Nuclear Facilities,” American Society of Mechanical Engineers, 2019.

9.1-6 ASME NOG-1-2020, “Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder),” American Society of Mechanical Engineers, 2020.

9.1-7 ASME BTH-1-2017, “Design of Below-The-Hook Lifting Devices,” American Society of Mechanical Engineers, 2017.

9.1-8 ASME NQA-1-2015, “Quality Assurance Requirements, for Nuclear Facility Applications,” American Society of Mechanical Engineers, 2015.

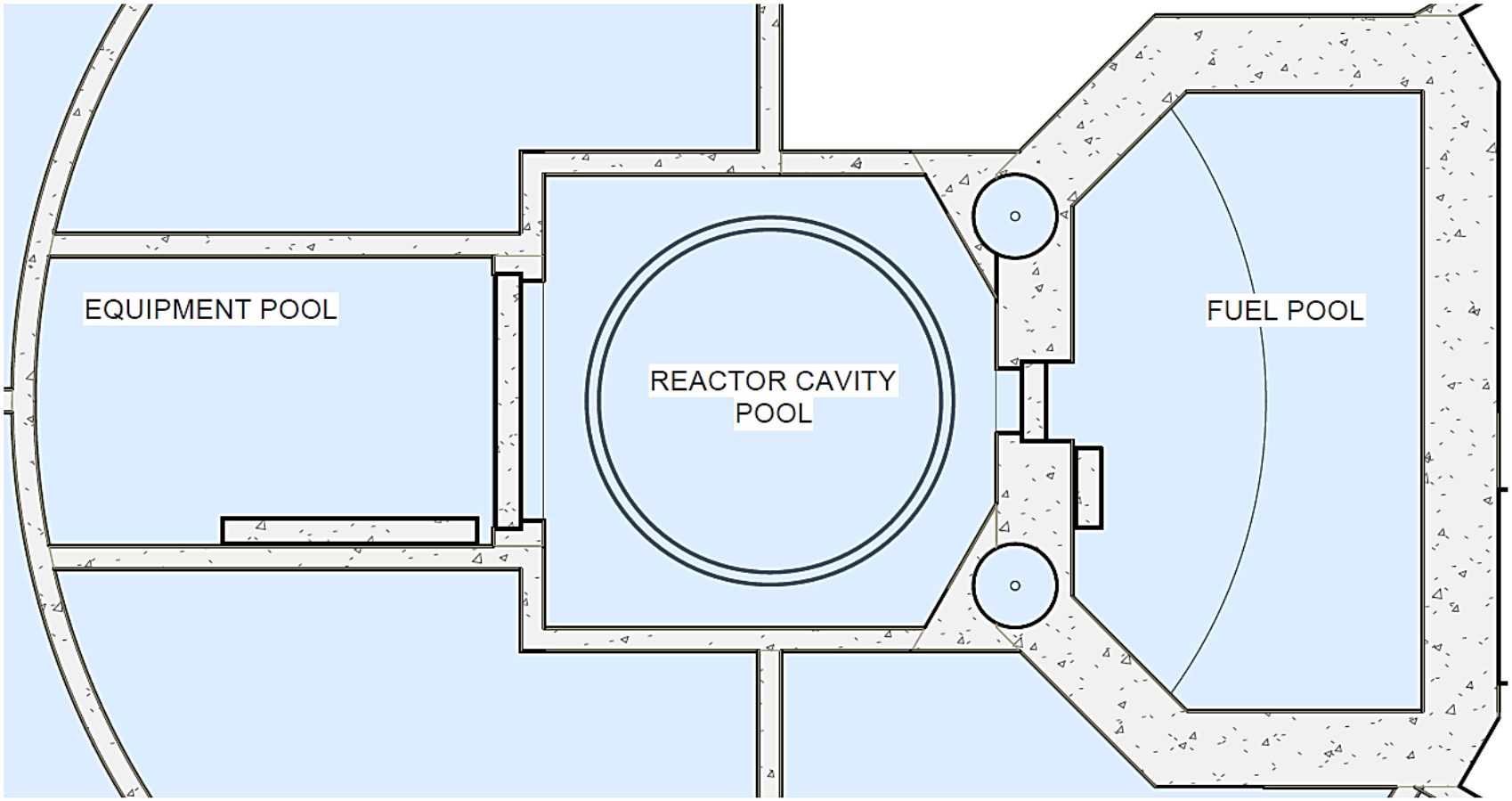


Figure 9.1-1: Arrangement of Fuel, Reactor Cavity and Equipment Pools

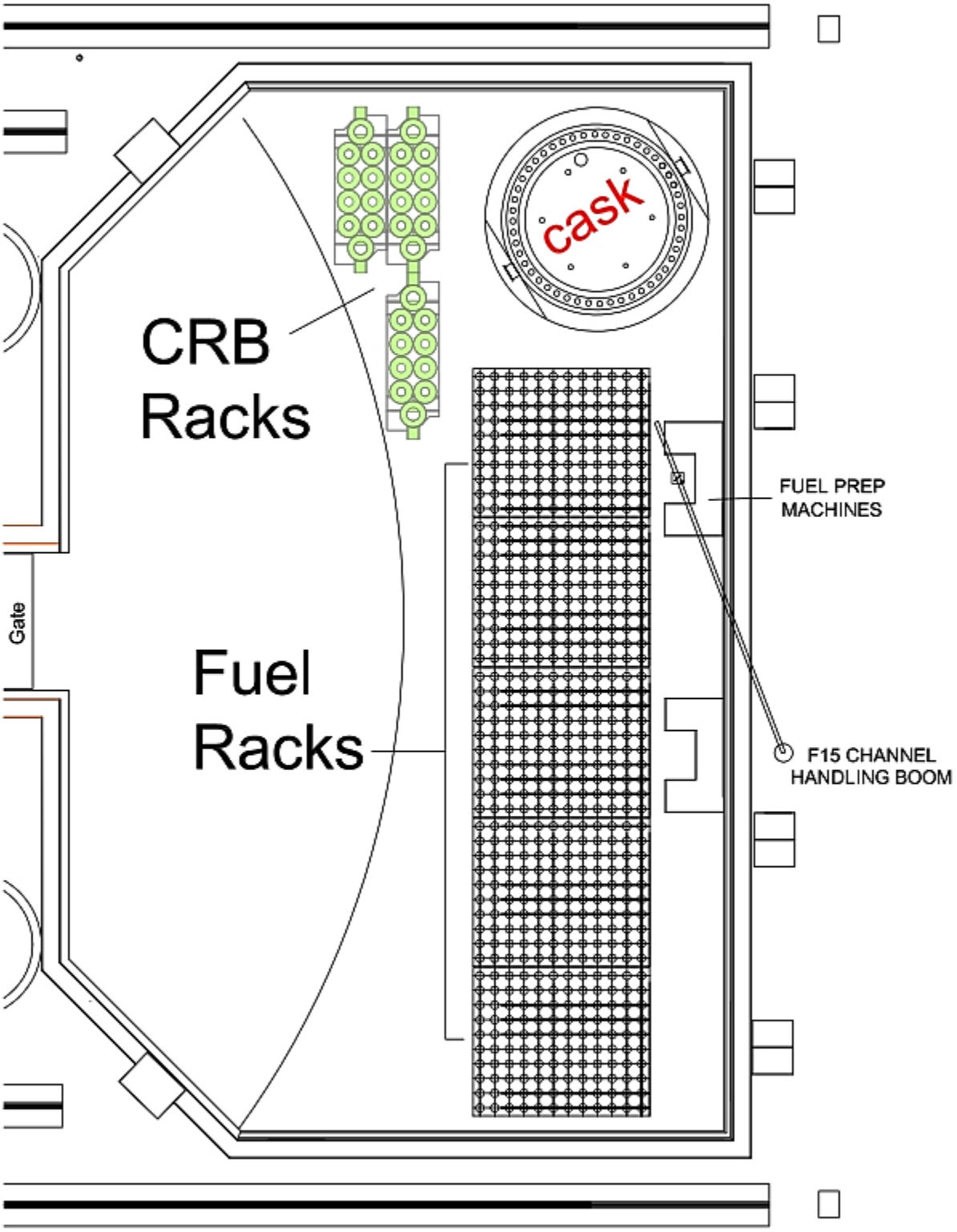


Figure 9.1-2: Fuel Pool Arrangement

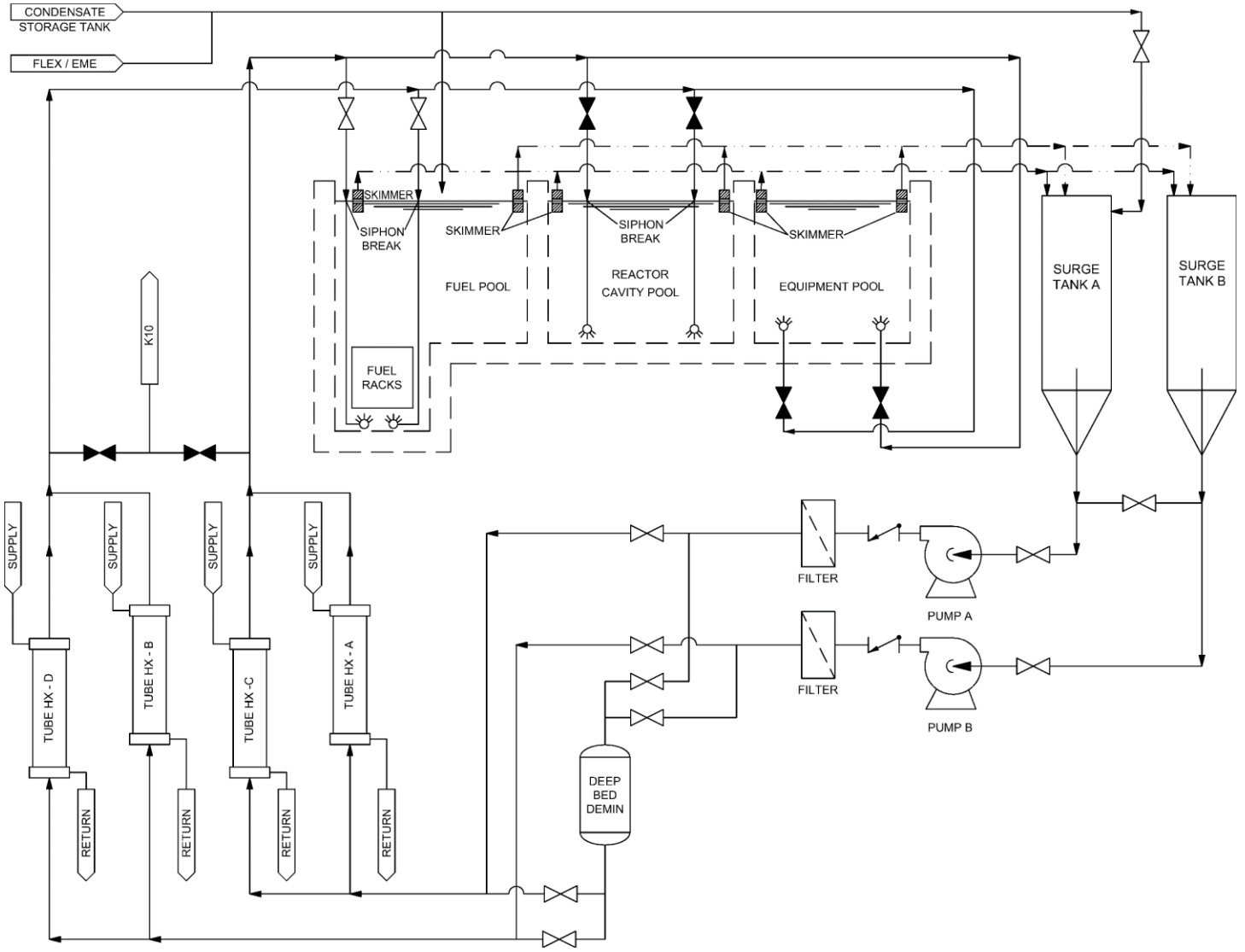


Figure 9.1-3: Fuel Pool Cooling System

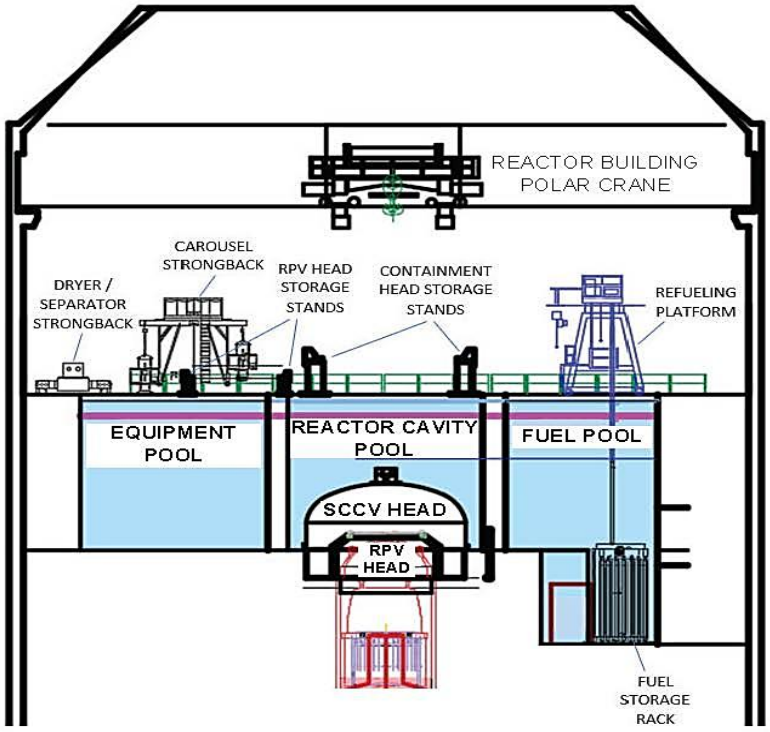


Figure 9.1-4: Location of Equipment on Refuel Floor

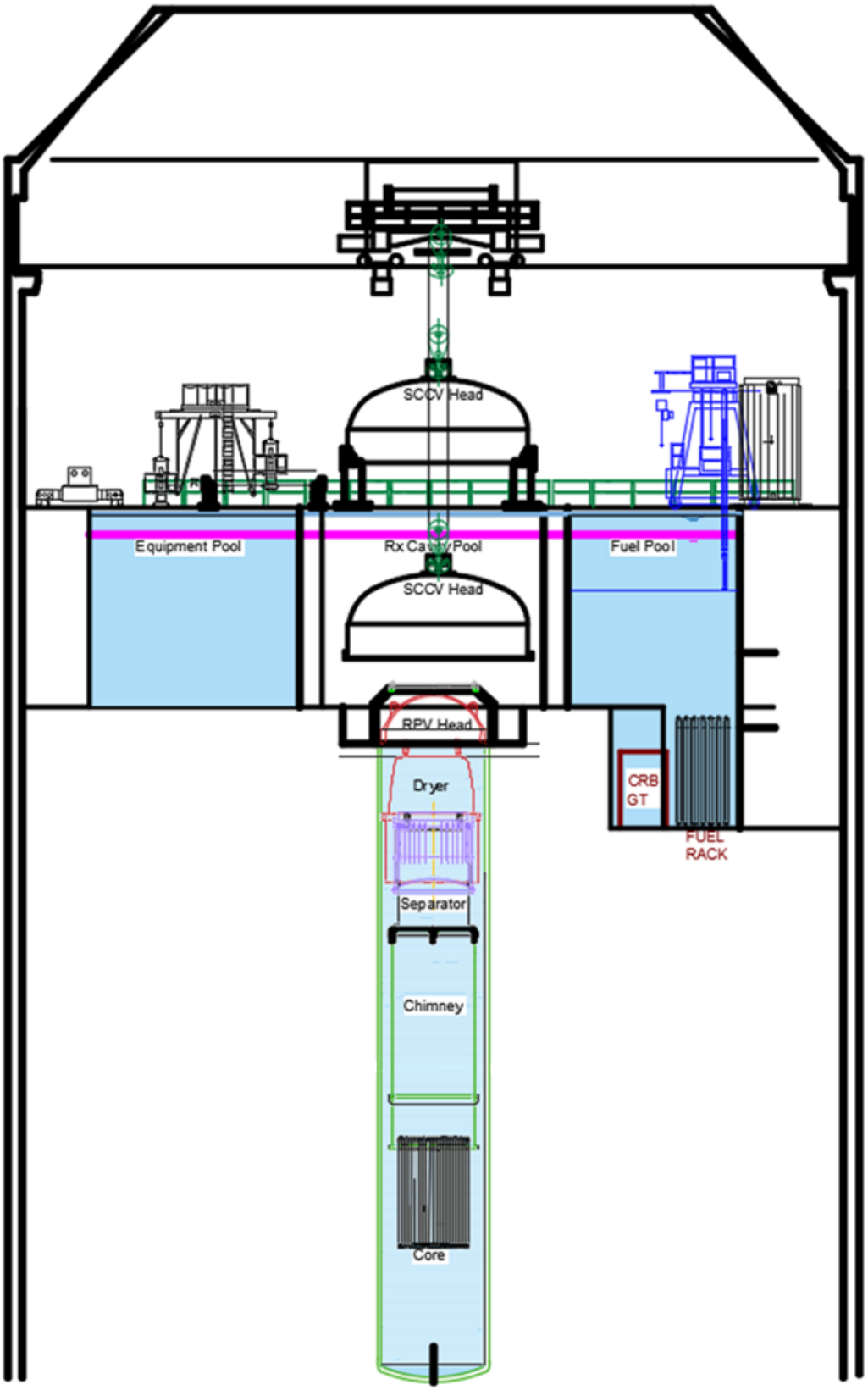


Figure 9.1-5: Containment Closure Head Removal

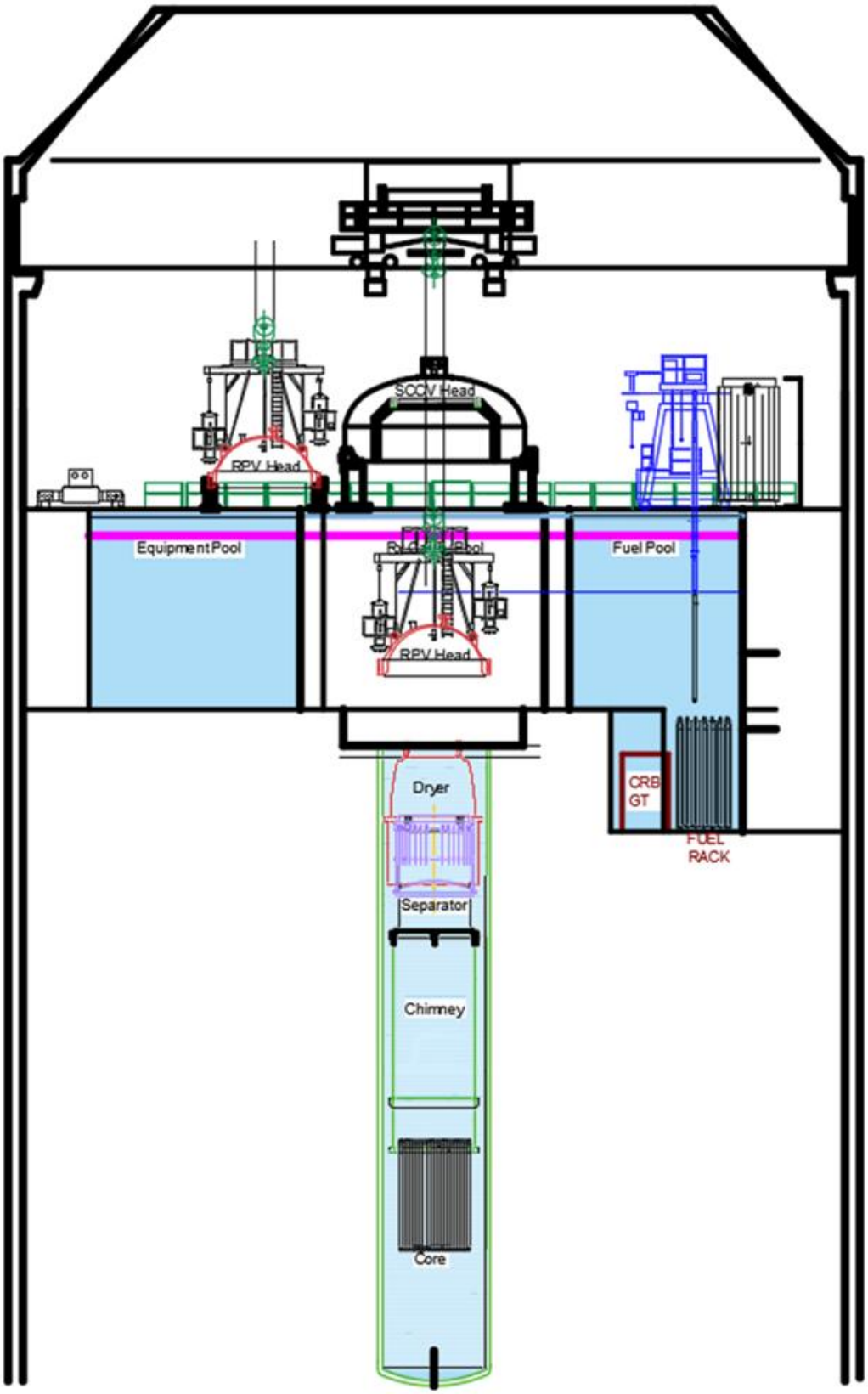


Figure 9.1-6: Reactor Pressure Vessel Head Removal

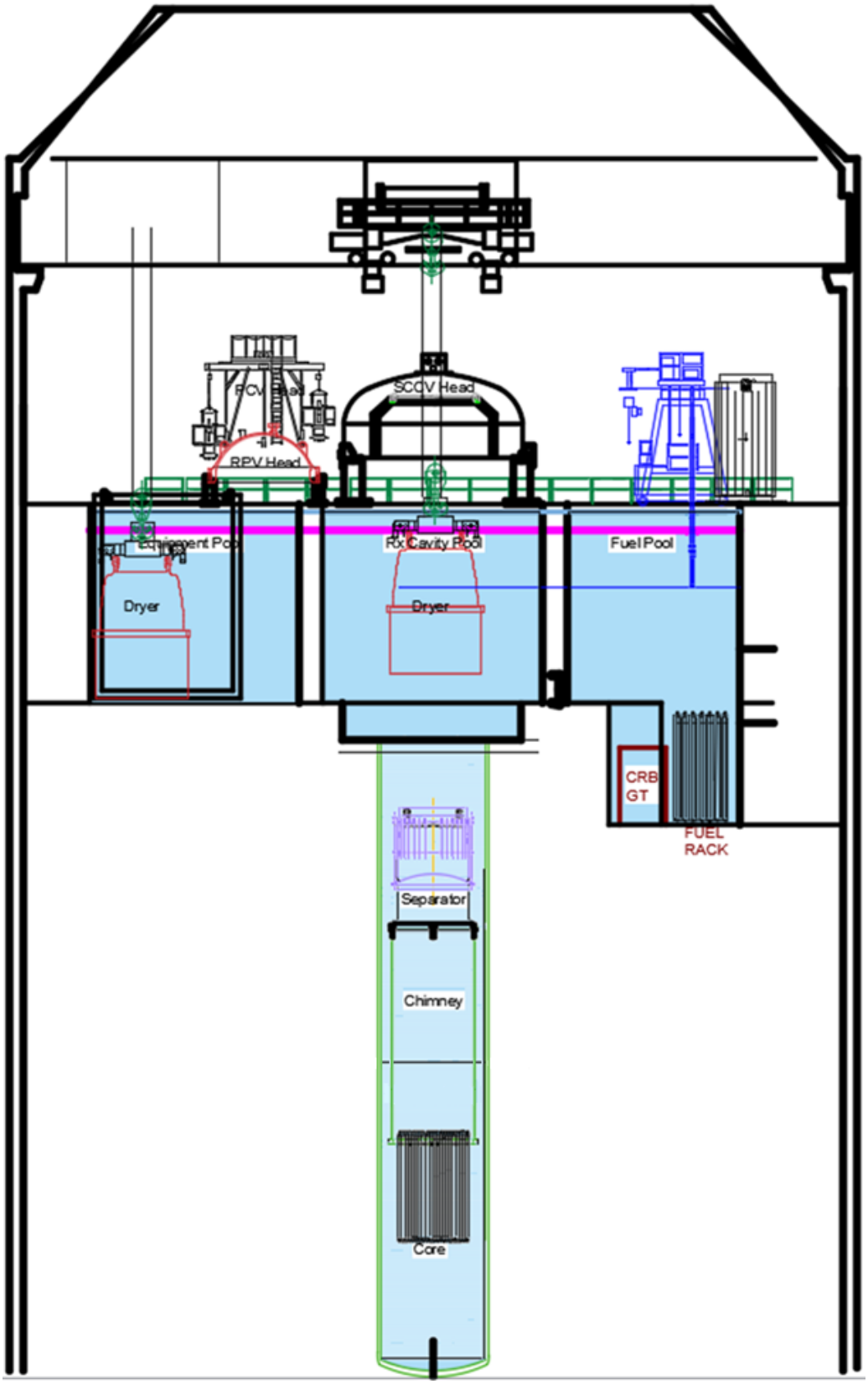


Figure 9.1-7: Steam Dryer Assembly Removal

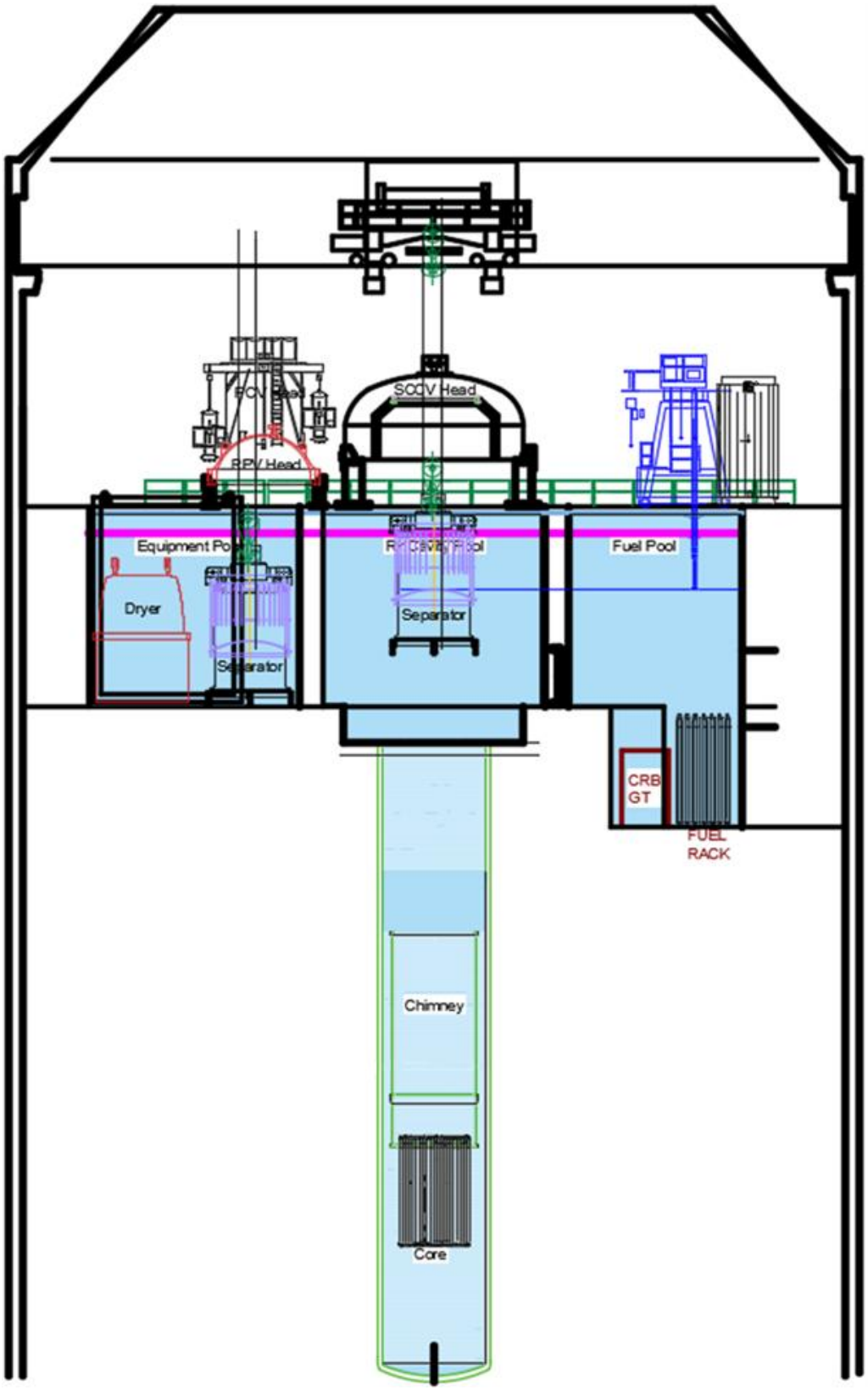


Figure 9.1-8: Chimney Head and Steam Separator Assembly Removal

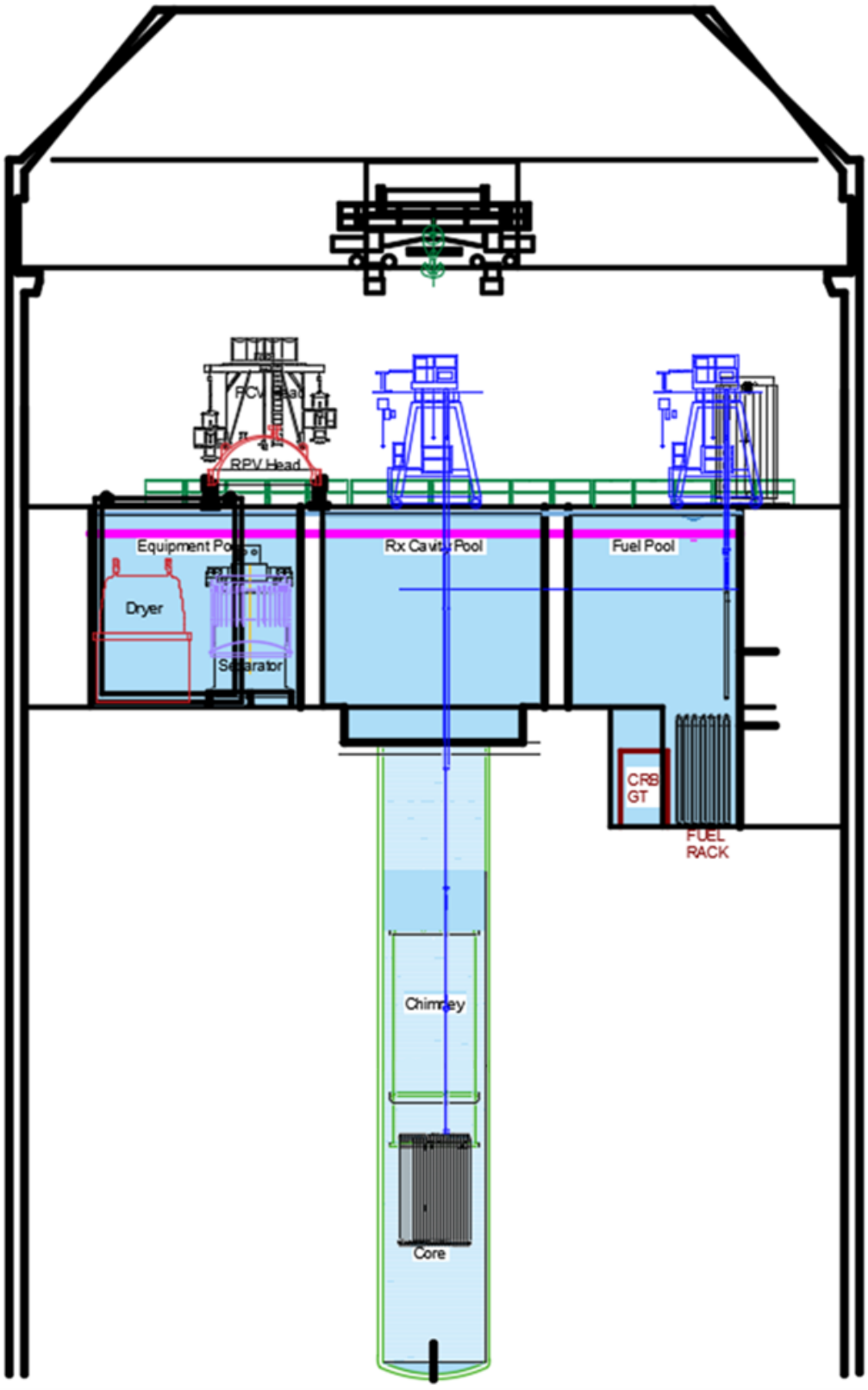


Figure 9.1-9: Fuel Handling

9.1.A Criticality Safety of New and Spent Fuel Storage

9.1.A.1 Design Bases

This section specifically demonstrates fuel storage facilities, including provisions necessary to maintain a subcritical array, are in conformance with the regulations in GDC 62 and in 10 CFR 50.68. The requirements of GDC 62 pertaining to criticality are met because the fuel remains subcritical under normal and credible abnormal conditions, by use of geometrically safe configurations. In addition, in lieu of a monitoring system capable of detecting a criticality, the fuel storage facilities comply with the additional design and analysis requirements specified in 10 CFR 50.68(b).

Regarding criticality safety of new and spent fuel storage, the following guidance documents are considered and applied as applicable in the design of the storage and handling for new and spent fuel assemblies:

- ANSI/ANS-57.1, American National Standards Institute, “Design Requirements for Light Water Reactor Fuel Handling Systems,” (Reference 9.1.A-1)
- ANSI/ANS-57.2, American National Standards Institute, “Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants,” (Reference 9.1.A-2)
- RG 1.13, “Spent Fuel Storage Facility Design Basis”
- Nuclear Energy Institute (NEI) 12-16, “Guidance for Performing Criticality Analyses of Fuel Storage at Light-Water Reactor Power Plants,” (Reference 9.1.A-3)
- RG 1.240, “Fresh and Spent Fuel Pool Criticality Analyses”

Subsection 9.1.3 describes the protection of the fuel storage racks from natural phenomena. Non-seismic SSCs in the vicinity of the spent fuel storage racks are evaluated to confirm that their failure could not result in an increase in the effective multiplication factor (k_{eff}) beyond the maximum allowable k_{eff}.

9.1.A.2 Facilities Description

Refer to Subsection 9.1.2.2 for a description of the fuel storage rack design. As discussed in Subsection 9.1.1, the BWRX-300 design does not include separate new fuel storage.

Subsection 9.1.3 describes the FPC which manages water in the fuel pool, Subsection 9.1.4 describes the fuel handling equipment, and Subsection 9.1.5 describes heavy load handling.

9.1.A.3 Safety Evaluation

In accordance with GDC 62, the fuel handling equipment and fuel storage racks, designed in accordance with RG 1.13, use geometrically safe configurations to prevent criticality. Fuel handling procedures place controls on the movement of each fuel assembly and the designated storage location in a rack. Consistent with ANSI/ANS-57.1 (Reference 9.1.A-1) and ANSI/ANS-57.2 (Reference 9.1.A-2) the design of the fuel storage racks does not require loading patterns or zones for storage of the fuel assemblies. The design of the fuel storage racks eliminates the possibility of an inadvertent criticality occurring, due to the selection of an inappropriate storage location or region.

Plant programs and procedures track fuel assembly storage locations in accordance with special nuclear material regulations. In addition, the design of the Light Load Handling System precludes system malfunctions or failures that could cause a criticality event.

The fuel handling system (i.e., Light Load Handling System) conforms to the applicable guidance of ANSI/ANS-57.1 (Reference 9.1.A-1) to handle fuel units and control components in a safe and reliable manner. As described in Subsection 9.1.4, safety devices such as interlocks on the fuel handling equipment assist operators to prevent damage to an assembly and help minimize mishandling and movements not allowed by plant approved procedures.

The fuel storage racks comply with the applicable guidance, including scope of the nuclear criticality safety analysis of ANSI/ANS-57.2 (Reference 9.1.A-2). The design of the geometrically safe fuel storage racks maintains stored fuel assemblies in subcritical arrays during normal and credible abnormal conditions. The BWRX-300 design and criticality analysis of the fuel storage racks is consistent with the racks modeled and described in Section 3.5 of NEDE-24011-P-A-31, "General Electric Standard Application for Reactor Fuel (GESTAR II)," (Reference 9.1.A-4). Applying the GESTAR II criteria provides reactor limits consistent with fuel provided by Global Nuclear Fuel (GNF) and therefore is bounded by this analysis. Upon selection of the fuel rack vendor, the fuel rack storage design is evaluated to verify that the parameters used remain bounding. Relevant information includes materials, dimensional data, spacing between storage racks, material/dimensional details of structures near the racks, which may provide neutron reflection (such as pool walls and floors). The fuel rack is made of stainless steel and borated stainless steel based on Type 304 stainless steel. The final rack configuration and analysis is completed and approved prior to arrival of fuel on site.

A monitoring system capable of detecting a criticality is not required, because the fuel storage racks are constructed of neutron absorbing material (i.e., no credit is taken for soluble boron), meeting 10 CFR 50.68(b)(4) requirements, and have a k_{eff} no greater than 0.95, at a 95 percent probability and a 95 percent confidence level, for storage of new and spent fuel assemblies.

9.1.A.3.1 Analysis Code and Validation

The BWRX-300 fuel racks are analyzed using the Monte Carlo Neutron Transport Program and ENDF/B-VII.0 cross-section library. The methodology used in this analysis is the peak standard cold core geometry incore eigenvalue ($k\infty$) criterion. Monte Carlo Neutron Transport Program has been compared to critical experiments for validation purposes using ENDF/B-VII.0 nuclear cross-section data. The experiments cover a number of moderator-to-fuel ratios and poison materials that represent material and geometric properties similar to that of Boiling Water Reactor (BWR) fuel lattices, both in and out of fuel racks. All benchmarks are either low-enriched uranium dioxide or Mixed Uranium-Plutonium Oxide (MOX) pin lattice in water experiments. Using the methodology in NUREG/CR-6698, "Guide for Validation of Nuclear Criticality Safety Calculational Methodology," the associated bias and bias uncertainty terms at the 95/95 confidence level were determined for Monte Carlo Neutron Transport Program.

9.1.A.3.2 Analysis Conditions

The criticality analysis incorporates the design bases for the fuel storage racks to ensure the true system reactivity is always less than the calculated reactivity.

The analysis considers the following fuel storage configurations to determine the maximum reactivity effect:

- Normal Configuration Models:
  - Storage of non-channeled assemblies
  - Eccentric loadings
  - Pool moderator temperature variation

- Abnormal Configuration Models
- Accident Scenarios

The treatment of modeling assumptions and conservatisms, and the modeling of design and manufacturing tolerances for abnormal conditions are the same as for normal conditions. This similar treatment is acceptable, consistent with the guidelines of SRP Section 9.1.1.

9.1.A.3.3 Criticality Analysis Results

In cases described in Subsection 9.1.A.3, the maximum k_{eff} is to remain below the applicable limits of 10 CFR 50.68(b) and no normal or credible abnormal condition is to cause an inadvertent criticality. Specifically, the analysis is to result in a storage rack maximum k_{eff} less than 0.95 at a 95% probability, 95% confidence level, for normal and credible abnormal operation with tolerances and computational uncertainties considered.

9.1.A.4 References

9.1.A-1 ANSI/ANS-57.1, “Design Requirements for Light Water Reactor Fuel Handling Systems,” American Nuclear Society, R2019.

9.1.A-2 ANSI/ANS 57.2, “Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants,” American National Standards Institute/American Nuclear Society, 1983.

9.1.A-3 NEI 12-16, “Guidance for Performing Criticality Analyses of Fuel Storage at Light-Water Reactor Power Plants,” Nuclear Energy Institute, Rev. 4, September 2019.

9.1.A-4 NEDE-24011-P-A-31, “General Electric Standard Application for Reactor Fuel (GESTAR II),” Global Nuclear Fuel – Americas, LLC, Revision 31, November 2020.

Blue Energy Global Inc.

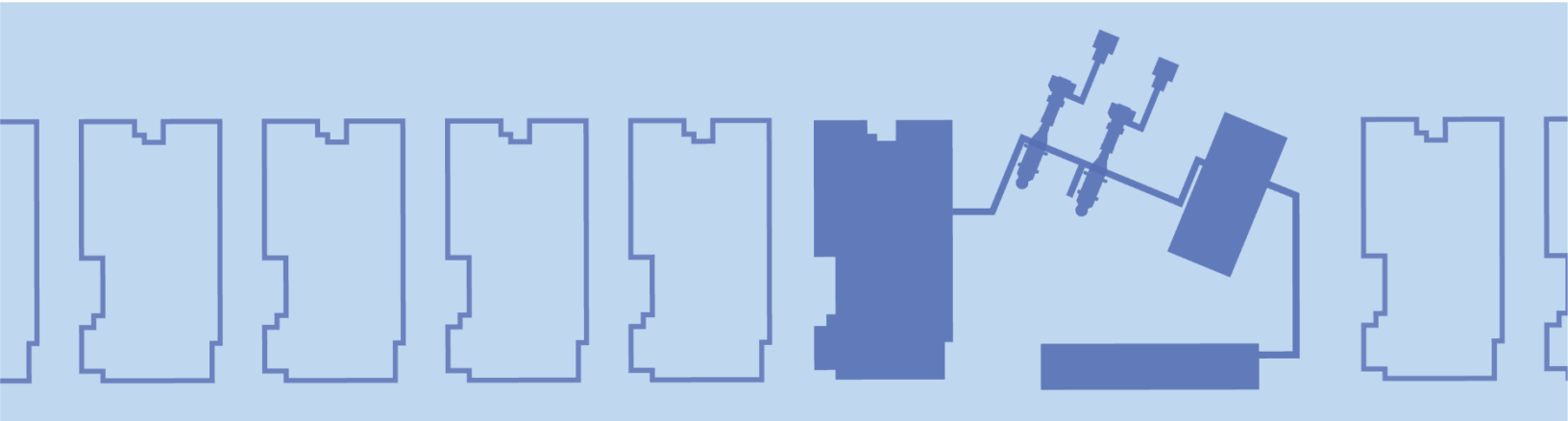
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Chapter 9
Section 9.2 – Water Systems



ACRONYM LIST

Acronym	Explanation
AC	Alternating Current
AOO	Anticipated Operational Occurrence
BOP	Balance of Plant
BTP	Branch Technical Position
D-in-D	Defense-in-Depth
DBA	Design Basis Accident
DC	Direct Current
DCIS	Distributed Control and Information System
ECCS	Emergency Core Cooling System
EDS	Electrical Distribution System
EMC	Electromagnetic Compatibility
EMI	Electromagnetic Interference
FLEX	Diverse and Flexible Coping Strategy
FMCRD	Fine Motion Control Rod Drive
GDC	General Design Criteria
GSU	Generator Step Up Transformer
HV	High Voltage
I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency
IC	Isolation Condenser
ICS	Isolation Condenser System
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronics Engineers
LG	Load Group
LOCA	Loss-of-Coolant Accident
LOOP	Loss-of-Offsite Power
LOPP	Loss-of-Preferred Power
LV	Low Voltage
MCC	Motor Control Center
MCR	Main Control Room
MTS	Manual Transfer Switch

Acronym	Explanation
MV	Medium Voltage
NERC	North American Electric Reliability Corporation
NFPA	National Fire Protection Association
NUREG	Nuclear Regulatory Report
PAM	Post-Accident Monitoring
PCCS	Passive Containment Cooling System
PDC	Principal Design Criterion
PWR	Pressurized Water Reactor
RAM	Reliability, Availability, and Maintainability
RAT	Reserve Auxiliary Transformer
RB	Reactor Building
RFI	Radio Frequency Interference
RG	Regulatory Guide
RVT	Regulating Voltage Transformer
SBO	Station Blackout
SC1	Safety Class 1
SC2	Safety Class 2
SC3	Safety Class 3
SCN	Non-Safety Class
SCR	Secondary Control Room
SDG	Standby Diesel Generator
SSCs	Structures, Systems, and Components
STS	Static Transfer Switch
TB	Turbine Building
TMI	Three Mile Island
TVSS	Transient Voltage Surge Suppressor
UAT	Unit Auxiliary Transformer
UPS	Uninterruptable Power Supply
USNRC	U.S. Nuclear Regulatory Commission
VAC	Volts Alternating Current
VDC	Volts Direct Current

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9.2 Water Systems

9.2.1 Service Water Subsystem

The Service Water (SW) Subsystem, which is a subsystem of the Circulating Water System (CWS) (Subsection 10.4.5), provides cooling water to the Plant Cooling Water System (PCW) (Subsection 9.2.2) heat exchangers. The heat from the SW Subsystem is transferred to the Normal Heat Sink (NHS) System (Subsection 9.2.9).

9.2.1.1 Design Bases

9.2.1.1.1 Safety Category Functions

Safety Category 1 Functions

None.

Safety Category 2 Functions

None.

Safety Category 3 Functions

- The SW Subsystem rejects heat from the PCW heat exchangers to the environment through the NHS System during normal conditions for Safety Class 3 (SC3) equipment.

Safety Category N Functions

- The SW Subsystem provides heat rejection for Non-Safety Class (SCN) loads serviced by PCW during normal operating and shutdown conditions.

The Safety Classification, Seismic Categories, and Quality Groups are presented in Table 3A-1.

9.2.1.2 System Description

The SW Subsystem has two 100 percent capacity pumps to provide cooling water to the PCW heat exchangers for normal and abnormal operating modes. Each SW line has a strainer at the pump discharge upstream of the PCW heat exchangers. Screens are included between the cooling tower basin and the SW pump suction lines. During normal operation, only one PCW heat exchanger is in service, and the corresponding SW pump is used to provide cooling water flow. During a plant shutdown, both PCW heat exchangers and SW pumps are typically in service to facilitate a plant shutdown. In the event of a LOPP, the SW pumps can be powered by the SDGs to provide continuous cooling support to the PCW. The SW pumps provide sufficient cooling water to meet the performance requirements of the PCW heat exchangers.

Two individual SW pump discharge lines each contain pressure instrumentation, air release valves, and an isolation valve. Each line supplies SW independently to its related PCW heat exchanger. SW piping is designed to American Society of Mechanical Engineers (ASME), B31.1, “Code for Pressure Piping, Power Piping,” (Reference 9.2-1). Below ground piping also meets applicable American Water Works Association (AWWA) standards based on pipe material and size.

A radiation monitor is installed on SW Subsystem piping downstream of the PCW heat exchanger to detect leakage from the PCW to the SW Subsystem.

A fill line connecting the SW Subsystem pumps to the CWS is provided to fill the main condenser and associated piping prior to startup. This is done to mitigate water hammer effects that could be caused by starting the Circulating Water (CW) Supply (Subsection 10.4.5) pumps.

A simplified diagram of the SW Subsystem is provided in Figure 9.2-1.

9.2.1.3 Safety Evaluation

The SW Subsystem is not credited for mitigation of a Design Basis Accident (DBA) and has no safe shutdown functions.

Regulatory Requirements

The SW Subsystem complies with the following General Design Criteria (GDC), which has plant-wide applicability, as discussed in Section 3.1:

GDC 5 – Sharing of structures, systems, and components

The following GDC have aspects specific to the SW Subsystem and compliance is addressed as follows:

GDC 2 – Design bases for protection against natural phenomena. This GDC is considered in the design of the SW Subsystem. No Safety Class 1 (SC1) SSCs are affected by this subsystem from the effects of natural phenomena such as earthquakes. The design and layout of the SW Subsystem include provisions to ensure that a failure of the subsystem will not adversely affect the functional performance of SC1 systems or components.

GDC 4 – Environmental and dynamic effects design bases. None of the equipment in the SW Subsystem, or cooled by the subsystem, is required to mitigate DBAs and, therefore, the SW Subsystem is not required to be qualified for adverse environmental conditions or be protected from potential dynamic effects of equipment failures.

GDC 44 – Cooling water. The SW Subsystem transfers heat from the PCW to the NHS System under normal operating conditions. The system has sufficient redundancy to perform this function assuming the failure of a single train or during a LOPP if the SDGs are available. However, the SW Subsystem is not required to provide cooling to any SC1 components and is not credited as the ultimate heat sink in the Chapter 15 safety analyses. The ultimate heat sink is described in Subsection 9.2.5.

GDC 45 – Inspection of cooling water system. The SW Subsystem is designed to permit appropriate periodic inspection of pumps and piping, to assure the integrity and capability of the system. However, the SW Subsystem is not required to provide cooling to any SC1 components, so no specific inspection requirements are implemented for this GDC.

GDC 46 – Testing of cooling water system. The SW Subsystem is designed to permit appropriate periodic pressure and functional testing to assure the structural and leak-tight integrity of its components. However, the SW Subsystem is not required to provide cooling to any SC1 components, so no specific testing requirements are implemented for this GDC.

GDC 60 – Control releases of radioactive materials to the environment. The PCW heat exchangers are designed to reduce the possibility of leakage from PCW to the SW Subsystem. If leakage were to occur, the SW Subsystem is monitored for radioactivity and can be isolated if high activity level is detected.

GDC 64 – Monitoring radioactive releases. Process sampling collects representative liquid samples for analysis and provides analytical information required for monitoring plant and equipment performance. Radiation monitoring of the SW Subsystem is provided downstream of each PCW heat exchanger. Therefore, the SW Subsystem meets the requirements of this GDC.

Additional information on GDC compliance is presented in Section 3.1.

Regulatory Guidance

Generic Letter 89-13 and Generic Letter 89-13, Supplement 1, “Service Water System Problems Affecting Safety-Related Equipment,” are considered in the design of the SW Subsystem. However, because the SW Subsystem has no Safety Category 1 functions, no specific guidance from these documents is implemented.

9.2.1.4 Testing and Inspection Requirements

Equipment and components of the SW Subsystem are designed for inspection and maintenance during plant operation. The SW Subsystem includes features (flow and temperature measure) to facilitate performance testing of the PCW heat exchangers.

9.2.1.5 Instrumentation Requirements

System Instrumentation and Control (I&C) logic provides controls for startup, monitoring, and shutdown of the SW pumps and associated valves. Radiation monitors on SW piping downstream of each PCW heat exchanger allow for the detection of high radiation levels.

SW pumps and remote valves are operated and monitored from the MCR. If an SW pump trips while in automatic control, the backup pump will auto-start and the PCW will re-align and put the associated heat exchanger in service. The trip of the pump will alarm in the MCR.

Additional information about the SW Subsystem I&C is found in Subsection 7.3.3.

9.2.2 Plant Cooling Water System

The PCW provides cooling water to SCN and SC3 components and provides a barrier against radioactive contamination of the SW Subsystem (Subsection 9.2.1) and NHS System (Subsection 9.2.9). The PCW has two closed-loop piping subsystems, the Reactor Component Cooling Water Piping Distribution, and the Turbine Component Cooling Water Piping Distribution, which provide cooling water to various heat exchangers.

9.2.2.1 Design Bases

9.2.2.1.1 Safety Category Functions

Safety Category 1 Functions

None.

Safety Category 2 Functions

None.

Safety Category 3 Functions

- Provides cooling water to the FPC heat exchangers
- Provides cooling water to Isolation Condenser Pools Cooling and Cleanup System (ICC) heat exchangers
- Provides cooling water for the Shutdown Cooling System (SDC) heat exchangers
- Provides cooling water for Plant Pneumatics System (PPS) compressor coolers
- Provides a barrier against radioactive contamination of the SW Subsystem and NHS System
- Provides cooling water to SC3 Turbine Building (TB) components and coolers

Safety Category N Functions

- Circulates cooling water removing heat from the SCN components in the Reactor Component Cooling Water Piping Distribution and the Turbine Component Cooling Water Piping Distribution during normal operating and shutdown conditions.

The Safety Classifications, Seismic Categories, and Quality Groups are presented in Table 3A-1.

9.2.2.2 System Description

Plant Cooling Water Subsystems

The PCW consists of two piping subsystems; Reactor Component Cooling Water Piping Distribution and Turbine Component Cooling Water Piping Distribution which provide cooling water to various heat exchangers.

PCW cooling water is circulated through various auxiliary equipment heat exchangers and rejects the heat to the SW Subsystem. The Turbine Component Cooling Water Piping Distribution is a single piping distribution that serves equipment in the TB and does not support reactor cooling functions. The Reactor Component Cooling Water Piping Distribution consists of two redundant piping distributions that support Reactor Building (RB) equipment cooling functions, PPS, and equipment located outside the RB associated with reactor cooling functions. Some systems cooled by the PCW contain fluid that has the potential to contaminate the PCW with radioactivity. The PCW is designed as a closed-loop system to act as a buffer between potentially radioactive systems and the nonradioactive SW Subsystem that transfers the heat to the environment.

A simplified diagram of the PCW is provided in Figure 9.2-2.

Plant Cooling Water System Components

Each cooling equipment train consists of a 100 percent capacity pump, 100 percent capacity heat exchanger, one surge tank, connecting piping, control valves, and instrumentation. Both trains share a chemical pot feeder. The two trains are normally cross connected at the supply and return headers during operation but may be isolated for individual train operation or maintenance. The PCW pumps and heat exchangers are in the TB. The pumps in each train are powered from separate buses. During a LOPP, the pumps are powered from the SDGs.

Surge tanks provide adequate pump suction head and allow for thermal expansion of the PCW inventory. Makeup water is provided to each PCW surge tank from the Demineralized Water Storage and Distribution System described in Subsection 9.2.3. Vents are placed at system high points and drains are placed in system low points. This ensures that the system is filled with water and that there are no air pockets. Vents also reduce the chance for water hammer after a pump start.

The design of the PCW heat exchangers follows the applicable subsection of ASME BPVC, Section VIII, Division 1, “Rules for Construction of Pressure Vessels,” (Reference 9.2-2). The chemical pot feeder is also designed to meet ASME BPVC, Section VIII requirements. Piping follows the applicable subsections of ASME B31.1 (Reference 9.2-1). The PCW surge tanks are atmospheric tanks and are designed to meet the requirements of AWWA D100, “Welded Carbon Steel Tanks for Water Storage,” (Reference 9.2-3). The PCW pumps are designed to meet the requirements of American National Standards Institute (ANSI)/Hydraulic Institute (HI) 1.3, “Rotodynamic Centrifugal Pumps for Design and Application,” (Reference 9.2-4).

Turbine Component Cooling Water Piping Distribution Heat Loads

- Pump motors and components in the TB
- Vacuum Pump Skid
- Generator Coolers
- Isophase Cooler
- Turbine Lube Oil Coolers
- Electro-Hydraulic Control Coolers

Reactor Component Cooling Water Piping Distribution Heat Loads

- Fuel Pool Cooling and Cleanup System (FPC) Heat Exchangers
- Isolation Condenser Pools Cooling and Cleanup System (ICC) Heat Exchangers
- SDC Heat Exchangers
- PPS Compressor Coolers

System Operation

During normal operations, plant cooling water flows to heat exchangers and coolers throughout the TB and RB by way of the Turbine Closed Cooling Water Piping Distribution and Reactor Closed Cooling Water Piping Distribution. One train of PCW is normally in operation while the other is in standby. The heat is rejected to the NHS System by SW that interfaces with PCW through the main PCW heat exchangers. The PCW is backed up by SDGs to support the system during a LOPP. Manual crosstie provisions placed upstream and downstream of each pump are available to provide operational flexibility.

9.2.2.3 Safety Evaluation

The PCW is not credited for mitigation of a DBA and has no safe shutdown functions.

Regulatory Requirements

Compliance with 10 CFR 50, Appendix A

The PCW complies with the following GDC, which has plant-wide applicability, as discussed in Section 3.1:

- GDC 5 – Sharing of structures, systems, and components

The following GDC have aspects specific to PCW and compliance is addressed as follows:

GDC 2 – Design bases for protection against natural phenomena. None of the equipment in the PCW, or cooled by the PCW, is required to mitigate DBAs and therefore, the system is not designed to withstand the effects of the most severe natural phenomena. In areas where PCW

piping or equipment may be in close proximity to SC1 equipment, the piping or equipment is designed to withstand the effects of natural phenomena only to protect nearby SC1 equipment.

GDC 4 – Environmental and dynamic effects design bases. PCW equipment is designed for the plant normal operating environmental conditions specified for its location within the TB and RB. None of the equipment in the PCW, or cooled by the PCW, is required to mitigate DBAs and, therefore, the PCW is not required to be qualified for adverse environmental conditions or be protected from potential dynamic effects of equipment failures.

GDC 44 – Cooling water. PCW removes heat from the SDC, fuel pool and Isolation Condenser (IC) pools under normal operating and normal cooldown conditions. The PCW also removes heat loads following an LOPP event once power to the PCW pumps is restored from the SDGs. However, the PCW is not credited for heat removal during DBAs. The ultimate heat sink is described in Subsection 9.2.5.

GDC 45 – Inspection of cooling water system. The PCW is designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system. The PCW is not required to provide cooling during DBAs. Therefore, there are no specific inspection requirements implemented for this GDC.

GDC 46 – Testing of cooling water system. The PCW is not required to provide cooling during DBAs. Therefore, no specific testing requirements are developed to meet this GDC.

Additional information on GDC compliance is presented in Section 3.1.

9.2.2.4 Testing and Inspection Requirements

Sampling provisions are available for detection of radioactivity and monitoring water quality in the PCW. The PCW is designed and constructed so that the radiation exposures to plant personnel during operation and maintenance are minimized.

Equipment and components of the PCW system are designed for inspection during plant operation. The manual crosstie provisions upstream and downstream of the PCW pumps provide operational flexibility if maintenance is required on a system pump or heat exchanger.

Preoperational and startup testing are conducted by the Initial Test Program, which is described in Chapter 14.

9.2.2.5 Instrumentation and Controls

The PCW is controlled and monitored from the MCR. During normal operation, one cooling train is operating, while being controlled by the Distributed Control and Information System (DCIS), to achieve the desired target temperature for each cooled line in the Reactor Component Cooling Water Piping Distribution and Turbine Component Cooling Water Piping Distribution. A low discharge pressure on the operating pump will initiate automatic transfer to the standby cooling train.

The SC3 I&C system provides monitoring and processing for Safety Category 3 function related I&C controls (e.g., logics, pressure, flow, temperature). This system also controls the PCW pumps and connects signals from different interfacing systems with their temperature control valves for the PCW to provide its cooling function. The SCN I&C system provides control and monitoring of SCN PCW equipment.

Level is monitored in the surge tanks for indication of system leakage.

Additional information about the PCW I&C is found in Subsection 7.3.3.

9.2.3 Demineralized Water Storage and Distribution Subsystem

The Demineralized Water Storage and Distribution Subsystem provides purified water for systems throughout the facility. The Demineralized Water Storage and Distribution Subsystem has Containment Isolation Valves (CIVs), which are not described in this section. The CIVs, their safety functions, and testing are described in Subsection 6.2.4.

9.2.3.1 Design Bases

Safety Category 1 Functions

None.

Safety Category 2 Functions

None.

Safety Category 3 Functions

None.

Safety Category N Functions

- Provides demineralized water storage and distribution for various systems in the Control Building (CB), TB, Radwaste Building (RWB), and RB

The Safety Classifications, Quality Groups, and Seismic Categories of the Demineralized Water Storage and Distribution Subsystem SSCs are listed in Table 3A-1.

9.2.3.2 System Description

The Demineralized Water Storage and Distribution Subsystem is comprised of the demineralized water storage tank, the demineralizer trailer which processes incoming raw water and provides makeup to the tank, redundant demineralized transfer pumps and associated piping, valves, and instrumentation. A piping tie-in is provided to the demineralized water storage tank for FLEX connections.

Demineralized water demands include but are not limited to: ICC, Boron Injection System (BIS), Chilled Water Equipment (CWE), Liquid Waste Management System (LWM), and PCW.

9.2.3.3 Safety Evaluation

The Demineralized Water Storage and Distribution Subsystem is not credited for mitigation of a DBA and has no safe shutdown functions.

Regulatory Requirements

Compliance with 10 CFR 50, Appendix A

The Demineralized Water Storage and Distribution Subsystem complies with the following GDC, which has plant-wide applicability, as discussed in Section 3.1:

- GDC 5 – Sharing of structures, systems, and components.

The Demineralized Water Storage and Distribution Subsystem is not designed to meet any GDCs. Additional information on GDC compliance is presented in Section 3.1.

9.2.3.4 Testing and Inspection Requirements

Equipment and components of the Demineralized Water Storage and Distribution Subsystem are designed for inspection and maintenance during plant operation. The system can be sampled to ensure that water quality requirements are satisfied.

9.2.3.5 Instrumentation and Controls

The I&C system provides controls for startup, monitoring, and shutdown of the demineralized water pumps and associated valves. Instrumentation monitors Demineralized Water Storage Tank level and pump discharge pressure.

9.2.4 Potable Water Distribution and Sanitary Sewer Handling Subsystems

The Potable Water Distribution and Sanitary Sewer Handling Subsystems provide potable water and remove wastewater from areas within the Power Block, but outside of the radiologically controlled area.

9.2.4.1 Design Bases

Safety Category 1 Functions

None

Safety Category 2 Functions

None

Safety Category 3 Functions

None

Safety Category N Functions

- Supplies potable water and sanitary sewer needs during normal operations to Power Block areas outside of the RCA.
- Supplies a source of make-up water to the FPS storage tanks.

9.2.4.2 System Description

The Potable Water Subsystem provides potable water to facility, including the restroom within the Main Control Room envelope, and other locations within the Power Block. The subsystem provides water to areas outside of the Power Block as well. Potable water is supplied by the municipal potable water source.

The Potable Water Subsystem also provides a source of make-up water to the Fire Protection System (FPS) make-up water tanks.

The Potable Water Distribution and Sanitary Sewer Subsystems Handling Subsystems are adequately sized for potable water, and sanitary sewer demands from the areas served. Above ground piping and instrumentation are designed to have heat tracing.

The Sanitary Sewer Handling Subsystem collects sanitary water from the facility restrooms and break areas and transfers it off-site to a connection with the municipality.

The Potable Water Distribution and Sanitary Sewer Handling Subsystems are not shared with the nearby combined-cycle gas turbine (CCGT) plant. Potable water is drawn from, and sanitary sewer waste is discharged to, the same municipal source as the CCGT plant.

9.2.4.3 Safety Evaluation

The Potable Water Distribution and Sanitary Sewer Handling Subsystems do not perform any Safety Category functions and are not credited for mitigation of an AOO or DBA and have no safe shutdown functions.

The Potable Water Distribution and Sanitary Sewer Handling Subsystems comply with the following GDC, which have plant-wide applicability, as discussed in Section 3.1:

GDC 5 – Sharing of structures, systems, and components.

The Potable Water Distribution and Sanitary Sewer Handling Subsystems comply with the following GDC, as described:

GDC 2 – Design Bases for Protection Against Natural Phenomena. Conformance with GDC 2 will be demonstrated in the FSAR internal flooding analysis in Section 3.4.1. The FSAR will describe how flooding due to failure of non-Seismic Category piping does not affect the ability of the plant to shut down safely, and remain in a safe shutdown condition.

GDC 60 – Control of releases of radioactive materials to the environment. The Potable Water Distribution and Sanitary Sewer Handling Subsystems do not connect with systems containing radioactivity.

Compliance with the GDC is further discussed in Section 3.1.

9.2.4.4 Inspection and Testing

Inspection and testing are performed consistent with applicable codes and standards.

9.2.4.5 Instrumentation and Controls

Provisions for monitoring and controls are provided to ensure the Potable Water Distribution Subsystem supplies an adequate quantity of potable water for plant operations.

9.2.5 Ultimate Heat Sink

The Ultimate Heat Sink (UHS) provides the Safety Category 1 function heat sink for DBAs and off-normal conditions when the NHS System is not available. The UHS consists of the water in the IC pools, the reactor cavity pool, the equipment pool, and the fuel pool. The UHS does not perform any functions under normal operating conditions except for water in the fuel pool. Normal cooling functions for the UHS pools are performed by the ICC for the IC pools and the FPC for the fuel pool, equipment pool, and reactor cavity pool.

9.2.5.1 Design Bases

Safety Category 1 Functions

- Water in IC pools provides heat removal from the Isolation Condenser System (ICS) under off-normal conditions.
- Water in the reactor cavity pool and the equipment pool provides heat removal from the Passive Containment Cooling System (PCCS) under off-normal conditions.
- Water in the fuel pool provides heat removal from the fuel in the pool.
- The IC pools, reactor cavity pool, equipment pool, and fuel pool are part of the RB and are Seismic Category I structures to support the SC1 UHS.

- Other Safety Category 1 functions related to the ICC are described in Subsection 9.2.8.

Safety Category 2 Functions

None.

Safety Category 3 Functions

- The FPC provides heat removal and makeup capacity for the fuel pool, reactor cavity pool, and equipment pool to ensure the proper volume and temperature of water are available for the UHS function for these pools. See Subsection 9.1.3 for additional information on the FPC.

Safety Category N Functions

None (see Subsection 9.2.8 for ICC functions).

The Seismic Category of RB structures is listed in Table 3A-2. The Safety Classifications, Quality Groups, and Seismic Categories of the UHS SSCs are listed in Table 3A-1 under the respective systems they belong to (such as ICS, ICC, or PCCS).

9.2.5.2 System Description

During plant operation and normal shutdown, the NHS System transfers heat from plant equipment to the environment. If the NHS System is not available, the Safety Category 1 function for heat removal is performed by the UHS. The UHS is a passive safety feature of the BWRX-300. Figure 9.2-3 shows the layout for the IC pools, reactor cavity pool, fuel pool, and the equipment pool. The UHS inventory is sufficient for 7 days without replenishment.

The IC pool volumes are independent from the equipment pool, fuel pool, and reactor cavity pool. During normal operation, the equipment pool and reactor cavity pool volumes are merged as a single pool volume, with the fuel pool separated by an installed gate. During refueling, the fuel pool gate is open, and all three pools function as a single volume.

During power operation, the water in the IC pools performs the UHS function for the ICS. When the ICS is actuated, the water in the IC pools absorbs heat from the ICS, thereby cooling the reactor core. As IC pool water is heated, steam is released through atmospheric vents to the outside atmosphere. The IC pools and atmospheric vents are described in Subsection 9.2.8. The design and operation of the ICS is discussed in Section 6.3. The IC pools also perform the UHS function for the reactor core during shutdown while the reactor vessel head is installed and the water level in the Reactor Pressure Vessel (RPV) is below the ICS steam lines.

For accident conditions that release fluid to containment, the water in the equipment pool and reactor cavity pool provides a heat sink for the PCCS. The PCCS is described in Subsection 6.2.2. The PCCS removes heat from the containment and reduces the containment temperature and pressure in accident conditions, such as Loss-of-Coolant Accidents (LOCAs) and other line breaks. As the reactor cavity pool and equipment pool heat up, vapor is released to the RB.

During refueling, the reactor vessel head is removed, and the reactor cavity is flooded. During this condition, the water in the RPV, reactor cavity pool, the equipment pool, and fuel pool (when gate is open) serves as the UHS. The water in the fuel pool is the UHS for fuel in the pool at all times.

If NHS is lost during refueling, or if fuel pool cooling is lost, water in UHS heats and vapor is released to the RB. Management of the transition from ICS to these flooded pools as the UHS is to be described in the FSAR.

The RB is normally vented to atmosphere. The vent can be isolated in response to a high radiation alarm, or due to a LOPP. Design features to manage vapor released from the fuel pool, reactor cavity pool, and equipment pool will be described in the FSAR.

9.2.5.3 Safety Evaluation

Regulatory Requirements

The UHS complies with the following GDC, which have plant-wide applicability, as discussed in Section 3.1:

GDC 1 – Quality standards and records

GDC 2 – Design bases for protection against natural phenomena

GDC 4 – Environmental and dynamic effects design bases

GDC 5 – Sharing of structures, systems, and components

The following GDC have aspects specific to the UHS and compliance is addressed as follows:

GDC 44 – Cooling water – Requirements for heat transfer to the UHS are met. In the event of a DBA during operational modes, heat is transferred from the fuel in the reactor and containment atmosphere to the water in the IC pools, equipment pool, and reactor cavity pool. Water in the IC inner pools is allowed to heat up and boil. The resulting steam is vented to the environment. The water volume in the smallest IC inner pool and the connected outer pools is sufficient to perform the Safety Category 1 function of transferring heat to the atmosphere for the initial 72-hours of an accident. The water volume of the larger IC inner pool (IC Inner Pool B+C) and the volume of the outer pools is sufficient to supply heat removal for 7 days. After 7 days, supplemental water can be provided through the IC pool FLEX/EME connection discussed in Subsection 9.2.8. The water volume in the equipment pool and reactor cavity pool is adequate to support PCCS for 7 days. The water volume in the fuel pool is also adequate to cool the fuel in the pool for 7 days.

GDC 45 – Inspection of cooling water system – The IC pools, equipment pool, reactor cavity pool, and fuel pool are designed to permit appropriate periodic ISI of important components, such as ICS and PCCS piping, to assure the integrity and capability of the systems.

GDC 46 – Testing of cooling water system – The IC pools, equipment pool, reactor cavity pool, and fuel pool are designed to permit appropriate Inservice Testing (IST) to assure the structural and leak-tight integrity of its components.

Additional information on GDC compliance is presented in Section 3.1.

Regulatory Guidance

Regulatory Guide (RG) 1.27, “Ultimate Heat Sink for Nuclear Power Plants,” is used as guidance for determining GDC applicability and a definition for the UHS. However, much of this RG is not applicable because the BWRX-300 is a passive plant and RG 1.27 provides guidance for plants that utilize active systems as the UHS.

9.2.5.4 Testing and Inspection Requirements

The IC pools, equipment pool, reactor cavity pool, and fuel pool, which support the UHS function, are part of the RB structure. An ISI and testing program is established for the RB to ensure the structure can fulfill its intended functions throughout its design service life.

Levels and temperatures in the IC pools, equipment pool, reactor cavity pool, and fuel pool are monitored and maintained within administrative and Technical Specification limits, as appropriate.

Preoperational and startup testing is performed by the Initial Test Program, which is described in Chapter 14.

9.2.6 Condensate Storage and Transfer Subsystem

The Condensate Storage and Transfer Subsystem supplies condensate to systems in the TB, RWB and the RB. It is a subsystem of the Liquid Waste Management System. The Condensate Storage and Transfer Subsystem has CIVs, which are not described in this section. The CIVs, their safety functions, and testing are described in Subsection 6.2.4.

9.2.6.1 Design Bases

Safety Category 1 Functions

None.

Safety Category 2 Functions

None.

Safety Category 3 Functions

None.

Non-Safety Category Functions

- Supplies condensate to systems in the TB, RWB, and the RB.

The Safety Classifications, Quality Groups, and Seismic Categories of the Condensate Storage and Transfer Subsystem SSCs are listed in Table 3A-1.

9.2.6.2 System Description

The Condensate Storage and Transfer Subsystem is part of the LWM and consists of the Condensate Storage Tank (CST), two transfer pumps, and distribution headers which supply condensate to the TB, RWB, and the RB for various systems. A simplified diagram of the subsystem is provided in Figure 9.2-4.

The Condensate Storage and Transfer Subsystem supplies condensate for the following uses:

- Makeup water to the Condenser Hotwell (Section 10.4.1)
- Alternate supply to the Control Rod Drive (CRD) pumps (Section 4.6)
- Makeup supply and backwash/flushing water for the FPC (Section 9.1.3)
- Flush water for the Solid Waste Management System (SWM) (Section 11.4)
- Flush water, filter backwash water and water for transferring resin and filter media for the LWM (Section 11.2)
- Condensate for resin transfer, resin washing, and filter backwash processes for the Condensate Filters and Demineralizers System (CFD) (Section 10.4.6)
- Water supply to Containment for under-vessel washdown during outages
- CST provides additional makeup water to the reactor cavity pool during cavity refill if required.

During normal operation, when the condenser hotwell reaches low water level, the Condensate Storage and Transfer Subsystem provides makeup water to the hotwell. When the hotwell level is high, CST receives condensate reject from the Condensate and Feedwater Heating System (CFS). During shutdown, the reactor temperature decreases causing the water volume

in the reactor to shrink. To maintain the normal water level in the reactor, the condensate pumps in the CFS provide the additional volume needed from the hotwell. If additional condensate is required, makeup to the hotwell is provided from the CST.

The CST can provide additional makeup water to the reactor cavity pool during the cavity refill to compensate for removal of the dome. CST makeup to the cavity is through the FPC or by makeup to the Refueling Water Storage Tank (RWST).

The CST is a stainless-steel atmospheric tank constructed in accordance with American Petroleum Institute (API)-650, "Welded Tanks for Oil Storage," (Reference 9.2-5). Piping from the CST is designed to meet the requirements of ASME B31.1 (Reference 9.2-1).

9.2.6.3 Safety Evaluation

The Condensate Storage and Transfer Subsystem complies with the following GDC, which has plant-wide applicability, as discussed in Section 3.1:

GDC 5 – Sharing of structures, systems, and components.

The following GDC have aspects specific to the Condensate Storage and Transfer Subsystem and compliance is addressed as follows:

GDC 2 – Design bases for protection against natural phenomena – This GDC is considered in the design of the Condensate Storage and Transfer Subsystem. The washdown connection has CIVs and is designed to meet this GDC as described in Subsection 6.2.4. The design and layout of the Condensate Storage and Transfer Subsystem include provisions to ensure that a failure of the system will not adversely affect the functional performance of SC1 components. The Condensate Storage and Transfer Subsystem is not required to operate during or after an accident.

GDC 44 – Cooling water – The Condensate Storage and Transfer Subsystem does not provide any cooling functions; therefore, there are no requirements implemented for this GDC.

GDC 45 – Inspection of cooling water system – The Condensate Storage and Transfer Subsystem does not provide any cooling functions; therefore, there are no requirements implemented for this GDC.

GDC 46 – Testing of cooling water system – The Condensate Storage and Transfer Subsystem does not provide any cooling functions; therefore, there are no requirements implemented for this GDC.

GDC 60 – Control releases of radioactive materials to the environment – The Condensate Storage and Transfer Subsystem does not have direct liquid release paths to the environment. Water from this subsystem is processed through the LWM described in Section 11.2. The LWM complies with GDC 60.

Additional information on GDC compliance is presented in Section 3.1.

10 CFR 50.63 – Station Blackout (SBO). Water in the Condensate Storage and Transfer Subsystem is not credited for the mitigation or coping time for an SBO event.

9.2.6.4 Testing and Inspection Requirements

Provisions are made for periodic inspection of major components to ensure capability and integrity of the systems. The system design provides adequate equipment removal paths and personnel access for replacement.

9.2.6.5 Instrumentation Requirements

The Condensate Storage and Transfer Subsystem is operated and monitored from the MCR. Major system parameters are indicated and alarmed in the MCR and locally, as required, to provide operational information and performance assessment. The following instrumentation is provided:

- CST Level
- Condensate transfer pump discharge pressure
- Condensate distribution header temperature

Additional information about the Condensate Storage and Transfer Subsystem I&C is found in Subsection 7.3.4.

9.2.7 Chilled Water Equipment

The CWE provides chilled water to various equipment to support cooling functions. The CWE has CIVs, which are not described in this section. The CIVs, their safety functions, and testing are described in Subsection 6.2.4.

9.2.7.1 Design Bases

Safety Category 1 Functions

None.

Safety Category 2 Functions

None.

Safety Category 3 Functions

- The CWE provides chilled water to SC3 equipment.
- The CWE supplies chilled water to the Containment Cooling System (CCS) Air Handling Units (AHUs) (Subsection 9.4.7).
- During a LOPP, one chiller/pump set from each chiller train can be powered using backup power provided by the SDGs to provide chilled water to equipment in the RB and CB.

Safety Category N Functions

- The CWE provides chilled water to SCN equipment.

The Safety Classification, Seismic Categories and Quality Groups are presented in Appendix 3A.

9.2.7.2 System Description

The CWE provides chilled water cooling to the Heating, Ventilation, and Cooling System (HVS) throughout the plant, to the offgas cooler condenser, and to the CCS in the RB.

The CWE is a closed-loop chilled water system that supplies chilled water to various AHU cooling coils and plant equipment coolers in the TB, RWB, RB, and CB. Heat absorbed by the CWE is rejected from the CWE chiller condensers to atmosphere. The expansion tanks create positive pressure throughout the system.

The CWE is split into two trains which allows for the isolation of the two different sets of AHUs and Fan Coil Units (FCUs) supporting redundant Defense-in-Depth (D-in-D) equipment. Both CWE trains are cross-tied together by valves, which are normally open, to allow for the three

active chillers to evenly share the heat load; however, these valves can be closed for train separation.

The CWE chillers utilize specific site-approved refrigerant to exchange heat with the propylene glycol and water mixture. Because these units are mounted outside, the use of refrigerant does not pose a risk to plant personnel.

Four chiller pumps are provided to deliver chilled water from the CWE return header to the dedicated chiller. During normal operation, three pumps are in service while the fourth is in standby mode. Air separators are provided to remove entrained air, which can cause system noise, excess wear, and pump cavitation.

The two expansion tanks are connected to the two main chilled water return lines and provide a reservoir of demineralized makeup water to account for small amounts of system leakage and accommodate thermal expansion and contraction of water within the system while maintaining the system pressure.

Chilled water supply to the AHUs, the offgas coolers, and CCS AHU fan cooling coils is regulated by remote operated valves that are controlled by the HVS, Offgas System (OGS), and CCS operation, respectively. To prevent water hammer from occurring in the CWE pipeline, the chillers are incrementally slowed to their minimum operating flow before being switched off one by one.

A single chemical bypass feeder supplies corrosion/scale inhibitors and pH adjustment and biocide chemicals, such as sodium hypochlorite, into the CWE system to prevent the development of organics within the closed chilled water loop.

Refrigerant piping meets the requirements of ASME Code B31.5, “Refrigeration Piping and Heat Transfer Components,” (Reference 9.2-6). Non-refrigerant piping is designed in accordance with the requirements of ASME B31.1 (Reference 9.2-1), with the exception of the containment penetration portion. The CIVs and lines that supply and return chilled water to the CCS cooling coils are described in Subsection 6.2.4. The CWE piping that penetrates containment is provided with relief valves to protect against overpressure caused by the thermal expansion.

A simplified diagram of the CWE is provided in Figure 9.2-5.

During a LOPP, one chiller and pump from each chiller train can be powered using backup power provided by the SDGs. One chiller and pump restarts upon receiving power from the SDG. The CWE supply header isolation valves to the TB and RWB automatically close. Chilled water flows only through the equipment in the RB and CB to support heat loads as required with the available CWE capacity.

9.2.7.3 Safety Evaluation

The CWE complies with the following GDC, which has plant-wide applicability, as discussed in Section 3.1:

GDC 5 – Sharing of structures, systems, and components

The following GDC have aspects specific to the CWE and compliance is addressed as follows:

GDC 2 – Design bases for protection against natural phenomena – None of the equipment in the CWE, or cooled by the CWE, is required to mitigate DBAs. In areas where the CWE piping or equipment may be in close proximity to SC1 equipment, the piping or equipment is designed to withstand the effects of natural phenomena, but only to protect nearby SC1 equipment.

GDC 4 – Environmental and dynamic effects design bases – The CWE is designed for the plant normal operating environmental conditions specified for its location within the TB and RB. None of the equipment in the CWE, or cooled by the CWE, is required to mitigate DBAs and, therefore,

CWE is not required to be qualified for adverse environmental conditions or be protected from potential dynamic effects of equipment failures.

GDC 44 – Cooling water – None of the equipment in the CWE is required to mitigate a DBA or is credited for cooling equipment required to mitigate a DBA. Therefore, no requirements are implemented for this GDC.

GDC 45 – Inspection of cooling water system – The CWE is designed for appropriate periodic inspection of important components, such as heat exchanges and piping, to assure the integrity and capability of the system. No SC1 equipment is cooled by the CWE, and no specific inspection requirements are established for this GDC.

GDC 46 – Testing of cooling water system – The CWE is designed for appropriate periodic testing of important components, such as heat exchangers and piping, to assure the integrity and capability of the system. No SC1 equipment is cooled by the CWE, and no specific testing requirements are established for this GDC.

Additional information on GDC compliance is presented in Section 3.1.

10 CFR 20.1406(a) – Some of the heat exchangers cooled by the CWE are in radiologically controlled areas. Programs are in place to minimize the spread of contamination as discussed in Section 12.1.

9.2.7.4 Testing and Inspection Requirements

The CWE chillers are tested in accordance with Air-Conditioning, Heating, and Refrigeration Institute (AHRI) Standard 550/590, “Performance Rating of Water-chilling and Heat Pump Water-heating packages using the Vapor Compression Cycle,” (Reference 9.2-7) and American Society of Heating, Refrigerating, and Air-Conditioning Engineers (ASHRAE) Standard 30, “Method of Testing Liquid Chillers,” (Reference 9.2-8). Chillers and pumps are arranged to provide adequate floor space and unobstructed clearance to permit monitoring, maintenance, and inspection.

9.2.7.5 Instrumentation Requirements

The CWE contains instrumentation designed to measure system temperatures, pressures, and flow, which can be monitored from the MCR. Additional information about the CWE I&C is found in Subsection 7.3.3.

As described during a LOPP, CWE controls close the supply header isolation valves to the TB and RWB automatically to divert available chilled water capacity to the RB and CB to support heat loads as required.

9.2.8 Isolation Condenser Pools Cooling and Cleanup System

The ICC maintains IC pool temperature, level, and water quality.

9.2.8.1 Design Bases

Safety Category 1 Functions

- The water in the IC pools is the UHS, which removes heat from the reactor core through the ICS during adverse conditions and DBAs. The ICS functions are described in Section 6.3.
- The IC pools atmospheric vents minimize pressurization of the IC pool compartments by providing a low-resistance flow path for steam to be released to the environment during IC deployment.

- The unidirectional flow conduits allow makeup water to flow from the IC outer pools into the IC inner pools following extended IC deployment with significant loss of pool water inventory due to boiloff.

Safety Category 2 Functions

None.

Safety Category 3 Functions

- The FLEX/EME Makeup Lines provide long-term makeup water to the IC pools from an external and independent clean water source following extended IC deployment with significant loss of pool water inventory due to boiloff.
- The Filter/Storage Pool vent directs steam and other containment gases to the environment from the CIS overpressure vent to limit Filter/Storage Pool compartment pressurization and to maintain integrity of the RB.

Safety Category N Functions

- The ICC maintains the level of the IC pools within a nominal range during normal operations that ensures sufficient volume for 7 days of passive heat removal during off-normal operation.
- The ICC extracts heat from the IC pool water to maintain IC pool temperature within administrative limits during normal operation.
- The ICC removes impurities from the IC pool water to comply with plant water quality requirements during normal operation.

The Safety Classifications, Quality Groups, and Seismic Categories of the ICC SSCs are listed in Table 3A-1.

9.2.8.2 System Description

The ICC maintains the water in the IC pool complex which consists of the IC inner pools, IC outer pools, and the filter/storage pool. Functions of the ICC include maintaining water temperature within prescribed administrative limits, maintaining the cleanliness of the pools, and providing the capability to add clean makeup water during normal reactor operations to offset the routine and minor loss of water inventory due to evaporation. A simplified diagram of the ICC is provided in Figure 9.2-6.

The IC pools are required to retain water so that the ICS System can perform its Safety Category 1 function. During adverse conditions, the IC pools remove heat from the ICs and reject it to the environment. The IC pools do not perform any functions under normal conditions.

The IC pools are physically separated by walls which are part of the RB structure. IC A is located within IC Inner Pool A (surrounded by IC Outer Pool A). IC B and IC C are located within IC Inner Pool B+C (surrounded by IC Outer Pool B+C). The water inside the pools is managed by ICC.

There are three types of connections between the IC outer pools and inner pools which provide limited fluid communication between the pools:

- Water conduits, which are located in the partition wall and bidirectional. They are at a higher elevation that maintains consistent level in the inner and outer pool provided the water conduits are submerged.

- Unidirectional pool makeup conduits, which are located in the partition wall at a lower elevation (below the water conduits and also below the reference point for minimum usable IC pool volume). The unidirectional pool makeup conduits only permit flow from the IC outer pool to the IC inner pool, preventing inner pool inventory losses due to an outer pool being ruptured and drained.
- Steam conduits, which are located in the partition wall above the pool water level.

Water conduits are also provided between the IC outer pools so that there is a common outer pool water level, and the outer pool water is available to share with the two IC inner pools. Normal IC pool makeup is provided by a demineralized water supply (Subsection 9.2.3). A drainage network allows each pool to be drained individually for maintenance or inspection.

Atmospheric vents for the IC pools are located in the outer pool ceiling near the RB exterior wall. Generated steam during extended ICS deployment flows from the inner pool to the outer pool through the steam conduits and then through the vent to the outside environment. The atmospheric vents are designed to limit pressurization within the IC pools.

The filter/storage pool functions as a suppression pool for the Containment Inerting System (CIS) containment overpressure vent and subsequent exhaust path to the atmosphere in the event containment venting is required. The filter/storage pool is also used to temporarily store IC pool water when the IC pools are drained for maintenance or inspection. The filter/storage pool is structurally isolated from the IC pools but does interface with ICC system piping for draining, recirculation, and replenishment. The filter/storage pool does not support ICS operation in any way.

An atmospheric vent is also installed in the filter/storage pool to direct containment gases from the filter/storage pool to the environment. The vent is designed to limit filter/storage pool compartment pressurization to maintain the integrity of the RB. The atmospheric vent penetrates the outer RB wall near the filter/storage pool ceiling and extends above building structure. Check valves are provided to prevent air intake during accident scenarios in which the filter/storage pool may be a hydrogen-rich environment.

The component requirements for the IC inner and outer pools are listed below:

- The smaller IC inner pool (IC Inner Pool A) and the full volume of the outer pools shall be capable of cooling the reactor for 72 hours after ICS initiation without replenishment.
- The larger IC inner pool (IC Inner Pool B+C) and the full volume of the outer pools shall be capable of cooling the reactor for 7 days after ICS initiation without replenishment.

FLEX/EME connections are provided outside the RB for long-term refilling of the IC pools and filter/storage pool. For redundancy, two external connections are provided, each on opposite sides of the RB. Either connection can be used to refill the IC inner pools and/or filter/storage pool with makeup water from an external and independent source. Check valves are installed to prevent backflow.

The ICC includes two cooling trains, each equipped with a centrifugal pump and heat exchanger to transfer energy from the IC pools to the PCW (Subsection 9.2.2). A booster pump is installed in the drain line from the filter/storage pool for injection into the IC pools suction line. The booster pump provides the head required to overcome the static head of the IC pools for recirculation of filter/storage pool water. The ICC also includes a demineralizer to remove impurities from the IC pool and filter/storage pool water. The two independent ICC trains allow maintenance to be performed on one train during reactor operation without having to disable the entire ICC. The demineralizer can be isolated to permit resin exchange and other maintenance activities to be

performed during reactor and ICC operation. The demineralizer is bypassed to prevent degrading the resin media if the heat exchanger discharge temperature exceeds the maximum recommended service temperature for the resin media. The demineralizer is also isolated during system preventive maintenance in which chemicals are injected into the process water to inhibit corrosion and biological activity in the IC and filter/storage pools and ICC components. Vents at high locations in the piping system ensure pipes are filled with process water and eliminate air pockets. The ICC SCN piping is designed in accordance with the requirements of ASME B31.1 (Reference 9.2-1). The ICC SC1 piping is designed in accordance with ASME BPVC Section III, Division 1, “Rules for Construction of Nuclear Facility Components”, Subsection NCD “Class 2 and Class 3 Components,” (Reference 9.2-9). The ICC pumps are designed to comply with the requirements of ANSI/HI 1.3 (Reference 9.2-4) or API 610, “Centrifugal Pumps for Petroleum, Petrochemical, and Natural Gas Industries,” (Reference 9.2-10).

The ICC draws water for subsequent cooling and/or purification from the Suction Surge Tank located in the IC Outer Pool B+C. The ICC can also take suction from the IC pool drains and the filter/storage pool drain. The Suction Surge Tank is a watertight, open-top, cylindrical enclosure surrounding the ICC main suction pipe and the makeup water supply pipe. The top of the Suction Surge Tank terminates between the normal minimum and maximum water level of the IC pools. This design prevents IC outer pool water level from falling below the top of the Suction Surge Tank (and subsequently, normal minimum IC pool water level) in the event of a non-isolable pipe break below the IC pools.

Processed water is returned to the IC inner pools through a piping network that ends at distribution spargers mounted near the ICs and the bottom of the filter/storage pool. The Return Guard Tank is a dry, watertight, cylindrical enclosure surrounding the individual return pipes to the IC inner pools and the filter/storage pool. The top of the Return Guard Tank extends above the maximum water level of the IC pools. The return lines to the IC inner pools and filter/storage pool can be individually isolated, as needed, to accommodate normal plant operations. Each return line includes an anti-siphon device to prevent the inadvertent draining of an IC inner pool in the event of pipe failure outside of the IC Outer Pool B+C. The Suction Surge Tank and Return Guard Tank are designed to prevent draining the IC outer pools in the event a pipe break occurs below the IC pool elevation.

The water contained in the IC and filter/storage pools is maintained as high purity water that meets or exceeds plant water quality specifications.

Most ICC components have no safety category functions and are therefore classified as SCN. Specific exceptions to this classification apply to designed safety features of the IC pools including: (1) the Suction Surge Tank, Return Guard Tank (including enclosed piping and anti siphon device), which serves as the primary physical interface with the Reactor Building IC pool structure; (2) the IC pools atmospheric vents, which minimize pressurization of the IC pool compartments; (3) the unidirectional pool makeup conduits, which allow makeup water to flow from the outer pools to the IC inner pools during off-normal conditions; and (4) IC pool isolation valves, which are attached to the pool penetrations that are contained within the Suction Surge Tank and Return Guard Tank. These components are classified as SC1, consistent with the overall IC pool structures classification to ensure the fundamental safety function of the ICS is maintained. The long-term FLEX/EME pool makeup lines, which provide a means to add makeup water from an external source, and filter/storage pool atmospheric vent are SC3.

9.2.8.3 Safety Evaluation

The Safety Category 1 core cooling function is accomplished by the IC pools water volume and atmospheric vents during DBAs. ICC cooling of the pool water is not required during DBAs.

ICC complies with the following GDC, which has plant-wide applicability, as discussed in Section 3.1:

GDC 5 – Sharing of structures, systems, and components

The following GDC have aspects specific to ICC and compliance is addressed as follows:

GDC 2 – Design bases for protection against natural phenomena – The IC pools are part of the RB and are Seismic Category I structures that meet GDC 2. The Suction Surge Tank and the Return Guard Tank are designed meet GDC 2 to prevent draining of the IC outer pools in the event of a break in ICC piping below the IC pools. ICC is designed such that there is no adverse interaction with either the SC1 IC pools or SC1 IC (and its connected ICS piping) during a seismic event.

GDC 4 – Environmental and dynamic effects design bases – The IC Pools are part of the RB and are Seismic Category I structures that protect SSCs against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures and from events and conditions outside the nuclear power unit. The Suction Surge Tank and the Return Guard Tank are evaluated for environmental and dynamic effects of design basis events.

GDC 44 – Cooling water – ICC cools the water in the IC pools under normal conditions. During off-normal conditions, ICS transfers heat to the IC pools. The safety function GDC compliance for the IC pools is addressed in Subsection 9.2.5. The normal cooling function of the ICC is not assumed for accident conditions, so the requirements of this GDC are not applicable to the ICC normal cooling functions.

GDC 45 – Inspection of cooling water system – The ICC is designed to allow appropriate periodic inspection of important components, such as heat exchanges and piping, to assure the integrity and capability of the ICC System. However, the cooling function of the ICC is not required for DBAs and therefore no specific inspection requirements are implemented for this GDC.

GDC 46 – Testing of cooling water system – The ICC is designed to allow appropriate periodic pressure and functional testing to assure structure and leak-tight integrity of ICC components. However, the cooling function of the ICC is not required for DBAs and therefore no specific testing requirements are implemented for this GDC.

Additional information on GDC compliance is presented in Section 3.1.S.9.2

9.2.8.4 Testing and Inspection Requirements

Mechanical systems, components, and equipment, in which continued availability of their function is necessary to assure that the plant design availability requirement is met, are designed to permit inspection, and testing under conditions of normal operation to the greatest extent practical. Service platforms are installed to facilitate inspection of the Surge Tank and Return Guard Tank. The design of the ICC allows for the inspection and testing of components under normal operating conditions to the greatest extent practicable.

Isolation valves are provided near the IC pool penetrations to permit maintenance and repair activities to be performed on pipe sections below the IC pools. Valves are installed with flanged connections to facilitate removal as needed for testing or repair.

9.2.8.5 Instrumentation Requirements

The ICC contains instrumentation designed to measure local process flow variables: water level, temperature, pressure, differential pressure, and conductivity. Additional information about the ICC I&C is described in Subsection 7.3.3.

9.2.9 Normal Heat Sink

The NHS System provides heat rejection for the CW Supply (Subsection 10.4.5) and SW Subsystem during normal operation and shutdown.

9.2.9.1 Design Bases

Safety Category 1 Functions

None.

Safety Category 2 Functions

None.

Safety Category 3 Functions

The NHS System provides heat rejection for the SW Subsystem SC3 loads (Subsection 9.2.1).The NHS System provides heat rejection for CW Supply SC3 loads (Subsection 10.4.5).**Safety Category N Functions**

- The NHS System provides heat rejection for the SW Subsystem SCN loads during normal operating and shutdown conditions.

The Safety Classifications, Quality Groups, and Seismic Categories of the NHS System SSCs are listed in Table 3A-1.

9.2.9.2 System Description

The NHS System provides support services for the power block and site facilities, which includes cooling of the main condenser via the CW Supply Subsystem, and the SW Subsystem. A simplified diagram of the NHS System is provided in Figure 9.2-7.

The NHS Subsystem includes, but is not limited to, the following components:

- Mechanical draft wet cooling tower cells;
- Cooling tower make-up tanks and associated pumps;
- Piping and associated components to route cooling tower basin blow down off-site to the wastewater treatment plant;
- Instrumentation and provisions to control water chemistry; and
- Various connections to other systems.

The NHS supports heat removal during normal operations and normal decay heat removal during shutdown.

The NHS System provides adequate pumping capacity to provide makeup water to the cooling tower basin. Makeup water pumps are provided to maintain cooling tower basin level. Basin level is monitored and controlled, as is the makeup tank level. The NHS System contains equipment to provide chemical treatment, as necessary, and blowdown lines. Blowdown is discharged to the off-site wastewater treatment plant. Makeup water can also be provided to the fire water storage tanks.

A local deep well aquifer serves as the makeup water source for the NHS cooling tower basin, replenishing water lost to evaporation, drift, and blowdown. Water from the aquifer is processed and treated off-site and then pumped to the site. Blowdown is routed to the off-site water treatment plant.

9.2.9.3 Safety Evaluation

Regulatory Requirements

Compliance with 10 CFR 50, Appendix A

The NHS System complies with the following GDC, which has plant-wide applicability, as discussed in Section 3.1:

GDC 5 – Sharing of structures, systems, and components

The following GDC have aspects specific to the NHS System and compliance is addressed as follows:

GDC 2 – Design bases for protection against natural phenomena – This GDC is considered in the design of the NHS. No SC1 SSCs require cooling by this system to mitigate DBAs. The design and layout of the NHS include provisions to ensure that a failure of the system will not adversely affect the functional performance of SC1 systems or components.

GDC 4 – Environmental and dynamic effects design bases – None of the equipment in the NHS, or cooled by the system, is required to mitigate DBAs and, therefore, the NHS is not required to be qualified for adverse environmental conditions or be protected from potential dynamic effects of equipment failures.

GDC 44 – Cooling water – The NHS System transfers heat from plant equipment to the environment during normal power and shutdown operation. However, the NHS System function is not relied upon in the safety analyses. The UHS, discussed in Subsection 9.2.5, provides the Safety Category 1 heat sink for equipment credited in safety analyses. Therefore, no specific requirements for the NHS System are implemented for this GDC.

GDC 45 – Inspection of cooling water system – Similar to GDC 44, the NHS System transfers heat from plant equipment to the environment during normal power and shutdown operation. However, the NHS function is not relied upon in the safety analyses. Therefore, no specific requirements for the NHS System are implemented for this GDC.

GDC 46 – Testing of cooling water system – Similar to GDC 44, the NHS System transfers heat from plant equipment to the environment during normal power and shutdown operation. However, the NHS System function is not relied upon in the safety analyses. Therefore, no specific requirements for the NHS System are implemented for this GDC.

GDC 64 – Monitoring radioactivity releases – The NHS System design controls radioactive material releases to the environment. The input streams to the NHS System from potentially radioactive systems are monitored for radioactivity. The LWM is equipped with a discharge path to the CWS, for dilution and release to the environment. The discharge path contains a radiation monitor and valves that automatically isolate if radiation greater than a preset limit is detected in the flow stream. The SW Subsystem is also monitored for radioactivity before discharging to the NHS. Therefore, the NHS System meets the requirements of this GDC.

Additional information on GDC compliance is presented in Section 3.1.

9.2.9.4 Testing and Inspection Requirements

Equipment and components in the NHS System are designed and configured to allow for inspection and maintenance. Testing and inspection will be performed consistent with applicable codes and standards for the NHS System. The cooling tower basin allows for sampling to monitor water chemistry parameters.

9.2.9.5 Instrumentation Requirements

The following parameters will be monitored:

- Cooling tower basin level, which ensures adequate volume to perform required functions;
- Makeup water storage tank level to monitor any disruption in makeup water supply;
- Water quality is monitored to control blowdown rate; and
- Temperature and pressure of makeup water are measured.

The NHS System is operated and monitored from the MCR.

9.2.10 Refueling Water Storage and Cleanup Subsystem

The Refueling Water Storage and Cleanup Subsystem provides and stores water to support refueling outage activities. It is a subsystem of the Liquid Waste Management System.

9.2.10.1 Design Bases

Safety Category 1 Functions

None.

Safety Category 2 Functions

None.

Safety Category 3 Functions

- Confinement of water that contains radioactivity.

Safety Category N Functions

- Transfer, hold, and filter water used to fill the reactor cavity pool and other connected systems.

The Safety Classifications, Quality Groups, and Seismic Categories of the Refueling Water Storage and Cleanup Subsystem SSCs are listed in Table 3A-1.

9.2.10.2 System Description

The Refueling Water Storage and Cleanup Subsystem consists of the RWST, two transfer pumps, a filter, and associated piping and valves. The RWST is used to hold the reactor cavity pool volume to permit draining and filling the reactor cavity to support refueling outage activities. The piping between the reactor cavity pool and RWST contains a filter to maintain water purity during movement activities. Water is sampled to ensure water quality meets requirements before being pumped back into the reactor cavity pool. The RWST also accepts overboarding flow from the Reactor Water Cleanup System (CUW) and the SDC (Section 5.4). The Refueling Water Storage and Cleanup Subsystem serves no functions for mitigation of a DBA and has no safe shutdown functions. A simplified diagram of the Refueling Water Storage and Cleanup Subsystem is provided in Figure 9.2-8.

The RWST is a stainless steel atmospheric tank sized to accommodate the reactor cavity pool volume during an outage. The RWST is constructed in accordance with API-650, “Welded Tanks for Oil Storage” (Reference 9.2-5).

9.2.10.3 Safety Evaluation

The Refueling Water Storage and Cleanup Subsystem is not credited for the mitigation of any DBAs. RWST is Safety Category 3 for confinement of radioactivity.

The Refueling Water Storage and Cleanup Subsystem complies with the following GDC, which has plant-wide applicability, as discussed in Section 3.1:

GDC 5 – Sharing of structures, systems, and components

The following GDC have aspects specific to the Refueling Water Storage and Cleanup Subsystem and compliance is addressed as follows:

GDC 2 – Design bases for protection against natural phenomena. This GDC is considered in the design of the Refueling Water Storage and Cleanup Subsystem. This subsystem is not required to operate during or after a design basis accident. It conforms to GDC 2 such that a failure of the system will not adversely affect the functional performance of SC1 components. No specific requirements for SC1 SSC interactions have been identified at this time, but is to be confirmed in the FSAR considering final piping layout and the internal flooding analysis. The RWST is located in the Turbine Building (TB) and the TB structural design requirements are discussed in Table 3.7-1 and Subsection 6.1.1 of NEDO-33914-A, "BWRX-300 Advanced Civil Construction and Design Approach," (Reference 9.2-11). The structure is designed per the International Building Code (Reference 9.2-12) in combination with ASCE/SEI 7-16, "Minimum Design Loads and Associated Criteria for Buildings and Other Structures," (Reference 9.2-13) to meet Risk Category IV requirements. Subsystem piping is designed to the requirements of ASME B31.1. Refer to the GDC 60 evaluation for discussion related to maintaining concentrations in the Refueling Water Storage and Cleanup Subsystem below RW-IIa limits in RG 1.143.

GDC 60 – Control releases of radioactive materials to the environment. The Refueling Water Storage and Cleanup Subsystem complies with GDC 60. Level is monitored in the RWST and high-level alarms are provided both locally at the tank and in the MCR to minimize risk of potential tank overflow. In the event of accidental tank overflow, an overflow connection is provided below the tank vent and above the high-level alarm setpoint which can be directed to the Equipment and Floor Drain System (EFS). EFS is described in Subsection 9.3.3. Walls/curbs are provided around the RWST to collect leaks and prevent release to groundwater. Acceptance of overboarding flow from CUW/SDC to the RWST is administratively controlled. Sampling of the contents is performed to ensure radioactive concentrations are below preset limits. Locked valves in the piping from the CUW/SDC overboard implement this administrative control. Water in the Refueling Water Storage and Cleanup Subsystem can be filtered and sampled to ensure that water quality meets requirements. If required, RWST contents can be transferred to the collection tanks in LWM described in Section 11.2 for further processing. There is no direct discharge path from the Refueling Water Storage and Cleanup Subsystem to the environment.

Additional information on GDC compliance is presented in Section 3.1.

9.2.10.4 Testing and Inspection Requirements

Provisions are made for periodic inspection of major components to ensure capability and integrity of the systems. The system design provides adequate equipment removal paths and personnel access for replacement. Adequate space is provided for maintenance of the pumps, valves, and instruments.

9.2.10.5 Instrumentation Requirements

The Refueling Water Storage and Cleanup Subsystem is operated and monitored from the MCR. Major system parameters are indicated and alarmed in the MCR and locally, as required, to provide operational information and performance assessment. The instrumentation includes:

- RWST level
- RWST transfer pump common suction and discharge pressure

- RWST filter differential pressure

9.2.11 References

9.2-1 ASME B31.1, “Code for Pressure Piping, Power Piping,” American Society of Mechanical Engineers, 2022.

9.2-2 ASME BPVC, Section VIII, Division 1, “Rules for Construction of Pressure Vessels,” American Society of Mechanical Engineers, 2021.

9.2-3 AWWA D100, “Welded Carbon Steel Tanks for Water Storage,” American Water Works Association, 2021.

9.2-4 ANSI/HI 1.3, “Rotodynamic Centrifugal Pumps for Design and Application,” American National Standards Institute, Hydraulic Institute, 2013.

9.2-5 API 650, “Welded Tanks for Oil Storage,” Thirteenth Edition, American Petroleum Institute, 2021.

9.2-6 ASME Code B31.5, “Refrigeration Piping and Heat Transfer Components,” American Society of Mechanical Engineers, 2019.

9.2-7 AHRI Standard 550/590 (I-P), “Performance Rating of Water-chilling and Heat Pump Water-heating Packages Using the Vapor Compression Cycle,” Air Conditioning, Heating, & Refrigeration Institute, 2023.

9.2-8 ANSI/ASHRAE Standard 30, “Method of Testing Liquid Chillers,” American National Standards Institute/American Society of Heating, Refrigerating, and Air-Conditioning Engineers, 2019.

9.2-9 ASME BPVC, Section III, Division I, “Rules for Construction of Nuclear Facility Components, Subsection NCD – Class 2 and Class 3 Components,” American Society of Mechanical Engineers, 2021.

9.2-10 API 610, “Centrifugal Pumps for Petroleum, Petrochemical, and Natural Gas Industries,” 12th Edition, American Petroleum Institute, 2021.

9.2-11 NEDO-33914-A, “BWRX-300 Advanced Civil Construction and Design Approach,” GE-Hitachi Nuclear Energy Americas, LLC, Revision 2, June 2022.

9.2-12 IBC, “International Building Code,” International Code Council, 2021.

9.2-13 ASCE/SEI 7-16, “Minimum Design Loads and Associated Criteria for Buildings and Other Structures,” American Society of Civil Engineers, 2016.

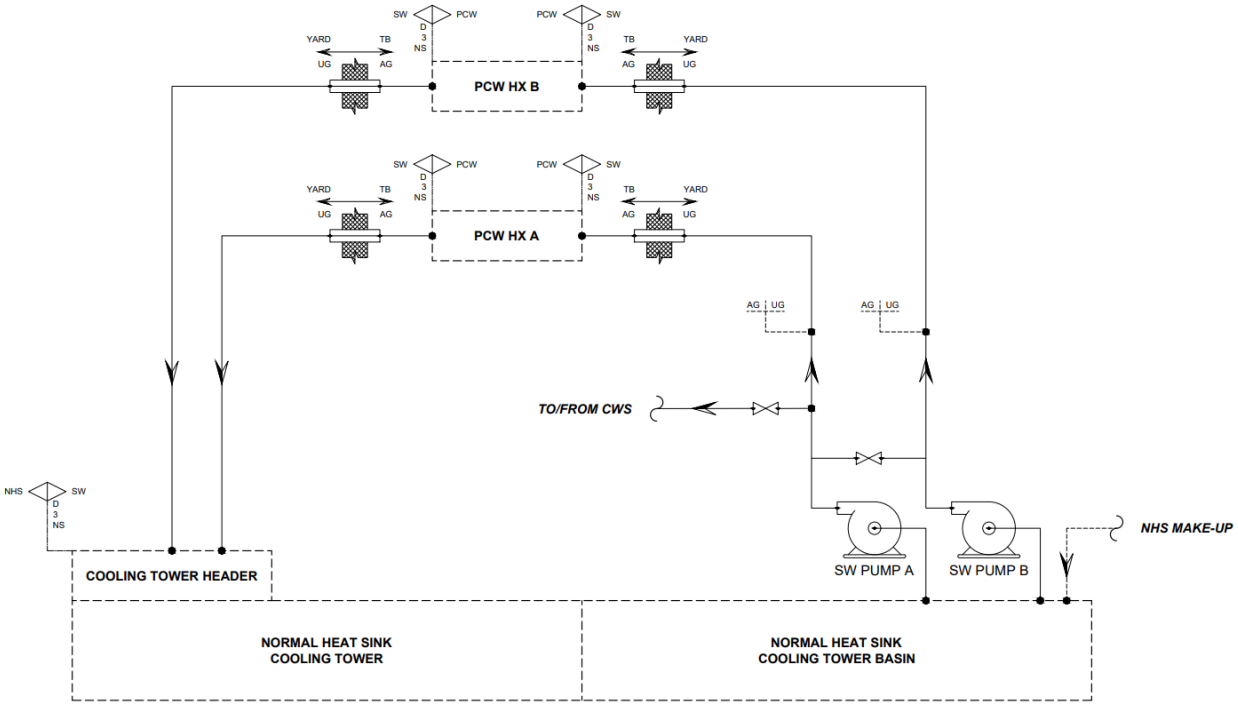


Figure 9.2-1: Service Water Subsystem Simplified Diagram

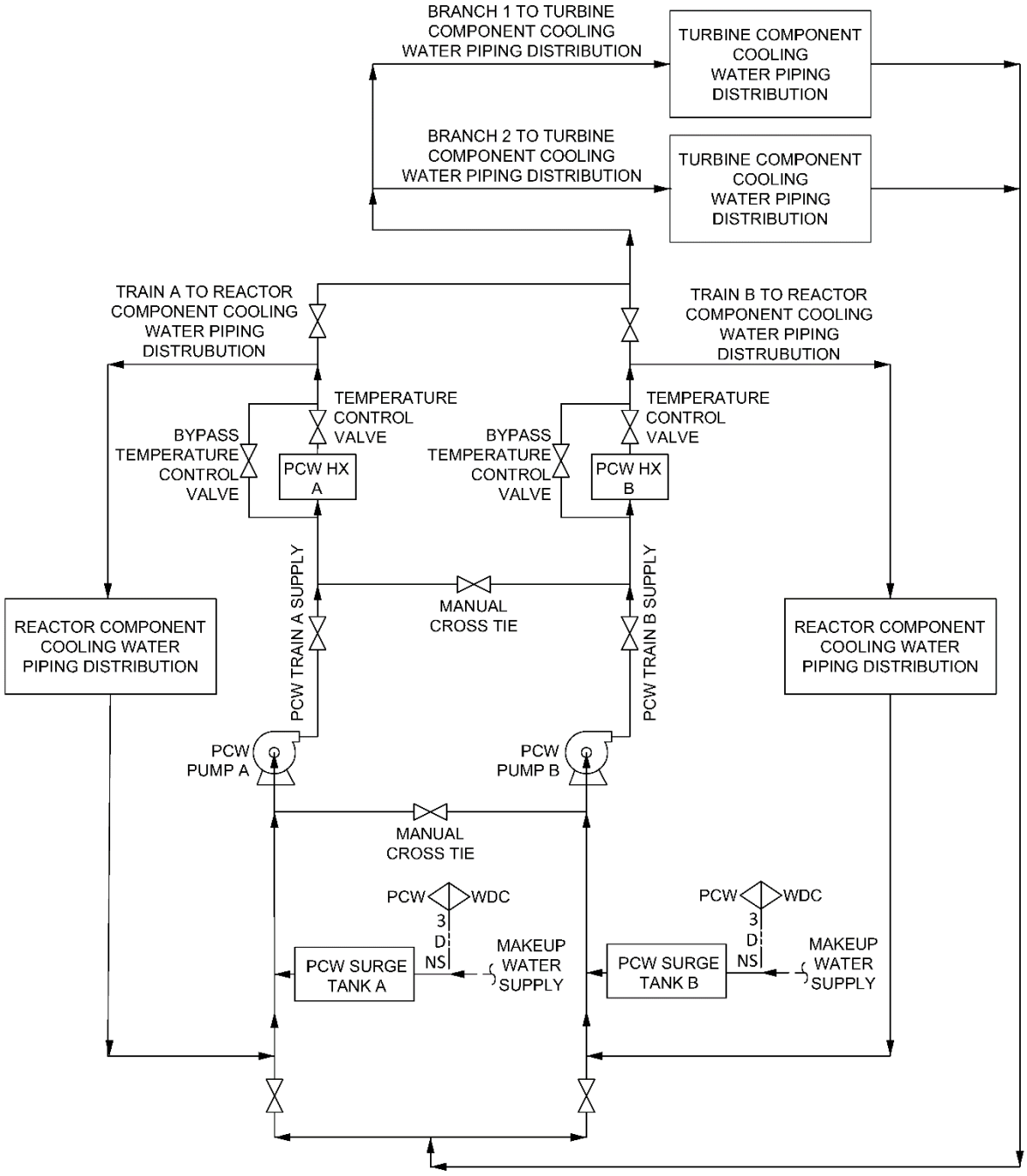


Figure 9.2-2: Plant Cooling Water System Simplified Diagram

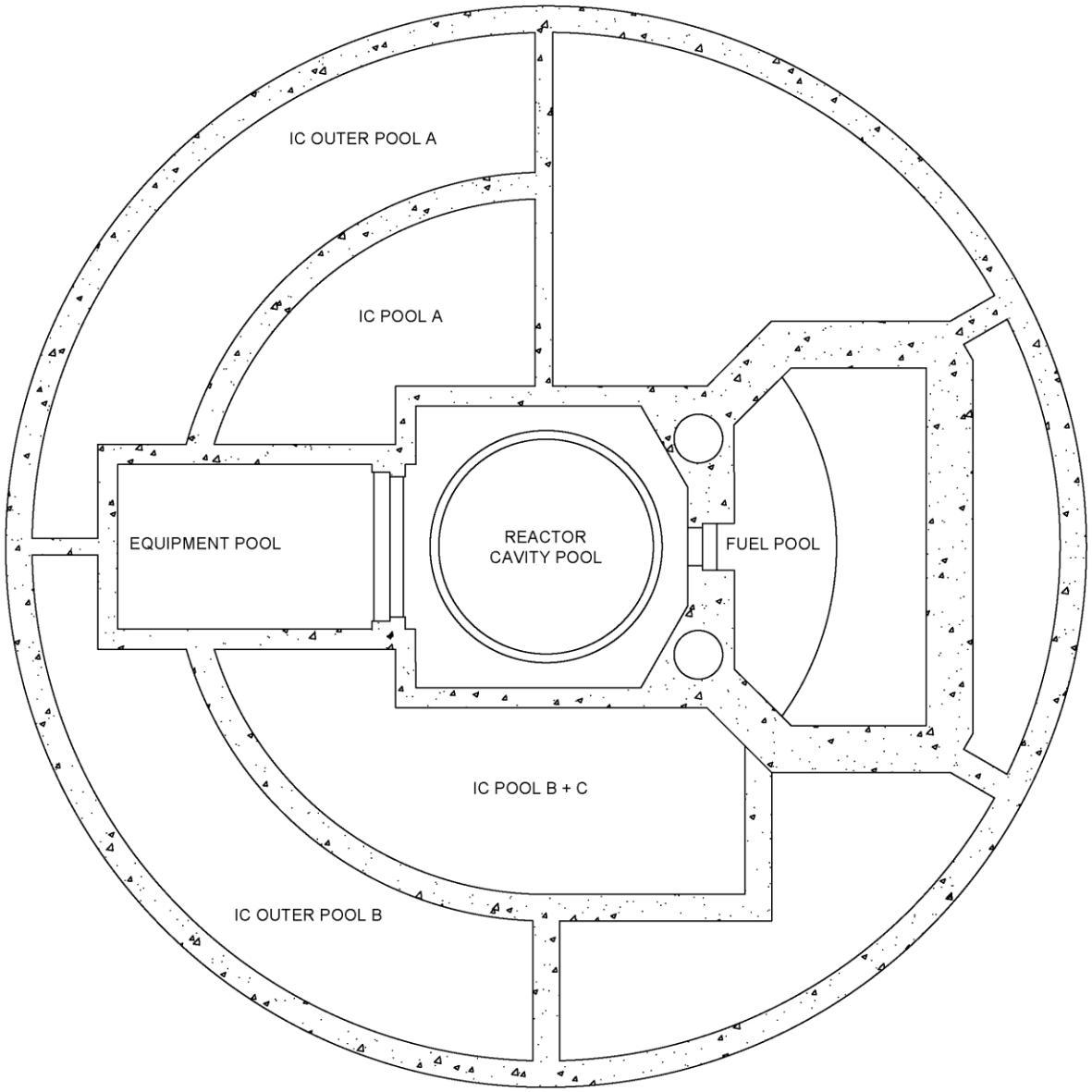


Figure 9.2-3: Ultimate Heat Sink Pools Simplified Diagram

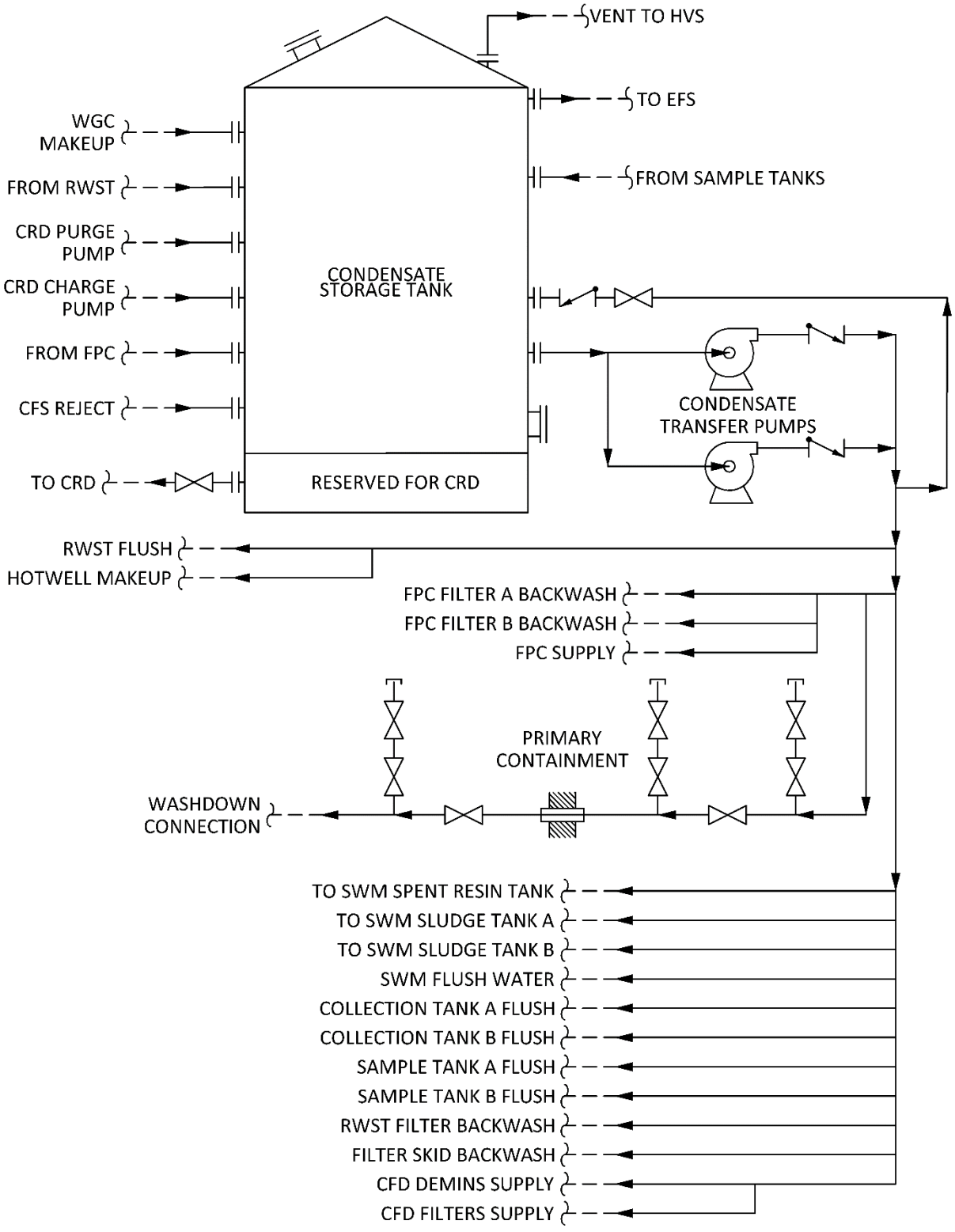


Figure 9.2-4: Condensate Storage and Transfer Subsystem Simplified Diagram

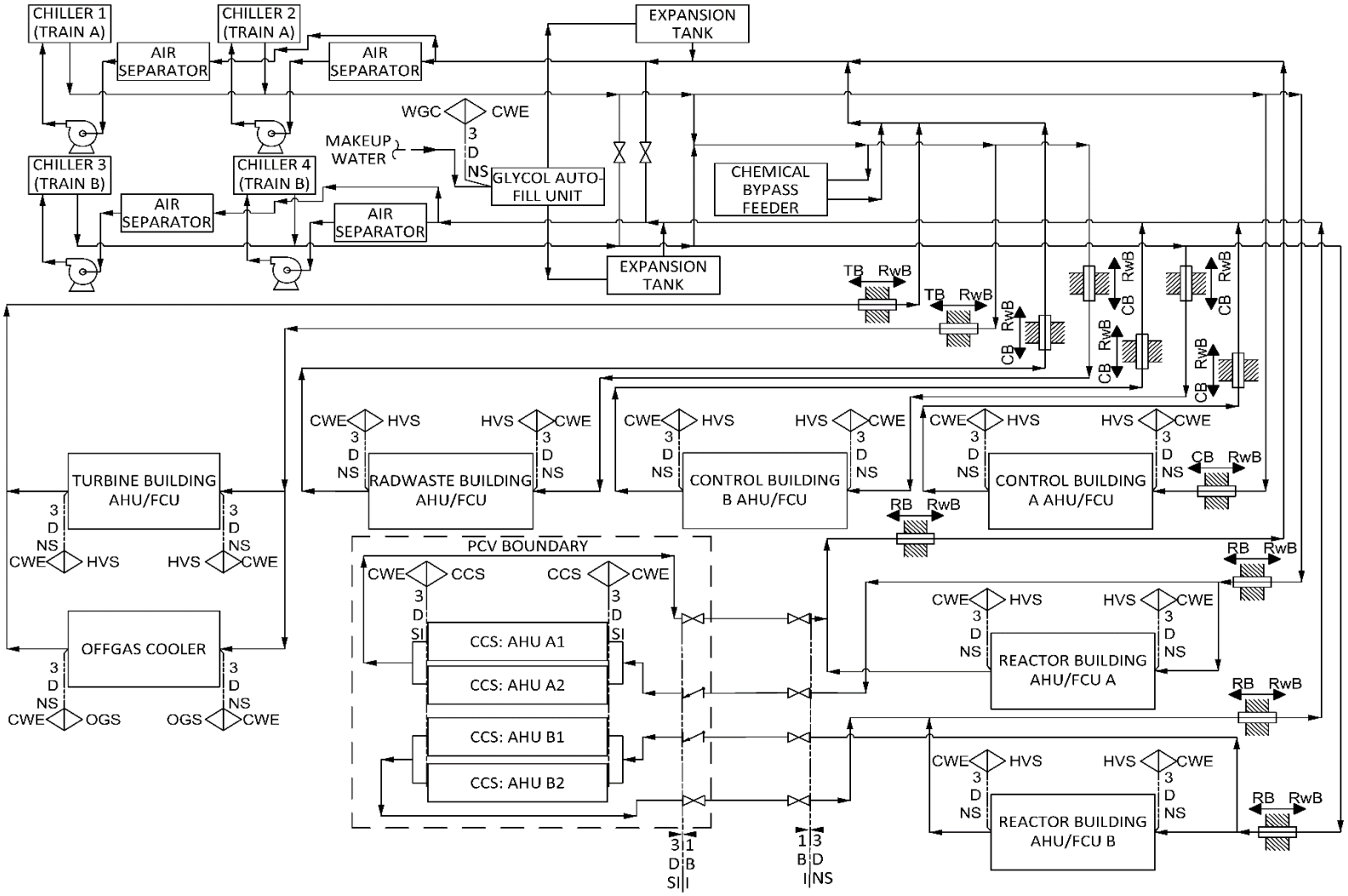


Figure 9.2-5: Chilled Water Equipment Simplified Diagram

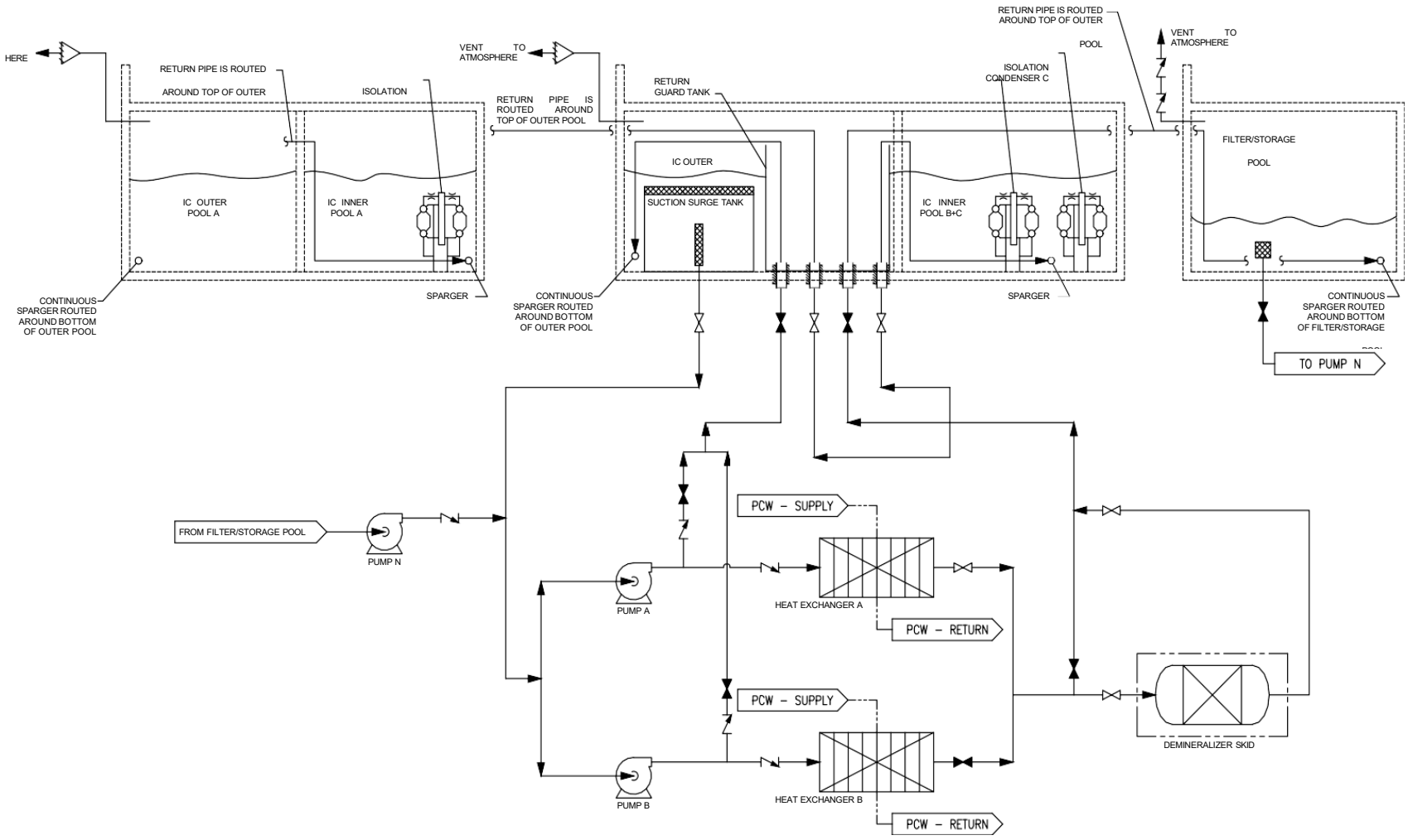


Figure 9.2-6: Isolation Condenser Pools Cooling and Cleanup System Simplified Diagram

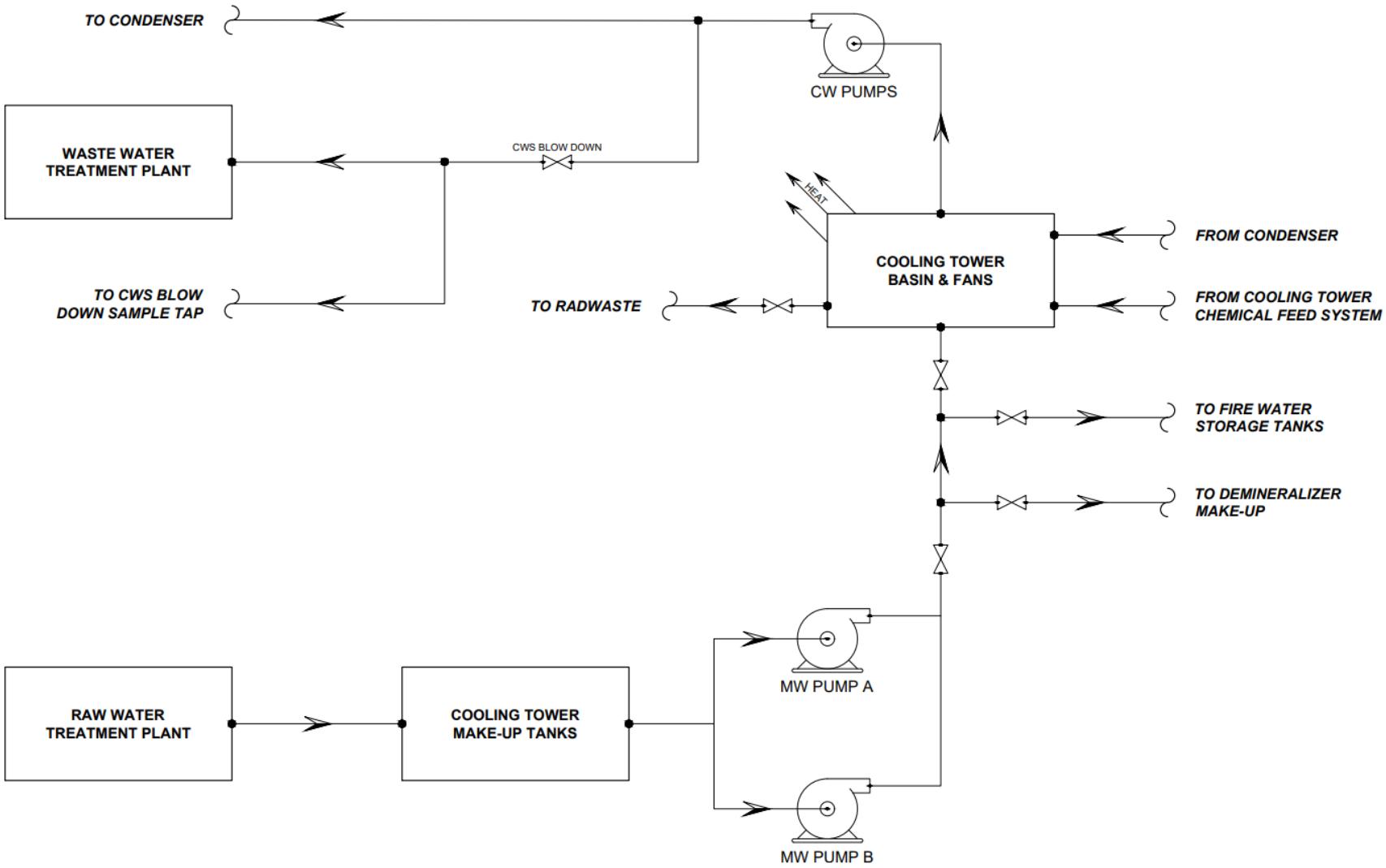


Figure 9.2-7: Normal Heat Sink System Simplified Diagram

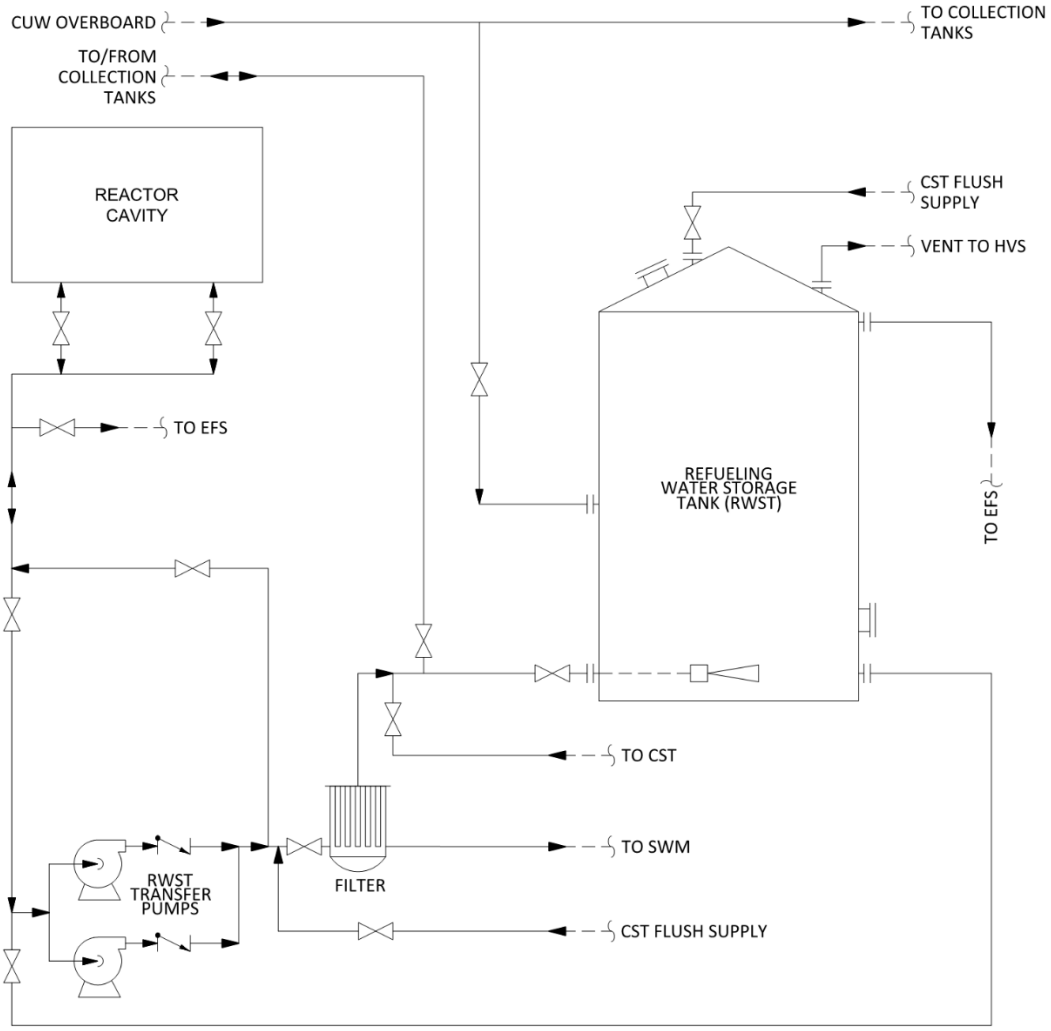


Figure 9.2-8: Refueling Water Storage and Cleanup Subsystem Simplified Diagram

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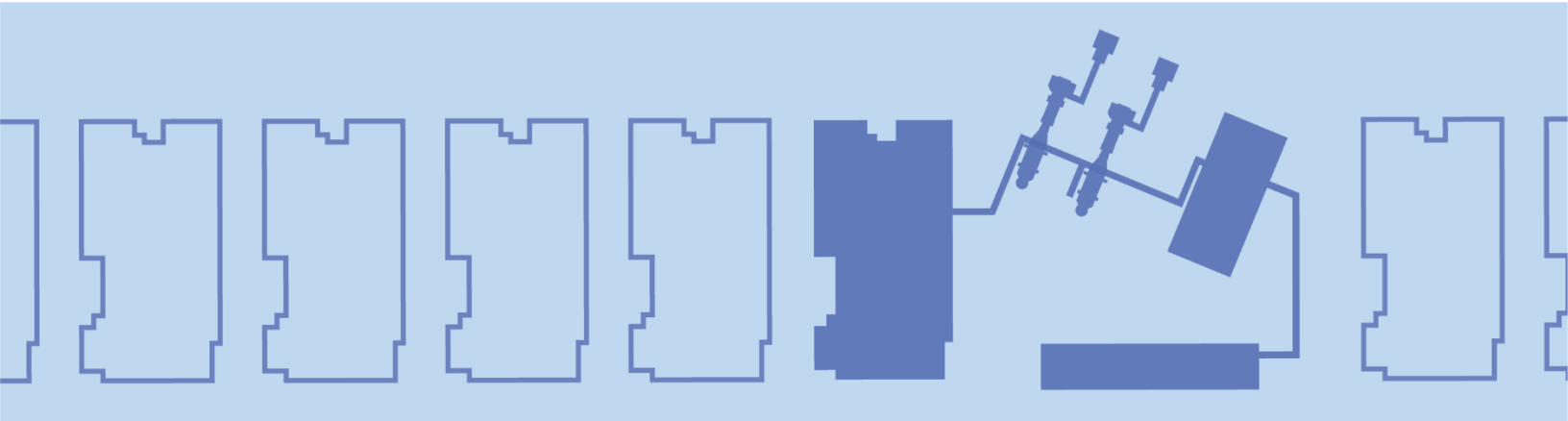
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Non-Proprietary Information

Chapter 9
Section 9.3 – Process Auxiliaries



ACRONYM LIST

Acronym	Explanation
ALARA	As Low As Reasonably Achievable
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
BIS	Boron Injection System
BECPV	Blue Energy Center Port of Victoria
BPVC	Boiler and Pressure Vessel Code
BWR	Boiling Water Reactor
CCS	Containment Cooling System
CFR	Code of Federal Regulations
CFS	Condensate and Feedwater Heating System
CIS	Containment Inerting System
CIV	Containment Isolation Valve
CRD	Control Rod Drive
CUW	Reactor Water Cleanup System
D-in-D	Defense-in-Depth
DBA	Design Basis Accident
DCIS	Distributed Control and Information System
DEC	Design Extension Conditions
EFS	Equipment and Floor Drain System
EPRI	Electric Power Research Institute
FPC	Fuel Pool Cooling and Cleanup System
FSAR	Final Safety Analysis Report
GDC	General Design Criteria
HCW	High Conductivity Waste
HVS	Heating, Ventilation, and Cooling System
HWC	Hydrogen Water Chemistry
I&C	Instrumentation and Control
IA	Instrument Air
ICC	Isolation Condenser Pools Cooling and Cleanup System
ICS	Isolation Condenser System
IGSCC	Intergranular Stress Corrosion Cracking

Acronym	Explanation
ISA	International Society of Automation
ITP	Initial Test Program
LWM	Liquid Waste Management System
MCR	Main Control Room
OGS	Offgas System
OLNC	On-Line NobleChem™
PCW	Plant Cooling Water System
PDC	Principal Design Criteria
PPS	Plant Pneumatics System
PREMS	Process Radiation and Environmental Monitoring System
PS	Process Sampling Subsystem
PWR	Pressurized Water Reactor
RB	Reactor Building
RCPB	Reactor Coolant Pressure Boundary
RG	Regulatory Guide
RWB	Radwaste Building
SA	Service Air
SBO	Station Blackout
SC3	Safety Class 3
SCCV	Steel-Plate Composite Containment Vessel
SCR	Secondary Control Room
SDC	Shutdown Cooling System
SDG	Standby Diesel Generator
SSCs	Structures, Systems, and Components
SSE	Safe Shutdown Earthquake
SWM	Solid Waste Management System
TB	Turbine Building
USNRC	United States Nuclear Regulatory Commission

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9.3 Process Auxiliaries

9.3.1 Plant Pneumatics Systems

During normal operation, the Plant Pneumatics System (PPS) provides a continuous supply of compressed air for various maintenance activities and other services. The PPS also provides dried and filtered compressed air to valve actuators, instrument control functions, and general instrumentation and valve services. The PPS includes piping connecting the Containment Inerting System (CIS) to pneumatic valves and other compressed gas users inside the Containment. The PPS has Containment Isolation Valves (CIV), which are not described in this section. The CIVs, their safety functions, and testing are described in Subsection 6.2.4.

9.3.1.1 Design Bases

Safety Category 1 Functions

None.

Safety Category 2 Functions

None.

Safety Category 3 Functions

- Provide a supply of service air or instrument air to other systems for tank sparging, filter/demineralizer backwashing, air-operated valves, and other services.
- Provide distribution of pressurized nitrogen to other systems in the Containment, where the atmosphere is inerted during operation.

Safety Category N Functions

- Provide a supply of service air or instrument air to other systems for tank sparging, filter/demineralizer backwashing, air-operated valves, and other services.

The Safety Classifications, Quality Groups, and Seismic Categories of the PPS Structures, Systems, and Components (SSCs) are listed in Table 3A-1.

9.3.1.2 System Description

The configuration of the PPS is shown in Figure 9.3-1. The piping penetrating the Containment, and associated supports are designed to Seismic Category I. The balance of the PPS is designed as Non-Seismic, except where a component could adversely interact with Seismic Category I SSCs during a Safe Shutdown Earthquake (SSE).

One 100% capacity air compressor operates to supply both Service Air (SA) receivers, one 100% capacity dryer, both Instrument Air (IA) receivers, and the system distribution piping. The PPS contains two 100% compressors in parallel with one selected as lead and the other as the standby. The standby compressor automatically starts on trip of the lead compressor, or if the SA header pressure drops below a predetermined setpoint.

Service Air Receivers

There are two SA receivers cross-tied in parallel, which provide surge volume and pressure spike damping for the SA supply. The receivers meet the requirements of ASME BPVC, Section VIII, Division 1, “Rules for Construction of Pressure Vessels,” (Reference 9.3-1). Both SA receivers are connected to a common header that distributes air to the Radwaste Building (RWB), Turbine Building (TB), Reactor Building (RB), Containment, and IA portion of the PPS via the air dryers. A pressure-relief safety valve is installed on each receiver to protect against system over

pressurization. The SA receivers also serve as a backup to the IA subsystem upon a loss of both compressors.

Air Dryers

The PPS consists of two 100% capacity dryer trains in parallel, one normally operating and the other in standby; however, the two trains are cross-tied together for operational flexibility. The standby air dryer auto starts on low discharge pressure or high outlet dewpoint of the lead dryer, or when both compressors are running. High differential pressure across a dryer filter is provided with an alarm.

The dryers feed two IA receivers, cross-tied in parallel which provide surge volume and pressure spike dampening for the IA supply. Both IA receivers are connected to a common header that distributes IA to the RWB, TB, RB, and the Containment.

Instrument Air

The IA subsystem of the PPS also contains piping and Containment penetrations that connect both the PPS and CIS to compressed gas users inside Containment. During normal operation, the air supply is isolated from the Containment via double block and bleed valve configuration, so that nitrogen from the CIS can be supplied to nitrogen users inside the Containment using the PPS containment piping penetrations. The pressure-relief safety valve is installed on each receiver to protect against system over pressurization. The block and bleed valves on the air supply prevent air from entering the Containment during normal operation and posing an explosion and fire risk.

During plant shutdown, the nitrogen supply is isolated from the Containment so that in the case of a leak in the PPS supply piping to the Containment, portions of the Containment do not turn into a confined space or pose an asphyxiation risk to plant personnel.

The PPS IA subsystem is designed and maintained to the standards of American National Standards Institute (ANSI)/International Society of Automation (ISA) 7.0.01, “Quality Standard for Instrument Air,” (Reference 9.3-2) which supersedes ANSI/ISA-S7.3-R1981. Air dryers and filter trains are used to maintain the dewpoint, remove oil droplets, and remove rust scale, dust, and other solid objects as well.

Breathing Air

The service air subsystem also connects to a separate utility containment penetration to provide service air to the service boxes inside Containment. These service boxes are used to supply air to portable breathing air filtration systems to provide breathing air hookups to personnel entering the Containment when the atmosphere is not yet fully habitable.

Configuration Information

Two outdoor air intakes, that are mounted outside the TB, supply clean air via air ducts to the air compressors. The intakes are located a minimum of 50 feet from any exhaust outlets or potential source of release of contaminants and are no less than 10 feet above grade. Compressed air and breathing air connections are provided in close proximity to radioactive equipment that is maintained in place.

Air service taps are connected to the top or sides of the main header wherever possible to avoid carryover of the moisture. Condensate drain line valves are provided at the bottom of the compressor coolers/moisture separators, air receiver tanks, dryers, and low point on the dryer bypass line, and other locations to permit removal of moisture. Condensate from the various points throughout the PPS are drained to the Equipment and Floor Drain System (EFS). Refer to Subsection 9.3.3.

Automatic or remotely actuated valves in the PPS fail in the position most likely to keep the system operational and controllable, with the exception of the PPS CIVs, which fail closed on loss of air or signal. The PPS control functions are designed to keep the system operational and supplying air to the IA system as a priority.

The piping, other than that required for containment isolation, is designed to meet the requirements of ASME B31.1, “Power Piping,” (Reference 9.3-3).

The PPS is designed to be operational following power loss to equipment. When a power loss occurs, the operating compressor and dryer both trip and restart when power is received from the Standby Diesel Generator (SDGs).

9.3.1.3 Safety Evaluation

The PPS complies with the following General Design Criteria (GDC), which have plant-wide applicability, as discussed in Section 3.1:

GDC 1 – Quality standards and records

GDC 2 – Design bases for protection against natural phenomena

GDC 5 – Sharing of structures, systems, and components

Additional information GDC compliance is found in Section 3.1.

The PPS is designed to comply with the following regulation:

10 CFR 50.63, as it relates to loss of all alternating current power. The design of the PPS considered 10 CFR 50.63; however, the PPS is not required to achieve safe shutdown. The PPS does not support safety category functions pertaining to maintaining the ability to actuate or control equipment necessary for core cooling and decay heat removal or maintaining containment integrity following a Station Blackout (SBO). Refer to Section 8.4 for information pertaining to the SBO coping analysis.

9.3.1.4 Inspection and Testing

Isolation valves permit maintenance activities of major PPS components without having to shut down the PPS. Crossties promote flexibility for maintenance, testing and operation. System operational tests for components normally closed to airflow are performed periodically to ensure system capability and integrity. Filters are periodically inspected for cleanliness, and the desiccant in the air dryers is periodically sampled to verify its useful life. Periodic testing of air quality and system functionality is performed to ensure compliance with ANSI/ISA 7.0.01 (Reference 9.3-2). PPS equipment is arranged to provide adequate pull space and clearance for personnel and testing equipment access to facilitate testing, repair, and maintenance.

9.3.1.5 Instrumentation

PPS is automatically and manually controlled by the Distributed Control and Information System (DCIS). Refer to Section 7.3 for description of the DCIS.

Local control panels for the compressors and dryers are used to operate the PPS during testing, calibration, and maintenance. Pressure, temperature, compressor status, and dryer status are monitored to verify system operation.

9.3.2 Process Sampling System

The Process Sampling Subsystem (PS) is part of the Process Radiation and Environmental Monitoring System (PREMS). The PREMS is comprised of several subsystems which monitor radiation, contamination, and environmental conditions throughout the plant. In addition to the PS, PREMS includes the following subsystems:

- Process Radiation Monitoring Subsystem, see Section 11.5
- Area Radiation Monitoring Subsystem, see Subsection 12.3.4.1
- Containment Monitoring Subsystem, see Section 6.8

9.3.2.1 Design Bases

Safety Category 1 Functions

None.

Safety Category 2 Functions

None.

Safety Category 3 Functions

None. Although PS does not perform Safety Category 3 functions, PS equipment suspected to contain radioactive materials is classified Safety Class 3 (SC3) to support confinement of radioactivity.

Safety Category N Functions

- Collect representative liquid and gaseous samples for analysis
- Provide analytical information for monitoring plant and equipment performance

The Safety Classifications, Quality Groups, and Seismic Categories of the PS SSCs are listed in Table 3A-1.

9.3.2.2 System Description

The PS collects representative liquid and gaseous samples for analysis and provides the analytical information required for monitoring plant and equipment performance.

For processes requiring continuous or frequent monitoring, sample tubing is routed from the process stream to automated sample panels. Sample conditioning equipment is provided within each panel for pressure, temperature, and flow adjustment. Online monitoring equipment is used to the greatest extent practicable, and provided for measurements, such as conductivity, dissolved oxygen, dissolved hydrogen, total organic carbons, and silica. Grab sample taps are provided at the panels for infrequent/diagnostic sampling, online analyzer checks, and instances in which online analyzers are unavailable. Test connections are provided for equipment calibration or installation of temporary analyzers.

Most processes requiring infrequent or diagnostic-only sampling are collected via local grab sample stations. If necessary, sample conditioning equipment is provided with each station for pressure, temperature, and flow adjustment.

PS equipment is installed throughout the plant. The automated sample panels are located as close as practical to the chemistry lab for personnel efficiency. Local grab sample stations are located throughout the plant as close as practicable to the process stream.

The PS provides online chemistry monitoring for the following:

- Control Rod Drive (CRD) system purge water
- Shutdown Cooling System (SDC)
- Reactor Water Cleanup System (CUW)
- Fuel Pool Cooling and Cleanup System (FPC)
- Liquid Waste Management System (LWM)
- Condensate and Feedwater Heating System (CFS)
- Condensate Filters and Demineralizers System (CFD)
- Main Condenser and Auxiliaries (MCA)

These samples are routed to an automated sample panel.

The PS provides local grab sampling equipment for the following:

- Isolation Condenser Pools Cooling and Cleanup System (ICC)
- Solid Waste Management System (SWM)
- CFS
- Plant Cooling Water System (PCW)
- EFS

Refer to Chapter 7 for design of related Instrumentation and Control (I&C) systems.

9.3.2.3 Safety Evaluation

Regulatory Requirements

The PS complies with the following GDC, which have plant-wide applicability, as discussed in Section 3.1:

GDC 1 – Quality standards and records

GDC 2 – Design bases for protection against natural phenomena

The following GDC and Principal Design Criteria (PDC) have aspects specific to the PS:

GDC 13 – Instrumentation and Control, as it relates to monitoring variables that can affect the fission process, the integrity of the reactor core, and the Reactor Coolant Pressure Boundary (RCPB).

The design of PS considers GDC 13 by monitoring variables that can affect the fission process, the integrity of the reactor core, and the RCPB. The PS provides samples from auxiliary systems containing reactor coolant during operation. Satisfying the requirements of GDC 13 is discussed in Section 7.1.

GDC 14 – Reactor coolant pressure boundary, as it relates to assuring the integrity of the RCPB.

The PS monitors parameters that are indicative of RCPB leakage. Containment fission product monitoring samples are physically collected and analyzed for indication of reactor coolant leakage within the Containment.

PDC 26 – Reactivity control system redundancy and capability, as it relates to the applicability of sampling boron concentration with regards to having two diverse means of controlling reactivity.

PDC 26 has no applicability with regards to the PS, as the Boron Injection System (BIS) does not interface with PS. Refer to Subsection 9.3.10 for a description of the BIS.

GDC 60 – Control of releases of radioactive materials to the environment, as it relates to the capability of the PS to control the release of radioactive materials to the environment.

The PS collects representative liquid and gaseous samples for analysis and provides the analytical information required for monitoring plant and equipment performance. As a result, the PS may contain radioactive material. Equipment suspected to contain radioactive materials is classified SC3 to support confinement of radioactivity. Meeting the requirements of GDC 60, as it relates to the PS, enhances safety by providing a means in the design to control the release of radioactive material to the environment. Application of GDC 60 provides reasonable assurance that the PS is designed, constructed, installed, and operated on a level commensurate with the need to protect the health and safety of the public and plant operating personnel.

GDC 63 – Monitoring fuel and waste storage, as it relates to detecting conditions that may result in excessive radiation levels in the fuel storage and radioactive waste systems.

The PS design considers GDC 63 by monitoring conditions that may result in excessive radiation levels in the fuel pool and radioactive waste systems. The PS also includes sampling capability for the LWM as discussed in Section 11.2 and SWM in Section 11.4. This enables analyses to be performed to detect conditions related to fuel storage and radioactive waste systems that could result in excessive radiation levels and excessive personnel exposure.

GDC 64 – Monitoring radioactivity releases, as it relates to monitoring the containment atmosphere and plant environs for radioactivity.

The PS provides online monitoring and grab sampling. Therefore, GDC 64 for monitoring radioactive releases is not applicable to this subsystem of PREMS.

Additional information on GDC compliance is found in Section 3.1. The PS is designed to consider the following regulation:

10 CFR 20.1101(b), as it relates to the use of engineering controls in maintaining doses “As Low As Reasonably Achievable” (ALARA).

Regulatory Guidance

Conforming to Regulatory Guide (RG) 1.21, “Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste,” the PS monitors process and effluent streams for potentially hazardous contamination such as radioactivity, noble gases, air particulates, and halogens. The PS design features, and configuration, consider ALARA program goals and objectives with regard to minimizing dose and contamination. The objective of the ALARA requirements is to ensure that reasonable efforts associated with the planning, design, and operation of the PS maintains exposures to radiation as far below the limits of 10 CFR Part 20 as is reasonably achievable. Station layout and design features are developed in support of reducing the potential doses to personnel who operate, service, or inspect the PS. Refer to Chapter 12 for additional information pertaining to ALARA.

9.3.2.4 Inspection and Testing

Sampling racks and electronic modules are serviced and maintained in accordance with industry standards, or in accordance with the operational instructions to ensure reliable operation. Periodic testing or calibration checks are performed as part of the maintenance plan. If work is performed that affects the calibration of the instrument, a re-calibration is performed following the maintenance operation.

9.3.2.5 Instrumentation and Control

Refer to Chapter 7 for description of the BWRX-300 I&C systems.

The system design accommodates access to sensors, instruments, and panels for maintenance. Instrument modules are designed to facilitate calibration checks and troubleshooting.

Instrument data is displayed at the panel. Sampling equipment either operates continuously or in grab sample mode.

9.3.3 Equipment and Floor Drain System

The EFS is designed to collect radioactive or potentially radioactive liquid wastes generated in the Power Block and transfer these wastes to an appropriate disposal or collection system. The EFS has CIVs, which are not described in this section. The CIVs, their safety functions, and testing are described in Subsection 6.2.4.

9.3.3.1 Design Bases

Safety Category 1 Functions

None.

Safety Category 2 Functions

None.

Safety Category 3 Functions

- Provide equipment to drain, collect, and transport potentially radioactive liquid waste to a storage location

Safety Category N Functions

- Provide equipment to drain, collect, and transport oily waste and clean waste to an appropriate storage location

The Safety Classifications, Quality Groups, and Seismic Categories of the EFS SSCs are listed in Table 3A-1.

9.3.3.2 System Description

The EFS consists of the drainpipe systems, collection sumps, sump pumps, discharge piping, and instrumentation to facilitate the collection, removal, and measurement of liquid waste in the plant. During a Design Basis Accident (DBA) and assuming the unavailability of normal power, EFS does not operate, nor is it required to operate.

The EFS is comprised of the following four types of sump collection systems which are described below:

- General EFS
- Pressurized sump to the general floor drain system
- Oily waste sumps
- Dedicated potentially clean drain systems

Figure 9.3-2 through Figure 9.3-4 are simplified diagrams related to the EFS.

General Equipment and Floor Drain System

The EFS continuously collects the liquid waste that is generated throughout the plant. Liquid waste collected in the radioactive drains is treated as High Conductivity Waste (HCW). Liquid waste includes floor drain, equipment drain, and process drain wastes collected throughout radiologically controlled areas. This includes wastes from equipment and personnel decontamination, eyewash stations, and fire protection water. Equipment drain hubs are not utilized, and equipment or process drains are either hard piped or temporarily piped/hosed to a floor drain fixture. Liquid waste is collected by floor drains and drain headers and routed by gravity drainage piping to one of the various sumps, where it is stored and eventually pumped to the LWM as discussed in Section 11.2 for processing.

Pressurized Sump to the General Floor Drain System

General sumps are included in the basement floor of the RB. One of the sumps is a pressurized sump vessel designed to meet the requirements of ASME BPVC Section VIII, Division 1 (Reference 9.3-1). The piping upstream of the pressurized sump penetrates the Containment, necessitating double isolation valves for containment isolation events. During modes of normal operation, any leakage that collects inside the Containment is drained to the pressurized sump vessel. This sump operates the same way as the general equipment and floor drain sumps and is equipped with a pressure relief valve, flush water line, redundant sump pumps, and level transmitters. The vessel is located inside a separate sump. Any leakage from the vessel or pumps is confined within this secondary sump. Due to its interface with a dry, inerted containment, the vessel is leak-tight. Redundant level transmitters accurately measure and trend containment leakage.

The floor drain line in the Containment connects directly to the containment atmosphere and penetrates the Containment. Refer to Subsection 6.2.4 for discussion of related containment isolation functions.

Oily Waste Sumps

Dedicated oily waste collection sumps are provided at the base of the Turbine Electro-Hydraulic Control unit, lube oil skid, and SDG skids to collect and identify leakage. A temporary pump and hose/piping is required to pump oily waste from the sump pit to a collection barrel for transport.

Dedicated Potentially Clean Subsystems

There are two interfacing systems with EFS, the PPS as discussed in Subsection 9.3.1, and PCW in Subsection 9.2.2, that have a dedicated drainage subsystem because the water is potentially clean. Water from these interfacing systems can be analyzed to determine the chemical and radiological contents prior to being pumped to its appropriate destination.

The PCW has a dedicated sump because the PCW water is expected to be non-radioactive. A temporary sump pump and piping/hose is required to empty the sump contents. Prior to emptying the sump, a grab sample is analyzed to determine the chemical and radiological contents. If the contents meet the demineralized makeup water storage tank quality criteria, then it can be pumped back to the PCW makeup water storage tank. If not, the contents are pumped to LWM via a general EFS sump floor drain.

9.3.3.3 Safety Evaluation

The EFS complies with the following GDC, which have plant-wide applicability, as discussed in Section 3.1:

GDC 2 – Design bases for protection against natural phenomena

GDC 4 – Environmental and dynamic effects design bases

The following GDC have aspects specific to the EFS, and compliance is addressed as follows:

GDC 60 – Control of releases of radioactive materials to the environment, as it relates to the capability of the EFS to control the release of radioactive materials to the environment.

Controlled collection and transfer to appropriate radioactive or nonradioactive treatment systems assures compliance with GDC 60. EFS continuously collects the liquid waste that is generated throughout the plant. Liquid waste collected in radioactive drains is treated as HCW. Liquid wastes are collected by floor drains and drain headers, routed by gravity drainage piping to one of the various sumps, where it is stored and eventually pumped to LWM as discussed in Section 11.2 for processing. The pipes carrying fluid to radioactive drains and clean drains are separate and independent.

The collection sumps are vented via connections to the Heating, Ventilation, and Cooling System (HVS) in the Power Block. Ventilation is provided as a means to preclude release of harmful gases into the plant.

Additional information on GDC compliance is found in Section 3.1.

9.3.3.4 Inspection and Testing

The equipment and components of the EFS are designed for inspection and maintenance during all modes of plant operation, except the pressurized sump. The pressurized sump is only available for maintenance and inspection during offline activities due to its interface with the Containment.

Liquid waste testing is utilized in order to test for radioactive or chemical contamination amongst the potentially clean collection samples. Grab sample points are provided on sump recirculation lines to sample sump contents when necessary.

9.3.3.5 Instrumentation and Control

Refer to Subsection 7.3.4 for description of balance of plant controllers related to EFS monitoring and control.

The EFS includes instrumentation to control and monitor (indicate and alarm) system operation, including reactor coolant leakage monitoring in accordance with RG 1.45, “Guidance on Monitoring and Responding to Reactor Coolant System Leakage.” The EFS instrumentation detects leaks or breaks by monitoring containment drain tank level, RB sump level, and TB sump level. An alarm is annunciated in the Main Control Room (MCR) and Secondary Control Room (SCR) upon high rate of level increase or high pump run times.

Pressure transmitters are provided in the discharge piping of each sump pump which displays the pressure in the MCR. The pressure readings enable the operator to monitor actual pump performance, and alarm on high discharge pressure.

9.3.4 Chemical and Volume Control System Pressurized Water Reactor (Including Boron Recovery System)

Not applicable to BWRX-300.

9.3.5 Standby Liquid Control System

As discussed in Section 4.0.1 of NEDC-33912P-A, “BWRX-300 Reactivity Control,” (Reference 9.3-4) a Standby Liquid Control System as a backup means for reactor shutdown is not required. The frequency of an Anticipated Transient Without Scram event is discussed in Subsection 15.6.1.4.5 that fulfills NEDC-33912P-A Limitation and Condition 5.1.

9.3.6 Containment Inerting System

The CIS is designed to establish and maintain an inert nitrogen atmosphere within the Containment. The principal objective of the CIS is to preclude the development of a combustible atmosphere by maintaining an oxygen deficient atmosphere inside the Containment. The CIS has CIVs, which are not described in this section. The CIVs, their safety functions, and testing are described in Subsection 6.2.4.

9.3.6.1 Design Bases

Safety Category 1 Functions

None.

Safety Category 2 Functions

None.

Safety Category 3 Functions

- Establish and maintain an inert atmosphere within the Containment for prevention of hydrogen combustion during normal plant operating modes, except plant shutdown
- Provide a supply of nitrogen to the PPS for the pneumatic users in the Containment
- Protect against containment over pressurization

Safety Category N Functions

- Maintain a positive pressure within the Containment to prevent the infiltration of non-inert air during all normal plant operating modes, except plant shutdown
- Establish a breathable de-inert atmosphere within the Containment during plant shutdown for refueling and maintenance
- Provide the capability to isolate the containment overpressure vent path when signaled to do so by I&C systems

The Safety Classifications, Quality Groups, and Seismic Categories of the CIS SSCs are listed in Table 3A-1.

9.3.6.2 System Description

The CIS is shown in Figure 9.3-5. The CIS is designed to establish and maintain an inert atmosphere (nitrogen) within the Containment. An inert atmosphere is maintained in all operating modes, except during plant shutdown for refueling, equipment maintenance, and during limited periods of time to permit access for inspection at low reactor power. The CIS can de-inert the Containment for safe personnel access without the need of a breathing apparatus. The CIS also provides a pathway for a fresh air supply into the Containment during outages to maintain a breathable atmosphere during outage work.

An inerted atmosphere with low oxygen concentration reduces the potential for a combustible environment inside the Containment following an accident. The oxygen concentration satisfies the guidance of 10 CFR 50.44, which defines an inerted atmosphere as less than 4% by volume.

The CIS consists of containment supply and containment exhaust.

Containment Supply Flow Path

The containment supply flow path is used to inject nitrogen into the Containment for inerting (large nitrogen flow) and makeup (small nitrogen flow), to inject breathable air into the Containment for

de-inerting, and to supply nitrogen to the PPS (Subsection 9.3.1) to support actuation of pneumatic devices located inside the Containment. The nitrogen supply consists of a pressurized liquid nitrogen storage tank, vaporizer to convert the liquid nitrogen to a gaseous phase, and electrical heater located in the yard. The containment supply flow path terminates in lower containment.

If the containment supply pressure control valve fails, a manual bypass valve can be opened to supply nitrogen to the Containment. A pressure relief valve is positioned downstream of the pressure control valve to protect the downstream piping and provide for pressure control during manual bypass of the pressure control valve. The manual bypass function allows for continuous operation of CIS while the pressure control valve is out of service.

Containment Exhaust Flow Path

The containment exhaust flow path is used to discharge the containment atmosphere during the inerting, makeup, and de-inerting process. The containment exhaust flow path consists of piping, valves, instruments, controls. All of the components are located within the RB. The exhaust flow path starts in upper containment on the opposite side of the Containment from the supply injection point and connects to the RB HVS as discussed in Subsection 9.4.6, before discharging to the Plant Vent Stack.

Containment Overpressure Vent Flow Path

The containment overpressure vent flow path is used in case of a severe accident where containment failure by overpressure is threatened. The containment overpressure vent flow path consists of a rupture disc, a remote-actuated bypass valve, isolation valve, check valve, sparger, and associated piping. The overpressure vent flow path penetrates containment and terminates in the filter/storage pool. By relieving pressure into the filter/storage pool, water scrubbing is expected to occur, thereby reducing the contaminants. Additionally, the filter/storage pool is located inside the RB with its own dedicated vent to the environment.

The filter/storage pool airspace is segregated from the rest of the Reactor Building and vented to the atmosphere through a series of check valves to prevent atmospheric oxygen from flowing into the pool airspace following initial venting. Venting containment to the filter/storage pool prevents contamination of the Reactor Building with fission products and combustible levels of hydrogen. The atmospheric vent associated with the filter/storage pool provides a designated release path to the environment.

9.3.6.2.1 Component Description

Liquid Nitrogen Storage Tank

The liquid nitrogen storage tank provides nitrogen supply to support CIS operations. The tank includes level and pressure indicators. Pressure is maintained automatically in the tank during nitrogen discharge. The tank is fabricated of material suitable for cryogenic service in order to withstand low liquid nitrogen temperatures. The tank is equipped with quick disconnect/connect or equivalent type of connection to support refilling.

Vaporizer

The vaporizer is used to convert the nitrogen from the liquid nitrogen storage tank from a liquid to a gaseous phase. The vaporizer is fabricated of material suitable for cryogenic services.

Heater

A trim heater operates in automatic mode to increase the temperature of the nitrogen. The heater is fabricated of material suitable for cryogenic services.

Rupture Disc

The rupture disc is part of the containment overpressure vent flow path and provides passive overpressurization protection for the Containment. The rupture disc is an ASME BPVC, Section III, Class 2 "Component" (Reference 9.3-5).

A bypass valve around the containment overpressure vent rupture disk can be used as an alternate method of venting. The bypass valve can be either remotely operated from the control room or locally by an operator. A pressure transmitter downstream of the rupture disk is in place to monitor premature disk rupture. An isolation valve is included after the rupture disk and bypass valve so that containment can be isolated after the rupture disk bursts.

9.3.6.2.2 System Operation

The CIS has several operational modes depending on the status of the plant, including inerting mode, makeup mode, de-inerting mode, and outage mode. A description of which CIS operational modes are used during which BWRX-300 plant operational modes are provided below.

System Startup

During plant startup, the containment hatches are closed, and the CIS is in service. Coming out of shutdown where the Containment was opened, or any condition where the oxygen concentration is above the acceptable level, the CIS is set to inerting mode. Liquid nitrogen from the nitrogen storage tank is vaporized, heated, and subsequently injected into the lower containment while exhausting from the upper containment, until the correct oxygen concentration is reached. The nitrogen is mixed with containment atmosphere by the Containment Cooling System (CCS) fans. Refer to Subsection 9.4.7. Once the targeted oxygen concentration and pressure levels are reached, the CIS CIVs close.

Once inerting is complete, the CIS can be set to makeup mode to maintain the required oxygen concentration and a slightly positive pressure inside the Containment to preclude oxygen infiltration from the RB. Liquid nitrogen from the nitrogen storage tank is vaporized, heated, and subsequently injected into the lower containment. During containment atmospheric heating conditions during plant startup, the exhaust line is opened to relieve excess containment pressure.

Operation

During power operations, the containment hatches are closed, and the CIS is in service. The CIS makeup mode is activated when the Containment pressure drops below the desired setpoint, or the oxygen concentration rises above the desired setpoint. Positive pressure is needed in the Containment to preclude oxygen infiltration from containment leakage. Liquid nitrogen from the nitrogen storage tank is vaporized, heated, and subsequently injected into lower containment. During the maximum containment atmospheric heating conditions, the containment exhaust line is opened to maintain containment internal pressure.

System Shutdown

During plant shutdown and refueling, if the containment hatches are required to be opened to support personnel entry, the CIS is set to de-inerting mode. De-inerting mode is used to achieve a safe, breathable atmosphere once the reactor reaches a setpoint temperature. In de-inerting mode, the CIS injects breathable air into the lower containment while exhausting from the upper containment until the correct oxygen concentration is reached. The breathable air is mixed with the containment atmosphere by the CCS fans as discussed in Subsection 9.4.7. Once the targeted oxygen concentration is reached for a breathable atmosphere, the containment hatches

may be opened for personnel entry. With the hatches open, the CIS may be placed into outage mode to provide a constant fresh air supply for personnel inside the Containment.

If the Containment is not required to be opened during plant shutdown, the CIS may be set to makeup mode to maintain an inerted atmosphere in the Containment.

9.3.6.3 Safety Evaluation

The following GDC have aspects specific to the CIS and compliance is addressed as follows:

GDC 41 – Containment atmosphere cleanup, as it relates to containment atmosphere cleanup. The BWRX-300 is a dry Containment and is nitrogen inerted and maintained during operation by a CIS. Fission products, hydrogen, oxygen, and other substances released from the reactor are contained within the low-leakage containment, and oxygen monitors are installed for monitoring during and after a DBA. Containment monitoring is discussed in Section 6.8.

The BWRX-300 design does not rely upon combustible gas control to maintain safe shutdown and Containment structural integrity. As stated in the “USNRC Safety Evaluation Report for NEDC-33911P-A” (Reference 9.3-6), the United States Nuclear Regulatory Commission (USNRC) finds that the BWRX-300 design is consistent with the requirements of GDC 41 for monitoring combustible gases in the Containment. The FSAR will reflect full compliance with GDC 41 based on the final design of the Containment and the combustible gas control and monitoring system, considering the results for DBAs and severe accidents. CIS additionally includes the Containment Overpressure Vent Flow Path to ensure that containment design pressure is not exceeded in a severe accident. Sizing is performed in detailed design.

GDC 42 – Inspection of containment atmosphere cleanup systems, as it relates to inspection of containment atmosphere cleanup systems. The CIS is designed with provisions for periodic inspection, operability testing, and leak rate testing of the systems or components. Refer to Subsection 9.3.6.4 for additional information pertaining to inspection of the CIS.

GDC 43 – Testing of containment atmosphere cleanup systems, as it relates to testing of containment atmosphere cleanup systems. The CIS is designed to permit periodic testing to ensure system integrity, and the operability of the systems and active components. Refer to Subsection 9.3.6.4 for detailed information pertaining to periodic testing of the CIS.

Additional information on GDC compliance is found in Section 3.1.

The CIS is designed to comply with the following regulations:

10 CFR 50.44, as it relates to combustible gas control. The BWRX-300 design is a dry, inerted Containment that does not rely upon combustible gas control to maintain safe shutdown and Containment structural integrity following a DBA. Subsection 6.2.5 addresses the functional capability of the BWRX-300 combustible gas control systems to ensure that the system maintains integrity. The approach for combustible gas control for the BWRX-300 as presented in Subsection 6.2.5 is consistent with the requirements of 10 CFR 50.44(c)(2) and, therefore, is acceptable for normal operating and DBA conditions. Refer to Subsection 6.2.5 for additional information on combustible gas control and Subsection 6.2.1.3 for compliance with 10 CFR 50.44 and conformance with RG 1.7 “Control of Combustible Gas Concentrations in Containment”. The CIS maintains a dry, inerted atmosphere in Containment during normal operations. The FSAR will reflect full compliance considering the results of severe accidents. Detailed design of the Containment Overpressure Vent Flow Path included in the CIS will consider impacts from the 100% fuel clad-coolant hydrogen generation analysis.

9.3.6.4 Inspection and Testing

During plant operation, CIS valves, instrumentation, wiring, and other components outside the Containment can be inspected visually at any time.

Periodic testing is performed to demonstrate the ability of the CIS to meet design requirements. System reference characteristics, such as pressure differentials and flow rates, are documented during the preoperational tests and are used as baselines for measurements in the subsequent operational tests. Preoperational testing requirements are defined by the Initial Test Program (ITP) (Chapter 14).

9.3.6.5 Instrumentation and Control

Refer to Subsection 7.3.3 for description of CIS monitoring and control for the generation and storage of nitrogen gas.

The CIS instrumentation monitors system pressure, temperature, and flow. The CIS system also receives indications from the Containment Monitoring Subsystem regarding containment pressure and containment oxygen concentration. Pressure control valves in the supply line are self-regulating. The control valves for makeup mode and inerting mode have different setpoints.

9.3.7 Hydrogen Water Chemistry Injection System

The Hydrogen Water Chemistry (HWC) Injection System prevents and reduces the growth rates of Intergranular Stress Corrosion Cracking (IGSCC) in reactor vessel internals. This mitigation of IGSCC is achieved from the reduction of oxygen and other oxidizing species (oxidants) in the reactor coolant. This is accomplished by injecting hydrogen into the feedwater. The injected hydrogen suppresses the radiolytic formation of oxidants in the reactor core.

To compensate for excess hydrogen, which may travel downstream and be removed from the Main Condenser by the Offgas System (OGS), a corresponding amount of oxygen (as a constituent of injected air) is injected into the OGS prior to the Offgas Recombiner.

9.3.7.1 Design Bases

Safety Category 1 Functions

None.

Safety Category 2 Functions

None.

Safety Category 3 Functions

None.

Safety Category N Functions

- Inject hydrogen into the reactor coolant system via the Feedwater System
- Inject air into the OGS upstream of the recombiner

The Safety Classifications, Quality Groups, and Seismic Categories of the HWC SSCs are listed in Table 3A-1.

9.3.7.2 System Description

The HWC Injection System receives compressed hydrogen from the Hydrogen Gas Delivery Subsystem. Compressed hydrogen from the Hydrogen Gas Delivery Subsystem is routed to a

hydrogen pressure regulation station, and then to the Power Block to provide hydrogen to the HWC Injection System. The flow rate is determined by I&Cs within the HWC Injection System.

The HWC Injection System injects hydrogen into the suction side of the feedwater pumps to mitigate IGSCC of reactor internals. The excess hydrogen being injected drives the recombination of hydrogen and oxygen in the vessel core region, resulting in a reduction of oxidants in the reactor coolant.

Hydrogen addition to the feedwater results in an excess ratio of hydrogen to oxygen at the entrance to the OGS. Refer to Section 11.3. In order to consume the excess hydrogen at this location, the HWC Injection System also controls the injection of air upstream of the Offgas recombiner to maintain a controlled percentage of oxygen at the exit downstream of the Offgas recombiner. This requires a monitoring system in the plant OGS at the Offgas recombiner exit consisting of hydrogen and oxygen monitors.

9.3.7.3 Safety Evaluation

The HWC Injection System does not perform a safety category function, nor does it contain any safety class components. However, the HWC Injection System is used, along with other measures, to reduce the likelihood of corrosion failures that would adversely affect plant availability. For storing and handling hydrogen, the guidelines in Electric Power Research Institute (EPRI) Report “Guidelines for Permanent Boiling Water Reactor Hydrogen Water Chemistry Installations,” (Reference 9.3-7) are utilized.

9.3.7.4 Inspection and Testing

The ITP is described in Chapter 14 and testing descriptions are provided in the FSAR. The system is designed for maintenance and inspection during plant operations. The system pressure boundary is pneumatically or hydrostatically tested in accordance with ASME B31.1 (Reference 9.3-3).

9.3.7.5 Instrumentation and Control

Refer to Subsection 7.3.4 for description of balance of plant controllers related to HWC Injection System monitoring and control.

9.3.8 On-Line NobleChem™

The purpose of the On-Line NobleChem™ (OLNC) System is to provide a means for the injection of a noble metal solution directly into the reactor coolant flow path to aid in the protection of reactor vessel internals from IGSCC, in combination with the addition of hydrogen by the HWC Injection System. An OLNC System and HWC Injection System mitigation strategy requires a much lower amount of hydrogen to be injected to achieve IGSCC mitigation levels, than plants that only use a HWC Injection System, and also have much lower main steam line dose rates.

9.3.8.1 Design Bases

Safety Category 1 Functions

None.

Safety Category 2 Functions

None.

Safety Category 3 Functions

None.

Safety Category N Functions

- Inject a noble metal solution to aid in the protection of reactor vessel internals

The Safety Classifications, Quality Groups, and Seismic Categories of the OLNC SSCs are listed in Table 3A-1.

9.3.8.2 System Description

The OLNC System provides a means for the injection of a noble metal solution directly into the reactor coolant flow path. A digital signal of the chemical solution weight loss (with a known metal concentration) is used by the data acquisition system to calculate the amount of noble metals injected into the reactor, as well as the rate of injection. Injections are performed approximately on an annual basis to maintain noble metal concentrations, and system operations occur for several days.

The OLNC Injection Skid consists of Noble Metal Solution Injection and Demineralized Water Injection flowpaths. There are two redundant lines feeding the noble metal injection pump. Each line consists of a tank that holds the noble metal solution, a weight scale to monitor the solution loss, a check valve to prevent back flow, and associated manual valves and fittings leading to the pump suction. Downstream of the injection pump discharge is an isolation valve to automatically isolate the injection line in case of an emergency. The noble metal solution and demineralized water are combined prior to exiting the skid.

9.3.8.3 Safety Evaluation

The OLNC System does not perform a safety category function, nor does it contain any safety class components, therefore a safety evaluation of the system is not applicable.

9.3.8.4 Inspection and Testing

Procedures are provided to perform typical maintenance and checks on the skid components prior to an application. If needed, the skid (and individual components) can also be isolated during an application to perform unplanned maintenance.

9.3.8.5 Instrumentation and Control

The system contains the appropriate components and instrumentation needed to inject the noble metal solution and to monitor the process.

9.3.9 Zinc Injection Passivation System

The Zinc Injection Passivation System provides a means for injection of soluble zinc to reduce out of core shutdown dose rates.

9.3.9.1 Design Bases

Safety Category 1 Functions

None.

Safety Category 2 Functions

None.

Safety Category 3 Functions

None.

Safety Category N Functions

- Provide a means for the injection of soluble zinc to reduce out of core shutdown dose rates. The Zinc Injection Passivation System is used only during power operation in its normal operating mode.

The Safety Classifications, Quality Groups, and Seismic Categories of the Zinc Injection Passivation System SSCs are listed in Table 3A-1.

9.3.9.2 System Description

The Zinc Injection Passivation System is designed to inject zinc ions in the reactor water via the CFS (Subsection 10.4.7). Feedwater taken from tie-in points downstream of the feedwater pump discharges is routed through a dissolution vessel containing sintered depleted zinc oxide pellets. The depleted zinc oxide pellets dissolve to produce soluble ionic zinc in the diverted feedwater stream, which is returned upstream of the feedwater pump common suction and is blended with the main feedwater flow. The feedwater transports the zinc to the reactor to deposit on the surface of the fuel and wetted reactor components, as well as the inner surfaces of external piping.

9.3.9.3 Safety Evaluation

The Zinc Injection Passivation System does not perform a safety category function, nor does it contain any safety class components, therefore a safety evaluation of the system is not applicable.

9.3.9.4 Inspection and Testing

The system is designed to be isolated from the plant to allow for maintenance while the plant is in operation without affecting the CFS. The system may be vented/depressurized while shut down for maintenance and zinc oxide pellet replenishment.

The Zinc Injection Passivation System pressure boundary is leak tested per ASME B31.1 (Reference 9.3-3).

9.3.9.5 Instrumentation and Control

Flow, temperature and differential pressure are measured for the Zinc Injection Passivation System.

9.3.10 Boron Injection System

The BIS introduces sufficient negativity into the reactor in order to assure a reactor shutdown from full power operating condition to a cold subcritical state.

9.3.10.1 Design Bases

The BIS has CIVs, which are not described in this section. The CIVs, their safety functions, and testing are described in Subsection 6.2.4. Additionally, a portion of BIS is included in the RCPB. Additional requirements associated with the RCPB are described in Section 5.2.

Safety Category 1 Functions

None.

Safety Category 2 Functions

None.

Safety Category 3 Functions

- Assure reactor shutdown from full power operation to cold subcritical without control rod movement by mixing a neutron absorber with the primary coolant

Safety Category N Functions

- Provisions for functional testing without contaminating the reactor with neutron absorber solution during each planned outage or shutdown of the reactor for refueling
- Functional testing of the BIS injection pump by circulating demineralized water with the injection valve closed

The Safety Classifications, Quality Groups, and Seismic Categories of the BIS SSCs are listed in Table 3A-1.

9.3.10.2 System Description

The BIS consists of a storage tank, a test tank, an injection pump, piping, valves, and instrumentation and controls necessary to prepare and inject a sodium pentaborate solution into the reactor vessel. The BIS simplified diagram is shown in Figure 9.3-6. The BIS does not require control rods to effect shutdown and is an independent and diverse Defense-in-Depth (D-in-D) backup system to the CRD system with respect to its capability of shutting down the reactor and maintaining it in shutdown conditions.

The BIS assures reactor shutdown from full power operation by mixing a neutron absorber with the primary coolant. This condition creates a conservative design requirement for a maximum demand of reactivity control from full power to cold shutdown conditions. The BIS mitigates Design Extension Conditions (DECs), i.e., beyond design basis accidents, in the event that the normal means for reactivity control is unable to shutdown the reactor. The BIS acts as an emergency backup to the insertion of control rods to provide a diverse means of making the reactor subcritical. The BIS is capable of injecting the sodium pentaborate solution at reactor pressures from zero to 120% of design pressure.

Electrical power needed for the injection pump, and control and instrumentation for the injection mode is made available from multiple sources: offsite power (normal supply), SDG power (backup supply), and permanently installed Emergency Mitigating Equipment connections for use with portable generators (external building connections).

The CRD system and the BIS do not share any components.

The injection location is into the condensate return line of the Isolation Condenser System (ICS) “C” loop downstream of the two ICS condensate return valves, providing a direct flow path into the reactor, see Section 6.3. This injection point was specifically selected as it is separate from the ICS loops that contain the two SDC loops discussed in Subsection 5.4.7. The BIS has its own Containment penetration and CIV upstream of the location where it injects into the ICS.

9.3.10.3 Safety Evaluation

The BIS complies with the following GDC, which have plant-wide applicability, as discussed in Section 3.1:

GDC 1 – Quality standards and records

GDC 2 – Design bases for protection against natural phenomena

GDC 4 – Environmental and dynamic effects design bases

GDC 5 – Sharing of structures, systems, and components

The following GDC and PDC have aspects related to the BIS:

PDC 26 – Reactivity control system redundancy and capability, as it relates to the requirement of having two diverse means of controlling reactivity.

The requirements of PDC 26 are met by the CRD system described in Section 4.6 and the BIS. The BIS is a diverse, D-in-D backup system to the CRD system with regards to its capability of shutting down the reactor and maintaining the reactor subcritical under cold conditions. It is a complementary design feature used to mitigate DECs when all methods of inserting control rods have failed.

The design of the BIS allows for periodic sampling and surveillance of boron concentration at the BIS storage tank as required. Details will be provided in the FSAR regarding assurance that the BIS can reliably mitigate DECs and maintain the reactor subcritical under cold conditions. Refer to Section 3.1 for more information regarding PDC 26 compliance

GDC 27 - Combined reactivity control systems capability, as it relates to the requirement that the reactivity control systems have a combined capability, in conjunction with poison addition by the ECCS, to reliably control reactivity changes under postulated accident conditions.

GDC 27 is satisfied for the BWRX-300 without the injection of soluble boron, therefore, this GDC is not applicable to BIS. Refer to Section 3.1.

The BIS considers the following regulation:

10 CFR 50.62(c)(4), as it relates to the SLCS being capable of reliably injecting a borated water solution into the reactor pressure vessel (RPV) at a boron concentration, boron enrichment, and flow rate that provides sufficient reactivity control and the need for the system to have automatic initiation, where required under the rule, to satisfy ATWS risk reduction requirements.

The BIS is used as a D-in-D measure and not considered a standby liquid control system as described in Subsection 9.3.5. Therefore, this regulation is not applicable to BIS.

9.3.10.4 Inspection and Testing

Provisions are provided in the design for both preoperational and periodic testing of the system. Periodic operational tests of the BIS pump may be conducted regardless of the state of reactor operation. BIS equipment is designed with adequate access to facilitate component maintenance.

Preoperational tests are performed to demonstrate system functionality. The ITP is discussed in Chapter 14.

9.3.10.5 Instrumentation and Control

The BIS control circuits enable the operator to control the operation of the system while the monitoring system provides the operator with information regarding status while in standby, and performance when in the injection mode. The BIS is capable of being initiated from the SCR if the MCR is not operational.

9.3.11 References

9.3-1 ASME BPVC, Section VIII, Division 1, “Rules for Construction of Pressure Vessels,” American Society of Mechanical Engineers, 2021.

9.3-2 ANSI/ISA-7.0.01-1996, “Quality Standard for Instrument Air,” American National Standards Institute/The Instrumentation, Systems, and Automation Society, 1996.

9.3-3 ASME B31.1 “Power Piping,” American Society of Mechanical Engineers, 2022.

9.3-4 NEDC-33912P-A, “BWRX-300 Reactivity Control,” GE-“Hitachi Nuclear Energy Americas, LLC, Revision 2, 2023.

9.3-5 ASME BPVC, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” Subsection NCD, “Class 2 and Class 3 Components,” American Society of Mechanical Engineers, 2021.

9.3-6 “Final Safety Evaluation for GE-Hitachi Licensing Topical Report NEDC-33911P, Revision 0, Supplement 1, BWRX-300 Containment Performance,” U.S. Nuclear Regulatory Commission, 3/12/2021.

9.3-7 EPRI NP-5283-SR-A, “Guidelines for Permanent BWR Hydrogen Water Chemistry Installations,” Electric Power Research Institute.

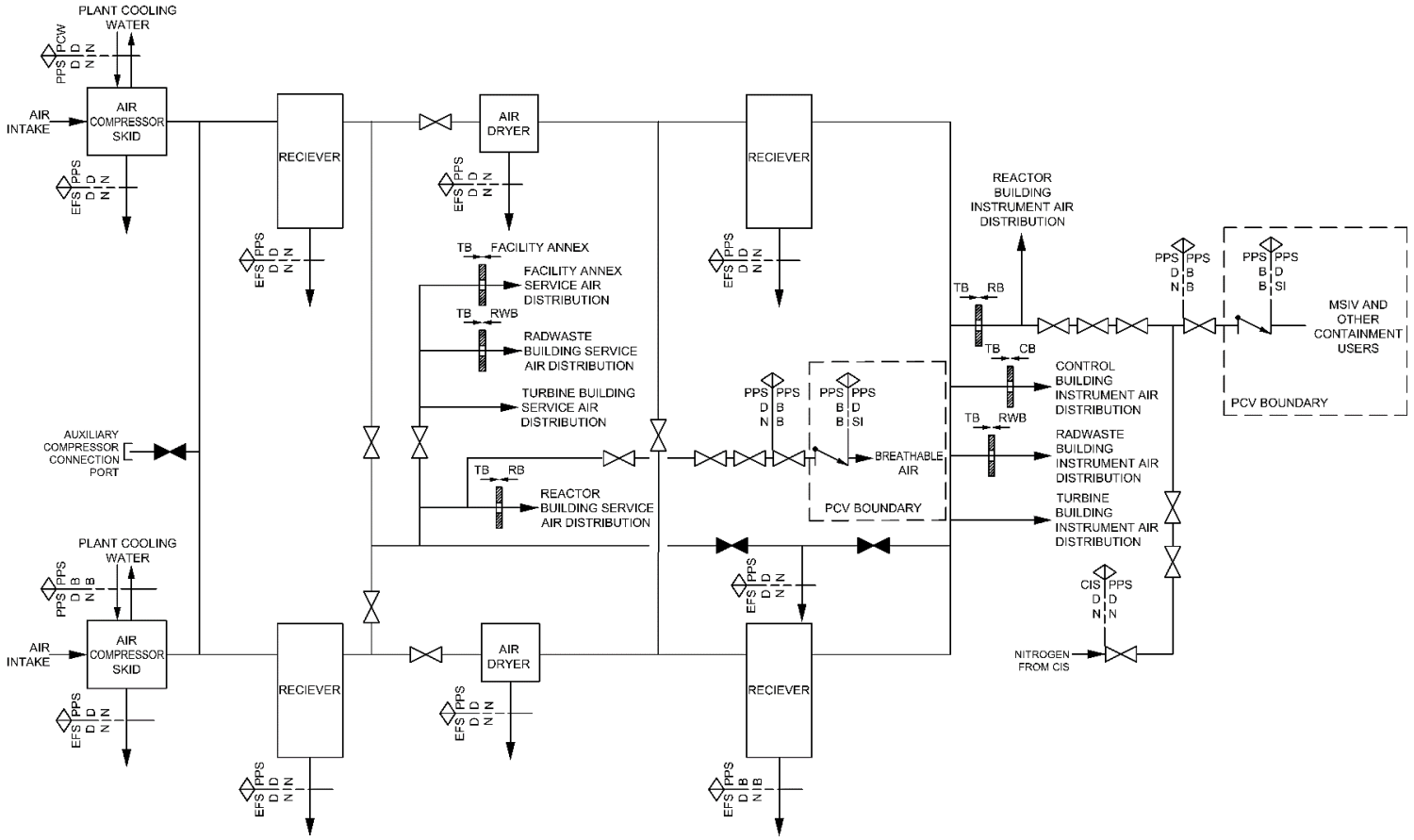


Figure 9.3-1: Plant Pneumatics System Simplified Diagram

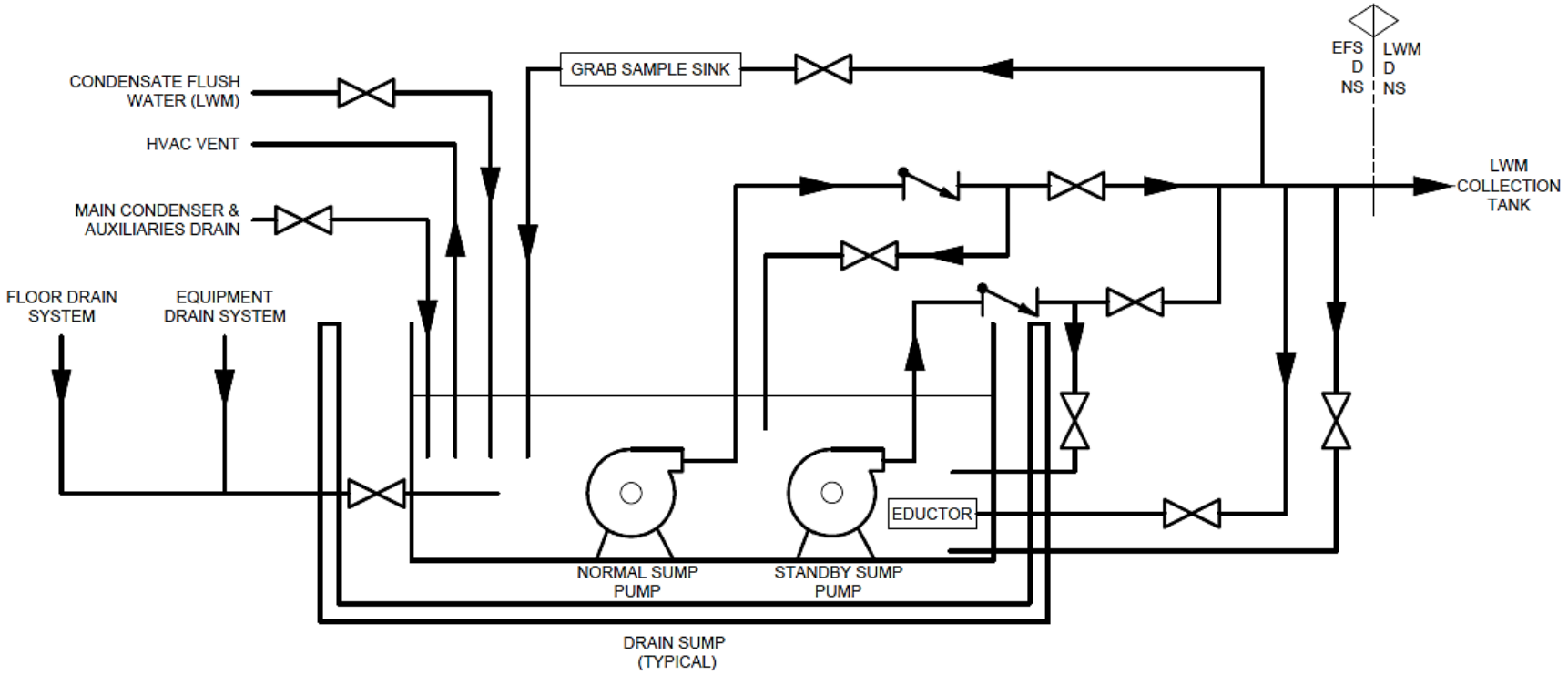


Figure 9.3-2: Normal Sump Simplified Diagram

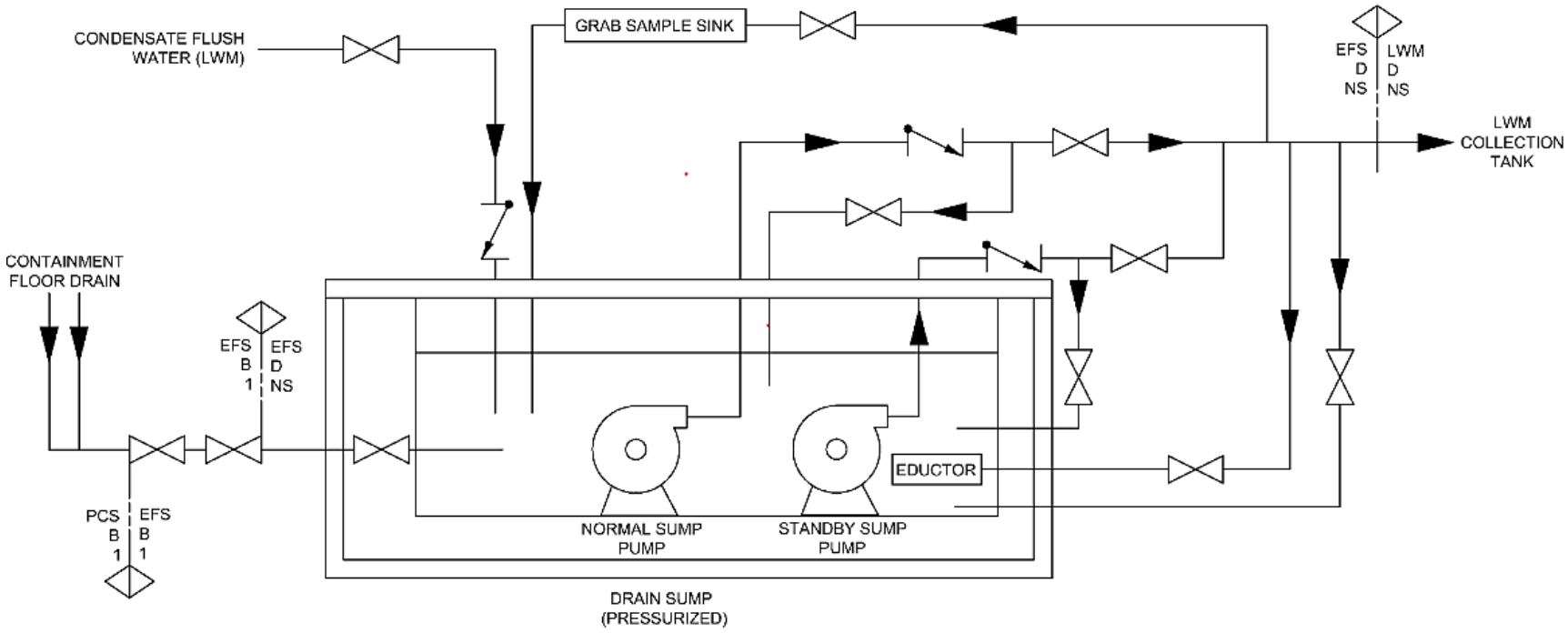


Figure 9.3-3: Pressurized Sump Simplified Diagram

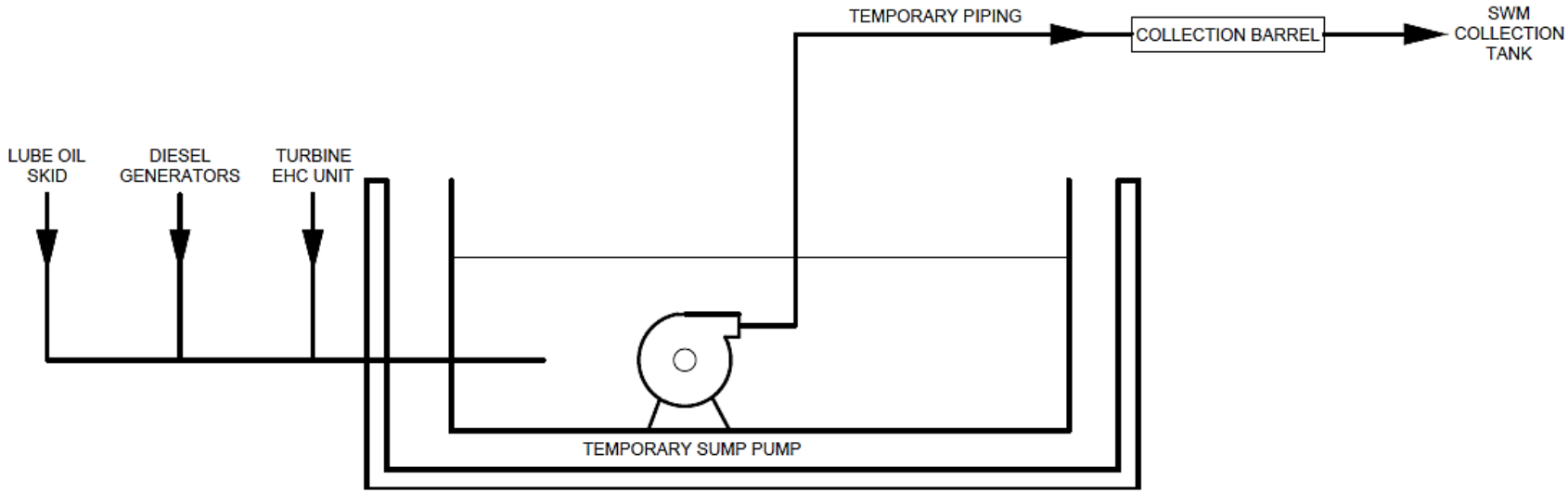


Figure 9.3-4: Oily Waste Sump Simplified Diagram

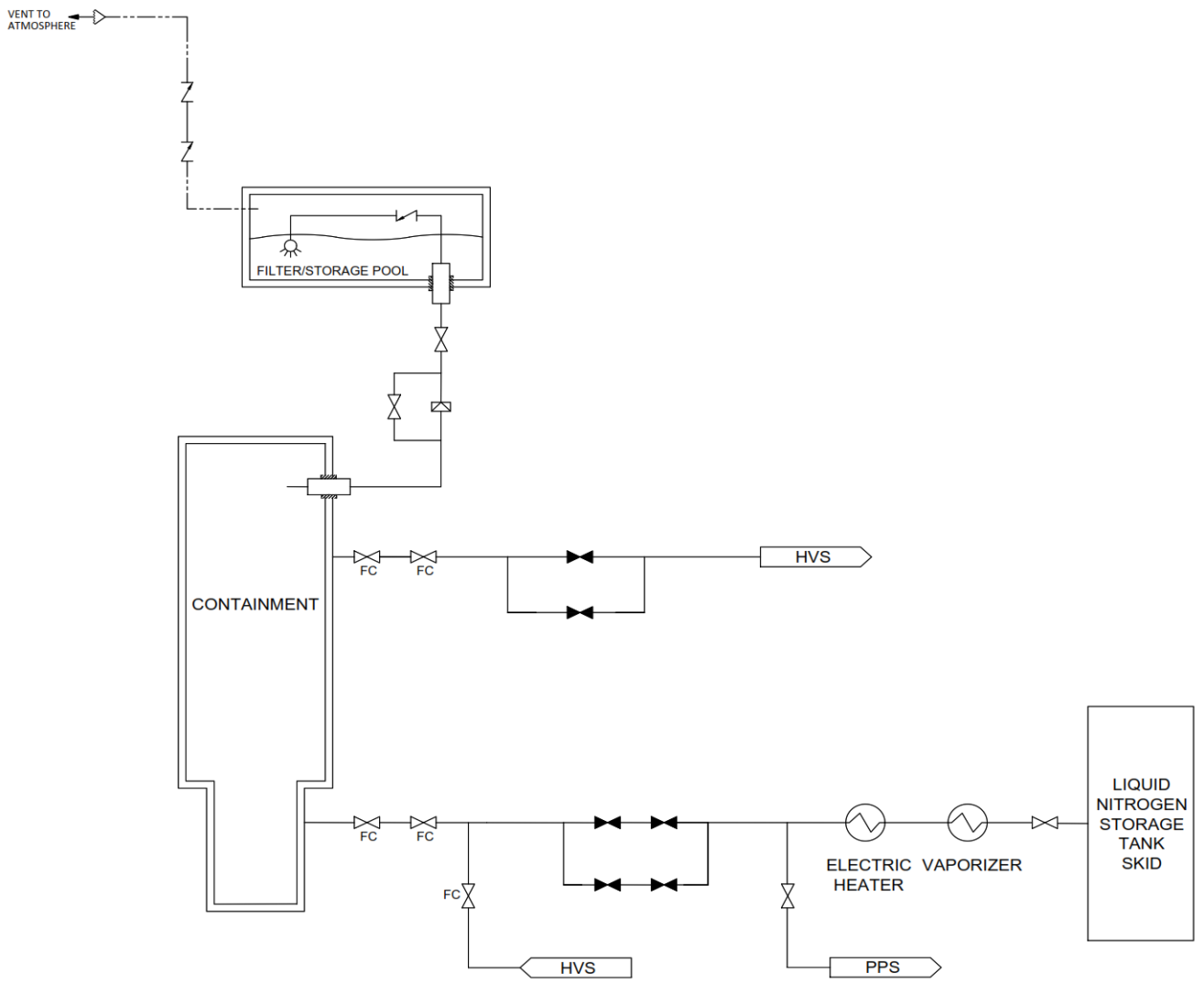


Figure 9.3-5: Containment Inerting System Simplified Diagram

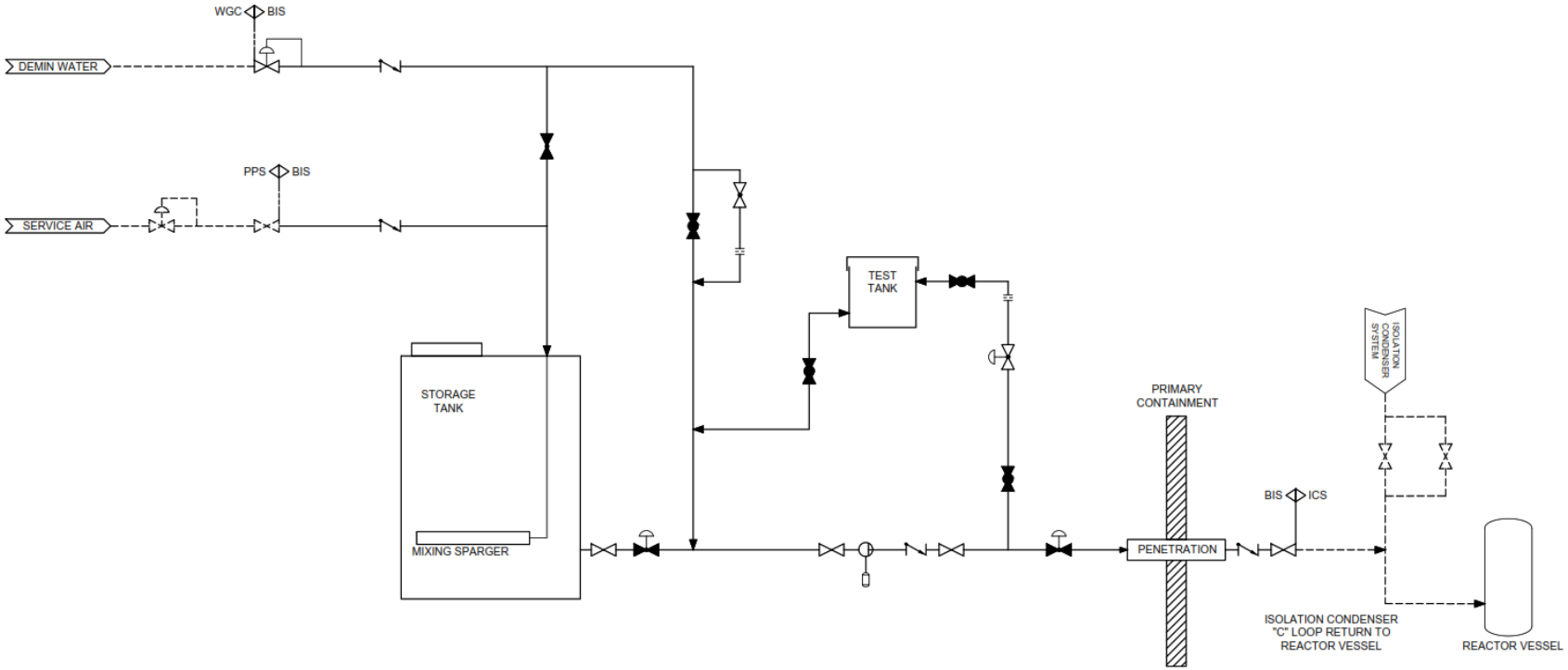


Figure 9.3-6: Boron Injection System Simplified Diagram

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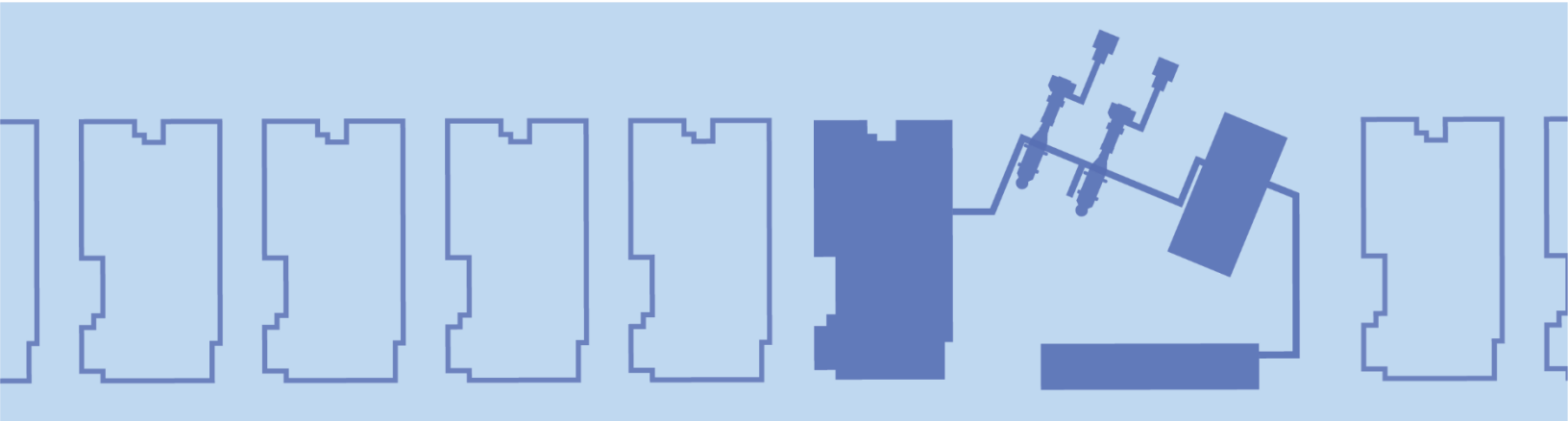
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Non-Proprietary Information

Chapter 9

Section 9.4 – Heating, Ventilation, and Cooling Systems



ACRONYM LIST

Acronym	Explanation
AHU	Air Handling Unit
ALARA	As Low As Reasonably Achievable
AMCA	Air Movement and Control Association
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
ARM	Area Radiation Monitoring Subsystem
ASD	Adjustable Speed Drive
ASME	American Society of Mechanical Engineers
BECPV	Blue Energy Center Port of Victoria
CB	Control Building
CCS	Containment Cooling System
CFR	Code of Federal Regulations
CIS	Containment Inerting System
CRD	Control Rod Drive
CRE	Control Room Envelope
CWE	Chilled Water Equipment
DBA	Design Basis Accident
DCIS	Distributed Control and Information System
EFS	Equipment and Floor Drain System
EFU	Emergency Filter Unit
EQ	Environmental Qualification
FCU	Fan Coil Unit
FPS	Fire Protection System
GDC	General Design Criteria
HVAC	Heating, Ventilation, and Air Conditioning
HVS	Heating, Ventilation, and Cooling System
I&C	Instrumentation and Control
ITP	Initial Test Program
LOPP	Loss-of-Preferred Power
MCR	Main Control Room

Acronym	Explanation
MCRE	Main Control Room Envelope
PDC	Principal Design Criterion
PREMS	Process Radiation and Environmental Monitoring System
PRM	Process Radiation Monitoring Subsystem
PVS	Plant Vent Stack
RB	Reactor Building
RG	Regulatory Guide
RWB	Radwaste Building
SBO	Station Blackout
SC1	Safety Class 1
SC2	Safety Class 2
SC3	Safety Class 3
SCCV	Steel-Plate Composite Containment Vessel
SCR	Secondary Control Room
SDG	Standby Diesel Generator
SSCs	Structures, Systems, and Components
SSE	Safe Shutdown Earthquake
TB	Turbine Building
TGFU	Toxic Gas Filtration Unit

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9.4 Heating, Ventilation, and Cooling Systems

9.4.1 Control Building Heating, Ventilation, and Cooling System

The Control Building (CB) has rooms that contain electrical equipment and associated heat dissipation loads. Other rooms, such as the Main Control Room (MCR), Radiation Protection offices, and conference room, contain areas of less heat density. The Battery Rooms, restrooms, break room, and janitor's closet are also low heat density rooms and are served by exhaust fans exhausting to the outdoors. Ductless mini-split systems supply conditioned air for the conference room, and other non-vital areas. Refer to Subsection 9.4.6 for the Heating, Ventilation, and Cooling System (HVS) subsystem related to the Secondary Control Room (SCR), which is located in the Reactor Building (RB).

9.4.1.1 Design Bases

Safety Category 1 Functions

None.

Safety Category 2 Functions

- Provide ventilation for the equipment room supporting DL4a functions

Safety Category 3 Functions

- Provide isolation to the MCR when signaled to do so by Instrumentation and Control (I&C) systems
- Maintain temperature, quality of air, and pressurization in required areas of the CB

Safety Category N Functions

- Maintain temperature, quality of air, and pressurization in other areas of the CB not classified as Safety Class 3 (SC3)

The Safety Classification, Quality Groups, and Seismic Categories of the CB HVS Structures, Systems, and Components (SSCs) are listed in Table 3A-1.

9.4.1.2 System Description

The CB HVS is shown in Figure 9.4-1.

The physical layout of the CB HVS integrates shielding, ventilation, monitoring instrumentation with traffic control, access control, and health physics aspects to assure the overall design, and minimizes radiation exposure to workers and to the public. Refer to Chapter 12 for additional information pertaining to As Low As Reasonably Achievable (ALARA).

The CB is provided heated, conditioned, and filtered supply air from one of two operating AHUs through supply ductwork, with a large percentage of the air recirculated back to the units. The second AHU is in standby, ready for auto-start should the operating unit fail. The remaining small percentage of the supply air is made up by outside air brought into the operating AHU. The normal supply AHUs are provided Adjustable Speed Drives (ASD) to be able to reduce flow as conditions change. AHU cooling in the high heat load rooms is supplemented by chilled water supplied FCUs. Ductless mini-split systems also supply conditioned air for various non-vital areas in the CB. The CB is also provided a series of exhaust fans. The exhaust fans provided for each Battery Room operate on timers, exhausting to the outdoors based on the manufacturer’s hydrogen off-gassing rate. Exhaust fans for the restrooms, break room, and janitors’ closet all operate continuously, exhausting to the outdoors. Smoke exhaust fans are also provided for the CB.

An electric duct heater in the supply duct to each Battery Room operates as required to maintain Battery Room temperatures near the optimal battery temperature as recommended by the battery manufacturer.

The CB HVS is provided tornado rated dampers as required to withstand high wind events.

Main Control Room

The Main Control Room Envelope (MCRE) is depicted within Figure 9.4-1. As discussed in Subsection 15.5.8 for the Main Control Room Model, the MCRE is not credited for mitigation of any Design Basis Accidents (DBA) because the MCR habitability is not a Safety Category 1 function. Instead, the analysis models the CB HVS only for the purpose of propagating contamination into the MCRE.

Appropriate measures protect occupants of the MCR against hazards such as high radiation levels resulting from DBAs, release of radioactive material, fire, or explosive or toxic gases.

The outside air intake is instrumented to analyze for toxic gases and for airborne radioactivity. To accommodate MCR habitability contingencies, the following additional AHUs are provided for the CB:

- CB Toxic Gas Filtration Units (TGFU) that operate automatically when toxic gas has been detected at the CB AHU outside air intakes
- MCRE Emergency Filter Units (EFU) that operate automatically upon detection of high radiation level at the operating CB supply AHU outside air intake. The MCRE goes into isolation mode.

These provide a pressurized envelope for the MCRE with respect to adjacent spaces and maintain MCR habitability during airborne radiation and toxic events.

Tornado dampers are provided on CB EFU air intake/exhaust openings and are designed to withstand the full negative pressure drop. They close when required to prevent the collapse of ductwork.

Off-Normal CB HVS Operation

In the event of a CB outside air intake radiation monitor alarm, the operating supply AHU and any associated TGFUs de-energize, and the same train MCRE EFU automatically starts. The MCRE isolation dampers close in conjunction with the start of the MCRE EFU. Standby Diesel Generator (SDG) backup power is provided to the MCRE EFU supply fans.

In the event of a CB outside air intake toxic gas detection, the operating AHU outside air intake closes, and the same train TGFU automatically starts and its associated motor control dampers automatically open, providing an alternate source of filtered outside air for CB pressurization.

SDG backup power is provided to the TGFU supply fans but not the heating coils in the supply AHU. Positive pressure is maintained in the CB by way of providing outside air through the supply AHU, the TGFU, or the MCRE EFU.

In the event of a fire, the CB HVS provides isolation of the MCRE (thereby preventing smoke injection) and exhaust of smoke from the CB and/or MCRE in coordination with the Fire Protection System (FPS) as described in Subsection 9.5.1.

In the event of a Loss-of-Preferred Power (LOPP), the restroom, break room, and janitors closet exhaust fans become inoperable.

In the event of a Station Blackout (SBO), power is lost to the CB AHUs, as well as electric duct heaters and cabinet heaters. The restroom, break room, and janitors closet exhaust fans become

inoperable. The Control Room Operators relocate to the SCR in the RB if the MCR becomes uninhabitable.

9.4.1.3 Safety Evaluation

The CB HVS complies with the following General Design Criteria (GDC), which have plant-wide applicability, as discussed in Section 3.1:

GDC 2 – Design bases for protection against natural phenomena

GDC 4 – Environmental and dynamic effects design bases

GDC 5 – Sharing of structures, systems, and components

The following GDC and PDC have aspects specific to the CB HVS, and compliance is addressed as follows:

Principal Design Criteria (PDC) 19 – Control room. This PDC is met as described in Section 3.1 and 15.5. None of the isolation functions of the Process Radiation Monitoring Subsystem (PRM) are credited to mitigate the dose consequences from DBAs. The analysis models the CB HVS only for the purpose of propagating contamination into the Main Control Room.

GDC 60 – Control of releases of radioactive materials to the environment, as it relates to system capability to suitably control release of gaseous radioactive effluents to the environment.

Additional information on GDC and PDC compliance is found in Section 3.1.

The CB HVS is designed to comply with the following regulation:

Compliance with 10 CFR 50.63 as it relates to loss of all Alternating Current power.

The CB HVS and its components use the guidance of Regulatory Guide (RG) 1.78, “Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release,” to assure meeting MCR occupancy protection requirements, as it relates to permitting access to and occupancy of the control room under accident conditions.

Mobile sources of hazardous chemicals will also be considered in an offsite transportation hazards screening in CPA Part 2. The outside air intake for the CB HVS is instrumented to analyze for toxic gases. The TGFUs operate automatically when toxic gas is detected at the CB AHU outside air intakes as described in Subsection 9.4.1.2. SDG backup power is provided to the TGFU supply fans if normal Alternating Current (AC) power is lost. This is an alternative approach to RG 1.78 conformance because the CB HVS pressurization and isolation functions are Safety Category 3 and Non-Seismic. Refer to Chapter 8 for the Safety Category of the SDGs and the Standby Power System. Other off-normal events that impact operation of the CB HVS active features (such as earthquake) were not considered as concurrent events. Refer to Section 6.4 for additional discussion regarding control room habitability.

In the event of a CB outside air intake radiation monitor alarm, the operating supply AHU and any associated TGFUs de-energize, and the same train MCRE EFU automatically starts. The MCRE isolation dampers close in conjunction with the start of the MCRE EFU. SDG backup power is provided to the MCRE EFU supply fans, and tornado dampers that are provided on EFU air intake/exhaust openings.

Refer to Table 3.7-1 and Section 3.8.4.4 for the description of the CB structure. A seismic event of sufficient magnitude may result in the MCR becoming functionally impaired. Under this scenario, the SCR is the assured shutdown location due to the seismic rating of the RB. The RB, which contains the SCR, is designed as a Seismic Category I structure. Refer to Subsection 9.4.6

for SCR HVS details. The design of the BWRX-300 is such that no event results in both Control Rooms being uninhabitable.

The design of the CB HVS complies with GDC 60 as it relates to control releases of radioactive materials to the environment. The CB does not house any portion of the nuclear steam supply process or other equipment that can act as a source of radioactive material, and therefore, has no postulated sources of radioactive materials in either particulate or gaseous form. Therefore, controls are provided relative to release of gaseous radioactive effluents to the environment. As stated in Section 6.4, the BWRX-300 does not utilize an engineered safety feature ventilation system, and therefore the guidance of RG 1.52, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants,” is not applicable. The CB HVS follows the guidance of RG 1.140, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants,” Criteria C.2 and C.3. The guidance of RG 1.143, “Design Guidance for Radioactive Waste management Systems, Structures, and Components Installed in Light-Water Cooled Nuclear Power Plants,” regarding gaseous radwaste management is considered consistent with Title 10 of the Code of Federal Regulations (10 CFR) Part 20, “Standards for Protection Against Radiation.” These provisions assure that the release of radioactive materials entrained in gaseous effluents during reactor operation, including Anticipated Operational Occurrences (AOO), is controlled.

The design of the CB HVS considers 10 CFR 50.63 as it relates to necessary support systems providing sufficient capacity and capability to ensure the capability to cope with a SBO event. SBO is discussed in Section 8.4. The Control Room Operators relocate to the SCR in the RB if the MCR becomes uninhabitable. The SCR in the RB is passively cooled for at least 72 hours and maintains a habitable environment for the event as the assured shutdown location as described in Subsection 9.4.6. The SDGs, when available, provide power for the CB AHUs, TGFUs, and EFUs.

Refer to Subsection 9.4.6.3 for the safety evaluation of the SCR environment located in the RB.

9.4.1.4 Inspection and Testing

System air balance tests and adjustment to design conditions are conducted during the Initial Test Program (ITP) (Chapter 14). The CB HVS is designed to permit periodic inspection and testing of major components, such as fans, motors, dampers, coils, filters, and ducts to verify their integrity, operability, and capability. CB HVS equipment and components are provided with proper access for initial and periodic inspection and maintenance activities.

Section AB of ASME AG-1, “Code on Nuclear Air and Gas Treatment,” (Reference 9.4-1) contains guidance for the testing of air treatment systems in nuclear facilities.

Testing and ratings of tornado dampers are in accordance with Air Movement and Control Association (AMCA) Standard 500-D, “Laboratory Methods of Testing Dampers for Rating,” (Reference 9.4-2).

9.4.1.5 Instrumentation

The CB HVS has dedicated temperature control panels that regulate the temperature of the spaces to within specified ranges. Instrumentation monitors the parameters using inputs to the control logic including temperature, moisture, differential pressure, damper position, and smoke detectors. Refer to Section 7.3 for a description of the Distributed Control and Information System (DCIS) functions.

9.4.2 Fuel Pool Area Ventilation

The fuel pool area is ventilated by the RB HVS. The BWRX-300 does not have a dedicated ventilation system for the fuel pool area. Information pertaining to the RB HVS is provided in Subsection 9.4.6. The RB HVS provides an exhaust function above the fuel pool.

9.4.3 Radwaste Building Heating, Ventilation, and Cooling System

The Radwaste Building (RWB) contains room temperature tanks and small to medium-sized pumps. Plant personnel can access the Chem Lab and dress out areas within the RWB.

9.4.3.1 Design Bases

Safety Category 1 Functions

None.

Safety Category 2 Functions

None.

Safety Category 3 Functions

None.

Safety Category N Functions

- Maintain temperature, quality of air, and pressurization in all required areas of the RWB

The Safety Classification, Quality Groups, and Seismic Categories of the RWB HVS SSCs are listed in Table 3A-1.

9.4.3.2 System Description

The RWB HVS is shown in Figure 9.4-2.

RWB ventilation is provided by supply AHUs and exhaust AHUs, functioning in a push-pull manner. Clean filtered outside air, heated when needed, is supplied through ductwork to the lobby and lab, pressurizing those areas. The exhaust AHUs take suction on the potentially contaminated tank and pump rooms, creating negative pressures in those spaces, resulting in air transferring from clean to “dirty” areas. Air is exhausted from the RWB through the operating exhaust AHUs, discharging to atmosphere from the Plant Vent Stack (PVS).

The RWB is provided with FCUs to supplement cooling provided by the AHUs in various areas such as the lobby area, the lab, and the upper elevation of the main area. These FCUs are provided chilled water for cooling. The lab is also provided a small cabinet heater, and the lobby area is provided with electric unit heaters. The Chem Lab Fume Hood exhaust is drawn into RWB Exhaust AHU suction.

During recovery from a fire, smoke is exhausted from the RWB.

9.4.3.3 Safety Evaluation

The RWB HVS complies with the following GDC, which have plant-wide applicability, as discussed in Section 3.1:

GDC 2 – Design bases for protection against natural phenomena

GDC 5 – Sharing of structures, systems, and components.

The following GDC have aspects specific to the RWB HVS, and compliance is addressed as follows:

GDC 60 – Control of releases of radioactive materials to the environment, as it relates to system capability to suitably control release of gaseous radioactive effluents to the environment.

The design of the RWB HVS complies with GDC 60 as it relates to control releases of radioactive materials to the environment. The RWB HVS air is exhausted from the RWB through operating exhaust AHUs, discharging to atmosphere from the PVS. Effluent from the PVS is monitored for radiation by the PRM as described in Section 11.5. The guidance of RG 1.143 regarding gaseous radwaste management is considered consistent with 10 CFR Part 20. These provisions assure that the release of radioactive materials entrained in gaseous effluents during reactor operation, including AOOs, is controlled.

Additional information on GDC compliance is found in Section 3.1.

9.4.3.4 Inspection and Testing

System air balance tests and adjustment to design conditions are conducted during the ITP. The RWB HVS is designed to permit periodic inspection and testing of major components, such as fans, motors, dampers, coils, filters, ducts, piping, and valves to verify their integrity and capability. Equipment layout provides access for inspection and maintenance. Instruments are calibrated during testing.

Section AB of ASME AG-1 contains guidance for the testing of air treatment systems in nuclear facilities.

9.4.3.5 Instrumentation

The RWB HVS has dedicated temperature control panels that regulate the temperature of the spaces to within specified ranges. Instrumentation monitors the parameters using inputs to the control logic including temperature, moisture, differential pressure, damper position, and smoke detectors. Refer to Section 7.3 for a description of the DCIS functions.

9.4.4 Turbine Building Heating, Ventilation, and Cooling System

The Turbine Building (TB) is a large building with significant cooling loads. It is divided into two areas: (1) potentially contaminated areas inside the bioshield area, and (2) non-contaminated areas outside the bioshield. Equipment inside the bioshield includes the Main Turbine, Main Condenser, Moisture Separator Reheater, Feedwater Heaters and other loads. Equipment outside the bioshield includes the Main Generator, assorted pumps, heat exchangers, and electrical equipment.

9.4.4.1 Design Bases

Safety Category 1 Functions

None.

Safety Category 2 Functions

None.

Safety Category 3 Functions

None.

Safety Category N Functions

- Maintain temperature, quality of air, and pressurization in all required areas of the TB

The Safety Classification, Quality Groups, and Seismic Categories of the TB HVS SSCs are listed in Table 3A-1.

9.4.4.2 System Description

The TB HVS, which includes the PVS, is shown in Figure 9.4-3.

The TB HVS is designed to maintain temperature, quality of air, and pressurization in all required areas of the TB. The TB HVS provides a controlled environment for personnel safety and comfort, and for the proper operation and integrity of equipment located in the TB. Materials used in the design of the TB HVS are selected in accordance with applicable codes, standards, and industry practice for the design, service, and test conditions, and expected ambient conditions. The physical layout of the TB relative to the TB HVS integrates shielding, ventilation, monitoring instrumentation with traffic control, access control, and health physics aspects to assure the overall design minimizes radiation exposure to workers and to the public. The TB stairwells are designed as smokeproof enclosures. Upon detection of smoke, the stairwells are pressurized to minimize ingress of smoke. The TB HVS equipment is operated in coordination with the FPS, as described in Subsection 9.5.1.

The TB is supplied with outside air using supply AHUs. These AHUs are provided with electric heating coils to provide a reasonable supply air temperature, and upstream prefilters to keep dirt off the coils, and for building cleanliness. The supply air is distributed through ductwork to areas both inside and outside the bioshield area. Most floors in the TB consist of metal grating, allowing supply air provided in the lower elevations of the TB to increase in temperature and elevate naturally. Air is exhausted from the building via AHUs, taking suction from roof penetrations or suction ductwork. Air flow rates to areas have been chosen to effectively transfer air from clean areas outside the bioshield area to potentially contaminated areas inside the bioshield area. TB supply and exhaust AHUs are all provided ASDs to be able to reduce flow as permitted by plant operation and reduced ambient temperature conditions.

The HVS air supply in the vicinity of the Main Steam piping is designed such that turbulence does not preclude a reliable high steam line turbine area temperature signal from being generated.

The TB Switchgear rooms are each provided with FCUs.

The TB Battery Room is ventilated and conditioned by two mini-split systems. One unit is provided backup power from SDG Train A and the other unit is provided backup power from SDG Train B. One of these two systems works with electric unit heaters as needed to maintain space temperature within design. Each unit brings a small amount of filtered, conditioned outside air into the room, maintaining a slight positive pressure in the space. Hydrogen off-gassing of batteries require exhaust ventilation based on the manufacturer's off-gassing rate, operated on a timer.

Plant Vent Stack

The TB HVS includes the scope of the PVS as shown in Figure 9.4-3.

Exhaust air from potentially contaminated areas, including within the TB, RB, and RWB is exhausted to the Continuous Exhaust Air Plenum. The Continuous Exhaust Air Plenum serves as a large mixing box where potentially contaminated air is mixed and diluted. During operation, the PVS fans take suction on the Continuous Exhaust Air Plenum and discharge to the nearby PVS. The PVS fans are provided ASDs to be able to maintain the Continuous Exhaust Air Plenum negatively pressurized relative to the outside and to adjust to varying flow inputs, which may vary based on the operation of the AHUs at each building.

9.4.4.3 Safety Evaluation

The TB HVS complies with the following GDC, which have plant-wide applicability, as discussed in Section 3.1:

GDC 2 – Design bases for protection against natural phenomena

GDC 5 – Sharing of structures, systems, and components

The following GDC have aspects specific to the TB HVS, and compliance is addressed as follows:

GDC 60 – Control of releases of radioactive materials to the environment, as it relates to system capability to suitably control release of gaseous radioactive effluents to the environment.

The design of the TB HVS complies with GDC 60 as it relates to control releases of radioactive materials to the environment. The TB HVS air is exhausted from the TB, discharging to atmosphere from the PVS. Effluent from the PVS is monitored for radiation by the PRM Subsystem as described in Section 11.5. The guidance of RG 1.143 regarding gaseous radwaste management is considered consistent with 10 CFR Part 20. These provisions assure that the release of radioactive materials entrained in gaseous effluents during reactor operation, including AOOs, is controlled.

Additional information on GDC compliance is found in Section 3.1.

9.4.4.4 Inspection and Testing

System air balance tests and adjustment to design conditions are conducted during the ITP. The TB HVS is designed to permit periodic inspection and testing of major components such as fans, motors, dampers, coils, filters, ducts, piping, and valves to verify their integrity and capability.

Equipment layout provides access for inspection and maintenance. Instruments are calibrated during testing.

Testing and ratings of tornado dampers are in accordance with AMCA Standard 500-D.

9.4.4.5 Instrumentation and Control

The TB HVS has dedicated control panels that regulate the temperature of the spaces to within specified ranges. Inputs to the control logic include temperature, moisture, differential pressure, damper position, smoke detectors.

9.4.5 Engineered Safety Feature Ventilation System

The BWRX-300 design does not utilize engineered safety features for ventilation systems to mitigate the offsite consequences of a DBA. Therefore, this section is not applicable to the BWRX-300.

9.4.6 Reactor Building Heating, Ventilation, and Cooling System

The RB is a multi-level building that has levels both above and below grade. Both the Fuel Pool and the SCR are located within the RB.

9.4.6.1 Design Bases

Safety Category 1 Functions

None.

Safety Category 2 Functions

- Maintain temperature, quality of air, and pressurization in the SCR and DCIS rooms of the RB

- Provide isolation to the SCR when signaled to do so by I&C systems
- Passively cool for the 72-hour battery coping time the SCR and Safety Class 1 (SC1) DCIS and electrical equipment rooms within the RB, such that the SCR remains habitable, and the DCIS and electrical equipment rooms do not exceed the temperatures for which their equipment has been qualified.

Safety Category 3 Functions

- Provide isolation to the Refueling Floor when signaled to do so by I&C systems

Safety Category N Functions

- Maintain temperature, quality of air, and pressurization in other areas of the RB not classified as Safety Class 2 (SC2) or SC3

The Safety Classification, Quality Groups, and Seismic Categories of the RB HVS are listed in Table 3A-1.

9.4.6.2 System Description

Figure 9.4-4 shows the RB HVS. HVS for the Containment is described in Subsection 9.4.7.

Portions of the RB HVS perform Safety Category 2 support functions to maintain the environment containing environmentally qualified SSCs within their prescribed limited. The physical layout of the RB HVS integrates shielding, ventilation, monitoring instrumentation with traffic control, access control, and health physics aspects to assure the overall design minimizes radiation exposure to workers and to the public. The RB HVS is operated in coordination with the FPS as described in Subsection 9.5.1.

The RB lower elevations are provided heated, conditioned, and filtered once-through supply air through ductwork from one of two operating AHUs. For each of the RB supply and exhaust AHUs, the second AHU is in standby, ready for auto-start should the operating AHU fail. These units are also provided ASDs to be able to adjust flow as necessary to maintain static pressure. An electric duct heater in the supply duct to each Battery Room and to the SCR operate as required to maintain Battery Room temperatures near the optimal battery temperature as recommended by the battery manufacturer, with cooling provided by supply ductwork from the operating chilled-water AHUs. Hydrogen off-gassing of batteries requires exhaust ventilation based on the manufacturer's off-gassing rate. Hydrogen off-gassing is operated on a timer.

AHU cooling is supplemented by cooling provided by chilled water supplied FCUs located in rooms that require extra cooling, with FCUs being controlled by room thermostats.

The RB HVS is designed to maintain the Electrical Distribution System Battery Room during normal operation within the optimal temperature range specified by the manufacturer.

The RB upper-level areas, consisting of the fuel handling area and operating deck, are provided heated, conditioned, and filtered once-through supply air from one of two AHUs. For each of the operating RB upper-level supply AHUs, the second supply AHU is in standby, ready for auto-start should the operating AHU fail. These units are provided ASDs to be able to vary flow to the refueling pool as needed. In the event of a fuel handling accident, a radiation monitor alarm initiates closure of the RB upper-level area supply and exhaust isolation dampers and securing of the RB upper level supply AHUs.

Air is exhausted from the RB upper and lower levels, including the Battery Rooms, from one of two operating exhaust AHUs, discharging to atmosphere from the PVS.

The RB exhaust receives air and inerting gases from the Containment Inerting System (CIS) described in Subsection 9.3.6, when required, to be exhausted to the PVS. The RB lower-level supply AHUs will also supply outside air to the CIS when required for containment de-inerting and to provide a habitable environment for maintenance personnel during refueling and maintenance operations.

The RB HVS is provided tornado dampers at all RB penetrations to withstand high wind events.

In the event of a LOPP, power is maintained using the SDG to one RB exhaust AHU fan, as well as one upper and one lower-level AHU supply fan, providing ventilation air to the RB. However, backup power is not provided to the heating coils of the supply AHUs nor the electric duct heaters for the Battery Rooms.

In the event of a SBO, power is lost to the RB exhaust AHUs, the upper and lower-level supply AHUs, as well as electric duct heaters for the Battery Rooms. No cooling is needed for at least the first 72 hours of the SBO event.

Secondary Control Room

Figure 9.4-5 shows the SCR HVS in the RB. The SCR is normally provided filtered, conditioned air from the operating RB lower level AHU. The SCR is provided with two trains of EFUs and pressurization fans that supply ventilation air to the operators when automatically placed into service upon detection of radiation, toxic gas, or smoke at the normal lower-level supply AHUs. A loss of power to both normal supply AHUs also initiates operation of the SCR EFUs and pressurization fans. The SCR EFUs and pressurization fans are powered by the Emergency Power System and backed up by battery power. The operating pressurization fan draws air through dedicated ducting located in the hallway and Truck Bay exterior walls. During SBO SCR operations, the SCR is passively cooled for 72 hours by the thermal mass of the surrounding concrete structure and heat transfer to the adjacent ground. When power and Chilled Water Equipment (CWE) (Subsection 9.2.7) are available, supplemental cooling is provided to the SCR by two FCUs. Electric duct heaters are provided in the SCR emergency supply air ducting to heat incoming outside air, if necessary.

9.4.6.3 Safety Evaluation

The RB HVS complies with the following GDC, which have plant-wide applicability, as discussed in Section 3.1:

GDC 2 – Design bases for protection against natural phenomena

GDC 4 – Environmental and dynamic effects design bases

GDC 5 – Sharing of structures, systems, and components

The following GDC and PDC have aspects specific to the RB HVS, and compliance is addressed as follows:

PDC 19 – Control room. This PDC is met as described in Section 3.1 and Section 15.5. None of the isolation functions of the PRM are credited to mitigate the dose consequences from DBAs.

GDC 60 – Control of releases of radioactive materials to the environment, as it relates to system capability to suitably control release of gaseous radioactive effluents to the environment.

GDC 61 – Fuel storage and handling and radioactivity control, as it relates to the systems capability to provide appropriate containment, confinement, and filtering to limit releases to airborne radioactivity under normal and postulated accident conditions.

Additional information on GDC and PDC compliance is found in Section 3.1.

The RB HVS is designed to comply with the following regulation:

Compliance with 10 CFR 50.63 as it relates to loss of all offsite or standby diesel generator AC power.

The Secondary Control Room Envelope is served by a ventilation system with the intent of maintaining the habitability of the SCR. The Secondary Control Room Envelope is shown in Figure 9.4-5 and consists of the SCR.

The design of the SCR, with emergency pressurization fans and EFUs, assures that adequate protection against radiation and hazardous chemical releases is provided such that access to and occupancy of a control room under accident conditions is afforded. The Emergency Power System provides battery-backed power to the SCR EFUs and pressurization fans to support SCR habitability functions.

The MCR, located in the CB, is the primary location for plant operations and habitability per PDC 19 and is discussed in Subsection 9.4.1. The SCR located in the RB is assured shutdown location for the plant after MCR evacuation if necessary. Section 6.4 discusses conditions that could warrant relocation from the MCR to the SCR. The design is such that no DBA can simultaneously affect both control rooms, so a habitable location is maintained. The hazardous chemicals screening performed per the guidance of RG 1.78 is described in Subsection 9.4.1 and does not consider a simultaneous event that would result in evacuation of the MCR to the SCR, although the SCR provides a habitable location under accident conditions. As discussed in Subsection 15.4.1, no human actions are required by a control room operator for DBAs.

The design of the RB HVS complies with GDC 60. Area and process radiation monitors in the RB measure levels of radiation within the building. If radiation monitoring is measured above the setpoint, the building is isolated. Complying with the requirements of GDC 60 provides assurance that release of radioactive materials entrained in gaseous effluents does not exceed specified 10 CFR Part 20 limits for normal operation and AOOs.

The design of the RB HVS complies with GDC 61 as it relates to radioactivity control of the refueling floor and fuel pool area. The RB HVS provides containment and confinement to limit releases of airborne radioactivity to the environment from the fuel pool under normal and postulated accident conditions. The portion of the HVS that services the Refueling Floor is capable of isolation when signaled to do so by monitoring instrumentation. In the event of a fuel handling accident, a radiation monitor alarm initiates closure of the RB isolation dampers and securing of the RB upper-level AHUs. The potential release of radioactive gases is contained. The Area Radiation Monitoring Subsystem (ARM), a subsystem of the Process Radiation and Environmental Monitoring System (PREMS), is responsible for area radiation monitoring, and provides the isolation signal.

The design of the RB HVS considers 10 CFR 50.63 as it relates to necessary support systems providing sufficient capacity and capability to ensure the capability for cope with a SBO event. SBO is discussed in Section 8.4.

In the event of a SBO, power is lost to the RB exhaust AHUs, the upper and lower-level supply AHUs, as well as electric duct heaters for the Battery Rooms. No cooling is needed for at least the first 72 hours of the SBO event.

The BWRX-300 SCR, DCIS, and electrical equipment rooms are passively cooled by the thermal mass of the surrounding concrete structure and adjacent ground for the 72-hour battery coping time such that the SCR remains habitable, and the DCIS and electrical equipment rooms do not exceed the temperatures for which their equipment has been qualified. The Emergency Power

System provides battery-backed power to support any accident monitoring and functions associated with SCR habitability.

9.4.6.4 Inspection and Testing

System air balance tests and adjustment to design conditions are conducted during the ITP. The RB HVS is designed to permit periodic inspection and testing of major components, such as fans, motors, dampers, coils, filters, ducts, piping, and valves to verify their integrity and capability.

Equipment layout provides access for inspection and maintenance. Instruments are calibrated during testing.

Charcoal is provided in the emergency filtration AHUs for either radioactive iodine removal, toxic gas removal, or both, based on the site-specific hazard analysis. Testing of the charcoal is performed via grab samples to be sent to a laboratory for testing.

Section AB of ASME AG-1 contains guidance for the testing of air treatment systems in nuclear facilities.

Testing and ratings of tornado dampers are in accordance with AMCA Standard 500-D.

9.4.6.5 Instrumentation and Controls

The RB HVS has dedicated temperature control panels that regulate the temperature of the spaces to within specified ranges. Instrumentation monitors the parameters using inputs to the control logic including temperature, moisture, differential pressure, damper position, and smoke detectors. Refer to Section 7.3 for a description of the DCIS functions.

9.4.7 Containment Cooling System

The Containment Cooling System (CCS) provides cooling for the atmosphere inside the Containment using AHUs with cooling coils that reject heat to the CWE (Subsection 9.2.7).

9.4.7.1 Design Bases

Safety Category 1 Functions

None.

Safety Category 2 Functions

None.

Safety Category 3 Functions

- Maintain the containment bulk average temperature within Environmental Qualification (EQ) limits of the related equipment located inside the Containment during run/startup/shutdown modes of operation
- CCS assists with containment cooldown following a LOPP and during periods of hot and cold shutdowns

Safety Category N Functions

- CCS maintains temperature for plant personnel entering the Containment during shutdowns and refueling activities

The Safety Classification, Quality Groups, and Seismic Categories of the CCS SSCs are listed in Table 3A-1.

9.4.7.2 System Description

Figure 9.4-6 depicts a simplified diagram of the CCS. The CCS maintains containment bulk average temperature inside the Containment within required limits.

The CCS is designed per the requirements of American National Standards Institute (ANSI)/American Nuclear Society (ANS) 56.7, “American National Standard for Boiling Water Reactor Containment Ventilation Systems,” (Reference 9.4-3).

The CCS is a closed loop air or nitrogen recirculating cooling system that consists of two fully redundant trains and a system of ducts and dampers to distribute the air as needed in different locations throughout the Containment. Each train is cooled by a corresponding chilled water train supplied from the CWE as described in Subsection 9.2.7, so that Containment cooling is possible even with loss of one train of CWE or one train of CCS. Each CCS train consists of two 50 percent AHUs. Each AHU is comprised of fans, cooling coils, and filter placement locations. Condensed liquid around the cooling coils is drained to the bottom of the Containment and then directed to the Equipment and Floor Drain System (EFS) as described in Subsection 9.3.3.

The AHUs and ductwork are all located in the Containment. Ducts are used to circulate air or nitrogen in the Containment to achieve the desired cooling and maintain bulk average Containment temperature. Supply ductwork is routed to different locations in Containment and the Control Rod Drive (CRD) area. Return ductwork originates from the upper area of the Containment and the upper CRD area and is utilized to provide ventilation to mitigate thermal stratification and circulate air back to the AHUs.

The CCS does not introduce any external air or nitrogen into the Containment in any mode of operation. The air inside the Containment is purged with nitrogen using the CIS, Subsection 9.3.6, during startup and kept inerted during plant operation.

Each AHU can be powered from the corresponding standby diesel generator of the Electrical Distribution System.

One of the two trains of CCS operates as lead while the other train is in standby. The lead train is initiated remotely from the MCR. The chilled water supply is controlled by the temperature control valve of the CWE to achieve the target temperature for supply air from the operating CCS train. The standby train is started automatically with the lead if the bulk temperature in the Containment exceeds a specific setpoint. The standby train is also automatically started upon tripping of the lead train.

During plant operation, the Containment is inerted with dry nitrogen, thus no air is introduced into the containment atmosphere during power operation. Nitrogen gas drawn from the upper containment space and the CRD area is directed to the AHU coils to be cooled and discharged by the AHU. The nitrogen is then released into a supply air header and its related ducts, which are distributed throughout the Containment and the CRD area in order to maintain the containment temperature at target values.

During outages that require Containment entry, the HVS supplies outside air into the Containment through the CIS, Subsection 9.3.6, interface piping. The CCS AHUs recirculate the mixed air and distribute the air throughout the Containment to facilitate establishing and maintaining the oxygen percentage in the Containment, support heat removal, and maintain temperature for plant personnel entering the Containment.

The CCS AHUs and supporting chillers are provided with backup power from the standby diesel generators. The CCS remains operational during LOPP conditions to keep the containment environment from exceeding design conditions. It assists with containment cooldown following a LOPP during the period from hot shutdown to cold shutdown.

9.4.7.3 Safety Evaluation

Regulatory Requirements

The CCS complies with the following GDC, which has plant-wide applicability, as discussed in Section 3.1:

GDC 2 – Design bases for protection against natural phenomena

Additional information on GDC compliance is found in Section 3.1.

9.4.7.4 Inspection and Testing

CCS equipment and components are designed for inspection and maintenance accessibility. Before the initial startup of the CCS, fans are inspected to assure they are ready for the initial startup. Each fan is tested separately to confirm its rotation is in the correct direction and that there are no abnormal noises or vibrations. The cooling coils and related piping are hydrotested. Preoperational testing of the CCS is performed by the ITP.

9.4.7.5 Instrumentation and Control

Refer to Section 7.3 for a description of the DCIS functions:

- Temperature elements are provided at the fan discharge to monitor on-line performance of CCS and control CWE flow using CWE temperature control valve to maintain supply air temperature constant
- Pressure transmitters are installed on supply air ducts to monitor for variation in supply air flow that could be due to AHU fan failure, accumulation of solids in filters during an outage, or other causes
- Flow sensors are placed on the drain lines to measure any associated leak rates inside the Containment during normal operations
- Temporary flow indicators can be used during system testing and flow balancing

9.4.8 References

9.4-1 ASME AG-1-2019, “Code on Nuclear Air and Gas Treatment,” American Society of Mechanical Engineers, 2019.

9.4-2 AMCA Standard 500-D-18, “Laboratory Methods of Testing Dampers for Rating,” Air Movement and Control Association, 2018.

9.4-3 ANSI/ANS-56.7-1978, “American National Standard for Boiling Water Reactor Containment Ventilation Systems,” American National Standards Institute/American Nuclear Society, R1987.

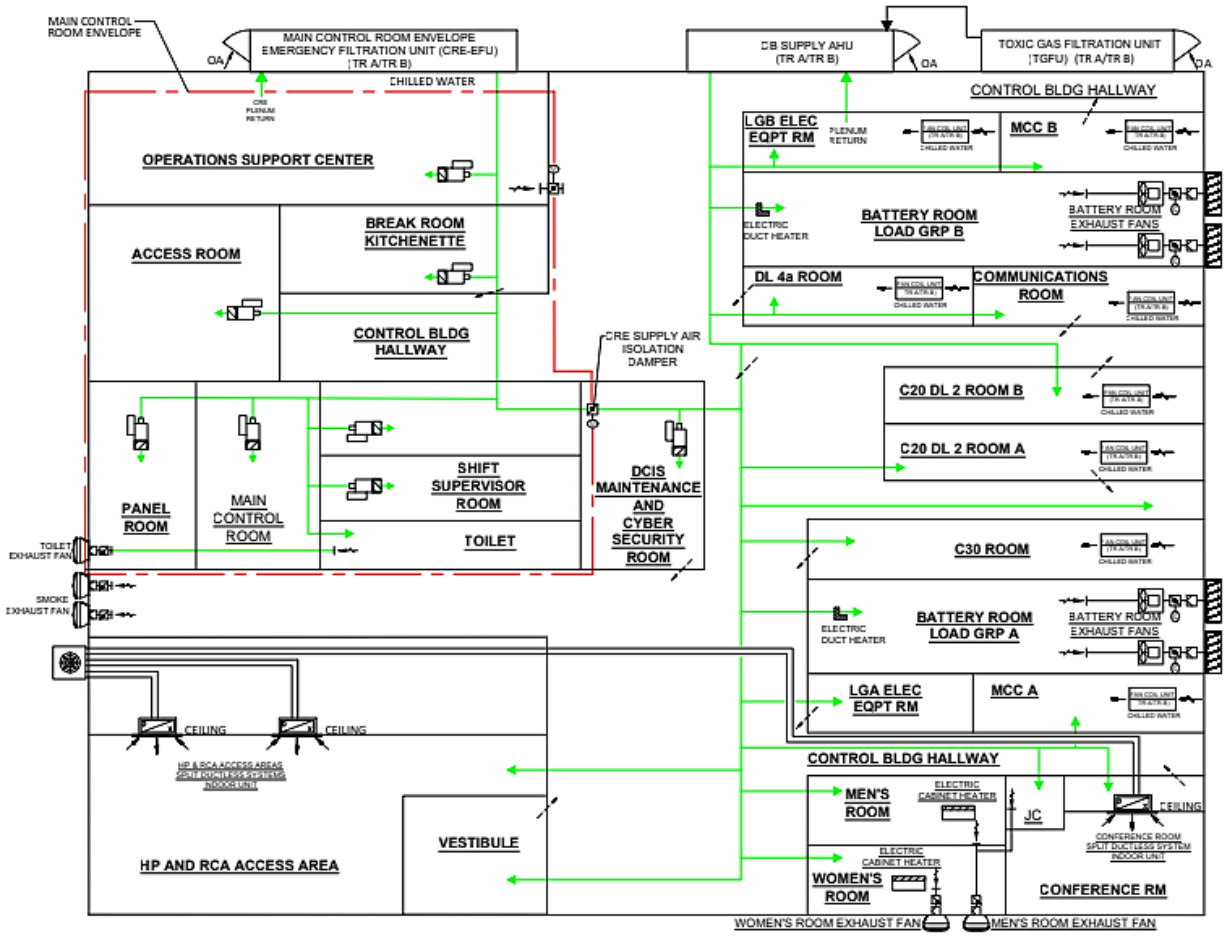


Figure 9.4-1: Control Building Heating, Ventilation, and Cooling Simplified Diagram

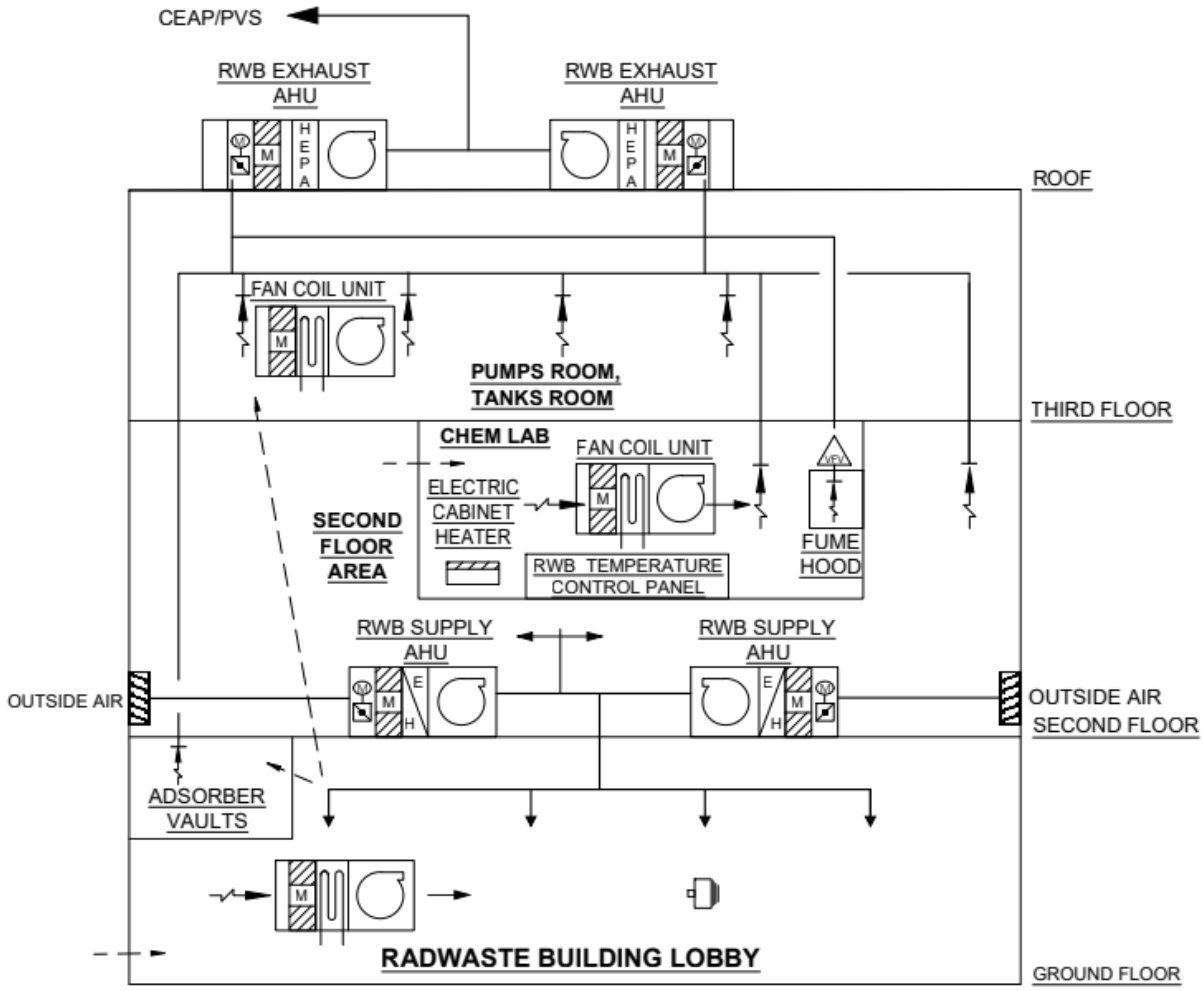


Figure 9.4-2: Radwaste Building Heating, Ventilation, and Cooling Simplified Diagram

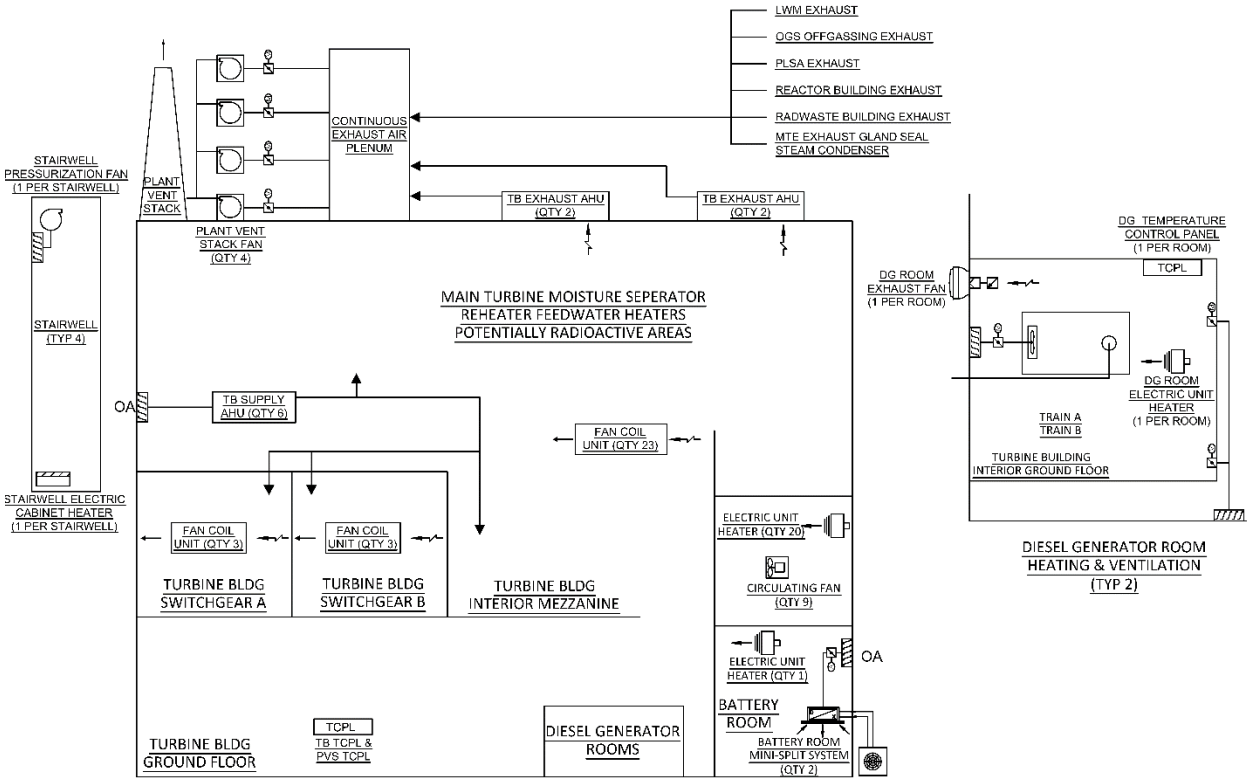


Figure 9.4-3: Turbine Building Heating Ventilation, and Cooling System and Plant Vent Stack Simplified Diagram

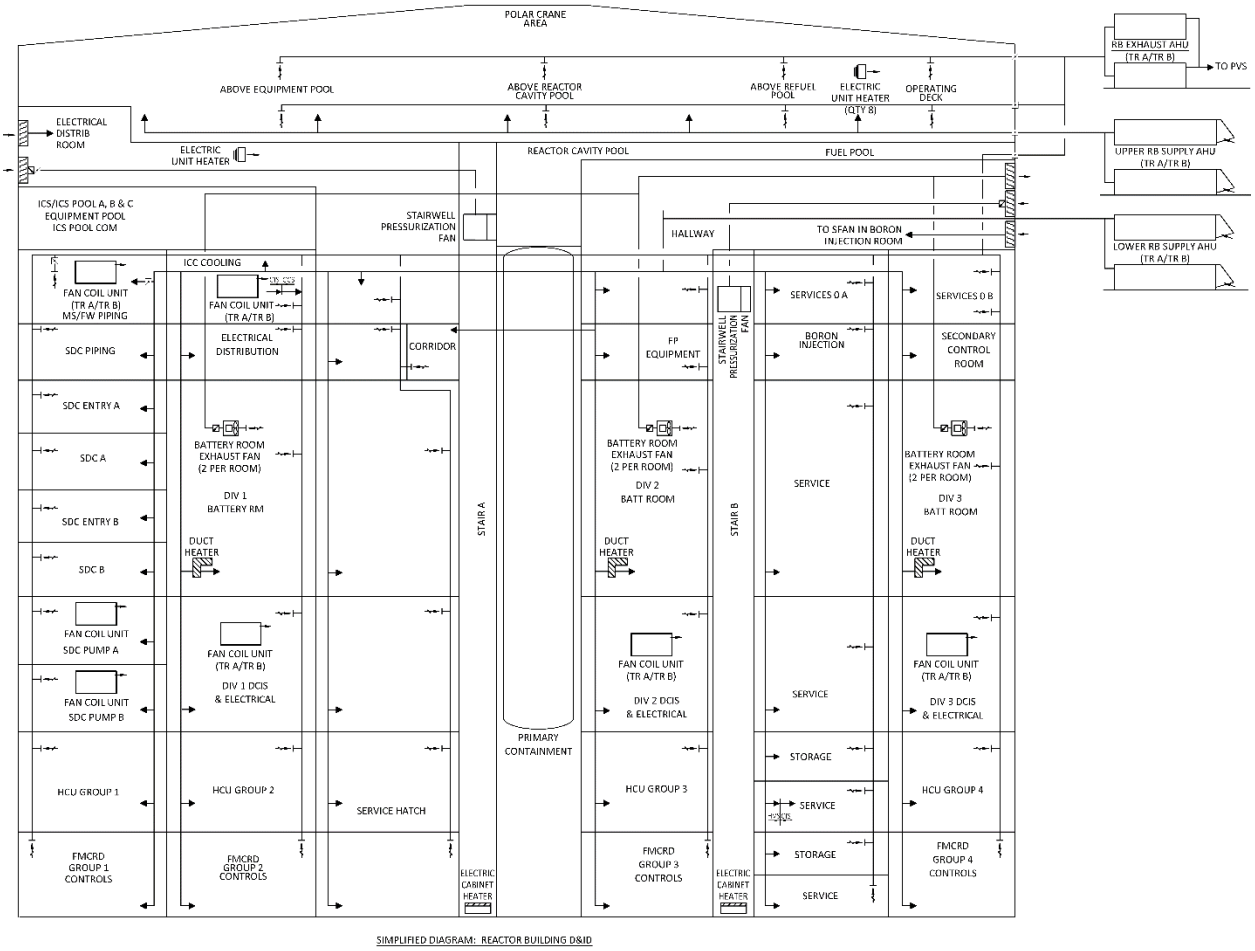


Figure 9.4-4: Reactor Building Heating, Ventilation, and Cooling Simplified Diagram

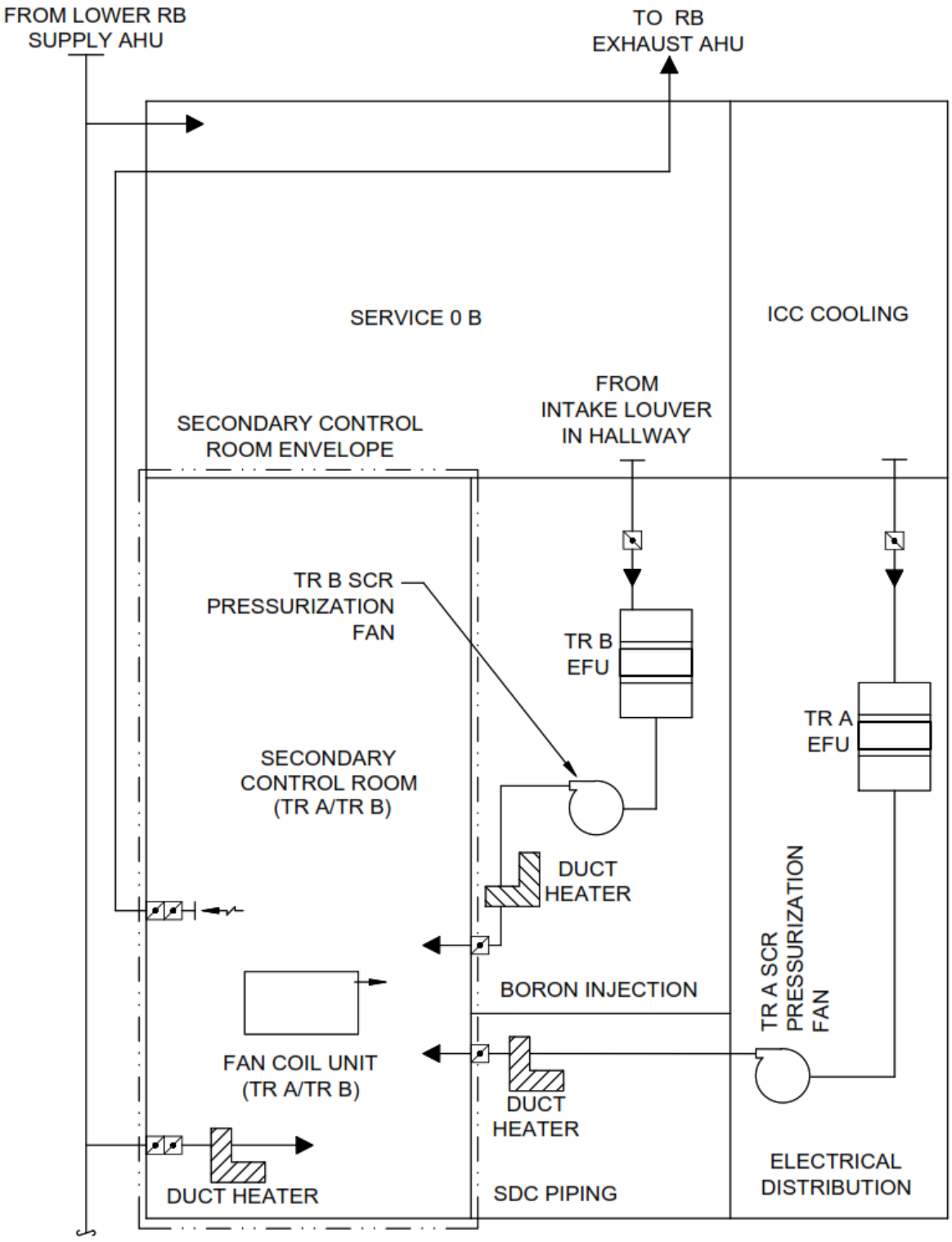


Figure 9.4-5: Secondary Control Room Heating, Ventilation, and Cooling Simplified Diagram

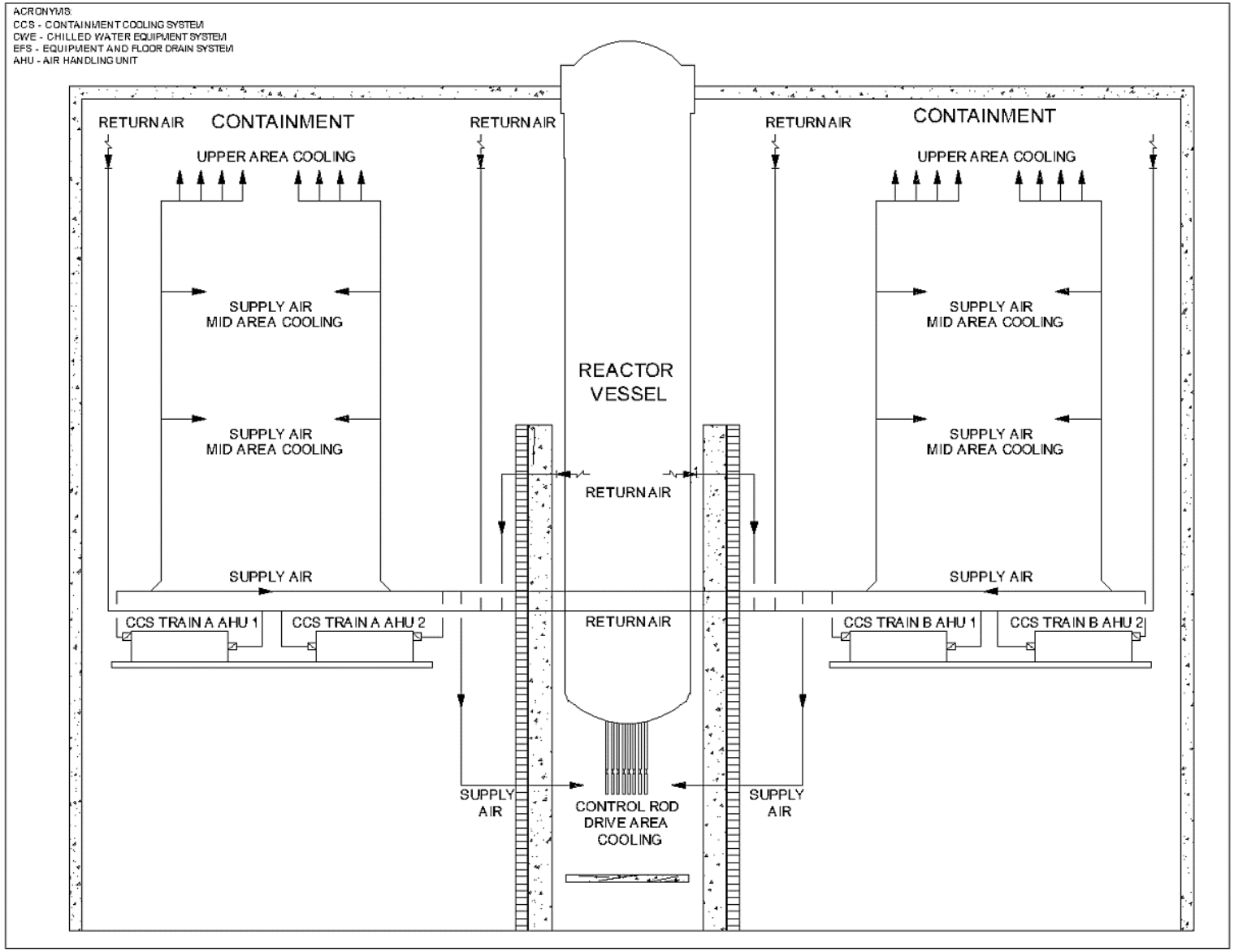


Figure 9.4-6: Containment Cooling System Simplified Diagram

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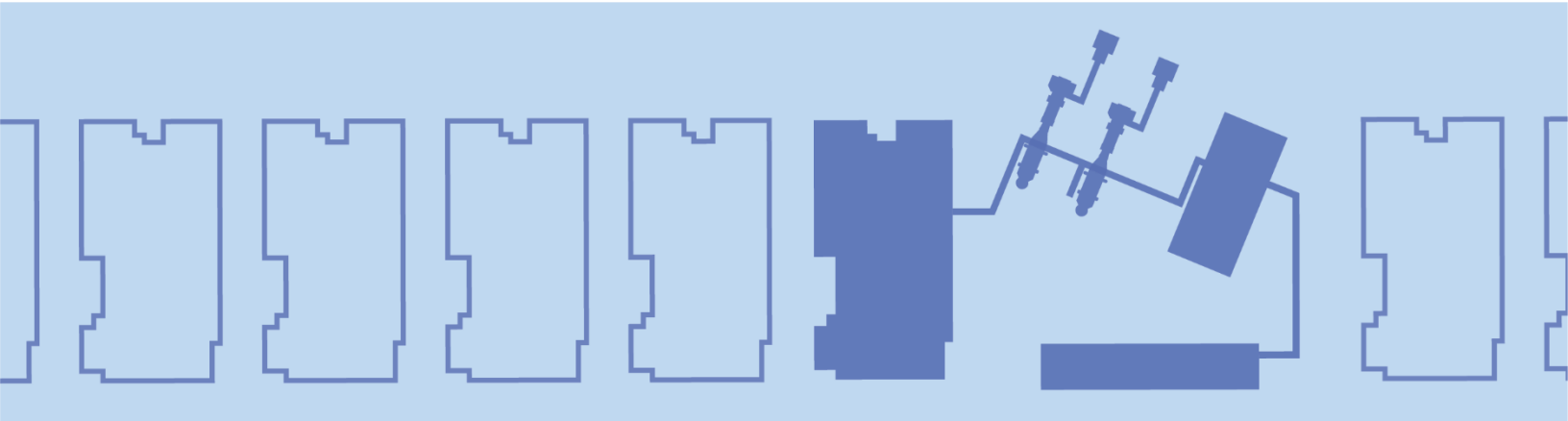
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Non-Proprietary Information

Chapter 9
Section 9.5 – Other Auxiliary Systems



ACRONYM LIST

Acronym	Explanation
AC	Alternating Current
AHU	Air Handling Unit
ASME	American Society of Mechanical Engineers
ASTM	American Society of Testing and Materials
BECPV	BWRX-300 Blue Energy Center Port of Victoria
CB	Control Building
CFR	Code of Federal Regulations
CHE	Cranes, Hoists, and Elevators
CIS	Containment Inerting System
CR	Control Room
CRE	Control Room Envelope
D-in-D	Defense-in-Depth
DCIS	Distributed Control and Information System
EFU	Emergency Filter Unit
FHA	Fire Hazards Analysis
FLEX	Diverse and Flexible Coping Strategies
FM	Factory Mutual
FPP	Fire Protection Program
FPS	Fire Protection System
FSAR	Final Safety Analysis Report
FSSA	Fire Safe Shutdown Analysis
GDC	General Design Criteria
HVS	Heating, Ventilation, and Cooling System
IEEE	Institute of Electrical and Electronics Engineers
ITP	Initial Test Program
LOPP	Loss-of-Preferred Power
MCR	Main Control Room
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NS	Non-Seismic
NUREG	Nuclear Regulatory Report

Acronym	Explanation
PDC	Principal Design Criterion
PSA	Probabilistic Safety Assessment
RB	Reactor Building
RG	Regulatory Guide
SC1	Safety Class 1
SCR	Secondary Control Room
SDG	Standby Diesel Generator
SSCs	Structures, Systems, and Components
SSE	Safe Shutdown Earthquake
TB	Turbine Building
TMI	Three Mile Island

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9.5 Other Auxiliary Systems

9.5.1 Fire Protection System

Fire Protection Program

The primary objectives of a Fire Protection Program (FPP) are to minimize both the probability of occurrence and the consequences of fire. The FPP consists of the integrated effort involving components, procedures, analyses, and personnel used in defining and performing activities of fire protection. It includes system and facility design, administrative controls, fire protection organization, inspection and maintenance, training and qualification, quality assurance, and testing.

Chapter 13 identifies the FPP as an Operational Program and identifies the schedule for its implementation.

The Fire Protection System (FPS) is part of the FPP which provides assurance through Defense-in-Depth (D-in-D), that the plant can achieve and maintain a safe shutdown condition in the event of a fire, and that radioactive releases to the environment in the event of a fire are limited. The FPS is described in this section, including the fire detection and fire suppression systems, in accordance with applicable codes and standards.

Appendix 9A presents the Fire Hazards Analysis (FHA) and Subsection 9A.3.2 describes the methodology for the Fire Safe Shutdown Analysis (FSSA). The FHA and FSSA will be completed for the Blue Energy Center – Port of Victoria Nuclear Unit 01 (BECPV-N01) prior to issuance of the Final Safety Analysis Report (FSAR).

Section 3.4 describes the internal flood analysis evaluating the effect of inadvertent actuation or breaks in the FPS water supply piping. Refer to Subsection 9.3.3 (Equipment and Floor Drain System) for removing fire water.

Section 9.4 describes the Heating, Ventilation, and Cooling System (HVS) design, as related to fire protection and smoke removal.

Subsection 9.5.2 presents the communication system design with respect to fire protection activities.

Subsection 9.5.3 presents the emergency lighting system design with respect to fire protection activities.

9.5.1.1 Design Bases

Safety Category 1 Functions

None.

Safety Category 2 Functions

None.

Safety Category 3 Functions

None.

Safety Category N Functions

- Detect and locate fires by fire area or fire zone and actuate alarms promptly (excluding the Containment)
- Provide automatic and manual fire suppression systems, where required by the FHA

- Provide manual backup to automatic fire suppression systems
- Limit the spread of fires
- Quickly suppress fires
- Ensure that at least one of the redundant divisions necessary to achieve fire safe shutdown is free of fire damage
- Limit radioactive or hazardous chemical exposure to personnel and their release to the environment as the result of a fire
- Provide firewater supply and distribution of firewater to suppression systems
- Provide fire extinguishers and smoke detection

The Safety Classifications, Quality Groups, and Seismic Categories of the FPS (Structures, Systems, and Components) SSCs are listed in Table 3A-1.

9.5.1.2 System Description

9.5.1.2.1 Fire Protection Program General Information

The FPS follows the applicable guidance of Regulatory Guide (RG) 1.189, “Fire Protection for Nuclear Power Plants,” and National Fire Protection Association (NFPA) 804, “Standard for Advanced Light Water Reactor Electric Generating Plants,” (Reference 9.5-1). Other applicable codes and standards are included in Table 9.5-1.

For cases where conflicting guidance is given in other codes or standards, the guidance from RG 1.189 prevails. Exceptions related to the design of the FPS are discussed in Subsection 9.5.1.9.

The FPP addresses aspects beyond the design and analysis of the FPS. RG 1.189, Sections 1.7, 1.8, 2.1, 2.2, 2.3, 2.4, 7.6, and 8.6 are not design related. Future implementation of the FPP as an Operational Program, as discussed in Chapter 13, will address compliance to RG 1.189 in these areas which are not design related.

9.5.1.2.2 Fire Protection System Design Features

Fire detection and fire suppression systems are provided and designed to quickly detect and extinguish or control fires. Fire separation or other measures are provided to limit the spread of fire and its effects.

9.5.1.2.3 Plant Arrangement

Building Compartmentalization

Compartmentalization is achieved by using passive fire barriers to subdivide the plant into separate areas. The walls, floors, and ceilings have fire resistance ratings of three-hours where required when an adjacent room contains equipment or systems necessary to ensure at least one fire safe shutdown path is available. This also includes fire doors, fire dampers, and penetration seals to provide separation of fire safe shutdown paths. Alternative fire safe shutdown features are provided where separation by three-hour barriers is not possible (e.g., main control room and secondary control room). The FHA defines the locations of fire areas and fire barriers. Fire areas can be sub-divided into fire zones. Equivalency evaluations may be required during detailed design (e.g., for a three-hour rated fire area boundary). These evaluations are performed by a fire protection engineer (assisted by others as needed) with justifications retained as part of the Fire Protection Report at the Operating License Application stage. Refer to Appendix 9A for a description of the FHA and FSSA content.

The plant layout provides means of access and egress to rooms and areas for manual firefighting and emergency escape. Stairwells serving as access or egress routes are enclosed within a barrier with a minimum fire rating of two hours.

Three-hour fire resistance ratings are also used where required based on high combustible loadings in the room (e.g., turbine lubrication oil storage).

Inside Containment

During normal operations, the Containment is inerted with nitrogen which provides insufficient oxygen to sustain combustion in the unlikely event that combustion initiation conditions occur. Subsection 9.3.6 describes the Containment Inerting System (CIS). Fire suppression is provided by hose stations just outside the Containment.

There are no reactor coolant pumps with associated lube oil systems located inside the Containment.

Main Control Room

The Main Control Room (MCR) complex is separated from the remainder of the building by a fire barrier having a fire rating of three hours.

The MCR is pressurized during normal conditions. With the detection of smoke in the Control Building (CB) HVS ducting (see Subsection 9.4.1), the smoke removal mode is actuated, and the dedicated smoke removal exhaust fans are turned on. If smoke is detected at the external Air Handling Unit (AHU) intake, the external AHU dampers close.

Secondary Control Room

The Secondary Control Room (SCR) is subject to the fire protection requirements per RG 1.189 for control rooms. The SCR is separated from the remainder of the Reactor Building (RB) by a fire barrier having a fire rating of three-hours. The SCR is pressurized under off-normal conditions in the event that the MCR is evacuated. Refer to Subsection 9.4.6 for details regarding the RB HVS which provides pressurization.

Electrical Rooms and Battery Rooms

Fire barrier rating, fire suppression type, and fire detection for these rooms are defined by the FHA. Electrical rooms and battery rooms are identified as requiring fire barriers, including:

- RB divisional electrical rooms
- RB divisional battery rooms
- CB electrical rooms
- Standby Diesel Generator (SDG) rooms

Battery rooms housing batteries that produce flammable off-gases have ventilation systems designed to limit concentrations of these gases. SDG rooms contain automatic fire suppression, and SDGs are separated from each other and from other areas of the plant by fire barriers that have a fire resistance rating of at least three-hours.

9.5.1.2.4 Fire Protection System

Figure 9.5-1 provides a simplified diagram of the FPS.

Water Source

The FPS includes two (2) firewater storage tanks. These tanks are designed in accordance with NFPA 22. The tanks meet NFPA 804 requirements for the total water demand for a period of 120 minutes (2 hours). The redundant storage tanks are physically separated.

Freeze protection is provided for the firewater storage tanks and exposed piping as necessary based on local climate conditions.

Fire Pumps

The fire pumps comply with applicable requirements of NFPA 20.

There are a total of three (3) firewater pumps: 2 electric and 1 diesel. The design flow rate is based on a turbine building fire, which is determined to be the bounding worst-case fire within the Power Block. A combination of 2 of the 3 pumps are capable of delivering the flow and pressure required to the location that has the most limiting combination of distance, line size and/or flow rate. The two electric fire pumps are in the fire-rated Fire Pump Enclosure, adjacent to the firewater storage tanks and near the Power Block. The diesel-driven fire pump is also in the Fire Pump Enclosure. The pumps are separated by a fire barrier having a fire rating of three-hours. The diesel-driven fire pump provides firewater in the event of failure of an electrically driven fire pump or Loss-of-Preferred Power (LOPP) as required.

The fuel oil day tank for a diesel-driven fire pump is designed in accordance with NFPA 20 and has a capacity sufficient to allow operation of the diesel engine for a minimum of 12 hours before refilling based on fuel consumption at rated pump capacity.

The electric pump is designed to start first followed by the second pump, if the first pump fails to start or extra capacity is needed. The third fire pump is designed to start in the event of a failure in either of the other two pumps. The fire pumps may be started manually from the Main Fire Alarm Panel in the MCR, from the SCR, or locally. The fire pumps can only be stopped manually.

The associated pressure maintenance (jockey) pump is used to maintain pressure in the system.

The diesel-driven fire pump engine meets the emissions requirements for the year that it is manufactured.

The Standby Power System (see Chapter 8) provides power to the motor-driven fire pumps. Remote signals of fire pump status are provided in the MCR and SCR in accordance with NFPA 20.

9.5.1.2.5 Firewater Main

The fire water supply piping consists of a buried yard main loop that feeds supplemental loops in the Power Block. Check valves are provided between each building connection from the main yard piping loop.

The yard main consists of buried Class 200 high-density polyethylene piping or other code compliant pipe material in accordance with NFPA 24.

The fire water supply piping located within Power Block structures consists of carbon steel and is in accordance with ASME B31.1 for the RB. Building isolation valves are provided to minimize loss of fire protection due to maintenance activities.

Fire Hydrants

Fire hydrants are located at approximately 250 ft intervals along the yard main loop to provide firefighting capability. The fire hydrants are located no closer than 40 ft from the buildings and structures protected by the hydrants.

9.5.1.2.6 Fire Suppression Systems

The FPS utilizes a combination of both automatic and manual fire suppression systems which are described below. The FHA determines the method of fire suppression for each fire area.

Fixed Automatic Water Extinguishing Systems

The sprinkler systems are designed and installed to meet the requirements of NFPA 13. The deluge systems are designed and installed to meet the requirements of NFPA 15. Additionally, fixed suppression piping located within the RB is designed and installed in accordance with ASME B31.1. Isolation valves, which, if closed, would prevent fixed suppression system operation, have tamper switches to monitor their normal position as required by NFPA 13. Fixed suppression systems have provisions to properly drain parts of the systems.

Wet Pipe Sprinkler System

Automatic sprinklers provide protection for the areas identified in the FHA.

Each wet pipe sprinkler system consists of an alarm check valve, thermally actuated closed-head sprinklers, and attached piping network containing water under pressure. Water discharges immediately from sprinklers opened by heat from a fire.

Preaction Sprinkler System

Preaction sprinkler systems are provided for plant areas where it is necessary to prevent serious water damage to SSCs, or contents that could occur with other sprinkler systems as a result of leaking automatic sprinkler heads, pipe breaks, or inadvertent actuation. Each preaction sprinkler system consists of a deluge valve, means to contain preaction air (such as a preaction check valve), thermally actuated closed-head sprinklers or spray nozzles, and attached piping containing air under pressure (the air pressure serves to monitor the integrity of the piping and provides a supervisory alarm in the event of trouble).

Deluge System

Deluge systems are provided as dictated by the FHA. Each deluge system consists of a deluge valve, open-head sprinklers or nozzles, and a dry piping network which is connected to the firewater supply.

Manual Suppression Means

Building standpipes and hose stations are provided in major buildings inside the protected area fence. The standpipes and hose stations are designed to NFPA 14 Class III Service. Rooms located within the protected area fence are within the reach of at least one effective hose stream from a Class III hose station unless evaluated as acceptable under the FHA.

Effective hose streams from two separate hose stations cover each room that contains SSCs required for fire safe shutdown that is not protected by a fixed fire suppression system. The need for two-hose-station coverage is also based upon the fire hazards present. Rooms not covered by a fixed fire suppression system and with coverage by only one hose station are furnished with portable fire extinguishers for secondary coverage.

Standpipes and hose connections in the RB are designed to ensure failure of the piping will not impact Safety Class 1 (SC1) equipment following a Safe Shutdown Earthquake (SSE). The piping in the RB is designed and installed in accordance with ASME B31.1. Drain valves are provided at the bottom of each standpipe to allow for drainage.

Hose stations are located outside of highly radioactive areas where possible. RB standpipes and hose stations are external to the Containment for fire protection during refueling and maintenance operations in the Containment. Portable fire extinguishers provide secondary coverage.

Fire Extinguishers

Areas of the plant are protected by portable fire extinguishers in accordance with NFPA 10, except in highly radioactive areas. Portable fire extinguishers are readily available outside the highly radioactive areas. The extinguishers will be listed (e.g., Underwriters Laboratories) for their intended use.

9.5.1.2.7 Floor Drains

Floor drains provide a means of removing fire water that is used to suppress a fire. Subsection 3.4.1 discusses flood protection for SSCs required to perform a safety function. Subsection 9.3.3 describes the Equipment and Floor Drain System.

9.5.1.2.8 Fire Detection, Instrumentation, and Fire Alarm System

Fire Detection System

Areas that contain SC1 SSCs or significant combustible loading according to the FHA have fire detection (except for Containment which is a normally inerted space) that alarms in the MCR.

The design of the fire detection system complies with the requirements of NFPA 72.

The type of detection used, and the location of the detectors are suitable for the type of fire hazard identified by the FHA.

Smoke detection in ventilation systems is provided to meet NFPA 90A/92 requirements, where applicable. Smoke detection for the elevator shaft is provided to meet ASME A17.1 requirements.

The Containment is inerted during normal operation. Portable detection equipment and fire watches, as required, are used inside the Containment during outages when the space is not inerted.

Primary and secondary power supplies exist for the fire detection and alarm system.

Instrumentation

Each fire suppression system automatically actuated by a fire detection system has the control logic and the capability for manual actuation available at the local fire panel for the protected room. Automatic sprinkler systems which do not require separate detection systems for actuation are generally not equipped with manual actuation means. Instrumentation for fire detection or automatically actuated fire suppression systems is either FM Approved or Underwriters Laboratories listed or both.

There are three primary types of FPS instrumentation:

1. Instrumentation supporting fire detection
2. Instrumentation supporting automatic suppression systems
3. Instrumentation supporting firewater delivery

Instrumentation for the fire detection system provides early detection and warning alarms of fires. Each room containing SC1 SSCs has a fire detection system comprised of multiple detectors, except for Containment which is a normally inerted space. Heat and smoke detectors are supervised by local fire panels per NFPA 72 guidelines.

Instrumentation for fixed fire suppression systems provide local and remote monitoring capability for the suppression system status.

Instrumentation supporting firewater delivery provides status indication in the MCR and SCR for low water level in tanks, power failure to fire pump motors or their controllers, running of fire pumps, fire pump trouble, low temperature in the Fire Pump Enclosure or tanks, abnormal position of isolation valves, and miscellaneous supervisory signals for other trouble indicators (e.g., abnormally high or low water pressures or failure of compressed air).

Fire Alarm System

A fire alarm system is provided and the need for it to include integrated, supervised, one-way communications within the protected area will be determined as part of the final design and addressed in the FSAR. Fire signals are distinctive and not easily confused with other alarm signals. Accessible spaces, except for the MCR and SCR, are equipped with audible and/or visual fire alarm signal devices, as required by the FHA.

Manual fire alarm station (pull box station) locations are determined by the FHA.

A Main Fire Alarm Panel is directly connected to the MCR panels to display common fire alarm, supervisory, and trouble conditions. A Secondary Fire Alarm Panel is also available, and it meets the same requirements. Refer to Section 7.3 for Distributed Control and Information System (DCIS) gateway interface to the FPS system panel.

9.5.1.2.9 Heating, Ventilation, and Cooling System Interface and Smoke Control

The FPS provides fire suppression for activated carbon beds and charcoal filters within ventilation systems. Refer to the:

- MCR toxic gas filtration units (Subsection 9.4.1)
- MCR and SCR emergency filtration units (Subsection 9.4.1 and Subsection 9.4.6)

FPS smoke detectors are mounted within HVS ductwork to detect smoke in various fire areas and transmit an alarm signal to the local fire control panel. Refer to Subsection 9.5.1.2.8 for fire detection.

The HVS removes smoke from fire areas, as required by the FHA. Refer to Subsection 9.5.1.3 for an operational description of the FPS in Fire Mode. Stairwell pressurization is provided in Power Block structures in accordance with NFPA 804 guidelines for safe egress routes, as required by the FHA. Additionally, the MCR and SCR are pressurized by the HVS, as smoke management is required in the control rooms. Refer to Subsection 9.5.1.2.3.

9.5.1.3 System Operation

The FPS operates continuously and reliably during plant modes. The following paragraphs describe the operation of the system.

Normal Standby Mode

Local fire panels continuously monitor suppression system status and fire detection system circuits. The local fire panels include initiating and indicating circuits. The FPS is maintained in a standby condition and is ready to function during the modes of plant operation, transients, and shutdown periods. The fire detection and alarm system detect and annunciate fire conditions. A local control panel and the Main Fire Alarm Panel annunciate and indicate fire conditions and location. If the Main Fire Alarm Panel is out-of-service, the Secondary Fire Alarm Panel located in the SCR performs the same function. The FPS is automatically activated by fire detection devices. Manually initiated systems are operated upon detection and confirmation of fire conditions. Hose

stations and portable fire extinguishers are available to provide secondary coverage of fixed fire suppression systems as determined by the FHA.

Fire Mode

Upon indication of a fire from any fire detector, the local fire panel activates a visual and audible fire alarm and annunciates the location of the fire. Fire alarms are transmitted to the Main Fire Alarm Panel and to remote annunciation panels via a dedicated FPS optical fiber cable. The Main Fire Alarm Panel interfaces directly with the MCR panels to display common fire alarm, supervisory, and trouble conditions. If the Main Fire Alarm Panel is out-of-service, the Secondary Fire Alarm Panel performs the same function within the SCR. Receipt of initiating signals from fire detectors or activation of an automatic sprinkler actuates the automatic suppression system in the room. A manual fire suppression system is used where automatic suppression systems are not provided per the FHA.

Smoke removal from areas containing SSCs required for fire safe shutdown is provided by the HVS system operating in smoke removal mode as required by the FHA. An HVS mode pressurizes fire-rated stairwells as required by the FHA.

9.5.1.4 Instrumentation and Control

The fire alarm system instrumentation used for fire detection and actuation of systems is described in Subsection 9.5.1.2.8.

9.5.1.5 Quality Assurance

The quality assurance requirements, under which the FPP is executed, are addressed in Chapter 17.

9.5.1.6 Safety Evaluation

The FPS complies with the following General Design Criteria (GDC), which has plant-wide applicability:

GDC 3 – Fire Protection

Additional information on GDC compliance is found in Section 3.1.

The FPS is designed to comply with the following regulations:

The FPS includes Title 10 of the Code of Federal Regulations (10 CFR) Section 50.48 in its design bases. Specifically, this includes applicable aspects of clauses (a)(1) and (a)(2) for design features that minimize the probability and effects of fires and require compliance with GDC 3. The fire suppression system is described in Subsection 9.5.1.2.6, and the fire detection system is described in Subsection 9.5.1.2.8. The methodologies for the FHA and the FSSA are discussed in Appendix 9A.

Regulatory Guidance

The FPS is designed to comply with the following regulatory guidance:

- RG 1.189 – “Fire Protection for Nuclear Power Plants,” Revision 5, October 2023

Table 9.5-1 lists the codes, standards and guidelines used in the fire protection FPS design. The list is preliminary, and not all inclusive. Additional design codes and standards are referenced within the items listed in Table 9.5-1.

9.5.1.6.1 Fire Hazards Analysis

Refer to Appendix 9A for a discussion of the FHA.

9.5.1.7 Inspection and Testing

The FPS is inspected and tested to meet codes and standards and NFPA requirements. Chapter 14 describes the Initial Test Program.

The following NFPA tests, which are required by code, are performed as a minimum part of the inspection and maintenance requirements, where applicable, for the system:

1. Waterflow alarm tests of sprinkler systems – NFPA 25
2. Deluge waterflow alarm and operational tests – NFPA 25
3. Fire detection and alarm system tests – NFPA 72
4. Smoke removal system/fire damper/smoke dampers – NFPA 92
5. Flow testing of fire pumps – NFPA 25

FPS SSCs are designed for inspection and maintenance during plant operation.

9.5.1.8 Personnel Qualification and Training

The fire protection program organizational structure, along with personnel training and qualifications will be addressed in the FSAR. Chapter 13 identifies the FPP as an Operational Program and its implementation schedule.

9.5.1.9 Exceptions and Alternative Requirements

The design for the BWRX-300 takes exception to RG 1.189 Position 3.2.1(j) which states that "Provisions should be made to supply water to at least two standpipes and hose connections for manual firefighting in areas containing equipment required for safe plant shutdown in the event of a safe-shutdown earthquake. The piping system serving such hose stations should be analyzed for safe-shutdown earthquake loading and should be provided with supports to ensure system pressure integrity."

FPS SSCs are primarily Seismic Category Non-Seismic (NS) and therefore meet seismic requirements of local building codes. The standpipe system piping in the RB up to and including the isolation valves supplying the sprinklers conform to ASME B31.1 and NFPA 13. The RB standpipe system piping is designed to ensure failure of the piping will not impact SC1 equipment following an SSE.

As justification, the BWRX-300 design achieves the fire protection goals by providing the ability to safely shut down the plant in the event of an SSE and a subsequent fire in the RB. The BWRX-300 achieves this through the use fire barriers having a fire rating of three-hours between fire safe shutdown SSC paths. Three-hour separation of the SSCs necessary to safely shutdown the reactor and maintain decay heat removal means the use of sprinklers or manual firefighting is not necessary to achieve safe shutdown. This will be demonstrated in the FSSA where the worst-case postulated fire is assumed to completely burn out the designated compartment without the application of fire suppression water.

The designation of the water supply to the standpipe system as Seismic Category NS does not adversely affect the ability of the plant to safely shutdown. The design of the standpipes in the RB ensures that the piping does not adversely affect SSCs required for fire safe shutdown.

Additionally, conformance to requirements of RG 1.189 not related to FPS design will be evaluated in the FSAR. Refer to Subsection 9.5.1.2.1 for conformance with RG 1.189 Regulatory Positions.

9.5.2 Non-Distributed Control and Information Communication System

9.5.2.1 Design Bases

Safety Category 1 Functions

None.

Safety Category 2 Functions

None.

Safety Category 3 Functions

- Support the coordination between plant personnel during accident or incident conditions, including transients, fire, security events, and accident conditions including LOPP events, under maximum potential noise levels.

Safety Category N Functions

- Intra-plant communications and plant-to-offsite communication during normal operation.

The Safety Classifications, Seismic Categories, and Quality Groups of Non-DCIS Communication System SSCs are listed in Table 3A-1.

9.5.2.2 System Description

The Non-DCIS Communication System consists of the following subsystems:

- Private branch exchange system
- Plant public address system
- Sound-powered system
- Plant radio system
- Security communication system

Designing for human factors is considered in the Non-DCIS Communication System and the applicable evaluation will be provided in the FSAR. Site-specific requirements determine the extent of communications subsystems required.

The Non-DCIS Communication System is designed to meet emergency plan requirements for accident conditions, including notification of personnel and implementation of evacuation procedures. This capability includes communications support to both onsite and offsite emergency centers. This capability includes at least one onsite and one offsite communications system, each with a backup power source.

The Non-DCIS Communication System is designed to support the coordination between plant personnel during accident or incident conditions under maximum potential noise levels. This capability also includes communications support for firefighting, including support of alternative and dedicated shutdown capabilities. Refer to Subsection 9.5.1 for information related to fire protection.

The Non-DCIS Communication System is designed to support coordination of security activities both within the plant and with external security and law enforcement organizations.

The Non-DCIS Communication System is designed to be sufficiently redundant and separated from one another, such that a failure in one system does not result in the inability to perform the required system function.

Key components of the Non-DCIS Communication System are designed to remain functional during a LOPP event.

Portions of the Non-DCIS Communication System are operated on battery-backed 8-hour uninterruptible power supplies.

9.5.2.3 Safety Evaluation

The following Principal Design Criterion (PDC) has aspects specific to the Non-DCIS Communication System and compliance is addressed as follows:

PDC 19 – Control room. The Non-DCIS Communication System does not contain any SC1 SSCs relied on for safe shutdown. Refer to Subsection 3.1.2.10 for evaluation against PDC 19.

Additional information on PDC compliance is found in Section 3.1.

The Non-DCIS Communication System considers the relevant requirements of the following regulations:

The design of the Non-DCIS Communication System will comply with the requirements in 10 CFR 50.160 and this will be confirmed in the FSAR. Provisions are made for emergency facilities and equipment, which include at least one onsite and one offsite communications system, with a backup power source. The Private Branch Exchange System, Public Address System, Radio System, and Security Communication System provide onsite communications capability. The Private Branch Exchange System, Radio System and Security Communication System are capable of providing offsite communications. These subsystems have a backup power source.

The design of the Non-DCIS Communication System considered 10 CFR 73.45(g)(4)(i) and 10 CFR 73.46(f), as it relates to communication networks and fixed site physical protection systems, subsystems, components, and procedures. The application of any communications systems for security purposes is to be described in the physical security plan which implements the requirements of 10 CFR 73.55. The schedule for implementation of the Physical Security Program is discussed in Chapter 13.

Refer to Section 13.3 for information pertaining to emergency planning.

Refer to Section 13.6 for the design of the Non-DCIS Communication System as it relates to performance capabilities for fixed site physical protection systems, and the application of the Non-DCIS Communication System for security purposes.

9.5.2.4 Inspection and Testing

Subsystems of the Non-DCIS Communication System required during emergency situations are tested at periodic intervals to demonstrate functionality when required.

9.5.3 Lighting

9.5.3.1 Design Bases

Safety Category 1 Functions

None.

Safety Category 2 Functions

- Provide lighting for both the MCR and SCR at acceptable levels of illumination.
- Provide emergency lighting at acceptable levels of illumination when the normal lighting is not available. Emergency lighting is to be available in areas where emergency operations are performed, including the access and egress routes to and from those areas, and areas used for fire response and safe shutdown.

Safety Category 3 Functions

None.

Safety Category N Functions

- Provide lighting throughout the plant at acceptable levels of illumination.

The Safety Classifications, Seismic Categories, and Quality Groups of Lighting SSCs are listed in Table 3A-1.

9.5.3.2 Lighting Description

The lighting is comprised of the following subsets of equipment:

- Normal lighting – Normal lighting provides illumination throughout the plant. Normal lighting is available under normal plant operating, maintenance, and testing conditions. Equipment used for normal lighting alone does not have a backup power source.
- Emergency lighting – Emergency lighting provides acceptable levels of illumination when normal lighting is not available. Emergency lighting is available in areas where emergency operations are performed, including the access and egress routes to and from those areas, in response to a fire, and safe shutdown. All emergency lighting derives power from a source backed by standby power. Lighting supporting egress and safe shutdown is maintained by standby power during LOPP and includes or is supplemented by self-contained battery pack units that provide power for sufficient lighting during a station blackout. The self-contained battery pack units are inspected and tested in accordance with manufacturer recommendations. For safe shutdown, access, and egress to safe shutdown areas, self-contained 8-hour battery pack units are provided for general egress, and life safety purposes to meet building codes. Self-contained 90-minute battery pack units are provided.
- Control room lighting – Includes lighting for both the MCR and SCR. Control room lighting equipment provides acceptable illumination under normal and emergency conditions. Power is derived from two divisions of the Emergency Power System which are backed by 72-hour seismically qualified batteries. Lighting supplied by a single division provides acceptable illumination. SCR lighting is seismically qualified to ensure sufficient lighting is available for post-accident monitoring following a seismic event.

The plant uses light emitting diodes operated from low voltage power or lighting control panels. These power panels are powered by the following sources:

- Preferred Power System: Normal Lighting
- Standby Power System: Emergency Lighting
- Uninterruptible Power Systems: MCR and SCR Lighting

Power panels are distributed as needed such that room lighting and service power can be easily obtained. Refer to Chapter 8 for additional information pertaining to power panels.

Lighting components are designed to remain functional during design basis events to which they are credited. Lighting fixtures located in the vicinity of equipment credited in responding to a seismic event, are supported so that they do not adversely affect the equipment when subjected to seismic loading of an SSE.

9.5.3.3 Safety Evaluation

Regulatory Requirements

There are no regulatory requirements that directly apply to the plant lighting systems.

Regulatory Guidance

Plant lighting is designed to comply with Nuclear Regulatory Report (NUREG)-0700, as it relates to lighting level recommendations.

Lighting is not essential for reactor shutdown, containment isolation, or containment and reactor heat removal. Lighting is not essential in preventing significant release of radioactive material to the environment. Failure of normal and emergency lighting does not compromise automatic actuation of safety class components or systems that perform a Safety Category function.

Emergency lighting and control room lighting provide illumination for operations during emergencies and anticipated operational occurrences.

Emergency lighting supporting egress and safe shutdown is maintained by standby power during LOPP and includes or is supplemented by self-contained battery pack units that provide power for sufficient lighting during egress.

Power associated with the MCR and SCR is derived from two divisions of the Emergency Power System which are backed by 72-hour seismically qualified batteries. SSCs in the vicinity of the batteries are designed such that, seismic event will not reduce the functional reliability of the batteries to perform their Safety Category function to unacceptable safety levels.

9.5.3.4 Inspection and Testing

Preoperational testing of lighting verifies that the normal lighting system provides illumination under normal plant operating, maintenance, and testing conditions, and that the emergency lighting system provides illumination throughout the station, including areas where emergency operations are performed. The Initial Test Program (ITP) is described in Chapter 14, and testing descriptions will be provided in the FSAR.

The Alternating Current (AC) lighting circuits are normally energized and require no periodic testing.

9.5.3.5 Instrumentation and Control

No special instrumentation is required for lighting. The emergency lighting automatically actuates on the loss of normal plant lighting.

9.5.4 Standby Diesel Generator Fuel Oil Storage and Transfer System

9.5.4.1 Design Bases

Safety Category 1 Functions

None

Safety Category 2 Functions

None

Safety Category 3 Functions

None

Safety Category N Functions

The SDG Fuel Oil Storage and Transfer System provides bulk fuel oil storage and transfer to the SDG day tanks.

The Safety Classification, Quality Groups, and Seismic Categories of SSCs are listed in Table 3A-1.

9.5.4.2 System Description

The SDGs provide electrical power to the Standby Power System as described in Chapter 8. The two SDGs are commercially available, self-contained skid-mounted power packages.

The SDG Fuel Oil Storage and Transfer System is a separate equipment and piping system used to supply diesel fuel to the SDGs by providing for storage and transfer of diesel fuel oil to the day tanks of the SDGs.

Figure 9.5-2 provides a simplified diagram of the Diesel Fuel Oil Storage and Transfer System.

The SDG Fuel Oil Storage and Transfer System includes one diesel fuel oil storage tank, and two fully redundant diesel fuel oil transfer pumps. The system also includes all necessary instrumentation, piping, and valves. The fuel oil storage tank is designed with features to prevent freezing, as needed, and has a recirculation line from the transfer pumps.

The SDG fuel oil storage tank will be designed to satisfy the requirements of API-650, “Welded Steel Tanks for Oil Storage” (Reference 9.5-2). The tank will be double-walled or have a means to contain a leak, if one develops.

Diesel fuel oil transfer pumps move diesel fuel oil from the storage tank to the day tanks, and provide recirculation, as appropriate. The day tanks are double walled to mitigate potential fuel oil spills. Underground fuel oil piping will be designed to allow for leak detection.

The SDG fuel oil storage tank and transfer pumps are located outside of the protected area. This increases the distance between the combustible fuel oil tank and portions of the plant that require protection from such hazards.

9.5.4.3 Safety Evaluation

Regulatory Requirements

The SDG Fuel Oil Storage and Transfer System complies with the following GDC, which has plant-wide applicability, as discussed in Section 3.1:

GDC 5 – Sharing of structures, systems, and components.

The Fuel Oil Storage and Transfer System is a supporting mechanical system for the SDGs. Refer to Chapter 8 for compliance with Principal Design Criterion (PDC) 17.

GDC 2 – Design bases for protection against natural phenomena. The SDG Fuel Oil Storage and Transfer system is a support system for the SDGs which are classified as SC3 (see Table 3A-1). Failure of the system does not impact SC1 SSCs. Equipment and piping layout will be provided in the FSAR. No SDG Fuel Oil Storage and Transfer system requirements are implemented for this GDC.

GDC 4 – Environmental and dynamic effects design bases. The SDG Fuel Oil Storage and Transfer system is a support system for the SDGs which are classified as SC3 (see Table 3A-1). Failure of the system does not impact SC1 SSCs. Equipment and piping layout will be provided in the FSAR. No SDG Fuel Oil Storage and Transfer system requirements are implemented for this GDC.

Additional information on GDC compliance is found in Section 3.1.

9.5.5 Standby Diesel Generator Cooling Water System

The SDG is a packaged system, and details will be provided in the FSAR. The SDGs are radiator cooled and do not require additional mechanical support for cooling such as an auxiliary cooling water system.

9.5.6 Standby Diesel Generator Starting System

The SDG is a packaged system, and details will be provided in the FSAR. The SDGs are electric start, and do not require additional mechanical support for starting such as an air system. SDG starting time is not material to it performing its safety category functions.

The SDGs may be started from the MCR, SCR or a local panel.

9.5.7 Standby Diesel Generator Lubrication System

The SDG is a packaged system, and details will be provided in the FSAR. Once started, the SDGs operate without requiring plant services other than those required to store and transfer fuel, with the capability of continuous operation for at least one week.

9.5.8 Standby Diesel Generator Combustion Air Intake and Exhaust System

The SDG is a packaged system, and details will be provided in the FSAR. Air is supplied from intakes located outside the building containing the SDGs.

The SDG rooms have separate exhausts for radiator heat removal and combustion air removal. The exhaust for combustion air is routed to a safe discharge location. Intake and exhaust design details including discharge locations will be provided in the FSAR.

Refer to Section 9.4 and Figure 9.4-3 for additional information on the HVS.

9.5.9 Cranes, Hoists, and Elevators

Cranes, Hoists, and Elevators (CHE) are located throughout the plant and provide the means to lift and lower equipment and materials and move them horizontally along safe load paths. The CHE provides a safe and effective means for transporting loads.

Refer to Subsection 9.1.4 for the Light Load Handling System, including equipment needed for handling of new and spent fuel.

Refer to Subsection 9.1.5 for a description of the Overhead Heavy Load Handling System (OHLHS) and overview of the OHLHS program requirements. The RB polar crane is also described in Subsection 9.1.5.

This subsection addresses the TB Overhead Bridge Crane and various equipment that does not handle fuel. Details will be provided in the FSAR.

9.5.9.1 Design Bases

Excluding the Light Load Handling System and the Overhead Heavy Load Handling System in the RB described in Section 9.1, the CHE does not support or provide Safety Category functions and as such has no safety design bases. As required per RG 1.244, CHE equipment is to be evaluated against OHLHS program requirements discussed in Subsection 9.1.5 for consequences of a potential load handling event considering properties of the lifted load and SSCs in the vicinity of the travel path.

9.5.10 References

9.5-1 NFPA 804, “Standard for Fire Protection for Advanced Light Water Reactor Electric Generating Plants,” National Fire Protection Associations, 2020.

9.5-2 API 650, “Welded Tanks for Oil Storage,” American Petroleum Institute, Thirteenth Edition, 2021.

Table 9.5-1: List of Applicable Codes, Standards and Regulatory Guidance for Fire Protection

Item	Applicable Codes, Standards, and Regulatory Guidance
1	American Society of Mechanical Engineers, ASME B31.1, “Power Piping,” 2022.
2	National Fire Protection Association, NFPA 10, Standard for Portable Fire Extinguishers, 2018.
3	National Fire Protection Association, NFPA 13, Standard for the Installation of Sprinkler Systems, 2022.
4	National Fire Protection Association, NFPA 14, Standard for the Installation of Standpipe and Hose Systems, 2024.
5	National Fire Protection Association, NFPA 15, Standard for Water Spray Fixed Systems for Fire Protection, 2022.
6	National Fire Protection Association, NFPA 20, Standard for the Installation of Stationary Pumps for Fire Protection, 2022.
7	National Fire Protection Association, NFPA 22, Standard for Water Tanks for Private Fire Protection, 2023.
8	National Fire Protection Association, NFPA 24, Standard for Installation of Private Fire Service Mains and Their Appurtenances, 2022.
9	National Fire Protection Association, NFPA 25, Standard for the Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems, 2023.
10	National Fire Protection Association, NFPA 72, National Fire Alarm and Signaling Code, 2022.
11	National Fire Protection Association, NFPA 90A, Standard for the Installation of Air-Conditioning and Ventilating Systems, 2024.
12	National Fire Protection Association, NFPA 92, Standard for Smoke Control Systems, 2024.
13	National Fire Protection Association, NFPA 804, Standard for Fire Protection for Advanced Light Water Reactor Electric Generating Plants, 2020.
14	Nuclear Energy Institute (NEI) 00-01, Guidance for Post Fire Safe Shutdown Circuit Analysis, Revision 4, December 2019.
15	Regulatory Guide 1.189 – Fire Protection for Nuclear Power Plants, Revision 5, October 2023.
16	American Society of Mechanical Engineers, ASME A17.1, “Handbook on Safety Code for Elevators and Escalators,” 2022.

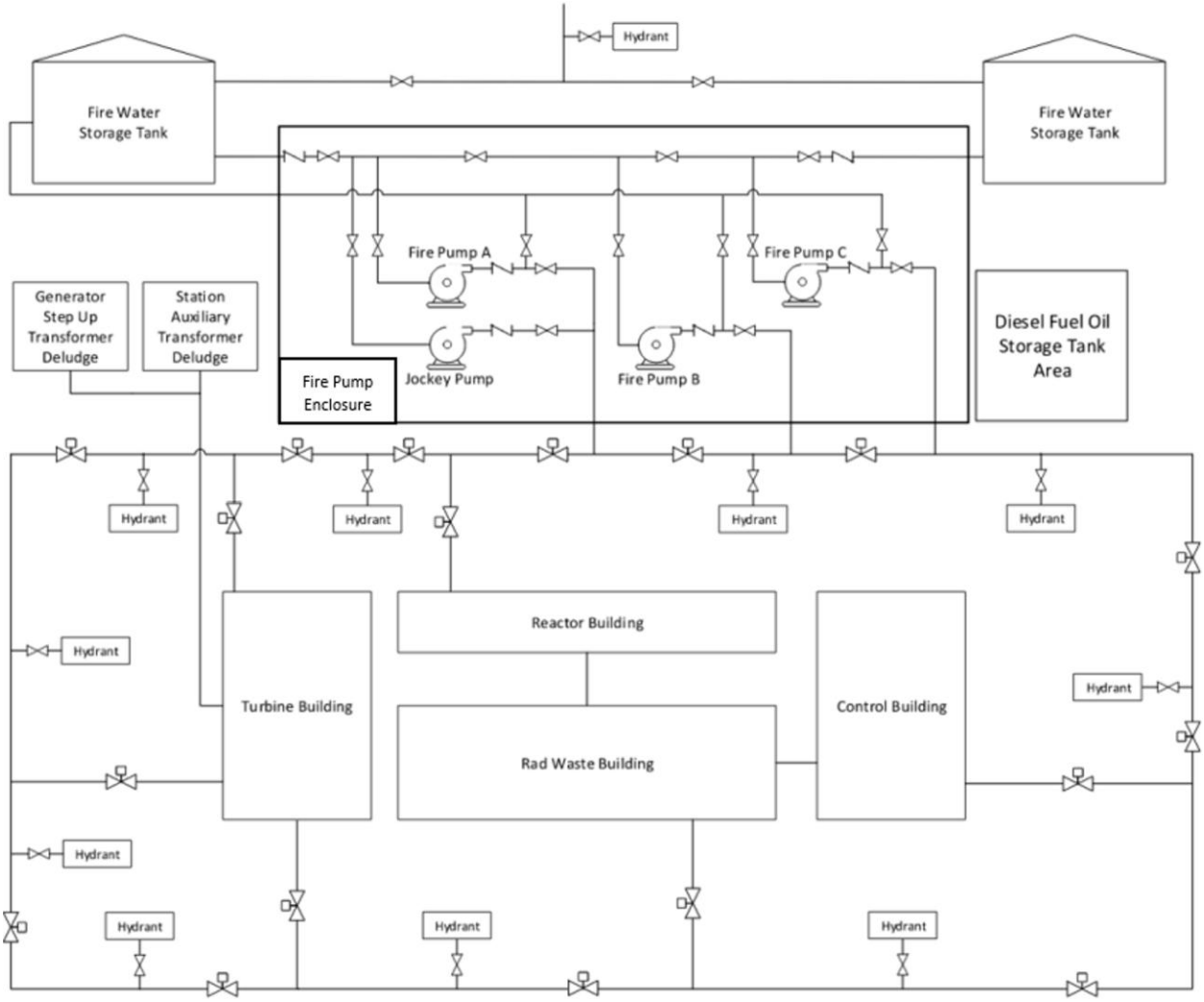


Figure 9.5-1: BWRX-300 Fire Protection System Schematic

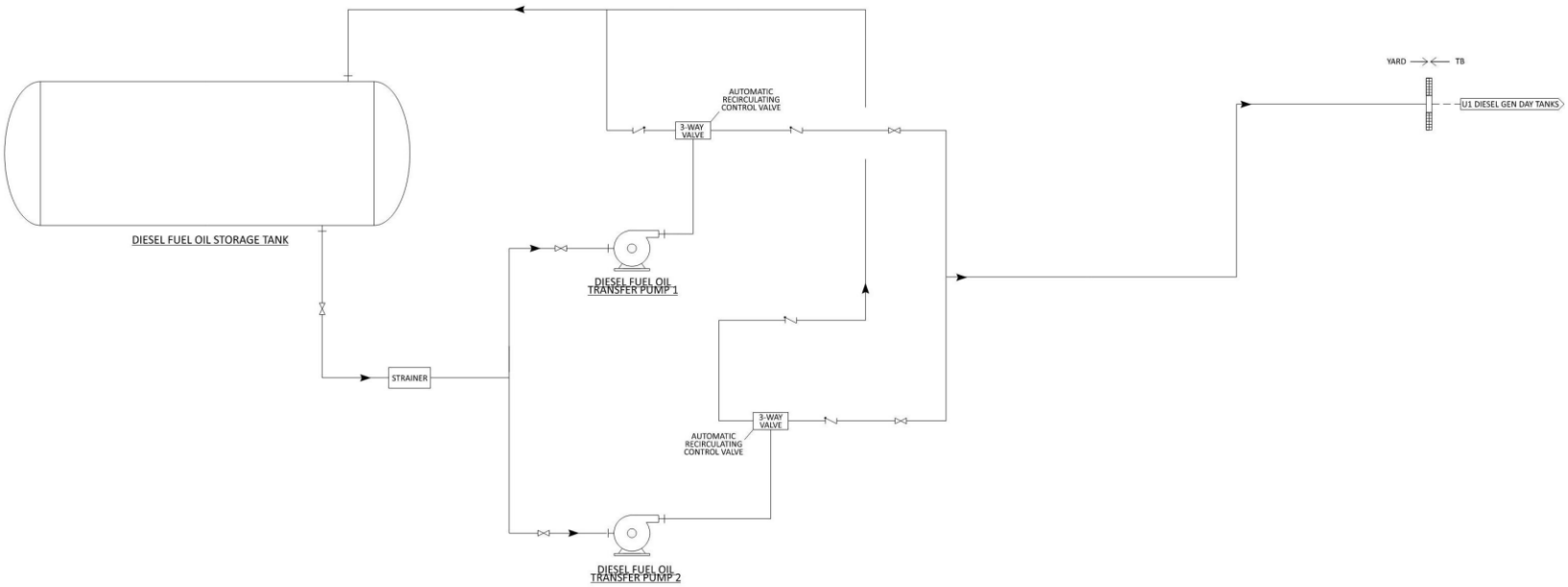


Figure 9.5-2: Diesel Fuel Oil Storage and Transfer System Schematic

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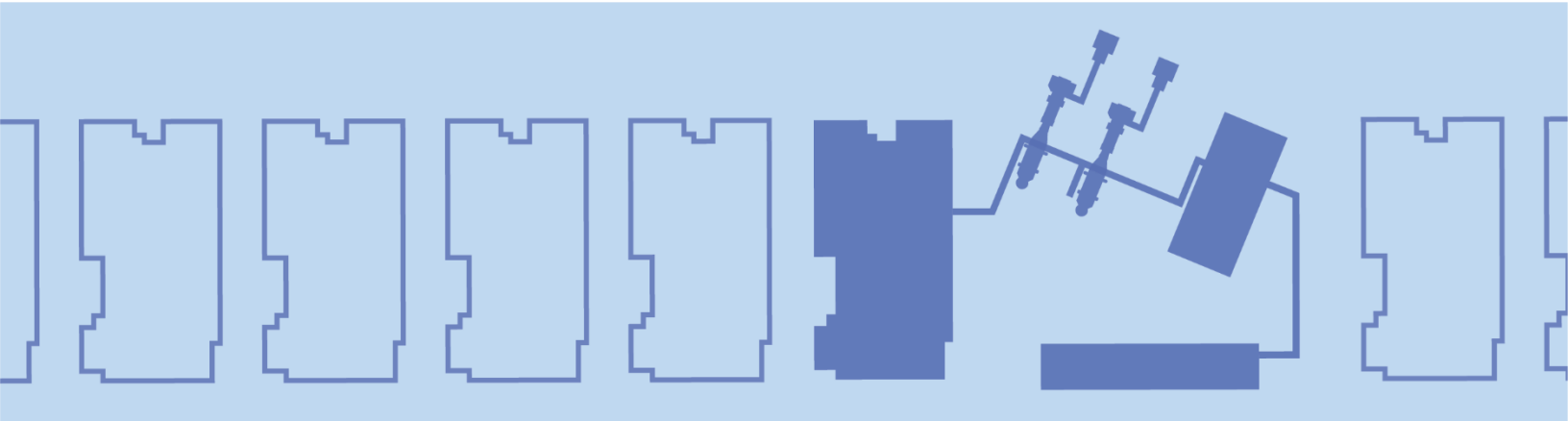
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Chapter 9
Section 9A – Fire Hazards Analysis



ACRONYM LIST

Acronym	Explanation
CB	Control Building
D-in-D	Defense-in-Depth
FHA	Fire Hazards Analysis
FP	Fire Protection
FPS	Fire Protection System
FSAR	Final Safety Analysis Report
FSSA	Fire Safe Shutdown Analysis
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NPP	Nuclear Power Plant
RB	Reactor Building
RG	Regulatory Guide
SSCs	Structures, Systems, and Components
TB	Turbine Building

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9A FIRE HAZARDS ANALYSIS

This section describes the content of the Fire Hazards Analysis (FHA) for the Blue Energy Center – Port of Victoria Nuclear Unit 01 (BECPV-N01). The FHA reviews the applicable codes and standards, defines acceptance criteria, informs the fire protection design, and reviews the preliminary design of other systems as related to the FHA and confirms suitability.

9A.1 Introduction

Performing a fire hazards analysis demonstrates that the plant can maintain the ability to perform fire safe shutdown functions and limit radioactive material releases to the environment in the event of a fire.

Regulatory Guide (RG) 1.189 Revision 5, “Fire Protection for Nuclear Power Plants,” provides guidance in the preparation of the FHA. The FHA includes fire protection requirements for the design, construction, commissioning, and operation of nuclear power plants, including Structures, Systems, and Components (SSCs) that directly support the plant.

9A.1.1 Design Bases

Refer to Subsection 9.5.1.1 for the design bases of the Fire Protection System (FPS).

9A.1.2 Fire Protection Defense-in-Depth Strategy

A Defense-in-Depth (D-in-D) strategy is used to achieve a high degree of fire protection by providing redundancy, diversity, and balance in fire protection. The layered elements of the Fire Protection D-in-D strategy for the FHA and design are as follows:

- Prevent fires - Design measures are put in place to reduce or eliminate, where practical, combustible materials and ignition sources
- Fire detection and suppression - Means are provided to quickly detect and extinguish or control fires
- Limit the effects of fire - Fire separations or other measures are provided to limit the spread of fire

9A.2 Analysis Criteria

9A.2.1 Objective and Scope

The FHA objective is to identify the specific hazards and fire protection capabilities in each area of the plant to demonstrate that potential damage is limited by various active and passive fire protection measures.

A fire hazards analysis is performed to demonstrate that the plant maintains the ability to perform fire safe shutdown functions and limits radioactive material releases to the environment in the event of a fire. The FHA is updated as necessary to reflect plant design and operational changes. The FHA has the following objectives:

- To consider potential in situ and transient fire hazards
- To determine the effects of a fire in any location in the plant on the ability to safely shut down the reactor or to limit the release of radioactivity to the environment
- To specify measures for fire prevention, detection, suppression, and containment for each fire area containing SSCs required for fire safe shutdown

Scope

The FHA establishes and evaluates distinct fire areas for the Power Block (Reactor Building (RB), Radwaste Building (RWB), Turbine Building (RB), Service Building (SB), and Control Building (CB)), and other structures or equipment in the protected area for the BWRX-300 plant. These other structures or equipment could include but are not limited to:

- Fire Pump Enclosure
- Outdoor Transformers

Additionally, equipment or buildings outside the protected area, such as the fuel oil storage tank(s), are in the scope. Equipment and piping layout is to be finalized at the FSAR.

9A.2.2 Codes and Standards

The Codes and Standards utilized in the FPS design are listed in Table 9.5-1.

9A.2.3 Safety Evaluation

Refer to Subsection 9.5.1.6 for the safety evaluation of the FPS.

9A.2.4 Room Data Sheets

A room data sheet is developed for each room in the Power Block identifying information, such as the room number, associated fire area, fire barriers, room contents, fire detection, fire suppression and fire impact (with and without fire protection systems).

9A.2.5 Acceptance Criteria

The acceptance criteria used in the design of the FPS are compliant with the requirements found in RG 1.189, and the applicable National Fire Protection Association (NFPA) guidance documents. Any exceptions or deviations from specific requirements are identified and justified.

9A.2.6 Systems Required to Achieve Fire Safe Shutdown

Systems that are required to achieve fire safe shutdown are evaluated in the FSSA. The methodology used to develop the FSSA is discussed in Section 9A.3.

The FHA and the FSSA evaluate SSCs to ensure that fire safe shutdown functions are maintained in the event of a fire. Results are presented in the FHA and FSSA as appropriate. Fire safe shutdown paths are identified, which are combinations of SSCs from systems that maintain these functions.

9A.3 Analysis Approach

The analysis approach used for the FHA and FSSA is based on the methodologies described below.

9A.3.1 Methodology for Fire Hazards Analysis

The BWRX-300 design utilizes the methodology described in RG 1.189 and NFPA 804, “Standard for Fire Protection for Advanced Light Water Reactor Electric Generating Plants,” (Reference 9A-1) for the development of the BWRX-300 FHA.

The content of the FHA complies with applicable guidance in RG 1.189, including elements and attributes listed in Regulatory Position 1.2. The FHA also separately identifies hazards and provides appropriate protection.

9A.3.2 Methodology for Fire Safe Shutdown Analysis

The BWRX-300 design utilizes the methodology described in RG 1.189 and NFPA 804 for the development of the FSSA. This methodology is illustrated in Figure 9A-1. The FSSA process includes the following steps further defined in Nuclear Energy Institute (NEI), NEI 00-01, "Guidance for Post Fire Safe Shutdown Circuit Analysis," (Reference 9A-2).

The steps for the fire safe shutdown analysis are as follows:

- Identify the fire safe shutdown functions
- Identify the SSCs necessary for fire safe shutdown and align those SSCs with the fire safe shutdown paths to achieve and maintain stable shutdown conditions
- Determine the cables associated with the SSCs identified and determine the routing of the cables by fire area. Cable selection includes both cables directly applicable to the SSCs and cables that address associated circuits regarding common power sources, common enclosure, and spurious operation of SSCs.
- Perform compliance assessment to determine that a fire safe shutdown path is available for a postulated fire in each fire area. The assessment assumes that equipment susceptible to fire damage in any one fire area will be rendered inoperable by fire and reentry is not possible. If there are fire areas that do not have a success path, determine methods for resolving the failure (e.g., changes to fire barrier separation in a fire area).

The FSSA demonstrates that at least one means of achieving nuclear safety objectives and performance criteria is available in the event of a fire. The FSSA scope is limited to plant areas where fires have a potential impact on SSCs required to perform the fire safe shutdown functions.

9A.3.3 Review Data

The FHA document contains the evaluation and analysis data for each fire area in BECPV-N01 and will be finalized as part of the Fire Protection Report at the operating license application stage.

9A.3.4 Exceptions

Exceptions are noted in the FHA and FSSA as applicable.

9A.4 Fire Hazards Analysis and Fire Safe Shutdown Analysis Summary

Fire Protection Analyses by Fire Area are provided in the FHA and FSSA documentation.

It is a general conclusion of the FSSA that the station has the capability to safely shutdown the unit in the event of a fire. The resulting analysis documented prior to the FSAR will show that the BWRX-300 plant is inherently safe with respect to internal fire events.

9A.5 Evaluation of Regulatory and Code Requirements

Regulatory requirements and code requirements frequently overlap and require some interpretation of requirements to ensure that either both requirements can be met, the more stringent requirement is met, or a combination of both.

Refer to Subsection 9.5.1.2.1 for cases of conflicting guidance.

The evaluation requires the completed FHA and FSSA documents and will be finalized as part of the Fire Protection Report.

9A.6 References

9A-1 NFPA 804-2020, "Standard for Fire Protection for Advanced Light Water Reactor Electric Generating Plants," National Fire Protection Association, 2020.

9A-2 NEI 00-01, “Guidance for Post Fire Safe Shutdown Circuit Analysis,” Nuclear Energy Institute, Revision 4, December 2019.

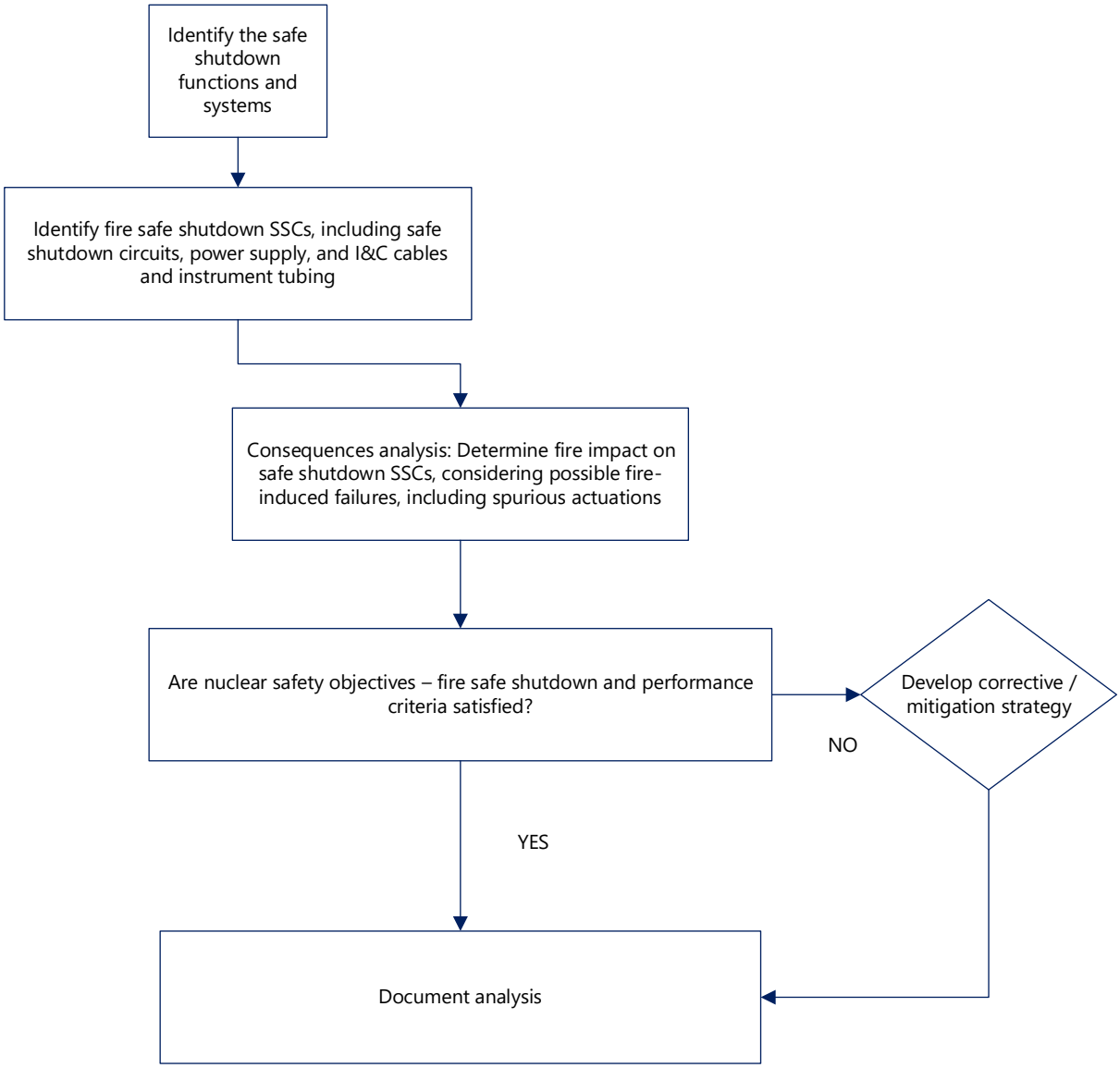


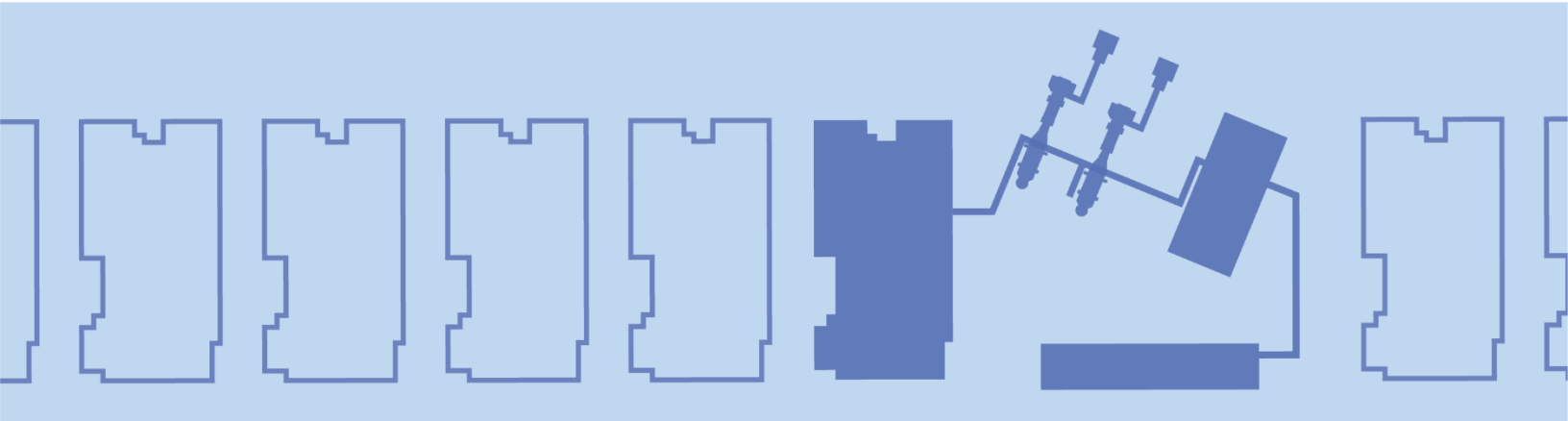
Figure 9A-1: BWRX-300 Fire Safe Shutdown Analysis Methodology

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Chapter 10
Steam and Power Conversion System



ACRONYM LIST

Acronym	Explanation
AC	Alternating Current
ALARA	As Low As Reasonably Achievable
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
ASTM	ASTM International
BPVC	Boiler and Pressure Vessel Code
BTP	Branch Technical Position
CFD	Condensate Filters and Demineralizers System
CFR	Code of Federal Regulations
CFS	Condensate and Feedwater Heating System
CST	Condensate Storage Tank
CUW	Reactor Water Cleanup System
CW	Circulating Water
CWS	Circulating Water System
DBA	Design Basis Accident
DC	Direct Current
DCIS	Distributed Control and Information System
DEC	Design Extension Condition
EHC	Electro-Hydraulic Control
EPRI	Electric Power Research Institute
FAC	Flow Accelerated Corrosion
FW	Feedwater
FWH	Feedwater Heater
GDC	General Design Criteria
GSC	Gland Seal Condenser
HEI	Heat Exchange Institute
HP	High Pressure
HVS	Heating, Ventilation, and Cooling System
HX	Heat Exchanger

Acronym	Explanation
I&C	Instrumentation and Control
ICS	Isolation Condenser System
ISI	Inservice Inspection
IST	Inservice Testing
LP	Low Pressure
LWM	Liquid Waste Management System
MCA	Main Condenser and Auxiliaries
MCR	Main Control Room
MS	Main Steam
MSCIV	Main Steam Containment Isolation Valve
MSL	Main Steam Line
MSR	Moisture Separator Reheater System
MSRIV	Main Steam Reactor Isolation Valve
MTE	Main Turbine Equipment
NDTT	Nil-Ductility Transition Temperature
NBS	Nuclear Boiler System
NC	Non-Condensable
NHS	Normal Heat Sink
NS	Non-Seismic
OGS	Offgas System
PRM	Process Radiation Monitoring Subsystem
PS	Process Sampling Subsystem
RB	Reactor Building
RG	Regulatory Guide
RLC	Reactor Level Control
RPV	Reactor Pressure Vessel
SBO	Station Blackout
SC1	Safety Class 1
SC2	Safety Class 2
SC3	Safety Class 3
SIR	Seismic Interface Restraint

Acronym	Explanation
SJAE	Steam Jet Air Ejector
SSCs	Structures, Systems, and Components
SW	Service Water
SWM	Solid Waste Management System
TB	Turbine Building
TBS	Turbine Bypass System
TBV	Turbine Bypass Valve
TCV	Turbine Control Valve
TG	Turbine Generator
TGCS	Turbine Generator Control System
TGSS	Turbine Gland Seal System
TLOS	Turbine Lube Oil System
TSV	Turbine Stop Valve
USNRC	United States Nuclear Regulatory Commission

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10.0 STEAM AND POWER CONVERSION SYSTEM

10.1 Summary Description

The steam and power conversion system for the BWRX-300 consists of the following systems: Turbine Generator (TG), Main Steam subsystem (MS) of the Nuclear Boiler System (NBS), Main Condenser and Auxiliaries (MCA), Moisture Separator Reheater System (MSR), Turbine Gland Seal System (TGSS), Turbine Bypass System (TBS), Circulating Water System (CWS), Condensate Filters and Demineralizers (CFD), and the Condensate and Feedwater Heating System (CFS).

A simplified system flow diagram and representative heat balance of the steam and power conversion system is provided in Figure 10.1-1. A summary of the important design and performance characteristics is provided in Table 10.1-1.

The TG includes the Main Turbine Equipment (MTE) components and subsystems shown below, generator and exciter. These are presented in Section 10.2:

- One High Pressure (HP) and two Low Pressure (LP) turbines, Turbine Stop Valves (TSVs), Turbine Control Valves (TCVs), non-return valves, intermediate stop valves and intercept valves (or combined intercept valves, depending on vendor-specific design), and turning gear
- Turbine Lube Oil System (TLOS)
- Extraction Steam Subsystem
- Electro-Hydraulic Control (EHC) Subsystem

Section 10.2 also details the turbine overspeed protection and turbine integrity that minimize the probability of turbine missile generation.

The MS subsystem, outboard of the Main Steam Containment Isolation Valves (MSCIVs), is presented in Section 10.3.

Other features of the steam and power conversion system are presented as follows in Section 10.4:

- MCA – Subsection 10.4.1
- MSR – Subsection 10.4.2
- TGSS – Subsection 10.4.3
- TBS – Subsection 10.4.4
- CWS – Subsection 10.4.5
- CFD - Subsection 10.4.6
- CFS – Subsection 10.4.7

Table 10.1-1: Design Features and Representative Performance Characteristics of the Steam and Power Conversion System

Parameter	Value
Reactor	
Core Thermal Power, MWt	870
Turbine Generator	
Nominal gross rating, MWe	300
Type	1 single-flow HP turbine 2 double-flow LP turbines (tandem)
Operating speed, rpm	3600
Pressure at turbine stop valve, psia	1000
Saturation temperature at turbine inlet, ºF	544.7
Moisture Separator Reheaters	
Number of MSRs per unit	1
Stages of moisture separation	1
Stages of reheating	2
Main Condenser	
Type	Single pressure, 2-shell
Circulating Water/Normal Heat Sink	
Number of circulating water pumps	4 – 33% capacity each
Circulating water flow rate, Mlbm/hr	132.2
Circulating water temperature rise, ºF	16.0
Normal heat sink type	Cooling tower
Condensate Pumps	
Number of pumps	2 – 100% capacity each
Feedwater Pumps	
Number of pumps	2 – 100% capacity each
Normal flow, Mlbm/hr	4.0
Feedwater Heaters	
Number of heaters	3 LP, 3 HP
Feedwater temperature, ºF	467.3

Note: Values provided in this table will be updated as appropriate following turbine vendor selection.

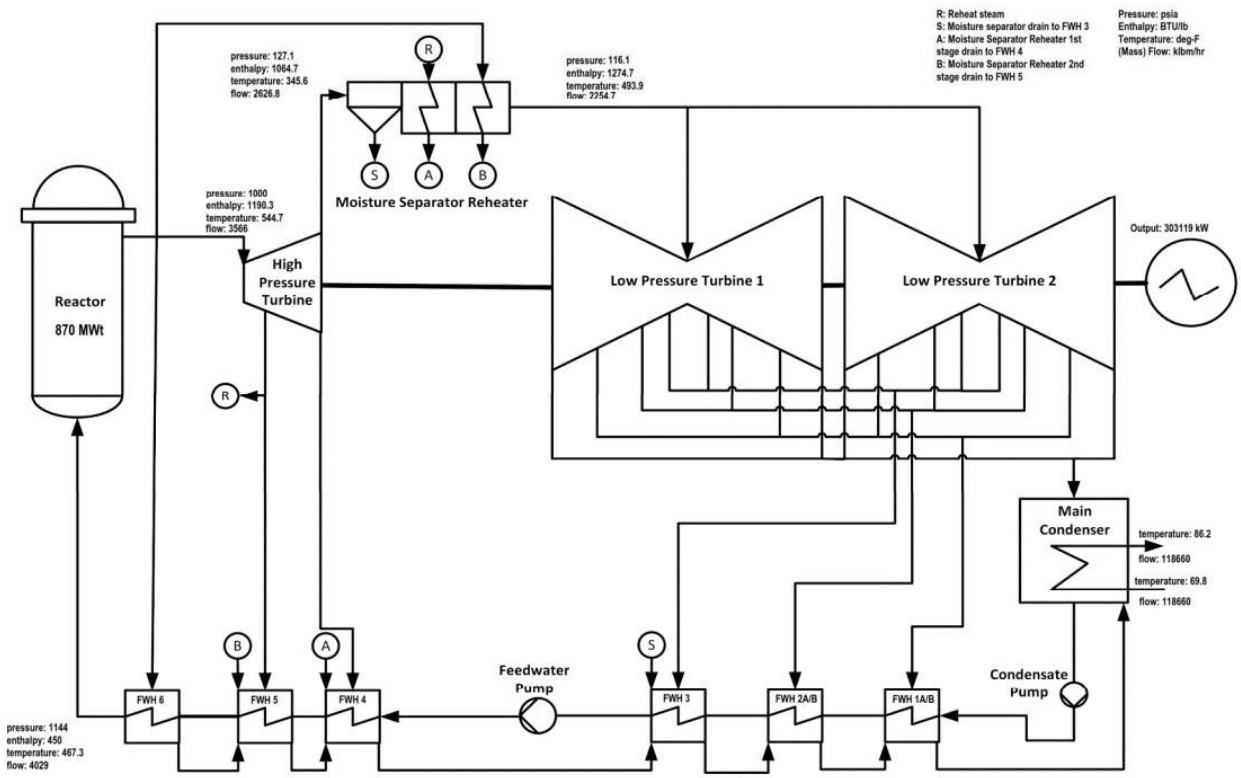


Figure 10.1-1: Simplified Flow Diagram with Representative Heat Balance of the Steam and Power Conversion System

Note: Values shown on this diagram will be updated as appropriate following turbine vendor selection.

10.2 Turbine Generator

10.2.1 Design Bases

Safety Category 1 Functions

None.

Safety Category 2 Functions

None.

Safety Category 3 Functions

- Convert thermal energy from the main steam and hot reheat steam into rotational torque on a shaft connected to the main generator based on the nominal conditions derived in the plant heat balance
- Convert mechanical energy from rotation of the main turbine into electrical power at a gross nominal output of 300 MWe
- Provide electrical power to the switchyard via the isophase bus ducts and generator step up transformer (described in Chapter 8)
- Provide electrical power to the Electrical Distribution System via the unit auxiliary transformer (described in Chapter 8)
- Throttle steam flow using the TCVs based on signals from Instrumentation and Control (I&C) systems
- Close the TSVs and TCVs when signaled to do so by I&C systems

Safety Category N Functions

None.

The Safety Classification, Quality Groups, and Seismic Categories of the TG Structures, Systems, and Components (SSCs) are listed in Table 3A-1.

10.2.2 System Description

The TG includes MTE (3600 rpm turbine, stop and control valves, and associated subsystems), generator, and exciter. The main turbine has a single-flow, HP turbine unit, and two double-flow LP turbine units in tandem. The main turbine converts steam energy from the reactor into rotational motion that the generator converts into electrical energy.

In addition to the main turbine, the MTE includes the following subsystems:

- TGSS (described in Subsection 10.4.3)
- TLOS
- Extraction Steam
- EHC

Figure 10.2-1 is a simplified single line diagram of the MTE.

Moisture separation is performed by the MSR, as described in Subsection 10.4.2.

Flow Accelerated Corrosion (FAC) resistant materials are used for components exposed to wet steam or flashing liquid flow where significant erosion could occur. The degree of FAC resistance

of the material is consistent with the temperature, moisture content and velocity of the wet steam to which the component is exposed.

The TG is capable of daily load following, during at least 90% of the plant design life, with the following electrical power profile (ramp rates are assumed to be ± 0.5% per minute):

- Starting at 100% power, power ramps down to 50% power in two hours
- Power remains at 50% for eight hours
- Power then ramps up to 100% in two hours
- Power remains at 100% for the remainder of the 24-hour cycle

Two Main Steam Lines (MSLs) from the Reactor Building (RB) are routed inside the Turbine Building (TB) to the TSVs and TCVs. The MTE and associated piping, valves, controls, and subsystems are located in the TB.

The TG consists of the following components:

- One single-flow, HP turbine
- Two hydraulically operated TSVs (one per MSL) and two hydraulically operated TCVs (one per MSL), four valves total
- Two double-flow, LP turbines
- One hydraulically operated reheat intermediate stop valve and one hydraulically operated intercept valve (or one combined intercept valve) per LP turbine
- Steam piping from TCVs to the HP turbine steam inlet nozzles
- HP turbine exhaust connections
- Extraction steam outlet connections from each LP turbine for the three LP Feedwater Heaters (FW)
- Extraction steam outlet connections from the HP turbine for HP Feedwater Heaters (FWHs) 4 and 5, and the 1st stage MSR
- LP turbine exhaust hoods
- Turbine shaft journal and thrust bearings and housings
- Turning gear
- Turbine to generator shaft coupling
- Heat insulation and lagging materials
- Drain and vent connections with associated valves and piping
- Exhaust hood sprays
- Generator rotor, stator, casing, windings, and bushings
- Excitation system, including an automatic voltage regulator
- Generator air coolers
- Generator protective relaying system, including current transformers, voltage transformers, and instrument transformers

- Generator neutral grounding and ground fault protection

The TLOS, shown in Figure 10.2-2, supplies the lube oil to the turbine, generator, and exciter bearings. The TLOS lube oil skid contains storage tanks, Heat Exchangers (HXs), and an oil conditioning system which supports the TLOS.

The Extraction Steam subsystem, shown in Figure 10.2-3, provides extraction steam from the TG to the MSR first stage reheater and to the FWHs for regenerative heating. It also protects the turbine against overspeed and water induction conditions. Extraction steam from the HP turbine supplies steam to FWHs 4 and 5, and from the LP turbines to FWHs 1, 2, and 3.

The EHC subsystem, shown in Figure 10.2-4, provides high-pressure hydraulic fluid to actuate the TSVs and TCVs, intermediate stop and intercept valves (or combined intercept valves), and the Turbine Bypass Valves (TBVs). The EHC subsystem is also used to actuate the trip devices in the trip and overspeed protection circuits for the turbine. The EHC includes a hydraulic power unit and trip manifold assembly.

The generator and exciter convert the rotational energy produced by the MTE into electrical power that is transmitted to the Electrical Distribution System, as described in Chapter 8. The generator consists of a synchronous electric generator (including rotor, stator, and stator frame), collector, generator shaft seals, generator bearings, generator HXs, neutral grounding resistor and associated neutral grounding transformer, potential transformers, current transformers, surge protection, generator protection panel and relaying, the generator field winding, and instrumentation. The exciter consists of a field exciter and voltage regulator that function to provide rated output voltage.

System Startup and Shutdown

The startup mode consists of cold start (ambient temperature), warm restart, and hot restart. The Turbine Generator Control System (TGCS) provides automatic and manual controls for preheating the turbine, synchronizing to the grid, loading the unit from no load to full load, and unloading the unit during controlled shutdowns.

Normal Operations

During normal operation, the TSVs and intermediate stop and intercept valves (or combined intercept valves) are fully open, with TCVs modulating as required to maintain reactor and main steam pressure. The TGCS generates the position signals for the TSVs, TCVs, and the intermediate stop and intercept valves (or combined intercept valves).

Off-Normal Operations

The TGCS trips the turbine in the case of unsafe operating conditions. These conditions include events classified as an Anticipated Operational Occurrence (AOO), Design Basis Accident (DBA), or Design Extension Condition (DEC). These event classifications correspond to American Society of Mechanical Engineers (ASME) Section III Division 1, “Rules for Construction of Nuclear Facility Components,” (Reference 10.2-5) service levels of upset, emergency, and faulted conditions, respectively.

The TG controller provides a triplicated signal to the Safety Class 3 (SC3) Anticipatory Trip System whenever the controller initiates a turbine trip. The trip demand is sent in parallel to the TSVs, Anticipatory Trip System and Reactor Pressure Control. The Anticipatory Trip System then initiates a hydraulic scram. In the unlikely event the Anticipatory Trip System fails to scram, the independent and diverse Safety Class 1 (SC1) I&C system scrams the reactor to maintain the plant safety functions. The TGCS and Anticipatory Trip System are part of the SC3 I&C System, as described in Subsection 7.3.3.2.

Turbine Overspeed Protection

The normal speed control system to guard against turbine overspeed includes the TGCS, TSVs and TCVs, intermediate stop and intercept valves (or combined intercept valves).

The normal speed control system comprises a first line of defense against turbine overspeed. The valve arrangements and closure times are such that a failure of any single valve to close does not result in unsafe turbine overspeed in case of a trip signal. An increase in speed above the setpoint closes the TCVs and intermediate stop and intercept valves (or combined intercept valves) in proportion to the speed increase.

The normal speed control system is designed to limit peak overspeed resulting from a loss of full load to at least 1% below the overspeed trip setpoint. Typically, this peak speed is in a range of 106-109% of rated speed, with the overspeed trip setpoint at 110% of rated speed.

In addition to the normal speed control function provided by the TGCS, a separate and redundant, highly reliable turbine overspeed protection system is included to minimize the possibility of turbine rotor failure and turbine missile generation. If the normal speed control fails, the overspeed trip devices close the steam admission valves, including the TSVs and intermediate stop valves, to ensure that turbine speed does not exceed 120% of rated speed. The logic uses two diverse and independent systems to provide sufficient reliability to eliminate the risk of turbine missile generation and to avoid the need for a mechanical overspeed trip.

The failure of any single component does not cause the rotor speed to exceed the emergency overspeed. The overspeed sensing devices are in the turbine front bearing standard and are therefore protected from the effects of missiles or pipe breakage. The hydraulic units are fail-safe, loss of hydraulic pressure results in a turbine trip. The TSVs and TCVs provide full redundancy in that these valves are in series and have independent control signals and operating mechanisms. Closure of both TSVs or both TCVs effectively shuts off all main steam flow to the HP turbine. The intermediate stop and intercept valves are in series and have independent control signals and operating mechanisms. Closure of either intermediate stop or intercept valve (or combined intercept valve) effectively shuts off steam flow to the corresponding LP turbine. This arrangement is such that failure of a single valve to close does not result in a maximum speed exceeding 120% of the rated speed.

10.2.3 Turbine Integrity

The failure of a turbine disk or rotor could produce a high-energy missile. To minimize this probability, turbine disk integrity is assured through the combined use of selected materials with suitable toughness, analyses, testing, inspections, and operating procedures.

Material specifications, fabrication history, and chemical analyses of the turbine disks and rotors will be available after a turbine vendor is selected and described in the Final Safety Analysis

Report (FSAR). Tests and Inspections are described in Section 10.2.5

The minimum requirements for fracture toughness, high temperature properties and turbine design to be met by the turbine vendor are specified below:

- Turbine material is suitable for the pressure, temperature, moisture, and environmental condition it is subjected to for all operating conditions
- Turbine casing material is suitable for the operating parameters it is subjected to. Casing material is designed for damage mechanisms such as creep, thermal fatigue, interaction of creep-fatigue, embrittlement, and FAC.

- Turbine blade material is designed for erosion/corrosion protection. Methods of protection can include induction hardening, flame hardening, shot peening, or coating, as applicable to the operating conditions of the blade.
- Turbine rotor material is forged or welded and made from a material and by a process that minimizes flaw occurrence and maximizes fracture toughness properties
- For the LP turbine rotors, the 50% Fracture Appearance Transition Temperature, as obtained from Charpy tests performed in accordance with ASTM International (ASTM) A-370-24a, "Standard Test Methods and Definitions for Mechanical Testing of Steel Products," (Reference 10.2-1), is not higher than 0º F. In lieu of Fracture Appearance Transition Temperature, Nil-Ductility Transition Temperature (NDTT) obtained in accordance with ASTM E-208, "Standard Test Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels," (Reference 10.2-2) is an alternative. NDTTs are no higher than -30º F.
- Each LP turbine rotor is tested using the Charpy V-notch at the minimum operating temperature in the tangential direction. The Charpy V-notch energy is at least 60 ft-lbs.
- The ratio of the fracture toughness of the LP turbine rotor material to the maximum tangential stress, at speeds ranging from normal to design overspeed, is to be at least 2√ in, at minimum operating temperature

10.2.4 Safety Evaluation

The following General Design Criterion (GDC) has aspects specific to the TG and compliance is addressed as follows:

GDC 4 - Environmental and dynamic effects design bases: Redundant, diverse, and testable turbine overspeed protection is provided to minimize the probability of turbine missile generation due to turbine overspeed. Turbine disk and rotor integrity are assured through material selection, testing, and inspection.

Additional information on GDC compliance is found in Section 3.1.

The design of the TG conforms with Regulatory Guide (RG) 1.115 "Protection Against Turbine Missiles." The TG's physical layout provides protection to essential systems and components from the effects of high and moderate energy TG system piping failures or failure of the connection(s) from the LP turbine exhaust hoods to the condenser. The TG is oriented within the TB such that the containment, as well as the Safety Class SSCs outside containment, are excluded from the low-trajectory hazard zone as described in RG 1.115. RG 1.115 low-trajectory hazard zones are based on low-pressure-stage wheels of 1800 rpm turbines. The BWRX-300 utilizes a 3600-rpm turbine. Staff Regulatory Position C.5 of RG 1.115 states that the United States Nuclear Regulatory Commission (USNRC) will review turbine designs that are significantly different from the current 1800 rpm machines on a case-by-case basis to determine the applicability of the strike zones. Refer to Section 3.5 for description of the missile protection design bases for SSCs.

Failure of TG equipment does not preclude safe shutdown of the reactor systems.

The main turbine steam flow shutoff characteristics, in collaboration with turbine bypass capacity and bypass opening response characteristics, are designed to prevent a reactor scram from occurring as a result of a load rejection or turbine trip up to approximately 25% output. The TBS is described in Subsection 10.4.4.

The TCVs are used to maintain the reactor target dome pressure within a specified error band during power operations. A turbine trip on high main condenser pressure is actuated during power operations.

The Process Radiation Monitoring Subsystem (PRM) provides radiation monitoring in the TB. Details of the PRM are provided in Section 11.5.

The anticipated sources of radiation and radiation levels in the TB, along with shielding and access control requirements to ensure occupational exposures are As Low As Reasonably Achievable (ALARA), are provided in Chapter 12.

The design of the TG complies with the following codes and standards:

- The piping and valves of the MTE are designed to meet the requirements of ASME B31.1, "Power Piping," (Reference 10.2-6). The turbine shell is designed to include attached piping loads (i.e., extraction, exhaust and drain piping) based on ASME B31.1 allowable stresses for primary plus secondary stresses.
- Extraction Steam subsystem non-return valves are located five pipe diameters downstream from flow disturbances, such as tees and elbows, and ten pipe diameters downstream from control valves, in accordance with Electric Power Research Institute (EPRI) NP-5479, "Application Guide for Check Valves in Nuclear Power Plants," (Reference 10.2-3), and turbine manufacturer recommendations.
- Extraction piping, valves, drains and associated controls are designed in accordance with ASME Standard TDP-2, "Prevention of Water Damage to Steam Turbines Used for Electric Power Generation: Nuclear-Fueled Plants," (Reference 10.2-4).

10.2.5 Tests and Inspections

Turbine Water Induction Prevention Testing is performed on the HP and LP FWHs and associated valves to ensure that a single failure does not result in water entering the turbine.

The piping arrangement is designed to permit nondestructive examination of welds, fittings, pipe bends and elbows.

Provisions are included for testing each of the following devices:

- TSVs and TCVs
- Intermediate stop and intercept valves (or combined intercept valves)
- Remote trip solenoids
- Primary and secondary Alternating Current (AC) bearing oil pumps and Direct Current (DC) emergency bearing oil pump while the unit is operating

The inspection program plan for the turbine assembly and valves is based on turbine missile probability calculations, operating experience of similar equipment, and inspection results. The valve testing program, including test frequency, will be determined when the turbine vendor is selected.

The turbine assembly inspection includes inspection of normally inaccessible parts, such as couplings, coupling bolts, turbine shafts, turbine blades and buckets, and LP and HP rotors. Turbine inspection is performed in sections during refueling outages. The recommended maintenance and inspection program plan for the turbine assembly, valves and controls ensures that the annual turbine generator missile probabilities are maintained at or below the acceptable level.

Routine testing of the TLOS is conducted in accordance with normal power plant requirements for demonstrating system and component operability and integrity.

Routine testing of the EHC Subsystem is conducted in accordance with normal power plant requirements for demonstrating system and component operability and integrity.

The description of the preservice inspection program, to demonstrate the initial integrity of the turbine disks and rotors, will be available after the turbine vendor is selected. This will include details on the finish machining, ultrasonic inspection, surface inspection, visual inspection, and spin testing. Additional details will be available in the FSAR.

10.2.6 Instrumentation and Controls

The TGCS provides automatic and manual controls in the Main Control Room (MCR) to preheat the turbine, start and load the unit from no-load to full load, adjust load during continuous operation, perform all normal periodic test functions, and to unload the unit for shutdown. TGCS uses triple redundant exhaust hood temperatures to control main turbine exhaust hood spray and to initiate a turbine trip on high temperature. The TGCS also monitors and controls the TGSS, TLOS, and EHC Subsystem.

The MTE contains sufficient instrumentation to monitor and control system operations, and to alarm abnormal system operation. The Turbine Supervisory Instrumentation includes the necessary instruments to monitor the following parameters:

- Vibration and eccentricity
- Thrust bearing wear
- Exhaust hood temperature
- Bearing metal temperatures
- Shell temperature
- Valve positions
- Shell and rotor differential expansion
- Shaft speed, and control valve inlet pressure indication
- Steam chest pressure

The following MTE system conditions are alarmed in the MCR:

- Loss of speed signal
- High thrust bearing wear
- Excessive rotor/shell/differential expansion
- High exhaust hood temperature
- High condenser pressure
- Overspeed
- High vibration
- High bearing temperature
- Zero rotor speed
- High rotor eccentricity

- Power Load Unbalance

The TSVs, TCVs, intermediate stop and control valves (or combined intercept valves) close to trip the turbine on the following signals:

- Emergency trip push buttons in control room
- High moisture-separator level
- High condenser pressure
- Low lube oil pressure
- High LP turbine exhaust hood temperature
- High reactor water level
- Excessive thrust bearing wear
- Overspeed (primary and emergency trip systems)
- Manual trip from the front standard
- Loss of stator coolant
- Low hydraulic fluid pressure
- Any generator trip
- Loss of TGCS electrical power
- Excessive turbine shaft vibration
- Loss of two speed probes – both within the primary set, or within the emergency set
- Loss of two pressure control channels
- Main Steam Reactor Isolation Valve (MSRIV)/MSCIV close trip signal

10.2.7 References

10.2-1 ASTM A-370-24a, “Standard Test Methods and Definitions for Mechanical Testing of Steel Products,” ASTM International.

10.2-2 ASTM E-208, “Standard Test Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels,” ASTM International.

10.2-3 EPRI NP-5479, “Application Guide for Check Valves in Nuclear Power Plants,” Electric Power Research Institute.

10.2-4 ASME Standard TDP-2, “Prevention of Water Damage to Steam Turbines Used for Electric Power Generation: Nuclear-Fueled Plants,” American Society of Mechanical Engineers.

10.2-5 ASME BPVC, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” American Society of Mechanical Engineers.

10.2-6 ASME B31.1, “Power Piping,” American Society of Mechanical Engineers.

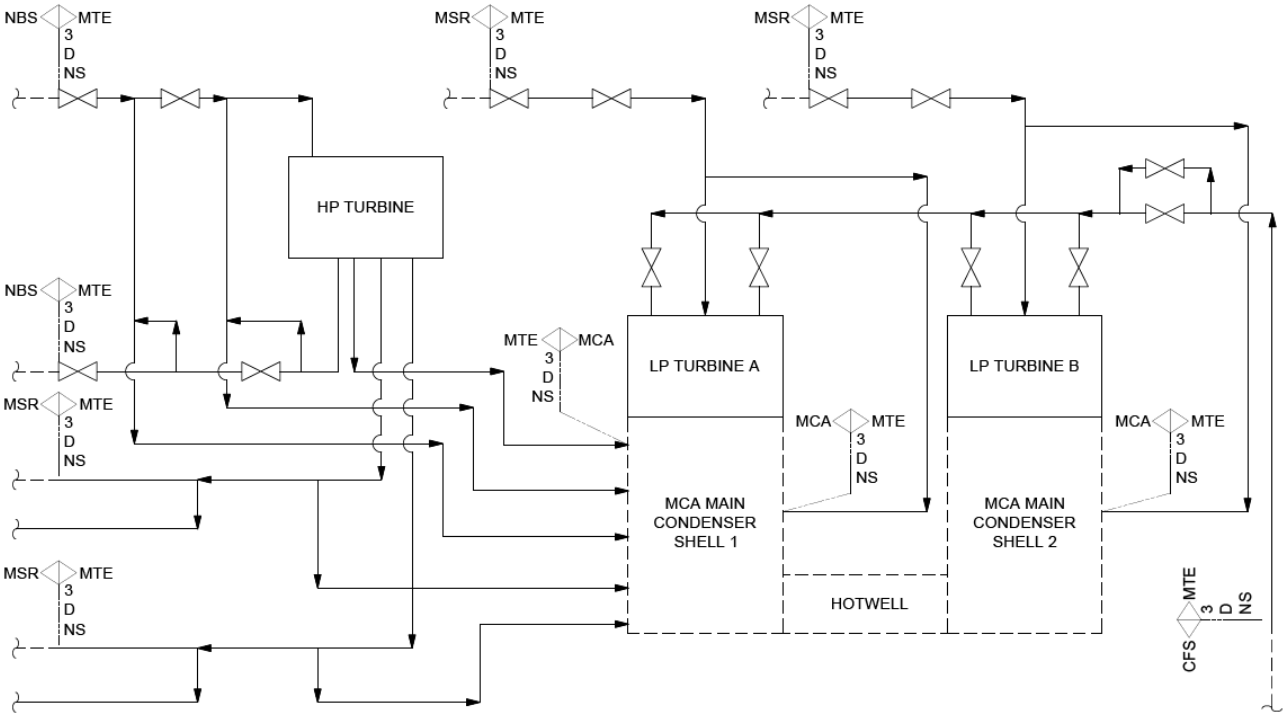


Figure 10.2-1: Main Turbine Equipment System Simplified Single Line Diagram

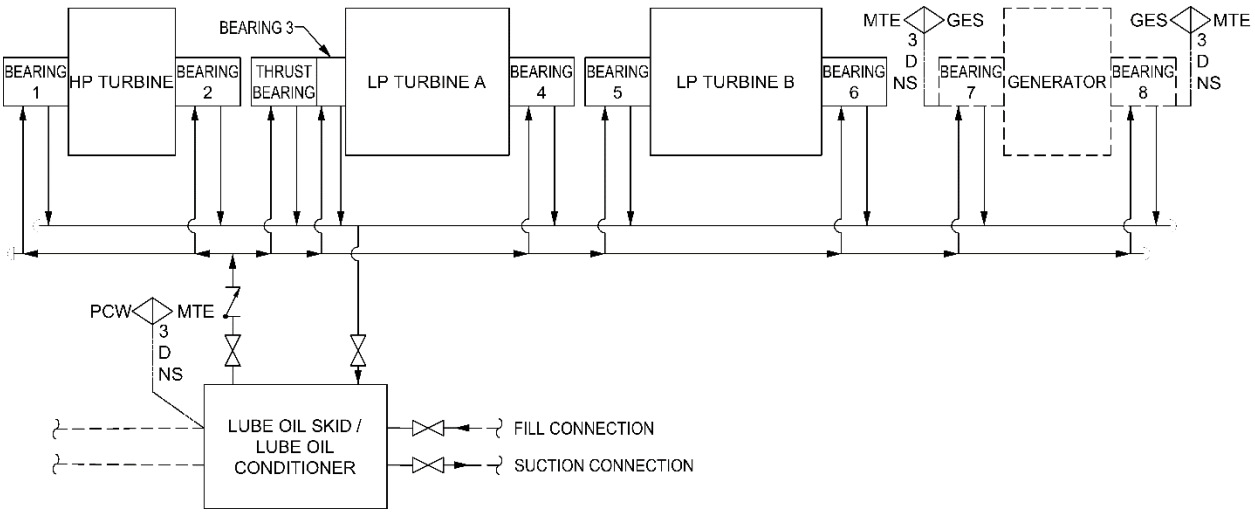


Figure 10.2-2: Turbine Lube Oil System Simplified Single Line Diagram

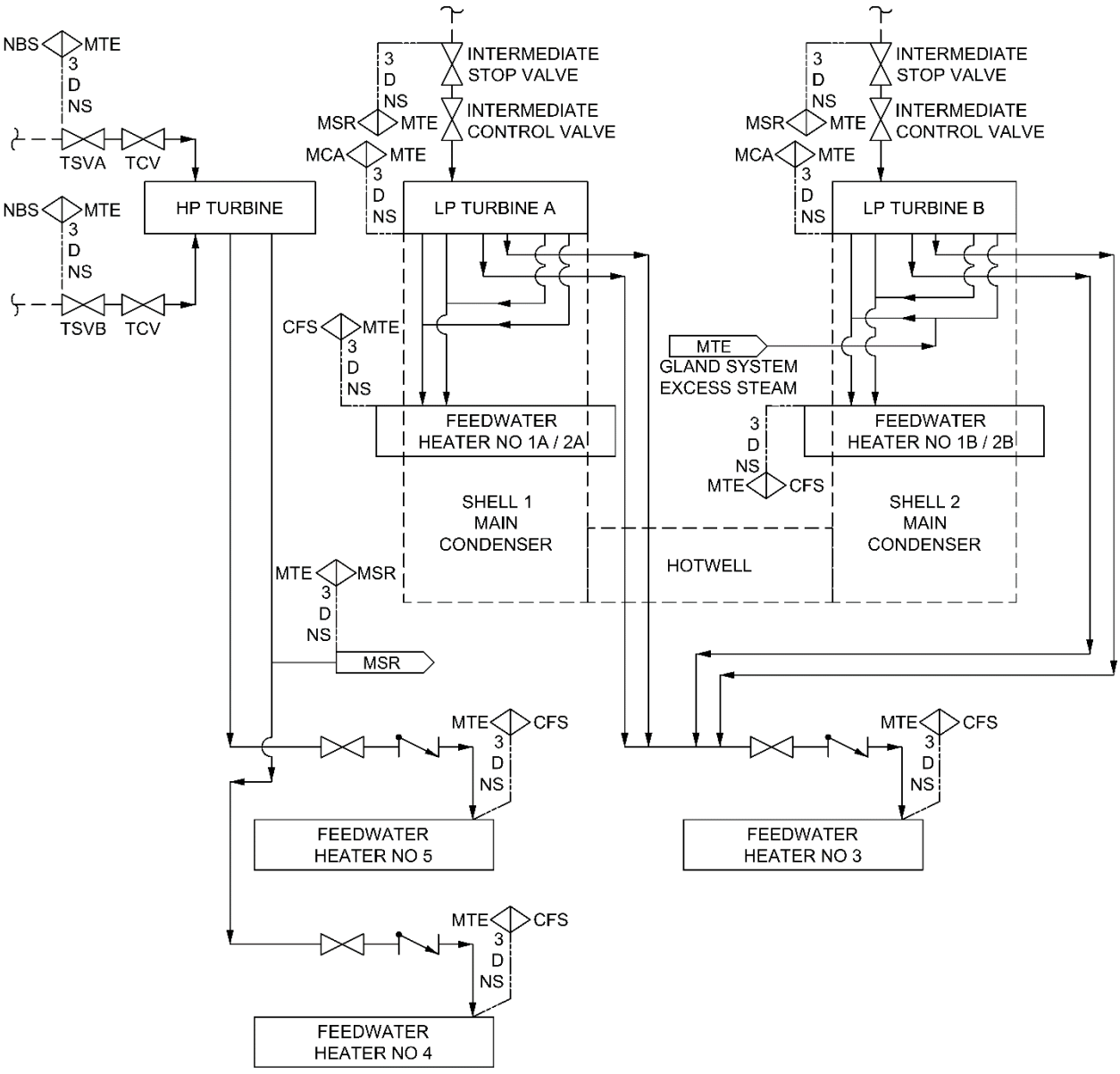


Figure 10.2-3: Extraction Steam Subsystem Simplified Single Line Diagram

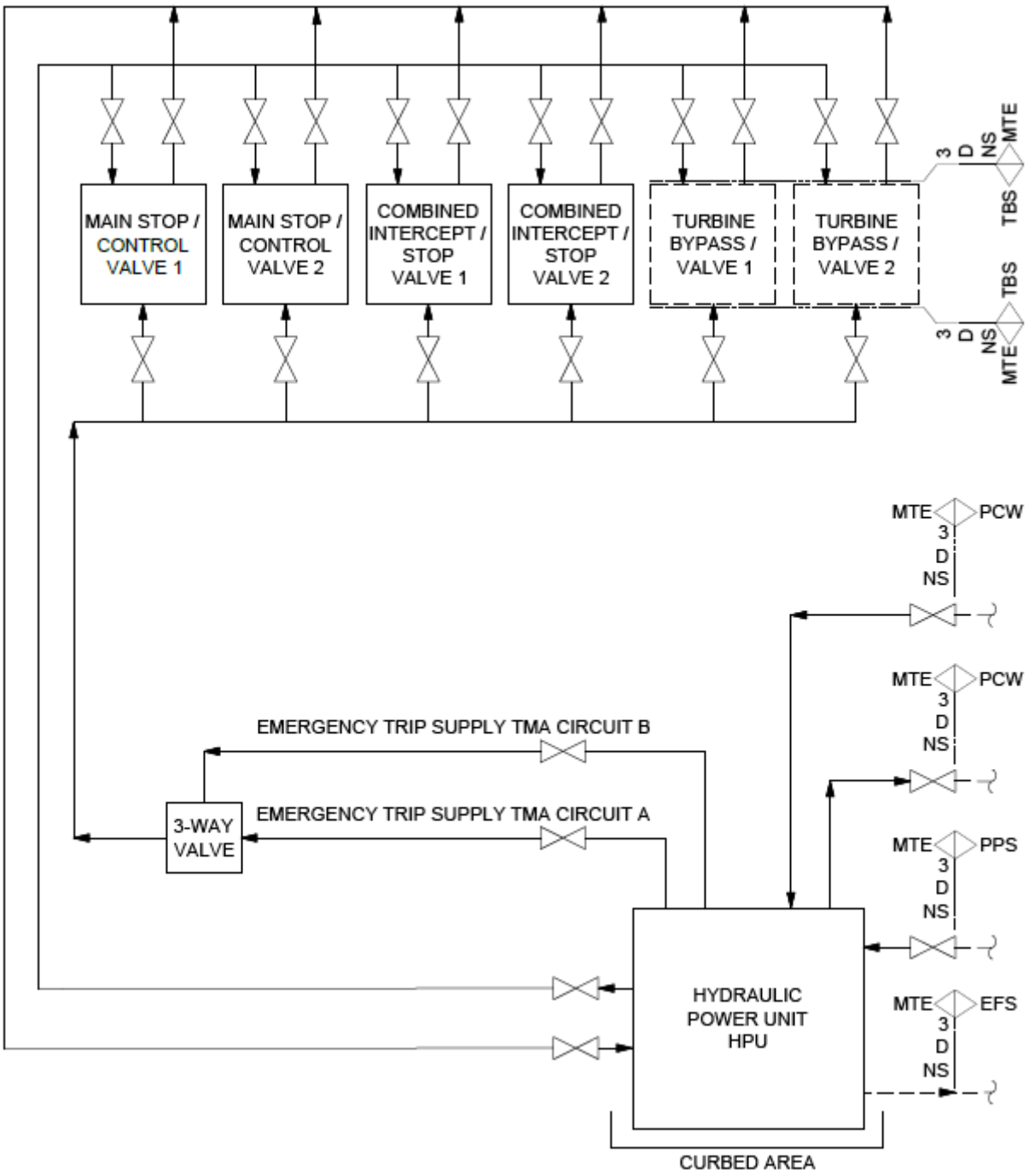


Figure 10.2-4: Electro-Hydraulic Control System Simplified Single Line Diagram

10.3 Main Steam Subsystem

The portion of the MS subsystem covered in this chapter is from the outboard side of the MSCIVs, through the RB, and in the TB.

10.3.1 Design Bases

Safety Category 1 Functions

The portion of the MS subsystem described in this chapter is designed to perform the following Safety Category 1 function:

- Provide the capability to detect a line break in main steam piping outside containment

Safety Category 2 Functions

The portion of the MS subsystem described in this chapter is designed to perform the following Safety Category 2 function:

- Provide the capability to detect a line break in main steam piping outside containment, using diverse means of detection from those used for Safety Category 1

Safety Category 3 Functions

The portion of the MS subsystem described in this chapter is designed to perform the following Safety Category 3 function:

- Convey steam from the reactor to the main turbine

Safety Category N Functions

None.

The Safety Classification, Quality Groups and Seismic Categories of the MS subsystem are listed in Table 3A-1.

The Safety Category functions associated with MSL reactor and system isolation are described in Subsection 5.4.5. The Safety Category functions associated with MSL containment isolation are described in Subsection 6.2.4.

10.3.2 System Description

The MS subsystem consists of two MSLs from the outboard MSRIVs to the TSVs, the TBVs, the MSL drains, and other load isolation/maintenance valves. The steam supply lines to these loads, all connecting branch lines up to and including their respective isolation/maintenance valves, and all associated piping supports are also part of the MS subsystem.

Figure 10.3-1 is a simplified line diagram showing this portion of the MS subsystem in the Reactor Building and Turbine Building. The portion of the MS subsystem in Containment, up to and including the MSCIVs, is shown in Figure 5.1-2.

The MS subsystem provides the means to convey steam from the reactor to the main turbine.

The MS equalizing header supplies HP steam via branch piping to the TBVs, the MSR, the No. 6 FWH, and the Turbine Auxiliary Steam Subsystem. The MS drain lines provide the ability to drain the condensate from the MS lines to the condenser in a controlled manner during startup, lower power operation (approximately 40% rated power for the bypass drain valves), normal power operation, and shutdown.

10.3.3 Safety Evaluation

The MS subsystem complies with the following GDCs, which have plant-wide applicability, as discussed in Section 3.1:

GDC 2 – Design bases for protection against natural phenomena

GDC 4 – Environmental and dynamic effects design bases

Additional information on GDC compliance is found in Section 3.1.

The design of the MS subsystem complies with Branch Technical Position (BTP) 3-3, as described in Section 3.6.

The design of the MS subsystem conforms with NUREG–0927, “Evaluation of Water Hammer Occurrence in Nuclear Power Plants,” to minimize the occurrence of water hammer. The horizontal process lines are sloped downward in the direction of flow to promote proper drainage. Where horizontal process lines cannot be continuously sloped, the lines are equipped with high point vents and accessible valves to ensure that the piping can be filled and vented. Normally stagnant horizontal process lines connected to the Reactor Pressure Vessel (RPV) are sloped downward from the RPV connection point to prevent the buildup of Non-Condensable (NC) gases. High point vents are included where lines cannot be continuously sloped.

The BWRX-300 design includes an outboard MSCIV on each MSL. The MSRIVs and MSCIVs provide isolation of the MSLs for line breaks, MSCIVs for containment isolation, and when required during plant shutdown condition.

Details regarding operation of the MSRIVs and MSCIVs after a DBA are provided in Subsections 5.4.5 and 6.2.4.

The PRM provides radiation monitoring for the MS subsystem in the RB and TB, as described in Section 11.5.

The anticipated sources of radiation and radiation levels in the RB and TB, along with shielding and access control requirements to ensure occupation exposures are ALARA, are provided in Chapter 12.

The break of a MSL does not result in offsite radiation exposures in excess of the site limits, as discussed in Subsection 15.5.9.1.1.

Plant response to a Station Blackout (SBO), as required by 10 CFR 50.63, is addressed in Section 8.4.

The design of the MS subsystem complies with the following codes and standards:

- The SIR is designed to ASME BPVC Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” Class NF (Reference 10.3-2) requirements
- The MS piping in the TB is designed ASME B31.1
- The portion of the MS subsystem located in the TB meets the requirements of ASME B31T “Standard Toughness Requirements for Piping,” Paragraphs 3, 4 and Appendix A (Reference 10.3-1)

10.3.4 Tests and Inspections

Preservice, ISI and testing requirements for the ASME Class 2 and 3 portions of the MS subsystem are discussed in Section 6.6.

The ASME B31.1 piping system testing is performed in accordance with ASME B31.1.

10.3.5 Instrumentation and Controls

Details of the instrumentation used for leak detection of the portions of the MS subsystem up to and including the MSCIVs are provided in Subsection 5.2.5.

MSL differential pressure transmitter instruments tapped off the MSL flow venturi sensing lines measure steam flow and detect conditions that are indicative of a line break. Instruments for sensing temperatures are provided in the RB steam tunnel and TB steam area.

10.3.6 Steam and Feedwater System Materials

Detailed discussion of the materials for components that are part of the Reactor Coolant Pressure Boundary is provided in Subsection 5.2.3. This includes materials specifications, compatibility with reactor coolant, fracture toughness properties and control of welding.

The MS piping from the outboard side of the MSCIV up to the SIR is designed to ASME BPVC Section III, Class 2 requirements.

The portion of the MS subsystem located in the TB is designed to ASME B31.1 and meets the requirements of ASME B31T Paragraphs 3, 4, and Appendix A (Reference 10.3-1).

See Subsection 10.4.7.1 for FW component classification.

10.3.7 References

10.3-1 ASME B31T, “Standard Toughness Requirements for Piping,” Paragraphs 3, 4, and Appendix A, American Society of Mechanical Engineers.

10.3-2 ASME BPVC, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” Subsection NF, “Supports,” American Society of Mechanical Engineers.

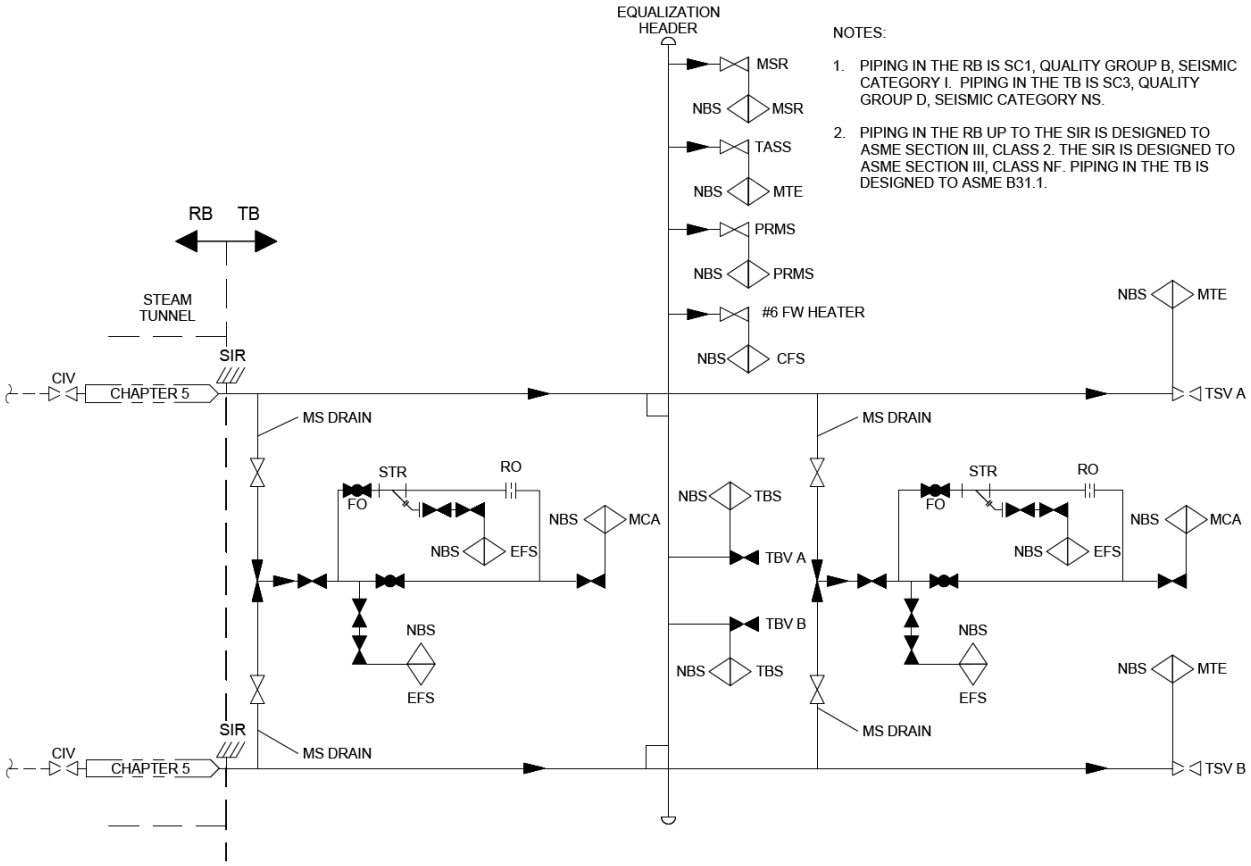


Figure 10.3-1: Main Steam Subsystem Simplified Single Line Diagram (RB & TB Portion)

10.4 Other Features of Steam and Power Conversion System

10.4.1 Main Condenser and Auxiliaries System

10.4.1.1 Design Bases

Safety Category 1 Functions

None.

Safety Category 2 Functions

- Measure main condenser pressure

Safety Category 3 Functions

- Measure main condenser pressure, using diverse means of detection from those used for Safety Category 2
- Provide a heat sink to condense steam exhausted from the main turbine or vented from other sources
- Provide a reservoir of condensed steam or condensate drained from other sources to allow for redistribution of water for other plant systems with a sufficient volume to provide a hold up time allowing for the decay of radioactivity to reduce dose to plant personnel
- Maintain the concentration of NC gases in the main condenser within an acceptable range for all conditions of plant operation

Safety Category N Functions

- Remove dissolved gases from the condensate during reactor startup

The Safety Classification, Quality Groups, and Seismic Categories of the MCA SSCs are listed in Table 3A-1.

10.4.1.2 System Description

During power operation, the main condenser is a heat sink for the steam cycle. LP turbine exhaust steam, turbine bypass steam, and other miscellaneous power cycle drains and vents are routed to the main condenser. The main condenser is a shell and tube type HX with two shells, each mounted beneath and connected to an LP turbine by an expansion joint. The CWS provides cooling water through tube bundles in the condenser shells. Leak detection troughs and sampling connections are located under each tube sheet to detect cooling water leakage. The condenser shell sides are protected from overpressurization by rupture diaphragms on the LP turbine exhaust hoods (steam side), which vent to the TB. Two stages of LP FWHs are located in the condenser neck. A vacuum breaker valve is provided on each condenser shell to break the vacuum and maintain atmospheric pressure within the condenser during shutdown.

The BWRX-300 design does not include a standalone Main Condenser Evacuation System. The vacuum pump and Steam Jet Air Ejectors (SJAEs), which are part of the MCA, perform this function and are described in the following paragraphs.

One 100% capacity vacuum pump is used as a rapid evacuator, to remove the air from the condenser shells and associated turbine, creating the initial vacuum which allows the startup of the turbine unit. The condenser vacuum pump takes suction from the shell side of the condenser and discharges the air extracted along with excess seal water to the seal water separator. The separator collects moisture in the mixture in the bottom and discharges the compressed air to the plant vent stack. The Plant Cooling Water system provides cooling water to the seal water HX.

A check valve at the outlet of the condenser vacuum pump prevents reverse flow to protect the condenser vacuum when the condenser vacuum pump stops.

Two 100% capacity SJAEs, one normally in operation and the other in standby, are used to maintain the turbine backpressure and remove the air in-leakage and NC gases, including radiolytic oxygen and hydrogen, from the main condenser. This helps to maintain the designed backpressure and improve plant efficiency. The SJAEs function by using main steam from the MTE. The SJAEs take suction from the shell side of the main condenser and discharge the motive steam and NC gases to the Offgas System (OGS). The air ejectors provide adequate motive force for the gaseous waste processing system. The steam supply to the SJAEs is maintained at minimum specified flow to ensure adequate dilution of hydrogen (less than 4% by volume), which prevents the NC gases from reaching the flammable limit of hydrogen.

Figure 10.4-1 is a simplified diagram of the MCA system. The main condenser and SJAEs are inside the shield wall area in the TB, and the condenser vacuum pump is located outside the shield wall.

Normal Operations

During system startup, the main condenser hotwell, if drained, is filled from the Condensate Storage Tank (CST) to the normal water level by automatic control of the normal and emergency makeup valves. Prior to placing the condenser vacuum pump in operation, the seal water separator is filled to the normal level, and cooling water flows through the tube side of the seal water HX. Once the TGSS has been placed into operation and the expansion joint seal water trough has been filled, the condenser vacuum pump may be started to evacuate the air from the condenser. Upon establishing a sufficient initial vacuum in the condenser, the condenser vacuum pump remains in service and holds the partial vacuum until motive steam is available for use by an SJAE skid. Motive steam should be provided by the MTE.

During normal operation, the main condenser condenses the LP turbine exhaust steam, stores condensate, acts as a condensate surge volume, and supplies condensate to the condensate pumps. The main condenser also accepts vents and drains from various points in the steam cycle. An SJAE unit removes air in-leakage and radiolytic gases from the condenser shells, and in conjunction with the CWS, maintains a vacuum in the condenser shells. The MTE supplies motive steam to one SJAE unit, while the other unit is typically in standby.

The main condenser acts as a heat sink in the initial phase of reactor cooldown during a normal plant shutdown. The condenser vacuum breaker valve may be opened to slow the turbine during shutdown.

Off-Normal Operations

The main condenser hotwell acts as a surge volume in the following events:

- Tube failure of feedwater heaters, SJAE intercondensers, MSRs, and other miscellaneous HXs
- High water level in the FWHs and reheater drain tanks
- Relief valve discharge for system overpressure protection

The condenser accepts steam and drains from various points in the turbine cycle, including those that require steam dump valves during TG load rejection. The condenser hotwell can receive overboard flow from the Shutdown Cooling System and the Reactor Water Cleanup System (CUW), as described in Subsections 5.4.7 and 5.4.8, respectively.

The main condenser can accept up to approximately 25% rated steam flow via the TBS in the event of a turbine trip or load reject. There is enough condensate stored in the hotwell to allow operation of the condensate pumps during this transient.

The pressure in the condenser may rise during normal operation due to low circulating water flow, high circulating water temperature, or high condenser duty. In these cases, as condenser pressure increases above normal levels, an alarm is annunciated. A further increase in pressure to the turbine trip limit results in a turbine trip and reactor scram. As pressure increases further beyond an established condenser limit, the TBVs are inhibited from opening and if open, close to prevent overpressurization of the condenser shell. If the pressure continues to increase, the MSCIVs close. The SJAE skid on standby may be brought online if a high absolute pressure exists in the condenser.

A condenser tube leak results in in-leakage from the CWS to the hotwell condensate, given the circulating water is at higher pressure than the condenser. At low in-leakage flowrates, a minor spike in conductivity could occur which can be accommodated by the CFD and hotwell level control. At higher in-leakage flowrates, vacuum pressure may not be maintained, resulting in a turbine trip.

10.4.1.3 Safety Evaluation

The following GDC has aspects specific to the MCA and compliance is addressed as follows:

GDC 60 - Control of releases of radioactive materials to the environment: During normal operation, radioactive leakage to the atmosphere via circulating water does not occur because the main condenser shells operate at a vacuum and air leakage is into the shell side of the main condenser. Condensate is retained in the main condenser to permit radioactive decay before entering the condensate system. A condenser vacuum pump removes the air from the condenser shells and associated turbine, creating the initial vacuum which allows the startup of the turbine unit. SJAEs maintain the turbine backpressure and remove NC gases from the main condenser, which are then exhausted to the OGS. Noble gas airborne concentrations are expected to be negligible throughout the TB except for inside the SJAE cubicles; the air from these cubicles is exhausted to the environment after high efficiency particulate air filtration and noble gas decay in activated carbon beds to eliminate the possibility of contamination of adjoining areas. Details of the OGS are provided in Section 11.3. The main condenser hotwell receives makeup water from the CST, and excess water in the hotwell is transferred to the CST, via the Condensate Storage and Transfer System, as described in Subsection 9.2.6. The Condensate Storage and Transfer System is a subsystem of the Liquid Waste Management System (LWM), which is described in Section 11.2.

Additional information on GDC compliance is found in Section 3.1.

The PRM provides radiation monitoring at the turbine building vent exhaust. The Process Sampling Subsystem (PS) obtains samples of each main condenser hotwell to monitor the condensate prior to entry into the CFS system to identify any potential condenser tube leakage, via continuous sampling. Details of the PRM are provided in Section 11.5. Details of the PS are provided in Subsection 9.3.2.

The steam supply to the second stage of the SJAE is maintained at minimum specified flow to ensure adequate dilution of hydrogen (less than 4% by volume), which prevents the NC gases from reaching the flammable limit of hydrogen. Any sections of pipe that may carry potentially explosive mixtures of hydrogen and oxygen are designed to contain the explosion.

The MCA has the capability to measure main condenser pressure (vacuum) and actuate a main turbine trip, TBV closure and/or a reactor scram as necessary on high main condenser pressure (i.e., low condenser vacuum).

The main condenser tubes and their tubesheets are made of titanium or low-cobalt stainless steel alloys which are compatible with the water chemistry to minimize corrosion. Provisions to prevent mechanical damage to the main condenser from turbine bypass flow are provided. Tube supports are designed to minimize tube vibrations. The condenser design assures that the inlet flow does not result in localized high velocities which could result in erosion of the tubes, tubesheets, plates, and other internal parts of the condenser. Precautions are taken to protect the tubes from the direct impingement of high velocity moisture-laden steam and high velocity condensate. The condenser is designed with provisions for either tube bundle replacement or individual tube replacement and individual tube plugging in place. The condenser is designed with an excess number of tubes to accommodate tube plugging.

The condenser hotwell is designed to store, at the normal operating water level, an amount of condensate equivalent to a short period with main steam valves wide open. In addition, the hotwell has an available standby surge storage capacity (i.e., maximum water level) to accommodate a brief period of normal, full load condensate flow. The main condenser can accommodate the turbine bypass flow for a short period following a trip of the circulating water pumps due to a loss of preferred power without exceeding the TBV isolation pressure setpoint in the condenser.

A flooding evaluation shows that the potential flooding which could result from failure of the condenser does not affect equipment required for the safe shutdown of the reactor.

The anticipated sources of radiation and radiation levels in the TB, along with shielding and access control requirements to ensure occupational exposures are ALARA, are provided in Chapter 12.

The MCA is designed in accordance with Heat Exchange Institute (HEI) 2629 “Standards for Steam Surface Condensers,” (Reference 10.4-1). MCA Piping is designed in accordance with ASME B31.1.

10.4.1.4 Tests and Inspections

The main condenser components are inspected and tested as part of the initial plant startup. Refer to Chapter 14 for a description of the initial test program. Detailed testing procedures are to be provided in the FSAR. Periodic inspections of the main condenser are performed in conjunction with the scheduled maintenance outages.

10.4.1.5 Instrumentation and Controls

Pressure instrumentation is provided as described below:

- Condenser vacuum is measured by SC3 narrow-range pressure transmitters. These triplicated transmitters are also used by the TBS, the TGCS, and for performance monitoring.
- Independent and redundant Safety Class 2 (SC2) wide-range pressure transmitters supply physically and electrically separate SC1 RPS condenser vacuum signals and leak detection and isolation signals for protection purposes
- Triple redundant pressure transmitters control the motive steam supply to the SJAEs

Instrumentation is provided to monitor temperature, level, flow, conductivity, and valve position for the system. The MCA is monitored and operated from the MCR. Package (skid)

instrumentation is provided to verify condenser vacuum pump and SJAE performance, in addition to normal operation and control. Alarms are provided in the MCR for major system parameters.

10.4.2 Moisture Separator Reheater System

10.4.2.1 Design Bases

Safety Category 1 Functions

None.

Safety Category 2 Functions

None.

Safety Category 3 Functions

- Provide heating to the steam exhaust from the HP turbine prior to it being supplied to the LP turbines

Safety Category N Functions

None.

The Safety Classification, Quality Groups, and Seismic Categories of the MSR system SSCs are listed in Table 3A-1.

10.4.2.2 System Description

The moisture separation function of the MTE is accomplished by the MSR system, which is located in the TB. Figure 10.4-2 is a simplified single line diagram for the MSR system.

Steam from the HP turbine exhaust enters the bottom of the MSR where steam separators remove entrained moisture before passing the steam to the reheater sections. Heating steam is supplied to the first stage reheater by the Extraction Steam subsystem, and to the second stage reheater by the MS subsystem. Hot reheat steam is then supplied to the two LP turbines. Drains are installed at each low point in the piping where water could collect. Moisture removed from the MSR shell is collected in the MSR shell drain tank. Steam condensed from the reheaters is collected in the reheater drain tanks. Condensate is routed through level control valves from the MSR shell drain tank to FWH 3, from the first stage reheater drain tank to FWH 4, and from the second stage reheater drain tank to FWH 5. An alternate condensate drain path to the condenser is provided from each drain tank to prevent overfilling the tanks and potentially reaching the main turbine.

During startup, the TSVs, TCVs, first stage reheater isolation valve, and second stage reheater control valve is initially closed. As the TSVs and TCVs begin to open, allowing steam to be admitted to the turbine, cold reheat (HP exhaust) steam enters the MSR. As turbine load increases, the first stage reheater isolation valve is opened allowing HP extraction steam to enter the first stage reheater. The second stage reheater control valve is then slowly opened to increase the hot reheat temperature.

During normal operations, the MSR receives steam from the HP turbine exhaust. Moisture is removed and the steam is reheated for use in the LP turbines. HP turbine extraction steam is directed through open isolation valves to the first stage reheater. Main steam is directed through open control valves to the second stage reheater.

The MSR System is designed to automatically control the reheat temperature under low turbine loads to avoid overheating the last stages of the LP turbine. Upon a turbine trip, the first stage reheater isolation valve and the second stage reheater control valve close. Hot reheat steam is

isolated by closing the reheat steam stop and intermediate control valves. Hot reheat drain lines open to drain condensate from the steam lines. Upon turbine shutdown, the TSVs and TCVs remain open and the MSR continues to receive cold reheat steam from the HP exhaust and supply hot reheat steam to the LP turbine until the main turbine falls below the manufacturer's specified output. At this point, the TSVs and TCVs, the first stage reheater isolation valves, and the second stage reheater control valves are closed.

10.4.2.3 Safety Evaluation

The MSR system performs the Safety Category 3 function of providing heating to the steam exhaust from the HP turbine prior to it being supplied to the LP turbines. The enthalpy loss from the HP turbine reduces the steam quality; therefore, it is necessary to reheat the exhaust steam to improve its quality before residual energy is converted in the LP turbines.

The PS collects representative liquid and gaseous samples for analysis and provides the analytical information required for monitoring plant and equipment performance. The PRM provides radiation monitoring for the MSR. Details of the PS are provided in Subsection 9.3.2. Details of the PRM are provided in Section 11.5.

The anticipated sources of radiation and radiation levels in the TB, along with shielding and access control requirements to ensure occupational exposures are ALARA, are provided in Chapter 12.

The MSR system is designed in accordance with the following codes and standards:

- The MSR HXs are designed, manufactured, installed, and tested in accordance with applicable codes and standards, including the Standards of Tubular Exchanger Manufacturers Association and ASME BPVC Section VIII Division 1, "Rules for Construction of Pressure Vessels," (Reference 10.4-3)
- Drains are installed at each low point in the piping where water could collect, per ASME TDP-2, "Prevention of Water Damage to Steam Turbines Used for Electric Power Generation: Nuclear-Fueled Plants," (Reference 10.4-2), Paragraph 3.1.3
- A power-operated non-return valve and motor-operated block valve is installed in the supply piping to the first stage reheater, as required by ASME TDP-2 (Reference 10.4-2) to minimize the potential of water damage to the turbines
- The MSR is protected by a relief valve, per ASME BPVC Section VIII, Division 1, UG-150 (Reference 10.4-3)
- Provisions are made to evacuate and purge all reheater tube bundles of moisture prior to steam piping pressurization to minimize distortion when heating steam is admitted to the tube bundle, per EPRI Moisture Separator Reheater Source Book (Reference 10.4-4)
- MSR system piping is designed per ASME B31.1 requirements

10.4.2.4 Tests and Inspections

The MSR HXs are tested in accordance with applicable codes and standards, including the Tubular Exchanger Manufacturers Association Standards and ASME BPVC Section VIII.

10.4.2.5 Instrumentation and Controls

Manual initiation, control, and shutdown of the MSR system is provided from the MCR. Instrumentation is provided for necessary status indication of key valves and annunciation of abnormal conditions during operation.

10.4.3 Turbine Gland Seal System

10.4.3.1 Design Bases

Safety Category 1 Functions

None.

Safety Category 2 Functions

None.

Safety Category 3 Functions

The TGSS is a subsystem of the MTE, which performs the Safety Category 3 functions described in Subsection 10.2.1.

Safety Category N Functions

None.

The Safety Classification, Quality Groups, and Seismic Categories of the TGSS SSCs are listed in Table 3A-1.

10.4.3.2 System Description

The TGSS supplies sealing steam to the turbine shaft/casing penetrations and to the valve stems of the TSVs, TCVs, TBVs, and Intermediate Stop and Intercept Valves (or combined intercept valves) to prevent the escape of radioactive steam and to prevent air in-leakage through sub-atmospheric turbine glands.

The TGSS is located in the TB and shown in Figure 10.4-3. The system consists of one Gland Steam Condenser (GSC), two 100% GSC exhaust blowers and associated valves, pressure regulators, a steam seal header, relief valves, isolation valves, piping, instrumentation, and control.

The GSC is a shell and tube HX designed to condense the maximum leakoff steam contained within the air-steam mixture withdrawn from the HP and LP turbine gland seals and the stem leak-offs of the TSVs, TCVs, Intermediate Stop and Intercept Valves (or combined intercept valves), and TBVs. The air-steam mixture is drawn into the GSC where steam is condensed on the outside of the tubes and air, and NC gases exit to the GSC exhaust blowers. The condensed steam is drained to the main condenser. Heat is rejected to the condensate flowing through the gland steam condenser tubes. The GSC is located far enough below the operating floor and forward of the main condenser so that the piping from the turbine to the condenser has a minimum pitch downward for proper drainage of the gland steam leakoff piping.

Two 100% capacity GSC exhaust blowers maintain a slight vacuum on the GSC shell to remove NC gases and seal the turbine prior to startup. NC gases are then exhausted to the TB Heating, Ventilation, and Cooling System (HVS).

The TGSS independent sources of steam are the MSLs during startup. At higher loads, steam from the pressure packings of the HP turbine is supplied to the vacuum packings of the LP turbine sections.

The MCA receives drained condensed steam from TGSS piping, the GSC, and steam traps for the GSC and GSC exhaust blowers. The MCA receives relief valve discharge from the GSC and steam seal header, as well as excess TGSS steam from the steam seal bypass dump valve.

Corrosion/erosion resistant materials are used for components exposed to wet steam or flashing liquid flow where significant erosion could occur. The degree of corrosion/erosion resistance of

the materials is consistent with the temperature, moisture content and velocity of the wet steam to which the components are exposed.

10.4.3.3 Safety Evaluation

The following GDC has aspects specific to the TGSS, and compliance is addressed as follows:

GDC 60 - Control of releases of radioactive materials to the environment: The system supplies sealing steam to the MTE to prevent the escape of radioactive steam from the MTE and to prevent air in-leakage through sub-atmospheric turbine glands. A partial vacuum is maintained in the GSC by redundant exhaust blowers to prevent steam leakage in the TB. The GSC is designed to prevent the entrainment of air in the gland steam condensate. Relief valves limit the steam seal header pressure to a safe level in the event of equipment failure which could result in high pressure. As stated in Subsection 10.4.3.2, the MCA receives steam from the TGSS. As stated in Subsection 10.4.1.3, excess water in the MCA hotwell is routed to the LWM. Details regarding the LWM are described in Section 11.2. The TB HVS receives NC gases from the GSC's blowers. As described in Subsection 9.4.4, TB HVS air is exhausted via the Plant Vent Stack, which is monitored for radiation by the PRM. Details of the PRM are provided in Section 11.5.

The potential of radioactivity leakage to the environment in the event of an equipment malfunction is provided in Chapter 15.

Additional information on GDC compliance is found in Section 3.1.

The TGSS is designed in accordance with the following codes and standards:

- The TGSS complies with the design requirements of EPRI TR 1013462, "Turbine-Generator Auxiliary Systems" (Reference 10.4-5)
- TGSS Piping is designed to meet the requirements of ASME B31.1

10.4.3.4 Tests and Inspections

Information regarding testing of the TGSS will be available after the turbine vendor is selected.

10.4.3.5 Instrumentation and Controls

Instrumentation for major system parameters is provided with indication and alarms in the MCR.

10.4.4 Turbine Bypass System

10.4.4.1 Design Bases

Safety Category 1 Functions

None.

Safety Category 2 Functions

None.

Safety Category 3 Functions

- TBVs fast open in response to signals from I&C systems
- TBVs close when signaled to do so by the I&C systems
- Modulate steam flow using the TBVs based on signals from I&C systems

Safety Category N Functions

None.

The Safety Classification, Quality Groups, and Seismic Categories of the TBS SSCs are listed in Table 3A-1.

10.4.4.2 System Description

The TBS provides the capability to discharge steam from the reactor directly to the main condenser. This minimizes step load reduction transient effects on the NBS. The TBS is used to assist in plant heat-up, cool-down, turbine trips and load rejection to maintain reactor pressure. The TBS, in combination with the reactor systems, provides the capability to provide for a 25% load rejection without a reactor scram. The ability to reject 25% of rated thermal power is intended to accommodate the plant conditions during startup, grid synchronization and shutdown sequences such that a turbine trip during startup would not cause a reactor scram. This supports the fundamental safety function of core heat removal.

The TBS is located in the TB and consists of two TBVs connected in parallel to the MS subsystem equalization header, and discharges to the main condenser. The system also includes TBV hydraulic actuators, supply and drain piping, servo valves, solenoid valves, check valves, and position transmitters. The TBS is shown in Figure 10.4-4.

The TBVs are control valves operated by hydraulic actuators equipped with springs capable of shutting the valves against the design steam pressure. The TBVs are designed to fail closed upon loss of control power or loss of hydraulic pressure to their operators.

During startup, the TSVs and TCVs are initially closed. The TBVs open as required, bypassing steam to the main condenser to allow the reactor to operate in a steady-state condition and to minimize pressure transients in the reactor during initial steam admission to the turbine. As steam is admitted to the turbine, the TBVs close as required to bring the turbine online while maintaining reactor pressure.

During normal operation, the TBVs are maintained in the closed position. The TBVs open as required to maintain reactor pressure following a load reduction or load rejection.

The MS piping and components downstream of the outboard MSCIVs are protected from system overpressure using the turbine bypass valves, the SC1 I&C system/insertion of control rods, the Isolation Condenser System (ICS), and the increased design pressure and temperature for the selected piping material, precluding the need for additional overpressure protection design features. Refer to Section 6.3 for information regarding RCPB overpressure protection.

A Main Steam Isolation Valve Leakage Control System is not part of the BWRX-300 design.

Upon a turbine trip, the TBVs initially open, then modulate as required to maintain reactor pressure. During shutdown, reactor power is reduced by rod insertion at specified rates and sequences. The TSVs and TCVs remain open until the MTE power falls below the manufacturer's specified output. At this point, the TSVs and TCVs close and the TBVs modulate to control steam pressure.

The TBVs open in response to redundant bypass demand signals received from the SC3 Distributed Control and Information System (DCIS) whenever the steam pressure exceeds the preset steam pressure. This occurs when the amount of steam generated by the reactor cannot be entirely used by the turbine. The number of TBVs required to open for bypass operation is determined by the SC3 DCIS and the TBVs modulate as required in response to changes in the bypass demand to discharge steam to the main condenser.

Material and equipment selection for the TBS is based on the range of normal environmental conditions that may exist at the location of the system's equipment and components.

10.4.4.3 Safety Evaluation

The TBS complies with the following GDC, which has plant-wide applicability, as discussed in Section 3.1:

GDC 4 – Environmental and dynamic effects design bases

The following GDC has aspects specific to the TBS and compliance is addressed as follows:

GDC 34 – Residual heat removal: The system can be used to remove decay heat from the reactor and discharge it to the main condenser.

Additional information on GDC compliance is found in Section 3.1.

The TBVs are initiated to fast open upon closure of the TSVs or TCVs. The TBS can accommodate an approximately 25% load rejection with the TBVs modulating as required to maintain reactor pressure and support operation of the turbine under reduced load.

Individual TBV hydraulic accumulators have the capability to stroke the valves with hydraulic power unit failure and meet the opening time and duration requirements. The accumulators, in combination with hydraulic check valves, ensure that no single hydraulic failure disables more than one of the TBVs. The accumulator hydraulic supply is designed to ensure that upon loss of off-site or station auxiliary power, TBVs can still be opened within a short time and re-closed. In the event electrical power is lost to the control system (regardless of whether the hydraulic unit has power or not), the TBVs automatically close and remain closed once all energy in the accumulator is expended.

The TBS is not credited to function following a DBA event. Failure of the TBS to operate does not preclude safe shutdown of the reactor. The ICS, in conjunction with the Reactor Isolation Valves, performs the safety functions of RPV overpressure protection, reactor coolant inventory maintenance, and decay heat removal. Refer to Section 6.3 for details of the ICS.

The TBS is designed in accordance with the following codes and standards:

- The TBS piping is designed and inspected to meet the requirements of ASME B31.1
- The TBVs are designated as American National Standards Institute (ANSI)/Fluid Controls Institute (FCI) 70-2 Class V, "Standard for Control Valve Seat Leakage," Testing," (Reference 10.4-6) leakage class to minimize wire drawing and other damage to the valves caused by leakage across the seat

10.4.4.4 Tests and Inspections

Routine testing of the TBS is conducted in accordance with normal power plant requirements for demonstrating system and component operability and integrity. Periodic inspections are performed on a rotating basis within a preventive maintenance program.

TBS system piping is inspected according to ASME B31.1.

10.4.4.5 Instrumentation and Controls

Automatic and manual control of the TBS is provided in the MCR. Triple redundant SC3 valve position transmitters are provided and installed at each TBV. Temperature elements installed in each TBV discharge pipe near the main condenser inlet provide temperature signals for bypass steam leakage detection.

10.4.5 Circulating Water System

The CWS has two subsystems: the main condenser Circulating Water (CW) Supply and the Service Water (SW) Subsystem. The SW Subsystem is described in Subsection 9.2.1.

10.4.5.1 Design Bases

The CW Supply Subsystem is designed to perform the following Safety Category functions:

Safety Category 1 Functions
None.

Safety Category 2 Functions
None.

Safety Category 3 Functions

- Provide the means to reject heat from the Main Condenser and Auxiliaries (MCA) to the environment through the Normal Heat Sink (NHS) System.
- During normal operations, provide the means to reject heat from the Plant Cooling Water (PCW) System to the environment through the NHS System.

Safety Category N Functions

- Remove accumulated air and other gases from the water boxes through water box vents.
- Provide the means to drain the condenser water boxes.

The Safety Classifications, Quality Groups, and Seismic Categories of the CW Supply SSCs are listed in Table 3A-1.

10.4.5.2 System Description

Figure 10.4-5 depicts the CW Supply, which delivers cooling water to the main condenser and, via the NHS System, transfers the resulting heat to the environment. Further description of the NHS System is provided in Subsection 9.2.9. Cooling water is supplied to the MCA by four CW Supply pumps, each rated for 33 percent of total design CWS flow; operating all four together yields full design flow, and the system remains capable of reduced-flow operation with any single pump removed for maintenance or inspection. Both the CW Supply and SW Subsystem pumps are housed in a pump house situated at or near the NHS System's cooling tower basin.

CW Supply pumps take suction from the cooling tower basin and deliver flow into a shared discharge header. Within the Turbine Building, this piping directs flow to the tube side of the main condenser, supplying the cooling water needed to condense the main turbine's exhaust steam. The condenser water boxes are divided so that each section may be isolated independently for online inspection or maintenance. Downstream of the main condenser, cooling water flows to the cooling tower header, tying into the NHS System's cooling tower distribution system.

Operation and monitoring of the CW Supply takes place in the BWRX-300 Main Control Room (MCR), as discussed in Subsection 10.4.5.5.

To prevent fouling, scaling, and corrosion of the condenser tubes, a tube cleaning system is provided.

Water box vents are incorporated into the CW Supply, serving both to assist in filling the condenser water boxes at startup and to release accumulated air and other gases from the

water boxes during normal operation. A drain pump and associated piping allow the condenser water boxes to be drained during outages or maintenance activities.

A cross-connection between the SW Subsystem pumps and the CW Supply piping fills the main condenser and its associated piping ahead of startup, a measure intended to prevent water hammer that could otherwise result from initial startup of the CW Supply pumps.

CW Supply materials and equipment are selected to suit the environmental conditions characteristics of the site.

Debris and small aquatic organisms in the CW Supply are controlled through strainers, filters, and chemical treatment provided by the NHS System . Additional chemical treatment limits biological growth, lowers alkalinity, and inhibits scale buildup. Together, these treatment chemicals maintain CW Supply water quality within water quality specifications.

Screens are included between the cooling tower basin and the CW Supply and SW Subsystem pumps. The mesh size and approach velocity are chosen to limit debris that could damage the pumps or condenser or add to the burden on the SW Subsystem strainers.

Safety Category 3 functions of the CW Supply remain available under off-normal conditions, including its design capability to sustain reactor operation through a turbine trip involving up to a 25 percent load rejection, during which 25 percent of main steam is bypassed around the turbine to the condenser.

10.4.5.3 Safety Evaluation

The CW Supply design complies with the following GDC, which has plant-wide applicability, as discussed in Section 3.1:

GDC 4 – Environmental and dynamic effects design bases. Additional information on GDC compliance is found in Section 3.1.

The CW Supply performs Safety Category 3 functions. It is not required to operate after a design basis event.

The CWS must be in operation before any steam or high energy sources are introduced into the condenser. Year-round full-power operation is supported by the circulating water the CW Supply delivers to the main condenser, and the CW Supply, in combination with the NHS System, furnishes circulating water sufficient to support the turbine-generator at its rated output.

Process Sampling (PS) collects representative liquid samples for analysis and provides information for monitoring plant and equipment performance. Details of the PS are provided in Subsection 9.3.2.

The CW Supply is designed in accordance with the following codes and standards:

- Pump qualification is performed in accordance with applicable industry codes and standards including, ANSI/HI 9.8, “Rotodynamic Pumps for Pump Intake Design,” (Reference 10.4-7);
- CW Supply piping is designed to meet the requirements of ASME B31.1; and

- Piping that is below ground meets applicable American Water Works Association standards based on pipe material and size.

10.4.5.4 Tests and Inspections

The CW Supply's equipment and components are configured to allow inspection and maintenance while the plant continues operating. Because pumps, cooling tower cells, and condenser water box trains are each provided with redundancy or divided into separate trains, individual components can be inspected and maintained without taking the plant offline. Testing of the CW Supply follows the routine requirements typical of power plant operation for confirming system and component operability and integrity, including periodic verification of cooling tower performance, as discussed in Subsection 10.4.5.3.

10.4.5.5 Instrumentation and Controls

The CW Supply is operated and monitored from the BWRX-300 MCR. Instrumentation for major system parameters provides indication and alarms in the applicable control room. Major CW Supply valves can also be operated locally.

10.4.6 Condensate Filters and Demineralizer System

10.4.6.1 Design Bases

Safety Category 1 Functions

None.

Safety Category 2 Functions

None.

Safety Category 3 Functions

- Provides a flow path to supply 100% rated flow of condensate to the CFS System

Safety Category N Functions

- Remove soluble and insoluble impurities from the condensate at 100% rated flow to meet or exceed the plant water quality requirements for FW

The Safety Classification, Quality Groups, and Seismic Categories of the CFD SSCs are listed in Table 3A-1.

10.4.6.2 System Description

The CFD purifies the condensate and the CUW water, as required, to maintain reactor FW purity. CFD filters remove suspended solids, including corrosion products. Demineralizers remove dissolved solids from various condensate sources of impurities including the CUW, the CST, and condenser tube leakage. The CFD is designed to ensure the effluent meets the water quality requirements described in the BWRX-300 Water Quality Design Specification.

The CFD includes three 50% capacity, high efficiency backwash-type filter elements and three 50% capacity demineralizer vessels. Auxiliary skids and tanks are provided to assist in cleaning the filters and replacing spent resin in the demineralizers. The system is located in the TB and is shown in Figure 10.4-6.

The CFD is used during plant startup to clean up the condensate, and to minimize the time required to achieve low power operation of the unit. The CFS provides recirculation flow paths to

assist in cleaning the condenser hotwell and in achieving the desired condensate water chemistry specifications prior to sending FW to the reactor.

The CFD normally operates with two filters and two demineralizers in service, while one filter and one demineralizer are in standby. Bypass facilities capable of bypassing full condensate flow around the CFD are included to provide continuous short-term condensate flow in the event of a condition within CFD which prevents treatment of the entire condensate flow. CFD bypass is also used during CFS system startup of the condensate pumps.

The CFD operates during and following a turbine trip, as well as under low load conditions.

The CFD limits the entry of dissolved solids into the CFS and, in the event of a large condenser tube break, permits a reasonable amount of time for orderly plant shutdown. In the event of a small condenser tube leak or ingress of impurities into the condensate system, the CFD allows for continued plant operation until the leak is repaired.

Material and equipment selection for the CFD is based on the normal operating environmental conditions that exist at the location of the system's equipment. The primary components of the system are constructed of corrosion resistant materials.

10.4.6.3 Safety Evaluation

The following GDC has aspects specific to the CFD system and compliance is addressed as follows:

GDC 14 – Reactor coolant pressure boundary: By maintaining water quality and using corrosion resistant materials, the design and operation of the CFD protects the reactor coolant pressure boundary.

Additional information on GDC compliance is found in Section 3.1.

The Solid Waste Management System (SWM) receives and stores condensate filter backwash waste, demineralizer resin washing waste, and spent resin waste. The water is separated and sent to LWM collection tanks. Details regarding the LWM are provided in Section 11.2. Details regarding the SWM are provided in Section 11.4.

The PRM performs radiation monitoring in the TB. Details of the PRM are provided in Section 11.5. The anticipated sources of radiation and radiation levels in the TB, along with shielding and access control requirements to ensure occupational exposures are ALARA, are provided in Chapter 12.

Process samples taken from the common condensate filter influent header, the effluent header of condensate filters, the effluent of each condensate demineralizer, the combined demineralizer effluent header, and the sluice water discharge header are piped to the PS. Details of the PS are provided in Subsection 9.3.2.

The CFD is designed in accordance with the following codes and standards:

- Where available, the maximum values in Table 5.2-5 are similar to, or the same as, EPRI BWRVIP-190 Revision 1: "BWR Vessel and Internals Project," (Reference 10.4-8) Action Level 3
- Piping and valves associated with resin and sludge systems are stainless steel designed to the requirements of ASME B31.1 (Reference 10.4-13), ASME B16.5 (Reference 10.4-14), and ASME B16.34 (Reference 10.4-15)

10.4.6.4 Tests and Inspections

Routine testing of the CFD is conducted in accordance with normal power plant requirements for demonstrating system and component functionality and integrity. The equipment and components of the CFD are designed for easy inspection and maintenance during plant operation.

10.4.6.5 Instrumentation and Controls

The CFD can be controlled either automatically or manually from the MCR. Instrumentation is provided for monitoring and controlling the filters and demineralizers in their various operating modes, and alarms are provided in the MCR for important system parameters.

10.4.7 Condensate and Feedwater Heating System

The portion of the CFS covered in this chapter is from the outboard side of the FW Containment Isolation Valves, through the RB, and in the TB.

10.4.7.1 Design Bases

The portion of the CFS described in this chapter is designed to perform the following Safety Category functions:

Safety Category 1 Functions

- Detect a line break in FW piping outside containment
- Detect loss of FW flow

Safety Category 2 Functions

- Detect a line break in FW piping outside containment
- Detect loss of FW flow
- FW and condensate pumps trip when signaled to do so by I&C systems

Safety Category 3 Functions

- Supply FW to the reactor vessel at the nominal conditions derived in the plant heat balance
- Modulate the flow of FW to the reactor for the purpose of reactor water level control, based on signals from I&C systems
- Detect loss of FW flow
- Prevent reverse flow in the FW lines
- The FW and condensate pumps trip when signaled to do so by I&C systems
- Monitor the operational status of the FW pumps
- FW pump in standby starts when signaled to do so by I&C systems
- Measure final FW temperature

Safety Category N Functions

- Monitor parameters necessary for measuring core flow and reactor heat balance

The Safety Classification, Quality Groups, and Seismic Categories of the CFS SSCs are listed in Table 3A-1.

The Safety Category functions associated with FW reactor and system isolation are described in Subsection 5.4.5. The Safety Category functions associated with FW containment isolation are described in Subsection 6.2.4.

10.4.7.2 System Description

The CFS supplies FW to the RPV at the required pressure, temperature, and flowrate to maintain RPV level in the required band. In addition, the CFS supplies water to various interface systems and responds to signals from the I&C systems. The CFS also provides the necessary indication, isolation functions, and design redundancy to support the required Safety Category functions.

The CFS is located in the TB, RB, and Containment. Figures 10.4-7 and 10.4-8 show the condensate portion of the CFS and the FW portion of the CFS, respectively. Figures 10.4-9 and 10.4-10 show the vents and drains for the CFS HP heaters and the LP heaters, respectively.

The CFS consists of the following components:

- Two (2) 100% capacity condensate pumps
- Three (3) LP FWHs arranged with two (2) 50% capacity duplex FWHs (stages one and two) in parallel and one (1) 100% capacity stage three FWH in series
- One (1) Low Flow Control Valve
- Two (2) 100% capacity reactor FW pumps
- Three (3) HP FWHs arranged in series (stages four, five, and six)
- Bypass valves around each FWH
- Piping, valves, and instrumentation

The CFS is designed to avoid FAC damage. The design and layout of the piping system considers the effects on the piping material of fluid velocity, bend location and the location of flash points. The majority of the CFS is constructed of FAC-resistant material. Material and equipment selection for the CFS is based on the range of normal environmental conditions that may exist at the location of the system's equipment and components.

Normal Operations

The condensate pumps and FW pumps are operated on separate buses to spread the load change in case of an electrical bus trip. The LP and HP FWH isolation valves are normally open. The LP and HP FWH bypass valves are normally closed. During power operation, reactor vessel water level is controlled by the Reactor Level Control (RLC) which controls FW flow by adjusting FW pump speed. The CFS modulates a control valve to receive MS to control final FW temperature entering the RPV. The CFS continuously drains the condensed HP and LP extraction steam in a cascading setup from FWH 6 to FWH 1. Normal and alternate drain line control valves maintain water level in each HP and LP FWH based on the signal from the corresponding level transmitters. The CFS continuously vents vapor and NC gases from the closed FWHs to the main condenser.

Off-Normal Operations

Condensate or FW Pump Trip

The CFS is normally operated with one condensate and one FW pump in service and one condensate and one FW pump on stand-by. In the event of a condensate or FW pump trip, the stand-by pump starts to continue supplying condensate to the FW pump suction, or FW to the RPV. The CFS has built in logic to immediately start the stand-by pump to minimize the system

transients to the system and RPV. The SC3 I&C trips the condensate pump trip on high reactor level. The SC1 I&C trips the FW pump on high reactor water level.

One LP or HP FWH Out of Service or Isolated

The CFS can be operated with one LP FWH or one HP FWH out of service by closing the FWH inlet and outlet isolation valves and opening the FWH bypass valve. The extraction steam HP or LP isolation valves for the affected heater close, the non-return valves for the affected heater close, and the extraction steam drain isolation valves open, providing a path to the main condenser. The SC3 DCIS provides an automatic reactor power reduction signal upon isolation of one of the LP or HP FWHs.

10.4.7.3 Safety Evaluation

The CFS complies with the following GDC, which have plant-wide applicability, as discussed in Section 3.1:

GDC 2 – Design bases for protection against natural phenomena

GDC 4 – Environmental and dynamic effects design bases

Additional information on GDC compliance is found in Section 3.1.

The design of the CFS complies with BTP 3-4 as described in Section 3.6.

Feedwater nozzle design complies with NUREG-0619, Revision 1 "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking." Details associated with FW nozzle design are provided in Subsection 3.9.3.

The design of the CFS complies with NUREG-0927. The layout of the CFS and the characteristics of the valves is such that water hammer loads are below the reactor design limits. Vents are located in appropriate high points and drains are located in appropriate low points ensuring the system is completely filled with water, with no air pockets. The FW control system is designed to ensure that there are no large sudden changes in FW flow that could induce water hammer.

Plant response to an SBO, as required by 10 CFR 50.63, is addressed in Section 8.4.

Process samples taken from the CFS are piped to the PS for sampling. Details regarding the PS are provided in Subsection 9.3.2.

Pumps located in radiation areas are designed to minimize the time required for maintenance. The anticipated sources of radiation and radiation levels in the RB and TB, along with shielding and access control requirements to ensure occupational exposures are ALARA, are provided in Chapter 12.

The CFS is designed to conform with the following codes and standards:

- The SIR is designed to ASME BPVC Section III Subsection NF requirements
- CFS pumps are designed to the general requirements of the Hydraulic Institute Standards for Centrifugal, Rotary, and Reciprocating Pumps (Reference 10.4-9)
- The FWHs are designed in accordance with ASME BPVC, Section VIII, Division 1 and HEI-2622, Standards for Closed Feedwater Heaters / HEI-2629 “Standards for Steam Surface Condensers” (References 10.4-10 and 10.4-1)
- The CFS piping in the TB, up to the SIR is designed to meet the requirements of ASME B31.1

10.4.7.4 Tests and Inspections

Routine testing of the CFS is conducted in accordance with normal power plant requirements for demonstrating system and component functionality and integrity. Suitable access and/or remote functions to permit Inservice Testing (IST) and inspection of the FW piping are provided. Inservice Inspection (ISI) accessibility is provided by appropriate arrangement of piping and major equipment. Accessible arrangement of vents and drains in the system comply with ASME BPVC Section XI requirements for the performance of ISI and testing for assessing operational readiness. Areas that require inspection are provided with access space and removable insulation.

CFS piping is subject to a program of nondestructive examination to verify structural integrity in accordance with EPRI NP-3944, “Erosion/Corrosion in Nuclear Power Plant Steam Piping: Causes and Inspection Guidelines,” (Reference 10.4-11), EPRI 3001000563, “Recommendations for an Effective Flow-Accelerated Corrosion Program (NSAC-202L-R2),” (Reference 10.4-12), and USNRC Generic Letter 89-08, “Erosion/Corrosion-Induced Pipe Wall Thinning.”

10.4.7.5 Instrumentation and Controls

The CFS is operated and monitored from the MCR in either automatic or manual modes. Instrumentation is provided to verify condensate and FW pump and FWH performance.

The CFS includes three divisions of SC1 instrumentation for measuring the necessary parameters required by the Reactor Isolation Function Logic (e.g., pressure and flow) to detect a line break.

10.4.8 References

10.4-1 HEI 2629 “Standards for Steam Surface Condensers, 12th Edition,” Heat Exchange Institute, 2017.

10.4-2 ASME TDP-2 “Prevention of Water Damage to Steam Turbines Used for Electric Power Generation: Nuclear-Fueled Plants,” American Society of Mechanical Engineers.

10.4-3 ASME BPVC Section VIII, Division 1, “Rules for Construction of Pressure Vessels,” American Society of Mechanical Engineers.

10.4-4 EPRI TR-106345, “Moisture Separator Reheater Source Book,” Electric Power Research Institute.

10.4-5 EPRI TR-1013462, “Turbine – Generator Auxiliary Systems,” Vol. 2., Electric Power Research Institute, 2006.

10.4-6 ANSI FCI 70-2, “Standard for Control Valve Seat Leakage,” American National Standards Institute.

10.4-7 ANSI/HI 9.8, “Rotodynamic Pumps for Pump Intake Design.” American National Standards Institute/Hydraulic Institute

10.4-8 EPRI "BWRVIP-190: “BWR Vessel and Internals Project, Revision 1, 2019 Interim Guidance,” Electric Power Research Institute.

10.4-9 “Standards for Centrifugal, Rotary, and Reciprocating Pumps,” Hydraulic Institute.

10.4-10 HEI-2622, “Standards for Closed Feedwater Heaters, 9th Edition,” Heat Exchange Institute.

10.4-12 EPRI NP-3944, “Erosion/Corrosion in Nuclear Power Plant Steam Piping: Causes and Inspection Guidelines,” Electric Power Research Institute.

10.4-13 EPRI 3001000563, “Recommendations for an Effective Flow-Accelerated Corrosion Program (NSAC-202L-R2),” Electric Power Research Institute.

10.4-14 ASME B16.5, “Pipe Flanges and Flanged Fittings: NPS 1/2 Through NPS 24, Metric/Inch Standard,” American Society of Mechanical Engineers.

10.4-15 ASME B16.34, “Valves Flanged, Threaded and Welding End,” American Society of Mechanical Engineers.

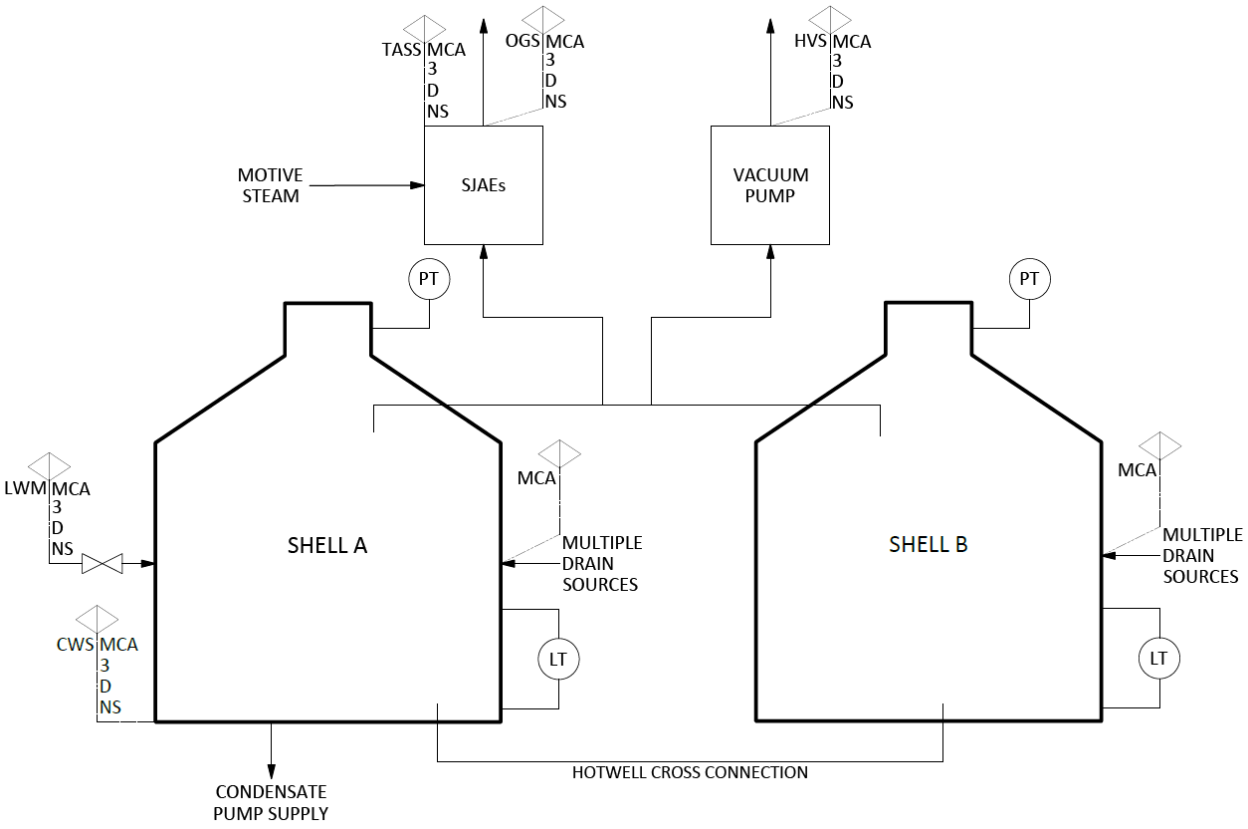


Figure 10.4-1: Main Condenser and Auxiliaries Simplified Diagram

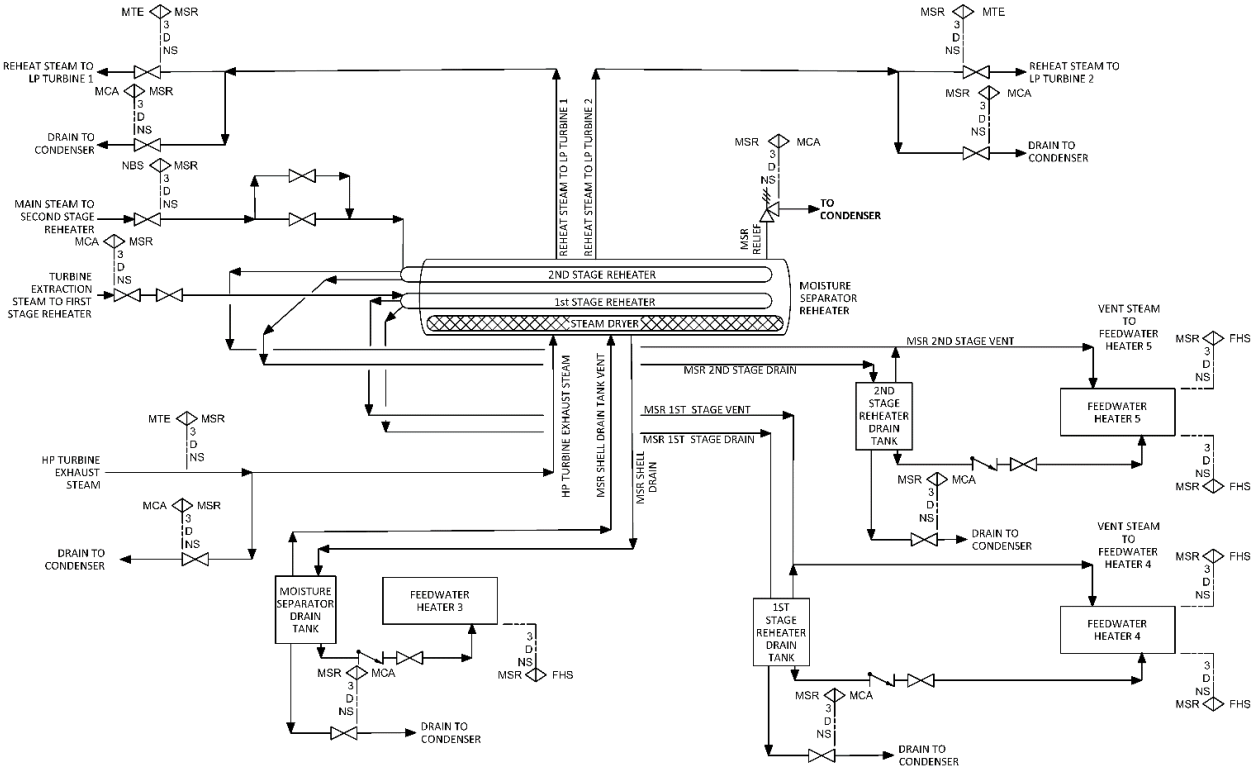


Figure 10.4-2: Moisture Separator Reheater System Simplified Diagram

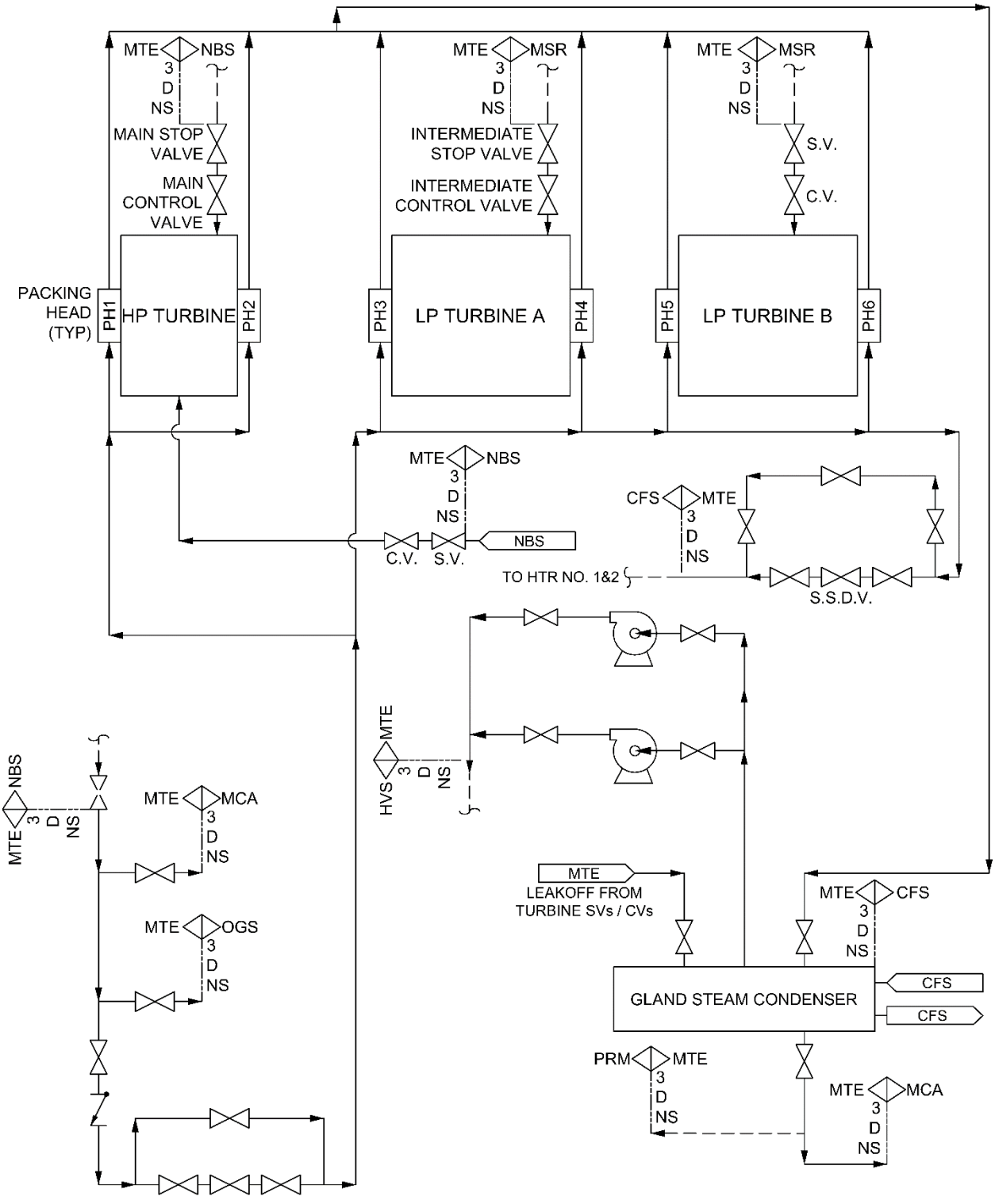


Figure 10.4-3: Turbine Gland Seal System Simplified Diagram

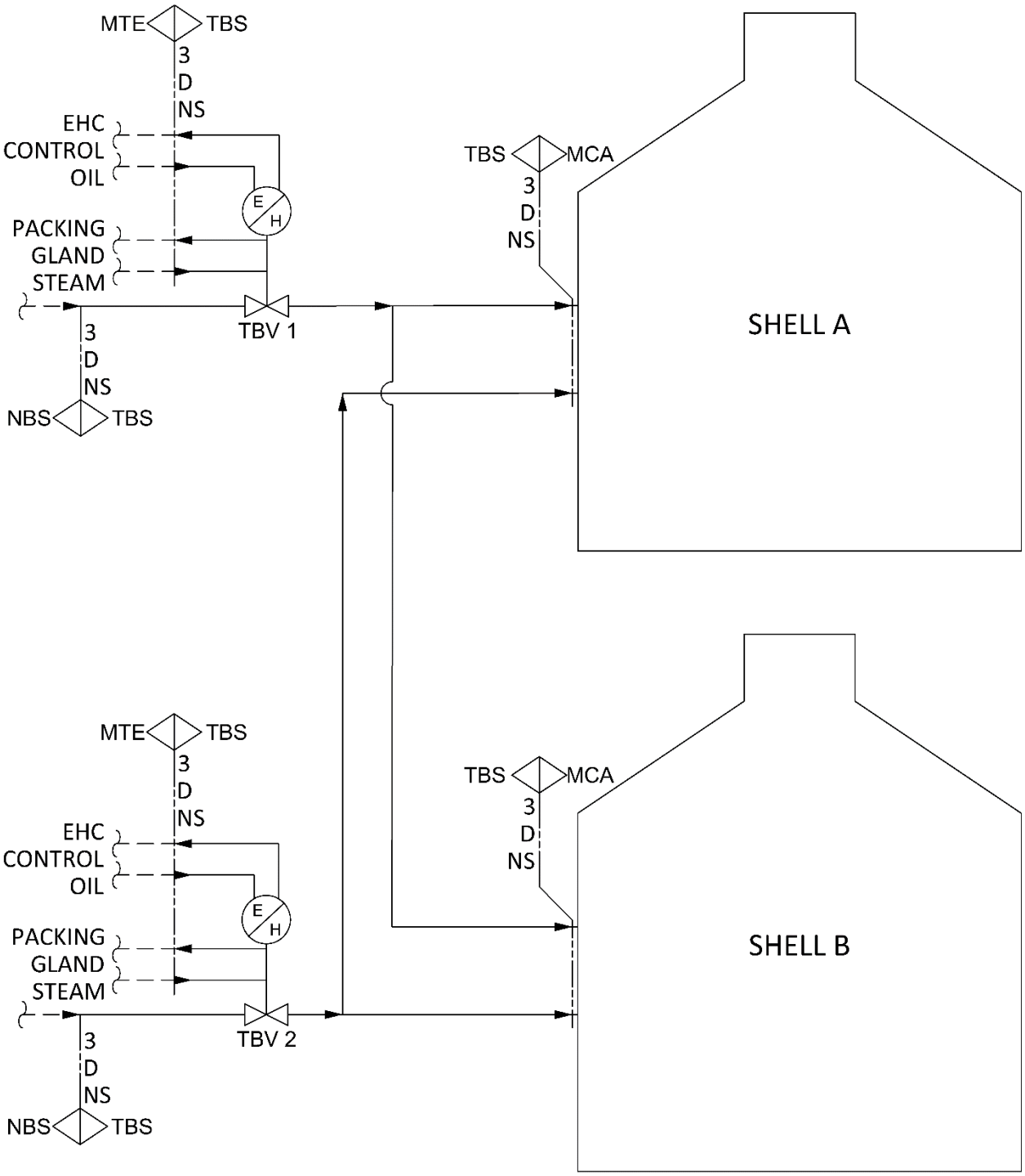


Figure 10.4-4: Turbine Bypass System Simplified Diagram

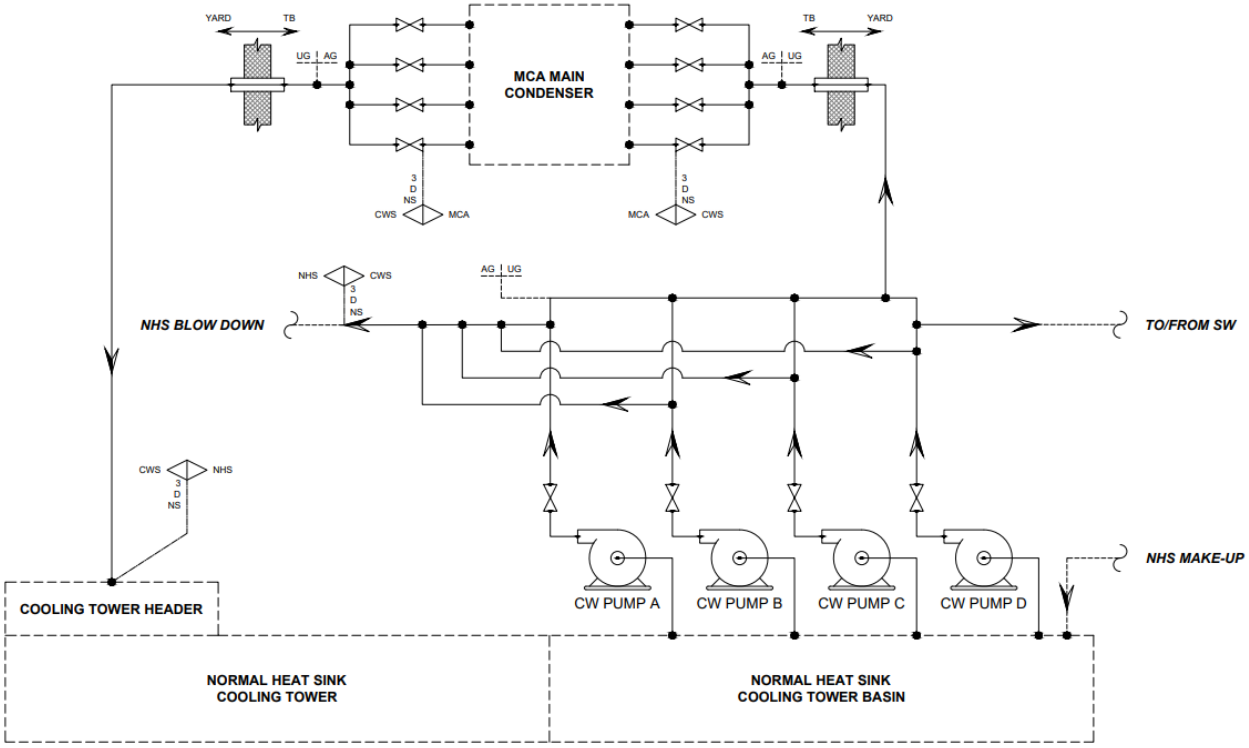


Figure 10.4-5: Circulating Water Supply Simplified Diagram

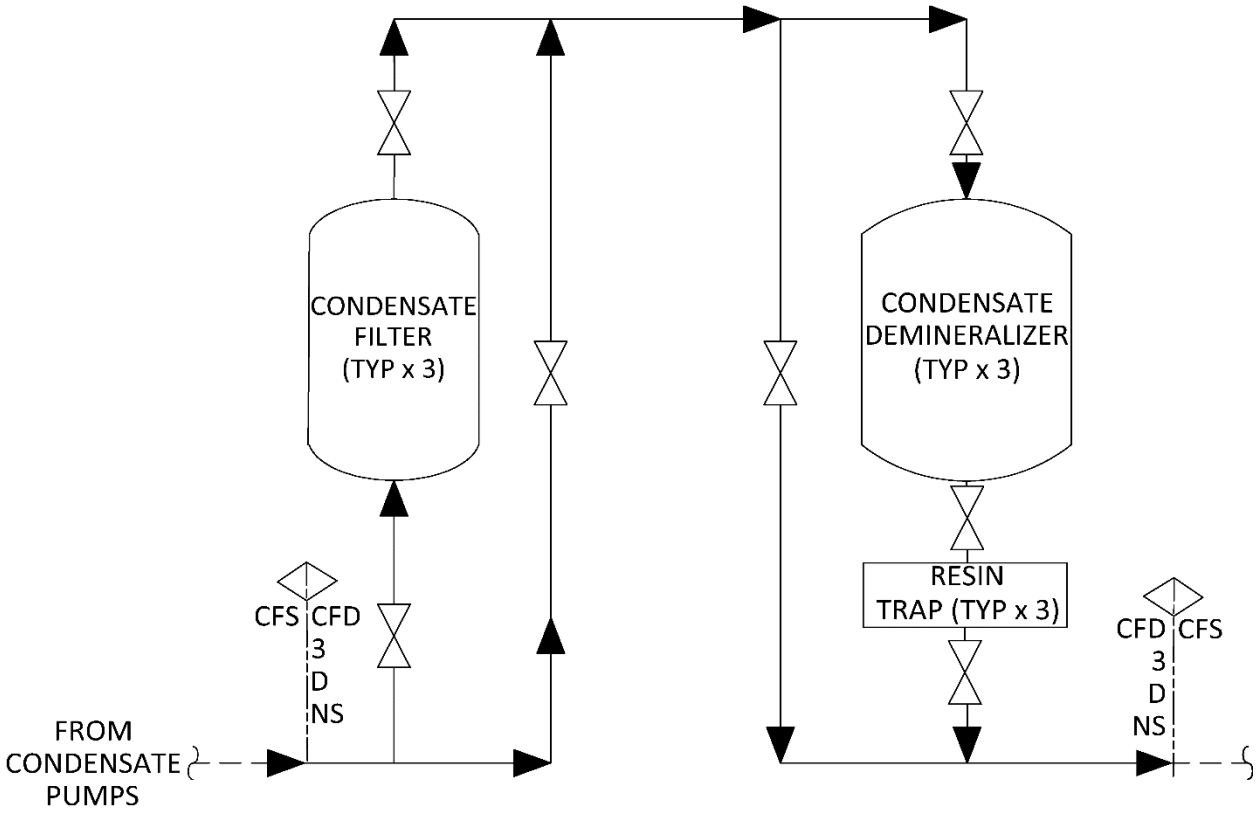


Figure 10.4-6: Condensate Filters and Demineralizers System Simplified Diagram

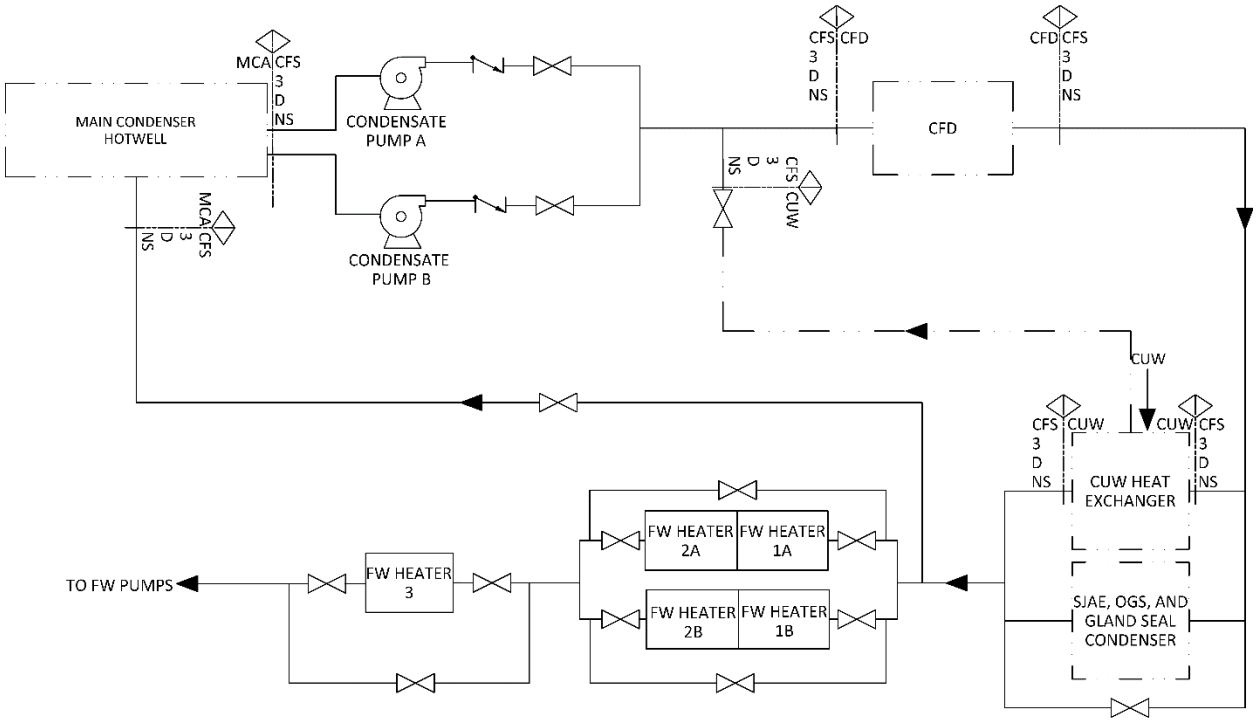
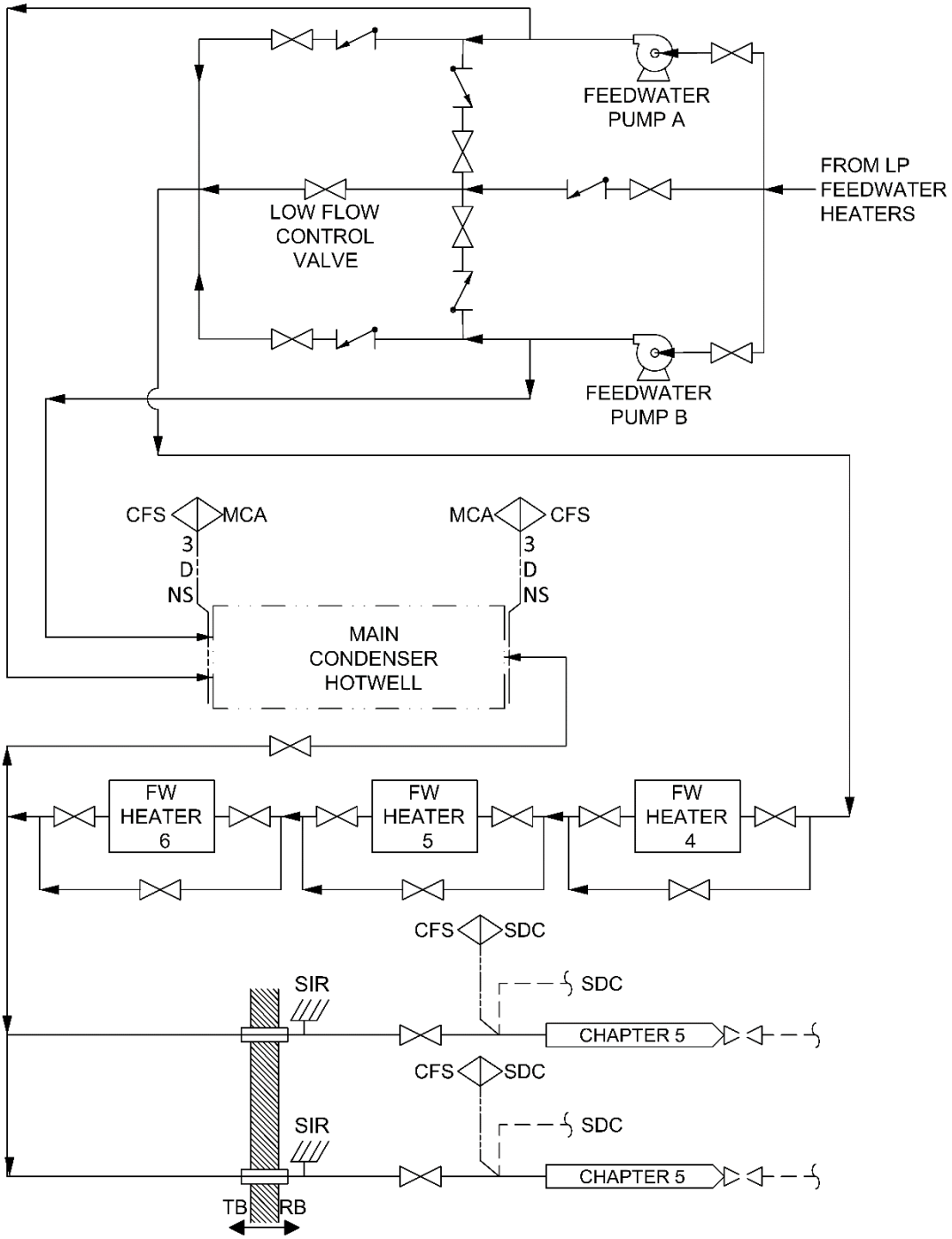


Figure 10.4-7: CFS Condensate Portion Simplified Diagram



1. PIPING IN THE RB IS SC1, QUALITY GROUP B, SEISMIC CATEGORY I. PIPING IN THE TB IS SC3, QUALITY GROUP D, SEISMIC CATEGORY NS.
2. PIPING IN THE RB UP TO THE SIR IS DESIGNED TO ASME SECTION III, CLASS 2. THE SIR IS DESIGNED TO ASME SECTION III, CLASS NF. PIPING IN THE TB IS DESIGNED TO ASME B31.1.

Figure 10.4-8: CFS Feedwater Portion Simplified Diagram

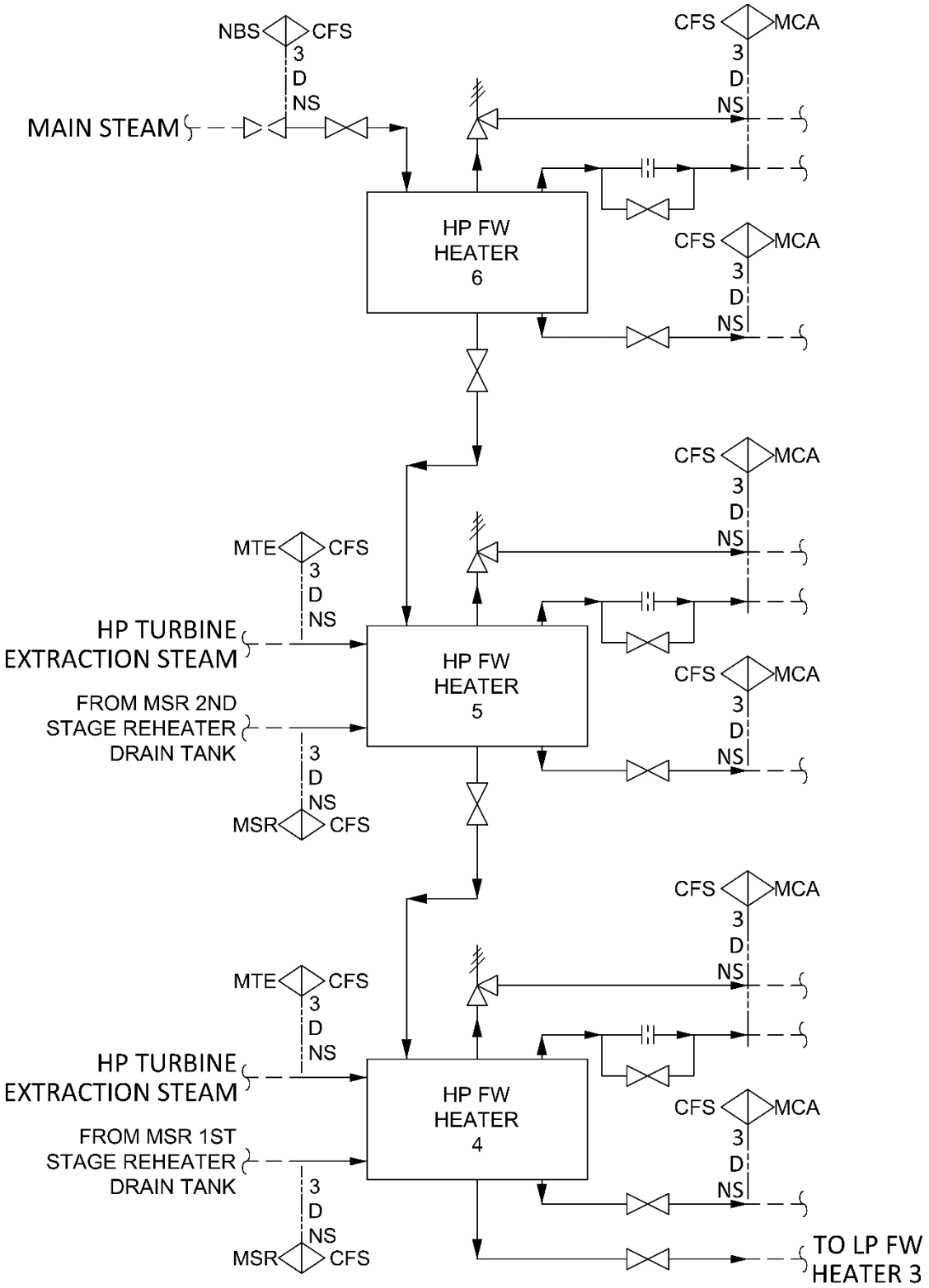


Figure 10.4-9: CFS HP Heaters and Drains Simplified Diagram

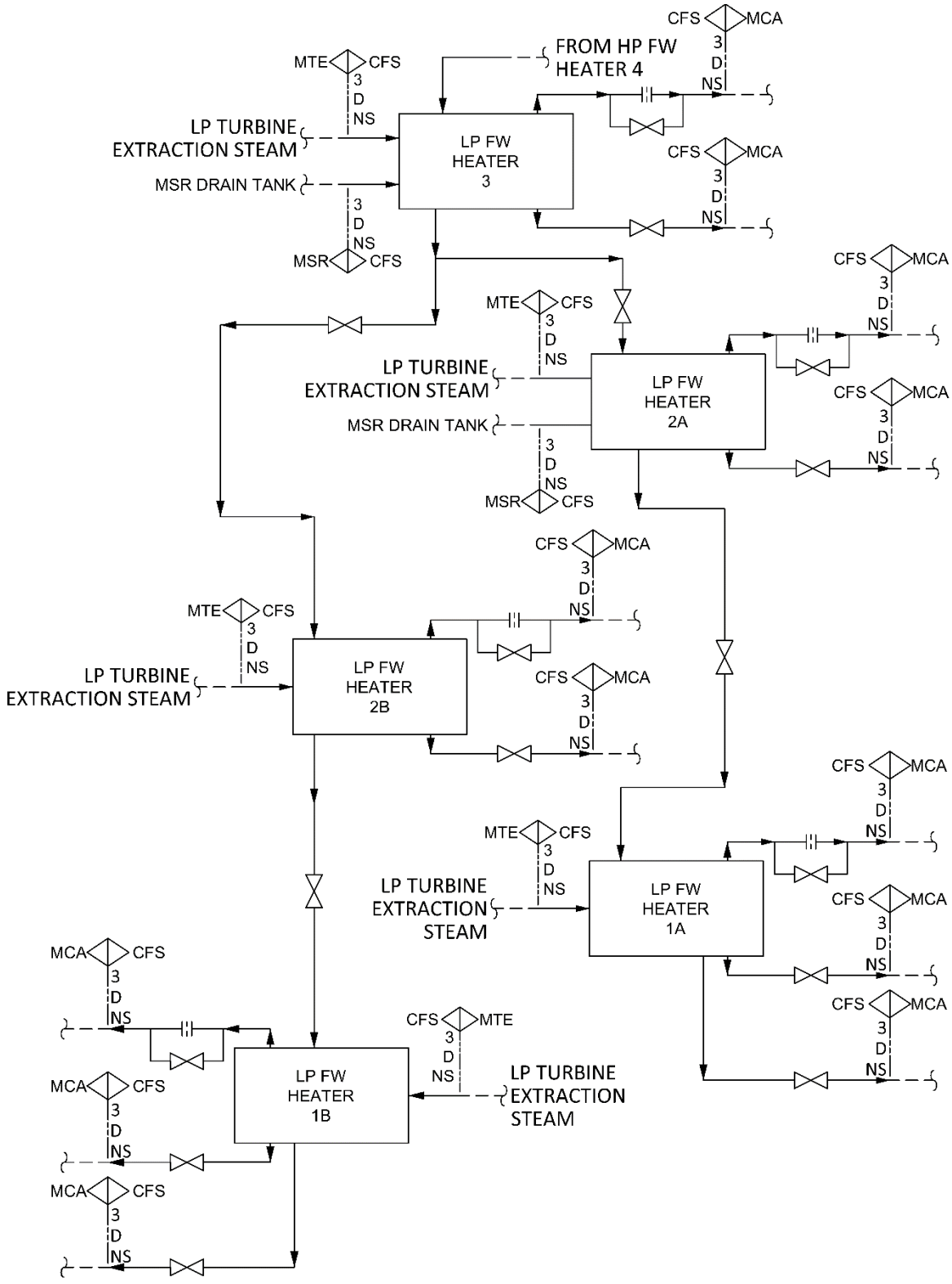


Figure 10.4-10: CFS LP Heaters and Drains Simplified Diagram

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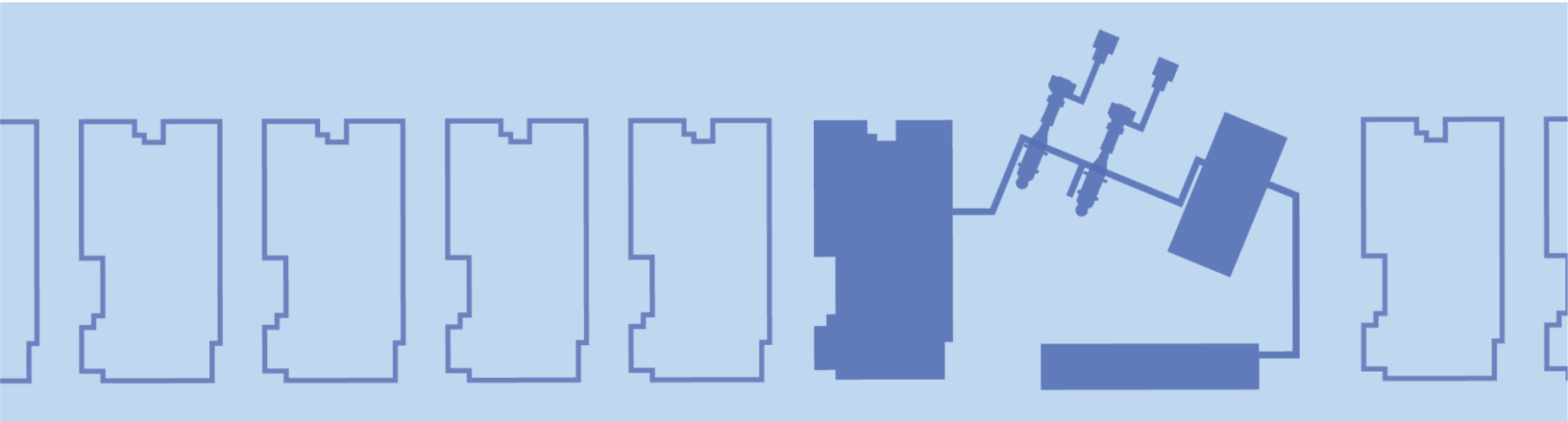
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September 2026

Non-Proprietary Information

Chapter 11
Radioactive Waste Management



ACRONYM LIST

Acronym	Explanation
AHU	Air Handling Unit
ALARA	As Low As Reasonably Achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
ARM	Area Radiation Monitoring Subsystem
ASME	American Society of Mechanical Engineers
BPVC	Boiler and Pressure Vessel Code
BTP	Branch Technical Position
BWR	Boiling Water Reactor
CFD	Condensate Filters and Demineralizer System
CFR	Code of Federal Regulations
CFS	Condensate and Feedwater Heating System
CIS	Containment Inerting System
CIV	Containment Isolation Valve
CMon	Containment Monitoring Subsystem
CO	Carbon Monoxide
CR	Control Room
CRE	Control Room Envelope
CST	Condensate Storage Tank
CUW	Reactor Water Cleanup System
CWE	Chilled Water Equipment
CWS	Circulating Water System
DBA	Design Basis Analysis
DCIS	Distributed Control and Information System
DF	Decontamination Factor
EAB	Exclusion Area Boundary
EFS	Equipment and Floor Drain System
EFU	Emergency Filter Unit
EPRI	Electric Power Research Institute
FPC	Fuel Pool Cooling and Cleanup System

Acronym	Explanation
GDC	General Design Criteria
GVH	GE Vernova Hitachi Nuclear Energy
GNF	Global Nuclear Fuel
GSC	Gland Seal Condenser
HEPA	High Efficiency Particulate Air
HIC	High Integrity Container
HVS	Heating, Ventilation, and Cooling System
HWC	Hydrogen Water Chemistry
ICC	Isolation Condenser Pools Cooling and Cleanup System
LWM	Liquid Waste Management System
MCA	Main Condenser and Auxiliaries
MCR	Main Control Room
MS	Main Steam
MSL	Main Steam Line
NHS	Normal Heat Sink
NUREG	Nuclear Regulatory Report
OGS	Offgas System
PCW	Plant Cooling Water System
PING	Particulate, Iodine and Noble Gas
PPE	Plant Parameter Envelope
PPS	Plant Pneumatics System
PREMS	Process Radiation and Environmental Monitoring System
PRM	Process Radiation Monitoring Subsystem
PS	Process Sampling Subsystem
PVS	Plant Vent Stack
RB	Reactor Building
RG	Regulatory Guide
RPV	Reactor Pressure Vessel
RWB	Radwaste Building
RWST	Refueling Water Storage Tank
SCR	Secondary Control Room
SDC	Shutdown Cooling System

Acronym	Explanation
SJAE	Steam Jet Air Ejector
SMR	Small Modular Reactor
SSCs	Structures, Systems, and Components
SWM	Solid Waste Management System
TB	Turbine Building
TGFU	Toxic Gas Filtration Unit
TMR	Triple Modular Redundant
USNRC	United States Nuclear Regulatory Commission

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11.0 RADIOACTIVE WASTE MANAGEMENT

11.1 Source Terms

Two source terms are presented in this section. The first source term is a realistic model used to predict expected long-term average concentrations of radionuclides in the principal fluid stream. The second is a conservative design basis model that utilizes an established GVH fuel clad defect model.

The realistic model, which is based on available measured nuclide concentrations during normal operation at operating Boiling Water Reactors (BWRs), is formulated as a standard for the American National Standards Institute (ANSI) Source Term Specifications and American Nuclear Society (ANS) 18.1, ANSI/ANS 18.1-2020, “Radioactive Source Term for Normal Operation of Light Water Reactors,” (Reference 11.1-1), and is the source term model used to estimate liquid and gaseous effluent releases. The realistic model is referred to as the normal operation source term.

The second source term conservatively employs a GVH fuel clad defect model and serves as a basis for system and shielding requirements and is referred to as the design basis coolant source term.

11.1.1 Sources of Waste

Liquid radioactive waste is generated by cleanup of the reactor coolant, spent fuel pool water, decontamination activities, and leakage from process streams. Leaked liquids are collected through floor drains, equipment drains, and process drains in the Reactor Building (RB), Turbine Building (TB), and Radwaste Building (RWB). These waste sources are collected by drains and drain headers, routed by gravity drainage piping to a sump, and pumped to Liquid Waste Management System (LWM) for processing.

Gaseous radioactive waste is generated as a result of activation of the reactor coolant during normal reactor operation, tramp uranium that may adhere to the outer surface of the fuel cladding during fabrication, and leakage of fission gas through miniscule defects in the fuel cladding.

Solid waste is created through plant operations, decontamination activities and through processing liquid radwaste. It includes the filter backwash sludges, charcoal media, and bead resins generated by the Fuel Pool Cooling and Cleanup System (FPC), Isolation Condenser Pools Cooling and Cleanup System (ICC), and Condensate Filters and Demineralizer Systems (CFDs). Contaminated solids such as High Efficiency Particulate Air (HEPA) and cartridge filters, rags, plastic, paper, clothing, tools, and equipment are also disposed of in the Solid Waste Management System (SWM).

11.1.2 Coolant Radiation Concentrations

The normal operation BWRX-300 reactor water and steam radiation concentrations are based on BWR operating data reported in the ANSI/ANS-18.1-2020 Standard. The standard provides bases for estimating typical concentrations of the principal radionuclides that may be anticipated over the lifetime of a BWR. The source term data is based on the cumulative industry experience at operating BWR plants, and measurements at several stations. It therefore reflects the influence of a number of observations made during the transition period from operation with fuel of older designs to operation with fuel of current improved designs. Normal operation coolant source terms are obtained by applying the procedures of ANSI/ANS-18.1-2020 for estimation of typical source terms and adjusting the results, as appropriate, to assure conservative bases for design.

The Condensate and Feedwater Heating System (CFS) provides a configuration of feedwater heater drains where one drain cascades to the next successively, ultimately draining to the Main

Condenser Hotwell, thus, corrosion products from these drains are filtered via condensate filter/demineralizers before returning to the Reactor Pressure Vessel (RPV). In addition, the BWRX-300 cleanup system configuration differs from the reference configuration in ANSI/ANS-18.1-2020, and typical BWRs, since the Reactor Water Cleanup System (CUW) does not include any filtering media. The CFD performs the cleanup of the reactor coolant in the BWRX-300 design. To account for the differing BWRX-300 cleanup configuration, the ANSI/ANS-18.1 removal rate equation (i.e., Subsection 11.1.4.2, Equation 11.1-3) is modified to account for the BWRX-300 cleanup configuration by sending the steam flow and CUW flow through the CFD as shown in Subsection 11.1.4.2, Equation 11.1-4.

The ANSI/ANS-18.1-2020 BWR Reference Plant reactor water and steam concentrations were updated based on data from operating domestic nuclear power plants collected from 1999 to 2015 and include contribution from normal operating events such as fuel leakers, implementation of different chemistry and mitigation strategies, and mid-cycle outages as established by the Electric Power Research Institute (EPRI) 3002009584, “Advanced Nuclear Technology: Reactor Coolant Radiological Source Terms for Normal Operation–Public Version: Technical Bases for Updating the ANSI/ANS 18.1-1999 Standard to Incorporate Contemporary Best-Estimate Radiological Source Terms in Principal Fluid Streams of Light Water Reactors,” (Reference 11.1-2).

While the BWRX-300 is a Small Modular Reactor (SMR), the fuel type, fuel burnup levels, fuel enrichment, and power to steam flow ratio are similar to the BWRs providing the operating data reviewed by EPRI 3002009584 used to generate the ANSI/ANS-18.1. Based on the similarities, ANSI/ANS-18.1 is applicable to the BWRX-300.

The radionuclides included in the normal operation and design basis coolant source term are categorized as fission products, activation products, and activated corrosion products. The lists do not necessarily include all radionuclides that may be detectable or theoretically predicted to be present. The principal nuclides identified in ANSI/ANS18.1 are included because they are potentially significant with respect to one or more of the following criteria:

- Plant equipment design
- Shielding design
- Understanding system operation and performance
- Measurement practicability

The design basis coolant source term employs a GVH fuel clad defect model that applies the following fuel leakage to the normal operation coolant source term which bound the expected fuel leakage rates for an operating BWRX-300 reactor:

- Noble gas fuel leak rate of 20,000 µCi/sec after a 30-minute decay
- I-131 fuel leak rate of 250 µCi/sec

The noble gas fuel leakage rate is intended for use in shielding models and ventilation design. The subsequent design basis noble gas radionuclide source term is provided in Table 11.1-1.

The GVH fuel clad defect model is developed based on more than 50 years of BWR fuel leakage operating data from several representative BWRs that experienced a variety of fuel defects. This review of modern fuel performance operating data identified two trends. First, the reliability of BWR fuel has increased dramatically over the last 25 years driven by improvements in fuel fabrication, design, plant operation, and chemistry while, at the same time, techniques for managing leaking fuel have been modernized. Second, BWR operators that utilize modern fuel

defect detection, characterization, and suppression techniques do not observe fuel leakage rates as high as those established the BWRX-300 clad defect model.

The GVH fuel clad defect model assumes the operator of any BWRX-300 will employ up to date modern BWR fuel failure detection, characterization, and suppression procedures similar to those documented in the Global Nuclear Fuel (GNF) “Fuel Operating Guidelines,” (Reference 11.1-3) or in the EPRI 3002016030, “Boiling Water Reactor Fuel Reliability Monitoring and Failure Evaluation Handbook,” (Reference 11.1-4). If these procedures, or similar techniques are employed by a BWRX-300 operator, the GVH clad defect model bounds fuel leakage rates that will be observed during power operation of the reactor.

11.1.2.1 Fission Products

Normal Operation Noble Gas Radionuclides

The BWRX-300 normal operation noble gas source term consists of the thirteen principal noble gas fission products as observed in steam flowing from the reactor vessel which are provided in ANSI/ANS-18.1-2020. Concentrations in the reactor water are considered negligible under normal power operation because the gases released to the coolant are assumed to be rapidly transported out of the vessel with the steam and removed from the system with the other non-condensable gases in the main condenser. Due to the rapid removal of the gases, the expected relative mix of gases does not depend on the reactor design. The normal operation principal noble gas radionuclides source terms are presented in Table 11.1-2.

Design Basis Noble Gas Radionuclides

The design basis noble gas source terms are developed by normalizing the ANSI/ANS-18.1 normal operation noble gas source term mixture to a fuel release rate of 87,000 µCi/sec at 30 minutes decay. The design basis noble gas radionuclide source term rate after 30-minute decay has been used as a conventional measure of the fuel leakage rate because it is conveniently measurable and is consistent with the nominal 30-minute offgas holdup system used on a number of early plants. The United States Nuclear Regulatory Commission (USNRC) Standard Review Plan, Branch Technical Position (BTP) 11-5, “Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure,” states that releases should be based on 100 µCi/sec per MWt with 30 minutes of decay.

The principal design basis noble gas radionuclide source terms are presented in Table 11.1-3. This source term is intended for use in the Offgas System (OGS) failure analysis presented in Subsection 11.3.3 and design basis accident analyses presented in Chapter 15.

Normal Operation Radioiodine

The BWRX-300 normal operation radioiodine concentrations of the five principal radioiodines in reactor water and steam are determined by adjusting BWR Reference Plant radioiodine concentrations using the methods in ANSI/ANS-18.1 and the BWRX-300 operating parameters in Table 11.1-4. The normal operation radioiodine concentrations are presented in Table 11.1-5.

Design Basis Radioiodine

The design basis radioiodine concentrations are based on the relative mix of radioiodines in reactor water predicted by the data of ANSI/ANS-18.1 with magnitudes increased by normalizing the I-131 concentrations to the GEH fuel clad defect model I-131 release rate of 250 µCi/sec. ANSI/ANS-18.1 specifies expected concentrations of the five principal radioiodines in reactor water for a BWR Reference Plant design and provides bases for adjusting the concentrations for plants with relevant plant parameters that do not match those of the Reference Plant. The design basis radioiodine concentrations in steam and water are presented in Table 11.1-6.

The ratio of concentration in reactor steam to concentration in reactor water (carryover ratio) is taken to be 0.02 for radioiodines, (Reference 11.1-1). Consequently, the design basis radioiodine concentrations in steam are calculated by multiplying the design basis radioiodine water concentrations in Table 11.1-5 by the factor 0.02.

Other Fission Products

This category includes fission products other than noble gases, and iodines and includes transuranic nuclides. Some of the fission products are noble gas daughter products that are produced in the steam and condensate system. The only detectable transuranic in significant concentrations is Np-239.

The BWRX-300 normal operation non-volatile fission product concentrations for fission products other than noble gases and iodine in reactor water and steam, are determined by adjusting BWR Reference Plant radioiodine concentrations using the methods in ANSI/ANS-18.1-2020 and the BWRX-300 operating parameters in Table 11.1-3. In order to assure conservative design basis concentrations for the BWRX-300, the normal operation concentrations of the non-volatile fission products are normalized to the I-131 leakage rate from the fuel reported in Table 11.1-7.

The normal operation non-volatile fission product concentrations are presented in Table 11.1-8, and the design basis concentrations are presented in Table 11.1-9.

11.1.3 Activation Products

11.1.3.1 Coolant Activation Products

The coolant activation product of primary importance in BWRs is N-16. Based on the ANSI/ANS-18.1 BWR Reference Plant and the BWRX-300 operating parameters, the N-16 levels in steam leaving the reactor vessel for a plant without Hydrogen Water Chemistry (HWC) is 50 µCi/gm, and 250 µCi/gm with high HWC injection rates. Under HWC conditions, the volatility of the N-16 in BWR water increases. As the feedwater hydrogen concentration is increased beyond approximately 0.4 ppm, the radiation levels in the main steam increase rapidly and then asymptotically approach levels 4 to 5 times the initial value at the higher feedwater hydrogen concentrations tested of approximately 2 ppm.

The BWRX-300 will employ On-line NobleChem™ technology along with the HWC which will allow the rate of hydrogen injection to be decreased such that the feedwater hydrogen concentration required is expected to be in the range of 0.2 to 0.4 ppm which implies no increase in operating dose rates due to increased concentrations of N-16 in the steam. The HWC system design maximum hydrogen injection rate corresponds to a feedwater hydrogen concentration of 0.5 ppm with the plant at the 100% power.

The normal operation source term is used to assess performance of radioactive waste management systems and other systems under normal operating conditions, the BWR Reference Plant concentrations of 60 µCi/gm for reactor water and 50 µCi/gm for reactor steam are adequately conservative for the normal operation source term. The normal operating N-16 concentrations are presented in Table 11.1-10.

The BWRX-300 design basis coolant concentrations are used for shielding design assessments. An increase in the HWC injection rate results in an increase in N-16 concentration in steam. The maximum HWC injection rate is assumed for conservatism. The BWRX-300 design basis N-16 concentrations are estimated to be 60 µCi/gm for reactor water and 130 µCi/gm for reactor steam as presented in Table 11.1-11.

11.1.3.2 Non-Coolant Activated Corrosion Products

Radionuclides produced in the coolant by neutron activation of circulating impurities and by corrosion of irradiated system materials are classified as activated corrosion products. Typical reactor water concentrations for activated corrosion products are contained in ANSI/ANS-18.1.

The BWRX-300 activated corrosion product concentrations are calculated by adjusting the BWR Reference Plant concentrations of non-coolant activation products to BWRX-300 conditions by using the methods in ANSI/ANS-18.1 and appropriate data from Table 11.1-4.

The BWRX-300 design basis concentrations of the activated corrosion products are not normalized using the fuel leakage model because they do not originate in the fuel but are produced by corrosion of irradiated system materials in the reactor and associated piping. Thus, the production of activated corrosion products is dependent on the inner surface area of the reactor and piping where gamma and neutron activation can occur, and the BWRX-300 has much less surface area than the large BWRs that provided the operating data bases for BWR Reference Plant concentrations.

The activated corrosion product concentrations in reactor water and steam are presented in Table 11.1-12. The Table 11.1-12 concentrations are used for both normal operation and design basis source terms.

Tritium

The BWRX-300 does not use boron in reactor water, so the production of tritium is limited to activation of naturally occurring deuterium in the primary coolant and, to a lesser extent, as a fission product in the fuel. In addition, the BWRX-300 design is capable of recycling all wastewater. The recycling of reactor water allows the BWRX-300 to operate with a zero liquid release strategy for radiological effluents that provides the capability to release tritium as a gaseous effluent instead of a liquid effluent. Zero liquid release for radiological effluents is a philosophy of processing wastewater in a way such that no wastewater generated at the power plant is released into local bodies of water and the environment in general.

Because tritium has a relatively long half-life (12 years) and is not affected by cleanup processes in the system, the concentration of tritium in reactor coolant is controlled by the rate of loss of water from the system by evaporation or leakage. The concentration of tritium is influenced by factors such as evaporation rates from fuel pool water and the addition of pure water.

Plant process water and steam have a common tritium concentration. ANSI/ANS-18.1 specifies a concentration of 0.014 µCi/gm in both reactor water and reactor steam based on BWR operating data. Under the zero liquid release strategy, tritium levels in the coolant are conservatively assumed to reach an equilibrium level of twice that of a typical BWR that discharges liquid effluents.

Thus, for the BWRX-300, the tritium concentration in reactor water and steam is 0.028 µCi/gm.

Argon-41

Argon-41 is produced in the reactor coolant as a consequence of neutron activation of naturally occurring Argon-40 in air that is entrained in the feedwater. The Argon-41 gas is carried out of the vessel with the steam and stripped from the system with the non-condensable gases in the main condenser. Observed Argon-41 levels are highly variable due to the variability in air inleakage rates into the system. NUREG-0016, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling-Water Reactors,” specifies a normal operation Argon-41 release rate from the vessel into the offgas treatment system of 40 µCi/sec. This value

is considered conservative as it bounds the available experimental database; this value is provided in Table 11.1-7.

11.1.4 Radionuclide Concentration Adjustments

The BWRX-300 normal operation coolant source term (or operational source term) is determined using ANSI/ANS-18.1 bulk operating parameters to calculate adjustment factors that are then applied to a set of BWR Reference Plant reactor water and steam concentrations provided in ANSI/ANS-18.1. The BWR Reference Plant adjustment factors are based on the following formula:

$$C = \frac{s}{m(\lambda + R)}$$

Equation 11.1-1

where:

C = nuclide concentration

s = nuclide input rate to coolant

m = reactor water mass

λ = nuclide decay constant

R = the sum of removal rates of the radionuclide

11.1.4.1 Class 1 Noble Gas

The noble gas concentrations in the reactor water are negligible under normal power operation. Due to the rapid removal of the gases, the expected relative mix of gases does not depend on the reactor design. The BWR Reference Plant concentrations of noble gas in reactor steam do not require adjustment for the BWRX-300 for normal operation (Reference 11.1-1).

11.1.4.2 Class 2 Halogens, Class 3 Cesium/Rubidium, and Class 6 Other Nuclides

For Class 2, 3, and 6 nuclides, the radionuclide input rate is taken to depend primarily on the reactor thermal power. The adjustment factors to be applied to the BWR Reference Plant water concentrations are given by:

$$Adjustment\ Factor = \frac{P \cdot WP_n \cdot (R_{ni} + \lambda)}{WP \cdot P_n \cdot (R_i + \lambda)}$$

Equation 11.1-2

Where:

R = removal rates of the radionuclide

The subscript “n” refers to the reference plant

Subscript “i” refers to the isotope class

P = the reactor thermal power

WP = weight of water in the reactor vessel

λ = nuclide decay constant

The ANSI/ANS-18.1 removal rate from the system is the sum of the removal rates due to the condensate demineralizers. The nuclide Class 1, 4, and 5 are not removed by the CUW and the CFD.

The removal rate, $R_{2,3,6}$ for nuclide Classes 2, 3, and 6 is given by:

$$R_{2,3,6} = \frac{(FA \cdot NA) + (NC \cdot FS \cdot NS \cdot NB)}{WP} \qquad \text{Equation 11.1-3}$$

where:

FA = CUW flow rate

NA = fraction of nuclide removed in CUW

NC = ratio of CFD System flow rate to steam flow rate (=1 for BWRX-300)

FS = steam flow rate

NS = ratio of nuclide concentration in steam to concentration in water

NB = fraction of nuclide removed by the CFD System

WP = weight of water in the reactor vessel

However, the BWRX-300 coolant cleanup system configuration differs from the ANSI/ANS-18.1 Reference BWR removal. Based on the differing configuration of the BWRX-300, Equation 11.1-3 requires modification for application to the BWRX-300.

The CUW does not have any filtering function. The CUW is not designed to clean up reactor coolant, but rather to optimize the pressure and temperature of reactor coolant for discharge to the CFD. The CUW takes blowdown flow from the reactor and sends it to the CFD which purifies the condensate, as required, to maintain reactor feedwater purity. The CFD uses filtration to remove suspended solids, including corrosion products, and ion exchange resin to remove dissolved solids from condenser tube leakage and other impurities. The CFD is a full flow system that consists of high efficiency backwash-type filters followed by mixed bed demineralizers.

With regard to the ANSI/ANS-18.1, BWR Reference Plant configuration, the adjustment models the reactor water cleanup flow (FA) through the system where it is subject to removal (NA). The BWRX-300 sends its blowdown flow (FA) to the CFD where it is subject to cleanup (NB). For this reason, Equation 11.1-3 required modification for application to the BWRX-300 cleanup pathway configuration. Equation 11.1-4 simplifies the ANSI/ANS-18.1 adjustment to model the removal rate, $R_{2,3,6}$ for nuclide Classes 2, 3, and 6. As shown in Equation 11.1-4, the CUW blowdown flow is now subject to the CFD cleanup (NB).

$$R_{2,3,6} = \frac{(FA + FS \cdot NS \cdot NC)NB}{WP} \qquad \text{Equation 11.1-4}$$

11.1.5 Process Leakage Sources

Process leakage results in potential release of noble gases and other volatile fission products via ventilation systems. Contaminated liquid from process leaks is collected and routed to the liquid or solid radwaste system. The OGS minimizes and controls the release of radioactive material into the atmosphere by delaying release to allow for decay of the offgas stream containing radioactive isotopes of krypton, xenon, iodine, nitrogen, and oxygen.

The Heating, Ventilation, and Cooling Systems (HVS) are designed to ensure that airborne radiation exposures to plant personnel and releases to the environment are As Low As Reasonably Achievable (ALARA). The Plant Vent Stack and the major streams feeding the stack are monitored by the Process Radiation Monitoring Subsystem (PRM) so that action may be taken to avoid releases in excess of regulatory limits.

Subsection 12.2.2 delineates the models, parameters, and sources required to evaluate the airborne concentrations of radionuclides during plant operations in various plant radiation areas due to process leakage.

Releases due to process leakage are reflected in the airborne release estimates discussed in Subsection 11.3.4.

11.1.6 References

11.1-1 ANSI/ANS 18.1, “Radioactive Source Term for Normal Operation of Light Water Reactors,” American Nuclear Society, July 2020.

11.1-2 EPRI 3002009584, “Advanced Nuclear Technology: Reactor Coolant Radiological Source Terms for Normal Operation–Public Version: Technical Bases for Updating the ANSI/ANS 18.1-1999 Standard to Incorporate Contemporary Best-Estimate Radiological Source Terms in Principal Fluid Streams of Light Water Reactors,” Electric Power Research Institute, August 2017.

11.1-3 GNF-0142-5151, “GNF Fuel Operating Guidelines,” Global Nuclear Fuel - Americas, LLC, Revision 2, November 2017.

11.1-4 EPRI 3002016030, “Boiling Water Reactor Fuel Reliability Monitoring and Failure Evaluation Handbook,” Electric Power Research Institute,” Revision 0, October 2019.

Table 11.1-1: Shielding and Ventilation Design: Design Basis Noble Gas Radionuclide Source Term

Isotope	Decay Constant (Per hour)	Steam Concentration (µCi/gm)
Kr-83m	3.79E-01	1.3E-02
Kr-85m	1.55E-01	3.6E-04
Kr-85	7.35E-06	8.9E-05
Kr-87	5.45E-01	2.5E-03
Kr-88	2.44E-01	1.4E-03
Kr-89	1.32E+01	4.7E-01
Xe-131m	2.44E-03	7.4E-05
Xe-133m	1.32E-02	4.5E-05
Xe-133	5.51E-03	6.7E-04
Xe-135m	2.72E+00	1.2E-02
Xe-135	7.58E-02	6.7E-03
Xe-137	1.09E+01	2.2E-02
Xe-138	2.95E+00	6.3E-02

Table 11.1-2: Normal Operation Noble Gas Radionuclide Source Term

Isotope	Decay Constant (Per hour)	Steam Concentration (µCi/gm)
Kr-83m	3.79E-01	5.9E-04
Kr-85m	1.55E-01	1.6E-05
Kr-85	7.35E-06	4.0E-06
Kr-87	5.45E-01	1.1E-04
Kr-88	2.44E-01	6.2E-05
Kr-89	1.32E+01	2.1E-02
Xe-131m	2.44E-03	3.3E-06
Xe-133m	1.32E-02	2.0E-06
Xe-133	5.51E-03	3.0E-05
Xe-135m	2.72E+00	5.4E-04
Xe-135	7.58E-02	3.0E-04
Xe-137	1.09E+01	9.9E-04
Xe-138	2.95E+00	2.8E-03

Table 11.1-3: Offgas System Failure and Design Basis Accidents: Design Basis Noble Gas Radionuclide Source Term

Isotope	Decay Constant (Per hour)	Source Term at t=30min (µCi/s)
Kr-83m	3.79E-01	2.4E+04
Kr-85m	1.55E-01	7.2E+02
Kr-85	7.35E-06	2.0E+02
Kr-87	5.45E-01	4.1E+03
Kr-88	2.44E-01	2.7E+03
Kr-89	1.32E+01	1.4E+03
Xe-131m	2.44E-03	1.6E+02
Xe-133m	1.32E-02	9.7E+01
Xe-133	5.51E-03	1.5E+03
Xe-135m	2.72E+00	6.8E+03
Xe-135	7.58E-02	1.4E+04
Xe-137	1.09E+01	2.1E+02
Xe-138	2.95E+00	3.1E+04

Table 11.1-4: Calculational Parameters for Source Term Adjustment

Plant Parameters for Source Term Adjustment		
Parameter	**Reference Plant**	**BWRX-300**
Thermal Power, MWt	3400	870
Reactor Water Mass, kg (lb)	1.70E+05 (3.80E+05)	1.47E+05 (3.25E+05)
CUW Flow Rate, kg/hr (lb/hr)	1.60E+01 (1.30E+05)	5.07E+00 (4.02E+04)
Steam Flow Rate, kg/hr (lb/hr)	1.90E+03 (1.50E+07)	5.03+02 (3.99E+06)
Ratio of Condensate Demineralizer Flow Rate to Steam Flow Rate	1	1

Table 11.1-5: Normal Operational Radioiodine in Reactor Water and Steam

Isotope	Decay Constant (per hour)	Water Concentration (µCi/gm)	Steam Concentration (µCi/gm)
I-131	3.60E-03	7.4E-05	1.5E-06
I-132	3.02E-01	7.4E-04	1.4E-05
I-133	3.33E-02	5.5E-04	1.1E-05
I-134	7.92E-01	2.1E-03	4.3E-05
I-135	1.06E-01	1.1E-03	2.0E-05

Table 11.1-6: Design Basis Radioiodine in Reactor Water and Steam

Isotope	Decay Constant (per hour)	Water Concentration (µCi/gm)	Steam Concentration (µCi/gm)
I-131	3.60E-03	1.8E-02	3.7E-04
I-132	3.02E-01	1.8E-01	3.5E-03
I-133	3.33E-02	1.3E-01	2.7E-03
I-134	7.92E-01	5.3E-01	1.1E-02
I-135	1.06E-01	2.6E-01	5.0E-03

Table 11.1-7: Source Term Design Basis Parameters

Parameter	Value
Total of the design basis release rates of the 13 noble gases (30-minute decay reference, t=30). For use in Offgas System failures and design basis accidents.	87,000 µCi/s
Total of the design basis release rates of the 13 noble gases (30-minute decay reference, t=30). For use in shielding and ventilation design.	20,000 µCi/s
Normal operation noble gas release rate	895 µCi/s
Design basis I-131 radioiodine core release rate	250 µCi/s
Normal operation I-131 radioiodine core release rate	1.0 µCi/s
Normal operation Ar-41 release rate into the Offgas System	40 µCi/s

Table 11.1-8: Normal Operation Non-Volatile Fission Products in Reactor Water and Steam

Isotope*	Decay Constant (per hour)	Water Concentration (µCi/gm)	Steam Concentration (µCi/gm)
Rb-89	2.75E+00	1.5E-03	1.5E-06
Sr-89	5.72E-04	2.5E-06	2.5E-09
Sr-90	2.75E-06	1.2E-07	1.2E-10
Y-90	2.75E-06	1.2E-07	1.2E-10
Sr-91	7.20E-02	1.5E-03	1.5E-06
Sr-92	2.61E-01	3.3E-03	3.3E-06
Y-91	4.94E-04	3.3E-05	3.3E-08
Y-92	1.96E-01	1.0E-03	1.0E-06
Y-93	6.81E-02	1.1E-04	1.1E-07
Zr-95/Nb-95	4.51E-04	6.9E-05	6.9E-08
Mo-99/Tc-99m	1.05E-02	3.2E-04	3.2E-07
Ru-103/Rh-103m	7.36E-04	1.7E-05	1.7E-08
Ru-106/Rh-106	7.73E-05	2.5E-06	2.5E-09
Te-129m	8.60E-04	3.3E-05	3.3E-08
Te-131m	2.31E-02	4.3E-05	4.3E-08
Te-132	9.01E-03	7.9E-06	7.9E-09
Cs-134	3.83E-05	2.2E-05	2.2E-08
Cs-136	2.20E-03	1.8E-05	1.8E-08
Cs-137/Ba-137m	2.62E-06	3.4E-05	3.4E-08
Cs-138	1.25E+00	1.7E-03	1.7E-06
Ba-140/La-140	2.27E-03	3.5E-04	3.5E-07
Ce-141	8.88E-04	1.7E-05	1.7E-08
Ce-144/Pr-144	1.01E-04	2.5E-06	2.5E-09
Np-239	1.23E-02	2.7E-04	2.7E-07

Note:

*Nuclides shown as pairs are assumed to be in secular equilibrium. The parent decay constant and concentration are shown.

Table 11.1-9: Design Basis Non-Volatile Fission Products in Reactor Water and Steam

Isotope*	Decay Constant (per hour)	Water Concentration (µCi/gm)	Steam Concentration (µCi/gm)
Rb-89	2.75E+00	3.8E-01	3.8E-04
Sr-89	5.72E-04	6.1E-04	6.1E-07
Sr-90	2.75E-06	3.1E-05	3.1E-08
Y-90	2.75E-06	3.1E-05	3.1E-08
Sr-91	7.20E-02	3.7E-01	3.7E-04
Sr-92	2.61E-01	8.0E-01	8.0E-04
Y-91	4.94E-04	8.1E-03	8.1E-06
Y-92	1.96E-01	2.5E-01	2.5E-04
Y-93	6.81E-02	2.7E-02	2.7E-05
Zr-95/Nb-95	4.51E-04	1.7E-02	1.7E-05
Mo-99/Tc-99m	1.05E-02	7.9E-02	7.9E-05
Ru-103/Rh-103m	7.36E-04	4.1E-03	4.1E-06
Ru-106/Rh-106	7.73E-05	6.1E-04	6.1E-07
Te-129m	8.60E-04	8.1E-03	8.1E-06
Te-131m	2.31E-02	1.1E-02	1.1E-05
Te-132	9.01E-03	1.9E-03	1.9E-06
Cs-134	3.83E-05	5.5E-03	5.5E-06
Cs-136	2.20E-03	4.3E-03	4.3E-06
Cs-137/Ba-137m	2.62E-06	8.4E-03	8.4E-06
Cs-138	1.25E+00	4.1E-01	4.1E-04
Ba-140/La-140	2.27E-03	8.5E-02	8.5E-05
Ce-141	8.88E-04	4.1E-03	4.1E-06
Ce-144/Pr-144	1.01E-04	6.1E-04	6.1E-07
Np-239	1.23E-02	6.5E-02	6.5E-05

Note:

*Nuclides shown as pairs are assumed to be in secular equilibrium. The parent decay constant and concentration are shown.

Table 11.1-10: Normal Operation N-16 Concentrations in Reactor Water and Steam

Isotope	Half-Life (s)	Water Concentration (µCi/gm)	Steam Concentration* (µCi/gm)
N-16	7.1	60	50

Note:

* Valid at core exit.

Table 11.1-11: Design Basis N-16 Concentrations in Reactor Water and Steam

Isotope	Half-Life (s)	Water Concentration (µCi/gm)	Steam Concentration* (µCi/gm)
N-16	7.1	60	130

Note:

* Valid at core exit.

Table 11.1-12: Activated Corrosion Products in Reactor Water and Steam

Isotope	Decay Constant (per hour)	Water Concentrations* (µCi/gm)	Steam Concentrations* (µCi/gm)
Na-24	4.63E-02	8.2E-04	8.2E-07
P-32	2.03E-03	3.3E-05	3.3E-08
Cr-51	1.04E-03	8.0E-04	8.0E-07
Mn-54	9.25E-05	4.0E-04	4.0E-07
Mn-56	2.69E-01	1.5E-03	1.5E-06
Fe-55	2.89E-05	8.3E-04	8.3E-07
Fe-59	6.49E-04	2.2E-04	2.2E-07
Co-58	4.08E-04	1.9E-04	1.9E-07
Co-60	1.50E-05	3.7E-04	3.7E-07
Ni-63	7.90E-07	8.3E-07	8.3E-10
Cu-64	5.46E-02	3.9E-03	3.9E-06
Zn-65	1.18E-04	1.7E-03	1.7E-06
Ag-110m	1.16E-04	8.3E-07	8.3E-10
W-187	2.92E-02	1.9E-04	1.9E-07

Note:

* Normal operational concentrations are the same as the design basis concentrations.

11.2 Liquid Waste Management Systems

The LWM consists of four subsystems as described below. The Waste Collection and Filtering and the Waste Sampling Subsystems are mostly relevant to radioactive waste management with contributions from the other subsystems as described in detail in this chapter. The LWM has Containment Isolation Valves (CIVs), which are not described in this section. The CIVs, their safety functions, and testing are described in Subsection 6.2.4. Detailed descriptions of the other subsystems of the LWM are described in other PSAR chapters as indicated:

- Waste Collection and Filtering Subsystem – The purpose of the Waste Collection and Filtering Subsystem is to collect and process wastewater primarily from the Equipment and Floor Drain System (EFS). The collection tanks may also receive overboarding flow from the Shutdown Cooling System (SDC) and the CUW.
- Waste Sampling Subsystem – The purpose of the Waste Sampling Subsystem is to collect processed liquid waste for sampling. The sample tanks may also receive overboarding flow from SDC and CUW.
- Condensate Storage and Transfer Subsystem – The purpose of the Condensate Storage and Transfer Subsystem is to supply condensate to the TB, RWB, and the RB for various systems. A detailed description of this subsystem is provided in Subsection 9.2.6.
- Refueling Water Storage and Cleanup Subsystem – The purpose of the Refueling Water Storage and Cleanup Subsystem is to hold the Reactor Cavity Pool volume to support refueling outage activities. A detailed description of this subsystem is provided in Subsection 9.2.10.

11.2.1 Design Bases

Safety Category 1 Functions

None.

Safety Category 2 Functions

None.

Safety Category 3 Functions

- Collect and transport potentially contaminated water received from other plant waste streams
- Provide a means to terminate waste rejection to the environment

Safety Category N Functions

- Reclaim, treat, and transport to storage treated water from waste streams for use by other plant systems
- Provide a means to reject and monitor treated waste streams to maintain concentrations of contaminants released to the environment within permissible limits

The Safety Classifications, Quality Groups, and Seismic Categories of the LWM Structures, Systems, and Components (SSCs) are listed in Table 3A-1.

11.2.2 System Description

The Waste Collection and Filtering Subsystem collects liquid radioactive wastes generated as the result of normal operation. Filtration skids process the waste and then send the processed water to the Waste Sampling Subsystem. If the plant's overall water inventory will not allow for the

water to be recycled, the filtered water can be discharged to the environment. The BWRX-300 is designed to be a zero-release facility, but in the case where a discharge is required, a release path to the environment is provided. The LWM safety classification, seismic classification, and Quality Group are presented in Appendix 3A. The Waste Collection and Filtering Subsystem is illustrated in Figure 11.2-1.

The Waste Sampling Subsystem is the end process point of the LWM where effluent from the filtering skids is sampled. Processed liquid waste that meets water quality requirements, is pumped into the Condensate Storage Tank (CST) for reuse within the plant. If the plants overall water inventory will not allow for the water to be recycled, the processed water can be discharged to the environment, following required sampling and analysis. The Waste Sampling Subsystem is illustrated in Figure 11.2-2.

The system is provided with adequate tank capacity to meet water movement requirements and is equipped with cross-connected pumps for the collection and sample tanks to maintain system availability in the event of any single component failure.

Methods to keep dose ALARA consist of, but are not limited to, concrete barriers surrounding LWM tanks, modular shield walls for the LWM skids, and locating pumps and controls in low dose areas.

11.2.2.1 Equipment, Arrangement, and Location

Simplified General Arrangement drawings for the RWB are provided in Subsection 1.2.3.

Waste Collection and Filtering Subsystem

The Waste Collection and Filtering Subsystem consists of two collection tanks, two transfer pumps, filtration skids, and associated piping and valves. The subsystem is located in the RWB.

Waste Sampling Subsystem

The Waste Sampling Subsystem consists of two sample tanks, two transfer pumps, and associated piping and valves. These components are located in the RWB.

11.2.2.2 Major Components Description

The design of the Waste Collection and Filtering Subsystem and the Waste Sampling Subsystem utilizes and conforms to the guidance provided in Regulatory Guide (RG) 1.143. The guidance does not apply to piping and valves downstream of the isolation valves that transport water from the Waste Sampling Subsystem to the CST and to Normal Heat Sink (NHS) for offsite discharge, as well as piping from the CST to either subsystem. The piping and valves of the portions of the subsystems in the scope of RG 1.143 are designed in accordance with American Society of Mechanical Engineers (ASME) B31.3, “Process Piping,” (Reference 11.2-1). Piping and valves outside the RG 1.143 boundary are designed in accordance with ASME B31.1, “Code for Pressure Piping, B31.1, Power Piping,” (Reference 11.2-2).

The applicable standards from RG 1.143, Table 1, are used in the design, construction, and testing of the components (see Table 11.2-1). The applicable design criteria from RG 1.143 Table 2, Table 3, and Table 4 are used in the design analysis of the LWM components. In addition to the design criteria in the RG 1.143 tables, the following design provisions and exceptions are incorporated:

- The RWB is categorized as Seismic Category RW-IIa, as described in Appendix 3A
- Components designated as inside the RG 1.143 boundary, with the exception of the containment penetration piping and associated valves (which are Seismic Category I), are

assigned Seismic Category RW-IIa. LWM components outside the RG 1.143 boundary are non-seismic.

- Requirements for the CST and Refueling Water Storage Tank (RWST) are discussed in Subsection 9.2.6, and Subsection 9.2.10, respectively
- RG 1.143 specifies API-650 for the Design and Construction of atmospheric tanks. API-650 is applicable to tanks with uniformly supported bottoms. The LWM collection and sample tanks have sloped bottoms not uniformly supported but potentially supported by support legs. The code used for inspection and testing of the tanks also deviates from that specified in RG 1.143 for atmospheric tanks. As an alternative, the requirements of ASME Section VIII may be used.
- LWM tanks have level sensors to monitor the tank liquid levels. Designated high liquid level conditions will actuate both local alarms and alarms in the Main Control Room (MCR), as described in Subsection 11.2.6.2.
- Tanks, pumps, and piping drain to EFS. LWM tank overflow piping is also routed to the EFS
- Mobile and temporary equipment are designed in accordance with the applicable mobile and temporary radwaste treatment systems guidance provided in RG 1.143, including the codes and standards listed in Table 1 of the Regulatory Guide. Seismic Category RW-IIa is applied to mobile process system components.

The subsystems, components and their functions are described in the following subsections.

11.2.2.2.1 Collection and Sample Tanks

Tanks are sized to accommodate the expected volumes of waste generated in the upstream systems that feed waste into the LWM for processing. Tanks are constructed of stainless steel. The tanks are provided with mixing eductors. These provide for mixing of the tank contents to maintain a representative, homogenous fluid sample, and to keep solids in suspension for removal during transfer of the tank contents. The capability exists to sample all Radwaste tanks. LWM tanks are vented into the HVS of the RWB. The vents are also large enough to accommodate the airflow associated with filling or pumping down the tanks at a maximum flowrate.

11.2.2.2.2 Pumps

The transfer pumps are constructed of materials suitable for their intended service. Each tank has a dedicated pump. Pumps are cross tied for flexibility of operation.

11.2.2.2.3 Mobile Skids

Provisions are made for mobile waste processing equipment skids to process radioactive liquid waste to condensate water quality requirements for plant reuse. Interconnections with permanent installed waste collection and sample tanks are provided to support remote operation of mobile waste processing equipment. Interconnections with EFS and SWM are also provided to support mobile process equipment draining and backwash capability.

11.2.2.2.4 Piping and Valves

LWM piping material is austenitic stainless-steel to minimize corrosion. RG 1.143 allows for the use of flexible hoses and hose connections in conjunction with Mobile Radwaste Processing Systems.

11.2.2.3 Interfaces with Other Equipment or Systems

An exterior liquid radioactive waste discharge pipeline is routed from the RWB to the NHS where it mixes with blowdown flow prior to discharge. The interface point for the discharge of blowdown and LWM System to the environment will be discussed in CPA Part 2. The LWM interfaces with other systems including:

- Process Radiation and Environmental Monitoring System (PREMS)
- SDC
- CUW
- FPC
- SWM
- HVS
- EFS
- Plant Pneumatics System (PPS)
- Demineralized Water Storage and Distribution Subsystem

11.2.2.4 System and Equipment Operation

Waste inputs are processed on a batch basis.

Potentially radioactive liquid wastes, including detergent and chemical wastes, enter the Waste Collection and Filtering Subsystem collection tanks via the EFS system. Clean, oily, and contaminated wastes are collected separately via the EFS system, which is described in Subsection 9.3.3. If the collection tank contents require pretreatment because they exceed the mobile skid influent requirements, the contents can be routed to SWM for pretreatment through the spent resin tank prior to processing through the mobile skids. Following pretreatment through SWM, the tank contents are routed to the collection tank.

The process effluents are collected in one of the two sample tanks for chemical and radioactivity analysis. The analysis is done in the sample laboratory located in the RWB. If acceptable, the tank contents are sent to the CST for plant reuse, or discharged to the environment, if the plants overall water inventory will not permit recycling. A recycle line from the sample tanks allows the sampled effluents that do not meet water quality requirements to be pumped back to one of the collection tanks for additional processing.

Interconnections between permanent installed waste collection and sample tanks support remote operation of mobile waste processing equipment.

The collection and sample tanks may also receive over boarding flow from the SDC and CUW.

The collection and sample tanks are provided with a flush line from the CST and are equipped with a tank wash nozzle for wash down of interior tank surfaces.

11.2.3 Safety Evaluation

LWM complies with the following General Design Criteria (GDC), which have plant-wide applicability, as discussed in Section 3.1:

GDC 2 - Design bases for protection against natural phenomena.

GDC 3 - Fire protection.

GDC 5 - Sharing of structures, systems, and components.

The following GDC have aspects specific to LWM, and compliance is addressed as follows:

GDC 60, as it relates to the design of the LWM to control releases of radioactive liquid effluents generated during normal reactor operations, including Anticipated Operational Occurrences (AOOs), as described in Subsection 11.2.4.

GDC 61, as it relates to radioactive waste systems being designed to provide for adequate safety under normal and postulated accident conditions and designed with suitable shielding for radiation protection as described in Section 12.3.

Additional information on GDC compliance is found in Section 3.1.

The Waste Collection and Filtering and the Waste Sampling Subsystems are designed to comply with the following regulations:

10 Code of Federal Regulations (CFR) 20.1101(b), as it relates to the use of engineering controls in maintaining doses to members of the public “As Low As Reasonably Achievable” is described in Subsection 11.2.2 and Section 11.5. Administrative controls are addressed in Section 12.5.

10 CFR 20.1301, 10 CFR 20.1302, and 10 CFR Part 20, Appendix B, Table 2 as they relate to radioactivity in liquid effluents released to unrestricted areas and doses to offsite receptors located in unrestricted areas is described in Subsection 11.2.4.

10 CFR 20.1406, as it relates to the design and operational procedures to minimize contamination, facilitate eventual decommissioning, and minimize the generation of radioactive waste as described in Chapter 12.

10 CFR Part 50, Appendix I, Section II.A, as it relates to the numerical guidelines for design objectives and limiting conditions for operation to meet the “As Low As Reasonably Achievable” criterion is described in Subsection 11.2.4.

40 CFR Part 190, “Environmental Radiation Protection Standards for Nuclear Power Operations,” as implemented under 10 CFR 20.1301(e), as it relates to limits on annual doses from all sources of radioactivity released in liquid effluent discharges and external radiation from site buildings and facilities is described in Subsection 11.2.4.

11.2.3.1 Radioactive Effluent Releases and Dose Calculations Due to Liquid Tank Leak or Failure

BTP 11-6 provides guidance and acceptance criteria for the consequences of the release of radioactive liquids following the postulated failure of a tank and its components, located outside of containment. Radioactive releases associated with the postulated failure of a tank and components should not do the following:

1. Result in radioactive material concentrations leading to a dose in excess of 100 mrem (1 mSv) at the point of entry into the nearest existing or a known future water supply when (1) used as a source of water for direct human consumption, or (2) used indirectly through livestock watering or irrigation of grazing pastures, consumption of animal products (meat and milk products), fish and invertebrate consumption, crop irrigation and consumption of such crops, or used as an ingredient in food products or food processing.
2. Result in a total dose in excess of 100 mrem (1 mSv) in instances where a scenario assumes the dual use of water, direct and indirect usage. In such an instance, the dose from each scenario must be derived separately and the resulting doses must be added, with the summation of doses compared to the SRP dose acceptance criteria.

Table 12.2-1 through Table 12.2-28 provide source inventories for tanks, components and buildings containing radioactive materials. The largest tank volume (RWST) along with the

maximum liquid source term concentrations (RPV water) from Tables 11.1-6, 11.1-9, 11.1-11, and 11.1-12 are used to create the most conservative release scenario.

Evaluation Results

CPA Part 2 will include a summary of the Plant Parameter Envelope (PPE) Evaluation and results relating to the assessment of the radioactive release from an accidental liquid tank leak or failure.

11.2.4 Radioactive Effluent Releases

Methodology

The analysis is performed with the GALE-BWR methodology (see Chapter 3, Appendix 3L) which implements the assumptions outlined in USNRC RG 1.112, Revision 1. The GALE-BWR methodology is described in USNRC NUREG-0016, Revision 2. GALE-BWR calculates expected releases based on:

1. Standardized coolant activities derived from ANS Standards 18.1 Working Group Recommendations.
2. Release and transport mechanisms that result in the appearance of radioactive material in liquid and gaseous waste streams.
3. Plant-Specific design features used to reduce the quantities of radioactive materials ultimately released to the environs.
4. Information received on the operation of nuclear power plants.

The calculations are based on data generated from operating reactors, field tests, laboratory tests, and plant-specific design considerations incorporated to reduce the quantity of radioactive materials that may be released to the environment. The calculation parameters are shown in Table 11.2-2.

The LWM is designed to eliminate the need for LWM routine release to the environment. Nearly all wastewater is assumed to be processed and recycled for use by plant systems. Consistent with this assumption, all tritium calculated to be available for release is included with the BWRX-300 annual average gaseous effluent release estimate presented in Subsection 11.3.4.

Release Evaluation Results

The BWRX-300 annual average liquid effluent release activities from various waste streams are shown in Table 11.2-3. Other nuclides with annual average liquid effluent release activities less than 1.0E-05 Ci/year are not listed in Table 11.2-3.

Processed liquid waste that meets the water quality requirements is transferred to the CST for reuse within the plant, or it can be discharged to the environment if the overall water inventory of the plant will not permit recycling.

A summary of the PPE evaluation of annual average liquid effluent release results will be included in CPA Part 2.

Dose Evaluation Results

A summary of dose evaluation results describing the radiological effects of liquid radioactive waste effluents from normal plant operation and annual collective doses to the population for liquid and gaseous releases will be provided in CPA Part 2.

The projected total annual doses per reactor from both liquid and gaseous radioactive effluents are within the design objectives of 10 CFR 50, Appendix I.

The projected total site doses due to liquid and gaseous radioactive effluents are within the environmental standards of 40 CFR 190.

Because 40 CFR 190 is more restrictive than 10 CFR 20.1301, compliance with the limits of 40 CFR 190 demonstrates compliance with the 0.1 rem (100 mrem) limit of 10 CFR 20.1301.

11.2.5 Testing and Inspection Requirements

The Initial Test Program is described in Chapter 14.

Provisions are made for periodic inspection of major components to ensure capability and integrity of the systems. The system design provides adequate equipment removal paths and personnel access for replacement per General Arrangement Drawings. Adequate space is provided in the RWB for maintenance of the pumps, valves, and instruments.

11.2.6 Instrumentation and Controls

The LWM provides the instrumentation to control and monitor (indicate and alarm) system operation.

11.2.6.1 Instrumentation

Major system parameters, (i.e., tank level, process flow rates, filter differential pressure, and filtration skid processing rate,) etc., are indicated and alarmed as required to provide operational information and performance assessment. Additional information about LWM instrumentation is included in Subsection 7.3.4.

11.2.6.2 Control

The Waste Collection and Filtering subsystem and the Waste Sampling subsystem are operated remotely from the MCR. Local controls are also provided for the Collection/Sample Pumps and the Filtration Skids.

Level signals can be used to start and stop the sump pumps based on high and low settings. A high LWM tank level initiates a high-level alarm in the MCR, and locally, for the affected tank. Closure of the tank inlet valves can be controlled remotely from the MCR to prevent overflow. If the operator doesn't take manual action to prevent tank overflow, the inlet valves to the receiving collection tank will automatically close on a high-high level to prevent overflow of the tank. Any overflow that would occur is directed to the EFS.

The environmental discharge controls consist of a flow control valve and two isolation valves in series that automatically close upon receiving a trip signal from the PRM (described in Section 11.5) and initiates a high radiation alarm in the MCR to control the release of radioactive liquid effluents within the allowed release limit.

11.2.7 References

11.2-1 ASME Code B31.3-2022, "Process Piping," American Society of Mechanical Engineers, 2022.

11.2-2 ASME Code for Pressure Piping, B31.1, Power Piping, American Society of Mechanical Engineers, 2020.

Table 11.2-1: Equipment Codes (from Table 1, RG 1.143, Rev. 2)

Component	Design and Construction	Materials[1]	Welding	Inspection and Testing
Piping and Valves	ANSI/ASME B31.3 [2,3]	ASME-Section II[4]	ASME, Section IX	ANSI/ASME B31.3
Atmospheric Tanks	API-650	ASME Section II	ASME, Section IX	API-650[7]
Tanks (0-15 psig)	API-620	ASME Section II	ASME, Section IX	API-620[7]
Pressure Vessels and Tanks (>15 psig)	ASME BPVC Section VIII, Division 1, or Division 2	ASME Section II	ASME, Section IX	ASME BPVC Section VIII, Division 1, or Division 2
Pumps	API-610; API-674; API-675; ASME BPVC Section VIII, Division 1, or Division 2	ASTM A571-84 (1997) or ASME Section II	ASME, Section IX	ASME BPVC Code Section III, Class 3[5]
Heat Exchangers	TEMA STD, 8th Edition; ASME BPVC Section VIII Division 1 or Division 2	ASTM B359-98 or ASME Section II	ASME, Section IX	ASME Section VIII, Division 1, or Division 2
Flexible Hoses and Hose Connections for Mobile Radwaste Processing Systems[6]	ANSI/ANS-40.37	ANSI/ANS-40.37	ANSI/ANS-40.37	ANSI/ANS-40.37

Notes:

1. Manufacturer's material certificates of compliance with material specifications may be provided in lieu of certified material test reports as discussed in Regulatory Position 1.1.2 of RG 1.143.
2. Class RW-IIa and RW-IIb Piping Systems are to be designed as category "M" systems.
3. Classes RW-IIa, RW-IIb and RW-IIc are discussed in Regulatory Position 5 of RG 1.143.
4. ASME Boiler and Pressure Vessel Code (BPVC) Section II required for Pressure Retaining Components.
5. ASME Code stamp, material traceability, and the quality assurance criteria of ASME BPVC, Section III, Division 1, Article NCA are not required. Therefore, these components are not classified as ASME Code Section III, Class 3.
6. Flexible Hoses should only be used in conjunction with Mobile Radwaste Processing Systems.
7. Alternative requirements such as those in ASME Section VIII may be used.

Table 11.2-2: Average Annual Liquid Release Calculation Parameters

Parameter	Input
Name of reactor	BWRX-300
Condensate demineralizer regeneration time	0 days
Copper tubing for condenser	0 (no)
Fraction feed water through the CFD	0.99
High purity waste input	0 gal/day
High purity: Fraction of reactor coolant activity	0.0
High purity: Decontamination Factor (DF) for iodine	1
High purity: DFs for Cs and Rb	1
High purity: DF for others	1
High purity: Collection time	0.0 days
High purity: Process and discharge time	0.0 days
High purity: Fraction discharged	0.0
Low purity waste input	4,320 gal/day
Low purity: Fraction of PCA	0.23
Low purity: DF for iodine	1000
Low purity: DFs for Cs and Rb	200
Low purity: DF for others	1000
Low purity: Collection time	4.9 days
Low purity: Decay and Process time	0.21 days
Low purity: Fraction discharged	1.0
Chemical waste input	0 gal/day
Chemical: Fraction of reactor coolant activity	0.0
Chemical: DF for iodine	1
Chemical: DF for Cs and Rb	1
Chemical: DF for others	1
Chemical: Collection time	0.0 days
Chemical: Process and discharge time	0.0 days
Chemical: Fraction discharged	0.0
Regenerant Solutions waste input	0 gal/day
Regenerant Solutions: DF for iodine	1
Regenerant Solutions: DF for Cs and Rb	1
Regenerant Solutions: DF for others	1
Regenerant Solutions: Collection time	0.0 days
Regenerant Solutions: Process and discharge time	0.0 days
Regenerant Solutions: Fraction discharged	0.0
Detergent waste stream partitioning factor	0.05

Table 11.2-3: Total Annual Average Liquid Effluent Release

(Sheet 1 of 2)

Nuclide	Annual Release (Ci/yr)	Annual Release (Bq/yr)
Class 2		
I-131	1.10E-03	4.07E+07
I-132	1.50E-04	5.55E+06
I-133	1.90E-03	7.03E+07
I-135	8.60E-04	3.18E+07
Class 3		
Cs-134	2.30E-03	8.51E+07
Cs-136	1.30E-03	4.81E+07
Cs-137	3.50E-03	1.30E+08
Class 6		
Na-24	1.90E-03	7.03E+07
P-32	4.70E-04	1.74E+07
Cr-51	1.20E-02	4.44E+08
Mn-54	6.50E-03	2.41E+08
Mn-56	2.00E-04	7.40E+06
Fe-55	1.40E-02	5.18E+08
Fe-59	3.50E-03	1.30E+08
Co-58	3.40E-03	1.26E+08
Co-60	6.60E-03	2.44E+08
Ni-63	1.00E-04	3.70E+06
Ni-65	4.00E-05	1.48E+06
Cu-64	7.40E-03	2.74E+08
Zn-65	2.70E-03	9.99E+07
Zn-69m	4.20E-03	1.55E+08
Br-83	6.70E-04	2.48E+07
Sr-89	5.00E-05	1.85E+06
Sr-91	2.00E-03	7.40E+07
Sr-92	4.80E-04	1.78E+07
Y-91	6.70E-04	2.48E+07
Y-92	1.50E-03	5.55E+07
Y-93	1.60E-04	5.92E+06
Zr-95	1.10E-03	4.07E+07
Zr-97	2.00E-05	7.40E+05
Nb-95	1.20E-03	4.44E+07
Nb-98	1.00E-05	3.70E+05
Mo-99	2.80E-03	1.04E+08
Tc-99m	2.80E-03	1.04E+08

Table 11.2-3: Total Annual Average Liquid Effluent Release

(Sheet 2 of 2)

Nuclide	Annual Release (Ci/yr)	Annual Release (Bq/yr)
Ru-103	2.70E-04	9.99E+06
Ru-105	7.90E-04	2.92E+07
Ru-106	4.80E-04	1.78E+07
Ag-110m	7.00E-05	2.59E+06
Te-129m	5.00E-04	1.85E+07
Te-131m	2.10E-04	7.77E+06
Te-132	7.00E-05	2.59E+06
Cs-137	3.50E-03	1.30E+08
Ba-139	2.20E-04	8.14E+06
Ba-140	4.90E-03	1.81E+08
La-142	1.80E-04	6.66E+06
Ce-141	3.30E-04	1.22E+07
Ce-143	1.60E-04	5.92E+06
Ce-144	2.30E-04	8.51E+06
Nd-147	4.00E-05	1.48E+06
Pr-143	5.90E-04	2.18E+07
W-187	7.40E-04	2.74E+07
Np-239	2.10E-03	7.77E+07

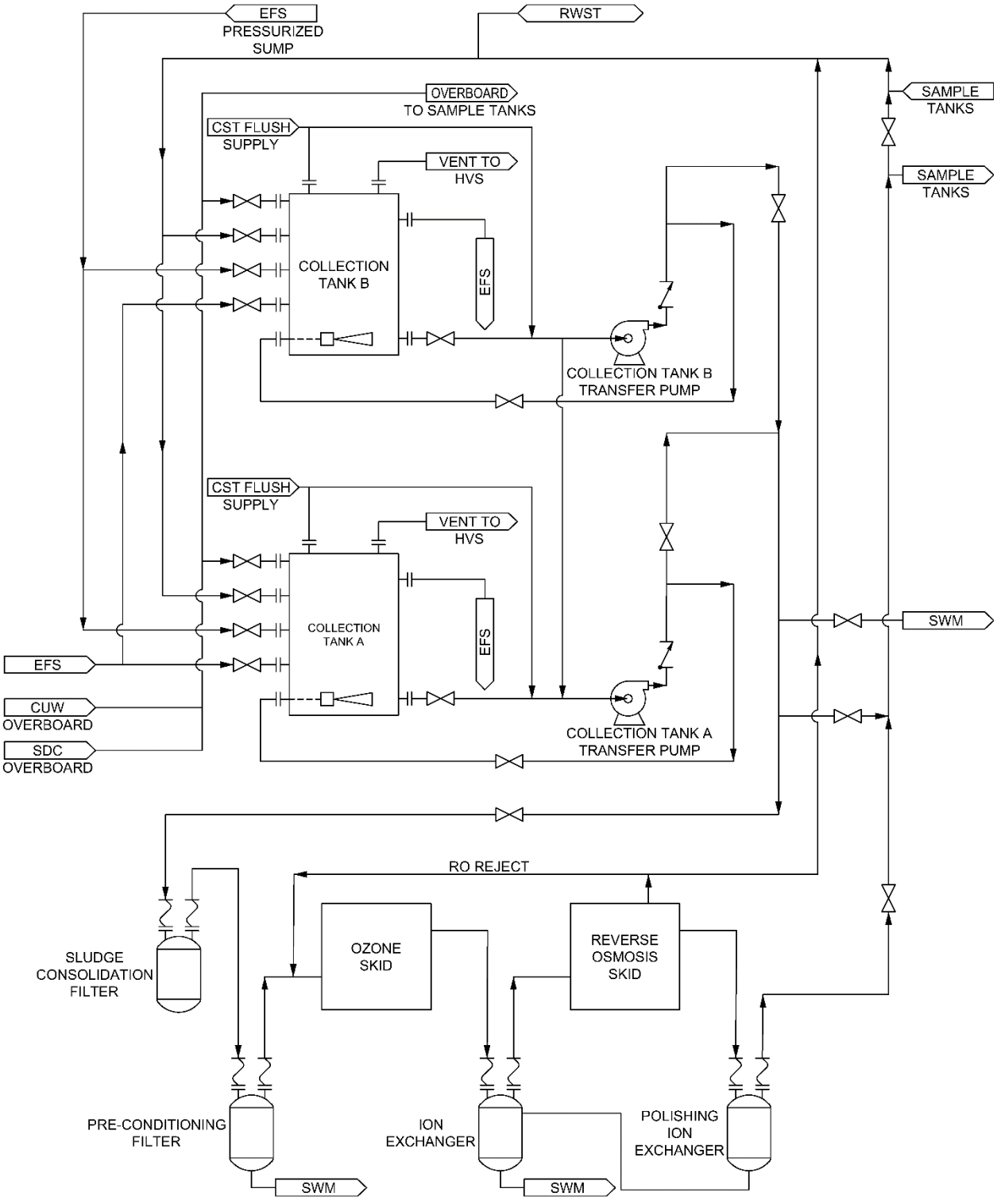


Figure 11.2-1: Waste Collection and Filtering Subsystem Simplified Diagram

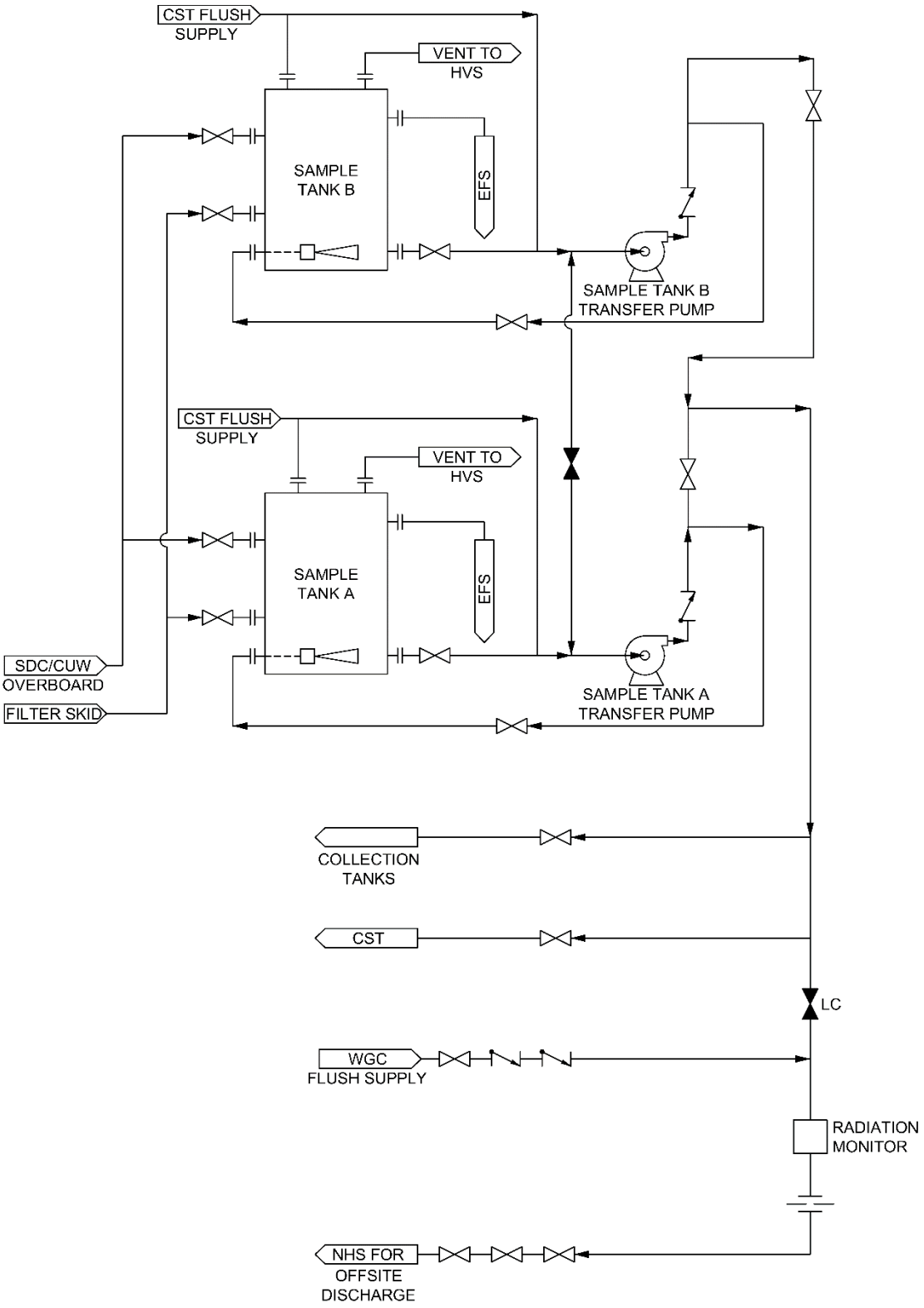


Figure 11.2-2: Waste Sampling Subsystem Simplified Diagram

11.3 Gaseous Waste Management Systems

Gaseous Waste Management is performed primarily by the OGS with contributions from the HVS. The HVS is described in detail in Section 9.4.

The purpose of the OGS is to process and control the release of gaseous radioactive effluents to the site environs to maintain the exposure of persons in unrestricted areas to radioactive gaseous effluents ALARA (within 10 CFR 50 Appendix I dose objectives) and below applicable regulatory limits. Offgas effluent consists of the entire primary flow stream through the OGS coming from the Steam Jet Air Ejectors (SJAEs), including steam, air, oxygen, hydrogen, noble gases, and other trace radioactive isotopes.

11.3.1 Design Bases

The OGS performs the following Safety Category Functions:

Safety Category 1 Functions

None.

Safety Category 2 Functions

None.

Safety Category 3 Functions

- Provides a flow path to discharge offgas from the Main Condenser SJAE to the HVS
- Recombines the hydrogen and oxygen in the offgas stream into water
- Removes radioisotopes in the offgas effluent stream to maintain dose to the public and plant personnel within permissible limits

11.3.1.1 Safety Category N Functions

- The OGS condenses steam in the offgas stream and returns it to the main condenser

The Safety Classifications, Quality Groups, and Seismic Categories of the OGS SSCs are listed in Table 3A-1.

11.3.2 System Description

The OGS processes non-condensable gases from the Main Condenser and Auxiliaries (MCAs) system that are produced through normal nuclear power operations. The main process influent to the system is a mix of steam, air, hydrogen, and radioactive noble gases from the MCA SJAE. The objective of OGS is to process this influent prior to release to the environment from the HVS. This processing is meant to reduce the release of gaseous radionuclides to maintain the exposure of persons in unrestricted areas to radioactive effluents ALARA and below applicable regulatory limits.

The OGS is designed to conform to the following regulatory guidance:

RG 1.143, Design guidance for radioactive waste management systems, structures, and components installed in light-water-cooled nuclear power plants, the design guidance for radioactive waste management systems is used for the OGS.

BTP 11-5, Postulated radioactive releases due to a waste gas system leak or failure, is used to determine the acceptable methods to evaluate the consequences associated with the postulated release of radioactive materials following the failure of a waste gas system component.

The processing includes two main functions:

- Recombination of hydrogen and oxygen into water to maintain plant water inventory and reduce hydrogen detonation risk
- Controlled adsorptive holdup of the radioactive isotopes of krypton, xenon, and argon to achieve decay, reducing effluent radioactivity releases from the plant

The OGS is comprised of the following components shown in Figure 11.3-1. Components that meet the design criteria of Table 11.2-1 and RG 1.143, Table 2, Table 3, and Table 4 are marked RW-IIa:

- Offgas Recombiner including:
  - Preheater section
  - Catalytic Recombiner section
  - Condenser section
- Cooler Condenser and Moisture Separator
- Refrigeration Dryers
- Gas Analyzer (hydrogen, oxygen) including Humidity Monitor
- Offgas Reheater (RW-IIa)
- Charcoal Adsorber Vault with ventilation equipment (RW-IIa)
- Charcoal Adsorber Tanks (four) (RW-IIa)
- Offgas HEPA Filter (RW-IIa)
- OGS Piping and valves located in the RWB (RW-IIa)

The OGS is designed with several features to increase system availability. The active components are duplicated to allow for continued operation in the case of a single SSC failure during power operations. There are redundant 100% cooling capacity Refrigeration Dryers with remote isolation valves such that adequate offgas effluent cooling for holdup continues if one of the dryers fails. Similarly, the Charcoal Adsorber Vault ventilation trains are duplicated to maintain the vault temperature stable in the event of failure of a single train.

The first Charcoal Adsorber Tank, known as the Guard Bed, has a bypass capability, which allows for continued offgas functionality and online maintenance of the first tank in the event of water intrusion or ignition of charcoal. Offgas flow can be directed through the other three charcoal tanks or bypass the Charcoal Adsorber Vault entirely.

The SSCs with automatic controlling instrumentation have three trains of instrumentation. This allows for the failure of any one controlling instrument without a failure of the function of the associated SSCs.

Valve failure positions have been selected to ensure that single point failures of power or air to any valve will not disrupt offgas functions.

11.3.2.1 Equipment, Arrangement, and Location

Equipment for the OGS is located within the TB and the RWB. The primary effluent line of the OGS begins at the valve interface with the MCAs system at the exit of the SJAEs within the TB. Simplified General Arrangement drawings for the TB and RWB are provided in Subsection 1.2.3.

The following components of the OGS are housed in the TB:

- Offgas Recombiner
- Cooler Condenser
- Moisture Separator
- Refrigeration Dryers
- Gas Analyzers

After exiting the Refrigeration Dryers, OGS process piping carries the offgas effluent from the TB to the RWB.

The following components of the OGS are housed in the RWB:

- Offgas Reheater
- Charcoal Adsorber Vault, including four Charcoal Adsorber Tanks
- Offgas Post Filter

The OGS is designed such that the Charcoal Adsorber Vault may be bypassed during startup or off-normal operation, as described in Subsection 11.3.2.2.6. This bypass piping and associated valves are in the RWB.

The OGS ends with the transfer of the offgas effluent to the HVS prior to final release.

11.3.2.2 Major Components Description

The design of the OGS conforms to the guidance provided in RG 1.143. The applicable standards from RG 1.143 Table 1 are used in the design, construction, and testing of the OGS components (see Table 11.2-1). The applicable design criteria from RG 1.143 Table 2, Table 3, and Table 4 are used in the design analysis of the OGS components.

The subsystems, components and their functions are described in the following subsections.

11.3.2.2.1 Offgas Recombiner

The Offgas Recombiner is a three-section vessel consisting of a Preheater section, Catalytic Recombiner section, and Condenser section. The purpose of the Offgas Recombiner is to perform the functions of oxygen and hydrogen recombination and to condense the resulting steam for return to the Main Condenser.

The Preheater section is a tube-and-shell heat exchanger. The purpose of the Preheater section is to raise the temperature of the offgas so that the water content reaches a superheated state, ensuring that it is fully boiled and gaseous prior to entering the Recombiner section. Effluent offgas, consisting of steam, air, hydrogen, oxygen, and other waste gases, from the SJAEs enters the Recombiner through the Preheater section and flows through the shell side. Steam from the Main Steam Line (MSL) flows through the tube side and is used to heat the offgas. The Main Steam (MS) is condensed from a saturated vapor state to a saturated liquid state. The condensed MS water is drained and returned to the main condenser, while the superheated offgas moves on to the Catalytic Recombiner section. The Preheater section contains a pressure relief line to the main condenser, a triplicated temperature elements to control steam inlet, a condensed steam drain line to the main condenser with a steam trap, and two manual maintenance drains.

The Catalytic Recombiner section uses a precious metal catalyst to cause the recombination of radiolytic oxygen and hydrogen gases into water. During this process, the temperature of the offgas rises, scaling with the reactor power level. Liquid water reduces the catalytic reaction

surface area, so superheated steam is maintained in the Preheater section. The Catalytic Recombiner section includes a catalyst bed with three temperature elements to monitor that effluent is superheated, a maintenance drain to empty the vessel when the OGS is not in operation, and a heating element fastened to the exterior to drive off moisture prior to the Offgas Recombiner being placed into service. Offgas effluent from the Catalytic Recombiner section flows into the Offgas Condenser section.

The Offgas Condenser section is a tube-and-shell heat exchanger. This component provides the first stage of condensation of the dilution steam in the effluent mix and the additional water vapor resulting from the recombination of the radiolytic gases. The Offgas effluent enters the shell side of the heat exchanger while the condensate water from the CFS enters the tube side of the heat exchanger. Dual mechanical float traps on the shell side outlet are used to ensure only the condensed liquid returns to the Main Condenser. The remaining gaseous offgas effluent flows to the Cooler Condenser. The Offgas Condenser section includes a pressure relief line to the main condenser. The outlet lines contain a flow element to monitor offgas condensate to the main condenser and a temperature element to monitor the exiting gaseous offgas effluent.

11.3.2.2.2 Cooler Condenser

The Cooler Condenser is a tube-and-shell heat exchanger. This component provides the second stage of offgas condensation. The offgas effluent enters the shell side of the heat exchanger while the chilled water from the Chilled Water Equipment (CWE) System enters the tube side of the heat exchanger. The chilled water is used to cool the offgas effluent and is controlled by CWE system valves. Chilled water passes through the Cooler Condenser and returns to the CWE System. As most moisture is condensed in the Recombiner Condenser section, the condensed liquid water from the Cooler Condenser is drained to local floor drain. A maintenance drain is included on the tube side. The remaining gaseous offgas effluent flows to the Moisture Separator.

11.3.2.2.3 Moisture Separator

The Moisture Separator is downstream of the Cooler Condenser. This is an in-line tank with a wire mesh or chevron screen-type media which serves to divert any moisture droplets remaining in the flow stream to a drain, pulling moisture from the effluent stream. Gaseous flow moves on to the Refrigeration Dryers, and removed liquid is drained to the local floor drain. A temperature element is located downstream of the moisture separator to monitor Cooler Condenser performance.

11.3.2.2.4 Refrigeration Dryers

The Refrigeration Dryers provide the final stage of cooling and condensation of the offgas effluent. Two 100% capacity dryers are provided in parallel, though only one is active at any given time. Dryer trains can be swapped manually or with an automatic control scheme during an equipment failure. This allows for operation of the OGS to continue if one dryer needs maintenance. The effluent is reduced in temperature, resulting in the condensation of water. Condensed water is drained to local floor drains, and gaseous flow flows to the Offgas Reheater or Charcoal Adsorber Vault bypass line depending on system mode.

11.3.2.2.5 Offgas Reheater

Downstream of the Refrigeration Dryers is the Offgas Reheater. In the case of a low temperature ambient environment, the process gas temperature may lower beyond its dew point causing condensation to occur. This negatively affects the charcoal adsorptive holdup of the noble radionuclides. When necessary, the Offgas Reheater heats the offgas effluent to a temperature with sufficient margin to prevent the process gas temperature from lowering and forming

condensation. This component is composed of a series of electrical resistance heaters mounted externally around the effluent pipe just upstream from the Charcoal Adsorber Vault.

11.3.2.2.6 Charcoal Adsorber Vault and Tanks

The OGS Charcoal Adsorber Vault is a shielded and isolated room in which radioactive isotopes are contained and held up to allow for decay prior to release. The Charcoal Adsorber Vault ventilation system maintains the ambient temperature of the vault above the dew point of the effluent and below a point which the dynamic adsorption coefficient of the charcoal is unacceptable. Vault temperature is continuously monitored, and the Charcoal Adsorber Vault ventilation system has a redundant train of heating and cooling equipment. The minimum temperature is set to prevent condensation from a temperature drop past the dew point, with margin. The maximum temperature is set from the degraded adsorption performance of the charcoal at higher temperatures.

The Charcoal Adsorber Tanks are made up of four vertical tanks of granulated activated charcoal designed for adsorptive holdup of the targeted krypton, xenon, and argon isotopes to reduce released effluent activity. These tanks (and their contained charcoal volume) provide the holdup for targeted noble gas isotopes. Multiple temperature elements monitor the temperatures within each tank. A Carbon Monoxide (CO) monitor is provided to detect combustion of the charcoal, and nitrogen may be injected into the vault piping to extinguish a fire if necessary. Radiation monitoring of the Charcoal Adsorber Vault ventilation exhaust is described in Section 11.5.

The first Charcoal Adsorber Tank, known as the Guard Bed, is configured to be bypassed if a problem manifests in it (e.g., high moisture or potential combustion issue) such that the system could operate on the latter three beds for short periods of time. As this is the first bed to process the effluent, it is most susceptible to any issues caused by a malfunction of upstream equipment. Bypassing the Guard Bed is acceptable for short periods of time until any issue with that bed or associated equipment can be remedied.

The offgas effluent flows through the Guard Bed, then through the other three tanks, before moving forward to the final Offgas Post Filter for release. The entire Charcoal Adsorber Vault may also be bypassed during offgas startup or off-normal operation.

11.3.2.2.7 Offgas Post Filter

Downstream from the charcoal beds is an Offgas Post Filter. This is a small filter contained in a tank which serves to pick up any fine pieces of charcoal and any other particulates which may pass downstream of the charcoal beds, prior to offgas release. Differential pressure across the filter is monitored to detect whether a buildup of particulate within the filter is occurring.

11.3.2.2.8 Gas Analysis Equipment

Hydrogen concentration, oxygen concentration, and moisture content is monitored at the outlet of the Refrigeration Dryers. Hydrogen and Oxygen monitors are triplicated to allow to allow continued detection in the event of any single failure. The annunciators of high hydrogen concentration are provided locally, signaling a potential detonation risk, and in the MCR for corrective action to be taken. High hydrogen can trigger a closure of the remote isolation valves connecting to HVS discharge and open the recirculation path back to the MCA. A Low oxygen concentration alarm is provided at the MCR and serves as an input to the Hydrogen Water Chemistry system to adjust the injection of hydrogen or air (containing oxygen) as needed.

CO monitoring is provided at the outlet of the Guard Bed and last charcoal tank to detect for fire. Radioactivity level in the OGS is monitored by the PRM as described in Section 11.5. In addition, there are high radioactivity level alarms in the MCR.

11.3.2.2.9 Piping and Valves

Piping and valves in the OGS are designed to minimize the risk of explosive overpressures by including the following design attributes:

Process piping in the system is designed to withstand detonation pressures per ANSI/ANS-55.4, "Gaseous Radioactive Waste Processing Systems for Light Water Reactors" (Reference 11.3-1).

Valve seats for valves exposed to process offgas are metal and spark-free. Gate valves are wedge-type. This reduces the chance of a valve spark that would cause an ignition of any hydrogen in the system.

11.3.2.3 System and Equipment Operation

The primary functions of the OGS are most applicable during power operations. Production of radiolytic H2 and O2, and of radioactive noble gas isotopes, are minimal except during power operations and as such, the OGS is not required to be in operation except during power operations.

11.3.2.3.1 Initial Configuration

The OGS is secured (inactive) except during power operations. There is no flow through the system and no components are active. At very low or no power, the OGS is not required to process radiolytic gases in the effluent stream, and so the OGS is bypassed. The MCA Mechanical Vacuum Pump is active and drawing main condenser vacuum, and valves controlling effluent flow to the OGS are closed.

11.3.2.3.2 Preparation

The Preparation Mode of the OGS occurs when the plant is in startup mode (Mode 2). During this time, there is no flow through the OGS, the system is isolated, and most equipment is inactive. However, a few pieces of equipment are active, in order to prepare the OGS for startup operations. The equipment active in this mode includes:

- Catalytic Recombiner section heater: activated to ensure a sufficiently high temperature of the Recombiner section prior to transition to startup operation
- Charcoal Adsorber Vault ventilation: activated to ensure desired atmospheric conditions in the Charcoal Adsorber Vault are achieved prior to power operations

11.3.2.3.3 System Startup

The Startup Operation Mode of the OGS occurs between the time that the SJAEs are placed into service (at around 5% reactor power) until reactor power has reached 10% power.

During this Mode, the effluent stream does not flow through the Charcoal Adsorber Vault, but rather instead flows through a bypass valve around the Charcoal Adsorber Vault, then releases to the ventilation system. This is to prevent damage to the Charcoal Tanks due to the greatly increased air inleakage rate during startup.

Note that during the startup mode there is little hydrogen and oxygen available for recombination as radiolysis is directly dependent on (and directly proportional to) reactor power to drive the process. As such, there is little temperature rise seen in the Catalytic Recombiner section during this mode (due to the minimal heat added by the exothermal process of recombination). Likewise, leakage rates of radioactive isotopes (krypton, xenon, argon) into the offgas effluent are minimal at this stage.

11.3.2.3.4 Normal Operation

The OGS normal operation mode begins at a point at or just beyond 10% reactor power, when main condenser vacuum is stable, much of the inleakage has subsided, and higher levels of radiation (in the flow stream) are being realized due to the rise in noble gas release from the fuel. The flow of effluent is routed through primary system components, including the Catalytic Recombiner, Cooler Condenser, Moisture Separator, Refrigeration Dryer, Charcoal Adsorber Vault and Charcoal Tanks, and the Post Filter before being released. Noble gas and iodine constituents are removed from the offgas process stream by adsorption on charcoal. The charcoal beds are also effective at retaining particulates (e.g., noble gas solid daughter products).

During normal operation, radiolytic gas production varies linearly with thermal power.

During the beginning of the normal mode of operation, the Catalytic Recombiner temperature steadily rises with reactor power up to 100% reactor power, at which point the Catalytic Recombiner temperature stabilizes.

One dryer is inservice, with the other on standby.

11.3.2.3.5 Off-Normal Operation

The only off-normal conditions that OGS is intended to function in is moisture intrusion or ignition of charcoal in the charcoal tanks. The first of the four in-series charcoal tanks, referred to as the Guard Bed, can be bypassed individually. The remaining three tanks can be isolated and bypassed as a whole. For conditions causing only a bypass of the Guard Bed, the OGS can continue operating at a reduced but still acceptable level of radioactive isotope holdup. For an off-normal condition affecting all four tanks, the OGS will not perform the function of holdup and decay of radioactive isotopes. The isolation of the charcoal tanks allows for corrective action to be taken to return the charcoal tanks back to normal conditions. Corrective action for moisture intrusion is accomplished with drains on each charcoal tank, and a nitrogen purge capability.

11.3.2.3.6 System Shutdown

OGS Shutdown operation essentially functions as a reversal of the startup operation. As the reactor powers down, the OGS will continue operating until the main condenser vacuum is switched to the Mechanical Vacuum Pump. When vacuum flow is no longer flowing through the OGS, a final purge flow of inert gas is sent through the OGS to eliminate any remaining gas in the piping. When the plant has reached Shutdown operation mode, the OGS is secured.

11.3.2.4 Ventilation System Description

The design of the HVS is described in detail in Section 9.4. Airborne radioactivity sources are described in Subsection 12.2.2. HVS radioactivity monitoring is performed by the PRM as described in Section 11.5. Section 11.5 also describes automatic actions supported by PRM.

The HVS maintains design temperatures, humidity, quality of air, and pressurization in areas of the power block, except containment, during normal operation. Containment is serviced by the containment cooling system as described in Subsection 9.4.7. Containment air is exhausted as required by the Containment Inerting System as described in Subsection 9.3.6 via the RB HVS.

The HVS consists of subsystems for fresh air and ventilation. With the exception of the Control Building subsystem, all of the potentially radioactive subsystem's exhaust to a common Plant Vent Stack during all normal operation modes. The Plant Vent Stack is monitored for radiation. The Control Building is outside the radiologically controlled area and, therefore, the exhaust does not require radioactive monitoring.

11.3.3 Safety Evaluation

Regulatory Requirements

OGS complies with the following GDC, which have plant-wide applicability, as discussed in Section 3.1:

GDC 2 – Design bases for protection against natural phenomena.

GDC 3 – Fire protection.

GDC 5 – Sharing of structures, systems, and components.

The following GDC have aspects specific to OGS and compliance is addressed as follows:

GDC 60, as it relates to the design of the OGS to control releases of radioactive gaseous effluents generated during normal reactor operations, including AOOs as described in Subsection 11.3.4. Section 9.4 discusses GDC 60 as it relates to HVS.

GDC 61, as it relates to the ability of the OGS design to ensure adequate safety under normal and postulated accident conditions, as provided in SRP Section 11.3 using guidance in BTP 11-5 and the analysis of RG 1.143 in assigning the safety classification to SSCs of the OGS for design purposes.

Additional information on GDC compliance is found in Section 3.1.

The OGS is designed to comply with the following:

- 10 CFR 20.1101(b), as it relates to the use of engineering controls in maintaining doses to members of the public “As Low As Reasonably Achievable” is described in Section 12.5
- 10 CFR 20.1301, 10 CFR 20.1302, and 10 CFR Part 20, Appendix B, Table 2 as they relate to radioactivity in gaseous effluents released to unrestricted areas and doses to offsite receptors located in unrestricted areas
- 10 CFR 20.1406, as it relates to the design and operational procedures to minimize contamination, facilitate eventual decommissioning, and minimize the generation of radioactive waste as described in Chapter 12
- 10 CFR Part 50, Appendix I, Section II.B and II.C, as they relate to the numerical guidelines for design objectives and limiting conditions for operation to meet the “As Low As Reasonably Achievable” criterion
- 40 CFR Part 190, “Environmental Radiation Protection Standards For Nuclear Power Operations,” as implemented under 10 CFR 20.1301(e), as it relates to limits on annual doses from all sources of radioactivity released in gaseous effluent discharges and external radiation from site buildings and facilities

11.3.3.1 Radioactive Effluent Releases and Dose Calculations Due to Gaseous Radioactive Waste System Leak or Failure

A site-specific Offgas System Failure analysis to demonstrate conformance with BTP 11-5 and RG 1.143 will be provided in CPA Part 2.

The radiological consequences are determined for an OGS accident as specified in Standard Review Plan Section 11.3, BTP 11-5. BTP 11-5 states that operator action could take up to two hours to terminate the event. The noble gas isotope releases are shown in Table 11.3-1. The dose results are presented in Table 11.3-2 and show the BWRX-300 design meets the acceptance criterion of BTP 11-5. Conservative passive plant analyses assume no operator action takes place for 72 hours. The BTP 11-5 analysis is also performed assuming the two-hour

operator action is not initiated for 72 hours (74 hours total release duration). For comparison, the results of this conservative analysis are presented in addition to the BTP 11-5 results for releases and dose in Table 11.3-1 and Table 11.3-2, respectively.

Methodology

The OGS equipment, piping, and valves are designed to be resistant to hydrogen detonations and seismic events. Thus, the OGS failure dose is determined using the methods and assumptions from BTP 11-5, Section B.2.B, which provides guidance for OGS failure dose consequence for BWRs containing charcoal delay beds, like the BWRX-300.

Three potential release paths were considered. The first release path would bypass the guard bed and pass through the remaining three follow-on charcoal beds. This flow path would have a minimal impact on the radiological release as most of the charcoal beds would remain in service. The second release path would bypass all the charcoal beds. This flow path has a closed manual valve in series with a closed automatic valve. A release through this path would require two independent failures that have no potential common cause and, therefore, was not analyzed. The third release path is through the guard bed and bypasses the three follow-on charcoal beds. This path has two normally closed automatic valves that could be subject to a single operator error. Therefore, the path through the guard bed bypassing the follow-on charcoal beds was evaluated for compliance with BTP 11-5.

Note that this event additionally requires the single active failure of the downstream automated isolation valve that is designed to isolate the OGS upon detection of high radiation in the effluent stream. Even with the failure of the downstream isolation valve, it is not anticipated nor assumed that the isolation instrumentation would fail but would instead alarm the Control Room (CR) with a high radiation alarm causing the operator to manually isolate the OGS (i.e., close the suction valves) within 2 hours of the alarm. However, this analysis differs from the BTP 11-5 on the following points:

1. There is no motive force to remove any inventory from the charcoal delay beds while in bypass, therefore no activity from these tanks is included in the final release calculation.
2. Flow is through a single “guard bed” charcoal tank with an evaluated hold up time given by NUREG-0016, Equation 4-7, using dynamic adsorption coefficients for Kr and Xe from NUREG-0016.
3. Additional case results are run assuming the two-hour operator action is not initiated for 72 hours after event initiation.

Evaluation Results

The evaluation assumptions are given in Table 11.3-3, the isotopic flows and releases in Table 11.3-1, and the dose results in Table 11.3-2.

11.3.4 Radioactive Effluent Releases

Methodology

The analysis is performed with the GALE-BWR methodology which implements the assumptions outlined in RG 1.112, Revision 1. The GALE-BWR methodology is described in NUREG-0016, Revision 2. GALE-BWR calculates expected releases based on:

- Standardized coolant activities derived from ANS Standards 18.1 Working Group recommendations
- Release and transport mechanisms that result in the appearance of radioactive material in liquid and gaseous waste streams

- Plant-specific design features used to reduce the quantities of radioactive materials ultimately released to the environs
- Information received on the operation of nuclear power plants

The calculations are based on data generated from operating reactors, field tests, laboratory tests, and plant-specific design considerations incorporated to reduce the quantity of radioactive materials that may be released to the environment. The GALE-BWR method determines the release of radioactive materials (noble gases, particulates, carbon-14, tritium, argon-41, and iodine) in gaseous effluents from normal operation including AOOs from the following sources: main condenser OGS, TGSS, mechanical vacuum pumps, and ventilation exhaust air from buildings in the BWRX-300 power block (RB, TB, and RWB). The BWRX-300 does not have a wetwell or suppression pool, so the entire containment is a drywell as defined in NUREG-0016, Revision 2.

Gaseous Release Evaluation Results

OGS design parameters used for effluent releases are listed in Table 11.3-4. The BWRX-300 annual average airborne release activities from various plant systems are shown in Table 11.3-5.

A PPE evaluation to show gaseous releases are within regulatory limits for all nuclides released will be provided in CPA Part 2.

Dose Evaluation Results

A description of the radiological effects of gaseous radioactive waste effluents from normal plant operation at BECPV on members of the public will be provided in CPA Part 2. An estimate of the maximum doses to the public will also be provided in CPA Part 2 as well as an evaluation of these doses compared to applicable regulatory limits.

The projected total annual doses per reactor from both liquid and gaseous radioactive effluents are within the design objectives of 10 CFR 50, Appendix I.

The projected total site doses due to liquid and gaseous radioactive effluents and direct radiation are within the environmental standards of 40 CFR 190.

Because 40 CFR 190 is more restrictive than 10 CFR 20.1301, compliance with the limits of 40 CFR 190 demonstrates compliance with the 0.1 rem (100 mrem) limit of 10 CFR 20.1301.

11.3.5 Testing and Inspection Requirements

The Initial Test Program is described in Chapter 14.

11.3.6 Instrumentation and Controls

11.3.6.1 Minimum Offgas Flow Rate Control and Air Injection

A minimum offgas flowrate is maintained while there is flow through the Offgas Recombiner to support normal function of the Catalytic Recombiner.

Flow detectors continuously measure the minimum offgas flowrate and transmit a control signal upstream to the MCAs to inject additional air into the effluent flow if this minimum is not met.

11.3.6.2 Instrumentation

The following are the key instrumentation features of the OGS:

1. Inlet System Flow – Flow is measured by a flow element prior to the Preheater section to monitor the combined offgas effluent stream from MCA system and HWC system. Low

Flow generates an alarm in the MCR to alert the operators to potential degraded catalyst performance due to below minimum flow, line blockage, or degraded SJAE output.

2. Preheater Section – Temperature of offgas effluent leaving the Preheater section is monitored by the Distributed Control and Information System (DCIS) and available in the MCR by triplicated temperature elements measurements. These measurements are used to control preheater inlet steam and alarm on low gas temperature entering the Catalytic Recombiner section (which may affect the recombination operation.)
3. Catalytic Recombiner Section – Temperature of the Catalytic Recombiner section is measured at three centerline locations, the first is within the first few inches of the entrance and the other two are evenly spaced to provide a profile of the rest of the Catalyst. High/Low temperatures generate alarms in the MCR to alert the operator of deviations from normal recombination.
4. Condenser Section – Temperature of the offgas effluent stream is measured after the Offgas Condenser section with a temperature element to monitor the effluent for insufficient cooling to remove undesired moisture. A High Temperature alarm is annunciated in the MCR to indicate degraded performance of Condenser section and reduced moisture removal.
5. Offgas Recombiner Vessel – The differential pressure is monitored across the vessel to indicate the presence of fouling, blockage, or other abnormal function of the Offgas Recombiner. A High Differential Pressure alarm annunciates in the MCR.
6. Condenser Section Condensate Drain – Liquid flow is measured by a flow element provided on the condensate drain line of the Recombiner Condenser section to permit monitoring of performance of the Condenser and the amount of condensate water being returned to the Main Condenser. Insufficient Condenser function will generate a Low Flow alarm in the MCR. Two level indicators are used on the mechanical float traps that alarm in the MCR on High Level to indicate a failed trap, or abnormally high flow.
7. Moisture Separator – Temperature of the offgas effluent stream is measured by a temperature element after the Moisture Separator to monitor moisture separator performance. High temperature alarms in the MCR to alert operators to degraded moisture separator performance and a potential for charcoal vault moisture intrusion. Water level for the drain loop seal on the moisture separator is monitored by a level indicator. High and low water level alarms in the MCR provide indication of potential drain line issues.
8. Refrigeration Dryer – Flow is measured at the outlet of the Refrigeration Dryers by flow element to monitor charcoal vault flow. High Flow generates an alarm in the MCR to alert the operator of potential damage to charcoal tanks in normal operation. Low Flow generates an alarm in the MCR to alert the operator of a likely leak or blockage in the system. Dryer train failure generates an alarm in the MCR to alert the operator that one or both of the dryer trains have failed and become inactive. Water level for the drain loop seals on refrigeration dryer train lines are monitored by a level switch. High and Low water levels alarm in the MCR for potential drain line issues. Temperature is taken at the outlet of the Refrigeration Dryers to monitor performance with indication in the MCR. Manual isolation of dryers from operator action is provided for maintenance as needed. Indication of the status of the dryers is provided in the MCR.
9. Offgas Process Radiation Monitoring – Process radioactivity at two sample points, downstream of the Refrigeration Dryer, and downstream of the charcoal beds, are monitored by PRM and are available in the MCR. High Activity isolates the OGS discharge

valves, alarms in the MCR to indicate to operator so that action may be taken to avoid releases in excess of regulatory or plant specific limits and indicates potential fuel failure.

10. Gas Analyzer – Hydrogen concentration, oxygen concentration, and moisture content is monitored at the outlet of the Refrigeration Dryers by triplicated gas analyzers. The annunciators of high hydrogen concentration and high/low oxygen concentration alarm are provided at the MCR and also serve as an input to the HWC system to adjust its injection of hydrogen into the plant and oxygen into the OGS as needed. High Hydrogen –1% by volume, alarms in the MCR and locally to indicate HWC must inject more oxygen and potential Offgas Recombiner degraded performance. High-High Hydrogen, 4% by volume, isolate the OGS discharge valves and alarms both locally and in the MCR to indicate risk of hydrogen detonation is present. High moisture alarms in the MCR to indicate potential condenser or dryer issues.
11. Charcoal Adsorber Vault – Radioactivity in the Charcoal Adsorber Vault environment is monitored at the outlet of the Charcoal Adsorber Vault ventilation by PRM. High radioactivity alarms locally and in the MCR to indicate presence of leak in the Charcoal Adsorber Vault. Temperature is measured in the Charcoal Adsorber Vault by triplicated temperature elements. This provides input to the Charcoal Adsorber Vault ventilation to maintain vault temperature and High/Low Temperature alarms in the MCR indicated reduced adsorber performance.
12. Charcoal Beds – CO concentration is measured at the outlet of the Guard Bed and the fourth charcoal bed by two separate detectors. High CO concentration alarms in the MCR to indicate presence of combustion in the guard bed or the subsequent beds. Temperature is measured at three evenly spaced points along each tank. High Temperature alarms in the MCR to indicate potential for smoldering combustion or deep-seated charcoal fire. Differential Pressure is measured across the Charcoal Adsorber Vault to monitor for blockage or other abnormal function and alarms in the MCR.
13. Post Filter – Differential pressure is measured across the filter to monitor for blockage or other abnormal function. High differential pressure alarms in the MCR alert the operator to perform maintenance or bypass the filter.
14. Outlet Flow – Offgas effluent flow rate is measured prior to discharge of the offgas effluent to the HVS by a flow element. This indication is partially used to determine radioactive releases from the plant, and the air in-leakage into the MCA system.
15. Remote Operated Valves – Valve Position is indicated in the MCR for remote operated valves (Air and Motor).

Additional information about OGS instrumentation is included in Subsection 7.3.4.

11.3.6.3 Control

Controls for the OGS are available in the MCR to facilitate system operation. System and component operating status, including the state of any bypasses or manual overrides, are provided at the control console. Manual initiation and shutdown of the OGS is provided from the MCR.

OGS parameter indications and alarms are provided in the MCR.

11.3.7 References

11.3-1 ANSI/ANS-55.4 “Gaseous Radioactive Waste Processing Systems for Light Water Reactors,” American National Standards/American Nuclear Society, 2017.

Table 11.3-1: Offgas Failure Environmental Releases

To be provided in CPA Part 2.

Table 11.3-2: Offgas System Failure Dose Results

To be provided in CPA Part 2.

Table 11.3-3: Offgas System Failure Accident Parameters

To be provided in CPA Part 2.

Table 11.3-4: BWRX-300 Offgas System Design Parameters Used for Effluent Release

Calculation Bases	Input
Xe Dynamic Adsorption Coefficient	330.0 cm^3/g [9134.4 in^3/lb]
Kr Dynamic Adsorption Coefficient	18.5 cm^3/g [512.1 in^3/lb]
Ar Dynamic Adsorption Coefficient	6.4 cm^3/g [177.2 in^3/lb]
Charcoal Mass per Adsorber Bed (4 Beds)	11,158 kg, [24,600 lb]
Charcoal Adsorber Vault Temperature	21.1 °C (70.0 °F)
Plant Capacity Factor	0.95
Plant Thermal Power	870 MWt
BWRX-300 Normal Operational Noble Gas Source Terms in Steam at t=0 min	Table 11.1-1

Table 11.3-5: BWRX-300 Annual Airborne Releases Including AOOs (Ci/yr) (Sheet 1 of 3)

Nuclide	Reactor Building	Plant Services Area	Turbine Building	Radwaste Building	Mechanical Vacuum Pump	Turbine Seal	Offgas System	Containment	Total[1]
Kr-83m	-	-	-	-	-	8.9E+00	1.3E-11	8.5E-03	8.9E+00
Kr-85m	2.4E-02	7.1E-02	5.9E-01	-	-	2.4E-01	2.1E-04	5.6E-04	9.3E-01
Kr-85	-	-	-	-	-	6.0E-02	6.0E+01	8.0E-03	6.0E+01
Kr-87	-	4.8E-02	1.4E+00	-	-	1.7E+00	7.3E-19	1.1E-03	3.2E+00
Kr-88	2.4E-02	7.1E-02	2.2E+00	-	-	9.3E-01	2.6E-07	1.4E-03	3.2E+00
Kr-89	-	4.8E-02	1.4E+01	6.9E-01	-	3.2E+02	-	8.7E-03	3.3E+02
Xe-131m	-	-	-	-	-	5.0E-02	9.8E-01	4.3E-03	1.0E+00
Xe-133m	-	-	-	-	-	3.0E-02	1.8E-08	8.2E-04	3.1E-02
Xe-133	6.4E-01	2.0E+00	3.6E+00	5.2E+00	3.1E+01	4.5E-01	6.4E-02	2.6E-02	4.3E+01
Xe-135m	3.6E-01	1.1E+00	9.5E+00	1.3E+01	-	8.1E+00	-	1.1E-03	3.2E+01
Xe-135	7.8E-01	2.2E+00	7.8E+00	6.7E+00	1.2E+01	4.5E+00	4.7E-50	2.2E-02	3.4E+01
Xe-137	1.1E+00	3.2E+00	2.4E+01	2.0E+00	-	1.5E+01	-	5.0E-04	4.5E+01
Xe-138	4.8E-02	1.4E-01	2.4E+01	4.8E-02	-	4.2E+01	-	5.2E-03	6.6E+01
I-131	1.9E-04	9.8E-04	6.7E-03	4.2E-04	2.4E-03	6.0E-05	-	3.4E-03	1.4E-02
I-132	1.9E-03	9.8E-03	6.7E-02	4.2E-03	-	-	-	5.6E-04	8.3E-02
I-133	1.4E-03	7.3E-03	5.0E-02	3.1E-03	-	1.2E-04	-	3.8E-03	6.5E-02
I-134	5.5E-03	2.8E-02	1.9E-01	1.2E-02	-	-	-	6.1E-04	2.4E-01
I-135	2.9E-03	1.5E-02	9.9E-02	6.3E-03	-	-	-	2.4E-03	1.3E-01
H-3	1.3E+01	-	1.3E+01	-	-	-	-	-	2.6E+01
C-14	-	-	-	-	-	-	1.1E+01	-	1.1E+01
Na-24	-	-	-	-	-	-	-	3.9E-05	3.9E-05
P-32	-	-	-	-	-	-	-	1.9E-05	1.9E-05

Table 11.3-5: BWRX-300 Annual Airborne Releases Including AOOs (Ci/yr) (Sheet 2 of 3)

Nuclide	Reactor Building	Plant Services Area	Turbine Building	Radwaste Building	Mechanical Vacuum Pump	Turbine Seal	Offgas System	Containment	Total[1]
Ar-41	-	-	-	-	-	-	8.8E-03	-	8.8E-03
Cr-51	1.2E-08	5.5E-08	5.5E-08	4.3E-08	-	-	-	5.3E-04	5.3E-04
Mn-54	2.4E-08	6.1E-08	3.6E-08	2.4E-07	-	-	-	3.1E-04	3.1E-04
Mn-56	-	-	-	-	-	-	-	1.2E-05	1.2E-05
Fe-55	-	-	-	-	-	-	-	6.6E-04	6.6E-04
Fe-59	5.5E-09	1.8E-08	6.1E-09	1.8E-08	-	-	-	1.6E-04	1.6E-04
Co-58	6.1E-09	1.2E-08	6.1E-08	1.2E-08	-	-	-	1.4E-04	1.4E-04
Co-60	6.1E-08	2.4E-07	6.1E-08	4.3E-07	-	-	-	2.9E-04	3.0E-04
Ni-63	-	-	-	-	-	-	-	6.6E-07	6.6E-07
Cu-64	-	-	-	-	-	-	-	1.6E-04	1.6E-04
Zn-65	6.1E-08	2.4E-07	3.6E-07	1.8E-08	-	-	-	1.3E-04	1.3E-04
Rb-89	-	-	-	-	-	-	-	1.2E-06	1.2E-06
Sr-89	1.8E-09	1.2E-09	3.6E-07	-	-	-	-	1.8E-06	2.2E-06
Sr-90	1.8E-10	4.3E-10	1.2E-09	-	-	-	-	9.6E-08	9.8E-08
Y-90	-	-	-	-	-	-	-	2.4E-08	2.4E-08
Sr-91	-	-	-	-	-	-	-	4.6E-05	4.6E-05
Sr-92	-	-	-	-	-	-	-	2.8E-05	2.8E-05
Y-91	-	-	-	-	-	-	-	2.4E-05	2.4E-05
Y-92	-	-	-	-	-	-	-	1.1E-05	1.1E-05
Y-93	-	-	-	-	-	-	-	3.5E-06	3.5E-06
Zr-95	1.8E-08	4.3E-08	2.4E-09	4.9E-08	-	-	-	5.1E-05	5.1E-05
Nb-95	6.1E-08	5.5E-07	3.6E-10	2.4E-10	-	-	-	4.8E-05	4.8E-05
Mo-99	3.6E-07	3.6E-06	1.2E-06	1.8E-10	-	-	-	6.5E-05	7.0E-05

Table 11.3-5: BWRX-300 Annual Airborne Releases Including AOOs (Ci/yr) (Sheet 3 of 3)

Nuclide	Reactor Building	Plant Services Area	Turbine Building	Radwaste Building	Mechanical Vacuum Pump	Turbine Seal	Offgas System	Containment	Total[1]
Tc-99m	-	-	-	-	-	-	-	6.1E-06	6.1E-06
Ru-103	1.2E-08	2.4E-07	3.0E-08	6.1E-11	-	-	-	1.2E-05	1.2E-05
Rh-103m	-	-	-	-	-	-	-	5.0E-08	5.0E-08
Ru-106	-	-	-	-	-	-	-	2.0E-06	2.0E-06
Rh-106	-	-	-	-	-	-	-	2.8E-07	2.8E-07
Ag-110m	-	-	-	-	-	-	-	6.5E-07	6.5E-07
Sb-124	1.2E-09	1.8E-09	6.1E-09	4.3E-09	-	-	-	-	1.3E-08
Te-129m	-	-	-	-	-	-	-	2.3E-05	2.3E-05
Te-131m	-	-	-	-	-	-	-	4.1E-06	4.1E-06
Te-132	-	-	-	-	-	-	-	1.8E-06	1.8E-06
Cs-134	4.3E-08	2.4E-07	1.2E-08	1.5E-07	-	-	-	1.7E-05	1.8E-05
Cs-136	6.1E-09	2.4E-08	6.1E-09	-	-	-	-	9.9E-06	9.9E-06
Cs-137	6.1E-08	3.0E-07	6.1E-08	2.4E-07	-	-	-	2.7E-05	2.8E-05
Cs-138	-	-	-	-	-	-	-	3.0E-06	3.0E-06
Ba-140	1.2E-07	1.2E-06	6.1E-07	2.4E-10	-	-	-	1.9E-04	1.9E-04
La-140	-	-	-	-	-	-	-	4.4E-05	4.4E-05
Ce-141	1.2E-08	4.3E-08	6.1E-07	4.3E-10	-	-	-	1.2E-05	1.2E-05
Ce-144	-	-	-	-	-	-	-	2.0E-06	2.0E-06
Pr-144	-	-	-	-	-	-	-	2.3E-09	2.3E-09
W-187	-	-	-	-	-	-	-	1.4E-05	1.4E-05
Np-239	-	-	-	-	-	-	-	4.8E-05	4.8E-05

Cells marked with – represents no release.

Note 1: Effluent releases are based on OE from NUREG-0016

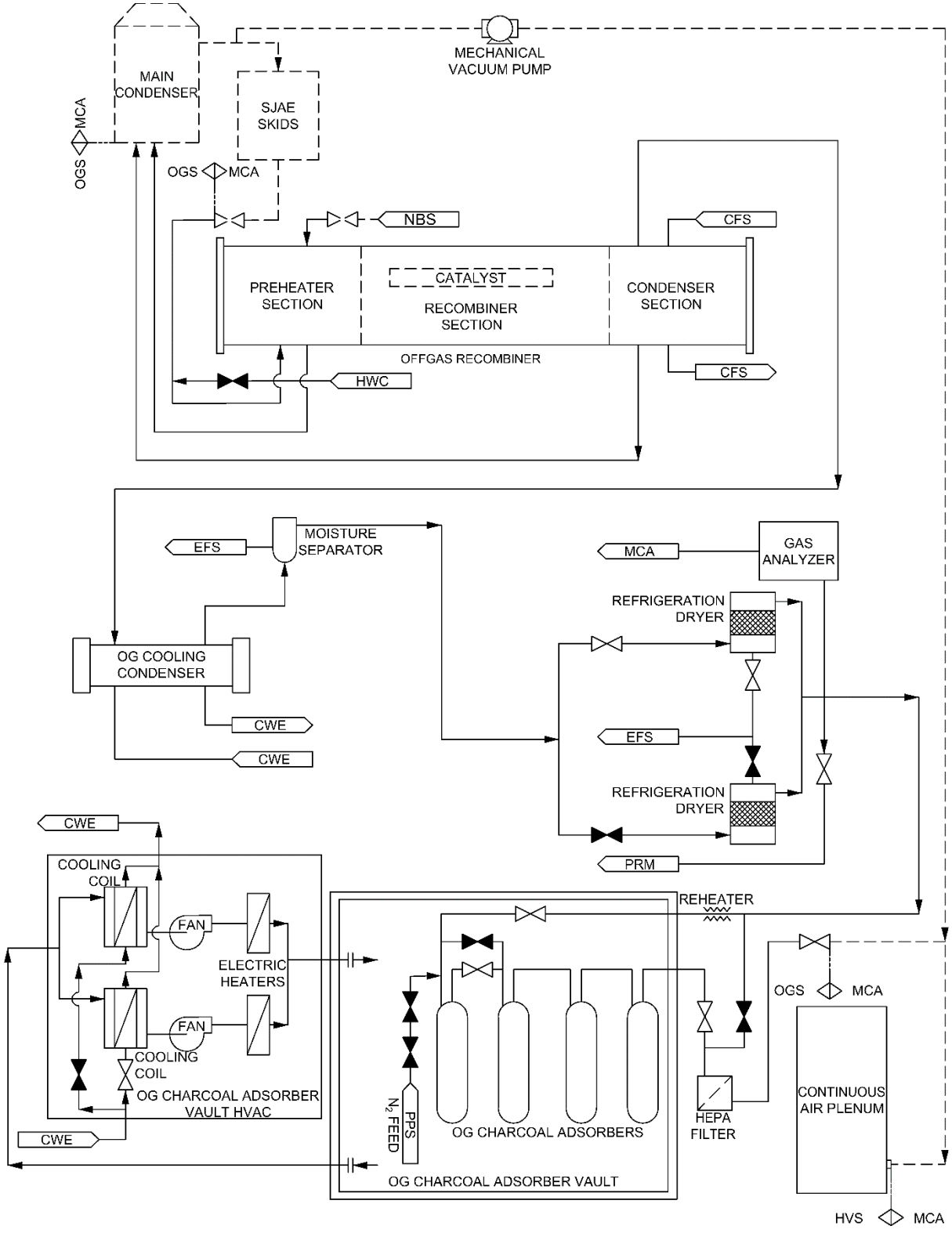


Figure 11.3-1: Offgas System Simplified Diagram

11.4 Solid Waste Management System

The SWM system controls, collects, handles, processes, packages, and temporarily stores solid waste generated by the plant prior to shipping the waste offsite. The SWM processes the filter backwash sludges, and bead resins generated by the LWM, FPC, ICC, and CFD systems. Contaminated solids such as HEPA filters, cartridge filters, rags, plastic, paper, clothing, tools, and equipment are also disposed of in the SWM. Other sources of waste including solids from lab samples, sump waste, and oily waste from the EFS is processed through the SWM. To the greatest extent possible, discarded chemicals (including those classified as hazardous) are kept out of the SWM.

11.4.1 Design Bases

Safety Category 1 Functions

None.

Safety Category 2 Functions

None.

Safety Category 3 Functions

- The SWM safely controls, collects, handles, processes, packages, and temporarily stores solid radioactive waste generated by the plant prior to shipping the waste offsite.

Safety Category N Functions

- Portions of the SWM that do not support the fundamental safety function of containing radioactive material are Safety Category N.

The Safety Classifications, Quality Groups, and Seismic Categories of the SWM SSCs are listed in Table 3A-1.

11.4.2 System Description

The SWM is primarily located in the RWB with individual collection locations around the plant, and consists of the following subsystems:

- Spent Resin Processing
- Sludge Processing
- Solid Waste Storage

Figure 11.4-1 provides a simplified single line diagram for the SWM. The subsystems, components and their functions are described in the following subsections.

Dry Active Waste is stored onsite prior to packaging e.g., contaminated clothing, paper, and plastic. These wastes are collected in containers located in appropriate areas throughout the plant. The filled containers are sealed and moved to controlled-access enclosed areas for temporary storage, sorting, packaging, and shipment to either a licensed offsite processor or for ultimate disposal.

11.4.2.1 Equipment, Arrangement, and Location

The SWM major components, which include the spent resin tank, sludge tanks, High Integrity Containers (HICs) (containers designed or engineered for radioactive waste), dewatering pump, sludge tank transfer pumps, spent resin transfer pump, and drum evaporator are located in the RWB. Adequate space is provided for on-site storage of the HICs, shielded filter containers, and

drums. Shielded HIC loading stations are located to allow for easy installation and removal. Simplified General Arrangement drawings for the RWB are provided in Subsection 1.2.3.

Dry active waste is collected at locations where generated and transferred to a low dose area for sorting, packaging, and storage prior to offsite transport. Sufficient space is available in low dose areas for dry active waste storage and handling.

11.4.2.2 Major Components Description

The design of the SWM utilizes and conforms to the guidance provided in RG 1.143. The applicable standards from RG 1.143 Table 1 are used in the design, construction, and testing of the SWM components (see Table 11.2-1). The applicable design criteria from RG 1.143, Table 2, Table 3, and Table 4 are used in the design analysis of the SWM components. The spent resin tank, sludge tanks, pumps, valves, and piping are classified as RW-IIa. These components are classified as non-seismic per provisions in RG 1.143, Section 3.4.

The following sections describe the major components in the SWM.

11.4.2.2.1 Spent Resin Processing

Spent Resin Tank

The spent resin tank is a cylindrical stainless steel atmospheric tank fitted with a liquid level sensor and transmitter to prevent overfilling of the tank. The spent resin tank is also fitted with an overflow connection at least the size of the largest tank inlet and installed below the tank vent and above the high-level alarm setpoint. The tank overflow line is routed to the EFS. A mixing eductor is fitted to the tank to facilitate mixing of the heavier solids and liquid and prevent heavier solids from settling at the bottom of the tank. Tank capacities are selected to provide sufficient storage capacity for waste generation at normal generation rates and to allow sufficient time for the decay of shorter-lived radionuclides before shipping to an off-site facility.

Spent Resin Tank Transfer Pump

The SWM is designed with pumping capable of transferring the spent resin from the spent resin tank to the HICs.

11.4.2.2.2 Sludge Processing

The following equipment is designed to exclusively handle sludge.

Sludge Tank

The sludge tanks are cylindrical stainless steel atmospheric tanks fitted with liquid level sensors and transmitters to prevent overfilling of the tanks. The sludge tanks are also fitted with an overflow connection at least the size of the largest tank inlet and installed below the tank vent and above the high-level alarm setpoint. The tank overflow lines are routed to the EFS system. Mixing eductors are fitted to the tanks to facilitate the creation of a slurry prior to the sludge transfer to the HICs.

Tank capacities were selected to provide sufficient storage capacity for waste generation at normal generation rates and to allow sufficient time for the decay of shorter-lived radionuclides before shipping to an off-site facility.

Sludge Tank Transfer Pump

The SWM is designed with pumps capable of transferring the sludge from the sludge tanks to the HICs.

11.4.2.2.3 Solid Waste Storage

The following equipment is designed to dewater the spent resin coming from the spent resin tank and the collected sludge from the sludge tanks.

High Integrity Container Filling Stations

In accordance with 10 CFR 61.56, the SWM is designed with HICs configured with dewatering internals which remove standing liquid to less than 1% by volume. The HICs are equipped with level sensors, level transmitters, passive vents, and sample monitoring.

Dewatering Pump

The SWM is designed with a pump capable of removing water from the HICs.

Drum Evaporator

The drum evaporator is loaded with a drum filled with solids such as lab samples, sump waste, and oily waste from EFS. The drum evaporator is capable of removing excess moisture, through evaporation, which is routed to the RWB HVS. The dried waste remaining in the drum is sent offsite to an appropriately permitted vendor processor.

11.4.2.2.4 Piping and Valves

The piping and valves associated with resin and sludge systems are corrosion resistant stainless steel. The valves minimize crud buildup by utilizing full port valves for transferring, and diaphragm style valves to control flow. Full port valves reduce the potential for solidification due to minimal flow restrictions, and diaphragm valves can handle slurry fluids without the potential of contaminating working parts of the valve. Bottom valves on slurry containing tanks are designed to be flush mounted to eliminate pockets.

11.4.2.3 System and Equipment Operation

The SWM operates continuously during normal operating modes of the plant. The SWM carries out the following functions:

- Continuously remove dissolved and suspended solids, corrosion products, and spent resin to minimize the potential for entering the EFS
- Store high activity spent resin for decay
- Dewater and package the waste for temporary on-site storage prior to shipping the waste offsite
- Dry active waste is continuously collected, stored, and shipped offsite for disposal or processing through an appropriately permitted third party processor

11.4.2.3.1 Spent Resin Processing

Spent resin slurry from the ICC, CFD, FPC, and LWM systems is sent to the dedicated spent resin tank for collection and decay. When sufficient bead resins have been collected in the spent resin tank, resin is transferred to HICs for dewatering and eventual disposal.

The associated tanks, piping, pumps, and valves can be flushed using CST water to minimize trace solids and reduce dose in the system.

11.4.2.3.2 Sludge Processing

Filter and demineralizer backwash from the FPC and CFD systems is collected in the two sludge tanks to allow the solids to settle out. Backwash water from the RWST Filter and Waste Collection and Filtering Subsystem skids can also be routed to the SWM sludge tanks for processing and

liquid removal. Once enough sludge has been collected, the sludge is transferred by the sludge tank transfer pumps to the HICs for dewatering and eventual disposal.

11.4.2.3.3 Solid Waste Storage and Handling

The SWM solid waste storage subsystem consists of three HICs for solid waste storage and a combined dewatering station for high activity spent resin and concentrated waste. The dewatering pump discharge is routed back to the sludge tanks. Each HIC is equipped with a passive screened vent to prevent pressure build up. The HIC vents are routed to the HVS.

Dry solid wastes consist of items such as HEPA filters, air filters, miscellaneous paper, and rags from contaminated areas; contaminated clothing, tools, and equipment parts that cannot be effectively decontaminated; and solid laboratory low-level waste. These wastes are collected in containers located in designated areas throughout the plant, as dictated by volume of wastes generated during operation and maintenance. The filled containers are sealed and moved to controlled-access enclosed areas for temporary storage.

To the greatest extent possible, discarded chemicals (including those classified as hazardous) are kept out of the SWM system. Waste is collected primarily in collection drums and sent offsite to an appropriately permitted vendor processor. Solids from lab samples, sump waste, and oily waste from the EFS are collected in drums and loaded in the drum evaporator.

11.4.3 Safety Evaluation

SWM complies with the following GDC, which have plant-wide applicability, as discussed in Section 3.1:

GDC 3, Fire protection.

GDC 5, Sharing of structures, systems, and components.

The following GDC have aspects specific to SWM, and compliance is addressed as follows:

GDC 60, Control releases of radioactive materials to the environment, as it relates to the design of the SWM to control the release of radioactive materials in liquid and gaseous effluents from the SWM and to handle solid wastes produced during normal plant operation.

GDC 61, Fuel storage and handling and radioactivity control, as it relates to the ability of systems that may contain radioactivity to assure adequate safety under normal and postulated accident conditions. Compliance with RG 1.143 provides assurance that this criterion is met.

GDC 63, Monitoring fuel and waste storage, as it relates to the storage of solid wastes. Decay heat from activity in this system is low such that it does not require heat removal. Area radiation monitors protect against excessive personnel exposure. Therefore, the requirements of this GDC are met.

Additional information on GDC compliance is found in Section 3.1.

The SWM is designed to comply with the following regulatory requirements:

10 CFR 20.1101(b) as it relates to the use of engineering controls to achieve occupational doses that are ALARA is described in Subsection 11.4.2 and Section 12.3. Administrative controls are addressed in Section 12.5.

10 CFR 20.1406, as it relates to the design and operational procedures to minimize contamination, facilitate eventual decommissioning, and minimize the generation of radioactive waste as described in Chapter 12.

40 CFR Part 190, “Environmental Radiation Protection Standards for Nuclear Power Operations,” as implemented under 10 CFR 20.1301, as it relates to limits on total annual doses from all sources of radioactivity released in effluent streams and external radiation from site buildings and facilities.

10 CFR 61.55 and 10 CFR 61.56, as they relate to waste classifications and characteristics, processing, volume and activity inventories, onsite storage, offsite storage at licensed facilities, and disposal of dry solid and wet wastes.

10 CFR Part 71 and 49 CFR Parts 171–180, as they relate to the use of approved containers and packaging methods for the shipment of radioactive materials.

The SWM is designed to conform to the following regulatory guidance:

RG 1.143, Design guidance for radioactive waste management systems, structures, and components installed in light-water-cooled nuclear power plants, as it relates to the design guidance for radioactive waste management systems.

RG 4.21, Minimization of contamination and radioactive waste generation: life-cycle planning.

BTP 11-3, Design guidance for solid radioactive waste management systems installed in light-water-cooled nuclear power reactor plants, as it relates to guidance on SWM design and operation.

11.4.4 Testing and Inspection Requirements

The Initial Test Program is described in Chapter 14.

The equipment and components of the SWM are designed for easy inspection and maintenance during plant operation.

11.4.5 Instrumentation and Controls

11.4.5.1 Instrumentation

Level transmitters are provided on each tank or HIC for high-level indication and alarm to prevent overflow of the tanks and the HICs.

Other parameters indicated include pump pressures, remote valve positions, process mode, evaporator temperature and level.

Additional information about SWM instrumentation is included in Subsection 7.3.4.

11.4.5.2 Control

The majority of operations take place remotely in the MCR. Local controls are also provided for operation of the HICs and SWM pumps. System interlocks prevent tank overflow and prevent simultaneous transfer of spent resin and sludge to the HICs. System logic includes a flushing cycle to clean the tanks and transfer lines of residual particulates which are then transferred to the HICs for further dewatering. Flexibility in system operation exists to allow backflushing of filters into the empty sludge tank while the full sludge tank is being processed to the HICs. This flush consists of clean water addition from the CST. The addition of the flush water to the tanks allows complete mixing and forwarding of suspended solids minimizing the solids depositing within the transfer piping. Manual initiation is required to transfer the spent resin and sludge to the HICs.

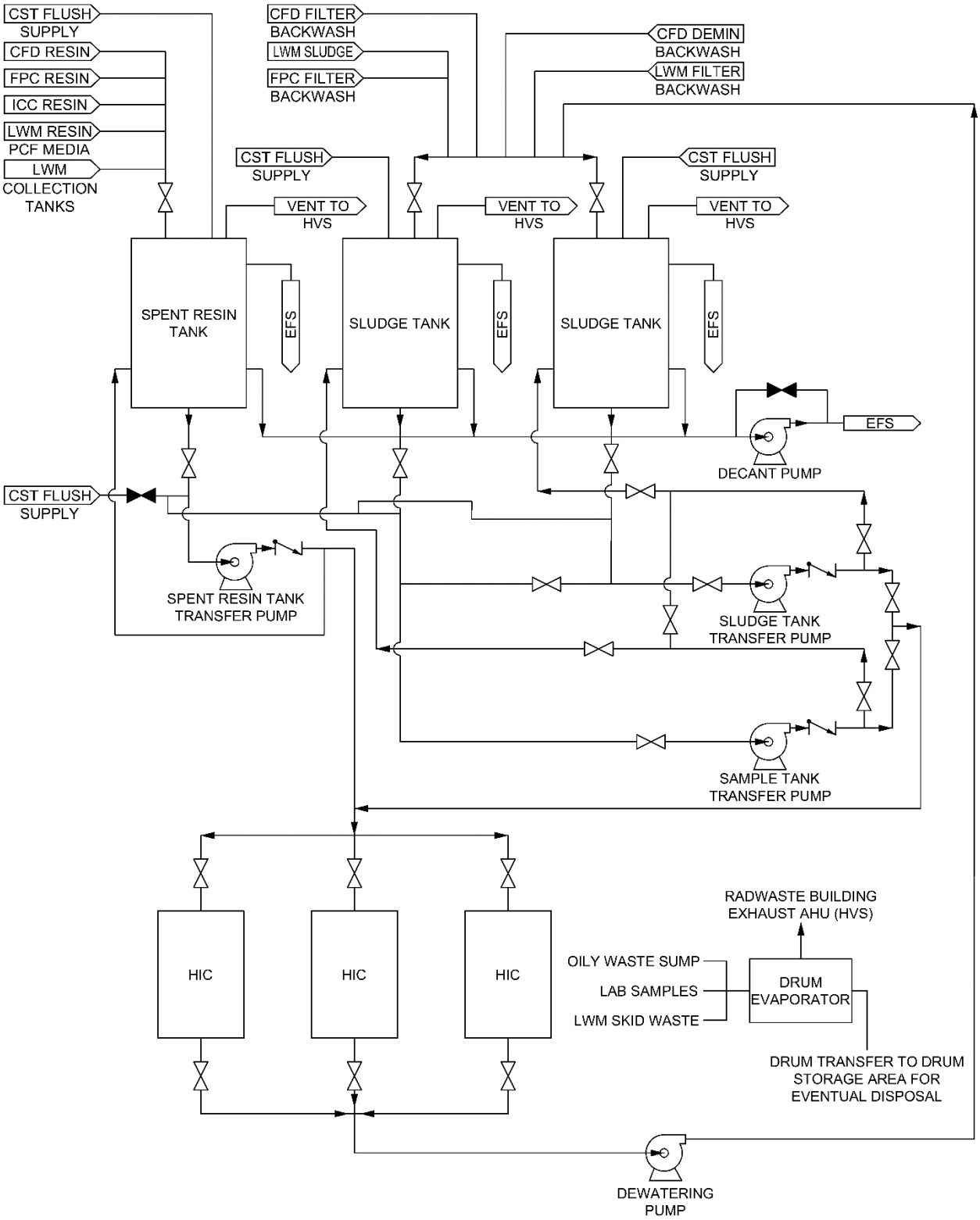


Figure 11.4-1: Solid Waste Management System Simplified Diagram

11.5 Process Radiation Monitoring

The PRM Subsystem is part of the PREMS. Descriptions of the other subsystems of the PREMS are included in other PSAR sections as indicated. Some sample lines in PREMS have CIVs, which are not described in this section. The CIVs, their safety functions, and testing are described in Subsection 6.2.4.

The PREMS includes the following sub systems:

- PRM, Section 11.5
- Area Radiation Monitoring Subsystem (ARM), Subsection 12.3.4
- Containment Monitoring Subsystem (CMon), Subsections 5.2.5 and 12.3.4
- Process Sampling Subsystem (PS), Subsection 9.3.2

11.5.1 Design Bases

Safety Category 1 Functions

None.

Safety Category 2 Functions

None.

Safety Category 3 Functions

- Monitors process and effluent streams for potentially radioactive noble gases, air particulates, and halogens

The following automatic functions are supported by the PRM:

- LWM Discharge: LWM discharge to the NHS is secured if high radiation is detected at the offsite discharge line
- RB HVS Intake: HVS Secondary Control Room (SCR) Emergency Filter Units (EFUs) and pressurization fans are aligned to provide safe breathing air to the SCR if high radiation or toxic gas is detected at either of the lower RB supply Air Handling Units (AHUs)
- OGS Discharge: OGS discharge to the HVS is secured if high radiation is detected downstream of the Charcoal Adsorber Beds
- CB HVS Intake: HVS Control Room Envelope (CRE) EFUs (and/or Toxic Gas Filtration Unit (TGFU)) are aligned to provide safe breathing air to the MCR if high radiation and toxic gas is detected at either of the CB supply AHUs

An evaluation of the potential for toxic gas hazards and determination if toxic gas monitors are necessary will be provided in CPA Part 2.

Safety Category N Functions

- Monitors process and effluent streams to permit control functions to maintain system parameters within established limits for normal operation

The Safety Classifications, Quality Groups, and Seismic Categories of the PRM SSCs are listed in Table 3A-1.

11.5.2 System Description

11.5.2.1 Process Radiation Monitoring Subsystem

PRM provides an assortment of monitoring equipment, with each device being tailored to the process/effluent stream according to fluid, location, and radiological hazard. In general, the following types of monitoring equipment are used:

- Offline Radiation Monitors: Radiation monitoring assemblies that are installed external to and away from the process stream. A continuous sample is extracted from the process and/or effluent stream and conveyed to the assembly. For example, combination skids that measure Particulate, Iodine and Noble Gas (PING) activities are commonly known as PING skids. PING skids typically include three independent analyzers (3 channels), flow extraction/control equipment, grab sampling taps, a local control panel, and audible/visual alarms. Particulate filters, iodine filters (charcoal cartridges), and grab samples are periodically collected from the skids for isotopic analysis.
- Inline Radiation Monitors: Single radiation detection element installed in the process stream. Detector can be of the scintillation or ionization chamber style and used to measure beta or gamma radiation.
- Adjacent-to-Line Radiation Monitors: Single radiation detection element mounted externally to the process line. Detector can be of the scintillation or ionization chamber style and used to measure gamma radiation.
- Area Radiation Monitors: General area radiation monitor. Detector can be of the scintillation or ionization chamber style and is used to measure gamma radiation.

The PRM also includes toxic gas monitoring. Toxic gas monitoring functions of the PRM are discussed in Section 9.4.

Real-time measurements are displayed in the MCR with alarms to notify personnel when measurements exceed preset limits. Local indication and alarms are provided where necessary to notify personnel of hazardous conditions.

Figure 11.5-1 provides a simplified single line diagram for the PRM.

PRM Instrumentation Characteristics are shown in Table 11.5-1.

The PRM interfaces with other equipment and systems as a function of system operation. The interfaces are summarized below.

11.5.2.1.1 Distributed Control and Information System

PRM instrumentation communicates with and receives electrical power from the DCIS.

11.5.2.1.2 Isolation Condenser System Monitoring

The PRM provides radiation monitoring of the IC pool areas for IC leakage detection.

11.5.2.1.3 Liquid Waste Management System

The PRM provides radiation monitoring of the LWM. One Triple Modular Redundant (TMR) set of radiation monitors is installed on the LWM discharge to the NHS System. The DCIS/LWM includes logic to automatically secure discharge to the NHS System upon detection of high radiation.

11.5.2.1.4 Offgas System Monitoring

The PRM provides radiation monitoring of the OGS as follows:

- One radiation monitor is installed on the Charcoal Adsorber Vault ventilation exhaust. This monitor provides indication of potential OGS leakage within the Charcoal Adsorber Vault.
- One noble gas monitoring skid is installed upstream of the Charcoal Adsorber Tanks. Similar to a PING skid, the skid includes an online noble gas analyzer and grab sampling taps. The skid provides baseline radiological data for monitoring OGS performance and is also the primary point of monitoring/sampling for fuel leakage. Elevated radioactivity levels at this location are typically indicative of fuel failures within the core.
- One noble gas monitoring skid is installed downstream of the Charcoal Adsorber Vault, between the Offgas HEPA filter and final discharge to the Continuous Exhaust Air Plenum. The skid includes a TMR set of online noble gas analyzers (to support automated control logic) and integral grab sampling taps. The DCIS/OGS includes logic to automatically secure OGS discharge to the HVS and open the recirculation line back to the main condenser upon detection of high radiation.

11.5.2.1.5 Main Turbine Equipment

The PRM provides radiation monitoring for the Main Turbine Gland Seal Condenser (GSC) Exhaust. One radiation monitor is installed on the GSC exhaust upstream of the Continuous Exhaust Air Plenum.

11.5.2.1.6 Circulating Water System Service Water Subsystem

The PRM provides radiation monitoring of the Circulating Water System (CWS) Service Water Subsystem. One radiation monitor is installed on the common header downstream of the Plant Cooling Water System (PCW) heat exchangers.

11.5.2.1.7 Containment Inerting System

The PRM provides radiation monitoring of the Containment Inerting System (CIS) exhaust. One radiation monitor is installed upstream of the CIS discharge isolation valve.

11.5.2.1.8 Heating, Ventilation and Cooling System

Reactor Building Heating, Ventilation, and Cooling System Monitoring

The PRM provides radiation monitoring of the RB HVS supply. Toxic gas monitoring is also provided. One set of radiation monitors is installed on the intake of each lower RB supply AHU (total of six monitors). The DCIS/HVS includes logic to automatically align SCR EFUs and pressurization fans upon detection of high radiation (or toxic gas, as described in Section 9.4).

The PRM provides radiation monitoring of the lower RB HVS exhaust. One radiation monitor is installed upstream of the common RB exhaust AHUs, prior to where the lower RB HVS and Refuel Floor / Fuel Pool Area HVS exhaust ducts combine.

The PRM provides radiation monitoring of the Refuel Floor / Fuel Pool Area HVS exhaust. One radiation monitor is installed upstream of the common RB exhaust AHUs, prior to where the Refuel Floor / Fuel Pool Area and lower RB HVS exhaust ducts combine.

Turbine Building Heating Ventilation and Cooling System Monitoring

The PRM provides radiation monitoring of the TB HVS exhaust. One radiation monitor is installed upstream of each exhaust AHU set (total 2 monitors).

The PRM provides radiation monitoring upstream of the exhaust AHUs.

The PRM provides radiation monitoring of the Plant Vent Stack (PVS). One PING monitoring skid is provided. The skid includes a second noble gas monitor for accident range measurement.

Control Building Heating Ventilation and Cooling Monitoring

The PRM provides radiation monitoring of the CB HVS supply. One TMR set of radiation monitors is installed on the intake of each CB supply AHU (total of six monitors). Toxic gas monitoring is also provided. The DCIS/HVS includes logic to automatically align CRE EFUs upon detection of high radiation (and/or TGFUs upon detection of toxic gas). One radiation monitor is installed downstream of each CRE EFU (total of two monitors).

Radwaste Building Heating Ventilation and Cooling System Monitoring

The PRM provides radiation monitoring of the RWB HVS exhaust. One radiation monitor is installed upstream of the exhaust AHUs.

11.5.2.1.9 System and Equipment Operations

The PRM provides monitoring during all plant modes of operation, unless removed from service for maintenance. Sampling equipment either operates continuously (where applicable) or in grab sample mode. There are no unique startup protocols.

Implementation of the following programs is described in Section 13.4:

- Radiological Effluent Control Programs
- Offsite Dose Calculation Manual
- Radiological Environmental Monitoring Program

11.5.3 Safety Evaluation

In accordance with 10 CFR 20.1302 and 10 CFR 20.1301(e), the PRM is designed to monitor radioactivity in plant radiological effluents to unrestricted areas resulting from operation during normal plant operations and AOOs. The PRM provides monitoring of the effluent streams as described in Subsection 11.5.2.

PRM complies with the following GDC, which has plant-wide applicability, as discussed in Section 3.1:

GDC 2 – Design bases for protection against natural phenomena.

GDC 5 – Sharing of structures, systems, and components.

The following GDC have aspects specific to PRM, and compliance is addressed as follows:

PDC 19, Control room. None of the isolation functions of the PRM are credited to mitigate the dose consequences of the Design Basis Analysis (DBA). The analysis models the CB HVS only for the purpose of propagating contamination into the MCR. This PDC is met as described in Section 3.1 and Subsection 15.5.9.

GDC 60, Control releases of radioactive materials to the environment, as it relates to ensuring potentially significant radioactive discharge paths are equipped with a control system to automatically isolate the effluent on indication of a high radiation level. The PRM provides radiation monitoring and supports automatic isolation for the following discharge paths:

- Radiation monitoring of offsite discharge supports automatic LWM isolation in the event of high radiation
- Automatically secure OGS discharge to the HVS and open the recirculation line back to the main condenser upon detection of high radiation

GDC 63, Monitoring fuel and waste storage, as it relates to ensuring fuel storage and radioactive waste systems and their associated handling areas are monitored for excessive radiation levels. The PRM provides monitoring of the LWM, OGS, and refuel/fuel pool area HVS Exhaust as described in Subsection 11.5.2.

GDC 64, Monitoring radioactive releases, as it relates to ensuring radiation levels in the containment atmosphere, IC cubicles, effluent discharge paths and important process streams are monitored for radioactivity, as described in Subsection 11.5.2. Monitoring of radiation levels in the containment atmosphere is described in Section 6.8.

Additional information on GDC compliance is found in Section 3.1.

The PRM is designed to comply with the following regulations:

10 CFR 20.1101(b) as it relates to the use of engineering controls in maintaining doses to members of the public “As Low As Reasonably Achievable”. Administrative controls are addressed in Section 12.5.

10 CFR 20.1301(e) and 10 CFR 20.1302 as they relate to the monitoring of radioactivity in plant radiological effluents to unrestricted areas.

10 CFR 20.1406 as it relates to the design and operational procedures to minimize contamination, facilitate eventual decommissioning, and minimize the generation of radioactive waste as described in Chapter 12.

10 CFR Part 50, Appendix I as it relates to the numerical guidelines for design objectives and limiting conditions for operation to meet the ALARA criterion.

The PRM is designed to conform to the following regulatory guidance:

RG 1.97, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants,” as it relates to satisfying criteria for accident monitoring instrumentation in Nuclear Power Plants.

RG 1.21, “Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous and Effluents and Solid Waste,” as it relates to measuring and evaluating radioactive material in liquid and gaseous effluents and solid waste.

11.5.4 Testing and Inspection Requirements

The system design accommodates easy access to sensors, instruments, and panels for maintenance. Additionally, devices for lifting or hoisting the units are provided to facilitate replacement if required.

11.5.5 Instrumentation and Controls

Instrument modules are designed to facilitate calibration checks and troubleshooting. Accessibility for power supply adjustments is provided.

There are no control functions supporting PRM operation or performance monitoring. In most cases, equipment is provided from the manufacturer with onboard diagnostic software and displays for fault indication. Instrumentation not including these features are inspected and calibrated periodically.

Additional information about PRM instrumentation is included in Subsection 7.3.3.

Table 11.5-1: PRM Instrumentation Characteristics

Monitored Process	Location	Monitoring Type	Number of Channels	Supports Control Logic
IC Leakage	Above IC Inner Pool A	Area	1	No
	Above IC Inner Pool B	Area	1	No
	Above IC Inner Pool C	Area	1	No
LWM Discharge	Upstream of discharge to NHS	Adjacent-to-Line	3 (TMR)	Yes
OGS Pre-Treatment	Upstream of the Charcoal Adsorber Vault	Offline (Noble Gas skid)	1	No
OGS Post-Treatment	Downstream of the Charcoal Adsorber Vault	Offline (Noble Gas skid)	3 (TMR)	Yes
OGS Charcoal Adsorber Vault Ventilation	Charcoal Adsorber Vault HVAC	Adjacent-to-Line	1	No
GSC Exhaust	Upstream of the Continuous Exhaust Air Plenum	Adjacent-to-Line	1	No
Service Water Discharge	Downstream of PCW System Heat Exchangers	Adjacent-to-Line	1	No
Containment Inerting Exhaust	Upstream of the Continuous Exhaust Air Plenum	Adjacent-to-Line	1	No
RB HVS Intake	Upstream of AHU	Adjacent-to-Line	3 (TMR)	Yes
Lower RB HVS Exhaust	Upstream of AHU	Adjacent-to-Line	1	No
Refuel Floor/Fuel Pool Area HVS Exhaust	Upstream of AHU	Adjacent-to-Line	1	No
TB HVS Exhaust	Upstream of AHU	Adjacent-to-Line	1	No
PVS	At the PVS	Offline (PING skid)	4 (skid + 1 high range NG)	No
CB HVS Intake	Upstream of AHU	Adjacent-to-Line	3 (TMR)	Yes
	Downstream of AHU	Adjacent-to-Line	1	No
RWB HVS Exhaust	Upstream of AHU	Adjacent-to-Line	1	No

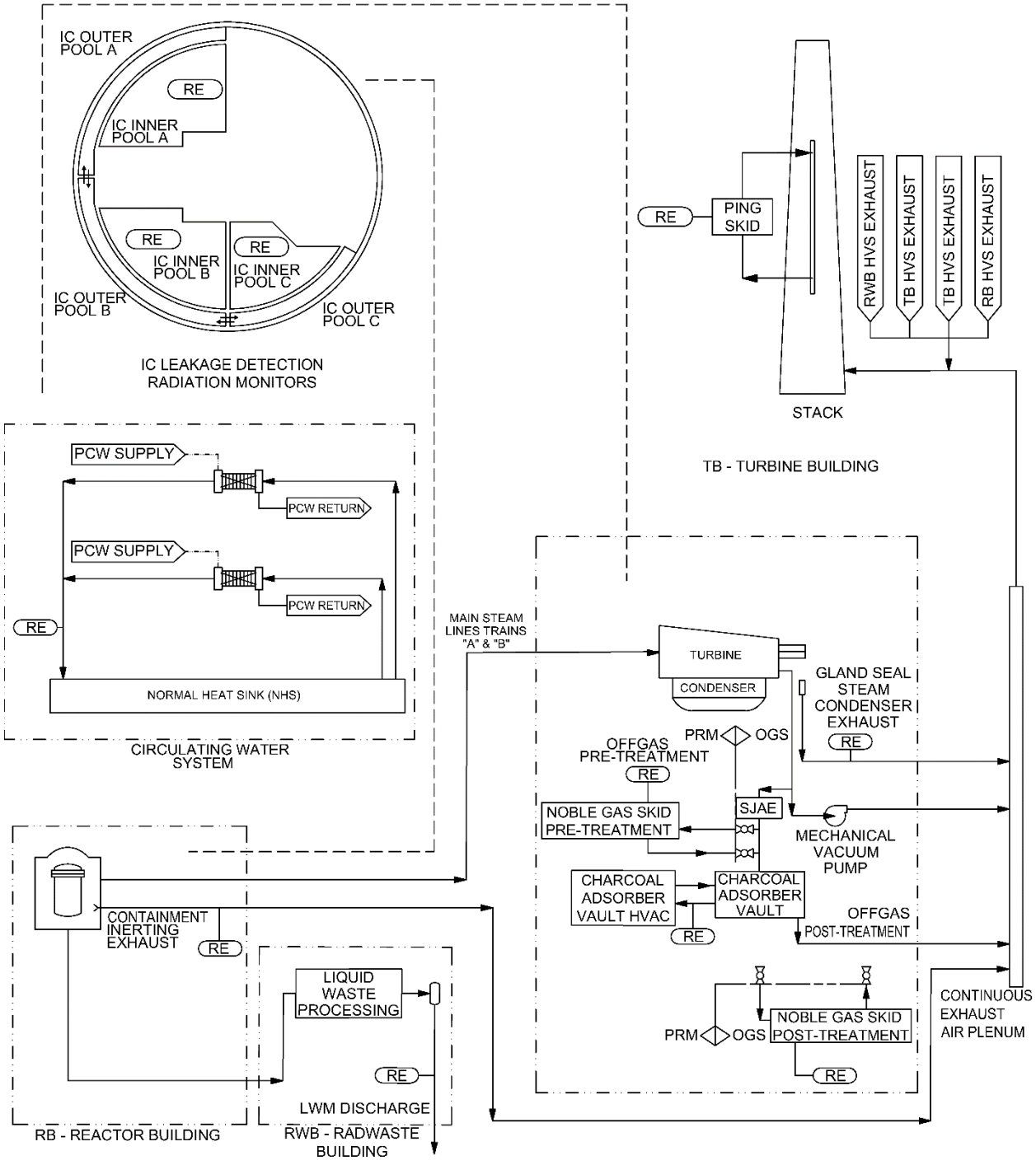


Figure 11.5-1: Process Radiation Monitoring Subsystem Simplified Diagram

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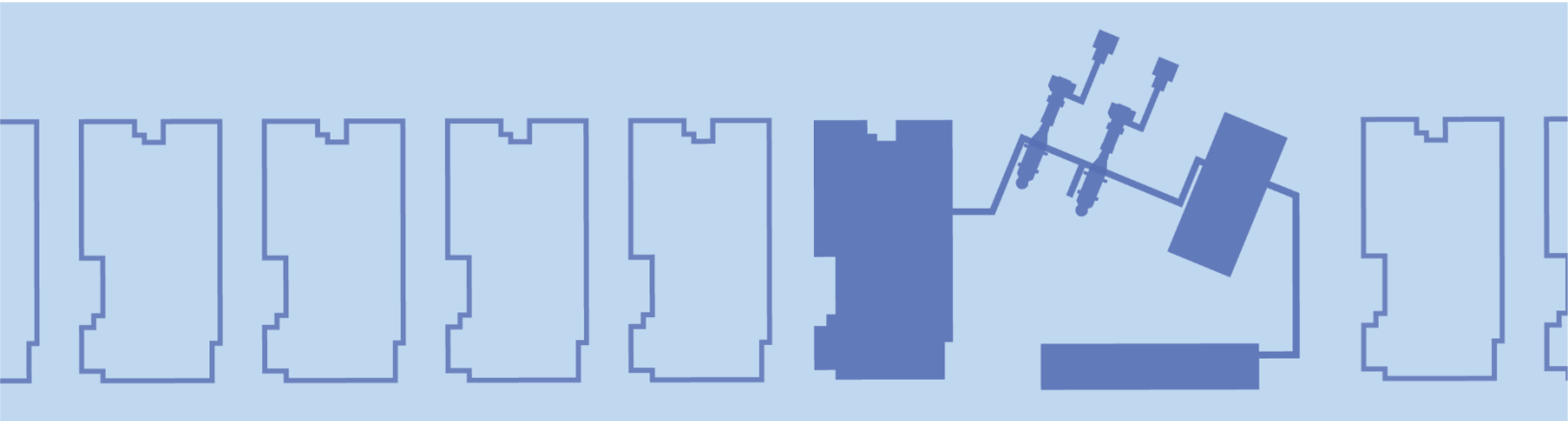
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Non-Proprietary Information

Chapter 12
Radiation Protection



ACRONYM LIST

Acronym	Explanation
ALARA	As Low As Reasonably Achievable
AOO	Anticipated Operational Occurrence
ARM	Area Radiation Monitoring Subsystem
BIS	Boron Injection System
BWR	Boiling Water Reactor
CB	Control Building
CFD	Condensate Filters and Demineralizers System
CFR	Code of Federal Regulations
CIV	Containment Isolation Valve
CMon	Containment Monitoring Subsystem
CRD	Control Rod Drive
CST	Condensate Storage Tank
CUW	Reactor Water Cleanup System
DAC	Derived Air Concentration
DAW	Dry Active Waste
DBA	Design Basis Accident
DEC	Design Extension Conditions
EFS	Equipment and Floor Drain System
FMCRD	Fine Motion Control Rod Drive
FPC	Fuel Pool Cooling and Cleanup System
GDC	General Design Criteria
GVH	GE Vernova Hitachi Nuclear Energy
GT	Gamma Thermometer
HCU	Hydraulic Control Unit
HIC	High Integrity Container
HVS	Heating, Ventilation, and Cooling System
HX	Heat Exchanger
IC	Isolation Condenser
ICC	Isolation Condenser Pools Cooling and Cleanup System
ICS	Isolation Condenser System
ISI	Inservice Inspection

Acronym	Explanation
LPRM	Local Power Range Monitor
LWM	Liquid Waste Management System
MCA	Main Condenser and Auxiliaries
MCR	Main Control Room
OGS	Offgas System
OPEX	Operating Experience
PAM	Post-Accident Monitoring
PCW	Plant Cooling Water System
PREMS	Process Radiation and Environmental Monitoring System
PRM	Process Radiation Monitoring Subsystem
PS	Process Sampling Subsystem
RB	Reactor Building
RG	Regulatory Guide
RHX	Regenerative Heat Exchanger
RIV	Reactor Isolation Valve
RPV	Reactor Pressure Vessel
RWB	Radwaste Building
RWST	Refueling Water Storage Tank
SCR	Secondary Control Room
SDC	Shutdown Cooling System
SJAE	Steam Jet Air Ejector
SSCs	Structures, Systems, and Components
SWM	Solid Waste Management System
TB	Turbine Building
NRC	United States Nuclear Regulatory Commission

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12.0 RADIATION PROTECTION

12.1 Ensuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable

Administrative programs and procedures, in conjunction with facility design, ensure that the occupational radiation exposure to personnel is kept As Low As Reasonably Achievable (ALARA).

12.1.1 Policy Considerations

The systematic application of the ALARA principle is applied during the design and construction phase of the BWRX-300 such that occupational exposure can be maintained ALARA during plant operation, maintenance, and decommissioning activities. The ALARA principles for BWRX-300 design conform with guidance in United States Nuclear Regulatory Commission (NRC) Regulatory Guide (RG) 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable," and RG 4.21, "Minimization of Contamination and Radioactive Waste Generation: Life Cycle Planning." The ALARA principles for plant personnel and operational program, including conformance with RG 8.8, RG 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable," and RG 1.8, "Qualification and Training of Personnel for Nuclear Power Plants," are described in Section 12.5. The organizations implementing these policies are described in Chapter 13.

12.1.2 Design Consideration

ALARA principles are applied during the initial design of the BWRX-300 and implemented via internal design reviews. The design is reviewed in detail for ALARA considerations and is reviewed, updated, and modified during the design phase to apply best practices and Operating Experience (OPEX). Consistent with RG 8.8 (Regulatory Position C.2), the plant design integrates features such as the layout, shielding, ventilation, lighting, instrumentation, equipment selection, material control, and radiation monitoring with traffic control, security, access control, operations, maintenance, decommissioning, and health physics aspects.

Consistent with 10 Code of Federal Regulations (CFR) 20.1101(b) and the guidance of RG 8.8, general design considerations and methods employed to maintain occupational radiation exposures ALARA have two main objectives:

- Reducing the necessity for and amount of personnel time spent in radiation areas
- Reducing radiation levels in routinely occupied plant areas in the vicinity of plant equipment that require personnel attention

Equipment and facility designs are evaluated for maintaining exposures ALARA during plant operations. Plant operations include tasks such as normal maintenance and repairs, special maintenance, refueling operations and fuel storage, Inservice Inspection (ISI), calibrations, radioactive waste handling and disposal, and equipment decommissioning activities.

BWRX-300 design criteria have been established to maintain exposures ALARA and minimize radioactive contamination and facilitate decommissioning, consistent with 10 CFR 20.1406 and RG 4.21. These design criteria are tracked from identification to validation and testing. These design criteria are based on the following ALARA objectives:

- Minimize the generation of radioactive contamination during operation
- Minimize leaks and spills of contaminated liquids and provide containment in areas where such events might occur
- Minimize the spread of contamination, and planned movement of radioactive materials

- Provide adequate shielding from direct and scattered radiation
- Minimize time spent by workers in radiation areas and in the vicinity of radiation sources
- Reduce the need to decontaminate equipment and structures by decreasing the probability of any release, and reducing any amounts released
- Design the plant to facilitate the removal or replacement of equipment or components during facility operation or decommissioning
- Provide adequate leak detection capability to provide prompt detection of leakage from any structure, system, or component that has the potential for leakage
- Use of remote operation to reduce external exposure
- Provide adequate capability for surveillance of plant system radiation levels
- Provide appropriate decontamination facilities for personnel
- Provide adequate ventilation and filtering to control airborne radioactive materials
- Facilitate decommissioning by minimizing embedded piping, sumps, or buried equipment
- Monitor and record dose to plant workers and personnel
- Control access to radiation areas and limit the time workers occupy areas with elevated radiation levels
- Control, monitor, and mitigate radiological releases to the environment

These objectives are used to develop detailed ALARA design criteria, which conform to RG 8.8 and RG 4.21. ALARA design criteria are integrated throughout the plant design. Engineers responsible for system design review and document conformance with ALARA design criteria, and each system is reviewed by engineers responsible for radiation protection.

12.1.3 Operational Considerations

As described in Section 12.5, operating procedures and techniques are developed and implemented under the Radiation Protection Program to maintain occupational radiation exposures ALARA.

12.2 Radiation Sources

This section describes the radiation sources, both contained and airborne, that provide input to the radiation protection design basis (e.g., shielding, ventilation, radwaste system design, personnel protection). At this stage of plant design, BWRX-300 source strengths are based on available information; accordingly, assumptions and/or preliminary information are used, and updates are expected for the Final Safety Analyses Report to adequately represent the source terms, including to appropriately account for daughter progeny and any other significant updates.

12.2.1 Contained Sources

Contained radiation sources are considered in radiation protection design and evaluation. Shielding design features are initially established based on Boiling Water Reactor (BWR) OPEX. Contained source term input to be applied in shielding design are described for specific contained sources below. Contained radiation source terms for process systems are determined by propagating design basis reactor water and steam concentrations described in Section 11.1 through BWRX-300 systems.

The contained radiation sources provided are shielding source terms which are used as the bases for the radiation protection design. The contained source term inputs to the shielding design are the maximum design basis activities that can be present in major plant components and radioactive waste management systems. Thus, where reactor water/coolant or reactor steam are used within this subsection, it is referring to the design basis radiation concentrations in reactor water and reactor steam, as described in Section 11.1. The contained radiation sources are determined by propagating the design basis coolant and steam concentrations through the BWRX-300 systems using conservative assumptions.

Reactor Vessel Core Sources

The reactor vessel core source is described in the manner needed as input to the shield design calculation. To determine dose rates external to shielding and within occupied areas, a reactor core source model equivalent to the neutron Watt Fission spectrum scaled to the BWRX-300 power density at full power is applied. This spectrum is representative of the energy dependent probabilities of fission-borne neutrons and neutron leakage expected in a core of this size. The effect of (n, gamma) interactions on the reactor vessel core source is considered.

Control Rod Drive System Sources

The Control Rod Drive (CRD) System is composed of three major elements: the electro-hydraulic Fine Motion Control Rod Drive (FMCRD) mechanisms, Hydraulic Control Units (HCUs), and the CRD hydraulic subsystem. The FMCRDs provide electric motor-driven positioning of control rods for normal insertion and withdrawal of the control rods and hydraulic-powered rapid insertion of control rods during abnormal operating conditions. The radioactive sources in the CRD system for the FMCRD mechanisms are shown in Table 12.2-1.

Shutdown Cooling System Sources

The Shutdown Cooling System (SDC) reduces the Reactor Pressure Vessel (RPV) pressure and temperature during shutdown and provides decay heat removal capability. The radiation contained in this system is in the SDC Heat Exchanger (HX) tube side, which contains reactor coolant. There are two SDC HXs, which are located in rooms 1451 and 1461 shown on Figure 12.3-4. The analysis assumes both SDC HXs are completely full. The source term activities are reported in Table 12.2-2.

Fuel Pool Sources

The fuel pool holds fresh and spent fuel. The BWRX-300 design utilizes GNF2 fuel bundles, and the fuel pool gamma source is calculated, as shown in Table 12.2-3.

Fuel Pool Cooling and Cleanup System Sources

The Fuel Pool Cooling and Cleanup System (FPC) provides cooling of the water volume in the fuel pool. Two FPC surge tanks receive inventory from the equipment pool, reactor cavity pool, and fuel pool, and inventory returning to the fuel pool is pumped through one of two trains from the surge tank through two HXs. Radiation concentrations in the fuel pool water are conservatively assumed to be 1% of the reactor water concentrations. The FPC surge tanks are located within the fuel pool wall as shown on Figure 12.3-8, and the FPC HXs are located in room 1556, which is shown in Figure 12.3-6. The source term activities are reported in Table 12.2-4 and Table 12.2-5.

Isolation Condenser Cooling and Cleanup System Sources

The Isolation Condenser Pools Cooling and Cleanup System (ICC) processes water from the Isolation Condenser (IC) inner pools and surrounding outer pools to maintain water temperature and plant water quality standards. The ICC includes two trains that transfer energy from the IC

pools to the Plant Cooling Water System (PCW) via a pump and HX. The water contained in the IC pools is maintained as high-purity water, and risk of contamination of IC pool water is low. The IC pool water is conservatively assumed to contain a small fraction (1E-5) of the radiation in the reactor steam. The IC pools are located in rooms 17P1 through 17P6, as shown on Figure 12.3-8, and the two HXs are located in room 1650, which is shown on Figure 12.3-7. The source term activities are reported in Table 12.2-6 and Table 12.2-7.

Reactor Water Cleanup System Sources

The Reactor Water Cleanup System (CUW) provides a blowdown-type cleanup flow for the RPV during reactor power operations and directs flow to the Condensate Filters and Demineralizers System (CFD). The radioactive source in this system is from the CUW Regenerative Heat Exchanger (RHX) tube side, which contains reactor coolant concentrations, and the CUW RHX shell side, which contains CFD activities. There is a single CUW RHX which is located in room 2180 shown on Figure 12.3-11. The analysis assumes the CUW RHX is completely full. The resultant radionuclide activity is shown in Table 12.2-8 and Table 12.2-9.

Main Condenser and Auxiliaries System Sources

The Main Condenser and Auxiliaries System (MCA) is the heat sink for the steam cycle. Two Steam Jet Air Ejectors (SJAEs) maintain turbine backpressure and remove the air inleakage and non-condensable gases from the main condenser. The radioactive sources in this system are from the two condenser shells during steady-state operation of the reactor, which contain reactor steam concentrations, the first and second stage SJAEs, and the SJAE intercondenser. The analysis assumes that the flow entering the main condenser is equal to the total steam flow leaving the reactor. The main condenser is shown on Figures 12.3-11 and 12.3-12. The SJAEs and intercondenser are located in rooms 2150 and 2160, which are shown on Figure 12.3-11. The source term activities are reported in Table 12.2-10 and Table 12.2-11.

Offgas System Sources

The Offgas System (OGS) processes non-condensable gases from the MCA that are produced through normal power operations. The main process influent to the system is a mix of steam, air, hydrogen, and radioactive noble gases from the MCA SJAE. The objective of the OGS is to process this influent prior to release to the environment to reduce the release of gaseous radionuclides to maintain the exposure of persons in unrestricted areas to radioactive effluents ALARA. The radioactive sources in the OGS are from the recombiner, which consists of a preheater section, recombiner section, and condenser section, the cooler condenser, moisture separator, two dryers, and four charcoal tanks. The recombiner, cooler condenser, moisture separator, and dryers are located in room 2181, which is shown on Figure 12.3-11. The four charcoal tanks are located in room 4100, which is shown on Figure 12.3-14. The source term activities are reported in Table 12.2-12.

Equipment and Floor Drain System Sources

The Equipment and Floor Drain System (EFS) collects liquid waste streams generated in the BWRX-300 power block and transfers these wastes to the appropriate disposal or collection system. The radioactive sources in this system are from sumps within Containment, the Reactor Building (RB), the Radwaste Building (RWB), and the Turbine Building (TB). When evaluating the source term within the Containment, RB, RWB, and TB sumps, the analysis assumes a single sump is collecting fluid within each building (and a single pressurized sump within containment), providing a conservative estimate as the inventory for a single sump is to be applied for all sumps located in the same building. The EFS Containment Sump is in room 1100A, which is shown in Figure 12.3-1. The RB, RWB, and TB sumps are located at the elevations shown on

Figure 12.3-1, Figure 12.3-14, and Figure 12.3-11. The source term activities are reported in Table 12.2-13 through Table 12.2-16.

Liquid Waste Management System Sources

The Liquid Waste Management System (LWM) reclaims, treats, and stores treated water from waste streams for use by other plant systems. The contaminated influent streams are fed to the LWM from the EFS sumps. The radioactive sources in this system are from the following components:

- Two LWM Collection Tanks - contain activities from the EFS sumps from containment, RB, TB, and RWB, and overboard flow from CUW/SDC
- LWM Sample Tanks A & B - contain activities from the LWM Collection Tanks after processing through filter skid and from overboard flow from CUW/SDC
- The LWM Condensate Storage Tank (CST) - contains activities from the LWM Sample Tanks A & B
- The LWM Refueling Water Storage Tank (RWST) - contains activities from the reactor cavity pool during pool drain down for refueling and overboard flow from CUW/SDC
- LWM Filter Skid Resin - contains activities from the two LWM collection tanks

The two collection tanks and two sample tanks are located in room 4171, which is shown on Figure 12.3-14. The CST and RWST are located in room 2183, which is shown on Figure 12.3-11. The source term activities are reported in Table 12.2-17 through Table 12.2-22.

Solid Waste Management System Sources

The Solid Waste Management System (SWM) processes, packages, and temporarily stores solid waste prior to shipment offsite. The significant contaminated influent streams are fed to the SWM from the LWM, the FPC, and the CFD. The accumulated radioactive sources in this system are from the spent resin tank, which contains activities from the LWM filter skid resin and the FPC and CFD demineralizer resins, and the two sludge tanks, which contain activities from the LWM RWST filter sludge, the FPC filter skid sludge, and the CFD filter sludge. The single spent resin tank and two sludge tanks are located in rooms 4180, 4150, and 4160, which are shown on Figure 12.3-14. The source term activities are reported in Table 12.2-23 and Table 12.2-24.

12.2.2 Airborne Radioactive Material Sources

Reactor Building

Sources of airborne activity in the RB include:

- Evaporation of fuel pool water containing low levels of radiation that becomes airborne on the operating deck. The fuel pool water does not contain noble gases, and there is no process equipment in the fuel handling area and operating deck.
- Leakage from process systems located in the lower levels of the RB. Process equipment in the RB lower levels include the SDC and the FPC. The normal flow into the SDC is single phase reactor coolant and does not include reactor steam or noble gases. Because the fuel pool contains no noble gases, there are none in the FPC. There is no other reactor coolant process equipment in the lower levels of the RB.

The RB upper-level area consisting of the fuel handling area and operating deck are provided with filtered once-through supply air from two 100% operating air handling units. The fuel handling area and operating deck are essentially one air volume extending from the containment top slab

to the RB roof. Accordingly, rooms serviced by these air handling units are considered collectively as the upper level of the RB for calculating airborne radiation activity.

The RB lower levels are provided with filtered once-through supply air from one of two operating air handling units. The airspace in the rooms on each RB lower level are connected through doors and passageways, and the RB levels in the lower RB are connected through stairwells, elevator shafts, and pipe/utility chases. Therefore, the lower levels are collectively treated as a single compartment for calculating airborne radiation activity.

The airborne concentrations for the RB upper level and the RB lower levels are compared to the Derived Air Concentration (DAC) limits established in 10 CFR Part 20, Appendix B, Table 1, Column 3. Table 12.2-25 and Table 12.2-26 provide these comparisons and demonstrate that airborne activity within the RB is under established limits.

Radwaste Building

Sources of airborne activity in the RWB include:

- As RWB tanks are filled from the top, splashing may occur, and entrained radioactivity could become airborne. This activity is not released directly into the atmosphere in the rooms; however, a small amount of leakage may occur.
- RWB pumps and valve leakage

RWB Ventilation is provided by two 50% capacity supply air handling units and two 50% capacity exhaust air handling units. Clean filtered outside air is supplied through ductwork to the RWB lobby and lab, pressurizing those areas. The exhaust air handling units take suction on the potentially contaminated tank and pump rooms, creating negative pressures in those spaces, resulting in air transferring from clean to potentially contaminated areas. The airspace in the rooms on each RWB level is connected through doors, passageways, and pipe/utility chases. Therefore, the separate levels and rooms are collectively treated as a single compartment for calculating airborne radiation activity.

The airborne concentrations for the RWB are compared to the DAC limits established in 10 CFR Part 20, Appendix B, Table 1, Column 3. Table 12.2-27 provides these comparisons and demonstrates that airborne activity within the RWB is under established limits.

Turbine Building

Sources of airborne activity in the TB include process leakage of steam and liquid in the turbine building.

The airborne concentrations for the TB are compared to the DAC limits established in 10 CFR 20, Appendix B, Table 1, Column 3. Table 12.2-28 provides this comparison and demonstrates that airborne activity within the TB is under established limits.

Table 12.2-1: Radioactive Sources in the Control Rod Drive System

Control Rod Drive Radiation Survey Data		
	Gamma Dose Rates Measured at Contact, mSv/hr (mrem/hr)**	
Component	**Before Cleaning**	**After Cleaning**
Rotating Ball Spindle	Not measured	2.0E-01 (2.0E+01)
Hollow Piston	5.0E-01 (5.0E+01)	2.5E-01 (2.5E+01)
Labyrinth Seal	4.0E-01 (4.0E+01)	4.0E-01 (4.0E+01)
Guide Tube	3.0E-01 (3.0E+01)	2.0E-01 (2.0E+01)
Outer Tube/Flange	2.2E+00 (2.2E+02)	2.0E-01 (2.0E+01)

Control Blade Principal Isotopes	
Isotopes	**MBq/Blade (Ci/Blade)**
Cr-51	5.2E+09 (1.4E+05)
Mn-54	3.4E+08 (9.1E+03)
Fe-55	5.9E+09 (1.6E+05)
Co-58m	2.8E+08 (7.7E+03)
Co-58	3.3E+08 (8.8E+03)
Co-60	4.1E+09 (1.1E+05)
Ni-63	1.9E+08 (5.0E+03)

**The gamma dose rates listed in this table are based on survey measurements taken for the FMCRD disassembly at a nuclear plant in 1988. No measurements were taken for the rotating balls spindle before cleaning.

Table 12.2-2: SDC Heat Exchanger Tube Side Activity

Isotope	Activity (MBq)
Class 2	
I-131	2.45E+02
I-132	2.45E+03
I-133	1.83E+03
I-134	6.95E+03
I-135	3.51E+03
Class 3	
Rb-89	5.12E+03
Cs-134	7.32E+01
Cs-136	5.86E+01
Cs-137	1.13E+02
Cs-138	5.49E+03
Ba-137m	1.13E+02
Class 4	
N-16	8.05E+05
Class 5	
H-3	3.66E+02
Class 6	
Na-24	1.10E+01
P-32	4.39E-01
Cr-51	1.10E+01
Mn-54	5.49E+00
Mn-56	2.09E+01
Fe-55	1.13E+01
Fe-59	2.93E+00
Co-58	2.60E+00
Co-60	5.12E+00
Ni-63	1.13E-02
Cu-64	5.49E+01
Zn-65	2.38E+00
Sr-89	8.42E+00
Sr-90	4.03E-01
Y-90	4.03E-01

Isotope	Activity (MBq)
Class 6 (continued)	
Sr-91	5.12E+03
Sr-92	1.10E+04
Y-91	1.10E+02
Y-92	3.33E+03
Y-93	3.62E+02
Zr-95	2.27E+02
Nb-95	2.27E+02
Mo-99	1.06E+03
Tc-99m	1.06E+03
Ru-103	5.49E+01
Rh-103m	5.49E+01
Ru-106	8.42E+00
Rh-106	8.42E+00
Ag-110m	1.13E-02
Te-129m	1.10E+02
Te-131m	1.43E+02
Te-132	2.64E+01
Ba-140	1.13E+03
La-140	1.13E+03
Ce-141	5.49E+01
Ce-144	8.42E+00
Pr-144	8.42E+00
W-187	2.56E+00
Np-239	8.78E+02

Table 12.2-3: Fuel Pool Gamma Source[1]

Photon Spectrum as a Function of Decay Time (photons/s per bundle)						
Upper Energy (MeV)	**0.0 Seconds**	**1.0 Second**	**30.0 Minutes**	**1.0 Hour**	**8.0 Hours**	**1.0 Day**
2.00E-02	3.50E+17	3.46E+17	1.92E+17	1.67E+17	1.24E+17	9.82E+16
3.00E-02	6.30E+16	6.21E+16	2.88E+16	2.41E+16	1.55E+16	1.18E+16
4.50E-02	5.47E+16	5.41E+16	2.75E+16	2.32E+16	1.66E+16	1.36E+16
7.00E-02	5.49E+16	5.40E+16	2.32E+16	1.94E+16	1.28E+16	9.94E+15
1.00E-01	9.31E+16	9.26E+16	4.86E+16	3.60E+16	2.30E+16	1.88E+16
1.50E-01	8.35E+16	8.29E+16	6.66E+16	6.37E+16	5.53E+16	4.61E+16
3.00E-01	1.12E+17	1.10E+17	6.36E+16	5.74E+16	4.55E+16	3.55E+16
4.50E-01	6.34E+16	6.29E+16	2.85E+16	2.06E+16	1.18E+16	9.58E+15
7.00E-01	1.01E+17	1.01E+17	6.46E+16	5.77E+16	4.06E+16	3.10E+16
1.00E+00	1.03E+17	1.02E+17	6.55E+16	5.62E+16	3.25E+16	2.65E+16
1.50E+00	5.53E+16	5.49E+16	2.83E+16	2.19E+16	8.58E+15	4.87E+15
2.00E+00	2.07E+16	2.06E+16	1.31E+16	1.12E+16	7.22E+15	6.13E+15
2.50E+00	1.00E+16	9.97E+15	5.38E+15	3.90E+15	8.00E+14	4.68E+14
3.00E+00	4.25E+15	4.23E+15	2.16E+15	1.57E+15	2.95E+14	2.28E+14
4.00E+00	1.97E+15	1.95E+15	5.58E+14	3.88E+14	1.73E+13	1.93E+12
6.00E+00	7.74E+14	7.56E+14	5.18E+12	3.43E+12	4.55E+11	9.26E+09
8.00E+00	8.10E+12	7.21E+12	3.77E+06	3.77E+06	3.77E+06	3.77E+06
1.10E+01	2.40E+09	1.97E+09	4.34E+05	4.34E+05	4.34E+05	4.34E+05
Total	1.17E+18	1.16E+18	6.59E+17	5.64E+17	3.95E+17	3.13E+17
MeV/s	3.74E+17	3.71E+17	2.14E+17	1.78E+17	1.04E+17	8.02E+16

1 A high burnup (58 GWd/MTU) is assumed.

Table 12.2-3: Fuel Pool Gamma Source[1] (continued)

Photon Spectrum as a Function of Decay Time (photons/s per bundle)						
Upper Energy (MeV)	**4.0 Days**	**30.0 Days**	**90.0 Days**	**180.0 Days**	**1.0 Year**	**3.0 Years**
2.00E-02	4.96E+16	1.13E+16	7.71E+15	5.90E+15	4.01E+15	1.41E+15
3.00E-02	7.42E+15	2.88E+15	1.81E+15	1.35E+15	9.18E+14	3.29E+14
4.50E-02	8.59E+15	2.96E+15	1.77E+15	1.31E+15	9.10E+14	3.49E+14
7.00E-02	6.21E+15	2.29E+15	1.57E+15	1.22E+15	8.27E+14	2.81E+14
1.00E-01	1.01E+16	1.68E+15	1.07E+15	8.38E+14	5.77E+14	1.94E+14
1.50E-01	2.24E+16	3.23E+15	1.62E+15	9.91E+14	6.35E+14	2.07E+14
3.00E-01	1.54E+16	1.46E+15	9.43E+14	7.37E+14	5.03E+14	1.61E+14
4.50E-01	6.14E+15	1.33E+15	5.28E+14	4.05E+14	2.83E+14	9.57E+13
7.00E-01	1.98E+16	1.01E+16	6.60E+15	5.14E+15	4.13E+15	2.42E+15
1.00E+00	1.95E+16	1.18E+16	7.52E+15	4.22E+15	1.99E+15	8.48E+14
1.50E+00	2.73E+15	9.76E+14	6.31E+14	5.52E+14	4.67E+14	2.96E+14
2.00E+00	5.10E+15	1.28E+15	7.85E+13	2.37E+13	1.59E+13	5.00E+12
2.50E+00	3.62E+14	1.21E+14	3.81E+13	2.68E+13	1.72E+13	3.14E+12
3.00E+00	1.88E+14	4.67E+13	2.49E+12	6.14E+11	4.22E+11	1.06E+11
4.00E+00	1.63E+12	4.77E+11	1.04E+11	7.59E+10	5.35E+10	1.36E+10
6.00E+00	3.54E+07	3.19E+07	3.05E+07	2.89E+07	2.68E+07	2.37E+07
8.00E+00	3.76E+06	3.68E+06	3.52E+06	3.33E+06	3.10E+06	2.73E+06
1.10E+01	4.32E+05	4.23E+05	4.05E+05	3.83E+05	3.56E+05	3.14E+05
Total	1.74E+17	5.15E+16	3.19E+16	2.27E+16	1.53E+16	6.59E+15
MeV/s	5.24E+16	2.15E+16	1.22E+16	8.06E+15	5.21E+15	2.66E+15

1 A high burnup (58 GWd/MTU) is assumed.

Table 12.2-4: FPC Surge Tank Activity Concentrations[1]

Isotope	Concentration (Bq/g)	Concentration (µCi/g)
Class 2		
I-131	6.70E+00	1.81E-04
I-132	6.70E+01	1.81E-03
I-133	5.00E+01	1.35E-03
I-134	1.90E+02	5.14E-03
I-135	9.60E+01	2.59E-03
Class 3		
Rb-89	1.40E+02	3.78E-03
Cs-134	2.00E+00	5.41E-05
Cs-136	1.60E+00	4.32E-05
Cs-137	3.10E+00	8.38E-05
Cs-138	1.50E+02	4.05E-03
Ba-137m	3.10E+00	8.38E-05
Class 5		
H-3	1.00E+01	2.70E-04
Class 6		
Na-24	3.00E-01	8.11E-06
P-32	1.20E-02	3.24E-07
Cr-51	3.00E-01	8.11E-06
Mn-54	1.50E-01	4.05E-06
Mn-56	5.70E-01	1.54E-05
Fe-55	3.10E-01	8.38E-06
Fe-59	8.00E-02	2.16E-06
Co-58	7.10E-02	1.92E-06
Co-60	1.40E-01	3.78E-06
Ni-63	3.10E-04	8.38E-09
Cu-64	1.50E+00	4.05E-05
Zn-65	6.50E-02	1.76E-06
Sr-89	2.30E-01	6.22E-06
Sr-90	1.10E-02	2.97E-07
Y-90	1.10E-02	2.97E-07
Sr-91	1.40E+02	3.78E-03
Sr-92	3.00E+02	8.11E-03
Y-91	3.00E+00	8.11E-05
Y-92	9.10E+01	2.46E-03

Isotope	Concentration (Bq/g)	Concentration (µCi/g)
Class 6 (continued)		
Y-93	9.90E+00	2.68E-04
Zr-95	6.20E+00	1.68E-04
Nb-95	6.20E+00	1.68E-04
Mo-99	2.90E+01	7.84E-04
Tc-99m	2.90E+01	7.84E-04
Ru-103	1.50E+00	4.05E-05
Rh-103m	1.50E+00	4.05E-05
Ru-106	2.30E-01	6.22E-06
Rh-106	2.30E-01	6.22E-06
Ag-110m	3.10E-04	8.38E-09
Te-129m	3.00E+00	8.11E-05
Te-131m	3.90E+00	1.05E-04
Te-132	7.20E-01	1.95E-05
Ba-140	3.10E+01	8.38E-04
La-140	3.10E+01	8.38E-04
Ce-141	1.50E+00	4.05E-05
Ce-144	2.30E-01	6.22E-06
Pr-144	2.30E-01	6.22E-06
W-187	7.00E-02	1.89E-06
Np-239	2.40E+01	6.49E-04

1 Contained source in each of the two surge tanks is the same. Results show the activity per surge tank.

Table 12.2-5: FPC Heat Exchanger Activity[1]

Isotope	Activity (Bq)	Activity (µCi)
Class 2		
I-131	3.99E+06	1.08E+02
I-132	3.99E+07	1.08E+03
I-133	2.98E+07	8.04E+02
I-134	1.13E+08	3.06E+03
I-135	5.71E+07	1.54E+03
Class 3		
Rb-89	4.17E+08	1.13E+04
Cs-134	5.95E+06	1.61E+02
Cs-136	4.76E+06	1.29E+02
Cs-137	9.23E+06	2.49E+02
Cs-138	4.46E+08	1.21E+04
Ba-137m	9.23E+06	2.49E+02
Class 5		
H-3	2.98E+07	8.04E+02
Class 6		
Na-24	3.57E+04	9.65E-01
P-32	1.43E+03	3.86E-02
Cr-51	3.57E+04	9.65E-01
Mn-54	1.79E+04	4.83E-01
Mn-56	6.79E+04	1.83E+00
Fe-55	3.69E+04	9.97E-01
Fe-59	9.52E+03	2.57E-01
Co-58	8.45E+03	2.28E-01
Co-60	1.67E+04	4.50E-01
Ni-63	3.69E+01	9.97E-04
Cu-64	1.79E+05	4.83E+00
Zn-65	7.74E+03	2.09E-01
Sr-89	2.74E+04	7.40E-01
Sr-90	1.31E+03	3.54E-02
Y-90	1.31E+03	3.54E-02
Sr-91	1.67E+07	4.50E+02
Sr-92	3.57E+07	9.65E+02
Y-91	3.57E+05	9.65E+00

Isotope	Activity (Bq)	Activity (µCi)
Class 6 (continued)		
Y-92	1.08E+07	2.93E+02
Y-93	1.18E+06	3.19E+01
Zr-95	7.38E+05	1.99E+01
Nb-95	7.38E+05	1.99E+01
Mo-99	3.45E+06	9.33E+01
Tc-99m	3.45E+06	9.33E+01
Ru-103	1.79E+05	4.83E+00
Rh-103m	1.79E+05	4.83E+00
Ru-106	2.74E+04	7.40E-01
Rh-106	2.74E+04	7.40E-01
Ag-110m	3.69E+01	9.97E-04
Te-129m	3.57E+05	9.65E+00
Te-131m	4.64E+05	1.25E+01
Te-132	8.57E+04	2.32E+00
Ba-140	3.69E+06	9.97E+01
La-140	3.69E+06	9.97E+01
Ce-141	1.79E+05	4.83E+00
Ce-144	2.74E+04	7.40E-01
Pr-144	2.74E+04	7.40E-01
W-187	8.33E+03	2.25E-01
Np-239	2.86E+06	7.72E+01

1 Each train in FPC has two HXs (four total), the activity per HX is shown.

Table 12.2-6: Isolation Condenser Pool Activity Concentrations

Isotope	Concentration (Bq/g)	Concentration (µCi/g)
Class 2		
I-131	1.40E-04	3.78E-09
I-132	1.30E-03	3.51E-08
I-133	1.00E-03	2.70E-08
I-134	3.90E-03	1.05E-07
I-135	1.80E-03	4.86E-08
Class 3		
Rb-89	1.40E-04	3.78E-09
Cs-134	2.00E-06	5.41E-11
Cs-136	1.60E-06	4.32E-11
Cs-137	3.10E-06	8.38E-11
Cs-138	1.50E-04	4.05E-09
Ba-137m	3.10E-06	8.38E-11
Class 5		
H-3	1.00E-02	2.70E-07
Class 6		
Na-24	3.00E-07	8.11E-12
P-32	1.20E-08	3.24E-13
Cr-51	3.00E-07	8.11E-12
Mn-54	1.50E-07	4.05E-12
Mn-56	5.70E-07	1.54E-11
Fe-55	3.10E-07	8.38E-12
Fe-59	8.00E-08	2.16E-12
Co-58	7.10E-08	1.92E-12
Co-60	1.40E-07	3.78E-12
Ni-63	3.10E-10	8.38E-15
Cu-64	1.50E-06	4.05E-11
Zn-65	6.50E-08	1.76E-12
Sr-89	2.30E-07	6.22E-12
Sr-90	1.10E-08	2.97E-13
Y-90	1.10E-08	2.97E-13
Sr-91	1.40E-04	3.78E-09
Sr-92	3.00E-04	8.11E-09
Y-91	3.00E-06	8.11E-11
Y-92	9.10E-05	2.46E-09

Isotope	Concentration (Bq/g)	Concentration (µCi/g)
Class 6 (continued)		
Y-93	9.90E-06	2.68E-10
Zr-95	6.20E-06	1.68E-10
Nb-95	6.20E-06	1.68E-10
Mo-99	2.90E-05	7.84E-10
Tc-99m	2.90E-05	7.84E-10
Ru-103	1.50E-06	4.05E-11
Rh-103m	1.50E-06	4.05E-11
Ru-106	2.30E-07	6.22E-12
Rh-106	2.30E-07	6.22E-12
Ag-110m	3.10E-10	8.38E-15
Te-129m	3.00E-06	8.11E-11
Te-131m	3.90E-06	1.05E-10
Te-132	7.20E-07	1.95E-11
Ba-140	3.10E-05	8.38E-10
La-140	3.10E-05	8.38E-10
Ce-141	1.50E-06	4.05E-11
Ce-144	2.30E-07	6.22E-12
Pr-144	2.30E-07	6.22E-12
W-187	7.00E-08	1.89E-12
Np-239	2.40E-05	6.49E-10

Table 12.2-7: ICC Heat Exchanger Activity

Isotope	Activity (Bq)	Activity (µCi)
Class 2		
I-131	1.79E+02	4.83E-03
I-132	1.66E+03	4.49E-02
I-133	1.28E+03	3.45E-02
I-134	4.98E+03	1.35E-01
I-135	2.30E+03	6.21E-02
Class 3		
Rb-89	1.79E+02	4.83E-03
Cs-134	2.55E+00	6.90E-05
Cs-136	2.04E+00	5.52E-05
Cs-137	3.96E+00	1.07E-04
Cs-138	1.91E+02	5.18E-03
Ba-137m	3.96E+00	1.07E-04
Class 5		
H-3	1.28E+04	3.45E-01
Class 6		
Na-24	3.83E-01	1.04E-05
P-32	1.53E-02	4.14E-07
Cr-51	3.83E-01	1.04E-05
Mn-54	1.91E-01	5.18E-06
Mn-56	7.28E-01	1.97E-05
Fe-55	3.96E-01	1.07E-05
Fe-59	1.02E-01	2.76E-06
Co-58	9.06E-02	2.45E-06
Co-60	1.79E-01	4.83E-06
Ni-63	3.96E-04	1.07E-08
Cu-64	1.91E+00	5.18E-05
Zn-65	8.30E-02	2.24E-06
Sr-89	2.94E-01	7.94E-06
Sr-90	1.40E-02	3.80E-07
Y-90	1.40E-02	3.80E-07
Sr-91	1.79E+02	4.83E-03
Sr-92	3.83E+02	1.04E-02
Y-91	3.83E+00	1.04E-04

Isotope	Activity (Bq)	Activity (µCi)
Class 6 (continued)		
Y-92	1.16E+02	3.14E-03
Y-93	1.26E+01	3.42E-04
Zr-95	7.92E+00	2.14E-04
Nb-95	7.92E+00	2.14E-04
Mo-99	3.70E+01	1.00E-03
Tc-99m	3.70E+01	1.00E-03
Ru-103	1.91E+00	5.18E-05
Rh-103m	1.91E+00	5.18E-05
Ru-106	2.94E-01	7.94E-06
Rh-106	2.94E-01	7.94E-06
Ag-110m	3.96E-04	1.07E-08
Te-129m	3.83E+00	1.04E-04
Te-131m	4.98E+00	1.35E-04
Te-132	9.19E-01	2.48E-05
Ba-140	3.96E+01	1.07E-03
La-140	3.96E+01	1.07E-03
Ce-141	1.91E+00	5.18E-05
Ce-144	2.94E-01	7.94E-06
Pr-144	2.94E-01	7.94E-06
W-187	8.94E-02	2.42E-06
Np-239	3.06E+01	8.28E-04

Table 12.2-8: CUW Regenerative Heat Exchanger Tube Side Activity

Isotope	Activity (MBq)
Class 2	
I-131	1.59E+02
I-132	1.59E+03
I-133	1.18E+03
I-134	4.50E+03
I-135	2.27E+03
Class 3	
Rb-89	3.31E+03
Cs-134	4.73E+01
Cs-136	3.79E+01
Cs-137	7.34E+01
Cs-138	3.55E+03
Ba-137m	7.34E+01
Class 4	
N-16	5.21E+05
Class 5	
H-3	2.37E+02
Class 6	
Na-24	7.10E+00
P-32	2.84E-01
Cr-51	7.10E+00
Mn-54	3.55E+00
Mn-56	1.35E+01
Fe-55	7.34E+00
Fe-59	1.89E+00
Co-58	1.68E+00
Co-60	3.31E+00
Ni-63	7.34E-03
Cu-64	3.55E+01
Zn-65	1.54E+00
Sr-89	5.44E+00
Sr-90	2.60E-01
Y-90	2.60E-01

Isotope	Activity (MBq)
Class 6 (continued)	
Sr-91	3.31E+03
Sr-92	7.10E+03
Y-91	7.10E+01
Y-92	2.15E+03
Y-93	2.34E+02
Zr-95	1.47E+02
Nb-95	1.47E+02
Mo-99	6.86E+02
Tc-99m	6.86E+02
Ru-103	3.55E+01
Rh-103m	3.55E+01
Ru-106	5.44E+00
Rh-106	5.44E+00
Ag-110m	7.34E-03
Te-129m	7.10E+01
Te-131m	9.23E+01
Te-132	1.70E+01
Ba-140	7.34E+02
La-140	7.34E+02
Ce-141	3.55E+01
Ce-144	5.44E+00
Pr-144	5.44E+00
W-187	1.66E+00
Np-239	5.68E+02

Table 12.2-9: CUW Regenerative Heat Exchanger Shell Side Activity

Isotope	Activity (MBq)
Class 2	
I-131	4.23E+01
I-132	4.23E+02
I-133	3.16E+02
I-134	1.20E+03
I-135	6.06E+02
Class 3	
Rb-89	4.42E+03
Cs-134	6.31E+01
Cs-136	5.05E+01
Cs-137	9.78E+01
Cs-138	4.73E+03
Ba-137m	9.78E+01
Class 4	
N-16	1.39E+06
Class 5	
H-3	6.31E+02
Class 6	
Na-24	1.89E+00
P-32	7.57E-02
Cr-51	1.89E+00
Mn-54	9.47E-01
Mn-56	3.60E+00
Fe-55	1.96E+00
Fe-59	5.05E-01
Co-58	4.48E-01
Co-60	8.84E-01
Ni-63	1.96E-03
Cu-64	9.47E+00
Zn-65	4.10E-01
Sr-89	1.45E+00
Sr-90	6.94E-02
Y-90	6.94E-02

Isotope	Activity (MBq)
Class 6 (continued)	
Sr-91	8.84E+02
Sr-92	1.89E+03
Y-91	1.89E+01
Y-92	5.74E+02
Y-93	6.25E+01
Zr-95	3.91E+01
Nb-95	3.91E+01
Mo-99	1.83E+02
Tc-99m	1.83E+02
Ru-103	9.47E+00
Rh-103m	9.47E+00
Ru-106	1.45E+00
Rh-106	1.45E+00
Ag-110m	1.96E-03
Te-129m	1.89E+01
Te-131m	2.46E+01
Te-132	4.54E+00
Ba-140	1.96E+02
La-140	1.96E+02
Ce-141	9.47E+00
Ce-144	1.45E+00
Pr-144	1.45E+00
W-187	4.42E-01
Np-239	1.51E+02

Table 12.2-10: Main Condenser Activity

Isotope	Activity (MBq)
Kr-83m	2.94E+04
Kr-85m	7.83E+02
Kr-85	1.99E+02
Kr-87	5.45E+03
Kr-88	3.07E+03
Kr-89	8.30E+05
Xe-131m	1.63E+02
Xe-133m	1.03E+02
Xe-133	1.51E+03
Xe-135m	2.60E+04
Xe-135	1.51E+04
Xe-137	4.15E+04
Xe-138	1.32E+05
I-131	8.45E+02
I-132	7.81E+03
I-133	6.04E+03
I-134	2.32E+04
I-135	1.09E+04
Rb-89	8.08E+02
Cs-134	1.21E+01
Cs-136	9.66E+00
Cs-137	1.87E+01
Cs-138	8.87E+02
Ba-137m	1.44E+01
N-16	2.38E+07
H-3	6.04E+04
Na-24	1.81E+00
P-32	7.25E-02
Cr-51	1.81E+00
Mn-54	9.06E-01
Mn-56	3.43E+00
Fe-55	1.87E+00
Fe-59	4.83E-01

Isotope	Activity (MBq)
Co-58	4.29E-01
Co-60	8.45E-01
Ni-63	1.87E-03
Cu-64	9.05E+00
Zn-65	3.92E-01
Sr-89	1.39E+00
Sr-90	6.64E-02
Y-90	6.64E-02
Sr-91	8.44E+02
Sr-92	1.80E+03
Y-91	1.81E+01
Y-92	5.48E+02
Y-93	5.97E+01
Zr-95	3.74E+01
Nb-95	3.74E+01
Mo-99	1.75E+02
Tc-99m	1.75E+02
Ru-103	9.06E+00
Rh-103m	8.95E+00
Ru-106	1.39E+00
Rh-106	4.67E-01
Ag-110m	1.87E-03
Te-129m	1.81E+01
Te-131m	2.35E+01
Te-132	4.35E+00
Ba-140	1.87E+02
La-140	1.87E+02
Ce-141	9.06E+00
Ce-144	1.39E+00
Pr-144	1.33E+00
W-187	4.22E-01
Np-239	1.45E+02

Table 12.2-11: SJAE Component Activity

Nuclide	First Stage Ejector (MBq)	Intercondenser (MBq)	Second Stage Ejector (MBq)
Class 1			
Kr-83m	1.8E+00	5.5E+01	8.8E-01
Kr-85m	4.8E-02	1.5E+00	2.3E-02
Kr-85	1.2E-02	3.7E-01	5.9E-03
Kr-87	3.4E-01	1.0E+01	1.6E-01
Kr-88	1.9E-01	5.7E+00	9.1E-02
Kr-89	6.3E+01	1.9E+03	3.0E+01
Xe-131m	1.0E-02	3.0E-01	4.8E-03
Xe-133m	6.3E-03	1.9E-01	3.0E-03
Xe-133	9.3E-02	2.8E+00	4.5E-02
Xe-135m	1.7E+00	5.0E+01	8.0E-01
Xe-135	9.3E-01	2.8E+01	4.5E-01
Xe-137	3.1E+00	9.2E+01	1.5E+00
Xe-138	8.6E+00	2.6E+02	4.1E+00
Class 2			
I-131	2.2E-01	6.5E+00	8.9E-02
I-132	2.0E+00	6.1E+01	8.3E-01
I-133	1.6E+00	4.7E+01	6.4E-01
I-134	6.1E+00	1.8E+02	2.5E+00
I-135	2.8E+00	8.4E+01	1.1E+00
Class 3			
Rb-89	2.2E-01	6.5E+00	8.9E-02
Cs-134	3.1E-03	9.4E-02	1.3E-03
Cs-136	2.5E-03	7.5E-02	1.0E-03
Cs-137	4.8E-03	1.4E-01	2.0E-03
Cs-138	2.3E-01	7.0E+00	9.5E-02
Ba-137m	4.8E-03	1.4E-01	2.0E-03
Class 4			
N-16	3.3E+00	2.6E+04	4.5E-01
Class 5			
H-3	1.6E+01	4.7E+02	6.4E+00

Nuclide	First Stage Ejector (MBq)	Intercondenser (MBq)	Second Stage Ejector (MBq)
Class 6			
Na-24	4.7E-04	1.4E-02	1.9E-04
P-32	1.9E-05	5.6E-04	7.6E-06
Cr-51	4.7E-04	1.4E-02	1.9E-04
Mn-54	2.3E-04	7.0E-03	9.5E-05
Mn-56	8.9E-04	2.7E-02	3.6E-04
Fe-55	4.8E-04	1.4E-02	2.0E-04
Fe-59	1.2E-04	3.7E-03	5.1E-05
Co-58	1.1E-04	3.3E-03	4.5E-05
Co-60	2.2E-04	6.5E-03	8.9E-05
Ni-63	4.8E-07	1.4E-05	2.0E-07
Cu-64	2.3E-03	7.0E-02	9.5E-04
Zn-65	1.0E-04	3.0E-03	4.1E-05
Sr-89	3.6E-04	1.1E-02	1.5E-04
Sr-90	1.7E-05	5.1E-04	7.0E-06
Y-90	1.7E-05	5.1E-04	7.0E-06
Sr-91	2.2E-01	6.5E+00	8.9E-02
Sr-92	4.7E-01	1.4E+01	1.9E-01
Y-91	4.7E-03	1.4E-01	1.9E-03
Y-92	1.4E-01	4.3E+00	5.8E-02
Y-93	1.5E-02	4.6E-01	6.3E-03
Zr-95	9.7E-03	2.9E-01	3.9E-03
Nb-95	9.7E-03	2.9E-01	3.9E-03
Mo-99	4.5E-02	1.4E+00	1.8E-02
Tc-99m	4.5E-02	1.4E+00	1.8E-02
Ru-103	2.3E-03	7.0E-02	9.5E-04
Rh-103m	2.3E-03	7.0E-02	9.5E-04
Ru-106	3.6E-04	1.1E-02	1.5E-04
Rh-106	3.6E-04	1.1E-02	1.5E-04
Ag-110m	4.8E-07	1.4E-05	2.0E-07
Te-129m	4.7E-03	1.4E-01	1.9E-03
Te-131m	6.1E-03	1.8E-01	2.5E-03
Te-132	1.1E-03	3.4E-02	4.6E-04

Nuclide	First Stage Ejector (MBq)	Intercondenser (MBq)	Second Stage Ejector (MBq)
Class 6 (continued)			
Ba-140	4.8E-02	1.4E+00	2.0E-02
La-140	4.8E-02	1.4E+00	2.0E-02
Ce-141	2.3E-03	7.0E-02	9.5E-04
Ce-144	3.6E-04	1.1E-02	1.5E-04
Pr-144	3.6E-04	1.1E-02	1.5E-04
W-187	1.1E-04	3.3E-03	4.4E-05
Np-239	3.7E-02	1.1E+00	1.5E-02

Table 12.2-12: OGS Normal Operation Equipment Activity

Nuclide	Preheater (MBq)	Recombiner (MBq)	Condenser (MBq)	Cooler Condenser (MBq)	Moisture Separator (MBq)	Dryer (MBq)	Guard Tank (MBq)	Tank 1 (MBq)	Tank 2 (MBq)	Tank 3 (MBq)
Class 1										
Kr-83m	1.5E-01	5.3E-01	6.8E+01	2.2E+00	1.5E+00	1.1E+02	2.3E+03	3.1E-01	4.1E-05	5.5E-09
Kr-85m	4.0E-03	1.4E-02	1.8E+00	6.0E-02	3.9E-02	3.0E+00	1.6E+02	4.9E+00	1.5E-01	4.6E-03
Kr-85	1.0E-03	3.5E-03	4.6E-01	1.6E-02	1.0E-02	7.8E-01	1.5E+02	1.5E+02	1.5E+02	1.5E+02
Kr-87	2.8E-02	9.8E-02	1.3E+01	4.1E-01	2.7E-01	2.0E+01	3.1E+02	1.6E-03	8.3E-09	4.3E-14
Kr-88	1.6E-02	5.5E-02	7.1E+00	2.4E-01	1.5E-01	1.2E+01	4.0E+02	1.6E+00	6.5E-03	2.6E-05
Kr-89	5.2E+00	1.8E+01	1.7E+03	3.4E+01	2.1E+01	8.4E+02	2.3E+02	8.6E-129	3.2E-259	0.0E+00
Xe-131m	8.3E-04	2.9E-03	3.8E-01	1.3E-02	8.2E-03	6.4E-01	1.4E+03	5.3E+02	2.0E+02	7.3E+01
Xe-133m	5.2E-04	1.8E-03	2.4E-01	8.0E-03	5.2E-03	4.0E-01	2.6E+02	1.2E+00	5.7E-03	2.7E-05
Xe-133	7.7E-03	2.7E-02	3.5E+00	1.2E-01	7.6E-02	5.9E+00	8.4E+03	9.5E+02	1.1E+02	1.2E+01
Xe-135m	1.4E-01	4.8E-01	5.8E+01	1.8E+00	1.1E+00	7.6E+01	2.0E+02	0.0E+00	0.0E+00	0.0E+00
Xe-135	7.7E-02	2.7E-01	3.5E+01	1.2E+00	7.6E-01	5.9E+01	6.6E+03	4.2E-10	2.6E-23	1.6E-36
Xe-137	2.5E-01	8.8E-01	8.6E+01	1.9E+00	1.2E+00	5.4E+01	2.1E+01	0.0E+00	0.0E+00	0.0E+00
Xe-138	7.1E-01	2.5E+00	3.0E+02	8.9E+00	5.7E+00	3.8E+02	9.2E+02	0.0E+00	0.0E+00	0.0E+00
Class 2										
I-131	1.4E-02	5.4E-02	6.5E+00	6.6E-04	4.3E-04	4.2E-03	4.1E-03	4.1E-03	4.1E-03	4.1E-03

Nuclide	Preheater (MBq)	Recombiner (MBq)	Condenser (MBq)	Cooler Condenser (MBq)	Moisture Separator (MBq)	Dryer (MBq)	Guard Tank (MBq)	Tank 1 (MBq)	Tank 2 (MBq)	Tank 3 (MBq)
I-132	1.3E-01	5.0E-01	6.0E+01	6.0E-03	3.9E-03	3.8E-02	3.5E-02	3.2E-02	2.9E-02	2.7E-02
I-133	1.0E-01	3.8E-01	4.7E+01	4.7E-03	3.1E-03	3.0E-02	2.9E-02	2.9E-02	2.9E-02	2.8E-02
I-134	4.0E-01	1.5E+00	1.8E+02	1.8E-02	1.1E-02	1.1E-01	8.9E-02	7.1E-02	5.7E-02	4.5E-02
I-135	1.9E-01	6.9E-01	8.4E+01	8.5E-03	5.5E-03	5.4E-02	5.1E-02	5.0E-02	4.8E-02	4.7E-02
Class 3										
Rb-89	1.4E-02	5.4E-02	6.0E+00	5.6E-04	3.6E-04	3.0E-03	1.7E-03	7.9E-04	3.6E-04	1.7E-04
Cs-134	2.1E-04	7.7E-04	9.3E-02	9.5E-06	6.1E-06	6.0E-05	5.9E-05	5.9E-05	5.9E-05	5.9E-05
Cs-136	1.7E-04	6.1E-04	7.5E-02	7.6E-06	4.9E-06	4.8E-05	4.7E-05	4.7E-05	4.7E-05	4.7E-05
Cs-137	3.2E-04	1.2E-03	1.4E-01	1.5E-05	9.5E-06	9.4E-05	9.2E-05	9.2E-05	9.2E-05	9.2E-05
Cs-138	1.5E-02	5.7E-02	6.7E+00	6.6E-04	4.2E-04	3.9E-03	3.0E-03	2.1E-03	1.4E-03	1.0E-03
Ba-137m	3.2E-04	1.2E-03	9.1E-02	5.2E-06	3.3E-06	1.4E-05	1.0E-06	9.7E-09	9.3E-11	8.9E-13
Class 4										
N-16	1.1E+02	2.5E+00	5.5E+04	6.4E-12	2.1E-12	3.9E-13	3.7E-31	1.2E-74	3.6E-118	1.1E-161
Class 5										
H-3	1.0E+00	3.8E+00	4.7E+02	4.7E-02	3.1E-02	3.0E-01	3.0E-01	3.0E-01	3.0E-01	3.0E-01
Class 6										
Na-24	3.1E-05	1.1E-04	1.4E-02	1.4E-06	9.2E-07	9.0E-06	8.7E-06	8.6E-06	8.5E-06	8.4E-06

Nuclide	Preheater (MBq)	Recombiner (MBq)	Condenser (MBq)	Cooler Condenser (MBq)	Moisture Separator (MBq)	Dryer (MBq)	Guard Tank (MBq)	Tank 1 (MBq)	Tank 2 (MBq)	Tank 3 (MBq)
P-32	1.2E-06	4.6E-06	5.6E-04	5.7E-08	3.7E-08	3.6E-07	3.5E-07	3.5E-07	3.5E-07	3.5E-07
Cr-51	3.1E-05	1.1E-04	1.4E-02	1.4E-06	9.2E-07	9.0E-06	8.9E-06	8.9E-06	8.9E-06	8.9E-06
Mn-54	1.5E-05	5.7E-05	7.0E-03	7.1E-07	4.6E-07	4.5E-06	4.4E-06	4.4E-06	4.4E-06	4.4E-06
Mn-56	5.9E-05	2.2E-04	2.6E-02	2.7E-06	1.7E-06	1.7E-05	1.5E-05	1.4E-05	1.3E-05	1.2E-05
Fe-55	3.2E-05	1.2E-04	1.4E-02	1.5E-06	9.5E-07	9.4E-06	9.2E-06	9.2E-06	9.2E-06	9.2E-06
Fe-59	8.3E-06	3.1E-05	3.7E-03	3.8E-07	2.5E-07	2.4E-06	2.4E-06	2.4E-06	2.4E-06	2.4E-06
Co-58	7.3E-06	2.7E-05	3.3E-03	3.4E-07	2.2E-07	2.1E-06	2.1E-06	2.1E-06	2.1E-06	2.1E-06
Co-60	1.4E-05	5.4E-05	6.5E-03	6.6E-07	4.3E-07	4.2E-06	4.1E-06	4.1E-06	4.1E-06	4.1E-06
Ni-63	3.2E-08	1.2E-07	1.4E-05	1.5E-09	9.5E-10	9.4E-09	9.2E-09	9.2E-09	9.2E-09	9.2E-09
Cu-64	1.5E-04	5.7E-04	7.0E-02	7.1E-06	4.6E-06	4.5E-05	4.4E-05	4.3E-05	4.2E-05	4.2E-05
Zn-65	6.7E-06	2.5E-05	3.0E-03	3.1E-07	2.0E-07	2.0E-06	1.9E-06	1.9E-06	1.9E-06	1.9E-06
Sr-89	2.4E-05	8.8E-05	1.1E-02	1.1E-06	7.1E-07	6.9E-06	6.8E-06	6.8E-06	6.8E-06	6.8E-06
Sr-90	1.1E-06	4.2E-06	5.1E-04	5.2E-08	3.4E-08	3.3E-07	3.3E-07	3.3E-07	3.3E-07	3.3E-07
Y-90	1.1E-06	4.2E-06	5.1E-04	5.2E-08	3.4E-08	3.3E-07	3.2E-07	3.2E-07	3.2E-07	3.2E-07
Sr-91	1.4E-02	5.4E-02	6.5E+00	6.6E-04	4.3E-04	4.2E-03	4.0E-03	4.0E-03	3.9E-03	3.8E-03
Sr-92	3.1E-02	1.1E-01	1.4E+01	1.4E-03	9.0E-04	8.8E-03	8.2E-03	7.6E-03	7.0E-03	6.5E-03
Y-91	3.1E-04	1.1E-03	1.4E-01	1.4E-05	9.2E-06	9.0E-05	8.9E-05	8.9E-05	8.9E-05	8.9E-05

Nuclide	Preheater (MBq)	Recombiner (MBq)	Condenser (MBq)	Cooler Condenser (MBq)	Moisture Separator (MBq)	Dryer (MBq)	Guard Tank (MBq)	Tank 1 (MBq)	Tank 2 (MBq)	Tank 3 (MBq)
Y-92	9.4E-03	3.5E-02	4.2E+00	4.3E-04	2.8E-04	2.7E-03	2.5E-03	2.4E-03	2.3E-03	2.1E-03
Y-93	1.0E-03	3.8E-03	4.6E-01	4.7E-05	3.0E-05	3.0E-04	2.9E-04	2.8E-04	2.8E-04	2.7E-04
Zr-95	6.4E-04	2.4E-03	2.9E-01	2.9E-05	1.9E-05	1.9E-04	1.8E-04	1.8E-04	1.8E-04	1.8E-04
Nb-95	6.4E-04	2.4E-03	2.9E-01	2.9E-05	1.9E-05	1.9E-04	1.8E-04	1.8E-04	1.8E-04	1.8E-04
Mo-99	3.0E-03	1.1E-02	1.4E+00	1.4E-04	8.9E-05	8.7E-04	8.6E-04	8.5E-04	8.5E-04	8.5E-04
Tc-99m	3.0E-03	1.1E-02	1.3E+00	1.4E-04	8.8E-05	8.6E-04	8.3E-04	8.0E-04	7.7E-04	7.5E-04
Ru-103	1.5E-04	5.7E-04	7.0E-02	7.1E-06	4.6E-06	4.5E-05	4.4E-05	4.4E-05	4.4E-05	4.4E-05
Rh-103m	1.5E-04	5.7E-04	6.8E-02	6.8E-06	4.4E-06	4.1E-05	3.5E-05	2.8E-05	2.3E-05	1.8E-05
Ru-106	2.4E-05	8.8E-05	1.1E-02	1.1E-06	7.1E-07	6.9E-06	6.8E-06	6.8E-06	6.8E-06	6.8E-06
Rh-106	2.4E-05	8.5E-05	2.2E-03	5.3E-09	2.9E-09	2.8E-09	7.5E-14	3.6E-24	1.7E-34	8.4E-45
Ag-110m	3.2E-08	1.2E-07	1.4E-05	1.5E-09	9.5E-10	9.4E-09	9.2E-09	9.2E-09	9.2E-09	9.2E-09
Te-129m	3.1E-04	1.1E-03	1.4E-01	1.4E-05	9.2E-06	9.0E-05	8.9E-05	8.9E-05	8.9E-05	8.9E-05
Te-131m	4.0E-04	1.5E-03	1.8E-01	1.8E-05	1.2E-05	1.2E-04	1.1E-04	1.1E-04	1.1E-04	1.1E-04
Te-132	7.4E-05	2.8E-04	3.4E-02	3.4E-06	2.2E-06	2.2E-05	2.1E-05	2.1E-05	2.1E-05	2.1E-05
Ba-140	3.2E-03	1.2E-02	1.4E+00	1.5E-04	9.5E-05	9.3E-04	9.2E-04	9.2E-04	9.2E-04	9.1E-04
La-140	3.2E-03	1.2E-02	1.4E+00	1.5E-04	9.5E-05	9.3E-04	9.1E-04	9.1E-04	9.0E-04	9.0E-04
Ce-141	1.5E-04	5.7E-04	7.0E-02	7.1E-06	4.6E-06	4.5E-05	4.4E-05	4.4E-05	4.4E-05	4.4E-05

Nuclide	Preheater (MBq)	Recombiner (MBq)	Condenser (MBq)	Cooler Condenser (MBq)	Moisture Separator (MBq)	Dryer (MBq)	Guard Tank (MBq)	Tank 1 (MBq)	Tank 2 (MBq)	Tank 3 (MBq)
Ce-144	2.4E-05	8.8E-05	1.1E-02	1.1E-06	7.1E-07	6.9E-06	6.8E-06	6.8E-06	6.8E-06	6.8E-06
Pr-144	2.4E-05	8.8E-05	1.0E-02	9.3E-07	6.0E-07	5.1E-06	3.2E-06	1.6E-06	7.9E-07	4.0E-07
W-187	7.2E-06	2.7E-05	3.3E-03	3.3E-07	2.1E-07	2.1E-06	2.1E-06	2.0E-06	2.0E-06	2.0E-06
Np-239	2.5E-03	9.2E-03	1.1E+00	1.1E-04	7.4E-05	7.2E-04	7.1E-04	7.0E-04	7.0E-04	7.0E-04

Table 12.2-13: EFS Containment Sump Activity[1]

Nuclide	Activity (MBq/m^3)	Activity (µCi/m^3)
Class 2		
I-131	6.63E+02	1.79E+04
I-132	6.43E+03	1.74E+05
I-133	4.78E+03	1.29E+05
I-134	1.80E+04	4.86E+05
I-135	9.48E+03	2.56E+05
Class 3		
Rb-89	1.06E+04	2.88E+05
Cs-134	2.03E+02	5.48E+03
Cs-136	1.58E+02	4.28E+03
Cs-137	3.10E+02	8.37E+03
Cs-138	1.33E+04	3.59E+05
Ba-137m	8.73E+01	2.36E+03
Class 5		
H-3	1.03E+03	2.79E+04
Class 6		
Na-24	3.01E+01	8.13E+02
P-32	1.22E+00	3.29E+01
Cr-51	2.95E+01	7.97E+02
Mn-54	1.47E+01	3.99E+02
Mn-56	5.38E+01	1.45E+03
Fe-55	3.06E+01	8.27E+02
Fe-59	8.11E+00	2.19E+02
Co-58	7.00E+00	1.89E+02
Co-60	1.36E+01	3.69E+02
Ni-63	3.06E-02	8.27E-01
Cu-64	1.43E+02	3.86E+03
Zn-65	6.27E+00	1.69E+02
Sr-89	2.25E+01	6.08E+02
Sr-90	1.14E+00	3.09E+01
Y-90	1.14E+00	3.09E+01
Sr-91	1.35E+04	3.66E+05
Sr-92	2.87E+04	7.76E+05
Y-91	2.99E+02	8.07E+03
Y-92	9.03E+03	2.44E+05

Nuclide	Activity (MBq/m^3)	Activity (µCi/m^3)
Class 6 (continued)		
Y-93	9.88E+02	2.67E+04
Zr-95	6.27E+02	1.69E+04
Nb-95	6.27E+02	1.69E+04
Mo-99	2.91E+03	7.86E+04
Tc-99m	2.88E+03	7.78E+04
Ru-103	1.51E+02	4.08E+03
Rh-103m	1.40E+02	3.78E+03
Ru-106	2.25E+01	6.08E+02
Rh-106	1.28E+00	3.45E+01
Ag-110m	3.06E-02	8.27E-01
Te-129m	2.99E+02	8.07E+03
Te-131m	4.05E+02	1.09E+04
Te-132	7.00E+01	1.89E+03
Ba-140	3.13E+03	8.47E+04
La-140	3.13E+03	8.45E+04
Ce-141	1.51E+02	4.08E+03
Ce-144	2.25E+01	6.08E+02
Pr-144	1.76E+01	4.77E+02
W-187	6.98E+00	1.89E+02
Np-239	2.39E+03	6.47E+04

1 Containment sump activity for a single sump with volume = 1.44 m^3.

Table 12.2-14: EFS Reactor Building Sump Activity[1]

Nuclide	Activity (MBq/m^3)	Activity (µCi/m^3)
Class 2		
I-131	4.31E+01	1.16E+03
I-132	1.98E+02	5.35E+03
I-133	2.84E+02	7.69E+03
I-134	2.61E+02	7.06E+03
I-135	4.63E+02	1.25E+04
Class 3		
Rb-89	5.45E+01	1.47E+03
Cs-134	1.33E+01	3.60E+02
Cs-136	1.03E+01	2.79E+02
Cs-137	2.03E+01	5.49E+02
Cs-138	1.30E+02	3.50E+03
Ba-137m	2.03E-01	5.48E+00
Class 5		
H-3	6.77E+01	1.83E+03
Class 6		
Na-24	1.73E+00	4.67E+01
P-32	7.93E-02	2.14E+00
Cr-51	1.93E+00	5.21E+01
Mn-54	9.67E-01	2.61E+01
Mn-56	1.78E+00	4.80E+01
Fe-55	2.01E+00	5.43E+01
Fe-59	5.31E-01	1.44E+01
Co-58	4.59E-01	1.24E+01
Co-60	8.95E-01	2.42E+01
Ni-63	2.01E-03	5.43E-02
Cu-64	8.02E+00	2.17E+02
Zn-65	4.11E-01	1.11E+01
Sr-89	1.47E+00	3.98E+01
Sr-90	7.50E-02	2.03E+00
Y-90	7.26E-02	1.96E+00
Sr-91	7.23E+02	1.95E+04
Sr-92	9.65E+02	2.61E+04
Y-91	1.96E+01	5.29E+02
Y-92	3.52E+02	9.50E+03

Nuclide	Activity (MBq/m^3)	Activity (µCi/m^3)
Class 6 (continued)		
Y-93	5.34E+01	1.44E+03
Zr-95	4.11E+01	1.11E+03
Nb-95	4.10E+01	1.11E+03
Mo-99	1.85E+02	5.00E+03
Tc-99m	1.37E+02	3.70E+03
Ru-103	9.90E+00	2.67E+02
Rh-103m	2.15E+00	5.82E+01
Ru-106	1.48E+00	3.99E+01
Rh-106	2.87E-03	7.75E-02
Ag-110m	2.01E-03	5.42E-02
Te-129m	1.95E+01	5.28E+02
Te-131m	2.48E+01	6.70E+02
Te-132	4.47E+00	1.21E+02
Ba-140	2.04E+02	5.52E+03
La-140	1.95E+02	5.27E+03
Ce-141	9.89E+00	2.67E+02
Ce-144	1.48E+00	3.99E+01
Pr-144	9.97E-02	2.70E+00
W-187	4.21E-01	1.14E+01
Np-239	1.51E+02	4.09E+03

1 RB sump activity for a single sump with volume = 1.27 m^3.

Table 12.2-15: EFS Radwaste Building Sump Activity[1]

Nuclide	Activity (MBq/m^3)	Activity (µCi/m^3)
Class 2		
I-131	1.22E+01	3.29E+02
I-132	5.98E+01	1.61E+03
I-133	8.11E+01	2.19E+03
I-134	8.16E+01	2.21E+03
I-135	1.35E+02	3.64E+03
Class 3		
Rb-89	1.71E+01	4.62E+02
Cs-134	3.76E+00	1.02E+02
Cs-136	2.92E+00	7.89E+01
Cs-137	5.74E+00	1.55E+02
Cs-138	4.06E+01	1.10E+03
Ba-137m	6.37E-02	1.72E+00
Class 5		
H-3	1.91E+01	5.17E+02
Class 6		
Na-24	4.94E-01	1.34E+01
P-32	2.24E-02	6.06E-01
Cr-51	5.45E-01	1.47E+01
Mn-54	2.73E-01	7.38E+00
Mn-56	5.33E-01	1.44E+01
Fe-55	5.67E-01	1.53E+01
Fe-59	1.50E-01	4.05E+00
Co-58	1.30E-01	3.50E+00
Co-60	2.53E-01	6.83E+00
Ni-63	5.67E-04	1.53E-02
Cu-64	2.30E+00	6.21E+01
Zn-65	1.16E-01	3.14E+00
Sr-89	4.16E-01	1.12E+01
Sr-90	2.12E-02	5.72E-01
Y-90	2.06E-02	5.55E-01
Sr-91	2.08E+02	5.63E+03
Sr-92	2.89E+02	7.82E+03
Y-91	5.52E+00	1.49E+02
Y-92	1.04E+02	2.82E+03

Nuclide	Activity (MBq/m^3)	Activity (µCi/m^3)
Class 6 (continued)		
Y-93	1.54E+01	4.15E+02
Zr-95	1.16E+01	3.13E+02
Nb-95	1.16E+01	3.13E+02
Mo-99	5.24E+01	1.42E+03
Tc-99m	3.99E+01	1.08E+03
Ru-103	2.79E+00	7.55E+01
Rh-103m	6.72E-01	1.82E+01
Ru-106	4.17E-01	1.13E+01
Rh-106	9.00E-04	2.43E-02
Ag-110m	5.67E-04	1.53E-02
Te-129m	5.52E+00	1.49E+02
Te-131m	7.05E+00	1.91E+02
Te-132	1.27E+00	3.42E+01
Ba-140	5.77E+01	1.56E+03
La-140	5.54E+01	1.50E+03
Ce-141	2.79E+00	7.55E+01
Ce-144	4.17E-01	1.13E+01
Pr-144	3.13E-02	8.46E-01
W-187	1.20E-01	3.24E+00
Np-239	4.29E+01	1.16E+03

1 RWB sump activity for a single sump with volume = 1.27 m^3.

Table 12.2-16: EFS Turbine Building Sump Activity[1]

Nuclide	Activity (MBq/m^3)	Activity (µCi/m^3)
Class 2		
I-131	6.58E-01	1.78E+01
I-132	2.74E+00	7.42E+01
I-133	4.28E+00	1.16E+02
I-134	3.48E+00	9.40E+01
I-135	6.77E+00	1.83E+02
Class 3		
Rb-89	7.22E-01	1.95E+01
Cs-134	2.03E-01	5.50E+00
Cs-136	1.58E-01	4.27E+00
Cs-137	3.11E-01	8.40E+00
Cs-138	1.72E+00	4.64E+01
Ba-137m	2.69E-03	7.27E-02
Class 5		
H-3	1.04E+00	2.80E+01
Class 6		
Na-24	2.59E-02	6.99E-01
P-32	1.21E-03	3.28E-02
Cr-51	2.95E-02	7.97E-01
Mn-54	1.48E-02	4.00E-01
Mn-56	2.48E-02	6.70E-01
Fe-55	3.07E-02	8.30E-01
Fe-59	8.12E-03	2.19E-01
Co-58	7.02E-03	1.90E-01
Co-60	1.37E-02	3.70E-01
Ni-63	3.07E-05	8.30E-04
Cu-64	1.20E-01	3.23E+00
Zn-65	6.29E-03	1.70E-01
Sr-89	2.25E-02	6.09E-01
Sr-90	1.15E-03	3.10E-02
Y-90	1.10E-03	2.98E-02
Sr-91	1.07E+01	2.90E+02
Sr-92	1.35E+01	3.65E+02
Y-91	2.99E-01	8.09E+00
Y-92	5.00E+00	1.35E+02

Nuclide	Activity (MBq/m^3)	Activity (µCi/m^3)
Class 6 (continued)		
Y-93	7.92E-01	2.14E+01
Zr-95	6.28E-01	1.70E+01
Nb-95	6.27E-01	1.70E+01
Mo-99	2.82E+00	7.61E+01
Tc-99m	2.00E+00	5.40E+01
Ru-103	1.51E-01	4.09E+00
Rh-103m	2.87E-02	7.76E-01
Ru-106	2.26E-02	6.10E-01
Rh-106	3.80E-05	1.03E-03
Ag-110m	3.07E-05	8.30E-04
Te-129m	2.99E-01	8.08E+00
Te-131m	3.75E-01	1.01E+01
Te-132	6.81E-02	1.84E+00
Ba-140	3.12E+00	8.43E+01
La-140	2.96E+00	8.00E+01
Ce-141	1.51E-01	4.09E+00
Ce-144	2.26E-02	6.10E-01
Pr-144	1.32E-03	3.57E-02
W-187	6.35E-03	1.72E-01
Np-239	2.30E+00	6.23E+01

1 TB sump activity for a single sump with volume = 1.27 m^3.

Table 12.2-17: LWM Collection Tank Activity[1]

Nuclide	Activity (MBq/m^3)	Activity (µCi/m^3)
Class 2		
I-131	2.16E+02	5.83E+03
I-132	1.92E+03	5.19E+04
I-133	1.54E+03	4.16E+04
I-134	4.87E+03	1.32E+05
I-135	2.98E+03	8.06E+04
Class 3		
Rb-89	2.12E+03	5.73E+04
Cs-134	6.60E+01	1.78E+03
Cs-136	5.16E+01	1.39E+03
Cs-137	1.01E+02	2.73E+03
Cs-138	3.33E+03	9.00E+04
Ba-137m	6.53E+00	1.76E+02
Class 5		
H-3	3.36E+02	9.09E+03
Class 6		
Na-24	9.65E+00	2.61E+02
P-32	3.96E-01	1.07E+01
Cr-51	9.60E+00	2.60E+02
Mn-54	4.80E+00	1.30E+02
Mn-56	1.62E+01	4.38E+02
Fe-55	9.97E+00	2.69E+02
Fe-59	2.64E+00	7.14E+01
Co-58	2.28E+00	6.17E+01
Co-60	4.44E+00	1.20E+02
Ni-63	9.97E-03	2.69E-01
Cu-64	4.57E+01	1.24E+03
Zn-65	2.04E+00	5.52E+01
Sr-89	7.32E+00	1.98E+02
Sr-90	3.72E-01	1.01E+01
Y-90	3.70E-01	1.00E+01
Sr-91	4.31E+03	1.16E+05
Sr-92	8.67E+03	2.34E+05
Y-91	9.72E+01	2.63E+03
Y-92	2.77E+03	7.49E+04

Nuclide	Activity (MBq/m^3)	Activity (µCi/m^3)
Class 6 (continued)		
Y-93	3.15E+02	8.50E+03
Zr-95	2.04E+02	5.52E+03
Nb-95	2.04E+02	5.52E+03
Mo-99	9.44E+02	2.55E+04
Tc-99m	9.03E+02	2.44E+04
Ru-103	4.92E+01	1.33E+03
Rh-103m	3.83E+01	1.03E+03
Ru-106	7.32E+00	1.98E+02
Rh-106	5.79E-02	1.56E+00
Ag-110m	9.97E-03	2.69E-01
Te-129m	9.72E+01	2.63E+03
Te-131m	1.31E+02	3.53E+03
Te-132	2.27E+01	6.14E+02
Ba-140	1.02E+03	2.76E+04
La-140	1.01E+03	2.74E+04
Ce-141	4.92E+01	1.33E+03
Ce-144	7.32E+00	1.98E+02
Pr-144	3.69E+00	9.97E+01
W-187	2.25E+00	6.09E+01
Np-239	7.76E+02	2.10E+04

1 Collection tank activity for a single tank with volume = 100 m^3. There are two LWM collection tanks.

Table 12.2-18: LWM Sample Tank A Activity[1]

Nuclide	Activity (MBq/m^3)	Activity (µCi/m^3)
Class 2		
I-131	2.13E-03	5.77E-02
I-132	8.61E-03	2.33E-01
I-133	1.39E-02	3.75E-01
I-134	9.72E-03	2.63E-01
I-135	2.18E-02	5.90E-01
Class 3		
Rb-89	6.15E-02	1.66E+00
Cs-134	3.30E-02	8.92E-01
Cs-136	2.56E-02	6.92E-01
Cs-137	5.04E-02	1.36E+00
Cs-138	2.13E-01	5.75E+00
Ba-137m	3.19E-05	8.61E-04
Class 5		
H-3	3.36E+02	9.09E+03
Class 6		
Na-24	8.37E-05	2.26E-03
P-32	3.93E-06	1.06E-04
Cr-51	9.57E-05	2.59E-03
Mn-54	4.80E-05	1.30E-03
Mn-56	7.82E-05	2.11E-03
Fe-55	9.97E-05	2.69E-03
Fe-59	2.64E-05	7.12E-04
Co-58	2.28E-05	6.16E-04
Co-60	4.44E-05	1.20E-03
Ni-63	9.97E-08	2.69E-06
Cu-64	3.87E-04	1.05E-02
Zn-65	2.04E-05	5.51E-04
Sr-89	7.31E-05	1.98E-03
Sr-90	3.72E-06	1.01E-04
Y-90	3.58E-06	9.68E-05
Sr-91	3.46E-02	9.36E-01
Sr-92	4.26E-02	1.15E+00
Y-91	9.71E-04	2.62E-02
Y-92	1.59E-02	4.31E-01

Nuclide	Activity (MBq/m^3)	Activity (µCi/m^3)
Class 6 (continued)		
Y-93	2.56E-03	6.92E-02
Zr-95	2.04E-03	5.51E-02
Nb-95	2.04E-03	5.50E-02
Mo-99	9.14E-03	2.47E-01
Tc-99m	6.43E-03	1.74E-01
Ru-103	4.91E-04	1.33E-02
Rh-103m	8.14E-05	2.20E-03
Ru-106	7.32E-05	1.98E-03
Rh-106	1.10E-09	2.97E-08
Ag-110m	9.96E-08	2.69E-06
Te-129m	9.70E-04	2.62E-02
Te-131m	1.22E-03	3.29E-02
Te-132	2.21E-04	5.97E-03
Ba-140	1.01E-02	2.74E-01
La-140	9.60E-03	2.59E-01
Ce-141	4.91E-04	1.33E-02
Ce-144	7.32E-05	1.98E-03
Pr-144	2.44E-06	6.59E-05
W-187	2.06E-05	5.56E-04
Np-239	7.47E-03	2.02E-01

1 Sample Tank A activity for a single tank with volume = 100 m^3.

Table 12.2-19: LWM Sample Tank B Activity[1]

Nuclide	Activity (MBq/m^3)	Activity (µCi/m^3)
Class 2		
I-131	2.14E-03	5.79E-02
I-132	1.15E-02	3.10E-01
I-133	1.45E-02	3.91E-01
I-134	1.55E-02	4.19E-01
I-135	2.46E-02	6.65E-01
Class 3		
Rb-89	1.02E-01	2.77E+00
Cs-134	3.30E-02	8.92E-01
Cs-136	2.57E-02	6.94E-01
Cs-137	5.04E-02	1.36E+00
Cs-138	3.51E-01	9.50E+00
Ba-137m	5.31E-05	1.43E-03
Class 5		
H-3	3.36E+02	9.09E+03
Class 6		
Na-24	8.85E-05	2.39E-03
P-32	3.94E-06	1.07E-04
Cr-51	9.58E-05	2.59E-03
Mn-54	4.80E-05	1.30E-03
Mn-56	1.02E-04	2.75E-03
Fe-55	9.97E-05	2.69E-03
Fe-59	2.64E-05	7.13E-04
Co-58	2.28E-05	6.16E-04
Co-60	4.44E-05	1.20E-03
Ni-63	9.97E-08	2.69E-06
Cu-64	4.13E-04	1.12E-02
Zn-65	2.04E-05	5.52E-04
Sr-89	7.32E-05	1.98E-03
Sr-90	3.72E-06	1.01E-04
Y-90	3.63E-06	9.81E-05
Sr-91	3.77E-02	1.02E+00
Sr-92	5.52E-02	1.49E+00
Y-91	9.72E-04	2.63E-02
Y-92	1.96E-02	5.30E-01

Nuclide	Activity (MBq/m^3)	Activity (µCi/m^3)
Class 6 (continued)		
Y-93	2.78E-03	7.50E-02
Zr-95	2.04E-03	5.51E-02
Nb-95	2.04E-03	5.51E-02
Mo-99	9.26E-03	2.50E-01
Tc-99m	7.33E-03	1.98E-01
Ru-103	4.91E-04	1.33E-02
Rh-103m	1.29E-04	3.47E-03
Ru-106	7.32E-05	1.98E-03
Rh-106	1.83E-09	4.95E-08
Ag-110m	9.96E-08	2.69E-06
Te-129m	9.71E-04	2.62E-02
Te-131m	1.25E-03	3.38E-02
Te-132	2.23E-04	6.04E-03
Ba-140	1.02E-02	2.74E-01
La-140	9.81E-03	2.65E-01
Ce-141	4.91E-04	1.33E-02
Ce-144	7.32E-05	1.98E-03
Pr-144	4.07E-06	1.10E-04
W-187	2.13E-05	5.76E-04
Np-239	7.59E-03	2.05E-01

1 Sample Tank B activity for a single tank with volume = 60 m^3.

Table 12.2-20: LWM Condensate Storage Tank Activity[1]

Nuclide	Activity (MBq/m^3)	Activity (µCi/m^3)
Class 2		
I-131	2.07E-03	5.61E-02
I-132	2.06E-03	5.57E-02
I-133	1.08E-02	2.92E-01
I-134	1.07E-03	2.88E-02
I-135	1.09E-02	2.94E-01
Class 3		
Rb-89	2.03E-03	5.50E-02
Cs-134	3.30E-02	8.92E-01
Cs-136	2.52E-02	6.80E-01
Cs-137	5.04E-02	1.36E+00
Cs-138	1.54E-02	4.16E-01
Ba-137m	1.78E-07	4.80E-06
Class 5		
H-3	3.36E+02	9.09E+03
Class 6		
Na-24	5.96E-05	1.61E-03
P-32	3.87E-06	1.05E-04
Cr-51	9.49E-05	2.57E-03
Mn-54	4.80E-05	1.30E-03
Mn-56	2.05E-05	5.54E-04
Fe-55	9.96E-05	2.69E-03
Fe-59	2.62E-05	7.09E-04
Co-58	2.27E-05	6.14E-04
Co-60	4.44E-05	1.20E-03
Ni-63	9.97E-08	2.69E-06
Cu-64	2.61E-04	7.05E-03
Zn-65	2.04E-05	5.51E-04
Sr-89	7.28E-05	1.97E-03
Sr-90	3.72E-06	1.01E-04
Y-90	3.29E-06	8.90E-05
Sr-91	2.09E-02	5.66E-01
Sr-92	1.14E-02	3.09E-01
Y-91	9.67E-04	2.61E-02
Y-92	5.30E-03	1.43E-01

Nuclide	Activity (MBq/m^3)	Activity (µCi/m^3)
Class 6 (continued)		
Y-93	1.58E-03	4.28E-02
Zr-95	2.03E-03	5.49E-02
Nb-95	2.02E-03	5.47E-02
Mo-99	8.42E-03	2.28E-01
Tc-99m	3.05E-03	8.23E-02
Ru-103	4.88E-04	1.32E-02
Rh-103m	9.45E-06	2.55E-04
Ru-106	7.32E-05	1.98E-03
Rh-106	1.19E-12	3.22E-11
Ag-110m	9.95E-08	2.69E-06
Te-129m	9.63E-04	2.60E-02
Te-131m	1.02E-03	2.76E-02
Te-132	2.06E-04	5.56E-03
Ba-140	9.94E-03	2.69E-01
La-140	8.41E-03	2.27E-01
Ce-141	4.87E-04	1.32E-02
Ce-144	7.32E-05	1.98E-03
Pr-144	9.21E-08	2.49E-06
W-187	1.65E-05	4.46E-04
Np-239	6.79E-03	1.84E-01

1 CST activity for a single tank with volume = 625 m^3.

Table 12.2-21: LWM Refueling Water Storage Tank Activity[1]

Nuclide	Activity (MBq/m^3)	Activity (µCi/m^3)
Class 2		
I-131	5.54E+01	1.50E+03
I-132	3.89E+02	1.05E+04
I-133	3.86E+02	1.04E+04
I-134	7.05E+02	1.90E+04
I-135	7.05E+02	1.91E+04
Class 3		
Rb-89	1.68E+02	4.54E+03
Cs-134	1.70E+01	4.60E+02
Cs-136	1.33E+01	3.59E+02
Cs-137	2.60E+01	7.02E+02
Cs-138	3.83E+02	1.04E+04
Ba-137m	6.26E-01	1.69E+01
Class 5		
H-3	8.66E+01	2.34E+03
Class 6		
Na-24	2.39E+00	6.47E+01
P-32	1.02E-01	2.75E+00
Cr-51	2.47E+00	6.68E+01
Mn-54	1.24E+00	3.34E+01
Mn-56	3.36E+00	9.08E+01
Fe-55	2.57E+00	6.94E+01
Fe-59	6.80E-01	1.84E+01
Co-58	5.87E-01	1.59E+01
Co-60	1.14E+00	3.09E+01
Ni-63	2.57E-03	6.94E-02
Cu-64	1.13E+01	3.04E+02
Zn-65	5.26E-01	1.42E+01
Sr-89	1.89E+00	5.10E+01
Sr-90	9.59E-02	2.59E+00
Y-90	9.46E-02	2.56E+00
Sr-91	1.05E+03	2.83E+04
Sr-92	1.81E+03	4.89E+04
Y-91	2.50E+01	6.77E+02
Y-92	6.09E+02	1.65E+04

Nuclide	Activity (MBq/m^3)	Activity (µCi/m^3)
Class 6 (continued)		
Y-93	7.67E+01	2.07E+03
Zr-95	5.26E+01	1.42E+03
Nb-95	5.25E+01	1.42E+03
Mo-99	2.41E+02	6.52E+03
Tc-99m	2.12E+02	5.72E+03
Ru-103	1.27E+01	3.42E+02
Rh-103m	5.70E+00	1.54E+02
Ru-106	1.89E+00	5.10E+01
Rh-106	8.85E-03	2.39E-01
Ag-110m	2.57E-03	6.94E-02
Te-129m	2.50E+01	6.76E+02
Te-131m	3.30E+01	8.93E+02
Te-132	5.81E+00	1.57E+02
Ba-140	2.62E+02	7.09E+03
La-140	2.57E+02	6.95E+03
Ce-141	1.27E+01	3.42E+02
Ce-144	1.89E+00	5.10E+01
Pr-144	3.07E-01	8.30E+00
W-187	5.66E-01	1.53E+01
Np-239	1.98E+02	5.35E+03

1 RWST activity for a single tank with volume = 625 m^3.

Table 12.2-22: LWM Filter Skid Resin Activity

Nuclide	Activity (Bq)
Class 2	
I-131	9.54E+11
I-132	1.01E+11
I-133	7.34E+11
I-134	9.79E+10
I-135	4.50E+11
Class 3	
Rb-89	1.23E+10
Cs-134	7.82E+12
Cs-136	3.74E+11
Cs-137	1.39E+13
Cs-138	4.26E+10
Ba-137m	6.37E+06
Class 5	
H-3	0.00E+00
Class 6	
Na-24	3.31E+09
P-32	3.11E+09
Cr-51	1.47E+11
Mn-54	4.59E+11
Mn-56	9.59E+08
Fe-55	1.23E+12
Fe-59	6.45E+10
Co-58	8.65E+10
Co-60	5.80E+11
Ni-63	1.38E+09
Cu-64	1.33E+10
Zn-65	1.77E+11
Sr-89	2.03E+11
Sr-90	5.13E+10
Y-90	5.45E+08
Sr-91	9.52E+11
Sr-92	5.29E+11
Y-91	3.09E+12
Y-92	2.25E+11

Nuclide	Activity (Bq)
Class 6 (continued)	
Y-93	7.35E+10
Zr-95	7.06E+12
Nb-95	3.93E+12
Mo-99	1.43E+12
Tc-99m	1.25E+11
Ru-103	1.06E+12
Rh-103m	8.21E+08
Ru-106	7.42E+11
Rh-106	1.10E+04
Ag-110m	8.73E+08
Te-129m	1.80E+12
Te-131m	9.00E+10
Te-132	4.01E+10
Ba-140	7.16E+12
La-140	9.36E+11
Ce-141	8.81E+11
Ce-144	6.77E+11
Pr-144	2.44E+07
W-187	1.23E+09
Np-239	1.01E+12

Table 12.2-23: SWM Spent Resin Tank Activity[1]

Nuclide	Activity (Bq/m^3)	Activity (µCi/m^3)
Class 2		
I-131	3.05E+11	8.24E+06
I-132	3.42E+10	9.25E+05
I-133	2.40E+11	6.49E+06
I-134	4.00E+10	1.08E+06
I-135	1.41E+11	3.82E+06
Class 3		
Rb-89	1.13E+09	3.04E+04
Cs-134	4.87E+11	1.32E+07
Cs-136	2.34E+10	6.33E+05
Cs-137	8.68E+11	2.35E+07
Cs-138	3.17E+09	8.56E+04
Ba-137m	2.73E+06	7.38E+01
Class 5		
H-3	0.00E+00	0.00E+00
Class 6		
Na-24	1.42E+08	3.83E+03
P-32	1.33E+08	3.59E+03
Cr-51	6.26E+09	1.69E+05
Mn-54	1.96E+10	5.30E+05
Mn-56	4.17E+07	1.13E+03
Fe-55	5.25E+10	1.42E+06
Fe-59	2.75E+09	7.44E+04
Co-58	3.70E+09	1.00E+05
Co-60	2.48E+10	6.71E+05
Ni-63	5.92E+07	1.60E+03
Cu-64	5.73E+08	1.55E+04
Zn-65	7.58E+09	2.05E+05
Sr-89	8.66E+09	2.34E+05
Sr-90	2.18E+09	5.90E+04
Y-90	2.32E+07	6.28E+02
Sr-91	4.09E+10	1.11E+06
Sr-92	2.29E+10	6.20E+05
Y-91	1.32E+11	3.57E+06
Y-92	9.71E+09	2.63E+05

Nuclide	Activity (Bq/m^3)	Activity (µCi/m^3)
Class 6 (continued)		
Y-93	3.15E+09	8.52E+04
Zr-95	3.01E+11	8.14E+06
Nb-95	1.68E+11	4.53E+06
Mo-99	6.10E+10	1.65E+06
Tc-99m	5.36E+09	1.45E+05
Ru-103	4.54E+10	1.23E+06
Rh-103m	3.64E+07	9.83E+02
Ru-106	3.17E+10	8.57E+05
Rh-106	8.25E+03	2.23E-01
Ag-110m	3.73E+07	1.01E+03
Te-129m	7.68E+10	2.08E+06
Te-131m	3.84E+09	1.04E+05
Te-132	1.72E+09	4.64E+04
Ba-140	3.06E+11	8.26E+06
La-140	4.00E+10	1.08E+06
Ce-141	3.76E+10	1.02E+06
Ce-144	2.89E+10	7.82E+05
Pr-144	1.18E+06	3.19E+01
W-187	5.24E+07	1.42E+03
Np-239	4.31E+10	1.16E+06

1 SWM Spent Resin Tank activity for a single tank with volume = 26.5 m^3.

Table 12.2-24: SWM Sludge Storage Tank Activity[1]

Nuclide	Activity (Bq/m^3)	Activity (µCi/m^3)
Class 2		
I-131	0.00E+00	0.00E+00
I-132	0.00E+00	0.00E+00
I-133	0.00E+00	0.00E+00
I-134	0.00E+00	0.00E+00
I-135	0.00E+00	0.00E+00
Class 3		
Rb-89	0.00E+00	0.00E+00
Cs-134	0.00E+00	0.00E+00
Cs-136	0.00E+00	0.00E+00
Cs-137	0.00E+00	0.00E+00
Cs-138	0.00E+00	0.00E+00
Ba-137m	0.00E+00	0.00E+00
Class 5		
H-3	0.00E+00	0.00E+00
Class 6		
Na-24	2.28E+10	6.17E+05
P-32	2.10E+10	5.68E+05
Cr-51	9.92E+11	2.68E+07
Mn-54	3.08E+12	8.32E+07
Mn-56	7.23E+09	1.95E+05
Fe-55	8.21E+12	2.22E+08
Fe-59	4.36E+11	1.18E+07
Co-58	5.85E+11	1.58E+07
Co-60	3.88E+12	1.05E+08
Ni-63	9.24E+09	2.50E+05
Cu-64	9.26E+10	2.50E+06
Zn-65	1.19E+12	3.22E+07
Sr-89	1.37E+12	3.71E+07
Sr-90	3.41E+11	9.21E+06
Y-90	3.69E+09	9.97E+04
Sr-91	6.65E+12	1.80E+08
Sr-92	3.97E+12	1.07E+08
Y-91	2.09E+13	5.65E+08
Y-92	1.65E+12	4.45E+07

Nuclide	Activity (Bq/m^3)	Activity (µCi/m^3)
Class 6 (continued)		
Y-93	5.12E+11	1.38E+07
Zr-95	4.76E+13	1.29E+09
Nb-95	2.66E+13	7.18E+08
Mo-99	9.70E+12	2.62E+08
Tc-99m	8.85E+11	2.39E+07
Ru-103	7.18E+12	1.94E+08
Rh-103m	7.14E+09	1.93E+05
Ru-106	4.97E+12	1.34E+08
Rh-106	9.43E+06	2.55E+02
Ag-110m	5.86E+09	1.58E+05
Te-129m	1.22E+13	3.29E+08
Te-131m	6.13E+11	1.66E+07
Te-132	2.73E+11	7.37E+06
Ba-140	4.84E+13	1.31E+09
La-140	6.37E+12	1.72E+08
Ce-141	5.95E+12	1.61E+08
Ce-144	4.54E+12	1.23E+08
Pr-144	3.28E+08	8.86E+03
W-187	8.40E+09	2.27E+05
Np-239	6.85E+12	1.85E+08

1 Sludge Storage tank activity for a single tank with volume = 41.64 m^3. There are two sludge storage tanks.

Table 12.2-25: Reactor Building Upper-Level Airborne Activity

Isotope	Airborne Concentration [µCi/mL]	10 CFR 20 DAC Value [µCi/mL]	DAC Fraction
Class 2			
I-131	6.5E-12	2E-08	3.3E-04
I-132	5.3E-11	3E-06	1.8E-05
I-133	4.7E-11	1E-07	4.7E-04
I-134	1.2E-10	2E-05	5.9E-06
I-135	8.8E-11	7E-07	1.3E-04
Class 3			
Rb-89	4.4E-11	6E-05	7.3E-07
Cs-134	2.0E-12	4E-08	5.0E-05
Cs-136	1.6E-12	3E-07	5.3E-06
Cs-137	3.2E-12	6E-08	5.3E-05
Cs-138	7.6E-11	2E-05	3.8E-06
Ba-137m	2.3E-13	1E-07	2.3E-06
Class 5			
H-3	1.3E-07	2E-05	6.3E-03
Class 6			
Na-24	7.0E-11	2E-06	3.5E-05
P-32	2.9E-12	2E-07	1.5E-05
Cr-51	7.3E-11	8E-06	9.1E-06
Mn-54	3.6E-11	3E-07	1.2E-04
Mn-56	1.1E-10	6E-06	1.9E-05
Fe-55	7.5E-11	8E-07	9.4E-05
Fe-59	1.9E-11	1E-07	1.9E-04
Co-58	1.7E-11	3E-07	5.8E-05
Co-60	3.4E-11	1E-08	3.4E-03
Ni-63	7.5E-14	3E-07	2.5E-07
Cu-64	3.5E-10	9E-06	3.9E-05
Zn-65	1.6E-11	1E-07	1.6E-04
Sr-89	2.2E-13	6E-08	3.7E-06
Sr-90	1.1E-14	2E-09	5.6E-06
Y-90	1.1E-14	3E-07	3.7E-08
Sr-91	1.3E-10	1E-06	1.3E-04
Sr-92	2.4E-10	3E-06	8.1E-05
Y-91	2.9E-12	5E-08	5.8E-05

Isotope	Airborne Concentration [µCi/mL]	10 CFR 20 DAC Value [µCi/mL]	DAC Fraction
Class 6 (continued)			
Y-92	7.8E-11	3E-06	2.6E-05
Y-93	9.2E-12	1E-06	9.2E-06
Zr-95	6.1E-12	5E-08	1.2E-04
Nb-95	6.1E-12	5E-07	1.2E-05
Mo-99	2.9E-11	6E-07	4.8E-05
Tc-99m	2.7E-11	6E-05	4.5E-07
Ru-103	1.5E-12	3E-07	4.9E-06
Rh-103m	9.3E-13	5E-04	1.9E-09
Ru-106	2.2E-13	5E-09	4.5E-05
Rh-106	3.3E-15	1E-07	3.3E-08
Ag-110m	7.5E-14	4E-08	1.9E-06
Te-129m	2.9E-12	1E-07	2.9E-05
Te-131m	3.8E-12	2E-07	1.9E-05
Te-132	7.0E-13	9E-08	7.8E-06
Ba-140	3.2E-11	6E-07	5.3E-05
La-140	3.1E-11	5E-07	6.2E-05
Ce-141	1.5E-12	2E-07	7.4E-06
Ce-144	2.2E-13	6E-09	3.7E-05
Pr-144	7.7E-14	5E-05	1.5E-09
W-187	1.7E-11	4E-06	4.2E-06
Np-239	2.4E-11	9E-07	2.6E-05
		Total DAC Fraction	1.2E-02

Table 12.2-26: Reactor Building Lower Levels Airborne Activity

Isotope	Airborne Concentration [µCi/mL]	10 CFR 20 DAC Value [µCi/mL]	DAC Fraction
Class 2			
I-131	9.7E-13	2E-08	4.9E-05
I-132	8.5E-12	3E-06	2.8E-06
I-133	7.1E-12	1E-07	7.1E-05
I-134	2.0E-11	2E-05	1.0E-06
I-135	1.3E-11	7E-07	1.9E-05
Class 3			
Rb-89	8.7E-12	6E-05	1.5E-07
Cs-134	3.0E-13	4E-08	7.5E-06
Cs-136	2.3E-13	3E-07	7.8E-07
Cs-137	4.7E-13	6E-08	7.8E-06
Cs-138	1.4E-11	2E-05	6.9E-07
Ba-137m	5.2E-14	1E-07	5.2E-07
Class 5			
H-3	1.9E-10	2E-05	9.4E-06
Class 6			
Na-24	1.1E-11	2E-06	5.3E-06
P-32	4.3E-13	2E-07	2.2E-06
Cr-51	1.1E-11	8E-06	1.4E-06
Mn-54	5.4E-12	3E-07	1.8E-05
Mn-56	1.8E-11	6E-06	3.0E-06
Fe-55	1.1E-11	8E-07	1.4E-05
Fe-59	2.9E-12	1E-07	2.9E-05
Co-58	2.6E-12	3E-07	8.5E-06
Co-60	5.1E-12	1E-08	5.1E-04
Ni-63	1.1E-14	3E-07	3.7E-08
Cu-64	5.3E-11	9E-06	5.9E-06
Zn-65	2.3E-12	1E-07	2.3E-05
Sr-89	3.3E-14	6E-08	5.5E-07
Sr-90	1.7E-15	2E-09	8.3E-07
Y-90	1.7E-15	3E-07	5.5E-09
Sr-91	2.0E-11	1E-06	2.0E-05
Sr-92	3.8E-11	3E-06	1.3E-05
Y-91	4.3E-13	5E-08	8.7E-06

Isotope	Airborne Concentration [µCi/mL]	10 CFR 20 DAC Value [µCi/mL]	DAC Fraction
Class 6 (continued)			
Y-92	1.2E-11	3E-06	4.1E-06
Y-93	1.4E-12	1E-06	1.4E-06
Zr-95	9.0E-13	5E-08	1.8E-05
Nb-95	9.0E-13	5E-07	1.8E-06
Mo-99	4.3E-12	6E-07	7.2E-06
Tc-99m	4.1E-12	6E-05	6.8E-08
Ru-103	2.2E-13	3E-07	7.3E-07
Rh-103m	1.6E-13	5E-04	3.2E-10
Ru-106	3.3E-14	5E-09	6.6E-06
Rh-106	7.8E-16	1E-07	7.8E-09
Ag-110m	1.1E-14	4E-08	2.8E-07
Te-129m	4.3E-13	1E-07	4.3E-06
Te-131m	5.7E-13	2E-07	2.9E-06
Te-132	1.0E-13	9E-08	1.2E-06
Ba-140	4.7E-12	6E-07	7.8E-06
La-140	4.7E-12	5E-07	9.3E-06
Ce-141	2.2E-13	2E-07	1.1E-06
Ce-144	3.3E-14	6E-09	5.5E-06
Pr-144	1.5E-14	5E-05	3.0E-10
W-187	2.5E-12	4E-06	6.2E-07
Np-239	3.5E-12	9E-07	3.9E-06
		Total DAC Fraction	9.0E-04

Table 12.2-27: Radwaste Building Airborne Activity

Isotope	Airborne Concentration µCi/mL]	10 CFR 20 DAC Value [µCi/mL]	DAC Fraction
Class 2			
I-131	1.4E-13	2E-08	7.1E-06
I-132	1.4E-12	3E-06	4.7E-07
I-133	1.1E-12	1E-07	1.1E-05
I-134	4.0E-12	2E-05	2.0E-07
I-135	2.0E-12	7E-07	2.9E-06
Class 3			
Rb-89	2.5E-12	6E-05	4.2E-08
Cs-134	4.4E-14	4E-08	1.1E-06
Cs-136	3.4E-14	3E-07	1.1E-07
Cs-137	6.9E-14	6E-08	1.1E-06
Cs-138	3.0E-12	2E-05	1.5E-07
Ba-137m	3.3E-14	1E-07	3.3E-07
Class 5			
H-3	2.7E-11	2E-05	1.4E-06
Class 6			
Na-24	1.6E-12	2E-06	7.9E-07
P-32	6.3E-14	2E-07	3.2E-07
Cr-51	1.6E-12	8E-06	2.0E-07
Mn-54	7.9E-13	3E-07	2.6E-06
Mn-56	3.0E-12	6E-06	4.9E-07
Fe-55	1.6E-12	8E-07	2.0E-06
Fe-59	4.2E-13	1E-07	4.2E-06
Co-58	3.8E-13	3E-07	1.3E-06
Co-60	7.4E-13	1E-08	7.4E-05
Ni-63	1.6E-15	3E-07	5.5E-09
Cu-64	7.9E-12	9E-06	8.8E-07
Zn-65	3.4E-13	1E-07	3.4E-06
Sr-89	4.9E-15	6E-08	8.1E-08
Sr-90	2.4E-16	2E-09	1.2E-07
Y-90	2.4E-16	3E-07	8.1E-10
Sr-91	2.9E-12	1E-06	2.9E-06
Sr-92	6.2E-12	3E-06	2.1E-06

Isotope	Airborne Concentration µCi/mL]	10 CFR 20 DAC Value [µCi/mL]	DAC Fraction
Class 6 (continued)			
Y-91	6.3E-14	5E-08	1.3E-06
Y-92	1.9E-12	3E-06	6.4E-07
Y-93	2.1E-13	1E-06	2.1E-07
Zr-95	1.3E-13	5E-08	2.6E-06
Nb-95	1.3E-13	5E-07	2.6E-07
Mo-99	6.3E-13	6E-07	1.1E-06
Tc-99m	6.3E-13	6E-05	1.0E-08
Ru-103	3.2E-14	3E-07	1.1E-07
Rh-103m	3.1E-14	5E-04	6.1E-11
Ru-106	4.9E-15	5E-09	9.7E-07
Rh-106	7.3E-16	1E-07	7.3E-09
Ag-110m	1.6E-15	4E-08	4.1E-08
Te-129m	6.3E-14	1E-07	6.3E-07
Te-131m	8.4E-14	2E-07	4.2E-07
Te-132	1.5E-14	9E-08	1.7E-07
Ba-140	6.9E-13	6E-07	1.1E-06
La-140	6.9E-13	5E-07	1.4E-06
Ce-141	3.2E-14	2E-07	1.6E-07
Ce-144	4.9E-15	6E-09	8.1E-07
Pr-144	4.2E-15	5E-05	8.4E-11
W-187	3.7E-13	4E-06	9.2E-08
Np-239	5.2E-13	9E-07	5.8E-07
		Total DAC Fraction	1.3E-04

Table 12.2-28: Turbine Building Airborne Activity

Isotope	Airborne Concentration [µCi/mL]	10 CFR 20 DAC Value [µCi/mL]	DAC Fraction
Class 1			
Kr-83m	2.5E-09	1E-02	2.5E-07
Kr-85m	7.6E-11	2E-05	3.8E-06
Kr-85	2.1E-11	1E-04	2.1E-07
Kr-87	4.2E-10	5E-06	8.4E-05
Kr-88	2.8E-10	2E-06	1.4E-04
Kr-89	1.0E-08	1E-07	1.0E-01
Xe-131m	1.7E-11	4E-04	4.3E-08
Xe-133m	1.0E-11	1E-04	1.0E-07
Xe-133	1.6E-10	1E-04	1.6E-06
Xe-135m	9.6E-10	9E-06	1.1E-04
Xe-135	1.5E-09	1E-05	1.5E-04
Xe-137	5.9E-10	1E-07	5.9E-03
Xe-138	4.5E-09	4E-06	1.1E-03
Class 2			
I-131	7.7E-12	2E-08	3.8E-04
I-132	6.3E-11	3E-06	2.1E-05
I-133	5.6E-11	1E-07	5.6E-04
I-134	1.4E-10	2E-05	7.1E-06
I-135	1.0E-10	7E-07	1.5E-04
Class 3			
Rb-89	2.7E-12	6E-05	4.5E-08
Cs-134	1.2E-13	4E-08	3.0E-06
Cs-136	9.2E-14	3E-07	3.1E-07
Cs-137	1.9E-13	6E-08	3.1E-06
Cs-138	4.6E-12	2E-05	2.3E-07
Ba-137m	1.4E-14	1E-07	1.4E-07
Class 5			
H-3	7.4E-08	2E-05	3.7E-03
Class 6			
Na-24	4.1E-12	2E-06	2.1E-06
P-32	1.7E-13	2E-07	8.5E-07
Cr-51	4.3E-12	8E-06	5.3E-07
Mn-54	2.1E-12	3E-07	7.1E-06

Isotope	Airborne Concentration [µCi/mL]	10 CFR 20 DAC Value [µCi/mL]	DAC Fraction
Class 6 (continued)			
Mn-56	6.8E-12	6E-06	1.1E-06
Fe-55	4.4E-12	8E-07	5.5E-06
Fe-59	1.1E-12	1E-07	1.1E-05
Co-58	1.0E-12	3E-07	3.4E-06
Co-60	2.0E-12	1E-08	2.0E-04
Ni-63	4.4E-15	3E-07	1.5E-08
Cu-64	2.1E-11	9E-06	2.3E-06
Zn-65	9.3E-13	1E-07	9.3E-06
Sr-89	1.3E-14	6E-08	2.2E-07
Sr-90	6.6E-16	2E-09	3.3E-07
Y-90	6.5E-16	3E-07	2.2E-09
Sr-91	7.6E-12	1E-06	7.6E-06
Sr-92	1.4E-11	3E-06	4.8E-06
Y-91	1.7E-13	5E-08	3.4E-06
Y-92	4.6E-12	3E-06	1.5E-06
Y-93	5.4E-13	1E-06	5.4E-07
Zr-95	3.6E-13	5E-08	7.1E-06
Nb-95	3.6E-13	5E-07	7.1E-07
Mo-99	1.7E-12	6E-07	2.8E-06
Tc-99m	1.6E-12	6E-05	2.6E-08
Ru-103	8.7E-14	3E-07	2.9E-07
Rh-103m	5.6E-14	5E-04	1.1E-10
Ru-106	1.3E-14	5E-09	2.6E-06
Rh-106	1.3E-14	1E-07	1.3E-07
Ag-110m	4.4E-15	4E-08	1.1E-07
Te-129m	1.7E-13	1E-07	1.7E-06
Te-131m	2.2E-13	2E-07	1.1E-06
Te-132	4.1E-14	9E-08	4.6E-07
Ba-140	1.8E-12	6E-07	3.1E-06
La-140	1.8E-12	5E-07	3.7E-06
Ce-141	8.7E-14	2E-07	4.3E-07
Ce-144	1.3E-14	6E-09	2.2E-06
Pr-144	4.8E-15	5E-05	9.5E-11
W-187	9.8E-13	4E-06	2.4E-07
Np-239	1.4E-12	9E-07	1.5E-06
		Total DAC Fraction	1.17E-01

12.3 Radiation Protection Design Features

The design features incorporated in the BWRX-300 are based on GE Vernova Hitachi Nuclear Energy (GVH) BWR experience and consistent with the policy and design considerations discussed in Section 12.1.

12.3.1 Facility Design Features

The BWRX-300 plant design features, including equipment design, material selection, plant layout, and shielding, are provided consistent with the design considerations described in Subsection 12.1.2. Together with implementation of the Radiation Protection Program described in Section 12.5, these plant design features ensure occupational radiation exposures are maintained ALARA.

12.3.1.1 Equipment Design Features

BWRX-300 equipment is designed to aid in maintaining exposure to plant personnel during system operation and maintenance ALARA and to minimize radioactive contamination as described in Subsection 12.1.2. The equipment design, including material selection, is consistent with the features described within this section.

Tanks

The overflow pipes of the tanks that hold radioactive liquid runs to the appropriate radioactive drain collection system, and the vents run to the building's ventilation exhaust system, but neither is hard connected to their respective drain or vent systems to prevent the uncontrolled transport of contamination through those systems. Non-corrosion-resistant material tanks that contain radioactive material have a corrosion resistant lining. Walls or curbs are provided around the tank to collect leaks. Room collection capacity is sufficient to contain the maximum amount of liquid for which the tank is designed. The spent resin, phase separator, and backwash tanks are located in enclosures free of flammable or combustible substances. Bottom valves on slurry containing tanks are flush mounted to eliminate pockets.

Heat Exchangers in Radioactive Service

Tube and shell HXs are designed with an excess of tubes to accommodate tube plugging, when necessary. The pressure of the clean fluid is greater to prevent active fluid leaks from entering the clean side for highly contaminated fluid. HX design allows for drainage of fluids from the exchanger and avoid pooling that could lead to radioactive crud deposition. HX drains are provided and are hard piped to floor drains. An air gap is provided at the connection of the drain lines and floor drains. There are cleaning connections for condensate, demineralized water, and/or chemical solutions as necessary.

Piping

Piping is selected to provide a service life equivalent to the design life of the plant, with consideration given to corrosion allowances and environmental conditions. Contaminated piping systems are welded to the extent possible. Backing rings or consumable inserts are not used for systems that contain highly radioactive fluids or that contain high solids content (i.e., spent resin or filter sludge). The piping, where possible, is constructed of seamless pipe as a means to reduce radioactivity accumulation on seams. Radioactive piping is sized to produce turbulent flow to maintain solids suspension without plugging. Drains and flushing provisions are employed as necessary to reduce the effect of “dead legs” and low points. In systems where the settling of contamination can occur, the drains are on the side of the piping, not the bottom of the pipe, or not installed at all. Systems that contain multiple pumps generally are designed with pumps in

separate alcoves with piping routed into shielded pipe chases as necessary. Radioactive piping is run in a manner that minimizes radiation exposure to plant personnel. This involves:

- Radioactive pipe routing in corridors is minimized. For situations in which radioactive piping are routed through corridors or other low radiation areas, an analysis is conducted to confirm this routing does not compromise the existing radiation zoning.
- Piping that may contain post-accident radiation sources is not routed in access routes, including stairways, hallways, and are not routed through emergency evacuation routes
- Use of shielded pipe trenches and pipe chases, where routing of high-activity pipes in low-level areas cannot be avoided
- Separating radioactive and non-radioactive pipes for maintenance purposes
- The arrangement of piping incorporates safety division separation, logical grouping of components within a system, system operability, and maintenance access requirements
- Lines for non-radioactive systems are not run inside biologically shielded compartments except where needed to supply services inside the compartment
- Piping configurations are designed to minimize the number of “dead legs” and low points to avoid accumulation of radioactive crud and fluids in the line

Provisions are made in radioactive systems for venting and flushing with condensate or chemical cleaning agents to reduce crud buildup. Instrument lines, except those for the reactor vessel, are designed with provisions for backflushing and maintaining a clean fill in the sensing lines, where appropriate. When needed (i.e., not continuously), the reactor vessel sensing lines are flushed with condensate from the instrument rack when the reactor vessel is depressurized. A distinction is made between piping conveying radioactive and non-radioactive fluids, and separate routing through shielded pipe chases is provided whenever possible. Piping conveying highly contaminated fluids is usually routed through shielded pipe chases and shielded cubicles. However, when these options are not feasible, the radioactive pressurized process piping is sleeved when embedded in concrete walls and floors. Pipes connected to floor drains are butt-welded stainless steel if embedded. Contaminated piping is cleanable or removable for decommissioning. “Clean” services, such as compressed air and demineralized water are not routed through shielded pipe chases, where possible. Where piping penetrates through walls (e.g., feedthroughs with short sections), the designs minimize the use of embedded piping to the extent possible to facilitate the dismantlement of the systems and the decommissioning of the facility. When radioactive and non-radioactive pipes are collocated, drain provisions are provided to remove radioactive fluids, and the valves are controlled remotely, without need for entering the cubicle. Penetrations for piping through shield walls are designed to minimize shine on surrounding areas. Underground piping is not installed unless it is only used for clean service. Otherwise, it is either enclosed within a guard pipe and monitored for leakage, or accessible for visual inspections via a trench or tunnel. Threaded and flanged connections are kept to a minimum. Other joints are welded or otherwise permanently bonded depending on the piping material.

Ion Exchange and Activated Carbon Beds and Filters

Filters in purification and cleaning systems are operated and cleaned remotely. For purification systems, the preparation and changing of resins or removal of activated carbon is performed remotely. Remotely adding new activated carbon is not required if access to the filter housing does not pose a significant risk (i.e., after contaminated filters have been removed). A mesh to trap resins or particles is installed downstream of vessels containing resins or activated carbon. Sampling points are installed at the inlet and outlet of, and differential pressure is measured

across filters, ion exchange vessels, and activated carbon beds. Where cartridge-type filters are used, they are sized such that each can independently provide the needed filtration for one fuel cycle with no changeout of the cartridges or using duplex filters with on-line exchanges using shielded housing and special ALARA tools.

Pumps and Valves in Radioactive Service

Pumps located in radiation areas are designed to minimize the time required for maintenance. Features such as quick-change cartridge-type seals on pumps, and pumps with back pullout features that permit removal of the pump impeller or mechanical seals without disassembly of attached piping, are employed to minimize exposure time during pump maintenance. Where two or more pumps conveying highly radioactive fluids are to be located adjacent to each other, shielding is provided between them to maintain exposure levels ALARA. Pumps adjacent to other highly radioactive equipment are also shielded to reduce the maintenance exposure (e.g., radwaste system). Pump control instrumentation for process equipment is located outside high radiation areas in separate alcoves, and motor or pneumatic-operated valves or valve extension stems are employed to allow operation from these areas. Operation of the pumps and associated valves for radioactive systems is accomplished remotely through reach rods or electric controls. Pump casing drains are provided on radioactive systems whenever possible to remove fluids from the pump prior to disassembly. Pump drains are piped to the radioactive drain system. Applications where valves with packing are used have cartridge-type packing that can be easily replaced, otherwise valves without packing are utilized, like diaphragm valves or quarter-turn ball valves. Instrumentation and valves in shielded cubicles are remotely operable to the maximum extent possible to reduce the need for entering these high radiation areas. Straight-through valve configurations are selected where practical, over those that exhibit flow discontinuities or internal crevices to minimize crud trapping. Valves are preferably placed in horizontal runs to facilitate dismantling of their top parts. For systems in high radiation areas, gasket materials are as degradation-mechanism resistant as possible. Degradation mechanisms (e.g., stress corrosion, thermal aging, embrittlement, fatigue, thermal fatigue) are considered jointly during the material selection process. Wherever possible, valves in systems containing radioactive fluids are separated from those for “clean” services to reduce the radiation exposure from adjacent valves and piping during maintenance. Remotely operated valves are employed in high radiation areas, whenever possible, to minimize the need for entering these areas, otherwise provisions are available to remove the source term before the valves are operated. Resources are provided for flushing the valves in radioactive systems, in order to reduce exposure to personnel during maintenance. Safety and relief valves and depressurization valves are designed with flange connections to allow whole valve removal or reinstallation. Isolation valves are located in such a way that, once they are closed, fluid contained in dead branch-pipes is minimized. Back seat valves are used wherever possible to reduce leaks through the packing when the valve is full open. Pipe insulation around valves is easily removable.

Radioactive Drains

For systems containing highly radioactive fluids, drains that are used frequently are hard piped to the appropriate drain. An air gap exists between the system drain and the floor or equipment drain hub. For system drains used less frequently, only a drain connection with a valve is provided so plant personnel can connect a hose and direct the outflow to the nearest appropriate drain. Sump vents are located near the room exhaust vent register to control airborne radioactivity released from discharges to the sump. Sumps are stainless steel or stainless steel lined to reduce corrosion and to provide easily decontaminated surfaces. The pipes carrying fluid to radioactive drains and clean drains are separate and independent. The resin slurry drains are separated from other drains because of their high-activity level, and their contents are routed to the RWB for subsequent treatment. To the extent possible, drain lines are sloped to drain by gravity. Drain

lines having a potential for containing highly radioactive fluids are routed through pipe chases or shielded cubicles. Pipes are used to drain fluids from equipment and component decontamination areas. Floor drains are one piece to minimize possibility of liquid penetrating at embedment boundaries. Non-porous alternatives to grout such as epoxy is considered for the installation of floor drains. Drain lines contain traps to prevent the flow of radioactive gas from compartment to compartment through the drain system. The drains from rooms that house spent resin tanks, phase separator tanks, or demineralizer vessels are equipped with normally closed isolation valves. Sumps are located at the lowest level of the buildings.

Sludge and Resin Systems

System design and implementation prevent accumulating and retention of radioactive sludge and resins in transfer pipes. For this purpose:

- The length of the pipe runs is reduced as much as possible
- Low points and blind sections are avoided
- The flow, as far as possible, is by gravity, so lines are designed with the appropriate slope, typically 1/4 to 1/2 in./ft
- Flow restrictions in the pipes are avoided
- The use of short 90º elbows is avoided whenever possible. They are replaced by large-radius curved sections at least five times the diameter or two 45º elbows spaced apart from each other.
- Whenever piping Tee connections or fittings are required, the main flow runs through the straight section. If the secondary flow is into the tee, then the connection is made above the axis of the pipe. If the secondary flow is out of the tee, then the connection is on the side (lateral connector).

Material Selection

Materials are selected to aide with decontamination and decommissioning efforts and maintain worker exposures ALARA. Key material design considerations include:

- The reactor coolant system materials are selected to prevent the formation of corrosion products. Reactor internal components, except for the zircaloy in the reactor core, are stress corrosion-resistant stainless steels or other high alloy steels.
- Carbon steel used in systems processing or storing reactor coolant or steam is low in nickel content and contains only very small amounts of cobalt impurity
- Main condenser system tubes and closed loop cooling HXs (tubes and their tube sheets) are made of low-cobalt stainless steel or titanium alloys and low-cobalt materials
- Materials in contact with reactor coolant and subject to significant wear, corrosion, erosion, or neutron activation, except the fuel assemblies, use the lowest cobalt content available with a target cobalt content of 0.02 weight percent or less for stainless steel or nickel base alloys, and with no stainless steel or nickel base alloys exceeding a cobalt content of 0.05 weight percent

- Nickel-Chrome-Iron alloys such as Alloy 600M and Alloy X-750, which have high nickel content, are used in some reactor vessel internal components. These materials are used in applications where special requirements (possessing specific thermal expansion characteristics along with adequate corrosion resistance) are necessary, and where no suitable alternative low-nickel material is available. Cobalt content in the Ally X-750 used in the fuel assemblies is limited to 0.05%.
- The BWRX-300 is designed to limit the use of cobalt-bearing materials on moving components that have historically been identified as major sources of reactor coolant contamination. Stellite is typically used for hard facing of components that are extremely wear resistant. Use of high cobalt alloys such as Stellite is restricted to those applications where no satisfactory alternative material is available.
- Control rod blade material is selected to use the lowest cobalt content available with a target cobalt content of 0.02 weight percent or less for stainless steel or nickel base alloys, and with no stainless steel or nickel base alloys exceeding a cobalt content of 0.05 weight percent. The average niobium content for the handle and absorber section, less boron carbide and hafnium, is less than 0.1 weight percent.
- Non-stainless-steel equipment has an adequate finish and is protected with a non-corrosive material to aid in decontamination
- For systems in contact with reactor coolant (e.g., pumps, valves), component materials are as degradation-mechanism resistant as possible. Degradation mechanisms (e.g., stress corrosion, thermal aging, embrittlement, fatigue, thermal fatigue) are considered jointly during the material selection process.
- System components are made of materials that are qualified to withstand the pressure, temperature, and radiation to which they are subject and thus minimize maintenance
- Pipes are designed for a 60 year life, based on the environmental conditions and corrosion in order to prevent maintenance, unless documented otherwise (e.g., abrasive slurry lines). Pipes with less than a 60 year life are replaceable (i.e., not embedded in concrete).
- Long life materials that can withstand the operating pressure, temperature, and radiation are used for component joints and seals in order to minimize maintenance
- HXs are constructed of stainless steel or titanium tubes to minimize the possibility of failure and reduce maintenance requirements
- Material selection minimizes the creation of corrosion products with the potential to be activated to the extent possible

12.3.1.2 Plant Layout Design Features

The layout of the BWRX-300 is designed to meet the ALARA design considerations described in Subsection 12.1.2. A preliminary plant layout is shown on Figures 12.3-1 through 12.3-19. Structures are designed to provide adequate shielding, as described in Subsection 12.3.2. The layout uses both distance and shielding to reduce potential dose rates in work locations, and areas are identified by radiation zones, as described in Subsection 12.3.1.3.

Facility layout is consistent with the following design features:

- From the point of view of radiological protection, the layout follows the concept of separation into three different and isolable areas: uncontrolled access areas, controlled (limited) access areas, and inaccessible areas. Table 12.3-1 further defines the different plant areas.

- The controlled access areas encompass radioactive systems, equipment, components, and materials
- The routes between buildings of the controlled (limited) access area and their accessible rooms are as short as possible, designed for an easy passage, and are run through low radiation areas
- Rooms that need to be accessed are reached easily from the service corridors
- The floor-ceiling height of accessible areas are adequate for comfortable passage of personnel and the necessary tools for inspections and maintenance
- There are areas for storing removable elements and mobile shielding that do not hinder personnel passage and mobility
- The areas of the plant are arranged in such a way that service operations, monitoring or inspection of highly radioactive equipment can be carried out manually or remotely
- Access to High Radiation Areas is suitably signposted and blocked by physical barriers
- Access to enclosures is sized to enable the entry and exit of the necessary components and tools
- Equipment, instruments, and sampling stations that require routine maintenance, calibration, operation, or inspection, are located for ease of access and minimum required occupancy time in radiation areas
- Plant areas are laid out to allow remote or mechanical operation, service, monitoring, or inspection of highly radioactive equipment
- Where possible, transportation equipment is provided (e.g., monorails, carts, hoists) to lower radiation areas for equipment or components requiring service
- Means to control contamination and to facilitate decontamination of potentially contaminated areas are provided where practicable
- Exposure of personnel during inspection and maintenance is minimized by locating equipment and instrumentation as far as possible from potential sources of high radiation
- Penetrations through outer walls of the building containing radiation sources are sealed to prevent leaks into the environment
- Appropriately sloped floor drains are provided in shielded cubicles and other areas where the potential for a spill exists to limit the extent of contamination
- Doors are installed, as necessary, to prevent open passage to High Radiation Areas
- Doors have the appropriate radiation area signs and locks to prevent inadvertent access to high radiation rooms
- The paint and finish of room walls, floors, staircases, and platforms are easy to decontaminate. The paint provides a non-porous surface.

12.3.1.3 Radiation Zones and Access Requirements

The BWRX-300 radiation zone design dose rate objectives are established and assigned based on:

1. Expected frequency of access and,
2. Estimated relative source strengths in the various plant rooms.

Operating activities, inspection requirements of equipment, maintenance activities, and abnormal operating conditions are considered in determining the appropriate radiation zone design dose rate objectives for a given area.

Areas of the plant have at least two assigned radiation zone classifications. The first classification is based on normal operation radiation fields, and the second classification is based on the normal shutdown radiation fields. Radiation zone classifications are described within this subsection and shown on Table 12.3-1, and the radiation zone design dose rate objectives established for plant areas are shown on Figures 12.3-1 through 12.3-19. Plant features are not depicted to scale and areas not shown in these figures are non-radiological control zones.

The dose rate applicable for a particular zone is based on OPEX and represents design dose rate objective. BWR plants have been operating for several decades, and OPEX with similar design basis numbers shows that only a small fraction of the permissible dose is received from radiation sources controlled by equipment layout, or the structural shielding provided. Accordingly, the dose rates for a particular zone should not be interpreted as the expected dose rates that apply in all portions of that zone, or for all types of work within that zone, or at all periods of entry into the zone.

Access to areas in the plant is controlled and regulated by the given area zone in accordance with 10 CFR 20.1601 and 10 CFR 20.1602 requirements. Areas with dose rates such that an individual could receive a dose in excess of 100 mrem in a period of one hour are locked and posted with "High Radiation Area" signs. Areas where an individual could receive a dose of 500 rads or more within a period of one hour at one meter from a radiation source or one meter from any surface that the radiation penetrates are posted with "Very High Radiation Area" signs.

The following classifications for radiation zones are used:

Class A Radiation Zones

These zones are non-radiological controlled zones with and have free access and unlimited stay time. The areas outside the controlled area, including electrical equipment rooms, Main Control Room (MCR), Secondary Control Room (SCR) and administrative areas are Class A radiation zones. The radiation dose rate in these areas can reach a maximum of 0.6 mrem/hr.

Class B Radiation Zones

These zones have restricted access and unlimited stay time. This zone corresponds to the corridors, staircases, control panel areas, and areas around rooms with radioactive material. These areas do not have radioactive equipment or components. Shielding is provided to allow for continuous occupancy. The radiation dose rate in these areas can reach 1.0 mrem/hr, and there is no radioactive or environmental contamination. Access to these areas is through the entry control point, and rooms with a higher radiation zone class are accessed from this control point.

Class C Radiation Zones

These zones have restricted access and stay time limited to 20 hr/wk. These zones are areas of frequent access, such as sampling stations and valve operating rooms. The dose rate in contact with the existing equipment is not greater than 5.0 mrem/hr, and there is a possibility of radioactive contamination and/or airborne radiation.

Class D Radiation Zones

These zones have restricted access and stay time is limited to 4 hr/wk. These areas are infrequently accessed, such as rooms for valves or equipment where the dose rate in contact is

no greater than 25 mrem/hr. There is a possibility of radioactive contamination and/or airborne radiation.

Class E Radiation Zones

These zones have restricted access and stay time is limited to 1 hr/wk. This zone corresponds to areas of occasional access such as rooms for pumps, valves, HXs, and tanks that require limited access for the actuation of various devices. The contact dose rate with equipment is no greater than 100 mrem/hr, but the radiation level can be reduced without significantly altering the plant operating mode. There is a possibility of radioactive contamination and/or airborne radiation.

Classes F to I Radiation Zones

Classes F to I radiation zones are high radiation zones that are areas accessible to individuals where radiation levels from radiation sources external to the body could result in an individual receiving a dose equivalent in excess of 0.1 rem in 1 hour at 1 ft. from the radiation source or 1 ft. from any surface that the radiation penetrates.

Class J Radiation Zone

Class J radiation zone is a very high radiation zone in which radiation levels from radiation sources external to the body could result in an individual receiving a dose equivalent 500 rem or more within a period of one hour at one meter from the radiation source or one meter from any surface that the radiation penetrates. Such areas are inaccessible during power operations.

Access to Class F and higher zones is not permitted without specific authorization. This corresponds to equipment rooms that do not require attention for actuation, inspection, and/or maintenance, such as tanks, filters, and pipes with radiation levels greater than 100 mrem/hr. Their radiation level cannot be reduced without significantly altering plant operation.

Typical non-accessible Class F to J, High and Very High radiation areas include the reactor vessel area, containment during power operation, the areas of the RB with numerous or large unshielded containment penetrations, the areas of spent fuel transfer whenever it is present, the fuel pool, the steam pipe tunnel during power operation, and areas for demineralizer filters and resin wash water collection tanks.

12.3.2 Shielding

The permanent shielding analysis is completed during the detailed design phase. The evaluations identify those areas experiencing elevated dose rates during power and shutdown conditions and are not mitigated by structure or component design or location. The anticipated dose rates and frequency of access are included in the shielding analyses. Using these results and system interaction and structural/component loading analyses, permanent shielding is installed in target areas/components. Permanent shielding is preferred to temporary shielding.

12.3.2.1 Shielding Design Method

The primary objective of radiation shielding is protecting operating personnel and the general public from radiation emanating from the reactor vessel, the power conversion systems, the radwaste process systems, and the auxiliary systems, while maintaining appropriate access for operation and maintenance. Source terms in various pieces of plant equipment are cited in Section 12.2. Shutdown conditions, such as fuel transfer operation, as well as accident conditions are considered in designing shielding for the plant. Radiation shielding keeps radiation doses to equipment below levels where disabling radiation damage occurs. Systems containing radioactivity are identified and shielded based on access and exposure level requirements of surrounding areas. The radiation zone areas described in Subsection 12.3.1.3 indicate design

radiation levels where equipment shielding contributes to achieving the dose rate in the area (i.e., radiation zones are shielding requirements).

The shielding requirements in the plant perform the following functions:

- Limit the general public, plant personnel, contractors, and visitors' exposure ALARA and within 10 CFR 20.1201 and 10 CFR 20.1301 limits
- Limit critical components radiation exposure within specified radiation tolerances assuring component performance and design life are not impaired

The source terms used in the shielding calculations are analyzed conservatively. Transient conditions, shut-down, and normal operating conditions are used ensuring a conservative source is used in the analysis. Shielding is based on design basis fission product concentrations in the coolant in addition to activation products. For components where N-16 is the major radiation source, the design basis N-16 concentrations in reactor steam and coolant in Table 11.1-11 are used.

The mathematical models used to represent a radiation source, associated equipment and shielding are established to produce conservative calculation results. Depending on the versatility of the applicable computer program, various degrees of complexity for the actual physical situation are incorporated.

12.3.2.2 Shielding Design Criteria

Consistent with Subsection 12.1.2, design criteria are used in establishing the BWRX-300 shielding design to meet ALARA objectives.

- Containment - The major shielding structures consist of the RPV pedestal and containment walls:
  - The RPV pedestal consists of steel plate composite modules with diaphragm plates of appropriate thickness.
  - The containment outer wall is a reinforced concrete cylinder of steel plate composite modules with diaphragm plates, of appropriate thickness, that totally surrounds containment. A reinforced concrete top slab, of appropriate thickness, covers containment.
  - The shielding for the RB is designed to maintain areas requiring regular daily access to dose rates less than 6 μSv/h (0.6 mrem/h.
  - The design includes necessary shielding provisions in the upper containment in order to reduce the dose ALARA during transfer of irradiated spent fuel assemblies.
  - The design includes applicable shielding provisions to minimize dose rates in case of a fuel handling mishap resulting in dropping a fuel assembly across the reactor vessel flange.
- Penetrations through the containment wall are shielded to reduce radiation streaming. Localized dose rates outside these penetrations are limited to less than 50 μSv/h (5 mrem/h). The penetrations through interior shield walls of the RB are shielded using a modified high-density silicone elastomer, or equivalent absorber, to reduce radiation streaming.

- The dose rate in the Main Control Room (MCR) and SCR is limited to 6 μSv/h (0.6 mrem/h) during normal reactor operating conditions. The outer walls of the Control Building (CB) are designed to attenuate radiation from radioactive materials contained within the RB and from possible airborne radiation in the environment surrounding the CB following an accident.
- Processing equipment is located as far as possible from personnel dwelling and passage areas minimizing shielding. When required to meet ALARA objectives, shielded labyrinths are used to eliminate radiation streaming through access ways from sources located in cubicles.
- Penetrations through shield walls are shielded reducing radiation streaming through the penetrations.
- Wherever possible, radioactive piping is run in a manner minimizing plant personnel radiation exposure that includes:
  - Minimizing radioactive pipe routing in corridors
  - Avoiding routing high-activity pipes through low-radiation zones
  - Using shielded pipe trenches and pipe chases where routing of high-activity pipes in low-level areas cannot be avoided
  - Separating radioactive and non-radioactive pipes for maintenance
- Maintaining acceptable levels at valve stations by using motor-operated or diaphragm valves. Minimizing worker radiation exposure by providing draining and flushing provisions in the valve design. Operator shielding is provided for manual valve applications using shield walls or valve stem extensions.
- Shielding is provided for the MCR to mitigate plant personnel exposure resulting from radiation sources in containment following an accident
- Provisions are made for shielding major radiation sources during ISI to reduce personnel exposure. Steel platforms are provided for ISI of the RPV nozzle welds and associated piping.
- The primary material used for shielding is concrete. Concrete used for shielding purposes is designed in accordance with RG 1.69, “Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants.” Where special circumstances dictate, steel, lead, water, a modified high density silicone elastomer, or a boron-laced refractory material is used.

12.3.3 Ventilation

The Heating, Ventilation, and Cooling Systems (HVSs) are designed to ensure that airborne radiation exposures to plant personnel are ALARA. The HVS design is described in Section 9.4. The OGS and Process Radiation Monitoring Subsystem (PRM) are described in Section 11.3 and Section 11.5.

12.3.3.1 Ventilation Design Criteria

Consistent with Subsection 12.1.2, design criteria are used in establishing the BWRX-300 HVS design to meet ALARA objectives and for personnel protection:

- HVS for the fuel pool minimizes airborne radioactivity by sweeping the pool surface and preventing evaporative pool losses and gases from mixing with the area atmosphere
- The shielding wall thickness between HVS filter cubicles are sized so that the dose contribution in any cubicle from the filter in the adjacent one does not exceed 25 mrem/hr under normal operation
- The HVS limits the extent of airborne contamination by providing airflow patterns from areas of low contamination to more contaminated areas by using a slight negative pressure in the more contaminated area
- The HVS service in contaminated areas of the BWRX-300 plant conditions and circulates air through the contaminated areas of the buildings in a once-through fashion. Flow is generally directed from the corridors (point of highest pressure) to the equipment alcove rooms, then to the rooms themselves, then to the air ducts, and finally to the exhaust ductwork.
- The passage of ventilation ducts through walls and structures does not significantly reduce the efficiency of the shielding. HVS duct penetrations used for exhaust are routed above personnel head height to minimize personnel exposure by streaming effects. HVS duct penetrations are not located in line of sight between radiation sources and occupied areas.
- Buildings that contain radioactive materials are equipped with radiological monitoring connections (sample ports) in the exhaust ductwork prior to the combined installed monitoring instrumentation as a means to target radiological system leakages
- Fume hoods or similar devices are installed in laboratories, sampling rooms and component decontamination rooms
- HVS filters that are expected to become highly contaminated are shielded from one another and from areas that are accessible by personnel, with the shielding specified to minimize their exposure during maintenance

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

The Process Radiation and Environmental Monitoring System (PREMS) provides monitoring for both general area radiation and process and effluent streams to ensure that radiation and effluent exposures are ALARA and within regulatory limits. The PREMS consists of four subsystems: Area Radiation Monitoring Subsystem (ARM), Containment Monitoring Subsystem (CMon), PRM, and Process Sampling Subsystem (PS).

Fixed area radiation monitoring outside of containment is described within this subsection and is provided by the ARM. Airborne radioactivity in effluent releases and ventilation air exhausts are continuously sampled and monitored for noble gases, air particulates, and halogens in the PRM. The PRM is described in Section 11.5, and the PS is described in Subsection 9.3.2. Fixed area radiation monitoring within containment is provided by the Cmon.

The BWRX-300 design follows the guidance in RG 8.25, “Air Sampling in the Workplace” and has low levels of airborne radiation. Monitoring of airborne radiation inside the facility is accomplished with periodic grab sampling using portable air monitors, and when necessary, bioassay. Sources of airborne radioactivity are described in Subsection 12.2.2. Stairwells, corridors, areas routinely occupied by operating personnel, and areas of the facility designated for unlimited access

maintain airborne radioactivity concentrations well below 0.1 DAC (defined in 10 CFR 20 Appendix B Table 2). All controlled and limited access areas of the plant outside of containment maintain airborne radioactivity concentrations below 0.3 DAC using a combination of adequate fresh air supply to decrease concentrations of potential airborne radiation and by limiting sources of airborne activity in the area to the extent practical. In the unlikely event some area of the facility is unexpectedly subjected to high levels airborne radiation, the PRM ventilation exhaust monitors detect the condition.

12.3.4.1 Area Radiation Monitoring Subsystem

The ARM continuously measures, indicates and records gamma radiation levels at strategic locations throughout the plant, except within the primary containment, and activates alarms in the MCR, SCR, and in local areas to warn personnel to take appropriate protective measures.

12.3.4.1.1 Design Basis

Safety Category 1 Functions

None.

Safety Category 2 Functions

None.

Safety Category 3 Functions

The following functions meet the basis for a Safety Category 3 function because area radiation level is considered a Post-Accident Monitoring (PAM) Type E variable and radiation monitoring supports the guidelines of 10 CFR Part 20.

- The ARM monitors gamma radiation levels in strategic locations throughout the plant (excluding containment).
- The ARM monitors area radiation levels on the refueling floor.

Safety Category N Functions

None.

The Safety Classification, Quality Groups, and Seismic Categories of the ARM Structures, Systems, and Components (SSCs) are listed in Table 3A-1.

12.3.4.1.2 System Description

The ARM measures the gamma radiation levels at assigned locations within the plant, displays the measurements in the MCR, and initiates an MCR alarm, SCR alarm, and a local alarm (both visual and audible) when the radiation level exceeds a preset limit. Area radiation monitors are placed at key locations throughout the plant, where they can continuously monitor and assess gamma exposure conditions.

Area radiation monitors are located in accordance with the guidance in RG 8.8. ARM equipment is located in the RB, TB, RWB, CB, and service building.

The following criteria are considered for detector placement:

- Areas which require entry or exit, or both, that are normally or occasionally accessible which may experience significant increases in exposure rates
- Areas that are most likely to be occupied

- Areas where structural materials or equipment do not cause inadvertent shielding of the detector, ensuring accurate monitor response to increases in exposure rates
- Environmental conditions, including the range of temperature, pressure, and humidity are considered for the areas where the detector, or electronics, or both are located
- Placement in easy-access and uncluttered areas, such that minimal use of equipment is required to service the monitor and to allow for proper calibration

Area radiation monitor ranges are selected in accordance with RG 8.8. Ranges are selected based on expected exposure rates and radiation sources present. The following criteria are considered in selecting monitor ranges:

- The lowest expected exposure rate, and the desirability of monitoring such rate
- The expected exposure rate under normal operating conditions
- The expected maximum exposure rate under Anticipated Operational Occurrences (AOOs)
- The upper end of the scale is sufficiently high to have an on-scale reading for exposure rates up to ten times the expected maximum rate
- If using two channels to obtain a wide measurement range, their ranges overlap by at least one decade

12.3.4.1.3 Safety Evaluation

The ARM subsystem complies with the following General Design Criteria (GDC).

GDC 63 - Monitoring fuel and waste storage, as it relates to providing detection of radiation levels in areas associated with fuel storage and radioactive waste systems

The ARM subsystem monitors area radiation levels on the refueling floor and are located in the RB, TB, RWB, CB, and service building using the guidance in RG 8.8. The ARM subsystem displays the area radiation measurements in the MCR and initiates an MCR alarm and a local alarm (both visual and audible) when the radiation level exceeds a preset limit, providing means to initiate appropriate safety actions.

Additional information on GDC compliance is found in Section 3.1.

The ARM subsystem is designed to support compliance with the following regulations:

- 10 CFR 20.1101(b), as it relates to the use of engineering controls in maintaining doses ALARA
- 10 CFR Part 20, Subpart F, as it relates to radiological monitoring
- 10 CFR 50.34(f)(2)(xxvii), as it relates to monitoring in-plant radiation

These regulations are met, as the ARM subsystem is designed to conform with RG 8.2, “Administrative Practices in Radiation Surveys and Monitoring,” and RG 8.8, as it relates to information relevant to ensuring that occupational radiation exposures are ALARA, and RG 1.97, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants,” as it relates to providing indication of radiation levels on the refueling floor and measuring gamma radiation levels in areas throughout the plant.

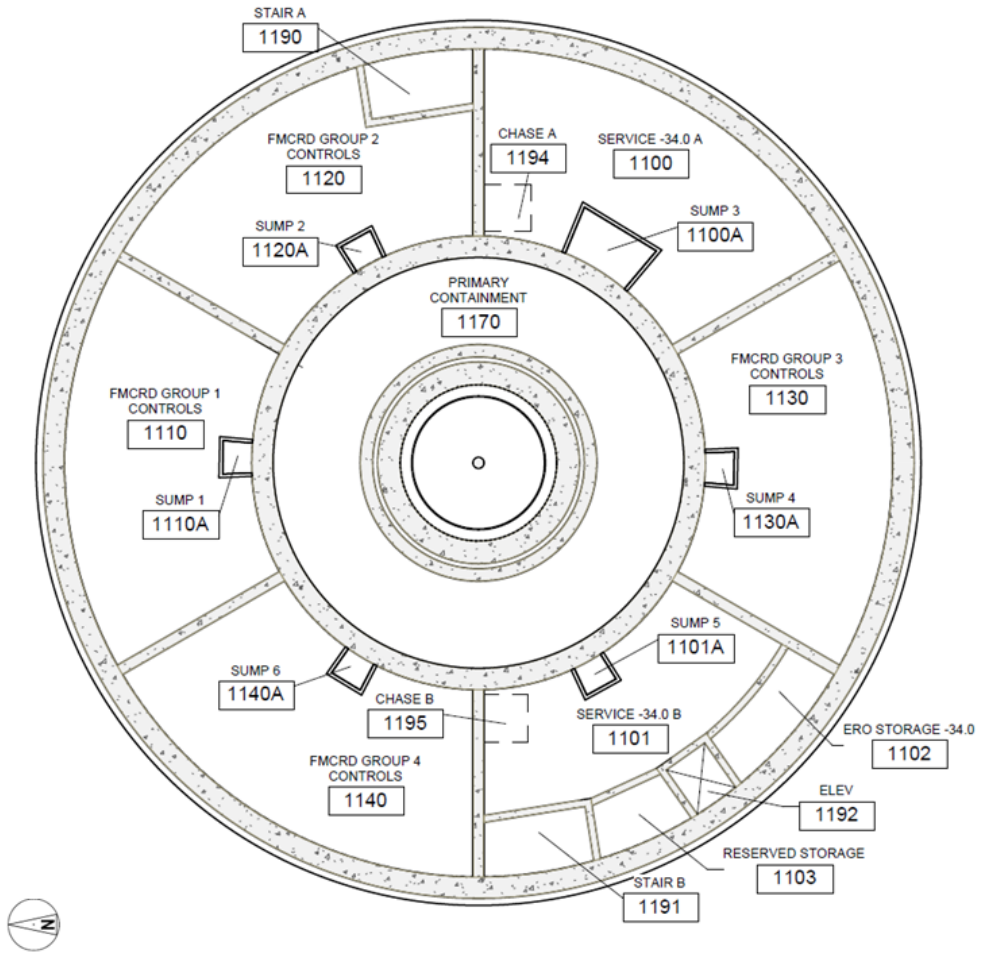
12.3.4.2 Containment Monitoring Subsystem

The CMon is described in Section 6.8.

Table 12.3-1: BWRX-300 Radiation Zone Classifications

Zone	Dose Rate	Stay Time Limit	Description	Zone Class
A	≤ 0.6 mrem/hr	None	Uncontrolled and unlimited access	Not a Radiation Area
B	≤ 1 mrem/hr	None	Controlled and unlimited access	
C	≤ 5 mrem/hr	20 hr/wk	Controlled and limited access	Limited Access Area
D	≤ 25 mrem/hr	4 hr/wk	Controlled and limited access	Radiation Area[1]
E	≤ 100 mrem/hr	1 hr/wk	Controlled and limited access	
F	≤ 1 rem/hr	5 hr/year	Controlled and limited access with special authorization permit required	High Radiation Area[1]
G	≤ 10 rem/hr	30 min/year	Controlled and limited access with special authorization permit required	
H	≤ 100 rem/hr	No access	Controlled and limited access with special authorization permit required	
I	≤ 500 rem/hr	No access	Controlled and limited access with special authorization permit required	
J	> 500 rem/hr	No access	Inaccessible	Very High Radiation Area[1]

1. The radiation area, high radiation area, and very high radiation area designations are assigned in accordance with the thresholds defined in 10 CFR 20.1003.

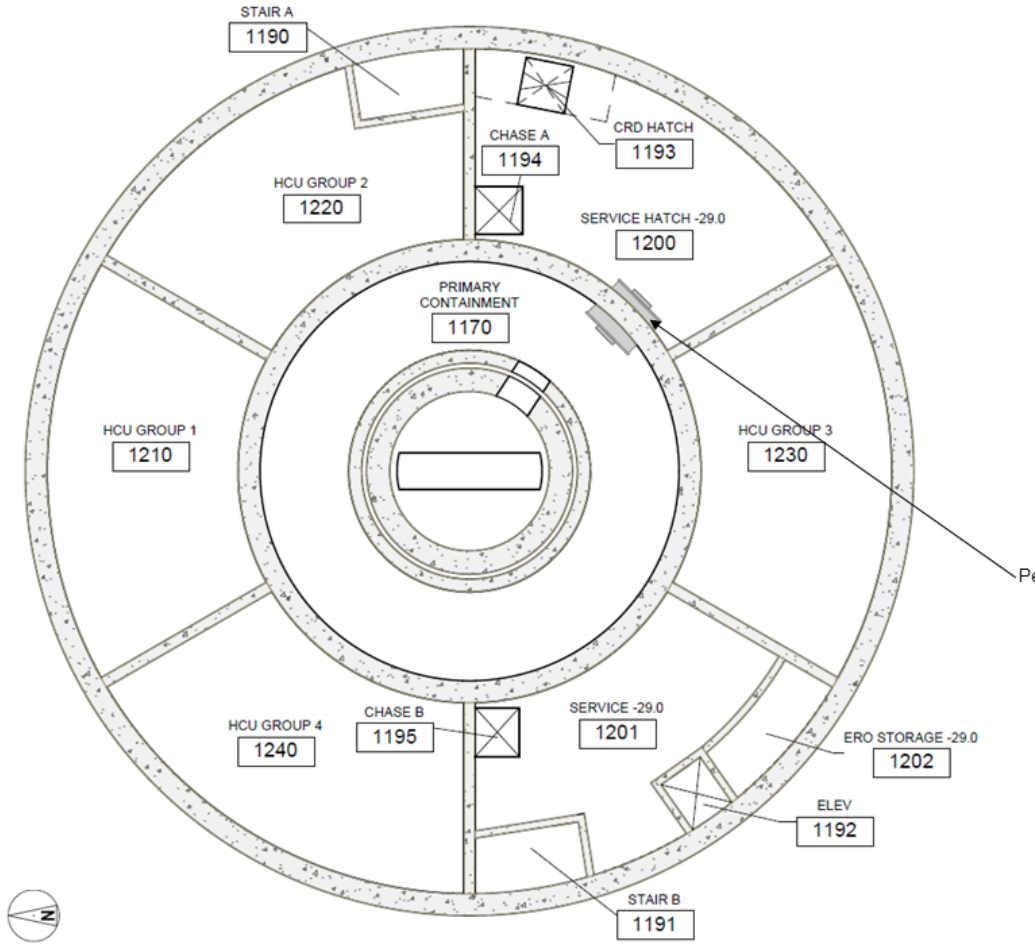


Radiation Zones Level -34.0m

ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
1100	SERVICE -34.0 A	B	A
1101	SERVICE -34.0 B	B	A
1102	STORAGE -34.0	B	A
1103	RESERVED STORAGE	B	A
1110	FMCRD GROUP 1 CONTROLS	B	A
1120	FMCRD GROUP 2 CONTROLS	B	A
1130	FMCRD GROUP 3 CONTROLS	B	A
1140	FMCRD GROUP 4 CONTROLS	B	A
1170	PRIMARY CONTAINMENT	J	G
1190	STAIR A	B	A
1191	STAIR B	B	A
1192	ELEVATOR	B	A
1194	CHASE A*	Unoccupied/Controlled Access	
1195	CHASE B*	Unoccupied/Controlled Access	
1100A	SUMP 3	D	C
1101A	SUMP 5	D	C
1110A	SUMP 1	D	C
1120A	SUMP 2	D	C
1130A	SUMP 4	D	C
1140A	SUMP 6	D	C

* Access to the chases requires an explicit authorization permit

Figure 12.3-1: Reactor Building Level -34.0 Meters Radiation Zones



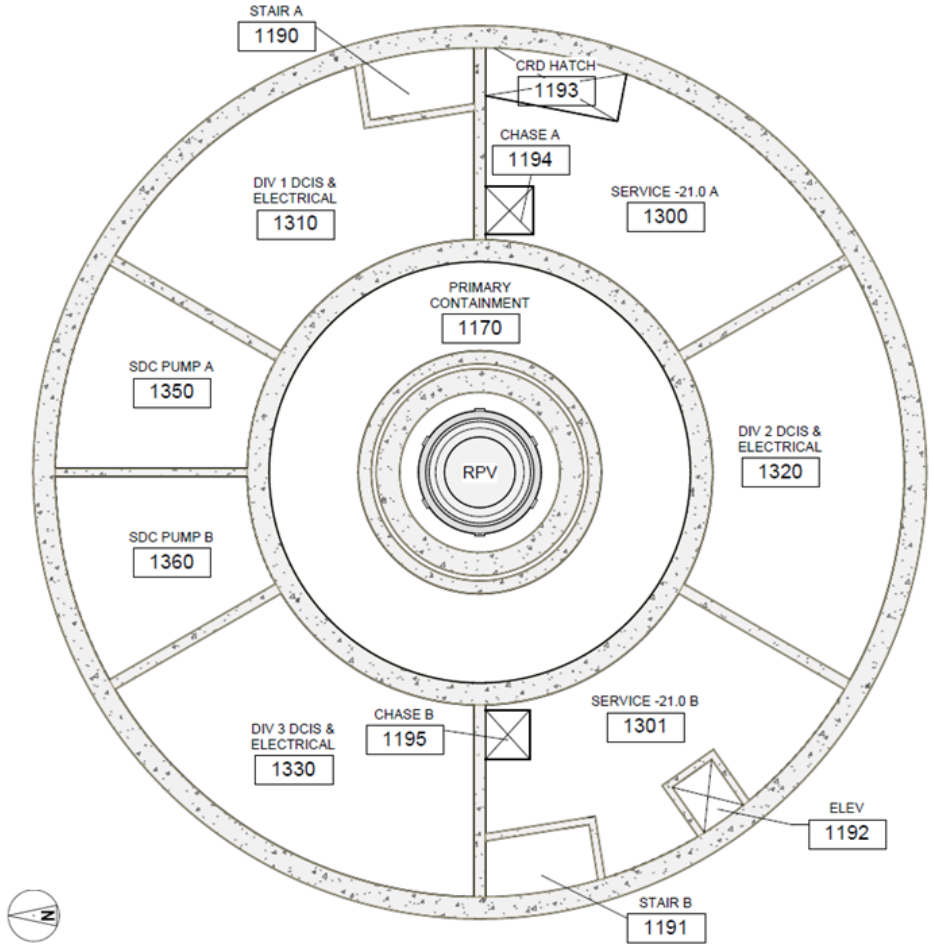
Personnel and Equipment Hatch is F / E zone.

Radiation Zones Level -29.0m

ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
1200	SERVICE HATCH -29.0	D	C
1201	SERVICE -29.0	C	B
1202	STORAGE	B	A
1210	HYDRAULIC CONTROL UNIT GROUP 1	B	A
1220	HYDRAULIC CONTROL UNIT GROUP 2	B	A
1230	HYDRAULIC CONTROL UNIT GROUP 3	B	A
1240	HYDRAULIC CONTROL UNIT GROUP 4	B	A
1170	PRIMARY CONTAINMENT	J	G
1190	STAIR A	B	A
1191	STAIR B	B	A
1192	ELEVATOR	B	A
1193	CONTROL ROD DRIVE HATCH	D	D
1194	CHASE A*	Unoccupied/Controlled Access	
1195	CHASE B*	Unoccupied/Controlled Access	

* Access to the chases requires an explicit authorization permit

Figure 12.3-2: Reactor Building Level -29.0 Meters Radiation Zones

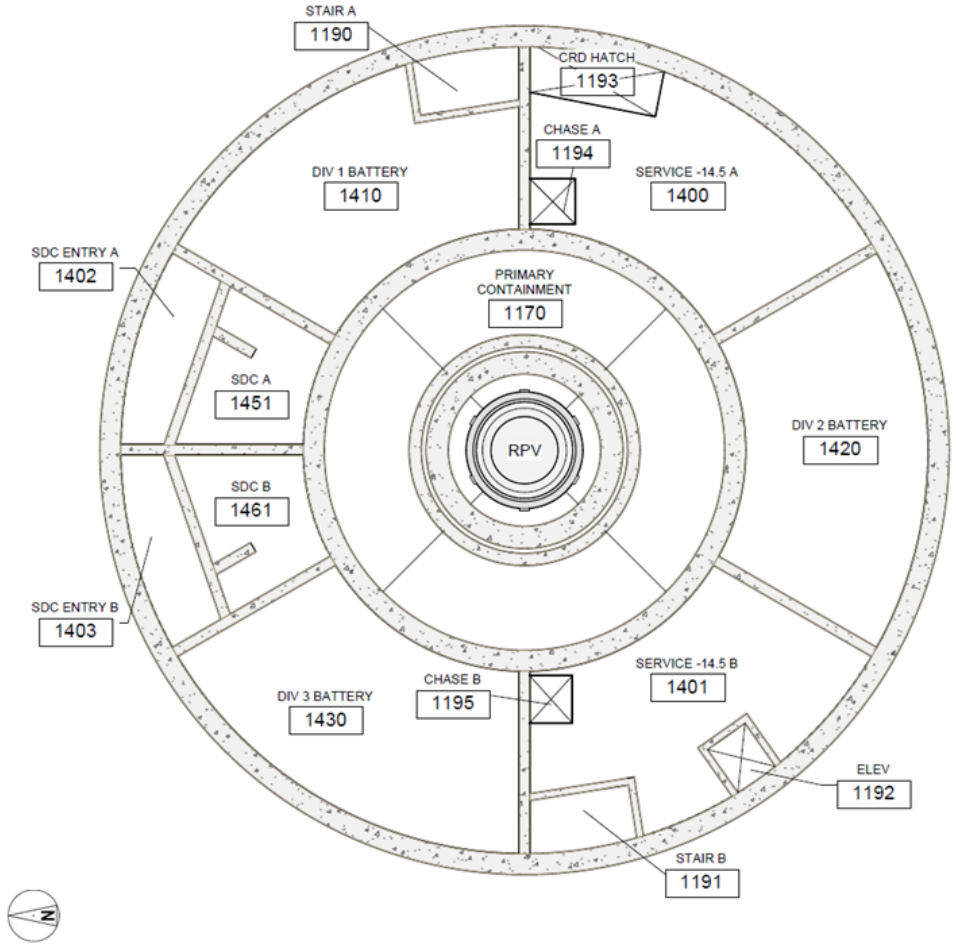


Radiation Zones Level -21.0m

ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
1300	SERVICE -21.0 A	B	A
1301	SERVICE -21.0 B	B	A
1310	DIV. 1 DCIS & ELECTRICAL	B	A
1320	DIV. 2 DCIS & ELECTRICAL	B	A
1330	DIV. 3 DCIS & ELECTRICAL	B	A
1350	SHUTDOWN COOLING PUMP A	C	E
1360	SHUTDOWN COOLING PUMP B	C	E
1170	PRIMARY CONTAINMENT	J	G
1190	STAIR A	B	A
1191	STAIR B	B	A
1192	ELEVATOR	B	A
1193	CONTROL ROD DRIVE HATCH	C	D
1194	CHASE A*	Unoccupied/Controlled Access	
1195	CHASE B*	Unoccupied/Controlled Access	

* Access to the chases requires an explicit authorization permit

Figure 12.3-3: Reactor Building Level -21.0 Meters Radiation Zones

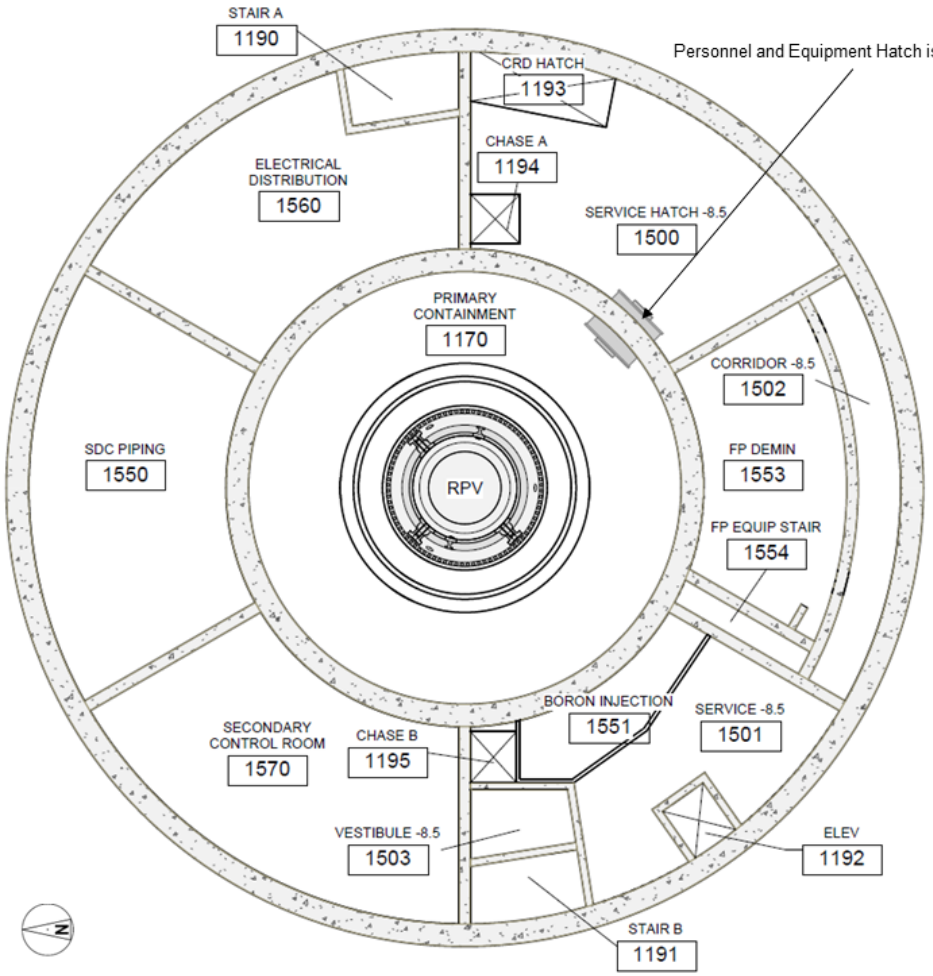


Radiation Zones Level -14.5m

ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
1400	SERVICE -14.5 A	B	A
1401	SERVICE -14.5 B	B	A
1402	SHUTDOWN COOLING ENTRY A	B	C
1403	SHUTDOWN COOLING ENTRY B	B	C
1410	DIVISION 1 BATTERY ROOM	B	A
1420	DIVISION 2 BATTERY ROOM	B	A
1430	DIVISION 3 BATTERY ROOM	B	A
1451	SHUTDOWN COOLING A	C	F
1461	SHUTDOWN COOLING B	C	F
1170	PRIMARY CONTAINMENT	J	G
1190	STAIR A	B	A
1191	STAIR B	B	A
1192	ELEVATOR	B	A
1193	CONTROL ROD DRIVE HATCH	C	D
1194	CHASE A*	Unoccupied/Controlled Access	
1195	CHASE B*	Unoccupied/Controlled Access	

* Access to the chases requires an explicit authorization permit

Figure 12.3-4: Reactor Building Level -14.5 Meters Radiation Zones



Radiation Zones Level -8.5m

ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
1500	SERVICE HATCH -8.5	D	C
1501	SERVICE -8.5	B	A
1502	CORRIDOR -8.5	B	A
1503	VESTIBULE -8.5	B	A
1550	SHUTDOWN COOLING PIPING	D	E
1551	BORON INJECTION	B	A
1553	FUEL POOL DEMIN	D	D
1554	FUEL POOL EQUIP STAIR	C	C
1560	ELECTRICAL DISTRIBUTION	B	A
1570	SECONDARY CONTROL ROOM	B	A
1170	PRIMARY CONTAINMENT	J	F
1190	STAIR A	B	A
1191	STAIR B	B	A
1192	ELEVATOR	B	A
1193	CONTROL ROD DRIVE HATCH	D	D
1194	CHASE A*	Unoccupied/Controlled Access	
1195	CHASE B*	Unoccupied/Controlled Access	

* Access to the chases requires an explicit authorization permit

Figure 12.3-5: Reactor Building Level -8.5 Meters Radiation Zones

Radiation Zones Level -4.8m

ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
1556	FUEL POOL EQUIP MEZZANINE	E	E

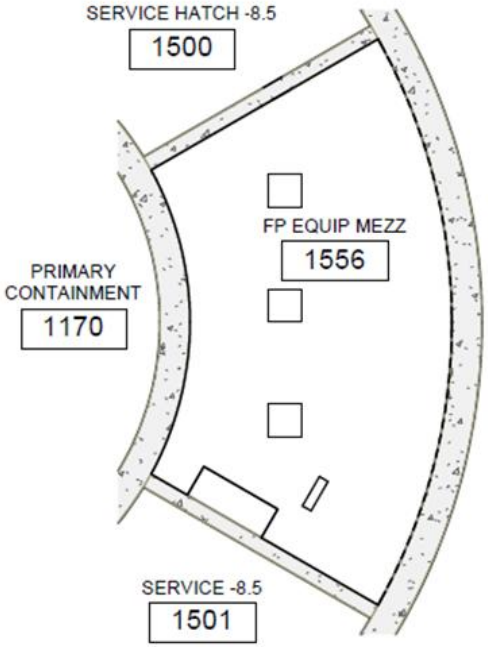
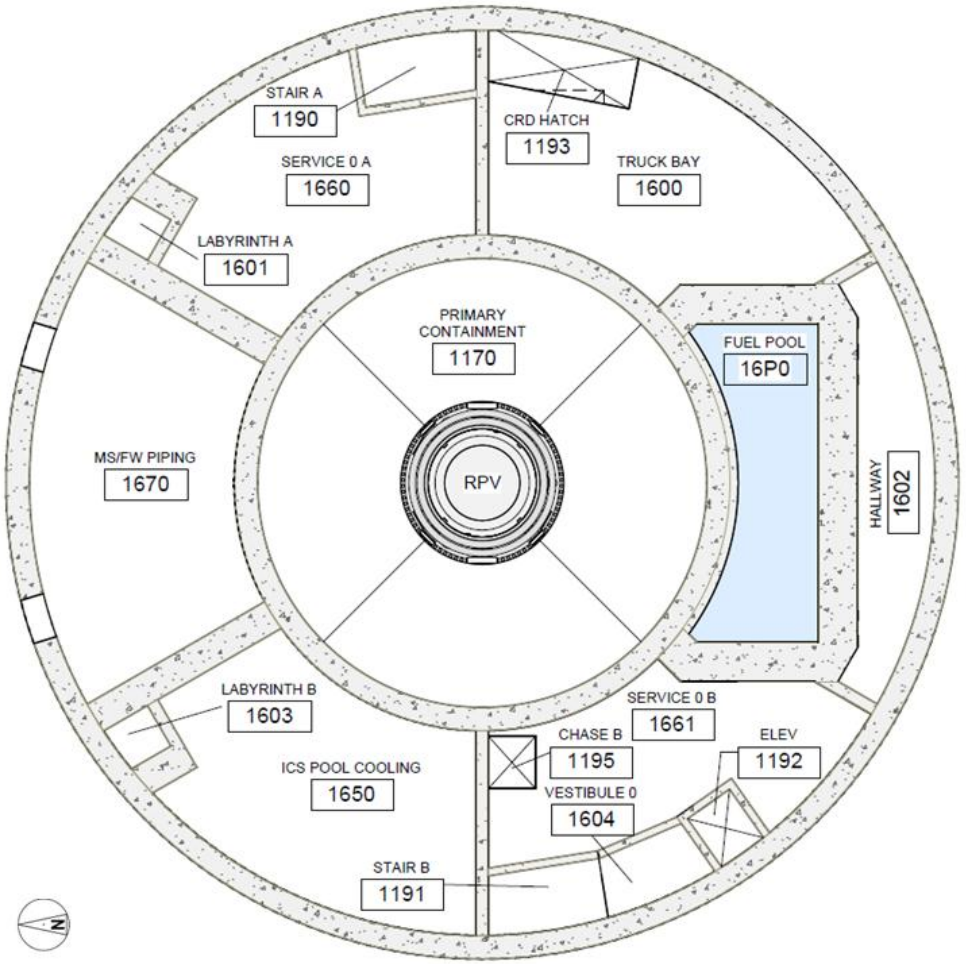


Figure 12.3-6: Reactor Building Level -4.8 Meters Radiation Zones

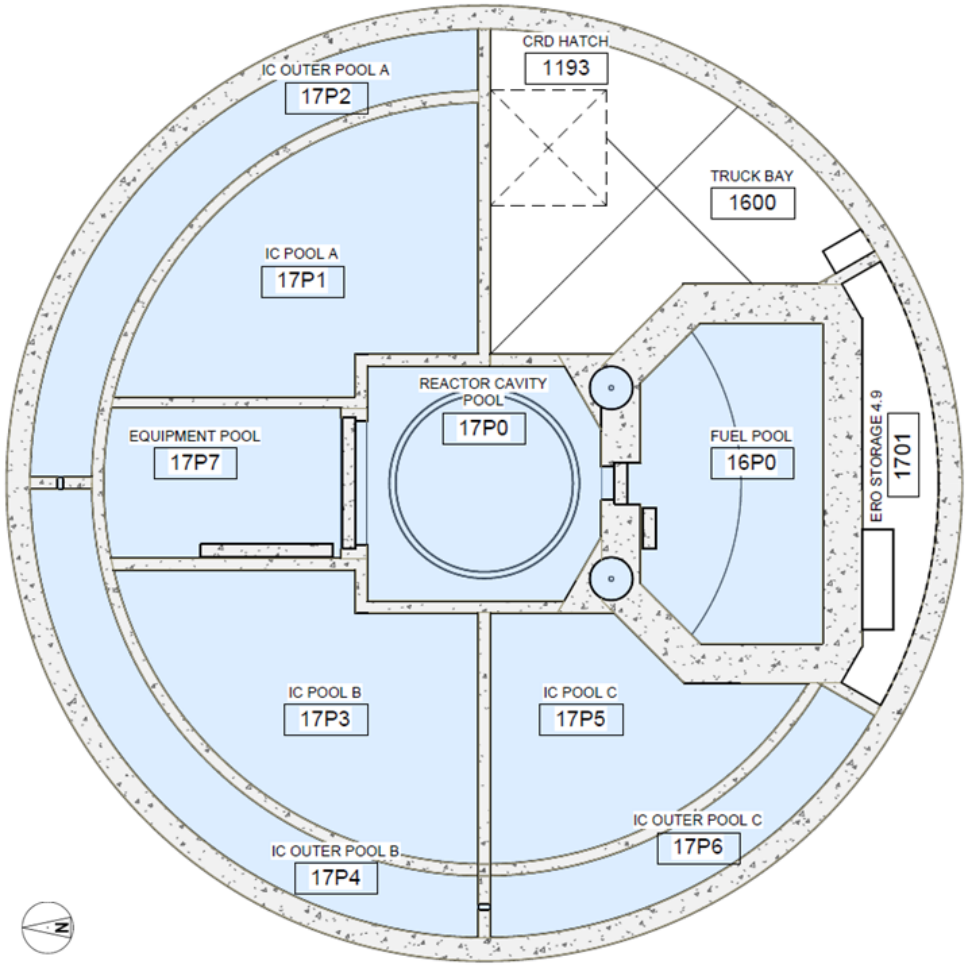


Radiation Zones Level 0m

ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
1600	TRUCK BAY	B	A
1601	LABYRINTH A	F	C
1602	HALLWAY	B	A
1603	LABYRINTH B	F	C
1604	VESTIBULE 0	B	A
1650	ICS POOL COOLING	D	C
1660	SERVICES 0 A	B	A
1661	SERVICES 0 B	B	A
1670	MAIN STEAM/FEEDWATER PIPING	H	C
16P0	FUEL POOL	I	I
1170	PRIMARY CONTAINMENT	I	F
1190	STAIR A	B	A
1191	STAIR B	B	A
1192	ELEVATOR	B	A
1193	CONTROL ROD DRIVE HATCH	C	D
1195	CHASE B*	Unoccupied/No Radiation Zone	

* Access to the chases requires an explicit authorization permit

Figure 12.3-7: Reactor Building Level 0.0 Meters Radiation Zones

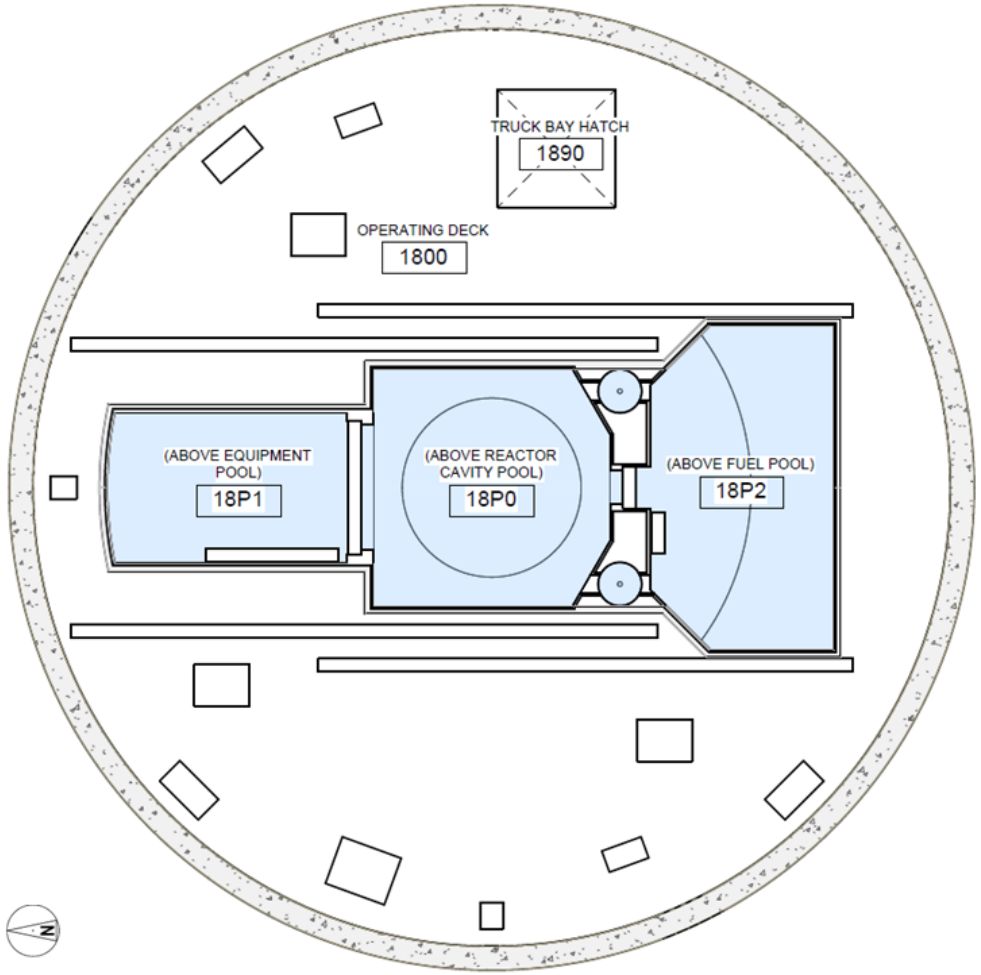


Radiation Zones Level 4.9m

ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
1600	TRUCK BAY	B	A
1701	STORAGE 4.9	B	A
16P0	FUEL POOL**	I	I
17P0	REACTOR CAVITY POOL	J	F
17P1	ISOLATION CONDENSER POOL A	C	C
17P2	ISOLATION CONDENSER OUTER POOL A	B	B
17P3	ISOLATION CONDENSER POOL B	C	C
17P4	ISOLATION CONDENSER OUTER POOL B	B	B
17P5	ISOLATION CONDENSER POOL C	C	C
17P6	ISOLATION CONDENSER OUTER POOL C	B	B
17P7	EQUIPMENT POOL**	C	D
1193	CONTROL ROD DRIVE HATCH	C	D

** Zone J during spent fuel transfers

Figure 12.3-8: Reactor Building Level 4.9 Meters Radiation Zones



Radiation Zones Level 13.0m

ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
1800	OPERATING DECK*	C	B
1890	TRUCK BAY HATCH*	C	B
18P0	(ABOVE REACTOR CAVITY POOL)*	C	B
18P1	(ABOVE EQUIPMENT POOL)**	C	B
18P2	(ABOVE REFUEL POOL)*	C	B

* Zone E during transfers of spent fuel or high activity components.
** Zone E in the presence of the vessel head

Figure 12.3-9: Reactor Building Level 13.0 Meters Radiation Zones

Radiation Zones Level 22.7m

ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
1990	POLAR CRANE AREA*	C	B

* Level 22.7 zone increases to D during transfers of spent fuel or high activity components.

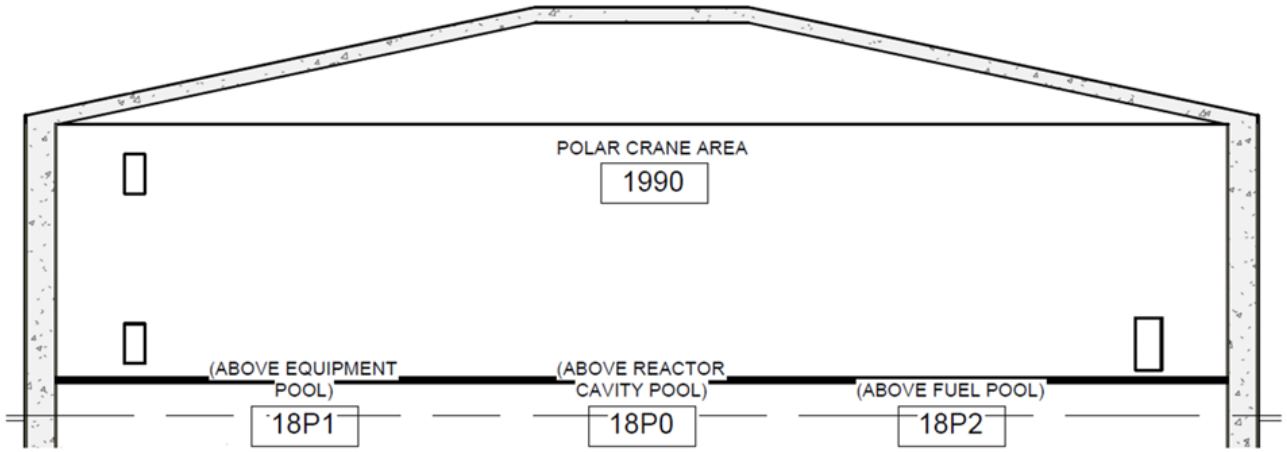
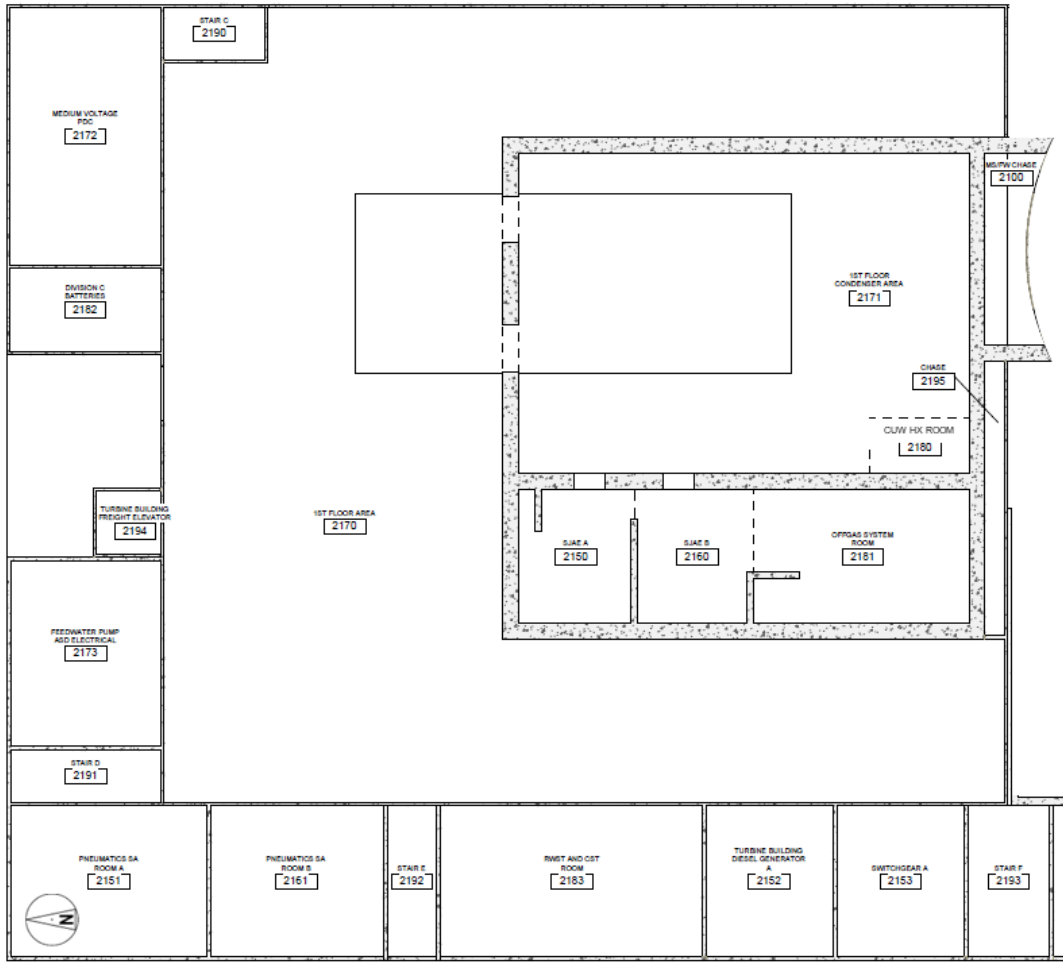


Figure 12.3-10: Reactor Building Level 22.7 Meters Radiation Zones

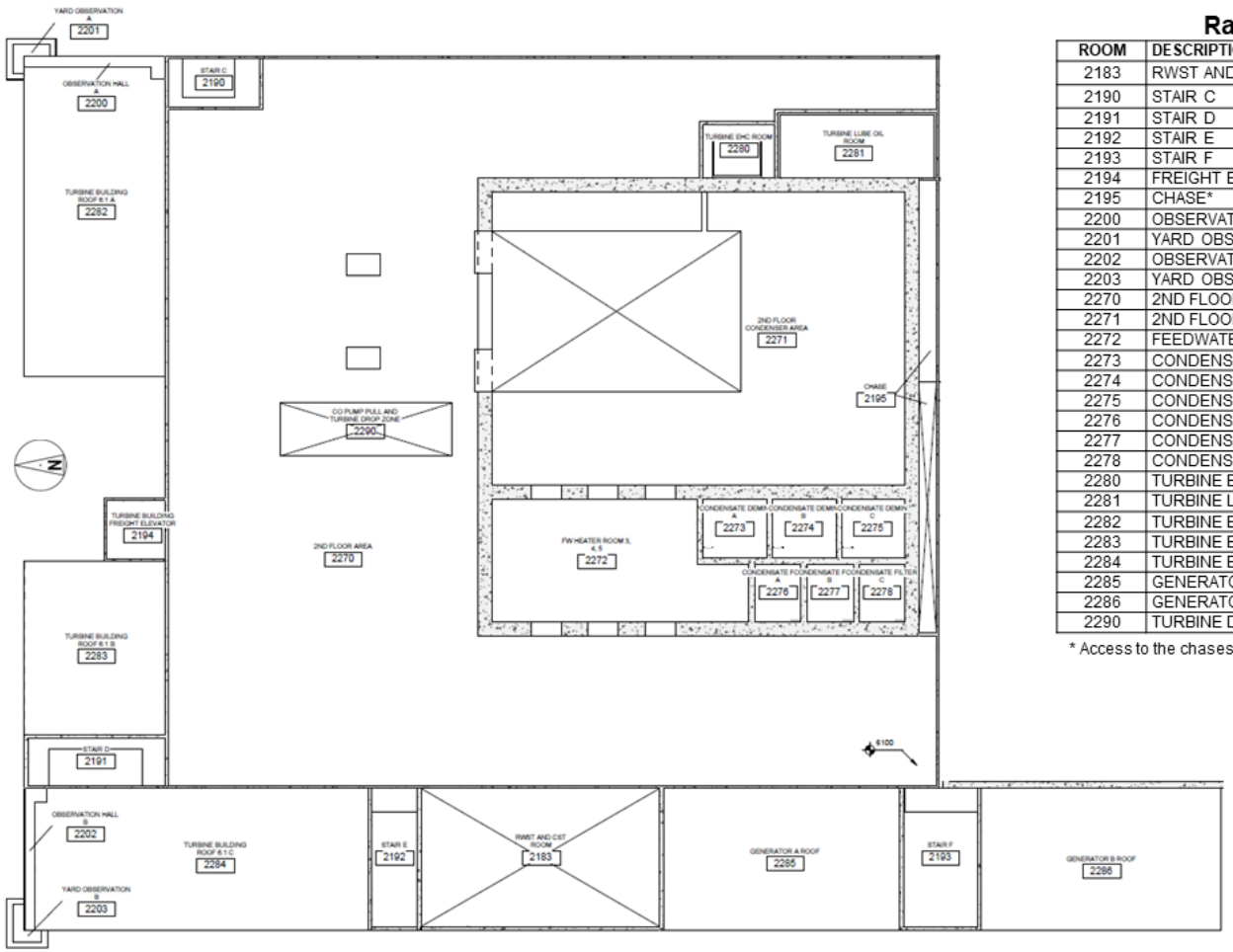


Radiation Zones Level 0m

ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
2100	MAIN STEAM/FEEDWATER CHASE	H	C
2150	STEAM JET AIR EJECTOR A	G	C
2151	PNEUMATICS SERVICE AIR ROOM A	A	A
2152	DIESEL GENERATOR A	A	A
2153	SWITCHGEAR A	A	A
2160	STEAM JET AIR EJECTOR B	G	C
2161	PNEUMATICS SERVICE AIR ROOM B	A	A
2162	DIESEL GENERATOR B	A	A
2163	SWITCHGEAR B	A	A
2170	1ST FLOOR AREA	B	B
2171	1ST FLOOR CONDENSER AREA	F	C
2172	MEDIUM VOLTAGE DISTRIBUTION	A	A
2173	FEEDWATER PUMP ASD ELECTRICAL	A	A
2180	CUW HEAT EXCHANGER ROOM	D	C
2181	OFFGAS SYSTEM ROOM	F	C
2182	DIVISION C BATTERIES	B	A
2183	RWST AND CST ROOM	C	C
2190	STAIR C	B	A
2191	STAIR D	B	A
2192	STAIR E	B	A
2193	STAIR F	B	A
2194	FREIGHT ELEVATOR	B	A
2195	CHASE*	Unoccupied/Controlled Access	

* Access to the chases requires an explicit authorization permit

Figure 12.3-11: Turbine Building Level 0.0 Meters Radiation Zones

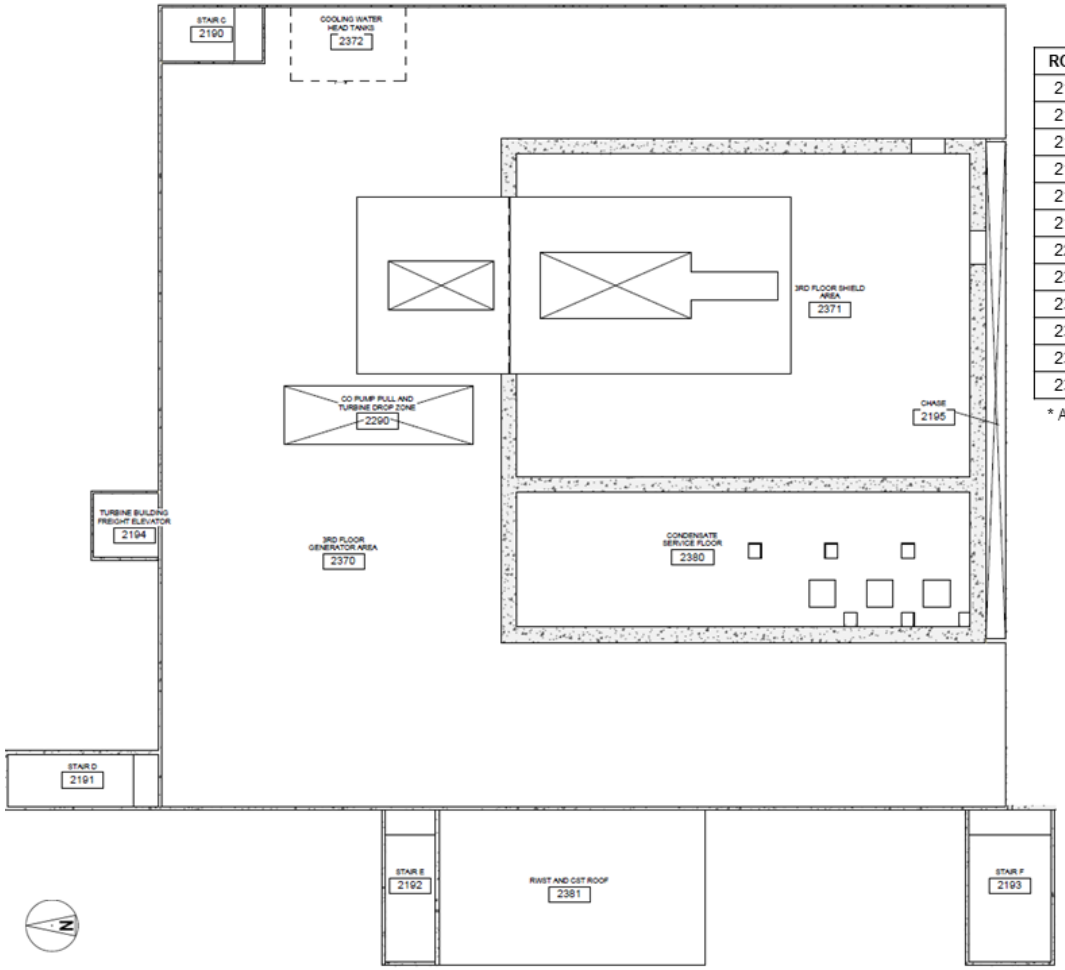


Radiation Zones Level 6.1m

ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
2183	RWST AND CST ROOM	C	C
2190	STAIR C	B	A
2191	STAIR D	B	A
2192	STAIR E	B	A
2193	STAIR F	B	A
2194	FREIGHT ELEVATOR	B	A
2195	CHASE*	Unoccupied/Controlled Access	
2200	OBSERVATION HALL A	B	A
2201	YARD OBSERVATION A	B	A
2202	OBSERVATION HALL B	B	A
2203	YARD OBSERVATION B	B	A
2270	2ND FLOOR AREA	B	B
2271	2ND FLOOR CONDENSER AREA	F	C
2272	FEEDWATER HEATER ROOM	G	C
2273	CONDENSATE DEMIN A	G	G
2274	CONDENSATE DEMIN B	G	G
2275	CONDENSATE DEMIN C	G	G
2276	CONDENSATE FILTER A	E	E
2277	CONDENSATE FILTER B	E	E
2278	CONDENSATE FILTER C	E	E
2280	TURBINE EHC ROOM	C	C
2281	TURBINE LUBE OIL ROOM	C	C
2282	TURBINE BUILDING ROOF 6.1 A	Unoccupied/Controlled Access	
2283	TURBINE BUILDING ROOF 6.1 B	Unoccupied/Controlled Access	
2284	TURBINE BUILDING ROOF 6.1 C	Unoccupied/Controlled Access	
2285	GENERATOR A ROOF	Unoccupied/Controlled Access	
2286	GENERATOR B ROOF	Unoccupied/Controlled Access	
2290	TURBINE DROP ZONE	B	B

* Access to the chases requires an explicit authorization permit

Figure 12.3-12: Turbine Building Level 6.1 Meters Radiation Zones

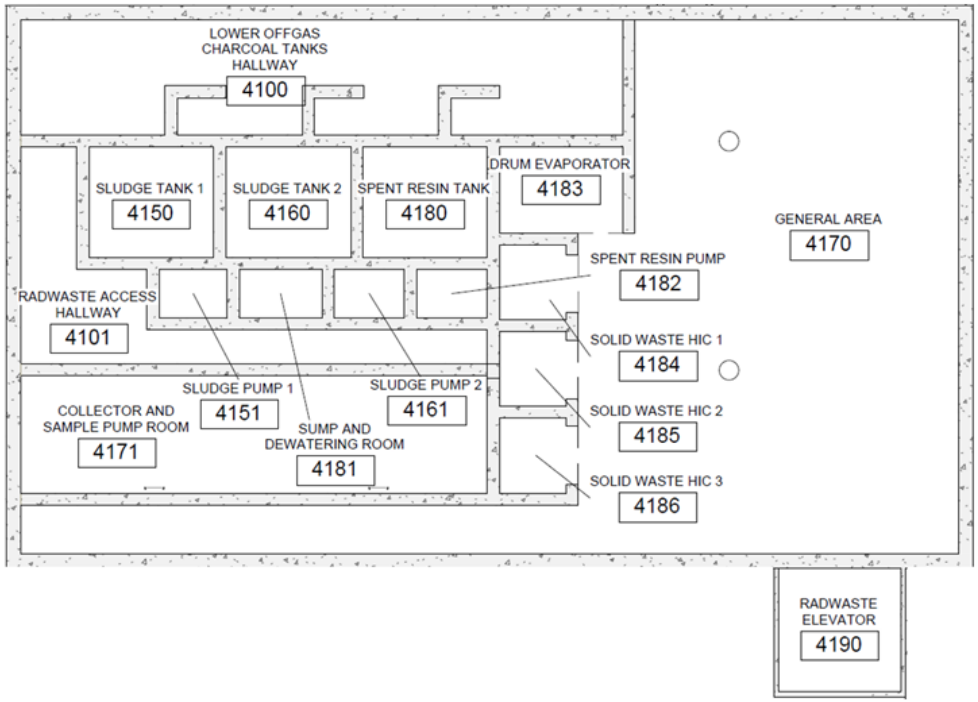


Radiation Zones Level 13.0m

ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
2190	STAIR C	B	A
2191	STAIR D	B	A
2192	STAIR E	B	A
2193	STAIR F	B	A
2194	FREIGHT ELEVATOR	B	A
2195	CHASE*	Unoccupied/Controlled Access	
2290	TURBINE DROP ZONE	B	B
2370	3RD FLOOR GENERATOR AREA	B	B
2371	3RD FLOOR SHIELD AREA	F	C
2372	COOLING WATER HEAD TANKS	B	B
2380	CONDENSATE SERVICE FLOOR	D	D
2381	RWST AND CST ROOF	Unoccupied/Controlled Access	

* Access to the chases requires an explicit authorization permit

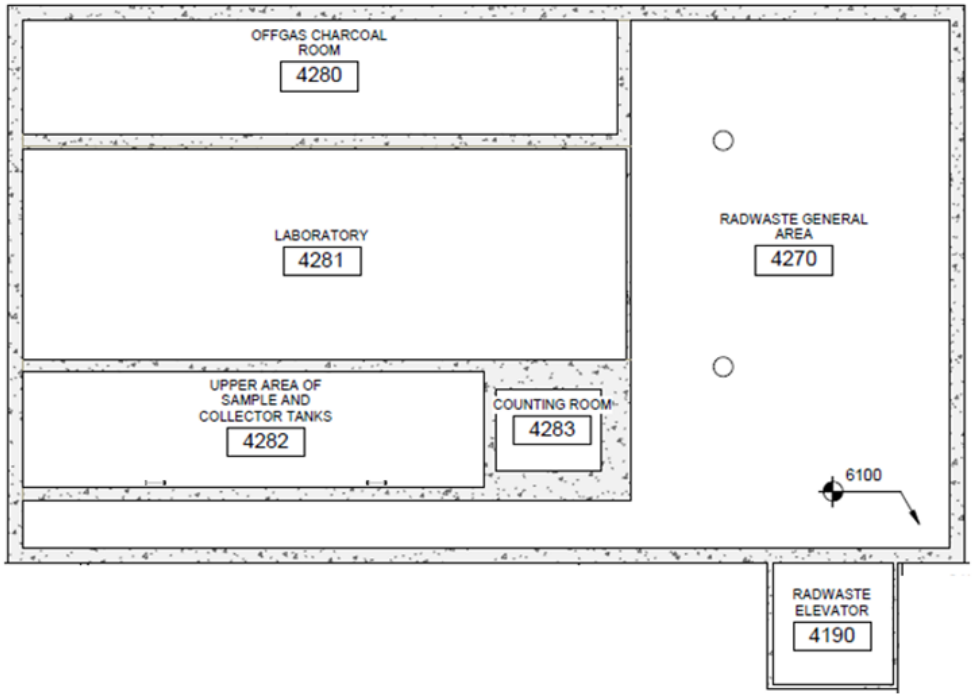
Figure 12.3-13: Turbine Building Level 13.0 Meters Radiation Zones



Radiation Zones Level 0m

ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
4100	LOWER OFFGAS CHARCOAL TANKS HALLWAY	B	B
4101	RADWASTE ACCESS HALLWAY	B	B
4150	SLUDGE TANK 1	H	H
4151	SLUDGE PUMP 1	F	C
4160	SLUDGE TANK 2	H	H
4161	SLUDGE PUMP 2	F	C
4170	GENERAL AREA	B	B
4171	COLLECTOR AND SAMPLE PUMP ROOM	C	C
4180	SPENT RESIN TANK	I	I
4181	SUMP AND DEWATERING ROOM	H	H
4182	SPENT RESIN PUMP	F	C
4183	DRUM EVAPORATOR	E	E
4184	SOLID WASTE HIC 1	G	G
4185	SOLID WASTE HIC 2	G	G
4186	SOLID WASTE HIC 3	G	G
4190	ELEVATOR	B	A

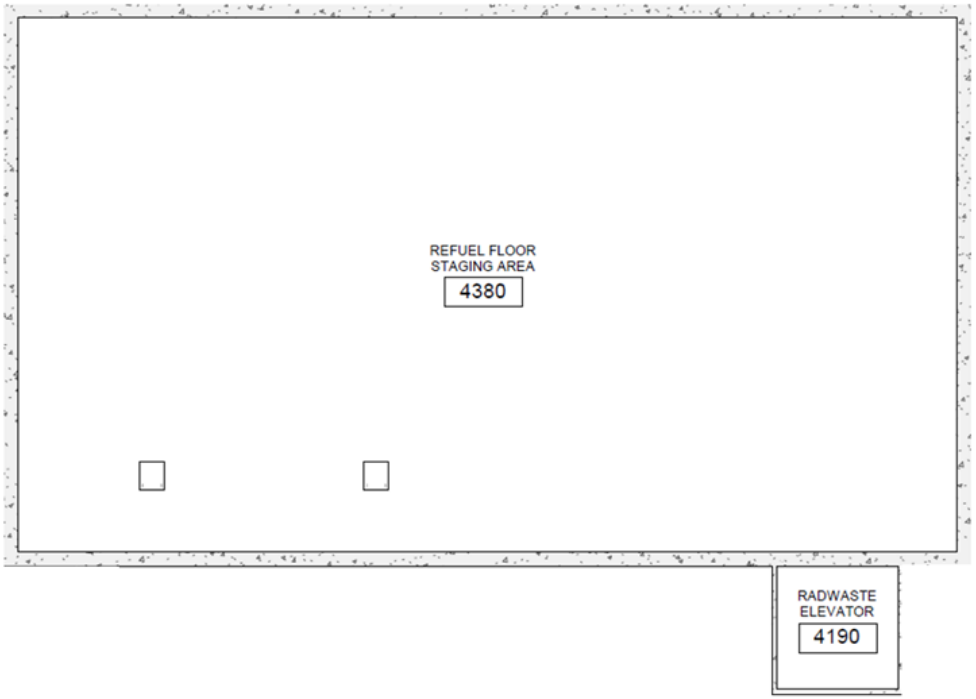
Figure 12.3-14: Radwaste Building Level 0.0 Meters Radiation Zones



Radiation Zones Level 6.1m

ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
4190	ELEVATOR	B	A
4270	RADWASTE GENERAL AREA	B	B
4280	OFFGAS CHARCOAL ROOM	E	C
4281	LABORATORY	B	B
4282	UPPER AREA OF SAMPLE AND COLLECTOR TANKS	C	C
4283	COUNTING ROOM	B	B

Figure 12.3-15: Radwaste Building Level 6.1 Meters Radiation Zones



Radiation Zones Level 13.0m

ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
4190	ELEVATOR	B	A
4380	REFUEL FLOOR STAGING AREA	B	B

Figure 12.3-16: Radwaste Building Level 13.0 Meters Radiation Zones

Radiation Zones Level 0m

ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
5101	TURBINE BUILDING ACCESS ARC	B	B
5102	FUEL AND REFUELING OUTAGE ACCESS ARC	B	B
5103	CONTROL BUILDING ACCESS ARC	B	B
5104	REACTOR BUILDING EGRESS PATHWAY	B	B
5105	RADWASTE BUILDING ACCESS ARC	B	B

Radiation Zones Level 0m

ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
5180	NEW FUEL / REFUELING OUTAGE STAGING	B	B
5190	STAIR H	A	A
5191	ELEVATOR	A	A
5192	STAIR I	B	A
5193	STAIR J	B	A

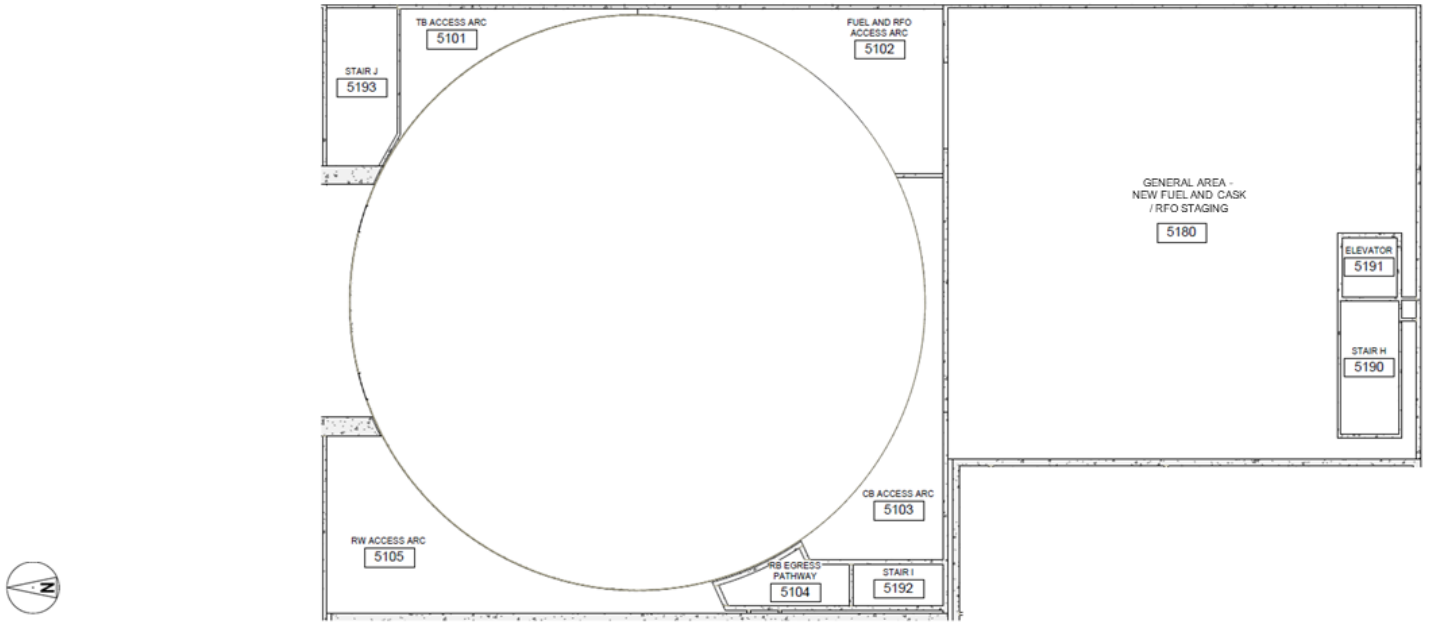


Figure 12.3-17: Service Building Level 0.0 Meters Radiation Zones

Radiation Zones Level 6.1m/6.7m

ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
5101	TURBINE BUILDING ACCESS ARC	B	B
5102	FUEL AND REFUELING OUTAGE ACCESS ARC	B	B
5103	CONTROL BUILDING ACCESS ARC	B	B
5105	RADWASTE BUILDING ACCESS ARC	B	B
5190	STAIR H	A	A
5191	ELEVATOR	A	A
5192	STAIR I	B	A
5193	STAIR J	B	A
5200	HALLWAY 6.7	A	A
5201	TB-SB HALLWAY	B	A

Radiation Zones Level 6.1m/6.7m

ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
5203	STAIR I VESTIBULE 6.1	B	B
5205	YARD OBSERVATION E	Unoccupied/Controlled Access	
5206	YARD OBSERVATION F	Unoccupied/Controlled Access	
5207	YARD OBSERVATION G	Unoccupied/Controlled Access	
5280	HEALTH PHSYICS AREAS	A	A
5281	COLD SHOP / MAINTAINENCE	A	A
5282	SPECIAL USE / FLEX	A	A
5283	WOMENS REST ROOM	A	A
5284	MENS REST ROOM	A	A

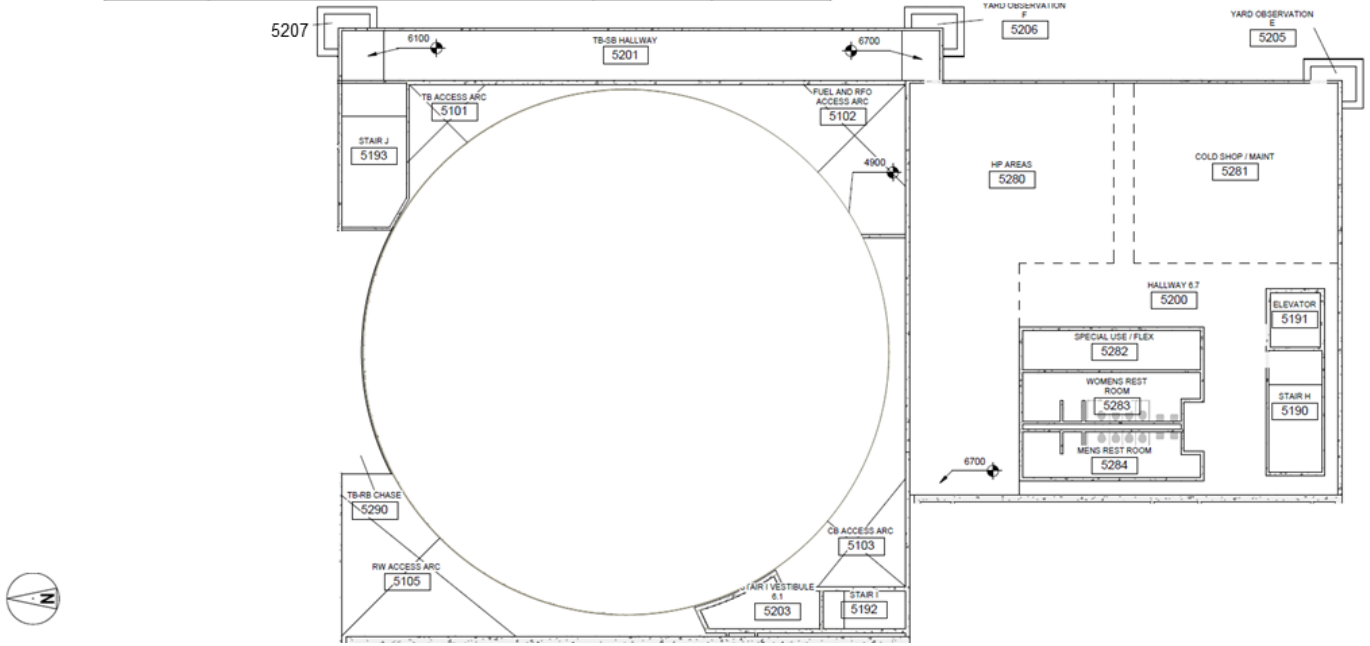


Figure 12.3-18: Service Building Level 6.1/6.7 Meters Radiation Zones

Radiation Zones Level 13.0m

ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
5101	TURBINE BUILDING ACCESS ARC	B	B
5102	FUEL AND REFUELING OUTAGE ACCESS ARC	B	B
5103	CONTROL BUILDING ACCESS ARC	B	B
5105	RADWASTE BUILDING ACCESS ARC	B	B
5190	STAIR H	A	A
5191	ELEVATOR	A	A
5192	STAIR I	B	A

Radiation Zones Level 13.0m

ROOM	DESCRIPTION	FULL POWER	SHUTDOWN
5193	STAIR J	B	A
5300	HALLWAYS	A	A
5302	STAIR J VESTIBULE 13.0	B	B
5380	OFFICES	A	A
5381	OUTAGE CENTER	A	A
5382	OTHER ROOMS	A	A

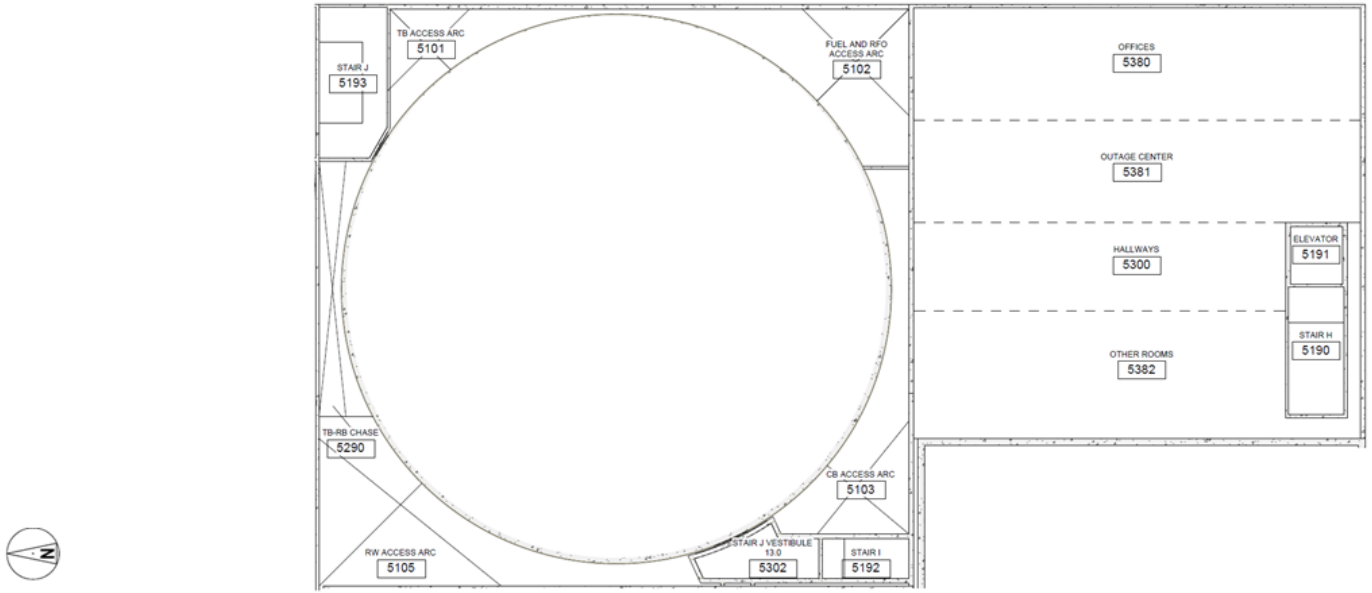


Figure 12.3-19: Service Building Level 13.0 Meters Radiation Zones

12.4 Dose Assessment

This section describes the occupational dose assessment for the BWRX-300 plant during normal operations, which is a significant element in the development of procedures and engineering controls necessary to achieve occupational radiation exposures consistent with the ALARA principle. The dose assessment is consistent with RG 8.19, "Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants – Design Stage Man-Rem Estimates," and demonstrates that the BWRX-300 design complies with 10 CFR Part 20. The estimated annual occupational radiation exposures are categorized into six collective occupational dose assessment categories:

- Reactor Operation and Surveillances
- Routine Maintenance
- Waste Processing
- Refueling
- ISI
- Special Maintenance

The dose assessment is dependent on estimates for dose rates in various occupied areas, frequency of operations, and the number of personnel participating in each of the activities for the six dose assessment categories. The estimates for these variables are developed using a variety of available methods including BWRX-300 radiation zoning levels, available technical reports, experiential data based on previous and current BWR plant designs and operational information, and the current expectations for operations, maintenance, waste processing, refueling and ISI activities for the BWRX-300 design. Accordingly, the dose assessment is expected to change as the design is refined.

Airborne radiation in the BWRX-300 is controlled to negligible levels during normal operation of the facility, so the dose from airborne activity is not a significant contributor to the total collective dose. As such, the estimated collective doses in this assessment do not assume contributions from inhalation or airborne radioactivity.

The dose assessment calculates specific doses for workers performing activities in areas of the facility within a significant (i.e., ≥ 1 μSv/hr) dose rate and sums those task specific doses to determine the total annual collective occupational dose estimate for normal operation of the plant in each of the six dose assessment categories. The task specific collective doses are calculated as the product of (a) the estimated effective dose rates where the activities are performed multiplied by (b) the collective man-hours/year estimates required to complete them. The results for each of the six dose assessment categories are shown in Tables 12.4-2 through 12.4-7, and the activities are summarized within this section. The BWRX-300 total occupational radiation exposure is shown in Table 12.4-1 and is the sum of the annual projected collective dose for the six categories.

Reactor Operation and Surveillances

The representative activities during routine reactor operations and surveillances considered in Table 12.4-2 include:

- Routine surveillances by Operations, Chemistry, Health Physics, and Security Personnel
- FMCRD, HCU Surveillance
- CUW Piping Surveillance

- SDC Surveillance
- Passive Systems Surveillance
- Instrumentation and Control Surveillance and Testing
- Outside Steam Tunnel
- Nonroutine Clean-up and Decontamination
- Routine Operating Deck/Pool Surveillances
- Fuel Receipt, Processing, and Channeling
- FPC Surveillances
- Nonroutine Fuel Sipping

Routine Maintenance

The representative activities during routine maintenance considered in Table 12.4-3 include:

- SDC Pipes/Valves and Pumps/Motors maintenance
- FMCRD HCUs maintenance
- Isolation Condenser System (ICS) valve and pool maintenance
- Instrumentation and Control maintenance
- FPC maintenance for filters, demineralizers, pumps, motors, and valves
- Maintenance on fuel pool, fuel storage racks, and fuel casks
- RWB maintenance related to reverse osmosis, demineralizers, tanks, pumps, valves, and instrumentation
- CUW HX, pipes, and valves maintenance and other miscellaneous work within the TBs accessible areas

Waste Processing

The representative activities during waste processing considered in Table 12.4-4 include:

- LWM / SWM, and Lab Operation
- Dry Active Waste (DAW) Sorting / Processing and shipments
- High Integrity Container (HIC) Processing/Shipments and other miscellaneous activities within the RWB

Refueling

The representative activities during refueling considered in Table 12.4-5 include:

- Containment and RPV Disassembly/Reassembly
- Refueling Process
- Fuel Transfer to Independent Fuel Storage Facility, if necessary

Inservice Inspection

The representative activities during ISI considered in Table 12.4-6 include:

- General ISI Activities

- RPV Welds/Nozzles
- Main Steam Piping/Valves
- Feedwater Piping/Valves
- CUW Piping/Valves
- SDC Piping/Valves
- Boron Injection System (BIS) Piping/Valves

Special Maintenance

The representative activities during special maintenance considered in Table 12.4-7 include:

- Containment Isolation Valve (CIV) and Reactor Isolation Valve (RIV) Rework
- FMCRD Under Vessel Maintenance
- Local Power Range Monitors (LPRMs) / Gamma Thermometers (GTs)
- Miscellaneous In-Containment Maintenance on Valves, Pumps, and Instrumentation
- Other In-Containment Outage Maintenance including Passive Systems
- SDC Pumps/Motors
- FMCRD HCU
- FMCRD Rebuild
- Condensate Treatment
- Special Maintenance in RB Instrumentation and other outage maintenance
- Main Turbine overhaul
- Turbine Valves/Pumps

Table 12.4-1: BWRX-300 Total Occupational Radiation Exposure Estimates

Activity	Estimated Collective Hours Annually (man-hours/year)	Projected Annual Collective Dose (man-mSv/year)	Percent of Total Collective Dose (%)
Reactor Operations and Surveillances	2,609	38	7.8
Routine Maintenance	7,085	90	18.2
Waste Processing	2,456	60	12.1
Refueling	2,580	47	9.6
ISI	759	32	6.4
Special Maintenance	11,624	225	45.8
Total	**27,113**	**491**	**100.0**

Table 12.4-2: Occupational Dose Estimates During Reactor Operations and Surveillances

Facility Area / Activity	Estimated Average Dose Rate (μSv/hr)	Estimated Collective Exposure Time (man-hours/year)	Collective Dose (man-mSv/year)
Reactor Building			
Routine Operation, Chemistry, Health Physics and Security Surveillance	8	1095	8.8
FMCRD HCU Surveillance	50	15	0.8
CUW Piping Surveillance	150	10	1.5
SDC Surveillance	150	30	4.5
Passive Systems Surveillance	30	125	3.8
Instrumentation and Control Surveillance and Testing	8	125	1.0
Outside Steam Tunnel	30	61	1.8
Nonroutine Clean-up and Decontamination	30	96	2.9
Routine Operating Deck/Pool Surveillances	8	122	1.0
Fuel Receipt, Processing, and Channeling	8	108	0.9
FPC Surveillances	150	12	1.8
Nonroutine Fuel Sipping	30	80	2.4
Radwaste Building			
Included in Table 12.4-4 estimate			
Turbine Building			
Routine Operation, Chemistry, Health Physics and Security Surveillance	10	730	7.3
Total		**2609**	**38.3**

Table 12.4-3: Occupational Dose Estimates During Routine Maintenance

Facility Area / Activity	Estimated Average Dose Rate (μSv/hr)	Estimated Collective Exposure Time (man-hours/year)	Collective Dose (man-mSv/year)
Reactor Building			
SDC Pipes/Valves	150	50	7.5
SDC Pumps/Motors	150	50	7.5
FMCRD HCUs	30	40	1.2
Passive System Valves (ICS)	30	50	1.2
Passive System Pools (ICS)	30	50	1.5
Instrumentation	30	150	4.5
FPC Filters/Demineralizers	150	60	9.0
FPC Pumps/Motors	30	75	2.3
FPC Valves	30	80	2.4
Fuel Pool, Fuel Racks, Fuel Casks	30	100	3.0
Radwaste Building			
Reverse Osmosis	150	75	11.3
Demineralizers	150	40	6.0
Tanks	150	100	15.0
Pumps	30	20	0.6
Valves	30	35	1.1
Instrumentation	30	60	1.8
Turbine Building			
CUW HX/Pipes/Valves	150	50	7.5
Miscellaneous TB Work in Accessible Areas	1	6000	6.0
Total		**7085**	**89.6**

Table 12.4-4: Occupational Dose Estimates During Waste Processing

Facility Area / Activity	Estimated Average Dose Rate (μSv/hr)	Estimated Collective Exposure Time (man-hours/year)	Collective Dose (man-mSv/year)
Radwaste Building			
LWM / SWM, and Lab Operation	10	2080	20.8
DAW Sorting/Processing	50	48	2.4
HIC Processing/Shipments	50	80	4.0
DAW Shipments	50	48	2.4
Miscellaneous Activities[1]	150	200	30.0
Total		**2456**	**59.6**

1. Miscellaneous activities in the RWB include activities in high dose rate areas, such as valve lineups or filter changes. These activities are assumed to be required approximately 200 person-hours per year (roughly 4 person-hours per week) in an average dose rate field of 150 μSv/hour.

Table 12.4-5: Occupational Dose Estimates During Refueling

Facility Area / Activity	Estimated Average Dose Rate (μSv/hr)	Estimated Collective Exposure Time (man-hours/year)	Collective Dose (man-mSv/year)
Reactor Building			
Containment and RPV Disassembly/Reassembly	15	740	11.1
Phase I Refueling[1]	25	440	11.0
Phase II Refueling[2]	25	900	22.5
Fuel transfer to Independent Fuel Storage Facility (if utilized)	5	500	2.5
Total		**2580**	**47.1**

1. Phase I of the refueling process includes removal of fuel from the core, in-vessel inspection and maintenance of reactor vessel components, and potential replacement of core components.
2. Phase II of the refueling process includes reloading the in-vessel maintenance fuel offload, loading new fuel assemblies, and shuffling of the core assemblies to complete the refueling and maintenance outage fuel transfer.

Table 12.4-6: Occupational Dose Estimates During Inservice Inspection

Facility Area / Activity	Estimated Average Dose Rate (μSv/hr)	Estimated Collective Exposure Time (man-hours/year)	Collective Dose (man-mSv/year)
Primary Containment			
General Activities	60	100	6.0
RPV Welds/Nozzles	60	100	6.0
Main Steam piping/valves	60	15	0.9
Feedwater piping/valves	60	15	0.9
CUW piping/valves	60	5	0.3
SDC piping/valves	60	12	0.7
Other Equipment (Passive systems)	60	50	3.0
Reactor Building			
General Activities	30	300	9.0
SDC	30	75	2.3
BIS piping and valves	30	25	0.8
FMCRD	30	12	0.4
Other Equipment (Passive systems)	30	50	1.5
Total		**759**	**31.7**

Table 12.4-7: Occupational Dose Estimates During Special Maintenance

Facility Area / Activity	Estimated Average Dose Rate (μSv/hr)	Estimated Collective Exposure Time (man-hours/year)	Collective Dose (man-mSv/year)
Primary Containment			
CIV Rework	18	1000	18.0
FMCRD Under Vessel	65	200	13.0
LPRMs/GTs	100	24	2.4
Miscellaneous Valves & Pumps	40	750	30.0
Miscellaneous Instrumentation	50	250	12.5
Other Outage Maintenance Including Passive Systems	50	650	32.5
Reactor Building			
RIV Rework	13	650	8.5
SDC Pumps/Motors	150	100	15.0
FMCRD HCU	30	150	4.5
FMCRD Rebuild	30	240	7.2
Instrumentation	8	300	2.4
Condensate Treatment	35	500	17.5
Other Outage Maintenance Including Passive Systems	8	1700	13.6
Turbine Building			
Major Turbine Overhaul	3	4200	12.6
Turbine Valves/Pumps	39	910	35.5
Total		**11624**	**225.1**

12.5 Health Physics Program

Any supplementation needed for the text below based on the site-specific program development will be provided in CPA Part 2.

12.5.1 Objectives

The BWRX-300 design, in conjunction with the health physics program, includes health physics facilities and features to administratively control:

- Plant personnel activities limiting personnel exposure to radiation and radioactive materials ALARA, and within the guidelines of 10 CFR Part 20
- Effluent releases to maintain the releases ALARA, and within the limits of 10 CFR 20 and the plant Technical Specifications
- Waste shipments in the plant to meet applicable shipping and receipt of material requirements at third party processors, or storage and burial sites

12.5.2 Equipment, Instrumentation, and Facilities

The health physics facilities are located in the Service Building, as shown on Figure 12.3-18.

12.5.3 Operational Considerations

BWRX-300 facilities and design features are such that an acceptable radiation protection program can be developed, documented, and implemented in accordance with 10 CFR 20.1101. A radiation protection program is developed, documented, and implemented through plant procedures that address quality requirements commensurate with the scope and extent sufficient to ensure compliance with the provisions of 10 CFR Parts 19, 20, 50, and 71 and consistent with the guidance in Regulatory Guides 1.8, 8.2, 8.7, 8.8, 8.9, 8.10, 8.13, 8.15, 8.27, 8.28, 8.29, 8.34, 8.35, 8.36, 8.38, and the consolidated guidance in NUREG-1736.

In accordance with 10 CFR 20, Subpart B, the purpose of the radiation protection program is to maintain occupational and public doses below regulatory limits and as low as is reasonably achievable (ALARA). To achieve this, the program will include:

I. A documented management commitment to keep exposures ALARA
II. A trained and qualified organization with sufficient authority and well-defined responsibilities
III. Adequate facilities, equipment, and procedures to effectively implement the program

The operational radiation protection program is implemented in stages consistent with the milestones described in Section 13.4.

Prior to initial receipt of by-product, source, or special nuclear materials (excluding Exempt Quantities as described in 10 CFR 30.18), and thereafter, when such radioactive materials are possessed under an applicable license, the following radiation protection program elements will be in place:

a. Organization – A radiation protection supervisor and at least one (1) radiation protection technician, each selected, trained, and qualified consistent with the guidance in Regulatory Guide 1.8.
b. Facilities – A facility or facilities to support the receipt, storage, and control of non-exempt radioactive sources in accordance with 10 CFR 20.1801, 20.1802, and 20.1906.

c. Instrumentation and Equipment –Adequate types and quantities of instrumentation and equipment will be selected, maintained, and used to provide for the appropriate detection capabilities, ranges, sensitivities, and accuracies to conduct radiation surveys and monitoring (in accordance with 10 CFR 20.1501 and 20.1502) for the types and levels of radiation anticipated for the non-exempt sources.

d. Procedures – Procedures will be established, implemented, and maintained sufficient to maintain adequate control over the receipt, storage, and use of radioactive materials and as necessary to assure compliance with 10 CFR 19.11 and 19.12 and the applicable portions of 10 CFR Part 20, commensurate with the types and quantities of radioactive materials received and possessed.

e. Training – Initial and periodic training will be provided to individuals responsible for the receipt, control, or use of non-exempt radioactive sources in accordance with 10 CFR 19.12 and consistent with the guidance in Regulatory Guides 1.8, 8.13, 8.27, and 8.29.

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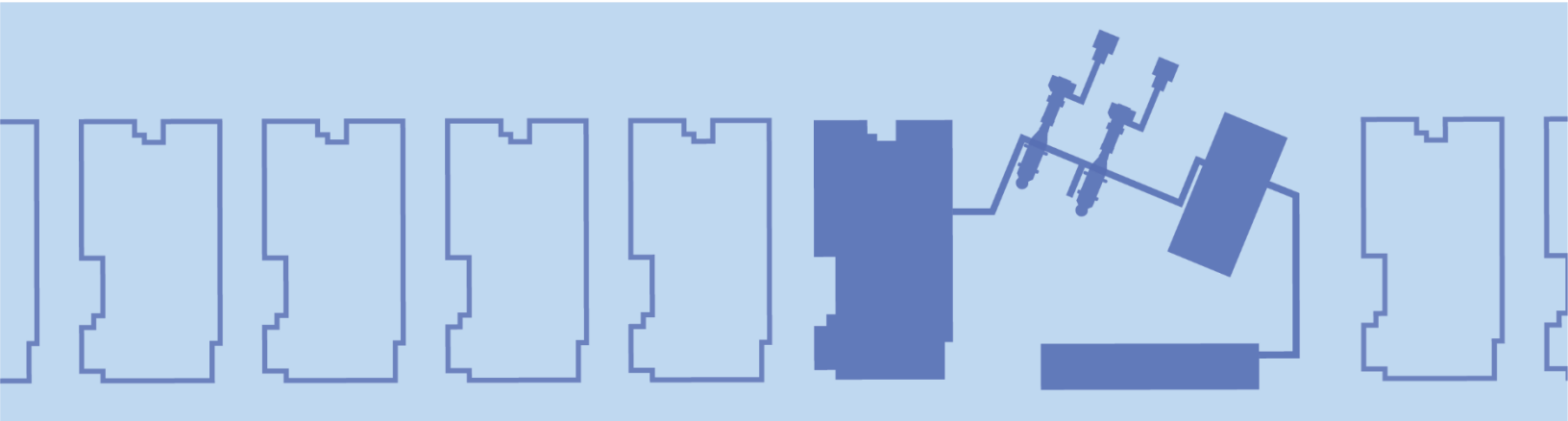
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Non-Proprietary Information

Chapter 13
Conduct of Operations



ACRONYM LIST

Acronym	Explanation
CFR	Code of Federal Regulations
ESPA	Early Site Permit Application
GEH	GE-Hitachi Nuclear Energy
OLA	Operating License Application
PSP	Plant Security Plan
SGI	Safeguards Information
SSAR	Site Safety Analysis Report
T&Q	Training and Qualification
USNRC	United States Nuclear Regulatory Commission

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None.

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None.

13.0 CONDUCT OF OPERATIONS

13.1 Organizational Structure

To be provided in CPA Part 2.

13.2 Training

To be provided in CPA Part 2.

13.3 Emergency Preparedness

To be provided in CPA Part 2.

13.4 Operational Programs

To be provided in CPA Part 2.

13.5 Plant Procedures

To be provided in CPA Part 2.

13.6 Physical Security

A site specific physical security plan will be provided in CPA Part 2.

A BWRX-300 Security Assessment has been performed that describes the security design bases and requirements for site characteristics, physical features, systems, and components incorporated into the BWRX-300 standard design to benefit or enhance the physical security capabilities of the site. The BWRX-300 Security Assessment addresses identification of vital equipment, vital areas, development of target sets, vulnerability assessments, defensive analyses, design features to enhance security, and the other security features of the BWRX-300 that establish the security system design. The BWRX-300 Security Assessment is designated and controlled as Safeguards Information (SGI) in accordance with 10 CFR 73.21, “Protection of Safeguards Information: Performance Requirements” and therefore is restricted from public release. The Operating License Application (OLA) will address site-specific design features of the GVH BWRX-300 that support site security, detailed security monitoring equipment design, and security methods for screening station operating personnel.

A security plan will be supplied to the USNRC to fulfill the requirements of 10 CFR 50.34(c) and 10 CFR 50.34(d). Implementing procedures, including testing and maintenance, will be developed, and maintained. The security plan is designated and controlled as Safeguards Information (SGI) in accordance with 10 CFR 73.21. Certain implementing procedures are designated and controlled as SGI in accordance with 10 CFR 73.21 and others are security-related and are restricted from public release under 10 CFR 2.390, “Public Inspections, Exemptions, Requests for Withholding.” The security plan consists of the following components:

- Physical Security Plan (PSP) which satisfies 10 CFR 50.34(c) and implements
- 10 CFR 73.55, “Requirements for Physical Protection of Licensed Activities in Nuclear Power Reactors Against Radiological Sabotage”
- Training and Qualification (T&Q) Plan which satisfies 10 CFR 50.34(c) and implements Section VI of Appendix B to Part 10 CFR 73
- Safeguards Contingency Plan which satisfies 10 CFR 50.34(d) and which implements Section II of Appendix C to Part 10 CFR 73

A cyber security plan, to be described in CPA Part 2, will be supplied to the USNRC to fulfill the requirements of 10 CFR 50.34(c) and 10 CFR 73.54, “Protection of Digital Computer and Communication Systems and Networks.”

GVH has developed a cyber program plan to ensure the BWRX-300 facility is designed to protect critical digital assets from cyber threats.

13.7 Fitness for Duty

To be provided in CPA Part 2.

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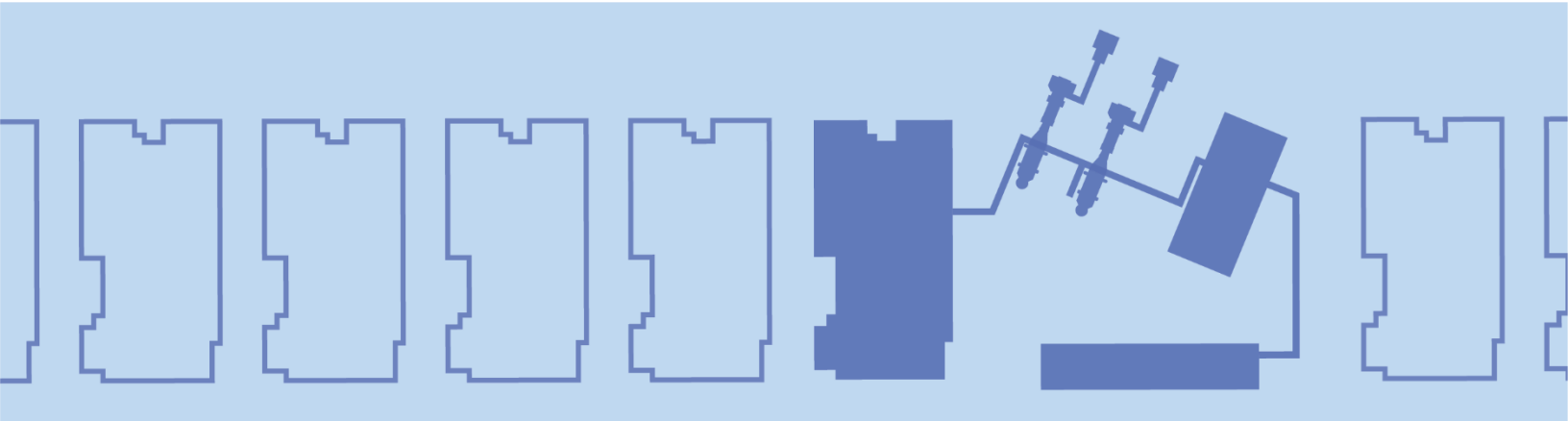
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Non-Proprietary Information

Chapter 14
Initial Test Program



ACRONYM LIST

Acronym	Explanation
BECPV	Blue Energy Center – Port of Victoria
BOP	Balance of Plant
CPA	Construction Permit Application
FOAK	First-of-a-Kind
FSAR	Final Safety Analysis Report
GVH	GE Vernova Hitachi Nuclear Energy
ICS	Isolation Condenser System
ITP	Initial Test Program
OL	Operating License
OPEX	Operating Experience
OPG	Ontario Power Generation
PSAR	Preliminary Safety Analysis Report
QAPD	Quality Assurance Program Description
RB	Reactor Building
RG	Regulatory Guide
SC	Safety Class
SC1	Safety Class 1
SC2	Safety Class 2
SC3	Safety Class 3
SCCV	Steel-Plate Composite Containment Vessel
SME	Subject Matter Expert
SSCs	Structures, Systems, and Components
TB	Turbine Building
TRACG	Transient Reactor Analysis Code General Electric
USNRC	United States Nuclear Regulatory Commission

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14.0 INITIAL TEST PROGRAM

The purpose of this chapter is to describe the test program that will be performed during the initial startup of the Blue Energy (Blue Energy) nuclear unit. The program describes the way testing will be performed, controlled, and documented for the applicable testing phases.

14.1 Preliminary Safety Analysis Report Content

14.1.1 Scope of Test Program

14.1.1.1 Summary of Test Program and Objectives

The Initial Test Program (ITP) comprises phases categorized as construction, preoperational, and startup tests. Construction tests serve as prerequisites for preoperational tests. Preoperational tests are tests conducted before fuel loading to demonstrate the capability of plant systems to meet performance requirements. Initial startup testing begins with fuel loading and includes precritical tests, initial criticality, low-power testing, and power-ascension testing. Initial startup testing demonstrates the integrated plant's capability to meet the required performance standards.

The ITP addresses Structures, Systems, and Components (SSCs) and design features for both the nuclear portion of the facility and the Balance of Plant (BOP). The classification of SSCs for the BWRX-300 is described in Section 3.2. ITP is applied to Safety Class 1 (SC1), Safety Class 2 (SC2), and Safety Class 3 (SC3) SSCs. The preliminary list of SSC classifications is provided in Table 3A-1. The SSCs described in Table 3A-1 as SC1, SC2, or SC3, together with the ITP descriptions herein, provide a high-level description of the types of testing to be conducted by the ITP. A graded approach is used in the development and implementation of ITP to ensure greater attention is given to SSCs based on Safety Class (SC) and risk significance. With this graded approach, documentation associated with testing is commensurate with the item's importance to safety.

The overall objectives of the ITP are to:

- Demonstrate that construction is complete and acceptable, including required testing
- Ensure the capability of SSCs to meet design performance requirements
- Provide additional assurance that the facility has been adequately designed
- Achieve fuel loading in a safe manner
- Demonstrate, where practical, that the unit is capable of withstanding anticipated transients and postulated accidents
- Validate, to the extent practical, that unit operating procedures and operator training are adequate to operate the facility safely
- Bring the unit to rated capacity and sustained power operation

Construction testing is the earliest testing performed to demonstrate that components and systems are correctly installed. Following construction or manufacturing, testing is performed to demonstrate the proper performance of SSCs and design features. These construction tests include, but are not limited to, flushing and cleaning, hydrostatic testing, initial calibration of instrumentation, checks of electrical wiring and equipment, valve testing, and initial energization. Component testing confirms fitness for use and readiness to support integrated testing through configuration control, calibration, and initial operation of the components. Component tests include control and display confirmation, functional checks, and mechanical run-in tests that do not require system operation. These may also include setting limit switches, stroke and cycle

timers, and component response parameters based on the component design. Completion of a system's component tests ensures it is ready to support preoperational testing. Construction testing may be performed offsite, provided that downstream activities do not invalidate prior test results.

Preoperational testing is conducted after construction testing and before fuel loading to demonstrate that plant systems are capable of operating safely and efficiently, in a manner compatible with the system design bases.

The general objectives of preoperational testing are to:

- Ensure design specification and test acceptance criteria are met
- Provide documentation of equipment and system performance
- Provide baseline test and operating data for future reference
- Ensure adequate integrated plant system operation
- Establish and evaluate normal, abnormal, and emergency operating procedures to the extent practical
- Establish and evaluate surveillance testing procedures whenever possible

Startup testing demonstrates that the nuclear unit operates as designed through the performance of normal plant operation and selected off-normal operations. Following completion of necessary prerequisites, fuel loading commences, making the unit an operating power plant. Startup testing consists of five test plateaus. Tests may be repeated throughout the plateaus to verify system performance or response at increasing temperatures or power levels:

- **Open Vessel Testing** - Includes baseline data taken before insertion of fuel into the reactor vessel, the fuel-load process through a fully loaded core, and shutdown-margin verifications
- **Heat-up Testing** - Begins after the reactor vessel is fully assembled and hydrostatically or leak tested and includes reactor startup to less than 5 percent power for heat-up to rated temperature and pressure. Testing may include thermal expansion, temperature monitoring, low-power nuclear instrument checks, process-controller tuning checks, and system performance tests at points along the heat-up and pressurization curve.
- **Low-Power Testing** - Includes testing between approximately 5 and 25 percent power and placing the turbine generator online for the first time. System performance testing, control-system tuning, and transient-event testing may be included.
- **Mid-Power Testing** - Includes testing between approximately 50 and 75 percent power, including system performance testing, control-system tuning, and typically transient-event testing
- **High-Power Testing** - Includes startup testing at 100 percent power (+0/-5 percent) and transient testing

Testing is performed at intermediate points if required by an approved procedure or if previous test results indicate the need for additional tests. Some tests may take data at power levels between the plateau definitions. Such tests are performed after permission has been granted to proceed to the higher power plateau.

14.1.1.2 Organization and Staffing

The Quality Assurance Program Description (QAPD) applicable to the Blue Energy project, provided in Section 17.5, describes the organization responsible for design, construction, and operations. Section 13.1 provides additional organizational and staffing information.

14.1.1.3 Startup Administration Manual

The startup administrative manual provides high-level organizational and administrative guidance intended to establish bounding requirements and an administrative framework applicable to preoperational and startup testing of a BWRX-300 plant. The manual provides minimum requirements and recommendations for the startup testing process to ensure that the test program is executed to validate that the nuclear unit has been constructed and functions in accordance with the approved design.

In accordance with the startup administrative manual, site procedures are developed that:

- Describe the responsibilities of the organization that conducts the test program, methods, and plans for providing the necessary staff, and the responsibilities, authorities, and qualifications of personnel conducting the ITP
- Delineate the development, review, and approval of test procedures consistent with Regulatory Guide (RG) 1.68, “Initial Test Programs for Water-Cooled Nuclear Power Plants”
- Delineate the utilization of reactor operating and testing experience in the development of test procedures
- Require development of plant operating and emergency procedures before fuel loading and their application during the test program, consistent with Section C.7 of RG 1.68
- Define requirements for the test program schedule consistent with Section C.5 of RG 1.68
- Identify the quality process used to control resolution of test failures, deficiencies, and oversights discovered in the ITP, including control of plant modifications required to resolve deficiencies

RG 1.68 specifies criteria for determining which SSCs and design features must be tested in accordance with the guidance therein.

14.1.1.4 Test Procedures

Preoperational and startup testing is conducted using detailed, step-by-step written procedures to control each test. The procedures include safety precautions and limits, as needed, to supplement those in normal operating procedures. Test procedures:

- Specify testing prerequisites
- Describe desired initial conditions
- Include methods to direct and control test performance, including sequencing
- Specify acceptance criteria by which the test is evaluated
- Provide for or specify the format by which data or observations are recorded

The procedures are developed and reviewed by qualified personnel with appropriate technical backgrounds and experience. This includes participation by principal design organizations to establish test performance requirements and acceptance criteria. Consideration of test requirements are incorporated into the BWRX-300 design early in the design process.

Preoperational and startup test specifications, which contain testing objectives and acceptance criteria, are used to generate test procedures. Test specifications consider potentially hazardous conditions and operating limitations during testing, such as large thermal gradients that may exceed stress limits or cause excessive fatigue usage under low-flow, low-heat conditions. Test procedures are reviewed and receive final approval by designated plant management personnel.

14.1.1.5 Conduct of Test Program

The ITP organization conducts preoperational and startup testing in accordance with the startup administrative manual. The manual outlines the administrative processes and requirements governing ITP activities and interfaces with other organizations. The manual defines the specific format and content of preoperational and startup test procedures, as well as the review and approval process. It also specifies the process for review and approval of test results; resolution of failures to meet acceptance criteria; and resolution of operational problems or design deficiencies. Site-specific controls ensure that the as-tested status of each system is known and track modifications, including retest requirements, deemed necessary for systems undergoing or having completed specified testing. Site-specific procedures govern the qualifications and responsibilities of positions within the ITP organization.

14.1.1.6 Review, Evaluation, and Approval of Test Results

Individual preoperational and startup test results are evaluated and reviewed, including test exceptions. Test results, including final resolutions, are reviewed and approved by designated ITP supervisory personnel. Final approval is obtained as defined in the startup administrative manual, including responsibility for final review and approval of overall test-phase results.

14.1.2 Plant Design Features that are Specific, Unique, or First of a Kind

The BWRX-300 contains design features that are new and have not been tested previously; therefore, these new features are treated as First-of-a-Kind (FOAK). Validation of FOAK design features can be demonstrated via analysis, appropriate test programs, experience, or a combination thereof. This subsection addresses FOAK tests performed by the ITP organization during preoperational and startup testing that are being reviewed for the first time by the United States Nuclear Regulatory Commission (USNRC). Methods of analysis, including the Transient Reactor Analysis Code General Electric (TRACG), and validation of construction techniques, including the Steel-Plate Composite Containment Vessel (SCCV) and Reactor Building (RB) structural design, are described elsewhere in the Preliminary Safety Analysis Report (PSAR) or in applicable topical reports.

The FOAK tests anticipated for the BWRX-300 based on the current design maturity are listed in Table 14.1-1. As testing plans are developed in preparation for the Operating License (OL) application, preoperational and startup tests may be added to this list.

14.1.3 Conformance of Test Programs with Regulatory Guides

The ITP is developed in conformance with RG 1.68. Additional RGs identified through RG 1.68 are considered during test development. Conformance to these additional RGs is addressed in the Final Safety Analysis Report (FSAR) when associated system test descriptions have been developed. Examples include:

- RG 1.20, “Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing”
- RG 1.32. “Criteria for Power Systems for Nuclear Power Plants”

- RG 1.52, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants”
- RG 1.68.2, “Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants”
- RG 1.68.3, “Preoperational Testing of Instrument and Control Air Systems”
- RG 1.79.1, “Initial Test Program of Emergency Core Cooling Systems for New Boiling-Water Reactors”
- RG 1.140, “Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants”

14.1.4 Utilization of Plant Operating and Testing Experiences at Other Reactor Facilities

Operating and testing experience from relevant reactor facilities is used to develop and implement the BWRX-300 testing program. Lessons learned from previous plant designs, construction, testing, and operation are incorporated into test-specification development and test procedures. Recent construction and startup experience, including experience from other advanced light-water reactor projects, may be considered where applicable.

Construction Permit Application (CPA) Part 2 will describe how Blue Energy will identify the specific sources of operating, construction, and testing experience available to Blue Energy; define the responsible organization; and confirm the mechanism for obtaining and incorporating BWRX-300 lessons learned.

14.1.5 Test Program Schedule

For SC1 SSCs, the schedule, relative to the initial fuel-load date, for conducting each major phase of the ITP is described dynamically. The test schedule remains under development. If time periods significantly shorter than nine months for preoperational testing and three months for startup testing are used, justification will be provided. To support USNRC inspection, test-procedure preparation is scheduled such that applicable approved procedures are available approximately 60 days before their intended use.

A detailed testing schedule will be developed and made available for regulatory awareness. The schedule may be updated and continually optimized to reflect actual progress and revised projections. Implementation milestones for the ITP, plans for hiring and training plant personnel, and development of plant operating and emergency procedures will be provided in CPA Part 2.

14.1.6 Trial Use of Plant Operating and Emergency Procedures

To the extent practicable throughout the preoperational and startup test phases, test procedures utilize operating, surveillance, emergency, and abnormal procedures where applicable. The use of these procedures provides the following:

- Provide the specific procedure or illustrate changes which may be required
- Provide training of plant personnel in the use of these procedures

Increase the level of knowledge of plant personnel on the systems being tested.

A test procedure may use a combination of references to operating, emergency, or abnormal procedures, or may repeat a series of steps from those procedures, to accomplish these objectives while efficiently performing the specified testing.

14.1.7 Augmenting Staff During the Test Program

CPA Part 2 will describe how Blue Energy develops its staffing-augmentation strategy, including anticipated use of an operating partner, experienced utility personnel, GVH resources, vendors, contractors, and other SMEs.

Table 14.1-1: Objectives of First-of-a-Kind Testing

System/Component Design Description	Test Objective
Isolation Condenser System (ICS) - The ICS consists of three independent trains, each containing a heat exchanger or isolation condenser submerged in a pool of water and connected to the reactor vessel by steam-supply and condensate-return piping. When operating, the ICS removes heat from reactor coolant and rejects it to the environment, providing overpressure protection by maintaining the reactor vessel below design limits. The use of the ICS for overpressure protection, instead of traditional safety or relief valves, is a new design feature not previously demonstrated at an operating nuclear station.	Full-scale prototype testing has been accomplished for the heat exchanger and condensate return to the reactor vessel. During the startup phase, an FOAK test places the ICS in service, determines heat-exchange capacity, and confirms prototype testing results within the limitations of the test program and site conditions.
ICS Condensate Return to the Chimney - When the ICS operates, the condensate-return line to the reactor vessel is open. Subcooled condensate stored in the ICS during standby enters the reactor-vessel chimney, thereby providing additional inventory, condensing steam, and increasing fluid density in the chimney. Placement of the ICS condensate returns in the chimney above the core, rather than in the downcomer as in previous designs, provides thermal-hydraulic benefits and is a new design feature.	The return line to the reactor-vessel chimney region was modeled and analyzed using computational fluid dynamics and TRACG. Scaled testing with steam/water at reactor conditions at the Hitachi Utility Steam Test Leading facility was used to benchmark the following key aspects of the Computational Fluid Dynamics and TRACG modeling of ICS condensate return to the chimney: • Interfacial shear characteristics between steam and liquid phases • Steady-state flow patterns of two-phase flows During the startup phase, a FOAK test confirms the accuracy of the predictions of the BWRX300 core flow when ICS is operating based on the measured temperatures at the core inlet.

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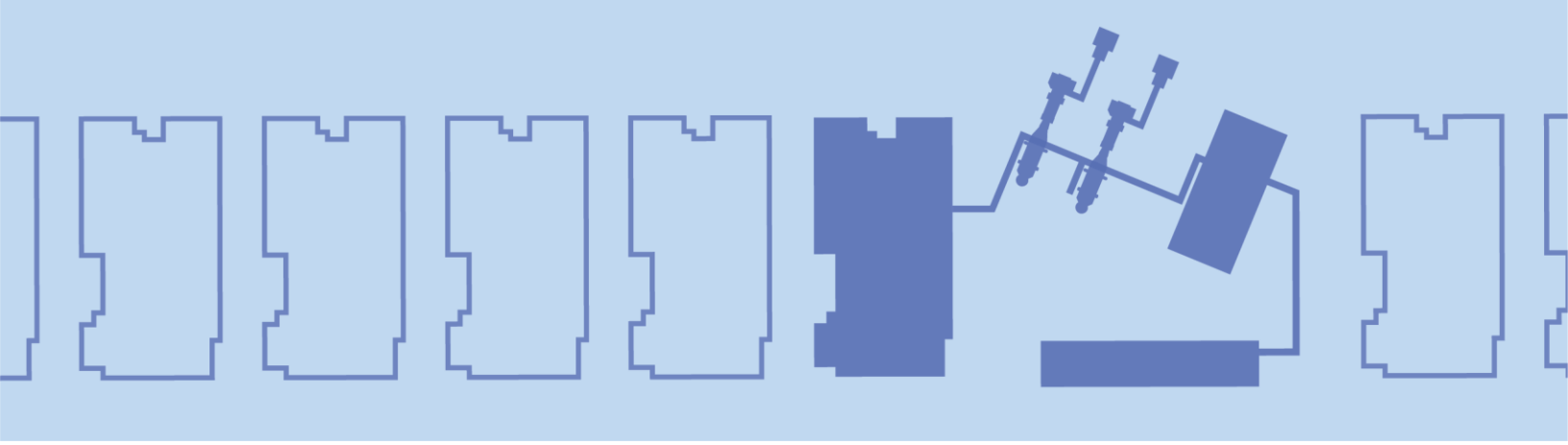
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Non-Proprietary Information

Chapter 15
Sections 15.1-15.4
Safety Analyses (Part 1)



ACRONYM LIST

Acronym	Explanation
AC	Alternating Current
ACRW	All Control Rod Withdrawal at Power
ALWR	Advanced Light Water Reactor
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
AST	Alternative Source Term
ATLM	Automatic Thermal Limit Monitor
ATS	Anticipatory Trip System
BDBA	Beyond Design Basis Accident
BE	Best Estimate
BIS	Boron Injection System
BL	Baseline
BL-AOO	Baseline Abnormal Operational Occurrence
BL-DBA	Baseline Design Basis Accident
BL-DSA	Baseline Deterministic Safety Analysis
BOP	Balance of Plant
BWR	Boiling Water Reactor
CAFTA	Computer Aided Fault Tree Analysis
CB	Control Building
CCDP	Conditional Core Damage Probability
CCF	Common Cause Failure
CDF	Core Damage Frequency
CET	Containment Event Tree
CIV	Containment Isolation Valve
CN	Conservative
CN-DBA	Conservative Design Basis Accident
CN-DSA	Conservative Deterministic Safety Analysis
CPR	Critical Power Ratio
CRD	Control Rod Drive

Acronym	Explanation
CRDA	Control Rod Drop Accident
CRDM	Control Rod Drive Motor
CSAU	Code Scaling, Applicability, and Uncertainty
CUW	Reactor Water Cleanup System
CWS	Circulating Water System
DBA	Design Basis Accident
DCF	Dose Conversion Factor
DCIS	Distributed Control and Information System
DEC	Design Extension Condition
DF	Decontamination Factor
DL	Defense Line
D-in-D	Defense-in-Depth
DSA	Deterministic Safety Analysis
EAB	Exclusion Area Boundary
ECCS	Emergency Core Cooling System
EHE	External Hazard Evaluation
EOC	End of Cycle
EOP	Emergency Operating Procedure
EOR	End of Rated Cycle
EPRI	Electric Power Research Institute
ESBWR	Economic Simplified Boiling Water Reactor
EX-DBA	Extended Design Basis Accident
EX-DEC	Extended Design Extension Condition
EX-DSA	Extended Deterministic Safety Analysis
FFA	Functional Failure Analysis
FHA	Fuel Handling Accident
FLEX	Diverse and Flexible Coping Strategy
FMCRD	Fine Motion Control Rod Drive
FMEA	Failure Modes and Effects Analysis
FSF	Fundamental Safety Function
FV	Fussell-Vesely
FW	Feedwater

Acronym	Explanation
FWCIV	Feedwater Containment Isolation Valve
FWFI	Feedwater Flow Increase
FWH	Feedwater Heater
FWPT	Feedwater Pump Trip
GDC	General Design Criteria
GVH	GE Vernova Hitachi Nuclear Energy
HCU	Hydraulic Control Unit
HEP	Human Error Probabilities
HFE	Human Factors Engineering
HFEA	Human Failure Event Analysis
HRA	Human Reliability Analysis
HVAC	Heating, Ventilation, and Air Conditioning
I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency
ICRW	Inadvertent Control Rod Withdrawal at Power – Single Rod
ICS	Isolation Condenser System
IE	Initiating Event
IHE	Internal Hazard Evaluation
II	Inventory Increase
IR	Inventory Reduction
LFWH	Loss of Feedwater Heating
LOCA	Loss-of-Coolant Accident
LOPP	Loss-of-Preferred Power
LPSD	Low Power Shutdown
LPZ	Low Population Zone
LRF	Large Release Frequency
LR-TT	Load Reduction – Turbine Trip
MAAP	Modular Accident Analysis Program
MCC	Motor Control Center
MCCI	Molten Core Concrete Interaction
MCPR	Minimum Critical Power Ratio
MCR	Main Control Room

Acronym	Explanation
MCS	Minimal Cutset
MOC	Middle of Cycle
MRBM	Multi-Channel Rod Block Monitor
MS	Main Steam
MSCIV	Main Steam Containment Isolation Valve
MSL	Main Steam Line
MSRIV	Main Steam Reactor Isolation Valve
NBR	Nuclear Boiler Rated
NBS	Nuclear Boiler System
NPP	Nuclear Power Plant
OBE	Operating-Basis Earthquake
OLMCPR	Operating Limit Minimum Critical Power Ratio
OPEX	Operating Experience
PA	Postulated Accident
PCCS	Passive Containment Cooling System
PCT	Peak Cladding Temperature
PCW	Plant Cooling Water System
PDC	Principal Design Criteria
PI	Pressure Increase
PI-AOO	Pressure Increase- Abnormal Operational Occurrence
PIE	Postulated Initiating Event
PIRT	Phenomena Identification and Ranking Table
POS	Plant Operating State
PPS	Plant Pneumatics System
PRA	Probability Risk Analysis
PSA	Probabilistic Safety Assessment
PSAR	Preliminary Safety Analysis Report
PSIG	Pounds per Square Inch Gauge
PWR	Pressurized Water Reactor
QHO	Quantitative Health Objectives
RAW	Risk Achievement Worth
RB	Reactor Building

Acronym	Explanation
RCPB	Reactor Coolant Pressure Boundary
RI	Reactivity Increase
RI-AOO	Reactivity Increase- Abnormal Operational Occurrence
RIV	Reactor Isolation Valve
RPC	Reactor Pressure Control
RPF	Radial Peaking Factor
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RWB	Radwaste Building
SA	Severe Accident
SAA	Severe Accident Analysis
SAFDL	Specified Acceptable Fuel Design Limits
SAM	Severe Accident Management
SAMG	Severe Accident Management Guideline
SC1	Safety Class 1
SC2	Safety Class 2
SC3	Safety Class 3
SCR	Secondary Control Room
SDC	Shutdown Cooling System
SDG	Standby Diesel Generator
SMR	Small Modular Reactor
SPSA	Seismic Probabilistic Safety Assessment
SRP	Standard Review Plan
SSC	Structures, Systems, and Components
SS-DBA	Safe-Shutdown Design Basis Analysis
STP	Simulated Thermal Power
TAF	Top of Active Fuel
TB	Turbine Building
TBV	Turbine Bypass Valve
TCV	Turbine Control Valve
TD	Temperature Decrease
TEDE	Total Effective Dose Equivalent

Acronym	Explanation
TRACG	Transient Reactor Analysis Code General Electric
TS	Technical Specifications
TSV	Turbine Stop Valve
URD	Utility Requirements Document
USNRC	United States Nuclear Regulatory Commission
ΔCPR/ICPR	Delta Critical Power Ratio Over Initial Critical Power Ratio

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None.

15.0 SAFETY ANALYSES

This chapter addresses BWRX-300 plant-specific safety analyses.

The BWRX-300 Safety Strategy uses a Defense-in-Depth (D-in-D) safety concept providing a layered and iterative approach to plant safety while meeting United States Nuclear Regulatory Commission (USNRC) regulatory requirements and guidance. The BWRX-300 Safety Analyses incorporate selected guidance from the International Atomic Energy Agency (IAEA) Safety Standards Specific Safety Requirements No. SSR-2/1, "Safety of Nuclear Power Plants: Design," (Reference 15.1-1) as described in NEDC-33934P, "BWRX-300 Safety Strategy," (Reference 15.1-2). Relevant areas in performing the safety analyses from NEDC-33934P include:

- How Defense Line (DL) functions are utilized to assure performance of Fundamental Safety Functions (FSFs)
- What fundamental safety properties and Deterministic Safety Analyses (DSA) acceptance criteria and Probabilistic Safety Assessment (PSA) safety goals are used
- How the safety strategy is implemented for DSA:
  - Potential Postulated Initiating Events (PIEs) are identified by Plant-Level Failure Analyses, which are:
    - Functional Failure Analysis (FFA)
    - Human Failure Event Analysis (HFEA)
  - The Fault Evaluation takes inputs from the Plant-Level Failure Analyses and organizes a comprehensive and representative set of PIE and event sequences that define the complete scope of the DSA performed across all plant modes and event categories by performing:
    - Deterministic PIE Selection
    - Complex Sequence Selection
- The PIEs and event sequences identified in the Fault Evaluation are included in the DSA, which demonstrates that applicable acceptance criteria are met based on the characteristics of the plant design. The applicable layers of DSA include:
  - Baseline DSA (BL-DSA)
  - Conservative DSA (CN-DSA)
  - Extended DSA (EX-DSA)
- What key safety design concepts are used in the design process:
  - Independence
  - Reliability and Redundancy
  - Diversity and Protection against Common Cause Failure
  - Simplicity in Design
  - Practical Elimination
  - Avoiding Cliff Edge Effects

- How Structures, Systems, and Components (SSCs) are Classified:
  - DL Correlation to Functional Safety Category
  - Assignment of Safety Class to SSCs
  - Reliability Targets for DL Functions
  - Seismic Categorization

The BWRX-300 Safety Strategy framework considers the functional and consequential safety-importance of each SSC to determine the design requirements and appropriate safety classification for each SSC. SSC classification is described in Section 3.2.

This chapter includes the following safety analyses complying with the requirements of 10 CFR Part 50, Domestic Licensing of Production and Utilization Facilities, especially 10 CFR 50.34 Contents of applications; technical information, 10 CFR 50.46, Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors, 10 CFR 50, Appendix A, General Design Criteria (GDC) for Nuclear Power Plants, and 10 CFR 100, Reactor Site Criteria:

- DSA
- PSA

The BWRX-300 design basis is achieved through an iterative safety process. The design is implemented to meet defined safety objectives that are confirmed via the safety analyses. Results of the safety analyses provide feedback to the design. If indicated by the results, the design may be modified until safety objectives are met.

This chapter documents the safety objectives and the safety analyses approach, description and results performed for the BWRX-300 design. The DSA descriptions in the chapter conform with the intent of NUREG-0800 Standard Review Plan (SRP) Chapter 15 Safety Analyses. However, NEDC-33934P justifies an alternative approach to SRP Section 15.0. Identification of Regulatory Treatment of Non-Safety Systems (RTNSS) components is not necessary, as the BWRX-300 Safety Strategy inherently identifies such components as Safety Class 1 (SC1), Safety Class 2 (SC2), or Safety Class 3 (SC3).

Section 15.1 General Considerations of the BWRX-300 Safety Analyses:

- Safety Analysis Objectives, Scope, and Approach
- Identification of Failures and Hazards - Scope and Approach
- Analysis of Design Basis Conditions - Scope and Approach
- Analysis of Design Extension Conditions (DECs) Without Core Damage - Scope and Approach
- Analysis of DECs with Core Damage (Severe Accident) - Scope and Approach

Section 15.2 Identification, Categorization, and Grouping of Postulated Initiating Events and Accident Scenarios discusses:

- Development of categorizing events according to their frequencies
- Grouping of events according to their types
- Selection of the bounding events for each group type

Section 15.3 Safety Objectives and Acceptance Criteria discusses:

- Acceptance criteria for Anticipated Operational Occurrences (AOOs), Design Basis Accidents (DBAs) and DECs

Section 15.4 Human Actions discusses:

- General Considerations
- Human Actions considered in the DSA
- Human Actions considered in the PSA

Section 15.5 Deterministic Safety Analysis discusses:

- Safety margins in the safety analysis and a description of codes and standards (Subsection 15.5.1)
- Analysis of Boiling Water Reactor (BWR) stability (Subsection 15.5.2)
- Analysis of AOOs (Subsection 15.5.3)
- Analysis of DBAs (Subsection 15.5.4)
- Analysis of DECs without core damage (Subsection 15.5.5)
- Analysis of DECs with core damage (Subsection 15.5.6)
- Analysis of PIEs and accident scenarios associated with the fuel pool (Subsection 15.5.7)
- Analysis of fuel handling accidents (Subsection 15.5.8)
- Analysis of radioactive release from a system or component during Loss-of-Coolant Accidents (LOCAs) outside containment (Subsection 15.5.9)

Section 15.6 Probabilistic Safety Assessment discusses:

- General approach to the PSA (Subsection 15.6.1)
- Probabilistic Safety Assessment Overview (Subsection 15.6.2)
- Level 1 PSA (Subsection 15.6.3)
- Level 2 PSA (Subsection 15.6.4)
- Severe Accident Methodology (Subsection 15.6.5)
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- Uncertainty and Sensitivity Analysis (Subsection 15.6.7)
- Model Integration and Quantification (Subsection 15.6.8)
- Results of the Level 1 Probabilistic Safety Assessment (Subsection 15.6.9)

Section 15.7 Summary Results of the Safety Analyses discusses:

- Analysis results of BWR stability (Subsection 15.7.1)
- Analysis results of AOOs (Subsection 15.7.2)
- Analysis results of DBAs (Subsection 15.7.3)
- Analysis results of DECs without core damage (Subsection 15.7.4)

- Analysis results of DECs with core damage (Subsection 15.7.5)
- Analysis results of fuel pool events (Subsection 15.7.6)
- Analysis results of fuel handling accident (Subsection 15.7.7)
- Analysis results of radioactive releases from LOCAs outside containment (Subsection 15.7.8)
- Analysis results of the PSA (Subsection 15.7.9)

Appendix 15A discusses practically eliminated claims and provisions.

Appendix 15B discusses complementary design features for mitigating DECs.

Appendix 15C provides an evaluation assuming a fission product release from the core into containment to assess compliance with 10 CFR 50.34(a)(1)(ii)(D). The evaluation is based on a maximum hypothetical accident (MHA) event sequence that results in a core melt rather than an event sequence meeting a defined frequency threshold.

The dose results following DBA PIEs are compared to the criteria in 50.34(a)(1)(ii)(D). Regulatory Guide (RG) 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” states that the maximum hypothetical accident is one “whose occurrence during the lifetime of the facility would appear to be credible.” The performance of the BWRX-300 design is assessed in Subsection 15.5.4.5 for two such DBA PIEs: large-break LOCAs for breaks in main steam piping and feedwater piping inside containment. Although these PIEs are similar to those events typically considered for Light Water Reactors (LWR), the safety analyses results show that there is no fuel damage as a result of these events. Due to the inherent safety of the BWRX-300 design, there is not an accident whose occurrence during the lifetime of the facility is credible that results in substantial meltdown of the core with subsequent release into containment of appreciable quantities of fission products.

15.1 General Considerations of the BWRX-300 Safety Analyses

The BWRX-300 Safety Strategy framework integrates the DLs provided by the implementation of the D-in-D concept with the safety analyses. The D-in-D concept uses insights gained from operating experience, deterministic analyses, and risk-informed analyses.

Design features, functions, and practices are organized into DLs that protect the integrity of the physical barriers against radioactive releases. Safety analyses are performed that demonstrate the effectiveness of the SSCs necessary to perform the functions assigned in various DLs that are credited to mitigate PIEs. The SSCs necessary to achieve the DL functions are described in Chapters 4 through 12. The PIEs analyzed are identified and selected through the systematic process of fault evaluation described in Section 15.2.

The safety analyses prove that the plant design meets the underlying safety objectives and acceptance criteria for event mitigation and confirms that the applicable regulatory safety objectives are met.

15.1.1 Safety Analysis Objectives, Scope, and Approach

The safety analysis objectives include:

- Demonstrating the design meets the acceptance criteria established following a graded approach for each plant state. The graded approach application may lead to acceptance criteria more restrictive for events with higher occurrence probability.
- Provide analytical basis to support the derivation of the plant Technical Specifications (TS) for normal operation

- Provide analytical basis for establishing and validating accident management procedures and guidelines

The safety analyses scope includes the following plant states:

- Normal operation
- Anticipated Operational Occurrences (AOOs)
- Design Basis Accidents (DBAs)
- Design Extension Conditions (DECs) with or without core damage (i.e., Beyond Design Basis Accidents (BDBAs))

The BWRX-300 DSA uses a layered analysis approach that includes three types of DSA evaluations:

- BL-DSA
- CN-DSA
- EX-DSA

This approach addresses initiating events and mitigating DL function failures in a more systematic and structured manner than past approaches.

These DSA acceptance criteria are discussed in Section 15.3. The DSA results are compared against the applicable plant state acceptance criteria and dose limits specified in Subsection 15.3.1.

The PSA is performed to complement the DSA. PSA estimates the overall risk presented by the facility that is compared to the regulatory safety goals specified in Subsection 15.3.2. The PSA is presented in Section 15.6.

15.1.2 Identification of Failures and Hazards

A critical part of the safety analysis is to systematically and comprehensively identify SSC functional and human failures that initiate a PIE, initiate a hazard that leads to a PIE or worsen a hazard that leads to a PIE. The BWRX-300 Safety Strategy process identifies two plant-level failure analyses to identify such failures: Functional Failure Analysis and Human Failure Event Analysis. The scope of each failure analysis encompasses the complete range of normal plant states (i.e., full power, low power, shutdown and refueling) as the type and consequence of each failure may differ depending on the plant state. The analyses also consider all sources of radioactivity (e.g., spent fuel, fuel being handled, radioactive waste) in addition to the reactor core itself.

In addition to failure analyses, hazard evaluations are performed to ensure that all potential hazards the plant might experience are identified and considered. The BWRX-300 Safety Strategy identifies two hazard evaluations: External Hazard Evaluation (EHE) and Internal Hazard Evaluation (IHE). A primary objective of each hazard evaluation is to identify hazards with the potential to initiate a PIE and pass that list of hazards to the relevant downstream analysis, PSA and/or deterministic hazards analyses (e.g., seismic hazard analyses, fire hazards assessment, pipe rupture hazards analysis). The focus of the EHE and IHE is identifying the list of credible hazards and defining the expected frequencies of those credible hazards. The plant is designed to withstand the hazards while maintaining performance of the FSFs. Subsection 15.6.2.3 summarizes the PSA screening process for hazards. The deterministic hazards analyses are described in other PSAR sections, as required. External events are site-specific and are specified in the site evaluation provided in Chapter 2. Examples of internal hazards include fires (discussed

in Section 9.5), explosions/missiles from rotating or pressurized equipment (discussed in Section 3.5), collapse of structures/falling objects (discussed in Sections 3.7 and 3.9), and pipe whip, jet effects and flooding (discussed in Section 3.6).

15.1.2.1 Functional Failure Analysis

The FFA identifies failures of plant systems or equipment with potential to cause a challenge to an FSF. Potential failures are identified in Failure Modes and Effect Analyses (FMEAs) performed on the plant systems.

The FFA is limited to random single failures and to Common Cause Failures (CCFs). Single failures identified during the FFA include single component failures occurring during normal operations that result in an initiating event. If a single component failure results in the loss of capability of other components, the resulting initiating event includes the effects of these failures.

The system FMEAs are reviewed to identify failures that cause challenges to FSFs. A consolidated list of relevant functional failures from all system FMEAs is generated and organized.

The FFA potential PIEs are organized by quantitative frequency using the frequency ranges defined in the BWRX-300 Safety Strategy.

15.1.2.2 Human Failure Event Analysis

The HFEA identifies those human failure events that could potentially initiate abnormal or accident event sequences that primarily involve errors made by personnel during normal operations while changing the state of plant equipment and errors made by personnel during a planned maintenance activity resulting in an unintended change in the state of plant equipment. If a single human failure event results in the loss of capability of multiple components, the resulting initiating event includes the effects of these failures.

Many human failure events produce the same effects as corresponding equipment failures, and the effects of these are included in the FFA described in Subsection 15.1.2.1. The HFEA identifies unique human failure events not identified by FFA, such as an operator initiating a group command on multiple actuators that is beyond what is considered in a single failure analysis of a particular system.

Inputs needed to complete the HFEA are incomplete at the PSAR stage. However, based on the analyses presented in Section 15.5 and the margin to acceptance criteria for these events, there is enough information about the BWRX-300 preliminary design and its ability to mitigate a wide range of events to assess the risk to public health and safety. By the Final Safety Analysis Report (FSAR) submittal the completed HFEA provides a list of potential PIE sequences based on human failure events (including both unique human failure events and human failure events that produce the same effect as corresponding equipment failures) with estimated quantitative frequencies that are evaluated in the fault evaluation (see Section 15.2). Also, the HFEA identifies human failure events which may be modeled further in the Human Reliability Analysis (HRA) discussed in Subsection 15.6.3.

15.1.3 Analysis of Design Basis Conditions

The BWRX-300 design basis conditions are normal operations, AOOs and DBAs described below:

1. **Normal Operation** is operation within specified TS (see Chapter 16) and includes the full range of plant operating modes.
2. **AOOs** are deviations from normal operation that are expected to occur at least once during the operating lifetime of the reactor facility.

3. **DBA** conditions are identified as deviations from normal operations that are less frequent and more severe than AOOs.

Acceptance criteria applicable to the DSA for each plant state is discussed in Subsection 15.3.1.

15.1.4 Analysis of Design Extension Conditions Without Core Damage

DECs are postulated accident conditions that are less frequent than DBAs. DECs may occur with or without core damage.

EX-DSA is performed for DECs without core damage demonstrating that radiological releases associated with reactor coolant are maintained within acceptable limits and support the PSA determination of no core damage.

DEC analysis include:

- AOO PIEs mitigated using DL2 and DL3 hydraulic scram functions in the BL-DSA and CN-DSA respectively. The same AOO PIE is analyzed assuming CCF of the mechanical equipment providing motive force for hydraulic scram.
- DBA PIEs caused by a single equipment failure, for which DL3 functions were credited in both the BL-DSA and CN-DSA. The same DBA PIE is analyzed assuming DL3 CCF.
- PIEs caused by single equipment failures in the DEC frequency category and complex sequences
- PIEs initiated by a CCF of DL3 function

The results of the EX-DSA for DECs without core damage are discussed in Section 15.7.

15.1.5 Analysis of Beyond Design Basis Accidents with Core Damage (Severe Accident)

Severe Accidents (SAs) involve a catastrophic failure, core damage, and fission product release. Representative DECs with core damage are postulated to provide inputs for the containment design and safety features ensuring containment functionality. This set of accidents is considered in the design of corresponding safety features for DECs and represents a set of bounding cases.

SA sequences are selected that identify representative core damage scenarios and corresponding plant damage states that are used as the basis for performing the Severe Accident Analysis (SAA). The scope of SA sequence selection corresponds to sequences involving significant core damage that could lead to a containment breach and radioactive release analyzed in the Level 2 PSA methodology, which will be described in Subsection 15.6.4. that will be provided in the FSAR. Selected SA sequences are included in the fault list (see Section 15.2).

The SAA goal is to provide input to accident management for terminating the progression of core damage, maintaining containment integrity as long as possible, and minimizing on-site and offsite radioactive material releases.

Design features are incorporated that minimize the risk of core damage event progression (see Appendix 15B). The response to SAs considers the use of permanent and temporary systems and equipment that are beyond their originally intended functions.

15.1.6 Practical Elimination

PIEs or event sequences leading to or involving core damage are considered for practical elimination based upon completing the Level 2 PSA methodology discussed in Subsection 15.6.4 that is provided in the FSAR. Practical elimination is discussed in NEDC-33934P and Appendices 15A and 15B.

Consistent with IAEA SSR 2/1, Paragraph 2.11, practical elimination is applied to events or sequences of events leading to or involving core damage (an SA) where confinement of radioactive materials cannot be reasonably achieved. Event sequences that are either physically impossible or extremely unlikely to occur are considered for practical elimination.

The practical elimination demonstration is performed with accident conditions and phenomena knowledge and is substantiated by relevant evidence (see Appendix 15A Practical Elimination Claims and Provisions). The practically eliminated event sequences with core damage will be finalized and provided in the FSAR.

15.1.7 References

15.1-1 IAEA SSR-2/1, “Safety of Nuclear Power Plants: Design,” International Atomic Energy Agency, 2016.

15.1-2 NEDC-33934P, “BWRX-300 Safety Strategy,” Revision 2, GE Vernova Hitachi Nuclear Energy Americas, LLC, March 2026.

15.2 Identification, Categorization and Grouping of Postulated Initiating Events and Accident Scenarios

A fundamental element of the safety analyses is the identification and selection of PIEs that is achieved through the systematic process of fault evaluation. The fault evaluation objective includes:

- Identification, categorization and grouping of PIEs
- Identification of the plant functions credited in the safety analyses and their assignment to a functional DL (DL2, DL3, DL4a, and DL4b). DL functions are shown in Table 15.5-5, Table 15.5-44, and Table 15.5-45.

The fault evaluation scope is the list of potential PIEs generated by the FFA and the HFEA (see Subsection 15.1.2) and includes:

- Complete range of operating modes
- All radioactivity sources (reactor core and outside core)
- Single failure PIEs and CCF PIEs caused by equipment failures or single human failure event. CCFs are analyzed as software platform or mechanical failures.
- Event sequences postulated to demonstrate the adequacy of the BWRX-300 D-in-D design—consisting of a PIE, an assumed CCF of an entire DL (e.g., DL2 for AOOs and DL3 for DBAs), and the DL function successes that mitigate the event sequence (e.g., DL3 for AOOs and DL4a for DBAs)

The output of the fault evaluation is documented in a fault list. The fault list establishes traceability between the plant design and the safety analysis. The fault evaluations start in parallel with or prior to DSA and PSA activities. DSA and PSA mature with the design and the fault list is updated accordingly.

The fault evaluation process used for PIE identification and selection produces results that are generally consistent with the events identified as AOOs and DBAs in NUREG-0800 SRP 15.0 Introduction – Transient and Accident Analysis and complies to the regulations referenced therein.

The fault evaluation includes the following activities in the BWRX-300 Safety Strategy Implementation Process:

- Deterministic PIE Selection
- Complex Sequence Selection
- SA Sequence Selection

Deterministic PIE Selection

The deterministic PIE selection is the systematic process in organizing and selecting events for deterministic safety analyses. The activities performed during the deterministic PIE selection are described in Subsection 15.2.1. Selected PIEs and event sequences are allocated to three DSA types in a fault list:

- PIE List for BL-DSA
- PIE/Event Sequence List for CN-DSA
- PIE/Event Sequence List for EX-DSA

Complex Sequence Selection

Complex sequences are event sequences involving failures of multiple mitigating features, which have not been included in the deterministic PIE selection but are identified in the Level 1 PSA as having the potential to lead to core damage with a frequency of occurrence or consequences judged to require analysis and DL mitigation function. These complex sequences are added to the fault list and analyzed in the EX-DSA.

Severe Accident Sequence Selection

The scope of SA sequence selection corresponds to those sequences involving significant core damage, which could lead to a breach of containment and radioactive release in the Level 2 PSA that is described in Subsection 15.6.4. To the extent that core damage is not practically eliminated, representative SA sequences (DECs with core damage) are postulated and analyzed in the SAA. The primary objective of the SA sequence selection is to identify representative core damage scenarios and define corresponding plant damage states that are used as the basis for performing the SAA. The selected SA sequences are documented in the fault list and are analyzed in the SAA.

15.2.1 Basis for Categorization of Postulated Initiating Events, Accident Scenarios and Fault Evaluation

The FFA and HFEA described in Subsection 15.1.2 results in the list of potential PIEs. The plant level failure analyses address a complete range of plant modes of operation, all sources of radioactivity, and any consequential failure that occurs because of the PIE. During design development, the initial fault evaluation is validated, and PIE selection is updated accordingly.

Ultimately, the set of evaluations and analyses demonstrate that the D-in-D design approach leads to an overall plant design that meets specific acceptance criteria per event category. The approach is a layered, comprehensive, and systematic way to evaluate events, apply DL functions, and specify related requirements for the design in an iterative, progressive manner. This process provides high confidence that once these evaluations and analyses are finalized, the plant design can be demonstrated to meet overall safety goals and has successfully implemented the BWRX-300 Safety Strategy.

The activities included in the deterministic PIE selection and their bases are presented below:

1. Event Sequences development – an Event Sequence is developed starting with a PIE and considers the success or failure of the required mitigating functions. The DL of each credited mitigating function is established for each Event Sequence.
2. Event sequences are grouped into fault groups based on similar effect on a certain plant parameter. For example, events that lead to pressure increase in the reactor such as inadvertent closure of the Turbine Stop Valves (TSVs) and/or Turbine Control Valves (TCVs) or inadvertent closure of the Main Steam Reactor Isolation Valves (MSRIVs) are grouped in the pressure increase fault group. Subsection 15.2.4 includes the output of the grouping activity.
3. Event sequences are categorized within each fault group as AOO, DBA or DEC based on their frequency of occurrence, with some exceptions described in NEDC-33934P, “BWRX-300 Safety Strategy,” (Reference 15.2-1). Subsection 15.2.2 describes this categorization.
4. Plant conditions are defined corresponding to each PIE supporting the scenario analysis.
5. Any exceptions are applied or justified to the standard PIE selection.

Event sequences are allocated to three types of DSAs:

- BL-DSA
- CN-DSA
- EX-DSA

A bounding set of PIEs and event sequences that result in the most significant challenge to the FSFs are selected for evaluation in the DSA. DSA layers and events categories are combined so that limiting baseline events are Baseline Anticipated Operational Occurrence (BL-AOO), the limiting Conservative Design Basis Accident (CN-DBA) and limiting Extended Design Extension Condition (EX-DEC). This notation is used to identify the layer and event category.

Subsection 15.2.4 describes the bounding event selection for each fault group that is captured in a fault list.

A description of the three deterministic safety analyses aligned with the functional DLs (DL2, DL3, and DL4a) is included below.

15.2.1.1 Baseline Deterministic Safety Analysis

The primary objective of the BL-DSA is to model the expected plant response to AOO and DBA PIEs assuming all functions, regardless of safety category, respond as designed (except for the initiating PIE) to mitigate the event. The scope of the BL-DSA includes single failure PIEs categorized as bounding BL-AOOs, and Baseline Design Basis Accidents (BL-DBAs) and DBA PIEs caused by a CCF in DL2 or DL4a functions. The BL-DSA models the expected response of the plant (no failure is postulated) to demonstrate that the event meets applicable acceptance criteria. The analysis end point is the controlled state condition. The protecting or mitigating DL functions credited in the BL-DSA for AOO PIEs are DL2 functions such as Anticipatory Trip System (ATS), maintain target level, power or pressure, or control rod block functions.

15.2.1.2 Conservative Deterministic Safety Analysis

The primary objective of the CN-DSA is demonstrating the effectiveness of DL3 functions, inherent safety features, or passive functions in mitigating AOO and DBA PIEs. The CN-DSA scope includes events categorized as bounding CN-DBAs:

- AOO PIEs caused by a single equipment failure or a single human failure event, plus an assumed CCF of DL2 functions credited for mitigation of the same PIE in the BL-DSA
- DBA PIEs caused by a single equipment failure plus an assumed CCF of any DL2 function(s) credited for mitigation of the same PIE in the BL-DSA
- DBA PIEs caused by a CCF in DL2 or DL4a functions, plus an assumed CCF of any non-DL3 functions credited for mitigation of the same PIE in the BL-DSA

The CN-DSA is performed using conservative initial conditions with established acceptance criteria and applying a graded approach in quantifying the uncertainties (see Subsection 15.5.1.1). Single failure criterion is applied to SC1 SSCs. CN-DSA credits only DL3 mitigation functions. DL2 and DL4a functions that make the event more severe are assumed to function unless failure of the function is a consequence of the PIE or event sequence. The end point of the analysis is a controlled state condition.

15.2.1.3 Extended Deterministic Safety Analyses

The EX-DSA primary objectives are demonstrating a second functional DL against DBA PIEs where DL3 functions were credited in both the BL-DSA and CN-DSA and demonstrating the plant's capability to avoid core damage in very unlikely event scenarios involving combinations or

types of mitigation failures that are beyond those deterministically postulated ("complex sequences"). DBAs initiated by CCF in lower class equipment are not deterministically included in the EX-DSA with an additional CCF in DL3 functions added as this would constitute two CCFs occurring simultaneously in independent and diverse technologies that is outside the deterministic plant design basis. The EX-DSA scope includes events categorized as DECs:

- AOO PIEs mitigated using DL2 and DL3 hydraulic scram functions in the BL-DSA and CN-DSA, respectively. The same AOO PIE is analyzed assuming a CCF of the mechanical equipment providing motive force for hydraulic scram.
- DBA PIEs caused by a single equipment failure where DL3 functions were credited in both the BL-DSA and CN-DSA. The same DBA PIE is analyzed assuming DL3 CCF.
- PIEs initiated by DL3 CCF
- PIEs caused by single equipment failures that are in the DEC frequency category
- Complex sequences

The EX-DSA can credit unaffected functions in DL2, DL3, and DL4a. Unaffected functions are those not impaired by the initiating failure, by a mitigation failure postulated as part of an event sequence, or by environmental conditions present during an event sequence.

15.2.2 Categorization of Events According to Their Frequencies

One fundamental element of the deterministic PIE selection and event sequence selection is the assignment of event sequences to categories based on their frequency of occurrence, with some exceptions described in NEDC-33934P Section 2.2.3:

- AOO (frequency greater than 1E-02 per reactor-year)
- DBA (frequency between 1E-02 and 1E-05 per reactor-year)
- DEC (frequency less than 1E-05 per reactor-year)

Qualitative frequencies are adopted as an interim measure and are used in the early design stages to progress the performance of deterministic analyses prior to availability of more mature PSA information. Qualitative frequencies are used to categorize events for the PSAR. Quantitative frequencies based on Level 1 PSA results are adopted as the final, governing measure of the event sequence category for the FSAR. The methodology for selection of event frequency is described in Subsection 15.6.3.1.

An event sequence consists of a PIE, an assumed failure of a mitigating function(s), and the DL function success that mitigates the PIE. The event sequence category is based on the sequence frequency, not the frequency of the PIE which initiates the sequence. The event category assigned to an event sequence may be different than the event category assigned to the PIE that initiated the sequence because the event sequence may include additional failures that make the sequence less likely to occur.

In addition to the event categorization frequency, the categorized events are allocated the following DSA types:

- BL-AOO
- CN-DBA
- EX-DEC

A fundamental concept of the BWRX-300 Safety Strategy is that an event sequence that assumes failure of DL functions capable of mitigating the sequence (i.e., a CCF of all functions in a DL that are credited to mitigate a given PIE) has a lower frequency of occurrence than if the DL functions were not assumed to fail. The assumed frequency of such an event sequence is the frequency of the PIE multiplied by the required maximum frequency of failure of the DL mitigation functions postulated to fail. For example, an event sequence for a PIE with a frequency in the AOO range where an assumed failure of DL2 functions to mitigate the PIE has a frequency equal to the PIE frequency multiplied by the target reliability value for DL2 functions results in a sequence frequency in the DBA range. Accordingly, such an event sequence is evaluated as a DBA using acceptance criteria associated with DBAs. The target reliability values used are:

- 1E-02 failures/demand for DL2
- 1E-04 failures/demand for DL3
- 1E-03 failures/demand for DL4a

For example, the reliability of DL2 is such that its assumed failure to mitigate the AOO PIE creates an event sequence in the DBA event category. A CN-DSA is then performed to demonstrate DL3 mitigation of the same PIE but assuming the DL2 failure occurs. Similarly, the reliability of DL3 is such that its assumed failure to mitigate a DBA PIE creates an event sequence in the DEC event category. An EX-DSA is then performed to demonstrate DL4a mitigation of the same PIE but assuming the DL3 failure occurs.

15.2.3 Grouping of Events According to Type

One of the steps in fault evaluation is grouping the events according to their type. The fault evaluation includes human failure events, functional failures, Level 1 PSA complex sequences, and Level 2 PSA SA sequences.

PIEs (faults) are grouped according to the resultant change in plant parameter:

- Temperature decreases events – decrease in core coolant temperature
- Pressure increases events – increase in reactor pressure
- Reactivity increases events – reactivity and power distribution anomalies
- Inventory increase events – increase in reactor coolant inventory
- Inventory reduction events – decrease in reactor coolant inventory
- Non-reactor events – these events are non-core related such as fuel handling accident

Once the fault groups are identified, the anticipated core physics response associated with each group is then selected. Once each response group is identified, the bounding event sequence from that group is then selected.

Table 15.2-1 provides the fault groups with an explanation of how anticipated core physics response (reactor response) was considered during development of the groups. Within these groups, event sequences with similar responses are compared and used to select the bounding events.

15.2.4 Postulated Initiating Events and Accident Scenarios

PIEs and event frequency are first determined qualitatively based on system conceptual design, previous similar designs, and operating experience. PIEs are evaluated in the fault evaluation where they are further screened for inclusion in the fault list.

The bounding event selection is performed for events that are initiated at full power conditions because they are expected to result in the most significant challenge to the fission product barriers (i.e., identification of the AOO and DBA with the most significant challenge to the fission product barriers). Identification of the most limiting scenarios under all conditions such as refueling or low power cannot confidently be performed until the FSAR submittal. The analyses presented in the PSAR are intended to provide confidence in the BWRX-300 design capabilities and design margins. The analyses to be presented in the FSAR are intended to set operational limits and include identification of the most limiting scenarios based on the final design.

Bounding events are selected in each fault group, for each event category (e.g., AOO, DBA, DEC without core damage) and for the applicable DSA layer (e.g., baseline, conservative and extended). The resulting events selected are listed in Table 15.2-2 and analyzed in Section 15.5. DEC events with core damage are part of the PSA and SAA.

The bounding event selection is performed for two event categories:

- Transient or non-Loss-of-Coolant-Accident (non-LOCA) described in Subsection 15.2.4.1 through Subsection 15.2.4.5
- LOCA scenarios described in Subsection 15.2.4.6

Bounding Event Selection for Transient Events

Bounding events are selected for the transient (non-LOCA) DSA that pose the most challenges in meeting the derived acceptance criteria.

The selected bounding events are summarized in Table 15.2-2. Table 15.2-2 also points to a complete description of the bounding event in the DSA described in Section 15.5.

15.2.4.1 Decrease in Core Coolant Temperature Bounding Event

Events that result in core coolant temperature decreases are grouped as Temperature Decrease (TD) faults. A reduction in coolant temperature (at the core inlet) has the potential to challenge the fuel cladding barrier due to increasing reactivity as a result of reducing the core coolant void fraction. The Reactor Coolant Pressure Boundary (RCPB) is not challenged because there is not a significant increase in steam flow and normal pressure control is not affected. Because this is a reactivity driven event, it is only of concern when the reactor is not shut-down.

The largest source of coolant supply is from the Feedwater (FW) pumps. FW flow enters the Reactor Pressure Vessel (RPV) downcomer through the FW piping. Extraction steam from the turbine is directed to heat FW in multiple stages of FW heaters. Failures such as loss of extraction steam can result in a reduction of the temperature entering the RPV.

The Isolation Condenser System (ICS), Control Rod Drive (CRD) system, Reactor Water Cleanup System (CUW) and the Shutdown Cooling System (SDC) also have inflows to or outflows from the RPV that have the potential to reduce the coolant temperature. Inadvertent ICS initiation PIEs are included in the increase in reactor coolant inventory fault group. These systems can only reduce the coolant temperature a small fraction relative to Feedwater Heating related PIEs. Therefore, only FW heater related PIEs are considered potentially bounding.

There is no fault group for increase in core coolant temperature for the following reasons:

1. The FW temperature is near the highest temperature that is feasible during normal operation.
2. Any increase in FW temperature would increase the core void fraction and reduce core power due to the decrease in void reactivity.

3. An increase in FW temperature does not result in an increase in core temperature because the core is boiling and remains at saturated conditions.

Other than an increase in the FW temperature, the core coolant temperature may increase due to the following conditions:

- Core power increase resulting in a small increase in pressure covered by the reactivity and power distribution anomalies fault group
- Reactor pressure increase that effects saturation temperature covered by increase in reactor pressure fault groups

Therefore, any possible increase in FW temperature is small and results in a small decrease in power and poses no threat to fission product barriers resulting in no need for an increase in core coolant temperature fault group.

The selected events result in the most significant challenge to the fission product barriers in the fault group. The selected events are summarized in Table 15.2-2.

Bounding AOO Event Selection

The event effect on the fuel Critical Power Ratio (CPR) and core power increase is used in selecting the bounding AOO events for this group.

The AOO that results in the largest reduction in FW temperature is the Loss of Feedwater Heating (LFWH) event. This BL-AOO event described in Subsection 15.5.3.1.1 is the bounding AOO event for this category.

Bounding DBA Event Selection

The bounding CN-DBA event described in Subsection 15.5.4.1.1 resulting in the largest postulated reduction in FW temperature is the CCF leading to loss of all FW heaters. The event effect on fuel Peak Cladding Temperature (PCT) is used in selecting the bounding DBA in this group.

Bounding DEC Without Core Damage Event Selection

There are no DEC events in this fault group because the DLs are established in the baseline and conservative DSA. No complex sequences in the fault group are identified.

15.2.4.2 Increase in Reactor Pressure Bounding Event Selection

Events that result in RCPB pressure increase are referred to as Pressure Increase (PI) faults. During full power operation, steam generated in the reactor exits the RPV through the Main Steam Lines (MSLs). There are normally open valves in each MSL: two MSRIVs and one main steam Containment Isolation Valve (CIV). Outside containment, there is a main steam header upstream of the TSVs. Downstream of the main steam header the TSVs are in series with the TCVs.

The MSRIVs and CIVs close to isolate the RPV or containment. The TSVs and TCVs close to protect the turbine. Turbine Bypass Valves (TBVs) located in the main steam equalizing line in the Turbine Building (TB) allow steam to bypass the closed TSV or TCV.

PI faults are generally caused by closure of valves in the steam flow path. Closure of a single MSRIV terminates steam flow in one of the two MSLs. Closure of any two of these valves in separate MSLs causes a complete isolation of the MSLs upstream of the TBVs. Closure of both TSVs and/or TCVs, terminates steam flow to the turbine. A turbine trip or load rejection signal initiates closure of the TSVs and TCVs, and fast opening of the TBVs. When RPV pressure increases, TBVs are opened by the normally operating Reactor Pressure Control (RPC), which allows up to ~25% of the rated steam flow to the main condenser.

The selected events result in the most significant challenge to the fission product barriers in the fault group. The selected events are summarized in Table 15.2-2.

There is no fault group for decrease in reactor pressure because a pressure decrease reduces reactivity. The decrease in reactor pressure is caused by either:

- Reactor power decrease that is covered by the reactivity at power distribution anomalies fault group
- Loss of reactor coolant inventory that is covered by the decrease in reactor coolant inventory fault groups

Therefore, there is no need for a decrease in reactor pressure fault group.

Bounding AOO Event Selection

The event effect on the fuel CPR, core power increase, and the RCPB pressure are used in selecting the bounding AOO events for this group.

There are several AOO events in the PI group in the fault list. Most of the AOOs are PIEs that result in closure of the TSVs, TCVs, or both. These events present similar challenges to the cladding and RCPB, and all of them are selected as potentially limiting (bounding) events:

- Turbine trip, load rejection
- Loss-of-Preferred Power (LOPP)
- Loss of condenser vacuum

Another AOO in this group is the BL-AOO closure of one MSRIV. The MSRIVs do not close as fast as the TCVs or TSVs. However, the MSRIVs are upstream of the TBVs, resulting in this event selected as potentially limiting.

Bounding DBA Event Selection

The event effect on PCT and the RCPB pressure are parameters used in selecting the bounding DBA events for this group.

There are several DBA events that result in increased pressure in the PI group. There are two main types assumed in the PI group:

1. AOO events where the TSV or TCV closes with a CCF of the DL2 mitigation equipment.
2. CCF results in closure of the MSRIV and Feedwater Reactor Isolation Valve (FWRIV).

Because the TCV fast closure is faster than the MSRIV closure, the closure of the TCVs with failure of the DL2 mitigation functions is a more significant pressure increase event. However, because several of these events result in similar challenges to acceptance criteria, several CN-DBA events (described in Subsection 15.5.4) are chosen as potentially bounding.

Bounding DEC Without Core Damage Event Selection

The event effects on PCT and the RCPB pressure are used in selecting the bounding DEC events for this group.

The main types of PI group DEC events are:

1. Bounding AOO events with failure of the hydraulic scram due to postulated CCF of the hydraulic components.
2. Complex sequence AOO event with half of the control rods fail to insert either by Hydraulic Control Units (HCUs) or with the CRD motors run-in.

3. CCF in DL3 functions initiating events.

The events assume a CCF of the hydraulic scram that significantly challenges the fuel cladding and RCPB because the CRD motor run-in that backs up the hydraulic scram results in slower negative reactivity insertion. Several of these events are selected as potentially bounding. Events resulting from CCF in DL3 functions are not concerning because sufficient non-DL3 functions initiate the hydraulic scram.

In the TSV and TCV closure events, pressure control is achieved (at least momentarily) via the TBVs to the main condenser. In the loss of condenser vacuum and LOPP AOO events with failure of the hydraulic scram, the main condenser is available for a limited amount of time. These events result in more severe pressurization and are selected as potentially bounding relative to the turbine trip and load rejection events. Closure of one MSRIV AOO with failure of the hydraulic scram event sequence is also selected for evaluation.

A complex sequence is also considered in this fault group based on the PSA evaluation. This event is selected as a potentially limiting event.

15.2.4.3 Reactivity and Power Distribution Anomalies Bounding Event Selection

Events that result in reactivity and power distribution anomalies are grouped as Reactivity Increase (RI) faults. These events include failures in reactivity control that challenge the fuel cladding or RCPB integrity. The BWRX-300 controls reactivity with control rod movement.

Normal control is accomplished using Fine Motion Control Rod Drives (FMCRDs), which are individually connected to the control rods. HCUs are used to quickly shutdown the reactor by inserting control rods, ensuring reactivity control during off normal operation. Increases in reactivity can result from control rod withdrawal.

The selected events result in the most significant challenge to the fission product barriers in the fault group. The selected events are summarized in Table 15.2-2.

SRP (NUREG-0800) 15.4.7 also describes evaluation of potential Fuel Loading Error (FLE) events. To meet the requirements of GDC 13, plant operating procedures should include a provision requiring that reactor instrumentation be used to search for potential fuel-loading errors after fueling operations. In the event the error is not detectable by the instrumentation, the acceptance criteria would be 10% of the 10 CFR Part 100 reference values. Discussion of fuel loading verification consistent with generic analysis will be provided at the FSAR stage.

Bounding AOO Event Selection

The fault list includes only one Reactivity Increase- Anticipated Operational Occurrence (RI-AOO). This event is Inadvertent Control Rod Withdrawal at Power - Single Rod (ICRW). However, this event is not evaluated because a protection function of Automatic Thermal Limits Monitor (ATLM) initiates a control rod block. The event conditions remain within normal operation conditions. Therefore, no analysis is needed. The design description of the ATLM is found in Subsection 7.3.3.2.

Bounding DBA Event Selection

The event effect on PCT and cladding integrity are the parameters used in selecting the bounding events for this group.

There are several DBA PIEs in the RI group to postulate in identifying the bounding event including the withdrawal of single or multiple control rods. The DBA PIEs are all mitigated by active DL2 functions ATLM and Multi-Channel Rod Block Monitor (MRBM) that result in minimal change to core power in the baseline sequence. The ATLM blocks rod motion before departure

from normal operation and MRBM blocks rod motion before significant fuel effects occur. These mitigation functions are reliable, resulting in a failure sequence in the DEC frequency range. These events with the failure of ATLM and MRBM are selected as bounding DECs without core damage events because of the low frequency of the sequence. The following event sequences are not selected as bounding DBA events:

1. CCF - All Control Rod Insertion at Power - (CCF-ACRI): For some core conditions, this event may result in a momentary increase in reactivity in the top part of the core because higher density water moves up the core faster than the rod motion. As the rods insert, high density coolant increases the reactivity above the control rods. This is a momentary effect, does not challenge fuel cladding criteria or any other fission product barrier, and is not selected as a bounding event.
2. Control Rod Drop Accident: (CRDA): This event postulates an event that allows separation of the control rod from the drive mechanism (design precludes a separation such as this due to a separation detection device). The control rod becomes stuck and remains in its position when the drive mechanism is attempted to be withdrawn. Before the stuck rod is detected and is inserted, the control rod becomes unstuck and falls. This event is prevented by the BWRX-300 bayonet-style FMCRD coupling, hollow piston latches, dual separation detection devices that limit separation of the rod from the drive. The event is discussed in Subsection 15.5.5.1.

Bounding DEC Without Core Damage Event Selection

The event effect on PCT and cladding integrity are the parameters used in selecting the bounding DEC events for this group. There are two control rod withdrawal error DEC events that are potentially limiting:

- Single rod withdrawal
- Inadvertent withdrawal of all rod groups

These DECs without core damage are compared to applicable acceptance criteria demonstrating that features to prevent core damage are adequate and no further complementary design features are required.

15.2.4.4 Increase in Reactor Coolant Inventory Bounding Event Selection

Events resulting in an increase in reactor coolant inventory are included as Inventory Increase (II) faults. These events may occur from a FW II. Inadvertent ICS initiation scenarios do not fit well into any fault group and are included in this increase in inventory event category. Inadvertent ICS initiation results in much less II than a FW flow increase. However, the dynamics of ICS flow into the chimney is different than an increase in FW flow. These scenarios are selected for analysis.

Other systems such as CRD, CUW, SDC or Boron Injection System (BIS) may result in inventory increases. However, these systems have low flow rates compared to FW and are not considered as bounding.

The events selected result in the most significant challenge to fission product barrier in this fault group. The selected events are summarized in Table 15.2-2.

Bounding AOO Event Selection

There is one BL-AOO event, Inadvertent Isolation Condenser Initiation – One Train, and it is selected for evaluation.

Bounding DBA Event Selection

The event effect on PCT and coolant inventory are the parameters used in selecting the bounding events in this group.

Two CN-DBA events are selected:

- Inadvertent initiation of all ICS trains (bounds one train) (Inadvertent Isolation Condenser Initiation – All Trains (CCF-IICI))
- Feedwater Flow Increase – All Pumps (CCF-FWFI). This event bounds the CN-DBA event for increase in flow of one FW pump

Bounding DEC Without Core Damage Event Selection

There are no DEC events identified in the II fault group because for each single failure AOO or DBA PIE, there are two DLs established that mitigate the event. No AOO events credit the hydraulic scram. For all CCF PIEs, a DL is established.

15.2.4.5 Decrease in Reactor Coolant Inventory Bounding Event Selection

Events that result in a decrease in reactor coolant inventory are included as Inventory Reduction (IR) events. These events may occur from failures that result in:

- Reduction or loss in FW flow
- Opening of TBVs
- Pipe breaks (LOCA events) (discussed in Subsection 15.2.4.6)
- Misalignment of systems connected to the RPV
- LOPP (this event is included in the PI group)

The IR bounding events selection only includes non-LOCA events. In this fault group, maintaining inventory above the Top of Active Fuel (TAF) ensures fuel cladding integrity. For non-LOCA events, there is no significant challenge to the RCPB or containment.

The selected events result in the most significant challenge to the fission product barriers in the fault group. The selected events are summarized in Table 15.2-2.

Bounding AOO Event Selection

The event effect on coolant inventory is the parameter used in selecting the bounding AOO events in this group. There are two BL-AOO events in this group:

- FW Pump Trip – one pump (FWPT)
- Inadvertent opening of one TBV

The FWPT event is selected as potentially limiting because the TBV opening represents less inventory loss than the FWPT and FW makes up inventory loss during the TBV opening.

Bounding DBA Event Selection

The event effects on coolant inventory are used in selecting the bounding DBA events for this group.

There are several PIE events in the IR group:

- Loss of all FW flow
- Opening of all TCVs and TBVs

- RPV Pressure Control Open

The selected bounding event is Loss of Feedwater Flow (LOFW) CN-DBA event. FW may also be lost by a FW isolation valve closure DL4a CCF BL-DBA but is not selected because this is less severe than LOFW. This is also less frequent and is categorized as a DEC when combined with a CCF of DL2 mitigation. The RPV Pressure Control Open (CCF-RPCO) is selected as potentially limiting for events where inventory is lost via the MSL.

LOCA events bound these events because they result in a more significant challenge to the RPV inventory (fuel cooling and long-term cooling) and challenge containment temperature and pressure. The non-LOCA DBA events are not bounding for this IR fault group.

Bounding DEC Without Core Damage Event Selection

The event effect on PCT and coolant inventory are the parameters used in selecting the bounding DEC events for this group.

The FW Isolation EX-DEC event is selected as a potentially limiting event. However, it is not expected to be a significantly different response than the CN-DBA event. LOCA events bound this event, and the non-LOCA events are not bounding for this IR fault group.

15.2.4.6 Bounding Event Selection for Loss-of-Coolant Accident Scenarios

The initiating events involving pipe breaks, scram, and trip initiation are identified in the fault list for evaluation in CN-DSA and EX-DSA. The evaluation scenarios are then selected to bound groups of pipe breaks. The consequences of postulated LOCAs are analyzed for fuel cladding and containment integrity and shown to meet the design basis acceptance criteria in Table 15.3-2. CN-DSA LOCA analyses demonstrate that the fuel cladding does not exceed the normal operating temperatures. Meeting these acceptance criteria assures the fuel cooling acceptance criteria in Table 15.3-2 are met.

The bounding scenarios for pipe breaks fall into two categories:

- Large breaks inside or outside containment
- Small breaks inside or outside containment

The breaks in each category may be in steam pipes or liquid pipes. LOCA PIEs are analyzed as DBAs even if their frequency of occurrence falls in the range of DEC events. CN-DSA sequences are mitigated by DL3 functions alone, assuming a CCF of DL2 functions. EX-DSA event sequences DECs are mitigated by DL2 and DL4a functions (see Subsection 15.2.4.6.4 for more details), assuming a CCF of DL3 functions.

Pipe breaks attached to the RPV may be as large as the complete rupture of the largest steam or feedwater pipes, or as small as leaks in smaller pipes attached to the RPV. Further, the largest breaks are instantaneous double-ended guillotine ruptures of the large pipes. This assumption is made to bound the thermal hydraulic response of the fuel, RPV and containment for all isolatable pipe breaks.

The bounding LOCA events provided in Table 15.2-2 and evaluated in Subsection 15.5.4.5 (LOCA inside containment) and Subsection 15.5.9.1 (LOCA outside containment) provide the assumptions and DL components used in mitigating the event with the corresponding signals, times, and other design parameters.

The BWRX-300 safety design philosophy for the mitigation of LOCAs includes features such as:

- Reliance on passive safety features such as ICS

- Utilization of inherent margins, such as larger water inventory, to reduce the severity of system challenges
- Reduction in the number and size of RPV nozzles
- Placement of the RPV nozzles high on the RPV
- Inclusion of redundant Reactor Isolation Valves (RIVs) attached directly to the RPV when needed to quickly isolate a break in pipe

As described in NEDC-33910P-A, “BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection,” (Reference 15.2-2), a break between the RPV and the Reactor Isolation Valves (RIV) is not evaluated as a DBA, so DBA breaks inside containment are postulated to occur at any arbitrary location between the RIV and the containment boundary. The BWRX-300 design has adequate justification for excluding a break at these locations from the design bases based on (1) the design's wholistic approach to LOCA mitigation, as summarized above and described in more detail in Reference 15.2-2, and (2) the demonstration that the likelihood of a break is highly unlikely, as described in Subsection 5.4.3. As such, breaks at these locations are excluded from 10 CFR 50.46(a)(1) and GDC 35 consideration. Subsection 15.2.4.6.4 describes the EX-DSA planned to evaluate DEC(s) involving unisolated breaks in the FSAR.

Large Breaks

Pipes that are larger than 19 mm inner diameter have two isolation valves attached directly to the RPV. The largest postulated pipe breaks are in the Main Steam (MS), FW, and ICS lines. These pipes have RIVs, which close in less than 5 seconds once they start closing. Another 5-second delay is assumed before a RIV starts closing to account for delays in break detection and signal development.

MS pipes are equipped with a flow limiter to prevent very large break flow prior to MSRIV closure. The flow limiters are placed inside main steam RPV nozzles. Breaks on the MS pipes are postulated to occur downstream of the RIV.

The analysis assumes the high containment pressure setpoint for break isolation is reached within 1 second for large breaks (steam and liquid). Peak containment pressure occurs at approximately the time RIVs are fully closed and the break is isolated. The largest steam pipe breaks, and the largest liquid pipe breaks are the most limiting for containment response because they have the highest mass and energy release until the break is isolated. For medium size breaks that are isolated on high containment pressure, there may be a delay in reaching the containment high pressure setpoint for isolation. However, containment pressure is increasing at a slower rate than it would for a larger break while the RIVs are closing. Because the RIV closure time is the same for all breaks, the containment peak pressure is smaller for a medium size break than it is for a large break.

Because the largest breaks are more limiting for the mass and energy release, fuel integrity and containment integrity, a break spectrum analysis is not required for isolatable breaks. This is also the case for breaks outside containment as they are isolated by the leak detection system in a similar manner.

The leakage detection system is designed to detect breaks in large pipes connected to the RPV. If the break is not detected, a conservative assumption is that the break remains unisolated for 72 hours for CN-DBA sequences because no operator action is credited.

For large break LOCAs, RIVs close rapidly and prevent significant loss of RPV inventory. The core remains covered throughout. The remainder of the event after the RIVs are closed is an

isolation event during which the ICS has ample capacity to remove the decay heat and depressurize the RPV and maintain fuel cooling for at least 72 hours. Fuel integrity is not a concern for large breaks. In the long-term, the Passive Containment Cooling System (PCCS) reduces containment pressure.

Small Breaks

Small pipes connected to the RPV with an inner diameter ≤ 19 mm do not have automatic RIVs. Breaks involving small pipes with an inner diameter ≤ 19 mm are considered unisolated breaks. Small unisolated breaks may also occur in larger pipes. The leakage detection system detects breaks in large pipes connected to the RPV but may not be capable of detecting breaks that are smaller than the area of a circle with a 19 mm diameter. If the break is not detected, it is assumed to remain unisolated for 72 hours for CN-DBA sequences because no operator action is credited for this duration.

Small breaks conservatively credit only two of the three isolation condensers even though a break of less than 19 mm equivalent diameter on an isolation condenser does not cause degradation in the isolation condenser heat removal rate. There is sufficient steam in the RPV to feed the condensation in the isolation condenser. Not having sufficient steam in the RPV to feed the isolation condenser can only occur if the RPV is depressurized so far that almost all of the steam is escaping the break. This would be the case if the RPV pressure is even lower than that calculated for an instrument pipe break. However, in this case the break flow is less than the break flow calculated for an instrument pipe break, making the isolation condenser small break less limiting than the instrument pipe break. It can be concluded that whether the break flow rates are the same for a small steam pipe break regardless of the break location, or the RPV pressure is too low to feed the isolation condenser, the breaks on isolation condenser steam pipes are no more limiting than a break on an instrument steam pipe break.

The largest of the unisolated breaks is most limiting with respect to RPV inventory. Peak containment pressure is also higher as the break size is increased for an unisolated break.

Large and Small Pipe Break Summary

The largest break sizes are the most limiting for isolatable (i.e., in the large break category) and un-isolatable (i.e., in the small break category) breaks.

In selecting the scenarios to evaluate for the pipe breaks, each of the sequences in the fault list is assessed with respect to the largest mass and energy release to containment and RPV inventory. Additional conservative assumptions that are described in Subsection 15.5.4.5 and Subsection 15.5.9.1 were made in constructing the bounding scenarios to reduce the number of analysis cases. Therefore, the bounding scenarios analyzed for pipe breaks in large and small break categories are scenarios bounding all scenarios in that category. However, the liquid and steam pipe breaks are analyzed separately.

The following common features are used in selecting the bounding sequences for pipe breaks:

1. A pipe break does not cause loss of FW or loss of normal containment cooling unless a direct or indirect effect of the pipe break causes the pump(s) to trip. FW and normal containment cooling are lost concurrent with the break for the sequences involving a pipe break concurrent with LOPP. This observation is used to determine whether the breaks are more limiting with or without LOPP.
2. All scrams credited in the LOCA analyses are direct scrams. A scram signal is initiated when the setpoint of the first scram function is reached for the scram functions that are available in the credited DL. Scram functions and setpoints in each DL and the trip parameters are provided.

3. In the CN-DBA sequences, a CCF of DL2 functions concurrent with the pipe break are assumed and credit only DL3 functions.
4. In the EX-DEC sequences, CCF of DL3 functions are assumed in the analysis and credit only DL2 and DL4a mitigation functions (see Subsection 15.2.4.6.4 for more details).

The guidelines for considering CCFs in the design and safety analyses are discussed in the NEDC-33934P.

The system responses, including the direct and indirect effects of large and small pipe breaks, are described in Subsection 15.5.4.5 and Subsection 15.5.9.1.

15.2.4.6.1 Large Steam Pipe Breaks

Large steam pipe breaks are postulated to occur in the following systems:

- MS pipes
- ICS steam supply pipes

For breaks inside containment, scram is initiated on high containment pressure (DL3-07). DL functions are shown in Table 15.5-5, Table 15.5-44, and Table 15.5-45. For breaks outside containment, scram is initiated on detection of a steam pipe break (DL3-09).

MSRIV closes on high containment pressure for breaks inside containment (DL3-22), and on MS pipe break detection for breaks outside containment (DL3-20). ICS RIVs close for an ICS train when an ICS break inside or outside containment is detected in the respective ICS train (DL3-27, 28, or 29).

Because the isolation valves close rapidly, the effect on isolation condenser availability is not significant for large breaks mass and energy releases. After break isolation, one ICS train is sufficient to remove decay heat and depressurize the RPV. As a bounding assumption, only one ICS train is credited for MS and ICS pipe breaks. This assumption is made so that an ICS steam supply break, whether the break is in the main line to the Isolation Condenser or in the ultimate pressure regulation branch line, is bounded by an analyzed MS pipe break. One ICS train initiates on high containment pressure (DL3-15) for breaks inside containment. One ICS train initiates on pipe break indication detection (DL3-16) in MS or ICS pipe breaks outside containment.

For a MS pipe break, the total break flow is the sum of break flows from both ends of the break. To bound all break locations, the break location is assumed as close to the RPV as possible, right outside the second or outboard RIV. Because two MS lines are connected through a header, the intact steam pipe also supplies the break location from the turbine side of the break. Break flow from the turbine side of the break is contributed by the flow from the RPV into the intact loop and the initial inventory in the piping. To maximize flow from the turbine side of the break, the isolation valves outside the containment are assumed to remain open, and TSVs and TCVs close rapidly. Because closure of CIVs outside containment are not credited, the calculated mass and energy release is applicable to breaks inside and outside containment.

If the break occurs when the plant is at very low power or hot shutdown, break flow from an MS pipe break may be higher due to carryover. At low power, there is more saturated water in the RPV to flash. This may increase the two-phase downcomer level much higher than that in the rated initial conditions case, contributing to the break flow due to increased liquid content although the break flow enthalpy is lower. In calculating the radiological consequences for breaks outside containment, the break mass flow rate for hot shutdown initial conditions may be more limiting. Both the rated initial conditions and the hot shutdown initial conditions are included in the bounding scenarios for MS pipe breaks.

The above scenario assumes LOPP concurrent with the break is bounding for the scenario when preferred power is available. If the preferred power is available, FW will continue to be injected into the RPV, which may increase mass release to the containment due to carryover. However, if preferred power is available, TSVs and TCVs also remain open, discharging much of the steam in the intact loop to the turbine rather than to the break from the turbine side of the break location. As a result, the break scenario with LOPP is the more bounding scenario.

Bounding CN Scenario for Large Steam Pipe Breaks

The bounding scenario analyzed MS pipe breaks inside and outside containment concurrent with LOPP.

The case is analyzed for rated initial power and hot shutdown conditions. The limiting break scenario is analyzed in Subsection 15.5.4.5.1 and Subsection 15.5.9.1.1.

15.2.4.6.2 Large Liquid Pipe Breaks

Large liquid breaks may occur in:

- FW pipe
- ICS condensate return pipe
- CUW pipe

The largest pipe from this list is the FW pipe and is analyzed in Subsection 15.5.4.5.2.

A large break in the ICS condensate return pipe is listed as a liquid break because the condensate return pipe is filled with water initially. However, this water is highly subcooled. The ICS condensate return valves are normally closed and the steam supply pipe is in communication with the RPV during normal operation. If a pipe break occurs at the condensate return pipe, the highly subcooled water in the condensate pipe is purged. The energy release to containment for a break inside containment or to the IC pool for a break outside containment from the purged highly subcooled water is insignificant. After the liquid inventory in the pipe is depleted, the break flow becomes steam flow supplied from the RPV through the steam supply pipe until the RIVs close for the broken ICS unit.

The ICS steam supply pipe is fitted with a 70 mm diameter orifice at the steam distribution pipes. Therefore, flow from a break in the condensate pipe is only steam flow through a 70 mm orifice after the subcooled water is purged. This is less limiting than a break in the ICS steam supply pipe included in the large steam pipe break cases. This pipe break requires no further analysis because it is included in the large steam pipe break category for CN-DBA sequences.

CUW pipe breaks may occur inside or outside the containment. CUW pipe is a smaller-bore pipe, and CUW breaks are isolated the same as the FW pipe breaks. Therefore, the CUW pipe breaks inside containment are bounded by the FW pipe breaks. CUW pipe breaks are routed through the same compartments that house the FW pipes in the Reactor Building (RB). Therefore, CUW pipe breaks outside containment are also bounded by the FW pipe breaks for RB pressure and temperatures.

The limiting liquid pipe breaks inside and outside the containment are the FW pipe breaks. The break isolation, scram, and ICS initiating are discussed below.

Bounding CN Scenario for Large Liquid Pipe Breaks

For a pipe break inside containment, scram is initiated on high containment pressure (DL3-07). For breaks outside containment, scram is initiated on pipe break indication in FW or ICS pipes (DL3-09). Scram does not occur for the CUW pipe breaks outside containment.

In accordance with the fault list, preferred power is available for CUW breaks outside containment and FW injection continues.

LOPP concurrent with a break is assumed for the FW pipe breaks inside the containment. The TSVs and TCVs are conservatively assumed to close rapidly with LOPP. In the LOPP cases, there is flow from the pump side of the break before the CIVs close outside containment at 10 seconds due to pump coast down, but more importantly, due to flashing of the water in the FW piping. If the FW pumps continue running despite the break when preferred power is available, there is some increase in flow from the pump side of the break. However, normal containment cooling also continues to run when preferred power is available, compensating for the effect on containment pressure from an increase in the flow from the pump side of the break.

For breaks outside containment in the RB, pressure and temperature calculation cases with and without FW running are included.

For FW pipe breaks outside containment, only the LOPP case is evaluated for radiological consequences. Only the water leaving the RPV is important in the radiological consequence analysis since feedwater is degassed and filtered upstream of the FW pumps.

For breaks inside containment, RIVs close on high containment pressure (DL3-22). CIVs outside containment also close on high containment pressure (DL3-22).

For FW pipe breaks outside containment, FW RIVs close on break detection. MSRIVs also close on break detection (DL3-21).

For CUW pipe breaks outside containment, only CUW RIVs close on break detection (DL3-26).

For breaks inside containment, ICS is initiated on high containment pressure (DL3-15).

For FW and ICS pipe breaks outside containment, ICS is initiated on-line break detection (DL3-16). For CUW breaks outside containment, ICS initiation is not needed because preferred power is available.

Bounding Scenario Summary for Large Liquid Pipe Breaks

The bounding liquid pipe break for containment response is analyzed for a FW pipe break inside containment:

- Double-ended guillotine FW pipe break inside containment concurrent with LOPP
- TSVs and TCVs are conservatively assumed to close rapidly retaining more energy
- FWPTs and coast down
- Scram initiation within 1 second after the break
- ICs initiate on high containment pressure
- MS and FW RIVs start closing in 5 seconds and are fully closed in 10 seconds
- FW CIVs outside containment start closing in 5 seconds and are fully closed in 10 seconds
- FW conservatively assumed to trip at time zero and coasts down with a 3 second time constant

The bounding liquid pipe break scenario outside containment is similar to breaks inside containment with the following differences:

- FW CIVs are conservatively assumed not to close
- FWPT may or may not occur. Area pressure and temperature calculations consider both cases. The radiological analyses conservatively assume a FWPT.

The above scenario does not rely on any scram or isolation function that is a result of the LOPP. Therefore, the same scenario bounds both preferred power available and LOPP cases.

15.2.4.6.3 Small Breaks

Small, unisolated steam or liquid pipe breaks may occur in instrument lines. Small breaks in the large pipes may also remain unisolated if they are below the leak detection system threshold, (i.e., less than 19 mm equivalent diameter). The lowest location for a liquid pipe break is 4 m above the TAF.

Small breaks are analyzed using conservative assumptions demonstrating that fuel and containment integrity are maintained for at least 72 hours using only passive systems after which the event is terminated.

The fault list includes unisolatable small breaks inside and outside the containment, with and without concurrent LOPP.

The bounding CN sequences in the fault list for a small break concurrent with LOPP are evaluated with respect to the fuel cladding and containment. For a break inside containment concurrent with LOPP, normal containment cooling system is also assumed to be lost. The energy discharged from the break to containment is removed by the PCCS (discussed in Subsection 6.2.2) and through the containment dome. PCCS does not require actuation because it is always in service.

When the preferred power is available, FW continues to run. Because the FW pump can make up for the break flow, fuel integrity is not a concern. If preferred power is available, normal containment cooling also continues to run. If the normal containment cooling and PCCS cannot keep up with the break flow and containment pressure increases, the reactor scrams, isolation condensers initiate, and RPV depressurizes, reducing the break flow. Containment cooling continues to operate, maintaining containment pressure at a lower value than PCCS alone maintains in the LOPP case.

The bounding small liquid and steam pipe break scenarios are the same, the only difference is the discharge from the small liquid pipe break is initially from the liquid water space. It becomes steam flow after water level falls below the RPV nozzle elevation of the broken pipe.

Following an unisolated break in an instrument pipe concurrent with LOPP, turbine pressure decreases rapidly resulting in a decrease in the steam pipe pressure and an increase in steam flow. Although a consequential closure of TSVs and TCVs may occur on LOPP, this is not credited in the analysis. TSVs and TCVs are assumed to remain in their initial position. Scram is initiated when the steam pipe pressure decreases to low steam pipe pressure setpoint (DL3-02) with a 1.7 second delay. Power is assumed to remain at the initial value for an additional 2 seconds to account for the time elapsed until prompt fission power is diminished after the control rods start inserting. MSRIVs also start closing on low steam pipe pressure (DL3-17) over 5 seconds after a delay of 5 seconds.

FW pumps trip concurrent with LOPP and coast down with a 3 second time constant.

All available ICS train condensate return valves start opening with a delay of 1 second when the water level falls to L2 level setpoint (DL3-14). Only two isolation condenser trains are assumed available. ICS condensate return valves are fully open 10 seconds after they start opening. There are no further actuations assumed for the remainder of the event.

Mass and energy releases from small pipe breaks do not credit containment back pressure for breaks inside containment. In addition, reactor scram and isolations initiated by high containment pressure are also not credited. Therefore, the above scenario applies to breaks outside containment as well as breaks inside containment. Because a break outside containment occurs from a longer pipe, break mass and energy releases calculated for a break inside containment bounds a small break outside containment.

Bounding Small Pipe Break

The bounding scenario is summarized as follows:

- Small steam or liquid pipe break concurrent with LOPP
- FWPTs and FW flow coasts down with a time constant of 3 seconds
- Pressure controller failure, TSVs and TCVs position remain open at their initial position
- Reactor scrams when steam pipe pressure decreases to low steam pipe pressure setpoint with a 1.7 second delay
- MSRIV closure when steam pipe pressure decreases to low steam pipe pressure setpoint with a 5-second delay
- ICS initiation when level is less than L2 setpoint

15.2.4.6.4 Bounding Scenarios for DEC Pipe Breaks

DEC pipe breaks assume DL3 CCF in addition to the pipe break. Only DL2 and DL4a functions are credited in DECs. There is either a DL2 or a DL4a backup function for credited DL3 functions in Table 15.5-44, except for those mitigating isolation condenser pipe breaks. Because the heat removal by the ICS is a higher-class safety function than isolation by a DL4a function, no DL4a associated function exists. An unisolated isolation condenser pipe break is like an unisolated MS pipe break and will be evaluated in the EX-DSA in the FSAR.

DL2 and DL4a functions performing the same function for the credited DL3 functions are listed in Table 15.5-45. The setpoints and timing of these functions are the same as or close to the DL3 functions. Therefore, except for the isolation condenser pipe breaks, the design basis LOCA analyses bound the DEC pipe breaks for the isolatable large pipe breaks and un-isolatable small breaks.

For the FSAR, EX-DSA will be performed for the most limiting, in terms of fuel cooling, unisolated large pipe break using best estimate or realistic analysis conditions. The event will be analyzed to the point of achieving stable, safe conditions. The EX-DSA will credit any DL function not affected by the breaks (i.e., using unaffected SC1, SC2, or SC3 SSCs); crediting human action is permitted with a justification of impracticability for implementing an automatic function. The EX-DSA will not assume a single failure or loss of power. The EX-DSA methodology used will be consistent with the TRACG thermal hydraulics model for mass and energy releases described in

Section 5 of NEDC-33922P-A, without the biases for the initial conditions and the modeling parameters. The acceptance criteria that will be applied to this EX-DSA are:

- Results demonstrate that the criteria in 10 CFR 50.46(b) are satisfied
- Offsite dose consequences of the event are demonstrated to meet the following criteria (either by realistic evaluation of the event or showing that the consequences are bounded by another event):
  - An individual located at any point on the boundary of the exclusion area for any 2 hour period following the onset of the postulated fission product release would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE).
  - An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 25 rem TEDE.
- Results demonstrate that the plant safety goals are met.

15.2.5 References

15.2-1 NEDC-33934P, “BWRX-300 Safety Strategy,” Revision 2, GE Vernova Hitachi Nuclear Energy Americas, LLC, March 2026.

15.2-2 NEDC-33910P-A, “BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection,” Revision 2, GE Hitachi Nuclear Energy Americas, LLC, June 2021.

Table 15.2-1: Fault Groups and Explanation of Core Reactivity Response Basis

Name (ID)	Description	Discussion of Reactivity Effects
Temperature Decrease (TD)	Decrease in Core Coolant Temperature	Void reactivity is key. The decrease in temperature results in an increase in the core inlet subcooling. More core thermal power goes into heating up the water and less into void production. The core void fraction decreases and causes the power to increase. This is a relatively slow increase in core power due to the thermal inertia of the coolant.
Pressure Increase (PI)	Increase in Reactor Pressure	Void reactivity is key. Pressurization results in decrease of core voids and increase in core power. For rapid pressurization, control rod scram is needed to mitigate. Core exposure effects on void reactivity and control rod position are important aspects that are modeled and included in the analysis.
Reactivity Increase (RI)	Reactivity and Power Distribution Anomalies	Control rod reactivity is key. Errors or failure in control rod movement are expected event initiators as these events add reactivity and change the local and core wide power.
Inventory Increase (II)	Increase in Reactor Coolant Inventory	Void reactivity is key. An increase in coolant inventory results in a reactor water level increase. The increase in reactor water level normally has the following effects: • Core flow increase (small reactivity effect at rated power conditions) • Core inlet subcooling increase (because the additional inventory is expected to be from lower temperature coolant, larger reactivity effect than the core flow increase) These effects tend to increase void reactivity.
Inventory Reduction (IR)	Decrease in Reactor Coolant Inventory	Void reactivity is key. The decrease in reactor water level has the following potential results: • Core flow decrease (small reactivity effect at rated power conditions) • Core pressure decrease (if break in coolant pressure boundary cannot be compensated for by pressure control) • Core inlet subcooling decrease (comes with pressure decrease) These effects tend to insert negative reactivity, resulting in core power decrease, due to void reactivity. It is typical in LOCA analysis to ignore these effects as the protection systems typically act quickly to insert control rods and shutdown the core before the negative void reactivity feedback has significant effect.
Non-Reactor Faults	Event specific. These events are not core related.	Fuel Handling Accident

Table 15.2-2: Bounding Events Transient (Non-LOCA) and LOCA

DSA Layer / Event Category	Event Sequence ID	Corresponding DSA Subsections 15.5.3 Through Subsection 15.5.9 Event Summary Results
Decrease in Core Coolant Temperature Bounding Event Summary		
BL-AOO	**Loss of Feedwater Heating (LFWH)** TD-LFWH_BL-AOO	Subsection 15.5.3.1.1
CN-DBA	**Common Cause Failure – Loss of All Feedwater Heating** CCF- LFWH, Passive CCF DL2 Digital Technology Platform; TD-CCF-LFWH_CCF-DL2_CN-DBA	Subsection 15.5.4.1.1
EX-DEC	None	N/A
Increase in Reactor Pressure Bounding Event Summary		
BL-AOO	**Generator Load Rejection or Turbine Trip** (LR-TT); PI-LR-TT_BL-AOO **Closure of One MSRIV** 1MSRIVC; PI-1MSRIVC_BL-AOO **Loss of Condenser Vacuum (LOCV)** PI-LOCV_BL-AOO **Loss-of-Preferred Power (LOPP)** PI-LOPP_BL-AOO	Subsection 15.5.3.2.1 Subsection 15.5.3.2.2 Subsection 15.5.3.2.3 Subsection 15.5.3.2.4
CN-DBA	**Generator Load Rejection or Turbine Trip** LR-TT, Passive CCF DL2 Technology Platform (CCF-DL2); PI-LR-TT_CCF-DL2_CN-DBA **Loss-of-Preferred Power** LOPP, Passive CCF DL2 Technology Platform (CCF-DL2); PI-LOPP_CCF-DL2_CN-DBA **RPV Pressure Control Downscale** CCF - RPV Pressure Control Downscale (CCF-RPCD), Passive CCF DL2 Technology Platform (CCF-DL2); PI-CCF-RPCD_CCF-DL2_CN-DBA **Closure of All MSRIVs and FW Isolation Valves** CCF - Closure of All MSRIVs and FW Isolation Valves (CCF-DL4a-MSRIVC-FWIV); PI-CCF-DL4a-MSRIVC-FWIV_CN-DBA	Subsection 15.5.4.2.1 Subsection 15.5.4.2.2 Subsection 15.5.4.2.3 Subsection 15.5.4.2.4

DSA Layer / Event Category	Event Sequence ID	Corresponding DSA Subsections 15.5.3 Through Subsection 15.5.9 Event Summary Results
EX-DEC	**Closure of One MSRIV** PI-1MSRIVC_CCF-Hydraulic-Scram_EX-DEC **Complex Sequence of Generator Load Rejection or Turbine Trip** Complex Sequence of LR-TT + CCF-Mechanical-CRD; CSS-LR-TT_CCF-Mechanical-CRD_EX-DEC **Loss of Condenser Vacuum** LOCV, CCF-Hydraulic-Scram; PI-LOCV_CCF-Hydraulic-Scram_EX-DEC **Loss-of-Preferred Power** LOPP, CCF-Hydraulic-Scram; PI-LOPP_CCF-Hydraulic-Scram_EX-DEC	Subsection 15.5.5.2.1 Subsection 15.5.5.2.2 Subsection 15.5.5.2.3 Subsection 15.5.5.2.4
Reactivity and Power Distribution Anomalies Bounding Event Summary		
BL-AOO	None	N/A
CN-DBA	None	N/A
EX-DEC	**CCF- All Control Rod Withdrawal at Power - All Rods (CCF-ACRW)** Passive Digital CCF DL2 Technology Platform (CCF-DL2); RI-CCF-ACRW_CCF-DL2_EX-DEC **Inadvertent Control Rod Withdrawal at Power - Single rod (ICRW)** RI-ICRW_DL2-CCF_EX-DEC	Subsection 15.5.5.3.1 Subsection 15.5.5.3.2
Increase in Reactor Coolant Inventory Bounding Event Summary		
BL-AOO	**Inadvertent Isolation Condenser Initiation – One Train (IICI-1)** II-IICI-1_BL-AOO	Subsection 15.5.3.4.1
CN-DBA	**Feedwater Flow Increase – All Pumps** CCF-FWFI with Passive CCF DL2 Digital Technology Platform (CCF-DL2); II-CCF_FWFI_CCF-DL2_CN-DBA **Inadvertent Isolation Condenser Initiation - All Trains** (CCF-DL4a-IICI), Passive CCF DL2 Technology Platform (CCF-DL2); II-CCF-IICI_CCF-DL2_CN-DBA	Subsection 15.5.4.3.1 Subsection 15.5.4.3.2
EX-DEC	None	N/A

DSA Layer / Event Category	Event Sequence ID	Corresponding DSA Subsections 15.5.3 Through Subsection 15.5.9 Event Summary Results
Decrease in Reactor Coolant Inventory Bounding Event Summary (non-LOCA)		
BL-AOO	**Feedwater Pump Trip – One Pump** FWPT; IR-FWPT_BL-AOO	Subsection 15.5.3.3.1
CN-DBA	**CCF Loss of Feedwater Flow** Passive CCF DL2 Technology Platform (CCF-DL2); IR-CCF-LOFW_CCF-DL2_CN-DBA **Reactor Pressure Vessel Pressure Control Open** CCF-RPCO, Passive CCF DL2 Technology Platform (CCF-DL2); IR-CCF-RPCO_CCF-DL2_CN-DBA	Subsection 15.5.4.4.1 Subsection 15.5.4.4.2
EX-DEC	**Feedwater Isolation** FW Isolation (CCF-FWI-DL3); IR-CCF-FWI-DL3_EX-DEC	Subsection 15.5.5.4.1
Decrease in Reactor Coolant Inventory Bounding Event Summary (LOCA)		
CN-DBA	**Main Steam Pipe Break Inside Containment, Conservative Case**	Subsection 15.5.4.5.1
CN-DBA	**Feedwater Pipe Break Inside Containment, Conservative Case**	Subsection 15.5.4.5.2
CN-DBA	**Large Isolation Condenser Pipe Breaks Inside Containment**	Subsection 15.5.4.5.3
CN-DBA	**Small Steam and Liquid Pipe Breaks Inside Containment**	Subsection 15.5.4.5.4
CN-DBA	**Main Steam Line Break Outside Containment**	Subsection 15.5.9.1.1
CN-DBA	**Large Feedwater Line Break Outside Containment**	Subsection 15.5.9.1.2
CN-DBA	**Shutdown Cooling System Line Break Outside Containment**	Subsection 15.5.9.1.3
CN-DBA	**Large Isolation Condenser Line Break Outside Containment**	Subsection 15.5.9.1.4
CN-DBA	**Small Breaks Outside Containment**	Subsection 15.5.9.1.5

15.3 Safety Objectives and Acceptance Criteria

The deterministic safety analysis AOO Event Sequences acceptance criteria are based on or derived from ensuring that the peak pressure limit associated with RCPB for 10 CFR 50, Appendix A, GDC 15 is met, and from ensuring Specified Acceptable Fuel Design Limits (SAFDLs) are met for the following 10 CFR 50, Appendix A GDCs or Principal Design Criteria (PDC):

- GDC 10, “Reactor Design”
- GDC 12, “Suppression of Reactor Power Oscillations”
- PDC 17, “Electric Power Systems”
- GDC 20, “Protection System Functions”
- GDC 25, “Protection System Requirements for Reactivity Control Malfunctions”
- PDC 26, “Reactivity Control System Redundancy and Capability”
- GDC 33, “Reactor Coolant Makeup”
- GDC 34, “Residual Heat Removal”

The deterministic safety analyses DBA Event Sequences acceptance criteria are based on or derived from ensuring that the 10 CFR 50.46(b) acceptance criteria for emergency core cooling systems are met.

15.3.1 Deterministic Safety Analysis Acceptance Criteria

The BWRX-300 design complies with the SAFDLs and 10 CFR 50.46(b) acceptance criteria.

Qualitative acceptance criteria are defined and met for each AOO and DBA to confirm the effectiveness of plant systems in maintaining the integrity of physical barriers against releases of radioactive material.

Derived qualitative and quantitative acceptance criteria are used to analyze AOOs or DBAs. Qualitative acceptance criteria are supported by experimental data, prescribed by regulatory requirements, or prescribed by applicable codes and standards. The results of the quantitative safety analysis confirm the derived acceptance criteria.

The acceptance criteria for DECs are plant safety goals for core damage frequency and large release frequency that are specified in Table 15.3-3. The analysis results for DECs without core damage described in Subsection 15.5.5 may be conservatively compared to DBA acceptance criteria to demonstrate that the plant safety goals would not be exceeded.

The Total Effective Dose Equivalent (TEDE) for an individual located at any point on the boundary of the exclusion zone does not exceed the dose acceptance criteria of 10 CFR 50.34(a)(1)(ii)(D), and PDC 19 for the control room personnel. The dose results provided in Section 15.7 from the limiting events identified in Section 15.5 demonstrate that the radiological consequences of the analyzed events do not exceed the dose acceptance criteria. There are no radiological consequences resulting from AOOs.

15.3.1.1 Acceptance Criteria for Analysis of Anticipated Operational Occurrences

The derived acceptance criteria for the DSA of AOOs are shown in Table 15.3-1.

15.3.1.2 Acceptance Criteria for Analysis of Design Basis Accidents

The derived acceptance criteria for the DSA of DBAs are shown in Table 15.3-2. These acceptance criteria are applied to the CN-DSA for all DBAs, including event sequences involving

an AOO PIE and an assumed CCF of DL2 functions, crediting only DL3 functions performed by SC1 SSCs. For LOCA DBAs, meeting the acceptance criteria for fuel integrity is demonstrated by showing that level does not fall below the TAF, or fuel cladding temperature does not exceed the fuel cladding temperature during normal operating conditions.

The limits for radiological consequences applied to DBAs are consistent with Table 1 of NUREG-0800 SRP Section 15.0.3 and are shown in Section 15.7. In addition, for DBA event sequences involving an AOO PIE and an assumed CCF of DL2, calculated off-site dose at any point on the exclusion area boundary and at any point on the outer boundary of the low population zone is not in excess of 2.5 rem total effective dose equivalent (TEDE).

15.3.2 Acceptance Criteria for Probabilistic Safety Assessment

The BWRX-300 standard plant design safety goals for the PSA of core damage frequency and large release frequencies are shown in Table 15.3-3 and are consistent with NRC “Safety Goals for the Operations of Nuclear Power Plants; Policy Statement,” Federal Register, Volume 51, No. 149, pp. 28044-28049, August 4, 1986. Corrected and reprinted at Federal Register, Volume 51, No. 162, pp. 30028-30033, August 21, 1986.

15.3.3 References

15.3-1 NEDC-33840P-A, “The PRIME Model for Transient Analysis of Fuel Rod Thermal-Mechanical Performance,” Revision 1, GE Hitachi Nuclear Energy Americas, LLC, August 2017.

15.3-2 NEDE-33005P-A, “TRACG Application for Emergency Core Cooling Systems / Loss-of-Coolant-Accident Analyses for BWR/2-6,” Revision 2, GE Hitachi Nuclear Energy Americas, LLC, May 2018.

15.3-3 NEDE-33885P-A, “GNF CRDA Application Methodology,” Revision 1, Global Nuclear Fuel -Americas, LLC, March 2020.

Table 15.3-1: Anticipated Operational Occurrence Deterministic Safety Analysis Acceptance Criteria

Fission Product Barrier or Fundamental Safety Function	Qualitative Acceptance Criteria	Quantitative Acceptance Criteria
General	An AOO will not escalate to a more serious accident condition unless other faults occur independently.	Not applicable
	There is no consequential failure or loss of function of any fission product barrier.	Not applicable
Fuel Rod	Fuel temperature results will not cause a loss of fuel rod mechanical integrity.	The calculated maximum fuel center temperature T_{center} remains below the fuel melting point $T_{melt.}$
	Fuel pellet-cladding results do not lead to loss of fuel rod mechanical integrity.	The cladding strain acceptance criteria defined in Section 5.0 of Reference 15.3-1.
	Fuel cladding temperature results will not cause a fuel rod failure.	The calculated core Minimum Critical Power Ratio (MCPR) ensures that 99.9% of the fuel rods in the core are not susceptible to boiling transition during AOO events. With the reactor steam dome pressure less than 685 psig (4.72 MPaG), the calculated reactor thermal power is less than 25% of rated thermal power.
Reactor Coolant Pressure Boundary	Design conditions of the reactor coolant pressure boundary are not exceeded during the most severe pressurization transient.	The calculated peak pressure associated with the reactor coolant pressure boundary does not exceed 110% of the design pressure or 1650 psig (11.38 MPaG).
	The reactor coolant pressure boundary maintains sufficient reactor coolant inventory for core cooling.	The calculated reactor water level is maintained at or above top of active fuel.

Fission Product Barrier or Fundamental Safety Function	Qualitative Acceptance Criteria	Quantitative Acceptance Criteria
Primary Containment	Containment integrity is maintained. If an AOO results in an energy release to the containment, or loss of containment heat removal, then containment stresses (i.e., pressure and temperature) are limited such that there is no loss of a containment barrier safety function, and thus, the containment remains within its design limit values.	No AOOs result in a significant energy release to containment, or prolonged loss of normal containment cooling. The normal operation limits and conditions are applied to containment, and no AOO containment quantitative criteria is needed.
Long-Term Heat Removal	SSCs important for preserving the integrity of the reactor core and the containment can remove residual heat for an extended period both during and after all applicable PIEs considered in all Operational States, including AOOs.	Following AOO events that do not result in shutdown, a controlled condition is achieved. Following AOO events that require shutdown, the core remains shutdown independent of operator action or offsite support for at least 72 hours. For AOO events that rely on ICS mitigation for long-term cooling, the plant can achieve and maintain safe-shutdown conditions with the average reactor coolant temperature below 420°F (215.6°C) within 36 hours and provide cooling for at least 72 hours without operator action or offsite support.

Table 15.3-2: Design Basis Accident Deterministic Safety Analysis Acceptance Criteria

Fission Product Barrier or Fundamental Safety Function	Qualitative Acceptance Criteria	Quantitative Acceptance Criteria
General	Except for fuel cladding, there is no loss of function of any fission product barrier.	Not applicable
Fuel Rod	The number of fuel rod failures is conservatively estimated for DBAs.	For the FSAR, fuel rod failure will be assumed if one of these thresholds is met as calculated by TRACG. For the PSAR, fuel rod failure was determined by criterion (1). (1) Cladding ductile rupture using the TRACG model in Reference 15.3-2, (2) Any local maximum fuel temperature (centerline or elsewhere) ≥ melting temperature, (3) Radially averaged increase in fuel enthalpy ≥ the limit shown in Figure 3-7 of Reference 15.3-3, or (4) Absolute value of peak radially averaged enthalpy ≥ the limit shown in Figure 3-8 of Reference 15.3-3. The calculated number of failed rods does not result in exceeding the applicable radiological dose acceptance criteria.
	Mechanical fracturing of a fuel assembly under DBA loading conditions does not result in losing the ability to cool the fuel assembly.	The mechanical integrity of the fuel is established from the mechanical and thermal fuel analysis described in Subsection 4.2.2.
Fuel Cooling	The calculated fuel cladding temperature is maintained at an acceptably low value and decay heat is removed for the extended period required by the long-lived radioactivity remaining in the core.	For Non-LOCA, the calculated PCT remains less than 2200°F (1204°C) for pre-transient hydrogen content ≤ 400 wppm. If 400 wppm is exceeded, the calculated PCT remains less than 2050°F (1121°C) or additional justification is provided. The calculated total oxidation of the cladding nowhere exceeds the embrittlement oxidation level specified in Figure 24 of NUREG/CR-7219, “Cladding Behavior during Postulated Loss-of-Coolant Accidents,” such that the capability to cool the core is challenged. For LOCA, the collapsed downcomer level is maintained at or above TAF, or fuel cladding temperature is maintained within normal operating temperature range.

Fission Product Barrier or Fundamental Safety Function	Qualitative Acceptance Criteria	Quantitative Acceptance Criteria
Reactor Coolant Pressure Boundary	Design conditions of the reactor coolant pressure boundary are not exceeded during the most severe pressurization transient because of a DBA.	The calculated peak pressure associated with the RCPB does not exceed 120% of the design pressure or 1800 psig (12.41 MPaG).
	The reactor coolant pressure boundary maintains sufficient reactor coolant inventory for core cooling.	Conformance is demonstrated by meeting the fuel cooling and long-term heat removal criteria.
Primary Containment	Containment pressures and temperatures are maintained below the design values.	The calculated containment pressure does not exceed the design pressure 60 psig (0.414 MPaG). The calculated containment shell temperature does not exceed the design temperature 330°F (165.6°C).
	The local combustible gas concentrations in the containment are within the range where deflagration or detonation cannot occur.	Containment atmosphere remains sufficiently mixed such that deflagration or detonation thresholds are not exceeded.
	Containment capability will be retained to reduce the containment pressure and temperature following a DBA to minimize the release of fission products to the environment and to preserve containment integrity and leak tightness.	The calculated containment pressure reduces to less than 50% of the calculated peak pressure for the most limiting LOCA within 24 hours.
Reactivity Control	Reactivity control required to bring the reactor to cold shutdown is maintained.	The core is capable of being made subcritical, with margin, in the most reactive condition throughout the operating cycle with the most reactive control rod fully withdrawn or rod pair associated with a single HCU postulated stuck in the full out position, with all other rods fully inserted. The subcriticality value is 0.38% Δk/k with the highest-worth control rod pair analytically determined.

Fission Product Barrier or Fundamental Safety Function	Qualitative Acceptance Criteria	Quantitative Acceptance Criteria
Long-Term Heat Removal	SSCs important for preserving the integrity of the reactor core and the containment are capable of removing residual heat for an extended period both during and after all applicable PIEs considered in all operational states, and DBAs.	Long-term cooling is maintained for a minimum of 72 hours independent of operator action and offsite support, and for 30 days with credit for operator actions and on-site resources. For DBA events that result in shutdown, the plant can achieve and maintain safe-shutdown conditions with the average reactor coolant temperature below 420°F (215.6°C) within 36 hours.

Table 15.3-3: Probabilistic Safety Goals

Qualitative Acceptance Criteria	Derived Quantitative Acceptance Criteria
Core damage frequency	The sum of frequencies of all event sequences that can lead to significant core degradation is less than 1E-5/Reactor-yr.
Large release frequency	The calculated sum of frequencies of all event sequences that can lead to any release to the environment that requires long-term relocation of the local population is less than 1E-6/Reactor-yr.

15.4 Human Actions

The BWRX-300 SSC design minimizes complexity while enhancing reliability and reducing the potential for human error.

15.4.1 Human Actions in Deterministic Safety Analysis

For PIEs and event sequences in the AOO or DBA event categories, the FSFs can be performed and maintained for 72 hours without operator action. This is substantiated by the CN-DSA that is provided in Subsection 15.5.4. The following BWRX-300 design features contribute to fulfilling the FSFs without operator actions:

1. Fail-safe (not reliant on external power) safety system actuations for DBAs.
2. Automatic, reliable actuation of the control rods with either stored energy or motors to shut down the reactor and maintain it in a guaranteed shutdown state via latching mechanisms.
3. Passive decay heat removal by ICS.
4. Fail-safe containment isolation and passive containment heat removal.

No reliance upon operator actions for 72 hours does not mean that operator actions are not performed in these event scenarios. Proper operator responses can lessen the severity of event sequences and prevent unnecessary challenges for equipment relied upon for safety. Operator responses in accordance with operator procedures are part of DL1 and are integral to the BWRX-300 D-in-D concept.

Operator actions are reflected in certain safety analyses, both deterministic and probabilistic, allowing understanding of when the actions are beneficial, and informing operating procedure development. Additionally, in certain types of very low likelihood event sequences, operator actions are credited supporting FSFs performance. Examples include complex sequences involving failure to shut down the reactor and external hazards probabilistically combined with unrelated yet simultaneous CCFs of systems protected from and qualified for the hazard.

The HFEA identifies failures that involve an erroneous decision or action taken by a human that can potentially initiate a PIE.

15.4.2 Human Actions in Probabilistic Safety Assessment

Human actions resulting from PSA event evaluations are discussed in Subsection 15.6.3.

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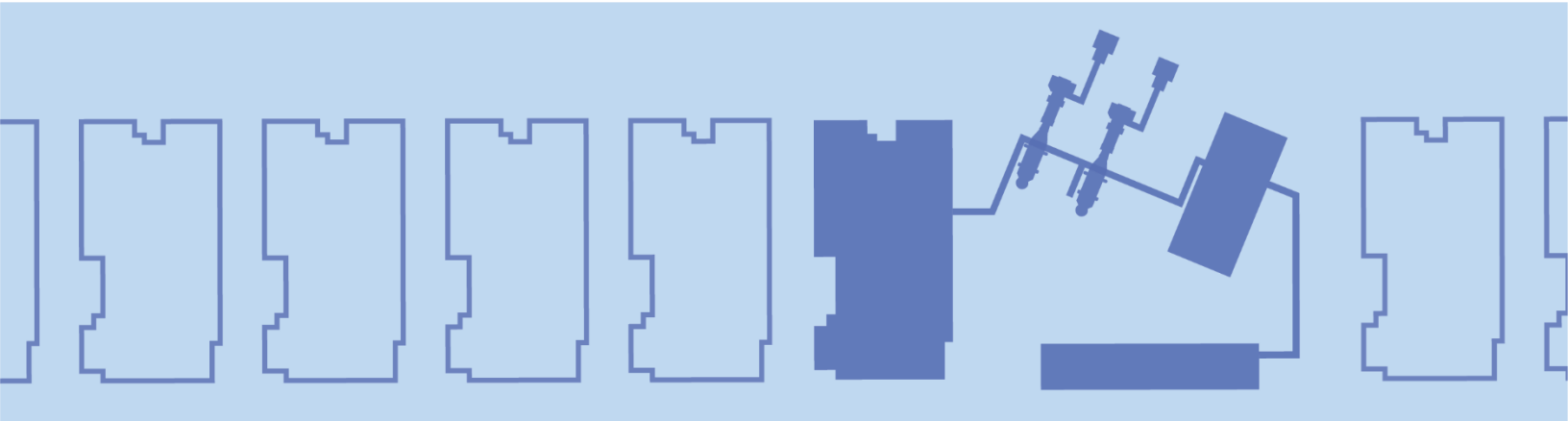
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Non-Proprietary Information

Chapter 15
Section 15.5
Safety Analyses (Part 2)



ACRONYM LIST

Acronym	Explanation
2D	Two-Dimensional
3D	Three-Dimensional
ABWR	Advanced Boiling Water Reactor
ACRW	All Control Rod Withdrawal at Power
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
AST	Alternative Source Term
ATLM	Automatic Thermal Limit Monitor
BDBA	Beyond Design Basis Accident
BECPV	Blue Energy Center - Port of Victoria
BL-AOO	Baseline Abnormal Operational Occurrence
BL-DSA	Baseline Deterministic Safety Analysis
BOC	Beginning of Cycle
BWR	Boiling Water Reactor
CB	Control Building
CCF	Common Cause Failure
CEDE	Committed Effective Dose Equivalent
CEM	Containment Evaluation Method
CFR	Code of Federal Regulations
CIV	Containment Isolation Valve
CN-DBA	Conservative Design Basis Analysis
CN-DSA	Conservative Deterministic Safety Analysis
COLR	Core Operating Limits Report
CPR	Critical Power Ratio
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
CSAU	Code Scaling, Applicability, and Uncertainty
CUW	Reactor Water Cleanup System
DBA	Design Basis Accident
DCF	Dose Conversion Factor

Acronym	Explanation
DDE	Deep Dose Equivalent
DEC	Design Extension Condition(s)
DF	Decontamination Factor
DL	Defense Line
DR	Decay Ratio
DSA	Deterministic Safety Analysis
EAB	Exclusion Area Boundary
ECCS	Emergency Core Cooling System
EFCV	Excess Flow Check Valve
EOR	End of Rated Power
EPRI	Electric Power Research Institute
ESBWR	Economic Simplified Boiling Water Reactor
EX-DEC	Extended Design Extension Condition
EX-DSA	Extended Deterministic Safety Analysis
FHA	Fuel Handling Accident
FMCRD	Fine Motion Control Rod Drive
FSAR	Final Safety Analysis Report
FSF	Fundamental Safety Function
FW	Feedwater
FWCIV	Feedwater Containment Isolation Valve
FWFI	Feedwater Flow Increase
FWLB	Feedwater Line Break
FWPT	Feedwater Pump Trip
FWRIV	Feedwater Reactor Isolation Valve
GEH	GE Hitachi Nuclear Energy
GNF	Global Nuclear Fuel
HEPA	High Efficiency Particulate Air
HVS	Heating, Ventilation, and Cooling System
IC	Isolation Condenser
ICLB	Isolation Condenser Line Break
ICRW	Inadvertent Control Rod Withdrawal at Power – Single Rod
ICS	Isolation Condenser System

Acronym	Explanation
ILB	Instrument Line Break
LFWH	Loss of Feedwater Heating
LHGR	Linear Heat Generation Rate
LOCA	Loss-of-Coolant Accident
LOCV	Loss of Condenser Vacuum
LOPP	Loss-of-Preferred Power
LPZ	Low Population Zone
LTR	Licensing Topical Report
MCPR	Minimum Critical Power Ratio
MCRE	Main Control Room Envelope
MOC	Middle of Cycle
MpaG	Mega Pascal Gauge
MRBM	Multi-Channel Rod Block Monitor
MS	Main Steam
MSCIV	Main Steam Containment Isolation Valve
MSL	Main Steam Line
MSLB	Main Steam Line Break
MSRIV	Main Steam Reactor Isolation Valve
NUREG	Nuclear Regulatory Report
OLMCPR	Operating Limit Minimum Critical Power Ratio
PCCS	Passive Containment Cooling System
PCT	Peak Clad Temperature
PI	Pressure Increase
PIE	Postulated Initiating Event
PIRT	Phenomenon Identification and Ranking Table
PRA	Probability Risk Assessment
PSA	Probabilistic Safety Assessment
PSAR	Preliminary Safety Analysis Report
PSIG	Pounds per Square Inch Gauge
PWR	Pressurized Water Reactor
RB	Reactor Building
RCPB	Reactor Coolant Pressure Boundary

Acronym	Explanation
RG	Regulatory Guide
RLC	Reactor Level Control
RPC	Reactor Pressure Control
RPF	Radial Peaking Factor
RPV	Reactor Pressure Vessel
SCRRI	Selected Control Rod Run-in
SDC	Shutdown Cooling System
SER	Safety Evaluation Report
SMR	Small Modular Reactor
SSC	Structure, System, or Component
STP	Simulated Thermal Power
TAF	Top of Active Fuel
TB	Turbine Building
TBV	Turbine Bypass Valve
TCV	Turbine Control Valve
TD	Temperature Decrease
TEDE	Total Effective Dose Equivalent
TRACG	Transient Reactor Analysis Code General Electric
TSV	Turbine Stop Valve
USNRC	U.S. Nuclear Regulatory Commission
ΔCPR/ICPR	Delta Critical Power Ratio Over Initial Critical Power Ratio

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15.5 Deterministic Safety Analyses

15.5.1 General Description of the Approach

The DSA is completed in two steps:

- The plant response to event sequences is evaluated and analyzed to confirm the performance of the fission product barriers against the derived acceptance criteria
- The event dose consequences resulting from a fission product release or other source of radiation, such as the reactor coolant, is radiologically analyzed

As described in Subsection 15.2.4, the DSA is performed based on the outputs of the fault evaluation. There are three layers of DSA performed (see Section 15.1 and Subsection 15.2.1 for additional details) that credit different sets of DLs where the bounding event sequence is performed for the transient or non-LOCA and LOCA events:

Deterministic Safety Analysis Approach for Non-LOCA Events

Non-LOCA or Transient DSA analyzes event sequences where the Reactor Coolant Pressure Boundary (RCPB) remains intact. These events are broken down into groups that result in similar core responses. Subsection 15.2.4 describes the core response during off-normal conditions, the groups determined for BWRX-300 (Table 15.2-1), and the selected bounding event scenarios for AOOs, DBAs and DECs without core damage analyses (summarized in Table 15.2-2).

The BWRX-300 scenarios are identified through the fault evaluation. The methods and assumptions described in NEDC-34043P, “BWRX-300 TRACG Application,” (Reference 15.5-1) are used to confirm the performance of the fission product barriers for the DSA non-LOCA events. The Transient Reactor Analysis Code General Electric (TRACG) application for both non-LOCA and LOCA event analysis is discussed in Subsection 15.5.1.2.1.

The TRACG Application for BWRX-300 also includes the stability analysis that evaluates potential coupled thermal hydraulic – neutronic instabilities in the reactor core. TRACG is used to perform transient safety analysis and stability analysis for both forced flow and natural circulation BWR designs. Previous TRACG applications as well as BWRX-300 use the systematic approach called Code Scaling, Applicability, and Uncertainty (CSAU) to confirm the applicability of a computer code for DSA that conforms to NUREG/CR-5249, “Quantifying Reactor Safety Margins: Application of Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large-Break, Loss-of-Coolant Accident,” (Reference 15.5-2). This approach involves systematic evaluation of the phenomena that are important for the plant design and accident scenarios identified. A qualitative process is used to identify and rank the importance of phenomena.

Through this process, a Phenomenon Identification and Ranking Table (PIRT) is established that conforms with RG 1.203, “Transient and Accident Analysis Methods.” The PIRT is used together with the TRACG documentation to systematically demonstrate the applicability of TRACG models and the qualification of the TRACG model to predict the phenomena. Defining the nodalization and evaluation of the effects of scale are included. In addition to code applicability and qualification, the PIRT is also used as the basis to perform quantitative uncertainty analysis of transient scenarios, if needed. Additional information regarding the approach for addressing uncertainty in the DSA is provided in Subsection 15.5.1.1.

The TRACG applicability to model phenomena also requires that the code capability be demonstrated to apply the code in the intended manner with a qualifying result achieved. TRACG capability to model phenomena is important to BWRX-300 simulation and is consistent with modern best practices. TRACG qualification is based upon proven practices for verification and validation using acceptable codes and standards. Experiments and plant events used to validate TRACG provide evidence that TRACG can be applied for the BWRX-300 design.

Integral to the capability of TRACG for transient DSA is the use of three-dimensional nuclear kinetics input. This input comes directly from and essentially uses the same methods as the steady state core simulator code, PANAC11. PANAC11 is used in the BWRX-300 as described in Subsection 4.3.3.1 and Subsection 15.5.1.2.3. Other code interfaces are described in NEDC-34043P.

Design control procedures require independent verification of safety analysis calculations to ensure that results are properly summarized from calculations, physically sound/correct, and consistent with expected results when compared to previous calculations. The results are then confirmed to meet the appropriate acceptance criteria. Table 15.5-43 provides additional insight in conservatisms used in the transient analysis.

Deterministic Safety Analysis Approach for LOCA Events

The methods and assumptions of the DSA confirming the performance of the fission product barriers for LOCAs are described in NEDC-33922P-A, “BWRX-300 Containment Evaluation Method,” (Reference 15.5-3) that is Incorporated by Reference.

TRACG calculates the mass and energy release from modeled breaks of various sizes and locations. Atmospheric pressure is used for the TRACG pressure boundary condition for any breaks. This approach provides no credit for the back pressure from containment. Consequently, the retained Reactor Pressure Vessel (RPV) inventory calculated by TRACG represents the minimum coolant volume. This modeling provides results as if the break occurred outside containment.

Breaks inside containment realistically experience back pressure from containment that reduces the mass and energy calculated by TRACG once the break flow becomes unchoked. However, this effect is not treated explicitly because it requires two-way coupling between the TRACG calculation and the GOTHIC containment calculation. Instead, the methodology has a one-way coupling with the mass and energy release rates conservatively calculated by TRACG that supplies inputs to the GOTHIC calculation up until the point in time when the containment and RPV pressures first equalize. Choked flow naturally satisfies the assumed one-way coupling because choked flow does not depend on the downstream pressure. Select TRACG inputs are specified so that mass and energy release rates are conservatively calculated.

Rapid mass and energy releases into containment occur before a large break is isolated. This leads to the highest containment peak pressure at approximately the same time that the break is isolated. For large breaks, the containment shell is the dominant short-term energy sink, and it causes containment pressure to decrease from its peak value after isolation of the break occurs.

Compared to large breaks, small unisolated breaks have a much slower mass and energy release rate from the RPV into containment. The lowest break on the RPV that remains unisolated and occurs outside containment produces the most limiting scenario for minimum RPV inventory. Regardless of break location and whether it is inside or outside containment, break flow slowly decreases with time because the RPV is being depressurized largely due to the ICS and to a much lesser extent by the break flow.

Containment pressure slowly increases and eventually equals the RPV pressure for breaks inside containment. It is not realistic to use the TRACG break flow that was calculated using an atmospheric pressure boundary condition as input to the GOTHIC containment calculation after the point in time when containment and RPV pressures first equal each other. A better approximation is to assume zero break flow after this point in time, but this could potentially be nonconservative with respect to the longer-term calculated containment pressures. The proposed methodology does not require the GOTHIC calculation to continue beyond the point where the

containment and RPV pressures equalize, because the longer-term containment pressure is bounded by the RPV pressure calculated.

The DL3 functions credited in the conservative DBA LOCA analyses and the DL2 and DL4a functions credited in the DEC LOCA analyses are described in Table 15.5-44 and Table 15.5-45, respectively. The methods and assumptions for radiological analyses are described for these events in Subsection 15.5.9.

15.5.1.1 Safety Margins in Safety Analyses

The safety margin is the result of the conservative assumptions used in the analysis and design rules applied to the SSC design capabilities. In addition, the DSA demonstrates that the challenges to the physical barriers do not exceed their physical capacity.

Uncertainties in initial conditions and methods are accounted for in the CN-DSA using RG 1.203. The BL-DSA and EX-DSA are performed using best-estimate methods based on the purpose of the BL-DSA and EX-DSA analyses. For CN-DSA thermal-hydraulic analysis, a graded approach is used combining uncertainties. The graded approach involves a qualitative assessment of the safety margin on a case-by-case basis and includes a review of the magnitude of results compared to acceptance criteria along with the judgment of conservatism in the derived acceptance criteria.

The DSA confirms the FSFs successfully keep plant radioactive material releases within the acceptance criteria with adequate safety margins.

15.5.1.1.1 Large Margin

For events with large margin or substantially non-limiting, there is no need to apply uncertainty to the analysis methodology. Judgment is used to establish what is “large margin” or “substantially non-limiting.” Instead of a quantitative evaluation of uncertainty, the event is dispositioned qualitatively based on the uncertainty evaluation performed for a limiting event of a similar type, historical analysis of similar type, or other qualitative based disposition.

Large Margin Examples

The inadvertent isolation condenser initiation in Subsection 15.5.4.3.2 and the closure of all MSRIVs and FWRIVs in Subsection 15.5.4.2.4 are examples of large margin events. There are many DBA events that have minimal effects compared to the acceptance criteria.

Inadvertent Isolation Condenser Initiation

This CN-DBA event results in a peak pressure of 7.32 MPa, much lower than the acceptance criteria of 12.41 MPaG and is bounded by pressure increase CN-DBA events in Subsection 15.5.4.2. Also, the PCT is 586.3°F (308.0°C), much lower than the acceptance criteria in Table 15.3-2 and is bounded by the generator load rejection CN-DBA event in Subsection 15.5.4.2.1. In this event, there is no concern for cladding oxidation, and there is no threat to the containment pressure boundary.

Closure of All MS Reactor Isolation Valves and FW Isolation Valves

This CN-DBA event results in a peak pressure of 8.73 MPa, much lower than the acceptance criteria of 12.41 MPaG. The peak pressure is bounded by other pressure increase CN-DBA events in Subsection 15.5.4.2. Also, the PCT is 594.9°F (312.7°C), much lower than the acceptance criteria in Table 15.3-2. This event has a large safety margin to the DBA acceptance criteria. In this event, there is no concern for cladding oxidation, and there is no threat to the containment pressure boundary. As a result, for the above events and any other events with similar margin, there is no need for any quantification of uncertainty, and a qualitative disposition is adequate.

15.5.1.1.2 Medium Margin

In medium margin scenarios, method uncertainty is addressed by biasing key important phenomena in a conservative direction (typically one or two sigma). Input parameters such as power, pressure, level, or temperature are based on using the most limiting normal operating values. In these cases, the selection of key important phenomena is dependent on the specific event evaluated. Important phenomena can be different for the output parameters when multiple output parameters are considered for selecting the bias direction. The selection of the key important phenomena and determining the bounding bias direction is considered for each output parameter that has medium margin and compared to the derived acceptance criteria.

Medium Margin Example

A medium margin event is the large break inside containment described in Subsection 15.5.4.5.

Large Pipe Breaks Inside Containment

There are multiple large pipe breaks inside containment event scenarios evaluated that have commonalities. These events result in no significant fuel cladding heatup and are not bounding with respect to maintaining inventory above the fuel (to ensure continued cooling). They represent the largest challenge to the containment fission product barrier. These events are treated as medium margin events and the initial conditions and modeling parameters are biased to ensure conservative containment conditions are calculated. The initial conditions used are provided in Table 15.5-1. The modeling parameters biased in this analysis are discussed in NEDC-33922P-A, that is incorporated by reference. The combination of the conservative biased inputs combined with the observation of the margin available results in conservative analyses.

15.5.1.1.3 Low Margin or Quantitative Evaluation of Uncertainty is Desired

A proven Monte Carlo technique is used to combine the individual biases and uncertainties into an overall bias and uncertainty for low margin events. This process is described in the NEDC-34043P. There are no events identified as low margin.

15.5.1.2 Description of the Computer Codes or Standards Used in the Safety Analyses

There is a large amount of data available from operating BWR plants and from the testing and licensing efforts to license the predecessor BWR/Advanced Boiling Water Reactor (ABWR)/Economic Boiling Water Reactor (ESBWR) designs and individual plants. The vast database of feature performance in licensed reactors, combined with the recent thorough licensing review of the ABWR and ESBWR, provides an extremely well qualified foundation from which to make the modest extrapolations to the BWRX-300. The following codes, methods, and accompanying assumptions are used in evaluating the performance of the BWRX-300. The results for DBAs and DECs reported in Section 15.7 are calculated using the RADTRAD radiological consequence code. Short-term atmospheric dispersion factors at the site boundary and control room intakes are calculated using the PAVAN and ARCON computer codes, respectively.

15.5.1.2.1 TRACG

TRACG is a GE Vernova Hitachi Nuclear Energy (GVH) proprietary version of the Transient Reactor Analysis Code. TRACG is the primary licensing analysis tool for LOCA and transient analyses for PIEs with a large range of frequencies up to events that do not involve significant core damage (severe accidents). TRACG has been used in a variety of applications for operating BWRs as well as design/analysis for the ESBWR.

TRACG uses advanced realistic one-dimensional and three-dimensional methods to model the phenomena that are important in evaluating the operation of BWRs. It is a best-estimate code for

analysis of BWR transients ranging from simple operational transients to design basis LOCAs and failure to scram transients. TRACG has an extensive qualification base for separate effects, BWR fuel and components, and integral tests. It has been reviewed and approved by the US NRC for several analysis applications such as AOOs, Emergency Core Cooling System (ECCS)/LOCA and failure to scram overpressure (a BDBA event for BWRs) analyses and has been reviewed and approved in the NEDC-33922P-A, that is incorporated by reference.

TRACG is used to analyze the challenges to the fuel, RPV, and the mass and energy releases to the containment, for LOCA and non-LOCA DSA. TRACG draws from the licensed BWR database, which includes design features of the BWRX-300 (albeit in various configurations) and appropriate testing and allows direct application to BWRX-300 design and analysis. TRACG is maintained and updated by GVH.

Scope and Capabilities

TRACG is based on a multi-dimensional two-fluid model for the reactor thermal hydraulics and a three-dimensional neutron kinetics model. The TRACG output is also used to provide mass and energy release input to the containment analysis code GOTHIC. GOTHIC solves the conservation equations for mass, momentum, and energy for the gas and liquid phases. TRACG does not include any assumptions of thermal or mechanical equilibrium between phases. The gas phase may consist of a mixture of steam and a non-condensable gas, and the liquid phase may contain dissolved boron. The thermal-hydraulic model is a multi-dimensional formulation for the vessel component and a one-dimensional formulation for all other components.

The conservation equations for mass, momentum, and energy are closed through an extensive set of basic models consisting of constitutive correlations for shear and heat transfer at the gas/liquid interface as well as at the wall. The constitutive correlations are flow regime-dependent and are determined based on a single flow regime map, which is used consistently throughout the code. In addition to the basic thermal-hydraulic models, TRACG contains a set of component models for BWR components, such as fuel channels, steam separators, and can simulate BWR steam dryers as part of its vessel model. TRACG also contains a control system model capable of simulating the major BWR control systems such as those for pressure and water level.

The neutron kinetics model is consistent with the GVH BWR core simulator code PANAC11. It solves a modified one-group diffusion model with six delayed neutron precursor groups.

Feedback is provided from the thermal-hydraulic model to the kinetics model for moderator density, fuel temperature, boron concentration and control rod position.

The TRACG structure is based on a modular approach. The TRACG thermal-hydraulic model contains a set of basic components, such as pipe, valve, tee, channel, steam separator, heat exchanger and vessel. System simulations are constructed using these components as building blocks. Any number of these components may be combined. The number of components, their interaction, and the detail in each component are specified through code input. Consequently, TRACG has the capability to simulate a wide range of facilities, ranging from simple separate effects tests to complete BWR plants.

TRACG has been extensively qualified against separate effects tests, component performance data, integral system effects tests and full-scale BWR plant data. A detailed documentation of the TRACG qualification is contained in NEDE-32177P, “TRACG Qualification,” (Reference 15.5-4). More recent qualification for passive BWR simulation tests is reported in NEDC-32725P, TRACG Qualification for SBWR, and NEDC-33080P, TRACG Qualification for ESBWR. TRACG is included as an approved methodology cited in NEDE-24011-P-A, “General Electric Standard Application for Reactor Fuel (GESTAR II).”

Scope of Application of TRACG to BWRX-300

The PIRT discussed in Reference 15.5-1 and Reference 15.5-3 identifies specific governing phenomena in predicting BWRX-300 transient and LOCA performance. Most of these phenomena are common to operating BWRs. This section examines specific Simplified Boiling Water Reactor (SBWR)/ESBWR-related tests and test facilities beyond the previous qualification database. Early in the SBWR program, it was identified that there was no information in the database for a heat transfer correlation for steam condensation in tubes in the presence of non-condensable gases. A Single Tube Condensation Test Program was conducted to secure this information. Test program results were reported to the USNRC in TRACG Qualification Reports for SBWR (NEDC-32725P) and ESBWR (NEDC-33080P). These reports are referenced in the TRACG model for the BWRX-300 as described in NEDC-33922P-A, that is incorporated by reference.

The test program was conducted to investigate steam condensation inside tubes in the presence of non-condensable gases. The work was independently conducted at the University of California at Berkeley and at the Massachusetts Institute of Technology (MIT). The work was initiated to obtain a database and a correlation for heat transfer and condensation inside tubes. Three researchers utilized three separate experimental configurations at the University of California at Berkeley, while two researchers utilized one configuration at MIT. The researchers ran tests with pure steam, steam/air, and steam/helium mixtures with representative and bounding flow rates and non-condensable mass fractions. The researchers found the system well-behaved for all tests, with either of the non-condensable gases. The results of the tests at the University of California at Berkeley are the basis for the condensation heat transfer correlation used in the TRACG computer code.

TRACG ICS modeling is qualified by the PANTHERS IC test using a representative configuration. The steady state heat exchanger performance was predicted by the PANTHERS-IC prototypical geometry full-scale test.

Because the BWRX-300 RPV and ICS are similar to those of the ESBWR, the TRACG method developed for the ESBWR RPV thermal hydraulics and mass energy release is also used for the BWRX-300 RPV thermal hydraulics and mass and energy release. The TRACG code and the ECCS-LOCA application method developed for ESBWR was reviewed and approved by the USNRC in NEDC-33083P-A, “TRACG Application for ESBWR,” (Reference 15.5-20). That application method was developed using the Code Scaling, Applicability and Uncertainty (CSAU) guidance. NEDC-33922P-A, that is incorporated by reference, provides an overview of the TRACG thermal hydraulics method for the mass and energy release and its applicability to the BWRX-300 RPV. As stated in Section 5.1 of NEDC-33922P-A, only the uncertainties relating to the RPV inventory and break flow need to be accounted for in the BWRX-300 application because like the ESBWR application, the core always remains well cooled.

BWRX-300 LOCA Evaluation Model

NEDC-33910P-A, Revision 2, “BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection,” (Reference 15.5-21) described the BWRX-300 approach for LOCA mitigation and established the BWRX-300 ECCS to be made up of the ICS, which cools the core, and the RIVs, which close quickly to limit coolant losses. The BWRX-300 is designed to mitigate LOCAs using inherent margins (e.g., reducing number and size of RPV nozzles, placing nozzles high on the RPV, increasing water volume above the core); therefore, no active or passive injection of additional water inventory is necessary. NEDC-33910P-A established LOCA acceptance criteria for the BWRX-300 to be reactor water level is maintained at or above top of active fuel or fuel cladding temperature is maintained within normal operating temperature range. The BWRX-300 design ensures that fuel overheat does not occur by satisfying the BWRX-300 LOCA acceptance

criteria. As discussed in NEDC-33910P-A, the BWRX-300 LOCA acceptance criteria bound the 10 CFR 50.46(b) acceptance criteria:

1. Peak cladding temperature—while this criterion would allow for maximum cladding temperature to reach up to 2200°F, the BWRX-300 LOCA acceptance criteria ensures fuel remains wetted or that peak cladding temperature is within normal operating range.
2. Maximum cladding oxidation—The BWRX-300 LOCA acceptance criteria ensure that fuel overheat does not occur; thus, cladding oxidation cannot occur.
3. Maximum hydrogen generation—The BWRX-300 LOCA acceptance criteria ensure that fuel overheat does not occur; thus, hydrogen generation is not a concern.
4. Coolable geometry—The BWRX-300 LOCA acceptance criteria ensure that fuel overheat does not occur; thus, maintaining coolable geometry is not a concern.
5. Long-term cooling—The BWRX-300 LOCA acceptance criteria ensure that fuel overheat does not occur, and the inventory in the ICS pools is adequate to ensure decay heat is removed for extended periods of time, as described in PSAR Subsection 9.2.8.2.

As the BWRX-300 LOCA acceptance criteria conservatively bound the 10 CFR 50.46(b) acceptance criteria, there is not a need to address uncertainty associated with each 10 CFR 50.46(b) criterion. Because the BWRX-300 LOCA acceptance criteria do not allow fuel overheat, and consequently excessive oxidation, the uncertainties of the method become limited to the uncertainties in maintaining inventory in the RPV, which would be the same as uncertainties in mass and energy release from the RPV.

NEDC-33922P-A is incorporated by reference and describes the TRACG model for calculating mass and energy releases from the RPV. The TRACG model described in NEDC-33922P-A, Section 5, is the BWRX-300 LOCA EM used to analyze LOCA DBAs identified in NEDC-33910P-A. Because the BWRX-300 LOCA acceptance criteria limit the uncertainties that need to be considered to those related to maintaining inventory in the RPV, there are no additional uncertainties for the BWRX-300 LOCA EM that are not discussed in NEDC-33922P-A, which already covers uncertainties related to TRACG qualification and those specific to BWRX-300 mass and energy releases. Section 5.1 of NEDC-33922P-A discusses the reviews and qualifications performed for TRACG, such as the TRACG qualification report (Reference 15.5-4) that includes comparisons of TRACG calculations with data from separate effects, component performance and integral system effects tests that directly support its use for BWR LOCA analyses. The use of TRACG in the ESBWR ECCS-LOCA EM was previously approved (Reference 15.5-20), and Section 5.2 of NEDC-33922P-A addresses the differences in the TRACG application between the ESBWR and the BWRX-300. Additionally, NEDC-33922P-A, Section 5 presents demonstration LOCA analysis cases for small and large breaks and shows that the BWRX-300 LOCA acceptance criteria established in NEDC-33910P-A are met. Because the BWRX-300 design can mitigate LOCAs without uncovering the core or resulting in significant fuel heat-up and oxidation, the majority of RG 1.157 does not apply to BWRX-300 LOCA analyses, as discussed in Section 1.3 of NEDC-33922P-A. Only the guidance related to general best practice elements of a rigorous method development are applicable, and RG 1.203 also addresses these best practice elements, by providing guidance for all elements and stages of developing an acceptable method and utilization of the CSAU framework. The application method presented in NEDC-33922P-A addresses conformance with RG 1.203 elements that are applicable to mature computer codes, which are utilized by the BWRX-300, and the CSAU methodology.

The TRACG method for mass and energy release described in NEDC-33922P-A is the method used as the BWRX-300 LOCA EM. The NEDC-33922P-A model for computing mass and energy

releases, which is incorporated by reference, along with conclusions of the assessment provided in this section provide the BWRX-300 LOCA EM compliance basis pursuant to 10 CFR 50.46(a)(1)(i).

1. **Novelty of the revised EM compared to the currently acceptable model**

   The evaluation model used as the BWRX-300 LOCA EM is the TRACG method for mass and energy release, as described in NEDC-33922P-A, Section 5. The BWRX-300 LOCA EM is essentially the same as the previously NRC-approved ESBWR LOCA EM, as discussed in Section 5.2 of NEDC-33922P-A, which already assessed differences between ESBWR and BWRX-300. Thus, this is not a novel method.

2. **Complexity of the event being analyzed**

   As discussed above, LOCA event progressions for BWRX-300 are relatively simple, as the breaks are isolated quickly by the RIVs and ICS performs well to meet the conservative BWRX-300 LOCA acceptance criteria. The LOCA events presented in PSAR Section 15.5.4.5 (main steam pipe break inside containment, feedwater pipe break inside containment, large ICS pipe breaks inside containment, and small steam and liquid pipe breaks inside containment) are the same set of events discussed in NEDC-33922P-A, Sections 5.3, 5.4 and 5.5.

3. **Degree of conservatism in the EM**

   The BWRX-300 LOCA EM conservatisms include the more stringent acceptance criteria for LOCA DBAs than those in 10 CFR 50.46(b), as described in NEDC-33910P-A, and the conservative assumptions and inputs associated with the TRACG mass and energy release model, as described in NEDC-33922P-A, Section 5.

4. **Extent of any plant design or operational changes that would require reanalysis**

   The BWRX-300 has not made substantive design operational changes that would require the EM to be updated. The limitations and conditions from NEDC-33922P-A are discussed in PSAR Section 15.5.4.5. As the detailed design is developed, LOCA analysis will be repeated to demonstrate that the BWRX-300 LOCA acceptance criteria continue to be met.

15.5.1.2.2 GOTHIC

Containment analysis is performed by using the GOTHIC code (Reference 15.5-5).

The GOTHIC computer code is a state-of-the-art program for modeling multiphase, multicomponent fluid flow for performing both containment DBA analyses and analyses to support equipment qualification. The GOTHIC code is developed by Numerical Applications Incorporated, and the development program is sponsored by the Electric Power Research Institute (EPRI).

The GOTHIC code has a noding structure that allows both lumped parameter and Three-Dimensional (3D) modeling capabilities. The multidimensional analysis capability facilitates the study of non-condensable gas and stratification and the calculation of flow field details within any given volume. The code has undergone extensive review and validation against a large test array. The validation program scope examines the code capability for predicting pressure and temperature as well as hydrogen distribution and mixing under various conditions.

GOTHIC is a continuously maintained and improved computer code. The GOTHIC code has been developed compliant with Title 10 Code of Federal Regulations (CFR) Part 50, Appendix B, Quality, meeting the GVH software quality requirements. The PSAR results generated use GOTHIC Version 8.3 that is the latest released version. Future BWRX-300 containment analyses may be performed using newer versions of the GOTHIC code provided the newer versions meet

the same 10 CFR 50, Appendix B quality standards, as well as changes in calculated results for the BWRX-300 containment application caused by any code changes that can be successfully dispositioned.

15.5.1.2.3 PANAC11

The BWR Core Simulator (PANAC11) is described in Subsection 4.3.3 and is a steady-state, 3D coupled nuclear-thermal-hydraulic computer program representing the BWR core exclusive of the external flow loop.

The analytical methods used in the design and analysis of the BWRX-300 are described in detail in NEDC-34039P, "BWRX-300 GNF2 Steady State Nuclear Methods TGBLA06/PANAC11 Application Methodology," (Reference 15.5-6).

15.5.1.2.4 ANSI/ANS-18.1-2020 Standard

Radiation concentrations in the reactor coolant and steam during normal operations are determined based on American National Standards Institute (ANSI)/American Nuclear Society (ANS)-18.1, "Radioactive Source Term for Normal Operation of Light Water Reactors," (Reference 15.5-7). This standard provides the bases for estimating typical concentrations of the principal radionuclides that may be anticipated over the lifetime of a BWR plant. The source term data is based on the cumulative industry experience at operating BWR plants, including measurements at several stations. The operating data reflects the influence of several observations made during the transition period from operation with fuel of older designs to operation with fuel of current improved designs such as the GNF2 fuel used in the BWRX-300.

15.5.1.2.5 PAVAN

NUREG/CR-2858, "PAVAN: An Atmospheric-Dispersion Program for Evaluating Design-Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations," (Reference 15.5-8) code is used in Section 2.3 to calculate offsite atmospheric dispersion factors using meteorological data (i.e., wind speed, wind direction, and measurements of atmospheric stability) recorded at the site, and the PAVAN uses Joint Frequency Distributions of wind direction, wind speed, and atmospheric stability class to estimate X/Q values for specific averaging time periods at specified distances. The PAVAN model is based on a straight-line Gaussian model that assumes the release rate is constant for the entire release period. PAVAN calculations implement the models in RG 1.145.

The RG 1.145 methods and the PAVAN code calculations incorporate the results of several field tracer tests and diffusion experiments for elevated and ground-level releases from various locations on reactor facility buildings during stable atmospheric conditions with low wind speeds. PAVAN calculated X/Q values are made for either assumed ground-level releases (e.g., through building penetrations and vents) or elevated releases from free-standing stacks. The dispersion theories applied, user instructions, code algorithms, FORTRAN source code, and test cases for the PAVAN code are documented in NUREG/CR-2858.

15.5.1.2.6 RADTRAD

The dose consequences of postulated DBAs are calculated using the RADTRAD Version 3.10 computer code NUREG/CR-6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation, (Reference 15.5-9). The RADTRAD code was developed by the Accident Analysis and Consequence Assessment Department at Sandia National Laboratories for the USNRC, Office of Nuclear Reactor Regulation, Division of Reactor Program Management.

RADTRAD is a simplified model for Radionuclide Transport and Removal and Dose estimation. The RADTRAD code uses a combination of tables and numerical models of source term reduction

phenomena to determine the time-dependent dose at specified locations both onsite and offsite for a given accident scenario. The RADTRAD code is used to assess occupational radiation exposures, typically in the control room and site boundaries. RADTRAD code is capable of estimating dose attenuation due to modification of a facility or accident sequence.

RADTRAD is a licensing analysis code used to show compliance with nuclear plant siting criteria for the radiation doses at onsite and offsite locations for various LOCA and non-LOCA DBAs. As radioactive material is transported through the containment, the user can account for sprays and natural deposition that may reduce the quantity of radioactive material. Material flow between buildings, from buildings to the environment, or into control rooms through High Efficiency Particulate Air (HEPA) filters, piping, or other connectors is modeled. An accounting of radioactive material amounts retained due to these tortuous pathways is maintained. Decay and in-growth of daughter isotopes can be calculated over time as the material is transported.

15.5.1.2.7 ARCON

ARCON, NUREG/CR-6331, "Atmospheric Relative Concentrations in Building Wakes," (Reference 15.5-10) is a computer code used to calculate atmospheric relative concentrations (X/Q) in support of control room habitability assessments required by 10 CFR Part 50, Appendix A, General Design Criterion 19. ARCON uses hourly meteorological data and the atmosphere's influence (i.e., dilution and dispersion) in the vicinity of buildings to calculate the relative concentration at control room air intakes. These concentrations would be exceeded no more than five percent of the time and calculated for averaging periods ranging from one hour to 30 days in duration.

The model is based on a straight-line Gaussian model that assumes the release rate is constant for the entire period of the release. ARCON can account for both plume meander under low wind speed conditions and the plume dispersion due to building wakes. An expanded description of ARCON code bases, capabilities, and limitations is documented in RG 1.194.

15.5.1.2.8 ORIGEN2

ORIGEN2, Version 2.1, "Isotope Generation and Depletion Code – Matrix Exponential Method," (Reference 15.5-11) is a computer code system for calculating the buildup, decay, and processing of radioactive materials. ORIGEN2 is typically used for estimating core inventories of nuclides for light water reactors at various stages of power operation.

15.5.2 Stability Analysis

Part of normal operational analysis is to confirm that the core will remain stable during normal operation. The stability considerations during normal operation and AOOs are described in Appendix 4A and meet the intent of Standard Review Plan 15.9, Boiling Water Reactor Stability.

The reference equilibrium cycle has been evaluated and confirmed to satisfy the stability core-wide Decay Ratio (DR) acceptance criterion of ≤ 0.80 as discussed in Appendix 4A. The equilibrium cycle is based on GNF2 fuel. Nominal stability analyses will be performed for an initial core and re-load cores. The DR acceptance criterion will be verified to be met for each cycle specific core. These considerations are further addressed in NEDC-34270P, "BWRX-300 Stability Analysis," (Reference 15.5-19). Parameters and values for use in performing a nominal stability analysis are captured in NEDC-34270P, with key parameters identified to confirm that the re-load core conforms with the DR acceptance criterion.

NEDC-34270P provides a statistical evaluation at the most limiting exposure condition for the equilibrium core under nominal full power operating conditions to establish the generic DR acceptance criterion of 0.80. The acceptance criterion is re-confirmed using the statistical evaluation when required as stipulated in NEDC-34270P.

As discussed below, the results from the stability analysis show that BWRX-300 is not susceptible to regional or radial modes of oscillation. The stability analysis includes the necessary provisions to address uncertainties associated with cycle specific conditions. NEDC-34270P identifies the requirements for cycle stability analyses.

The BWRX-300 240-bundle core is evaluated for Beginning of Cycle (BOC), Middle of Cycle (MOC) and End of Rated Power (EOR) exposures in determining both the core-wide decay ratios and regional mode oscillations. In addition, NEDC-34270P presents consistent, but more refined, results, such as the nominal stability response at the most limiting cycle exposure.

Core-Wide Decay Ratio

The core-wide DRs are evaluated using a step perturbation in pressure while the regional mode oscillations are evaluated using channel velocity perturbations. The primary stability evaluation is performed at nominal conditions including nominal Feedwater (FW) temperature 467.4°F (241.9°C).

Another stability evaluation is performed for the state that is reached after a Loss of Feedwater Heating (LFWH) AOO analysis as described in Subsection 15.5.3.1.1. A Selected Control Rod Run-In (SCRRI) (described in Subsection 7.3.3.2) is initiated as a mitigating response for a LFWH AOO. The analysis of LFWH AOO described in Subsection 15.5.3.1.1 assumes that the FW temperature is reduced to 377.4°F (191.9°C) at BOC, MOC and EOR. Once the reactor achieves a new steady-state condition, a step pressure perturbation is applied to evaluate core stability response.

Core-Wide Dominance

For the regional mode evaluation, based on the harmonic modes distribution of the core, the inlet velocities for all channels were perturbed by ±20% at time = 0. This harmonic power distribution is predicted by the steady state core simulator code PANAC11 and results in a line of symmetry between the two halves of the core with higher and lower predicted harmonic power. The velocity perturbations are made positive on one side of the line of symmetry and negative on the other side. This stimulates the potential harmonic oscillations (regional oscillations). The resulting channel power response of limiting channels is evaluated for susceptibility to regional mode oscillations. If the core is not susceptible to regional mode oscillations after a velocity perturbation, the initially symmetric, out of phase channel power responses come into phase after a short duration, confirming the dominance of core-wide oscillations.

Results

DR/SCRRI

The maximum nominal core-wide DR design limit is 0.80. The calculated core-wide DRs at nominal conditions are below the maximum DR allowed. The calculated DR values at the end of the LFWH event are below the maximum allowed DR and are lower than the DR values at nominal conditions. The calculated DR values at nominal temperature and LFWH with SCRRI conditions are presented in Table 15.5-2. The nominal stability response is presented on Figure 15.5-1 for the MOC exposure. The LFWH stability response with SCRRI is presented on Figure 15.5-2 for the MOC exposure.

Consistent with these results, NEDC-34270P provides a summary of nominal core-wide DR results at various cycle exposures and the nominal stability response at the most limiting cycle exposure. NEDC-34270P also provides the DR result, at the most limiting cycle exposure, for LFWH just prior to SCRRI.

Core Wide Dominance

A limiting core-wide dominance evaluation is performed at 115% rated power. At MOC, the limiting channel power values follow the expected behavior where initially symmetric, out of phase channel power responses come into phase after a short duration. The core is not susceptible to regional mode oscillations at limiting conditions, and this conclusion also applies to normal operation. The regional stability response using the limiting channels is presented on Figure 15.5-3 at the limiting nominal core exposure, or 5512 MWd/t-U.

Based on these stability evaluations, the following stability claims are supported:

1. Power oscillations that result in conditions exceeding specified acceptable fuel design limits are not possible.
2. Regional instability is not possible.
3. Design features prevent the loss of stability margin for upset events.

15.5.3 Analysis of Anticipated Operational Occurrences

15.5.3.1 Decrease in Core Coolant Temperature AOO

This section describes the bounding BL-AOO event for Temperature Decrease (TD) Group.

15.5.3.1.1 Loss of Feedwater Heating AOO

This event is designated as a BL-AOO event. The event sequence name is LFWH, and the event sequence ID is TD-LFWH_BL-AOO.

Additional LFWH AOO cases support the detailed evaluation and demonstrate the thermal hydraulic stability of the BWRX after a LFWH AOO.

Postulated Initiating Event

The event assumes a LFWH from a single failure of either the closure of one extraction steam valve or the inadvertent bypass of a FW heater. This failure is conservatively modeled as an instantaneous decrease in FW temperature that bounds the maximum FW TD resulting from a single failure. A PIE with AOO frequency results in the maximum FW temperature reduction identified in Table 15.5-3 from a loss of one FW heater.

Sequence of Event

The event sequence summary:

- Reduction in FW temperature occurs instantly resulting in an increase in power
- Reactor Level Control (RLC) compensates initially by lowering flow rate minimizing the effect on power
- SCRRI inserts control rods on indication of FW temperature reduction
- Reactor Pressure Control (RPC) maintains pressure and RLC maintains level
- A new controlled steady-state condition achieved with a new power distribution

Table 15.5-6 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL2 functions:

- DL2-27 – SCRRI on FW TD
- DL2-02 – Maintain Target Level
- DL2-01 – Maintain Target Pressure

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by decreasing the FW temperature. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5. The analysis is performed using an equilibrium core design. The event is run at BOC, MOC and EOR cycle exposure conditions.

Results

The results of the simulated loss of FW heating event are presented on Figure 15.5-4 through Figure 15.5-9. The results are presented in Table 15.7-1 for the exposure with the limiting Critical Power Ratio (CPR) response.

The reduced temperature FW enters the core and causes an increase in core inlet subcooling. This increases core power due to the negative void reactivity coefficient. RLC compensates initially by lowering FW flow rate and a SCRRI is initiated to minimize the core power increase and decrease the final steady state power. SCRRI is modeled with four control rods inserted; there is margin in the design which allows up to eight control rods to be selected for SCRRI. Steam flow and FW flow then stabilize at a lower level. The RPV water level decreases and then returns to normal level. The pressure and level remain well within the RPV water level and RCPB pressure acceptance criteria in Subsection 15.5.3.1.

The core power increase is limited. Thermal-mechanical evaluations confirm there is significant margin to centerline fuel temperature or cladding strain acceptance criteria in Subsection 15.3.1.1. Limits on the Linear Heat Generation Rate (LHGR) are included each operating cycle ensuring the centerline fuel temperature and cladding strain acceptance criteria are met. The limits on LHGR are included in the Core Operating Limits Report (COLR).

The calculated Delta Critical Power Ratio Over Initial Critical Power Ratio (ΔCPR/ICPR) is provided. This is used to set an Operating Limit Minimum Critical Power Ratio (OLMCPR) ensuring the CPR remains within the Minimum Critical Power Ratio (MCPR) acceptance criterion. This event is potentially limiting for OLMCPR and is evaluated each operating cycle to determine the core OLMCPR. The resulting limiting event OLMCPR is included in the COLR.

Sensitivity studies were performed on maximum FW pump flow (120% of rated), initial FW temperature (-6°C), and FW controller settings. The sensitivity studies demonstrated no significant change in the event sequence or results.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient more than the derived acceptance criteria for the fuel, pressure vessel, or containment. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

Because this event does not result in fuel failures or release of primary coolant to the environment, there are no radiological consequences associated with this event.

15.5.3.2 Increase in Reactor Pressure AOOs

15.5.3.2.1 Generator Load Rejection or Turbine Trip AOO

This event is in the Pressure Increase (PI) Group and is designated a BL-AOO event. The event sequence name is Generator Load Rejection or Turbine Trip (LR-TT), and the Event Sequence ID is PI-LR-TT_BL-AOO.

Postulated Initiating Event

The initiating event is either a generator load rejection or a turbine trip. Turbine Control Valves (TCVs) have a fast closure function to protect the turbine during a generator load rejection. The Turbine Stop Valves (TSVs) close at a fast rate following a turbine trip. RPC remains unaffected and demands the Turbine Bypass Valves (TBVs) open to control reactor pressure.

Sequence of Event

The event sequence summary:

- TCVs and/or TSVs close quickly causing pressure increase
- Anticipatory scram occurs on indication of a load rejection or turbine trip
- RPC opens TBVs to control pressure
- RLC maintains level
- Controlled state achieved

Table 15.5-7 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL2 functions:

- DL2-02 – Maintain Target Level
- DL2-01 – Maintain Target Pressure
- DL2-08 – Anticipatory Hydraulic Scram on Generator Load Rejection or Turbine Trip Demand
- DL2-09 – TBV Fast Open on Generator Load Rejection or Turbine Trip Demand

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by initiating a generator load rejection or turbine trip resulting in a fast closure of the TCVs or TSVs. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The simulated generator load rejection / turbine trip is presented on Figure 15.5-10 through Figure 15.5-15, and the results are presented in Table 15.7-1. The results are shown for the case with

the limiting CPR result. Automatic reactor scram occurs following indication of a generator load rejection or turbine trip. Pressure increases but is limited by the TBVs opening.

The core thermal power does not increase above the initial power and there is no concern in approaching the centerline fuel temperature or cladding strain acceptance criteria in Section 15.3. This event is not limiting and is not considered during LHGR limits development.

The calculated ΔCPR/ICPR is provided. This is used to set an OLMCPR ensuring the CPR remains within the MCPR Acceptance Criterion. This event is potentially limiting for OLMCPR and is evaluated each operating cycle for determining the core OLMCPR. The resulting limiting event OLMCPR is included in the COLR.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

Because this event does not result in fuel failures or release of primary coolant to the environment, there are no radiological consequences associated with this event.

15.5.3.2.2 Closure of One Main Steam Reactor Isolation Valve AOO

This event is in the PI Group and is designated as a BL-AOO event. The event sequence name is Closure of one MSRIV and the Event Sequence ID is PI-1MSRIVC_BL-AOO.

Postulated Initiating Event

There are two Main Steam Lines (MSLs). The event is an inadvertent closure of one MSRIV that terminates flow in one of the MSLs. A minimum MSRIV closure time results in the most severe event.

Sequence of Event

The event sequence summary:

- One MSRIV closes causing RPV pressure and power to increase
- Anticipatory Trip System (ATS) scram occurs on MSRIV position
- RLC controls levels
- Second MSRIV in the second steam line closes on leak detection indication (this is assumed because it makes the event more severe)
- One ICS train initiates on high RPV pressure (the simulation is ended before any ICS initiation because the key mitigation DL functions have already been demonstrated and a single ICS train can control pressure and removing decay heat as demonstrated in the pressure increase DBA analysis)
- Controlled state achieved

Table 15.5-8 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL2 functions:

- DL2-02 – Maintain Target Level
- DL2-21 – Anticipatory Hydraulic Scram on MSRIV/Main Steam Containment Isolation Valve (MSCIV) Position
- DL2-31 – ICS Pressure Control on High Reactor Pressure

Core and System Performance

The event is simulated by initiating a closure of one MSRIV. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5.

Input Parameters and Initial Conditions

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The simulated closure of one MSRIV is presented on Figure 15.5-16 through Figure 15.5-21, and the results are presented in Table 15.7-1. The results are shown for the case with the limiting CPR and reactor pressure response. The neutron flux and PI resulting from the closure of one MSRIV are limited by an anticipatory scram on MSRIV position. The PI is also initially limited because the MSRIV remains open in the second steam line. The second steam line is then assumed to close on MSL break indication. This conservative assumption makes the event more severe. Pressure then increases and ICS is initiated on high RPV pressure.

The core thermal power increase is not significant and there is no concern for approaching the centerline fuel temperature or cladding strain acceptance criteria in Subsection 15.3.1.1. This event is not limiting and does not need to be considered during development of limits on the LHGR.

The calculated ΔCPR/ICPR is provided. This is used to set an OLMCPR ensuring the CPR remains within the MCPR acceptance criterion. This event is potentially limiting for OLMCPR and is evaluated each operating cycle for determining the core OLMCPR. The resulting limiting event OLMCPR is included in the COLR.

Barrier Performance

This event does not result in any temperature or pressure transient more than the derived acceptance criteria for the fuel, pressure vessel, or containment. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

Because this event does not result in fuel failures or release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.5.3.2.3 Loss of Condenser Vacuum AOO

This event is in the PI Group and is designated a BL-AOO event. The event sequence name is Loss of Condenser Vacuum (LOCV), and the Event Sequence ID is PI-LOCV_BL-AOO.

Postulated Initiating Event

There are a few potential causes of a LOCV including loss of one or more circulating water pumps. The LOCV results in a turbine trip. The TSVs close at a fast rate following a turbine trip.

Sequence of Event

The event sequence summary:

- The TSVs close and main turbine trips on low main condenser vacuum causing pressure increase
- Anticipatory scram occurs on a turbine trip
- RPC opens TBVs to control pressure
- RLC maintains level
- TBVs close on high main condenser pressure and pressure increases slowly due to decay heat (the simulation is ended before TBV closure because the key mitigation DL functions are demonstrated, and a single ICS train is capable of controlling pressure and removing decay heat)
- One ICS train initiates on high RPV pressure (the simulation is ended before any ICS initiation because the key mitigation DL functions have already been demonstrated and a single ICS train is capable of controlling pressure and removing decay heat as demonstrated in the PI DBA analysis)
- Controlled state achieved

Table 15.5-9 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL2 functions:

- DL2-02 – Maintain Target Level
- DL2-01 – Maintain Target Pressure
- DL2-13 – Turbine Trip on High Main Condenser Pressure Setpoint 2
- Anticipatory Hydraulic Scram on Either:
  - DL2-08 – Generator Load Rejection or Turbine Trip Demand
  - DL2-37– High Main Condenser Pressure Setpoint 1
- DL2-09 – TBV Fast Open on Generator Load Rejection or Turbine Trip Demand
- DL2-14 – TBV Closure on High Main Condenser Pressure Setpoint 3
- DL2-31 – ICS Pressure Control on High Reactor Pressure

Core and System Performance

Input Parameters and Initial Conditions

The LOCV results in a turbine trip. The event is simulated by initiating a turbine trip resulting in a fast TSV closure. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC and EOR cycle exposure conditions.

Results

The simulated loss of condenser vacuum is presented on Figure 15.5-22 through Figure 15.5-27, and the results are presented in Table 15.7-1. The results are shown for the case with the limiting CPR result. The pressure increases due to the fast TSV closure and is limited by the anticipatory scram on indication of a turbine trip. A scram on high main condenser pressure may occur sooner but is conservatively not modeled. The PI is also initially limited by the TBVs opening.

Condenser vacuum loss is assumed to continue, resulting in TBVs closing. Once TBVs are closed, reactor pressure increases, and one ICS train initiates on high RPV pressure.

The core thermal power does not increase above the initial power and there is no concern for approaching the centerline fuel temperature or cladding strain acceptance criteria in Section 15.3. This event is not limiting and is not considered during LHGR limits development.

The calculated ΔCPR/ICPR is provided. This is used to set an OLMCPR ensuring the CPR remains within the MCPR acceptance criterion. This event is potentially limiting for OLMCPR and is evaluated each operating cycle for determining the core OLMCPR. The resulting limiting event OLMCPR is included in the COLR.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

Because this event does not result in fuel failures or release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.5.3.2.4 Loss of Preferred Power AOO

This event is in the PI Group and is designated a BL-AOO event. The event sequence name is Loss-of-Preferred Power (LOPP), and the Event Sequence ID is PI-LOPP_BL-AOO.

Postulated Initiating Event

A LOPP is initiated by offsite power supply failure. The loss of power results in the generator output breakers opening and the TCVs fast closure.

Sequence of Event

The event sequence summary:

- LOPP occurs
- TCVs close quickly causing pressure increase
- FW pumps lose power, FW pump discharge check valves maintain coolant inventory
- Circulating water pumps lose power
- Anticipatory scram occurs on generator load rejection
- RPC opens TBVs to control pressure
- TBVs close on LOPP

- One ICS train initiates on high RPV pressure (the simulation is ended before any ICS initiation because the key mitigation DL functions have already been demonstrated and a single ICS train can control pressure and removing decay heat as demonstrated in the pressure increase DBA analysis)
- Controlled state achieved

Table 15.5-10 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL2 functions:

- DL2-01 – Maintain Target Pressure
- Anticipatory Hydraulic Scram on Either:
  - DL2-08 – Generator Load Rejection or Turbine Trip Demand
  - DL2-26 – Low Electric Bus Voltage
- DL2-09 – TBV Fast Open on Generator Load Rejection or Turbine Trip Demand
- DL2-31 – ICS Pressure Control on High Reactor Pressure
- DL2-43 – FW Check Valve Closure on Reverse FW Flow

Core and System Performance

Input Parameters and Initial Conditions

The LOPP results in the generator output breakers opening and a loss of power to the FW pumps. The event is simulated by initiating a FW trip and a load rejection resulting in fast TCVs closure. Anticipatory scram occurs on a generator load rejection. A scram on low bus voltage may occur sooner but is conservatively not modeled. This scram timing is the same as the anticipatory scram on a load rejection. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions. The results are shown for the case with the limiting result for CPR.

Results

The simulated LOPP is presented on Figure 15.5-28 through 15.5-33, and the results are presented in Table 15.7-1. The results are shown for the case with the limiting result for CPR. The PI due to TCV closure is limited by the anticipatory scram on the generator load rejection. Scram on a low electric bus voltage may occur sooner but is conservatively not credited. The PI is also initially limited by the TBVs opening. The TBVs later close on loss of power. Once the TBVs are closed, reactor PIs and ICS initiates on high RPV pressure. The ICS continues to limit the PI.

The core thermal power increase is not significant and there is no concern for approaching the centerline fuel temperature or cladding strain acceptance criteria in Subsection 15.3.1.1. This event is not limiting and does not need to be considered during development of limits on the LHGR.

The calculated ΔCPR/ICPR is provided. This is used to set an OLMCPR ensuring the CPR remains within the MCPR acceptance criterion. This event is potentially limiting for OLMCPR and is evaluated each operating cycle for determining the core OLMCPR. The resulting limiting event OLMCPR is included in the COLR.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

Because this event does not result in fuel failures or release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.5.3.3 Decrease in Reactor Coolant Inventory AOO

15.5.3.3.1 Feedwater Pump Trip – One Pump

The section analyzes the bounding BL-AOO event for the Inventory Reduction (IR) group. The event sequence name is FW Pump Trip – One Pump (FWPT) and the Event Sequence ID is IR-FWPT_BL-AOO.

Postulated Initiating Event

There is one FW pump normally operating and a second FW pump in standby. This event assumes a failure resulting in a trip of the operating FW pump. The RLC remains unaffected by the failure and increases the flow demand on the standby FW pump to maintain RPV water level.

Sequence of Event

The event sequence summary:

- One FW pump trips causing RPV water level decrease
- Standby FW pump starts and increases to rated FW flow
- Power decreases temporarily from a reduction in core flow and core inlet subcooling
- RPC controls pressure
- RLC maintains level
- RPV water low level scram and high-level FW isolation are avoided
- Controlled state achieved

Table 15.5-11 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL2 functions:

- DL2-25 – Start Standby FW pump on Loss of Operating FW pump
- DL2-02 – Maintain Target Level
- DL2-01 – Maintain Target Pressure

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by initiating a FW pump trip. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The FWPT event is presented on Figures 15.5-34 through 15.5-39, and the results are presented in Table 15.7-1 for the exposure with the limiting CPR response. Reduction in FW flow results in a reduction of vessel inventory, causing the vessel water level to drop. The standby FW pump starts on confirmed low FW flow conditions, the RPC throttles TCVs to control pressure, and the RLC increases the FW pump flow to rated conditions to maintain level. Low reactor water level scram is avoided.

The core thermal power does not increase and there is no concern in approaching the centerline fuel temperature or cladding strain acceptance criteria in Subsection 15.3.1.1. This event is not limiting and is not considered during development of LHGR limits.

The calculated ΔCPR/ICPR is provided. The event is not limiting and is not considered in the OLMCPR development.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

Because this event does not result in fuel failures or release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.5.3.4 Increase in Reactor Coolant Inventory AOO

15.5.3.4.1 Inadvertent Isolation Condenser Initiating – One Train

This event is designated as a BL-AOO event. The event sequence ID is II-IICI-1_BL-AOO. This event assumes a failure causes a single Isolation Condenser (IC) condensate return valve to open. The event assumes that RLC remains unaffected by the failure and is able to maintain level. The event also assumes that RPC remains unaffected by the failure and is able to maintain pressure.

Postulated Initiating Event

The ICs are normally in standby mode. This event assumes spurious opening of a single IC condensate return valve, resulting in the introduction of cold water into the reactor. The event assumes that RLC and RPC remain unaffected by the failure and are available to control reactor level and reactor pressure.

Sequence of Event

The event sequence summary:

- ICS condensate return valve on one train opens
- Cold ICS condensate water drains into the chimney

- RLC maintains level
- RPC maintains pressure
- Controlled state achieved

Table 15.5-12 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL2 functions:

- DL2-02 – Maintain Target Level
- DL2-01 – Maintain Target Pressure

Input Parameters and Initial Conditions

The event is simulated by opening the IC condensate return valve on one ICS train. The initial conditions are provided in Table 15.5-3 through Table 15.5-5.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The inadvertent initiation of one ICS train event is presented on Figure 15.5-40 through Figure 15.5-45. Table 15.7-1 shows the limiting results for CPR response. When the IC condensate return valve is opened, cold water is introduced into the chimney region. After an initial small perturbation, the increased density in the chimney reduces core flow, water level, and power temporarily. After the initial reduction, core flow, level, and power increase. When the increase in level is sensed, the FW controller starts to demand the operating FW pump to reduce flow. After the initial surge, as condensate water drains into the chimney and IC flow reduces, the FW controller demands the operating FW pump to increase flow. The RPV water level, core flow, and core power settle back to their initial values. RPV pressure increases insignificantly. The level and pressure remain well within the RPV water level and RCPB pressure acceptance criteria in Subsection 15.3.1.1.

The core power increase is limited. There is no concern for approaching the centerline fuel temperature or cladding strain acceptance criteria in Subsection 15.3.1.1. This event is not limiting and does not need to be considered during development of limits on the LHGR.

The calculated ΔCPR/ICPR is provided. This is used to set an OLMCPR ensuring the CPR remains within the MCPR acceptance criterion. This event is potentially limiting for OLMCPR and is evaluated each operating cycle to determine the core OLMCPR. The resulting limiting event OLMCPR is included in the COLR.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

Because this event does not result in fuel failures or release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.5.4 Analysis of Design Basis Accidents

This section evaluates the bounding non-LOCA and LOCA PIEs. Subsections 15.5.4.1 through 15.5.4.4 describe the DSA non-LOCA DBAs, while Subsection 15.5.4.5 describes the DSA LOCAs inside containment. Subsection 15.5.9.1 describes the DSA for LOCAs outside containment.

15.5.4.1 Decrease in Reactor Coolant Temperature Design Basis Accident

15.5.4.1.1 Loss of All Feedwater Heating

This event is in the TD group and is designated as a CN-DBA event. The event sequence name is CCF-LFWH, Passive Digital CCF DL2 Technology Platform (CCF-DL2), and the Event Sequence ID is TD-CCF-LFWH_CCF-DL2_CN-DBA.

Postulated Initiating Event

A CCF results in the loss of all FW heating. Any CCF that results in the loss of all FW heating occurs gradually because of the thermal inertia inherent in the FW heaters. The FW temperature lowers to the main condenser temperature with the assumed time constant shown in Table 15.5-3.

Sequence of Event

The event sequence summary:

- CCF results in the loss of all FW heating
- FW temperature decreases causing positive reactivity insertion
- RLC and RPC fail as-is
- Scram on high Simulated Thermal Power (STP) causing negative reactivity insertion
- RPV pressure decreases. The downcomer level decreases temporarily because of void collapse
- MS isolation occurs on low RPV pressure
- FW continues at initial flow causing RPV water level to increase
- FW isolation occurs on high RPV water level
- An ICS train initiates on high RPV pressure. The first ICS train fails to actuate (assumed single failure). One of the two remaining ICS trains sufficiently controls pressure (the simulation is ended before any ICS initiation because the key mitigation DL functions have already been demonstrated and a single ICS train can control pressure and removing decay heat as demonstrated in the pressure increase DBA analysis).
- Controlled state achieved

Table 15.5-13 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL3 functions:

- DL3-05 – Hydraulic Scram on High Simulated Thermal Power
- DL3-23 – FW Isolation on High RPV Water Level

- DL3-17 – MSRIV/MSCIV Isolation on Low RPV Pressure
- DL3-12 – ICS Train 2 Initiation on High RPV Pressure

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by decreasing the FW temperature. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The simulated loss of all FW heating event is presented on Figure 15.5-46 through Figure 15.5-51, and the results are presented in Table 15.7-2 for the exposure with the limiting PCT response. The reduced FW temperature enters the core and causes an increase in core inlet subcooling. This increases core power due to the negative void reactivity coefficient, and a scram occurs on high STP. MSRIV isolates on low RPV pressure.

FW flow remains at 100% due to the RLC failure, and RPV water level rises until FW isolates on high RPV water level. Decay heat causes RPV pressure to rise and an ICS train initiates. Only one ICS train is needed to prevent RPV pressure increase and maintain long-term cooling. A single failure of an ICS train starting on high RPV pressure does not affect event mitigation.

The long-term core cooling capability is assured by meeting the acceptance criteria for fuel cladding and the RCPB. The reactor integrity is assured by meeting the pressure criteria provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature at which significant oxidation occurs due to metal water reaction.

Because there is significant margin to the acceptance criteria, this event is considered to have large margin, and no additional conservatism is applied.

Barrier Performance

This event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

Because this event does not result in fuel failures or release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.5.4.2 Increase in Reactor Pressure Design Basis Accidents

15.5.4.2.1 Generator Load Rejection or Turbine Trip

This event is in the PI group and is designated a CN-DBA event. The event sequence name is Generator Load Rejection or Turbine Trip (LR-TT), Passive Digital CCF DL2 Technology Platform (CCF-DL2) and the Event Sequence ID is PI-LR-TT_CCF-DL2_CN-DBA.

Postulated Initiating Event

The PIE is the same as the BL-AOO event. The event sequence assumes a passive CCF of the DL2 functions. The CCF results in RPC and RLC failing as-is, which are continually operating, and failure of the anticipatory scram.

Sequence of Event

The event sequence summary:

- TCVs and/or TSVs close quickly causing pressure and power increase
- RLC fails as-is at initial condition, the TBVs remain closed, and anticipatory scram fails
- Scram occurs on high neutron flux
- After scram, no immediate challenge to cladding and RCPB integrity
- RPV pressure continues to increase because RPC fails as-is
- RPV water level reduces due to the PI
- One ICS train initiates on high RPV pressure. First ICS train fails to actuate (assumed single failure). One of the two remaining trains is sufficient to control pressure.
- With RLC failing as-is, initial FW flow continues causing RPV water level to increase
- FW isolates on high RPV water level
- Controlled state achieved

Table 15.5-14 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL3 functions:

- DL3-04 – Hydraulic Scram on High neutron flux
- DL3-23 – FW Isolation on High RPV Water Level
- DL3-12 – ICS Train 2 Initiation on High RPV Pressure

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by initiating a fast closure of the TCVs or TSVs. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5. This event is limiting for PCT. Therefore, additional initial conditions and phenomena that affect PCT described in NEDC-34043P are applied in the limiting direction by at least one standard deviation consistent with the approach for an event with medium margin:

- Core void coefficient
- Channel interfacial shear
- Chimney interfacial shear
- Separator steam carry under

- Critical quality used in boiling transition correlation
- Channel radial peaking factor
- Hot rod power
- Total initial power

Analysis is performed as needed to confirm the conservative PCT direction. Then a bounding case is created with all inputs in the conservative direction.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The simulated generator load rejection / turbine trip is presented on Figure 15.5-52 through Figure 15.5-57, and the results are presented in Table 15.7-2. The results are presented for the bounding case described above. Reactor scram occurs following high neutron flux. The neutron flux increases rapidly because of the void reduction caused by the PI. TBVs fail to open; however, ICS limits the PI. Only one ICS train is needed to prevent PI and maintain long-term cooling. A single failure of an ICS train to start on high RPV pressure does not affect the event mitigation.

The long-term core cooling capability is assured by meeting acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the pressure criteria provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains below the temperature at which significant oxidation occurs due to metal water reaction.

Because this event is considered to have medium margin to acceptance criteria (PCT), additional conservatisms are applied as described above to create a bounding case. The results from applying these conservatisms have little effect on peak pressure, because peak pressure is primarily driven by the ICS initiation setpoint. Once the pressure setpoint is reached, ICS initiates, and pressure rapidly reduces. For PCT, these uncertainties result in a PCT 91°C (163°F) higher than the base case with conservatisms similar to other DBA analyses. The results are within the acceptance criteria.

Barrier Performance

The effect of this event does not result in any challenge to the temperature or pressure transient derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

Because this event does not result in fuel failures or release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.5.4.2.2 Loss of Preferred Power

This event is in the PI group and is designated a CN-DBA event. The event sequence name is LOPP, and the Event Sequence ID is PI-LOPP_CCF-DL2_CN-DBA.

Postulated Initiating Event

The PIE is the same as the BL-AOO event. The event sequence assumes a passive CCF of the DL2 functions. The CCF results in RPC and RLC failing as-is at the initial condition, which are continually operating, and failure of the anticipatory scram.

Sequence of Event

The event sequence summary:

- TCV closes slowly due to loss of turbine control hydraulic pumps
- FW pumps lose power and coast down
- RLC fails as-is at the initial condition, TBVs remain closed, and the anticipatory scram fails
- Scram occurs on high neutron flux
- After scram, no immediate challenge to cladding and RCPB integrity
- RPV pressure continues to increase because TBVs remain closed
- An ICS train initiates on high RPV pressure. The first ICS train fails to actuate (assumed single failure). One of the remaining two trains is sufficient to control pressure.
- Controlled state achieved

Table 15.5-15 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL3 functions:

- DL3-04 – Hydraulic Scram on High Neutron Flux
- DL3-12 – ICS Train 2 Initiation on High RPV Pressure
- DL3-39 – FW Isolation on Loss of Normal FW Flow

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by initiating a slow closure of the TCVs and FWPT. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The simulated LOPP is presented on Figure 15.5-58 through Figure 15.5-63, and the results are presented in Table 15.7-2. The results are shown for the case with the limiting result for cladding temperature and reactor pressure response. The neutron flux increases rapidly because of the void reduction caused by the PI. Reactor scram occurs following high neutron flux. TBVs fail to open; however, the PI is limited by the ICS initiation. Only one ICS train is needed to prevent PI and maintain long-term cooling. Therefore, a single failure of an ICS train to start on high RPV pressure does not affect the event mitigation. The closure of the TCVs in the LOPP AOO is due to DL2 active mitigation. In the LOPP CN-DBA sequence, the TCVs close because of the PIE. If there is no power to maintain hydraulic pressure, the valves slowly close.

The long-term core cooling capability is assured by meeting the acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the pressure criteria provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding

temperature remains well below the temperature at which significant oxidation occurs due to metal water reaction.

Because there is significant margin to the acceptance criteria, this event is considered to have large margin, and no additional conservatism is applied.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient challenge to the derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

Because this event does not result in fuel failures or release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.5.4.2.3 RPV Pressure Control Downscale

This event is in the PI group and is designated a CN-DBA event. The event sequence name is CCF – RPV Pressure Control Downscale (CCF-RPCD), Passive Digital CCF DL2 Technology Platform (CCF-DL2) and the Event Sequence ID is PI-CCF-RPCD_CCF-DL2_CN-DBA.

Postulated Initiating Event

The PIE is a spurious CCF of the RPV pressure control. This failure results in a demand to close the TCVs (normal servo closure). This PIE also prevents the TBVs from opening. The event sequence assumes a passive CCF of the DL2 functions. The CCF results in the RLC failing as-is at initial conditions, and failure of the anticipatory scram.

Sequence of Event

The event sequence summary:

- RPC demands TCVs to slow close and the TBVs remain closed
- RLC fails as-is at the initial condition and the anticipatory scram fails
- Scram occurs on high neutron flux
- After scram, no immediate challenge to cladding and RCPB integrity
- With RLC failing as-is at the initial condition, the initial FW flow continues causing RPV water level to increase
- FW isolates on high RPV water level
- One ICS train initiates on high RPV pressure. The first ICS train fails to actuate (assumed single failure). One of the two remaining ICS trains is sufficient to control pressure.
- Controlled state achieved

Table 15.5-16 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL3 functions:

- DL3-04 – Hydraulic Scram on High Neutron Flux
- DL3-23 – FW Isolation on High RPV Water Level
- DL3-12 – ICS Train 2 Initiation on High RPV Pressure

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by initiating a slow closure of the TCVs. The initial conditions and plant parameters are provided in Table 15.5-3 through Table 15.5-5. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The simulated RPV Pressure Control Downscale is presented on Figure 15.5-64 through Figure 15.5-69, and the results are presented in Table 15.7-2. The results are shown with the limiting result for cladding temperature and reactor pressure response. The neutron flux increases rapidly because of the void reduction caused by the PI. Reactor scram occurs following high neutron flux. TBVs fail to open, and FW pump trips on high RPV water level.

Natural circulation continues at a rate consistent with decay heat power. Only one ICS train is needed to prevent PI and maintain long-term cooling. Therefore, a single failure of an ICS train to start on high RPV pressure does not affect the event mitigation.

The long-term core cooling capability is assured by meeting the acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the pressure criteria provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature at which significant oxidation occurs due to metal water reaction.

Because there is significant margin to the acceptance criteria, this event is considered to have large margin, and no additional conservatism is applied.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

Because this event does not result in fuel failures or release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.5.4.2.4 Closure of All Main Steam Reactor Isolation Valves and FW Isolation Valves

This event is in the PI group and is designated a CN-DBA event. The event sequence name is CCF- Closure of All MSRIVs and FW isolation valves (CCF-DL4a-MSRIVC-FWIV) and the Event Sequence ID is PI-CCF-DL4a-MSRIVC-FWIV_CN-DBA.

Postulated Initiating Event

The PIE is a spurious CCF that affects all MSRIVs and FW isolation valves.

Sequence of Event

The event sequence summary:

- Closure of all MSRIVs and FW isolation valves
- Scram occurs on high neutron flux
- After scram, no immediate challenge to cladding and RCPB integrity
- An ICS train initiates on high RPV pressure. The first ICS train fails to actuate (assumed single failure). One of the two remaining trains is sufficient to control pressure.
- Controlled state achieved

Table 15.5-17 lists the event sequence:

Identification of Operator Actions

- No operator action is required to mitigate the event.

Systems Operation

Credited DL3 functions:

- DL3-04 – Hydraulic Scram on High Neutron Flux
- DL3-12 – ICS Train 2 initiation on High RPV Pressure

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by initiating a closure of the MSRIVs and FW isolation valves. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The simulated closure of MSRIVs and FW isolation valves is presented on Figure 15.5-70 through Figure 15.5-75, and the results are presented in Table 15.7-2. The results are shown for the limiting case for cladding temperature and reactor pressure response. Reactor scram occurs following high neutron flux. The PI is limited by ICS initiation. Only one ICS train is needed to prevent PI and maintain long-term cooling. A single failure of an ICS train to start on high RPV pressure does not affect event mitigation.

The long-term core cooling capability is assured by meeting the acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the pressure criteria provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature at which significant oxidation occurs due to metal water reaction.

Because there is significant margin to the acceptance criteria, this event is considered to have large margin, and no additional conservatism is applied.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient that challenges the derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

Because this event does not result in fuel failures or release of primary coolant to the environment, there are no radiological consequences associated with this event.

15.5.4.3 Increase in Reactor Coolant Inventory DBAs

This event group is in the Inventory Increase (II) group.

15.5.4.3.1 Feedwater Flow Increase – All Pumps

This event is designated as a CN-DBA event. The event sequence name is Feedwater Flow Increase – All Pumps (CCF-FWFI), Passive Digital CCF DL2 Technology Platform (CCF-DL2) and the Event Sequence ID is II-CCF_FWFI_CCF-DL2_CN-DBA.

Postulated Initiating Event

One FW pump is normally operating and a second FW pump in standby. The RLC adjusts the pump speed to adjust FW flow to maintain RPV water level. This event assumes a spurious CCF that causes both FW pumps to increase flow to maximum speed that results in the maximum FW flow. Although not possible, the increase in flow is assumed to occur instantaneously.

Sequence of Event

The event sequence summary:

- Both FW pumps increase to maximum flow causing RPV water level increase
- RPC remains as-is at initial condition
- Level, pressure, and power increase
- Automatic FW isolation on high RPV water level
- Scram on high simulated thermal power
- RPV pressure and level decrease
- MS isolation on low RPV pressure
- An ICS train initiates on high RPV pressure. The first ICS train fails to actuate (assumed single failure). One of the two remaining trains is sufficient to control pressure and remove decay heat as demonstrated in the pressure increase DBA analysis (the simulation is ended before any ICS initiation because the key mitigation DL functions have already been demonstrated).
- Controlled state achieved

Table 15.5-18 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL3 functions:

- DL3-05 – Hydraulic Scram on High Simulated Thermal Power
- DL3-23 – FW Isolation on High RPV Water Level
- DL3-17 – MSRIV/MSCIV Isolation on Low RPV Pressure
- DL3-12 – ICS Train 2 Initiation on High RPV Pressure

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by increasing both FW pumps flow of to the maximum speed, resulting in the maximum FW flow. The initial conditions and total maximum flow for both FW pumps is provided in Tables 15.5-3 through 15.5-5. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The simulated maximum FW pumps flow event is presented on Figure 15.5-76 through Figure 15.5-81, and the results are presented in Table 15.7-2. The results are shown for the case with the limiting PCT response results. The increase in FW flow causes reactor level, pressure, and power to increase as RPC and RLC are unavailable. FW isolation occurs on high reactor level. Scram occurs on high STP. Pressure decreases until MSRIV isolation initiation occurs on low RPV pressure. Pressure then increases and one ICS train is initiated on high RPV pressure. Only one ICS train is needed to prevent RPV pressure from increasing and maintaining long-term cooling. A single failure of an ICS train to start on high RPV pressure does not affect the event mitigation.

The long-term core cooling capability is assured by meeting the acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the pressure criteria provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature at which significant oxidation occurs due to metal water reaction.

Because there is significant margin to the acceptance criteria, this event is considered to have large margin, and no additional conservatism is applied.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the design criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

Because this event does not result in fuel failures or release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.5.4.3.2 Inadvertent Isolation Condenser Initiation – All Trains

This event is designated as a CN-DBA event. The event sequence name is Inadvertent Isolation Condenser Initiation – All Trains (CCF-DL4a-IICI), Passive Digital CCF DL2 Technology Platform (CCF-DL2) and the Event Sequence ID is II-CCF-IICI_CCF-DL2_CN-DBA.

Postulated Initiating Event

The ICS is normally in standby mode. This event assumes a spurious CCF that causes all IC condensate return valves to open, resulting in introducing cold water into the reactor.

Sequence of Event

The event sequence event summary:

- All IC condensate return valves open
- RLC and RPC fail as-is at the initial condition
- Cold ICS condensate water drains into the chimney
- Core flow, reactor pressure, and power decrease
- RPV water level increases due to RLC failing as-is
- FW isolation occurs on high RPV water level
- Scram occurs on low RPV pressure
- MS isolation occurs on low RPV pressure
- Controlled state achieved

Table 15.5-19 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL3 functions:

- DL3-23 – FW Isolation on High RPV Water Level
- DL3-02 – Hydraulic Scram on Low RPV Pressure
- DL3-17 – MSRIV/MSCIV Isolation on Low RPV Pressure

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by opening all IC condensate return valves in one second. The initial conditions are provided in Table 15.5-3 through Table 15.5-5 The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The inadvertent initiation of all ICS trains event is presented on Figure 15.5-82 through Figure 15.5-87, and the results are presented in Table 15.7-2. The results are shown for the case with the limiting PCT response results. When the IC condensate return valves are opened, cold water

is introduced into the chimney region, reducing reactor power, reactor pressure, and core flow. Reactor scram and MSRIV isolation initiation occurs on low RPV pressure. After an initial reduction in reactor level, level begins to rise as RLC is unavailable. FW isolation occurs on high reactor level.

The long-term core cooling capability is assured by meeting the acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the RCPB pressure criteria. The acceptance criteria are provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature where significant oxidation occurs due to metal water reaction.

Sensitivity studies were performed at the ICS minimum initial temperature in Table 15.5-3 and for an assumed increase in ICS return line volume of 50%. There was no significant change in the event sequence or results. Because there is significant margin to the acceptance criteria, this event is considered to have large margin, and no additional conservatism is applied.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

Because this event does not result in fuel failures or release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.5.4.4 Decrease in Reactor Coolant Inventory Design Basis Accidents

This event group is in the IR group.

15.5.4.4.1 Loss of Feedwater Flow

This event is designated as a CN-DBA event. The event sequence name is CCF Loss of FW Flow (CCF-LOFW), Passive Digital CCF DL2 Technology Platform (CCF-DL2) and the Event Sequence ID is IR-CCF-LOFW _CCF-DL2_CN-DBA.

Postulated Initiating Event

The event sequence assumes a spurious CCF causes the loss of all FW flow and a passive CCF of the DL2 function results in a fails as-is RPC.

Sequence of Events

The event sequence summary:

- Loss of FW flow causes RPV water level and power to decrease
- RPV pressure decreases due to frozen RPC
- Scram occurs on low RPV water level
- FW isolation on a loss of normal FW flow indication
- MS isolation on low RPV pressure
- All ICS trains initiate on low RPV water level
- Controlled state achieved

Table 15.5-20 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL3 functions:

- DL3-03 – Hydraulic Scram on Low RPV Level
- DL3-17 – MSRIV/MSCIV Isolation on Low RPV Pressure
- DL3-14 – ICS Initiation on Low RPV Water Level
- DL3-39 – FW Isolation on Loss of Normal FW Flow

Core and System Performance

Input Parameters and Initial Conditions

The event is conservatively simulated by initiating a trip of all FW pumps. The initial conditions and plant parameters are provided in Table 15.5-3 through Table 15.5-5. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The loss of FW flow is presented on Figure 15.5-88 through Figure 15.5-93, and the results are presented in Table 15.7-2 for the exposure with the limiting PCT response. The trip of all FW pumps results in a reduction of vessel inventory, causing the pressure and vessel water level to drop. Reactor scram occurs on low RPV water level. FW isolates on sustained loss of normal FW flow; while FW isolation is not directly simulated, the analysis indirectly credits FW isolation as no inventory losses are assumed to occur due to reverse leakage through the FW check valves.

Leakage via the FW check valves is not specifically modeled for the 30 seconds leading to FW isolation. During this time significant inventory is being transferred from the RPV to the main condenser via the steam lines. This flow from the steam lines is much larger than the expected leakage from the FW check valves, and the additional inventory loss via FW check valve leakage for 30 seconds would have an insignificant effect on the reactor water level, which has significant margin to top of active fuel, as shown on Figure 15.5-90. The MSRIVs close on low RPV pressure. RPV water level continues to decrease until ICS initiates. Three ICS trains are modeled to open. Only one ICS train is needed to prevent RPV pressure increase and maintain long-term cooling. A single failure of an ICS train to start on low RPV water level will not affect event mitigation.

The long-term core cooling capability is assured by meeting the acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the RCPB pressure criteria. The acceptance criteria are provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature where significant oxidation occurs from metal water reaction.

Because there is significant margin to the acceptance criteria, this event is considered to have large margin, and no additional conservatism is applied.

Barrier Performance

This event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because

there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

Because this event does not result in fuel failures or release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.5.4.4.2 Reactor Pressure Vessel Pressure Control Open

This event is designated as a CN-DBA event. The event sequence name is RPV Pressure Control Open (CCF-RPCO), Passive Digital CCF DL2 Technology Platform (CCF-DL2) and the Event Sequence ID is IR-CCF-RPCO_CCF-DL2_CN-DBA.

Postulated Initiating Event

The event assumes all TCVs and TBVs are fully opened by a spurious RPC CCF. The event sequence assumes a passive CCF of the DL2 functions resulting in the RLC failing as-is.

Sequence of Events

The event sequence summary:

- All TCVs and TBVs open causing RPV pressure and power to decrease
- FW flow remains at 100% due to DL2 CCF
- Reactor scram and MS isolation on low RPV pressure
- FW isolates on high RPV water level
- An ICS train initiates on high RPV pressure. The first ICS train fails to actuate (assumed single failure). One of the two remaining trains is sufficient to control pressure and removes decay heat as demonstrated in the PI DBA analysis (the simulation is ended before any ICS initiation because the key mitigation DL functions have already been demonstrated).
- Controlled state achieved

Table 15.5-21 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL3 functions:

- DL3-02 – Hydraulic Scram on Low RPV Pressure
- DL3-17 – MSRIV/MSCIV Isolation on Low RPV Pressure
- DL3-23 – FW Isolation on High RPV Water Level
- DL3-12 – ICS Train 2 Initiation on High RPV Pressure

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by fully opening all TCVs and TBVs. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The simulated RPV Pressure Control Open event is presented on Figure 15.5-94 through Figure 15.5-99, and the results are presented in Table 15.7-2 for the exposure with the limiting PCT response. The opening of the TCVs and TBVs results in a decrease in reactor pressure causing voids to increase and power to decrease. FW remains at 100% rated flow due to the RLC CCF. The reactor scrams and MS isolates on low RPV pressure. RPV water level rises until FW isolates on high RPV water level. RPV pressure then rises due to decay heat. An ICS train initiates on high RPV pressure after the first ICS train fails to actuate (assumed single failure).

One ICS train sufficiently controls pressure.

The long-term core cooling capability is assured by meeting the acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the RCPB pressure criteria. The acceptance criteria are provided in Table 15.3-2. The results demonstrate significant margin to the acceptance criteria. The cladding temperature remains well below the temperature where significant oxidation occurs due to metal water reaction.

Because there is significant margin to the acceptance criteria, this event is considered to have large margin, and no additional conservatism is applied.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

Because this event does not result in fuel failures or release of primary coolant to the environment, there are no radiological consequence associated with this event.

15.5.4.5 Loss of Coolant Accidents Design Basis Accidents

Discussion of design features to control radiolytic gas concentration will be provided in CPA Part 2.

The scenarios for LOCA developed in Subsection 15.2.4.6 bound the CN-DBA sequences, demonstrating the fuel and containment integrity acceptance criteria are met for at least 72 hours using only passive heat removal systems.

The LOCA method used in containment analyses is described in NEDC-33922P-A that is incorporated by reference. The BWRX-300 LOCA EM is described in Subsection 15.5.1.2.1. The initial conditions and the modeling parameters are biased to account for uncertainties. The defense lines credited in the conservative LOCA breaks inside containment are identified in Table 15.5-44. The LOCA DBA safety analyses contained herein demonstrate the following NEDC-33922P-A Limitations and Conditions (L&Cs) have been met:

1. L&C #1 – Isolation Condenser Radiolytic Gas Removal – *The use of this Containment Evaluation Method (CEM) is limited to a BWRX-300 design that limits the total volumetric fraction of radiolytic gases in the IC lower drum to a sufficiently low level throughout a 72-hour period following the event such that condensation heat transfer in the ICs is not adversely affected, and the hydrogen deflagration margin is maintained*. The analyses assume that the isolation condensers are equipped with features to maintain the radiolytic

gases in the lower drum sufficiently low to prevent deflagration of combustible gases and degradation of the IC heat removal rate. The design feature to control radiolytic gas concentration will be provided in the FSAR.

2. L&C #2 – Isolation Condenser Return Line Design and Further Demonstration of Transient Reactor Analysis Code General Electric Modeling Capability – *The use of this CEM is limited to a BWRX-300 design that a proper isolation condenser return line layout is chosen, such as a loop seal or a water trap, to prevents reverse flow from RPV into the IC return line throughout a 72-hour period following the event or where an applicant or licensee referencing this report demonstrates that the TRACG code is capable of conservatively modeling the overall ICs heat removal capacity when reverse flow occurs in the IC discharge lines*. A loop seal has been added to the design to prevent reverse flow from the RPV into the ICs through the condensate return line.
3. L&C #3 – Demonstration of No Safety-Significant Break Flow Reversal During the First 72-Hours into the Event – *The use of this CEM is limited to a BWRX-300 design in which the PCCS is sized sufficiently large such that a reverse flow from containment back to RPV does not occur during the first 72-hours into the event. The applicant or licensee referencing this report needs to demonstrate that no reverse flow could occur, or any reverse flow that occurs under the most bounding flow reversal conditions resulting in the degradation of ICs heat transfer is not safety-significant with respect to the acceptance criteria for the BWRX-300 CEM*. Reverse flow from containment to the RPV will not occur as demonstrated in the small break analyses presented in Subsection 15.5.4.5.4 and shown on Figure 15.5-115.
4. L&C #4 – Demonstration of the Applicability of the BWRX-300 Containment Evaluation Method to the Final Passive Containment Cooling System Licensing-Basis Analysis – *The use of this CEM was demonstrated for a BWRX-300 design with the specific concentric pipe PCCS design configuration and placement described in this LTR. For any alternate PCCS design configuration and placement, the applicability of this method and the PCCS modeling approach must be reviewed and found to be acceptable by the NRC for BWRX-300 licensing-basis analyses.* The updated PCCS design discussed in Subsection 6.2.2 assumes two of the three PCCS units are available. Although the PCCS design is different than the conceptual design provided in the NEDC-33922P-A, Revision 3, the flow, and heat transfer phenomena in the current design are similar and less complicated than that provided in the LTR. The key modeling element is the condensation heat transfer on the outside of the PCCS tubes. The tube height, vertical placement, and tube diameter are comparable to the design in NEDC-33922P-A Revision 3, and therefore, the condensation heat transfer correlations and the biases applied to it for conservatism are equally applicable to the updated PCCS design. Heat transfer inside the PCCS tubes is a simpler process in the updated design because the heat transfer across the inner wall is eliminated. Further, intake and discharge locations are farther apart in the pool eliminating the concerns for a short-circuit. Any uncertainties in the phenomena important to the calculation of the thermosyphon effect are either reduced or remain the same. Therefore, the method described in NEDC-33922P-A Revision 3 is applicable to the current PCCS design including the biases. The analyses presented in this section were performed using the updated PCCS design and confirm the ability of the PCCS to remove heat from containment.

The design of the PCCS and the ICS is discussed in Subsection 6.2.2 and Section 6.3, respectively.

The acceptance criteria for LOCAs are shown in Table 15.3-2. The collapsed downcomer level is used in the BWRX-300 LOCA acceptance criteria because if the collapsed downcomer level is

above TAF, the two-phase level inside the shroud must also be above TAF. Fuel may still be wetted even when the collapsed downcomer level falls below TAF. If the downcomer level falls below TAF, the BWRX-300 LOCA acceptance criteria assesses the effect on fuel cladding temperature. If the downcomer level falls below TAF, fuel cladding temperature is maintained within normal operating temperature range when the fuel cladding temperature remains less than the initial cladding temperature for a given event.

As discussed in Subsection 15.2.4.6, all large breaks are isolated rapidly (10 seconds). Therefore, RPV inventory loss does not threaten fuel integrity in a large break LOCA. After RPV isolation, decay heat is removed by the ICs directly from the RPV. The limiting parameters for large break LOCA events are containment pressure and temperature. Containment peak pressure reaches its peak value at approximately the time of RPV isolation.

The LOCA analyses demonstrate the core remains covered or fuel cladding temperature remains below the normal operating temperature for at least 72 hours using conservative assumptions for unisolated small break LOCAs. Therefore, fuel cladding temperature remains well below the fuel acceptance criteria, oxidation does not occur, and there is no hydrogen generation from cladding oxidation.

15.5.4.5.1 Main Steam Pipe Break Inside Containment, Conservative Case

Postulated Initiating Event

A break in the MS pipe occurs at an arbitrary location between the outside RIV and the containment penetration. The most limiting break is the double-ended instantaneous guillotine break of the MS pipe.

The break flow occurs from both ends of the break. Break flow from the RPV side of the break is choked at the flow limiter inside the main steam RPV nozzle. The TSVs and TCVs are conservatively assumed to close rapidly because this results in retaining more energy in containment. With this assumption, steam flows from RPV to the steam line header upstream of the TSV/TCV through the intact steam pipe and to the break location in reverse direction to the normal flow through the broken steam pipe. Steam flow from the RPV is choked at the flow limiters in both the broken and intact steam pipes. Steam inventory in the steam pipe also discharges into the containment without being restricted at the flow limiters. Containment Isolation Valve (CIV) closure is conservatively not credited, and the entire steam line volume inventory contributes to containment pressurization.

Sequence of Event

The bounding scenario analyzed:

- Double-ended guillotine break of MSL inside containment concurrent with LOPP
- FWPT and coast down
- TSVs and TCVs close rapidly
- Scram initiated from high containment pressure
- Control rods start to insert
- ICS condensate return valve starts opening
- CUW stops
- Control rods inserted sufficiently to diminish fission from prompt neutrons
- RIVs fully close

- Condensate return valve for one ICS train is fully open
- Peak containment pressure is reached
- Containment pressure starts decreasing

Table 15.5-22 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

As shown on Figure 15.5-102, break flow from the turbine side is generally higher than the break flow from the RPV side of the break because of the pipe inventory, and the break flow continues even after the RIVs are fully closed at 10 seconds.

FW pumps are assumed to trip concurrent with the break initiation because the bounding scenario assumes LOPP concurrent with the pipe break. One IC is started when the high containment pressure setpoint is reached, which occurs within 1 second. Although two ICs are available, only one IC is credited to bound the large break case in the isolation condenser steam supply pipe. Figure 15.5-100 shows the heat removal rate of one IC exceeds the power generation decay heat after 20 seconds. As a result, reactor pressure decreases rapidly even after the break isolation shown on Figure 15.5-101. RPV water level, labeled as “Collapsed Downcomer Level” on Figure 15.5-103 stabilizes well above the TAF in 3 hours. The decrease in the downcomer level during the first 3 hours is due to the gradual decrease in the void fraction in the core and chimney, not due to RPV water inventory losses. There is no RPV water inventory loss after the RIVs close. Fuel never heats up because the core remains covered throughout the event. After the fission power is diminished, fuel cladding temperature remains near saturation temperature.

Figure 15.5-104 shows the containment pressure in response to a large MS pipe break inside containment. The break location is assumed away from the containment shell at the lowest MS pipe elevation and directed upwards as discussed in NEDC-33922P-A that is incorporated by reference. This configuration maximizes containment pressure. The break location is assumed next to the containment shell and directed towards the containment shell in calculating the shell temperature on Figure 15.5-105 as discussed in NEDC-33922P-A. This maximizes shell temperature.

The initial containment pressure includes a bias to account for uncertainties and is assumed at the containment high pressure setpoint for scram, reactor isolation and IC initiation. Although it appears that the setpoint is reached as soon as the break occurs, this is an artifact of the conservative initial condition assumption. A finite amount of time would have to elapse for containment pressure to reach the setpoint if the containment initial pressure is at the nominal pressure in normal operation. The pressure trend on Figure 15.5-104 shows that the containment pressure increases by 18.5 kPa in less than 1 second, indicating that the containment high pressure setpoint is reached in less than 1 second when the initial containment pressure is at the nominal pressure in normal operation. This confirms the break flow calculation assumption that containment high pressure setpoint will be reached in less than 1 second.

The initial reactor power in the conservative case calculations is 102% of the rated power to account for the power uncertainty. However, hot shutdown conditions may be more limiting for the mass release from the break because the initial RPV void fraction is lower, resulting in higher liquid carryover to the break location. This also causes the break flow enthalpy to be lower. Both the rated initial conditions and hot shutdown initial conditions are analyzed. Figure 15.5-104 shows that the rated initial condition is more limiting for containment peak pressure.

PCCS does not require actuation, it is always in service and rejects heat to the equipment pool that is connected to the reactor cavity pool during normal operation. The calculations assume there is no heat loss from the containment shell to the concrete as discussed in the containment evaluation method NEDC-33922P-A that is incorporated by reference. Heat removal from containment atmosphere is by the containment shell heating up, by PCCS and through the containment dome to the pool. After RIV closure for large breaks, the only energy addition to the containment is due to the heat transfer from the RPV and hot piping walls. Because only one IC is sufficient to depressurize the RPV rapidly, heat load to containment also becomes small in the long term.

Core and Barrier Performance

Results

Key results are summarized in Table 15.7-3. The plots for RPV parameters and containment parameters are shown on Figure 15.5-100 through Figure 15.5-105. The peak pressure is less than the design pressure with more than 20% margin. Containment shell temperature is also well below the containment shell design temperature of 166°C. Peak accident pressure is approximately 322 kPaG (423 kPa) and half of the peak pressure is approximately 161 kPaG (262 kPa). As shown on Figure 15.5-104, containment is depressurized rapidly, and containment pressure is reduced to 185 kPa at 6-hours. This meets the acceptance criterion for the containment response to pipe breaks that the containment pressure should be reduced to less than half of the peak pressure in 24 hours.

Barrier Performance

There is no fuel damage as a result of an MSLB inside containment. The only activity available for release is normal reactor coolant concentration in the vessel and piping prior to the break.

Radiological Consequences

The radiological consequences for a MSLB inside containment are bounded by the consequences for MSLB outside containment presented in Subsection 15.5.9.1.1.

15.5.4.5.2 Feedwater Pipe Break Inside Containment, Conservative Case

Postulated Initiating Event

A double-ended guillotine break occurs in the larger diameter segment of one FW pipe. This is more limiting than a break occurring in the smaller diameter FW pipe segments closer to the RPV. The bounding scenario is described in Subsection 15.2.4.6.

Sequence of Event

The bounding scenario analyzed:

- Double-ended guillotine break of the FW pipe inside containment concurrent with LOPP
- FWPT and coast down
- TSVs and TCVs close rapidly
- Scram initiated from high containment pressure
- Control rods start inserting on scram initiation
- Condensate return valves on two ICS trains start opening
- Control rods are inserted sufficiently to diminish fission from prompt neutrons
- FWRIVs and CIVs are fully closed

- Peak containment pressure is reached
- IC valves are fully open

Table 15.5-23 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

System Operation

RPV and containment response to a large FW break is similar to the RPV and containment response to a large MS pipe break discussed in Subsection 15.5.4.5.1. Containment pressure reaches the containment high pressure setpoint for scram, reactor and containment isolation, and IC initiation in less than 1 second. Power generated by prompt fission is diminished in 3 seconds after the break. Condensate return valves in two of the three ICS trains start opening in 1 second and fully open in 11 seconds. As shown on Figure 15.5-161, heat removal rate is much larger than the decay heat. As a result (shown on Figure 15.5-162), RPV pressure decreases much faster than the MS pipe break case. Reactor water level is shown on Figure 15.5-163. The indicated water level stabilizes above the actual collapsed downcomer level. This is because the wide range level is off scale when the actual level falls below the lower tap and no longer indicates level. The actual collapsed downcomer level stabilizes well above TAF.

Break flow from the pump side decreases initially because the break location is far away from the pump and the enthalpy becomes saturated locally right after the break although the pump is still coasting down as shown on Figure 15.5-164. The break flow from the pump side recovers and exceeds that of the RPV break side. This is because the pipe water inventory and the pump coasting down. The break flow becomes zero when the CIV is closed at 10 seconds. Enthalpy value after this point is not meaningful because there is no break flow.

As in the MS pipe break cases, containment pressure on Figure 15.5-165 and temperature on Figure 15.5-166 are calculated for a break location maximizing pressure and temperature.

An additional FW pipe break case was included accounting for the lower initial FW temperature because the containment peak pressure may be higher if the FW pipe break occurs when the plant is operating at reduced FW temperature. Break flow rate is higher at higher subcooling. However, break flow enthalpy is also lower. Figure 15.5-165 shows the containment pressure for normal FW temperature and reduced FW temperature. Normal FW temperature results in a higher containment pressure. The peak pressure for both cases is bounded by the peak pressure resulting from a MS pipe break.

Core and Barrier Performance

Results

Plots for RPV and containment parameters are shown on Figures 15.5-161 through 15.5-166. Key results are summarized in Table 15.7-3 and show that the peak containment pressure and temperature resulting from FW pipe breaks are bounded by the MS pipe breaks. Containment pressure and temperatures are less limiting than the MS pipe cases and meet the acceptance criteria. Containment pressure calculated for FW pipe break at 6 hours is also less than half the peak containment pressure calculated for the MS pipe case and decreasing.

Barrier Performance

There is no fuel damage as a result of a Feedwater Line Break (FWLB) inside containment. The only activity available for release is normal reactor coolant concentration in the pipe prior to the break.

Radiological Consequences

The radiological consequences for a FWLB inside containment are bounded by the consequences for FWLB outside containment presented in Subsection 15.5.9.1.2.

15.5.4.5.3 Large Isolation Condenser Pipe Breaks Inside Containment

An ICS break larger than the area of a 19 mm equivalent diameter pipe is detected by the leakage detection for each ICS train separately. When a break is detected in one ICS train, both RIVs in the steam supply pipe and the condensate return pipe of the affected ICS train are closed. The stroke time and delay time assumed for the IC isolation valves in the analysis are the same as those for all other RIVs and bound all other equipment initiation delays starting from the time of the pipe break. The other two unaffected ICs are available to remove decay heat. For conservatism, the analysis assumes only one of the two remaining ICs is put in service on high containment pressure. Therefore, the number of ICs available in this case is only one, which is the same as the number of ICs available in the MS pipe break cases as analyzed for all breaks larger than a 19 mm equivalent diameter.

Although the IC steam supply pipe diameter may be as large as the MS flow limiter diameter, the break flow rate from an IC steam supply pipe break is less than the break flow rate from the MS pipe break. This is due to the much larger inventory in the MS piping connected to both ends of the break.

The liquid in the IC is subcooled and does not contribute to high energy discharge from the break. Therefore, the MS pipe break for containment response is more limiting than the IC steam supply pipe break.

ICS condensate return pipe diameter is much smaller than the FW pipe diameter used in the FWLB analysis. Therefore, large breaks in the IC condensate return pipe are bounded by large breaks in the FW pipe or MS pipe.

Because the IC pipe breaks are bounded by either the MS pipe or the FW pipe breaks, no further analysis of IC pipe breaks is needed.

15.5.4.5.4 Small Steam and Liquid Pipe Breaks Inside Containment

Postulated Initiating Event

A break area of ≤ 19 mm equivalent diameter remains unisolated. These breaks are analyzed for fuel integrity and containment integrity for at least 72 hours using conservative assumptions.

All liquid pipe break nozzles are at least 4 meters above TAF. A small pipe break on instrument lines may remain unisolated indefinitely.

A small liquid pipe break and a small steam pipe break have similar break flow rates after the level falls to 4 m above TAF. Because the ICs depressurize the RPV, the break flow becomes very small in a few hours. Fuel heat up does not occur even without injection to the RPV. Containment heat removal occurs through the PCCS to the equipment pool and through the containment head to the reactor cavity pool.

Sequence of Event

The bounding scenario analyzed:

- Small steam pipe break concurrent with LOPP
- Pressure controller remains as-is; TCV remains at initial position, turbine chest pressure decreases rapidly
- FW pump trips and coasts down

- MS pipe low pressure setpoint is reached
- Reactor scrams
- MSRIVs are fully closed
- RPV water level decreases to Level 2
- Condensate return valves on two ICS trains are fully open
- Peak containment pressure is reached

Tables 15.5-24 and 15.5-25 list the event sequences for small steam and liquid pipe breaks, respectively.

Identification of Operator Actions

No operator action is required to mitigate the event.

System Operation

The conservative cases assume LOPP concurrent with the pipe break, which is more limiting than the case where preferred power is available (discussed in Section 15.2). TCV and TSV closure is expected to occur because of the LOPP. However, TSVs and TCVs are assumed not to close on LOPP. Rather, MSRIV closure on low steam pipe pressure is conservatively credited in the analysis. The back pressure through TSVs and TCVs is assumed to decrease rapidly, maximizing the RPV water inventory loss to the turbine. Reactor scram also occurs on low steam line pressure accounting for the delays after the low steam pipe pressure is reached.

Small Liquid Pipe Break

IC condensate return valves start opening when the level falls to Level 2. As shown on Figure 15.5-112 and Figure 15.5-113, decay heat is removed by two ICs. The RPV depressurizes initially when the sum of the decay heat removal rate by the ICs and the energy discharge from the break exceeds the decay heat power. ICs remove less power at lower pressure because of the lower temperature difference between the RPV steam and pool water. Reactor pressure stabilizes at a low value, and the depressurization rate becomes very small. As shown on Figure 15.5-114, there is a rapid decrease in the collapsed downcomer level. This decrease is primarily due to the void collapse in the RPV. There are two small increases in level at approximately 69000 and 91000 seconds on Figure 15.5-114. These increases are due to void redistribution in the vessel. There is no increase in the RPV water inventory. As shown on Figure 15.5-114, downcomer collapsed level falls below TAF at 206000 seconds. However, the two-phase level in the core remains above TAF. Fuel remains wetted and thus never heats up.

The break mass and energy release are calculated assuming there is no back pressure. This assumption was made to bound breaks outside containment and accounts for the expected lower containment pressure than calculated because of the conservative assumptions used in the containment analyses. Even without break back pressure, Figure 15.5-116 shows that the break flow becomes very small in the long term.

Containment pressure calculated by using conservative assumptions and the small liquid pipe break flow without back pressure is shown on Figure 15.5-117. RPV pressure is also shown on the same figure. The calculated containment pressure increases to the RPV pressure at approximately 232800 seconds. Containment pressure is not higher than the RPV pressure because the break flow stops if the containment pressure becomes equal to the RPV pressure.

However, there is a potential that if ICS depressurizes the RPV faster than PCCS depressurizes containment in the absence of a break, reverse flow from containment to the RPV may occur.

Non-condensables ingested into the RPV may collect in the ICs and reduce their efficiency. Both the RPV and the containment could start repressurizing if back flow were to occur. To investigate this possibility, the containment pressure is calculated starting from 232800 seconds until the end of the 72-hour period assuming no break flow. The dashed line on Figure 15.5-115 shows that containment is rapidly depressurized in the absence of break flow. Because containment pressure decreases faster than the RPV pressure when there is no break flow, the RPV cannot depressurize below the containment pressure and reverse flow cannot occur. Energy released from the RPV through the break is a small fraction of the decay heat in the long term. A much larger fraction of the decay heat is removed by the ICs. Therefore, RPV pressure calculated with and without a break are approximately the same at the time the RPV depressurizes to near containment pressure.

Figure 15.5-117 shows that containment pressure remains below 262 kPa in the long term, which is 50% of the peak accident pressure calculated in Subsection 15.5.4.5.1.

Small Steam Pipe Break

Figure 15.5-108 shows that level remains above TAF for small steam pipe breaks. Containment response shown on Figure 15.5-110 and Figure 15.5-111 for a small steam pipe break is similar to the containment response in Figure 15.5-117 and Figure 15.5-118 for a small liquid pipe break.

A small pipe break on an IC steam pipe does not cause a more limiting core or containment response than an instrument line break. The small breaks conservatively credit only two of the three ICs even though a break of ≤ 19 mm equivalent diameter on an IC does not cause degradation in the IC. There is sufficient steam in the RPV to feed the condensation in the IC. Insufficient steam in the RPV to feed the IC only occurs if the RPV is depressurized to the point where almost all of the steam escapes the break. This is the case if the RPV pressure is lower than that calculated for an instrument pipe break. However, in this case, the break flow is also less than the break flow calculated for an instrument pipe break, resulting in the IC small break less limiting than the instrument pipe break. Breaks on IC steam pipes are no more limiting than a break on an instrument steam pipe break. This is because the break flow rates are the same for a small steam pipe break regardless of the break location, or the RPV pressure is too low to feed the IC.

Core and Barrier Performance

Results

Key results for small steam and liquid pipe breaks are summarized in Table 15.7-3. The plots for RPV parameters and containment for small steam pipe breaks are shown on Figure 15.5-106 through Figure 15.5-111.

The plots for RPV and containment parameters for small liquid pipe breaks are shown on Figure 15.5-112 through Figure 15.5-118. All liquid pipe break nozzles are at least 4 meters above TAF. A small pipe break on the instrument lines may remain unisolated indefinitely. Because the ICs depressurize the RPV, the break flow becomes very small in a few hours. Fuel heat up does not occur even without injection to the RPV.

Barrier Performance

There is no fuel damage because of a small liquid and FW break inside containment. The only activity available for release is normal reactor coolant concentration in the pipe prior to the break.

Radiological Consequences

The radiological consequences for a small liquid or FW break inside containment are bounding for the consequences for small liquid and steams pipe breaks outside containment presented in Subsection 15.5.9.1.5.

15.5.5 Analysis of Design Extension Conditions Without Core Damage

The bounding transient event selection in Subsection 15.2.3 determines the list of DEC events evaluated in the following subsections. Table 15.5-3 and Table 15.5-4 provide input parameters and initial conditions used in the DEC analyses. Table 15.5-5 provides the DLs used in the DEC analyses. The analysis is performed consistent with the DSA analysis approach described in Subsection 15.1.3 and Subsection 15.2.4..

15.5.5.1 Control Rod Drop Accident

NEDC-33912P-A, "BWRX-300 Reactivity Control," (Reference 15.5-12) explains that the BWRX-300 uses GNF2 fuel, with a core design that is like the BWR operating fleet. The approved Control Rod Drop Accident (CRDA) methodology of NEDE-33885P-A, "GNF CRDA Application Methodology" will be applied to the BWRX-300 to demonstrate that the cladding failures do not occur for the postulated CRDA. The results of the rod drop calculations will be discussed in the Probabilistic Risk Assessment (PRA) analysis that will be summarized in the FSAR.

The staff stated in Section 4.1.11 of "Final Safety Evaluation for GE-Hitachi Licensing Topical Report NEDC-33912P, Revision 0, BWRX-300 Reactivity Control" (ADAMS Accession Number ML20275A373):

> "NEDC-33912, Section 4.1.11, states that the safety analyses to demonstrate compliance with GDC 28, including each of the specified transients and accidents, will be provided during future licensing activities. The CRDA event applied to an equilibrium cycle will be analyzed using the approved GNF CRDA methodology (NEDE-33885P-A) following confirmation of its applicability to the final BWRX-300 design. In addition, GNF has indicated plans to deviate from NEDE-33885P-A and will provide information in future licensing activities to support a conclusion that additional cycle-by-cycle CRDA evaluations are not warranted. The staff will review this information and make a finding on the acceptability of this approach at that time.
>
> Based on the design features of the BWRX-300 and, specifically, the ability of the rod control system to implement control rod patterns and control rod blocks, along with the additional analyses described in NEDC-33912 (including a CRDA event analyzed consistent with NEDE-33885P-A and in conformance with Limitation and Condition 5.2 of this SE) that will be performed to support future licensing activities, the NRC staff finds this approach consistent with GDC 28 and, therefore, acceptable. The NRC staff will conduct a detailed evaluation to confirm that GDC 28 is satisfied and cycle-by-cycle CRDA evaluations are not warranted when the NRC receives an application for a BWRX-300 SMR."

The Limitation and Condition 5.2 of the Safety Evaluation Report states: "A CRDA design-basis safety analysis applied to an equilibrium cycle in accordance with an approved methodology, providing justification for any deviations (e.g., performing a one-time analysis to bound cycle-by-cycle variations), or request an exemption to justify the CRDA as a beyond-design-basis event and document the CRDA analysis results in the probabilistic risk assessment."

PDC 28 compliance ensures that the amount and reactivity rate increases resulting from steam line rupture, changes in reactor coolant temperature and pressure, and cold-water addition are limited and do not result in damage to the RCPB or disturb the core, its support structure, or other

reactor vessel internals. Control rod drop and ejection is prevented by positive means. A probability analysis supporting categorization of a CRDA as a DEC is presented below in this section under the “Probability Analysis” subheading.

The Limitation and Condition 5.2 of NEDC-33912P is satisfied by PDC 28 compliance and justification for categorizing the CRDA as a beyond-design-basis event. The CRDA will be analyzed as a DEC event in the FSAR. It should be noted that a CRDA event has never occurred in the history of BWR operation.

Technical Basis

The combined features of the CRD system and the rod control system incorporate appropriate limits on the potential amount and rate of reactivity increase. The fine motion movement capability of the FMCRD allows reactivity additions from rod withdrawal to be limited. Control rod withdrawal sequences and patterns are selected to achieve optimum core performance and low individual rod worth. The BWRX-300 design prevents rod drop and rod ejection events through positive design means.

The BWRX-300 uses GNF-2 fuel and has a core design that is similar to the BWR operating fleet.

Positive Means

The BWRX-300 features a FMCRD that uses a bayonet style coupling that requires a 45-degree rotation to uncouple. Because the FMCRD is firmly bolted into its position under the reactor vessel and the control rod is constrained from rotation by the fuel assemblies, it is not physically possible for the control rod to uncouple from the FMCRD during normal reactor operation.

The hollow piston is the component within the FMCRD coupled to the control rod. The hollow piston normally rests on the ball nut internal to the FMCRD. There are dual FMCRD separation detection switches that sense whether the hollow piston along with the associated control rod are resting on the ball nut. If the sensor detects that the hollow piston is no longer on the ball nut, then control rod withdrawal is blocked. Additionally, the hollow piston has latches that prevent inadvertent withdrawal of the assembly when not attached to the ball nut. This essentially limits possible separation such that it is not physically possible for a CRDA involving a single control rod falling completely out of the core to occur.

Control rod ejection is prevented by physical constraints including the attachment of the control rod guide tube to the core plate and the CRD connection to the control rod guide tube. The FMCRD includes a brake that further prevents inadvertent rod withdrawal. The FMCRD also includes an internal ball check valve, which reduces the likelihood of rapid rod withdrawal. The Safety Class 1 ball check valve performs a Safety Category 1 function in preventing:

- Reverse flow from the scram inlet port against the pressure and flow conditions caused by a break of the scram line
- Loss of pressure from the underside of the hollow piston
- Generation of loads on the drive that could cause a rapid rod withdrawal and associated reactivity insertion

The rod control system controls rod patterns and provides control rod blocks limiting the rate and amount of reactivity addition for control rod movement.

The combined features of the CRD system and the rod control system incorporate appropriate limits on the potential amount and rate of reactivity increase. The fine motion movement capability of the FMCRD allows limited reactivity additions from rod withdrawal. Control rod withdrawal sequences and patterns are selected to achieve optimum core performance and low individual rod worth. The BWRX-300 design prevents rod drop and rod ejection events through positive

design means. Control rod drop is prevented using a bayonet style coupling, CRD mechanism latches, and CRD separation switches.

Probability Analysis

The BWRX-300 design features two separate and independent functions to detect separation between a control rod and the CRDM. A block is initiated that cuts power to the CRDM, preventing further separation. As a result, if the rod were to drop at a later point in time, it would only drop a short distance (a few centimeters), causing a mild reactivity change compared to what is analyzed with the approved methodology of NEDE-33885P-A.

For a FMCRD, the frequency of a control rod drop is estimated to be 3.47E-6 per year. This estimate assumes the control rod gets stuck in the core with a frequency of 1.0 per year and that the drop can occur at any time in the reactor cycle, making it conservative and not completely representative of a CRDA event. Due to doppler and void feedback characteristics, CRDA is only of concern in low power conditions, i.e., below 5% rated thermal power and below 300 psig steam dome pressure.

Conclusion

The Limitation and Condition 5.2 of NEDC-33912P is satisfied by PDC 28 compliance and justification for categorizing the CRDA as a beyond-design-basis event. The CRDA will be analyzed as a DEC event in the FSAR.

15.5.5.2 Pressure Increase – DECs

15.5.5.2.1 Closure of One Main Steam Reactor Isolation Valve

This event is designated as an EX-DEC event. The event sequence name is Closure of One MSRIV (1MSRIVC), and the Event Sequence ID is PI-1MSRIVC_CCF-Hydraulic-Scram_EX-DEC.

Postulated Initiating Event

The PIE is the same as the BL-AOO event discussed in Subsection 15.5.3.2.2. The analysis assumes a CCF hydraulic scram failure. The control rods enter the core using the CRDM run-in function. This event demonstrates that the CRDM run-in function performs the FSF control of reactivity without hydraulic scram.

Sequence of Event

The event sequence summary:

- One MSRIV closes causing pressure and power to increase
- Hydraulic scram signal on MSRIV position fails; scram fails
- Hydraulic scram on any signal fails
- MSRIV in the second steam line closes on leak detection indication (assumed because it makes the event more severe)
- CRDM run-in initiation on high flux after scram signal
- All ICS trains initiate on high flux after scram signal
- FW and condensate pumps trip on high flux after scram signal
- Controlled state achieved

Table 15.5-26 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL2 functions:

- DL2-21 – Anticipatory Hydraulic Scram on MSRIV/MSCIV Position (scram fails)
- DL2-43 – FW Check Valve Closure on Reverse FW Flow

Credited DL4a functions:

- DL4a-40 – CRD Fast Motor Run-In on High Flux After Scram Signal
- DL4a-41 – FW Pump/Condensate Pump Trip on High Flux after Scram Signal
- DL4a-42 – ICS Trains 1, 2, and 3 Initiations on High Flux after Scram Signal

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by initiating closure of one MSRIV. The initial conditions and plant parameters are provided in Table 15.5-3 through Table 15.5-5. The initial CPR and hot rod power are consistent with the referenced core design.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The simulated closure of one MSRIV is presented on Figure 15.5-119 through Figure 15.5-124, and the results are presented in Table 15.7-2. The results are shown for the case with the limiting result for cladding temperature and reactor pressure response. The neutron flux increases rapidly because of the void reduction caused by the pressure increase. The pressure increase is limited due to the initiation of ICS, FWPT and CRDM run-in that occur on high flux after scram signal (i.e., indications of high-power level post scram initiation). The long-term core cooling capability is assured by meeting the DBA event acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the DBA event RCPB pressure criteria. The acceptance criteria are provided in Section 15.3. Although the DEC acceptance criteria are described in Subsection 15.3.2, the results are conservatively compared to the DBA acceptance criteria (described in Subsection 15.3.1) and demonstrate significant margin. The cladding temperature remains below the temperature at which significant oxidation occurs due to metal water reaction.

This event resulted in the highest peak cladding temperature and peak vessel pressure. Sensitivities are performed to examine cliff edge effects. Sensitivities on key initial conditions and phenomena that affect cladding temperature and peak vessel pressure described in the TRACG Application NEDC-34043P are applied separately by at least one standard deviation:

- Core void coefficient
- Channel interfacial shear
- Chimney interfacial shear
- Separator steam carry under
- Critical quality used in boiling transition correlation
- Hot rod power

Results indicate no significant cliff edge effects. No excessive vessel pressure and no core damage occurs.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient more than the DBA derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

DEC events do not have event-specific radiological acceptance criteria. The effects of DEC events are considered in the PSA (Section 15.6).

15.5.5.2.2 Complex Sequence of Generator Load Rejection or Turbine Trip

This event is designated as an EX-DEC event. The event sequence name is Complex Sequence Generator Load Rejection or Turbine Trip (LR-TT) plus CCF-Mechanical CRD, and the Event Sequence ID is CSS-LR-TT_CCF-Mechanical-CRD_EX-DEC. This event demonstrates that with multiple failures to insert independent control rods with diverse motive forces combined with a very frequent PIE, that the remaining control rods perform the FSF reactivity control.

Postulated Initiating Event

The PIE is the same as for the LR-TT AOO event. Additionally, the event assumes that half of the control rods with the highest rod worth fail to scram and the CRDM run-in fails to insert the rods that failed to scram. No other failures are assumed.

Sequence of Event

The event sequence summary:

- TCVs and/or TSVs close quickly causing pressure and power increase
- Anticipatory scram occurs on indication of a turbine trip or load rejection signal, but half of the control rods fail to insert
- RPC opens TBVs to control pressure
- RLC maintains level
- Controlled state achieved

Table 15.5-27 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL2 functions:

- DL2-02 – Maintain Target Level
- DL2-01 – Maintain Target Pressure
- DL2-08 – Anticipatory Hydraulic Scram on Generator Load Rejection or Turbine Trip Demand
- DL2-09 – TBV Fast Open on Generator Load Rejection or Turbine Trip Demand

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by initiating a generator load rejection or turbine trip that results in a fast closure of the TCVs and/or TSVs. The initial conditions and plant parameters are provided in Table 15.5-3 through Table 15.5-5. The initial CPR and hot rod power are consistent with the referenced core design.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The simulated generator load rejection or turbine trip is presented on Figure 15.5-125 through Figure 15.5-130, and the results are presented in Table 15.7-2. The results are shown for the case with the limiting result for reactor pressure response. Automatic anticipatory reactor scram occurs following indication of a generator load rejection or turbine trip with half of the rods failing to insert. The neutron flux increases rapidly because of the void reduction caused by the PI.

The long-term core cooling capability is assured by meeting the DBA event acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the DBA event RCPB pressure criteria. The acceptance criteria are provided in Section 15.3. Although the DEC acceptance criteria are described in Subsection 15.3.2, the results are conservatively compared to the DBA acceptance criteria (described in Subsection 15.3.1) and demonstrate significant margin. The cladding temperature remains well below the temperature at which significant oxidation occurs due to metal water reaction.

A controlled state is achieved with steam bypassed to the main condenser and fed back into the reactor by FW. This is maintained for a significant amount of time as long as power is available. Given the conditions, operators initiate additional CRDM run-in signals or manually insert CRDM to insert the remaining control rods into the core. If operator actions are unsuccessful, operators inject boron to shut the reactor down. Another option available to the operators is to decrease power by reducing FW flow. With reduced FW flow, reactor water level decreases, reducing core flow and reducing reactivity through void reactivity feedback until the steam flow matches the FW flow.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the DBA derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

DEC events do not have event specific radiological acceptance criteria. The effects of DEC events are considered in the PSA (Section 15.6).

15.5.5.2.3 Loss of Condenser Vacuum with CCF Hydraulic Scram

This event is designated as an EX-DEC event. The event sequence name is LOCV, CCF-Hydraulic-Scram and the Event Sequence ID is PI-LOCV_CCF-Hydraulic-Scram_EX-DEC. This event demonstrates that the CRDM run-in function performs the FSF controlling reactivity without the hydraulic scram.

Postulated Initiating Event

The PIE is the same as the BL-AOO event. A CCF results in failure of hydraulic scram. The control rods enter the core using the CRDM run-in function. No other failures are assumed.

Sequence of Event

The event sequence summary:

- Loss of vacuum results in a turbine trip
- TSVs close quickly causing pressure and power increase
- Hydraulic scram signal on either turbine trip or high main condenser pressure scram fails
- Hydraulic scram fails on any signal
- CRDM run-in initiation occurs on high flux after scram signal
- RPC opens TBVs to control pressure
- RLC maintains RPV water level
- ICS initiates on high flux after scram signal
- FW pumps trip on high flux after scram signal
- TBVs close on high main condenser pressure
- Controlled state achieved

Table 15.5-28 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited defense line functions:

DL2:

- DL2-02 – Maintain Target Level
- DL2-01 – Maintain Target Pressure
- DL2-09 – TBV Fast Open on Generator Load Rejection or Turbine Trip Demand
- DL2-13 – Turbine Trip on High Main Condenser Pressure Setpoint 2
- DL2-14 – TBV Closure on High Main Condenser Pressure Setpoint 3
- Anticipatory Hydraulic Scram on Either:
  - DL2-08 – Generator Load Rejection or Turbine Trip Demand (scram fails)
  - DL2-37 – High Main Condenser Pressure Setpoint 1 (scram fails)

DL4a:

- DL4a-40 – CRD Fast Motor Run-In on High Flux After Scram Signal
- DL4a-41 – FW Pump/Condensate Pump Trip on High Flux after Scram Signal
- DL4a-42 – ICS Trains 1, 2, and 3 Initiations on High Flux after Scram Signal

Core and System Performance

Input Parameters and Initial Conditions

The LOCV results in a turbine trip. The event is simulated by initiating a turbine trip that results in a fast closure of the TSVs. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5. The initial CPR and hot rod power are consistent with the referenced core design.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The simulated LOCV is presented on Figures 15.5-131 through 15.5-136, and the results are presented in Table 15.7-2. The results are shown for the case with the limiting result for cladding temperature and reactor pressure response. The neutron flux increases rapidly because of the void reduction caused by the PI. However, the PI is initially limited by the TBVs opening. The pressure increase is limited due to the initiation of ICS, FW trip, and CRDM run-in that occur on high flux after scram signal.

The long-term core cooling capability is assured by meeting the DBA event acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the DBA event RCPB pressure criteria. The acceptance criteria are provided in Section 15.3. Although the DEC acceptance criteria are described in Subsection 15.3.2, the results are conservatively compared to the DBA acceptance criteria (described in Subsection 15.3.1) and demonstrate significant margin. The cladding temperature remains below the temperature at which significant oxidation occurs due to metal water reaction.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the DBA derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

DEC events do not have event specific radiological acceptance criteria. The effects of DEC events are considered in the PSA (Section 15.6).

15.5.5.2.4 Loss of Preferred Power with CCF Hydraulic Scram

This event is designated as an EX-DEC event. The event sequence name is LOPP, CCF-Hydraulic-Scram and the Event Sequence ID is PI-LOPP_CCF-Hydraulic-Scram_EX-DEC. This event demonstrates that the CRDM run-in function performs the FSF controlling reactivity without the hydraulic scram.

Postulated Initiating Event

The PIE is the same as the BL-AOO event. A CCF results in failure of hydraulic scram. The control rods enter the core using the CRDM run-in function. No other failures are postulated in the event.

Sequence of Event

The event sequence summary:

- LOPP results in generator output breakers opening
- TCVs close quickly
- FW pumps lose power

- Hydraulic scram signal fails on either generator load rejection or low electric bus voltage
- Hydraulic scram fails on any signal
- CRDM run-in initiation on high flux after scram signal
- TBVs close on loss of power
- ICS initiates on high flux after scram signal
- Controlled state achieved

Table 15.5-29 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited defense line functions:

DL2:

- DL2-01 – Maintain Target Pressure
- Anticipatory Hydraulic Scram on Either:
  - DL2-08 – Generator Load Rejection or Turbine Trip Demand (scram fails)
  - DL2-26 – Low Electric Bus Voltage (scram fails)
- DL2-09 – TBV Fast Open on Generator Load Rejection or Turbine Trip Demand
- DL2-43 – FW Check Valve Closure on Reverse FW Flow

DL4a:

- DL4a-40 – CRD Fast Motor Run-In on High Flux After Scram Signal
- DL4a-42 – ICS Trains 1, 2, and 3 Initiations on High Flux after Scram Signal

Core and System Performance

Input Parameters and Initial Conditions

The LOPP results in the generator output breakers opening and a loss of power to the FW pumps. The event is simulated by initiating a FWPT and a load rejection that results in a fast closure of the TCVs. The initial conditions and plant parameters are provided in Tables 15.5-3 through 15.5-5. The initial CPR and hot rod power are consistent with the referenced core design.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The simulated loss of preferred power with CCF hydraulic scram failure is shown on Figures 15.5-137 through 15.5-142 and the results are presented in Table 15.7-2. The results are shown for the case with the limiting result for cladding temperature and reactor pressure response. The rapid closure of the TCVs results in a PI. The neutron flux increases rapidly because of the void reduction caused by the PI. However, the PI is initially limited by the opening of the TBVs. The TBVs later close, ICS initiates, and CRDMs run in on high flux after scram signal. The ICS continues to limit the PI.

The long-term core cooling capability is assured by meeting the DBA event acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the DBA event RCPB pressure criteria. The acceptance criteria are provided in Section 15.3. Although the DEC acceptance criteria are described in Subsection 15.3.2, the results are conservatively compared to the DBA acceptance criteria (described in Subsection 15.3.1) and demonstrate significant margin. The cladding temperature remains below the temperature at which significant oxidation occurs due to metal water reaction.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the DBA derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

DEC events do not have event specific radiological acceptance criteria. The effects of DEC events are considered in the PSA (Section 15.6).

15.5.5.3 Reactivity and Power Distribution Anomalies – DECs

15.5.5.3.1 All Control Rod Withdrawal at Power

This event is designated as an EX-DEC event. The event sequence name is CCF – All Control Rod Withdrawal at Power- All Rods (CCF-ACRW), Passive Digital CCF DL2 Technology Platform (CCF-DL2) and the Event Sequence ID is RI-CCF-ACRW_CCF-DL2_EX-DEC.

Postulated Initiating Event

All control rods inserted in the core start to withdraw due to rod control spurious CCF. A passive CCF of DL2 technology platform results in DL2 function failure.

Sequence of Event

The event sequence summary:

- All control rods start to withdraw resulting in a power increase
- Automatic Thermal Limit Monitor (ATLM) and Multi-Channel Rod Block Monitor (MRBM) fail to block rod withdrawal
- RPC and RLC fail as-is at the initial condition
- Scram on STP power or neutron flux
- After scram, no immediate challenge to cladding and RCPB integrity
- RPV pressure decreases
- RPV water level decreases temporarily due to the void collapse in the core and chimney
- Sensed level increases due to continuing FW flow and flashing in the downcomer
- FW isolation occurs on high RPV water level
- MS isolation occurs on low RPV pressure and pressure slowly increases
- One ICS train initiates on high RPV pressure (the simulation is ended before any ICS initiation because the key mitigation DL functions have already been demonstrated and a single ICS train can control pressure and removing decay heat as demonstrated in the pressure increase DBA analysis)

- Controlled state achieved

Table 15.5-30 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited DL3 functions:

- DL3-05 – Hydraulic Scram on High Simulated Thermal Power
- DL3-04 – Hydraulic Scram on High Neutron Flux
- DL3-23 – FW Isolation on High RPV Water Level
- DL3-17 – MSRIV/MSCIV Isolation on Low RPV Pressure
- DL3-11 – ICS Train 1 initiation on High RPV Pressure

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by withdrawing all the control rods in the core using the initial conditions, plant parameters, and control rod speed specified in Tables 15.5-3 through 15.5-5. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC and MOC cycle exposure conditions. EOC exposure is not run because all control rods are fully withdrawn.

Results

The simulated All Control Rod Withdrawal at Power (ACRW) is presented on Figures 15.5-143 through 15.5-148. The analysis results are presented in Table 15.7-2. The results are shown for the case with the limiting PCT response result.

When the control rods are withdrawn, the power increases and scram occurs on high simulated thermal power or neutron flux. The RPV water level increases and FW is isolated. RPV pressure decreases and the MSRIVs close. Eventually the RPV pressure increases, and one ICS train initiates. The pressure remains well within the DBA event RCPB pressure acceptance criteria in Subsection 15.3.1. There is no challenge to containment.

The long-term core cooling capability is assured by meeting the DBA acceptance criteria for fuel cladding and RCPB. The reactor integrity is assured by meeting the DBA event RCPB pressure criteria. The acceptance criteria are provided in Section 15.3. Although the DEC acceptance criteria are described in Subsection 15.3.2, the results are conservatively compared to the DBA acceptance criteria (described in Subsection 15.3.1) and demonstrate significant margin. The cladding temperature remains well below the temperature at which significant oxidation occurs due to metal water reaction.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the DBA derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed and there is no core damage.

Radiological Consequences

DEC events do not have event specific radiological acceptance criteria. The effects of DEC events are considered in the PSA (Section 15.6).

15.5.5.3.2 Inadvertent Single Control Rod Withdrawal at Power DEC

This event is designated as an EX-DEC event. The event sequence name is Inadvertent Control Rod Withdrawal at Power – Single Rod (ICRW), and the Event Sequence ID is RI-ICRW_DL2_CCF_EX-DEC.

Postulated Initiating Event

A control rod inserted in the core is withdrawn due to a failure. A passive CCF of DL2 results in failure of the DL2 functions.

Sequence of Event

The event sequence summary:

- Single rod (with highest reactivity worth) starts to withdraw
- ATLM and MRBM fail to block the rod withdrawal
- RPC and RLC are assumed to function normally because this prolongs the event and makes it more severe for cladding temperature effects
- Reactor power increases but the scram level is not reached
- Local power and cladding temperature increase
- Power and the cladding temperature reach a stable level
- Operator action to initiate scram is expected due to the high-power level (Not credited in analysis simulation as these DL functions demonstrate achieving and maintaining a controlled state)
- After the scram, no credit is taken for RPC or RLC (the simulation is ended because the key mitigation DL functions have already been demonstrated)
- RPV pressure decreases and the RPV water level increases (the simulation is ended because the key mitigation DL functions have already been demonstrated)
- FW isolation occurs on high RPV water level (the simulation is ended because the key mitigation DL functions have already been demonstrated)
- MS isolation occurs on low RPV pressure and pressure slowly increases (the simulation is ended because the key mitigation DL functions have already been demonstrated)
- One ICS train initiates on high RPV pressure (the simulation is ended before any ICS initiation because the key mitigation DL functions have already been demonstrated and a single ICS train can control pressure and removing decay heat as demonstrated in the pressure increase DBA analysis)
- Controlled state achieved

Table 15.5-31 lists the event sequence.

Identification of Operator Actions

No operator action is credited.

Systems Operation

Credited DL3 functions:

- DL3-17 – MSRIV/MSCIV isolation on Low RPV Pressure
- DL3-23 – FW Isolation on High RPV Water Level
- DL3-11 – ICS Train 1 Initiation on High RPV Pressure

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by withdrawing a single control rod in the core using the initial conditions, plant parameters, and control rod speed specified in Tables 15.5-3 through 15.5-5. The initial CPR and the hot rod power are consistent with the reference core design.

The analysis is performed using an equilibrium core design. The event is run at BOC and MOC cycle exposure conditions. EOC exposure is not run because all control rods are fully withdrawn.

Results

The simulated ICRW event is presented on Figure 15.5-149 through Figure 15.5-154. The analysis results are presented in Table 15.7-2. The results are shown for the case with the limiting PCT response result.

When the control rod is withdrawn, the power increases. The RPV water level and pressure vary insignificantly because RLC and RPC function to maintain level and pressure preventing scram, thus maximizing the effect of fuel cladding temperature increase.

The acceptance criteria are provided in Section 15.3. Although the DEC acceptance criteria are described in Subsection 15.3.2, the results are conservatively compared to the PCT values of the DBA acceptance criteria (described in Subsection 15.3.1) and demonstrate significant margin.

The long-term core cooling capability is assured because the effects are local. The reactor integrity is assured by meeting the DBA event RCPB pressure criteria.

Sensitivities are performed examining cliff edge effects. Key initial conditions are conservatively biased to cause transition boiling even though this does not occur at the nominal/realistic conditions associated with DEC conditions. With initial CPR conservatively biased low (by approximately 0.05), and the hot rod power (LHGR) conservatively biased high (approximately 30%), local high cladding temperatures occur in hot rods in a few high-power bundles located near the control rod withdrawn in error. This is expected to result in fuel cladding failure in a very limited number of rods. However, the fuel failures are localized, the core remains cooled, and no core damage occurs.

Barrier Performance

There is no challenge to the RCPB and containment. The fuel cladding may experience local failures if initial LHGR and CPR are more severe. The predicted number of rod failures is limited to high powered fuel rods in a few high-powered bundles near the control rod withdrawn in error. However, the fuel failures are localized, the core remains cooled, and no core damage occurs.

Radiological Consequences

DEC events do not have event specific radiological acceptance criteria. The effects of DEC events are considered in the PSA (Section 15.6).

15.5.5.4 Decrease in Reactor Coolant Inventory – DEC

This event is in the IR group and is designated as an EX-DEC event. The event sequence name is FW Isolation (CCF-FWI-DL3) and the Event Sequence ID is IR-CCF-FWI-DL3_EX-DEC.

Postulated Initiating Event

The event sequence assumes a spurious CCF isolates all FW flow and a passive CCF of the DL3 functions.

Sequence of Event

The event sequence summary:

- FW flow isolation causes RPV water level and power to decrease
- RPC maintains RPV pressure
- Scram and MSRIV isolation on sustained low FW flow
- ICS pressure control initiates on high RPV pressure
- Controlled state achieved

Table 15.5-32 lists the event sequence.

Identification of Operator Actions

No operator action is required to mitigate the event.

Systems Operation

Credited defense line functions:

DL2:

- DL2-01 – Maintains Target Pressure
- DL2-42 – Anticipatory Hydraulic Scram on Sustained Low FW Flow
- DL2-31 – ICS Pressure Control on High Reactor Pressure

DL4:

- DL4a -12 – MSRIV/MSCIV Isolation on Sustained Low FW Flow

Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by initiating a conservatively fast isolation of all FW flow. The initial conditions and plant parameters are provided in Table 15.5-3 through Table 15.5-5. The initial CPR is conservatively biased low, and the hot rod power is conservatively biased high.

The analysis is performed using an equilibrium core design. The event is run at BOC, MOC, and EOR cycle exposure conditions.

Results

The CCF FW isolation event is presented on Figure 15.5-155 through Figure 15.5-160, and the results are presented in Table 15.7-2 for the exposure with the limiting PCT response. The loss of FW flow results in a reduction of vessel inventory, causing the power and RPV water level to decrease. RPC maintains reactor pressure. Reactor scram and MS isolation occurs based on sustained low FW flow. ICS pressure control initiates based on high RPV pressure.

The long-term core cooling capability is conservatively assured by meeting the DBA acceptance criteria for fuel cladding and RCPB. The reactor integrity is conservatively assured by meeting the DBA event RCPB pressure criteria. The acceptance criteria are provided in Section 15.3.

Although the DEC acceptance criteria are described in Subsection 15.3.2, the results are conservatively compared to the DBA acceptance criteria (described in Subsection 15.3.1) and demonstrate significant margin. The cladding temperature remains well below the temperature at which significant oxidation occurs from metal water reaction.

Barrier Performance

The effect of this event does not result in any temperature or pressure transient in excess of the DBA derived acceptance criteria for the fuel, pressure vessel, or containment. No fuel failures occur because there is no significant cladding temperature increase. Therefore, these barriers maintain their integrity and function as designed.

Radiological Consequences

DEC events do not have event-specific radiological acceptance criteria. The effects of DEC events are considered in the PSA (Section 15.6).

15.5.6 Analysis of Design Extension Conditions with Core Damage

The analysis of DECs with core damage are addressed in the Level 2 PSA described in Subsection 15.6.4. Severe accidents identified in the Level 2 PSA will be described in the FSAR.

15.5.7 Analysis of Postulated Initiating Events and Accident Scenarios Associated with the Fuel Pool

The analysis description of those postulated events and accident scenarios associated with the fuel pool will be described in the FSAR.

15.5.8 Analysis of Fuel Handling Accident

The Fuel Handling Accident (FHA) is categorized as a non-reactor group DBA event.

Postulated Initiating Event

The event occurs because of a failure of the fuel assembly lifting mechanism, resulting in the drop of a raised irradiated fuel assembly on the top of the reactor core or into the fuel storage racks. The dropped irradiated fuel assembly results in cladding failure in the dropped and impacted bundles and a subsequent release of fission products from the water in 2 distinct release phases. The radiological release consequences are determined for Exclusion Area Boundary (EAB), Low Population Zone (LPZ), and the Main Control Room (MCR).

Sequence of Event

The event sequence summary:

- Reactor is shut down for refueling operations that begins 24 hours after shutdown. Over this period, the core isotopic inventory decays.
- During refueling operation, a fuel assembly is moved over the top of the core, or the fuel pool and fuel bundle, grapple, mast, and head fall on top of the core or the spent fuel racks
- Rods in the dropped bundle and impacted bundles fail, releasing fission gases and cesium iodide in the plenum and gap of the damaged rods to the reactor or fuel pool water that initiates a 2-phase fission product release

- In Phase 1 of the release that occurs over the first two hours after the event begins, fission gases rise through reactor or fuel pool water to refueling operation floor common airspace surrounding the top of the reactor cavity or fuel pool
- The Phase 1 fission gases initially released to the refueling floor are released to the environment
- The Phase 2 release begins two hours after event initiation, where the cesium iodide initially released to the pool evolves to form elemental iodine. Given the chemical conditions in the fuel pool, a portion of the elemental iodine becomes volatile and releases to the refueling floor airspace and subsequently to the environment.

To prevent the spread of a fission gas release from a FHA to rooms adjacent to the RB fuel handling area operating deck, the RB truck bay door remains closed during movement of irradiated fuel bundles. Administratively closing the RB truck bay door during movement of irradiated fuel bundles also ensures that the FHA release to the environment occurs from the RB as a diffuse source release.

The event sequence for the postulated FHA is provided in Table 15.5-33.

Methodology

The dose consequences of the postulated FHA are calculated using the RADTRAD Version 3.10 computer code. RADTRAD shows compliance with nuclear plant siting criteria for the radiation doses at offsite and Control Room receptors for various LOCA and non-LOCA DBAs.

The fission product release from the breached fuel resulting from the FHA is based on RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Regulatory Position 3.2, and the estimate of the number of fuel rods breached, including the fuel power and burnup limitations specified are based upon RG 1.183, Footnote 11. Radionuclides considered include xenon, krypton, halogens, cesium, and rubidium. The chemical form of radioiodine released from the fuel to the fuel pool is assumed to be 95 percent Cesium Iodide (CsI), 4.85 percent elemental iodine (I_2), and 0.15 percent organic iodide such as CH3I. All the gap activity from the damaged rods is assumed to be released over two Phases.

Phase 1: The instantaneous release from the rising bubbles of fission gas.

I_2 and organic iodine are conservatively assumed to be in vapor form and subsequently decontaminated by passage through the overlying pool of water into the building atmosphere. The retention of noble gases in the water in the fuel pool or reactor cavity is negligible (i.e., decontamination factor of 1). Particulate radionuclides, such as CsI, released from the fuel are assumed to be retained by the water in the fuel pool or reactor cavity (i.e., infinite decontamination factor).

Phase 2: The projected release of CsI re-evolving as I_2.

CsI is conservatively assumed to completely dissociate into the pool water. Due to the low pH of the pool water, CsI (and Phase 1 absorbed I_2) then re-evolves as I_2 into the building atmosphere.

Identification of Operator Actions

No operator action is credited.

Fuel Damage

Because of the complex nature of the impact and the resulting damage to fuel assembly components, predicting the number of failed rods is not possible. For this reason, a simplified energy approach is taken. Numerous conservative assumptions are made to assure that the number of failed rods is conservatively analyzed.

The key assumption for the FHA is that during a refueling operation, a fuel assembly is moved over the top of the reactor core or fuel pool. While the fuel grapple is in the over-hoist condition with the bottom of the assembly at the maximum height allowed when using the fuel handling equipment, the main hoist cable and a redundant cable fails. A drop height of 67.4 ft (20.544 m) above the core is assumed because it is bounding with respect to plant design. The grapple head and mast are fixed vertically to the dropped assembly so that all the kinetic energy is transferred through the dropped assembly to the group of impacted assemblies. The dropped assembly impacts the core at a slight angle and the rods in this assembly might bend. After the assembly impacts the core, the assembly, grapple head and mast fall onto the core horizontally without contacting the side of the pressure vessel or fuel pool.

Fuel rod failure is assumed at 1% circumferential strain. The associated axial strain is (0.01)/ υ, where υ is Poisson's ratio, plastic deformation is assigned a value of 0.5, and the energy per rod failure is expressed by:

$$E_f = \sigma_y \times \varepsilon \times V \qquad \text{(Equation 15.5-1)}$$

Where:

E_f = energy per rod failure

σ_y = yield stress

ε = axial cladding strain

V = volume of fuel cladding

Kinetic energy is calculated for the dropped fuel bundle that accounts for the influences of buoyancy and resistance from the reactor cavity pool water. Finite element analysis simulations are used to determine the kinetic energy based on the drop distance in air or water. The simulation results revealed that when the drop distance of a fuel bundle in air is greater than 7.5 ft (2.3 m), the kinetic energy of the fuel bundle drop in water is less than 50% of that in air. When the bundle drop height is 34.0 ft (10.4 m), the energy is approximately 22% of that in air. This analysis credits a 50% reduction in the kinetic energy of the dropped bundle although the limiting case drop height correlates to a larger reduction.

The fuel assembly wet weight is 600.03 lbf (272.17 kgf), and the mast wet weight is 430.00 lbf (195.04 kgf). Applying the 50% kinetic energy reduction to the fuel assembly due to dropping through water is expressed as:

$$E_1 = \frac{h_{drop} \times (W_{fuel} + W_{mast})}{2} \qquad \text{(Equation 15.5-2)}$$

Where:

E_1 = energy from initial drop

W_{fuel} = weight of fuel bundle

W_{mast} = weight of refueling mast

h_{drop} = drop height

Substituting numerical values into Equation E_1 yields 34,712.01 ft-lbf (4,799.2 kgf-m).

It is assumed that half of the energy is absorbed by the cladding. The ratio of the cladding to the non-fuel mass for GNF2 fuel is 0.4997. The energy per rod failure using the methodology described above (see Equation E_f) is 271.26 ft-lbf/rod (37.503 kgf-m/rod). Therefore, the number of failed rods in the impacted assemblies from the initial drop is:

$$\underline{(50\%)\ (4799.2\ \text{kgf-m})\ (0.4997)} = 31.97 \text{ rods or } 32 \text{ rods}$$

37.503 kgf-m/rod

Additional energy is generated in a secondary impact as the bundle falls over from a vertical orientation to a horizontal orientation, and damages additional rods in the impacted bundles. The fuel bundle is assumed to have a height of 160.0 in. (4.064 m). Incorporating the 50% reduction due to the kinetic energy in water is expressed as:

$$E_2 = 50\% \times (h_{fuel} \times W_{mast} + \tfrac{1}{2} \times h_{fuel} \times W_{fuel}) \qquad \text{(Equation 15.5-3)}$$

Where:

E_2 = energy of dropped bundle and mast from secondary impact

h_{fuel} = height of refueling mast

Fifty percent of the kinetic energy is absorbed by the impacted assemblies; therefore, the number of failed rods in the impacted assemblies from the secondary impact is five rods.

All the rods in the dropped assembly, which contains 92 rods, are assumed to fail and to be full length rods. GNF2 fuel assemblies contain both full-length and part-length rods, resulting in 85.6 effective full-length rods per bundle when accounting for the difference in lengths. Aside from the dropped assembly, the length of the damaged fuel rods is not considered in this evaluation. All the damaged rods are conservatively assumed to be full-length rods.

The number of failed rods used in determining the radiological consequences is:

92 rods + 32 rods + 5 rods = 129 failed fuel rods or 1.507 failed fuel bundles.

There are 240 fuel bundles in the BWRX-300 core. The fraction of the core damaged in an FHA is determined by:

$$\text{Core Damage Fraction} = \frac{1.507 \text{ bundles damaged}}{240 \text{ bundles in the core}} = 6.28\text{E-}03$$

Core Inventory of Isotopes

A BWRX-300 core inventory of isotopes is calculated using the Oak Ridge National Laboratory code ORIGEN2, Version 2.1 and the BWRUE.LIB cross-section library in units of Ci/MWth for the bulk operating parameters in Table 15.5-34.

A subset of the more than 700 isotopes from this inventory are used to model DBA dose consequences. The 60 isotopes used for DBAs are the dominant contributors to immersion and inhalation doses from airborne activity released during a DBA. This set of nuclides consist of 54 isotopes identified in WASH-1400 (NUREG-75/014) and 6 isotopes identified in

SAND-85-2575 (NUREG/CR-4467). This is the group of isotopes typically used for Alternate Source Term (AST) dose evaluations.

The BWRX-300 reactor is subcritical for at least 24 hours prior to initiating refueling operations. The BWRX-300 core inventory of the 60 dose-significant isotopes after 24 hours of decay are shown in Table 15.5-35 with an uncertainty of 1.02 applied to the gap inventory of the damaged fuel to comply with RG 1.183, Section 3.1.

Gap Fractions

For events that are non-LOCAs where some fuel damage is postulated like the FHA, the fractions of the core inventory assumed in the fuel rod gap region for the various radionuclides are taken from RG 1.183 and reported in Table 15.5-36. These gap fractions are subject to the power and burnup limitations specified in RG 1.183, Footnote 11.

Radial Peaking Factor

The radioactive material available for release in an FHA is assumed in the analysis to come from assemblies with peak inventory. To simulate this assumption, the inventory is scaled up by the maximum power Radial Peaking Factor (RPF). This represents the maximum achievable operational power history immediately preceding shutdown. Based on prior experience with GNF2 cores, a representative RPF value of 1.70 is assumed for this analysis.

Activity Released from the Fuel

All particulate isotopes are retained by the water in the fuel pool or reactor cavity water. Thus, only the noble gases and the gaseous form of iodine are available to escape the water. In accordance with RG 1.183 and "Alternative Fuel Handling Accident Transport Methodology," (Reference 15.5-13), the gaseous activity of iodine present in the fuel rod gap is in the form of 4.85% elemental and 0.15% organic iodine. In Phase 1 of the FHA, only 5% of the iodine in fuel rod gap is considered. The remaining 95% in the form of cesium iodide is retained in the pool water and is accounted for in the Phase 2 release. The total activity initially released from the fuel in Phases 1 and 2 is the mathematical product of the activity, core power, RPF, power uncertainty factor, gap fractions (Table 15.5-36), and core damage fraction. With this information, the release from the fuel is calculated as shown in Table 15.5-37.

Pool Scrubbing (Decontamination)

Credit is taken for retention of some released fission gas in the water after it is released from the damaged fuel rods. Because the depth of water above the reactor cavity pool and fuel pool is greater than 23.0 ft, the Decontamination Factor (DF) models from the "Alternative Fuel Handling Accident Transport Methodology" are conservatively applied without correction. The alternative iodine transport model described in the "Alternative Fuel Handling Accident Transport Methodology" evaluates the FHA released from the pool in two distinct Phases:

1. Phase 1 models the initial, instantaneous release of the gaseous activity available in the fuel rod gap to the airspace above the refueling pool. This Phase accounts for the release of the noble gases and the gaseous elemental and organic forms of iodine initially released from the damaged fuel rods.
2. Phase 2 models the subsequent release of elemental iodine that slowly re-evolves from the disassociation and volatilization of the initially released cesium iodine (CsI) and from the elemental iodine retained in the pool because of the Phase 1 release.

A new model for the overall iodine DF applied to the iodine release in Phase 1 is taken from the alternative iodine transport model described in Reference 15.5-13.

In the Phase 2 release, the volatilization rate of I_2 is used to determine the rate at which the I_2 is released from the pool to the airspace above the pool. In this Phase, essentially no further retention of I_2 is assumed. The DF is 1 in the transfer of I_2 from the water to the airspace above the pool during Phase 2.

Chemical Species

Consistent with RG 1.183 and Reference 15.5-13 guidance on FHA, the chemical form of the iodine released from the fuel to the pool water is:

- 95% CsI
- 4.85% I_2
- 0.15% organic iodine (methyl iodine)

An overall DF is applied uniformly to the total of the I_2 and organic iodine release during Phase 1 while the CsI is retained by the water. Thus, the chemical species of the iodine released from the pool to the airspace above the pool during Phase 1 is in the following form:

- 0% CsI
- 97% I_2
- 3% organic iodine (methyl iodine)

In Phase 2, the only iodine released from the pool is in the form of I_2, either from the I_2 retained in the water initially in Phase 1 or that which evolves from CsI. Therefore, the chemical form of the iodine that is released to the airspace above the pool during Phase 2 can be characterized as:

- 0% CsI
- 100% I_2
- 0% organic iodine (methyl iodine)

Other radionuclides such as cesium and rubidium are considered. However, they are assumed to be permanently retained by the water either in particulate form or in solution. This is because there are no mechanisms to cause these radionuclides to evolve from the pool and be released in an airborne form.

Phase 1 Release from the Pool: Initial Gaseous Release and Water Depth

The depth of water is greater than 23 ft, but conservatively assumed to be 23 ft, to allow for use of the alternate DF method. The overall iodine DF for the Phase 1 release, DF_1, is calculated based on the bubble size and rise time through the fuel pool. The bubble size and rise time are a function of the fuel pin pressure. The Phase 1 DF_1 is computed using a best estimate fuel pin pressure per (Reference 15.5-13). Therefore, a conservative pin pressure of 1200 psig is conservatively applied that corresponds to the conditions applicable to the FHA scenario. The DF_1 takes into consideration the buildup of fission gas in the fuel rod over the exposure life as well as the actual temperature condition of the pool water during the FHA. The equation for calculating the Phase 1 DF_1 is shown in Equation DF_1 is shown in Equation 15.5-4:

$$DF_1 = 81.046e^{0.305(t/d)} \qquad \text{(Equation 15.5-4)}$$

Where:

DF_1 = Decontamination Factor

t = bubble rise time (sec)

d = bubble diameter (cm)

The bubble rise time and diameter are calculated using the following Equations:

$$t = 9.2261e^{-6E-4*x}$$

$$d = -0.0002*x+1.0009$$

Where x is the internal rod pressure (psig)

The bubble rise time, bubble diameter and DF_1 are calculated using a bounding fuel rod pressure and the results are shown in Table 15.5-38c. The DF_1 value of 490 is applied to the total iodine released in Phase 1 (5% of the total iodine released to the pool) and is shown in Table 15.5-38a.

Phase 2 Release from the Pool: Re-evolution Phase

During Phase 2 the CsI initially released to the pool water is assumed to disassociate and dissolve in the water. This is assumed to occur instantaneously after release. The dissolved iodine subsequently evolves due to chemical reactions with hydrogen peroxide to form elemental iodine I_2.

The I_2 formed from the CsI (95% of the total iodine released to the pool) is in addition to the I_2 initially retained by the pool in Phase 1. The amount of iodine retained by the pool in Phase 1 is greater than 99% of the iodine released to the pool in Phase 1 (i.e., 1-1/490). As a result, about 5% of the total iodine released from the fuel rod gap inventory in Phase 1 is absorbed in the pool, and available for Phase 2 release. The iodine that remains in the pool for Phase 2 is released directly to the environment using the re-evolution rate. The initial iodine inventory in the pool at the beginning of Phase 2, without adjustment for decay, is shown in Table 15.5-38b.

Transport in the Reactor Building

The radioactivity released from the reactor cavity pool is assumed to mix instantaneously with the free air volume of the refueling outage floor and crane area, which is the intermediate volume between the FHA release from the water and the environment. The RB is not isolated until after the FHA occurs and there are no mechanical means to ensure the refueling outage floor and crane area airspace is well mixed and confined. As a result, no credit is taken for holdup and retention by the RB.

The analysis is not sensitive to the building volume because the release occurs over a 2-hour duration per RG 1.183, Appendix B in Phase 1, and is instantaneously released to the environment in Phase 2 as the I_2 volatizes.

Release Assumptions

Phase 1

The Phase 1 no holdup release transport to the environment simulated transporting 99.9% of the available activity for transport to the environment in 2 hours.

To simulate the leakage to the environment, the following equation is used to calculate a corresponding leakage rate.

$$\frac{C(t)}{C_{ss}} = e^{-Qt/V} \qquad \text{(Equation 15.5-5)}$$

Where:

- $C(t)$ = Volumetric concentration at time t
- C_{ss} = Steady state concentration at the beginning of the release
- Q = Constant flow rate out of space (m^3/min)
- V = RB airspace volume (m^3)
- t = Duration of the release period in minutes

To achieve a 99.9% release (0.1% retention) from the RB to the environment within 120 minutes, the following values are assumed:

- $C(t)/C_{ss}$ = 0.001 (fraction retained in the RB)
- V = 2.83E+02 m^3 (1.0E+04 ft^3)
- t = 120 minutes (2 hours)

Substituting these values and solving for Q in Equation 15.5-5 results in a Phase 1 flow rate of 5.753E+02 ft^3/min (1.629E+01 m^3/min). This Phase 1 flow transports 99.9% of the contamination

released to the RB refueling outage floor and crane area and is transported and released to the environment in two hours.

Phase 2

The Phase 2 iodine transport parameters are taken from Reference 15.5-13. A portion of the I_2 inventory volatilizes due to the chemistry conditions present in the water and slowly release from the water without reduction. The rate of release or evolution is characterized by the evolution removal coefficient that is effectively the rate of release of the volatized I_2 from the pool to the airspace above the pool.

The evolution removal coefficient is converted to a volumetric flow rate (Q_e(L/s)) for use in RADTRAD resulting in a volumetric flow rate of 3.77E-03 cfm.

Dose Calculation

Because the inventory of isotopes released from the surface of the reactor cavity pool is already determined in Tables 15.5-38a and 15.5-38b, the RADTRAD model used to calculate the dose consequences is a simple two compartment model that simulates transport from the RB or fuel pool to the environment.

Breathing Rates

The postulated FHA breathing rates used for the MCR and offsite EAB and LPZ locations are consistent with RG 1.183.

Decay and Daughtering

This analysis assumes a decay time of 24 hours prior to the removal of spent fuel during refueling, and credit for this decay period is taken in the initial core inventory shown in Table 15.5-35.

Decay and daughtering of nuclides during the FHA transport are credited in the RADTRAD model for the duration of the event (i.e., 30 days).

Dose Conversion Factors

Dose Conversion Factors (DCFs) used for inhalation of radioactive material comply with Federal Guidance Report (FGR) 11, “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” (Reference 15.5-14). DCFs used for submergence in a semi-infinite cloud are taken from FGR 12 “External Exposure to Radionuclides in Air, Water, and Soil,” (Reference 15.5-15).

Dispersion in the Environment

As described in Subsection 15.5.9.1, transport to the environment is simulated using offsite atmospheric dispersion factors (χ/Q values) that are established for the Blue Energy Center - Port of Victoria (BECPV) site in the “noting that the maximum EAB χ/Q value is used for the full event duration to ensure the worst 2-hour dose consequence:

Offsite Atmospheric Dispersion Factors (s/m^3)

To be provided in the CPA Part 2.

Transport to the Main Control Room Envelope (MCRE) from the BWRX-300 RB is simulated using onsite atmospheric dispersion factors (χ/Qs) calculated using BECPV site meteorological data. Several sets of dispersion factors were determined for potential RB release points for the FHA to the Control Building (CB) Heating, Ventilation, and Cooling System (HVS) intakes. The most

limiting set of dispersion factors determined for a RB diffuse source release to the CB HVS intakes are used in this analysis:

Main Control Room Envelope Atmospheric Dispersion Factors (s/m^3)

To be provided in the CPA Part 2.

Main Control Room Model

Volume

The MCRE has an approximate volume of 48,125 ft^3 (1,363 m^3) and is serviced by the CB HVS. To account for equipment and structures in the MCRE, this volume is conservatively reduced by 20% to a free-air volume of 38,500 ft^3 (1,090 m^3) for this analysis.

Unfiltered MCR Intake Flow

In the event of a FHA, the CB HVS will automatically switch from supplying unfiltered outside air through the CB normal ventilation to emergency filtration of the outside air supply. However, because the control room habitability is not a Safety Category 1 function in the BWRX-300 design, emergency filtration is not credited in the MCRE for mitigation of any DBA. This analysis does not take credit for filtration of the MCRE. Instead, the analysis models the CB HVS only for the purpose of propagating contamination into the MCRE.

The normal CB HVS intake flow rate and emergency CB HVS intake flow rate are the same. The supply of unfiltered outside makeup air that enters the MCRE is at the normal CB HVS flow rate of 1,870 ft^3/min (0.883 m^3/s).

In addition to the CB HVS flow, the potential exists for unfiltered inleakage of outside air to enter the MCRE from ingress and egress of plant personnel or through small openings in the MCRE such as the bathroom fans, penetrations, and small cracks in the walls. No MCRE unfiltered inleakage flow rate is established for the BWRX-300 MCRE because the control room habitability is not a Safety Category 1 function. The primary concern radiologically is that increased control room unfiltered inleakage could result in the reactor operator being exposed to a larger dose than predicted.

To ensure this analysis conservatively bounds unfiltered inleakage, the intake flow of contaminated outside air is assumed to be at an overall rate of one MCRE volume per minute for the duration of the event (720 hours). Thus, an unfiltered flow rate of 38,500 ft^3/min (18.2 m^3/s) is assumed for 30 days, and the MCRE exhaust flow is assumed equal to the intake flow.

This flow rate is a factor of more than 20 times the normal intake flow rate ensuring the limiting MCRE operator dose estimate is established. Use of an MCRE unfiltered air intake rate that bounds the actual CB HVS flow rate ensures the results are applicable to situations where the MCRE intake flow may vary and bounds any potential changes to the CB HVS flow rate that might occur during the preliminary design phase of the facility. In addition, the conservatism in this flow rate assumption bounds any unfiltered inleakage rate. Therefore, the assumed MCRE flow rate established by this design analysis does not require confirmation of the unfiltered inleakage through testing as it represents an open volume with a maximized inflow and exhaust rate.

Results

The dose results are conservatively compared to the criterion in Table 6 of RG 1.183 that establishes a 6.3 rem Total Effective Dose Equivalent (TEDE) dose limit. The dose consequences in the MCRE are compared to the PDC 19 dose limit of 5.0 rem TEDE for control

room operators. The FHA dose acceptance criteria are satisfied with considerable margin as shown in Table 15.7-4.

15.5.9 Analysis of Radioactive Releases from a Subsystem or a Component

15.5.9.1 Analysis of Loss of Coolant Accidents Outside Containment

As discussed in Subsection 15.5.4.5, the scram and RPV isolation trips occur for the large breaks outside containment within the same time as breaks inside containment. For LOCAs outside containment, the dose consequence analyses are performed using the NUREG/CR-6604 RADTRAD computer code Version 3.10 and the transport radioactivity assumptions from Appendix D of RG 1.183 to various offsite receptors. The results of the breaks outside containment show considerable margin in meeting the dose limits in 10 CFR 50.34(a)(1)(ii)(D), Table 1 of NUREG-0800, SRP 15.0.3, and PDC 19 as shown in Table 15.7-5 through Table 15.7-8. For large breaks, timing of the break detection is less than 1 s for breaks outside containment; that is the same as the time to reach the containment high pressure setpoint. Because reactor scram and isolation valve closures for breaks inside containment also occur for breaks outside containment, the event progression is no different for breaks outside containment than inside containment. For MS pipe breaks, the break flow rate calculated for breaks inside containment is also used for breaks outside containment because the MSCIV closure is not credited in the MS pipe break inside containment. For FW pipe breaks, the only difference between the pipe breaks inside and outside of containment is the closure of the Feedwater Containment Isolation Valve (FWCIV). For a FW pipe break outside containment, break flow includes flow from the reactor as well as the pipe inventory. The IC pipe break outside containment is limited to the flow passing through the orifices in the steam distribution pipes. During normal operation prior to the break, condensate return valves are closed and remain closed. Isolation steam supply pipe has a guard pipe outside containment so that break flow in the supply pipe upstream of the orifice is not discharged outside containment. A break in the IC is followed by a discharge of the subcooled water in the IC into the fuel pool, followed by steam break flow passing through the orifice in the steam supply pipe until the IC RIVs close on break detection. Because the small break analyses inside the containment do not credit containment back pressure, the mass and energy release calculated for breaks inside the containment are bounding for breaks outside the containment.

The following analytical inputs, assumptions, and criteria are used for all breaks outside containment:

Source Term

There is no fuel damage resulting from the following postulated breaks outside containment:

- MSLB
- FWLB
- Isolation Condenser Line Break (ICLB)
- Instrument Line Break (ILB)

Because there is no fuel damage from any of the breaks postulated outside containment, the activity available for release from the break is the activity present in the reactor coolant when the break occurs. As a result, the source term is the same for all breaks outside containment.

Thermal hydraulic LOCA analyses are performed for both small and large break scenarios that comply with the BWRX-300 LOCA acceptance criteria shown in Table 15.3-2.

Compliance with these criteria ensures the integrity of the cladding fission product barrier is maintained for small and large pipe break scenarios in containment that are bounding for breaks both inside and outside containment.

The only activity available for release from the break is that which is present in the reactor coolant and steam when the break occurs. Radiation concentrations in BWRX-300 reactor coolant and steam adequate for use in design basis calculations (such as shielding, equipment design) are determined based on ANSI/ANS-18.1.

Using Appendix D from RG 1.183, two cases are considered for conditions when the postulated accident occurs:

1. The concentration that is the maximum iodine pre-accident spike value, typically, 4.0 µCi/g dose equivalent I-131, permitted under plant operating limits and conditions.
2. The concentration that is the maximum equilibrium value, typically 0.2 µCi/g dose equivalent I-131, permitted under plant operating limits and conditions.

Dose Conversion Factors

The DCFs used to convert design basis iodine concentrations to dose equivalent I-131 concentrations for all breaks outside containment are those from FGR No. 11, Table 2.1, Column "effective."

TEDE doses are the sum of the Committed Effective Dose Equivalent (CEDE) from inhalation and the Deep Dose Equivalent (DDE) from external exposure. CEDE DCFs are taken from FGR No. 11. The DDE DCFs used from FGR No. 12 assume submergence in a semi-infinite cloud with appropriate credit for attenuation by body tissue. DCFs for Kr-89 and Xe-137 are taken from FGR No. 15, "External Exposure to Radionuclides in Air, Water, and Soil," (Reference 15.5-16). The DCFs used in the analyses are based on data provided by the International Commission on Radiological Protection.

Reactor or Turbine Building Volume

The breaks outside containment analyses are not sensitive to the volume of either building as the release to the environment is instantaneous. The use of a large volume generates a faster flow rate of a more diluted source, and a small volume results in a slower flow rate of a more concentrated source term. In either case, the activity available for release to the environment is transported in one minute, and the flow rate applied is calculated based on the assumed volume to ensure an instantaneous release.

Breathing Rate and Occupancy Factors

The postulated breathing rates and occupancy factors used are taken from RG 1.183 for onsite (MCRE) and offsite (EAB and LPZ) dose receptors.

Decay and Daughtering Nuclides

Decay and daughtering of nuclides are credited in the dose model for the 30-day duration of the events, except for the small line break outside containment.

Main Control Room Model

The MCR model is the same for all breaks outside containment and is used in assessing the dose to the operators and the technical support center personnel that are housed in the MCRE as the MCR operators and would receive the same dose as a result.

Volume

The MCRE has a volume of approximately 48,125 ft^3 (1,363 m^3) and is serviced by the CB HVS. To account for equipment and structures, this volume is conservatively reduced by 20% to a free-air volume of 38,500 ft^3 (1,090 m^3) for the analyses.

Unfiltered Intake Flow

In the event of a break outside containment, the CB HVS automatically switches from supplying unfiltered outside air through the CB normal ventilation to emergency filtration of the outside air supply. However, because the MCR habitability is not a Safety Category 1 function, emergency filtration is not credited for mitigation of any DBA. The analyses do not credit MCRE filtration. The analyses model the CB HVS only for the purpose of propagating contamination into the MCR.

The normal CB HVS intake flow rate and emergency CB HVS intake flow rate are the same. The supply of unfiltered outside makeup air that enters the MCRE is at the normal CB HVS flow rate of 1,870 ft^3/min (0.883 m^3/s).

In addition to the CB HVS flow, the potential exists for unfiltered inleakage of outside air to enter the MCRE from ingress and egress of plant personnel or through small openings in the MCRE such as the bathroom fans, penetrations, and small cracks in the walls. No MCRE unfiltered inleakage flow rate is established because the MCR habitability is not a Safety Category 1 function. The primary concern radiologically is that increased MCR unfiltered inleakage could result in the reactor operator being exposed to a larger dose than predicted.

To ensure the analyses conservatively bound unfiltered inleakage, the intake flow of contaminated outside air is assumed to be at an overall rate of one MCRE volume per minute for the duration of the event (720 hours). Thus, an unfiltered flow rate of 38,500 ft^3/min (18.2 m^3/s) is assumed for 30 days, and the MCRE exhaust flow is assumed equal to the intake flow.

This flow rate is a factor of more than 20 times the normal intake flow rate ensuring the limiting MCRE operator dose estimate is established. Use of an MCRE unfiltered air intake rate that bounds the actual CB HVS flow rate ensures the results are applicable to situations where the MCRE intake flow may vary and bounds any potential changes to the CB HVS flow rate that might occur during the preliminary design phase of the facility. In addition, the conservatism in this flow rate assumption bounds any unfiltered inleakage rate. Therefore, the assumed MCRE flow rate established by the design analysis does not require confirmation of the unfiltered inleakage through testing as it represents an open volume with a maximized inflow and exhaust rate.

No Holdup Release to the Environment Flow Rate and Release Duration

All activity is released from line breaks outside containment to the Turbine Building (TB) or RB (for ICLB and ILB events). The analyses are not sensitive to the TB or RB volume because the release is assumed instantaneously. No holdup releases to the environment are assumed with a total of 99.9% transport of the TB or RB airspaces to the environment over a period of one minute after the event at a flow rate of 6.91E+04 cfm (1.95E+03 m^3/min). The flow rate applied ensures an instantaneous release and is applied for the duration of the event (720 hours). For the line breaks outside containment, the radioactivity in the coolant released in each event is transported to the atmosphere instantaneously without credit for plate out, holdup, or dilution within facility buildings.

Dispersion to the Environment

Transport to the environment is simulated using offsite atmospheric dispersion factors (χ/Q values) that are established for the BECPV site. The PAVAN code was applied along with the site-specific data to develop offsite dispersion factors.

Acceptance Criteria

The offsite dose consequences for breaks outside containment are compared to dose limits in 10 CFR 50.34(a)(1)(ii)(D), Table 1 specified in NUREG-0800, SRP 15.0.3, and PDC 19. The offsite dose limit at the EAB and LPZ for a break with a pre-incident spike is 25 rem TEDE. The offsite dose limit of Table 1 from SRP 15.0.3 at the EAB and the LPZ for a break outside containment with equilibrium iodine activity is 2.5 rem TEDE.

The dose consequences for the MCRE are compared to the PDC 19 acceptance criteria of 5 rem TEDE for control room operators.

The analyzed breaks outside containment are discussed in the following subsections.

15.5.9.1.1 Main Steam Line Break Outside Containment

Main steam line break outside containment analysis will be provided in the FSAR.

Postulated Initiating Event

A MSLB accident occurs during power operation in the steam tunnel that is the interface between the RB and TB or in the TB. The postulated event assumes that a MSL instantaneously and circumferentially breaks outside containment and downstream of the outermost MSCIV. The plant is designed to immediately detect such an occurrence and initiate isolation of all MSLs, including the broken line. The energetic release of reactor steam and water from the break is assumed to blowdown into the TB airspace where it is released to the atmosphere instantaneously without credit for plate out, holdup, or dilution within the facility.

Mass Release

The total mass of coolant released is the amount in the line at the time of the break plus the amount that passes through the RIVs and the outboard CIV prior to closure. The masses of coolant and steam released from the postulated MSLB based on preliminary thermal hydraulic analysis are:

- Reactor Water Release = 29,659 lbm (13,453 kg)
- Reactor Steam Release = 13,241 lbm (6,006 kg)

Table 15.5-39 provides the activity release from this coolant volume.

This analysis assumes that the liquid released from the break flashes to steam and is available for transport to the environment along with the steam released from the break. The mass release duration from the MSLB is equal to the maximum closure time of the CIV of 10 seconds. A flashing fraction of 0.4 is conservatively applied to the liquid reactor coolant released as flashing steam.

Dispersion to the Environment

Transport to the EAB and LPZ from the TB is simulated using offsite atmospheric dispersion factors (χ/Q values) that are established for the BECPV site. The offsite atmospheric dispersion factors (s/m3) used in the analysis are:

Offsite χ/Q (s/m^3)

To be provided in the CPA Part 2.

Transport to the MCRE from the TB is simulated using onsite atmospheric dispersion factors χ/Q values calculated using BECPV site meteorological data which meets the intent of RG 1.23 for CPA Part 2, and will be collected in accordance with RG 1.23 for the FSAR.. The most limiting

set of dispersion factors (s/m³) determined for a TB release to the Control Building (CB) HVS intakes are used for a TB area source release to the closest MCR normal intake:

Main Control Room Envelope χ/Q (s/m³)

To be provided in the CPA Part 2.

Results

The results are listed in Table 15.7-5 for comparison to the radiological acceptance criteria.

The MSLB accident dose consequences satisfy the site DBA dose acceptance criteria with considerable margin as shown in Table 15.7-5.

15.5.9.1.2 Large Feedwater Line Break Outside Containment

Large feedwater line break outside containment analysis will be provided in the FSAR.

Flow from the RPV side of the break is bounded by FW breaks inside containment because of the longer pipe length. The higher-pressure losses occur for a break outside containment. No operator actions are required to mitigate the event.

Postulated Initiating Event

An instantaneous circumferential break of a FW line at a location downstream of the high-pressure feedwater heaters and upstream of the outermost CIV is conservatively assumed for the postulated event. The plant is designed to immediately detect such an occurrence and initiate FW line isolation. The energetic release of reactor water from the break is assumed to blowdown into the TB airspace where it is released to the atmosphere instantaneously as a ground-level release without credit for plate out, holdup, or dilution within facility. The mass release duration from the FWLB is equal to the maximum CIV closure time of 10 seconds.

Mass Release

The limiting mass release of coolant and steam from the postulated FWLB are:

- Reactor Water Release = 39,657 lbm (17,988 kg)
- Reactor Steam Release = 3,106 lbm (1,409 kg)

The total FWLB source term activity released to the environment is shown in Table 15.5-40. A flashing fraction of 0.4 is conservatively applied to the liquid reactor coolant released.

Dispersion to the Environment

Transport to the EAB and LPZ from the TB is simulated using offsite atmospheric dispersion factors (χ/Q values) that are established for the BECPV site. The offsite atmospheric dispersion factors (s/m3) used in the analysis are:

Offsite χ/Q (s/m³)

To be provided in the CPA Part 2.

Transport to the MCRE from the TB is simulated using onsite atmospheric dispersion factors χ/Q values calculated using BECPV site meteorological data which meets the intent of RG 1.23 for CPA Part 2, and will be collected in accordance with RG 1.23 for the FSAR.. The most limiting

set of dispersion factors (s/m^3) determined for a TB release to the Control Building (CB) HVS intakes are used for a TB shielded area source release to the closest CB HVS emergency intake:

Main Control Room Envelope χ/Q (s/m^3)

To be provided in the CPA Part 2.

Results

The results are listed in Table 15.7-6 for comparison to the radiological acceptance criteria.

The FWLB accident dose consequences satisfy the site DBA dose acceptance criteria with considerable margin as shown in Table 15.7-6.

15.5.9.1.3 Shutdown Cooling System Line Break Outside Containment

The SDC (Subsection 5.4.7) provides decay heat removal for refueling or maintenance. The SDC provides decay heat removal at normal and lower reactor operating pressures. The SDC is subjected to high energy conditions for a short time (less than 2% of the plant operating conditions). The system piping is assigned a medium energy line due to the short time that it is subjected to high energy conditions.

The SDC connects to the FW system between the FW containment isolation and FW isolation control valve outside containment. Due to the smaller SDC piping diameter, the FW pipe break outside containment discussed previously in Subsection 15.5.9.1.2 bounds the SDC pipe break outside containment with respect to the dose consequence.

15.5.9.1.4 Large Isolation Condenser Line Break Outside Containment

Large isolation condenser line break outside containment analysis will be provided in the FSAR.

The analysis for ICS break outside containment satisfies the Limitation and Condition #2 from NEDC-33911P-A, “BWRX-300 Containment Performance,” (Reference 15.5-18) that requires assessing pipe failures outside containment for closed-loop piping being used on the “other defined basis” for compliance to GDC 55.”

The mass and energy release from the IC pipe breaks outside containment are still bounded by the IC pipe breaks inside containment. Breaks inside containment remain bounding because the IC pipe breaks do not utilize any DL3 functions that depend on containment parameters, and the containment back pressure is not credited in any of the IC pipe breaks inside containment.

As is the case for IC pipe breaks inside containment, core response is not a concern because the break is isolated rapidly. The consequences of large IC pipe breaks outside containment require an evaluation for the loads, pressures, and temperatures outside containment, and radiological consequences resulting from normal operation coolant activity.

Postulated Initiating Event

The ICLB accident analyzed for offsite dose consequences is a postulated break of an ICS steam supply pipe in an IC pool on the operating deck of the RB due to the largest mass release. The plant is designed to immediately detect such an occurrence, initiate isolation of the broken line within 5 seconds, and fully isolate the break in 10 seconds. Blowdown steam from the break is directed to the two heat exchangers in one ICS unit and the liquid mass in the heat exchangers is expelled to an IC pool where it mixes with the pool water without flashing. Scrubbing in an IC pool is conservatively ignored. The mass release duration from the ICLB to the RB is 10 seconds accounting for the 10 sec closure of the CIVs. The energetic release of reactor steam from the

break is assumed released to the environment instantaneously as a ground-level release without holdup.

Mass Release

The limiting mass of steam released from the postulated ICLB is 6,684 lbm (3,032 kg). Table 15.5-41 provides the release source term for the ICLB.

Dispersion to the Environment

Transport to the EAB and LPZ from the BWRX-300 ICS airspace is simulated using offsite atmospheric dispersion factors (χ/Q values) that are established for the BECPV site. The offsite atmospheric dispersion factors (s/m^3) used in the analysis are:

Offsite χ/Q (s/m^3)

To be provided in the CPA Part 2.

Transport to the MCRE from the ICS airspace above the pools in the RB is simulated using onsite atmospheric dispersion factors χ/Q values calculated using data which meets the intent of RG 1.23 for CPA Part 2, and will be collected in accordance with RG 1.23 for the FSAR..

The most limiting set of dispersion factors (s/m^3) determined for an ICS vent release to the Control Building (CB) HVS intakes that are a point source release from the IC pools that conservatively bounds the IC pool vent closest to the MCR intake on the east side of the plant:

Main Control Room Envelope χ/Q (s/m^3)

To be provided in the CPA Part 2.

Results

The results are listed in Table 15.7-7 for comparison to the radiological acceptance criteria.

The ICLB accident dose consequences satisfy the site DBA dose acceptance criteria with considerable margin as shown in Table 15.7-7.

15.5.9.1.5 Small Breaks Outside Containment

Small line break outside containment analysis will be provided in the FSAR.

Because the small break analyses inside containment do not credit containment back pressure, the mass and energy release calculated for breaks inside containment are bounding for breaks outside containment.

Dose analysis is performed for breaks outside containment for the instrument line break.

Instrument Line Break Dose Consequences

Postulated Initiating Event

A circumferential rupture of an instrument line connected to the primary coolant system is postulated to occur outside primary containment in the RB. The ILB analysis assumes that the event cannot be isolated, and no fuel damage occurs. The resulting activity is released to the environment directly from the RB with no credit for holdup or filtration. There is no identified specific event or circumstance that results in the failure of a small line.

Instrument lines penetrating primary containment are required to have an automatically operated CIV, one that can be manually operated from a remote location, or an Excess Flow Check Valve (EFCV) (see Section 6.2 for a description of EFCV usage). In addition, instrument lines penetrating containment are sized or include a flow restricting orifice to ensure that any breach of the line outside containment during power operation reduces leakage to the maximum extent practical and ensures the rate and extent of coolant loss are within the capability of the normal reactor coolant makeup system.

The ILB analysis assumes the ruptured instrument line is equipped with a 0.25 in (6.4 mm) flow restricting orifice and a passively actuated EFCV that is typically configured for use in operating BWRs. For evaluating the dose consequences of a small line rupture outside, the EFCV is conservatively treated as an active component that fails to isolate the break, resulting in radiological release to the environment. The results of this analysis are not applicable to instrument lines that are larger than 0.25 in (6.4 mm) in diameter or do not contain a 0.25 in (6.4 mm) flow restricting orifice.

The ILB leakage may result in noticeable increases in radiation, temperature, humidity, or audible noise levels in the RB or abnormal indications of actuations caused by the break. Termination of the break flow is dependent on operator action. The action is initiated upon the discovery of the un-isolatable leak. The action consists of the orderly shutdown and depressurization of the reactor. Operator action is assumed to occur at 72 hours after the break occurs, after which a controlled shutdown of the reactor is assumed to occur over a 5.2-hour period.

Two cases are considered for coolant conditions that may exist when the postulated accident occurs:

1. The maximum equilibrium iodine concentration permitted for continued full power operations.
2. The iodine concentration corresponding to the conditions of an assumed pre-accident spike.

Mass Release

No credit is taken for operator action for the first 72 hours. After 72 hours, the control room operators begin a controlled shutdown of the plant that takes an additional 5.2 hours. The coolant mass release is 761,007 lbm (345,187 kg) calculated over the duration of the event.

The ILB Airborne Release Source Term for equilibrium and iodine spike is provided in Table 15.5-42a and Table 15.5-42b, respectively. Only the equilibrium concentrations of iodine and iodine spike activity releases differ.

Flashing of Reactor Coolant to Steam

The total iodine fraction in the liquid that becomes airborne assumed equal to the leakage fraction that flashes to vapor (flash fraction (FF)) is determined using a constant enthalpy process:

$$FF = \frac{h_{f1} - h_{f2}}{h_{fg}}$$

Where:

h_{f1} = Enthalpy of coolant liquid release at different pressure and temperature conditions during shutdown

h_{f2} = Enthalpy at the heat of vaporization at atmospheric pressure = 180.1 BTU/lbm (418.9 kJ/kg)

h_{fg} = Enthalpy at the heat of vaporization at 100 °C (212 °F) = 970.2 BTU/lbm (2,256.7 kJ/kg)

NEDO-32708, Figure 7-2 provides the reactor pressure during shutdown for the whole duration of the ILB event that is used to calculate the enthalpy of the saturated liquid (h_{f1}) at different times during the shutdown. The mass of coolant that flashes to steam at 72 hours is 279,745 lbm (126,890 kg). The mass of coolant that flashes to steam calculated over the duration of the whole event is 286,627 lbm (130,012 kg). The ILB water concentrations and the mass of coolant flashed are used to calculate the airborne activity released.

Release Duration

Operator action is assumed to occur at 72 hours. After 72 hours, the reactor is taken through a controlled shutdown where it is assumed that the instrument line cannot be isolated and primary coolant continues to flow through the ILB. The controlled shutdown process is assumed to occur over a 5.2-hour period. The coolant mass release occurs over a 77.2-hour period (i.e., 72 hour operator action delay + 5.2-hour controlled shutdown), approximately.

Dispersion to the Environment

Transport to the EAB and LPZ from the RB is simulated using offsite atmospheric dispersion factors (χ/Q values) that are established for the BECPV site. The offsite atmospheric dispersion factors (s/m^3) used in the analysis are:

Offsite χ/Q (s/m^3)

To be provided in the CPA Part 2.

Transport to the MCRE from the BWRX-300 RB is simulated using onsite atmospheric dispersion factors (χ/Qs) calculated using BECPV site meteorological data which meets the intent of RG 1.23 for CPA Part 2, and will be collected in accordance with RG 1.23 for the FSAR.

Several sets of dispersion factors were determined for potential RB release points for the ILB accident to the CB HVS intakes. The most limiting set of dispersion factors determined for a diffuse source release from the RB to the CB HVS intakes are used in this analysis:

MCRE Atmospheric Dispersion Factors (s/m^3)

To be provided in the CPA Part 2.

Results

The BWRX-300 ILB dose consequences satisfy the site DBA dose acceptance criteria as shown in Table 15.7-8. The integral activity releases from the RB to the environment are provided including equilibrium iodine and iodine spike source terms. The integral activity releases are reported up to 78 hours to cover the entire 77.2-hour coolant mass release period.

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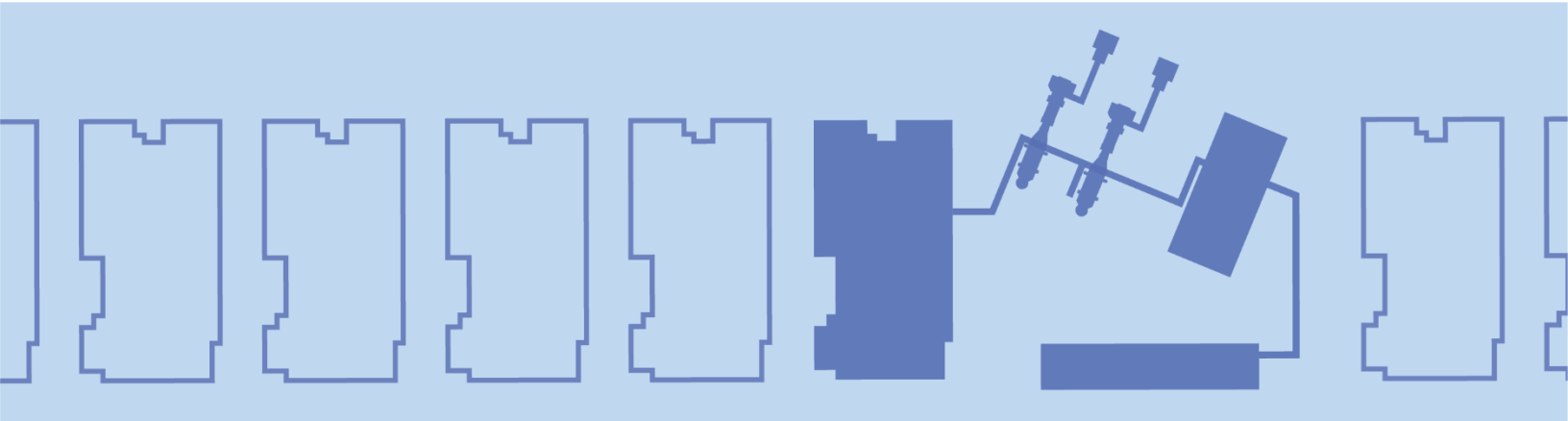
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Non-Proprietary Information

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Table 15.5-1: Key Initial Conservative LOCA Evaluations

Parameter	Value	Notes
Thermal Power	887.4 MW	102% of rated power. Hot shutdown initial power is also included in steam pipe break cases
Dome pressure	7308.4 kPa	Upper end of normal operating range
Initial feedwater temperature	241.9 °C for MSL Break	Initial temperature of 191.9 °C is also included in FW pipe break cases
Initial temperature of the ICS and reactor cavity pools	43.3 °C	
Initial containment pressure	119.7 kPa	Upper end of normal operating range
Initial containment temperature	43.3 °C	Lower end of normal operating range
Initial water level in downcomer	21.1 m (large breaks)	Lower initial level (0.152 m below normal level) is also evaluated for small breaks

Table 15.5-2: Summary of Core-Wide Decay Ratio Results

Cycle Exposure (GWd/ST)	Core-Wide Decay Ratio	
	Nominal (241.9 °C)	LFWH to 191.9 °C with SCRRI mitigation (AOO Conditions)
0 (BOC)	0.56	0.53
4 (MOC)	0.71	0.57
6 (EOR)	0.64	0.56

Table 15.5-3: Input Parameters and Initial Conditions and Assumptions Used in Non-LOCA Analyses

Parameter	Value
Heat Balance Related Parameters	
Rated thermal power level, MWt	870
Core flow, Mlbm/hr (kg/s) – analysis value	Calculated
Reference rated core flow, Mlbm/hr (kg/s)	15.0 (1890.0)
Steam flow, Mlbm/hr (kg/s) – analysis value	Calculated
Reference rated steam flow, kg/s	503.2
Feedwater (FW) flow, Mlbm/hr (kg/s) – analysis value	Calculated
Reference rated FW flow, kg/s	507.0
Nominal dome pressure, psia (MPa)	1040 (7.171)
Nominal FW temperature, °F	467
Normal Reactor Water Level (NWL), m	21.097 m from inside bottom of the RPV 12.22 m above TAF
Control Rod Drive / High Pressure Injection System Related Parameters	
Control rod position versus time	Table 15.5-4
Control Rod Drive Mechanism (CRDM) run-in fast speed (minimum requirement), mm/s	70
CRDM withdrawal (maximum speed), mm/s	28.0
Main Steam Line (MSL) Related Parameters	
Number of MSLs	2
Minimum MSL length (average of all lines): flow path from vessel to Turbine Stop Valve (TSV), ft (m)	194 (59.0)
Minimum MSL volume (total of all lines including header and piping to TBV), cubic ft (cubic m)	885.3 (25.07)
Minimum MSL pressure difference between the vessel dome pressure and the turbine throttle pressure at rated conditions, psi (kPa)	30.0 (207)
MSRIV minimum closure time, s	3.0
MSRIV closure profile for minimum closure time, s 100% open area 100% open area 1% open area 0% open area	 0.0 0.6 1.7 3.0
Feedwater Related Parameters	
Number of motor-driven FW pumps (One FW pump is operating, and one FW pump is in standby during power operation)	2

Parameter	Value
Maximum flow demanded is between, % Rated	105 - 115
Maximum flow of both FW pumps assuming a CCF, % Rated	220
FW temperature reduction for Loss of Feedwater Heating (LFWH) AOO, ° F (°C)	90 (50)
Minimum time constant of FW temperature response, s	60
FW pump trip coast down time constant, s	3
Isolation Condenser System (ICS) Related Parameters	
ICS heat removal capacity per train, MW	33.75
ICS return liquid and IC pool initial temperature, K	294
ICS condensate return valve maximum opening time, s	10
ICS train condensate return line internal diameter, mm	177.9
The elevation difference from centerline of the horizontal connections between the ICS steam line distribution header to the centerline of the ICS return injection to the chimney, minimum, in (m)	622.362 (15.808)
Balance of Plant Related Parameters	
TSV and TCV maximum fast closure steam flow shutoff rate from rated power, % rated steam flow / s	667
TCV maximum slow (servo) closure rate of change of steam flow shutoff from rated power % rated steam flow / s	40
TBV capacity at rated conditions % rated steam flow	25
Miscellaneous Parameters	
Biased initial CPR, (reduction versus nominal initial CPR) Note: the value is approximate because it is slightly different depending on the initial condition. This is used in CN-DBA events and some DEC events (as noted in event description)	~0.1

Table 15.5-4: CRD Scram Time

Reactor Vessel Bottom Gauge Pressure psig (MPaG)	Rod Insertion Position	Required Maximum Time (s) Note 1
≤ 1085 psig (7.48 MPaG)	10%	≤ 0.42
	40%	≤ 1.00
	60%	≤ 1.44
	100%	≤ 2.80
≤ 1269 psig (8.75 MPaG)	10%	≤ 0.46
	40%	≤ 1.20
	60%	≤ 1.71
	100%	≤ 3.70
≤ 1375 psig (9.48 MPaG) Note 2	10%	≤ 0.56
	40%	≤ 1.40
	60%	≤ 2.03
	100%	≤ 4.20

Notes:

1. The times include 0.2 s delay from de-energizing the scram pilot valves to control rods movement.
2. These times are used in pressurization increase events in the DBA category even if the pressure is not greater than 1269 psig (8.75 MPaG) during the rod insertion.

Table 15.5-5: Defense Line Functions Used in Non-LOCA Analyses

DL ID	Function Name	Implementation	Inputs	Setpoint / Delay Analytical Limits
DL2-01	Maintain Target Pressure (Performed by RPC)	RPC described in Subsection 7.3.3.2, Item 5	Steam flow demand is a function of reactor dome pressure	Target dome pressure is normal dome pressure in Table 15.5-3
DL2-02	Maintain Target Level (Performed by RLC)	RLC is described in Subsection 7.3.3.2, Item 3	RPV level error (function of core power, reactor water level, steam flow, steam flow enthalpy, FW enthalpy)	Target reactor level is NWL in Table 15.5-3
DL2-04	Control Rod Block on ATLM	ATLM described in described in Subsection 7.3.3.2, Item 19	Determination of the thermal limits or soft duty guidelines violation	Block applied prior to exceeding normal operating conditions as described in Subsection 7.3.3.2, Item 19
DL2-05	Control Rod Block on MRBM	MRBM described in Subsection 7.3.3.2, Item 20	Indication of potential fuel damage thermal limits being exceeded	Block applied prior to exceeding normal operating conditions as described in Subsection 7.3.3.2, Item 20
DL2-08	Anticipatory Hydraulic Scram on Turbine Trip or Generator Load Rejection Demand	ATS described in Subsection 7.3.3.2, Item 1	Turbine Trip or Generator Load Rejection Demand	Total time delay = 0.05 s after generator load rejection or turbine trip signal
DL2-09	TBV Fast Open on Turbine Trip or Generator Load Rejection Demand	ATS described in Subsection 7.3.3.2, Item 1	Turbine Trip or Generator Load Rejection Demand	Before or at the same time as TCV or TSV closure
DL2-13	Turbine Trip on High Main Condenser Pressure Setpoint 2	ATS described in Subsection 7.3.3.2, Item 1	High Main Condenser Pressure (HMCP) Setpoint 2	HMCP2 no greater than 30.5 kPaA Total time delay = 1 s
DL2-14	TBV Closure on High Main Condenser Pressure Setpoint 3	ATS described in Subsection 7.3.3.2, Item 1	High Main Condenser Pressure Setpoint 3	HMCP3 no greater than 71.1 kPaA Total time delay = 1 s
DL2-21	Anticipatory Hydraulic Scram on MSRIV/MSCIV Position	ATS described in Subsection 7.3.3.2, Item 1	MSRIV/MSCIV Position	<90% open Total time delay (scram) = 0.05 s
DL2-25	Start Standby FW Pump on Loss of Operating FW Pump	CFS Standby FW Start is described in Subsection 10.4.7.2	Loss of Operating FW Pump	Analysis allows 10 seconds to start standby pump. Analysis allows 15 seconds to ramp up the standby pump to full flow
DL2-27	Select Control Rod Run-In on FW Temperature Decrease	RC&IS described in Subsection 7.3.3.2, Item 8	FW Temperature Decrease	FW temperature reduction of 30°F (16.6°C) or more. Total time delay = 5 s

DL ID	Function Name	Implementation	Inputs	Setpoint / Delay Analytical Limits
DL2-31	ICS Initiation on High Reactor Pressure	DPS described in Subsection 7.3.2.2, Item 1	High Reactor Pressure	Settings are not important for Transient DSA results (not simulated)
DL2-43	FW Check Valve Closure on Reverse FW Flow	CFS reverse flow prevention is described in Subsection 10.4.7.1	Reverse FW Flow	No setpoint/analytical limit
DL3-01	Hydraulic Scram on High RPV Pressure	SC1 I&C described in Subsection 7.3.1.2, Item 1	High RPV Pressure (HP1)	1129.4 psig (7.787 MPaG) Total time delay = 0.7 s
DL3-02	Hydraulic Scram on Low RPV Pressure	SC1 I&C described in Subsection 7.3.1.2, Item 1	Low RPV Pressure (LP1)	800.0 psig (5.516 MPaG) Total time delay = 0.7 s
DL3-03	Hydraulic Scram on Low RPV Level	SC1 I&C described in Subsection 7.3.1.2, Item 1	Low RPV Level	773.425 in (19.645 m) from inside bottom of RPV Total time delay = 1.0 s
DL3-04	Hydraulic Scram on High Neutron Flux	SC1 I&C described in Subsection 7.3.1.2, Item 1	High Neutron Flux	125% of rated reactor power Total time delay = 0.09 s
DL3-05	Hydraulic Scram on High Simulated Thermal Power	SC1 I&C described in Subsection 7.3.1.2, Item 1	High Simulated Thermal Power	115% of rated thermal power Signal Time Constant = 7 s Total time delay = 0.09 s
DL3-11	ICS Train A Initiation on High RPV Pressure	SC1 I&C described in Subsection 7.3.1.2, Item 4	High RPV Pressure (HP2)	1202.2 psig (8.289 MPaG) Total time delay = 0.7 s
DL3-12	ICS Train B Initiation on High RPV Pressure	SC1 I&C described in Subsection 7.3.1.2, Item 4	High RPV Pressure (HP3)	1233.4 psig (8.504 MPaG) Total time delay = 0.7 s
DL3-13	ICS Train C Initiation on High RPV Pressure	SC1 I&C described in Subsection 7.3.1.2, Item 4	High RPV Pressure (HP4)	1264.6 psig (8.719 MPaG) Total time delay = 0.7 s
DL3-14	ICS Initiation on Low RPV Water Level	SC1 I&C described in Subsection 7.3.1.2, Item 4	Low RPV Water Level	560.0 in (14.224 m) from inside bottom of RPV Total time delay = 1.0 s
DL3-17	MSRIV/MSCIV Isolation on Low RPV Pressure	SC1 I&C described in Subsection 7.3.1.2, Item 3	Low RPV Pressure (LP1)	800.0 psig (5.516 MPaG) Total time delay = 0.7 s
DL3-23	FW Isolation on High RPV Water Level	SC1 I&C described in Subsection 7.3.1.2, Item 3	High RPV Water level	880.984 in (22.377 m) from inside bottom of RPV Total time delay = 1.0 s
DL3-39	FW Isolation on Loss of Normal FW Flow	SC1 I&C described in Subsection 7.3.1.2, Item 3	Sustained Loss of Normal FW Flow	30 seconds is sufficient to allow standby pump to start but short enough to limit RPV inventory losses

DL ID	Function Name	Implementation	Inputs	Setpoint / Delay Analytical Limits
DL4a-12	MSRIV/MSCIV Isolation on Sustained Low FW Flow	DPS described in Subsection 7.3.2.2, Item 1	Sustained Low FW flow	Analysis allows 70 seconds to confirm sustained low FW flow
DL4a-40	CRD Fast Motor Run-In on Control Rods Not Full-In Signal	DPS described in Subsection 7.3.2.2, Item 1	High Flux After Scram Signal	Total time delay = 5 s
DL4a-41	FW Pump/Condensate Pump Trip on Control Rods Not Full-In Signal	DPS described in Subsection 7.3.2.2, Item 1	High Flux After Scram Signal	Total time delay = 5 s
DL4a-42	ICS Trains 1, 2, and 3 Initiations on High Flux After Scram Signal	DPS described in Subsection 7.3.2.2, Item 1	High Flux After Scram Signal	Total time delay = 5 s

Table 15.5-6: Sequence of Events for Loss of Feedwater Heating (AOO)

Time (s)	Event
0	Initiate a 90°F (50°C) temperature reduction in the FW system
5.5	SCRRI inserts control rods on indication of FW temperature reduction
~60	Control rods stop moving
~400	New steady state achieved

Table 15.5-7: Sequence of Events for Turbine Trip (AOO)

Time (s)	Events
0.0	Reactor scrams on initiation on turbine trip signal
0.2	Control rods begin to move
0.25	TSVs begin to close
0.25	TBVs begin to open
0.40	TSVs are closed
>6.0	New steady state

Table 15.5-8: Sequence of Events for Closure of One Main Steam Reactor Isolation Valve (AOO)

Time (s)	Event
0.0	Initiate closure of one MSRIV
0.8	Anticipatory scram on MSRIV position
3.0	MSRIV in first steam line is closed
3.0	Closure of MSRIV in second steam line initiated on leak detection indication
6.0	MSRIV in second steam line is closed
>25.0	High RPV pressure reached, ICS train initiated (not simulated)
>25.0	New Steady state

Table 15.5-9: Sequence of Events for Loss of Condenser Vacuum (AOO)

Time (s)	Event
0.0	Anticipatory scram initiated on turbine trip signal caused by loss of condenser vacuum
0.2	Control rods begin to move
0.25	TSVs begin to close
0.25	TBVs begin to open
0.40	TSVs are closed
24.0	TBVs close on high main condenser pressure (not simulated)
>24.0	ICS initiation on high RPV pressure (not simulated)
>24.0	New steady state

Table 15.5-10: Sequence of Events for Loss-of-Preferred Power (AOO)

Time (s)	Event
0.0	FW pumps lose power
0.0	Anticipatory scram initiated on generator load rejection caused by generator output breakers opening on loss of power
0.2	Control rods begin to move
0.25	TCVs begin to close due to generator load rejection signal
0.25	TBVs begin to open
6.25	TBVs close following loss-of-preferred power
>20.0	ICS initiation on high RPV pressure (not simulated)
>20.0	New steady state

Table 15.5-11: Sequence of Events for Feedwater Pump Trip (AOO)

Time (s)	Event
0.0	Initiate FW pump trip
10.0	Standby FW pump starts
25.0	Standby FW pump at 100% rated FW flow
~200	New steady state achieved near 100% power and 100% FW flow

Table 15.5-12: Sequence of Events for Inadvertent Isolation Condenser Initiation One Train (AOO)

Time (s)	Event
0	Initiate opening of IC condensate return valve on one train
>150	New steady state achieved near initial conditions

Table 15.5-13: Sequence of Events for Loss of Feedwater Heating (DBA)

Time (s)	Event
0	Initiate FW temperature reduction
~90	High STP reached, scram initiated
~110	Low RPV pressure reached, Main Steam isolation initiated
~300	High RPV level reached, FW isolation initiated
> 400	High RPV pressure reached; ICS train initiated (not simulated)

Table 15.5-14: Sequence of Events for Generator Load Rejection (DBA)

Time (s)	Event
0.0	TCVs begin to close due to a generator load rejection signal
~0.0	TBVs begin to open
0.42	High neutron flux reached, Scram initiated
0.62	Control rods begin to move
152	High RPV pressure reached, ICS train initiated
166	High RPV level reached, FW isolation initiated
>300	A controlled state is achieved

Table 15.5-15: Sequence of Loss-of-Preferred Power (DBA)

Time (s)	Event
0.0	TCV begin to close, and FW pump trips
0.80	High neutron flux reached; Scram initiated
1.0	Control rods begin to move
152	High RPV pressure reached; ICS train initiated
>300	A controlled state is achieved

Table 15.5-16: RPV Pressure Control Downscale (DBA)

Time (s)	Event
0.0	TCVs begin to close, and TBVs remain closed
0.80	High neutron flux reached, Scram initiated
1.0	Control rods begin to move
155	High RPV pressure is reached, ICS train initiated
167	High RPV level reached, FW isolation initiated
>300	A controlled state is achieved

Table 15.5-17: Closure of All MSRIVs and FW Isolation Valves (DBA)

Time (s)	Event
0.0	MSRIVs begin to close, and FW isolation valves close
1.09	High Neutron Flux reached, Scram initiated
1.29	Control rods begin to move
122	High RPV pressure reached, ICS initiated
>300	A controlled state is achieved

Table 15.5-18: Sequence of Events for Feedwater Flow Increase – All Pumps

Time (s)	Event
0	Initiate instant increase in speed of both FW pumps
~20	High RPV level reached, FW isolation initiated
~30	High STP reached, Scram initiated
~90	Low RPV pressure reached, Main Steam Reactor Pressure Isolation Valve isolation initiated
>200	High RPV pressure reached, ICS train initiated (not simulated)

Table 15.5-19: Sequence of Events for Condenser Initiation – All Trains

Time (s)	Events
0	Initiate opening of all IC condensate return valves
~35	High RPV level reached, FW isolation initiated
~55	Low RPV pressure reached, scram, and MSRIV isolation initiated

Table 15.5-20: Sequence of Events for Loss of Feedwater (DBA)

Time (s)	Event
0.0	Initiate loss of FW flow
~30	Low RPV level reached, scram initiated
~85	Low RPV pressure reached, MSRIV isolation initiated
~130	Low RPV level reached, all IC Trains initiate
>300	New steady state achieved

Table 15.5-21: Sequence of Events for RPV Pressure Control Open (DBA)

Time (s)	Event
0.0	TBVs and TCVs open
~70	Low RPV pressure reached, reactor scram and MSRIV isolation initiated
~225	High RPV level reached, FW isolation initiated
>300	High RPV pressure reached, ICS train initiated (not simulated)

Table 15.5-22: Timing of Events for Main Steam Pipe Break Inside Containment, CN-DSA

Time (seconds)	Event	Notes
0.0	Double-ended guillotine rupture of main steam pipe break inside the containment concurrent with LOPP	
0.0	FW pump trip and coast down	This is a consequence of LOPP
0.0	TSV or TCV starts closing rapidly	Conservative assumption
1.0	Control rods start to insert on scram initiation	High containment pressure setpoint for scram, reactor isolation and isolation condenser initiation is reached in less than 1 second.
1.0	ICS condensate return valve starts opening	
1.0	CUW stops	
3.0	Control rods are inserted sufficiently to diminish fission from prompt neutrons	This is a conservatively long duration for fission from prompt neutrons to diminish. Fission power starts decreasing when the control rods are partially inserted. In addition, voiding in the core due to rapid depressurization also causes reactor power to decrease rapidly.
5.0	RIVs start to close	The delay time is significantly conservative given that containment pressure reaches the isolation setpoint in less than 1 second.
10.0	RIVs are fully closed	
11.0	ICS condensate return valve is fully open	
12.2	Peak containment pressure is reached	
>12.2	Containment pressure starts decreasing	

Table 15.5-23: Timing of Events for Feedwater Pipe Break Inside Containment, CN-DSA

Time (seconds)	Event	Notes
0.0	Double-ended guillotine rupture of FW pipe break inside containment concurrent with LOPP	
0.0	FW pump trip and coast down	This is a consequence of LOPP
0.0	TSV or TCV starts closing rapidly	Conservative assumption
1.0	Control rods start being inserted on scram initiation	High containment pressure setpoint for scram, reactor isolation and isolation condenser initiation is reached in less than 1 second
1.0	ICS-A and B condensate return valves start opening	
3.0	Control rods are inserted sufficiently to diminish fission from prompt neutrons	This is a conservatively long duration for fission from prompt neutrons
5.0	FW RIVs and CIVs start to close	
10.0	FW RIVs and CIVs are fully closed	
10.1	Peak containment pressure is reached	
11.0	Isolation condenser valves are fully open	

Table 15.5-24: Timing of Events for Small Steam Pipe Break Inside Containment, CN-DSA

Time (seconds)	Event	Notes
0.0	Small steam pipe break concurrent with LOPP	
0.0	Pressure controller freezes. TCV remains in initial position, turbine chest pressure decreases rapidly	Assuming continued steam discharge to the turbine is conservative. TCV closure as a consequence of LOPP is not credited.
0.0	FW pump trip and coast down	
10.6	Steam line low pressure setpoint is reached	
12.3	Reactor scram, 0.7 s delay after low pressure setpoint is reached and another 1 s for scram delay	
15.6	MSRIVs start closing 5 seconds after steam line low pressure	
20.6	MSRIVs are fully closed	
63.6	Level decreases to Level 2	
64.6	ICS-A and ICS-B condensate return valves start opening	
74.6	ICS-A and ICS-B condensate return valves are fully open	
234000	Peak containment pressure reached	

Table 15.5-25: Timing of Events for Small Liquid Pipe Break Inside Containment, CN-DSA

Time (seconds)	Event	Notes
0.0	Small liquid pipe break concurrent with LOPP	
0.0	Pressure controller freezes. TCV remains at initial position, turbine chest pressure decreases rapidly	Assuming continued steam discharge to the turbine is conservative. TCV closure as a consequence of LOPP is not credited.
0.0	FW pump trip and coast down	
10.6	Steam line low pressure setpoint is reached	
12.3	Reactor scram, 0.7 s delay after low pressure setpoint is reached and another 1 s for scram delay	
15.6	MSRIVs start closing 5 seconds after steam pipe low pressure	
20.6	MSRIVs are fully closed	
58.9	Level decreases to Level 2	
59.9	ICS-A and ICS-B condensate return valves start opening	
69.9	ICS-A and ICS-B condensate return valves are fully open	
232800	Peak containment pressure occurs	

Table 15.5-26: Sequence of Events for Closure of One Main Steam Reactor Isolation Valve (DEC)

Time (s)	Event
0.0	Initiate closure of one MSRIV
0.8	Anticipatory scram signal on MSRIV position and scram fails
3.0	MSRIV in first steam line is closed
3.0	Closure of MSRIV in second steam line initiated on leak detection indication
5.8	Feedwater pump trips on high flux after scram signal
5.8	All ICS trains initiate on high flux after scram signal
5.8	CRDM run-in initiation on high flux after scram signal
6.0	MSRIV in second steam line is closed
>300	A controlled state is achieved

Table 15.5-27: Sequence of Events for Complex Sequence Generator Load Rejection (DEC)

Time (s)	Event
0.0	Scram initiation on load rejection signal Half of the control rods fail to insert
0.2	Control rods begin to move
0.25	TCVs begin to close
0.25	TBVs begin to open
>200	A controlled state is achieved

Table 15.5-28: Sequence of Events for Loss of Condenser Vacuum (DEC)

Time (s)	Event
0.0	Scram fails on initiation on turbine trip signal caused by loss of condenser vacuum
0.25	TSVs begin to close
0.25	TBVs begin to open
0.40	TSVs are closed
5.0	CRDM run-in initiation on high flux after scram signal
5.0	ICS initiation on high flux after scram signal
5.0	Feedwater pump trips on high flux after scram signal
24.0	TBVs close on high main condenser pressure
>300	A controlled state is achieved

Table 15.5-29: Sequence of Events for Loss of Preferred Power (DEC)

Time (s)	Event
0.0	FW pumps lose power
0.0	Scram fails on generator load rejection caused by generator output breakers opening on loss of power
0.25	TCVs begin to close due to generator load rejection signal
0.25	TBVs begin to open
5.0	CRDM run-in initiation on high flux after scram signal
5.0	ICS initiation on high flux after scram signal
6.25	TBVs close following loss-of-preferred power
>300	A controlled state is achieved

Table 15.5-30: Sequence of Events for All Control Rod Withdrawal at Power (ACRW)

Time (s)	Event
0.1	Initiate withdrawal of all rods
19	High STP reached, scram initiated
63	High RPV Level reached, FW isolation initiated
84	Low RPV pressure reached, Main Steam isolation initiated
> 200	High RPV pressure reached; ICS train initiates on high pressure (not simulated)

Table 15.5-31: Sequence of Events for Inadvertent Control Rod Withdrawal at Power Single Rod (ICRW)

Time (s)	Event
0.1	Initiate withdrawal of one rod
>70	Power reaches a stable level
long term	Operators act to control reactor power

Table 15.5-32: Sequence of Events for Feedwater Isolation (DEC)

Time (s)	Event
0.0	Initiate FW Isolation
5	FW flow at zero
~30	Low RPV level reached, no scram (DL3 failure)
65	Reactor scram initiated on sustained low FW flow
70	MSRIV isolation initiated on sustained low FW flow
250	ICS Trains initiate on high RPV pressure
>400	Controlled state achieved

Table 15.5-33: Fuel Handling Accident Sequence of Events

Sequence of Events	Elapsed Time
The BWRX-300 reactor is shut down for refueling operations that begins 24 hours after shutdown. Over this period the core isotopic inventory decays.	24 hours
During a refueling operation a fuel assembly is moved over the top of the core or fuel pool, and the fuel bundle, grapple, mast, and head fall on top of the core or spent fuel racks.	0
Rods in the dropped bundle and impacted bundles fail, releasing the fission gases and cesium iodide in the plenum and gap of the damaged rods to the pool water.	0
Fission gases rise through the pool water to refueling operation floor common airspace surrounding the top of the reactor cavity and fuel pool.	0
The fission gases initially released to the refueling floor airspace are released to the environment.	2 hours
The cesium iodide initially released to the pool evolves to form elemental iodine. Given the chemical conditions in the pool, a portion of the elemental iodine becomes volatile and releases to the refueling floor airspace and subsequently to the environment. This phase begins 2 hours after the event.	718 hours

Table 15.5-34: BWRX-300 Core Parameters

Parameter/Description	Value
Thermal Power	870 MWth
Core Size (Number of Bundles)	240
Fuel Type	GNF2
Bundle Average Enrichment	3.84-4.68 w/o U235
Total Core Uranium Mass	44.79 Mt (49.37 St)
Core Average Exposure	38,000 MWd/Mt (41,888 MWd/St)

Table 15.5-35: BWRX-300 Core Inventory 24 Hours After Shutdown

Nuclide	Activity (Ci/MWth)	Activity (MBq/MWth)	Nuclide	Activity (Ci/MWth)	Activity (MBq/MWth)
Co-58	3.08E+02	1.14E+07	Te-131m	2.31E+03	8.55E+07
Co-60	6.51E+02	2.41E+07	Te-132	3.13E+04	1.16E+09
Kr-85	5.40E+02	2.00E+07	I-131	2.54E+04	9.40E+08
Kr-85m	1.78E+02	6.59E+06	I-132	3.23E+04	1.20E+09
Kr-87	2.91E-02	1.08E+03	I-133	2.54E+04	9.40E+08
Kr-88	5.56E+01	2.06E+06	I-134	1.36E-03	5.03E+01
Rb-86	7.40E+01	2.74E+06	I-135	4.16E+03	1.54E+08
Sr-89	2.56E+04	9.47E+08	Xe-133	5.16E+04	1.91E+09
Sr-90	4.48E+03	1.66E+08	Xe-135	1.49E+04	5.51E+08
Sr-91	5.70E+03	2.11E+08	Cs-134	9.46E+03	3.50E+08
Sr-92	7.64E+01	2.83E+06	Cs-136	2.60E+03	9.62E+07
Y-90	4.62E+03	1.71E+08	Cs-137	5.97E+03	2.21E+08
Y-91	3.32E+04	1.23E+09	Ba-139	3.20E-01	1.18E+04
Y-92	1.13E+03	4.18E+07	Ba-140	4.51E+04	1.67E+09
Y-93	8.00E+03	2.96E+08	La-140	4.88E+04	1.81E+09
Zr-95	4.73E+04	1.75E+09	La-141	7.08E+02	2.62E+07
Zr-97	1.84E+04	6.81E+08	La-142	1.03E+00	3.81E+04
Nb-95	4.80E+04	1.78E+09	Ce-141	4.44E+04	1.64E+09
Mo-99	4.00E+04	1.48E+09	Ce-143	2.55E+04	9.44E+08
Tc-99m	3.81E+04	1.41E+09	Ce-144	3.82E+04	1.41E+09
Ru-103	4.33E+04	1.60E+09	Pr-143	4.02E+04	1.49E+09
Ru-105	7.65E+02	2.83E+07	Nd-147	1.70E+04	6.29E+08
Ru-106	2.05E+04	7.59E+08	Np-239	4.53E+05	1.68E+10
Rh-105	2.15E+04	7.96E+08	Pu-238	2.28E+02	8.44E+06
Sb-127	2.63E+03	9.73E+07	Pu-239	1.87E+01	6.92E+05
Sb-129	1.95E+02	7.22E+06	Pu-240	2.70E+01	9.99E+05
Te-127	2.87E+03	1.06E+08	Pu-241	7.83E+03	2.90E+08
Te-127m	4.19E+02	1.55E+07	Am-241	1.47E+01	5.44E+05
Te-129	1.08E+03	4.00E+07	Cm-242	3.03E+03	1.12E+08
Te-129m	1.31E+03	4.85E+07	Cm-244	2.15E+02	7.96E+06

Table 15.5-36: Non-LOCA Fraction of Fission Product Inventory in Gap

Group	Gap Fraction
I-131	0.08
Kr-85	0.10
Other Noble Gases	0.05
Other Halogens	0.05
Alkali Metals	0.12

Table 15.5-37: BWRX-300 FHA Activity Released from Fuel

Nuclide	Activity Released from Fuel (Ci)	Activity Released from Fuel (MBq)
Kr-85m	8.43E+01	3.12E+06
Kr-85	5.12E+02	1.89E+07
Kr-87	1.38E-02	5.10E+02
Kr-88	2.63E+01	9.74E+05
I-131	1.92E+04	7.12E+08
I-132	1.53E+04	5.66E+08
I-133	1.20E+04	4.45E+08
I-134	6.44E-04	2.38E+01
I-135	1.97E+03	7.29E+07
Xe-133	2.44E+04	9.04E+08
Xe-135	7.06E+03	2.61E+08

Table 15.5-38a: BWRX-300 Fuel Handling Accident (FHA) Phase 1 Activity Released from the Reactor Cavity Pool

Nuclide	Activity Released from Fuel (Ci)	Activity Released from Fuel (MBq)	Phase 1 Decontamination Factor (DF)	Activity Released from the Pool (Ci)	Activity Released from the Pool (MBq)
Kr-85m	8.43E+01	3.12E+06	1	8.43E+01	3.12E+06
Kr-85	5.12E+02	1.89E+07	1	5.12E+02	1.89E+07
Kr-87	1.38E-02	5.10E+02	1	1.38E-02	5.10E+02
Kr-88	2.63E+01	9.74E+05	1	2.63E+01	9.74E+05
I-131	9.62E+02	3.56E+07	490	1.96E+00	7.27E+04
I-132	7.65E+02	2.83E+07	490	1.56E+00	5.78E+04
I-133	6.02E+02	2.23E+07	490	1.23E+00	4.54E+04
I-134	3.22E-05	1.19E+00	490	6.57E-08	2.43E-03
I-135	9.85E+01	3.65E+06	490	2.01E-01	7.44E+03
Xe-133	2.44E+04	9.04E+08	1	2.44E+04	9.04E+08
Xe-135	7.06E+03	2.61E+08	1	7.06E+03	2.61E+08

Table 15.5-38b: BWRX-300 FHA Phase 2 Activity Available for Release from the Reactor Cavity Pool

Nuclide	Activity Available for Release from the Pool (Ci)	Activity Available for Release from the Pool (MBq)
I-131	1.92E+04	7.12E+08
I-132	1.53E+04	5.66E+08
I-133	1.20E+04	4.45E+08
I-134	6.44E-04	2.38E+01
I-135	1.97E+03	7.29E+07

Table 15.5-38c: Key Parameters for DBA FHA Analysis

I. Data and Assumptions Used to Estimate Source Terms	
A. Power level, MWth	870
B. Core Radionuclide Inventory	Table 15.5-35
C. Plenum Activity	
Radioactivity for I-131, %	8
Radioactivity for Kr-85, %	10
Radioactivity for other noble gases, %	5
Radioactivity for other halogens, %	5
Radioactivity for alkali metals, %	12
D. Radial peaking factor for damaged rods	1.7
E. Duration of release, hr	
Phase 1, initial gaseous release	2
Phase 2, pool iodine re-evolution	718[1]
F. Total Number of Bundles in Core	240
G. Number of damaged bundles	1.507
H. Minimum time after shutdown to accident, hr	24
I. Average fuel exposure, MWd/MT (MWd/ST)	38,000 (41,888)
II. Data and Assumptions Used to Estimate Activity Released	
A. Species fraction, in percentage form released from fuel	
Organic iodine, %	0.15
Elemental iodine, %	4.85
Particulate iodine, %	95
Noble gas, %	100
Phase 1 Reactor Building/Fuel Building Atmosphere	
Organic iodine, %	3
Elemental iodine, %	97
Particulate iodine, %	0
Noble gas, %	100
Phase 2 Reactor Building/Fuel Building Atmosphere	
Organic iodine, %	0
Elemental iodine, %	100
Particulate iodine, %	0
Noble gas, %	0
B. Pool Water Level above the core, m (ft)	≥7.01 (23.0)
Pool Volume (L)	770,000

C. Pool Retention decontamination factor	
Phase 1 Total Iodine	490
Fuel Rod Pressure, psig	1200
Bubble Rise Time, sec	4.491
Bubble diameter, cm	0.761
Phase 1 Noble gas	1
Phase 2 Iodine	1
Phase 1 and 2 Particulates	Infinite
D. Release rate, %/hr	
Phase 1, 0 to 2 hours	345
Phase 2, 2 to 720 hours	8.33E-04
Pool, pH	4
Total Iodine in pool, mol	5.12E-02
Stable Iodine Concentration, mol/L	1.50E-08
Total Iodine Concentration, mol/L	6.65E-08
I_2 Volatile Fraction	6.32E-03
S_{pool}/V_{pool}, 1/m	0.1
III. Dispersion and Dose Data	
A. Atmospheric Dispersion Factors, s/m^3	
EAB[2]	
0-2 hours	To be provided in CPA Part 2.
2-8 hours	
8-24 hours	
1-4 days	
4-30 days	
LPZ	
0-8 hours	To be provided in CPA Part 2.
8-24 hours	
1-4 days	
4-30 days	
MCRE	
0-2 hours	To be provided in CPA Part 2.
2-8 hours	
8-24 hours	
1-4 days	
4-30 days	
IV. MCRE Parameters	
A. Volume m^3 (ft^3)	1,090 (38,500)
B. Flow Rate to Environment m^3/s (ft^3/min)	18.2 (38,500)

V. Breathing Rates m^3/s	
EAB and LPZ	
0-8 hours	3.5E-04
8-24 hours	1.8E-04
1-30 days	2.3E-04
MCRE	
0-30 days	3.5E-04
VI. Occupancy Factors	
MCRE	
0-24 hours	1.0
1-4 days	0.6
4-30 days	0.4

Notes:

[1] Phase 2 starts at a 2-hour time delay after the start of the Phase 1 release.

[2] The maximum EAB χ/Q value is used for full duration of the event to ensure the worst 2-hour dose consequence.

Table 15.5-39: Main Steam Line Break Accident Airborne Release Source Term[1]

Isotope	Equilibrium Iodine		Pre-Incident Iodine Spike	
	MBq	Ci	MBq	Ci
Kr-83m	1.3E+04	3.4E-01	1.3E+04	3.4E-01
Kr-85m	3.6E+02	9.6E-03	3.6E+02	9.6E-03
Kr-85	8.7E+01	2.3E-03	8.7E+01	2.3E-03
Kr-87	2.4E+03	6.6E-02	2.4E+03	6.6E-02
Kr-88	1.3E+03	3.6E-02	1.3E+03	3.6E-02
Kr-89	4.4E+05	1.2E+01	4.4E+05	1.2E+01
Xe-131m	7.1E+01	1.9E-03	7.1E+01	1.9E-03
Xe-133m	4.2E+01	1.1E-03	4.2E+01	1.1E-03
Xe-133	6.4E+02	1.7E-02	6.4E+02	1.7E-02
Xe-135m	1.2E+04	3.1E-01	1.2E+04	3.1E-01
Xe-135	6.4E+03	1.7E-01	6.4E+03	1.7E-01
Xe-137	2.1E+04	5.8E-01	2.1E+04	5.8E-01
Xe-138	6.0E+04	1.6E+00	6.0E+04	1.6E+00
I-131	1.3E+04	3.6E-01	2.6E+05	7.2E+00
I-132	1.3E+05	3.5E+00	2.6E+06	7.1E+01
I-133	9.8E+04	2.6E+00	2.0E+06	5.3E+01
I-134	3.9E+05	1.0E+01	7.7E+06	2.1E+02
I-135	1.9E+05	5.1E+00	3.9E+06	1.0E+02
Rb-89	7.6E+04	2.0E+00	7.6E+04	2.0E+00
Cs-134	1.1E+03	3.0E-02	1.1E+03	3.0E-02
Cs-136	8.6E+02	2.3E-02	8.6E+02	2.3E-02
Cs-137	1.7E+03	4.5E-02	1.7E+03	4.5E-02
Cs-138	8.2E+04	2.2E+00	8.2E+04	2.2E+00
Ba-137m	1.7E+03	4.5E-02	1.7E+03	4.5E-02
H-3	1.2E+04	3.2E-01	1.2E+04	3.2E-01
Na-24	1.6E+02	4.4E-03	1.6E+02	4.4E-03
P-32	6.6E+00	1.8E-04	6.6E+00	1.8E-04
Cr-51	1.6E+02	4.3E-03	1.6E+02	4.3E-03
Mn-54	8.0E+01	2.2E-03	8.0E+01	2.2E-03
Mn-56	3.0E+02	8.1E-03	3.0E+02	8.1E-03
Fe-55	1.7E+02	4.5E-03	1.7E+02	4.5E-03
Fe-59	4.4E+01	1.2E-03	4.4E+01	1.2E-03
Co-58	3.8E+01	1.0E-03	3.8E+01	1.0E-03

Isotope	Equilibrium Iodine		Pre-Incident Iodine Spike	
	MBq	Ci	MBq	Ci
Co-60	7.4E+01	2.0E-03	7.4E+01	2.0E-03
Ni-63	1.7E-01	4.5E-06	1.7E-01	4.5E-06
Cu-64	7.8E+02	2.1E-02	7.8E+02	2.1E-02
Zn-65	3.4E+01	9.2E-04	3.4E+01	9.2E-04
Sr-89	1.2E+02	3.3E-03	1.2E+02	3.3E-03
Sr-90	6.2E+00	1.7E-04	6.2E+00	1.7E-04
Y-90	6.2E+00	1.7E-04	6.2E+00	1.7E-04
Sr-91	7.4E+04	2.0E+00	7.4E+04	2.0E+00
Sr-92	1.6E+05	4.3E+00	1.6E+05	4.3E+00
Y-91	1.6E+03	4.4E-02	1.6E+03	4.4E-02
Y-92	5.0E+04	1.3E+00	5.0E+04	1.3E+00
Y-93	5.4E+03	1.5E-01	5.4E+03	1.5E-01
Zr-95	3.4E+03	9.2E-02	3.4E+03	9.2E-02
Nb-95	3.4E+03	9.2E-02	3.4E+03	9.2E-02
Mo-99	1.6E+04	4.3E-01	1.6E+04	4.3E-01
Tc-99m	1.6E+04	4.3E-01	1.6E+04	4.3E-01
Ru-103	8.2E+02	2.2E-02	8.2E+02	2.2E-02
Rh-103m	8.2E+02	2.2E-02	8.2E+02	2.2E-02
Ru-106	1.2E+02	3.3E-03	1.2E+02	3.3E-03
Rh-106	1.2E+02	3.3E-03	1.2E+02	3.3E-03
Ag-110m	1.7E-01	4.5E-06	1.7E-01	4.5E-06
Te-129m	1.6E+03	4.4E-02	1.6E+03	4.4E-02
Te-131m	2.2E+03	5.9E-02	2.2E+03	5.9E-02
Te-132	3.8E+02	1.0E-02	3.8E+02	1.0E-02
Ba-140	1.7E+04	4.6E-01	1.7E+04	4.6E-01
La-140	1.7E+04	4.6E-01	1.7E+04	4.6E-01
Ce-141	8.2E+02	2.2E-02	8.2E+02	2.2E-02
Ce-144	1.2E+02	3.3E-03	1.2E+02	3.3E-03
Pr-144	1.2E+02	3.3E-03	1.2E+02	3.3E-03
W-187	3.8E+01	1.0E-03	3.8E+01	1.0E-03
Np-239	1.3E+04	3.5E-01	1.3E+04	3.5E-01

Note:1.

The released activity shown in the table is decayed in the RADTRAD model over the one-minute release duration from the building assumed.

Table 15.5-40: FWLB Accident Airborne Release Source Term

Isotope	Equilibrium Iodine		Pre-Incident Iodine Spike	
	Ci	MBq	Ci	MBq
Kr-83m	8.03E-02	2.97E+03	8.03E-02	2.97E+03
Kr-85m	2.25E-03	8.34E+01	2.25E-03	8.34E+01
Kr-85	5.50E-04	2.03E+01	5.50E-04	2.03E+01
Kr-87	1.55E-02	5.73E+02	1.55E-02	5.73E+02
Kr-88	8.45E-03	3.13E+02	8.45E-03	3.13E+02
Kr-89	2.88E+00	1.06E+05	2.88E+00	1.06E+05
Xe-131m	4.51E-04	1.67E+01	4.51E-04	1.67E+01
Xe-133m	2.68E-04	9.91E+00	2.68E-04	9.91E+00
Xe-133	4.09E-03	1.51E+02	4.09E-03	1.51E+02
Xe-135m	7.33E-02	2.71E+03	7.33E-02	2.71E+03
Xe-135	4.09E-02	1.51E+03	4.09E-02	1.51E+03
Xe-137	1.36E-01	5.02E+03	1.36E-01	5.02E+03
Xe-138	3.80E-01	1.41E+04	3.80E-01	1.41E+04
I-131	4.70E-01	1.74E+04	9.39E+00	3.47E+05
I-132	4.62E+00	1.71E+05	9.39E+01	3.47E+06
I-133	3.47E+00	1.28E+05	6.93E+01	2.57E+06
I-134	1.37E+01	5.08E+05	2.74E+02	1.02E+07
I-135	6.72E+00	2.49E+05	1.37E+02	5.08E+06
Rb-89	2.73E+00	1.01E+05	2.73E+00	1.01E+05
Cs-134	3.96E-02	1.46E+03	3.96E-02	1.46E+03
Cs-136	3.09E-02	1.14E+03	3.09E-02	1.14E+03
Cs-137	6.05E-02	2.24E+03	6.05E-02	2.24E+03
Cs-138	2.95E+00	1.09E+05	2.95E+00	1.09E+05
Ba-137m	6.05E-02	2.24E+03	6.05E-02	2.24E+03
H-3	2.41E-01	8.91E+03	2.41E-01	8.91E+03
Na-24	5.90E-03	2.18E+02	5.90E-03	2.18E+02
P-32	2.37E-04	8.79E+00	2.37E-04	8.79E+00
Cr-51	5.76E-03	2.13E+02	5.76E-03	2.13E+02
Mn-54	2.88E-03	1.07E+02	2.88E-03	1.07E+02
Mn-56	1.08E-02	3.99E+02	1.08E-02	3.99E+02
Fe-55	5.97E-03	2.21E+02	5.97E-03	2.21E+02
Fe-59	1.58E-03	5.86E+01	1.58E-03	5.86E+01

Isotope	Equilibrium Iodine		Pre-Incident Iodine Spike	
	Ci	MBq	Ci	MBq
Co-58	1.37E-03	5.06E+01	1.37E-03	5.06E+01
Co-60	2.66E-03	9.85E+01	2.66E-03	9.85E+01
Ni-63	5.97E-06	2.21E-01	5.97E-06	2.21E-01
Cu-64	2.81E-02	1.04E+03	2.81E-02	1.04E+03
Zn-65	1.22E-03	4.53E+01	1.22E-03	4.53E+01
Sr-89	4.39E-03	1.62E+02	4.39E-03	1.62E+02
Sr-90	2.23E-04	8.25E+00	2.23E-04	8.25E+00
Y-90	2.23E-04	8.25E+00	2.23E-04	8.25E+00
Sr-91	2.66E+00	9.85E+04	2.66E+00	9.85E+04
Sr-92	5.76E+00	2.13E+05	5.76E+00	2.13E+05
Y-91	5.83E-02	2.16E+03	5.83E-02	2.16E+03
Y-92	1.80E+00	6.66E+04	1.80E+00	6.66E+04
Y-93	1.94E-01	7.19E+03	1.94E-01	7.19E+03
Zr-95	1.22E-01	4.53E+03	1.22E-01	4.53E+03
Nb-95	1.22E-01	4.53E+03	1.22E-01	4.53E+03
Mo-99	5.69E-01	2.10E+04	5.69E-01	2.10E+04
Tc-99m	5.69E-01	2.10E+04	5.69E-01	2.10E+04
Ru-103	2.95E-02	1.09E+03	2.95E-02	1.09E+03
Rh-103m	2.95E-02	1.09E+03	2.95E-02	1.09E+03
Ru-106	4.39E-03	1.62E+02	4.39E-03	1.62E+02
Rh-106	4.39E-03	1.62E+02	4.39E-03	1.62E+02
Ag-110m	5.97E-06	2.21E-01	5.97E-06	2.21E-01
Te-129m	5.83E-02	2.16E+03	5.83E-02	2.16E+03
Te-131m	7.92E-02	2.93E+03	7.92E-02	2.93E+03
Te-132	1.37E-02	5.06E+02	1.37E-02	5.06E+02
Ba-140	6.12E-01	2.26E+04	6.12E-01	2.26E+04
La-140	6.12E-01	2.26E+04	6.12E-01	2.26E+04
Ce-141	2.95E-02	1.09E+03	2.95E-02	1.09E+03
Ce-144	4.39E-03	1.62E+02	4.39E-03	1.62E+02
Pr-144	4.39E-03	1.62E+02	4.39E-03	1.62E+02
W-187	1.37E-03	5.06E+01	1.37E-03	5.06E+01
Np-239	4.68E-01	1.73E+04	4.68E-01	1.73E+04

Table 15.5-41: ICS Line Break Accident Airborne Release Source Term

Isotope	Equilibrium Iodine		Pre-Incident Iodine Spike	
	Ci	MBq	Ci	MBq
Kr-83m	1.73E-01	6.39E+03	1.73E-01	6.39E+03
Kr-85m	4.85E-03	1.79E+02	4.85E-03	1.79E+02
Kr-85	1.18E-03	4.38E+01	1.18E-03	4.38E+01
Kr-87	3.34E-02	1.23E+03	3.34E-02	1.23E+03
Kr-88	1.82E-02	6.73E+02	1.82E-02	6.73E+02
Kr-89	6.19E+00	2.29E+05	6.19E+00	2.29E+05
Xe-131m	9.70E-04	3.59E+01	9.70E-04	3.59E+01
Xe-133m	5.76E-04	2.13E+01	5.76E-04	2.13E+01
Xe-133	8.79E-03	3.25E+02	8.79E-03	3.25E+02
Xe-135m	1.58E-01	5.83E+03	1.58E-01	5.83E+03
Xe-135	8.79E-02	3.25E+03	8.79E-02	3.25E+03
Xe-137	2.92E-01	1.08E+04	2.92E-01	1.08E+04
Xe-138	8.19E-01	3.03E+04	8.19E-01	3.03E+04
I-131	3.94E-03	1.46E+02	8.19E-02	3.03E+03
I-132	3.64E-02	1.35E+03	7.58E-01	2.80E+04
I-133	2.91E-02	1.08E+03	5.76E-01	2.13E+04
I-134	1.15E-01	4.26E+03	2.27E+00	8.41E+04
I-135	5.46E-02	2.02E+03	1.09E+00	4.04E+04
Rb-89	1.15E-03	4.26E+01	1.15E-03	4.26E+01
Cs-134	1.67E-05	6.17E-01	1.67E-05	6.17E-01
Cs-136	1.30E-05	4.82E-01	1.30E-05	4.82E-01
Cs-137	2.55E-05	9.42E-01	2.55E-05	9.42E-01
Cs-138	1.24E-03	4.60E+01	1.24E-03	4.60E+01
Ba-137m	2.55E-05	9.42E-01	2.55E-05	9.42E-01
H-3	8.49E-02	3.14E+03	8.49E-02	3.14E+03
Na-24	2.49E-06	9.20E-02	2.49E-06	9.20E-02
P-32	1.00E-07	3.70E-03	1.00E-07	3.70E-03
Cr-51	2.43E-06	8.97E-02	2.43E-06	8.97E-02
Mn-54	1.21E-06	4.49E-02	1.21E-06	4.49E-02
Mn-56	4.55E-06	1.68E-01	4.55E-06	1.68E-01
Fe-55	2.52E-06	9.31E-02	2.52E-06	9.31E-02
Fe-59	6.67E-07	2.47E-02	6.67E-07	2.47E-02

Isotope	Equilibrium Iodine		Pre-Incident Iodine Spike	
	Ci	MBq	Ci	MBq
Co-58	5.76E-07	2.13E-02	5.76E-07	2.13E-02
Co-60	1.12E-06	4.15E-02	1.12E-06	4.15E-02
Ni-63	2.52E-09	9.31E-05	2.52E-09	9.31E-05
Cu-64	1.18E-05	4.38E-01	1.18E-05	4.38E-01
Zn-65	5.15E-07	1.91E-02	5.15E-07	1.91E-02
Sr-89	1.85E-06	6.84E-02	1.85E-06	6.84E-02
Sr-90	9.40E-08	3.48E-03	9.40E-08	3.48E-03
Y-90	9.40E-08	3.48E-03	9.40E-08	3.48E-03
Sr-91	1.12E-03	4.15E+01	1.12E-03	4.15E+01
Sr-92	2.43E-03	8.97E+01	2.43E-03	8.97E+01
Y-91	2.46E-05	9.09E-01	2.46E-05	9.09E-01
Y-92	7.58E-04	2.80E+01	7.58E-04	2.80E+01
Y-93	8.19E-05	3.03E+00	8.19E-05	3.03E+00
Zr-95	5.15E-05	1.91E+00	5.15E-05	1.91E+00
Nb-95	5.15E-05	1.91E+00	5.15E-05	1.91E+00
Mo-99	2.40E-04	8.86E+00	2.40E-04	8.86E+00
Tc-99m	2.40E-04	8.86E+00	2.40E-04	8.86E+00
Ru-103	1.24E-05	4.60E-01	1.24E-05	4.60E-01
Rh-103m	1.24E-05	4.60E-01	1.24E-05	4.60E-01
Ru-106	1.85E-06	6.84E-02	1.85E-06	6.84E-02
Rh-106	1.85E-06	6.84E-02	1.85E-06	6.84E-02
Ag-110m	2.52E-09	9.31E-05	2.52E-09	9.31E-05
Te-129m	2.46E-05	9.09E-01	2.46E-05	9.09E-01
Te-131m	3.34E-05	1.23E+00	3.34E-05	1.23E+00
Te-132	5.76E-06	2.13E-01	5.76E-06	2.13E-01
Ba-140	2.58E-04	9.54E+00	2.58E-04	9.54E+00
La-140	2.58E-04	9.54E+00	2.58E-04	9.54E+00
Ce-141	1.24E-05	4.60E-01	1.24E-05	4.60E-01
Ce-144	1.85E-06	6.84E-02	1.85E-06	6.84E-02
Pr-144	1.85E-06	6.84E-02	1.85E-06	6.84E-02
W-187	5.76E-07	2.13E-02	5.76E-07	2.13E-02
Np-239	1.97E-04	7.29E+00	1.97E-04	7.29E+00

Table 15.5-42a: Instrument Line Break Accident Airborne Iodine Equilibrium Source Term

Time	2.0 h		8.0 h		24.0 h		72.0 h		78.0 h	
Nuclide	Ci	MBq	Ci	MBq	Ci	MBq	Ci	MBq	Ci	MBq
I-131	2.3E-01	8.5E+03	9.2E-01	3.4E+04	2.7E+00	1.0E+05	8.2E+00	3.1E+05	8.5E+00	3.1E+05
I-132	2.3E+00	8.3E+04	9.0E+00	3.3E+05	2.7E+01	1.0E+06	8.1E+01	3.0E+06	8.3E+01	3.1E+06
I-133	1.7E+00	6.3E+04	6.8E+00	2.5E+05	2.0E+01	7.5E+05	6.1E+01	2.3E+06	6.2E+01	2.3E+06
I-134	6.7E+00	2.5E+05	2.7E+01	9.9E+05	8.0E+01	3.0E+06	2.4E+02	8.9E+06	2.5E+02	9.1E+06
I-135	3.3E+00	1.2E+05	1.3E+01	4.8E+05	3.9E+01	1.5E+06	1.2E+02	4.4E+06	1.2E+02	4.5E+06
Rb-89	1.3E+00	4.9E+04	5.4E+00	2.0E+05	1.6E+01	5.9E+05	4.8E+01	1.8E+06	4.9E+01	1.8E+06
Cs-134	1.9E-02	7.2E+02	7.8E-02	2.9E+03	2.3E-01	8.6E+03	7.0E-01	2.6E+04	7.2E-01	2.6E+04
Cs-136	1.5E-02	5.6E+02	6.1E-02	2.2E+03	1.8E-01	6.7E+03	5.5E-01	2.0E+04	5.6E-01	2.1E+04
Cs-137	3.0E-02	1.1E+03	1.2E-01	4.4E+03	3.6E-01	1.3E+04	1.1E+00	3.9E+04	1.1E+00	4.0E+04
Cs-138	1.4E+00	5.3E+04	5.8E+00	2.1E+05	1.7E+01	6.4E+05	5.2E+01	1.9E+06	5.3E+01	2.0E+06
Ba-137m	3.0E-02	1.1E+03	1.2E-01	4.4E+03	3.6E-01	1.3E+04	1.1E+00	3.9E+04	1.1E+00	4.0E+04
H-3	9.9E-02	3.6E+03	3.9E-01	1.5E+04	1.2E+00	4.4E+04	3.6E+00	1.3E+05	3.6E+00	1.3E+05
Na-24	2.9E-03	1.1E+02	1.2E-02	4.3E+02	3.5E-02	1.3E+03	1.0E-01	3.8E+03	1.1E-01	3.9E+03
P-32	1.2E-04	4.3E+00	4.7E-04	1.7E+01	1.4E-03	5.2E+01	4.2E-03	1.5E+02	4.3E-03	1.6E+02
Cr-51	2.8E-03	1.0E+02	1.1E-02	4.2E+02	3.4E-02	1.3E+03	1.0E-01	3.8E+03	1.0E-01	3.9E+03
Mn-54	1.4E-03	5.2E+01	5.6E-03	2.1E+02	1.7E-02	6.3E+02	5.1E-02	1.9E+03	5.2E-02	1.9E+03
Mn-56	5.3E-03	2.0E+02	2.1E-02	7.8E+02	6.3E-02	2.3E+03	1.9E-01	7.0E+03	2.0E-01	7.2E+03
Fe-55	2.9E-03	1.1E+02	1.2E-02	4.3E+02	3.5E-02	1.3E+03	1.1E-01	3.9E+03	1.1E-01	4.0E+03
Fe-59	7.7E-04	2.9E+01	3.1E-03	1.1E+02	9.3E-03	3.4E+02	2.8E-02	1.0E+03	2.9E-02	1.1E+03

Time	2.0 h		8.0 h		24.0 h		72.0 h		78.0 h	
Nuclide	**Ci**	**MBq**	**Ci**	**MBq**	**Ci**	**MBq**	**Ci**	**MBq**	**Ci**	**MBq**
Co-58	6.7E-04	2.5E+01	2.7E-03	9.9E+01	8.0E-03	3.0E+02	2.4E-02	8.9E+02	2.5E-02	9.1E+02
Co-60	1.3E-03	4.8E+01	5.2E-03	1.9E+02	1.6E-02	5.8E+02	4.7E-02	1.7E+03	4.8E-02	1.8E+03
Ni-63	2.9E-06	1.1E-01	1.2E-05	4.3E-01	3.5E-05	1.3E+00	1.1E-04	3.9E+00	1.1E-04	4.0E+00
Cu-64	1.4E-02	5.1E+02	5.5E-02	2.0E+03	1.6E-01	6.1E+03	4.9E-01	1.8E+04	5.1E-01	1.9E+04
Zn-65	6.0E-04	2.2E+01	2.4E-03	8.9E+01	7.2E-03	2.7E+02	2.2E-02	8.0E+02	2.2E-02	8.2E+02
Sr-89	2.1E-03	7.9E+01	8.6E-03	3.2E+02	2.6E-02	9.5E+02	7.7E-02	2.9E+03	7.9E-02	2.9E+03
Sr-90	1.1E-04	4.0E+00	4.4E-04	1.6E+01	1.3E-03	4.9E+01	3.9E-03	1.5E+02	4.0E-03	1.5E+02
Y-90	1.1E-04	4.0E+00	4.4E-04	1.6E+01	1.3E-03	4.9E+01	3.9E-03	1.5E+02	4.0E-03	1.5E+02
Sr-91	1.3E+00	4.8E+04	5.2E+00	1.9E+05	1.6E+01	5.8E+05	4.7E+01	1.7E+06	4.8E+01	1.8E+06
Sr-92	2.8E+00	1.0E+05	1.1E+01	4.2E+05	3.4E+01	1.3E+06	1.0E+02	3.8E+06	1.0E+02	3.9E+06
Y-91	2.9E-02	1.1E+03	1.1E-01	4.2E+03	3.4E-01	1.3E+04	1.0E+00	3.8E+04	1.1E+00	3.9E+04
Y-92	8.8E-01	3.3E+04	3.5E+00	1.3E+05	1.1E+01	3.9E+05	3.2E+01	1.2E+06	3.3E+01	1.2E+06
Y-93	9.5E-02	3.5E+03	3.8E-01	1.4E+04	1.1E+00	4.2E+04	3.4E+00	1.3E+05	3.5E+00	1.3E+05
Zr-95	6.0E-02	2.2E+03	2.4E-01	8.9E+03	7.2E-01	2.7E+04	2.2E+00	8.0E+04	2.2E+00	8.2E+04
Nb-95	6.0E-02	2.2E+03	2.4E-01	8.9E+03	7.2E-01	2.7E+04	2.2E+00	8.0E+04	2.2E+00	8.2E+04
Mo-99	2.8E-01	1.0E+04	1.1E+00	4.1E+04	3.3E+00	1.2E+05	1.0E+01	3.7E+05	1.0E+01	3.8E+05
Tc-99m	2.8E-01	1.0E+04	1.1E+00	4.1E+04	3.3E+00	1.2E+05	1.0E+01	3.7E+05	1.0E+01	3.8E+05
Ru-103	1.4E-02	5.3E+02	5.8E-02	2.1E+03	1.7E-01	6.4E+03	5.2E-01	1.9E+04	5.3E-01	2.0E+04
Rh-103m	1.4E-02	5.3E+02	5.8E-02	2.1E+03	1.7E-01	6.4E+03	5.2E-01	1.9E+04	5.3E-01	2.0E+04
Ru-106	2.1E-03	7.9E+01	8.6E-03	3.2E+02	2.6E-02	9.5E+02	7.7E-02	2.9E+03	7.9E-02	2.9E+03

Time	2.0 h		8.0 h		24.0 h		72.0 h		78.0 h	
Nuclide	**Ci**	**MBq**	**Ci**	**MBq**	**Ci**	**MBq**	**Ci**	**MBq**	**Ci**	**MBq**
Rh-106	2.1E-03	7.9E+01	8.6E-03	3.2E+02	2.6E-02	9.5E+02	7.7E-02	2.9E+03	7.9E-02	2.9E+03
Ag-110m	2.9E-06	1.1E-01	1.2E-05	4.3E-01	3.5E-05	1.3E+00	1.1E-04	3.9E+00	1.1E-04	4.0E+00
Te-129m	2.9E-02	1.1E+03	1.1E-01	4.2E+03	3.4E-01	1.3E+04	1.0E+00	3.8E+04	1.1E+00	3.9E+04
Te-131m	3.9E-02	1.4E+03	1.6E-01	5.7E+03	4.7E-01	1.7E+04	1.4E+00	5.2E+04	1.4E+00	5.3E+04
Te-132	6.7E-03	2.5E+02	2.7E-02	9.9E+02	8.0E-02	3.0E+03	2.4E-01	8.9E+03	2.5E-01	9.1E+03
Ba-140	3.0E-01	1.1E+04	1.2E+00	4.4E+04	3.6E+00	1.3E+05	1.1E+01	4.0E+05	1.1E+01	4.1E+05
La-140	3.0E-01	1.1E+04	1.2E+00	4.4E+04	3.6E+00	1.3E+05	1.1E+01	4.0E+05	1.1E+01	4.1E+05
Ce-141	1.4E-02	5.3E+02	5.8E-02	2.1E+03	1.7E-01	6.4E+03	5.2E-01	1.9E+04	5.3E-01	2.0E+04
Ce-144	2.1E-03	7.9E+01	8.6E-03	3.2E+02	2.6E-02	9.5E+02	7.7E-02	2.9E+03	7.9E-02	2.9E+03
Pr-144	2.1E-03	7.9E+01	8.6E-03	3.2E+02	2.6E-02	9.5E+02	7.7E-02	2.9E+03	7.9E-02	2.9E+03
W-187	6.7E-04	2.5E+01	2.7E-03	9.9E+01	8.0E-03	3.0E+02	2.4E-02	8.9E+02	2.5E-02	9.1E+02
Np-239	2.3E-01	8.5E+03	9.2E-01	3.4E+04	2.7E+00	1.0E+05	8.2E+00	3.1E+05	8.5E+00	3.1E+05

Table 15.5-42b: Instrument Line Break Accident Iodine Spike Release Source Term

Time	2.0 h		8.0 h		24.0 h		72.0 h		78.0 h	
Nuclide	**Ci**	**MBq**	**Ci**	**MBq**	**Ci**	**MBq**	**Ci**	**MBq**	**Ci**	**MBq**
I-131	4.6E+00	1.7E+05	1.8E+01	6.8E+05	5.5E+01	2.0E+06	1.6E+02	6.1E+06	1.7E+02	6.3E+06
I-132	4.6E+01	1.7E+06	1.8E+02	6.8E+06	5.5E+02	2.0E+07	1.6E+03	6.1E+07	1.7E+03	6.3E+07
I-133	3.4E+01	1.3E+06	1.4E+02	5.0E+06	4.1E+02	1.5E+07	1.2E+03	4.5E+07	1.2E+03	4.6E+07
I-134	1.3E+02	4.9E+06	5.4E+02	2.0E+07	1.6E+03	5.9E+07	4.8E+03	1.8E+08	4.9E+03	1.8E+08
I-135	6.7E+01	2.5E+06	2.7E+02	9.9E+06	8.0E+02	3.0E+07	2.4E+03	8.9E+07	2.5E+03	9.1E+07
Rb-89	1.3E+00	4.9E+04	5.4E+00	2.0E+05	1.6E+01	5.9E+05	4.8E+01	1.8E+06	4.9E+01	1.8E+06
Cs-134	1.9E-02	7.2E+02	7.8E-02	2.9E+03	2.3E-01	8.6E+03	7.0E-01	2.6E+04	7.2E-01	2.6E+04
Cs-136	1.5E-02	5.6E+02	6.1E-02	2.2E+03	1.8E-01	6.7E+03	5.5E-01	2.0E+04	5.6E-01	2.1E+04
Cs-137	3.0E-02	1.1E+03	1.2E-01	4.4E+03	3.6E-01	1.3E+04	1.1E+00	3.9E+04	1.1E+00	4.0E+04
Cs-138	1.4E+00	5.3E+04	5.8E+00	2.1E+05	1.7E+01	6.4E+05	5.2E+01	1.9E+06	5.3E+01	2.0E+06
Ba-137m	3.0E-02	1.1E+03	1.2E-01	4.4E+03	3.6E-01	1.3E+04	1.1E+00	3.9E+04	1.1E+00	4.0E+04
H-3	9.9E-02	3.6E+03	3.9E-01	1.5E+04	1.2E+00	4.4E+04	3.6E+00	1.3E+05	3.6E+00	1.3E+05
Na-24	2.9E-03	1.1E+02	1.2E-02	4.3E+02	3.5E-02	1.3E+03	1.0E-01	3.8E+03	1.1E-01	3.9E+03
P-32	1.2E-04	4.3E+00	4.7E-04	1.7E+01	1.4E-03	5.2E+01	4.2E-03	1.5E+02	4.3E-03	1.6E+02
Cr-51	2.8E-03	1.0E+02	1.1E-02	4.2E+02	3.4E-02	1.3E+03	1.0E-01	3.8E+03	1.0E-01	3.9E+03
Mn-54	1.4E-03	5.2E+01	5.6E-03	2.1E+02	1.7E-02	6.3E+02	5.1E-02	1.9E+03	5.2E-02	1.9E+03
Mn-56	5.3E-03	2.0E+02	2.1E-02	7.8E+02	6.3E-02	2.3E+03	1.9E-01	7.0E+03	2.0E-01	7.2E+03
Fe-55	2.9E-03	1.1E+02	1.2E-02	4.3E+02	3.5E-02	1.3E+03	1.1E-01	3.9E+03	1.1E-01	4.0E+03
Fe-59	7.7E-04	2.9E+01	3.1E-03	1.1E+02	9.3E-03	3.4E+02	2.8E-02	1.0E+03	2.9E-02	1.1E+03
Co-58	6.7E-04	2.5E+01	2.7E-03	9.9E+01	8.0E-03	3.0E+02	2.4E-02	8.9E+02	2.5E-02	9.1E+02
Co-60	1.3E-03	4.8E+01	5.2E-03	1.9E+02	1.6E-02	5.8E+02	4.7E-02	1.7E+03	4.8E-02	1.8E+03
Ni-63	2.9E-06	1.1E-01	1.2E-05	4.3E-01	3.5E-05	1.3E+00	1.1E-04	3.9E+00	1.1E-04	4.0E+00

Time	2.0 h		8.0 h		24.0 h		72.0 h		78.0 h	
Nuclide	**Ci**	**MBq**	**Ci**	**MBq**	**Ci**	**MBq**	**Ci**	**MBq**	**Ci**	**MBq**
Cu-64	1.4E-02	5.1E+02	5.5E-02	2.0E+03	1.6E-01	6.1E+03	4.9E-01	1.8E+04	5.1E-01	1.9E+04
Zn-65	6.0E-04	2.2E+01	2.4E-03	8.9E+01	7.2E-03	2.7E+02	2.2E-02	8.0E+02	2.2E-02	8.2E+02
Sr-89	2.1E-03	7.9E+01	8.6E-03	3.2E+02	2.6E-02	9.5E+02	7.7E-02	2.9E+03	7.9E-02	2.9E+03
Sr-90	1.1E-04	4.0E+00	4.4E-04	1.6E+01	1.3E-03	4.9E+01	3.9E-03	1.5E+02	4.0E-03	1.5E+02
Y-90	1.1E-04	4.0E+00	4.4E-04	1.6E+01	1.3E-03	4.9E+01	3.9E-03	1.5E+02	4.0E-03	1.5E+02
Sr-91	1.3E+00	4.8E+04	5.2E+00	1.9E+05	1.6E+01	5.8E+05	4.7E+01	1.7E+06	4.8E+01	1.8E+06
Sr-92	2.8E+00	1.0E+05	1.1E+01	4.2E+05	3.4E+01	1.3E+06	1.0E+02	3.8E+06	1.0E+02	3.9E+06
Y-91	2.9E-02	1.1E+03	1.1E-01	4.2E+03	3.4E-01	1.3E+04	1.0E+00	3.8E+04	1.1E+00	3.9E+04
Y-92	8.8E-01	3.3E+04	3.5E+00	1.3E+05	1.1E+01	3.9E+05	3.2E+01	1.2E+06	3.3E+01	1.2E+06
Y-93	9.5E-02	3.5E+03	3.8E-01	1.4E+04	1.1E+00	4.2E+04	3.4E+00	1.3E+05	3.5E+00	1.3E+05
Zr-95	6.0E-02	2.2E+03	2.4E-01	8.9E+03	7.2E-01	2.7E+04	2.2E+00	8.0E+04	2.2E+00	8.2E+04
Nb-95	6.0E-02	2.2E+03	2.4E-01	8.9E+03	7.2E-01	2.7E+04	2.2E+00	8.0E+04	2.2E+00	8.2E+04
Mo-99	2.8E-01	1.0E+04	1.1E+00	4.1E+04	3.3E+00	1.2E+05	1.0E+01	3.7E+05	1.0E+01	3.8E+05
Tc-99m	2.8E-01	1.0E+04	1.1E+00	4.1E+04	3.3E+00	1.2E+05	1.0E+01	3.7E+05	1.0E+01	3.8E+05
Ru-103	1.4E-02	5.3E+02	5.8E-02	2.1E+03	1.7E-01	6.4E+03	5.2E-01	1.9E+04	5.3E-01	2.0E+04
Rh-103m	1.4E-02	5.3E+02	5.8E-02	2.1E+03	1.7E-01	6.4E+03	5.2E-01	1.9E+04	5.3E-01	2.0E+04
Ru-106	2.1E-03	7.9E+01	8.6E-03	3.2E+02	2.6E-02	9.5E+02	7.7E-02	2.9E+03	7.9E-02	2.9E+03
Rh-106	2.1E-03	7.9E+01	8.6E-03	3.2E+02	2.6E-02	9.5E+02	7.7E-02	2.9E+03	7.9E-02	2.9E+03
Ag-110m	2.9E-06	1.1E-01	1.2E-05	4.3E-01	3.5E-05	1.3E+00	1.1E-04	3.9E+00	1.1E-04	4.0E+00
Te-129m	2.9E-02	1.1E+03	1.1E-01	4.2E+03	3.4E-01	1.3E+04	1.0E+00	3.8E+04	1.1E+00	3.9E+04
Te-131m	3.9E-02	1.4E+03	1.6E-01	5.7E+03	4.7E-01	1.7E+04	1.4E+00	5.2E+04	1.4E+00	5.3E+04
Te-132	6.7E-03	2.5E+02	2.7E-02	9.9E+02	8.0E-02	3.0E+03	2.4E-01	8.9E+03	2.5E-01	9.1E+03
Ba-140	3.0E-01	1.1E+04	1.2E+00	4.4E+04	3.6E+00	1.3E+05	1.1E+01	4.0E+05	1.1E+01	4.1E+05

Time	2.0 h		8.0 h		24.0 h		72.0 h		78.0 h	
Nuclide	**Ci**	**MBq**	**Ci**	**MBq**	**Ci**	**MBq**	**Ci**	**MBq**	**Ci**	**MBq**
La-140	3.0E-01	1.1E+04	1.2E+00	4.4E+04	3.6E+00	1.3E+05	1.1E+01	4.0E+05	1.1E+01	4.1E+05
Ce-141	1.4E-02	5.3E+02	5.8E-02	2.1E+03	1.7E-01	6.4E+03	5.2E-01	1.9E+04	5.3E-01	2.0E+04
Ce-144	2.1E-03	7.9E+01	8.6E-03	3.2E+02	2.6E-02	9.5E+02	7.7E-02	2.9E+03	7.9E-02	2.9E+03
Pr-144	2.1E-03	7.9E+01	8.6E-03	3.2E+02	2.6E-02	9.5E+02	7.7E-02	2.9E+03	7.9E-02	2.9E+03
W-187	6.7E-04	2.5E+01	2.7E-03	9.9E+01	8.0E-03	3.0E+02	2.4E-02	8.9E+02	2.5E-02	9.1E+02
Np-239	2.3E-01	8.5E+03	9.2E-01	3.4E+04	2.7E+00	1.0E+05	8.2E+00	3.1E+05	8.5E+00	3.1E+05

Table 15.5-43: Conservatisms Used in the Non-LOCA DSA

Analysis Type	Event Type	Conservatism	
		Code	Plant Parameters and System Performances
BL-DSA	AOO	Best Estimate	• Rated power initial conditions • Conservatism in the plant parameters and the derived acceptance criteria is established conservatively such that there is no need to account for uncertainties in the analysis method • DL2 functions are primarily credited to meet acceptance criteria. • If a DL3 function is necessary in the BL-DSA to mitigate a less frequent AOO PIE, a demonstration has to be provided that it is not practicable to implement a DL2 mitigation function, and an EX-DSA has to also be performed assuming CCF of DL3 mitigating functions.
CN-DSA	DBA	Graded Approach	• Conservative initial conditions are established for events that are limiting • Conservative setpoints and plant performance parameters • Analysis method uncertainties are addressed with a graded approach depending on margin to the derived acceptance criteria as described in Subsection 15.5.1.1 • DL3 functions are credited in the analyses • Application of limiting single active component failures among equipment performing DL3 functions
EX-DSA	DEC	Best Estimate	• Rated power initial conditions are used • Nominal setpoints and plant performance parameters are used (conservative setpoints and plant performance may be used for convenience/simplification) • Sensitivity analyses performed to understand cliff edge effects

Analysis Type	Event Type	Conservatism	
		Code	Plant Parameters and System Performances
			• Any available DL function that is not disabled by the PIE or assumed failures in the event sequences may be credited

Table 15.5-44: DL3 Functions Credited in Conservative LOCA Analyses

Credited Function	Action (Note 3)	Signal	Analytical Setpoint
DL3-02	Hydraulic scram	Low steam RPV pressure	5.617 MPa
DL3-07	Hydraulic scram	High containment pressure	2.8 psig (18.5 kPaG)
DL3-09	Hydraulic scram	Line Break Indication (MS, FW, ICS)	For breaks larger than 19 mm in diameter (Note 1)
DL3-14	ICS initiation	Low RPV water level	14.22 m
DL3-15	ICS initiation	High Containment pressure	2.8 psig (18.5 kPaG)
DL3-16	ICS initiation	Line Break Indication (MS, FW, ICS)	For breaks larger than 19 mm in diameter (Note 1)
DL3-17	MSRIV closure (Note 2)	Low steam RPV pressure	5.617 MPa
DL3-18	MSRIV closure (Note 2)	Low RPV water level	14.22 m
DL3-22	Reactor and Containment Isolation Valve closure (Note 2)	High Containment pressure	2.8 psig (18.5 kPaG) (Note 1)
DL3-20 DL3-21	MSRIV closure	Line break indication in MSL, FW or SDC	Within 1 second for breaks larger than 19 mm in diameter
DL3-25	FW and SDC RIV closure	Line break indication in FW or SDC	Within 1 second for breaks larger than 19 mm in diameter
DL3-26	CUW RIV closure	Line break indication in CUW	Within 1 second for breaks larger than 19 mm in diameter
DL3-27 DL3-28 DL3-29	ICS RIV closure of the broken ICS train	Line break indication in the respective ICS trains	Within 1 second for breaks larger than 19 mm in diameter
DL3-36	RIV and Containment Isolation Valve closure	Low RPV water level	14.22 m

(1) This setpoint is reached in less than 1 s in large break cases.

(2) For large breaks, isolation valves are assumed to start closing with a 5 second delay from the time of pipe break and are fully closed in another 5 seconds. For small breaks, the isolation valves start closing with a 5 second delay after the setpoint is reached and are fully closed in another 5 seconds. Containment isolation is credited only for FW pipe breaks. Isolation functions may also include CIVs. This table shows only the valves credited to close by the isolation signal.

(3) The SC1 I&C system performing these DL3 functions is described in Subsection 7.3.1.2, items 1 (hydraulic scram), 3 (valve closures except for ICS RIV), 4(ICS initiation), and 5(ICS RIV closure).

Table 15.5-45: DL2 and DL4a Functions Credited in DEC LOCA Analyses

DL3 Function Failure	Action	DL2 or DL4a Credited Function (Note 1)	Notes
DL3-02	Hydraulic scram on low steam pipe pressure	DL2-08 on turbine trip demand	DL2-08 is initiated on LOPP, and a faster scram than DL3-02 credited in CN-DSA. ATS is described in Subsection 7.3.3.2, Item 1.
DL3-07	Hydraulic scram on high containment pressure	DL4a-08	
DL3-09	Hydraulic scram on Line Break Indication (MS, FW, ICS)	DL4a-26	
DL3-14	ICS initiation on Low RPV water level	DL4a-33	
DL3-15	ICS initiation on High Containment pressure	DL4a-11	
DL3-16	ICS initiation on Line Break Indication (MS, FW, ICS)	DL4a-27 and DL4a-28	
DL3-17	MSRIV closure on Low steam pipe pressure	DL2-13 and DL2-41	DL3-17 is credited in small break cases for limiting inventory losses to the turbine. In DL2, the pressure controller throttles the flow until DL2-13 (turbine trip on high main condenser pressure setpoint 2) and DL2-14 (TBV closure on high main condenser pressure setpoint 3) stop steam flow to the condenser. This combination of functions results in less inventory losses than the CN-DSA sequences. ATS is described in Subsection 7.3.3.2, Item 1. Subsections 10.2.4 and 10.4.1.3 describe DL2-13 and DL2-4.

DL3 Function Failure	Action	DL2 or DL4a Credited Function (Note 1)	Notes
DL3-18	MSRIV closure on Low RPV water level	DL4a-34	
DL3-22	Reactor and Containment Isolation Valve closure on High Containment pressure	DL4a-16	
DL3-20 DL3-21	MSRIV closure on Line break indication in MSL, FW or SDC	DL4a-14 and DL4a-15	
DL3-25	FW and SDC RIV closure on line break indication in FW or SDC	DL4a-17	
DL3-27 DL3-28 DL3-29	Isolation Condenser RIV closure of the broken ICS train line break indication in the respective ICS train	None	See discussion in Subsection 15.2.4.6.4.

Note:

1. The Diverse Protection System performing the DL4a functions are described in Subsection 7.3.2.2, Item 1. The DL2 and DL4a functions identified perform the same action on the same setpoints as the DL3 functions, unless noted otherwise.

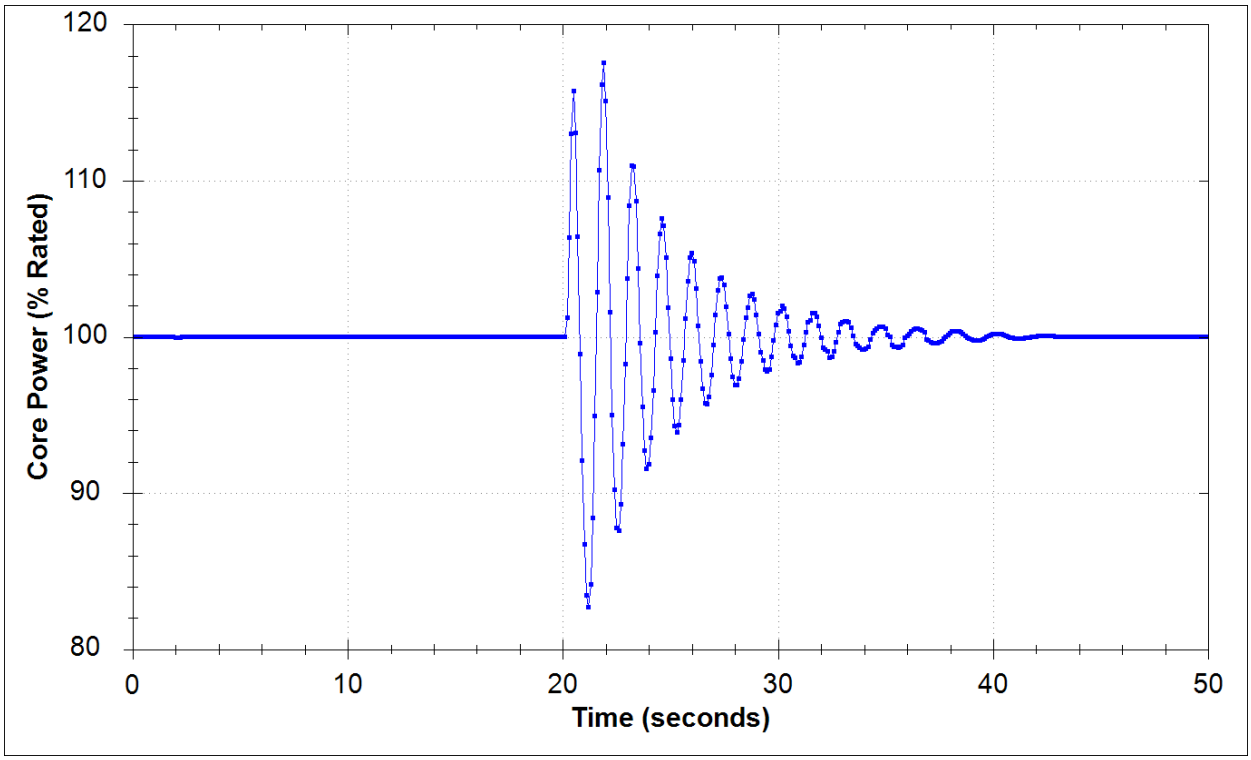


Figure 15.5-1: MOC Transient Results for Core-Wide Stability Analysis

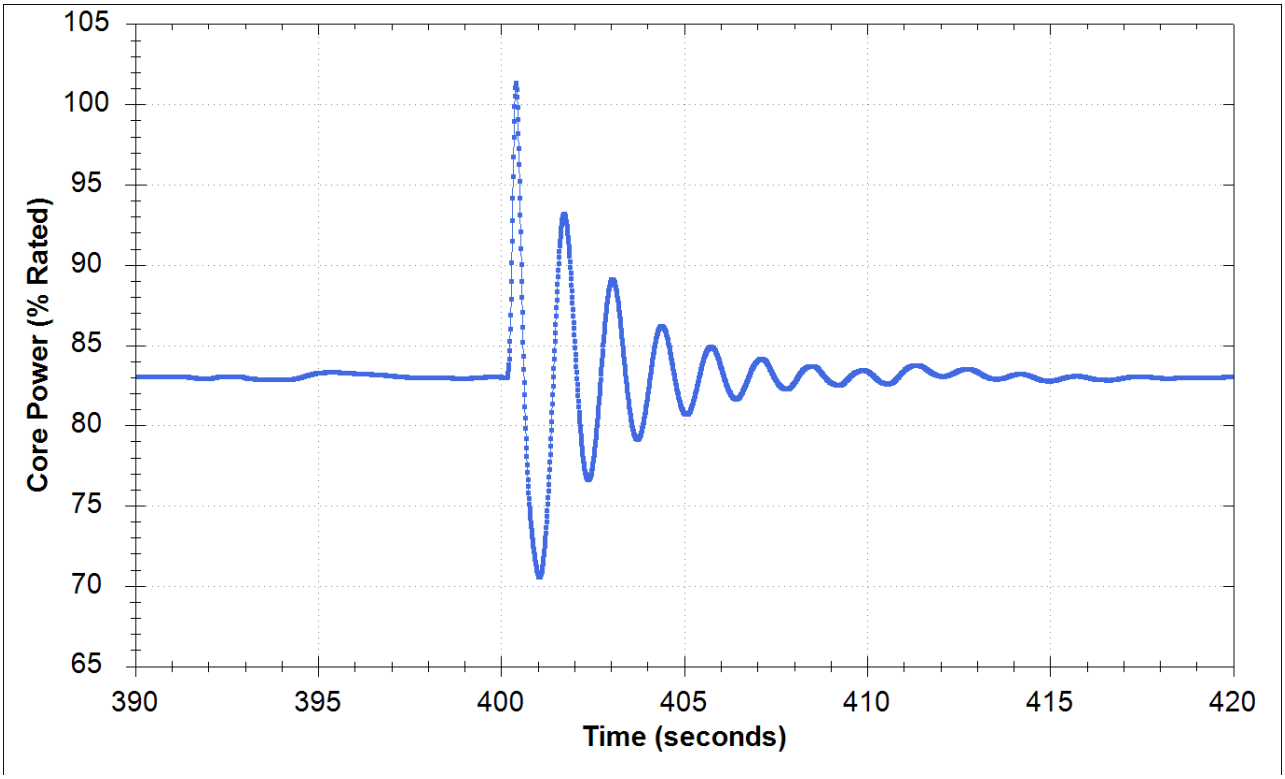


Figure 15.5-2: MOC Transient Results for LFWH with SCCRI Core-Wide Stability Analysis

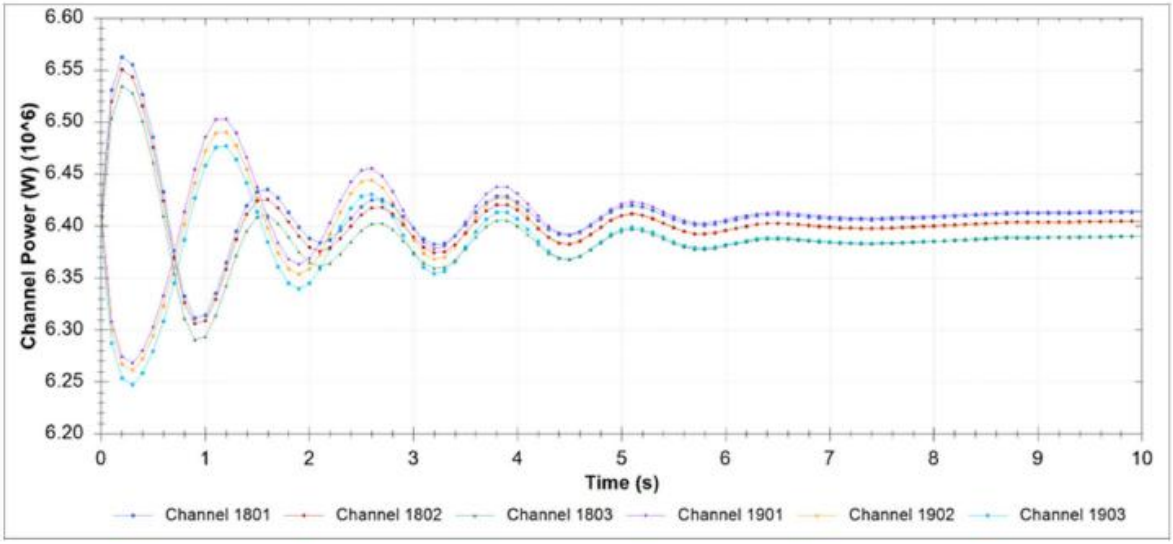


Figure 15.5-3: Regional Mode Stability Response at 115% Power

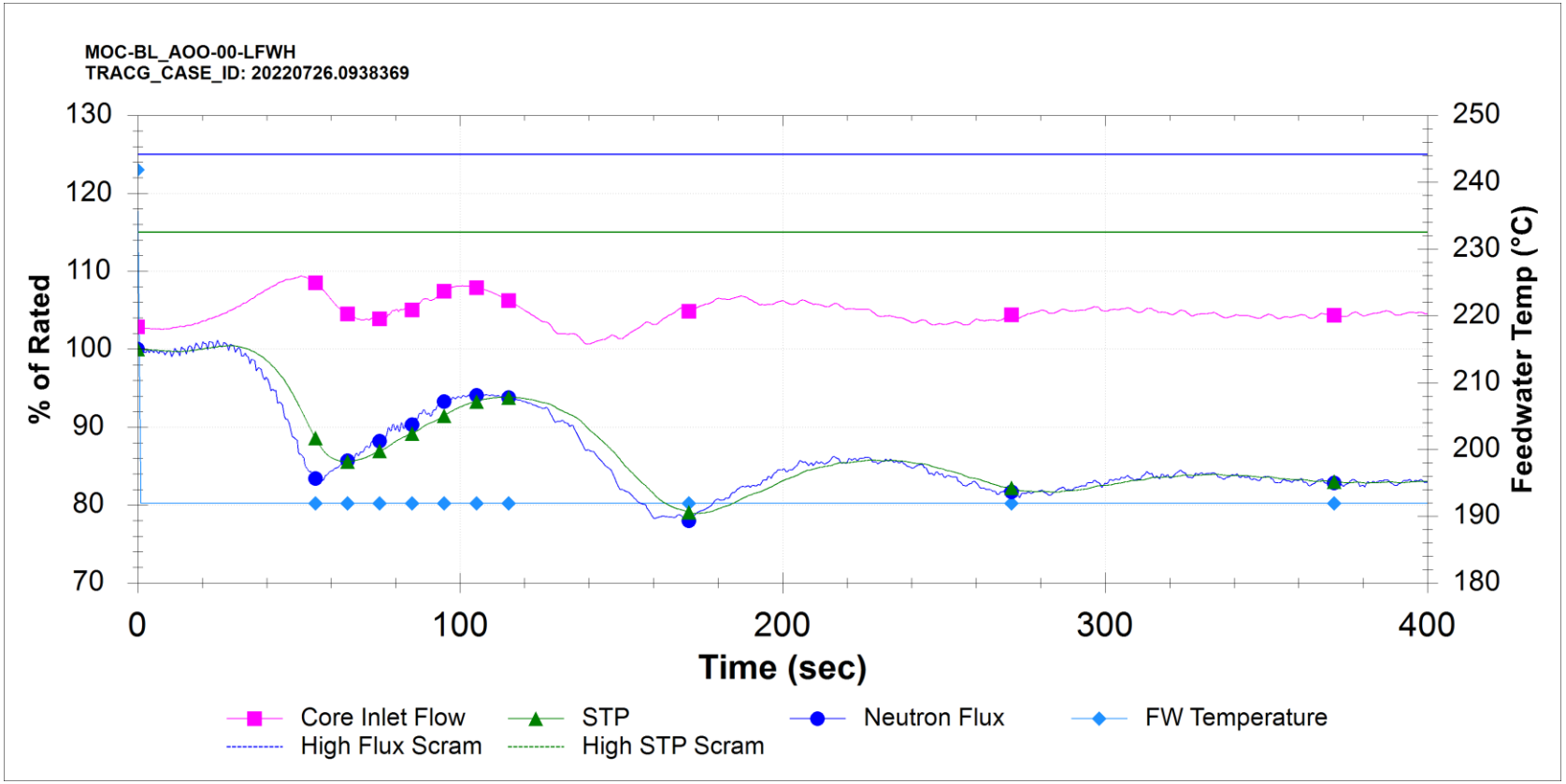


Figure 15.5-4: Loss of Feedwater Heating (AOO)

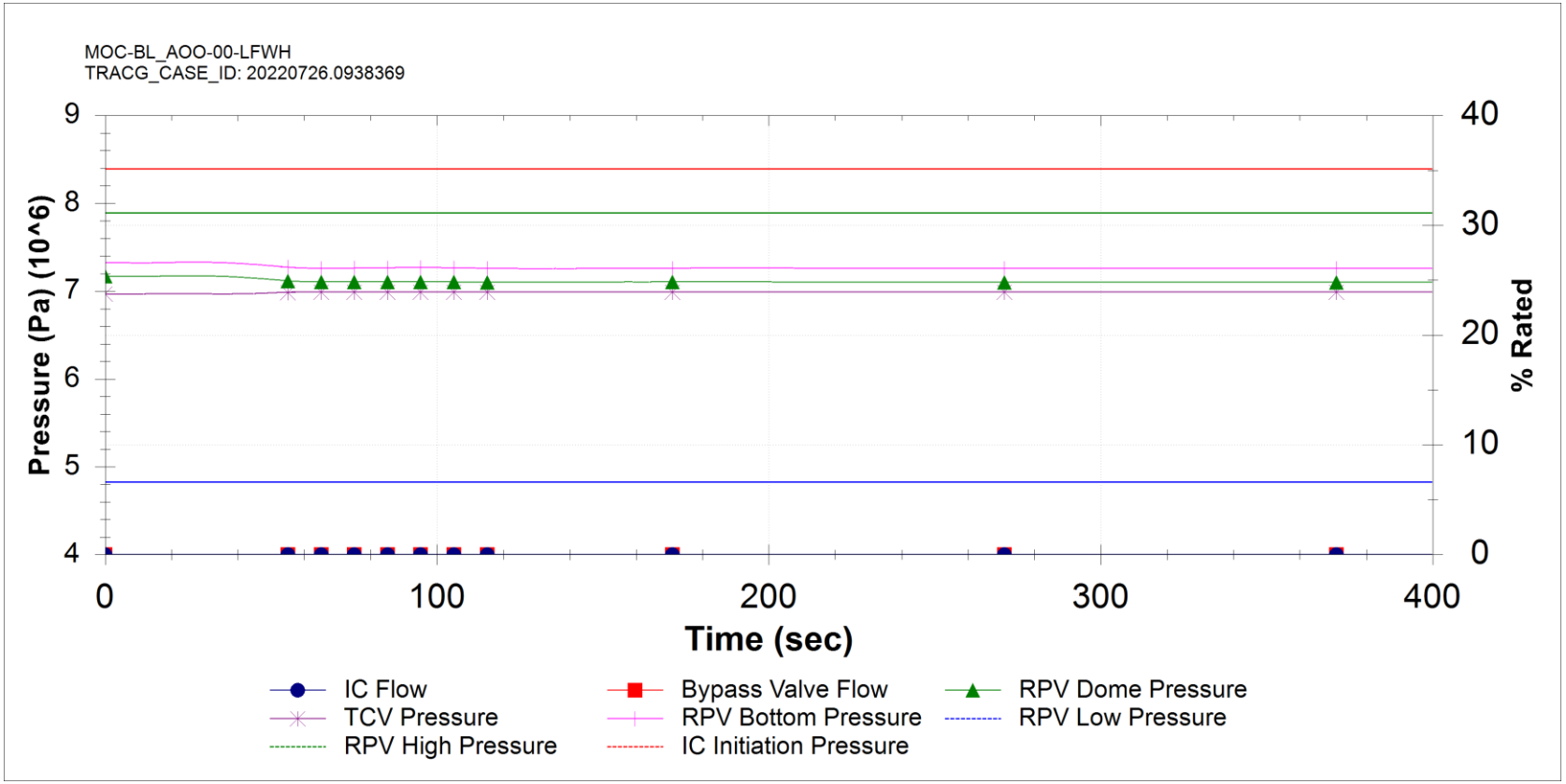


Figure 15.5-5: Loss of Feedwater Heating (AOO)

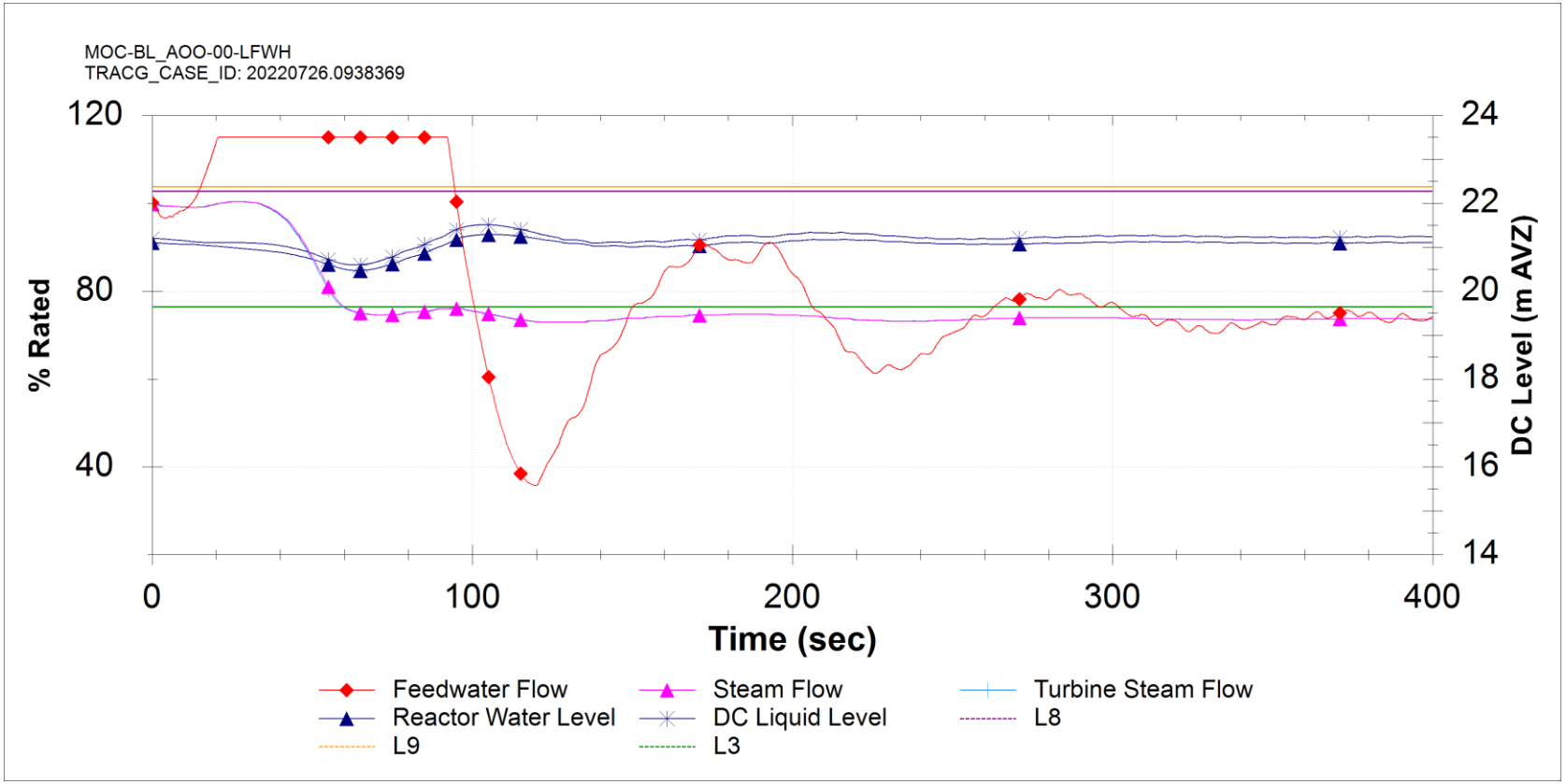


Figure 15.5-6: Loss of Feedwater Heating (AOO)

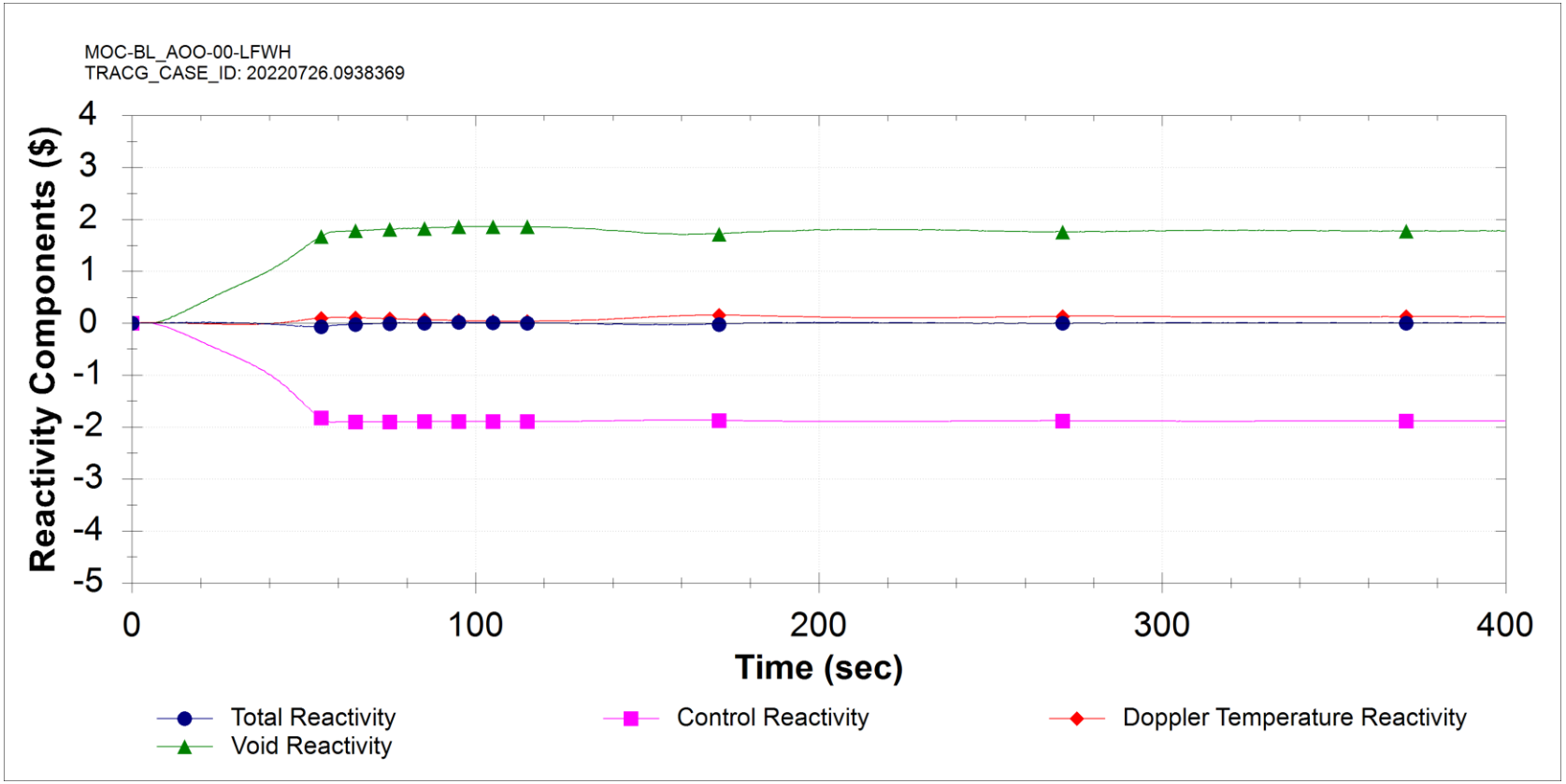


Figure 15.5-7: Loss of Feedwater Heating (AOO)

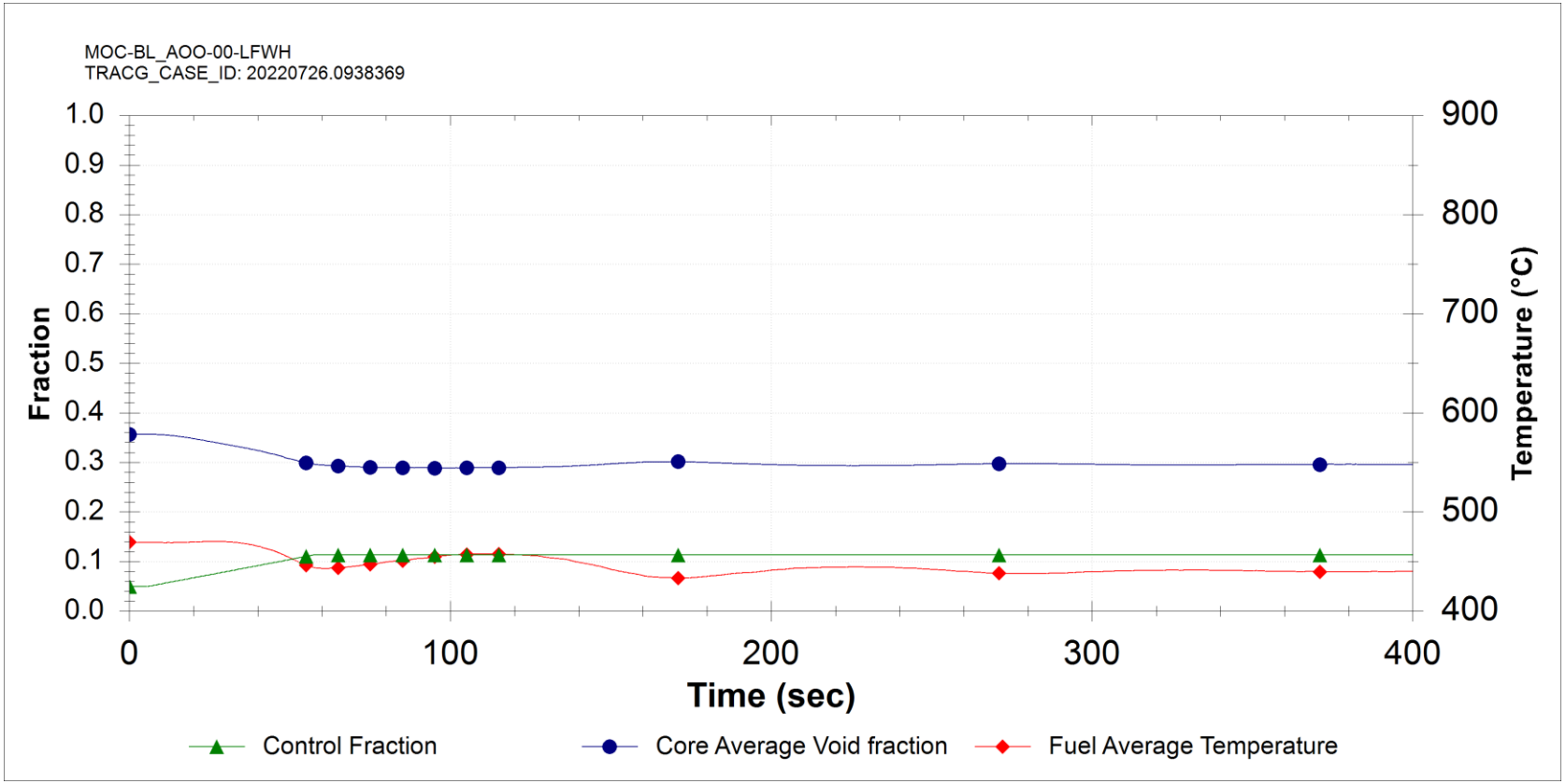


Figure 15.5-8: Loss of Feedwater Heating (AOO)

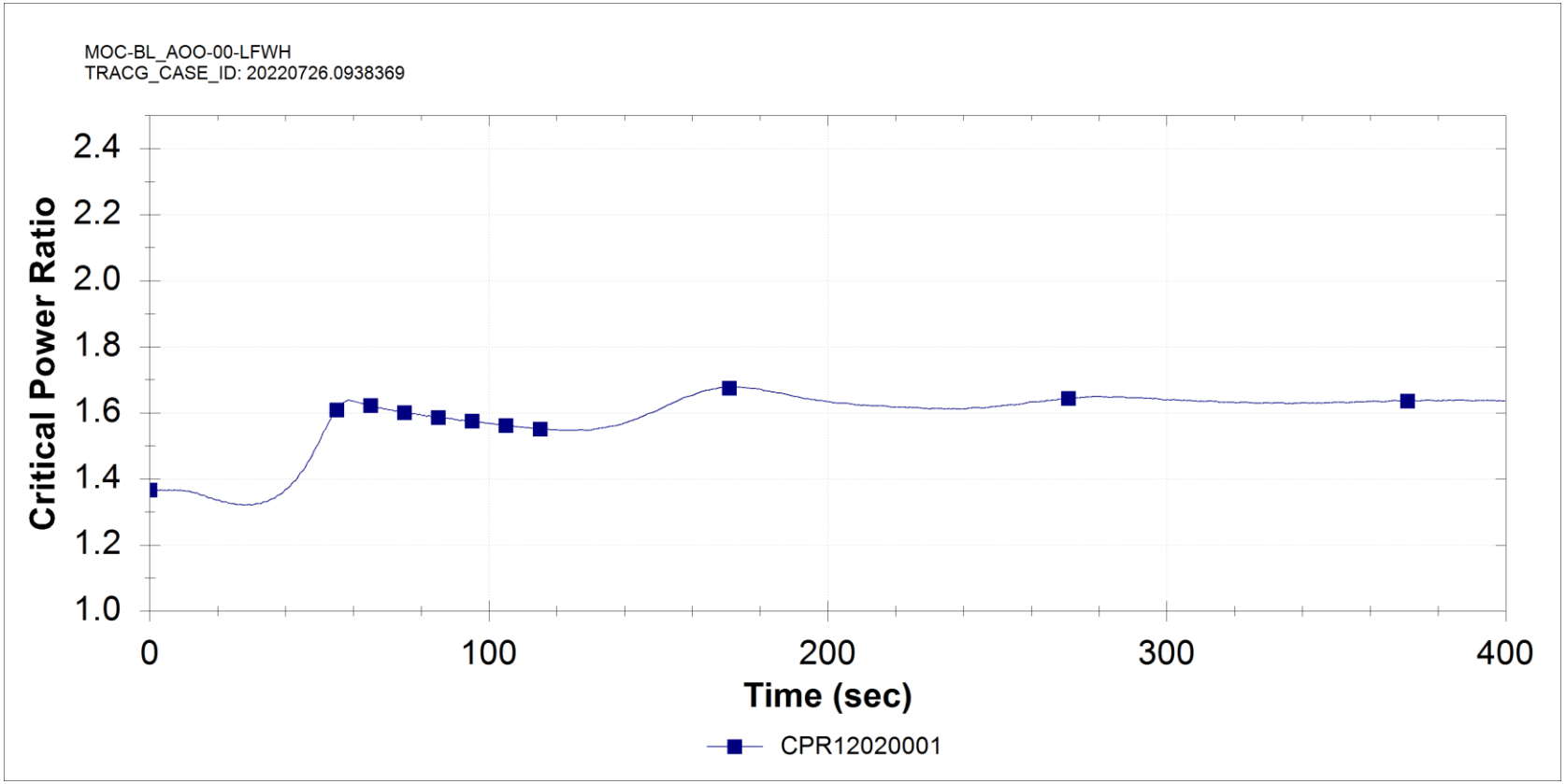


Figure 15.5-9: Loss of Feedwater Heating (AOO)

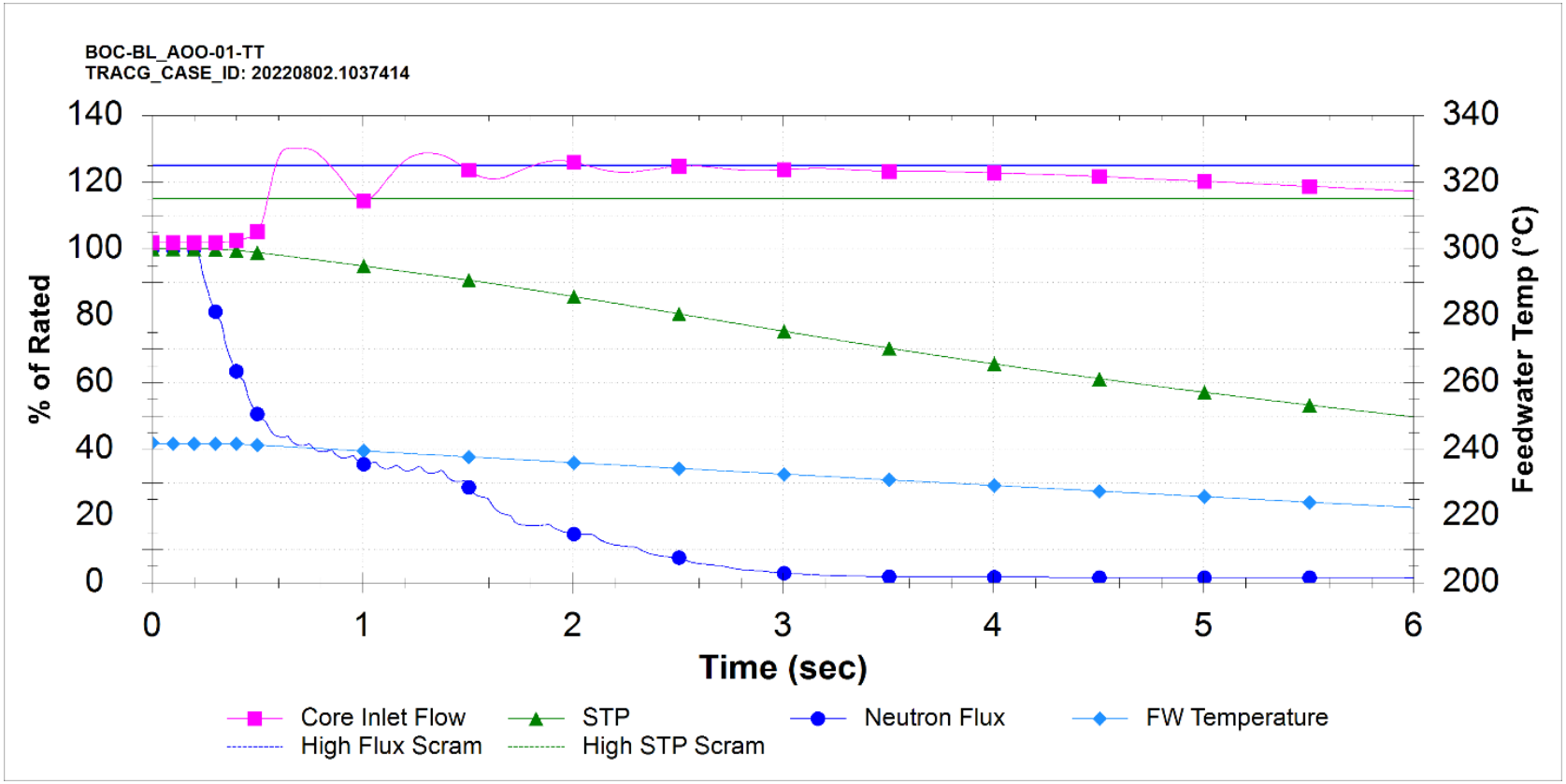


Figure 15.5-10: Turbine Trip (AOO)

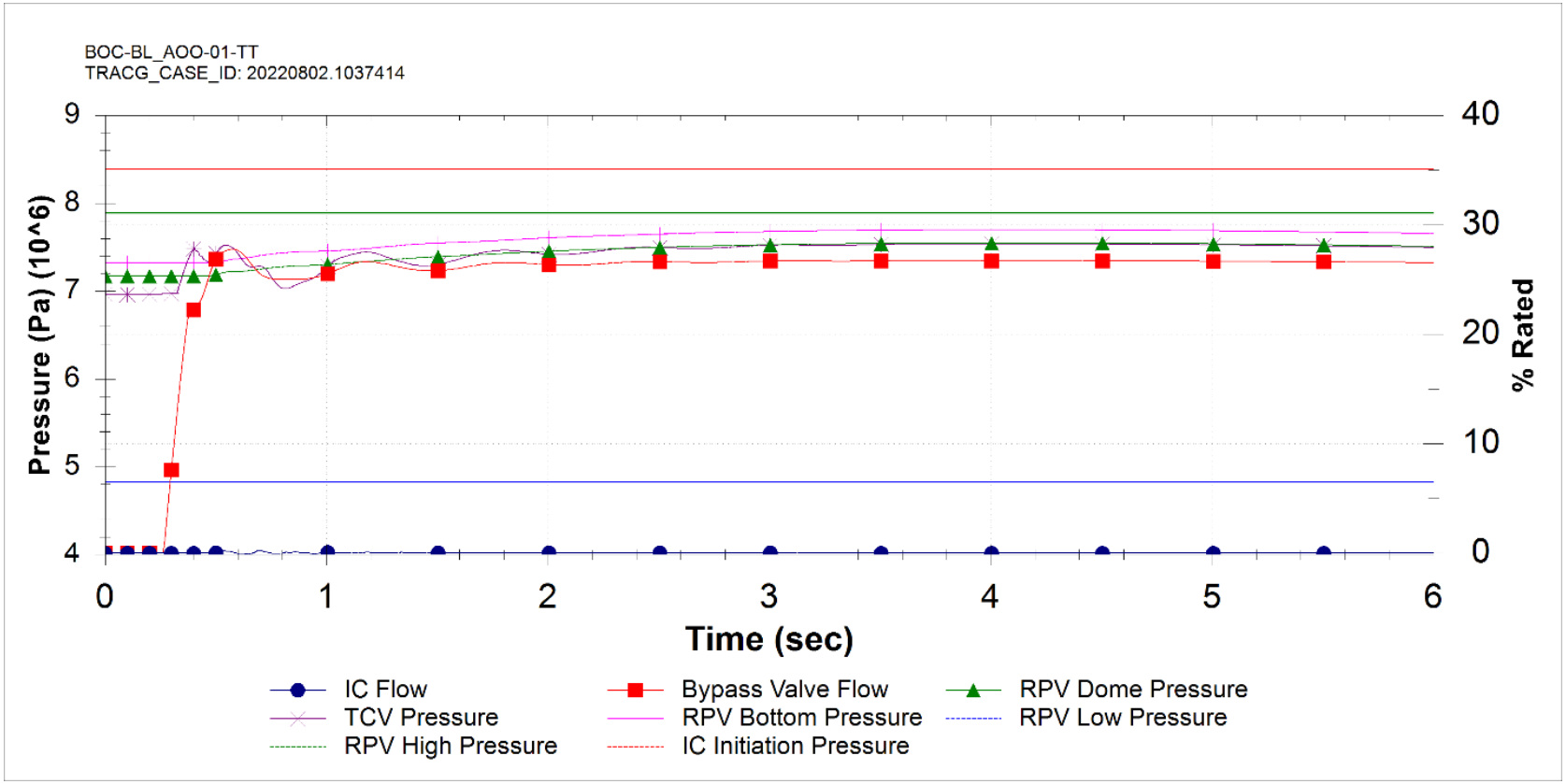


Figure 15.5-11: Turbine Trip (AOO)

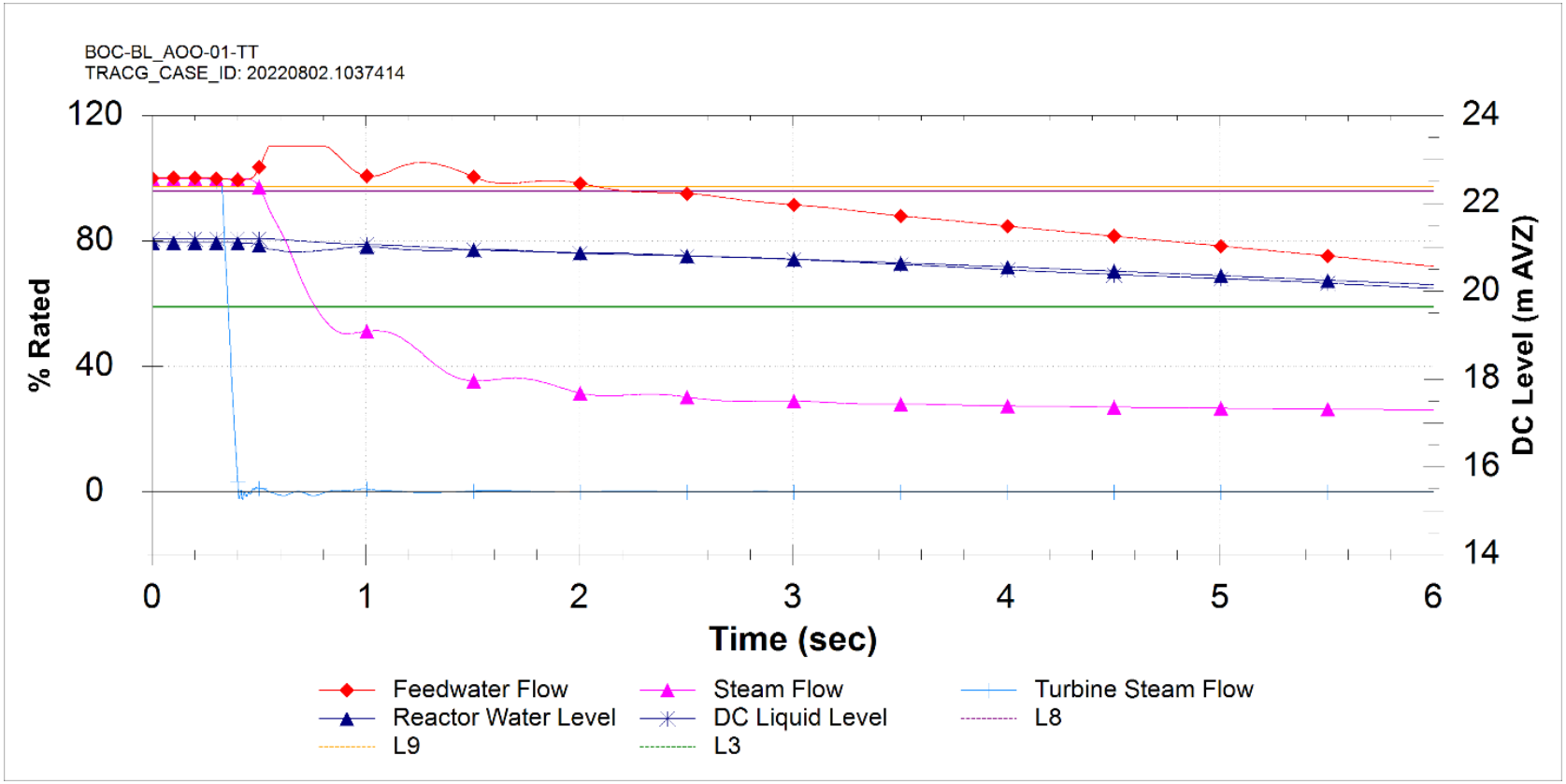


Figure 15.5-12: Turbine Trip (AOO)

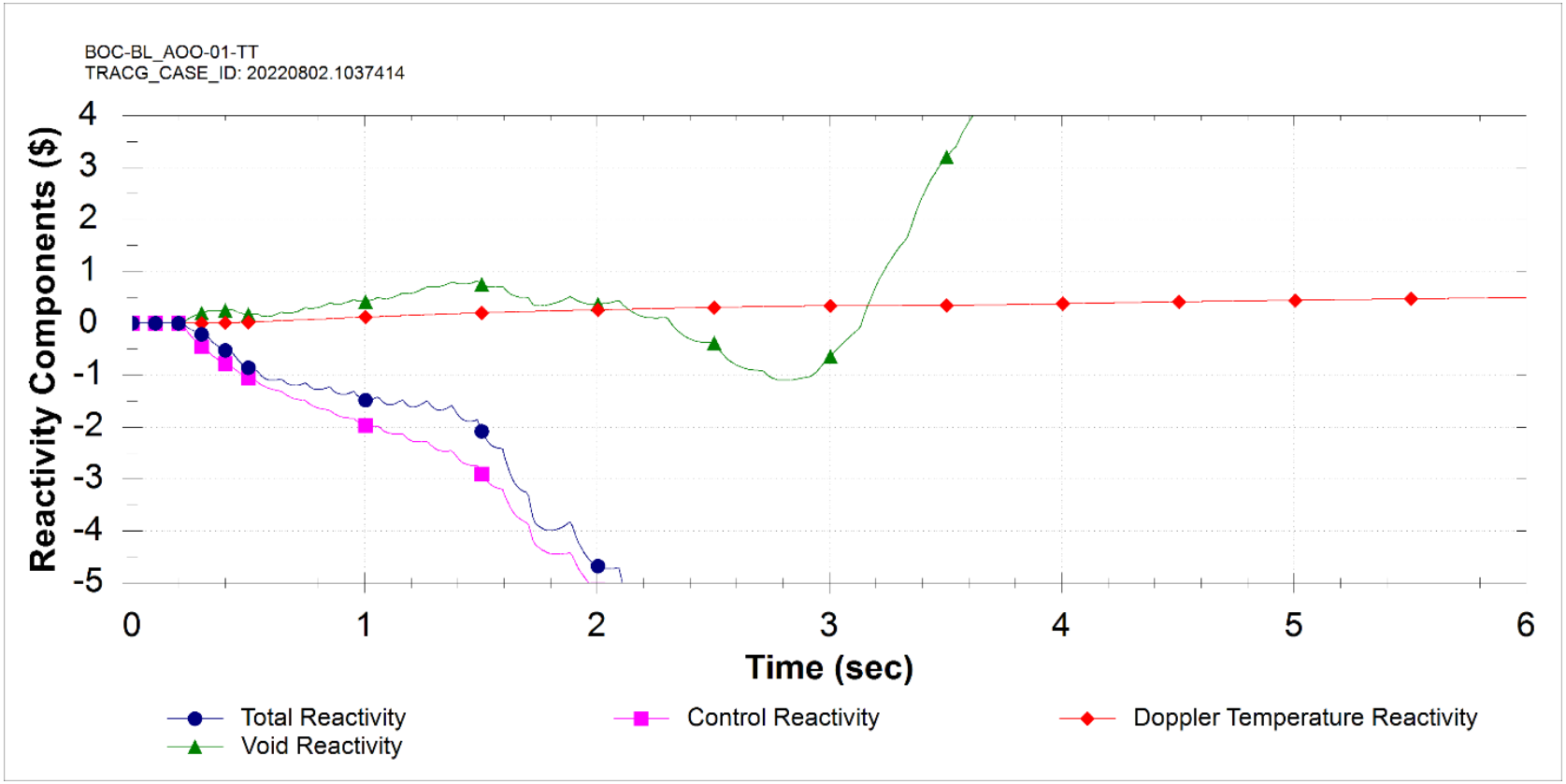


Figure 15.5-13: Turbine Trip (AOO)

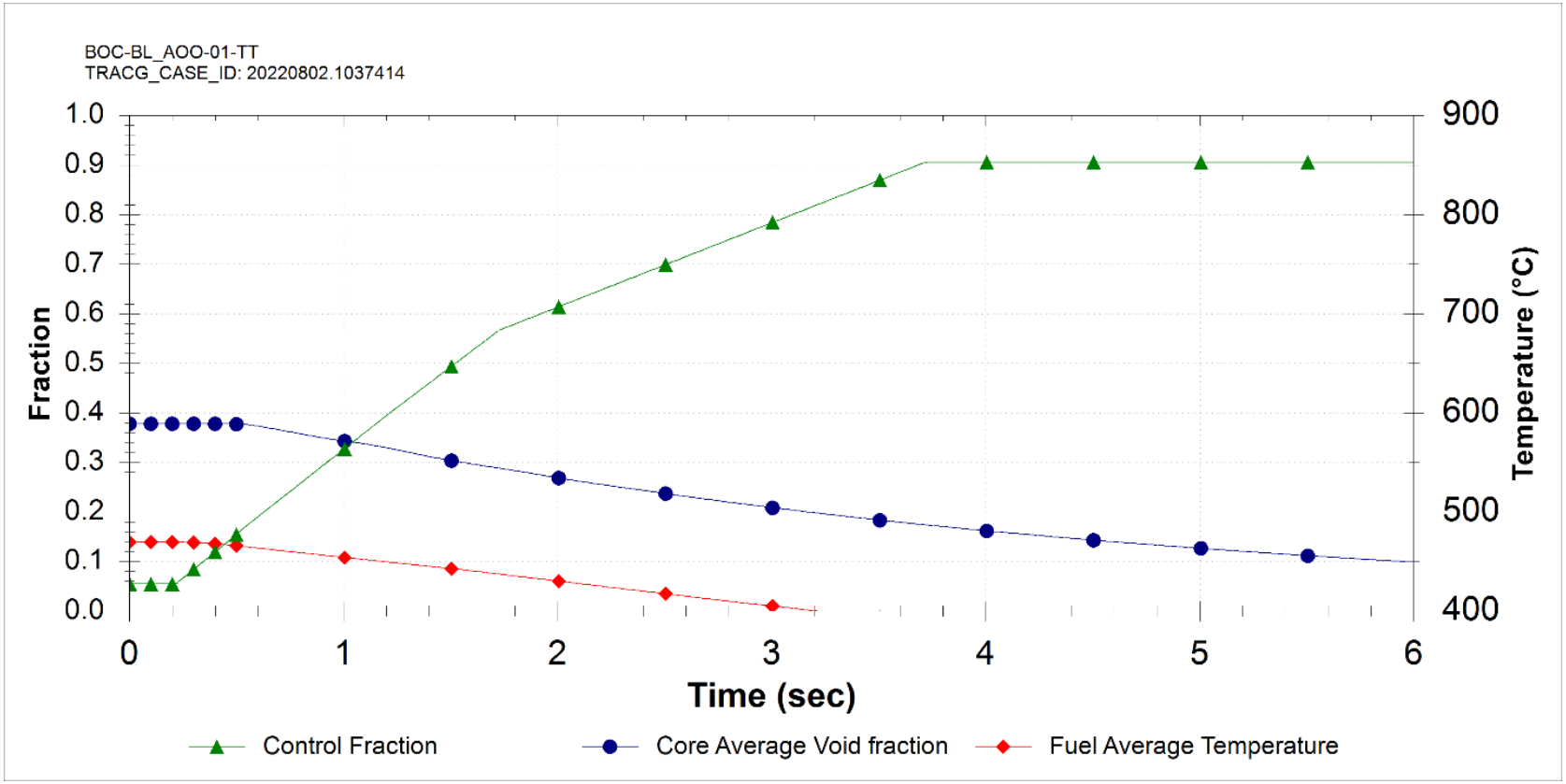


Figure 15.5-14: Turbine Trip (AOO)

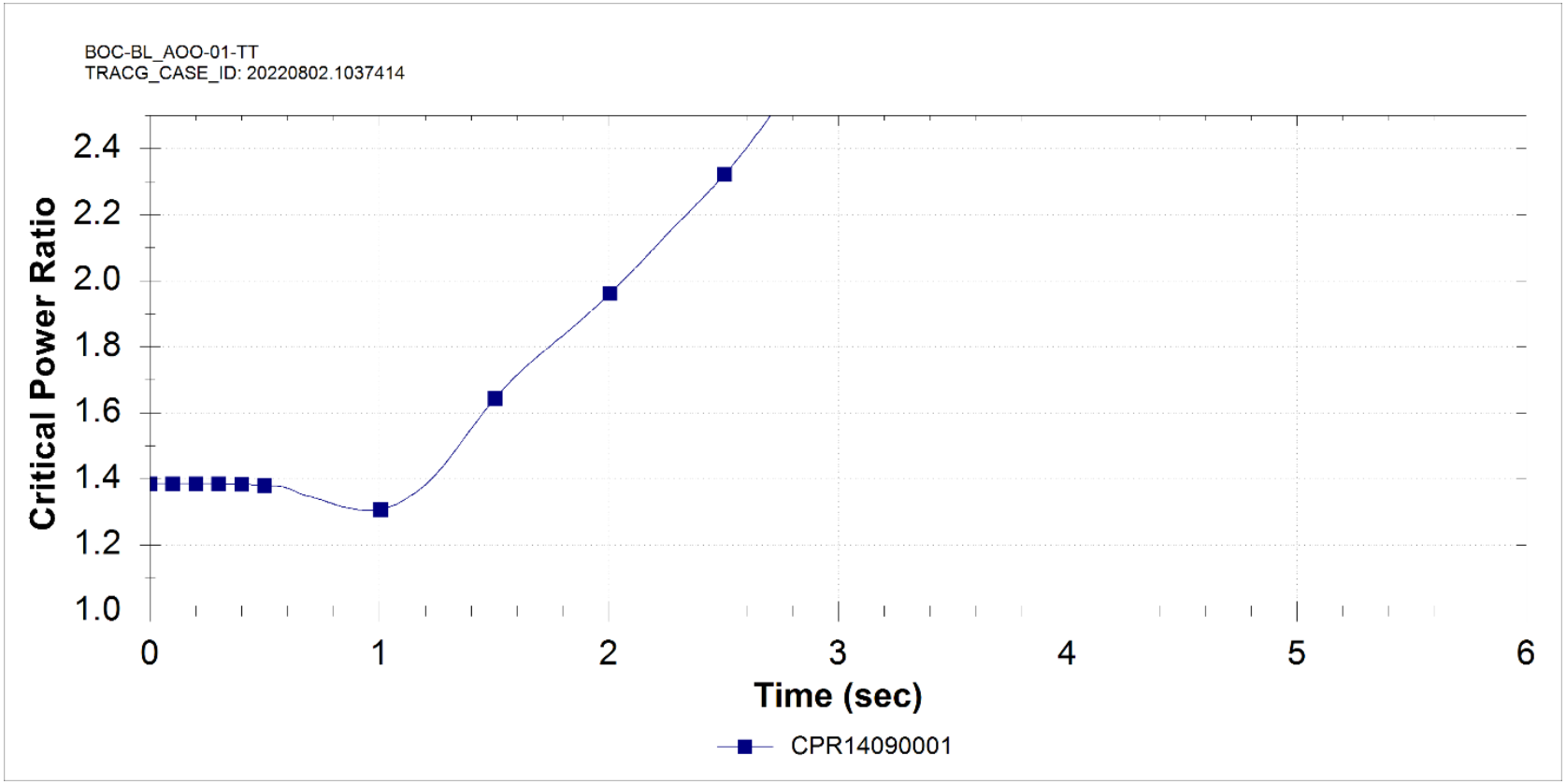


Figure 15.5-15: Turbine Trip (AOO)

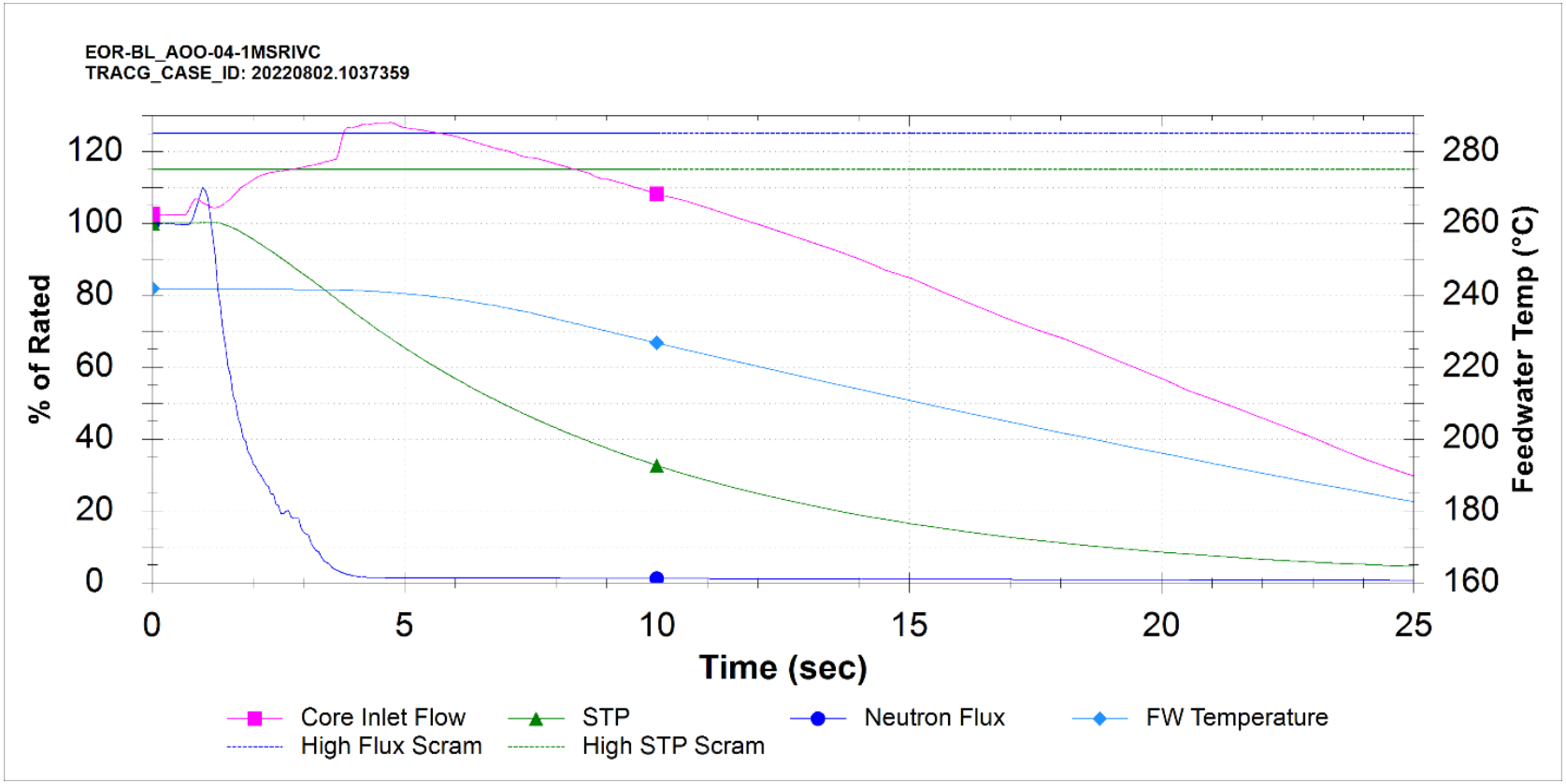


Figure 15.5-16: Closure of One Main Steam Reactor Isolation Valve (AOO)

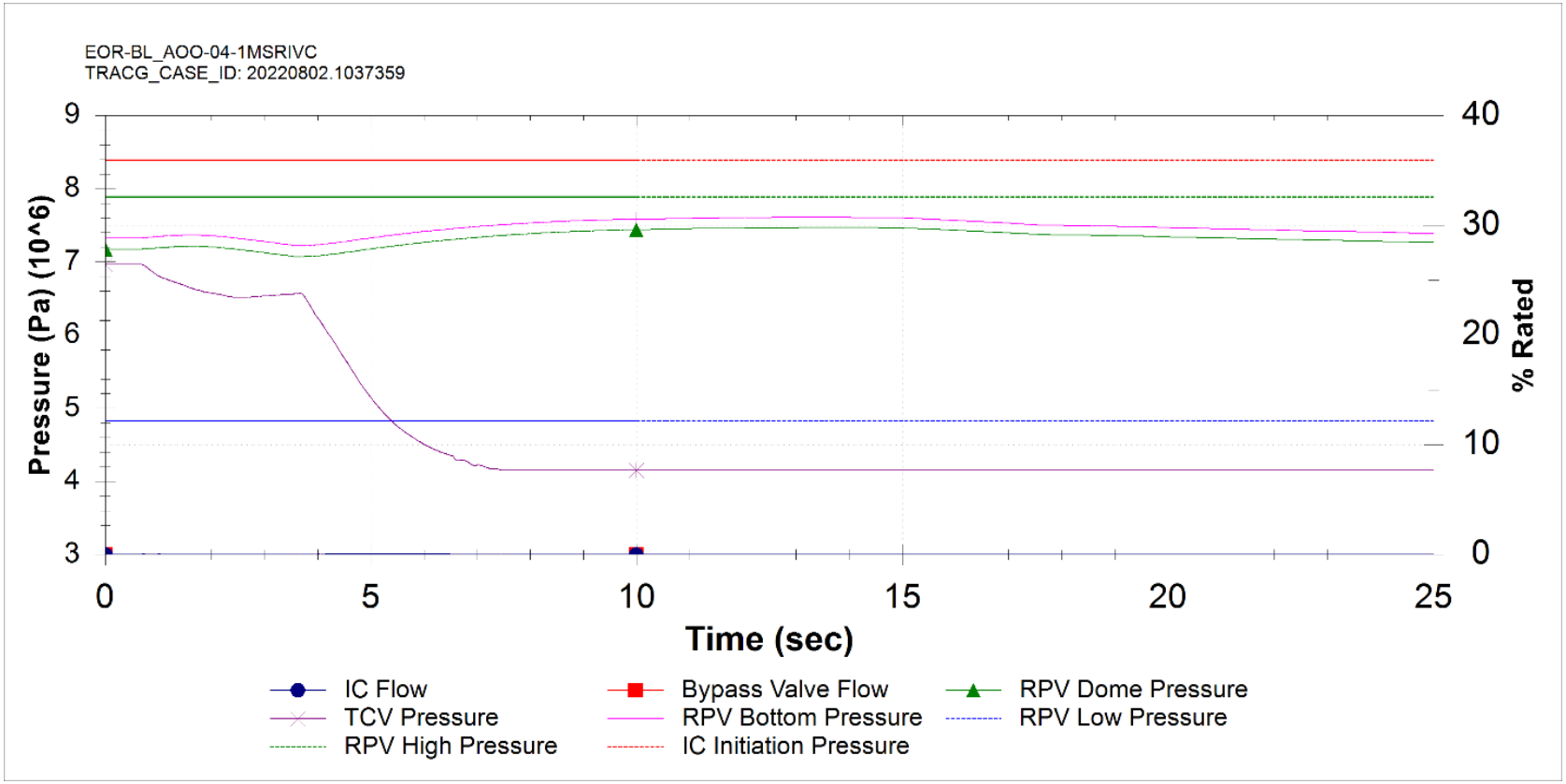


Figure 15.5-17: Closure of One Main Steam Reactor Isolation Valve (AOO)

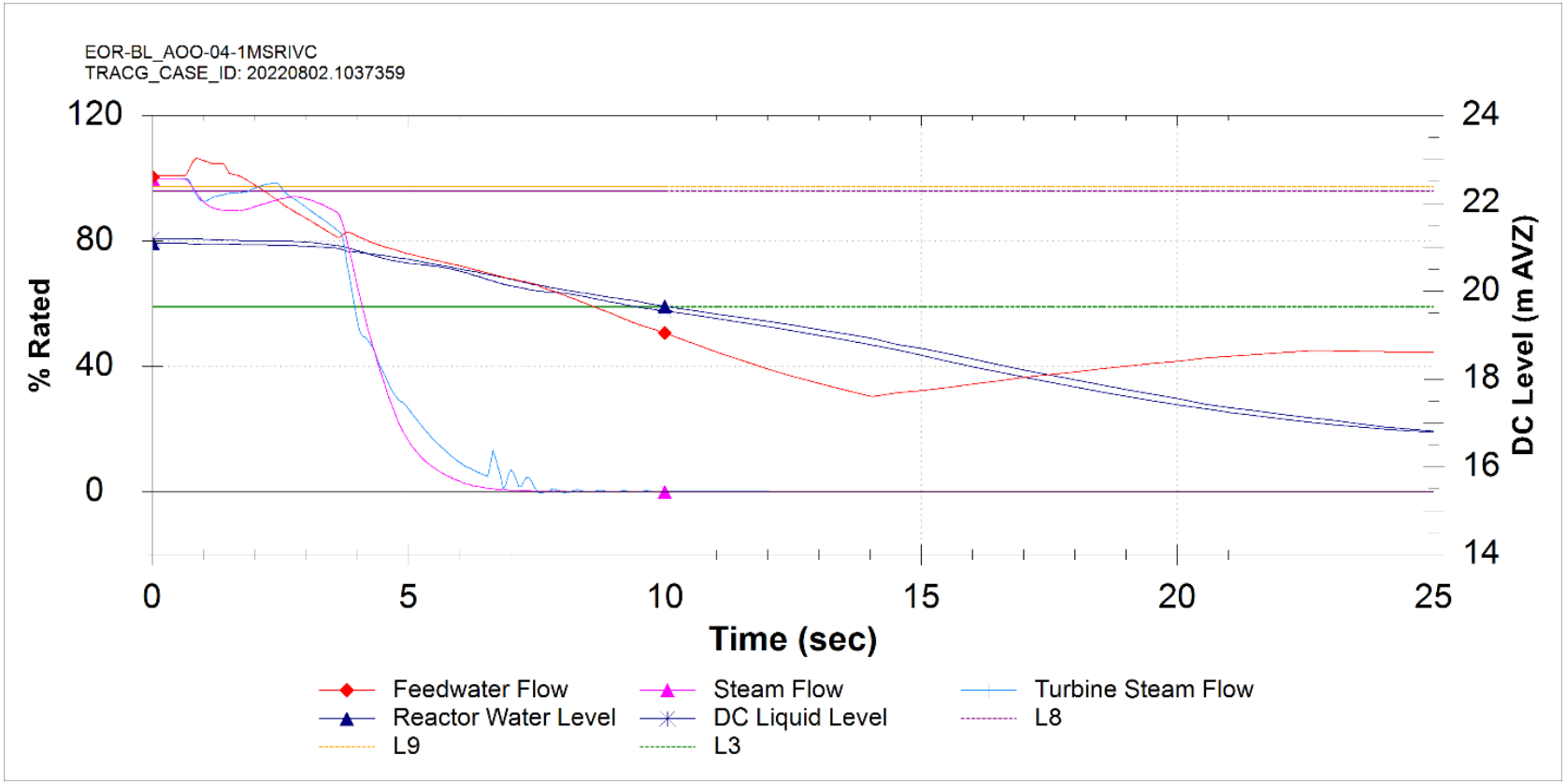


Figure 15.5-18: Closure of One Main Steam Reactor Isolation Valve (AOO)

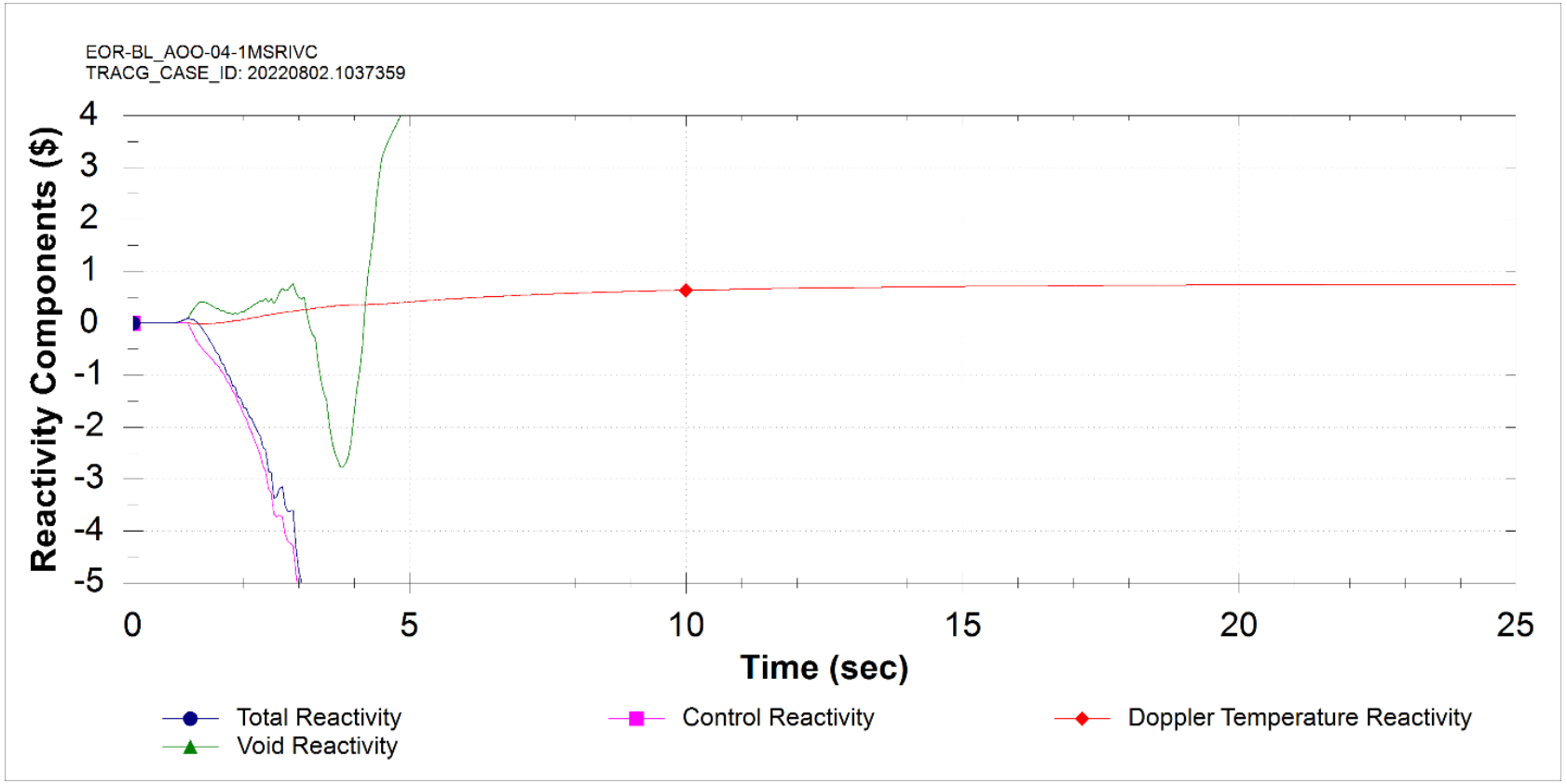


Figure 15.5-19: Closure of One Main Steam Reactor Isolation Valve (AOO)

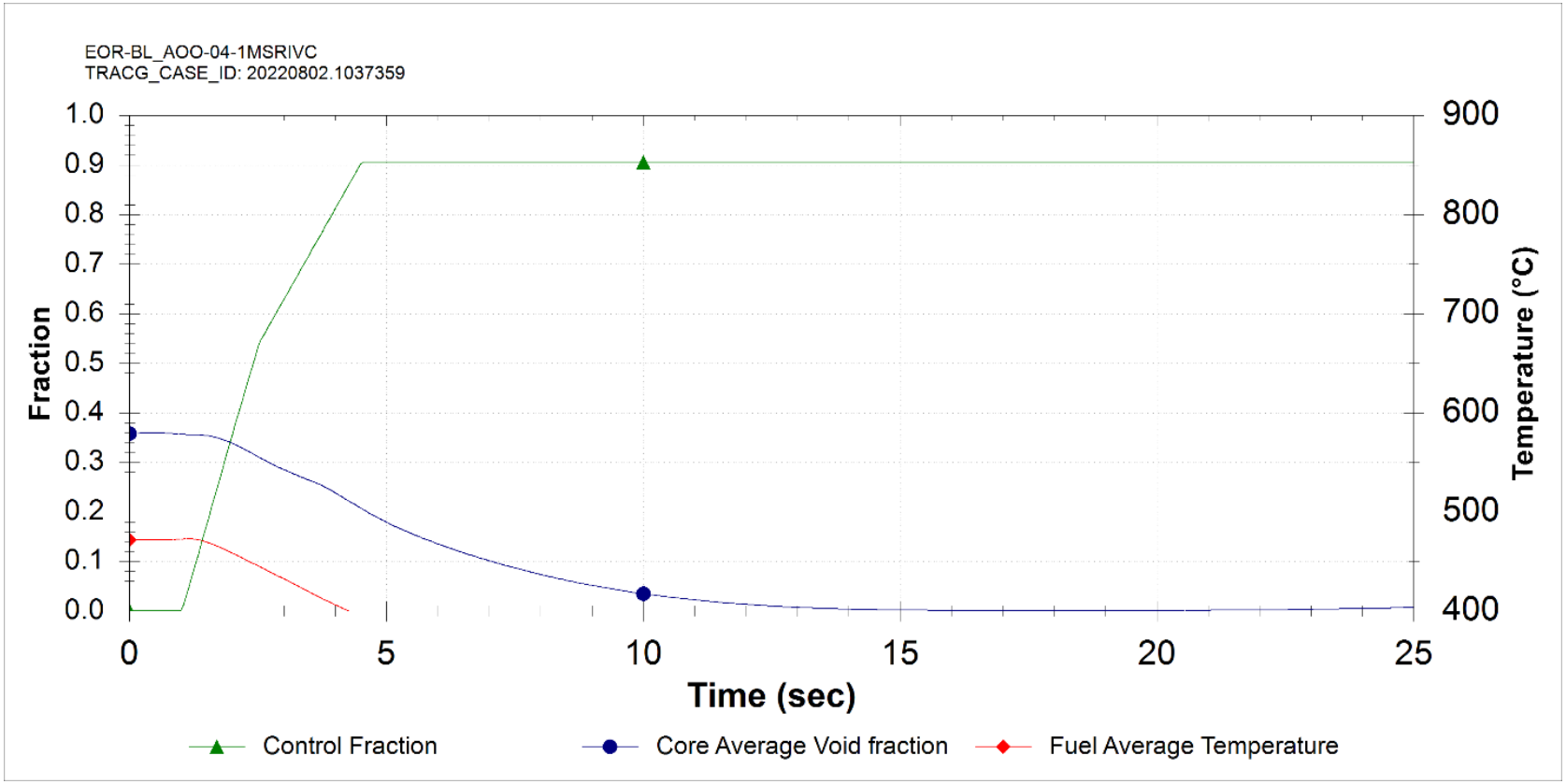


Figure 15.5-20: Closure of One Main Steam Reactor Isolation Valve (AOO)

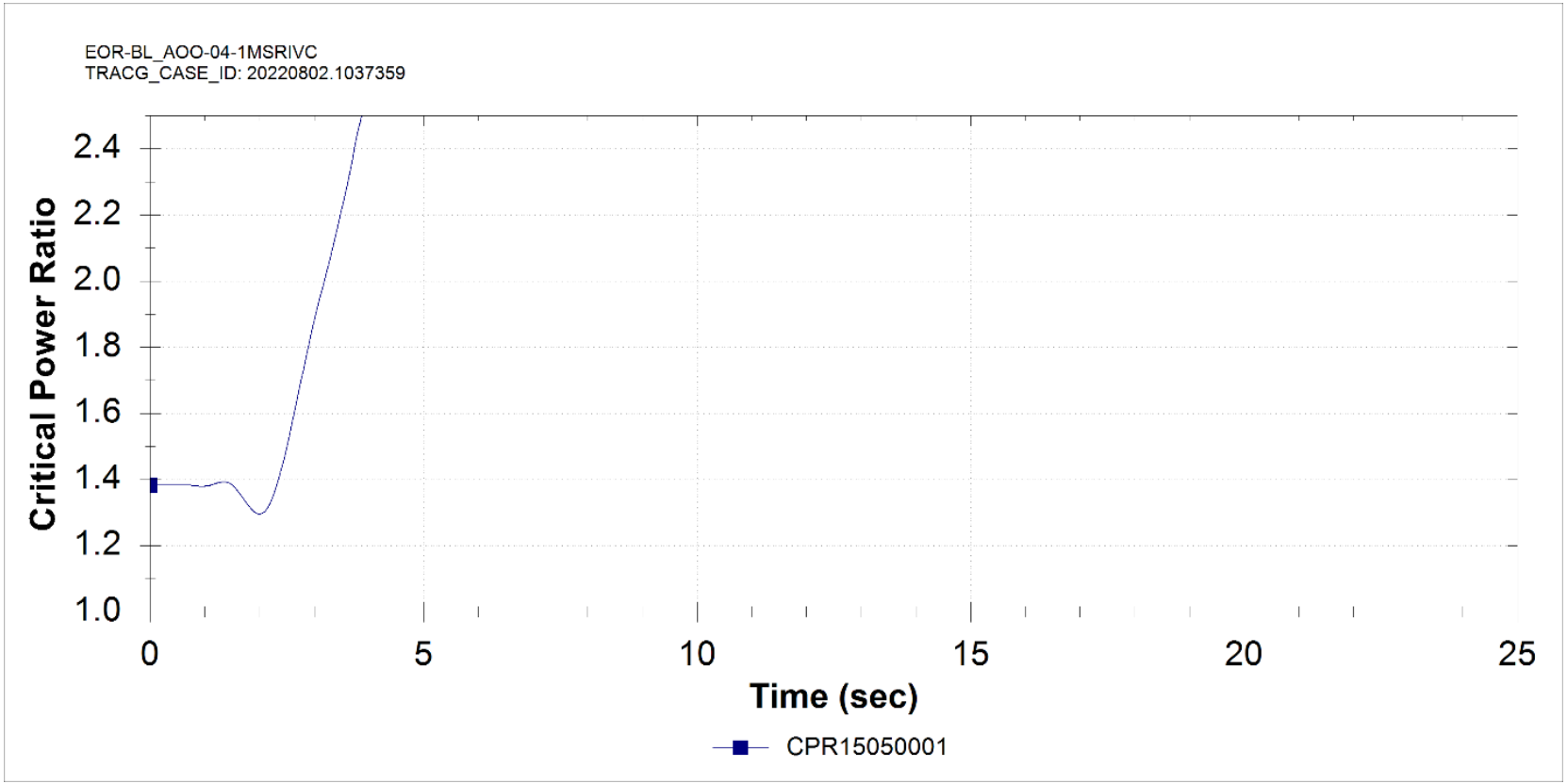


Figure 15.5-21: Closure of One Main Steam Reactor Isolation Valve (AOO)

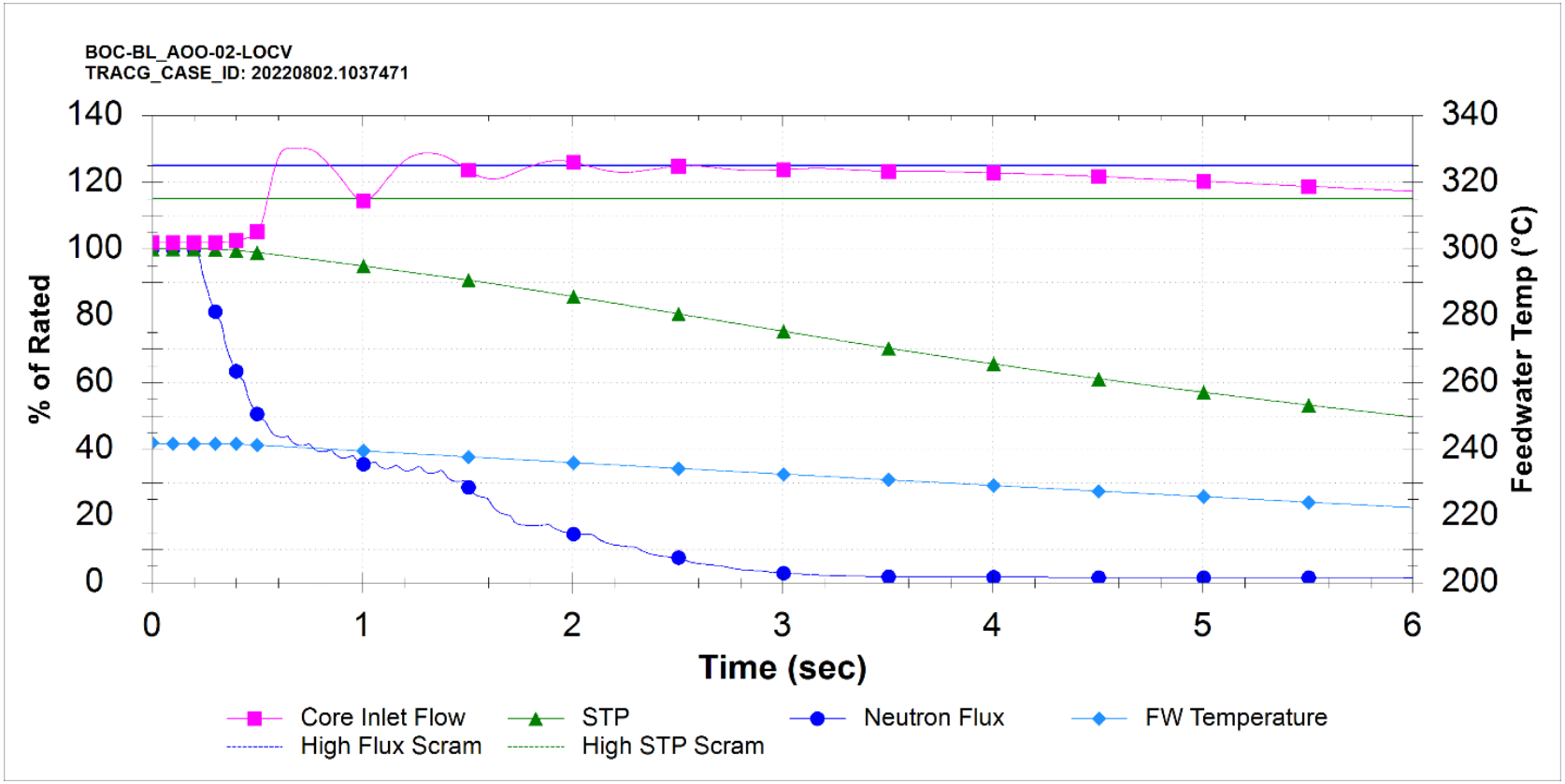


Figure 15.5-22: Loss of Condenser Vacuum (AOO)

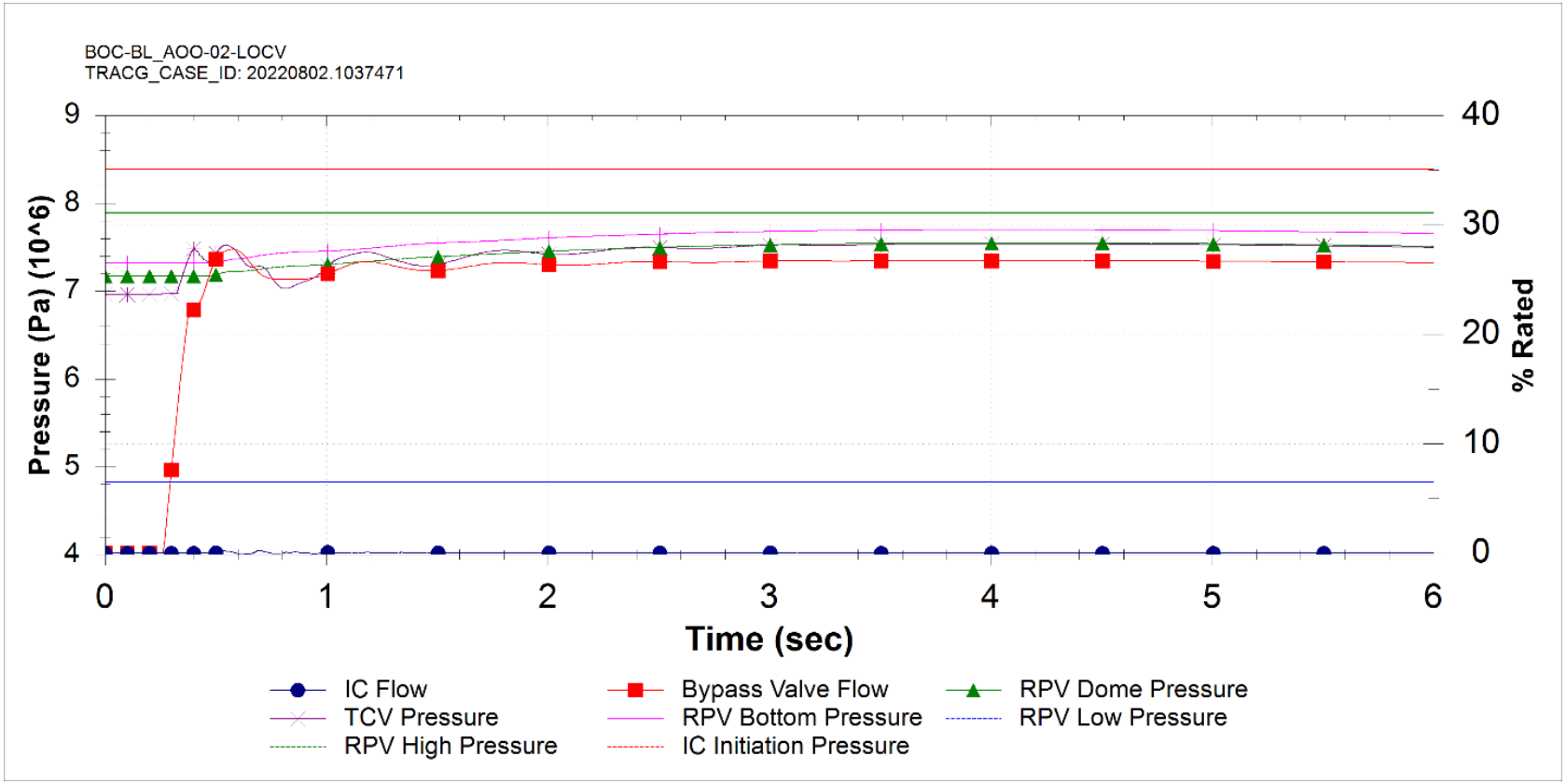


Figure 15.5-23: Loss of Condenser Vacuum (AOO)

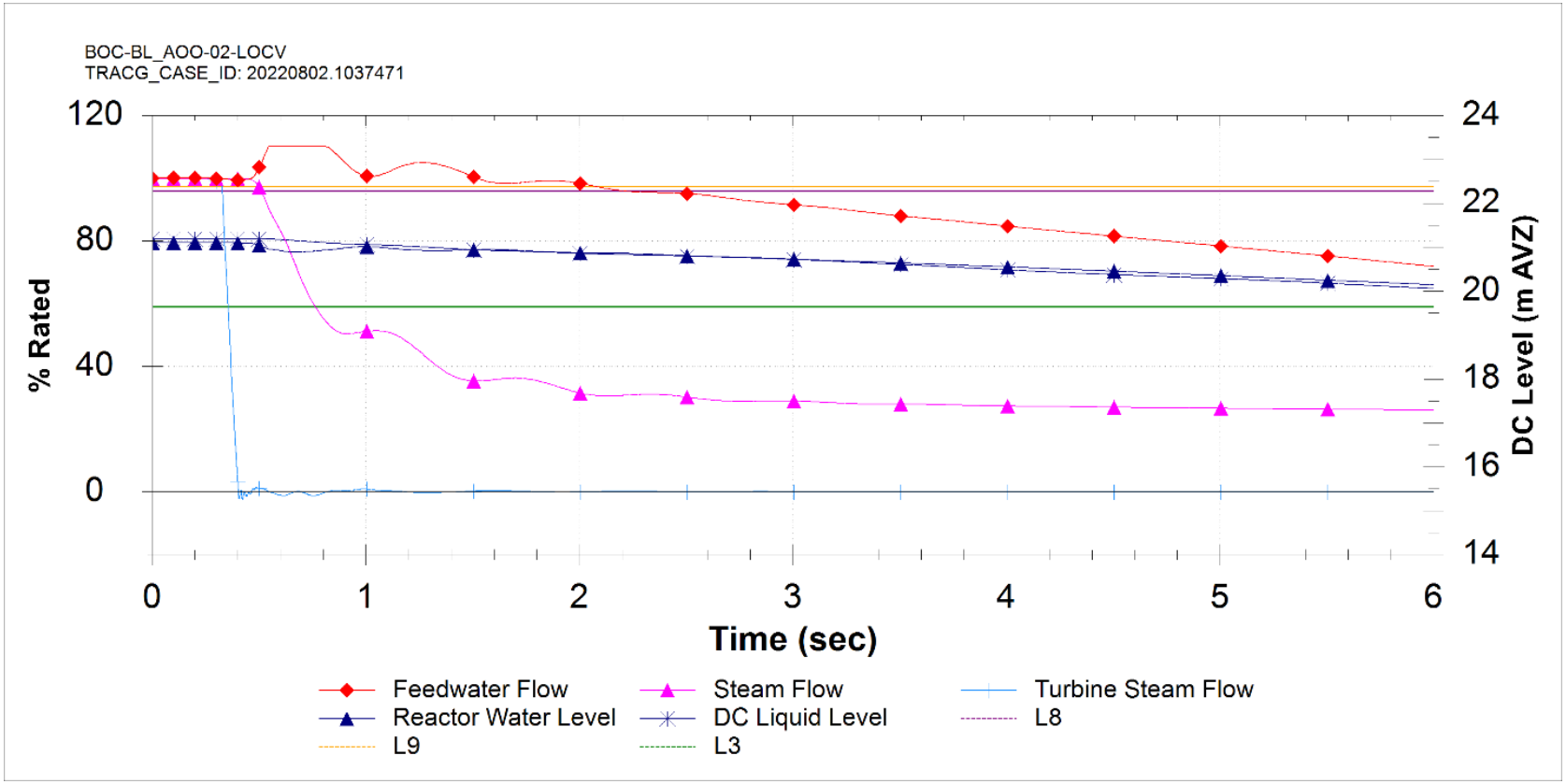


Figure 15.5-24: Loss of Condenser Vacuum (AOO)

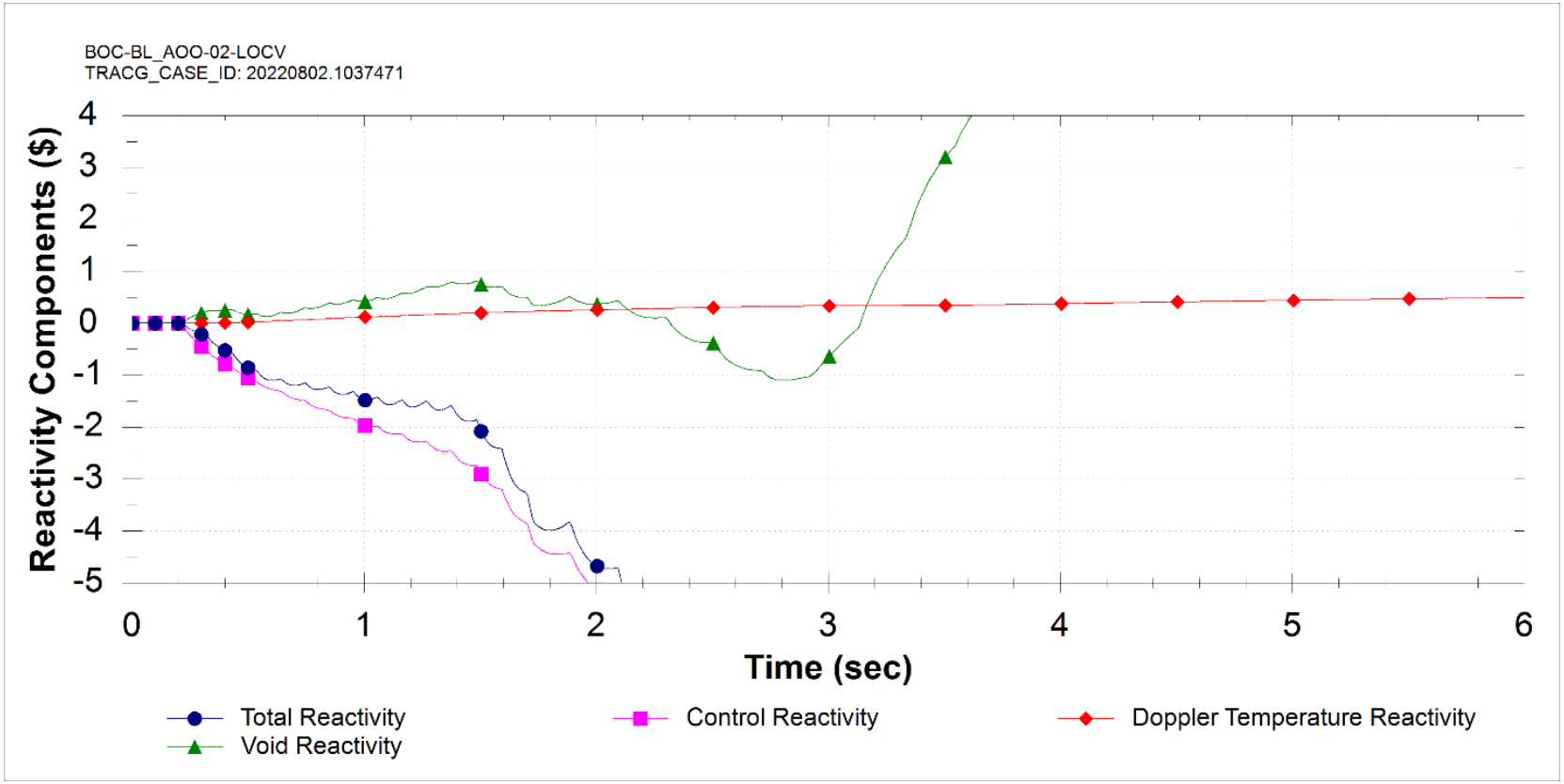


Figure 15.5-25: Loss of Condenser Vacuum (AOO)

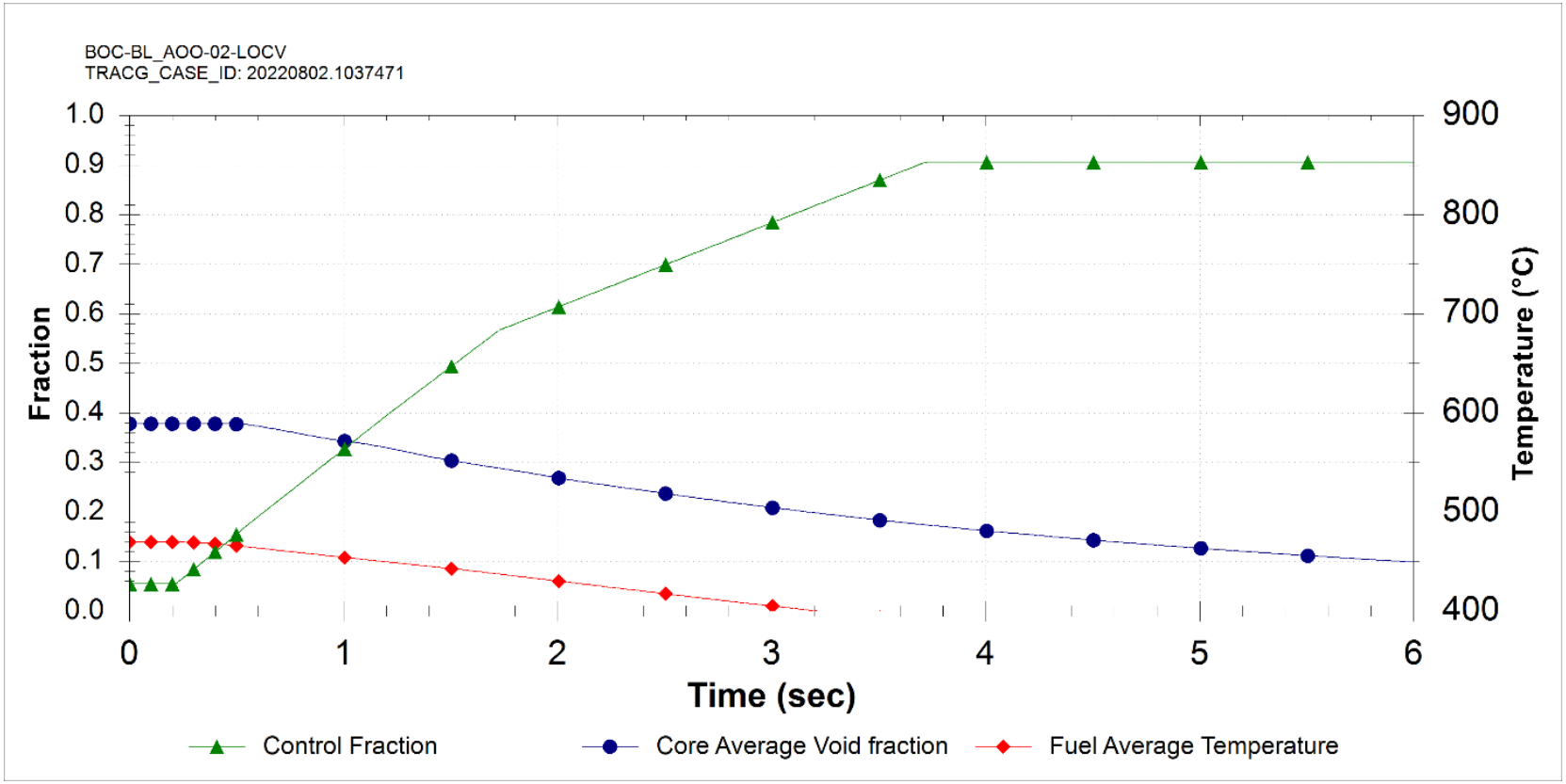


Figure 15.5-26: Loss of Condenser Vacuum (AOO)

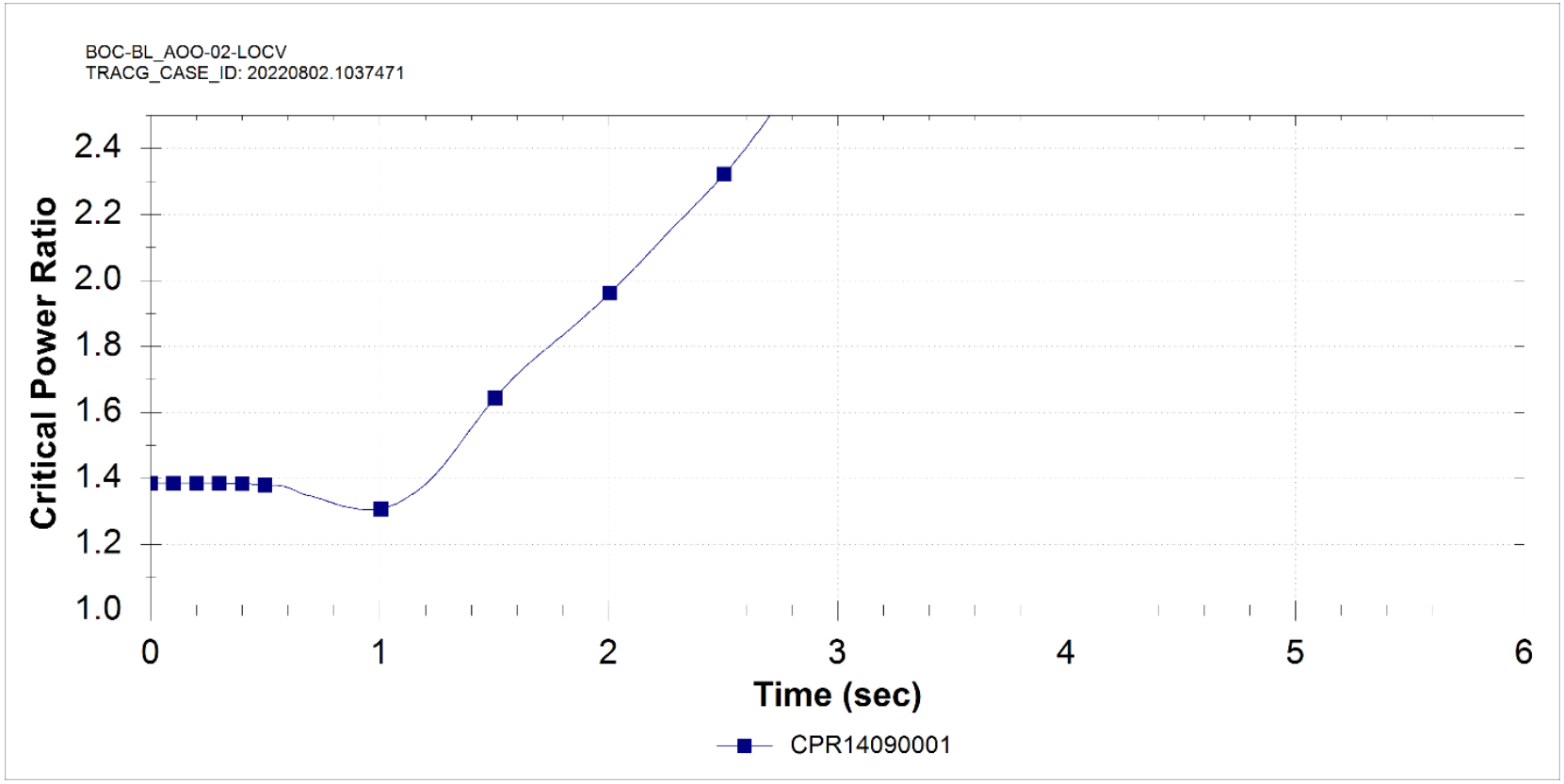


Figure 15.5-27: Loss of Condenser Vacuum (AOO)

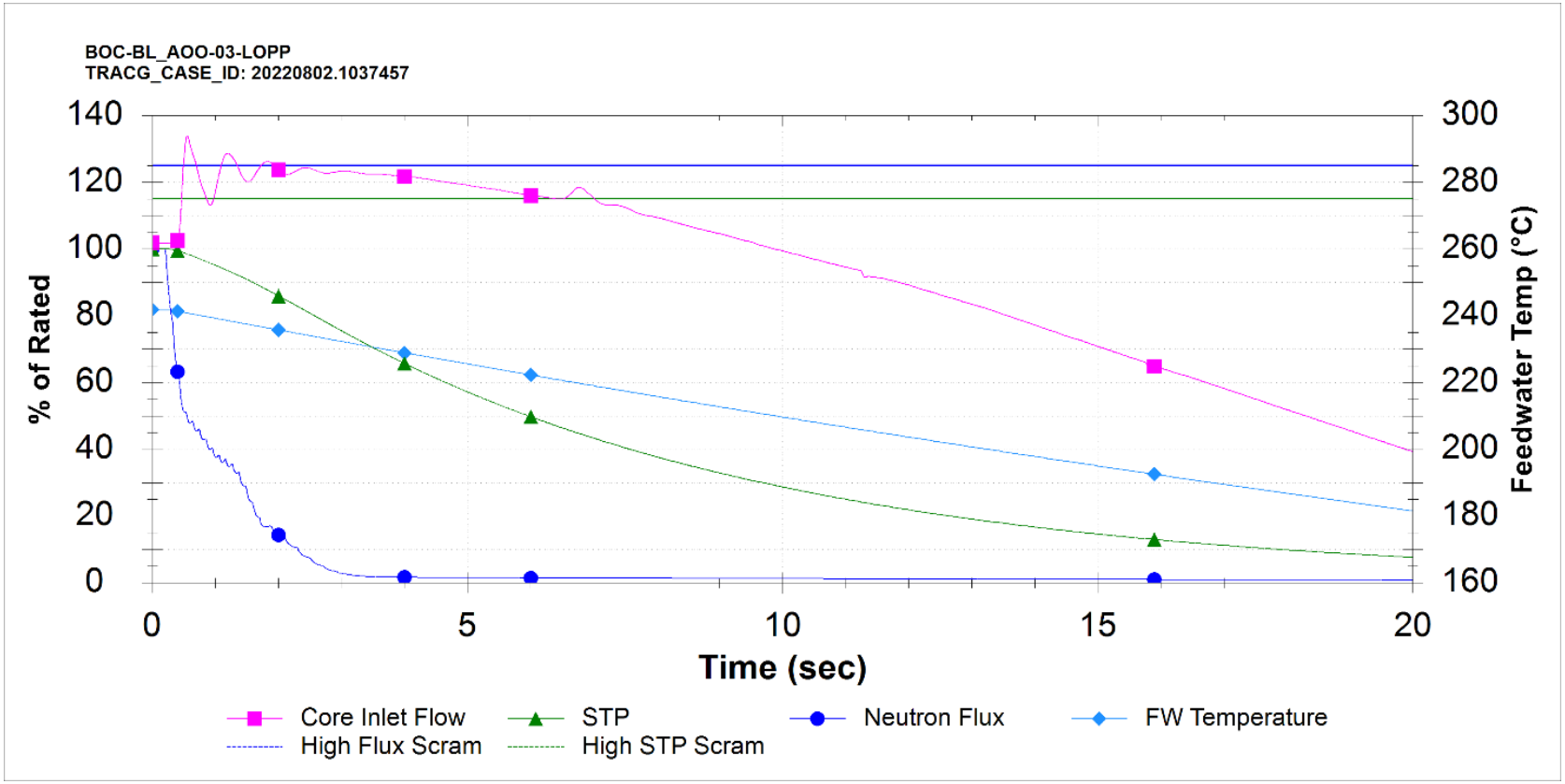


Figure 15.5-28: Loss of Preferred Power (AOO)

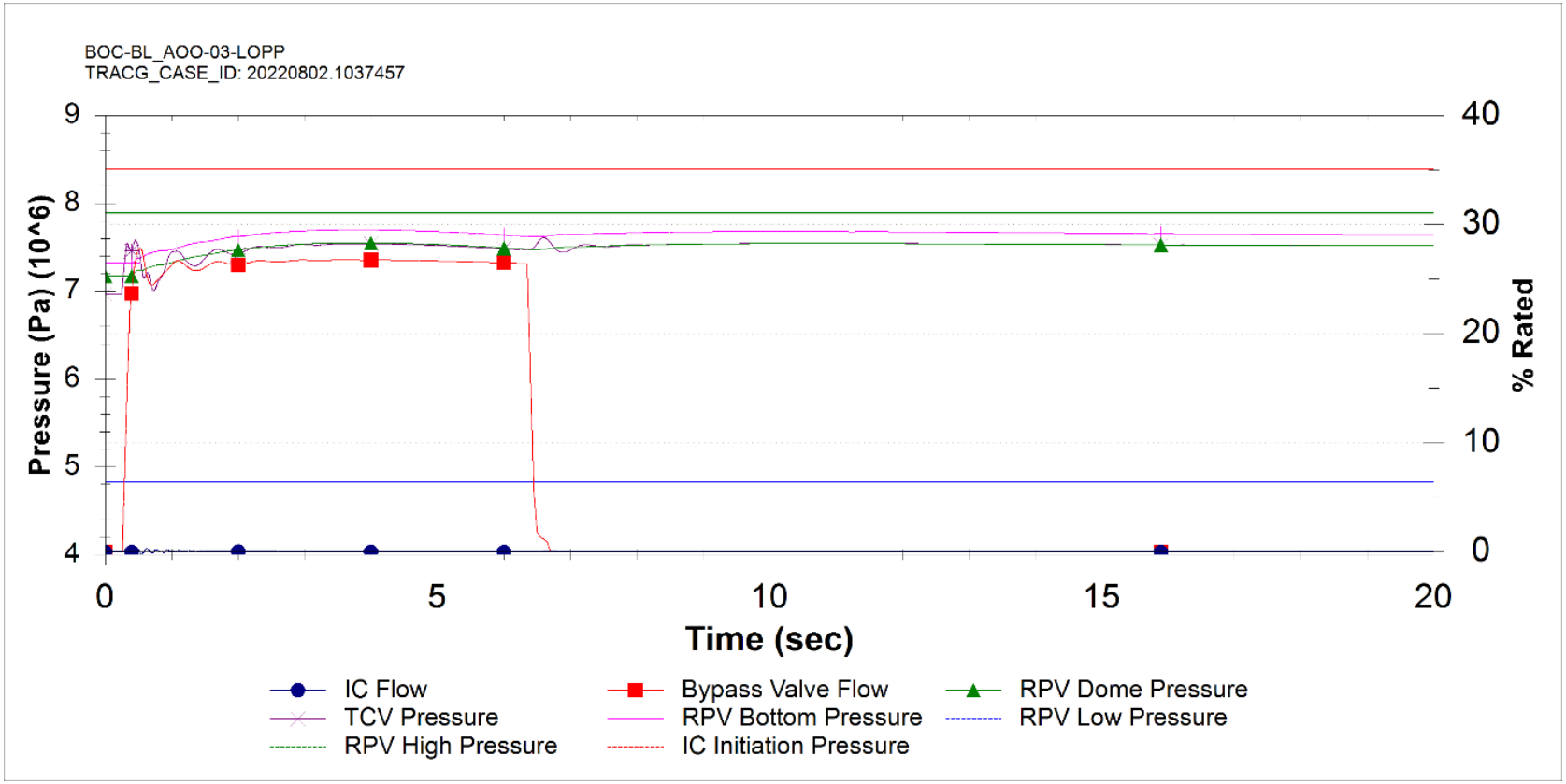


Figure 15.5-29: Loss of Preferred Power (AOO)

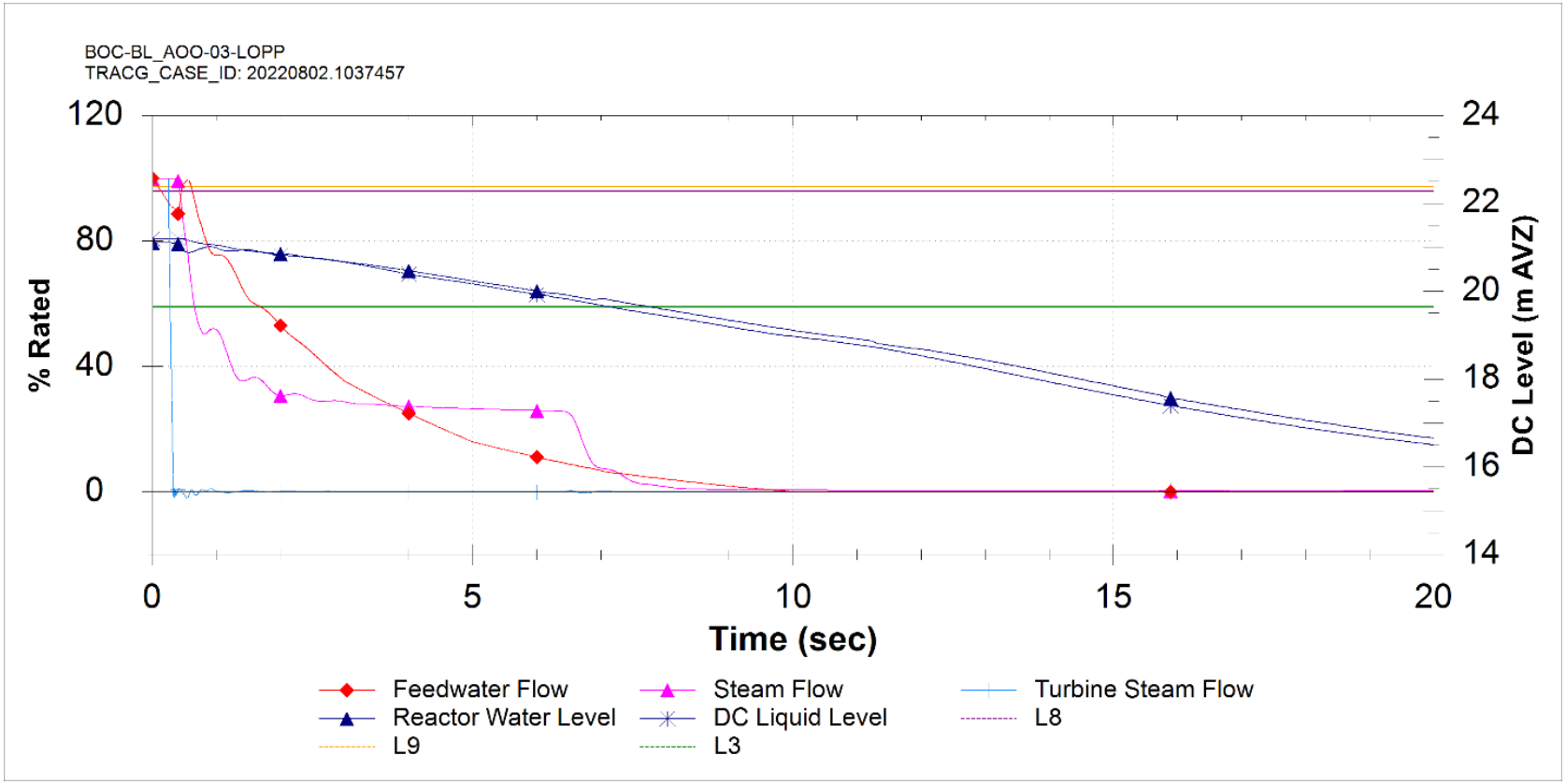


Figure 15.5-30: Loss of Preferred Power (AOO)

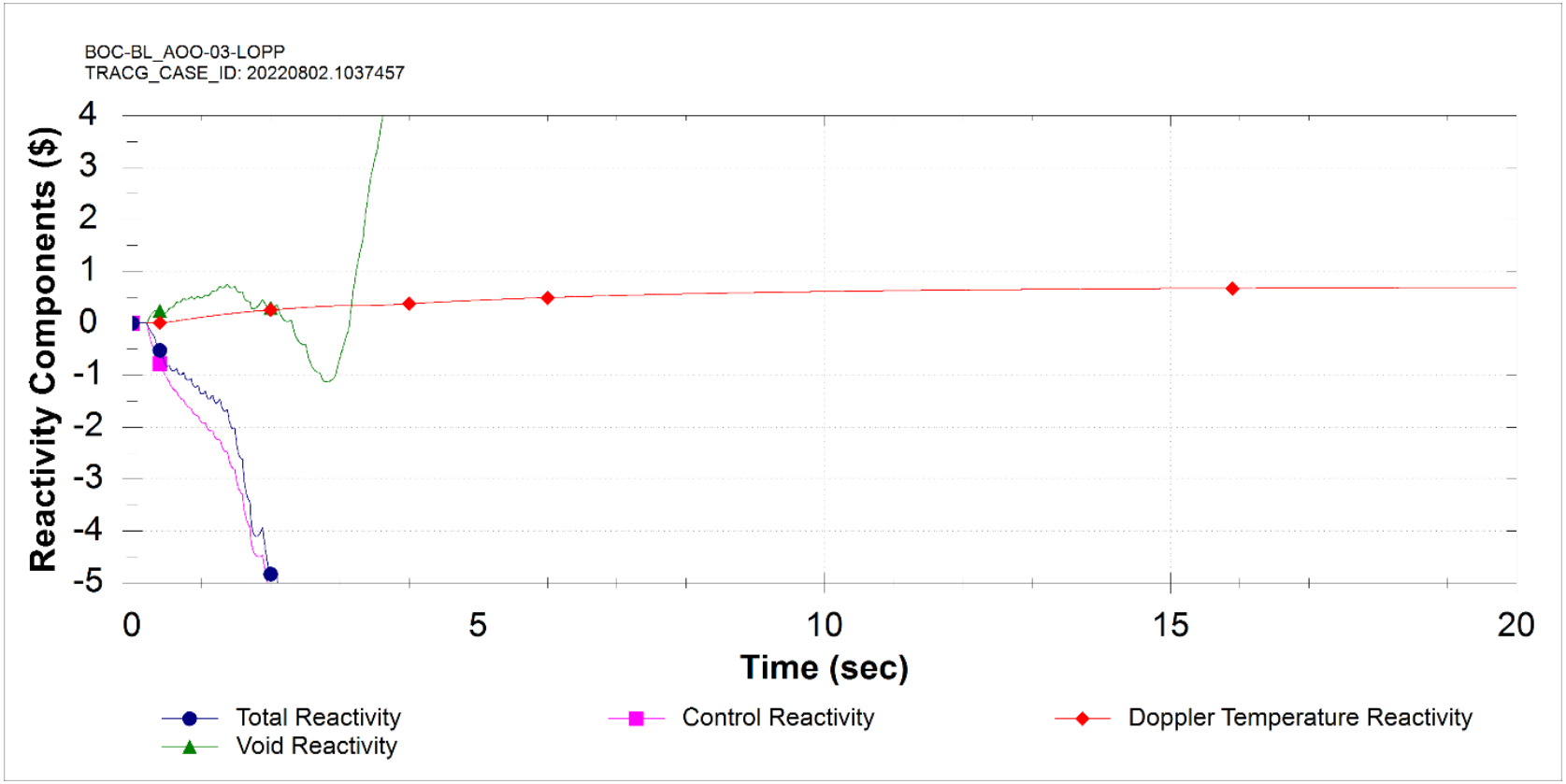


Figure 15.5-31: Loss of Preferred Power (AOO)

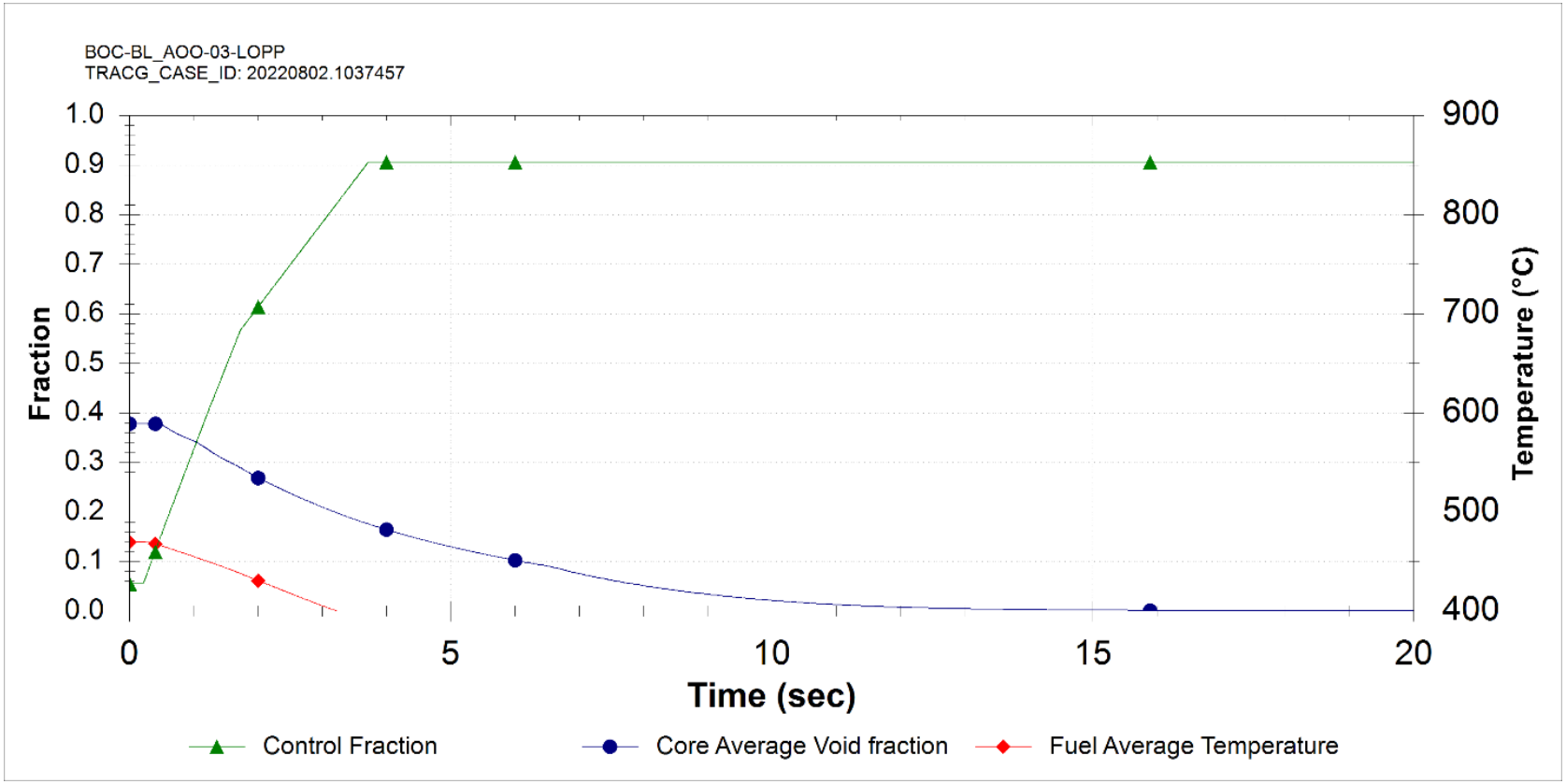


Figure 15.5-32: Loss of Preferred Power (AOO)

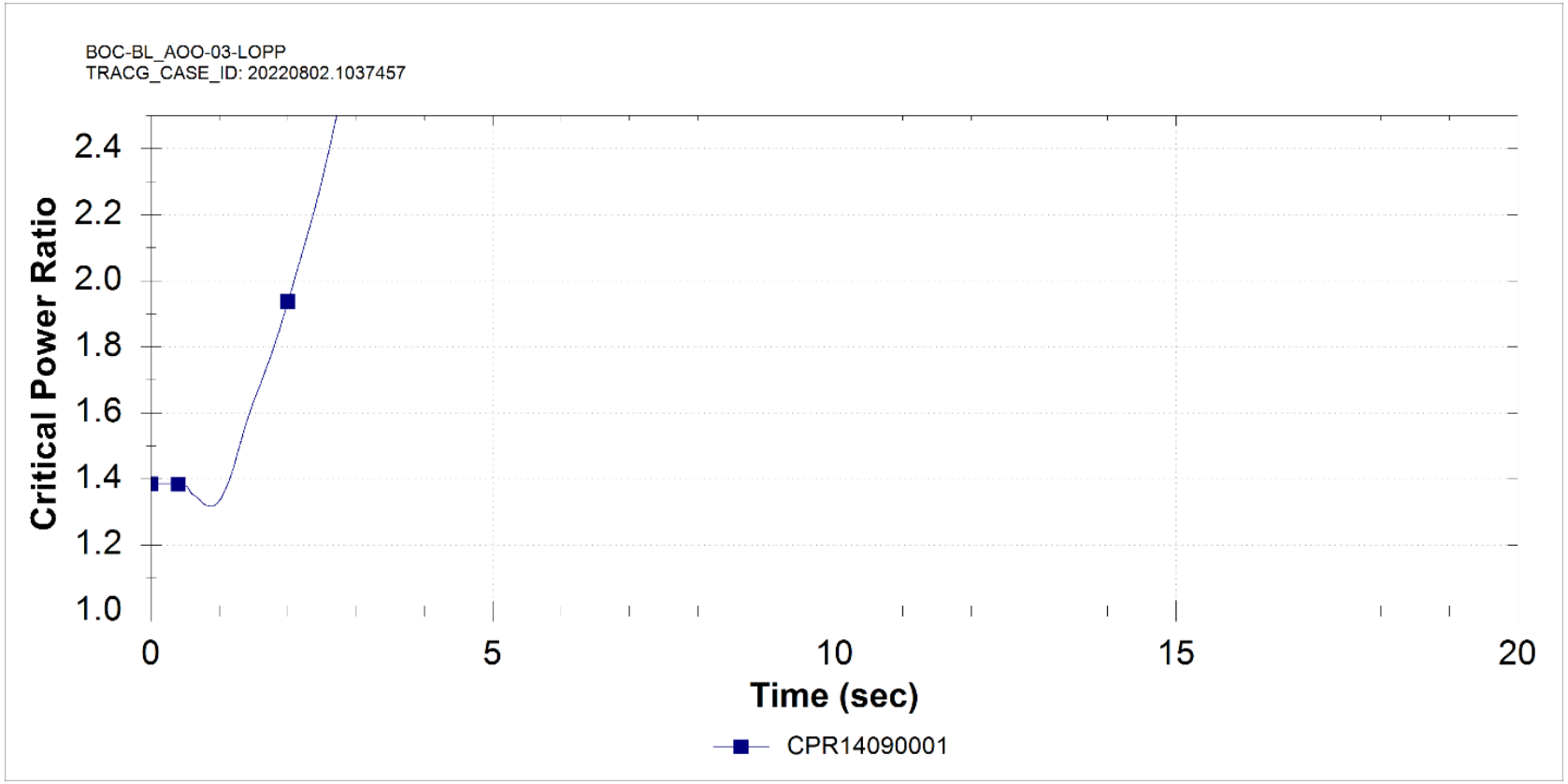


Figure 15.5-33: Loss of Preferred Power (AOO)

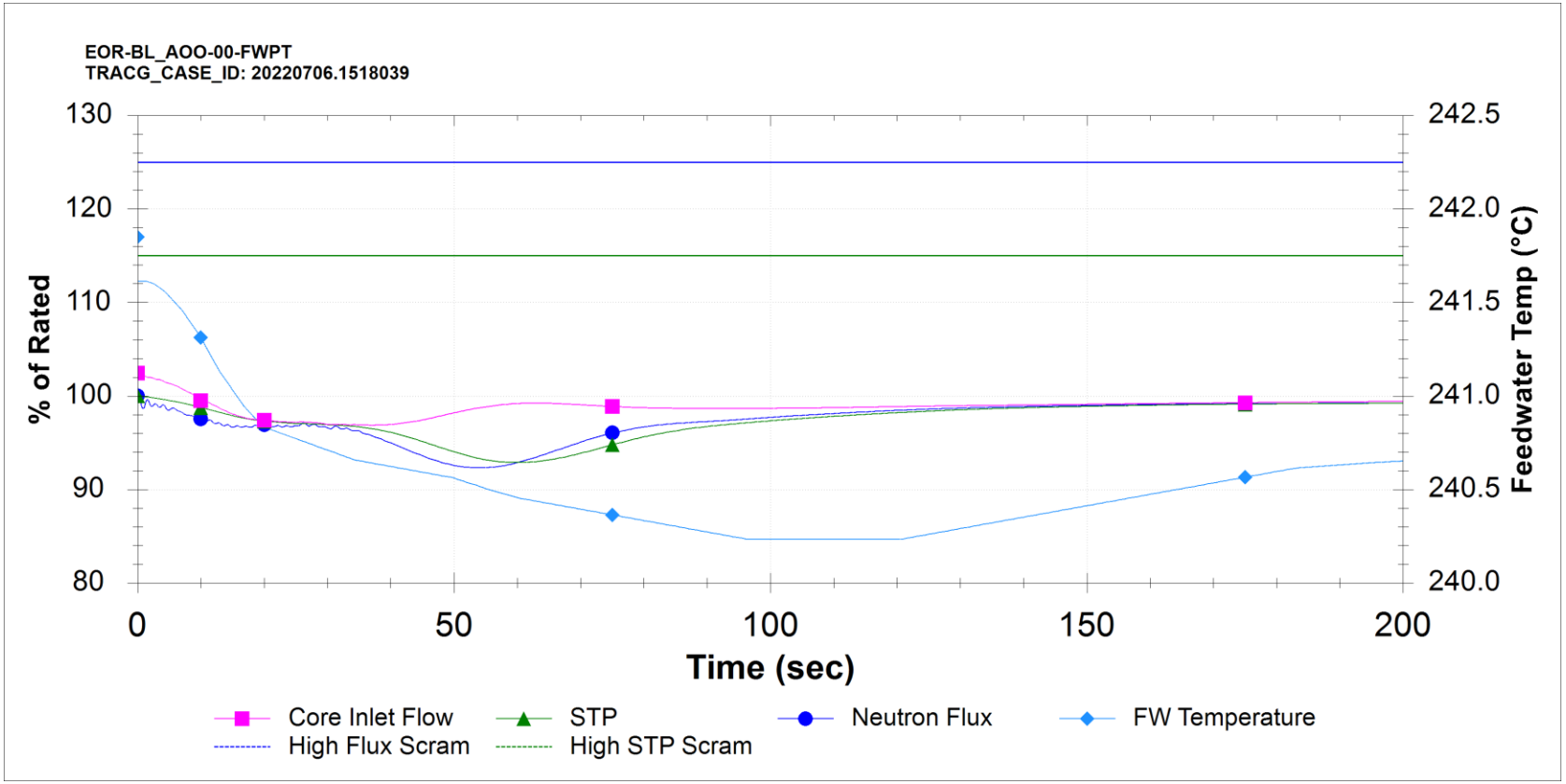


Figure 15.5-34: Feedwater Pump Trip – One Pump (AOO)

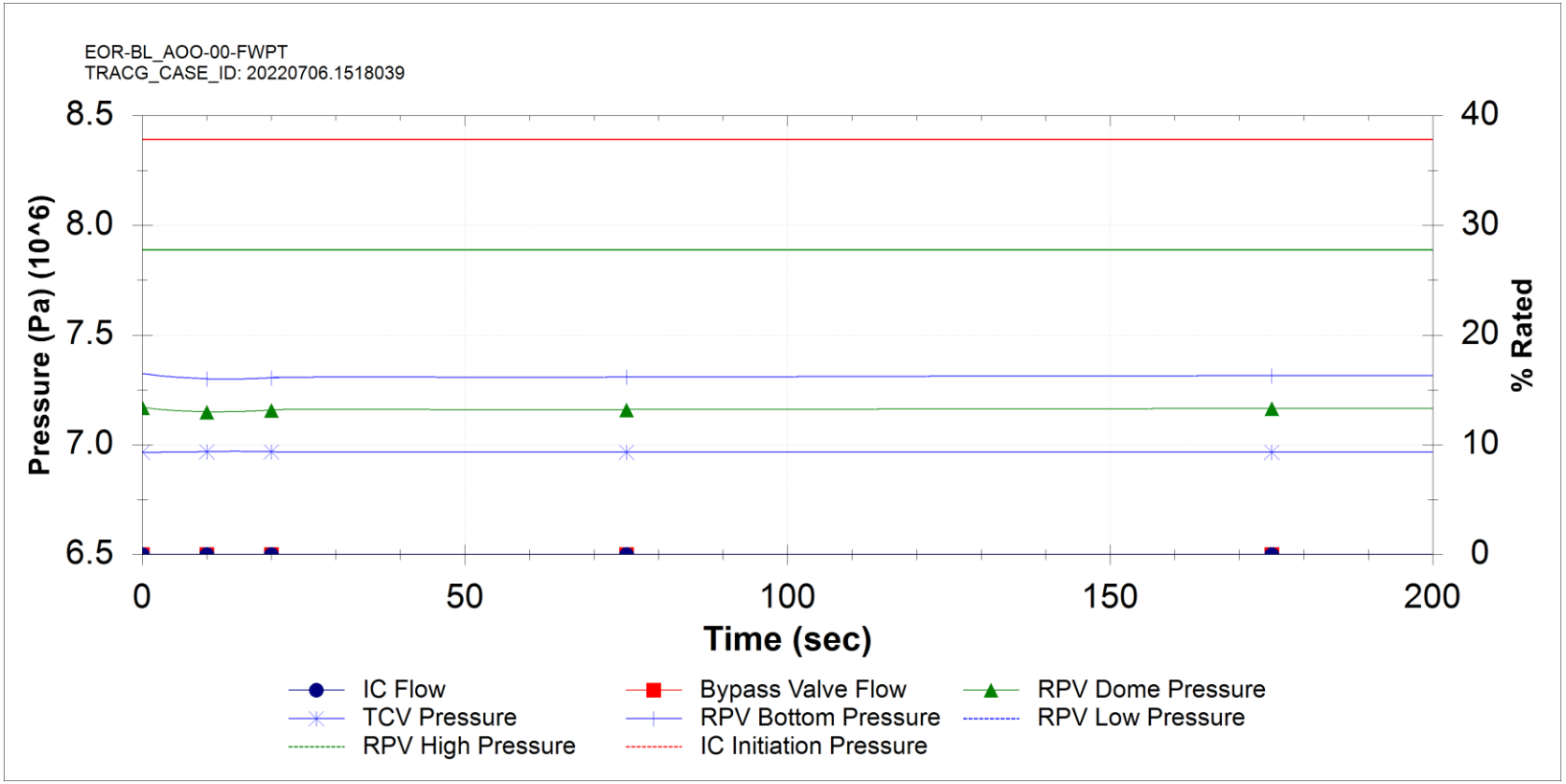


Figure 15.5-35: Feedwater Pump Trip – One Pump (AOO)

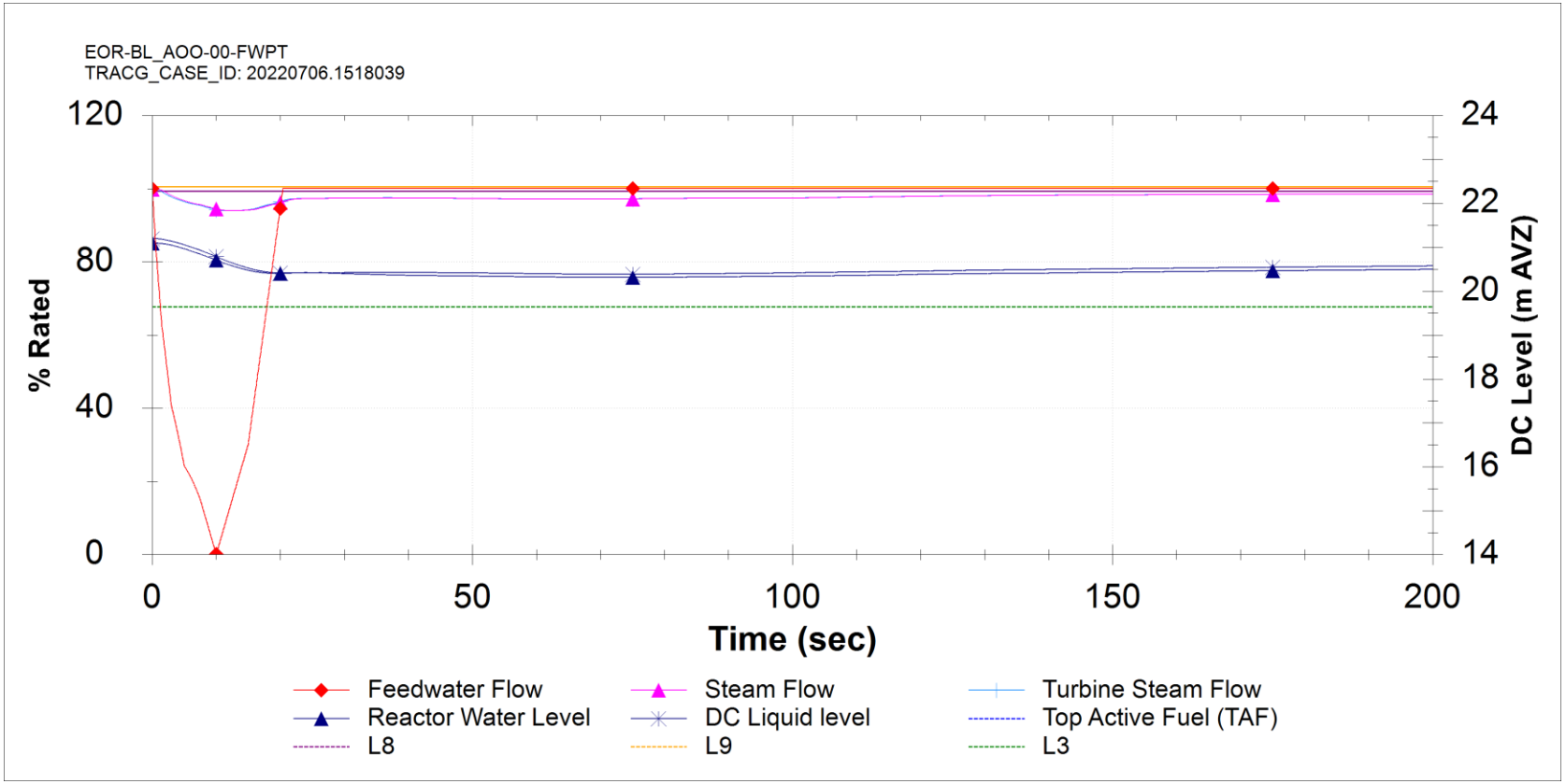


Figure 15.5-36: Feedwater Pump Trip – One Pump (AOO)

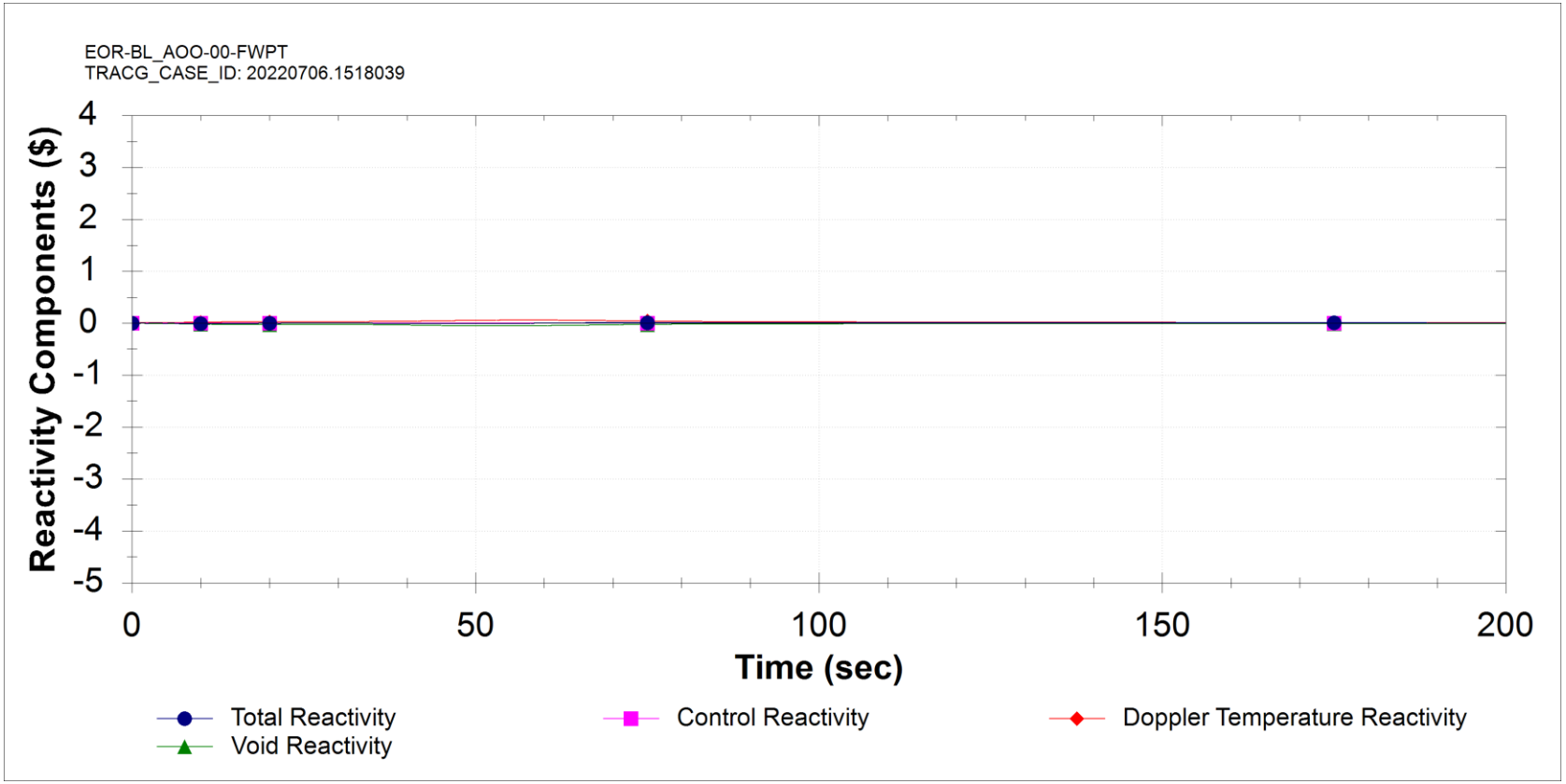


Figure 15.5-37: Feedwater Pump Trip – One Pump (AOO)

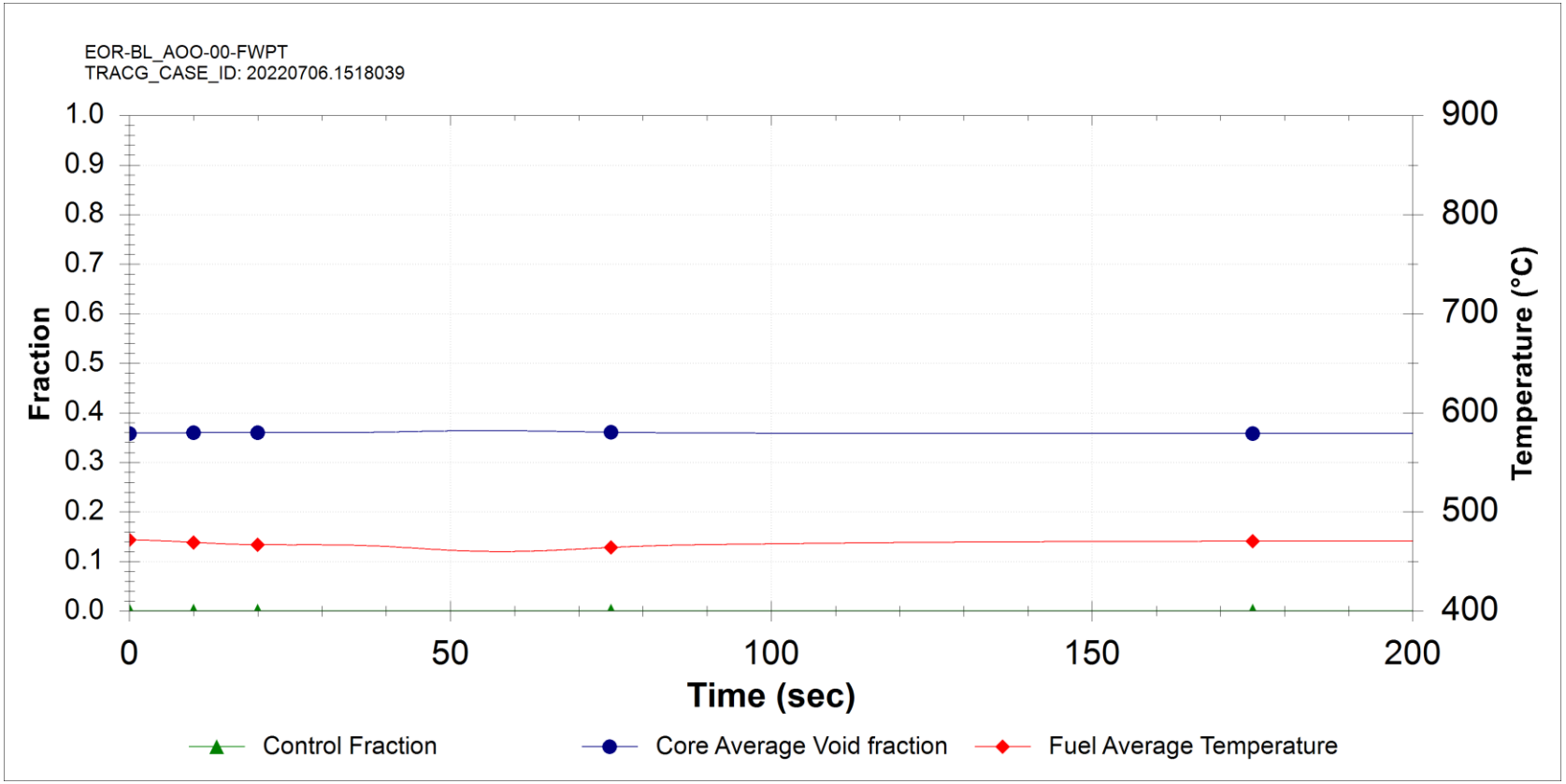


Figure 15.5-38: Feedwater Pump Trip – One Pump (AOO)

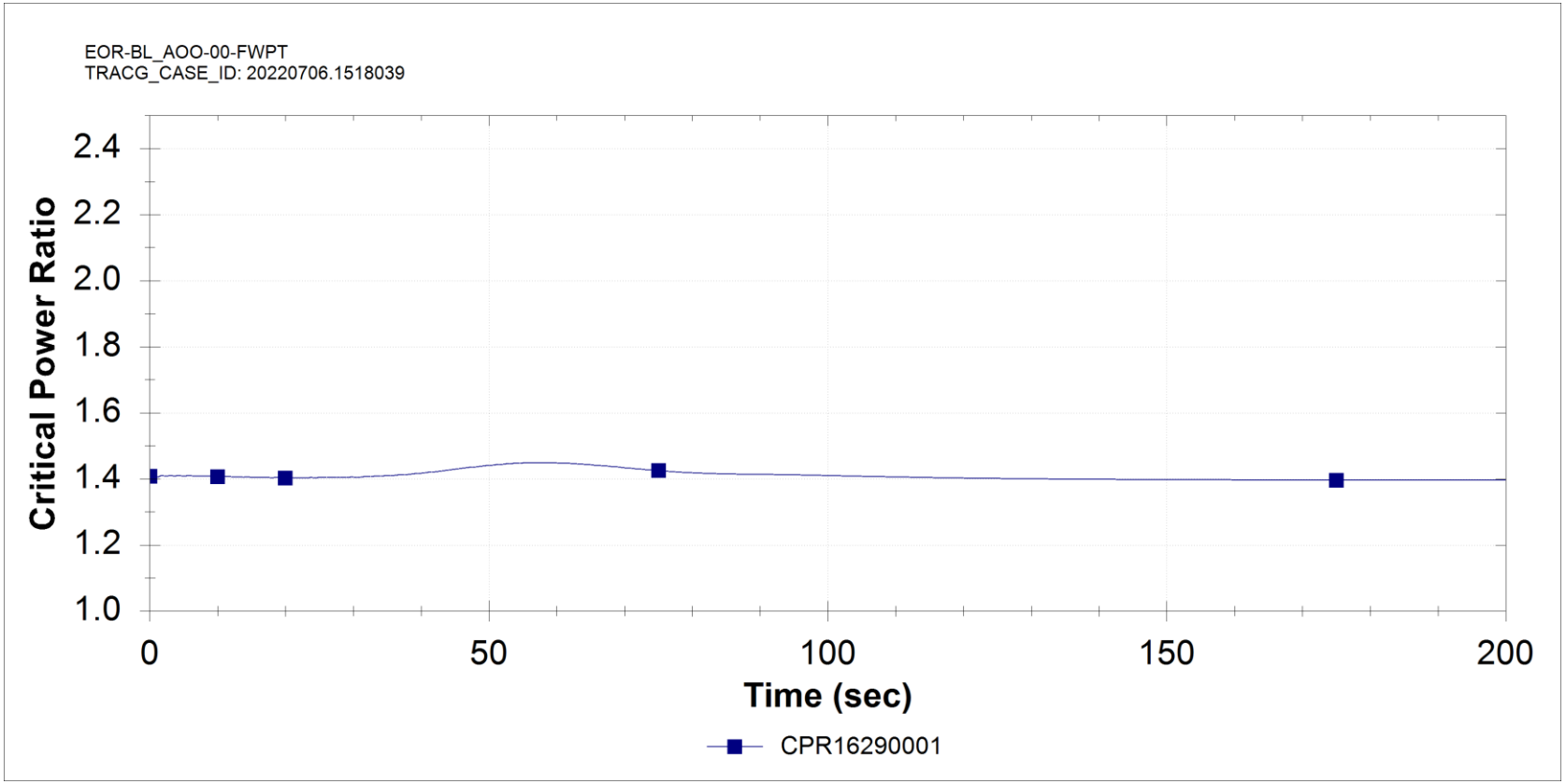


Figure 15.5-39: Feedwater Pump Trip – One Pump (AOO)

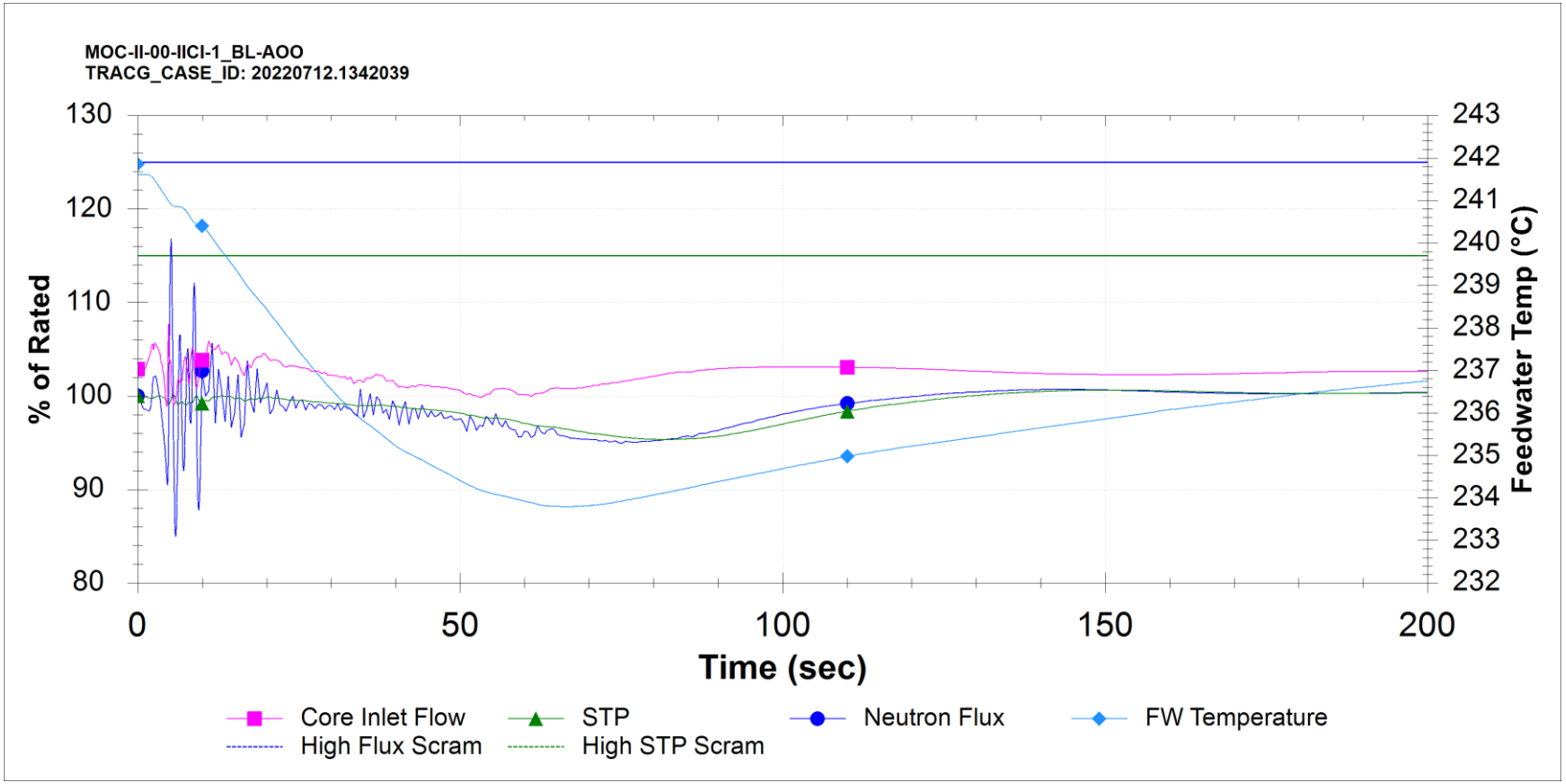


Figure 15.5-40: Inadvertent Isolation Condenser Initiation – One Train (AOO)

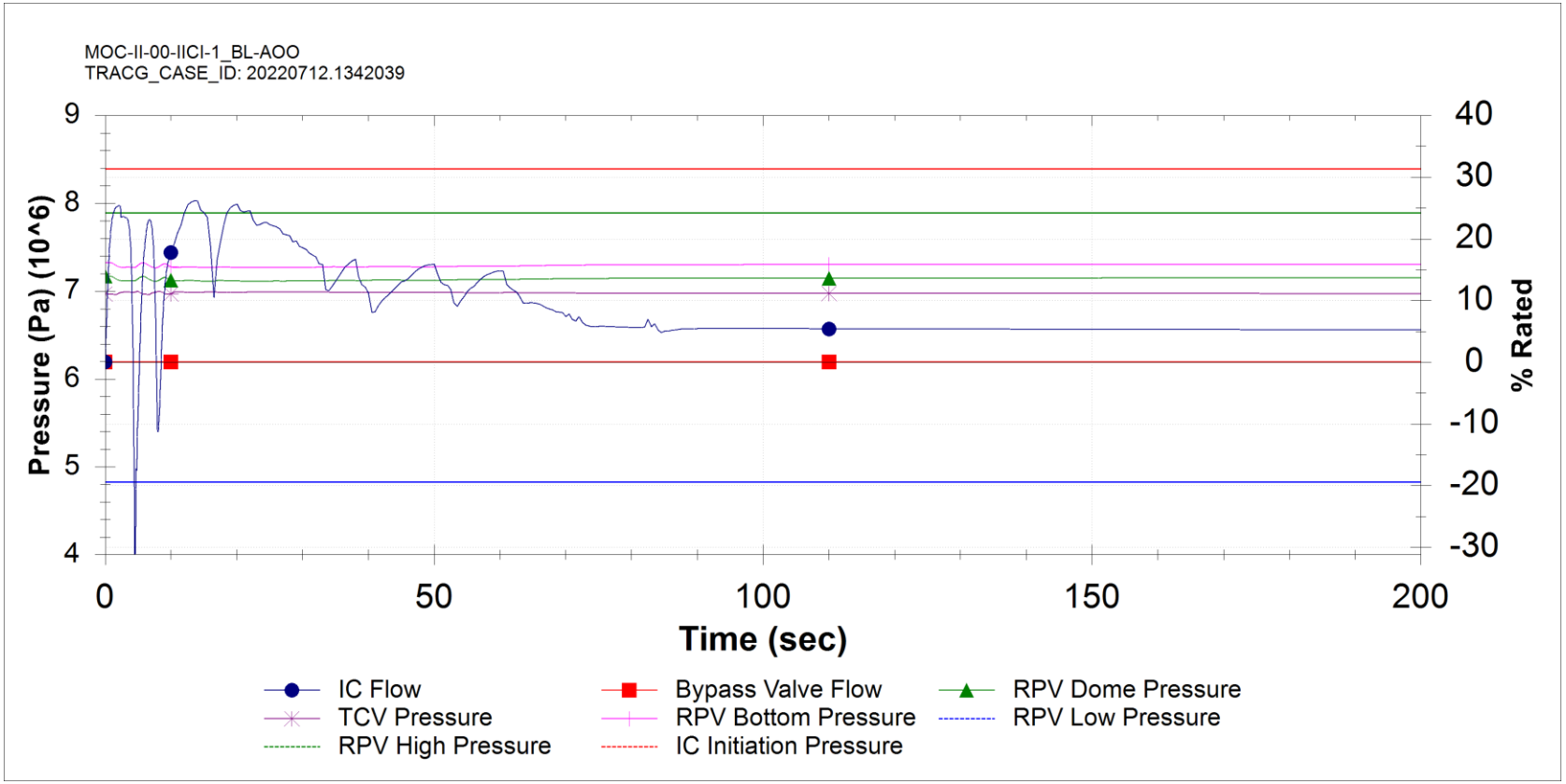


Figure 15.5-41: Inadvertent Isolation Condenser Initiation – One Train (AOO)

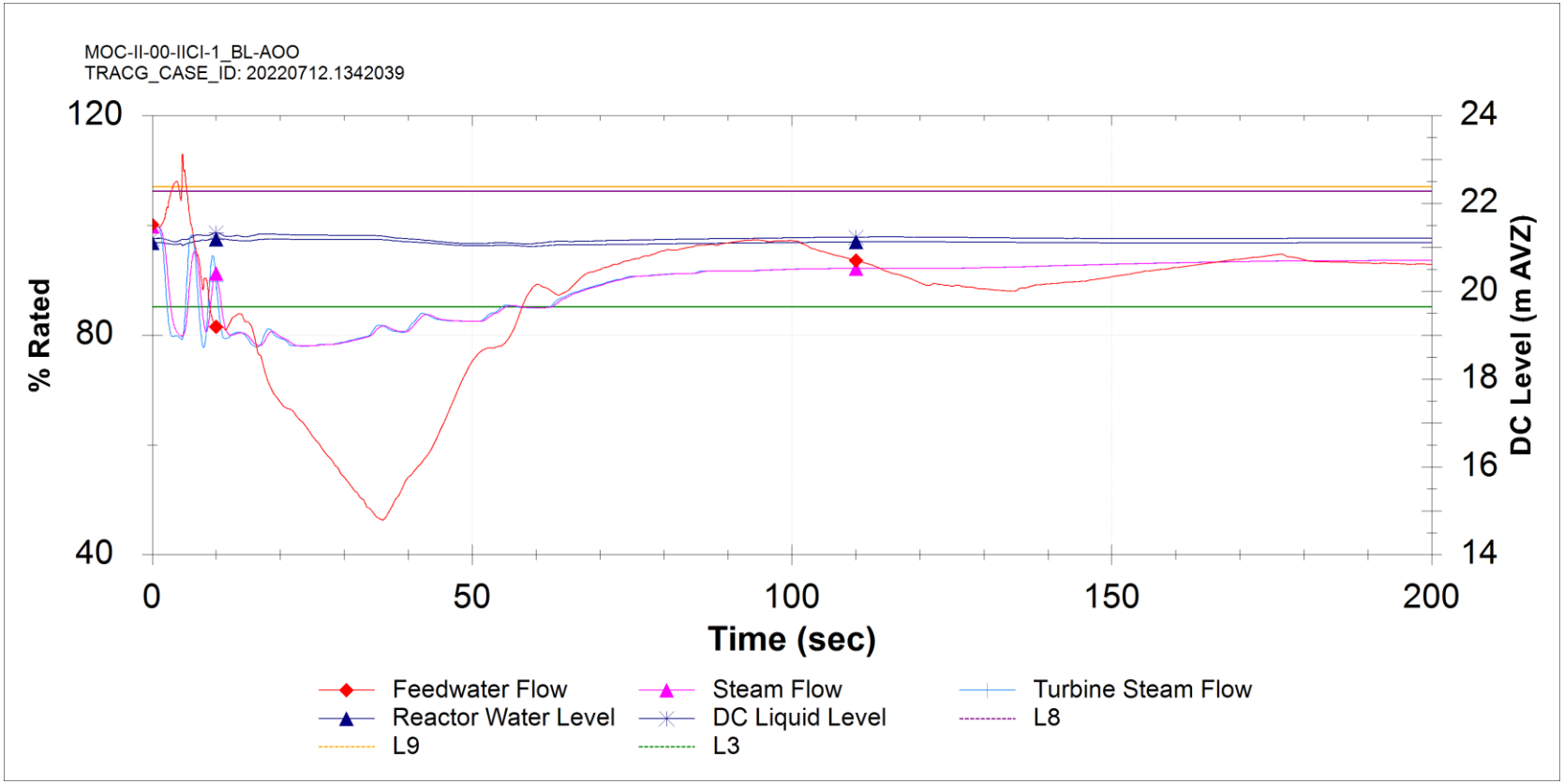


Figure 15.5-42: Inadvertent Isolation Condenser Initiation – One Train (AOO)

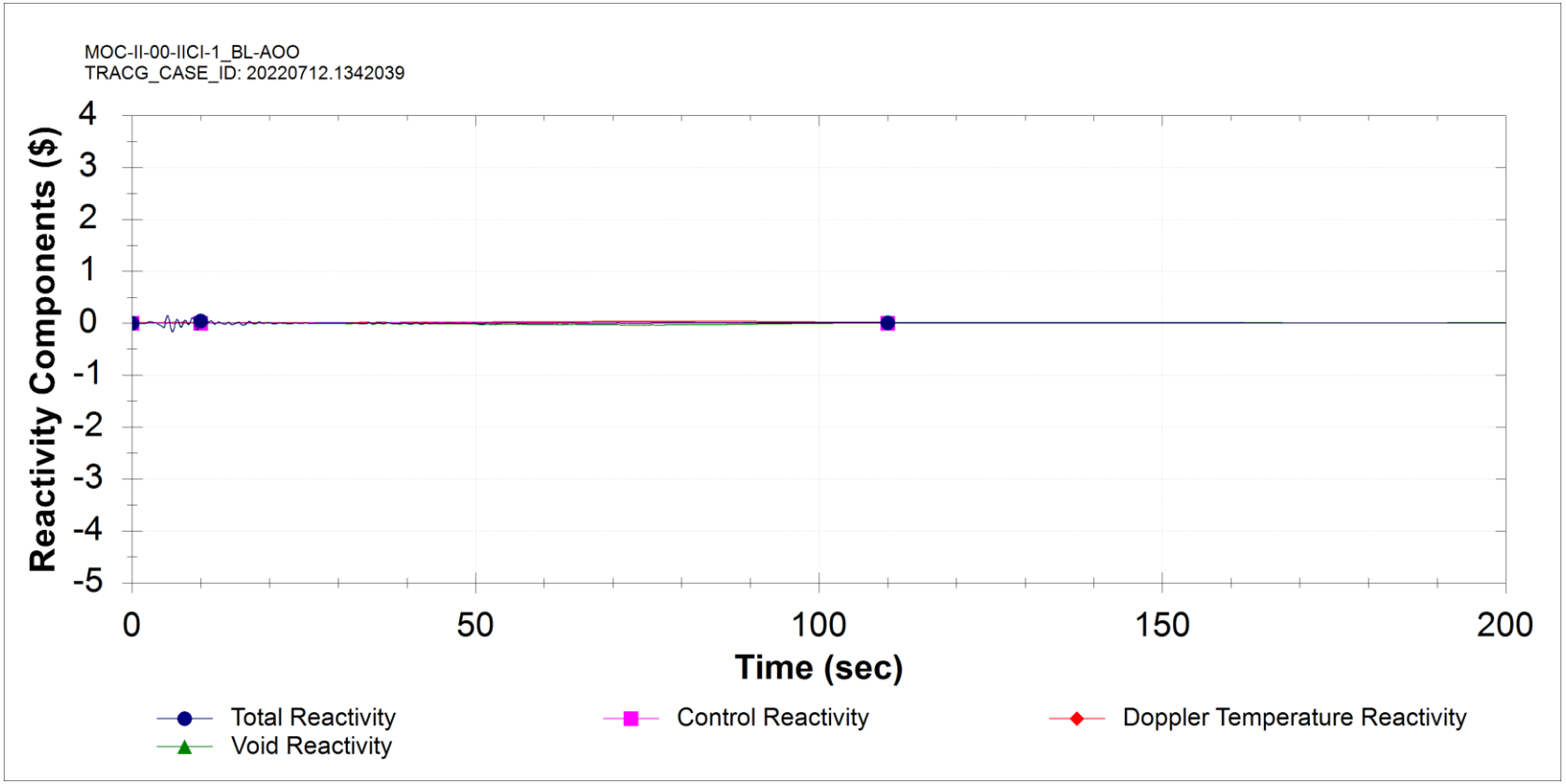


Figure 15.5-43: Inadvertent Isolation Condenser Initiation - One Train (AOO)

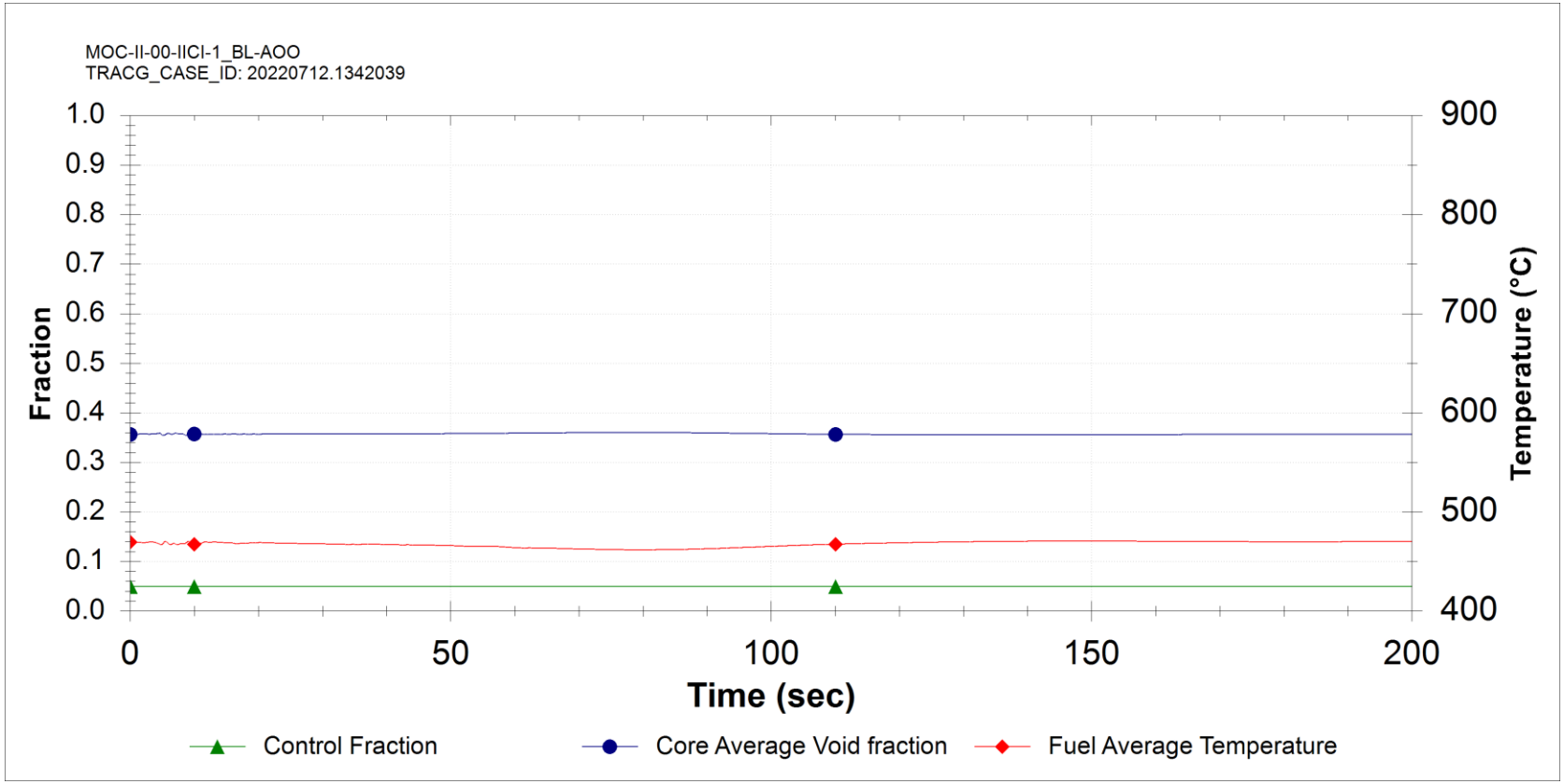


Figure 15.5-44: Inadvertent Isolation Condenser Initiation - One Train (AOO)

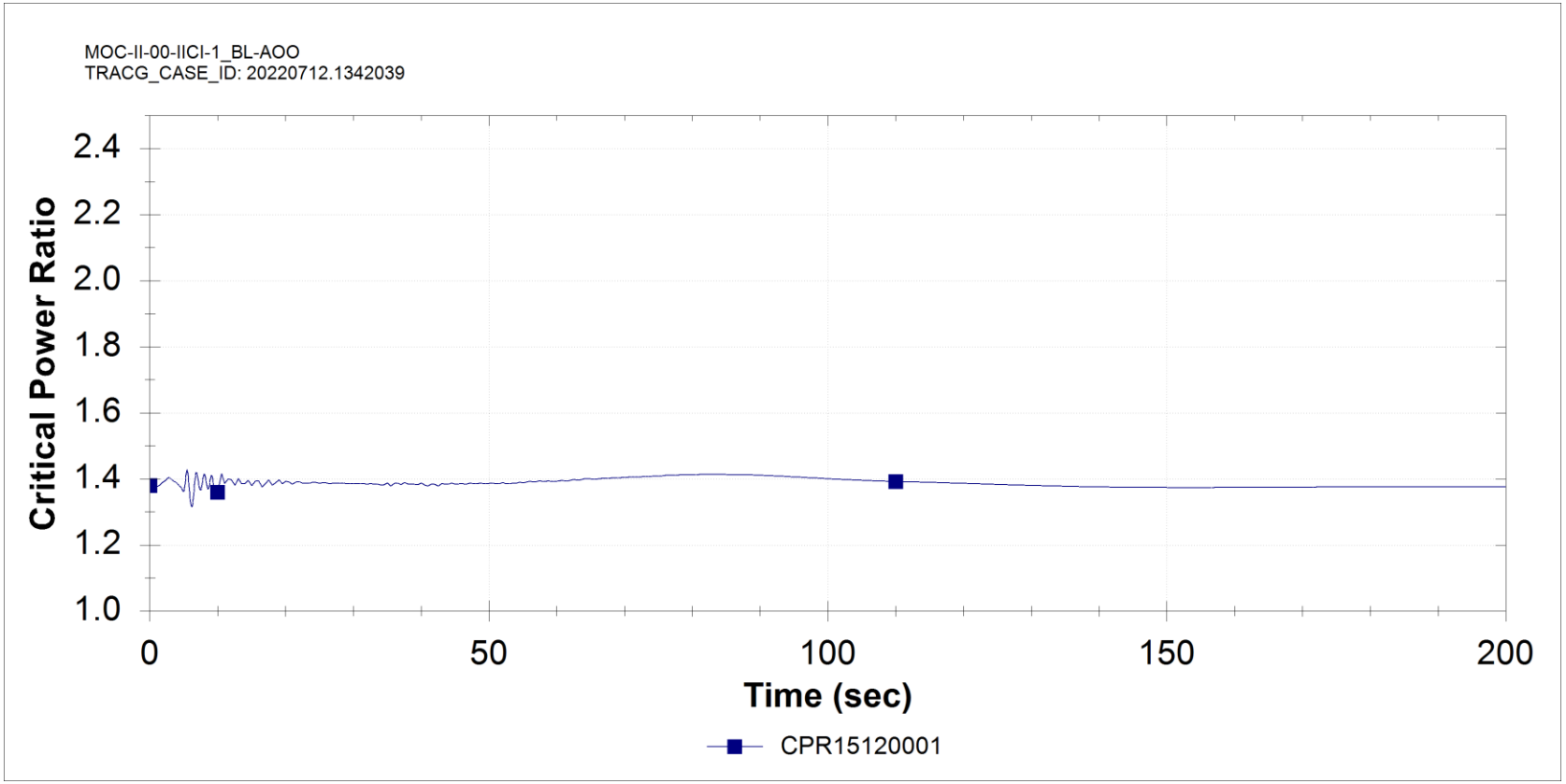


Figure 15.5-45: Inadvertent Isolation Condenser Initiation - One Train (AOO)

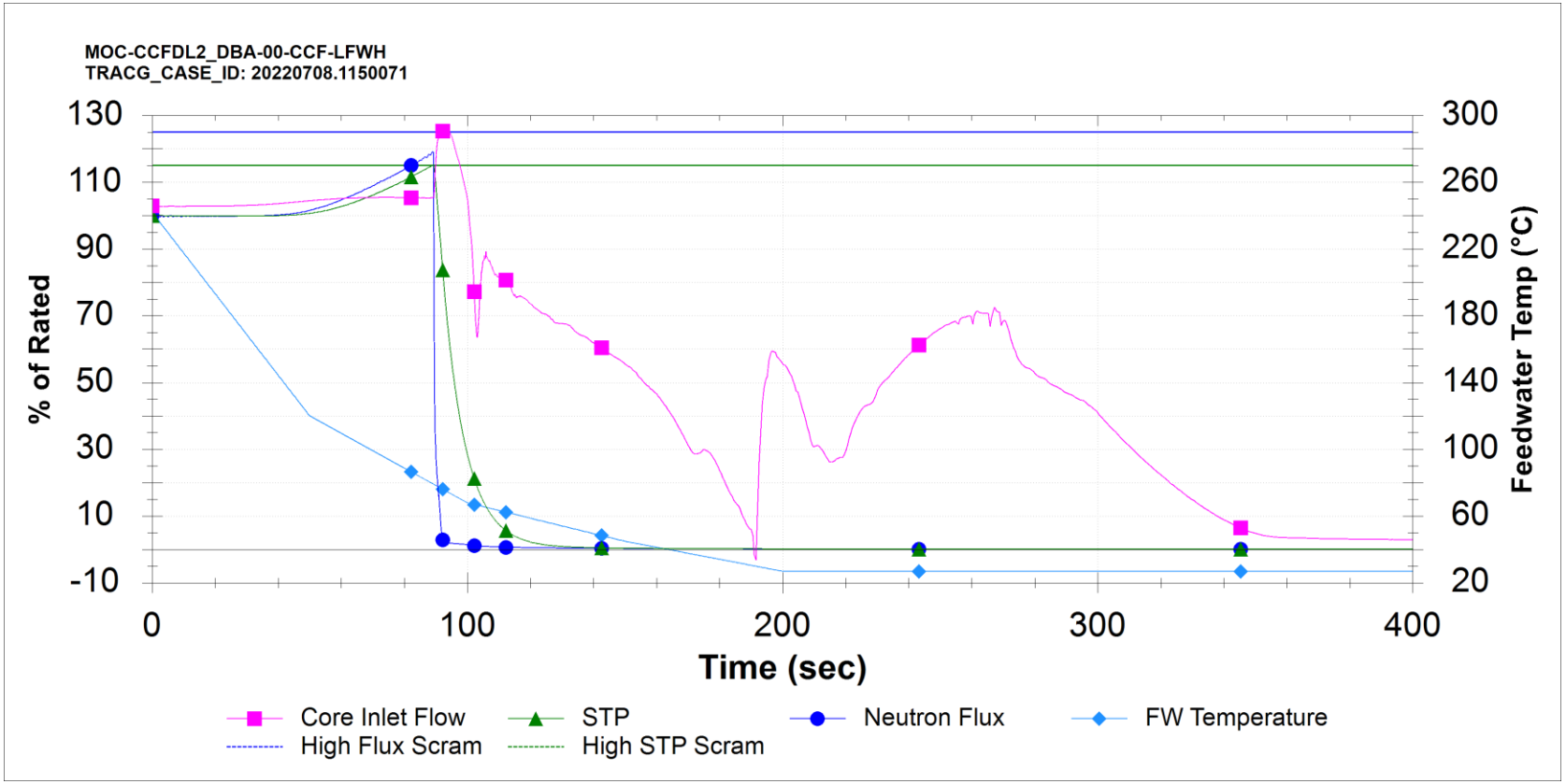


Figure 15.5-46: Loss of Feedwater Heating (DBA)

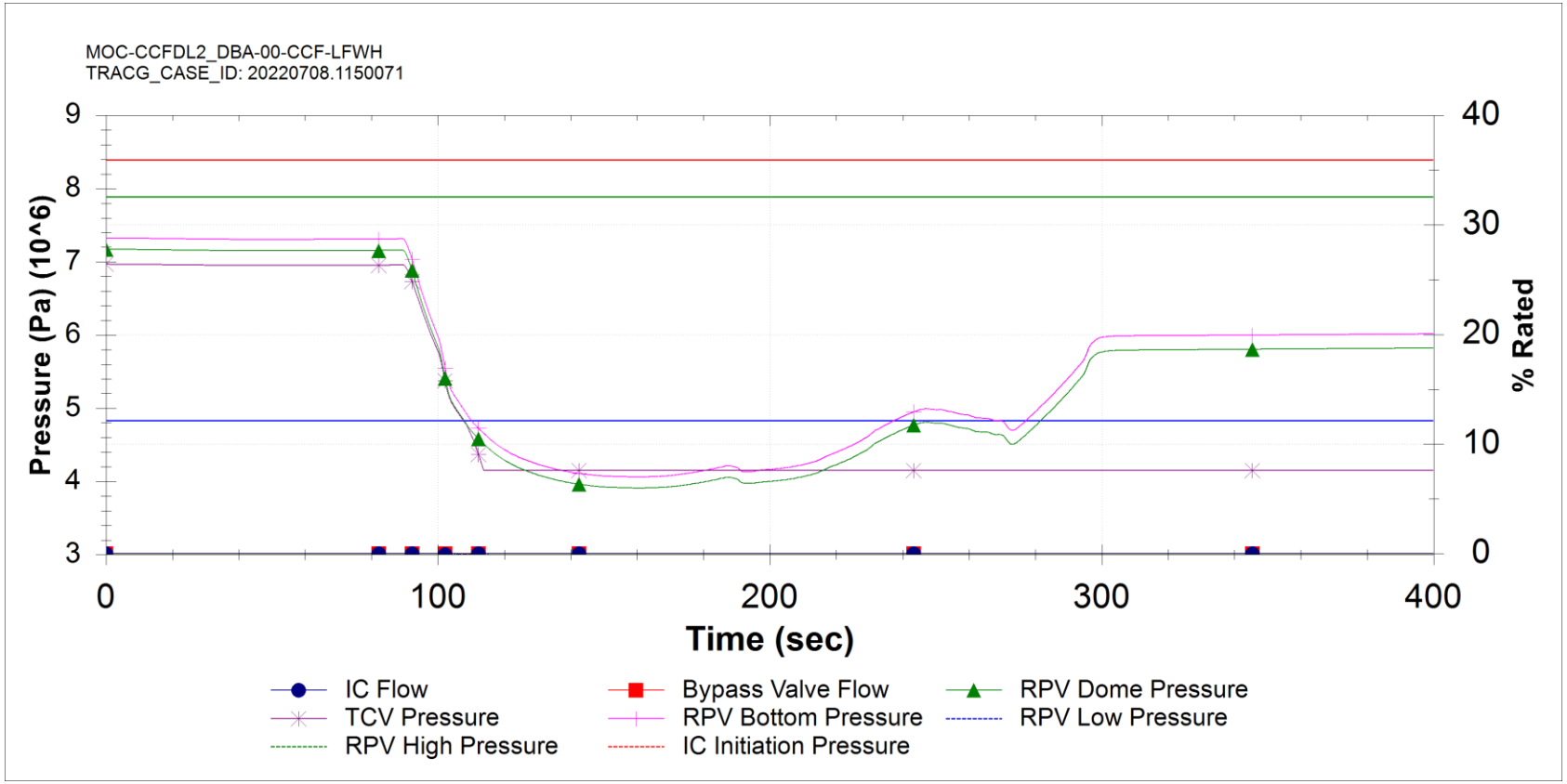


Figure 15.5-47: Loss of Feedwater Heating (DBA)

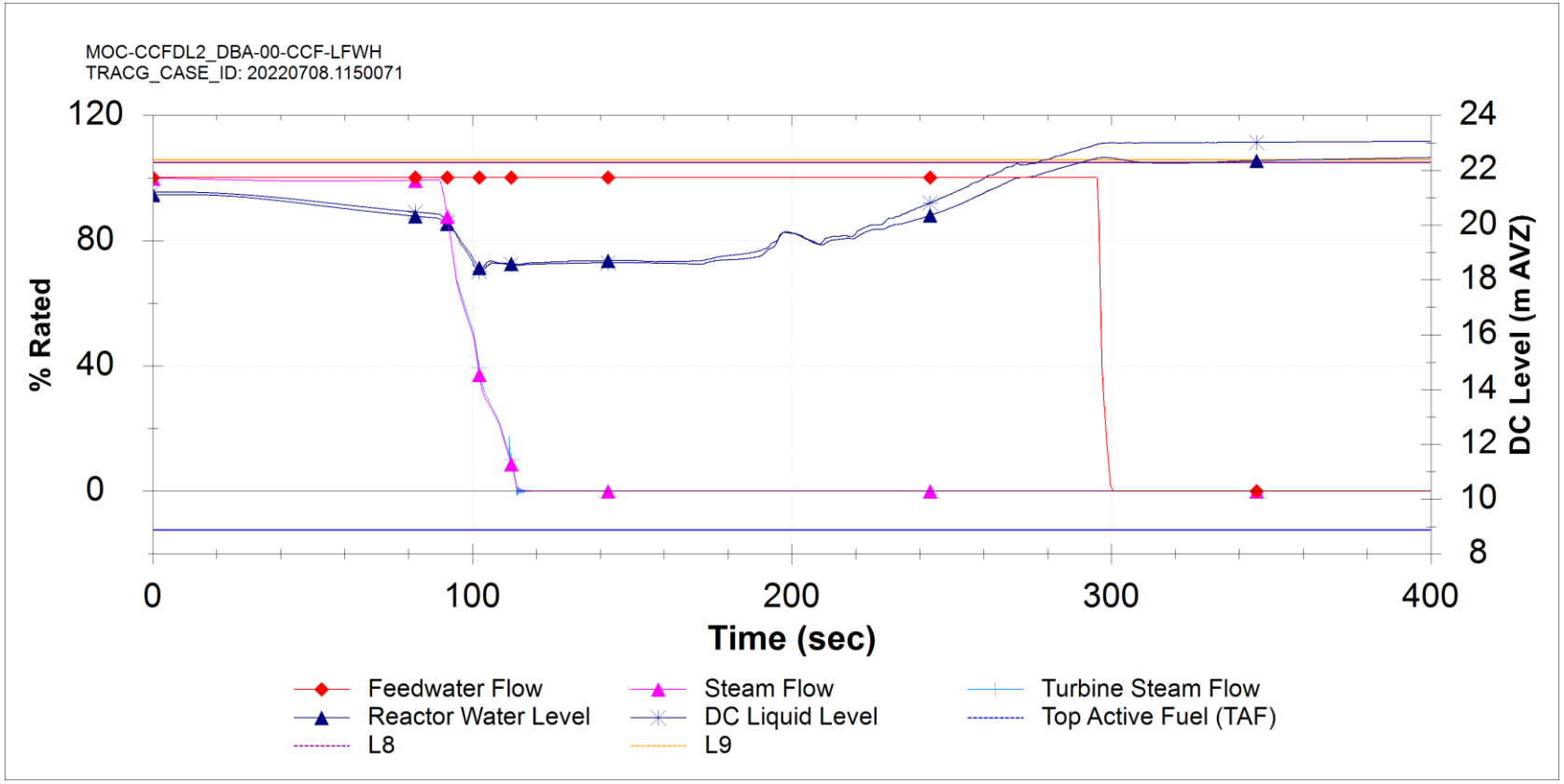


Figure 15.5-48: Loss of Feedwater Heating (DBA)

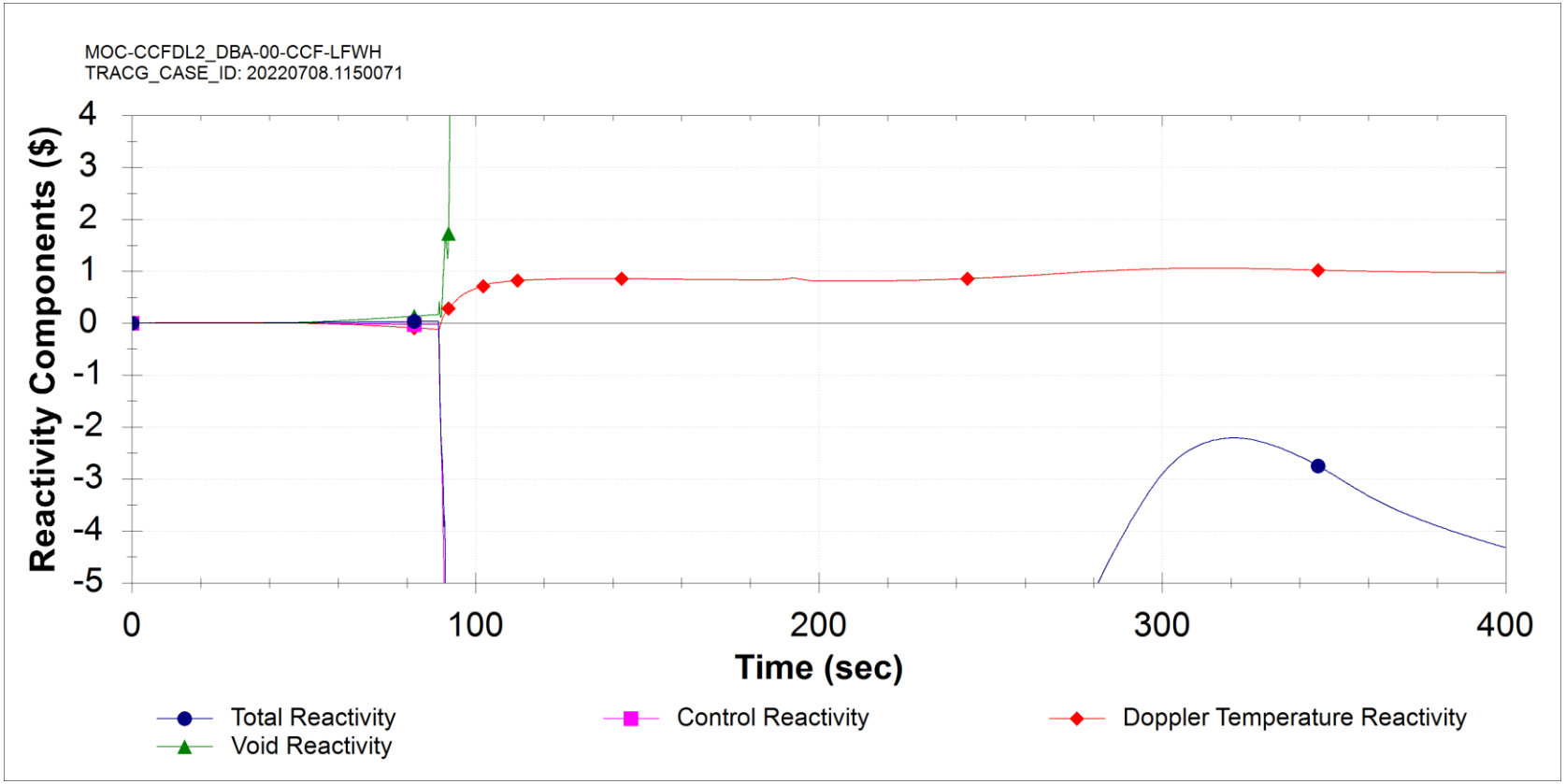


Figure 15.5-49: Loss of Feedwater Heating (DBA)

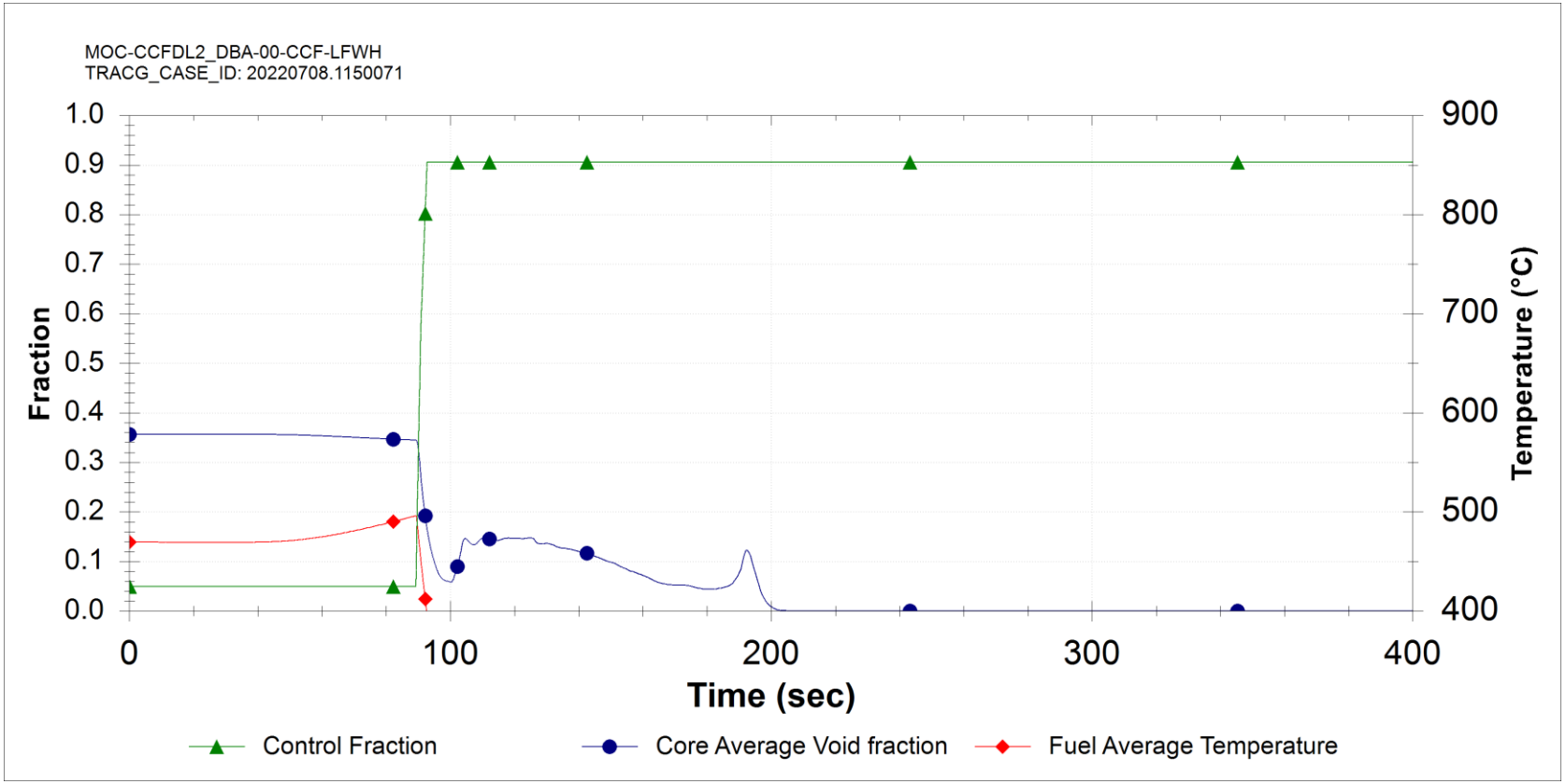


Figure 15.5-50: Loss of Feedwater Heating (DBA)

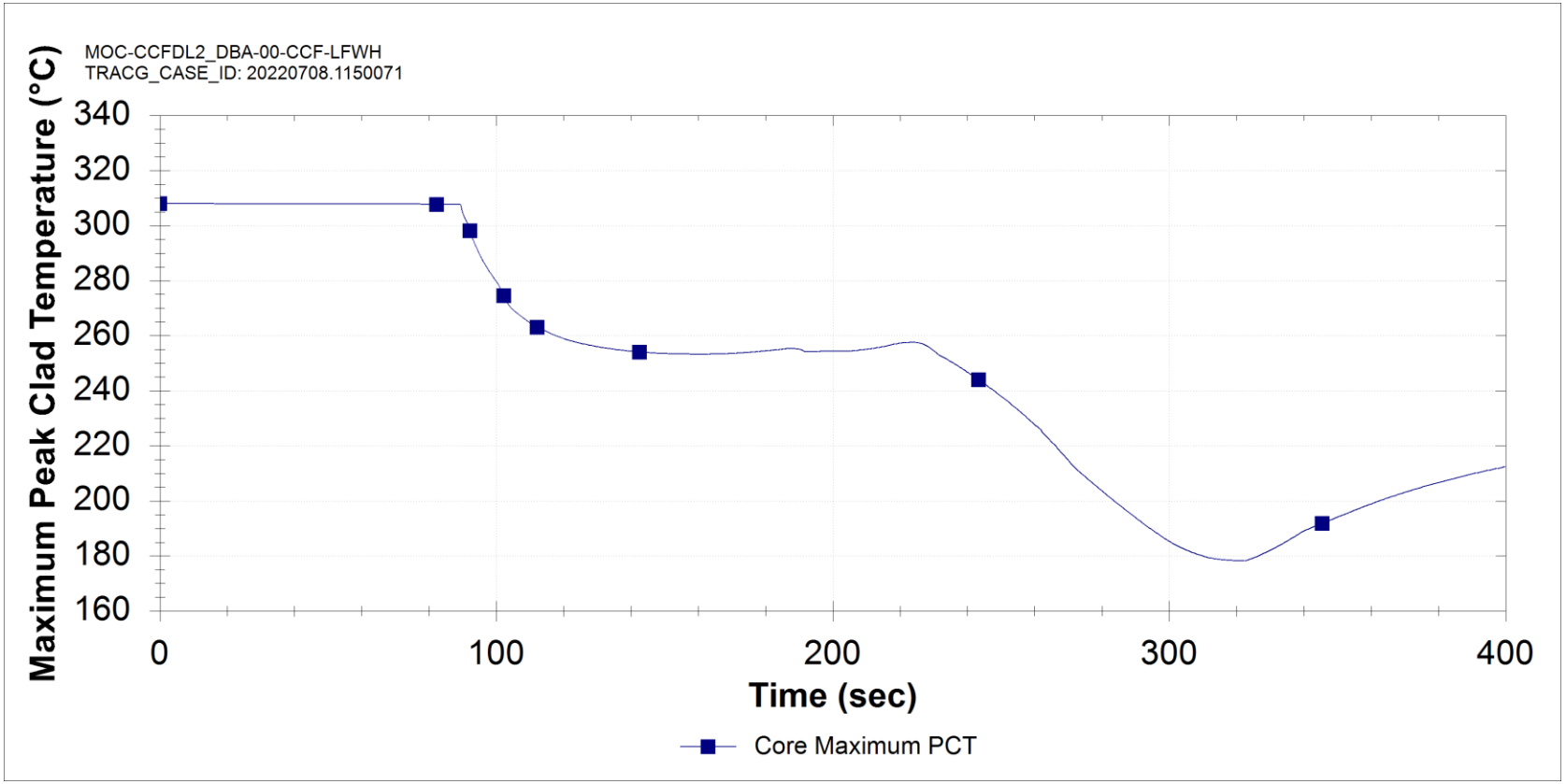


Figure 15.5-51: Loss of Feedwater Heating (DBA)

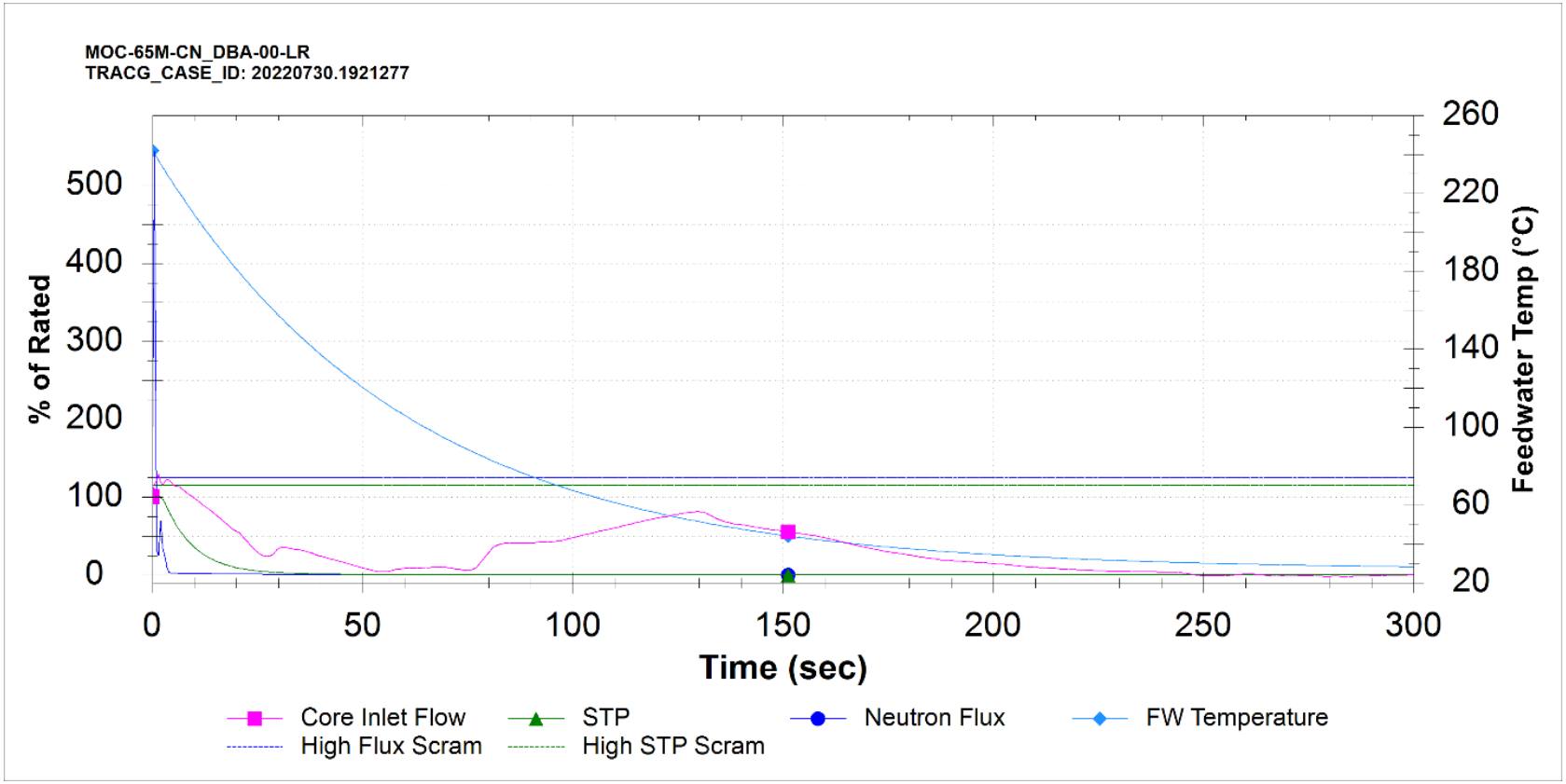


Figure 15.5-52: Generator Load Rejection (DBA)

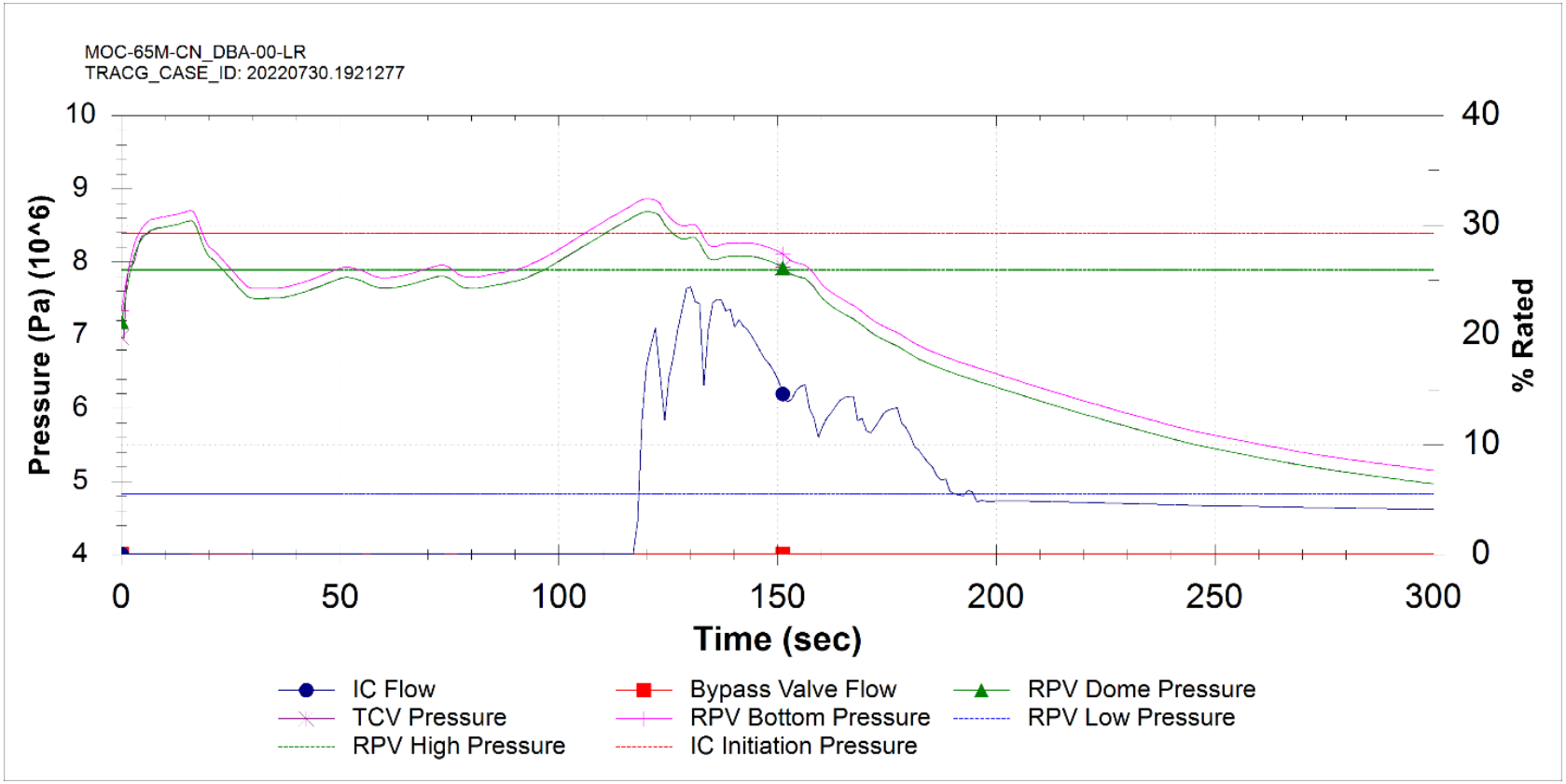


Figure 15.5-53: Generator Load Rejection (DBA)

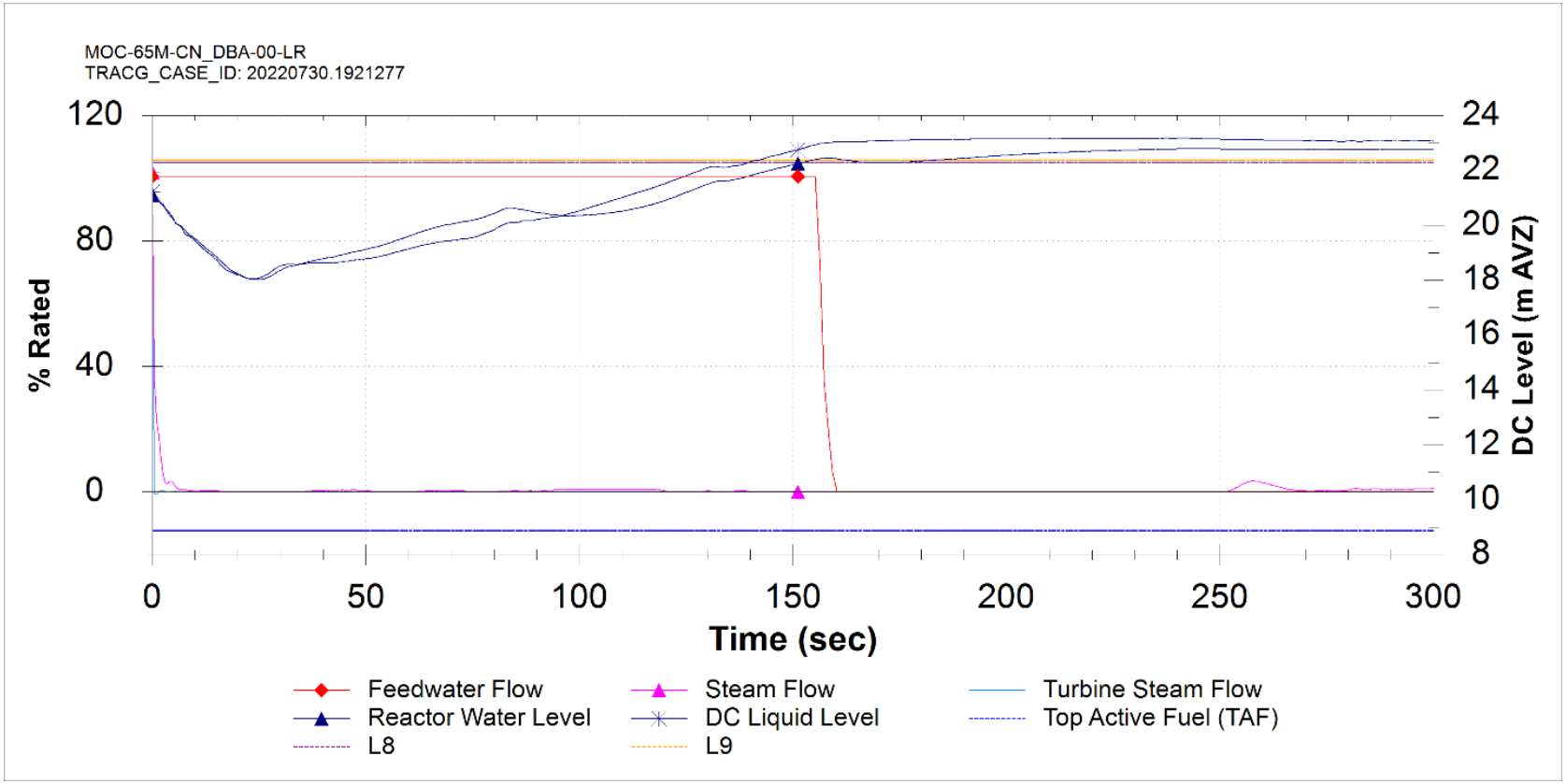


Figure 15.5-54: Generator Load Rejection (DBA)

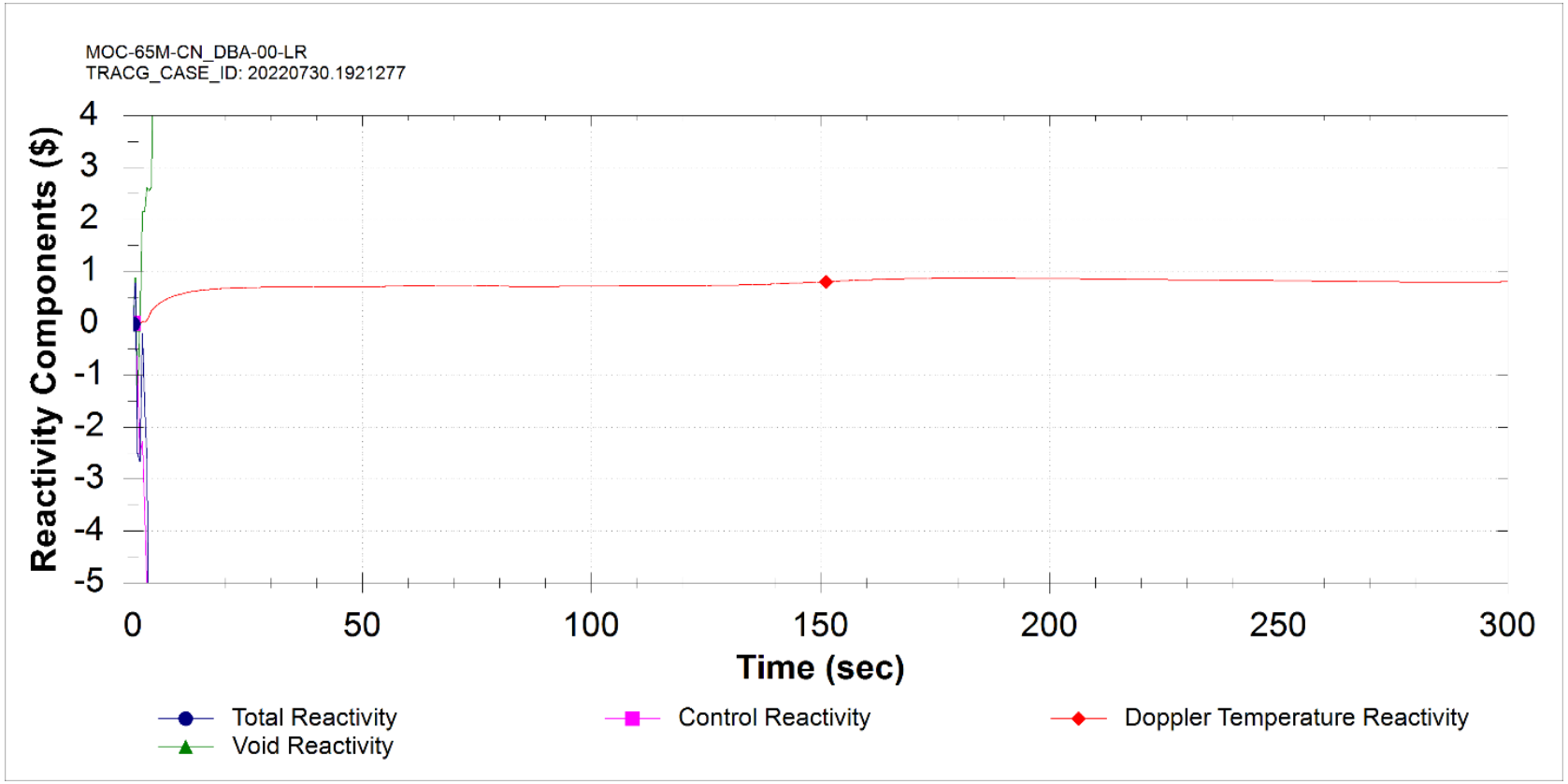


Figure 15.5-55: Generator Load Rejection (DBA)

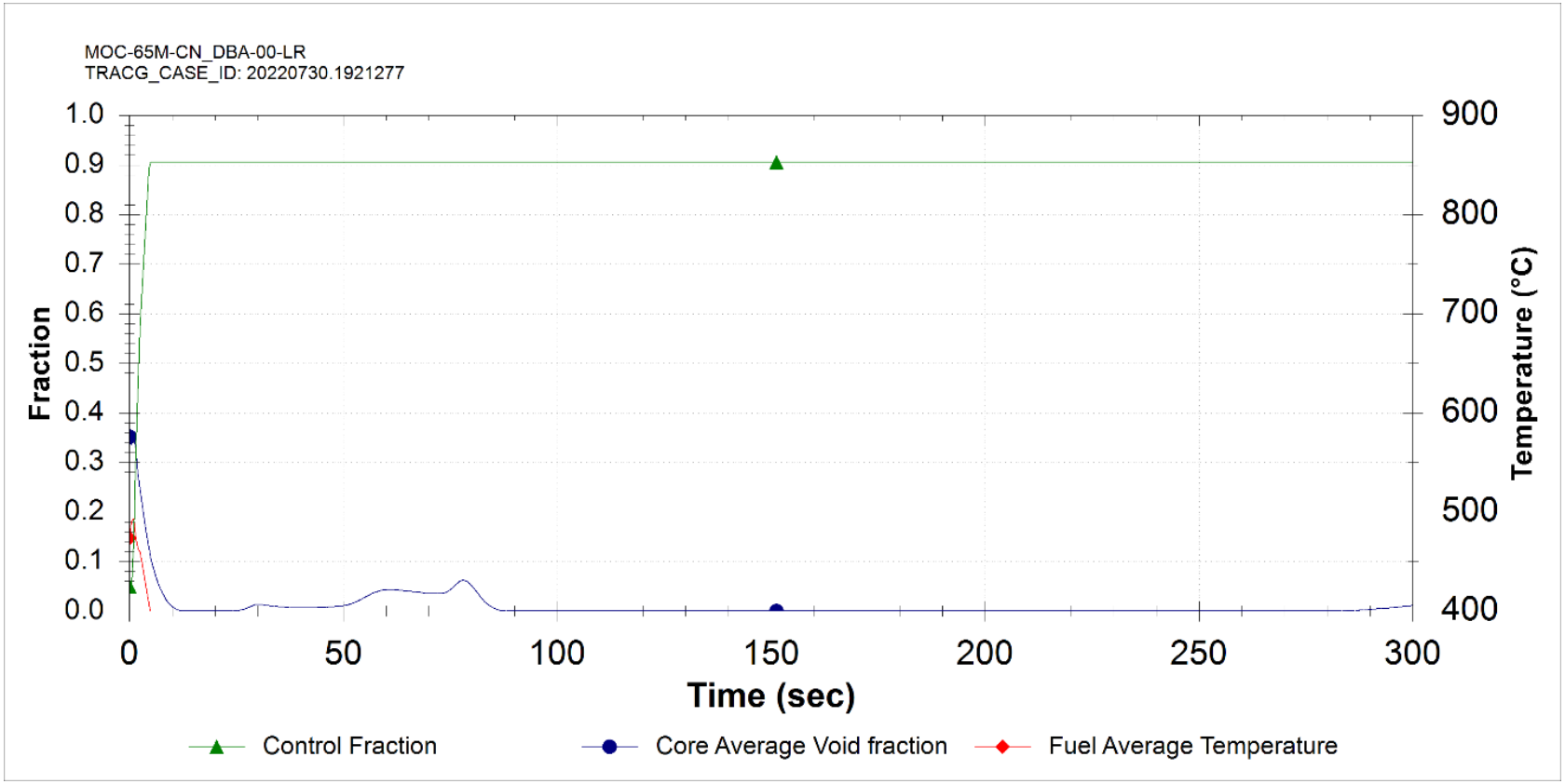


Figure 15.5-56: Generator Load Rejection (DBA)

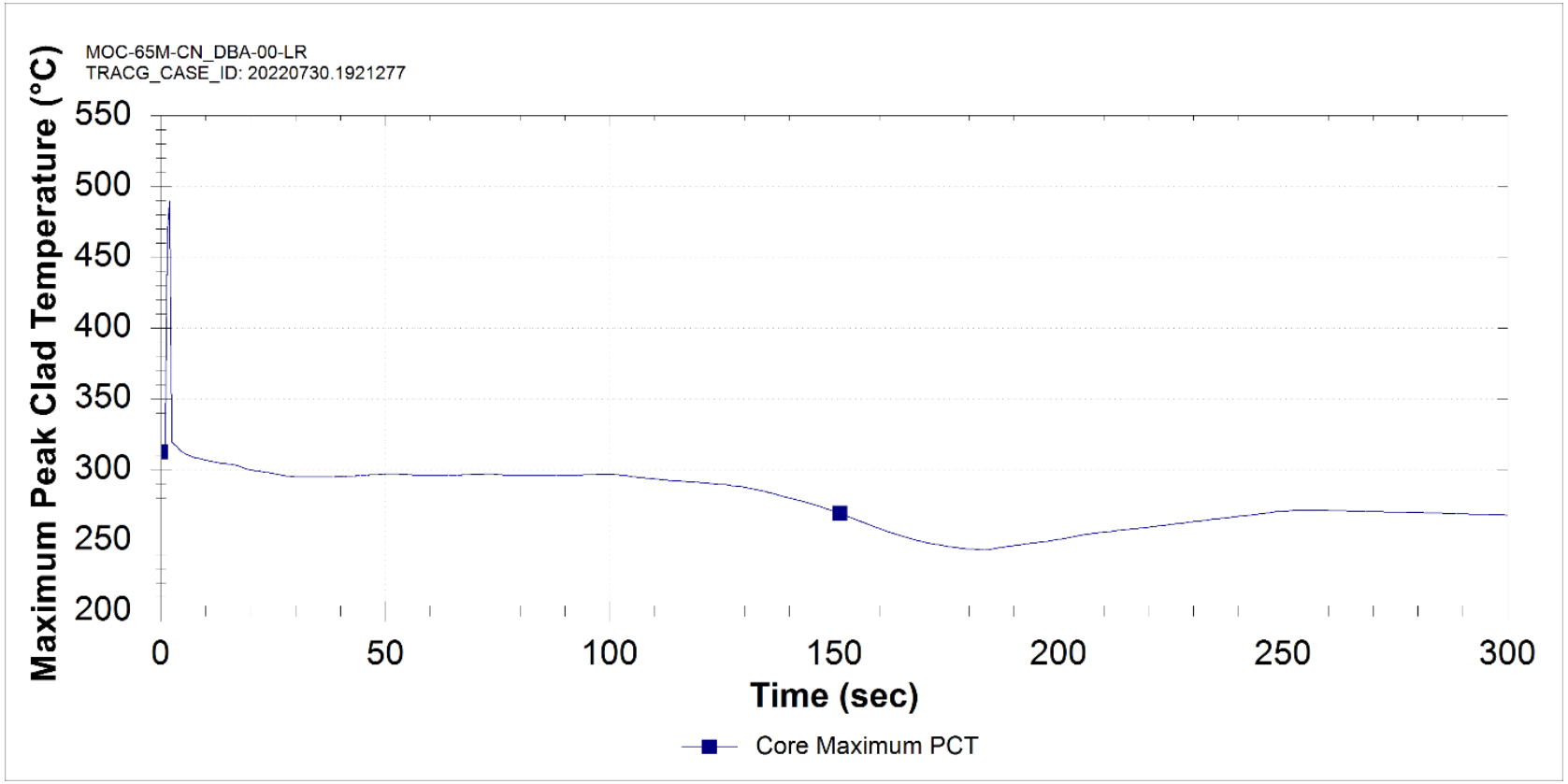


Figure 15.5-57: Generator Load Rejection (DBA)

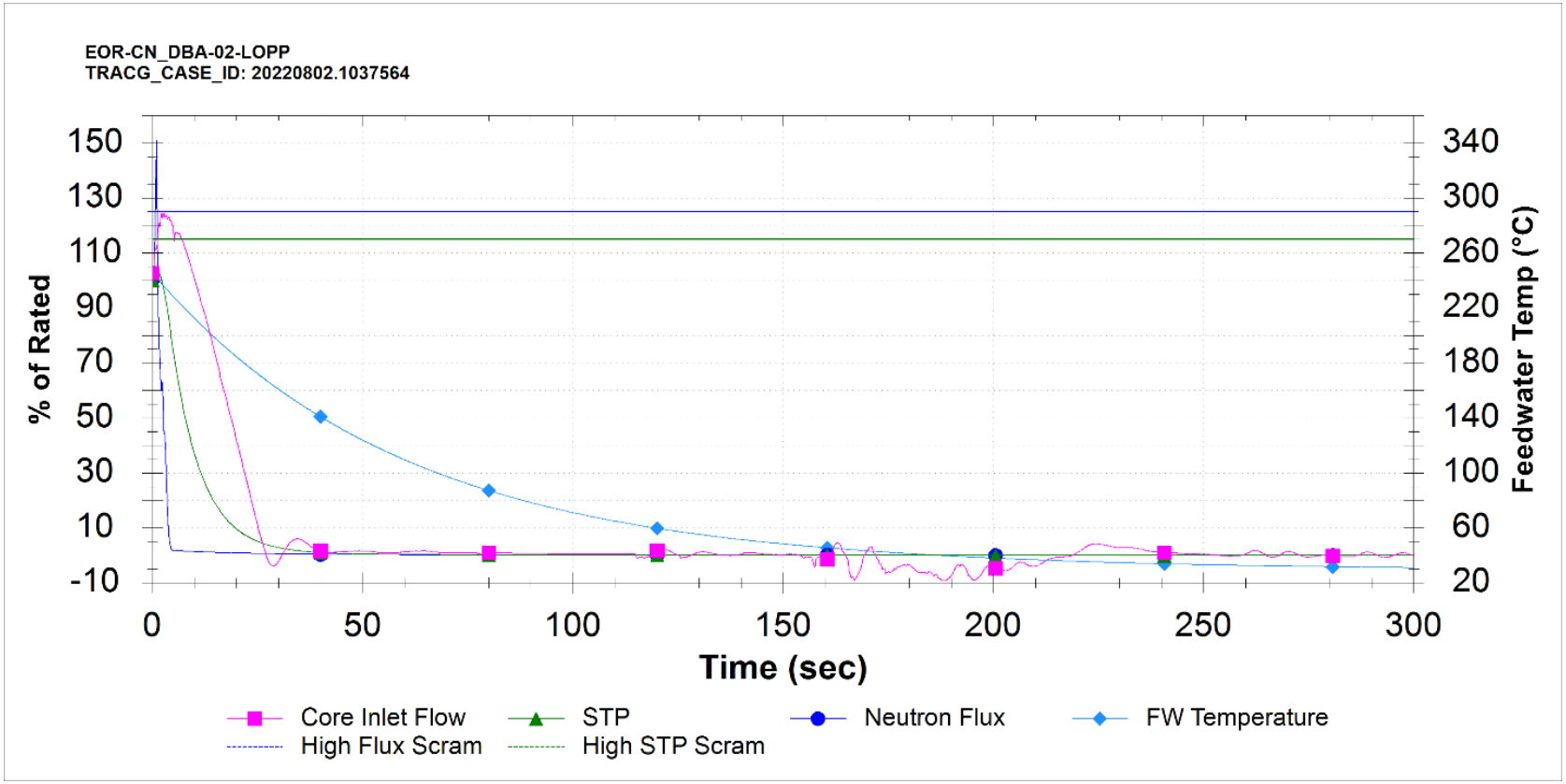


Figure 15.5-58: Loss of Preferred Power (DBA)

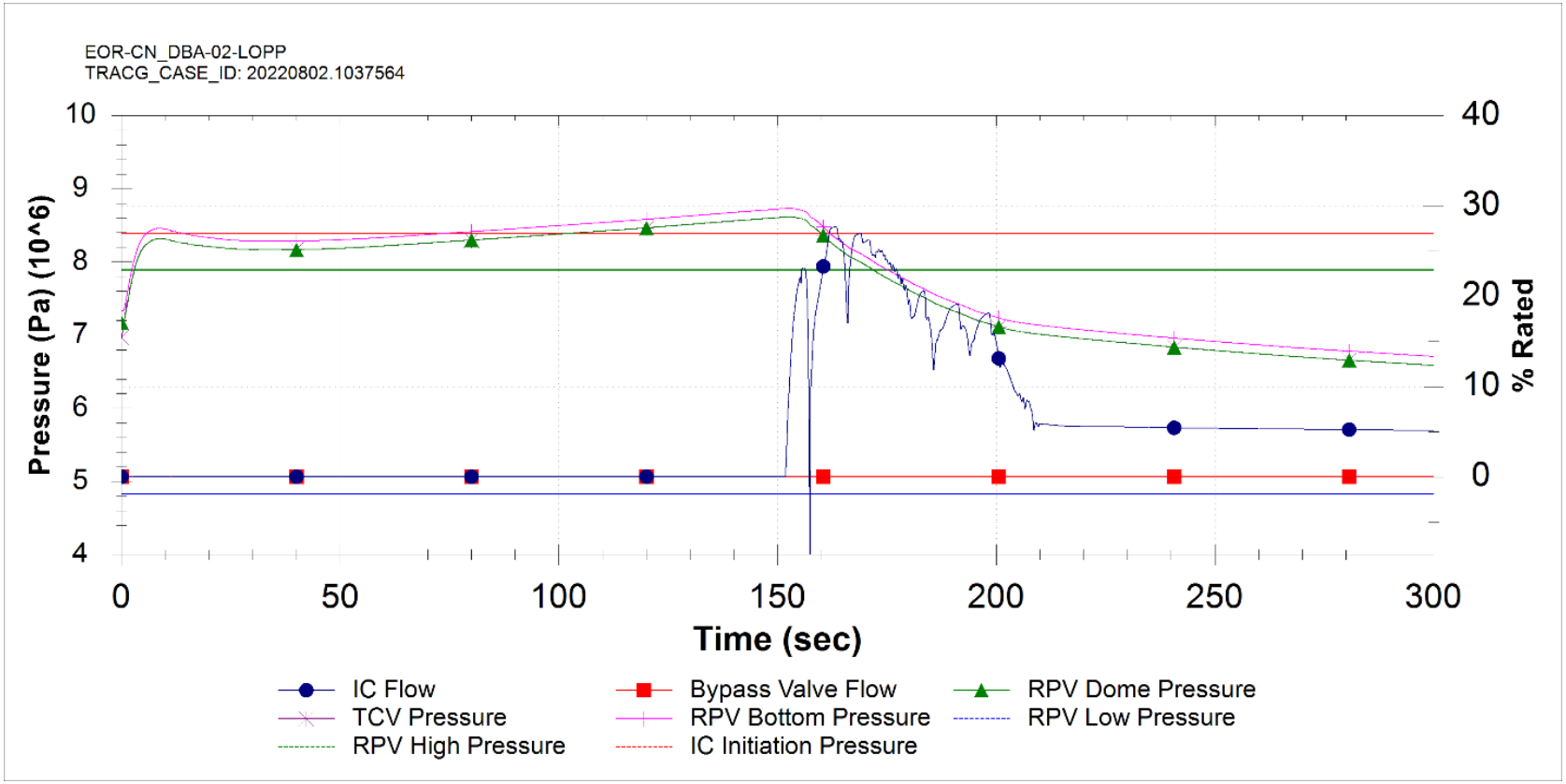


Figure 15.5-59: Loss of Preferred Power (DBA)

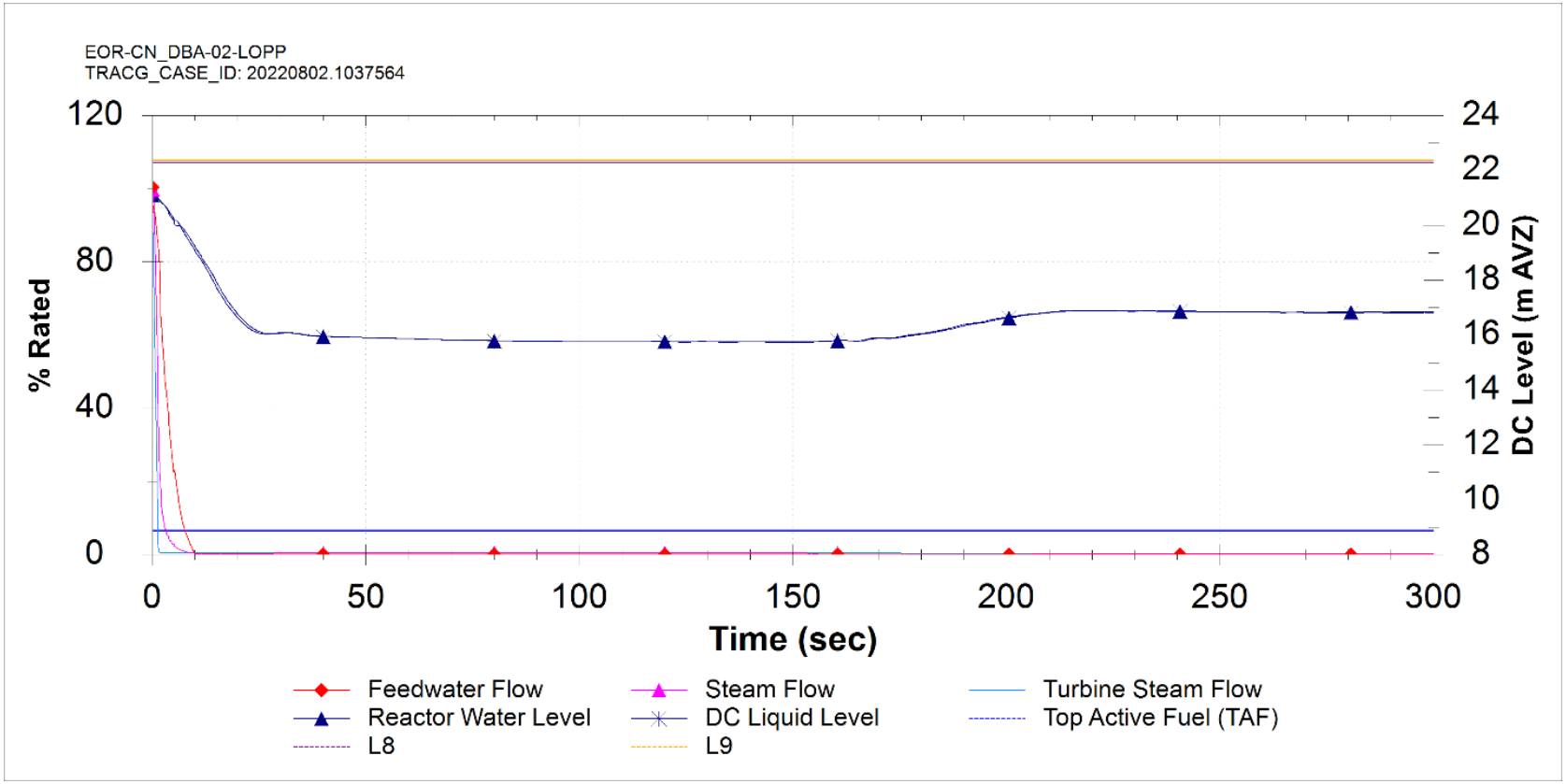


Figure 15.5-60: Loss of Preferred Power (DBA)

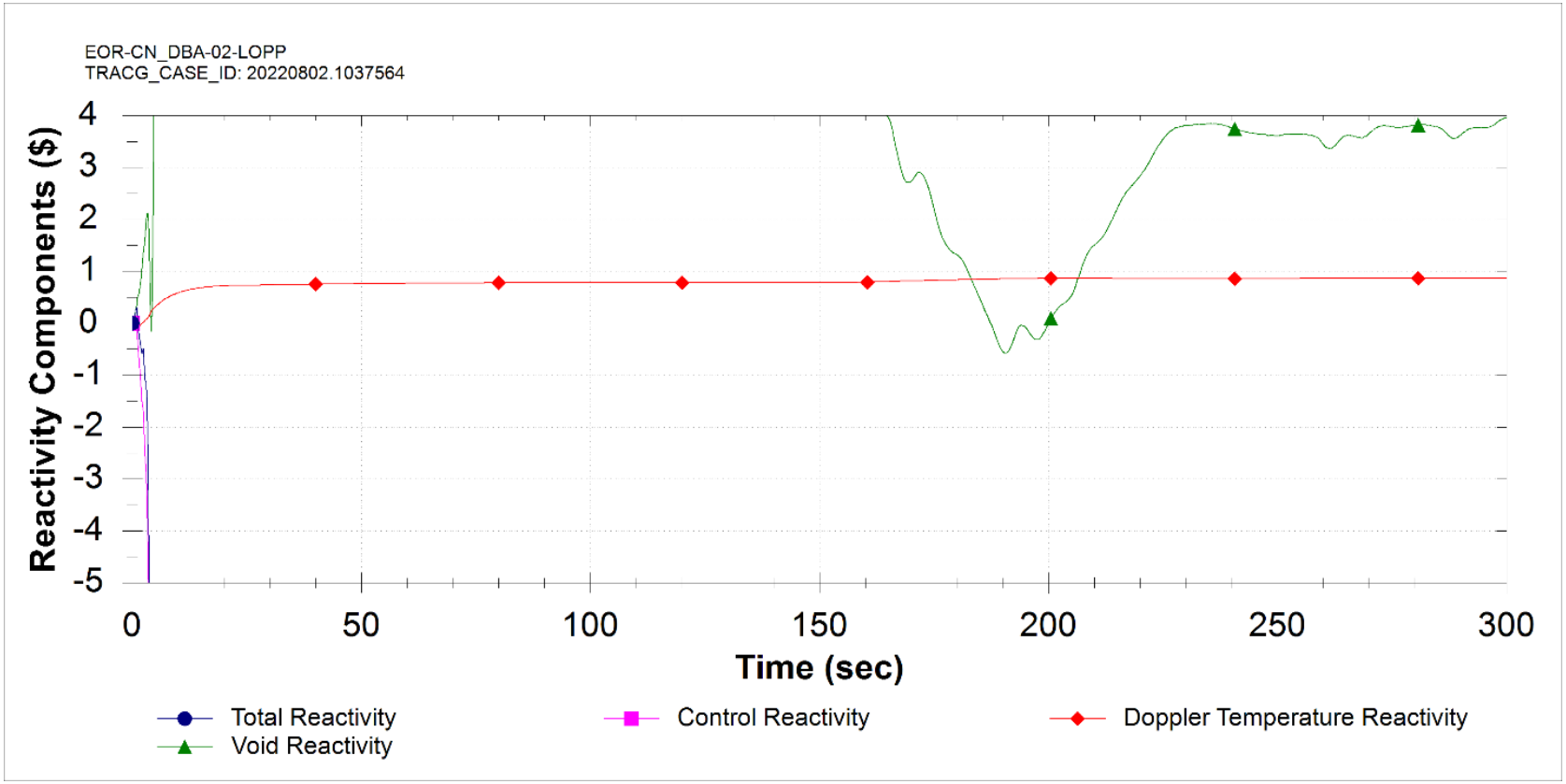


Figure 15.5-61: Loss of Preferred Power (DBA)

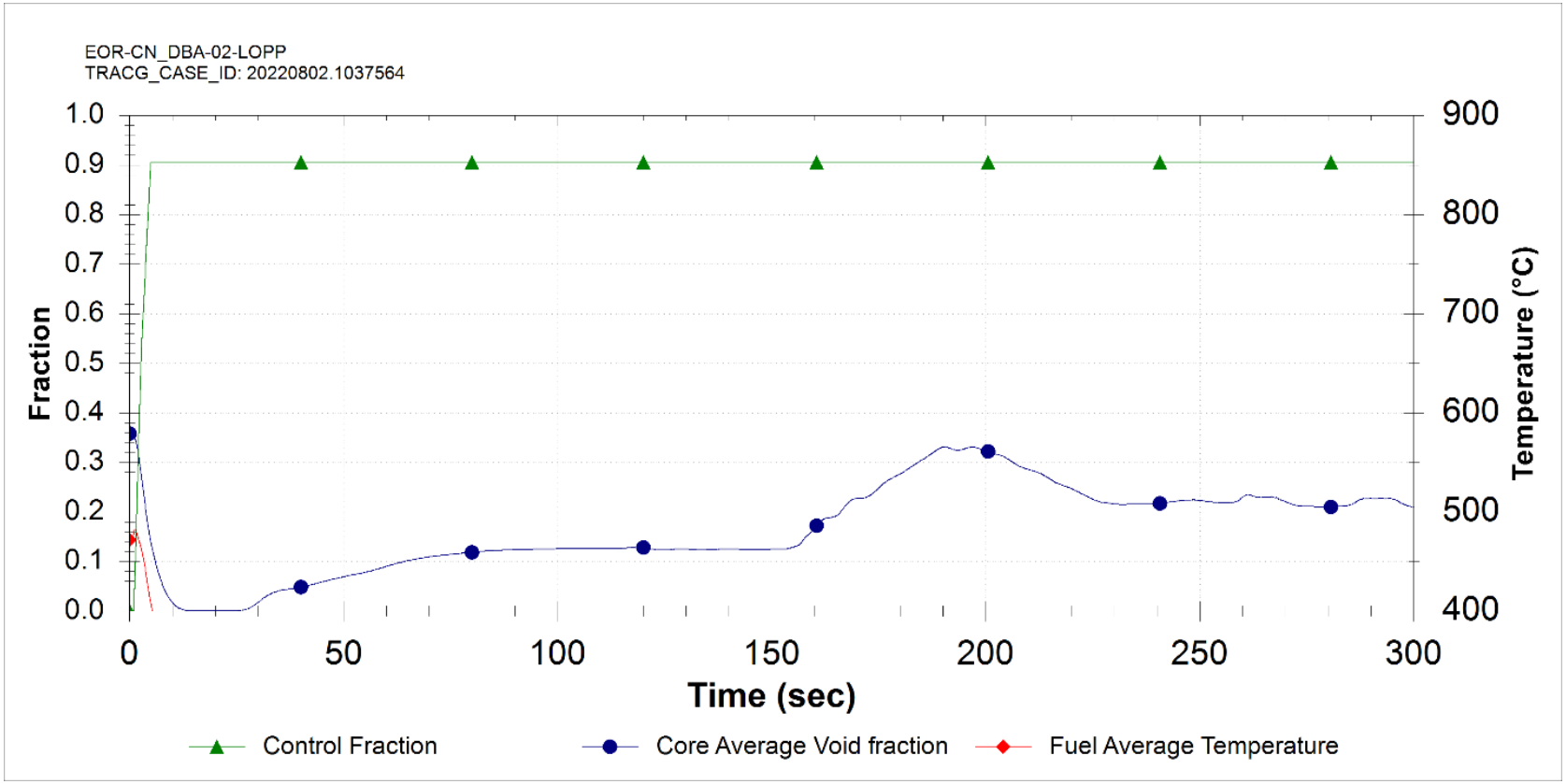


Figure 15.5-62: Loss of Preferred Power (DBA)

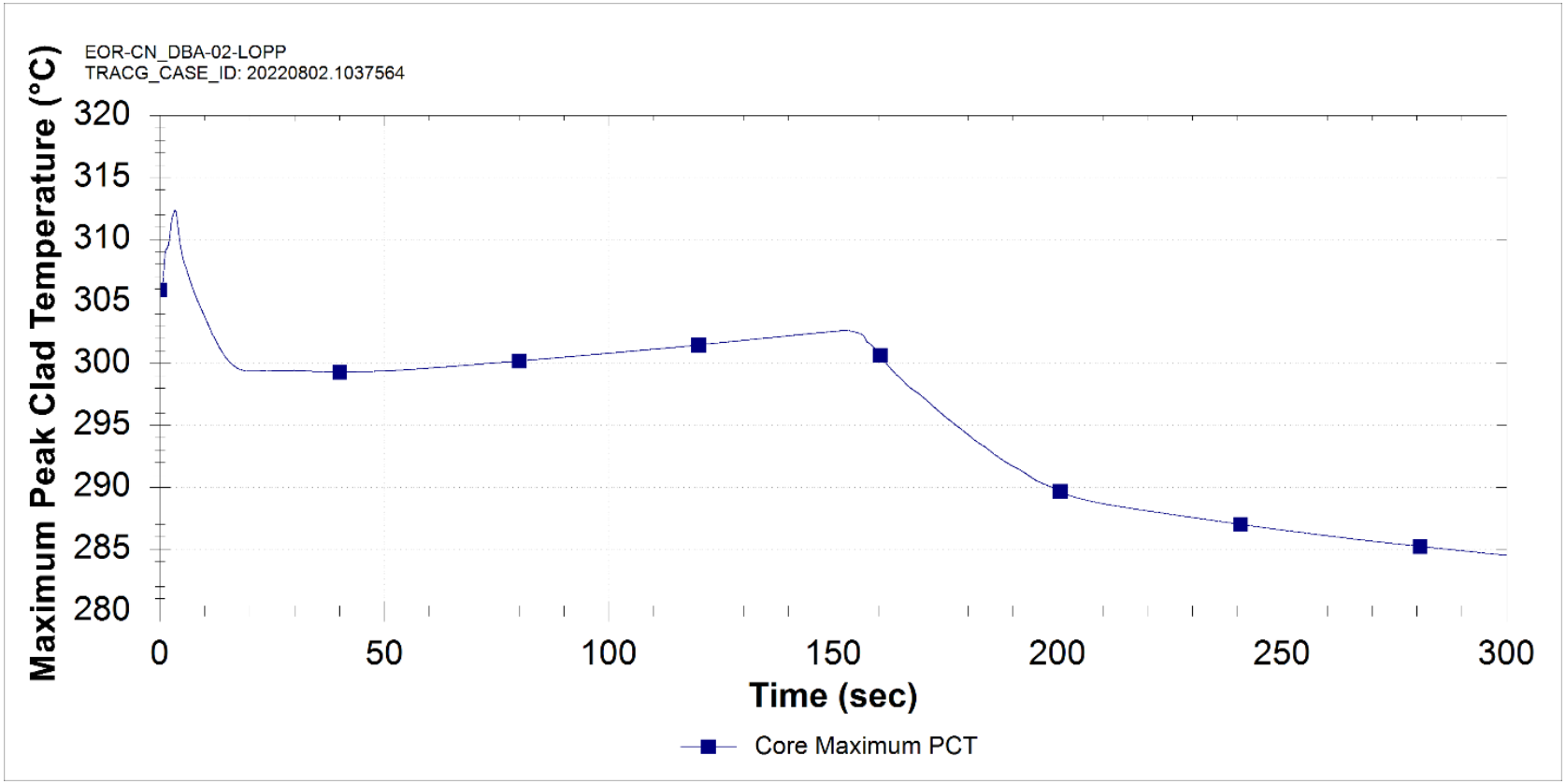


Figure 15.5-63: Loss of Preferred Power (DBA)

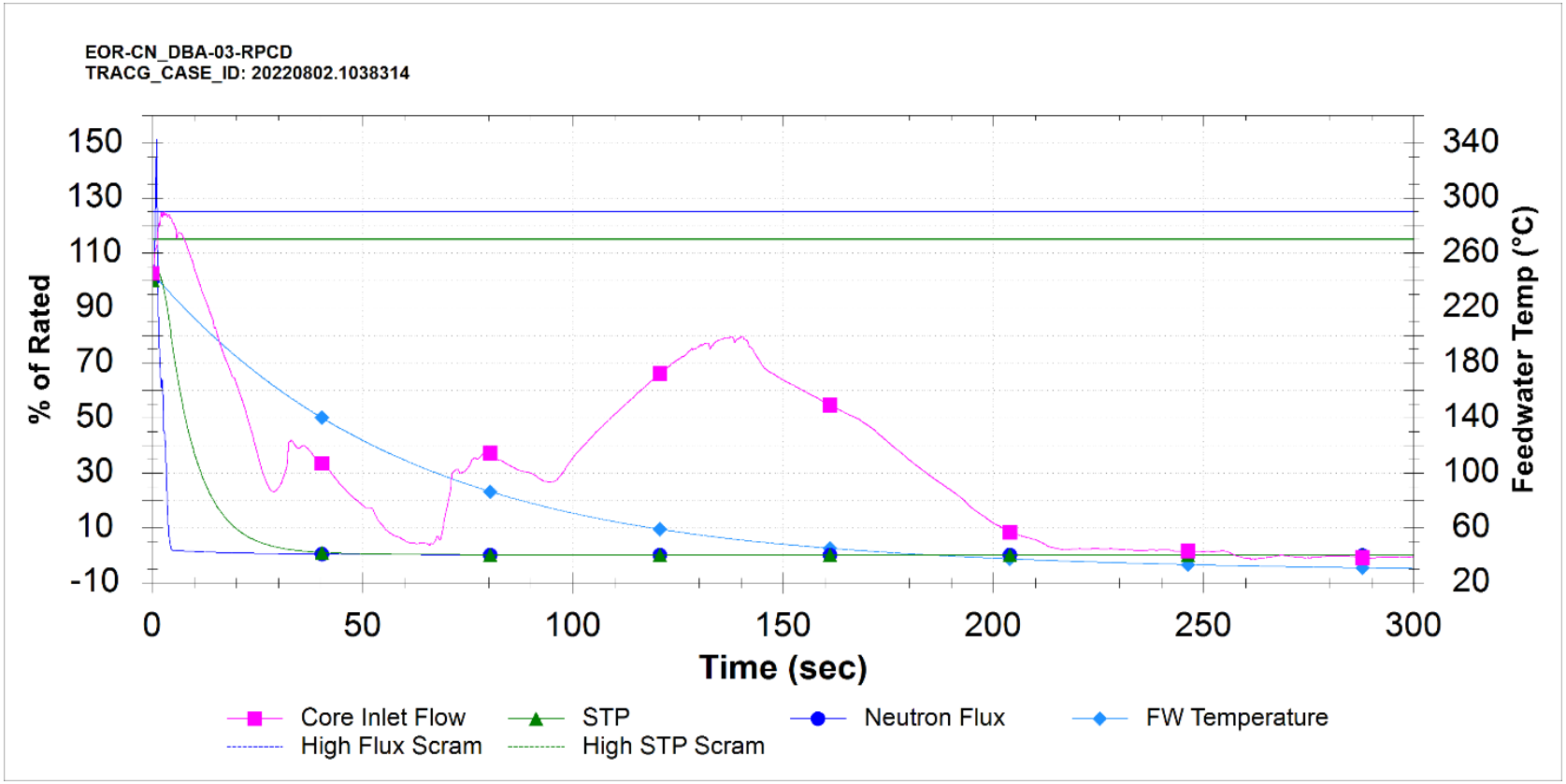


Figure 15.5-64: RPV Pressure Control Downscale (DBA)

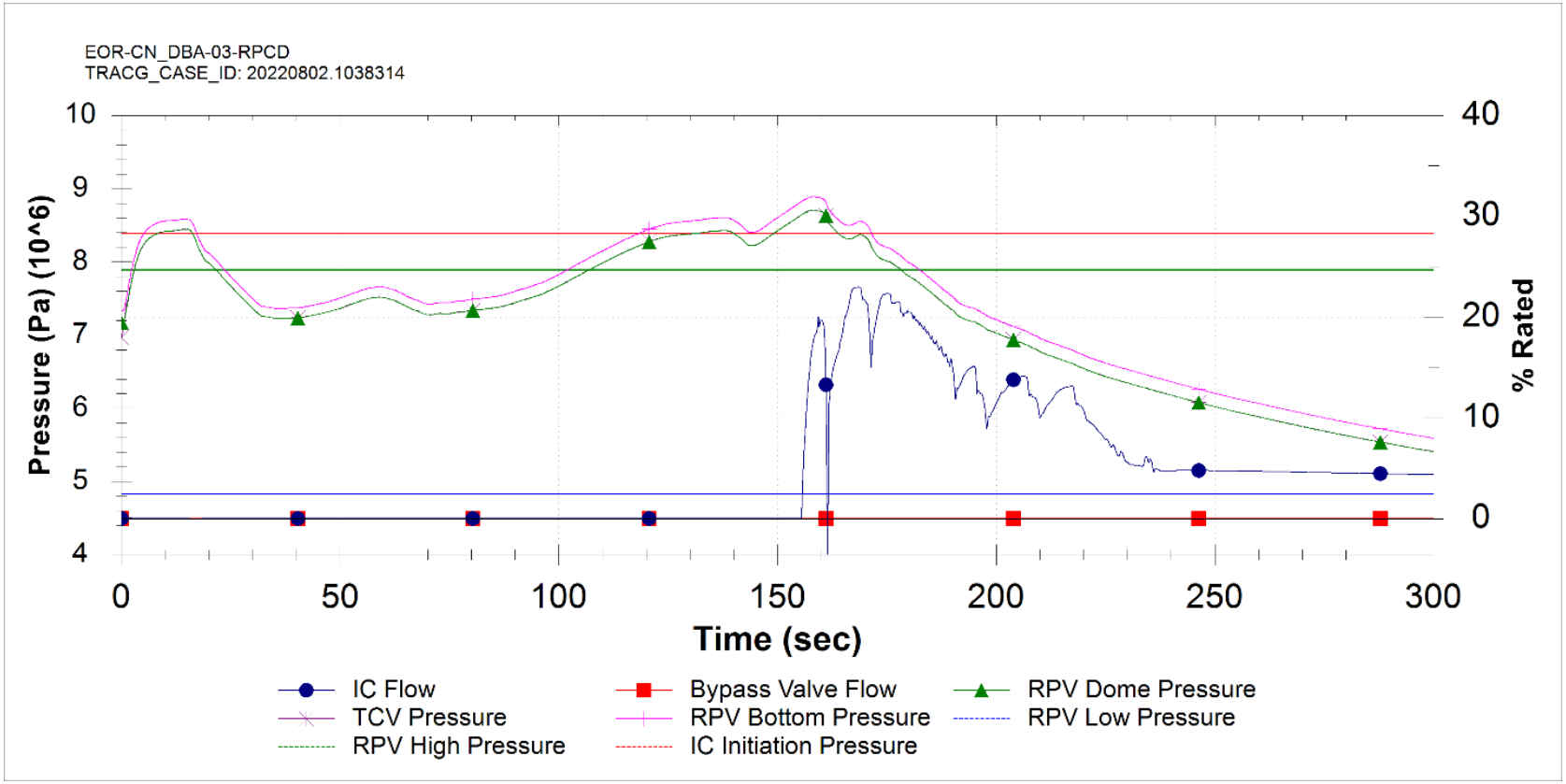


Figure 15.5-65: RPV Pressure Control Downscale (DBA)

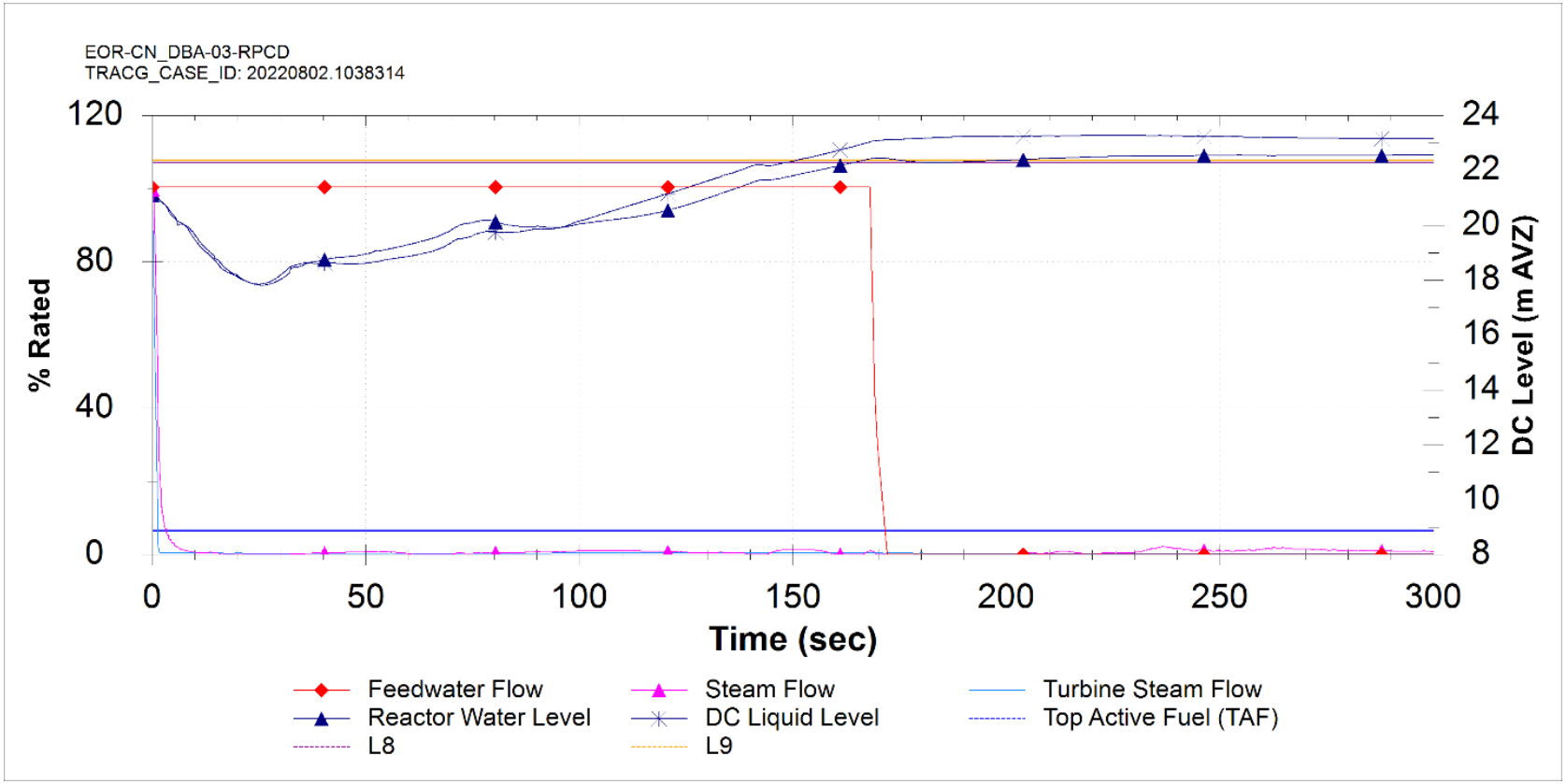


Figure 15.5-66: RPV Pressure Control Downscale (DBA)

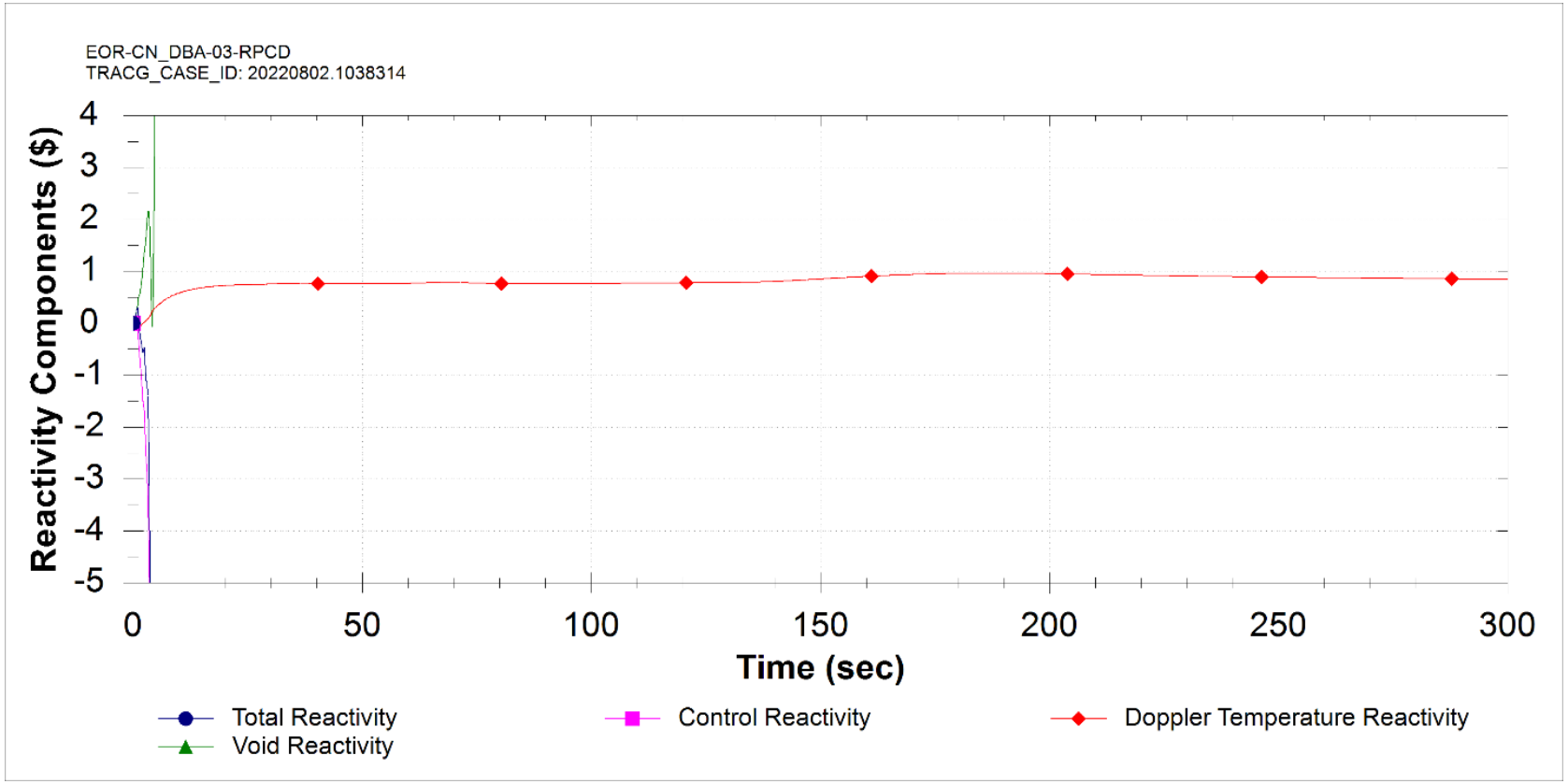


Figure 15.5-67: RPV Pressure Control Downscale (DBA)

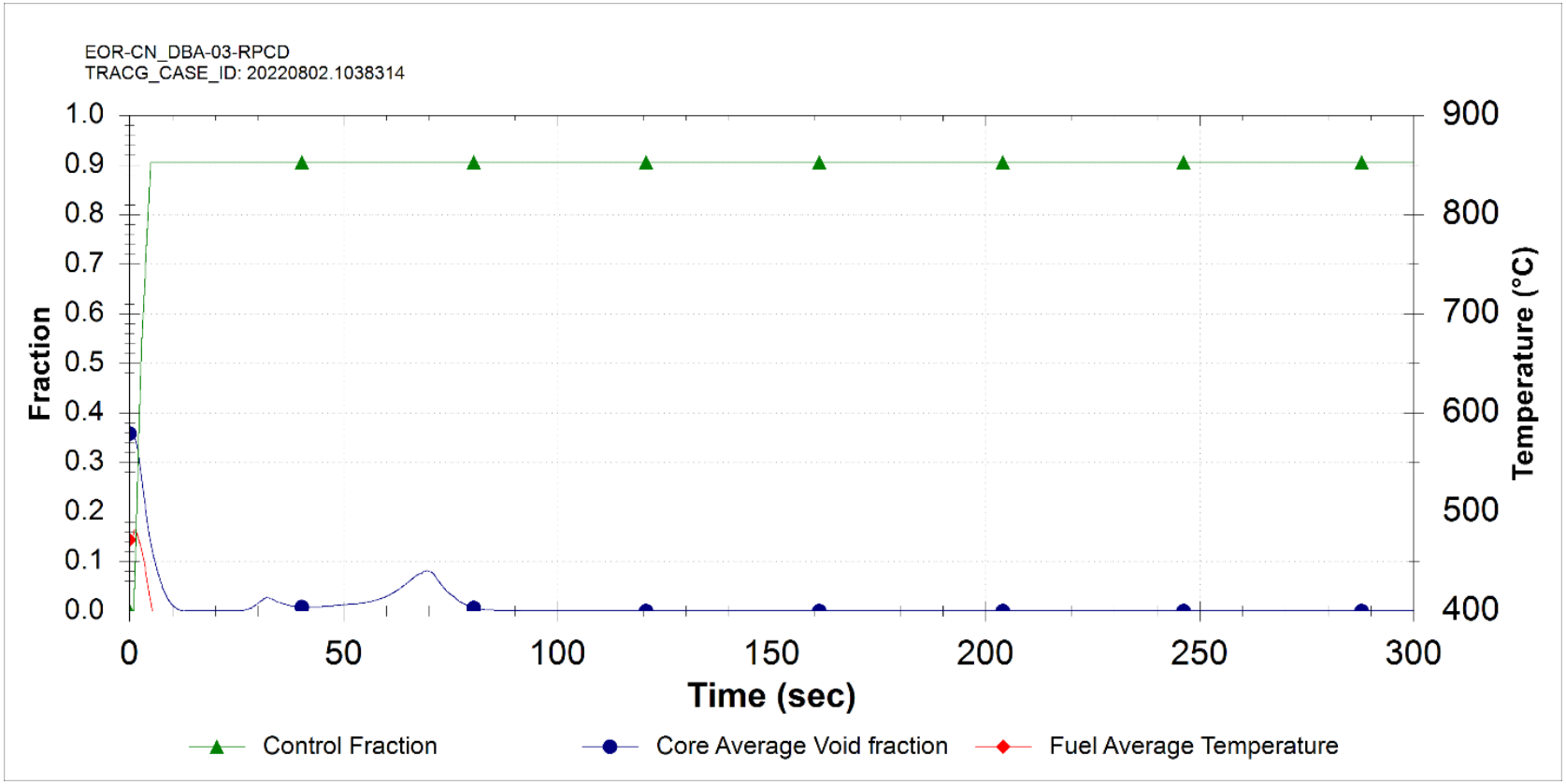


Figure 15.5-68: RPV Pressure Control Downscale (DBA)

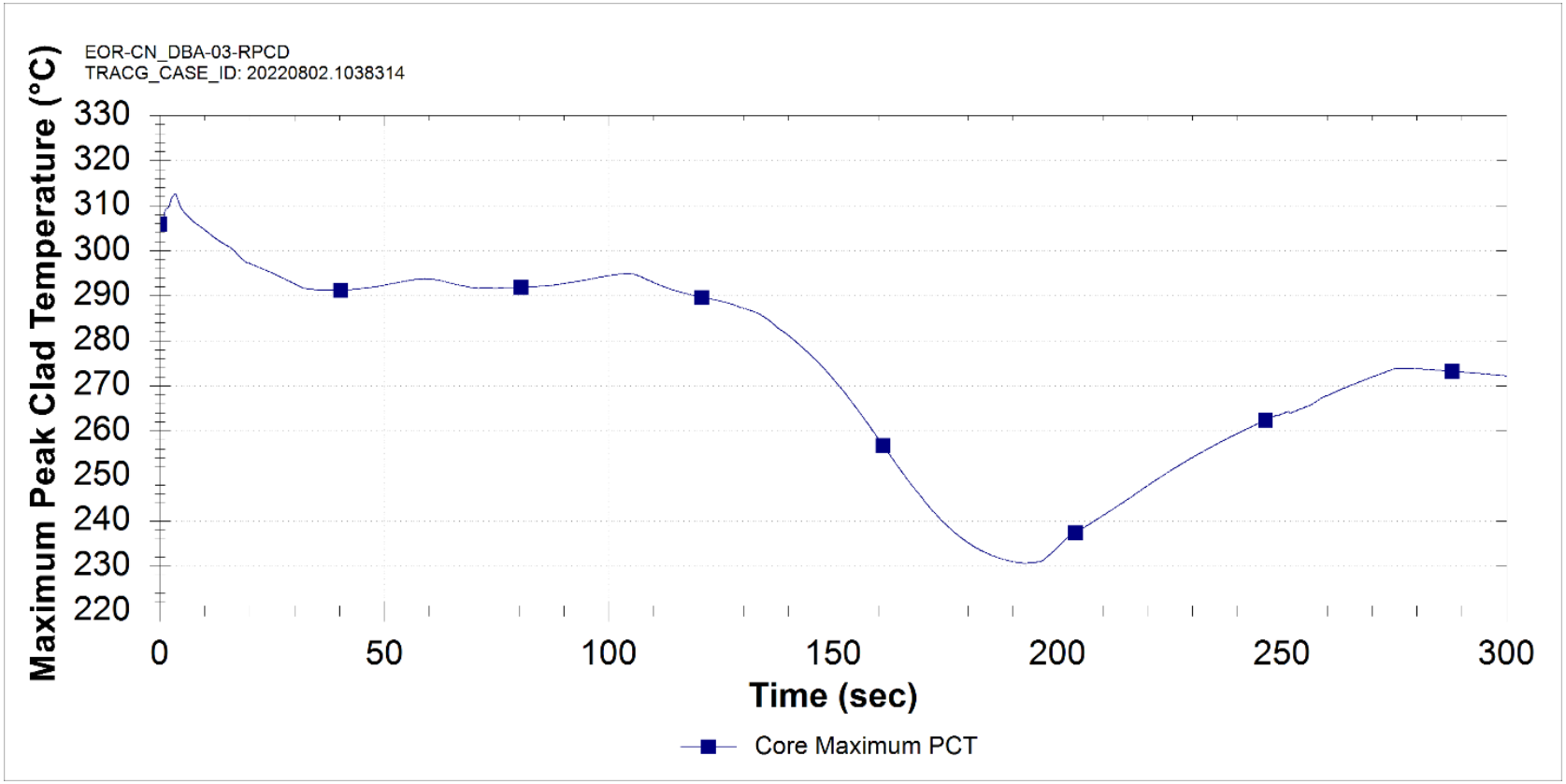


Figure 15.5-69: RPV Pressure Control Downscale (DBA)

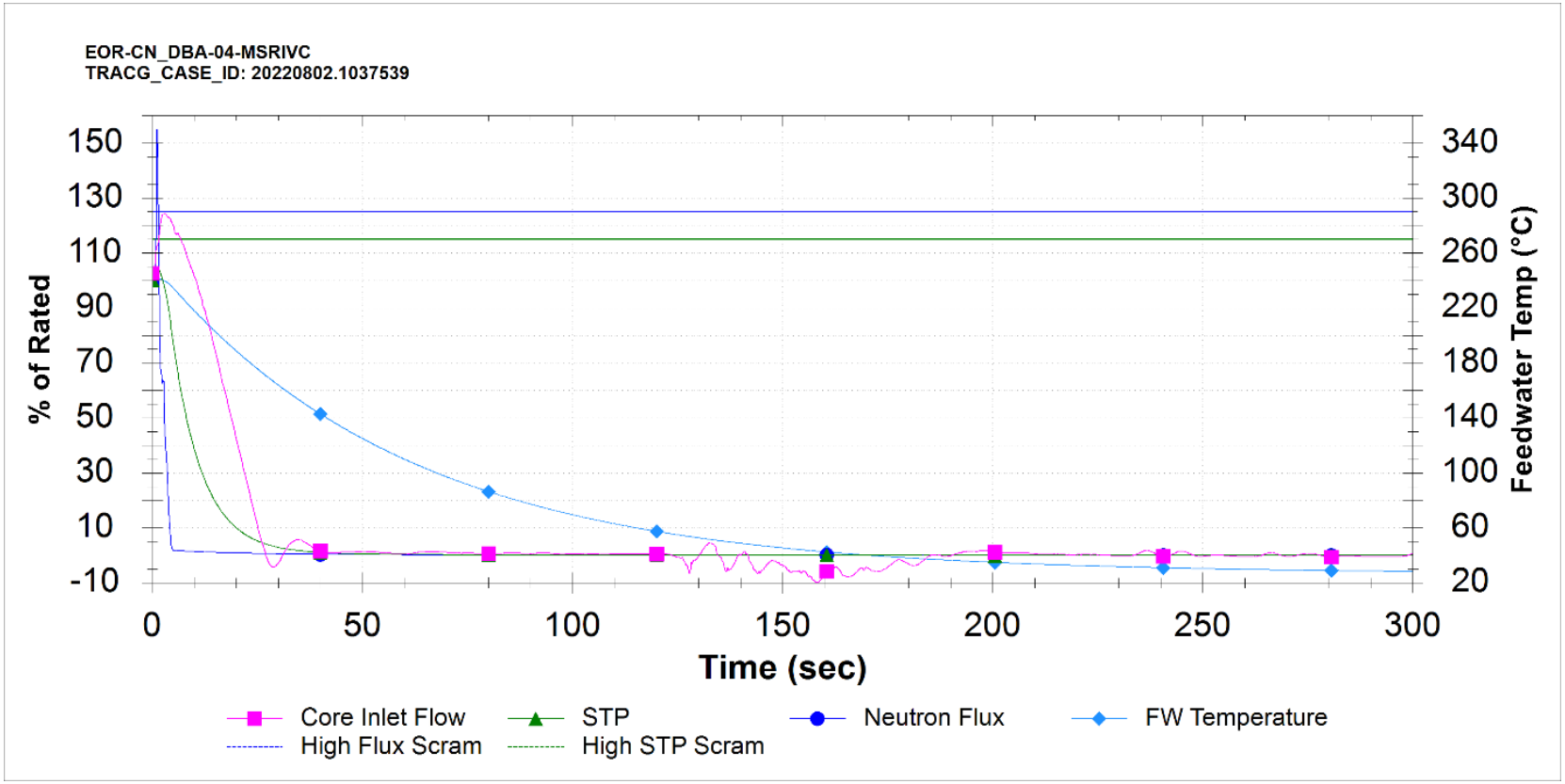


Figure 15.5-70: Closure of All MSRIVs and FW Isolation Valves (DBA)

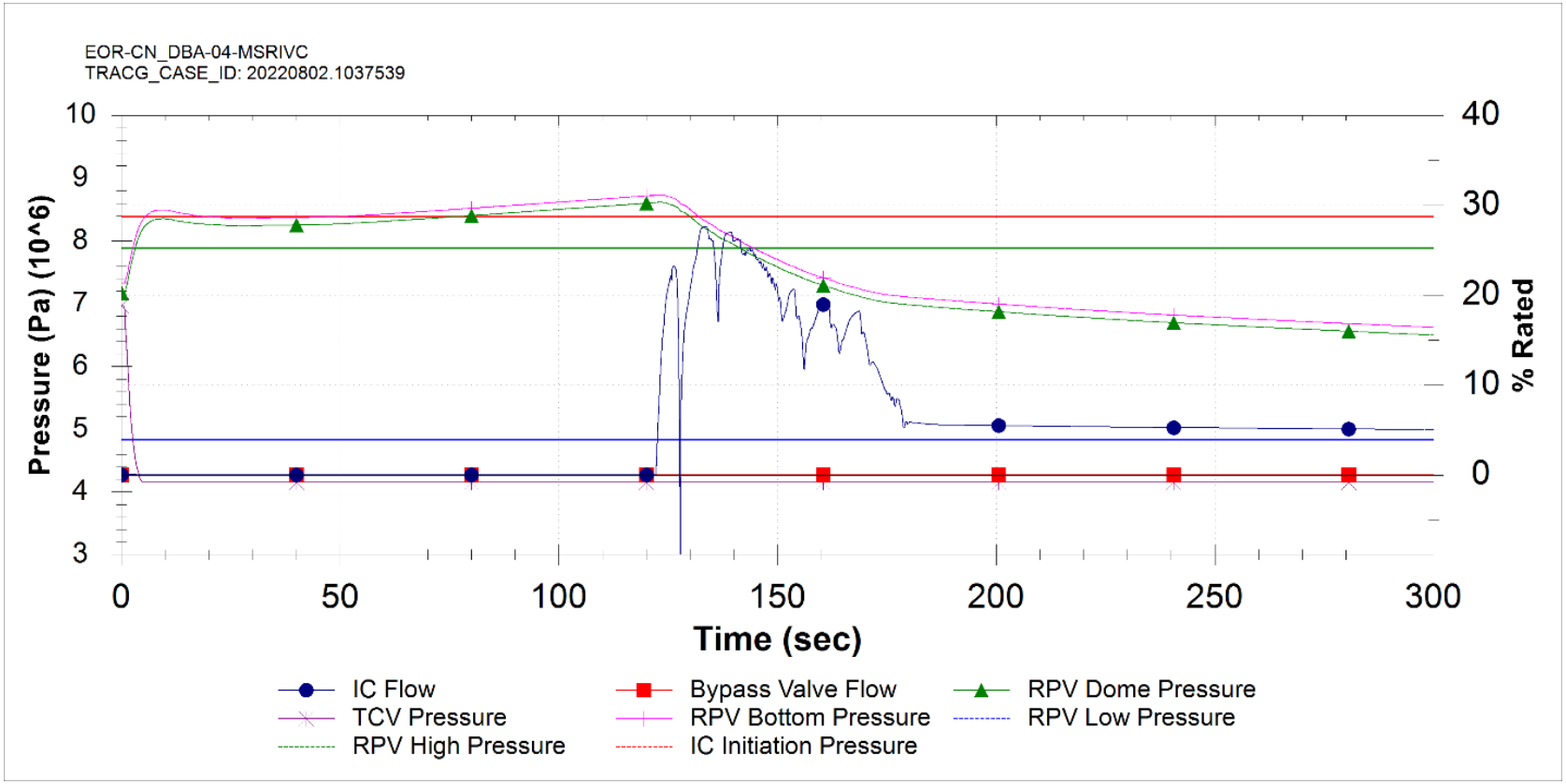


Figure 15.5-71: Closure of All MSRIVs and FW Isolation Valves (DBA)

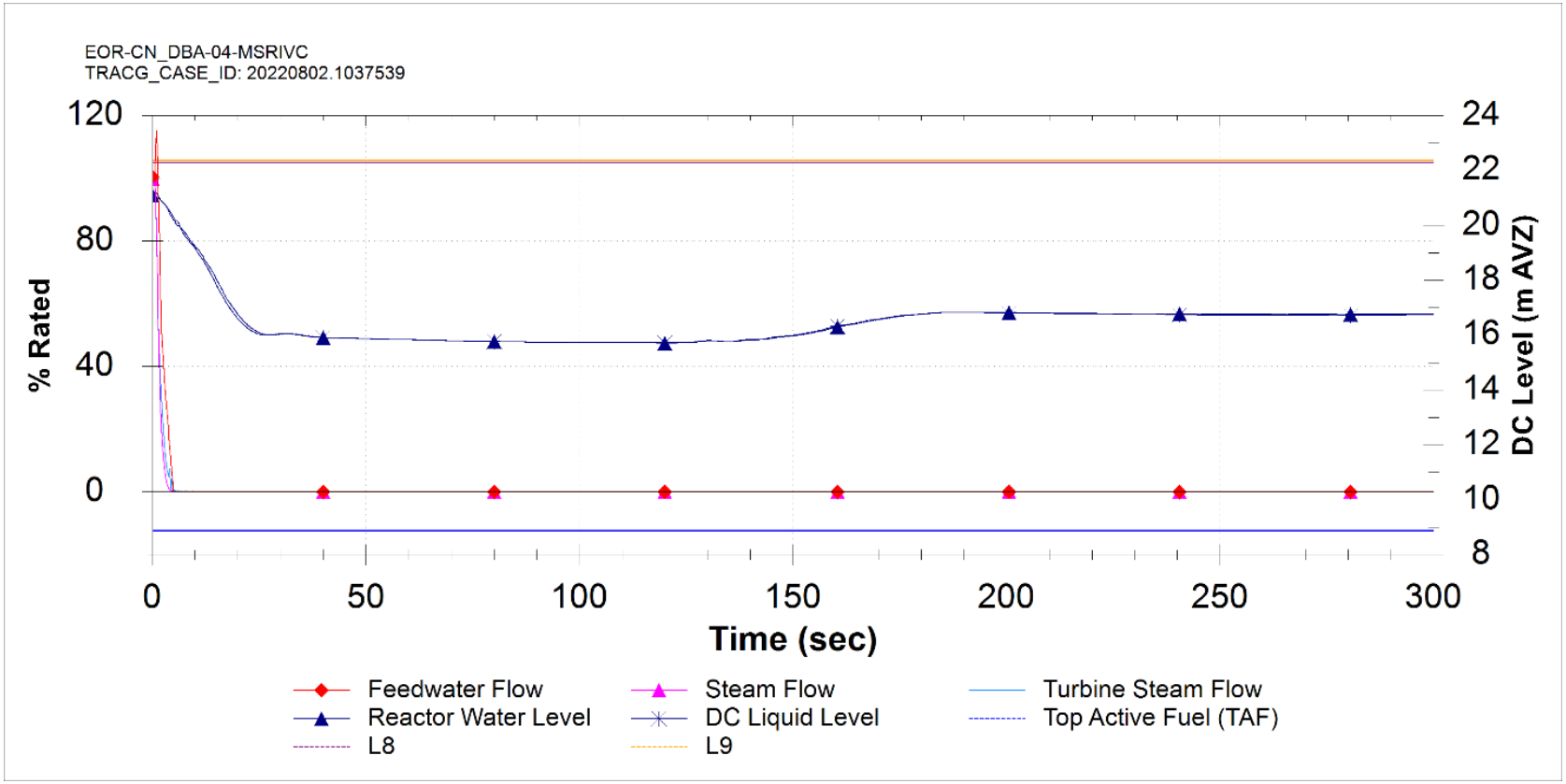


Figure 15.5-72: Closure of All MSRIVs and FW Isolation Valves (DBA)

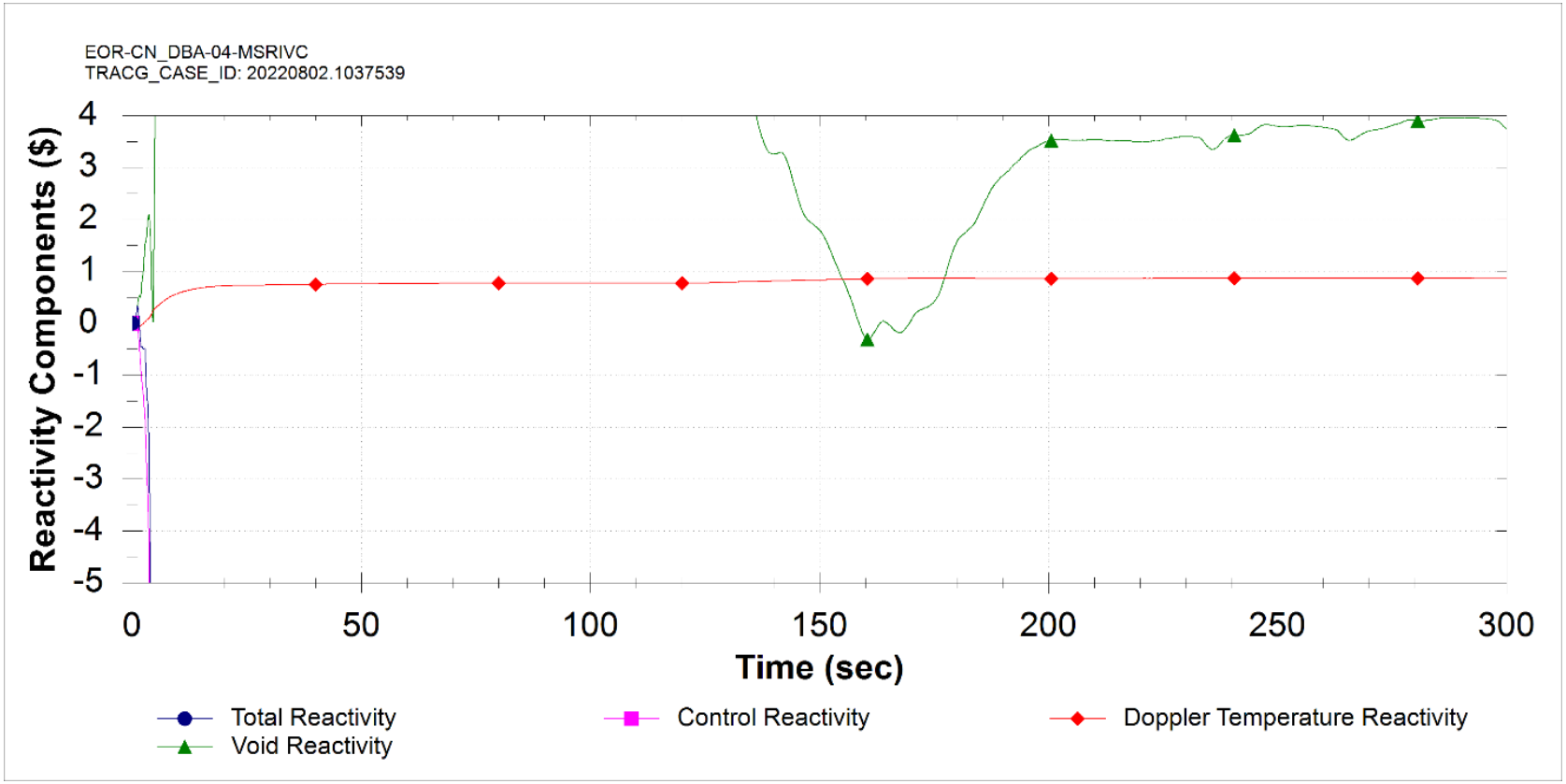


Figure 15.5-73: Closure of All MSRIVs and FW Isolation Valves (DBA)

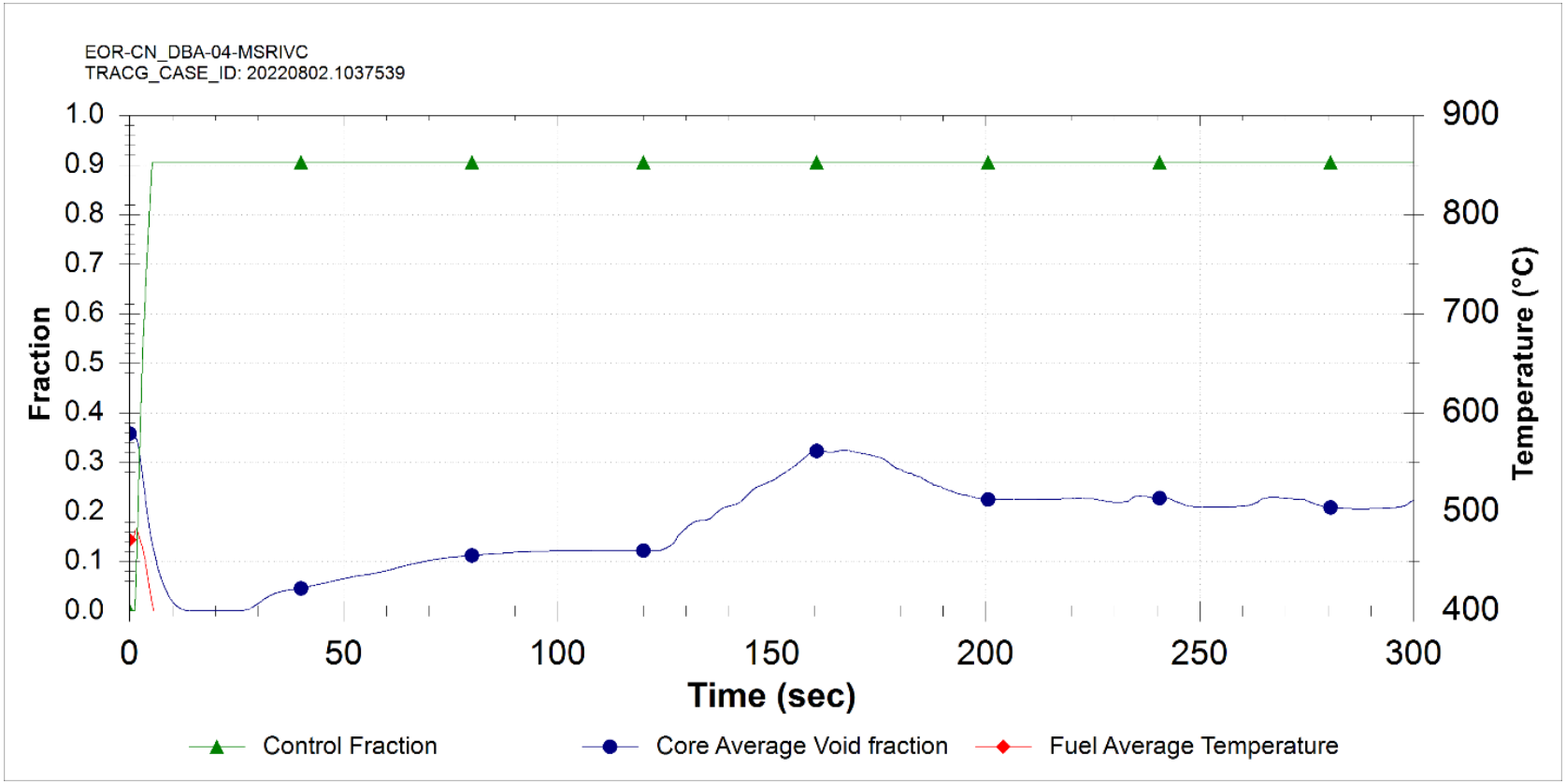


Figure 15.5-74: Closure of All MSRIVs and FW Isolation Valves (DBA)

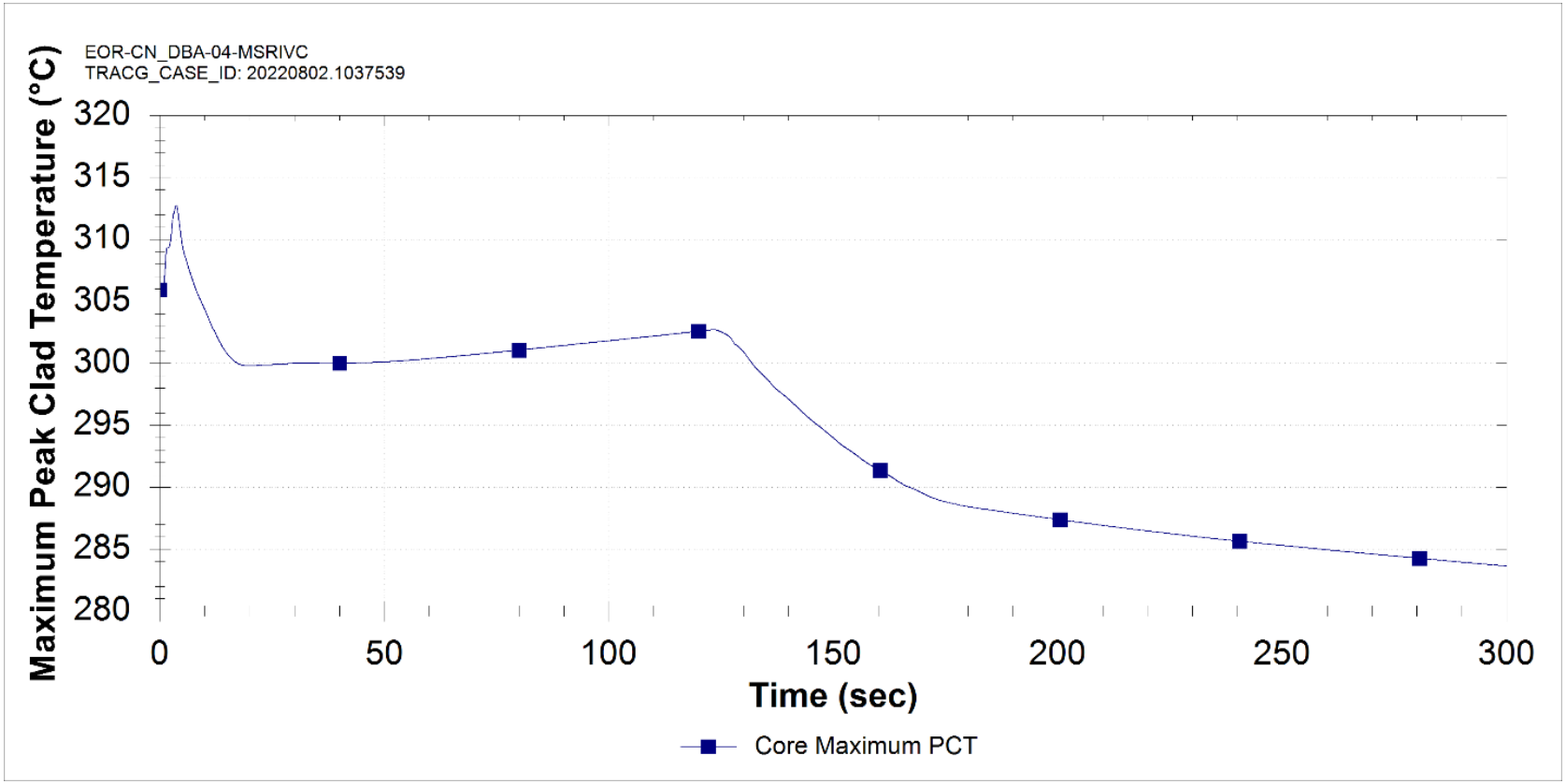


Figure 15.5-75: Closure of All MSRIVs and FW Isolation Valves (DBA)

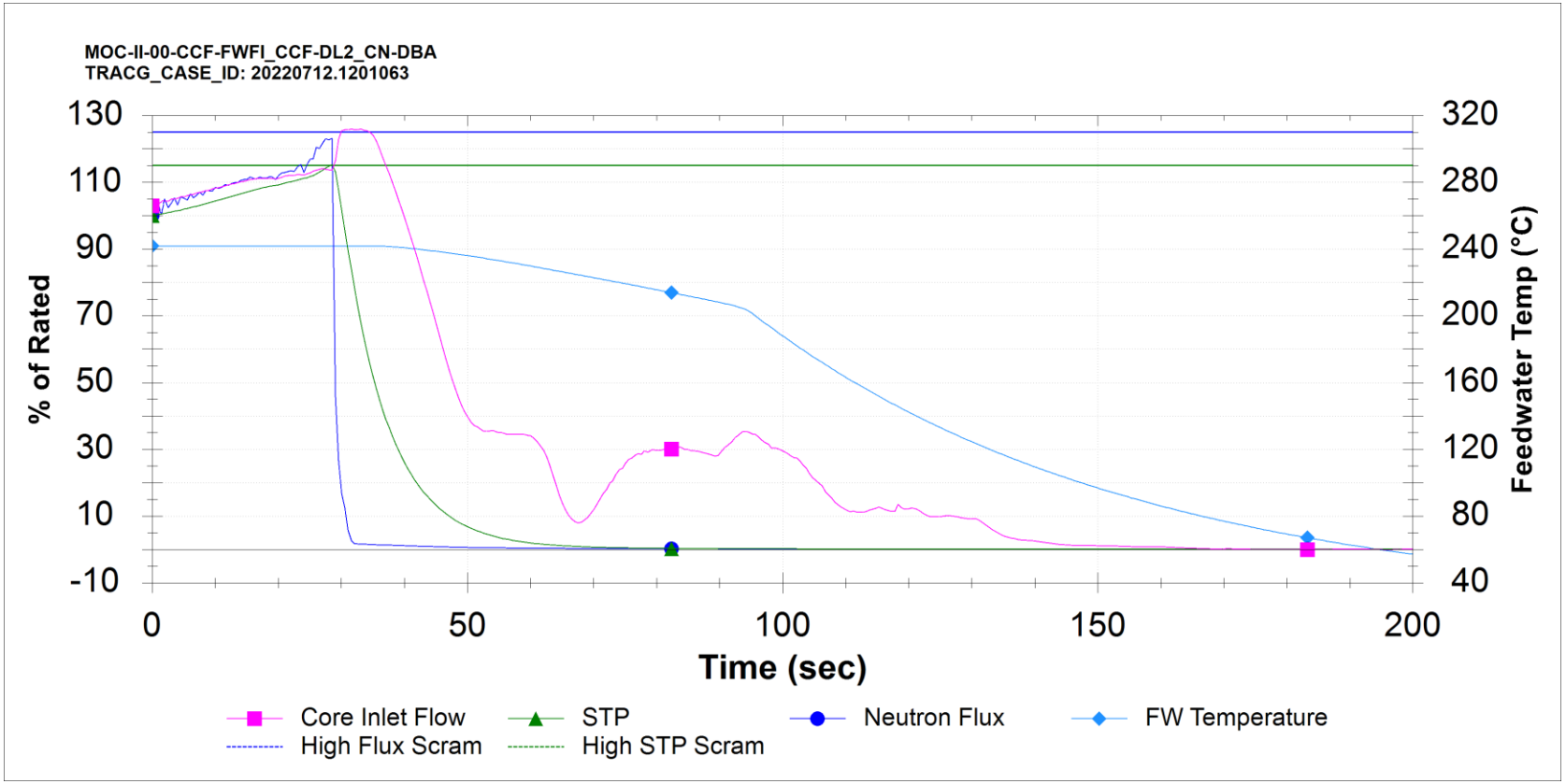


Figure 15.5-76: Feedwater Flow Increase (DBA)

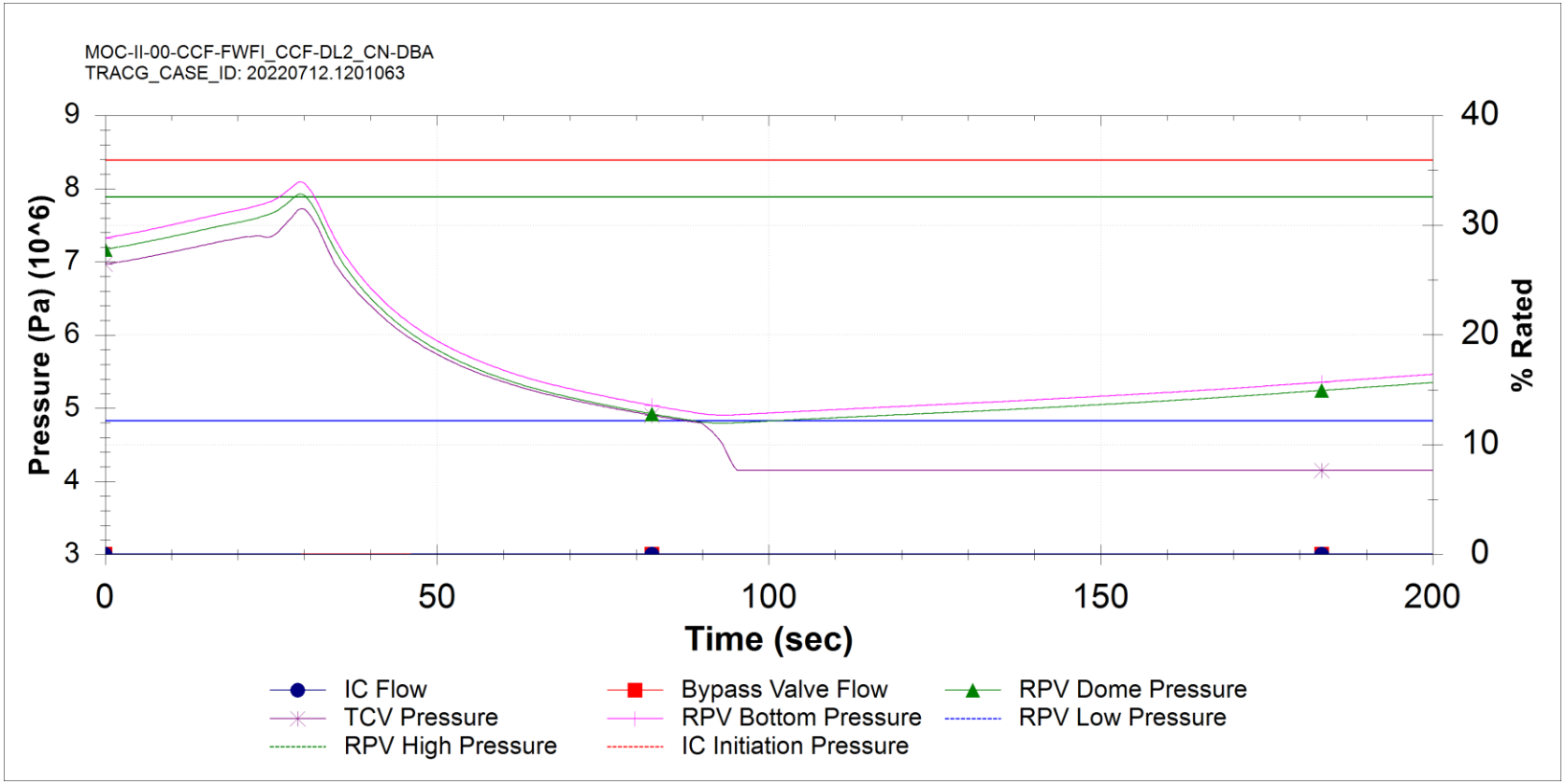


Figure 15.5-77: Feedwater Flow Increase (DBA)

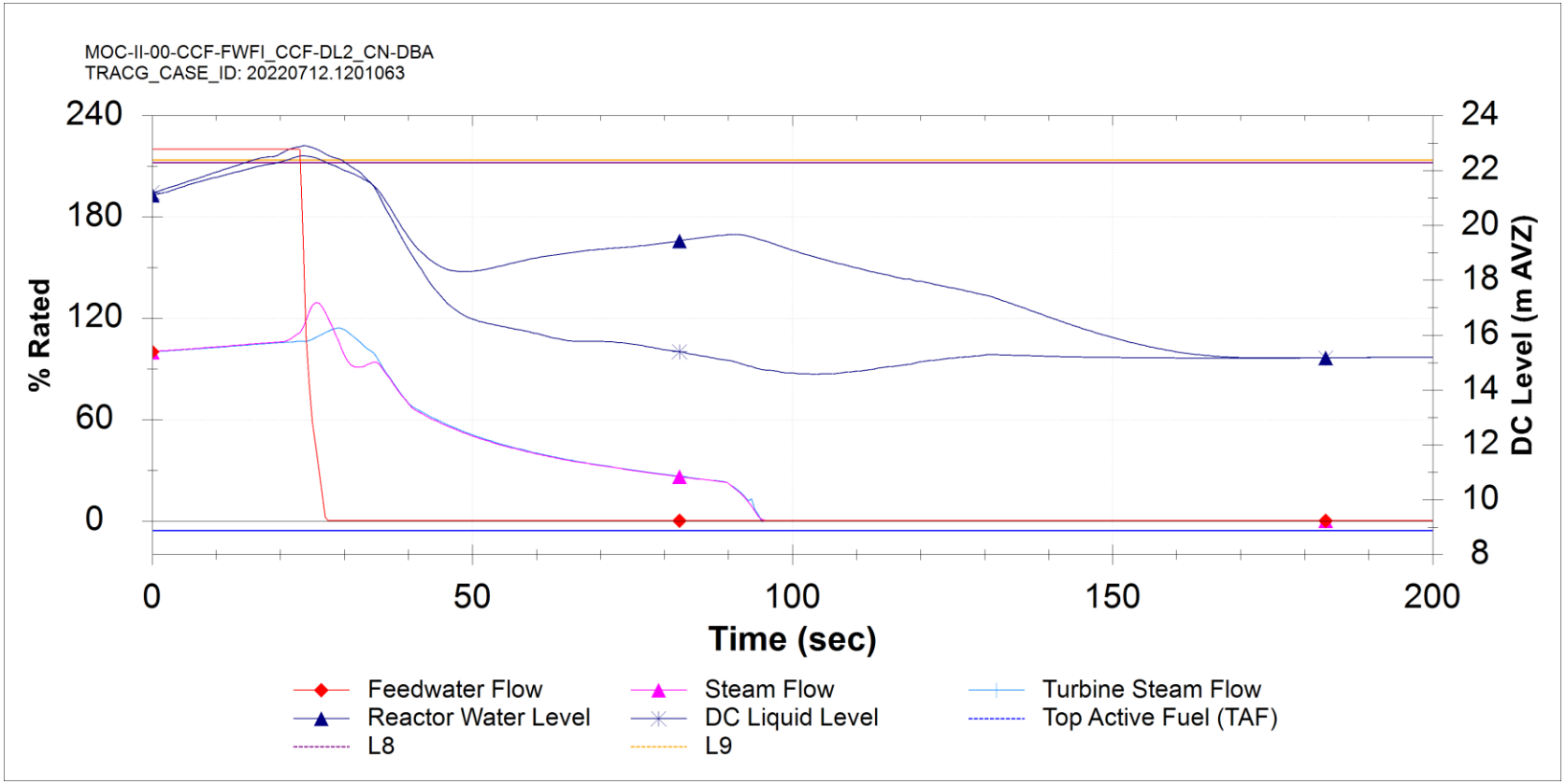


Figure 15.5-78: Feedwater Flow Increase (DBA)

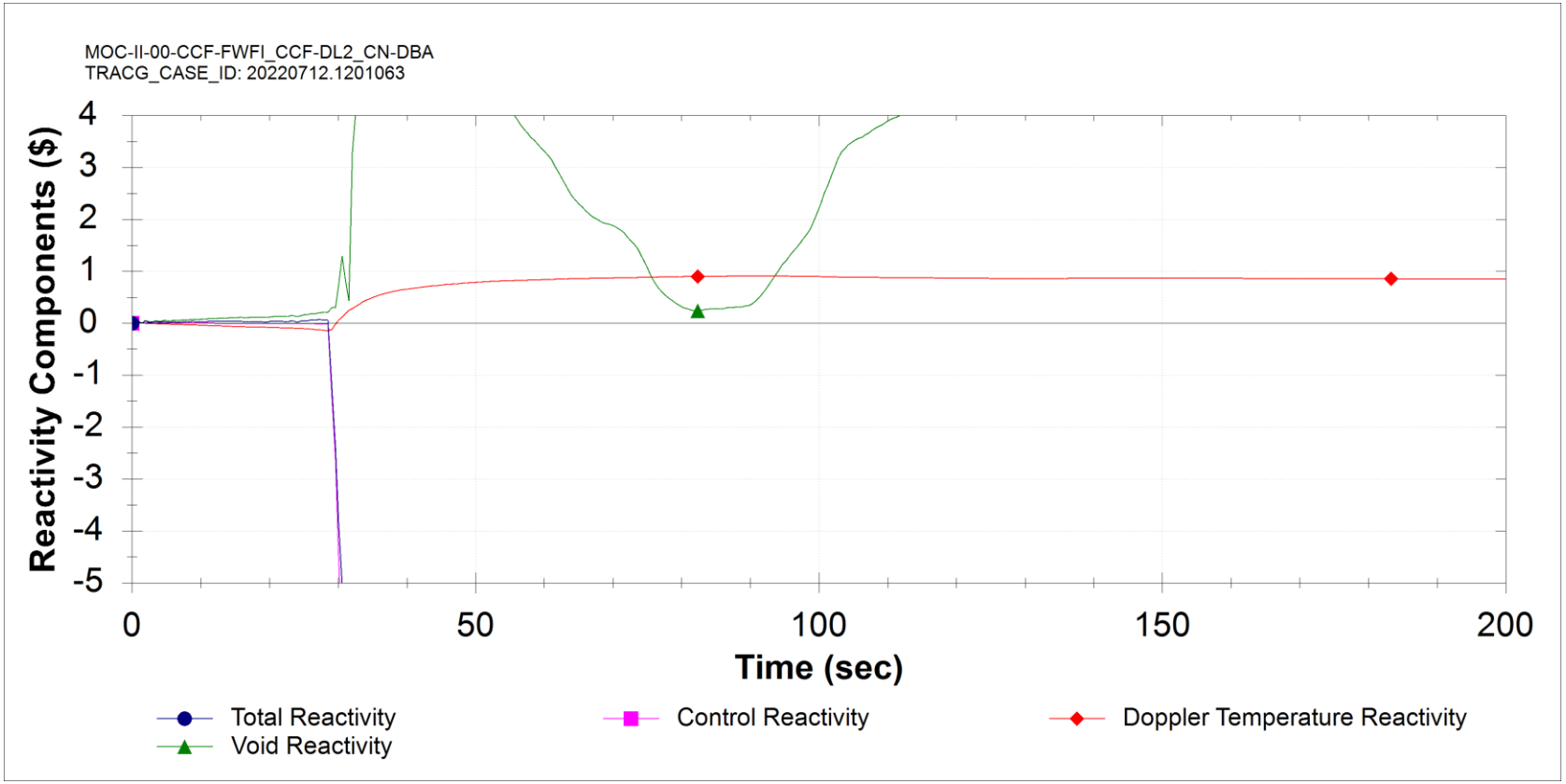


Figure 15.5-79: Feedwater Flow Increase (DBA)

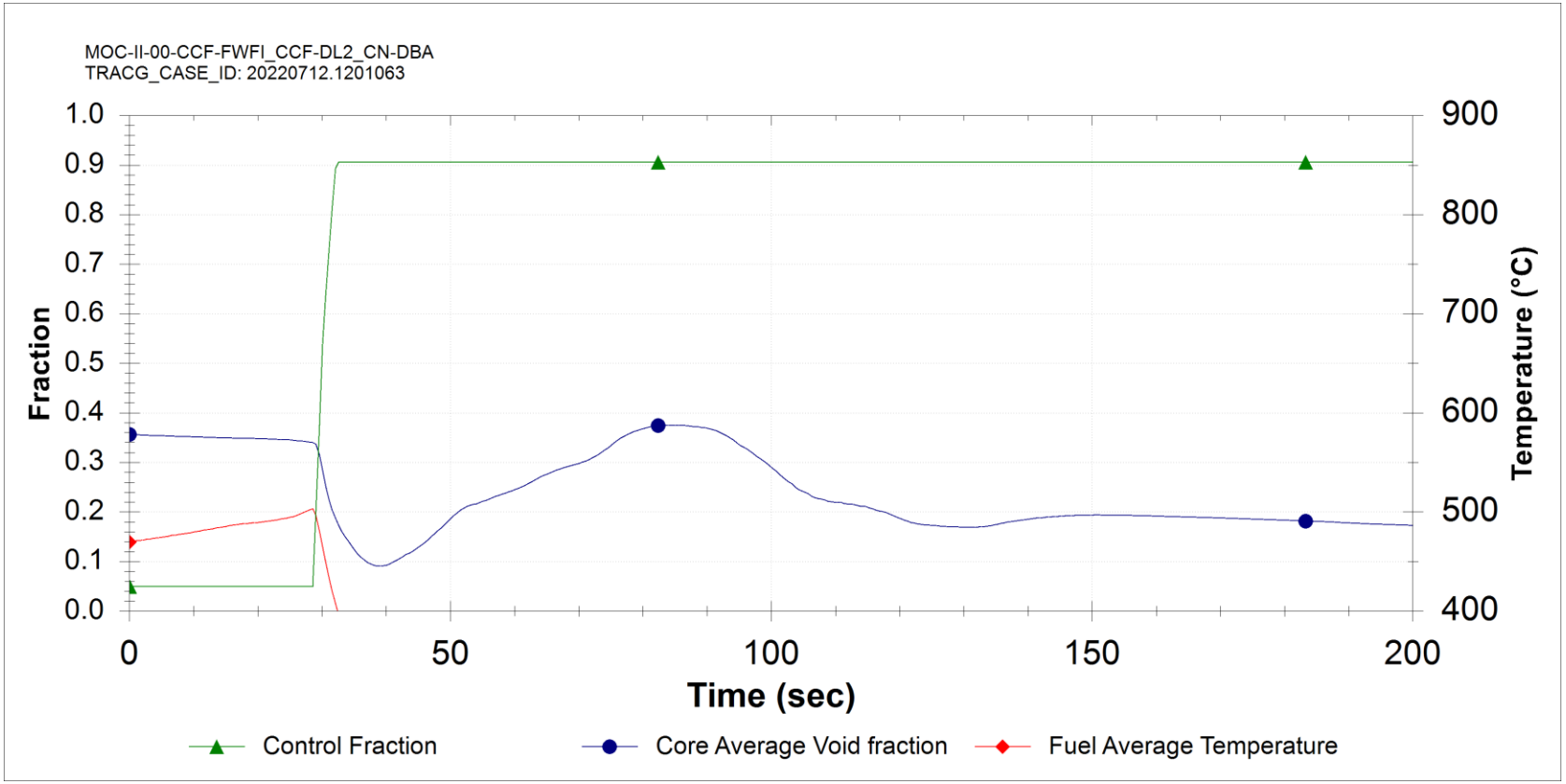


Figure 15.5-80: Feedwater Flow Increase (DBA)

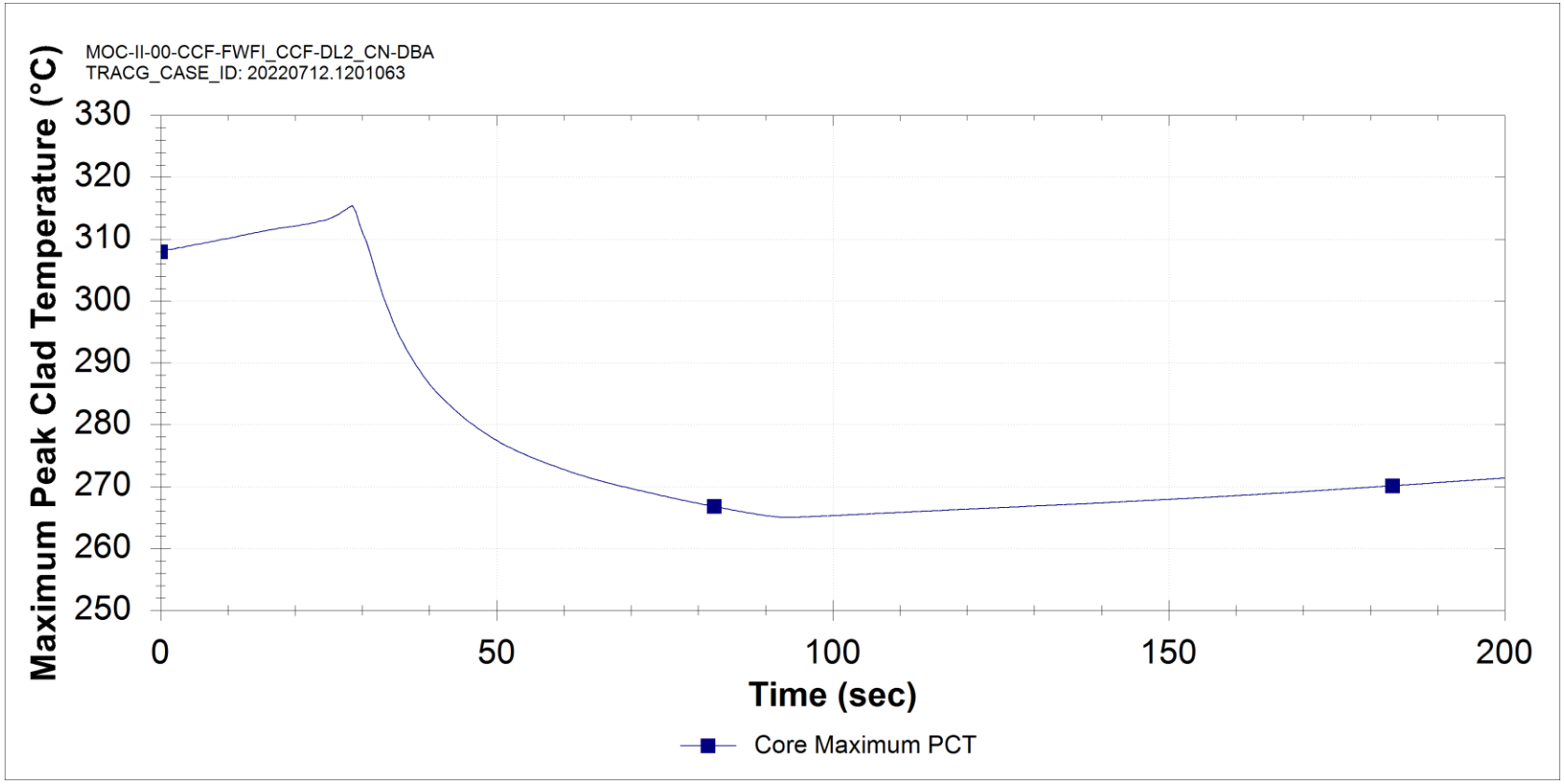


Figure 15.5-81: Feedwater Flow Increase (DBA)

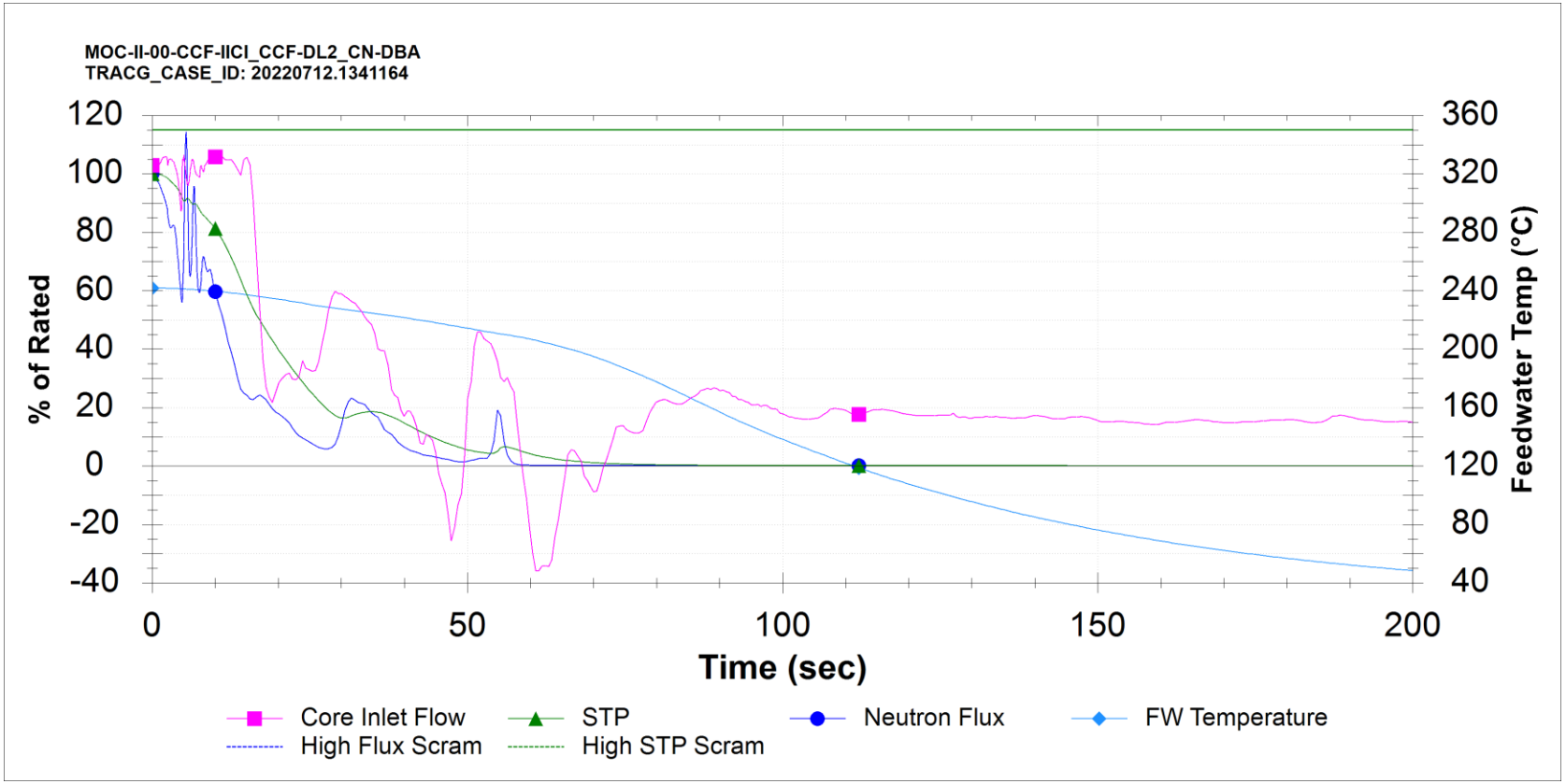


Figure 15.5-82: Inadvertent Isolation Condenser Initiation - All Trains (DBA)

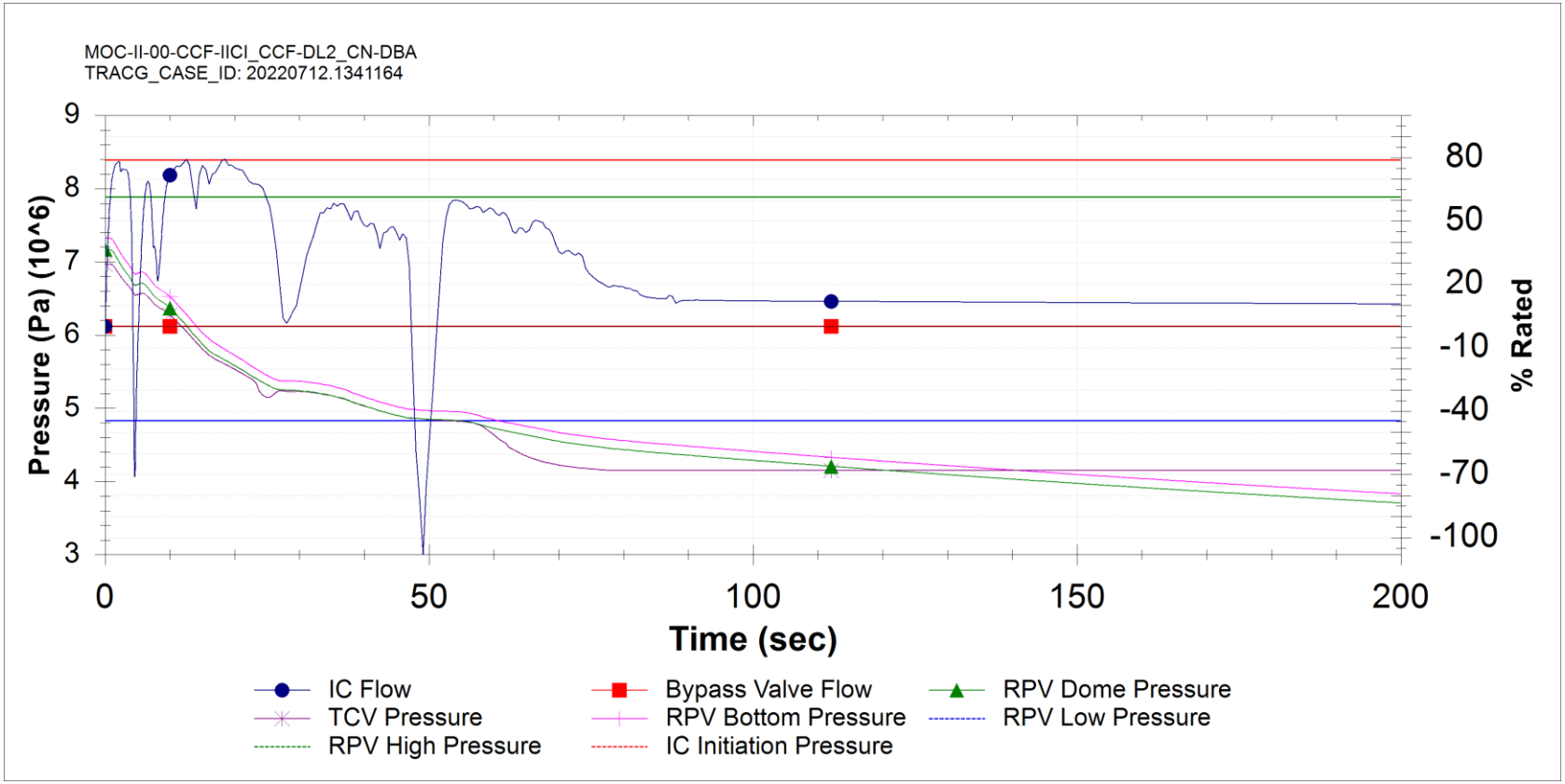


Figure 15.5-83: Inadvertent Isolation Condenser Initiation - All Trains (DBA)

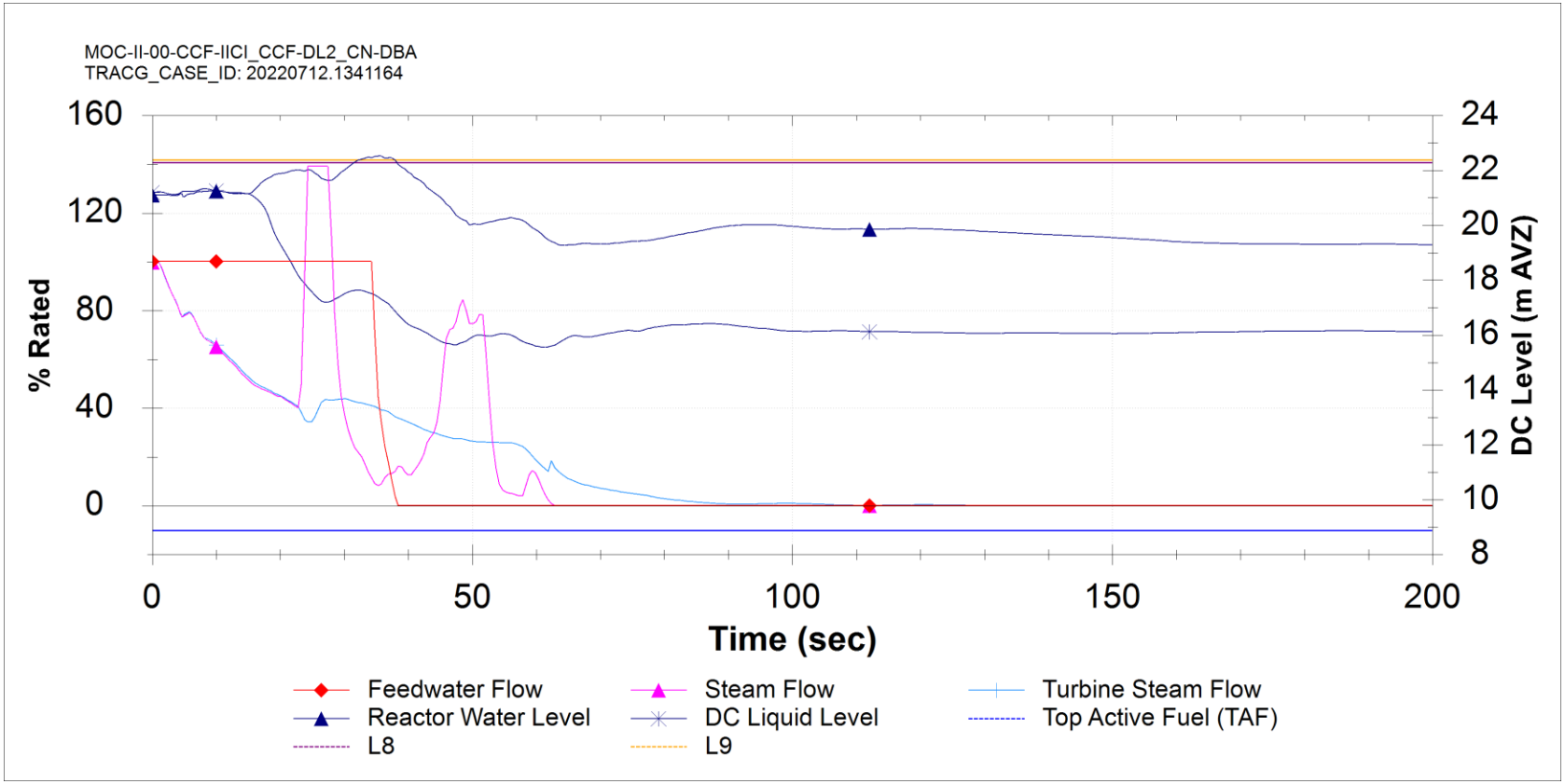


Figure 15.5-84: Inadvertent Isolation Condenser Initiation - All Trains (DBA)

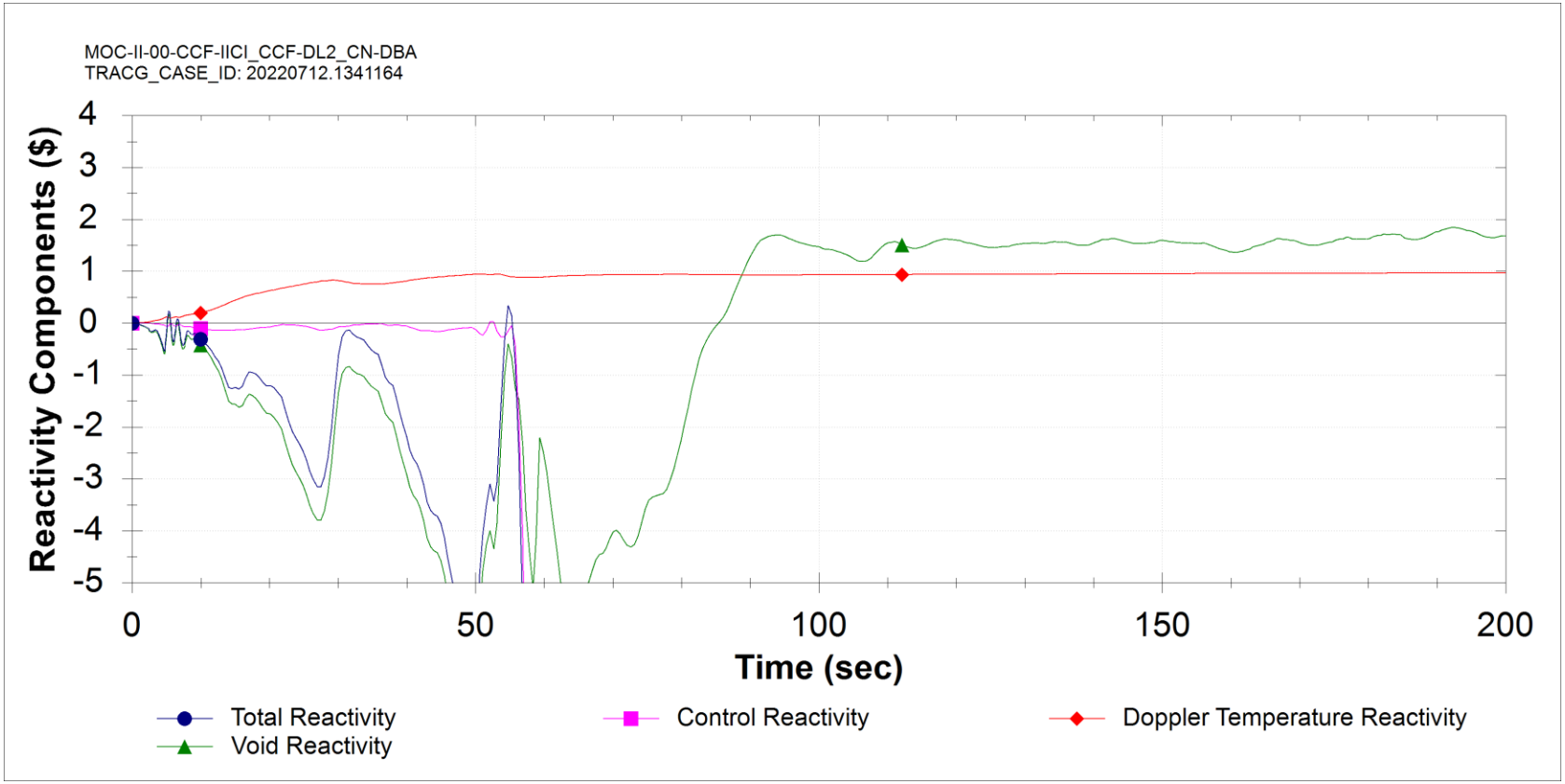


Figure 15.5-85: Inadvertent Isolation Condenser Initiation - All Trains (DBA)

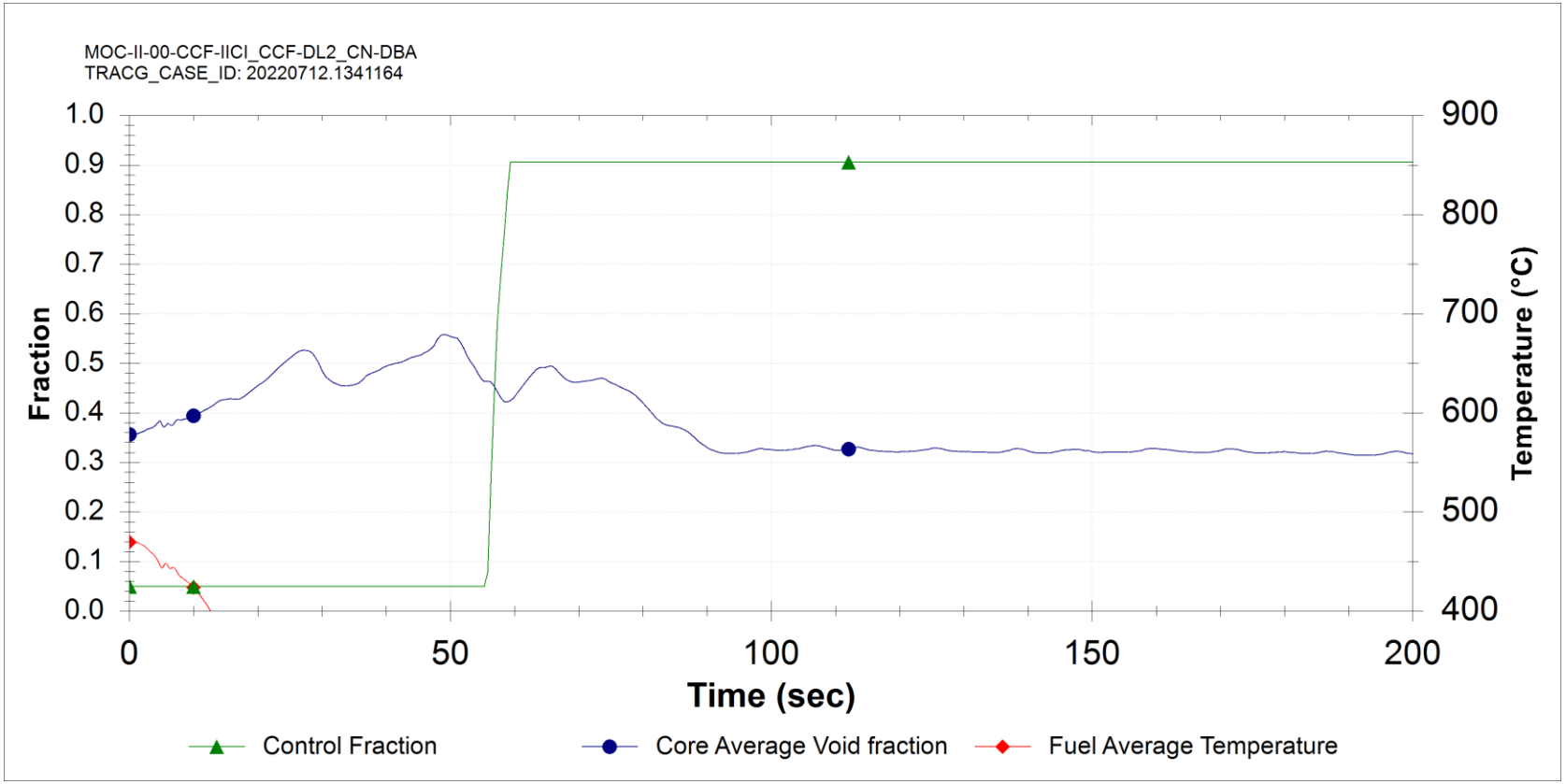


Figure 15.5-86: Inadvertent Isolation Condenser Initiation - All Trains (DBA)

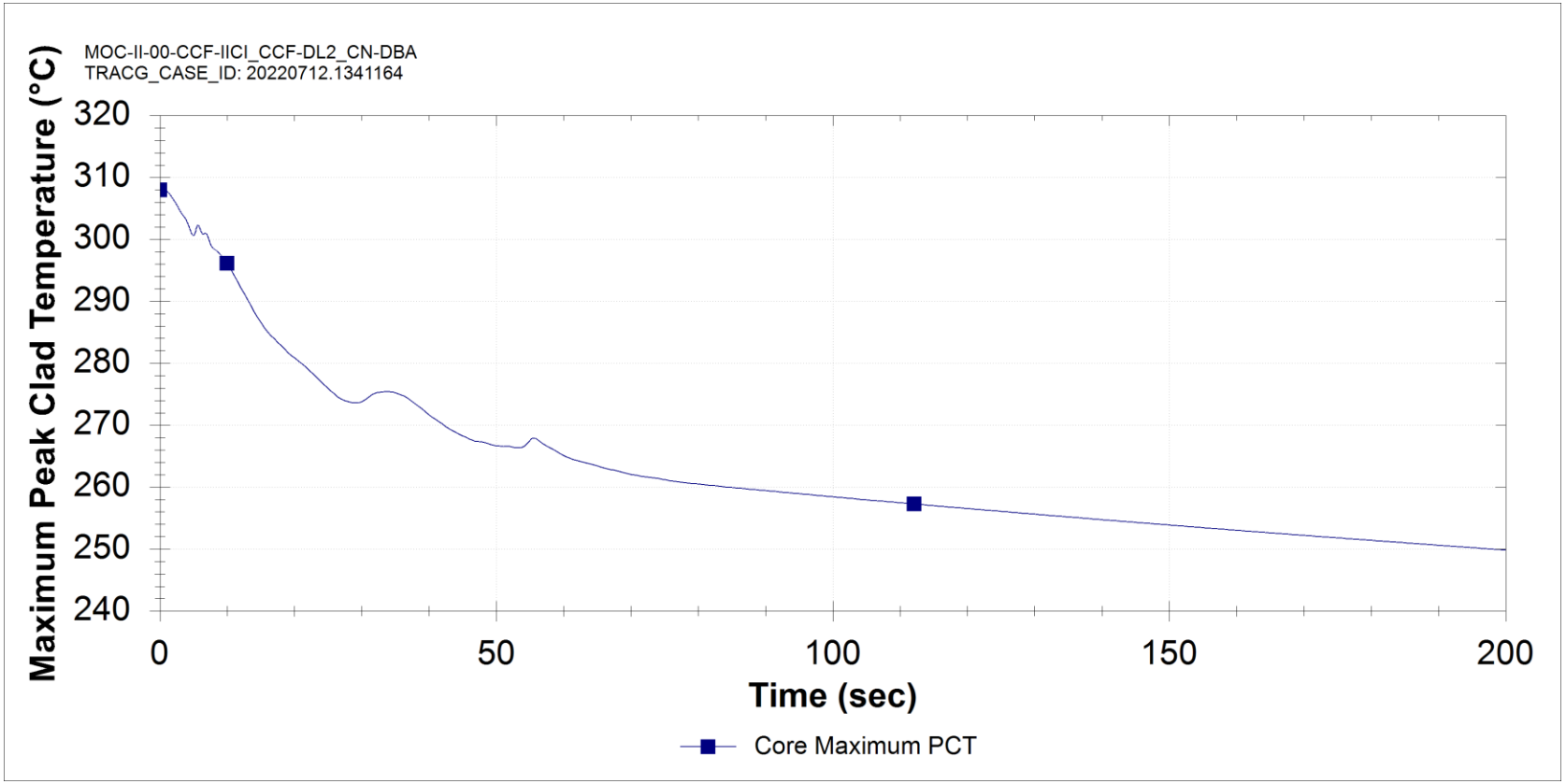


Figure 15.5-87: Inadvertent Isolation Condenser Initiation - All Trains (DBA)

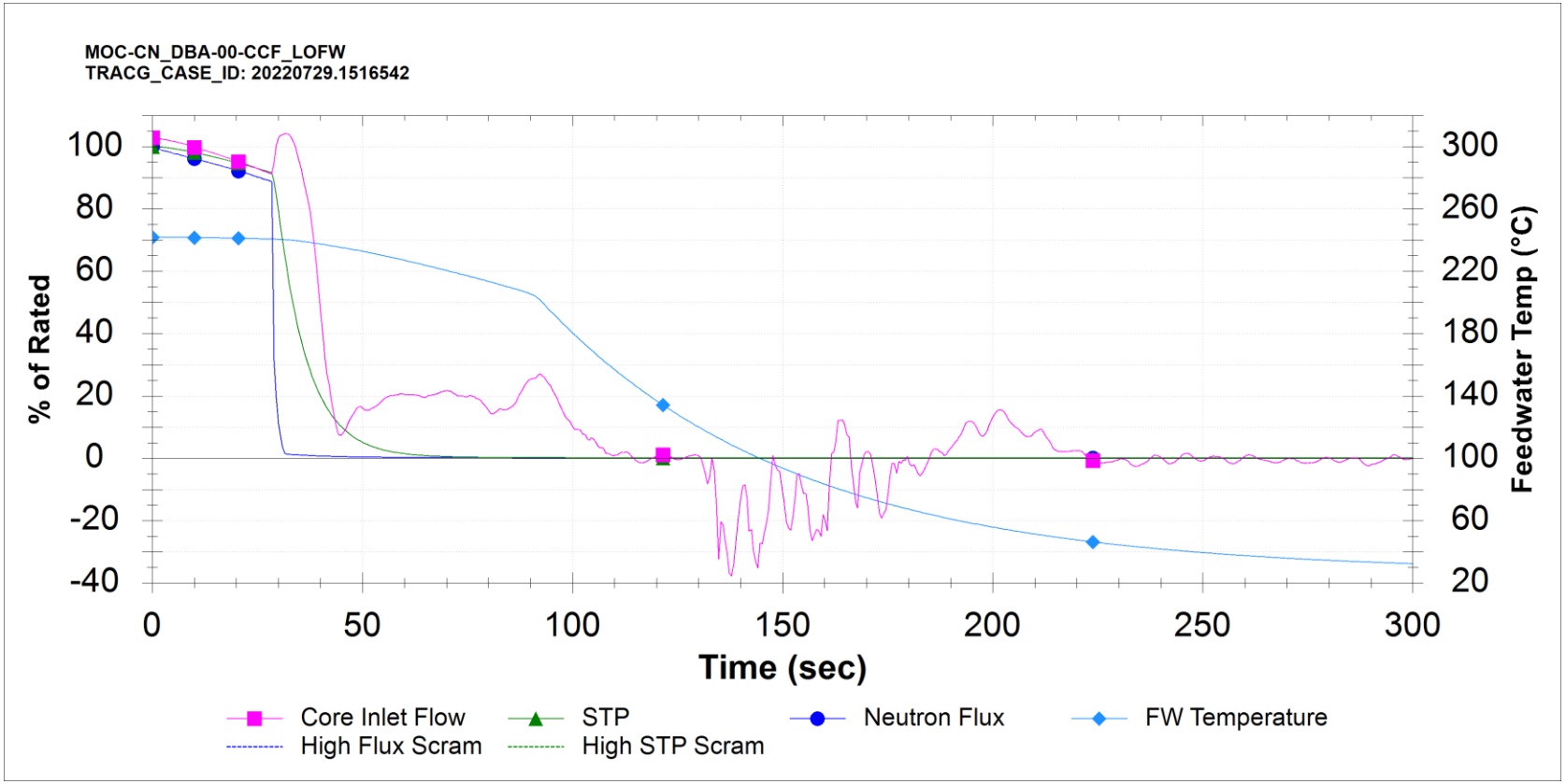


Figure 15.5-88: Loss of Feedwater Flow (DBA)

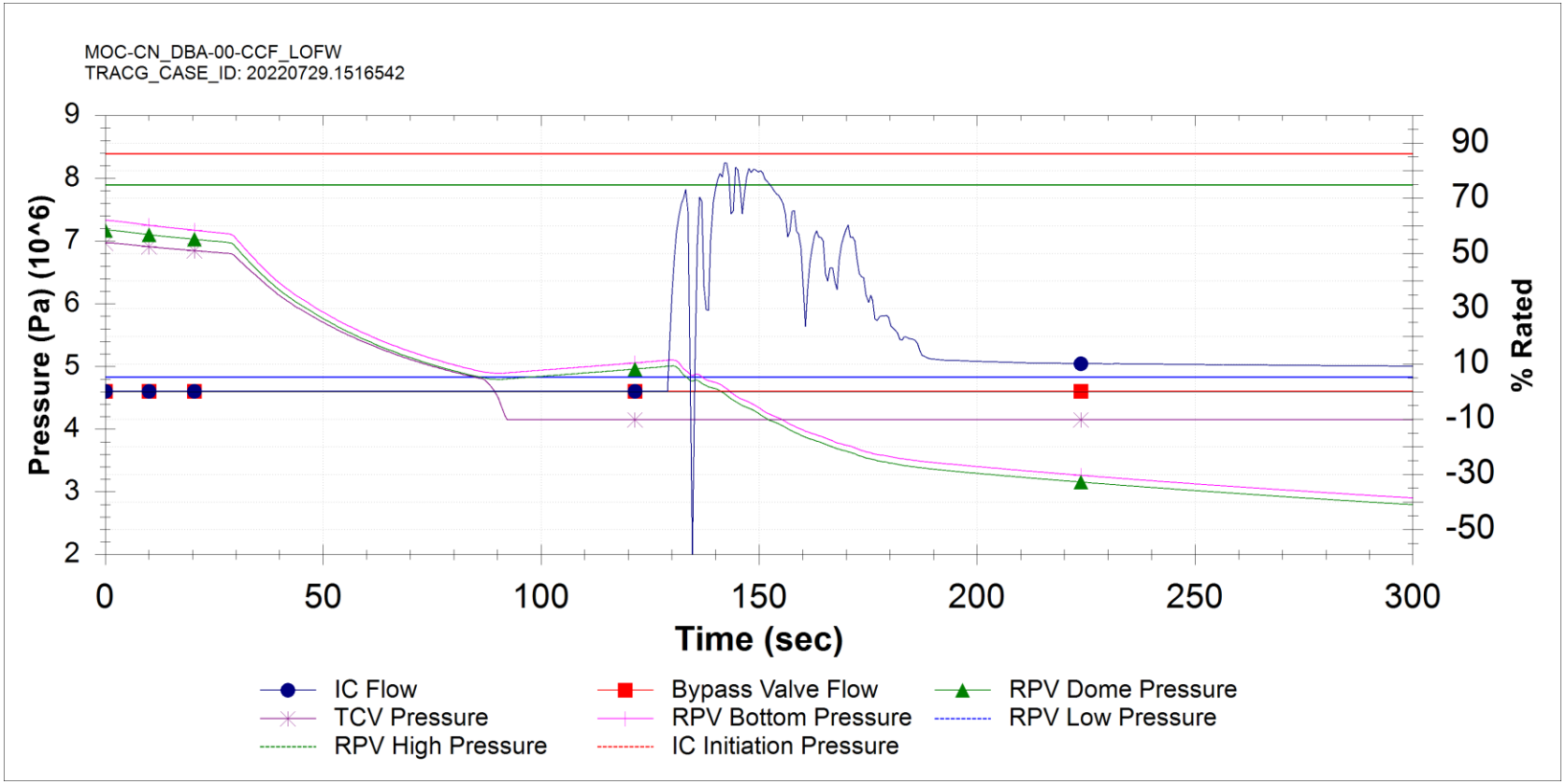


Figure 15.5-89: Loss of Feedwater Flow (DBA)

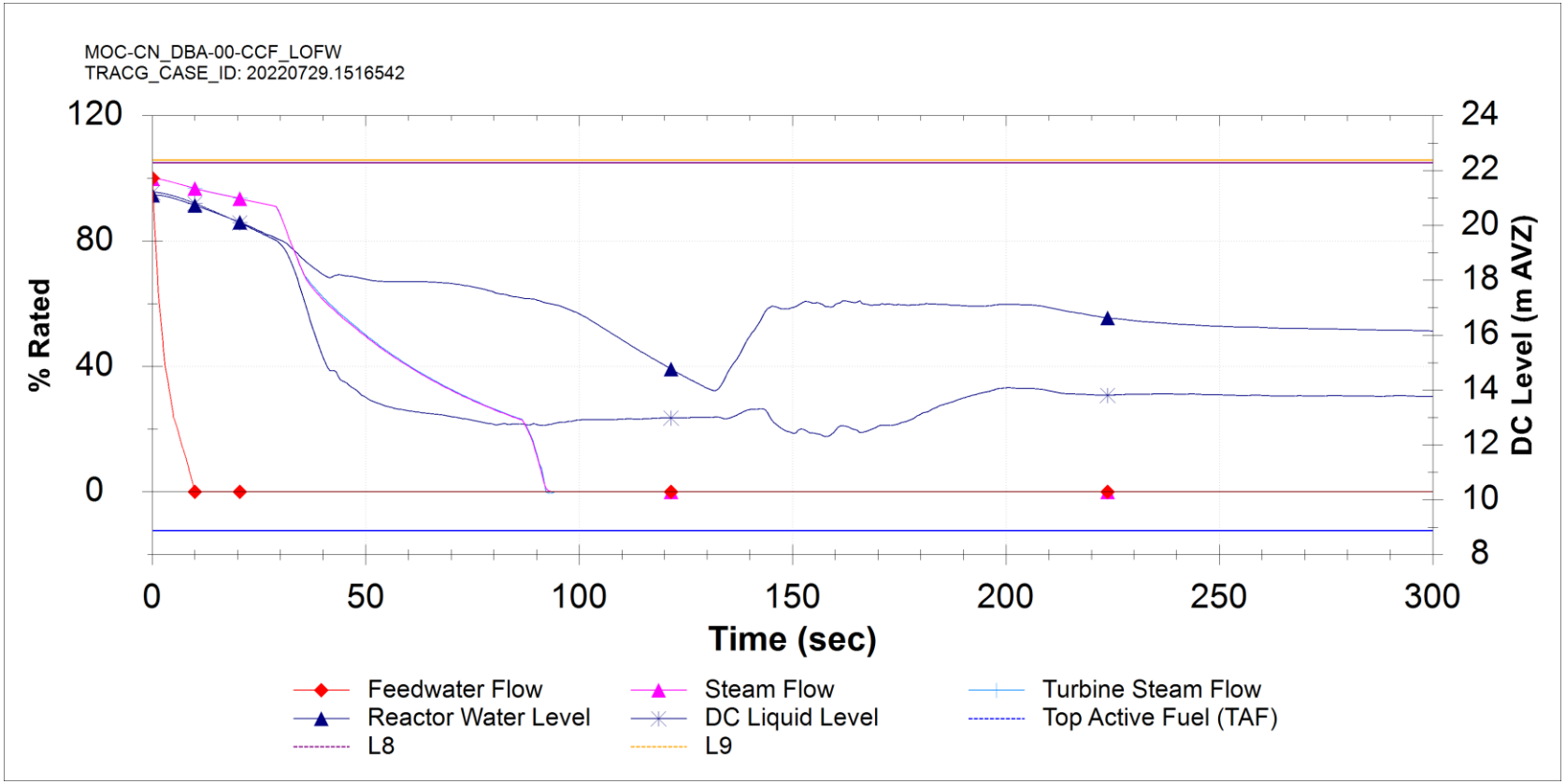


Figure 15.5-90: Loss of Feedwater Flow (DBA)

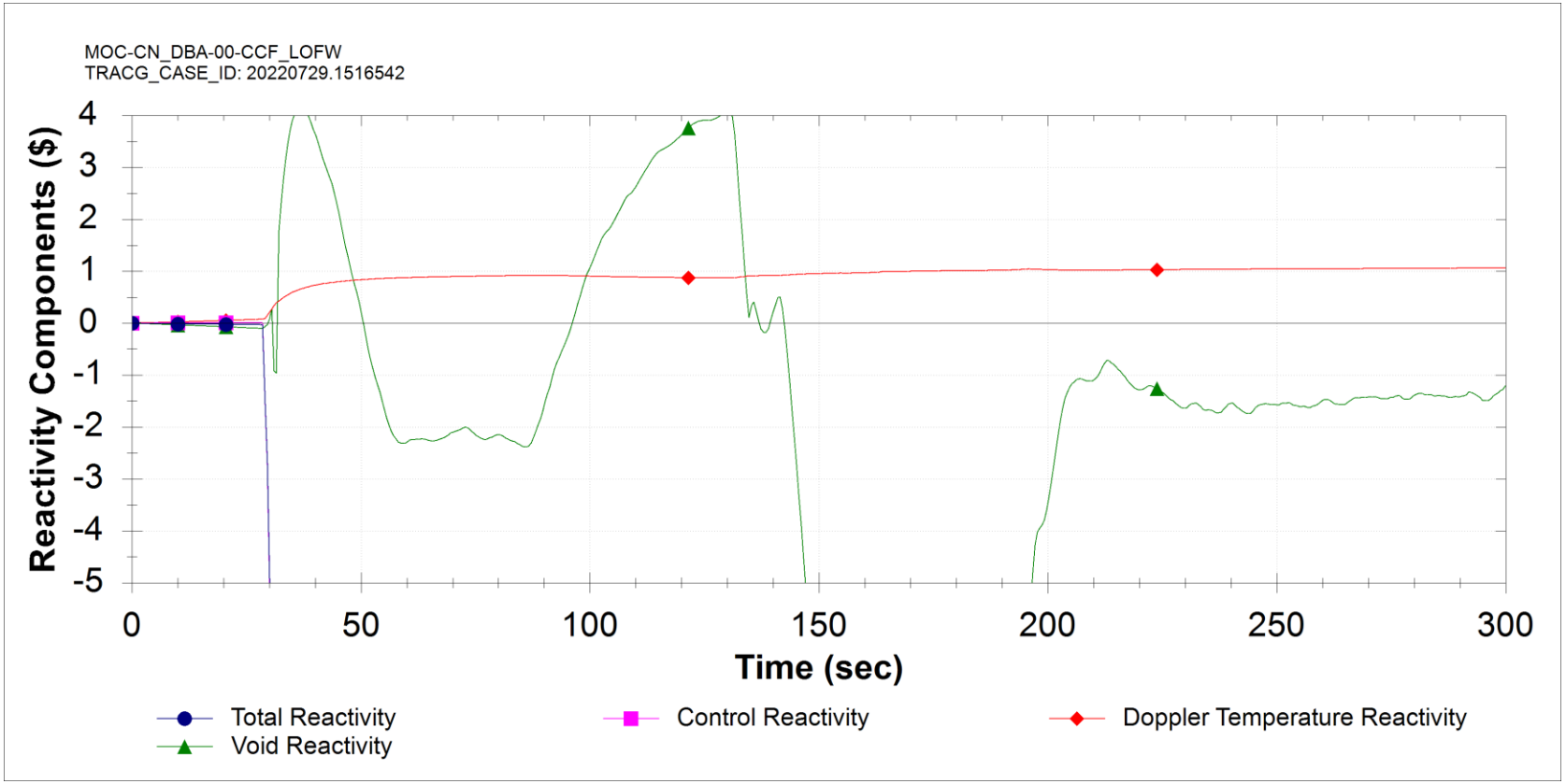


Figure 15.5-91: Loss of Feedwater Flow (DBA)

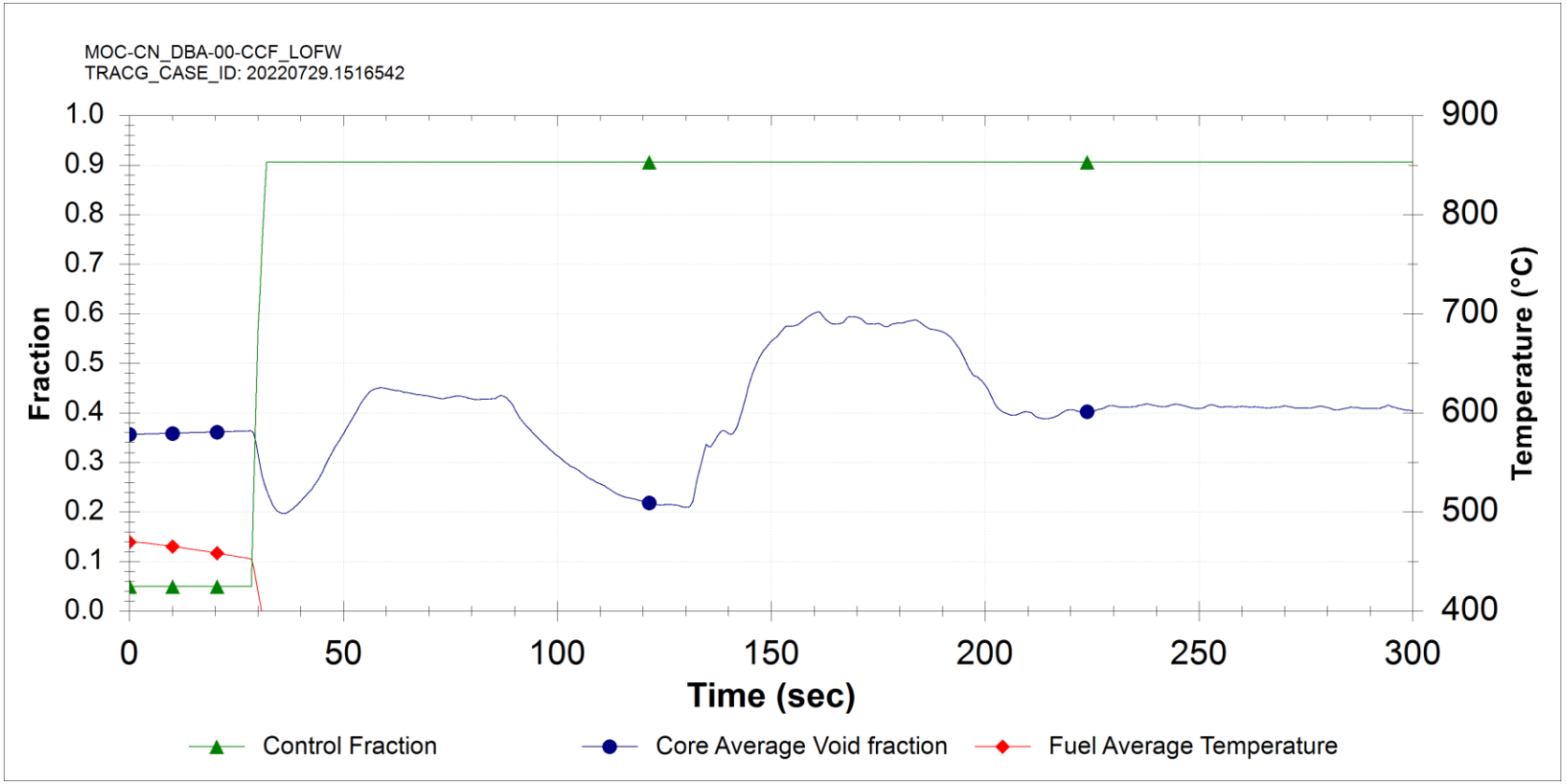


Figure 15.5-92: Loss of Feedwater Flow (DBA)

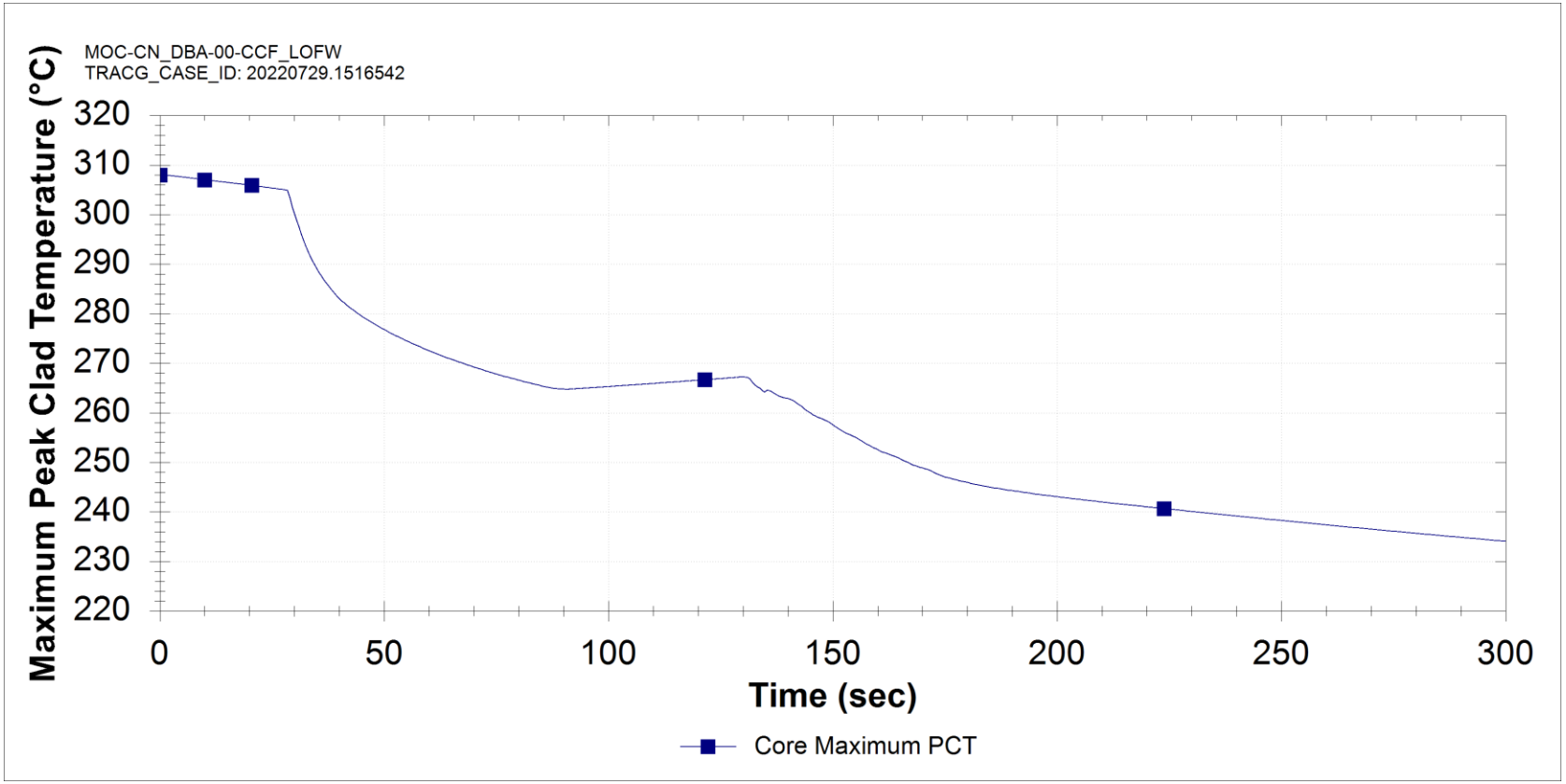


Figure 15.5-93: Loss of Feedwater Flow (DBA)

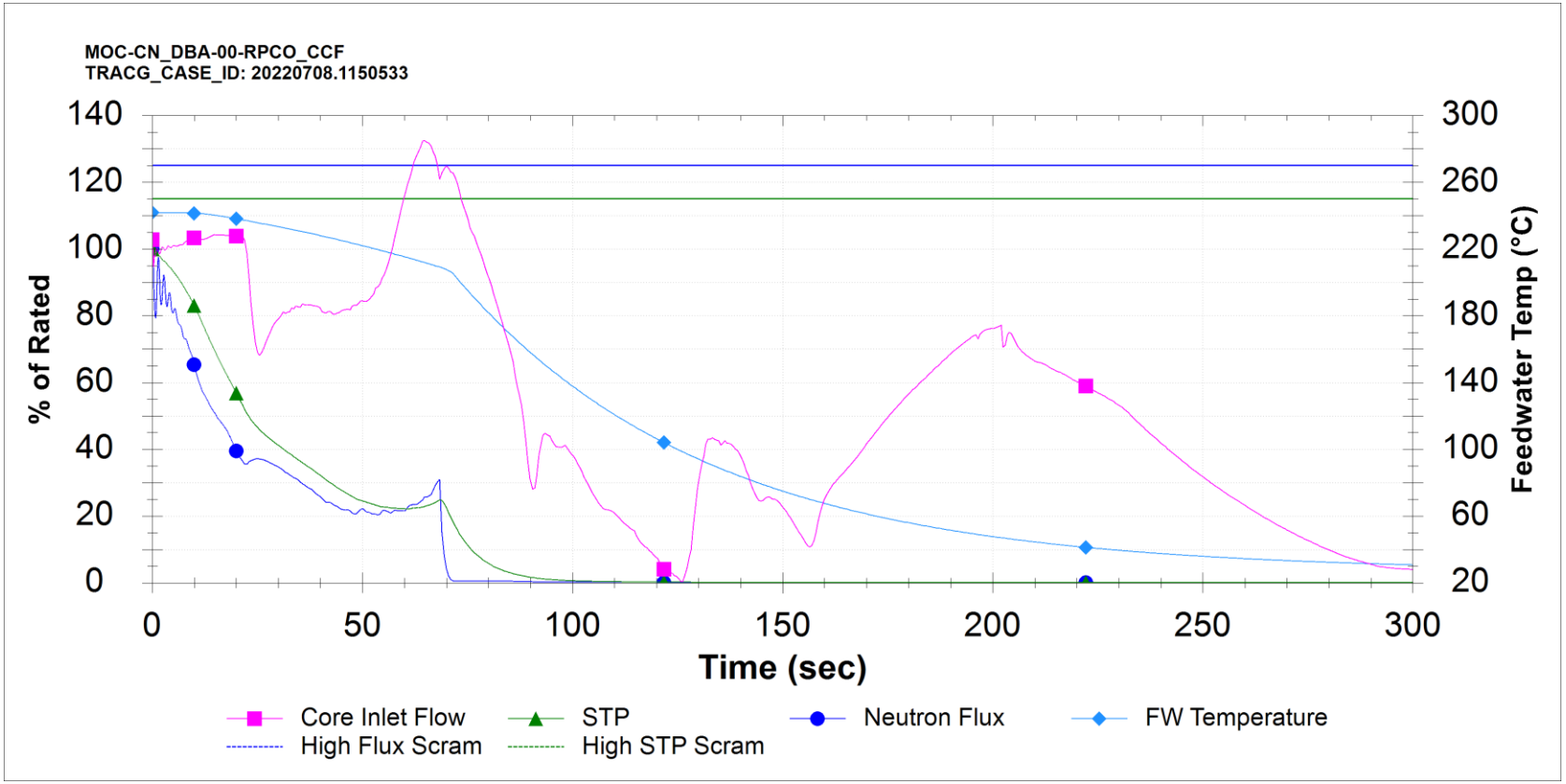


Figure 15.5-94: RPV Pressure Control Open (DBA)

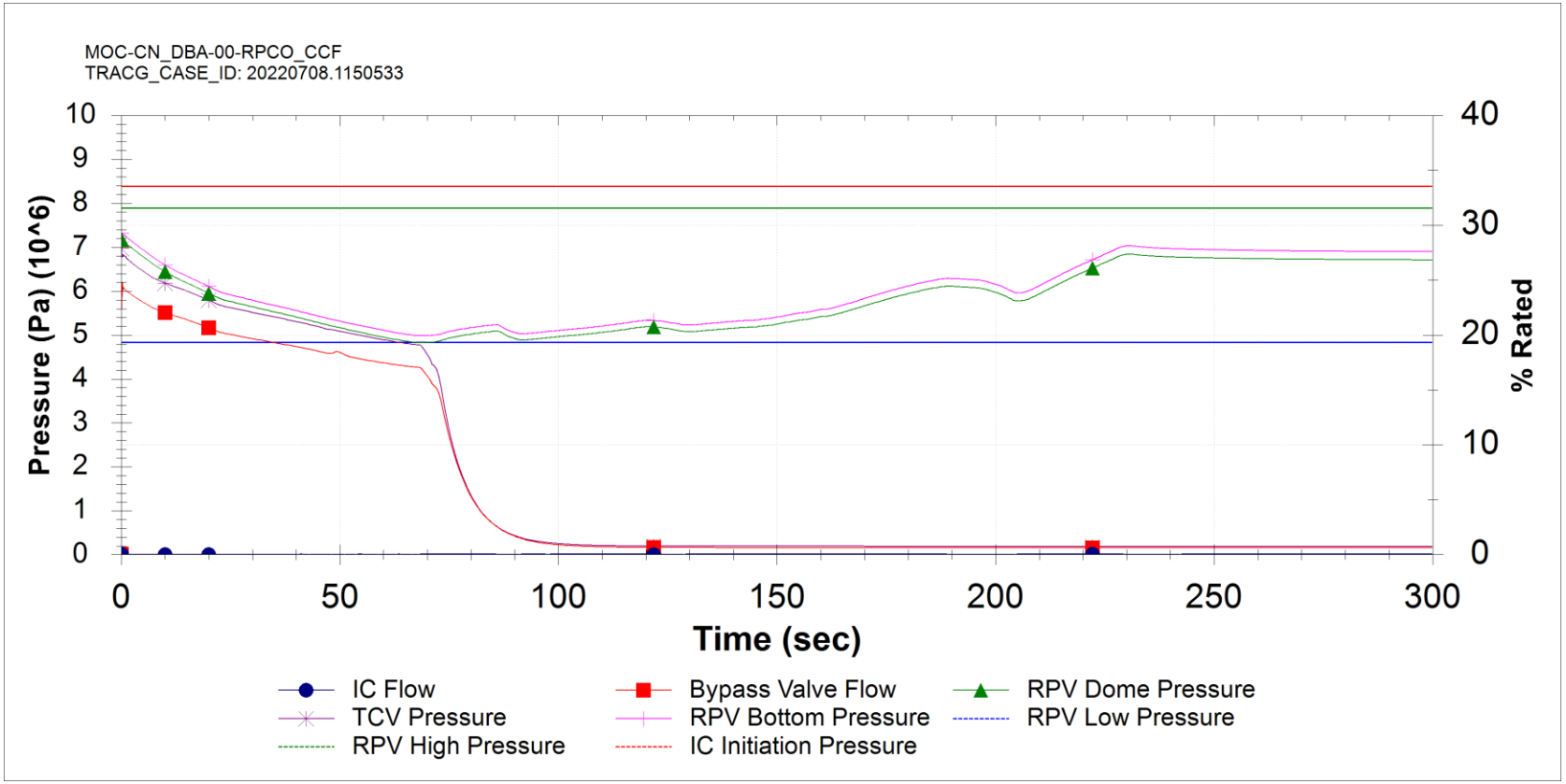


Figure 15.5-95: RPV Pressure Control Open (DBA)

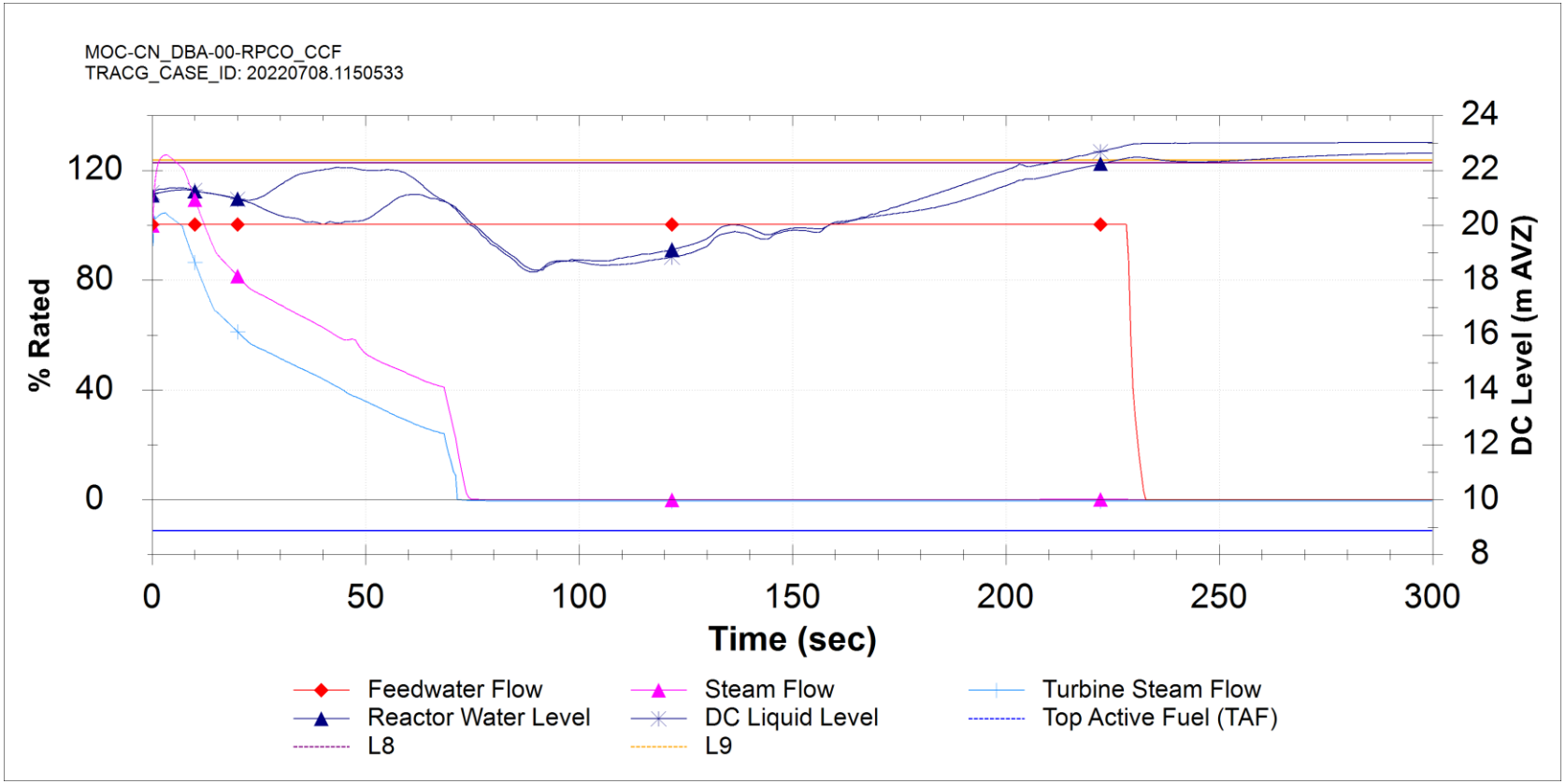


Figure 15.5-96: RPV Pressure Control Open (DBA)

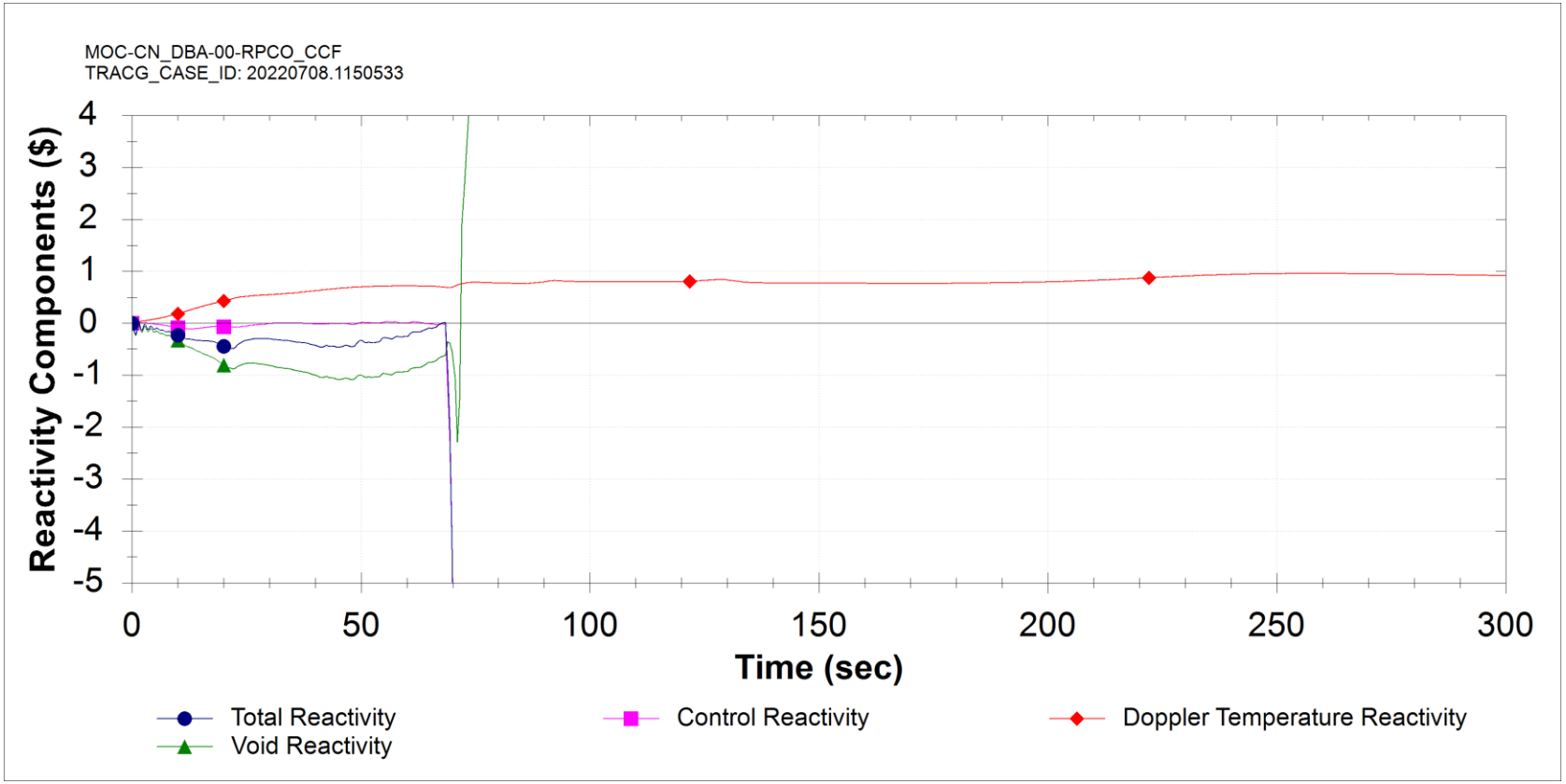


Figure 15.5-97: RPV Pressure Control Open (DBA)

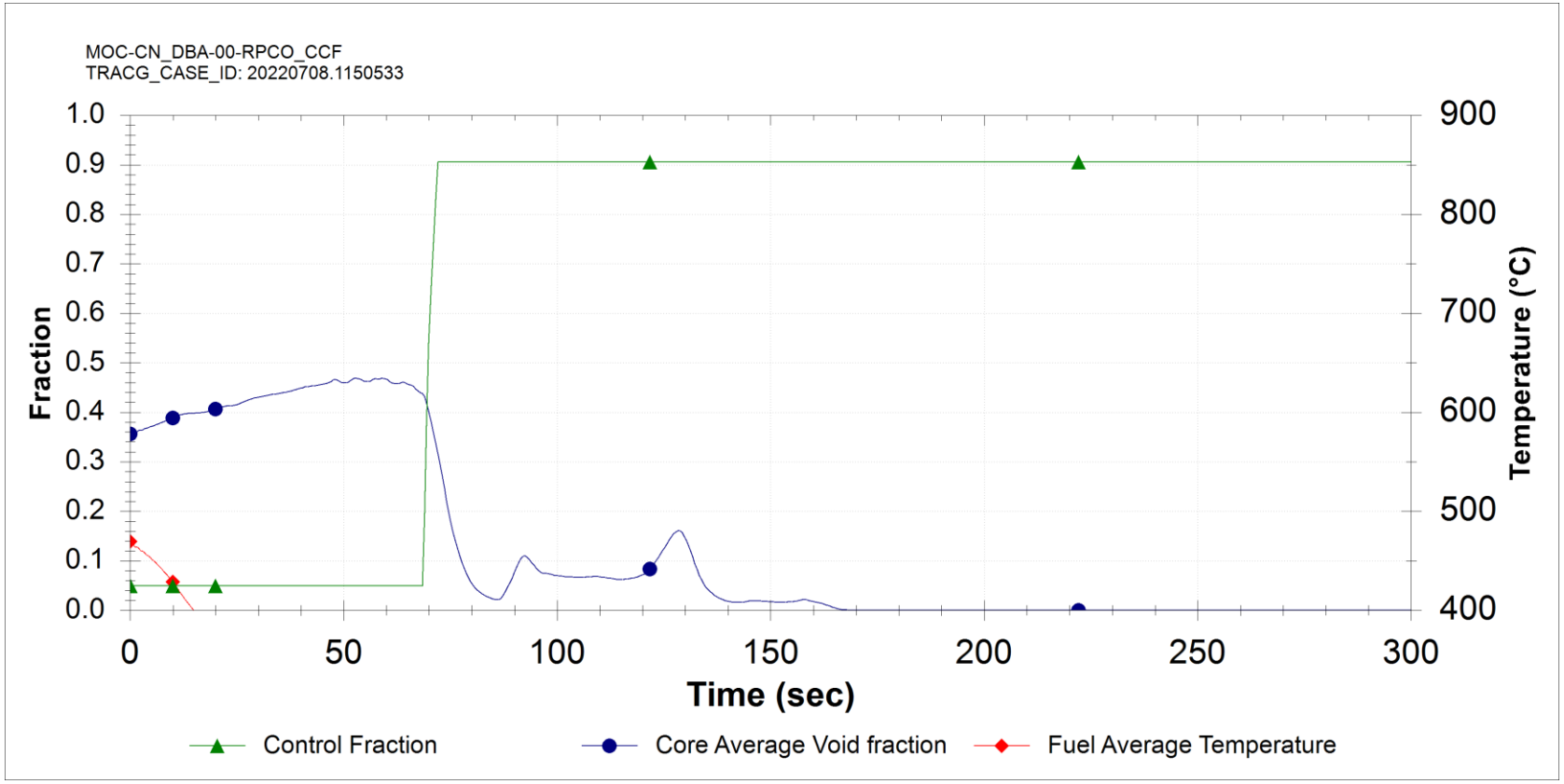


Figure 15.5-98: RPV Pressure Control Open (DBA)

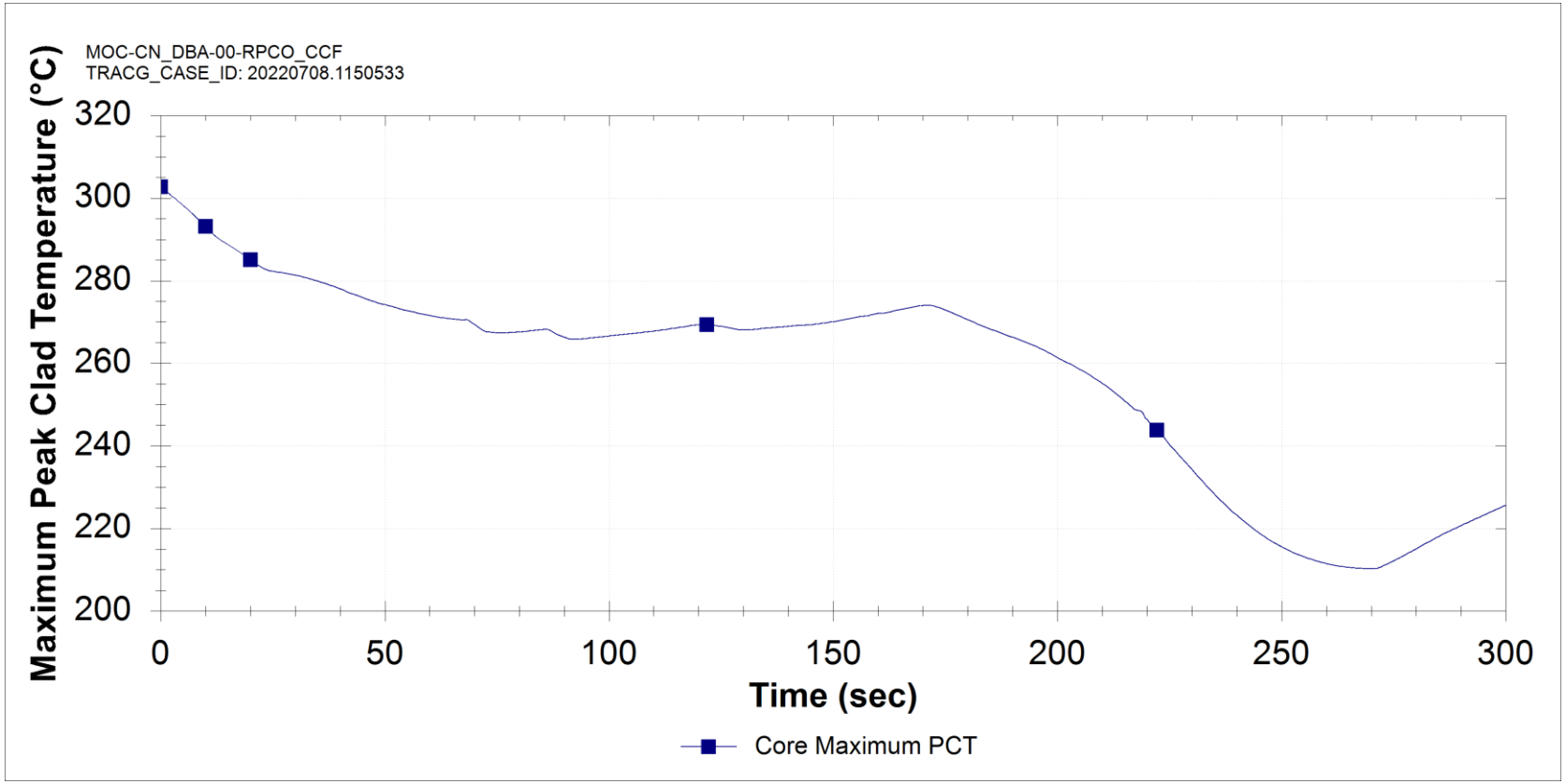


Figure 15.5-99: RPV Pressure Control Open (DBA)

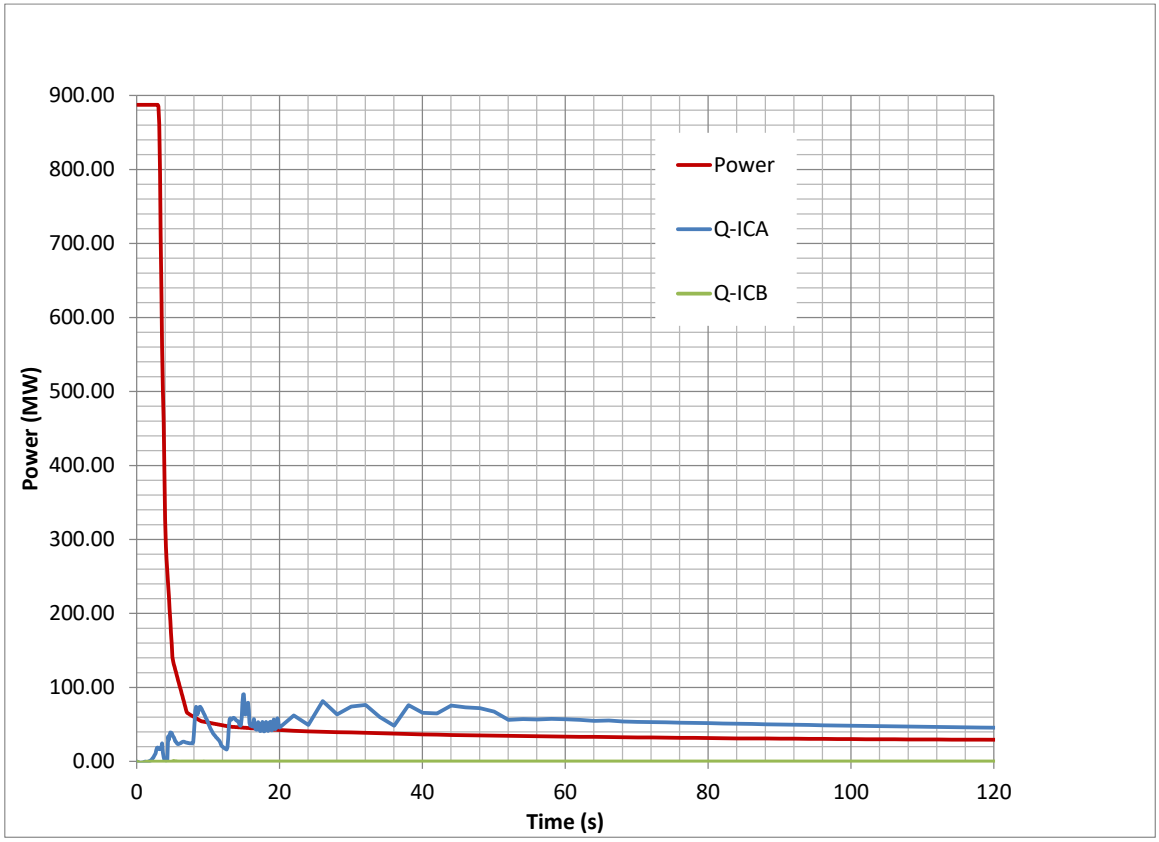


Figure 15.5-100: Reactor Power, Large Main Steam Pipe Break, Conservative Case

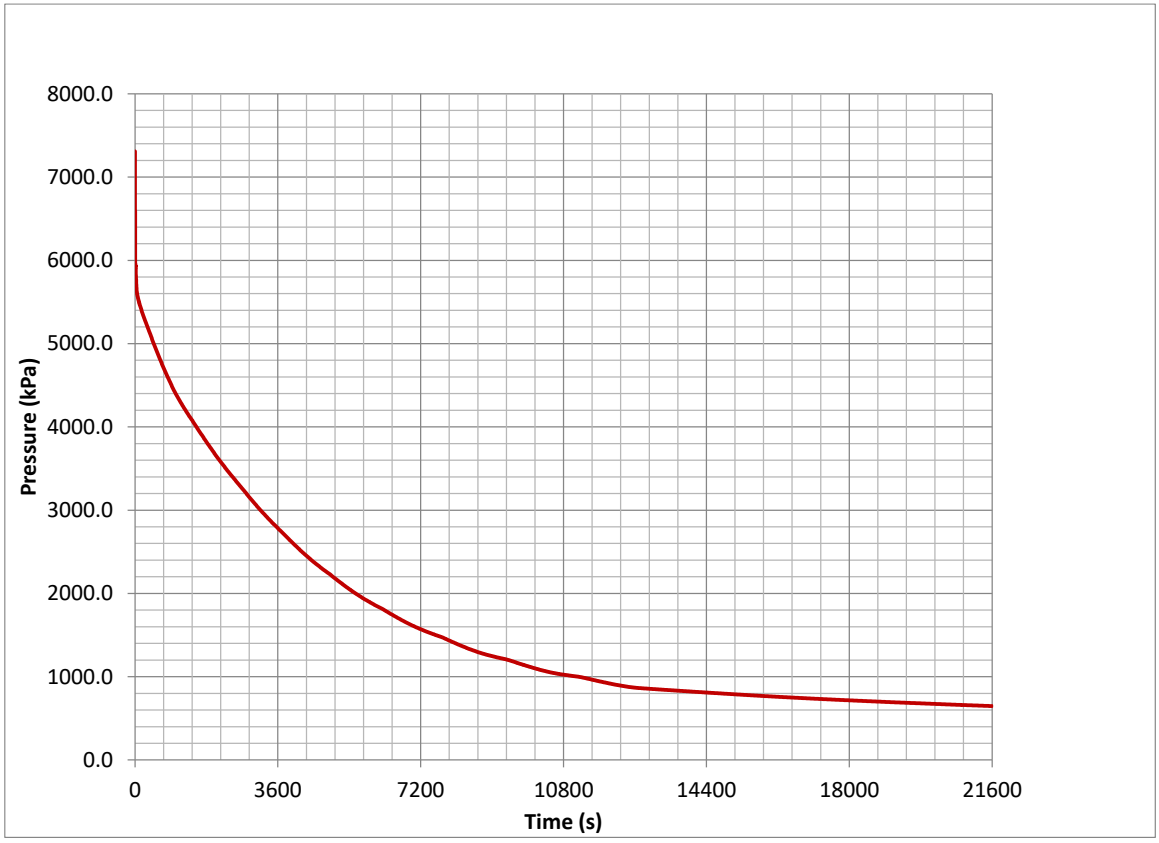


Figure 15.5-101: Reactor Pressure, Large Main Steam Pipe Break, Conservative Case

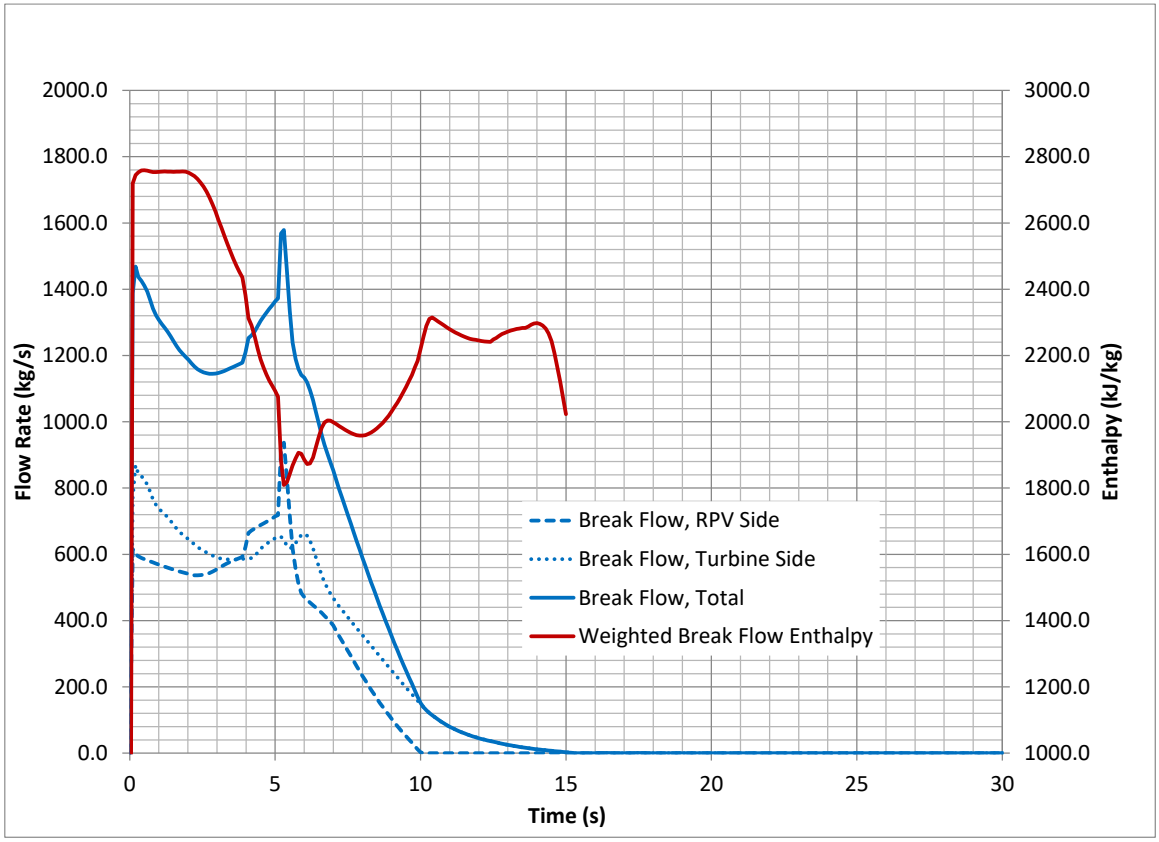


Figure 15.5-102: Break Flow Rate and Enthalpy, Large Main Steam Pipe Break, Conservative Case

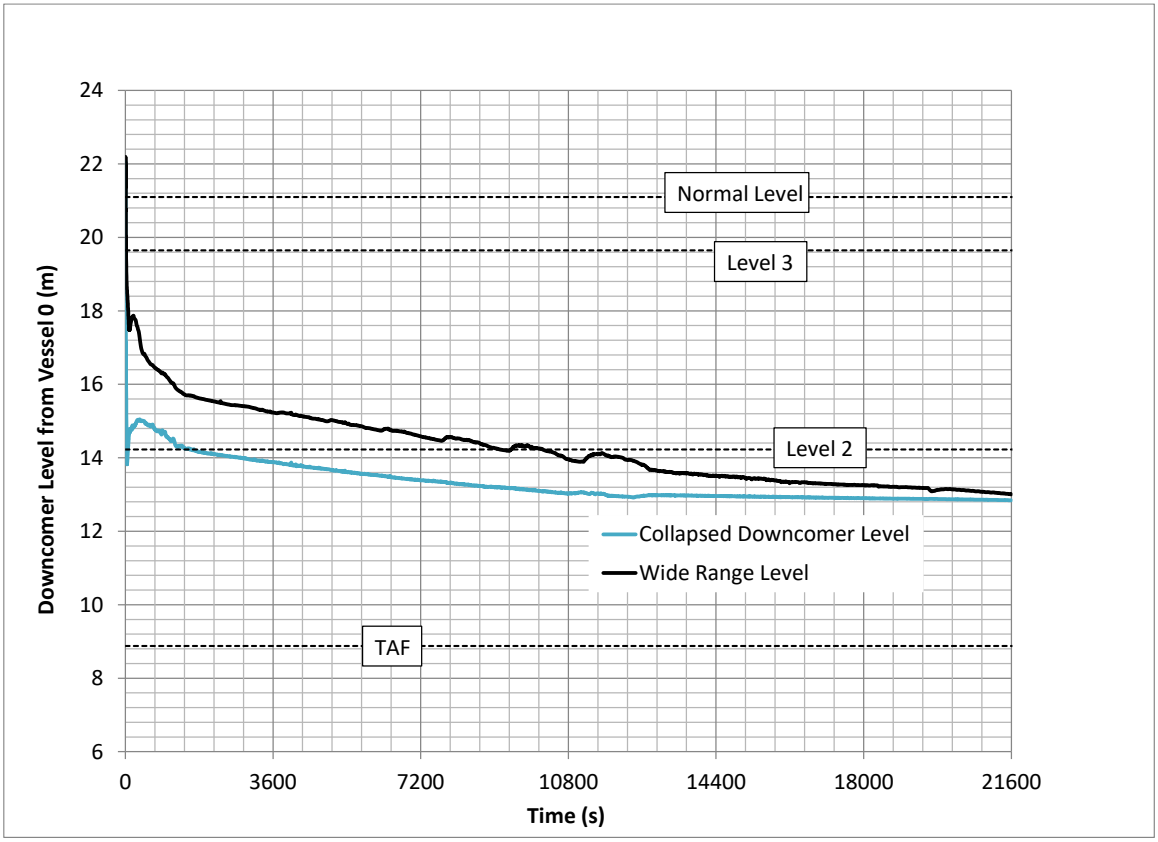


Figure 15.5-103: Reactor Water Level, Large Main Steam Pipe Break, Conservative Case

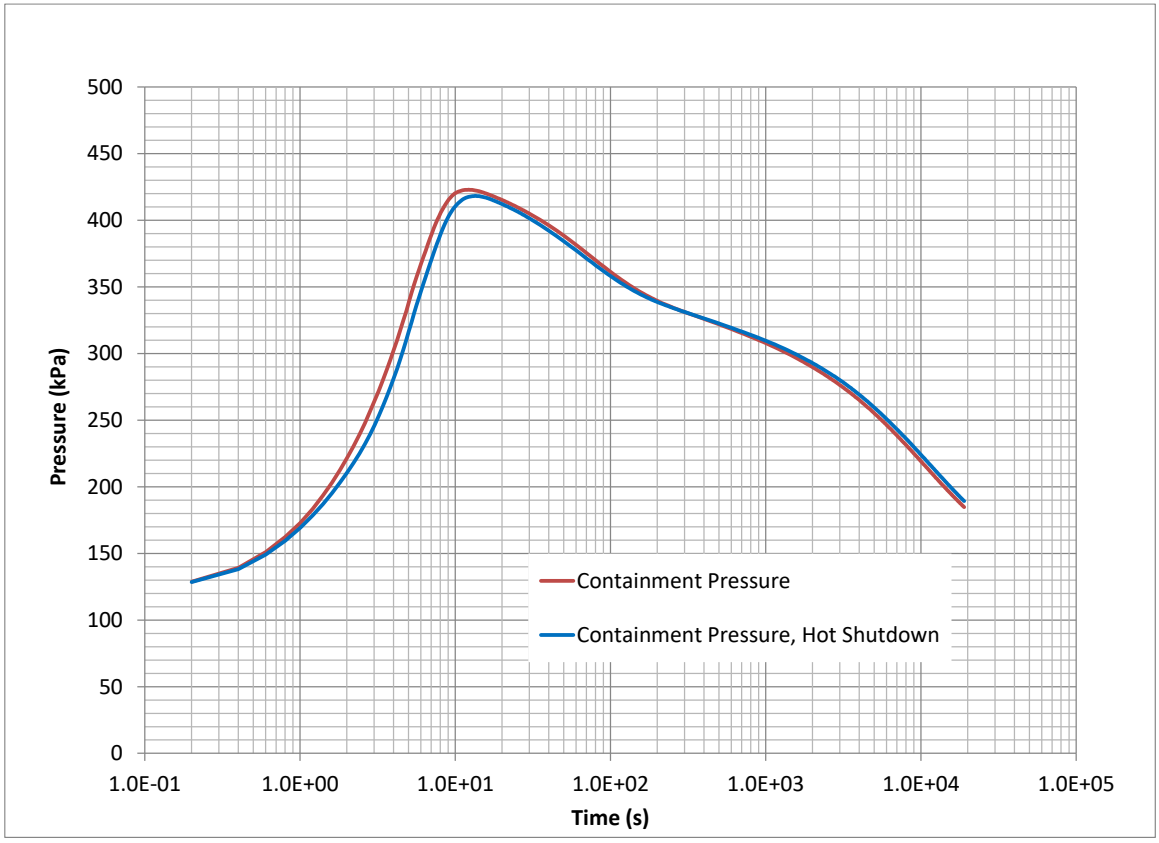


Figure 15.5-104: Containment Pressure, Large Main Steam Pipe Break, Conservative Case

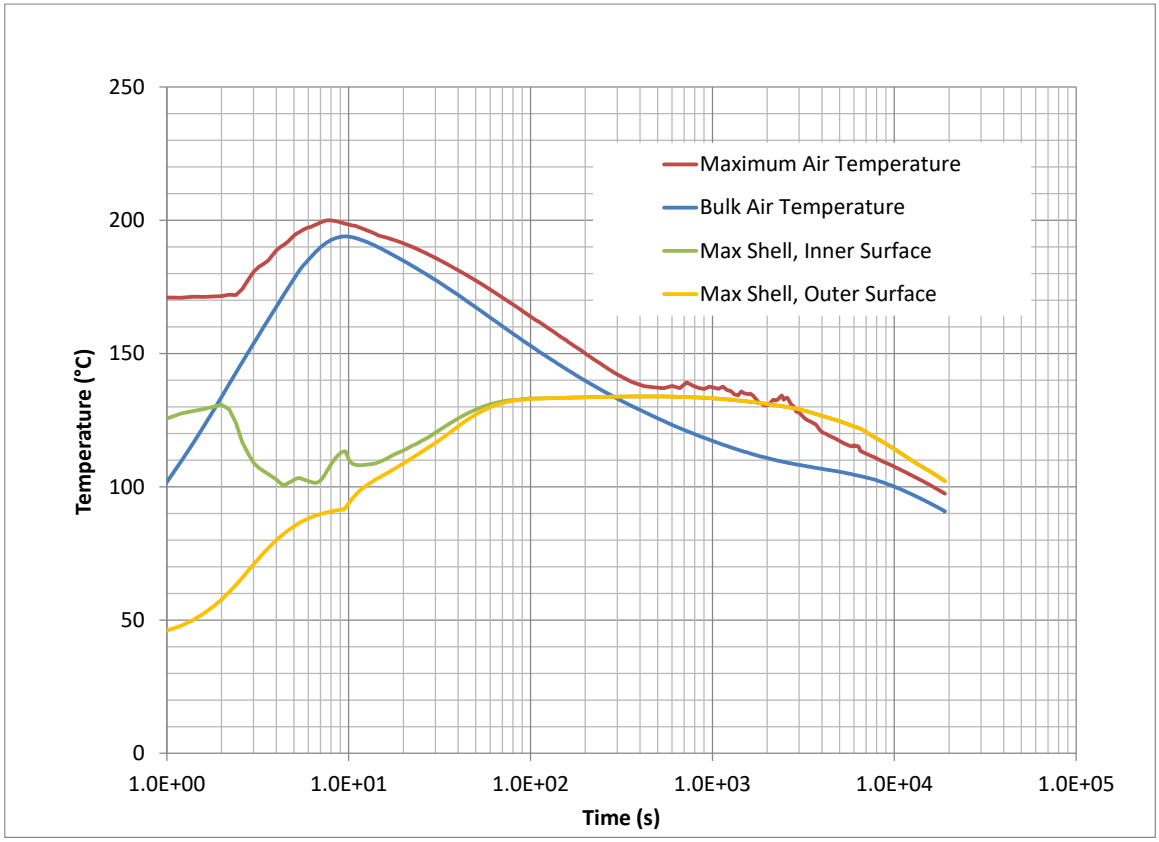


Figure 15.5-105: Containment Temperatures, Large Main Steam Pipe Break, Conservative Case

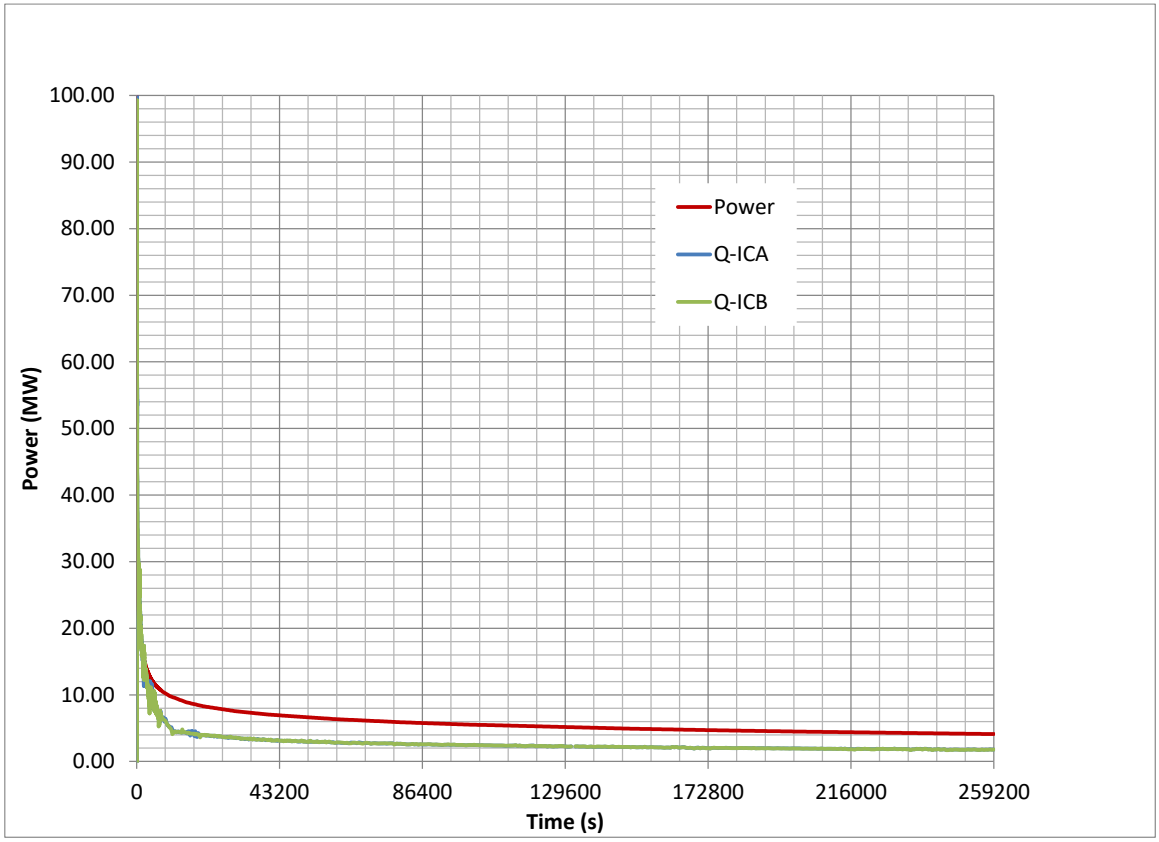


Figure 15.5-106: Reactor Power, Small Steam Break With LOPP, Conservative Case

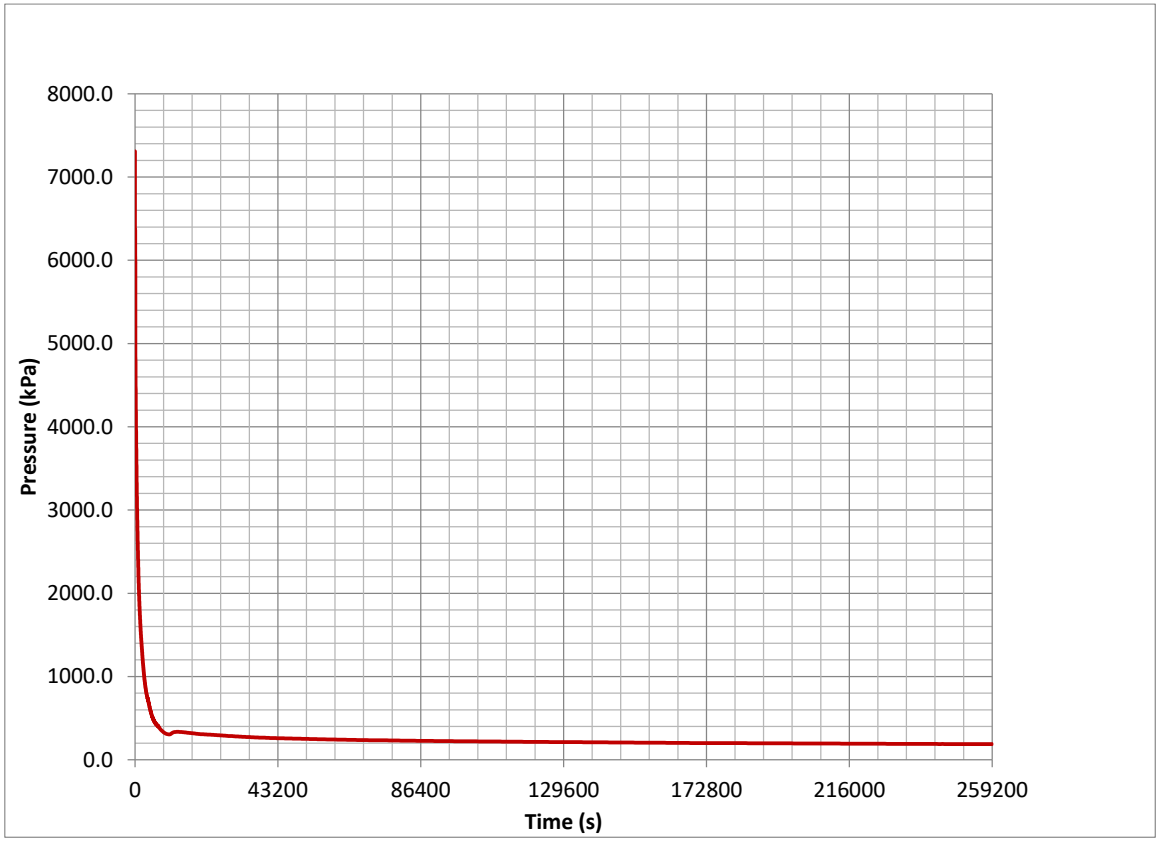


Figure 15.5-107: Reactor Pressure, Small Steam Pipe Break With LOPP, Conservative Case

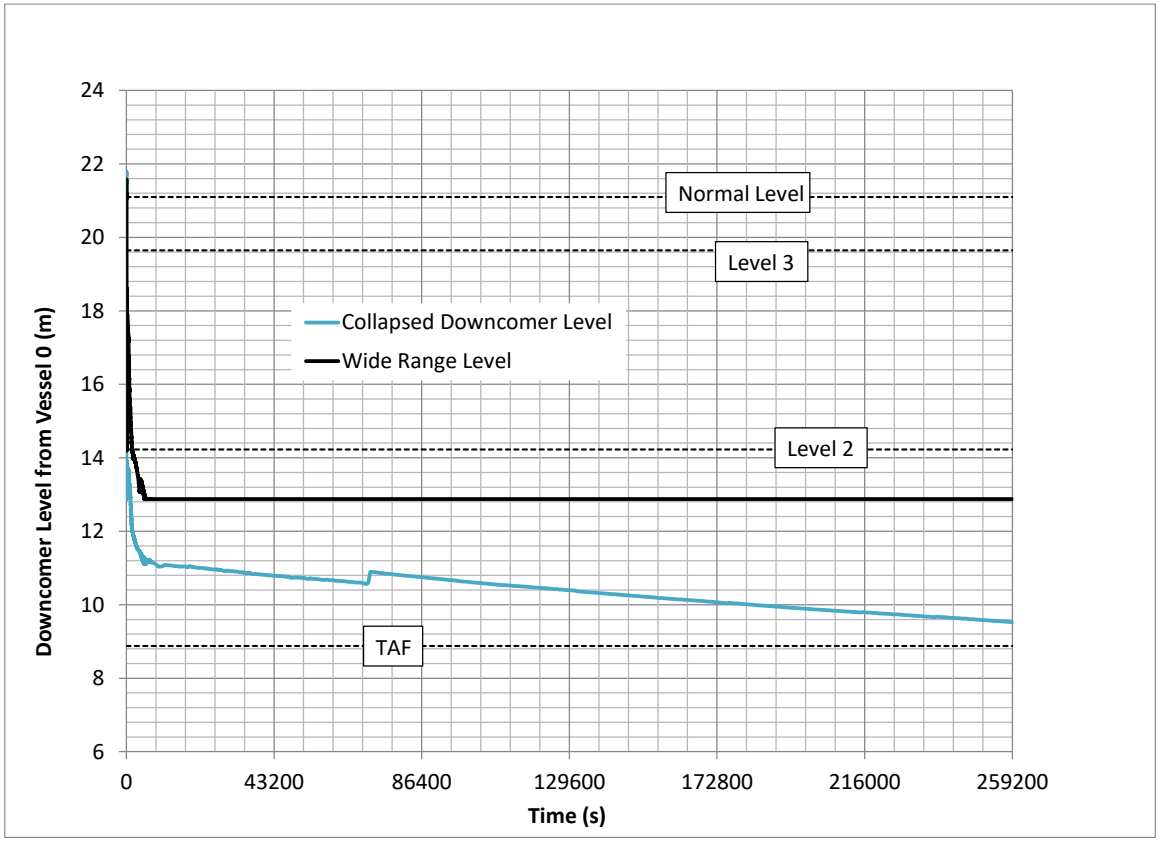


Figure 15.5-108: Reactor Water Level, Small Steam Pipe Break With LOPP, Conservative Case

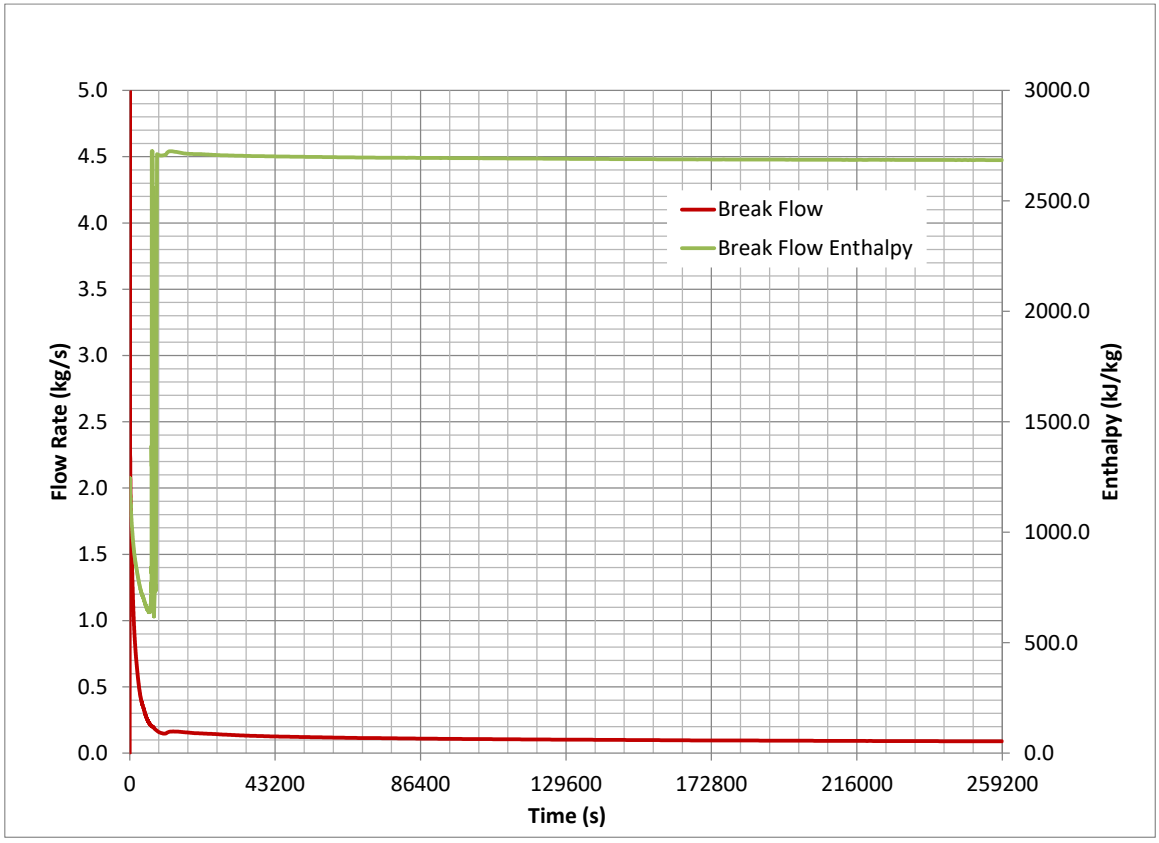


Figure 15.5-109: Break Flow Rate and Enthalpy, Small Steam Pipe Break With LOPP, Conservative Case

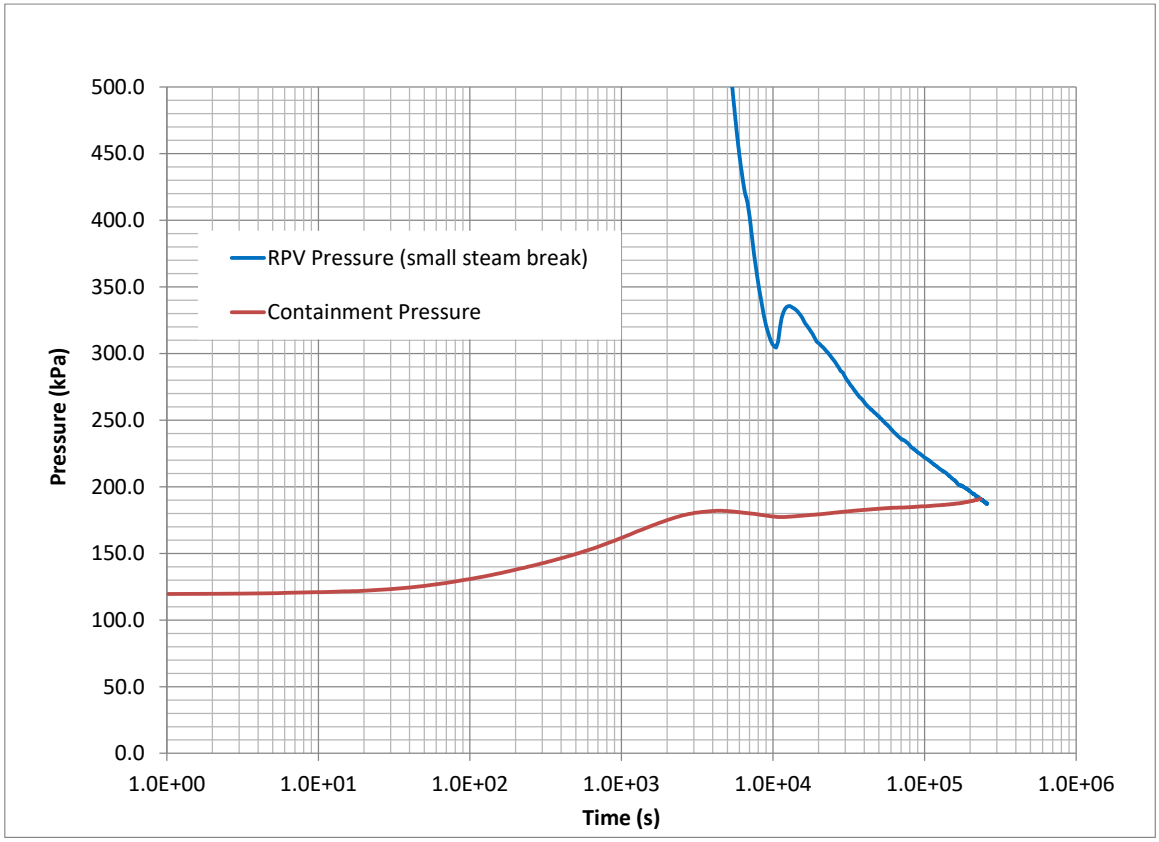


Figure 15.5-110: Containment Pressure, Small Steam Pipe Break With LOPP, Conservative Case

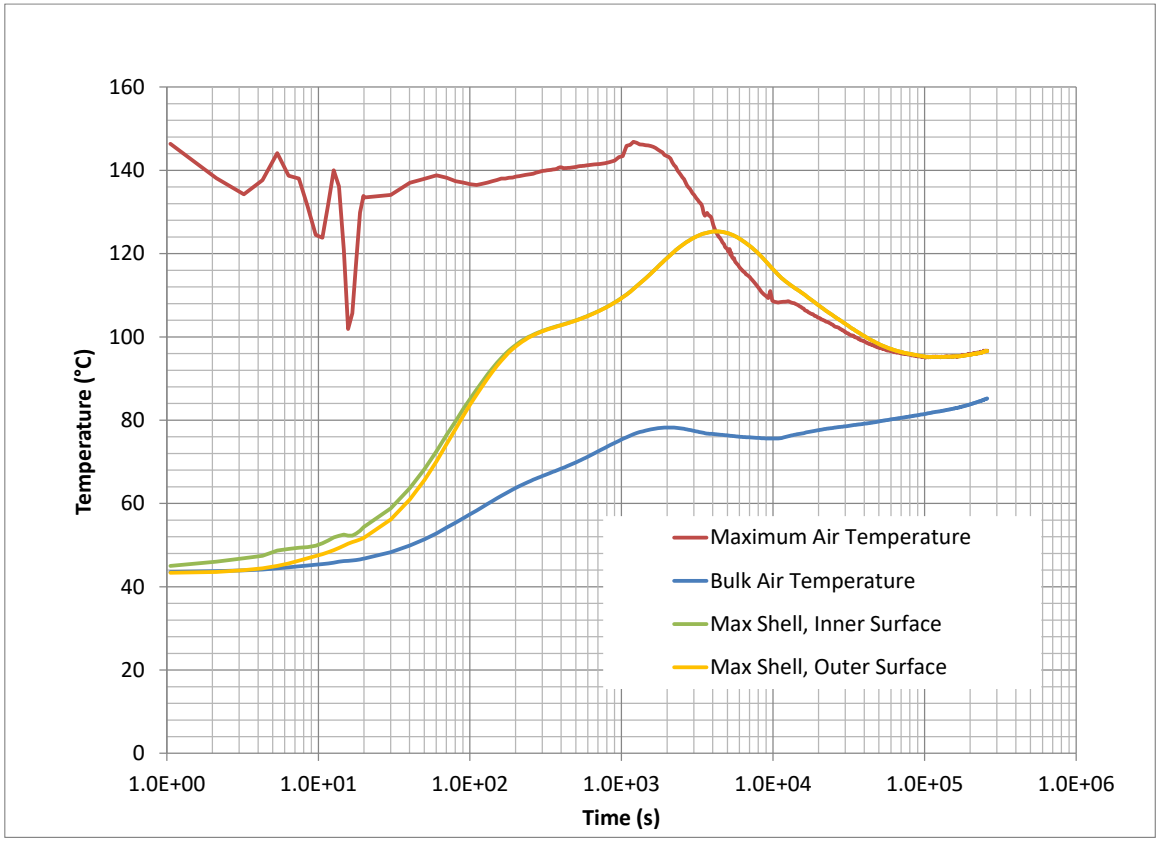


Figure 15.5-111: Containment Temperature, Small Steam Pipe Break with LOPP, Conservative Case

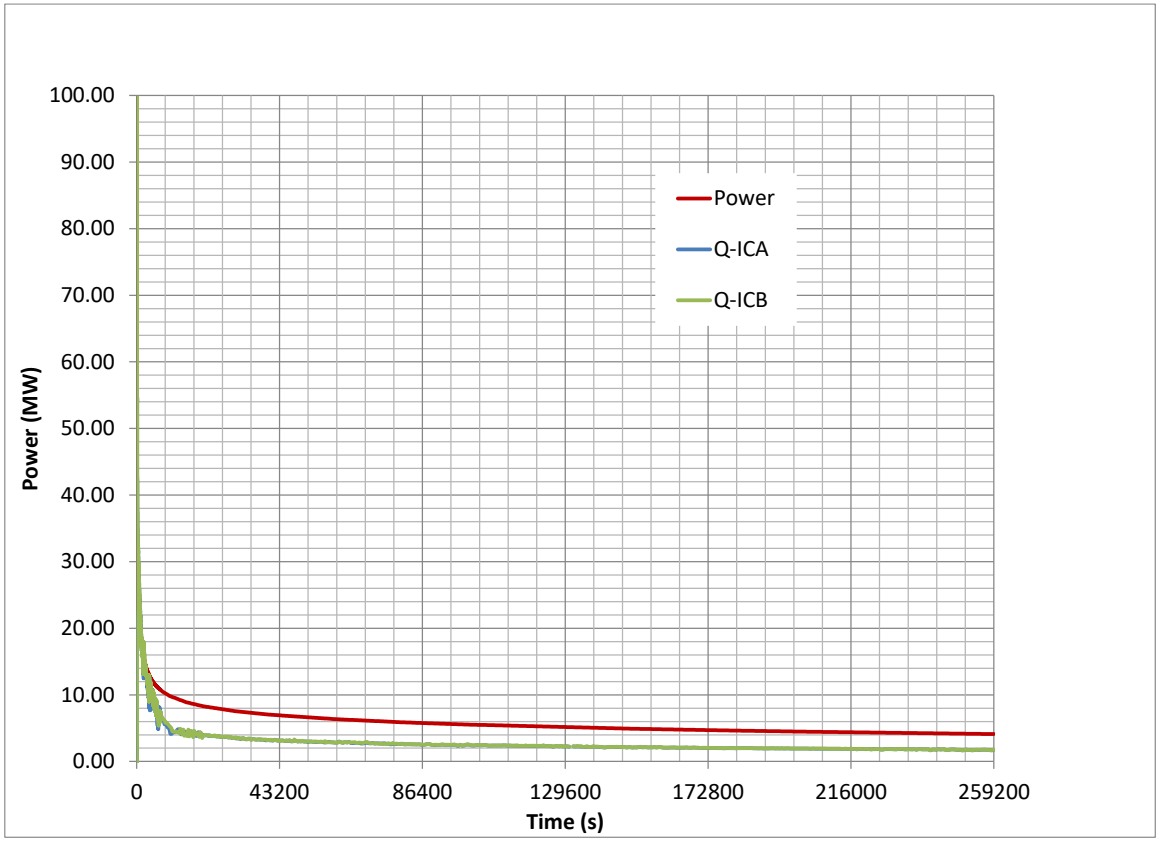


Figure 15.5-112: Reactor Power, Small Liquid Pipe Break, Conservative Case

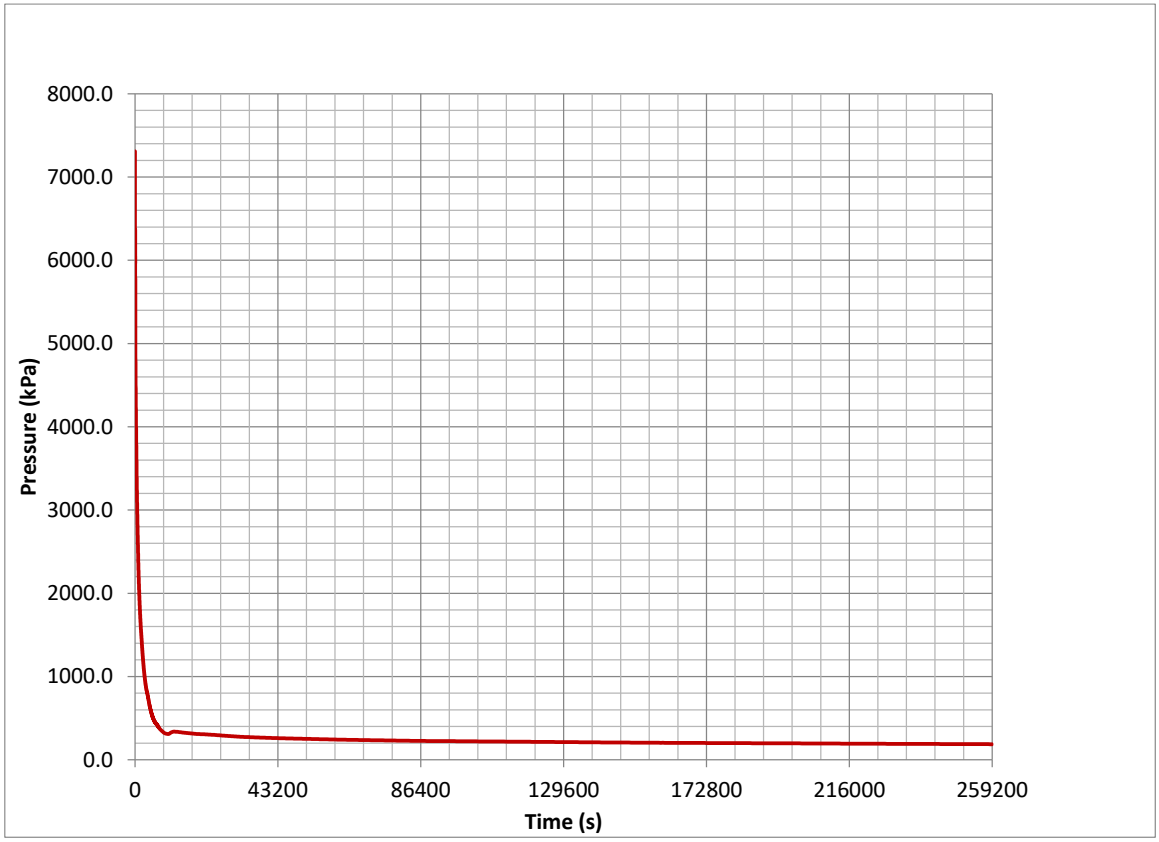


Figure 15.5-113: Reactor Pressure, Small Liquid Pipe Break, Conservative Case

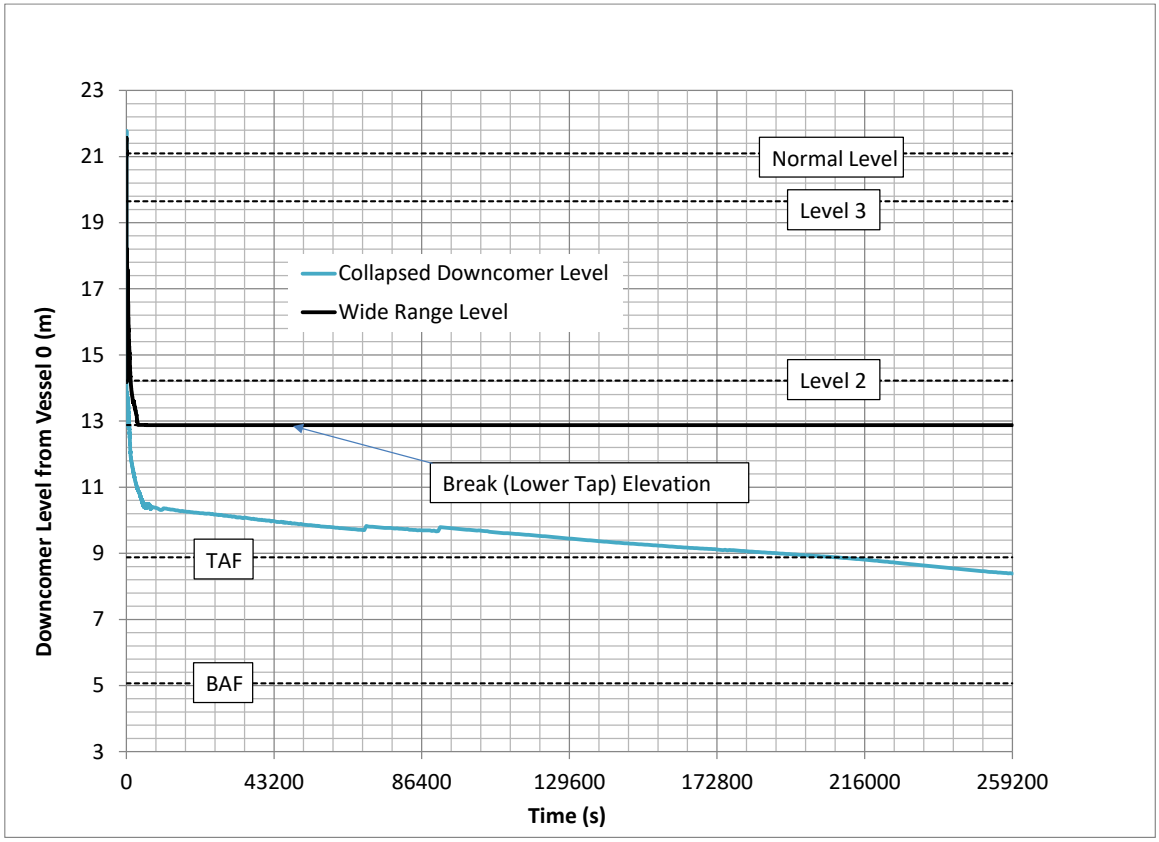


Figure 15.5-114: Reactor Water Level, Small Liquid Break, Conservative Case

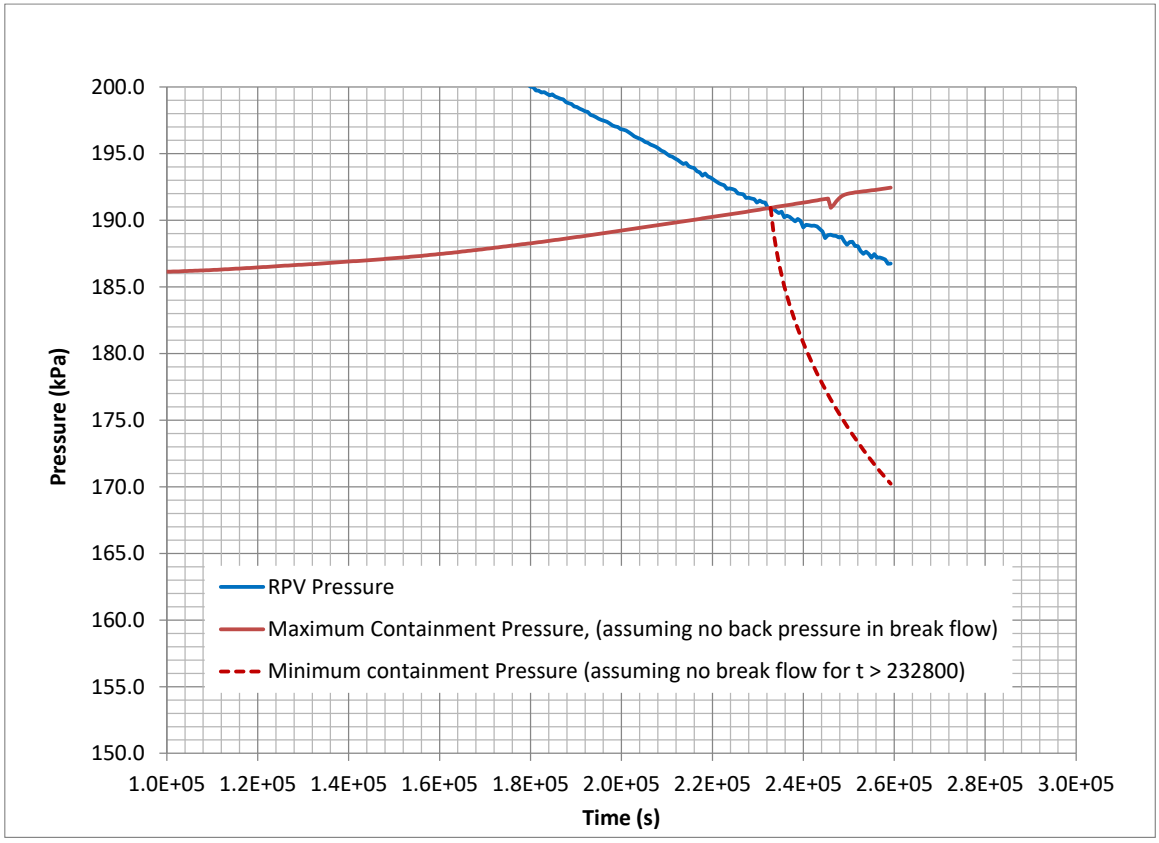


Figure 15.5-115: Containment Pressure After RPV Depressurization, Small Liquid Pipe Break, Conservative Case

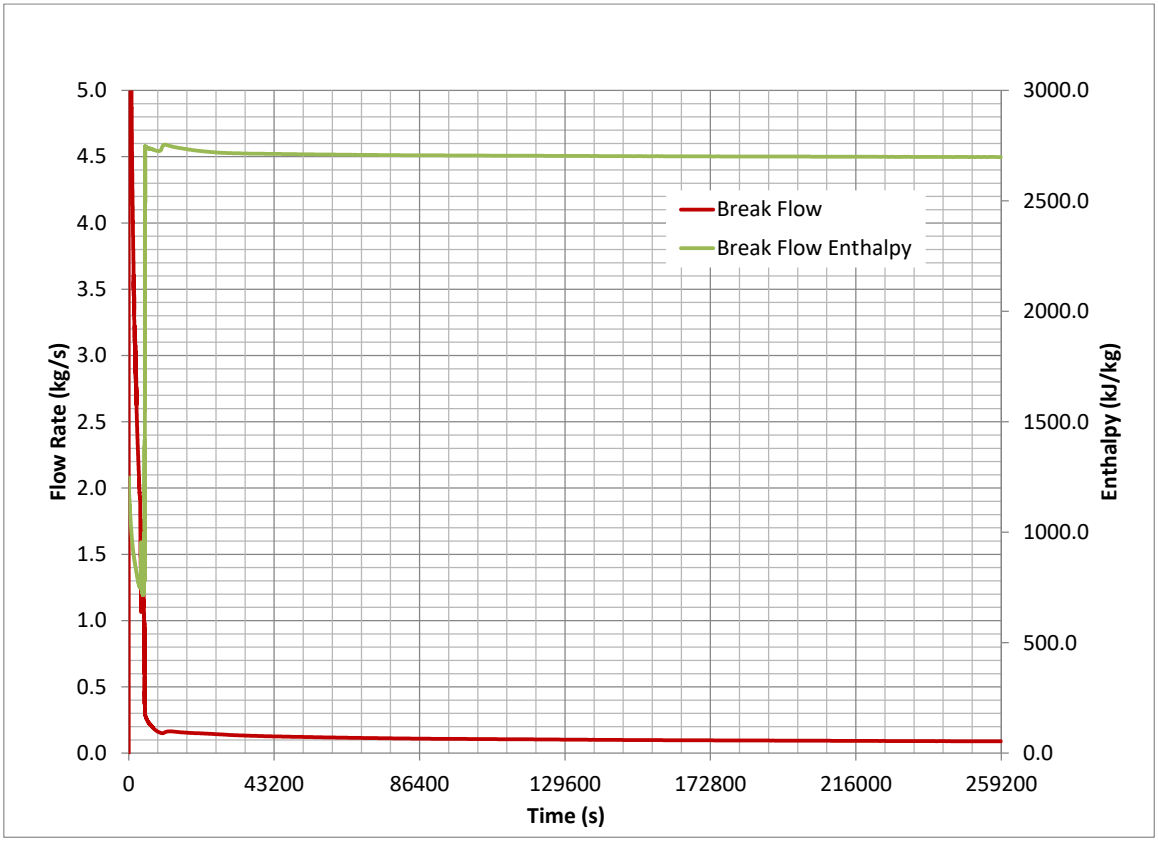


Figure 15.5-116: Break Flow Rate and Enthalpy, Small Liquid Pipe Break, Conservative Case

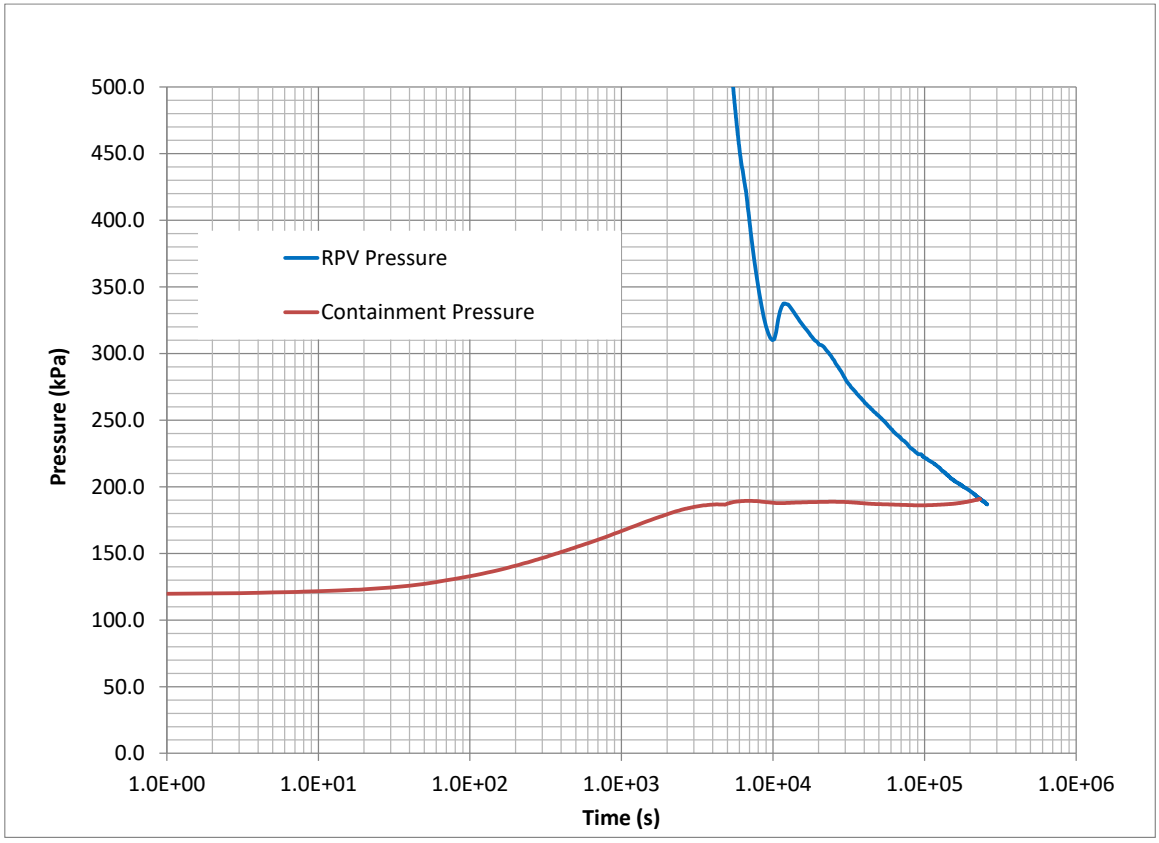


Figure 15.5-117: Containment Pressure, Small Liquid Pipe Break, Conservative Case

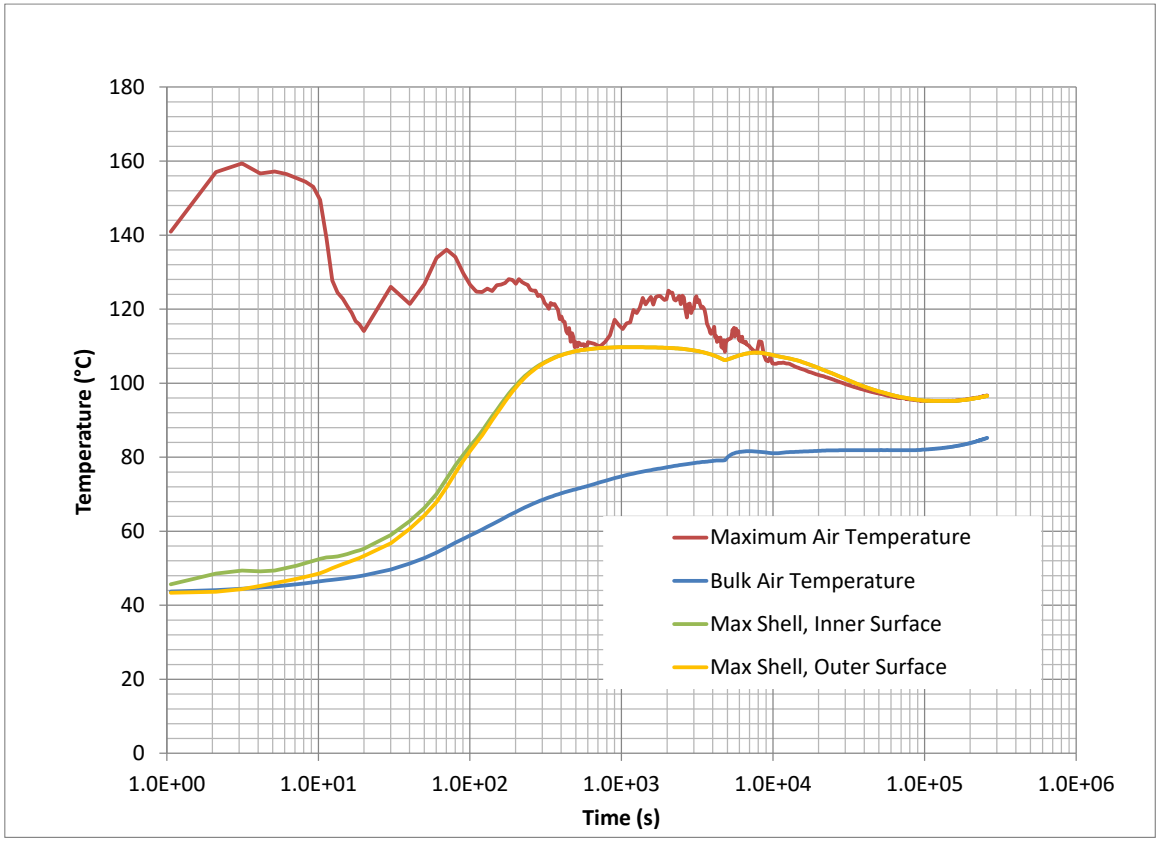


Figure 15.5-118: Containment Temperature, Small Liquid Pipe Break, Conservative Case

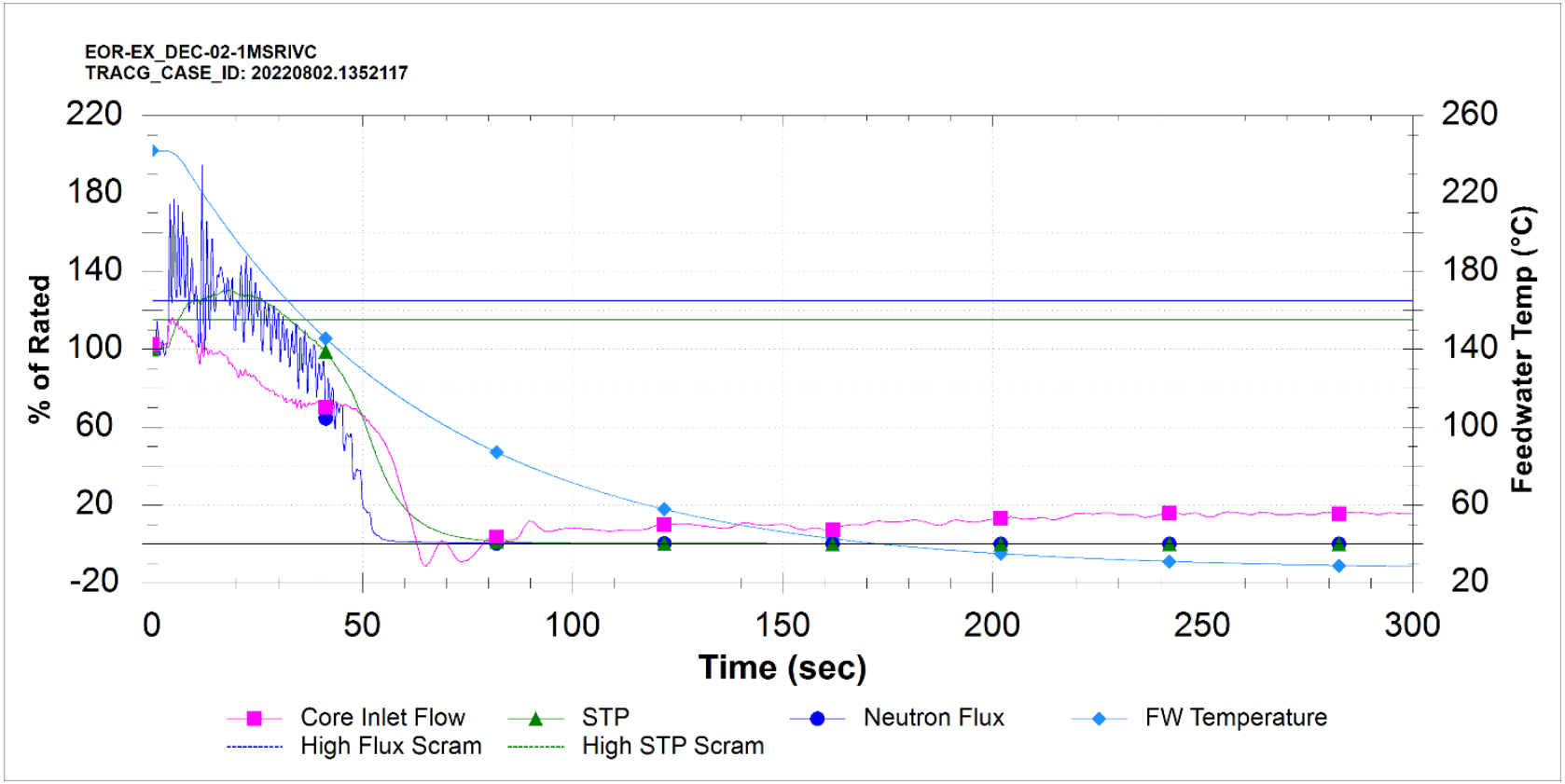


Figure 15.5-119: Closure of One Main Steam Reactor Isolation Valve with Failure to Scram (DEC)

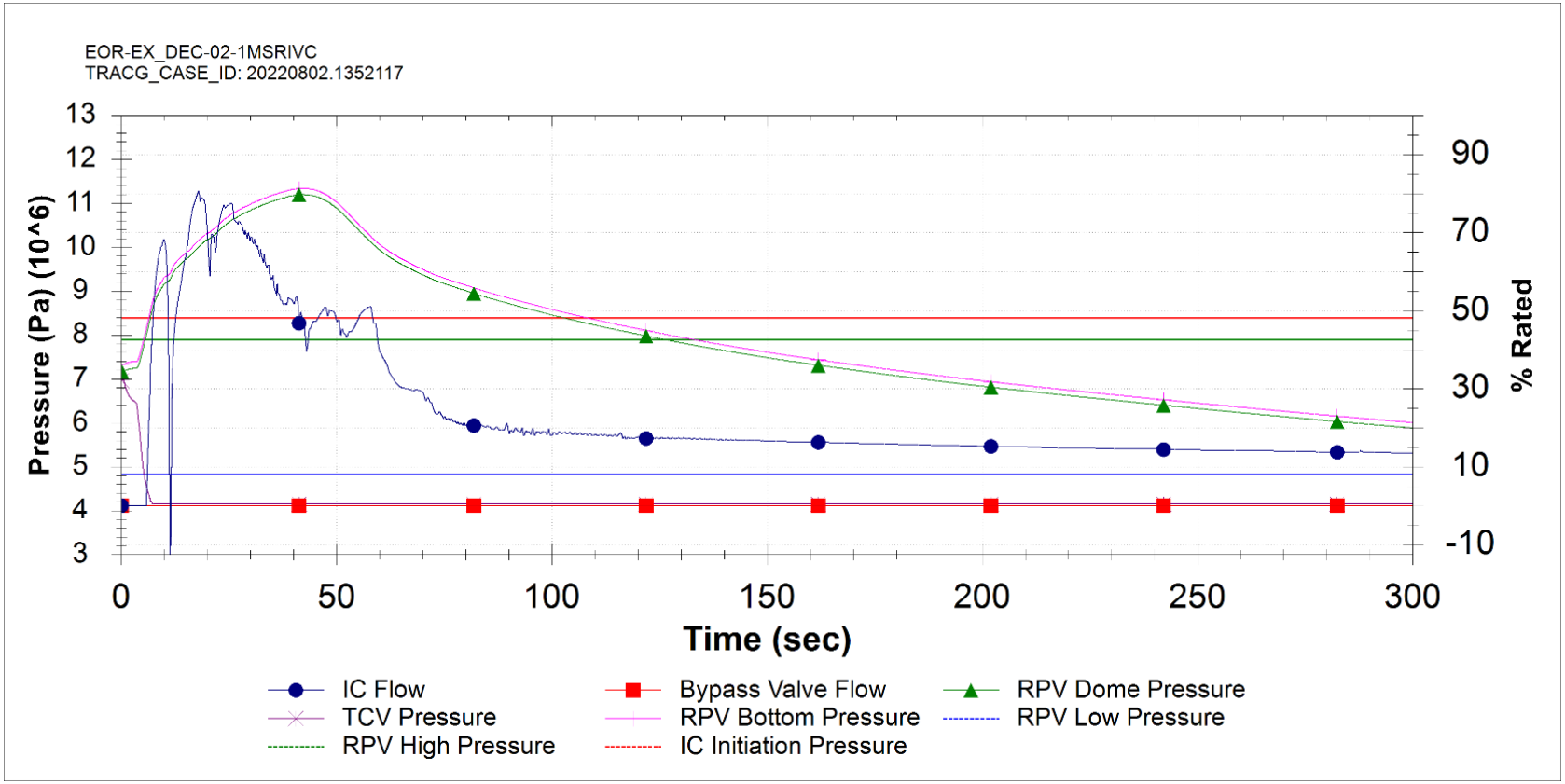


Figure 15.5-120: Closure of One Main Steam Reactor Isolation Valve with Failure to Scram (DEC)

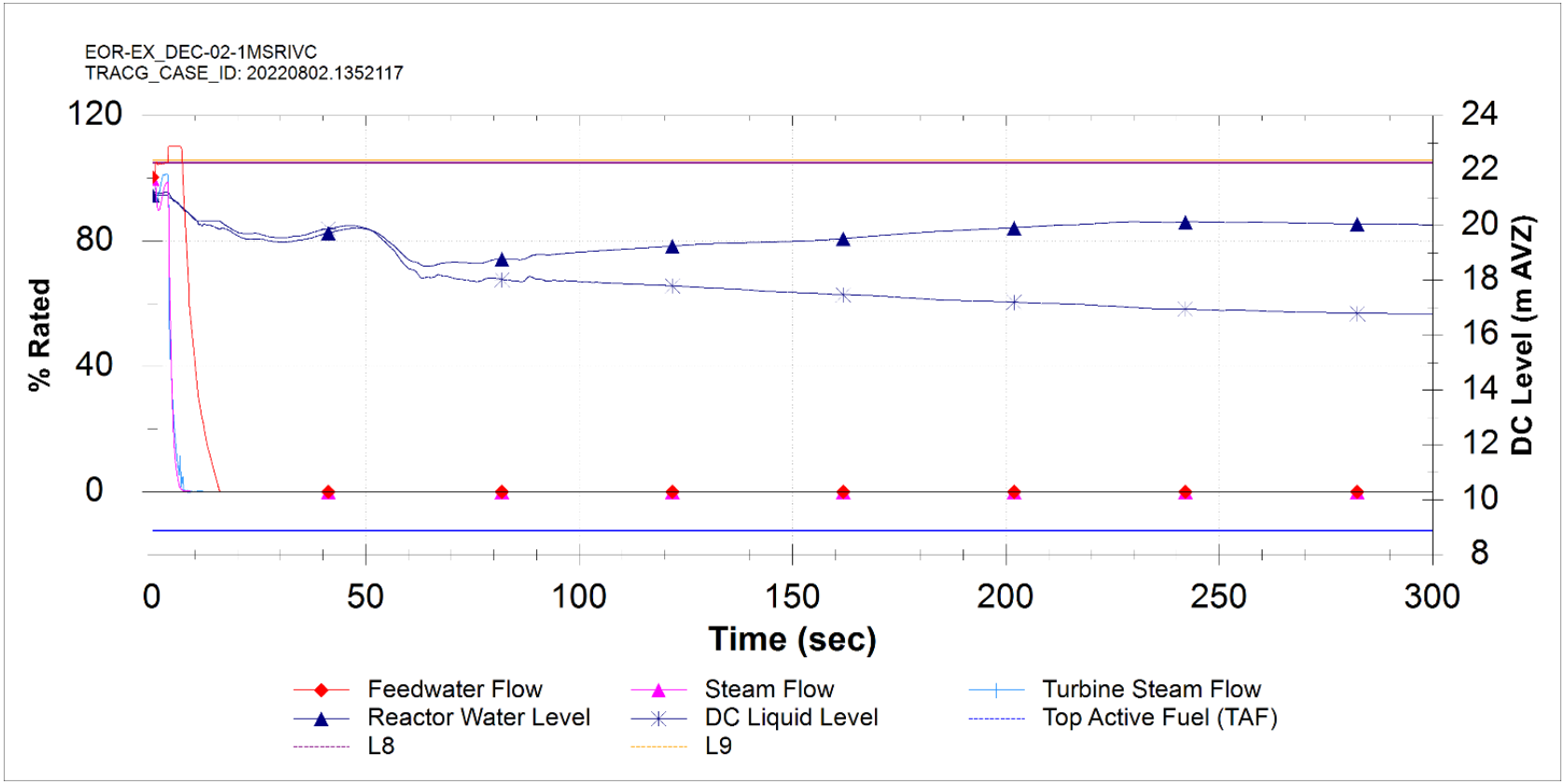


Figure 15.5-121: Closure of One Main Steam Reactor Isolation Valve with Failure to Scram (DEC)

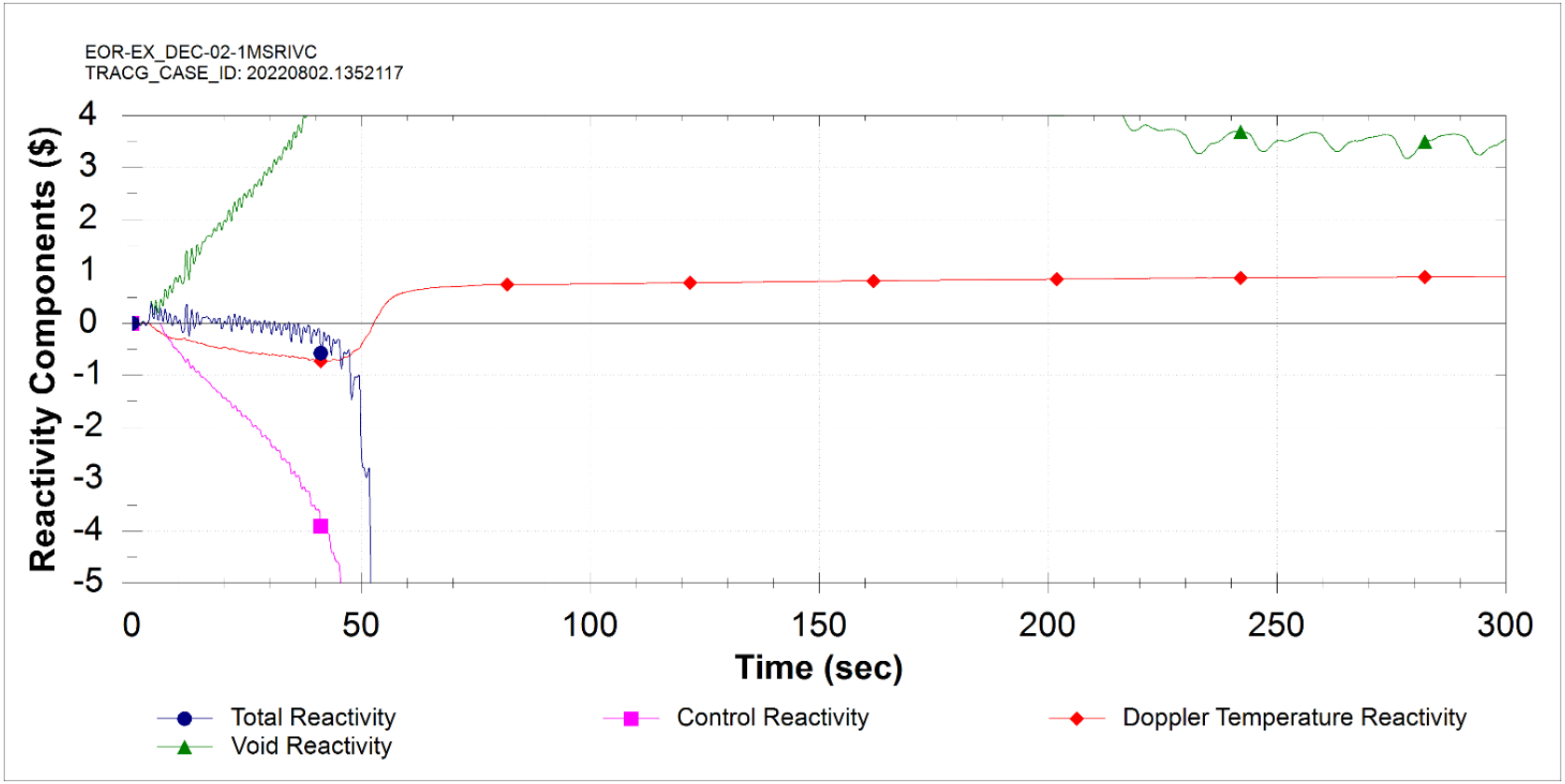


Figure 15.5-122: Closure of One Main Steam Reactor Isolation Valve with Failure to Scram (DEC)

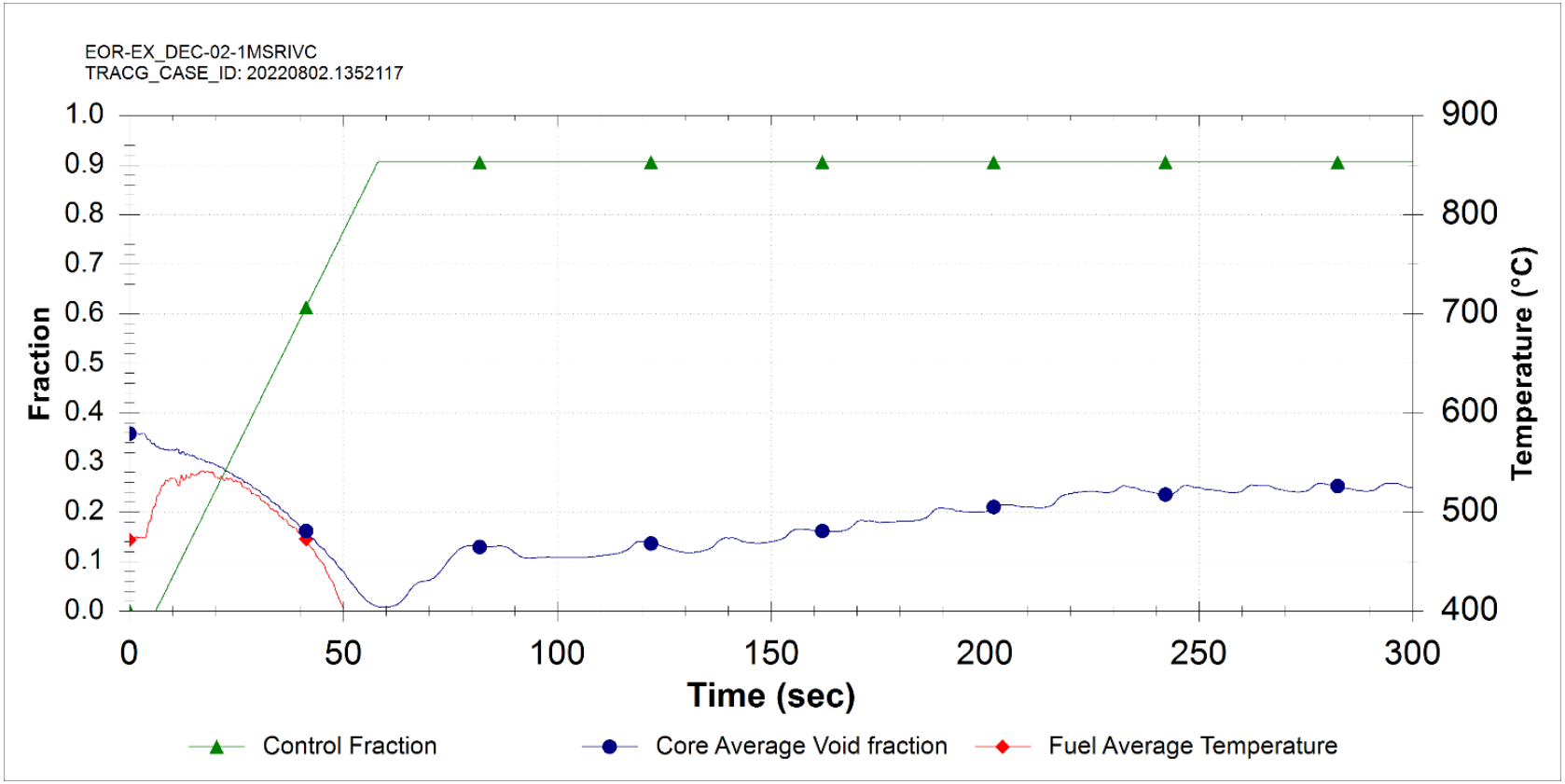


Figure 15.5-123: Closure of One Main Steam Reactor Isolation Valve with Failure to Scram (DEC)

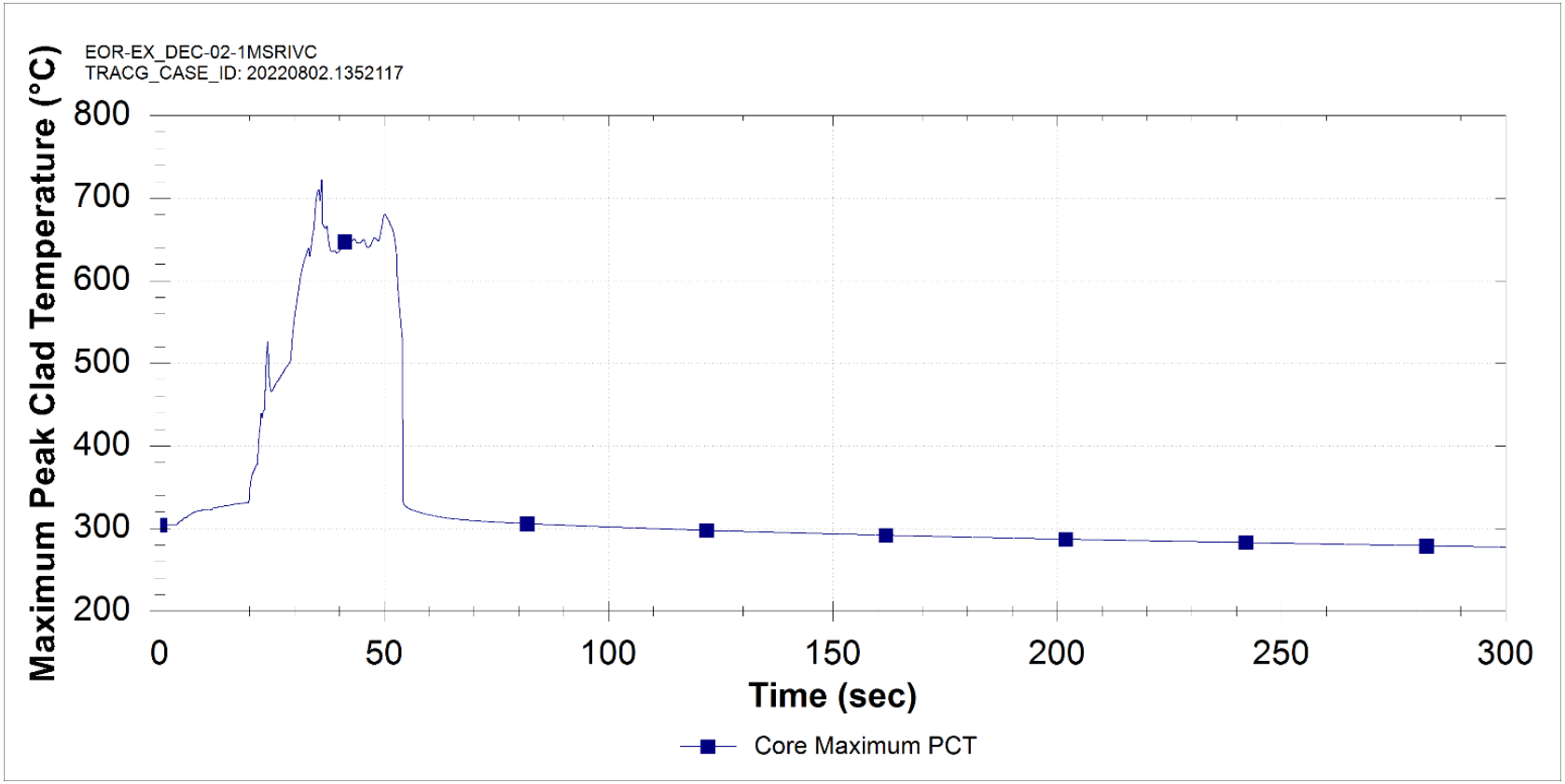


Figure 15.5-124: Closure of One Main Steam Reactor Isolation Valve with Failure to Scram (DEC)

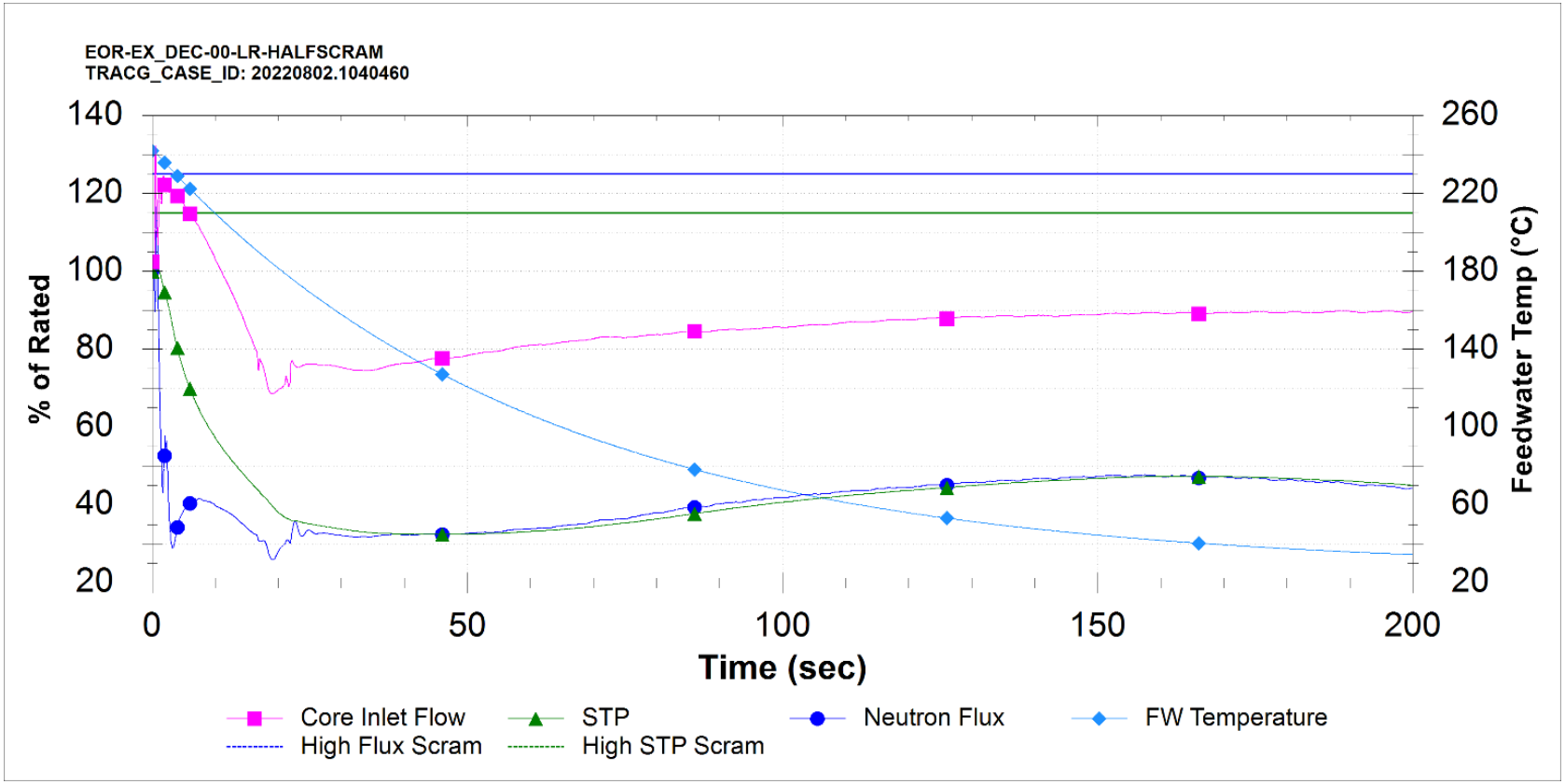


Figure 15.5-125: Complex Sequence Generator Load Rejection (DEC)

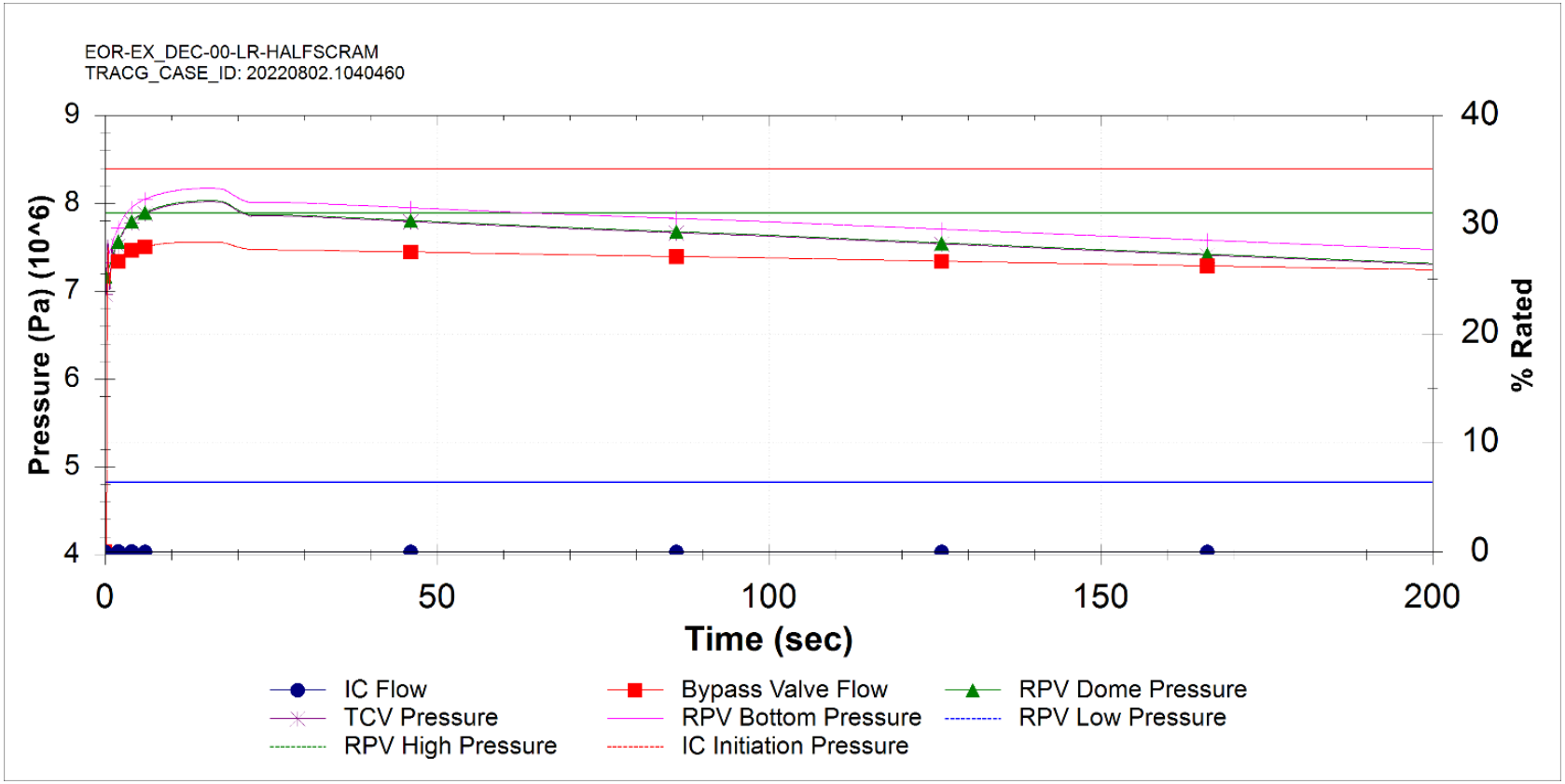


Figure 15.5-126: Complex Sequence Generator Load Rejection (DEC)

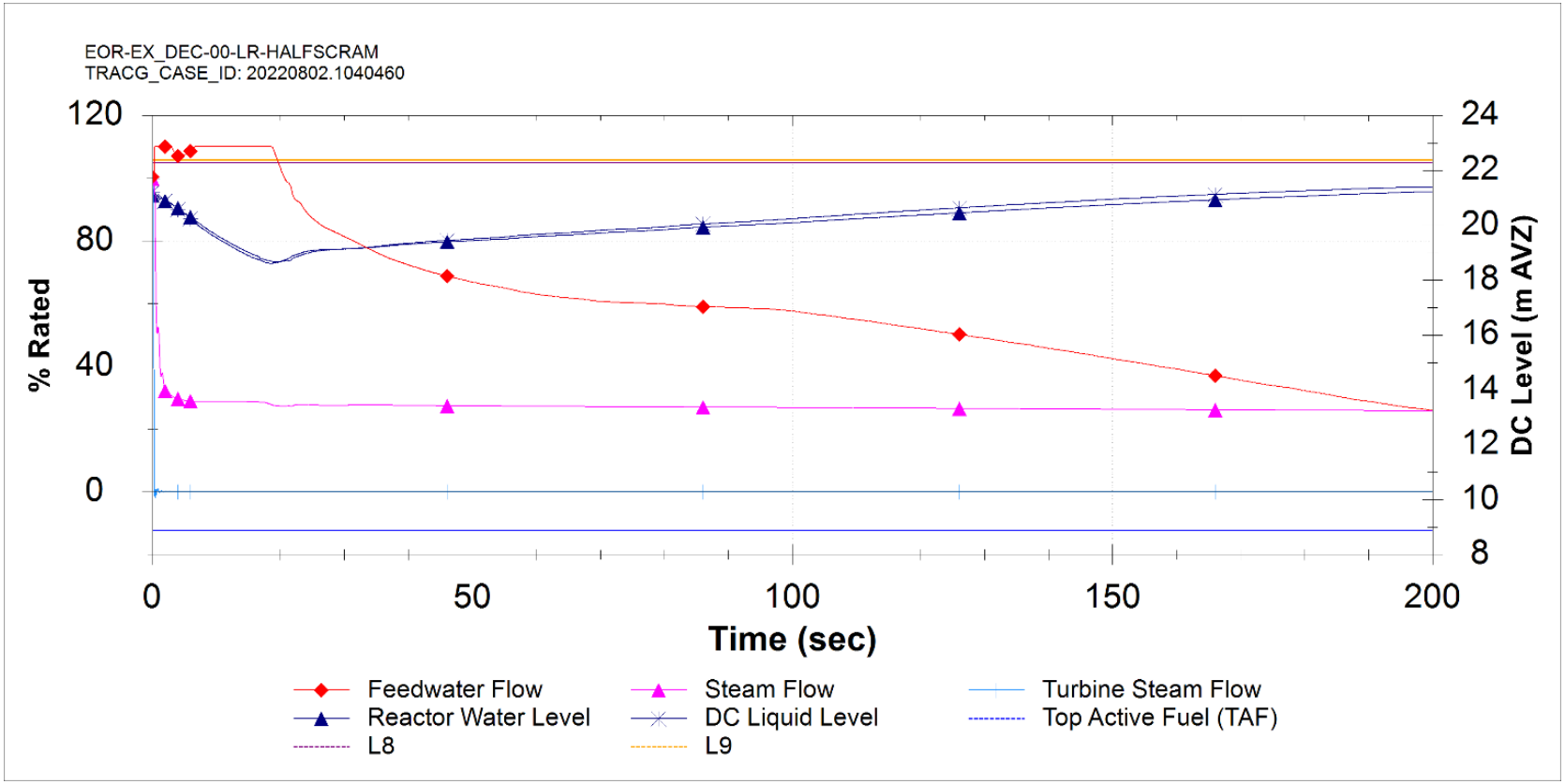


Figure 15.5-127: Complex Sequence Generator Load Rejection (DEC)

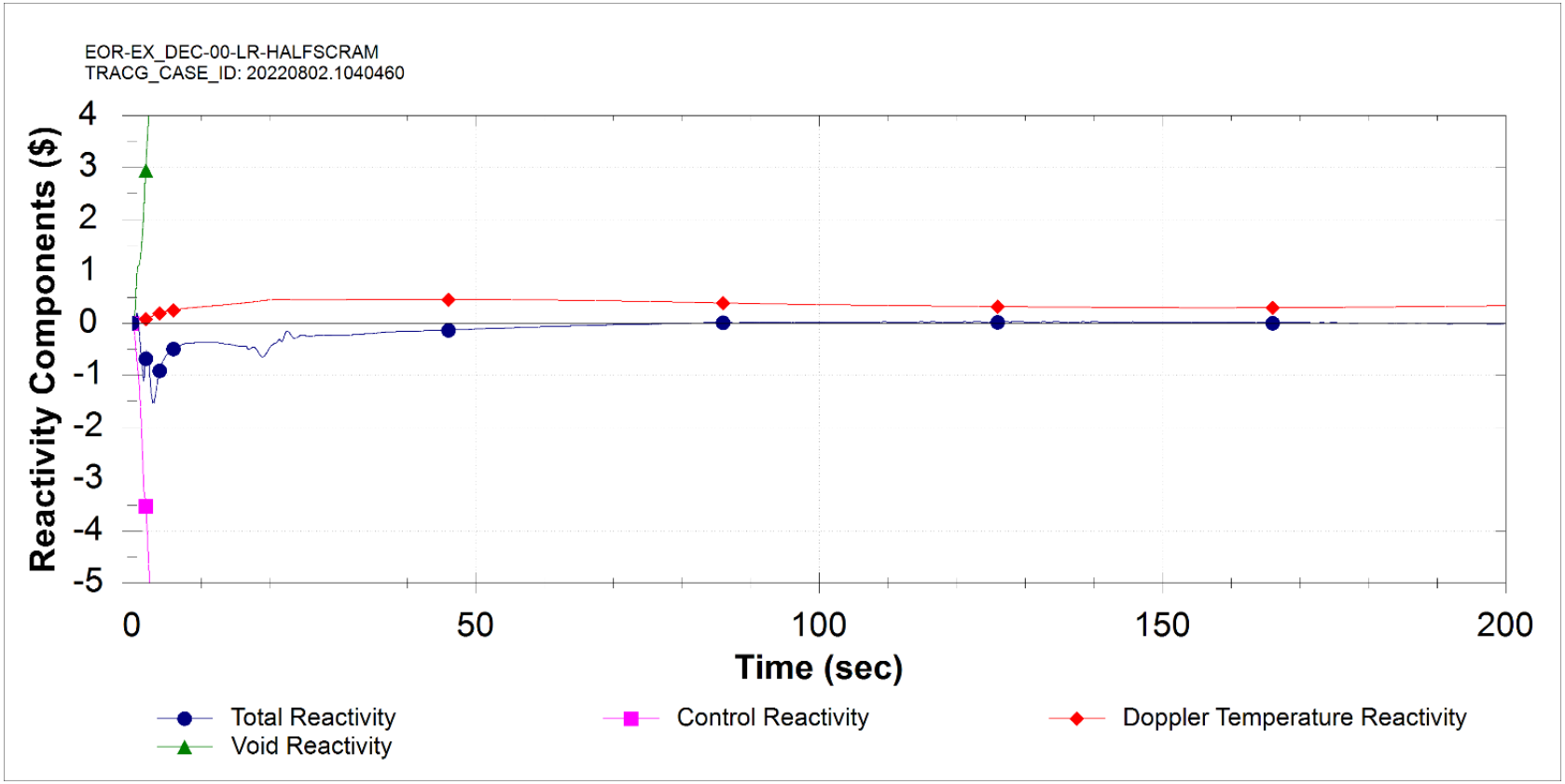


Figure 15.5-128: Complex Sequence Generator Load Rejection (DEC)

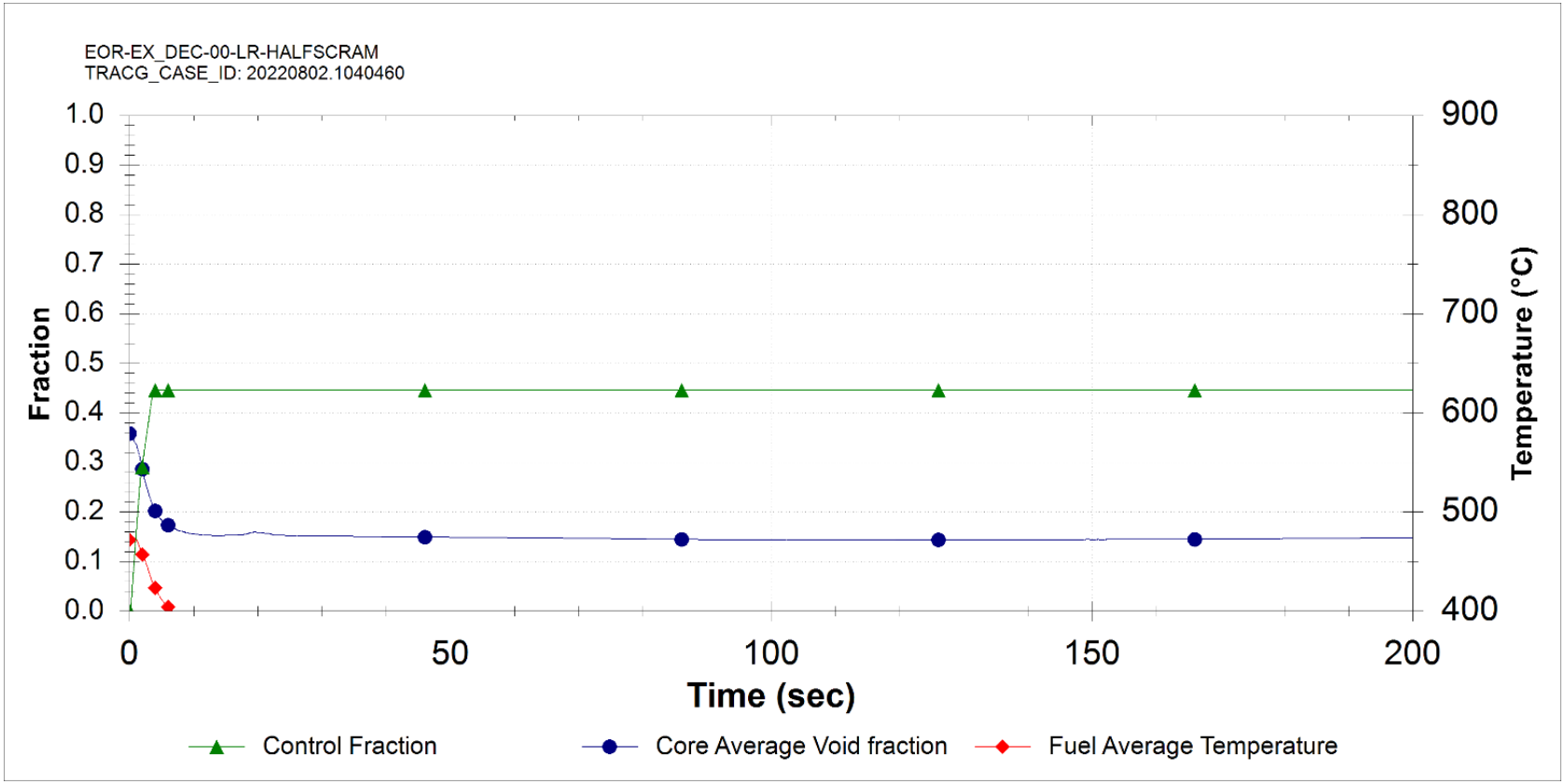


Figure 15.5-129: Complex Sequence Generator Load Rejection (DEC)

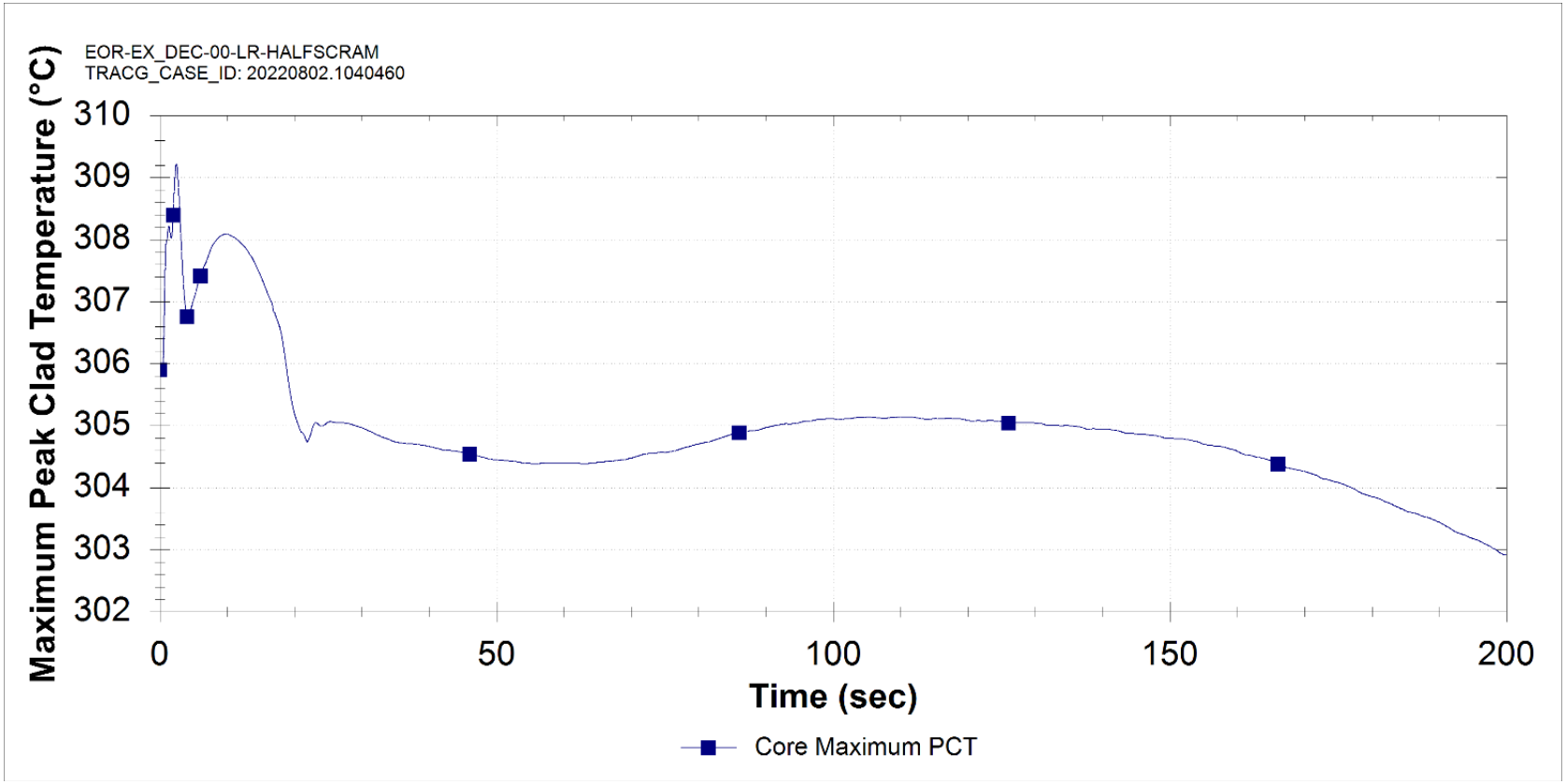


Figure 15.5-130: Complex Sequence Generator Load Rejection (DEC)

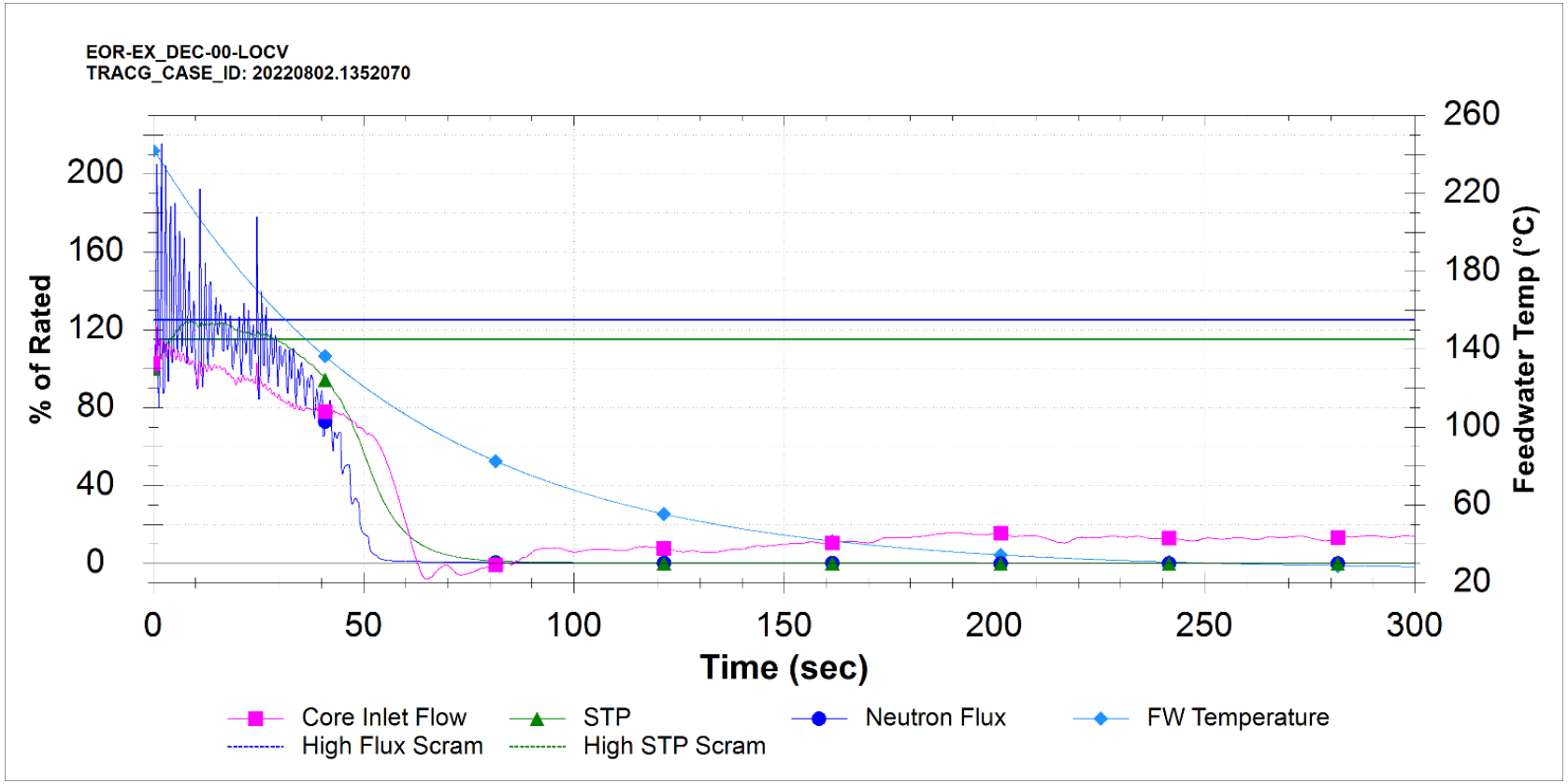


Figure 15.5-131: Loss of Condenser Vacuum (DEC)

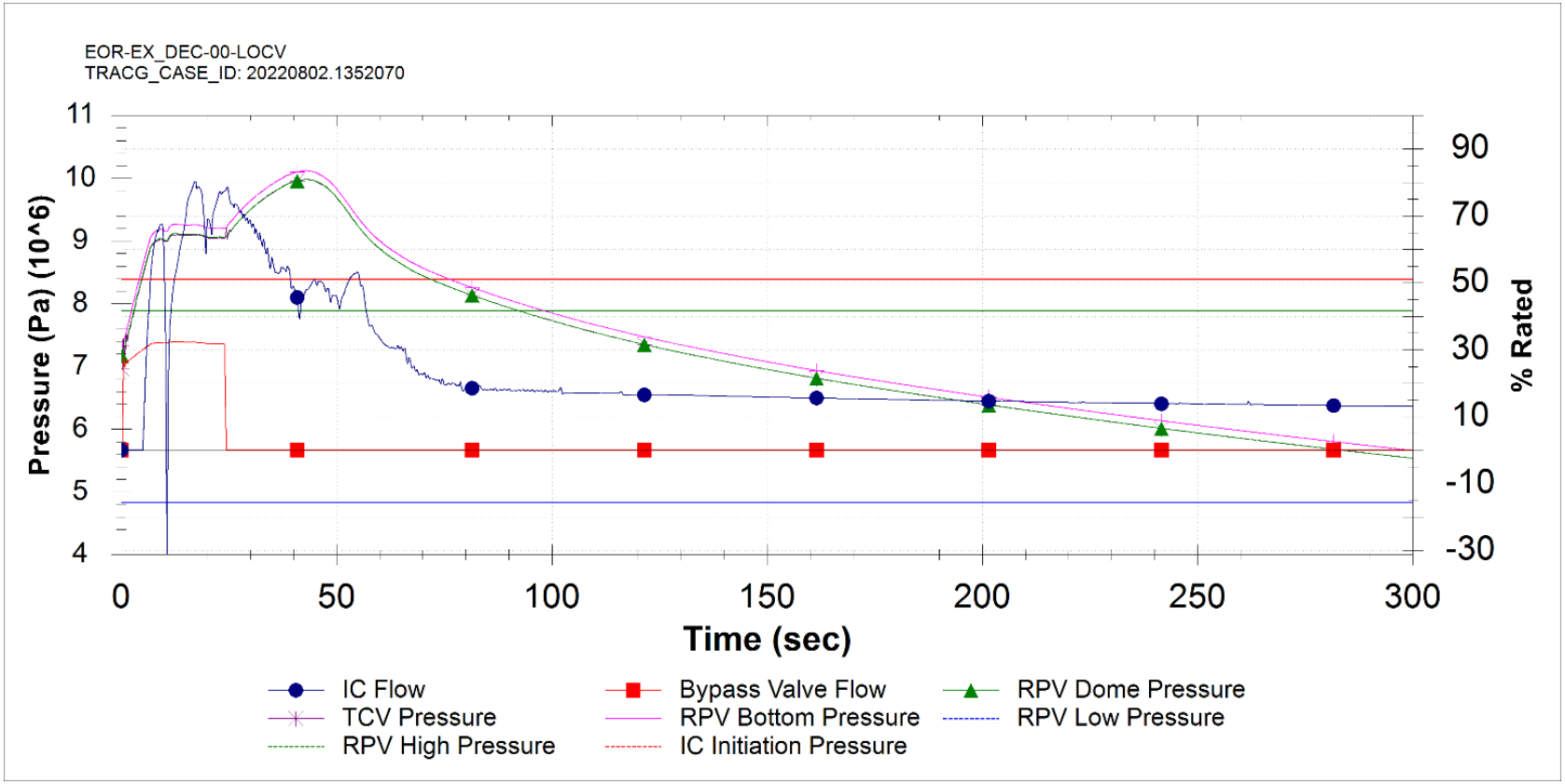


Figure 15.5-132: Loss of Condenser Vacuum (DEC)

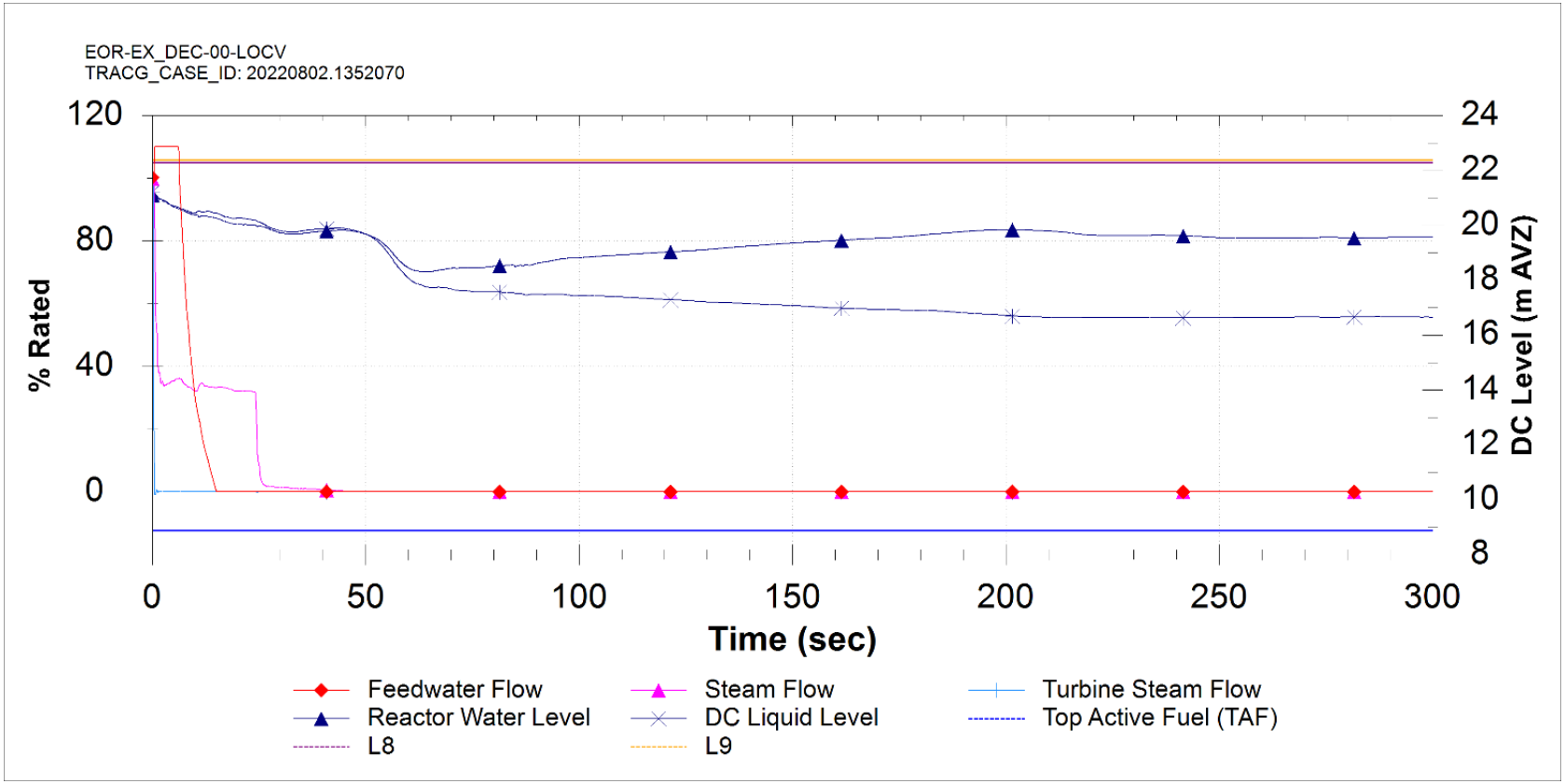


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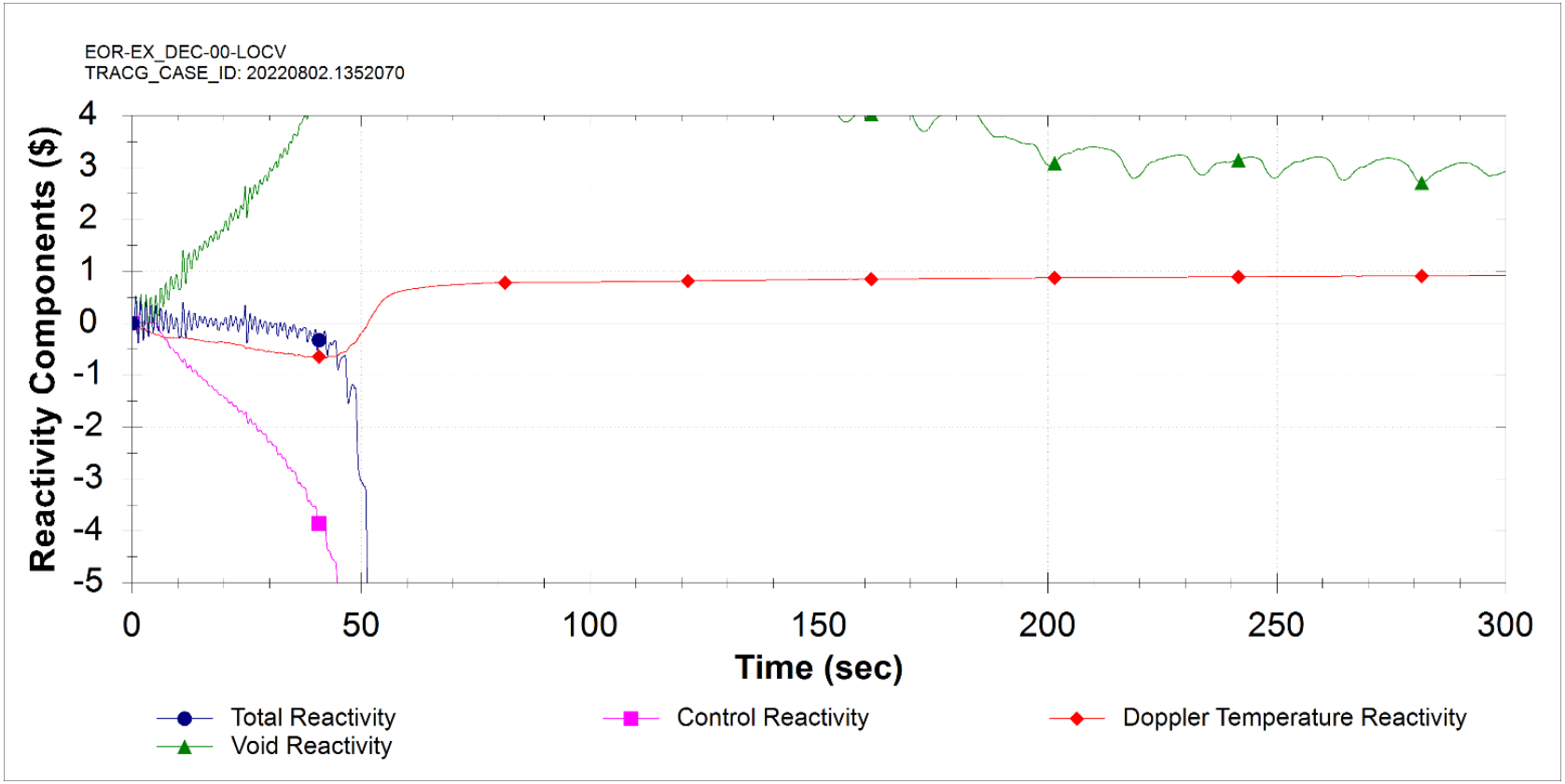


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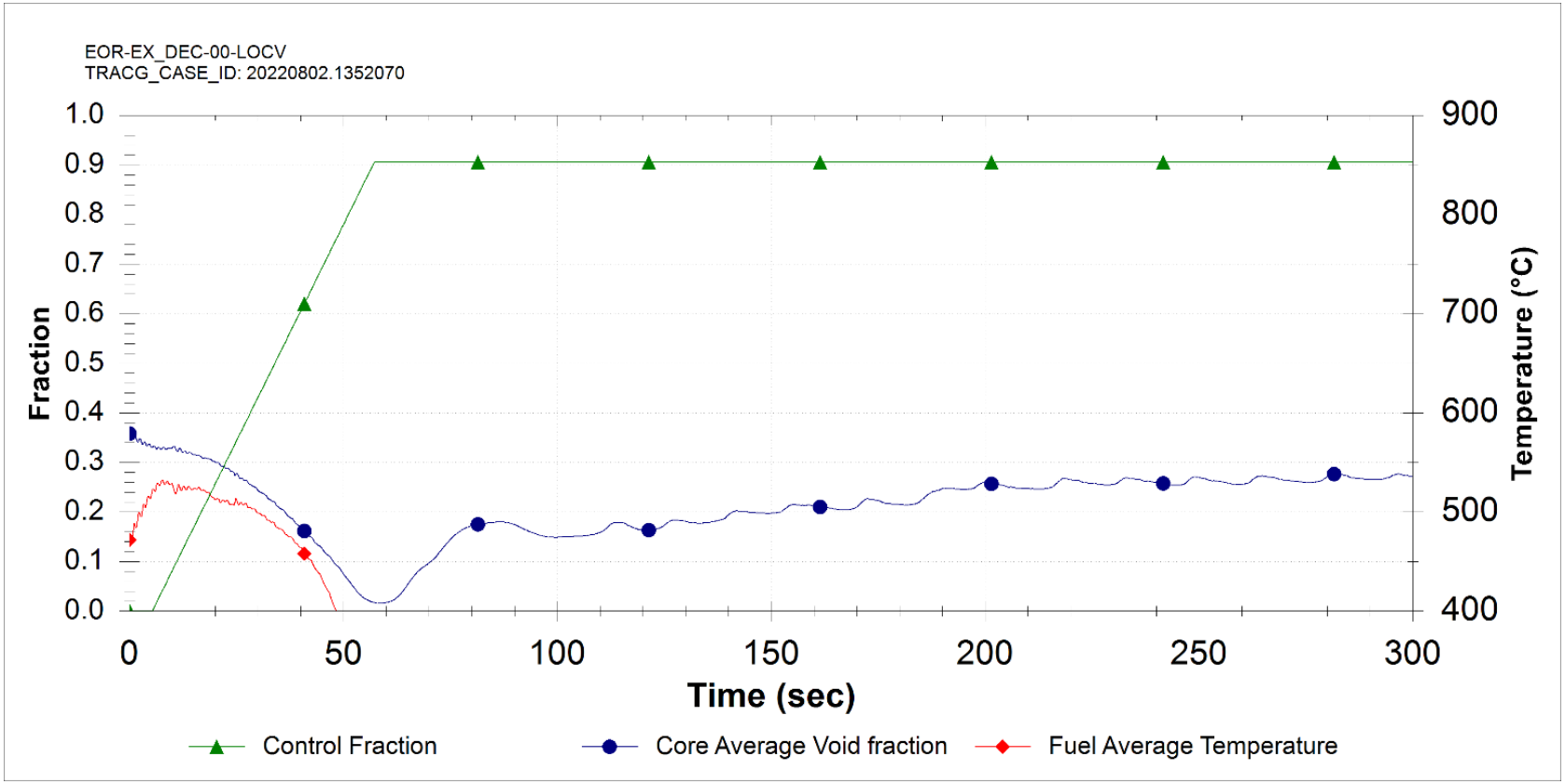


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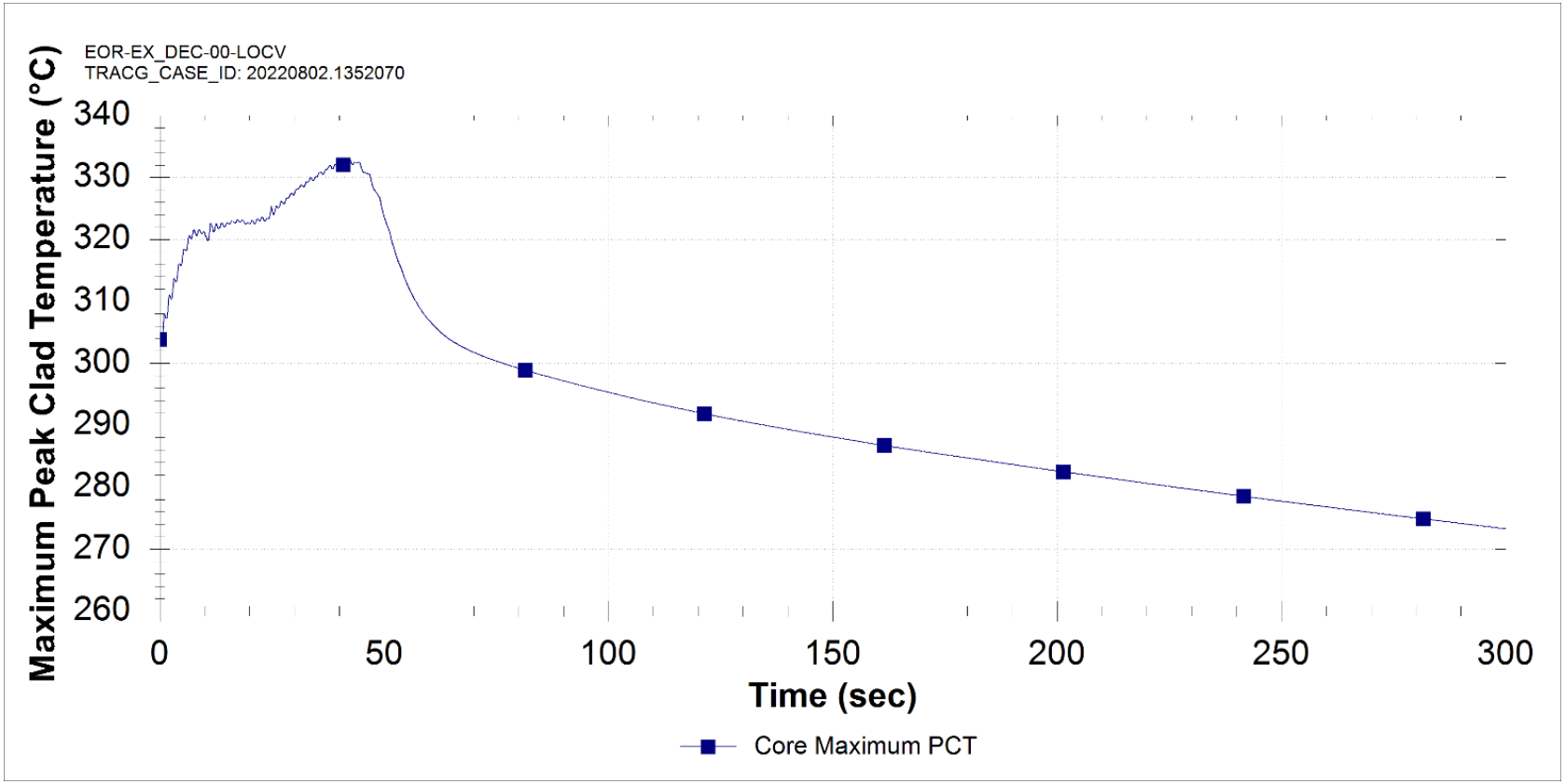


Figure 15.5-136: Loss of Condenser Vacuum (DEC)

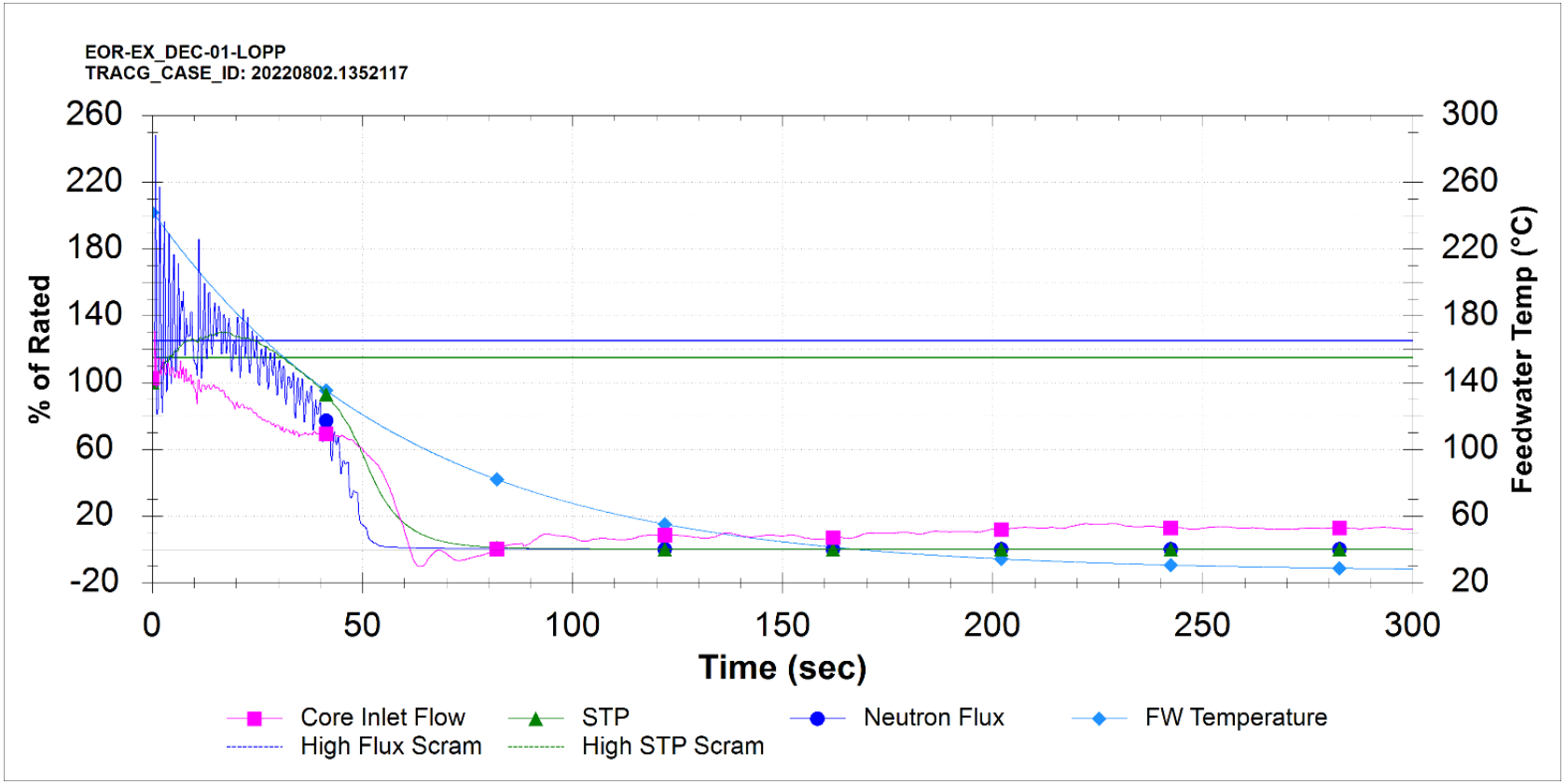


Figure 15.5-137: Loss of Preferred Power (DEC)

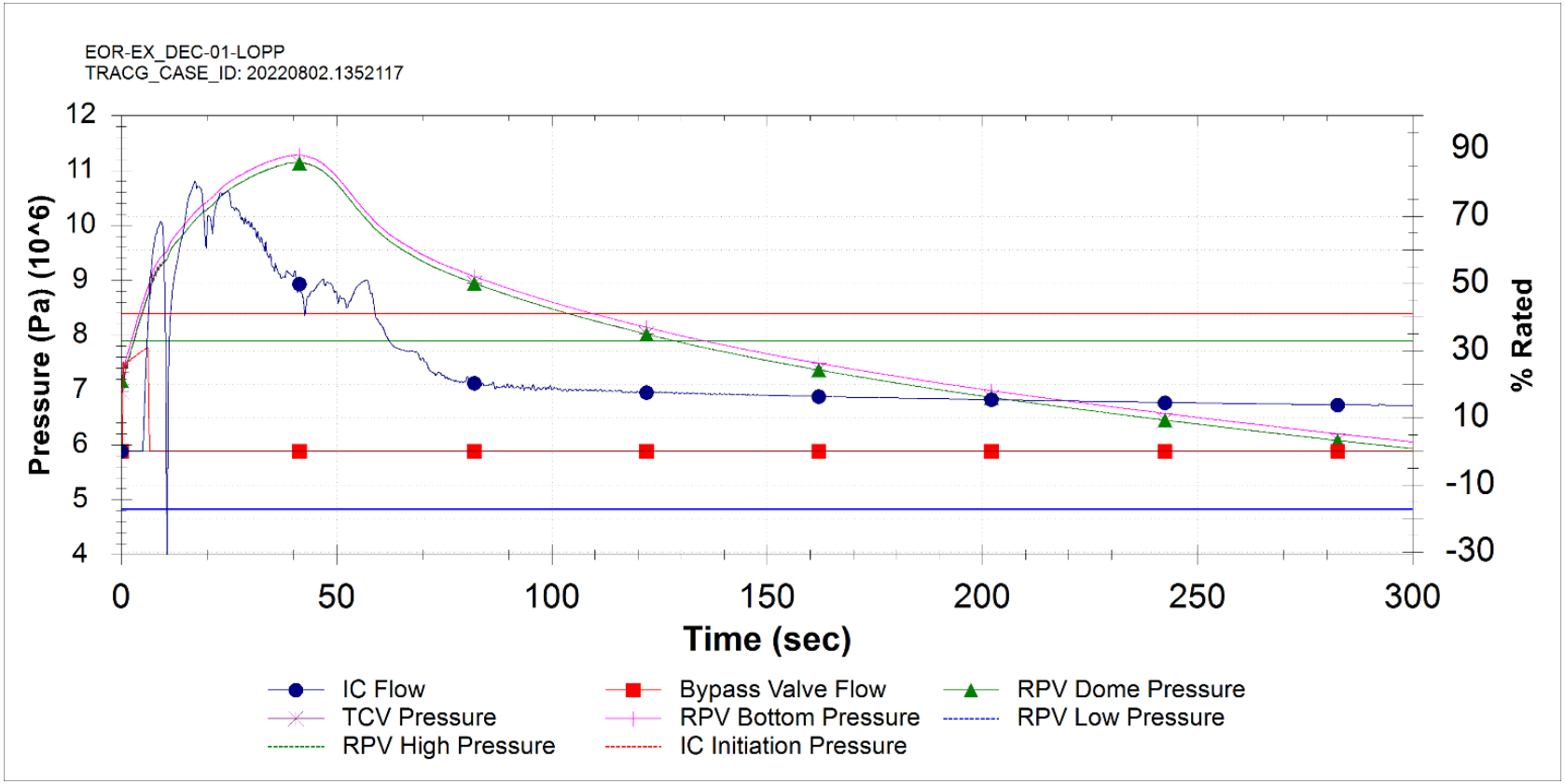


Figure 15.5-138: Loss of Preferred Power (DEC)

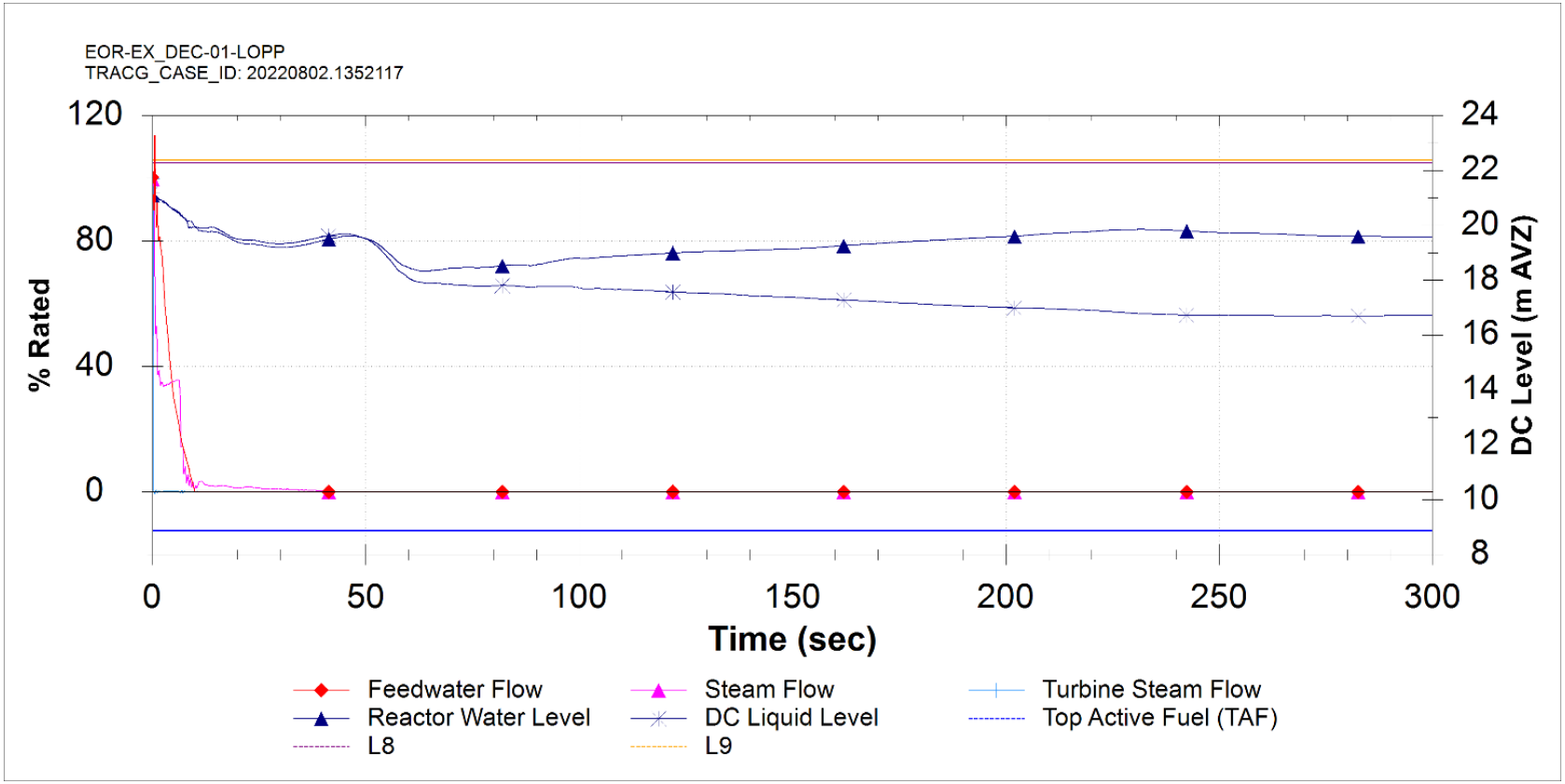


Figure 15.5-139: Loss of Preferred Power (DEC)

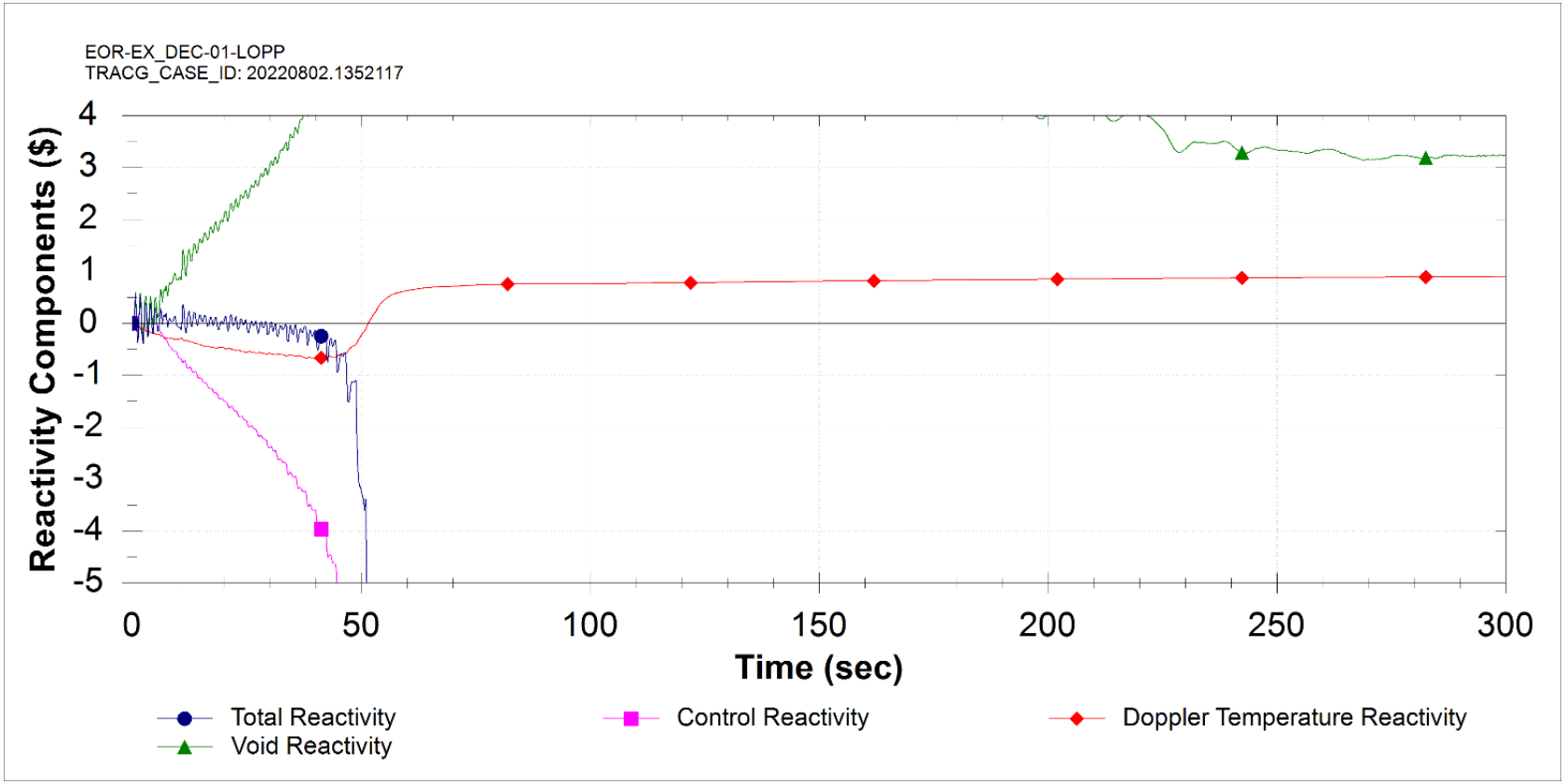


Figure 15.5-140: Loss of Preferred Power (DEC)

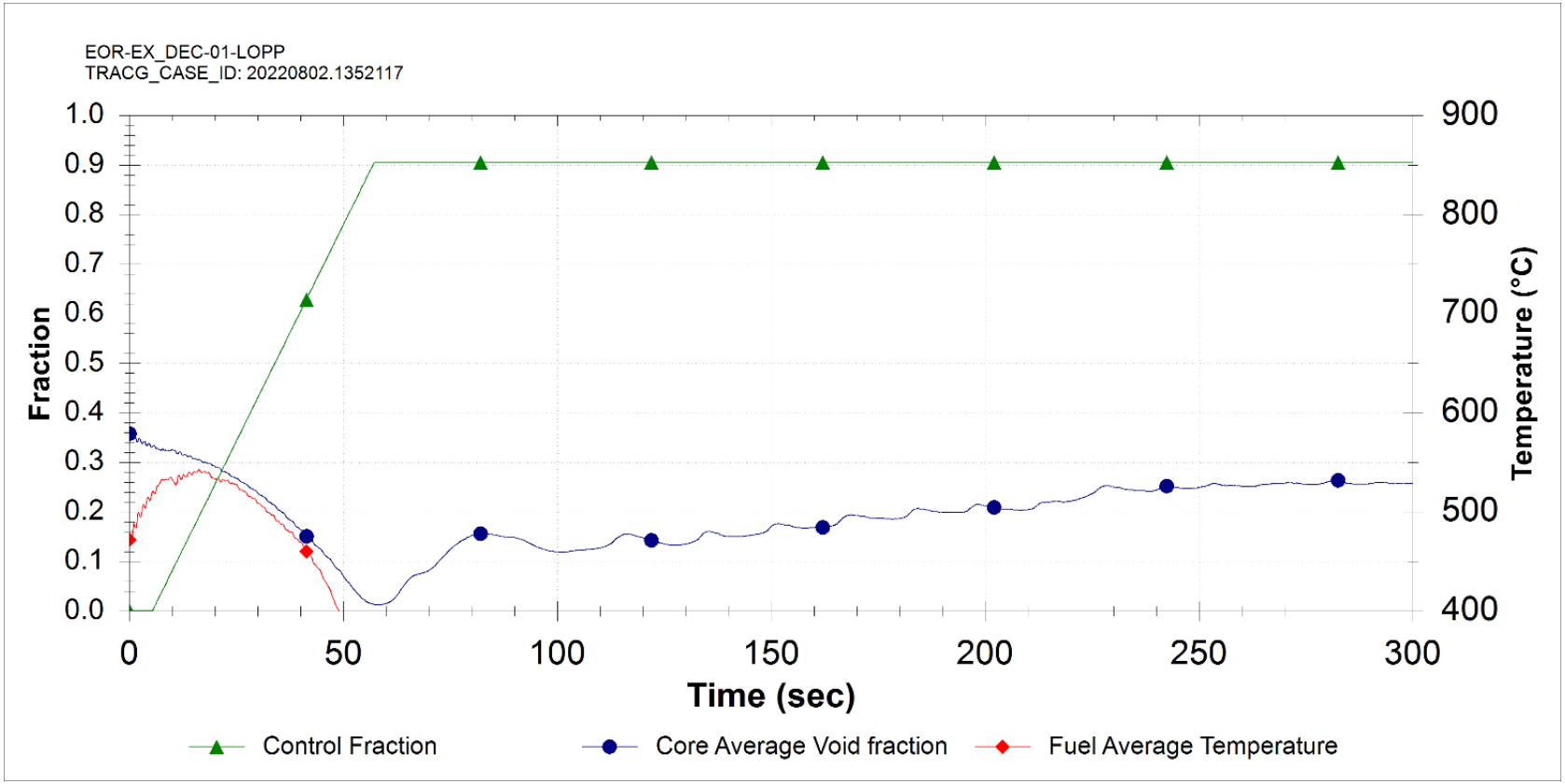


Figure 15.5-141: Loss of Preferred Power (DEC)

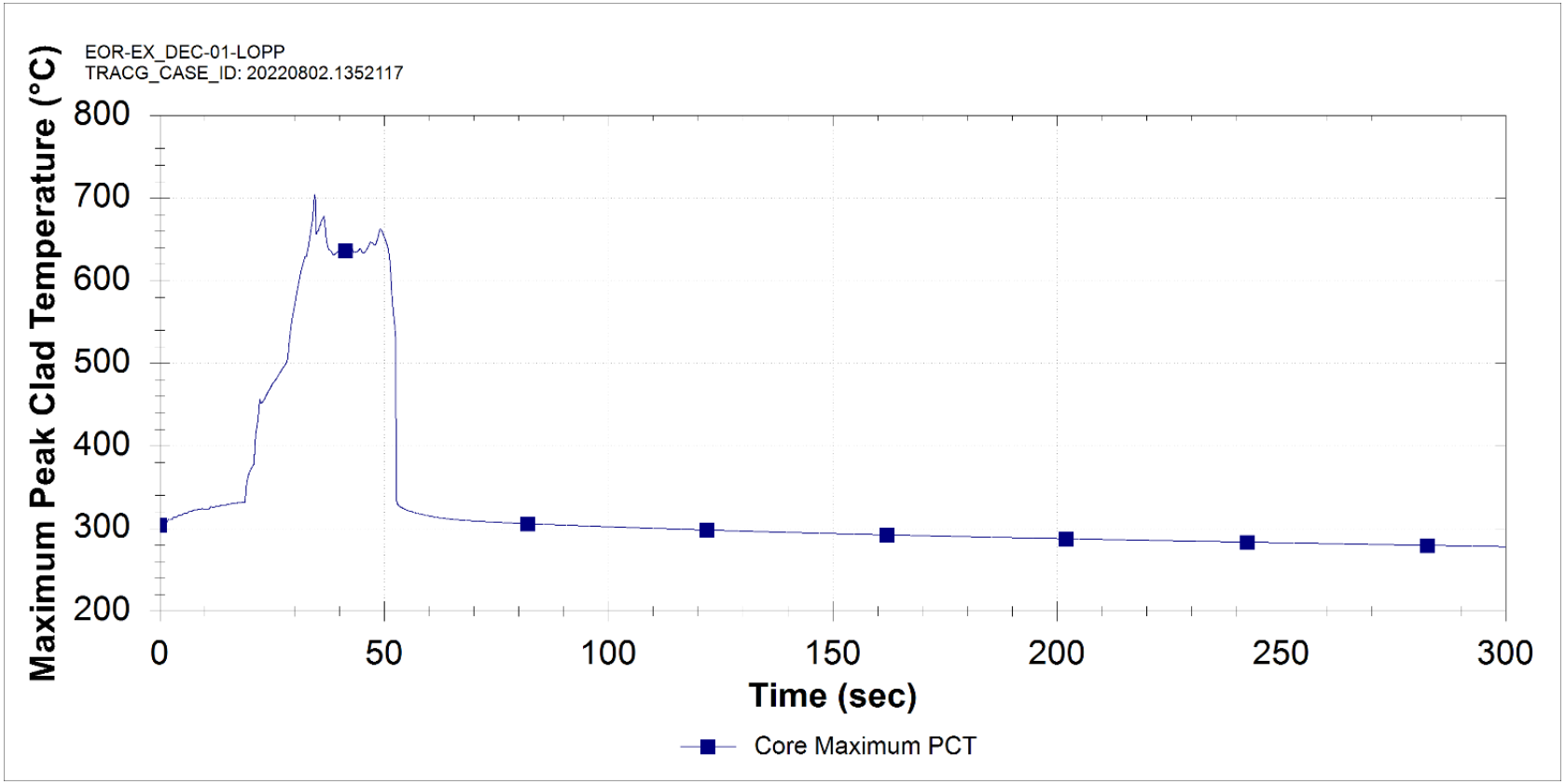


Figure 15.5-142: Loss of Preferred Power (DEC)

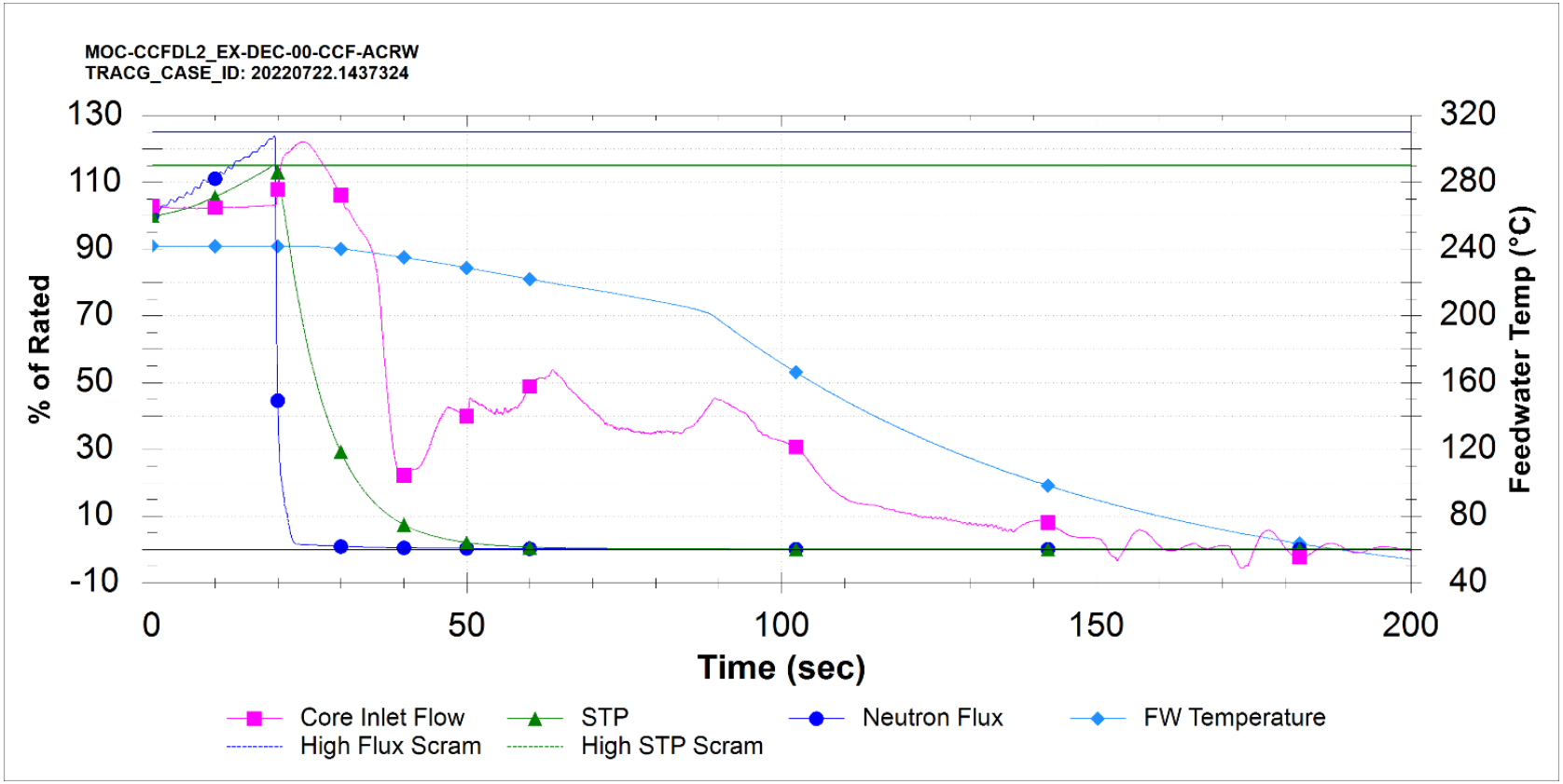


Figure 15.5-143: All Control Rod Withdrawal at Power (ACRW)

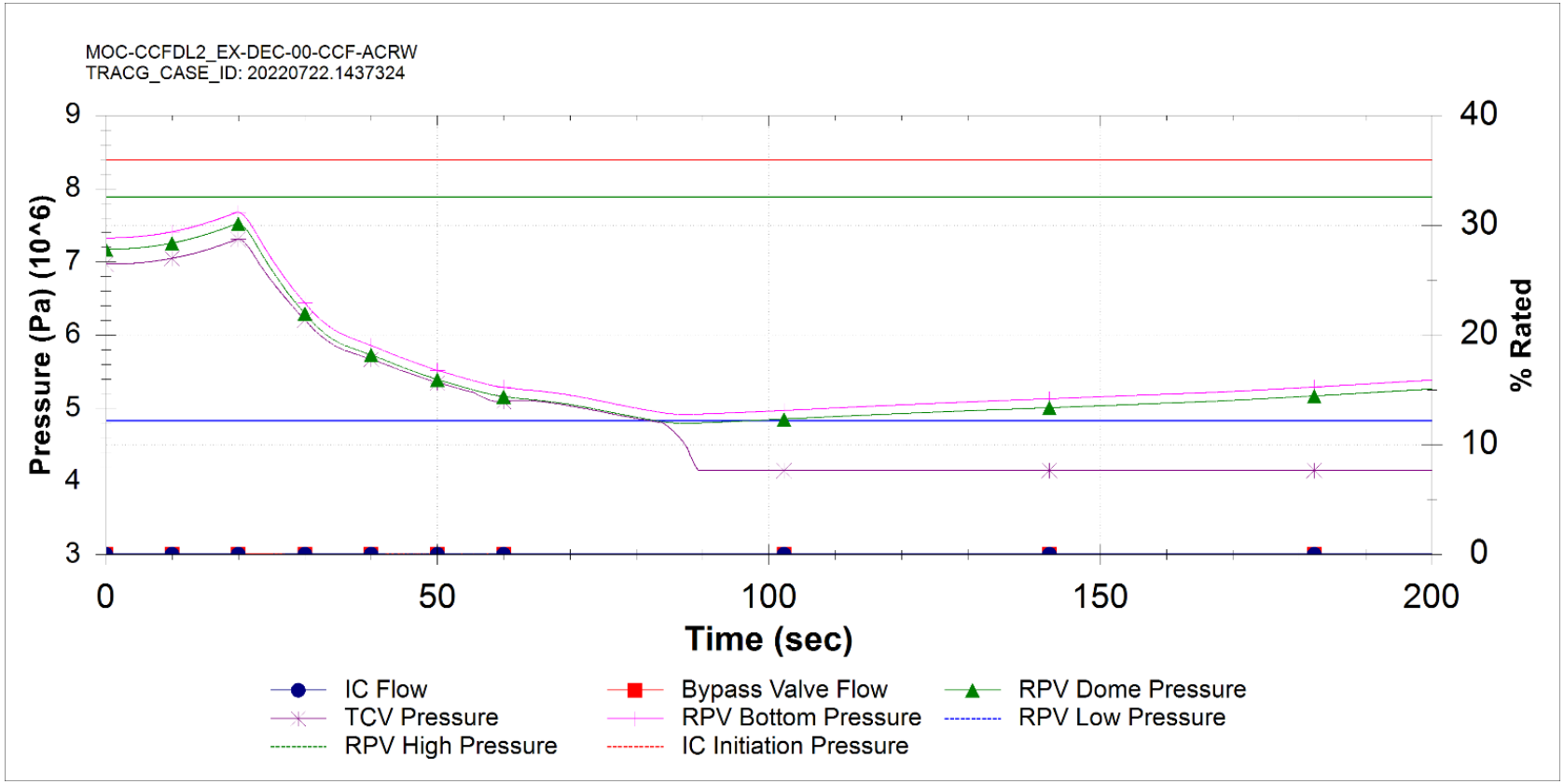


Figure 15.5-144: All Control Rod Withdrawal at Power (ACRW)

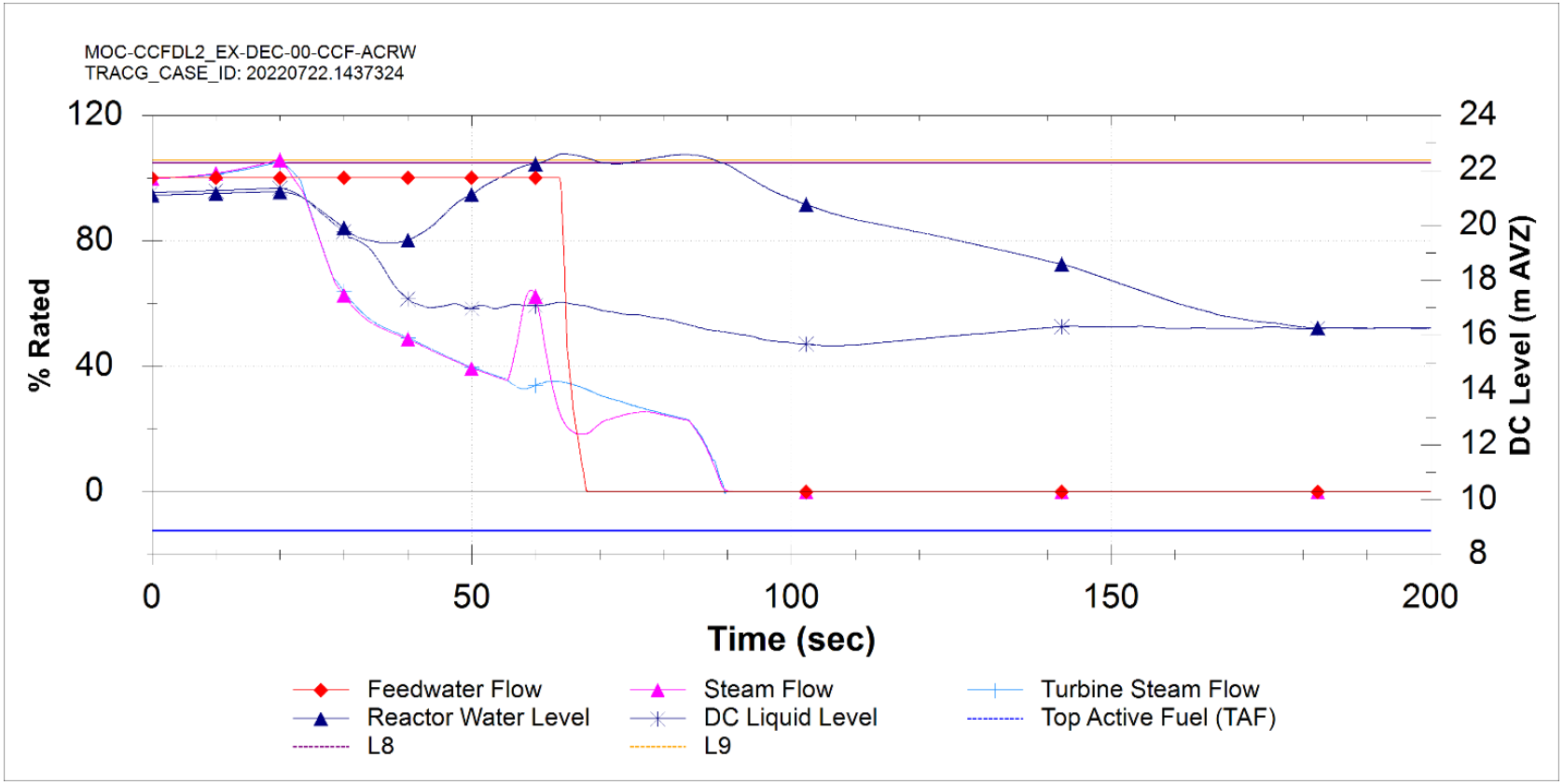


Figure 15.5-145: All Control Rod Withdrawal at Power (ACRW)

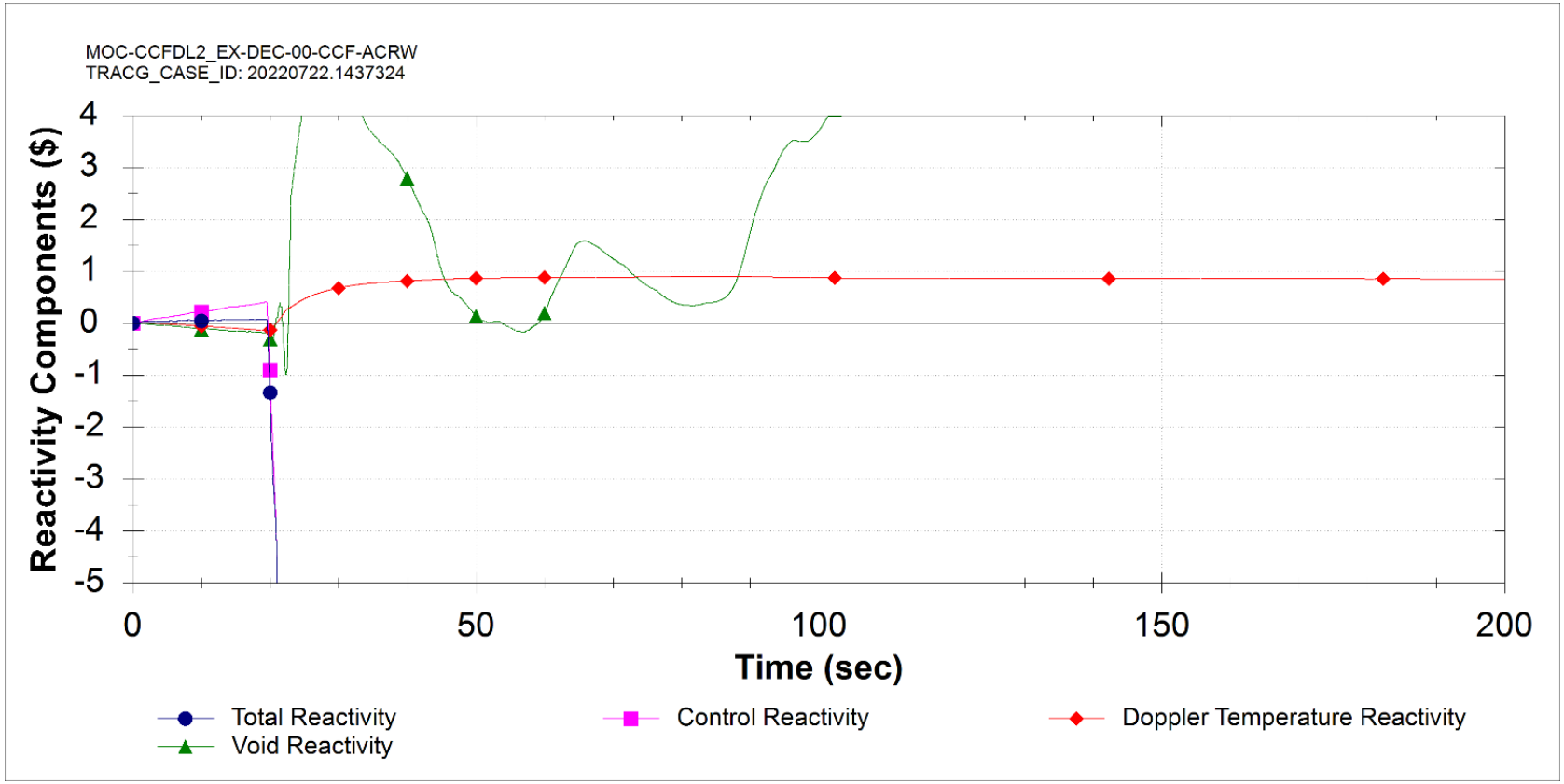


Figure 15.5-146: All Control Rod Withdrawal at Power (ACRW)

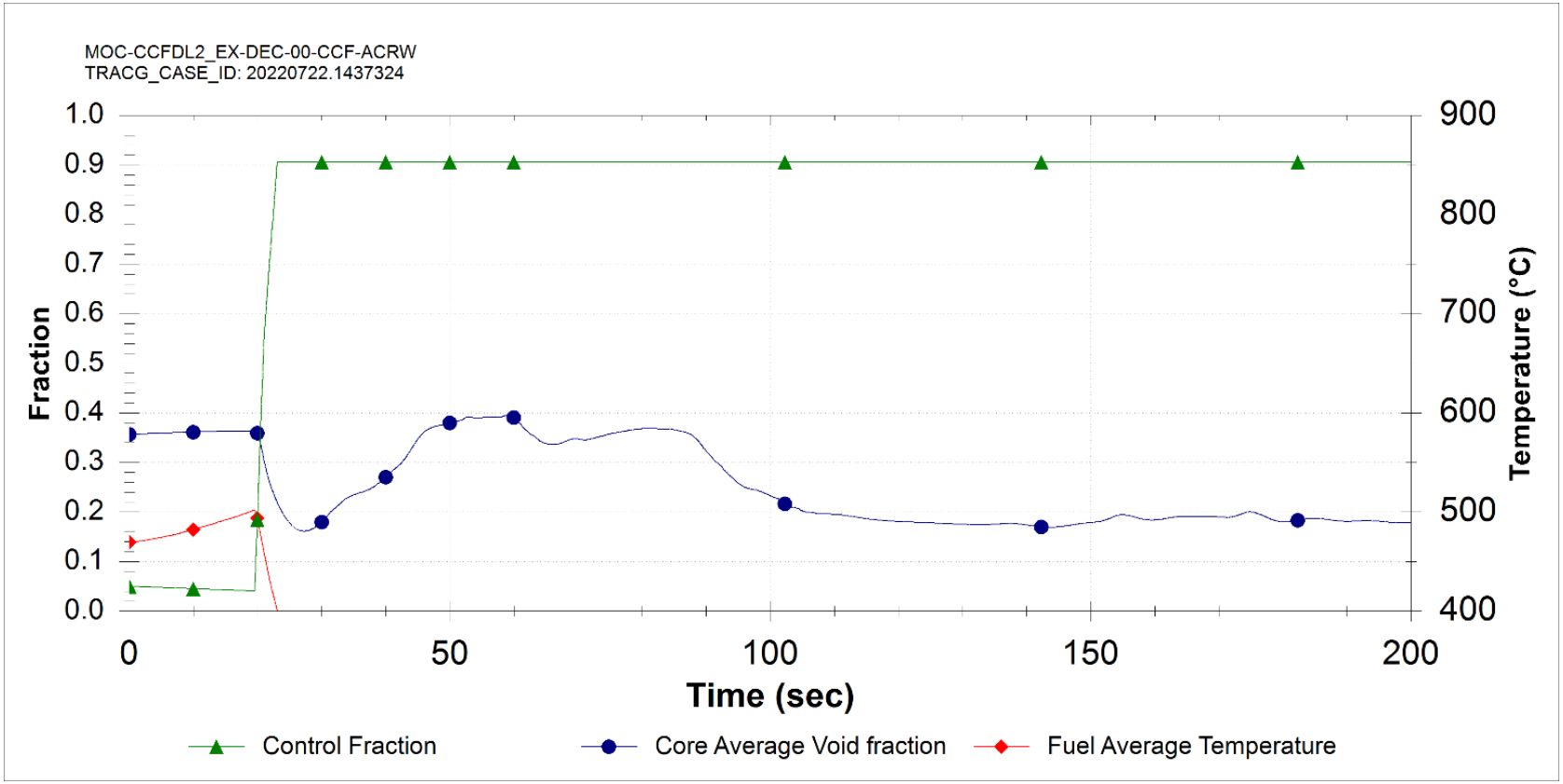


Figure 15.5-147: All Control Rod Withdrawal at Power (ACRW)

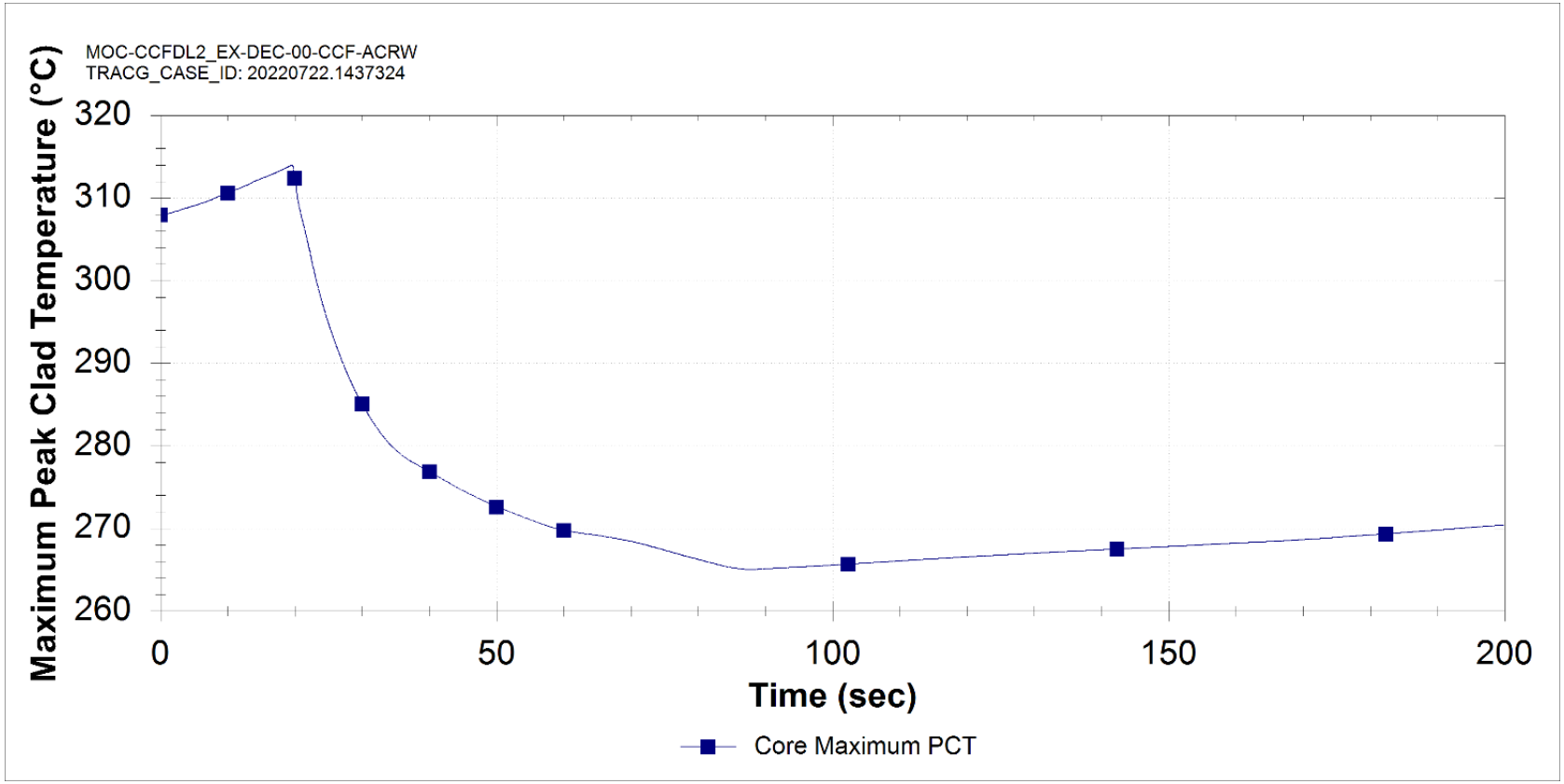


Figure 15.5-148: All Control Rod Withdrawal at Power (ACRW)

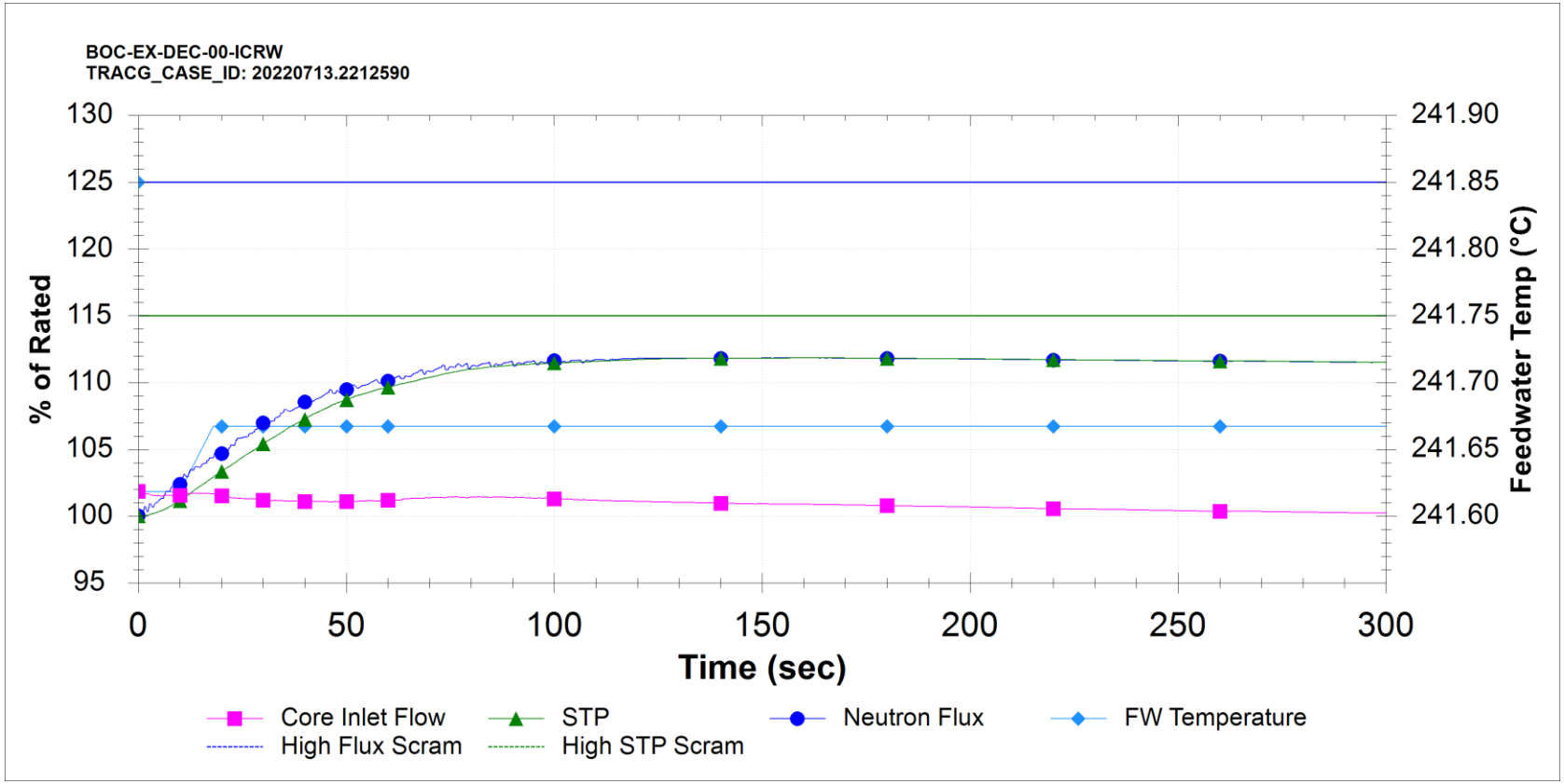


Figure 15.5-149: Inadvertent Control Rod Withdrawal at Power - Single Rod (ICRW)

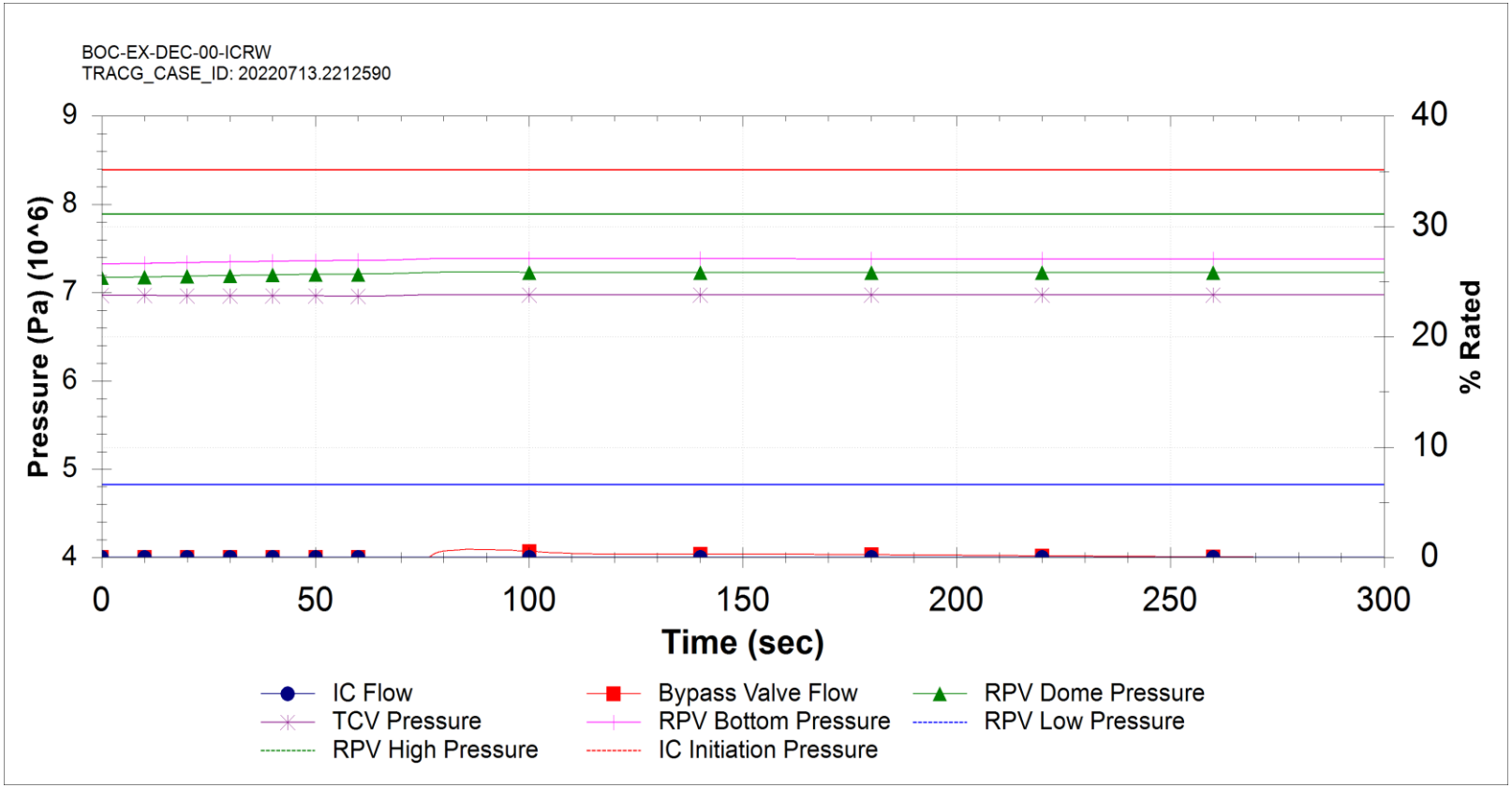


Figure 15.5-150: Inadvertent Control Rod Withdrawal at Power - Single Rod (ICRW)

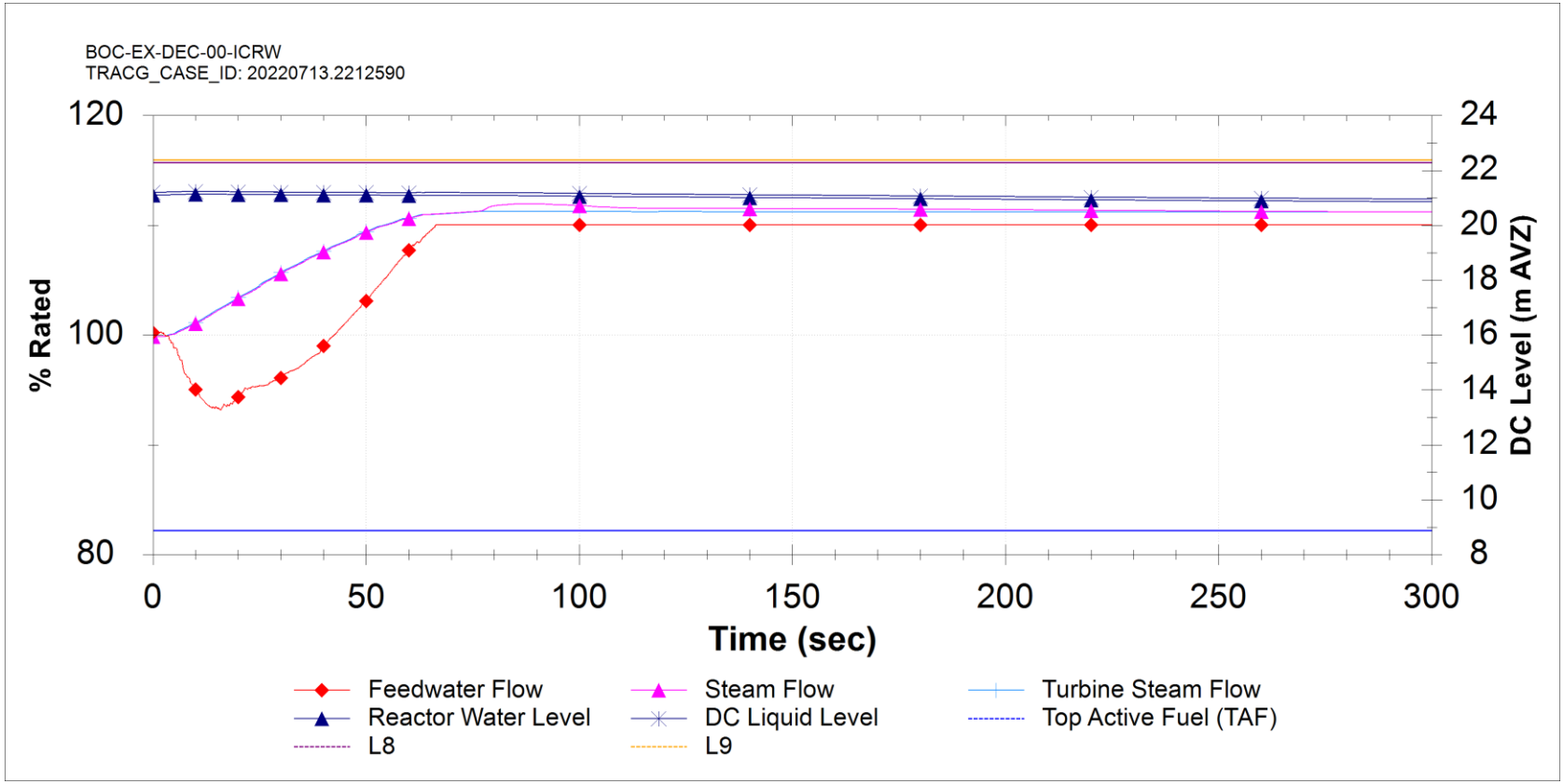


Figure 15.5-151: Inadvertent Control Rod Withdrawal at Power - Single Rod (ICRW)

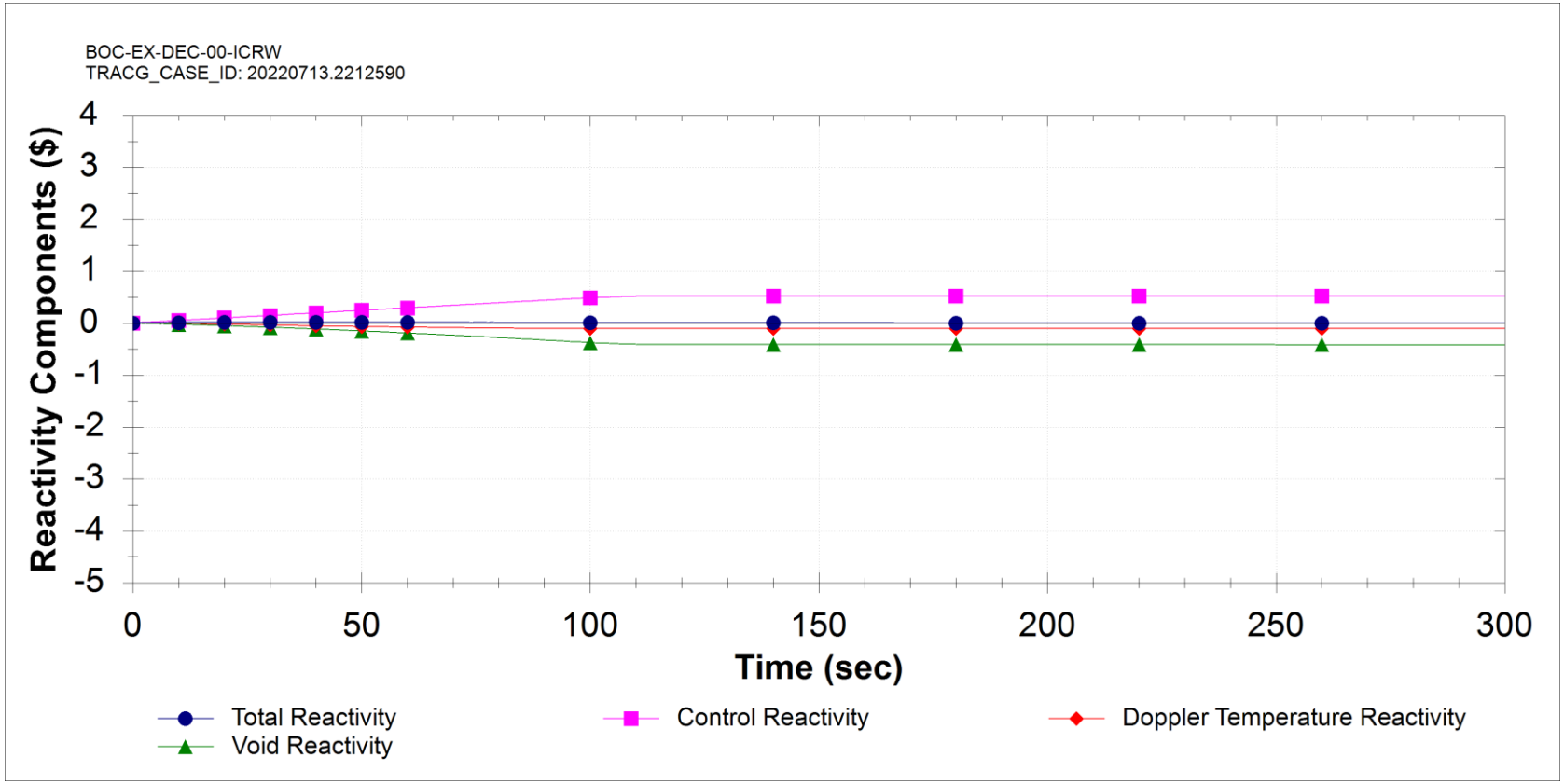


Figure 15.5-152: Inadvertent Control Rod Withdrawal at Power - Single Rod (ICRW)

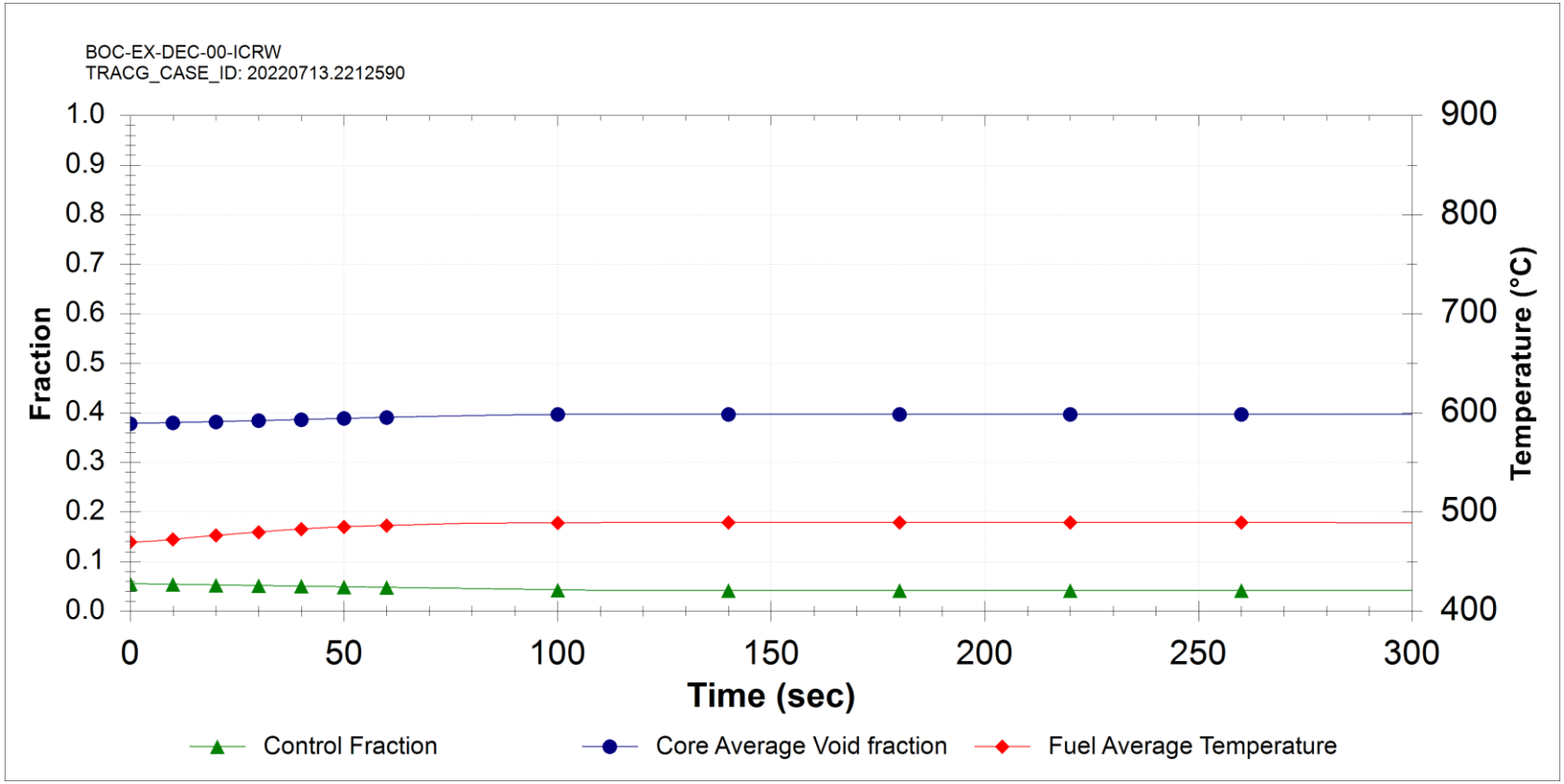


Figure 15.5-153: Inadvertent Control Rod Withdrawal at Power - Single Rod (ICRW)

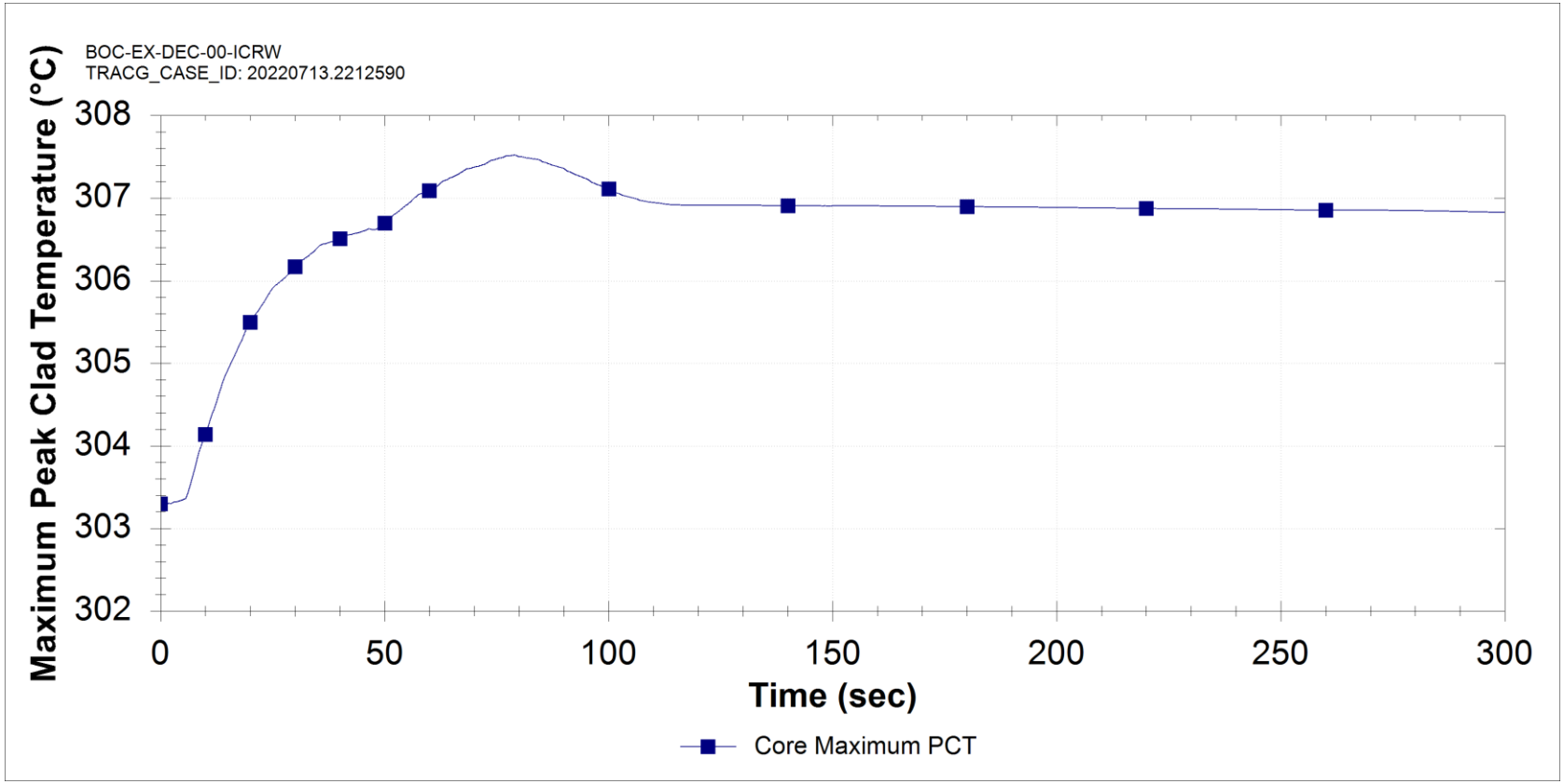


Figure 15.5-154: Inadvertent Control Rod Withdrawal at Power - Single Rod (ICRW)

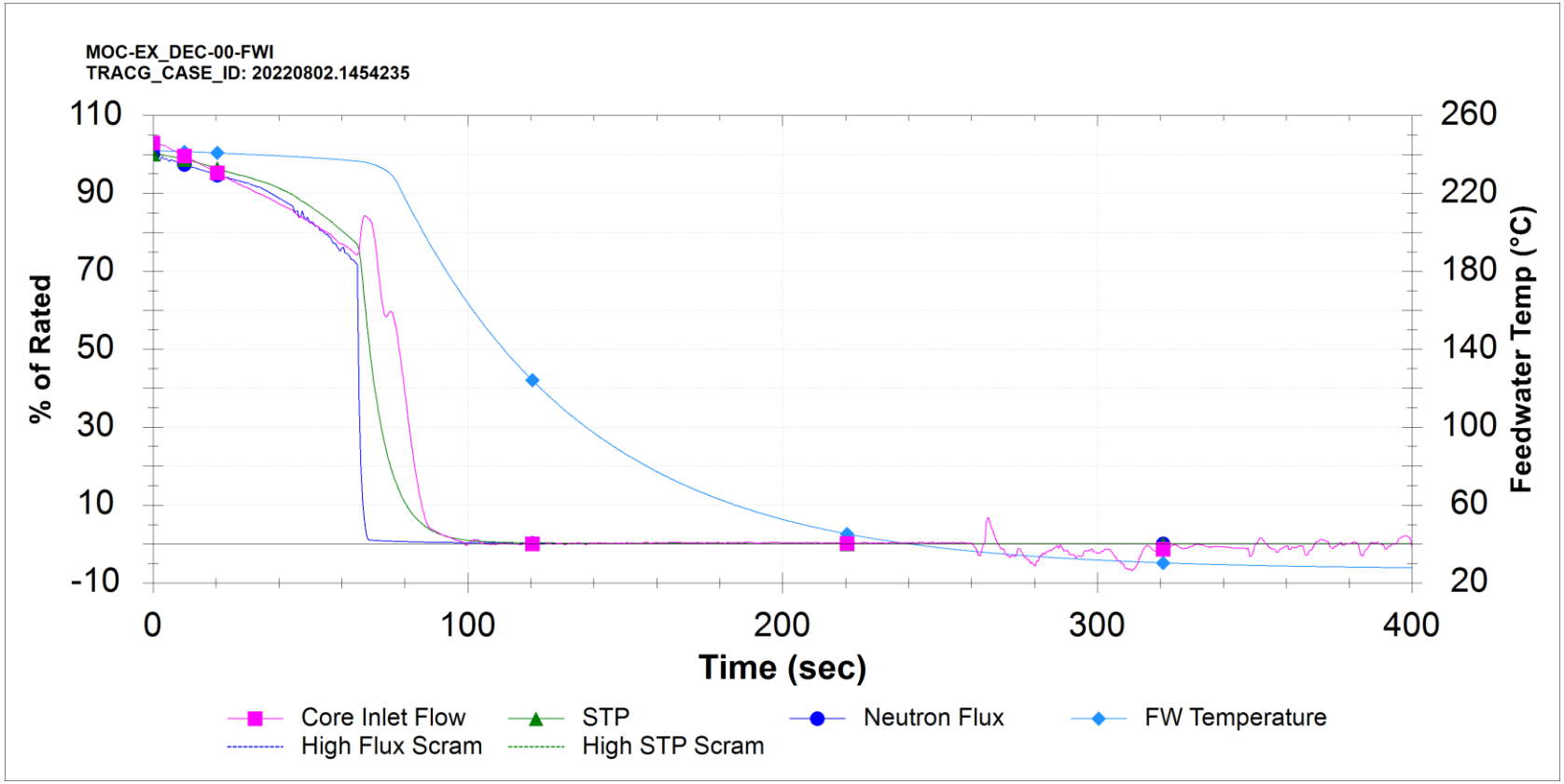


Figure 15.5-155: Feedwater Isolation (DEC)

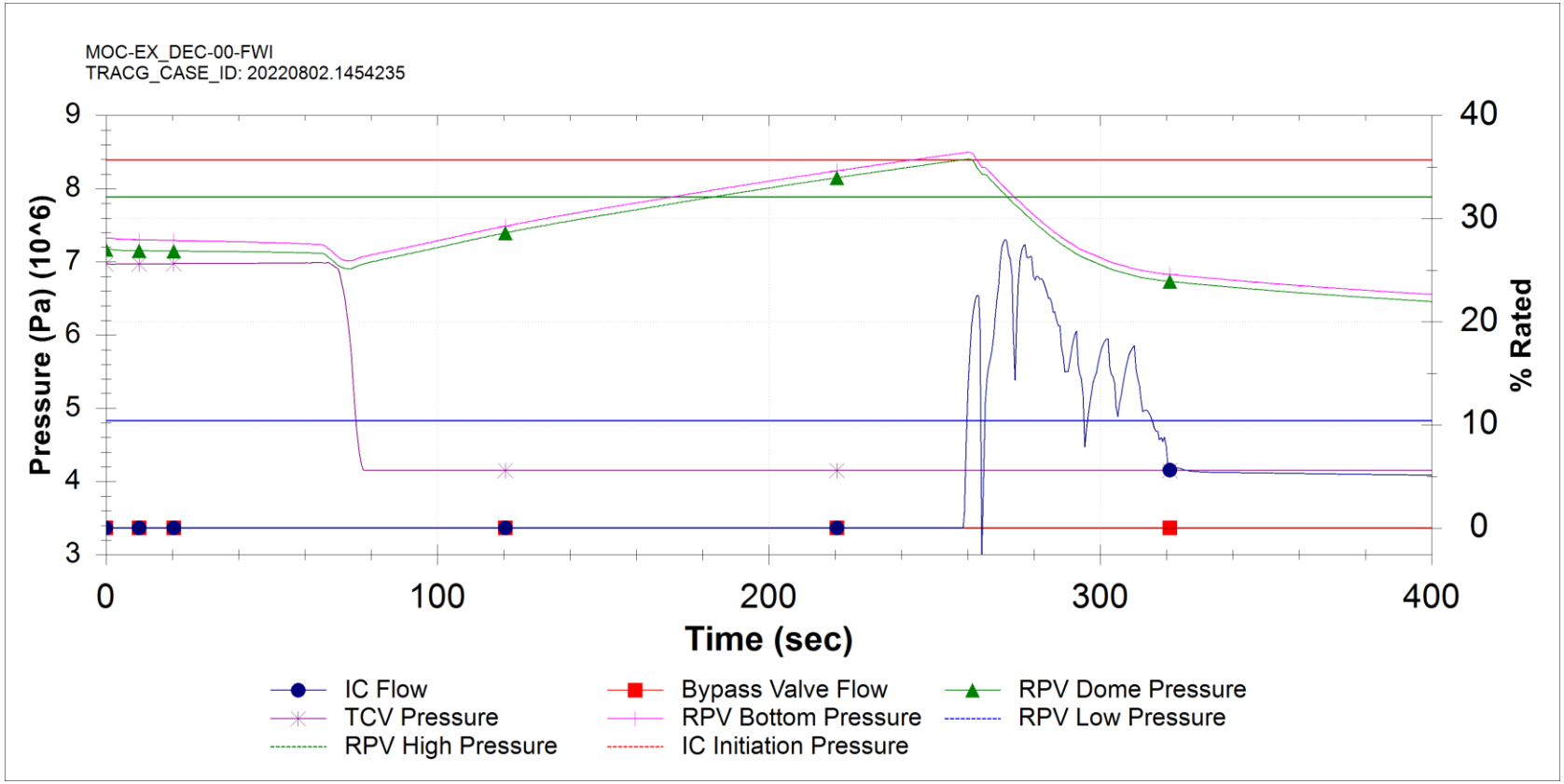


Figure 15.5-156: Feedwater Isolation (DEC)

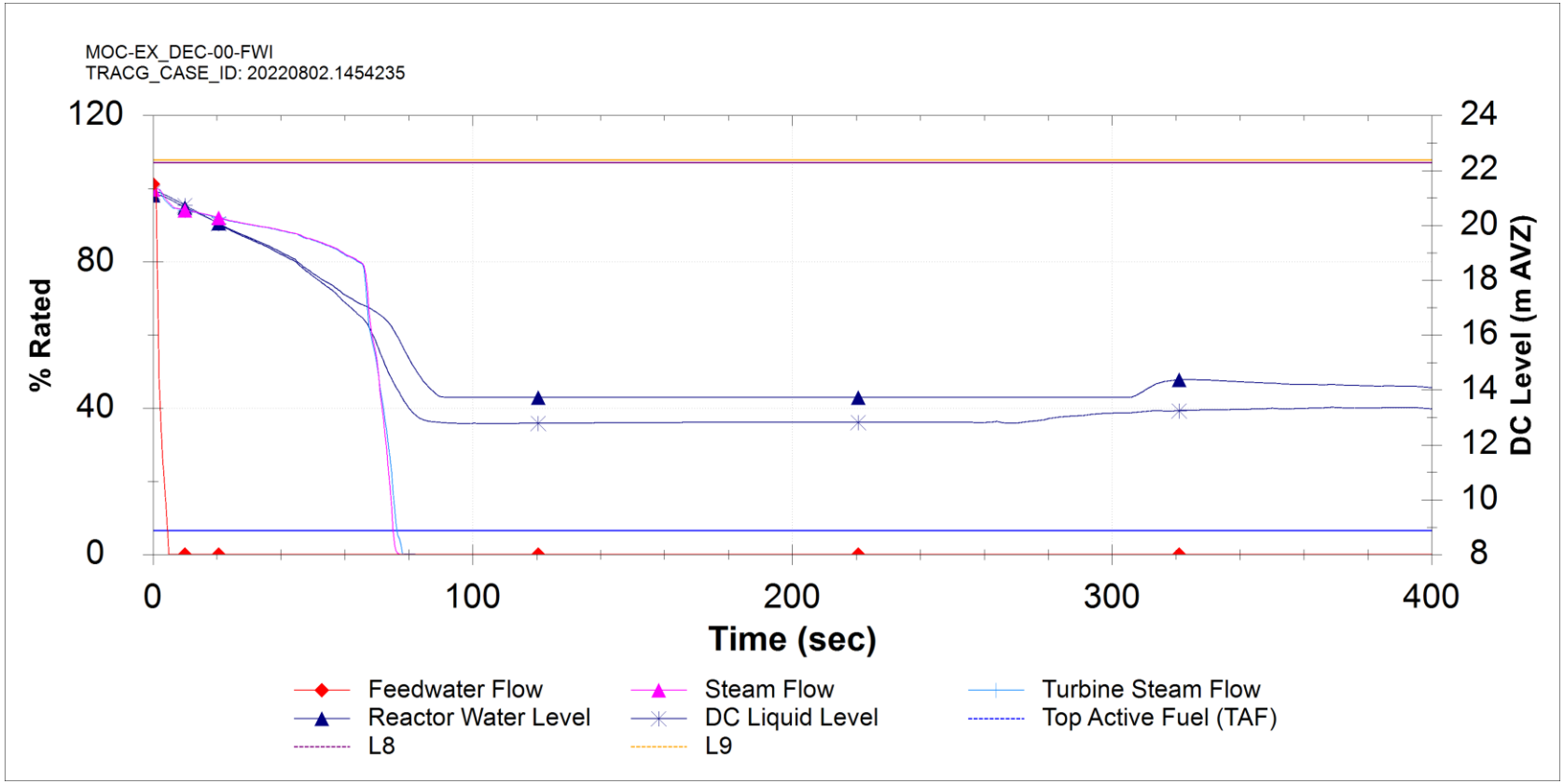


Figure 15.5-157: Feedwater Isolation (DEC)

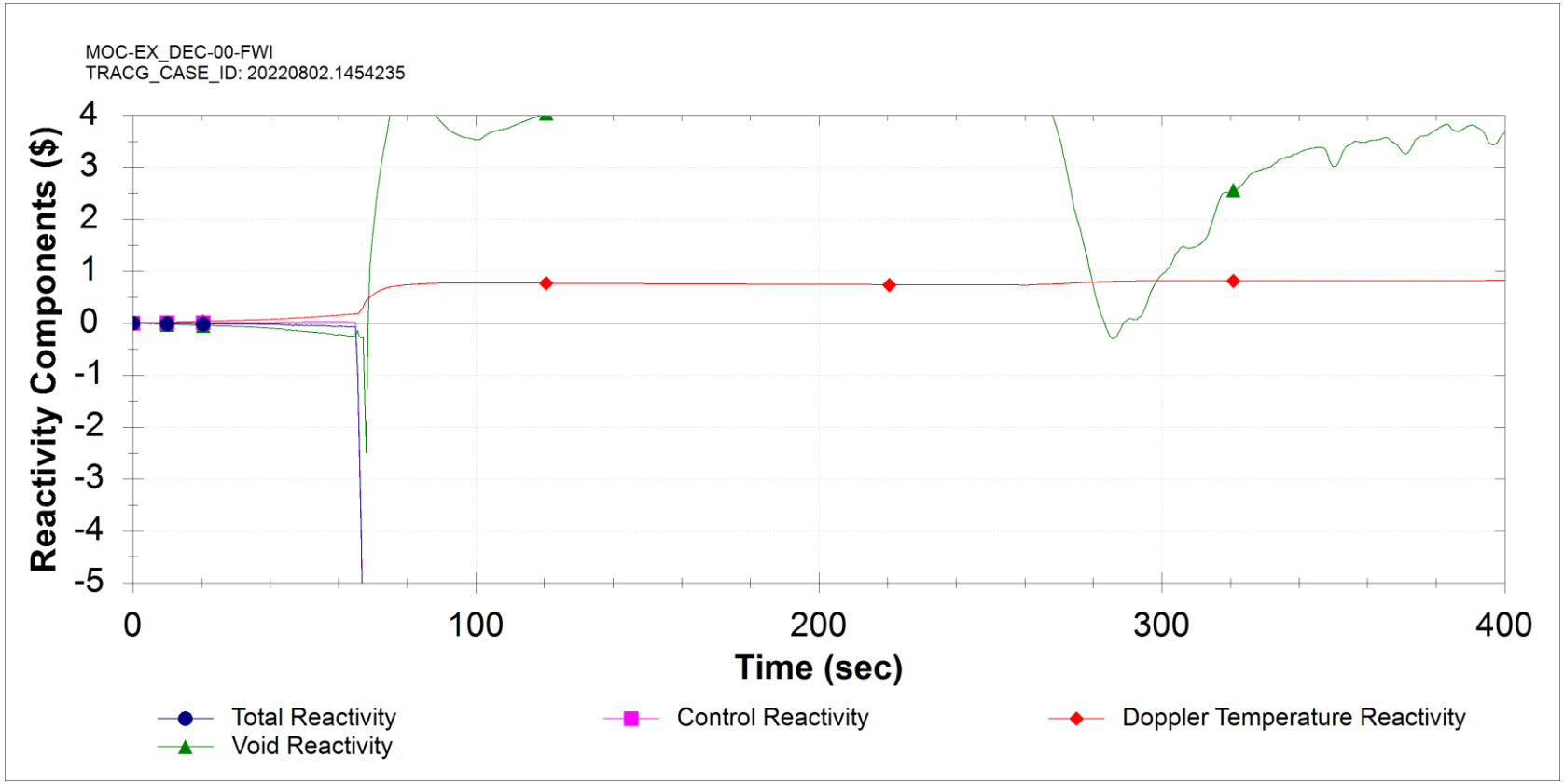


Figure 15.5-158: Feedwater Isolation (DEC)

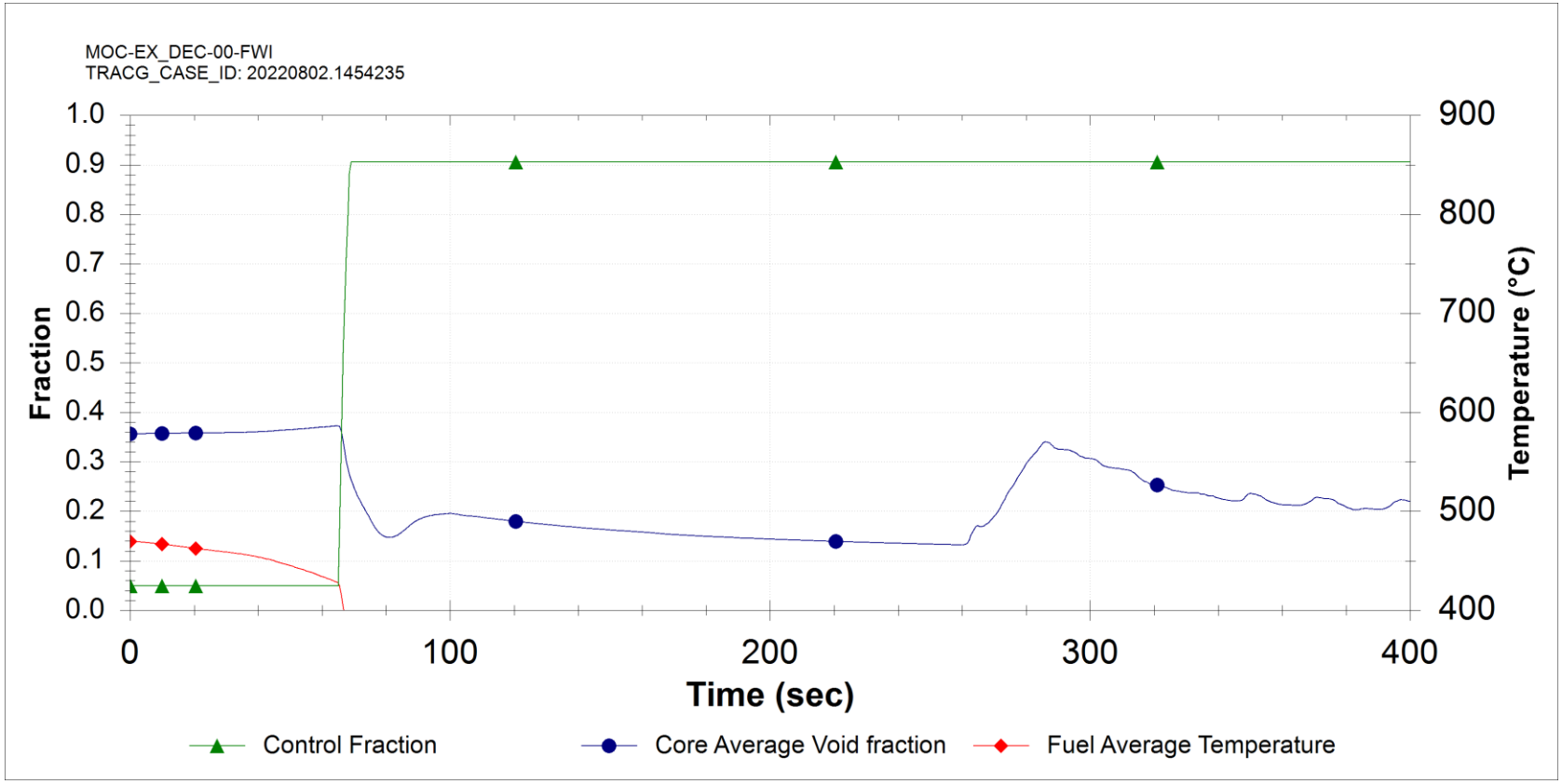


Figure 15.5-159: Feedwater Isolation (DEC)

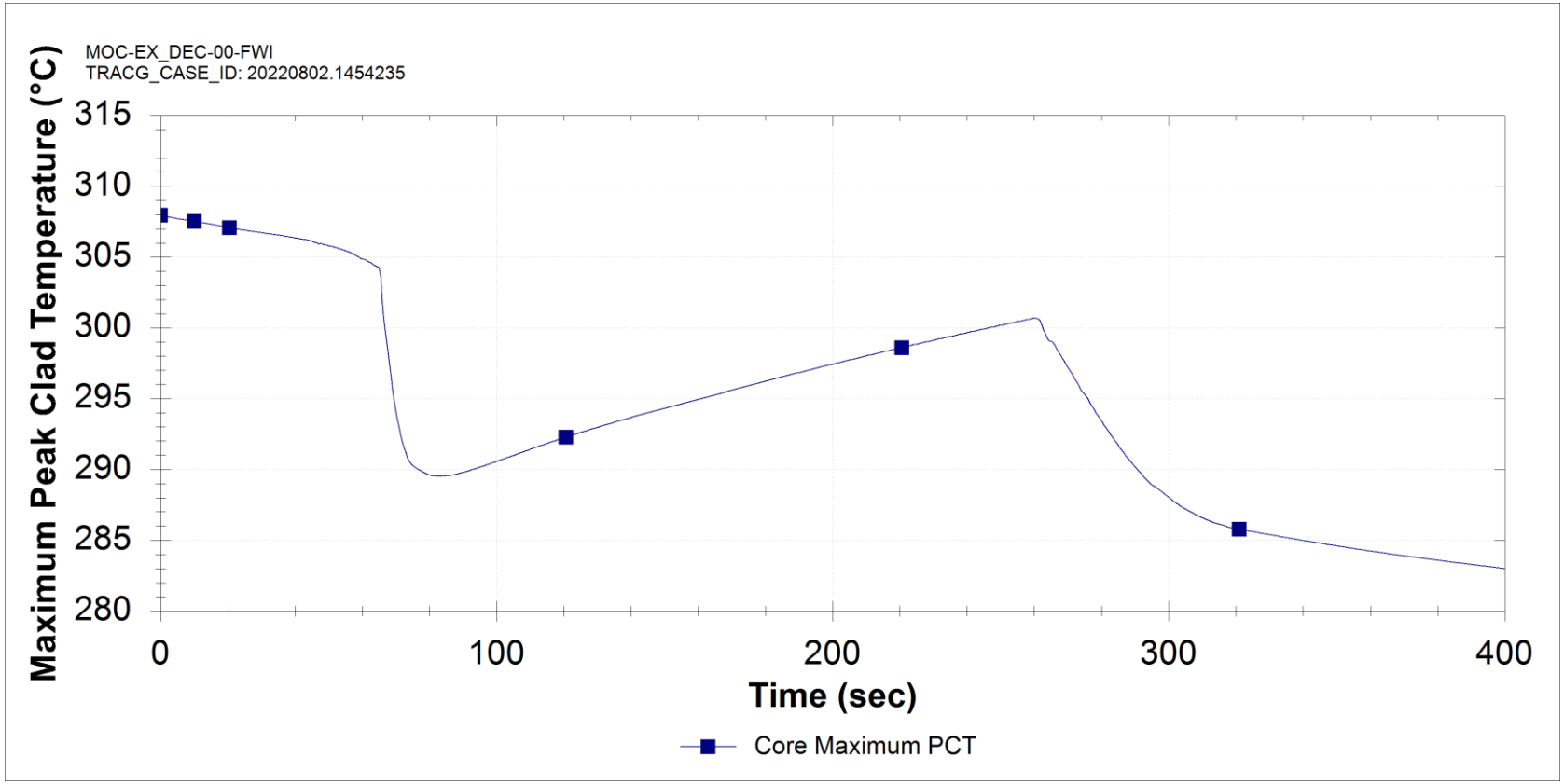


Figure 15.5-160: Feedwater Isolation (DEC)

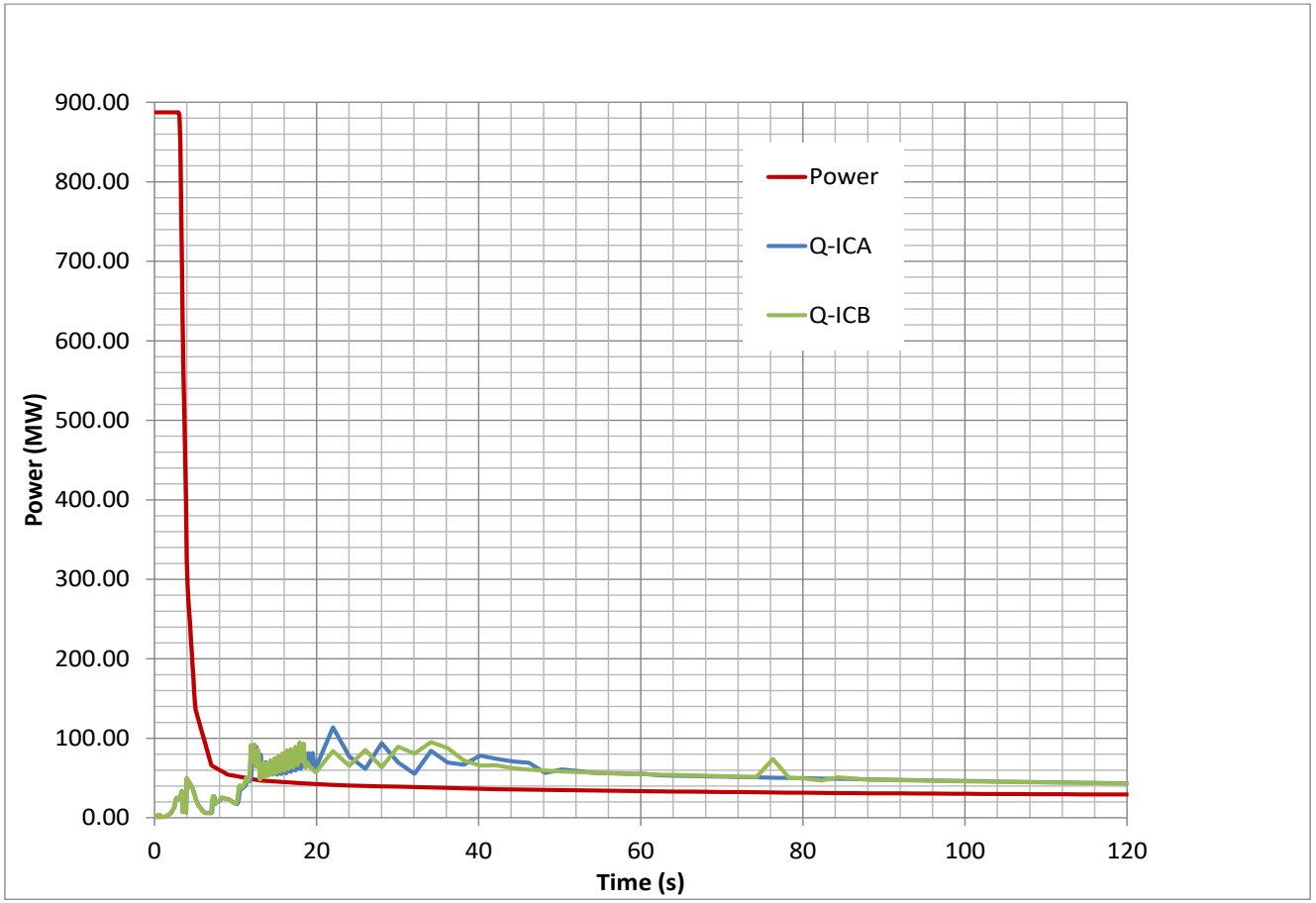


Figure 15.5-161: Reactor Power, Large FW Pipe Break, Conservative Case

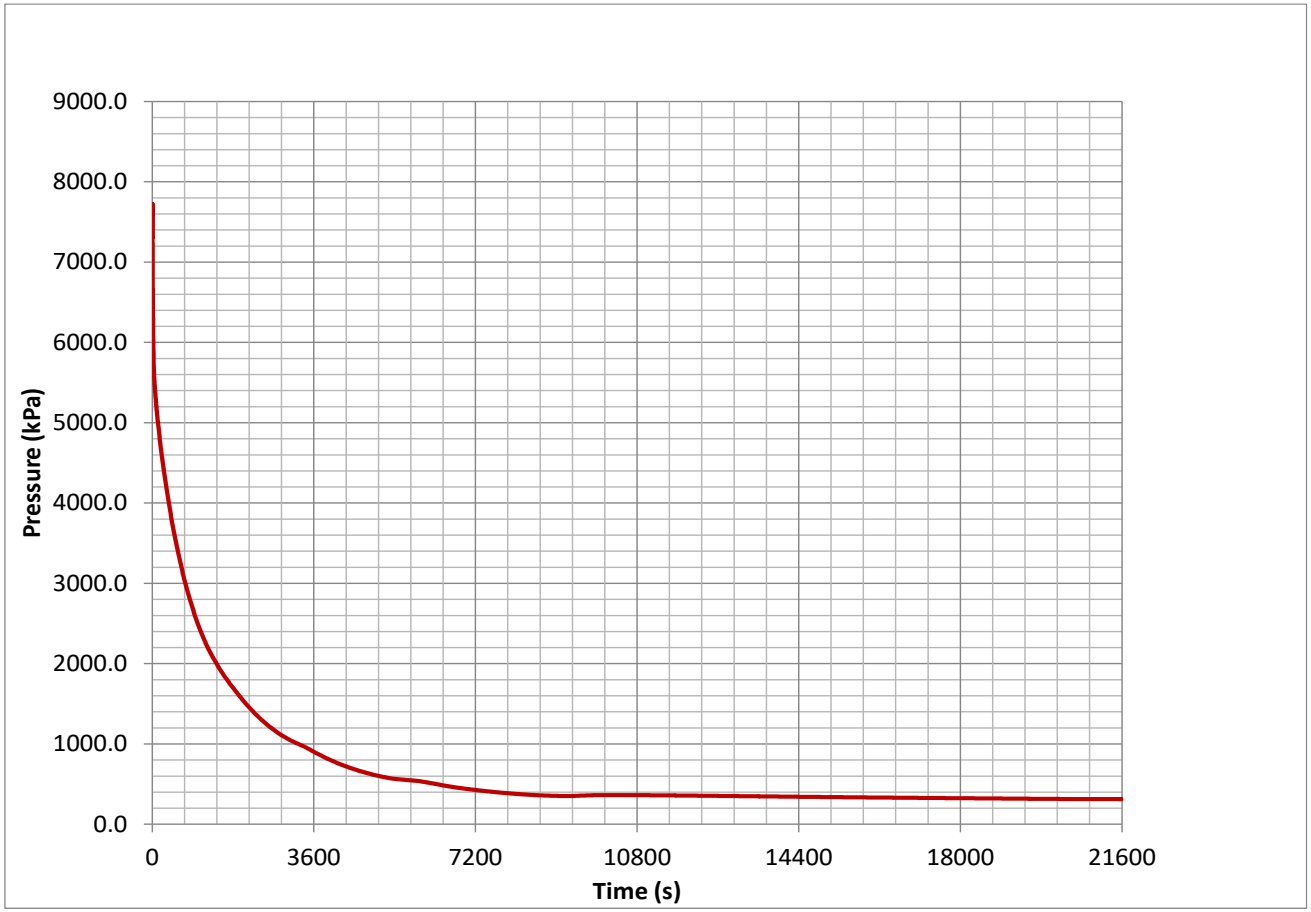


Figure 15.5-162: Reactor Pressure, Large FW Pipe Break, Conservative Case

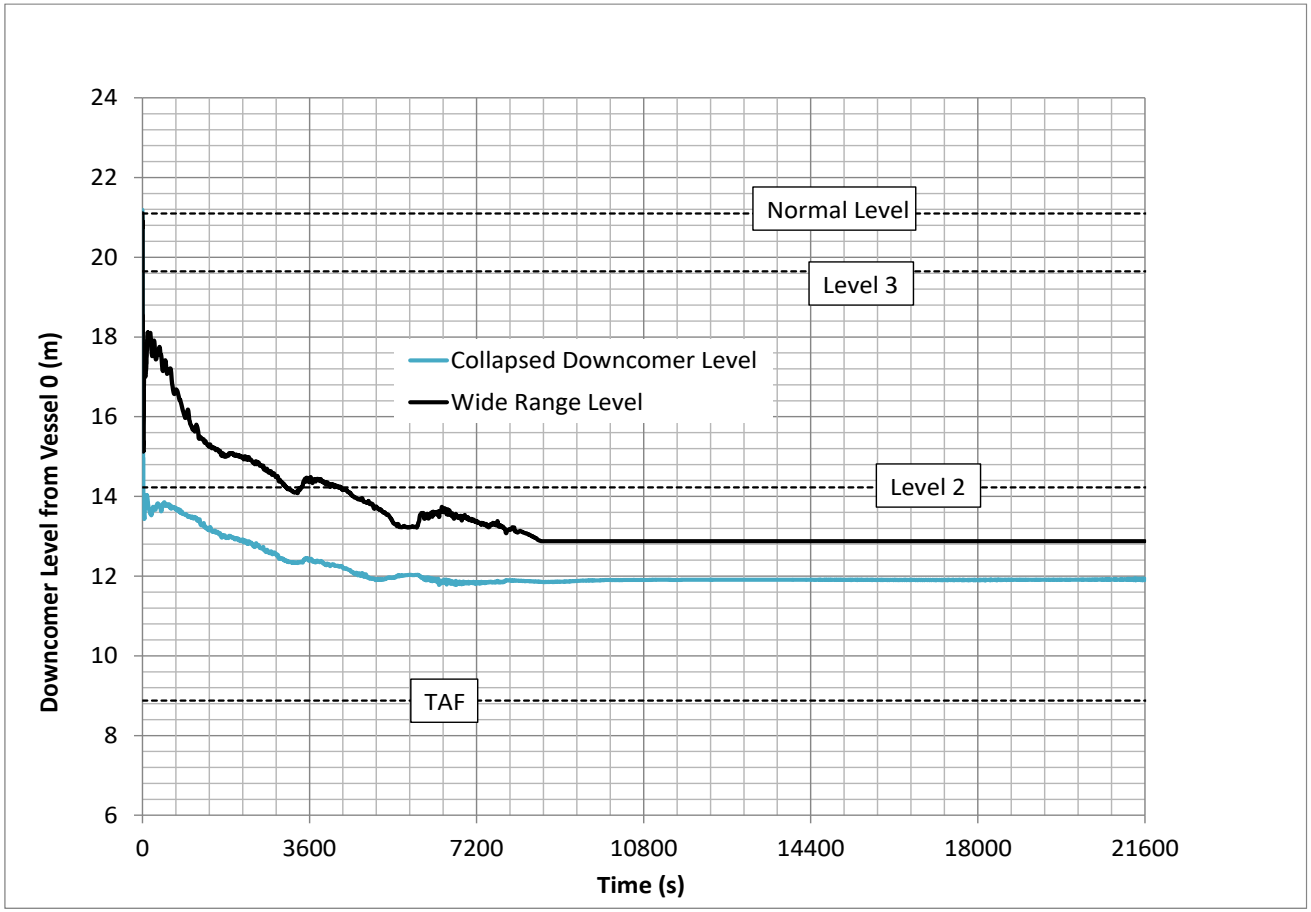


Figure 15.5-163: Reactor Water Level, Large FW Pipe Break, Conservative Case

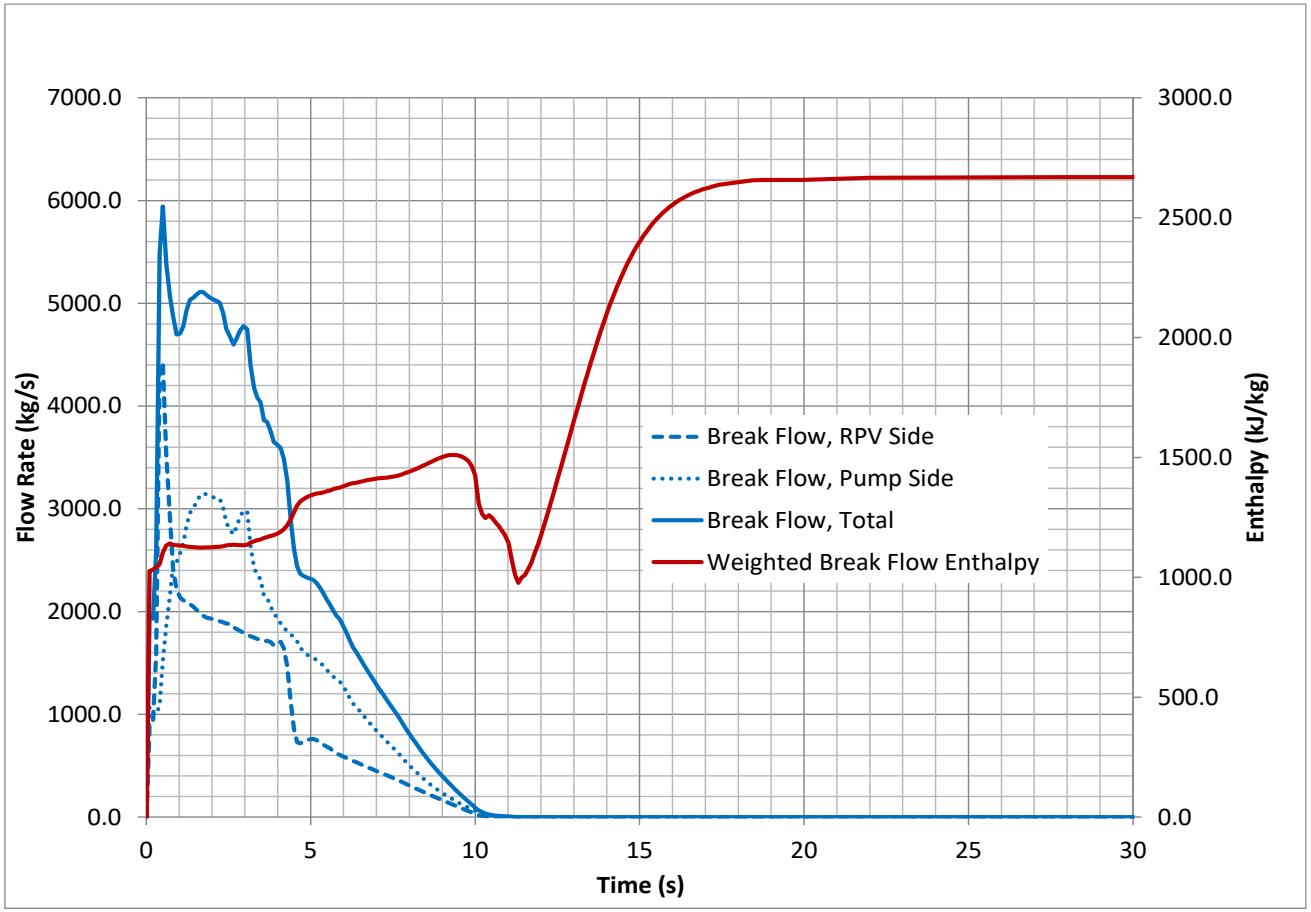


Figure 15.5-164: Break Flow Rate and Enthalpy, Large FW Pipe Break, Conservative Case

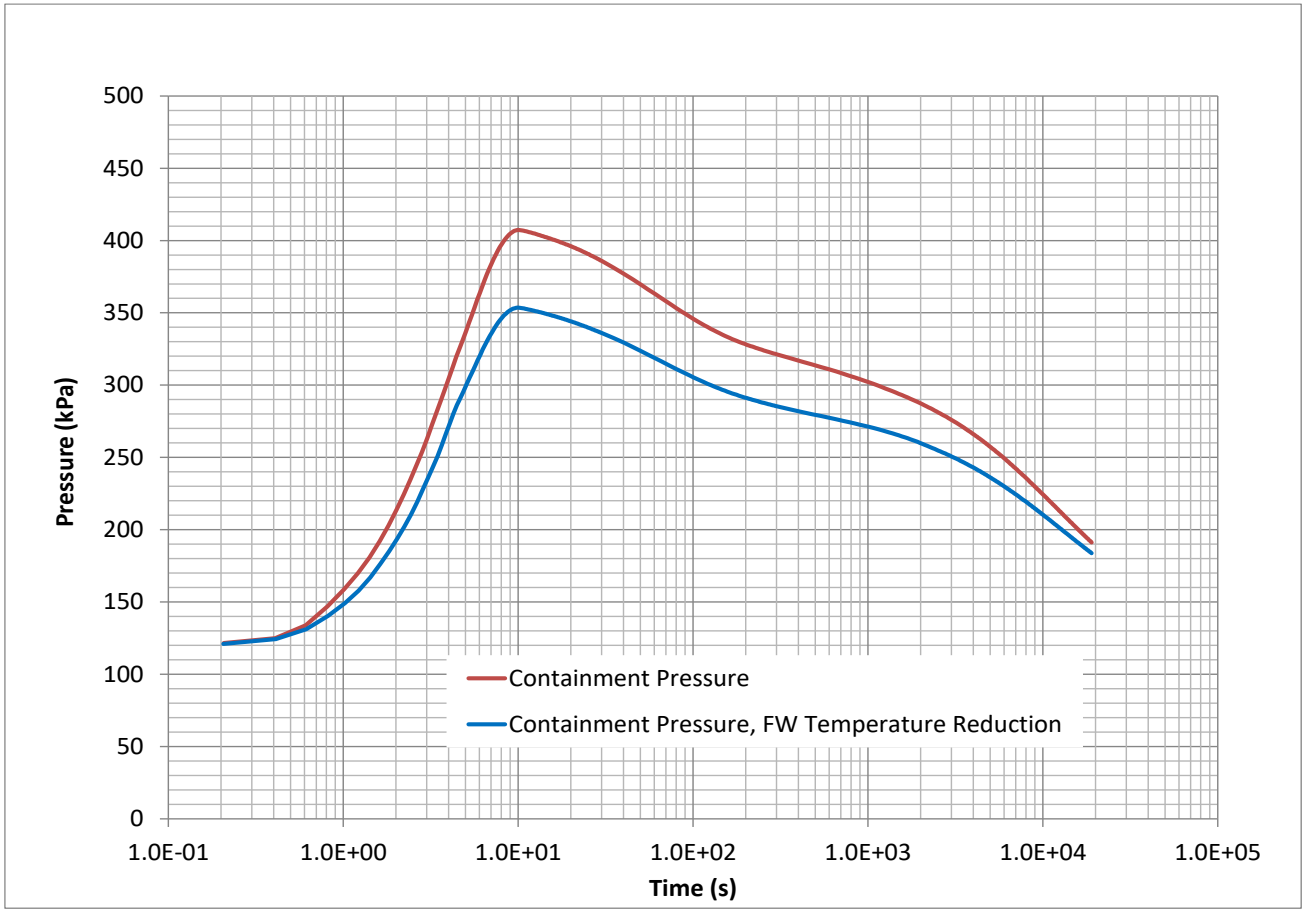


Figure 15.5-165: Containment Pressure, Large FW Pipe Break, Conservative Case

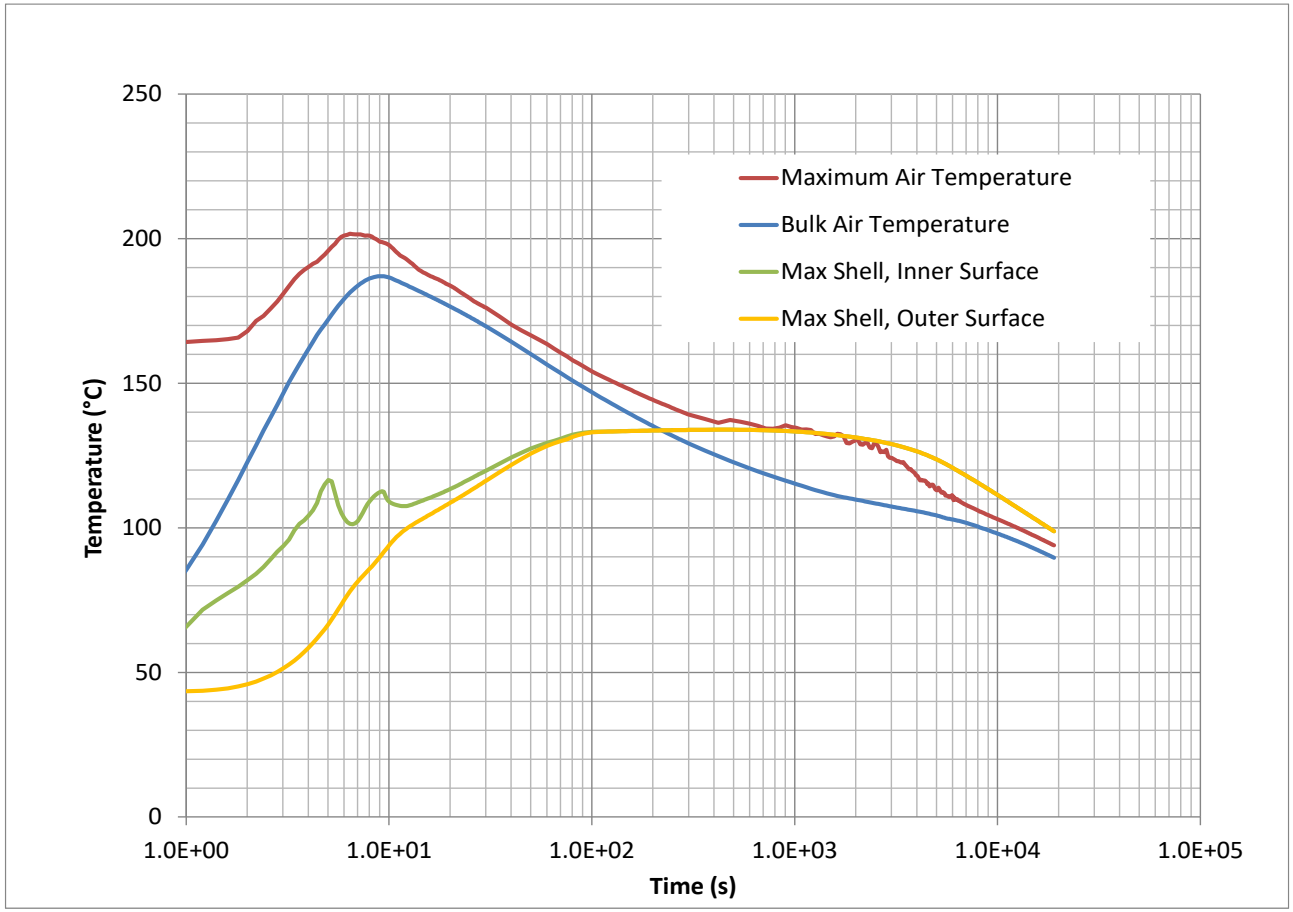


Figure 15.5-166: Containment Temperature, Large FW Pipe Break, Conservative Case

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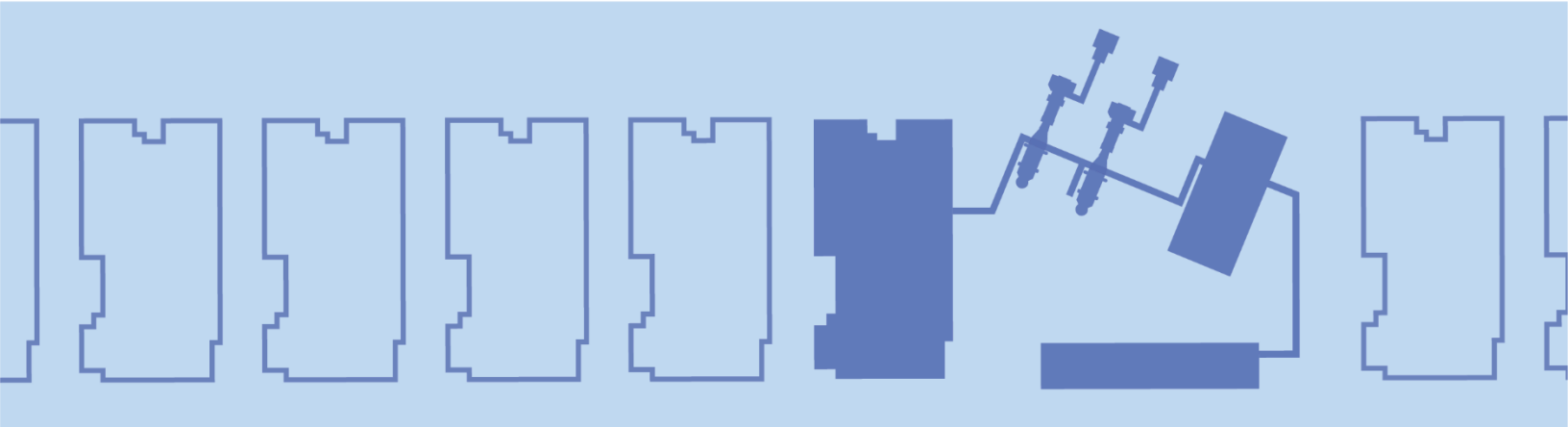
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Non-Proprietary Information

Chapter 15
Section 15.6
Safety Analyses (Part 4)



ACRONYM LIST

Acronym	Explanation
AC	Alternating Current
ADAMS	Agencywide Documents Access and Management System
ALWR	Advanced Light Water Reactor
ANS	American Nuclear Society
ANSI	American National Standards Institute
API	Application Programming Interface
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
BDBA	Beyond Design-Basis Accident
BE	Basic Event
BIS	Boron Injection System
BOC	Break Outside Containment
BOP	Balance of Plant
BWR	Boiling Water Reactor
CAFTA	Computer Aided Fault Tree Analysis
CB	Control Building
CCCG	Common Cause Component Group
CCDP	Conditional Core Damage Probability
CCF	Common Cause Failure
CCI	Common Cause Initiators
CDF	Core Damage Frequency
CET	Containment Event Tree
CRD	Control Rod Drive
CUW	Reactor Water Cleanup System
CWS	Circulating Water System
D-in-D	Defense-in-Depth
DBA	Design Basis Accident
DC	Direct Current
DCIS	Distributed Control and Information System
DEC	Design Extension Condition
DF	Decontamination Factor

Acronym	Explanation
DL	Defense Line
DSA	Deterministic Safety Analysis
EE	External Events
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
ES	Event Sequence
FDF	Fuel Damage Frequency
FLEX	Diverse and Flexible Coping Strategy
FMEA	Failure Modes and Effects Analysis
FSAR	Final Safety Analysis Report
FSER	Final Safety Evaluation Report
FTREX	Fault Tree Reliability Evaluation Expert
FV	Fussell-Vesely
FW	Feedwater
HCU	Hydraulic Control Unit
HEE	Human Error Event
HELB	High Energy Line Break
HEP	Human Error Probability
HRA	Human Reliability Analysis
HVS	Heating Ventilation and Cooling System
I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency
ICS	Isolation Condenser System
IE	Initiating Event
ISLOCA	Interfacing System LOCA
LOCA	Loss-of-Coolant Accident
LOPP	Loss-of-Preferred Power
LPSD	Low-Power and Shutdown
LRF	Large Release Frequency
LTR	Licensing Topical Report
LWR	Light Water Reactor
MAAP	Modular Accident Analysis Program

Acronym	Explanation
MCC	Motor Control Center
MCS	Minimal Cutset
MOV	Motor Operated Valve
NBS	Nuclear Boiler System
NEA	Nuclear Energy Agency
NPP	Nuclear Power Plant
PAU	Physical Analysis Unit
PCCS	Passive Containment Cooling System
PCV	Primary Containment Vessel
PCW	Plant Cooling Water System
PDS	Plant Damage States
PIE	Postulated Initiating Event
POS	Plant Operation State
PPS	Plant Pneumatic System
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
PSAR	Preliminary Safety Analysis Report
PSF	Performance Shaping Factors
QHO	Quantitative Health Objectives
RAW	Risk Achievement Worth
RB	Reactor Building
RES	NRC Office of Nuclear Regulatory Research
RG	Regulatory Guide
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
SA	Severe Accident
SAA	Severe Accident Analysis
SAM	Severe Accident Management
SDC	Shutdown Cooling System
SDG	Standby Diesel Generator
SPSA	Seismic Probabilistic Safety Assessment
SRM	Staff Requirements Memoranda

Acronym	Explanation
SSCs	Structures, Systems, and Components
TB	Turbine Building
URD	Utility Requirements Document
USNRC	United States Nuclear Regulatory Commission

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None.

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None.

15.6 Probabilistic Safety Assessment

[Reserved. To be addressed in CPA Part 2 or the operating license application, as necessary.]

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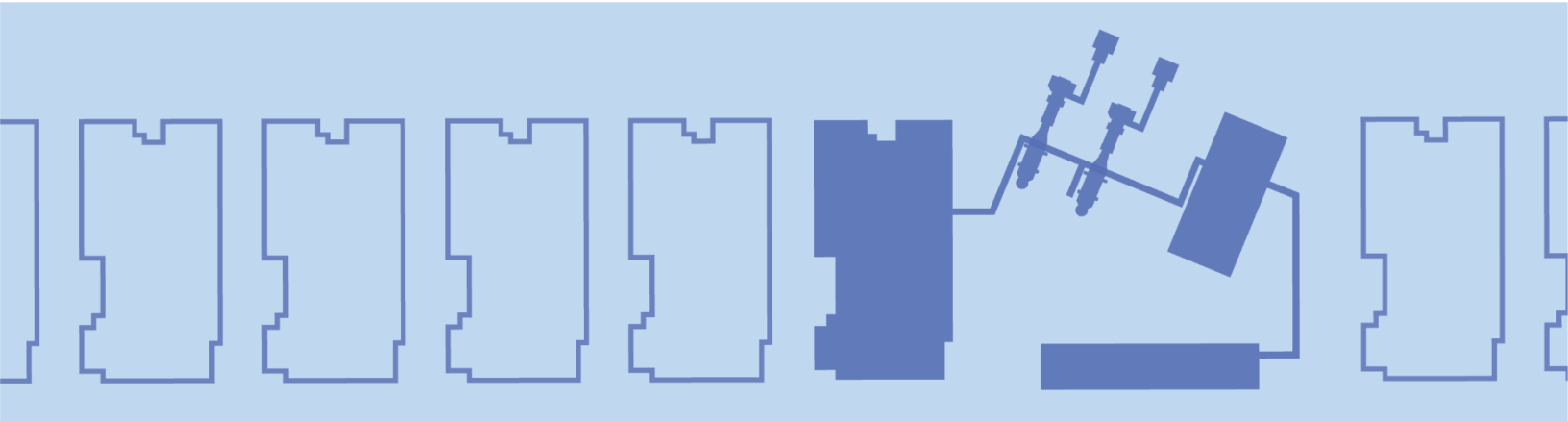
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Chapter 15
Section 15.7
Safety Analyses (Part 5)



ACRONYM LIST

Acronym	Explanation
ACRW	All Control Rod Withdrawal at Power
AOO	Anticipated Operational Occurrence
BL-AOO	Baseline Abnormal Operational Occurrence
BOC	Beginning of Cycle
CCF	Common Cause Failure
CDF	Core Damage Frequency
CN-DBA	Conservative Design Basis Analysis
CRD	Control Rod Drive
D-in-D	Defense-in-Depth
DBA	Design Basis Accident
DCF	Dose Conversion Factor
DEC	Design Extension Condition
DF	Decontamination Factor
DSA	Deterministic Safety Analysis
EAB	Exclusion Area Boundary
ECCS	Emergency Core Cooling System
EHE	External Hazard Evaluation
EOR	End of Rated Cycle
FFHE	Functional Failure Hazard Evaluation
FHA	Fuel Handling Accident
FMEA	Failure Modes and Effects Analysis
FSF	Fundamental Safety Function
FV	Fussell-Vesely
FWFI	Feedwater Flow Increase
FWPT	Feedwater Pump Trip
IAEA	International Atomic Energy Agency
ICRW	Inadvertent Control Rod Withdrawal at Power - Single Rod
ICS	Isolation Condenser System
IE	Initiating Event
IHE	Internal Hazard Evaluation
LFWH	Loss of Feedwater Heating

Acronym	Explanation
LOCA	Loss-of-Coolant Accident
LOCV	Loss of Condenser Vacuum
LOPP	Loss-of-Preferred Power
LPZ	Low Population Zone
LRF	Large Release Frequency
LR-TT	Load Reduction – Turbine Trip
MOC	Middle of Cycle
MSRIV	Main Steam Reactor Isolation Valve
NBR	Nuclear Boiler Rated
ODE	Ordinary Differential Equation
PA	Postulated Accident
PI-AOO	Pressure Increase-Abnormal Operation Occurrence
PIE	Postulated Initiating Event
PSA	Probabilistic Safety Assessment
PSAR	Preliminary Safety Analysis Report
QHO	Quantitative Health Objectives
RAW	Risk Achievement Worth
RB	Reactor Building
RCPB	Reactor Coolant Pressure Boundary
RI-AOO	Reactivity Increase-Abnormal Operational Occurrence
RPF	Radial Peaking Factor
SDC	Shutdown Cooling System
SMR	Small Modular Reactor
SS-DBA	Safe-Shutdown Design Basis Analysis
STP	Simulated Thermal Power
USNRC	U.S. Nuclear Regulatory Commission
ΔCPR/ICPR	Delta Critical Power Ratio Over Initial Critical Power Ratio

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15.7 Results of the Deterministic Safety Analyses and Probabilistic Safety Assessment

The DSA results for the bounding BWRX-300 events meet the acceptance criteria in Table 15.3-1 and Table 15.3-2. Results Summary for the DSA is provided in the following tables:

Table 15.7-1 Results Summary of AOO Events

Table 15.7-2 Results Summary of the DBA and DEC Events Non-LOCA

Table 15.7-3 Results Summary of DBA and DEC Events for LOCA

Table 15.7-4 Fuel Handling Accident Dose Consequences

Table 15.7-5 BWRX-300 Main Steam Line Break Outside Containment Accident Dose Consequences

Table 15.7-6 BWRX-300 Feedwater Line Break Outside Containment Accident Dose Consequences

Table 15.7-7 BWRX-300 Isolation Condenser System Line Break Outside Containment Accident Dose Consequences

Table 15.7-8 BWRX-300 Instrument Line Break Outside Containment Accident Dose Consequences

The results for low power and shutdown DSA will be provided in the FSAR. Preliminary results for the beyond design basis accidents evaluated in the PSA are provided in Table 15.7-9 and Table 15.7-10. Final results will be provided in the FSAR.

Implementation of the D-in-D concept ensures multiple, independent layers of protection against unacceptable radiation releases. None of the bounding AOOs, DBAs, or DEC Events Without Core Damage analyzed approach the regulatory limits for radioactive releases.

15.7.1 Results of Boiling Water Reactor Stability Analysis

The analysis of normal operations for BWR stability is described in Appendix 4A and Subsection 15.5.2.

15.7.2 Results of Analysis of Anticipated Operational Occurrences

The analysis of AOOs is described in Subsection 15.5.3. The resulting maximum neutron flux, maximum dome pressure, maximum RPV bottom pressure, maximum simulated thermal power, and ΔCPR/ICPR for AOOs is provided in Table 15.7-1.

15.7.3 Results of Analysis of Design Basis Accidents

The analysis of DBAs is described in Subsection 15.5.4. The summary results of maximum neutron flux, maximum dome pressure, maximum RPV bottom pressure, maximum simulated thermal power, and peak clad temperature from non-LOCA DBAs are provided in Table 15.7-2.

The LOCA DBA summary results for peak cladding temperature, peak containment pressure, and peak containment shell temperature are provided in Table 15.7-3.

15.7.4 Results of Analysis of Design Extension Conditions without Core Damage

The analysis of DEC events without core damage is described in Subsection 15.5.5. The summary results of maximum neutron flux, maximum dome pressure, maximum RPV bottom pressure, maximum simulated thermal power, and peak clad temperature from non-LOCA DECs are provided in Table 15.7-2.

See Subsection 15.2.4.6.4 for a discussion of LOCA DEC results.

15.7.5 Results of Analysis of Design Extension Conditions with Core Damage

The analysis of DECs with core damage are addressed in Subsection 15.6.5. Severe accidents will be described in the CPA Part 2.

15.7.6 Results of Analysis of Postulated Initiating Events and Accident Scenarios Associated with the Fuel Pool

The analysis description of those postulated events and accident scenarios associated with the fuel pool will be described in the CPA Part 2.

15.7.7 Results of Analysis of Fuel Handling Accident

The analysis of FHA is described in Subsection 15.5.8. The resultant dose consequences from an FHA are provided in Table 15.7-4.

15.7.8 Results of Analysis of LOCA Breaks Outside Containment

LOCA from pipe breaks outside containment described in Subsection 15.5.9.1 result in normal coolant isotopic concentration releases that are within allowable limits.

15.7.8.1 Results of Main Steam Line Break Outside Containment

The analysis of a MSL break outside containment is described in Subsection 15.5.9.1.1 and the results are provided in Table 15.7-5.

15.7.8.2 Results of Large Feedwater Pipe Break Outside Containment

The analysis of a large feedwater pipe break outside containment is described in Subsection 15.5.9.1.2, and the results are provided in Table 15.7-6.

15.7.8.3 Results of Large Isolation Condenser Pipe Break Outside Containment

The analysis of a large ICS pipe breaks outside containment is described in Subsection 15.5.9.1.4, and the results are provided in Table 15.7-7.

15.7.8.4 Results of Small Breaks Outside Containment

The analysis of a small pipe break outside containment is described in Subsection 15.5.9.1.5, and the results are provided in Table 15.7-8.

15.7.9 Results of Probabilistic Safety Assessment

The Level 1 and Level 2 PSA are described in Subsection 15.6.3 and Subsection 15.6.4, respectively. The general PSA approach and the insights and applications are described in Subsection 15.6.1.3 and Subsection 15.6.10. Preliminary PSA results for CDF and LRF are provided in Table 15.7-9 and Table 15.7-10, respectively. Final CDF and LRF results will be provided in the FSAR.

Table 15.7-1: Results Summary of AOO Events

Description	Exposure	Max. Neutron Flux, % NBR	Max. Dome Pressure, MPa (psia)	Max. Vessel Bottom Pressure, MPa (psia)	Max. Simulated Thermal Power, % NBR	ΔCPR/ ICPR
Decrease in Core Coolant Temperature AOOs						
LFWH AOO (Subsection 15.5.3.1.1)	MOC	101.1	7.17 (1040.1)	7.33 (1062.6)	100.4	0.0341
Pressure Increase AOOs						
LR-TT AOO (Subsection 15.5.3.2.1)	BOC	100.0	7.55 (1094.6)	7.70 (1116.3)	100.0	0.0583
1MSRIVC AOO (Subsection 15.5.3.2.2)	EOR	110.7	7.47 (1083.2)	7.61 (1103.1)	100.3	0.0631
LOCV AOO (Subsection 15.5.3.2.3)	BOC	100.0	7.55 (1094.6)	7.70 (1116.3)	100.0	0.0583
LOPP AOO (Subsection 15.5.3.2.4)	BOC	100.0	7.55 (1094.6)	7.70 (1116.2)	100.0	0.0495
Inventory Reduction AOOs						
FWPT AOO (Subsection 15.5.3.3.1)	EOR	100.0	7.17 (1039.7)	7.32 (1062.1)	100.0	0.0086
Increase in Reactor Coolant Inventory AOOs						
Inadvertent Isolation Condenser Initiation One Train (Subsection 15.5.3.4.1)	MOC	116.8	7.17 (1039.7)	7.32 (1062.1)	100.6	0.0464

Legend:

NBR: Nuclear Boiler Rated

Table 15.7-2: Results Summary of the DBA and DEC Events – Non-LOCA

Description	Exposure	Max. Neutron Flux, % NBR	Max. Dome Pressure, MPa (psia)	Max. Vessel Bottom Pressure, MPa (psia)	Max. Simulated Thermal Power, % NBR	PCT °C (°F)
Decrease in Reactor Coolant Temperature Event DBA						
LFWH DBA (Subsection 15.5.4.1.1)	MOC	119.1	7.17 (1039.7) Note 1	7.32 (1062.1) Note 1	115.2	308.0 (586.3)
Increase in Reactor Pressure Events DBAs						
LR DBA Note 2 (Subsection 15.5.4.2.1)	MOC	544.3	8.69 (1259.8)	8.86 (1285.1)	111.8	511.8 (953.3)
LOPP DBA (Subsection 15.5.4.2.2)	EOR	151.0	8.61 (1249.2)	8.73 (1266.1)	103.5	312.4 (594.3)
RPCD DBA (Subsection 15.5.4.2.3)	EOR	151.4	8.70 (1262.3)	8.89 (1288.9)	103.5	312.6 (594.7)
MSRIVC-FWIV DBA (Subsection 15.5.4.2.4)	EOR	158.9	8.61 (1249.4)	8.73 (1266.3)	103.7	312.7 (594.9)
Increase in Reactor Coolant Inventory DBAs						
FWFI – All Pumps (Subsection 15.5.4.3.1)	MOC	123.3	7.93 (1150.0) (Note 1)	8.09 (1173.6) (Note 1)	115.5	315.5 (599.9)
Inadvertent Isolation Condenser Initiation – All Trains (Subsection 15.5.4.3.2)	MOC	114.3	7.17 (1039.7)	7.32 (1062.1)	100.0	308.0 (586.3)
Decrease in Reactor Coolant Inventory DBAs						
LOFW DBA (Subsection 15.5.4.4.1)	MOC	100.0	7.17 (1039.7)	7.32 (1062.1)	100.0	308.0 (586.3)
RPCO DBA (Subsection 15.5.4.4.2)	MOC	100.0	7.17 (1039.7) Note 1	7.32 (1062.1) Note 1	100.0	302.8 (577.0)

Description	Exposure	Max. Neutron Flux, % NBR	Max. Dome Pressure, MPa (psia)	Max. Vessel Bottom Pressure, MPa (psia)	Max. Simulated Thermal Power, % NBR	PCT °C (°F)
Analysis of Design Extension Conditions Without Core Damage						
Pressure Increase DECs						
1MSRIVC DEC (Subsection 15.5.5.2.1)	EOR	208.9	11.19 (1623.2)	11.33 (1643.3)	130.1	727.1 (1340.8)
Complex Sequence LR DEC (Subsection 15.5.5.2.2)	EOR	118.4	8.03 (1164.3)	8.17 (1184.5)	100.3	309.2 (588.6)
LOCV DEC (Subsection 15.5.5.2.3)	EOR	245.0	9.98 (1447.3)	10.12 (1467.5)	124.5	332.9 (631.2)
LOPP DEC (Subsection 15.5.5.2.4)	EOR	250.1	11.14 (1615.6)	11.28 (1635.5)	130.4	328.9 (624.0)
Reactivity and Power Distribution Anomalies – DECs						
CCF ACRW (Subsection 15.5.5.3.1)	MOC	123.8	7.52 (1091.0)	7.68 (1113.3)	115.4	314.0 (597.2)
ICRW (Subsection 15.5.5.3.2)	BOC	111.8	7.23 (1048.5)	7.38 (1070.9)	111.8	307.5 (585.5)
Decrease in Reactor Coolant Inventory – DEC						
FW Isolation DEC (Subsection 15.5.5.4)	MOC	100.0	8.40 (1218.5)	8.50 (1232.1)	100.0	308.0 (586.3)

Notes:

1. The simulation of this event is ended before the RPV pressure increase resulted in an ICS train initiation.
2. Results are from a bounding case with combined conservatisms.

Table 15.7-3: Results Summary of DBA Events For LOCA

Parameter	Value
Results for Main Steam Pipe Break Inside Containment, Conservative Case (Subsection 15.5.4.5.1)	
Peak cladding temperature	Less than normal operating temperature
Peak containment pressure	423 kPa
Peak containment shell temperature	134 °C
Results for Feedwater Pipe Break Inside Containment, Conservative Case (Subsection 15.5.4.5.2)	
Peak cladding temperature	Less than normal operating temperature
Peak containment pressure	407 kPa
Peak containment shell temperature	134 °C
Results for Small Steam Pipe Break Inside Containment, Conservative Case (Subsection 15.5.4.5.4)	
Peak cladding temperature	Less than normal operating temperature
Peak containment pressure	191 kPa
Peak containment shell temperature	125 °C
Results for Small Liquid Pipe Break Inside Containment, Conservative Case (Subsection 15.5.4.5.4)	
Peak cladding temperature	Less than normal operating temperature
Peak containment pressure	191 kPa
Peak containment shell temperature	110 °C

Table 15.7-4: Fuel Handling Accident Dose Consequences

To be provided in CPA Part 2.

Table 15.7-5: BWRX-300 Main Steam Line Break Outside Containment Accident Dose Consequences

To be provided in CPA Part 2.

Table 15.7-6: BWRX-300 Feedwater Line Break Outside Containment Accident Dose Consequences

To be provided in CPA Part 2.

Table 15.7-7: BWRX-300 Isolation Condenser System Line Break Outside Containment Accident Dose Consequences

To be provided in CPA Part 2.

Table 15.7-8: BWRX-300 Instrument Line Break Outside Containment Accident Dose Consequences

To be provided in CPA Part 2.

Table 15.7-9: Core Damage Frequency Preliminary Results

(Final Results Provided in FSAR)

To be provided in CPA Part 2 or the FSAR.

Table 15.7-10: Large Release Frequency Preliminary Results

(Final Results Provided in FSAR)

To be provided in CPA Part 2 or the FSAR.

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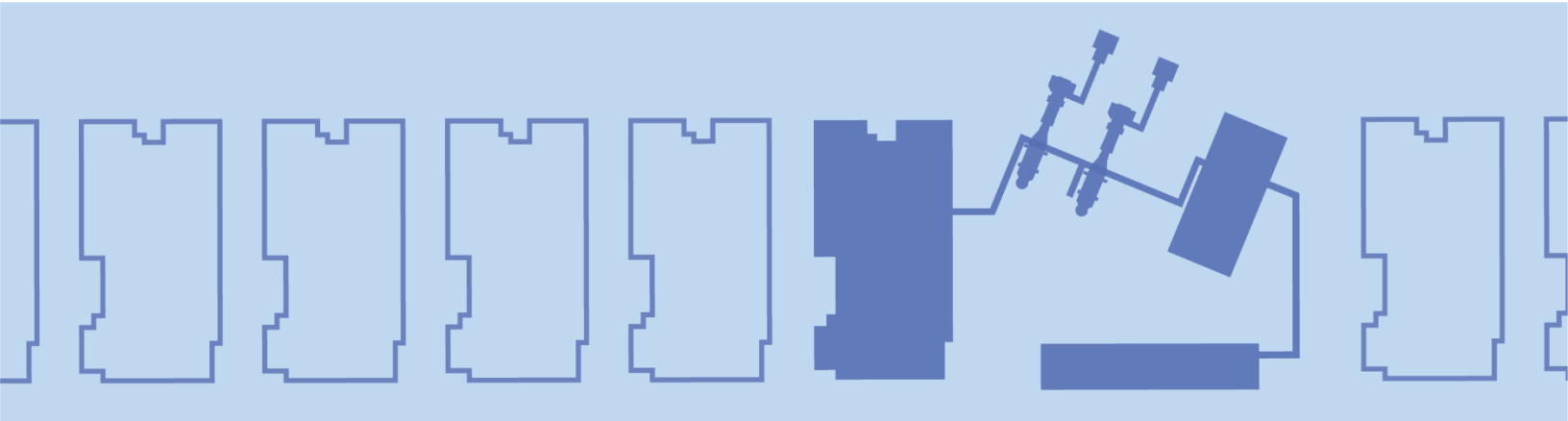
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Chapter 15
Appendix 15A & 15B
Safety Analyses



ACRONYM LIST

Acronym	Explanation
ACRW	All Control Rod Withdrawal at Power
AOO	Anticipated Operational Occurrence
BL-AOO	Baseline Abnormal Operational Occurrence
BOC	Beginning of Cycle
CCF	Common Cause Failure
CDF	Core Damage Frequency
CN-DBA	Conservative Design Basis Analysis
CRD	Control Rod Drive
D-in-D	Defense-in-Depth
DBA	Design Basis Accident
DCF	Dose Conversion Factor
DEC	Design Extension Condition
DF	Decontamination Factor
DSA	Deterministic Safety Analysis
EAB	Exclusion Area Boundary
ECCS	Emergency Core Cooling System
EHE	External Hazard Evaluation
EOR	End of Rated Cycle
FFHE	Functional Failure Hazard Evaluation
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FMEA	Failure Modes and Effects Analysis
FSF	Fundamental Safety Function
FV	Fussell-Vesely
FWFI	Feedwater Flow Increase
FWPT	Feedwater Pump Trip
IAEA	International Atomic Energy Agency
ICRW	Inadvertent Control Rod Withdrawal at Power - Single Rod
ICS	Isolation Condenser System
IE	Initiating Event
IHE	Internal Hazard Evaluation
LFWH	Loss of Feedwater Heating

Acronym	Explanation
LOCA	Loss-of-Coolant Accident
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LOPP	Loss-of-Preferred Power
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MSRIV	Main Steam Reactor Isolation Valve
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ODE	Ordinary Differential Equation
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PSAR	Preliminary Safety Analysis Report
QHO	Quantitative Health Objectives
RAW	Risk Achievement Worth
RB	Reactor Building
RCPB	Reactor Coolant Pressure Boundary
RI-AOO	Reactivity Increase-Abnormal Operational Occurrence
RPF	Radial Peaking Factor
SDC	Shutdown Cooling System
SMR	Small Modular Reactor
SS-DBA	Safe-Shutdown Design Basis Analysis
STP	Simulated Thermal Power
USNRC	U.S. Nuclear Regulatory Commission
ΔCPR/ICPR	Delta Critical Power Ratio Over Initial Critical Power Ratio

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None.

15.0A APPENDIX 15A: PRACTICAL ELIMINATION CLAIMS AND PROVISIONS

[Reserved. To be addressed in CPA Part 2 or the operating license application, as necessary.]

15.0B APPENDIX 15B: COMPLEMENTARY DEFENSE LINE 4 FUNCTIONS FOR MITIGATING DECS

[Reserved. To be addressed in CPA Part 2 or the operating license application, as necessary.]

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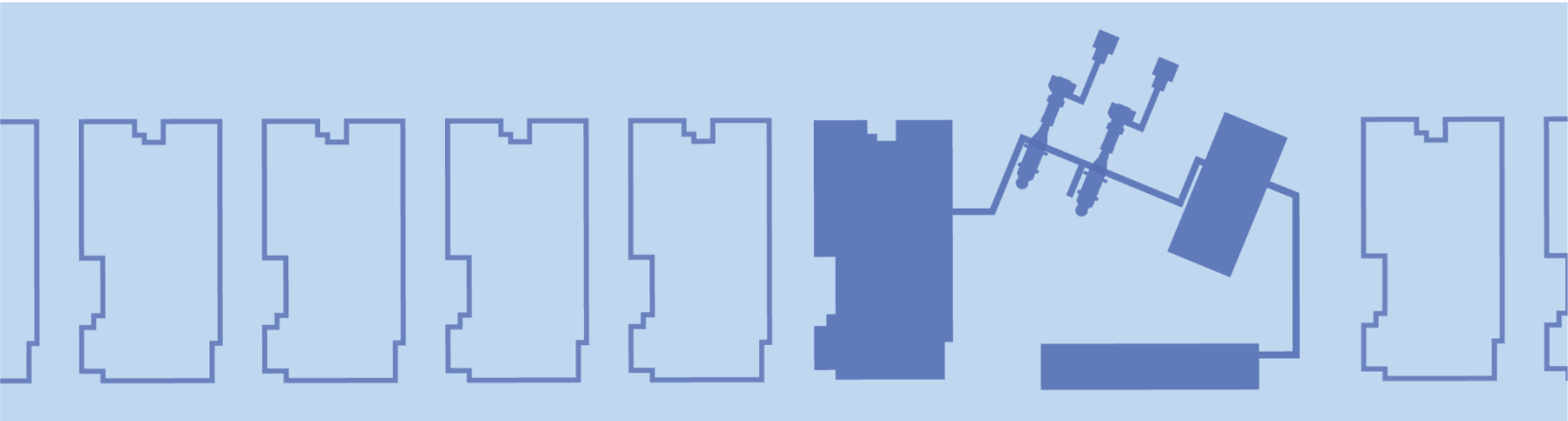
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Chapter 15
Appendix 15C
Safety Analyses



15.0C SAFETY ANALYSES

[Reserved]. To be addressed in CPA Part 2.

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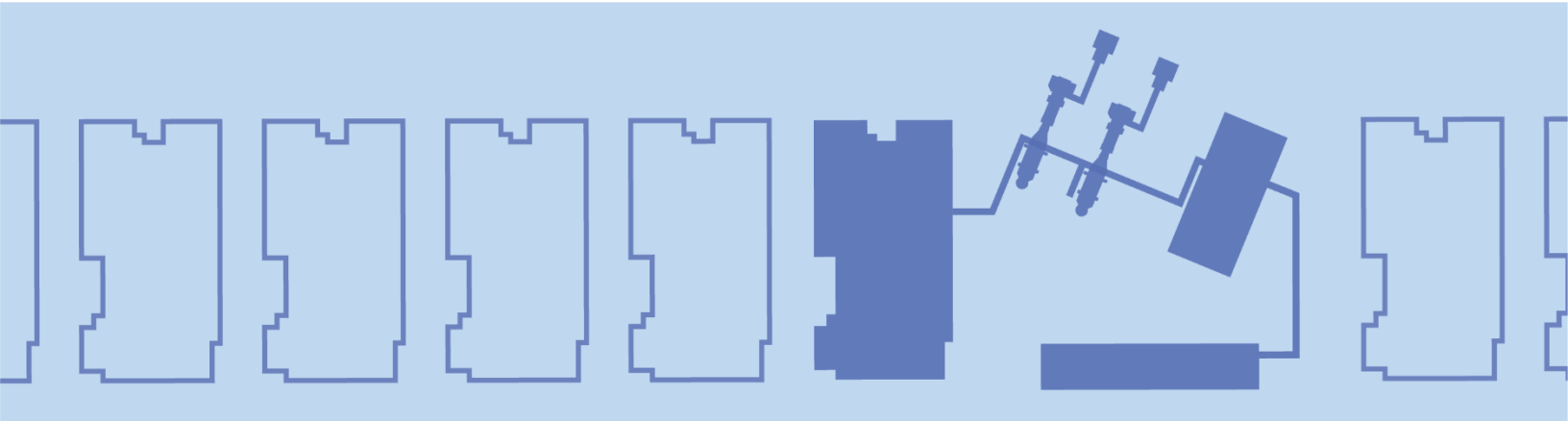
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Chapter 16
Technical Specifications



ACRONYM LIST

Acronym	Explanation
AOO	Anticipated Operational Occurrence
BL-DSA	Baseline Deterministic Safety Analysis
BWR	Boiling Water Reactor
CCF	Common Cause Failure
CFR	Code of Federal Regulations
CN-DSA	Conservative Deterministic Safety Analysis
BECPV-N01	Blue Energy Center – Port of Victoria (BECPV) Nuclear Unit 1
D-in-D	Defense-in-Depth
DBA	Design Basis Accident
DL	Defense Line
DL2	Defense Line 2
DL3	Defense Line 3
DSA	Deterministic Safety Analysis
ESBWR	Economic Simplified Boiling Water Reactor
FSAR	Final Safety Analysis Report
ICS	Isolation Condenser System
LCO	Limiting Condition for Operation
OL	Operating License
PIE	Postulated Initiating Event
PSAR	Preliminary Safety Analysis Report
RCS	Reactor Coolant System
RG	Regulatory Guide
SC1	Safety Class 1
SC3	Safety Class 3
SL	Safety Limit
SR	Surveillance Requirement
STS	Standard Technical Specification
TS	Technical Specifications
TSTF	Technical Specifications Task Force
USNRC	United States Nuclear Regulatory Commission

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None.

16.0 TECHNICAL SPECIFICATIONS

16.1 Preliminary Safety Analysis Report and Construction Permit Application Requirements

The requirements for Preliminary Safety Analysis Report (PSAR) Technical Specifications (TS) and Bases are specified in 10 Code of Federal Regulations (CFR) 50.34, “Contents of Applications; Technical Information,” Paragraph (a)(5). This section specifically requires:

> “An identification and justification for the selection of those variables, conditions, or other items which are determined as the result of preliminary safety analysis and evaluation to be probable subjects of TS for the facility, with special attention given to those items which may significantly influence the final design.”

Section 16.5 satisfies this requirement for the Blue Energy Center – Port of Victoria Nuclear Unit 1 (BECPV-N01) design by identifying preliminarily selected variables, conditions, and items that were identified as the result of the descriptions and safety analyses contained elsewhere in this PSAR.

16.2 Regulatory Guidance for Preliminary Technical Specification Contents

Regulatory Guide (RG) 1.70, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants” provides guidance regarding expected preliminary TS submittal in support of an application for a construction permit.

Section 16.5 satisfies this requirement for the BECPV-N01 design by identifying preliminarily selected variables, conditions, and items that were identified as the result of the descriptions and safety analyses contained elsewhere in this PSAR.

16.3 Conformance with Industry Standards and Practices

16.3.1 Writer's Guide Conformance

The TS content and format align with TSTF-GG-05-01, "Writer's Guide for Plant-Specific Improved Technical Specifications," (Reference 16.3-1) to the extent appropriate for the BECPV-N01 design and safety analyses details. A complete Bases for each Safety Limit (SL) and each Limiting Condition for Operation (LCO) is provided in the Final Safety Analysis Report (FSAR) consistent with the guidance in NUREG-1434, "Standard Technical Specifications General Electric BWR/6 Plants," (Reference 16.3-2), NUREG-1433, "Standard Technical Specifications – General Electric BWR/4 Plants," (Reference 16.3-3), and TSTF-GG-05-01, Writer's Guide for Plant-Specific Improved Technical Specifications" (Reference 16.3-1).

Another example is that programmatic controls in the Standard Technical Specification (STS), Section 5.5 are only identified when associated requirements are applicable to the design or analyses.

16.3.2 Alignment with U.S. Nuclear Regulatory Commission and Industry Harmonization of Rules for Applicability and Definitions

The administrative structure and rules for application of TS requirements are primarily specified in Chapter 1, LCO 3.0, and Surveillance Requirement (SR) 3.0 in modern STS. The use of consistent rules has improved understanding of TS requirements by users, compliance during operations, and communications between the United States Nuclear Regulatory Commission (USNRC) staff, plant operators, and between facilities when sharing operating experience. The BECPV-N01 TS align with the STS content to the extent applicable.

16.3.3 Recent Industry Experience Lessons Learned

The content of legacy plant STS are constantly subject to review and modification, largely through the efforts of the industry Technical Specifications Task Force (TSTF) in cooperation with USNRC staff. Changes are issued in the form of approved travelers, which are available for adoption by the specified designs, and are incorporated in future revisions to the associated NUREG STS.

The BWRX-300 design is not reflected or considered in the development or review of travelers by the TSTF. However, some changes identified in travelers, such as administrative structure, rules of application, and certain programs may be pertinent to the content of the BECPV-N01 TS.

In the past decades, new passive plant and small modular reactor designs have been reviewed and licensed by the USNRC. Experience and lessons learned during the review of those TS submittals (i.e., Economic Simplified Boiling Water Reactor (ESBWR)) are considered in the development of the BECPV-N01 TS.

16.3.4 References

16.3-1 TSTF-GG-05-01, "Writer's Guide for Plant-Specific Improved Technical Specifications," Revision 1, August 2010.

16.3-2 NUREG-1434, "Standard Technical Specifications General Electric BWR/4 Plant," Revision 5.0, U.S. Nuclear Regulatory Commission, September 2021.

16.3-3 NUREG-1433, "Standard Technical Specifications General Electric BWR/4 Plants," Revision 5.0, U.S. Nuclear Regulatory Commission, September 2021.

16.4 Methodology for Selection of Preliminary Technical Specification Contents

16.4.1 Generic and Programmatic Content

NUREG-1433, "Standard Technical Specifications General Electric BWR/4 Plants," (Reference 16.4-1) and NUREG-1434, "Standard Technical Specifications General Electric BWR/6 Plants," (Reference 16.4-2) define STS for Boiling Water Reactor (BWR) designs. The certified ESBWR Standard Design TS provide generic TS for a more recent BWR design. Although the STS and ESBWR generic TS are not applicable to the BWRX-300, portions of Chapter 1, Section 2.2, LCO 3.0, SR 3.0, Chapter 4, and Chapter 5 of the NUREGs are incorporated into the BECPV-N01 TS and are similar in format and content. This ensures regulatory conformance, and that the context of the TS is written and interpreted consistently with regulatory and industry expectations and practices.

Some content may reflect anticipated changes that are under discussion between the USNRC and the operating plants, regardless of the generic change status.

16.4.2 Application of BWRX-300 Safety Strategy

The BWRX-300 safety strategy is described in NEDC-33934P, "BWRX-300 Safety Strategy," (Reference 16.4-3) which implements a systematic process for ensuring Defense-in-Depth (D-in-D). Providing layered protection against unacceptable releases of radiation is the basic premise of the BWRX-300 D-in-D concept by utilizing Defense Lines (DLs). For each Anticipated Operational Occurrences (AOOs) and Design Basis Accident (DBA) Postulated Initiating Event (PIE) caused by a single failure with a frequency greater than 1E-05 per reactor-year, two independent and diverse DL functions can mitigate the PIE. The layers of defense are demonstrated via multiple types of Deterministic Safety Analysis (DSA). Based on this D-in-D approach, it is necessary to establish which set of DL functions is considered the primary success path pursuant to LCO Criterion 3 for AOOs and DBAs and which DSA should be used pursuant to LCO Criterion 2 in accordance with the 10 CFR 50.36(c)(2)(ii) criteria.

AOOs

Defense Line 2 (DL2) and Defense Line 3 (DL3) functions provide layered protection for mitigating AOO PIEs. As described in Section 5.1 of NEDC-33934P, DL3 functions performed by Safety Class 1 (SC1) Structures, Systems, and Components (SSCs) mitigate AOOs to within the 10 CFR 50.2 "safety-related" criteria, as demonstrated by a Conservative Deterministic Safety Analysis (CN-DSA). The Baseline Deterministic Safety Analysis (BL-DSA) is performed to demonstrate the capability of Safety Category 3 functions. DL3 functions performed by SC1 SSCs are designated as the primary success path for mitigating AOO PIEs with DL2 Common Cause Failure (CCF) pursuant to LCO Criterion 3. CN-DSA are the appropriate set of analyses to establish LCOs pursuant to LCO Criterion 2. These designations are reasonable and ensure that "safety-related" SSCs (i.e., SC1 SSCs) are the subject of TSs.

Safety-Class 3 (SC3) SSCs performing DL2 functions are subject to TSs if the SSCs are determined to be risk-significant in accordance with LCO Criterion 4. Excluding SC3 SSCs that are not risk-significant from TSs is justified as follows:

- SC3 SSCs that are not risk-significant are subject to the Availability Controls Manual (also may be called Technical Requirements Manual). The initial TSs and initial Availability Controls Manual are subject to USNRC review and approval during the Operating License (OL) application review prior to OL issuance. The primary difference between Availability Controls Manual and TSs is that certain changes to the Availability Controls Manual after initial issuance can be made by the licensee without prior USNRC approval if permitted by 10 CFR 50.59. However, all changes to TSs require prior USNRC approval. The

BWRX-300 Safety Strategy's treatment of SSCs ensures that TS changes are the most safety-significant, as these changes are related to SC1 SSCs and risk-significant SSCs. Additionally, the criteria in 10 CFR 50.59 are adequate to maintain acceptable levels of safety for the SSCs monitored in the Availability Controls Manual. Inclusion of SC3 SSCs performing DL2 functions that are not risk-significant in the Availability Controls Manual provides monitoring of function availability under a change control process more appropriate for non-SC1 SSCs. Including SSCs that are not risk-significant and are not "safety-related" in the Availability Controls Manual exceeds USNRC requirements described in SECY-94-084 and SRP 19.3.

- The primary difference of interest between the AOO acceptance criteria, which are applied to BL-DSA that credit DL2 functions to mitigate AOOs, and the DBA acceptance criteria, which are applied to CN-DSA that credit DL3 functions to mitigate AOOs along with an assumed failure of DL2 functions, is the potential for fuel rod failure. The following criterion will be used to establish LCOs to ensure that SSCs necessary to ensure fuel cladding integrity is maintained following AOOs are included in TSs:
  - -If CN-DSA results for an event sequence involving an AOO PIE with an assumed CCF of DL2 functions conclude that fuel cladding failures would occur, the corresponding BL-DSA initial conditions and SSCs credited to mitigate that same AOO PIE will be subject to Technical Specification LCOs.

This criterion, along with establishing LCOs for CN-DSA initial conditions and SSCs performing DL3 functions to mitigate AOO PIEs with an assumed DL2 CCF, ensure that TSs provide adequate protection of public health and safety following an AOO.

In summary, a graded approach to monitoring the two sets of DL functions that are capable of mitigating AOO PIEs is being applied as part of the BWRX-300 Safety Strategy. DL3 functions performed by SC1 SSCs mitigate AOO PIEs to within the 10 CFR 50.2 "safety-related" criteria and are included in TSs, along with non-SC1 SSCs determined to be risk-significant. Additionally, if CN-DSA results for an event sequence involving an AOO PIE with an assumed CCF of DL2 functions conclude that fuel cladding failures would occur, the corresponding BL-DSA initial conditions and SSCs credited to mitigate that same AOO PIE will be subject to TS LCOs. The remaining DL2 functions performed by SC3 SSCs are included in and surveilled by the Availability Controls Manual. This approach ensures that SSCs performing DL2 or DL3 functions are adequately monitored based on their safety importance and risk-significance, while reducing the potential for license amendment requests for changes that do not exceed the criteria in 10 CFR 50.59 involving non-SC1, non-risk-significant SSCs. Therefore, there is adequate justification for excluding DL2 functions performed by SC3 SSCs from TS and identifying DL3 functions performed by SC1 SSCs as the primary success path for LCO criterion 3.

<u>DBAs</u>

DL2, DL3, and DL4a functions provide layered protection for mitigating DBA PIEs, depending on the scenario. In all cases, DL3 functions mitigate the consequences of the DBA PIE, ensuring the DBA acceptance criteria are met, which satisfy the 10 CFR 50.2 "safety-related" criteria as described in Section 5.1 of NEDC-33934P. These acceptance criteria ensure adequate protection of public health and safety, and the analysis that demonstrates the acceptance criteria are met is the CN-DSA. Therefore, DL3 functions performed by SC1 SSCs are designated as the primary success path for mitigating DBA PIEs pursuant to LCO Criterion 3, and CN-DSA are the appropriate set of analyses to establish LCOs pursuant to LCO Criterion 2.

16.4.3 Specification of Completion Times and Surveillance Frequencies

Final Conditions, Required Actions, Completion Times, and Surveillance Frequencies are not available because of the preliminary nature of the design and safety analyses at the PSAR stage. However, the intent is to specify conditional logic, identify required actions, and establish completion times and surveillance frequencies consistent with the plant design, industry experience and practice for similar variables, limits, structures, systems, and components in existing TS. Surveillance frequencies are controlled in accordance with a Surveillance Frequency Control Program consistent with the scope of existing plant implementations. Initial frequencies and the bases for those frequencies are provided at the OL application stage.

16.4.4 References

16.4-1 NUREG-1433, “Standard Technical Specifications General Electric BWR/4 Plants,” Revision 5.0, U.S. Nuclear Regulatory Commission, September 2021.

16.4-2 NUREG-1434, “Standard Technical Specifications General Electric BWR/6 Plants,” Revision 5.0, U.S. Nuclear Regulatory Commission, September 2021.

16.4-3 NEDC-33934P, “BWRX-300 Safety Strategy,” Revision 2, GE Vernova Hitachi Nuclear Energy Americas, LLC, March 2026.

16.5 Results of Selection Methodology Application

Section 16.5 describes the reason for inclusion of the individual specifications. The selection methodology was described in Section 16.4. Table 16.5-1 provides a preliminary table of contents for the BWRX-300 TS, including the reason for inclusion or selection of that content.

16.5.1 Use and Application

Chapter 1 (Use and Application) is included to ensure consistent and common understanding of the contents of the TS, and to align BWRX-300 with industry standards for that content. It aligns with the rules for applying the TS with the rules used for writing, formatting, implementing, and with other STS to the extent appropriate for the BWRX-300 design. The following sections are included in TS Chapter 1:

- 1.1 - Definitions

  Definitions are generally consistent with industry practice and previously licensed BWR designs, however with modifications to reflect the BECPV-N01 design, analyses, and content of the balance of the TS.

- 1.2 - Logical Connectors

  The Logical Connectors section is generally aligned with industry practice and previously licensed BWR designs, however with modifications to reflect the BECPV-N01 TS contents and usage.

- 1.3 - Completion Times

  The Completion Times section generally aligns with industry practice and previously licensed BWR designs, however with modifications to reflect the BECPV-N01 TS contents and usage.

- 1.4 - Frequency

  The Frequency section generally aligns with industry practice and previously licensed BWR designs, however with modifications to reflect the BECPV-N01 TS content and usage.

16.5.2 Safety Limits

Chapter 2 (Safety Limits) is included to satisfy USNRC requirements and align with industry standards for content. SLs are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity. They are required to be included in the TS by 10 CFR 50.36(c)(1)(i)(A). The BECPV-N01 SLs provide protection for the integrity of the Reactor Core and the Reactor Coolant System (RCS). The following sections are included in TS Chapter 2:

- 2.1.1 - Reactor Core SLs

  Reactor Core SLs protect the integrity of the reactor fuel and its cladding. These SLs ensure integrity of a fission product barrier to limit the release of radioactive material from the nuclear fuel into the RCS.

- 2.1.2 - RCS Pressure SL

  The RCS Pressure SL ensures the integrity of the RCS which acts as a fission product barrier limiting the release of radioactive material from the reactor core to the containment and environment.

- 2.2 - SL Violations

  This section describes the actions required to be completed if any SL is violated, and the time in which the actions must be completed.

16.5.3 Limiting Conditions for Operation and Surveillance Requirements

TS Chapter 3 (Limiting Conditions for Operation and Surveillance Requirements) is included to satisfy USNRC requirements and align with industry standards for content. 10 CFR 50.36(c)(2) requires LCOs that establish the lowest functional capability or performance levels of equipment required for safe operation of the facility. Additionally, 10 CFR 50.36(c)(3) specifies inclusion of requirements relating to tests, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation is within SLs, and that the LCOs are met.

This chapter begins with rules for the applicability of the LCOs and SRs that ensure proper and consistent application of the requirements listed, including default actions if requirements cannot be met. The rules support the technical and regulatory content in the individual LCO and SR that follow. They are described in:

- 3.0 - LCO Applicability:
  - Applicability rules for the LCO portions of individual specifications in Chapter 3
- 3.0 - SR Applicability:
  - Applicability rules for the SR portions of individual specifications in Chapter 3

The balance of Chapter 3 contains LCOs, and SRs divided similarly to existing BWR TS, into ten sections as follows:

- 3.1 - Reactivity Control Systems
- 3.2 - Power Distribution Limits
- 3.3 - Instrumentation
- 3.4 - Reactor Coolant System
- 3.5 - Emergency Core Cooling Systems
- 3.6 - Containment Systems
- 3.7 - Plant Systems
- 3.8 - Electrical Power Systems
- 3.9 - Refueling Operations
- 3.10 - Special Operations

Contents of these ten sections are selected using four criteria listed in 10 CFR 50.36(c)(2)(ii).

The results of the methodology described in Subsection 16.4 are provided in Table 16.5-1, with the criteria used to select the content listed. The table provides additional information by reference to a regulation, industry guidance, PSAR section, subsection, or table that supports the Bases for inclusion of the LCO into the TS.

16.5.4 Design Features

Chapter 4 (Design Features) is included to satisfy USNRC requirements and align with industry standards for content. 10 CFR 50.36(c)(4) requires inclusion of those features of the facility such

as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described paragraphs (c)(1), (2) and (3) of 10 CFR 50.36. Based on USNRC guidance on content and industry practice, the following three sections are included in this chapter:

- 4.1 - Site Location

  This section describes details of the plant location. It contains information for the plant site, similar to the level of detail in other TS.

- 4.2 - Reactor Core

  This section describes the reactor fuel and control rod assemblies. It contains plant-specific information similar to the level of detail in other TS.

- 4.3 - Fuel Storage

  This section describes features of fuel storage related to prevention of inadvertent criticality, ensuring adequate cooling of spent fuel, and the fuel storage capacity. It contains plant-specific information similar to the level of detail in other TS.

16.5.5 Administrative Controls

Chapter 5 (Administrative Controls) is included to satisfy USNRC requirements and align with industry standards for content. 10 CFR 50.36(c)(5) requires inclusion of provisions relating to organization and management, procedures, recordkeeping, review, and audit, and reporting necessary to assure operation of the facility in a safe manner. This chapter also includes the administrative controls that satisfy the requirements of 10 CFR 50.36a. Based on USNRC guidance on content and industry practice, the following seven sections are included in this chapter:

- 5.1 - Responsibility

  This section describes who has overall responsibility for unit operation and who is responsible for the control room command function. It aligns with recent USNRC and industry guidance for content.

- 5.2 - Organization

  This section describes the onsite, offsite, and unit organization including shift crew requirements. It contains information similar to the level of details in other TS.

- 5.3 - Unit Staff Qualifications

  This section describes the overall qualifications of plant staff. It contains information similar to the level of detail in other TS.

- 5.4 - Procedures

  This section describes requirements that written procedures are established, implemented, and maintained for activities listed, including plant operations. It aligns with recent USNRC and industry guidance for content.

- 5.5 - Programs and Manuals

  This section requires programs to be established, implemented, and maintained, including programs referenced in LCOs and SRs. The programs and requirements described are consistent with recent guidance. Content is adjusted to reflect the design and analyses applicable to the BWRX-300.

- 5.6 - Reporting Requirements

  This section establishes requirements for reports including radioactive effluents, core operating limits, and pressure and temperature limits on the RCS. The programs and requirements described are consistent with recent guidance. Content is adjusted to reflect the design and analyses applicable to the BWRX-300.

- 5.7 - High Radiation Area

  This section establishes alternatives to requirements of 10 CFR 20, “Standards for Protection Against Radiation” for the control of high radiation areas. The controls are similar to those in the industry standard TS.

16.5.6 Generic Changes Assessment

The TS content is subject to review as generic changes are considered by the industry/USNRC traveler process. The consideration of travelers for inclusion to the extent appropriate for the design and safety analyses occur at the OL application stage.

16.5.7 References

16.5-1 TSTF-GG-05-01, “Writer’s Guide for Plant-Specific Improved Technical Specifications,” Revision 1, August 2010.

Table 16.5-1: Preliminary Table of Contents and Bases for Inclusion

Specification	Title	LCO Selection Criteria	Bases for Inclusion
1.0	**USE AND APPLICATION**		
1.1	Definitions	N/A	TSTF-GG-05-01 (Reference 16.5-1)
1.2	Logical Connectors	N/A	
1.3	Completion Times	N/A	
1.4	Frequency	N/A	
2.0	**SAFETY LIMITS**		
2.1	SLs	N/A	10 CFR 50.36(c)(1)
2.2	SL Violations	N/A	
3.0	**LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS**		
3.0	**LIMITING CONDITIONS FOR OPERATION APPLICABILITY**	N/A	TSTF-GG-05-01 (Reference 16.5-1)
3.0	SURVEILLANCE REQUIREMENT (SR) APPLICABILITY	N/A	
3.1.1	Shutdown Margin (SDM)	2	Subsection 4.3.4.5, Table 15.3-2
3.1.2	Reactivity Anomalies	2	Subsection 4.3.4.3
3.1.3	Control Rod Operability	3	Subsection 15.5.4, Table 15.5-4, Table 15.5-5, Table15.5-44, and Section 4.6
3.1.4	Control Rod Scram Times	3	Subsection 15.5.4, Table 15.5-4, Table 15.5-5, Table15.5-44
3.1.5	Control Rod Scram Accumulators	3	Subsection 15.5.4, Table 15.5-4, Table 15.5-5, Table15.5-44, and Section 4.6
3.2.1	Linear Heat Generation Rate	2	Subsections 4.3.1.3, 4.3.1.3.1, and 4.4.1.2
3.2.2	Minimum Critical Power Ratio	2	Subsections 4.3.1.3 and 4.3.1.3.2
3.3.1 Function 1	Power Range Neutron Monitoring	3	Subsections 15.5.4.1 and 15.5.4.2, Table 15.5-5

Specification	Title	LCO Selection Criteria	Bases for Inclusion
3.3.1 Function 2	Reactor Pressure Vessel Pressure	3	Subsection 15.5.4.2, Tables 15.5-5, and 15.5-44
3.3.1 Function 3	Reactor Pressure Vessel Level	3	Subsection 15.5.4.4.1, Tables 15.5-5 and 15.5-44
3.3.1 Function 4	Containment Pressure	3	Subsection 15.5.4.5 and Table 15.5-44
3.3.1 Function 5	Loss of Normal Feedwater Flow	3	Subsections 15.5.4.2.2 and 15.5.4.4.1
3.3.1 Function 6	Feedwater Line Break Detection	3	Subsection 15.5.4.5 and Table 15.5-44
3.3.1 Function 7	Main Steam Break Detection	3	Subsections 15.5.4.5 and Table 15.5-44
3.3.1 Function 8	Shutdown Cooling Line Break Detection	3	Subsection 15.5.4.5, and Table 15.5-44
3.3.1 Function 9	Reactor Water Cleanup Line Break Detection	3	Subsection 15.5.4.5, and Table 15.5-44
3.3.1 Function 10	Isolation Cooling Line Break Detection	3	Subsection 15.5.4.5, and Table 15.5-44
3.4.1	RCS Operational Leakage	2	Subsection 5.2.5.4
3.4.2	RCS Specific Activity	2	Subsection 15.5.9
3.4.3	RCS Pressure and Temperature Limits	2	Subsection 5.3.2
3.4.4	Reactor Pressure Vessel Steam Dome Pressure	2	Subsection 15.5.1.1.2, Table 15.5-1, Subsection15.5.4, and Table 15.5-3
3.4.5	RCS Leakage Detection Instrumentation	1	Subsection 5.2.5.8 Position 2.3 and 5.2.5.5.
3.5.1	Isolation Condenser System (ICS)	3	Section 6.3 and Subsection 15.5.4
3.5.2	ICS Pool Complex	3	Subsections 6.3.2, 15.5.4, and 9.2.8 and Tables 15.5-1 and 15.5-3
3.6.1	Containment	3	Section 6.2 and Appendix 15C.1.3
3.6.2	Containment Air Locks	3	Section 6.2 and Appendix 15C.1.3

Specification	Title	LCO Selection Criteria	Bases for Inclusion
3.6.3	Containment Isolation Valves and Reactor Isolation Valves	3	Subsection 6.2.4, Table 6.2-2, Section 5.4.3, and Subsection15.5.4
3.6.4	Primary Containment Pressure	2	Table 15.5-1
3.6.5	Containment Air Temperature	2	Table 15.5-1
3.6.6	Passive Containment Cooling System and Equipment Pool	3	Subsections 6.2.2, and 15.5.4.5.1
3.7.1	Offgas System	2	Subsection 11.3.3.1 and Table 11.3-3
3.7.2	Fuel Pool Water Level and Temperature	2	Subsections 9.1.2.3, 9.2.5, and 15.5.8
3.8.1	Emergency Electrical Power	3	Subsection 8.3.2.1
3.8.2	Battery Parameters	3	Subsection 8.3.2.1
3.9	Refueling Operations		Provided in the FSAR
3.10	Special Operations	Note 1	Provided in the FSAR
4.0	**DESIGN FEATURES**		
4.1	Site Location	N/A	10 CFR 50.36(c)(4)
4.2	Reactor Core	N/A	
4.3	Fuel Storage	N/A	
5.0	**ADMINISTRATIVE CONTROLS**		
5.1	Responsibility	N/A	10 CFR 50.36(c)(5) 10 CFR 50.36(a)
5.2	Organization	N/A	
5.3	Unit Staff Qualifications	N/A	
5.4	Procedures	N/A	
5.5	Programs and Manuals	N/A	
5.6	Reporting Requirements	N/A	
5.7	High Radiation Area	N/A	

Note:

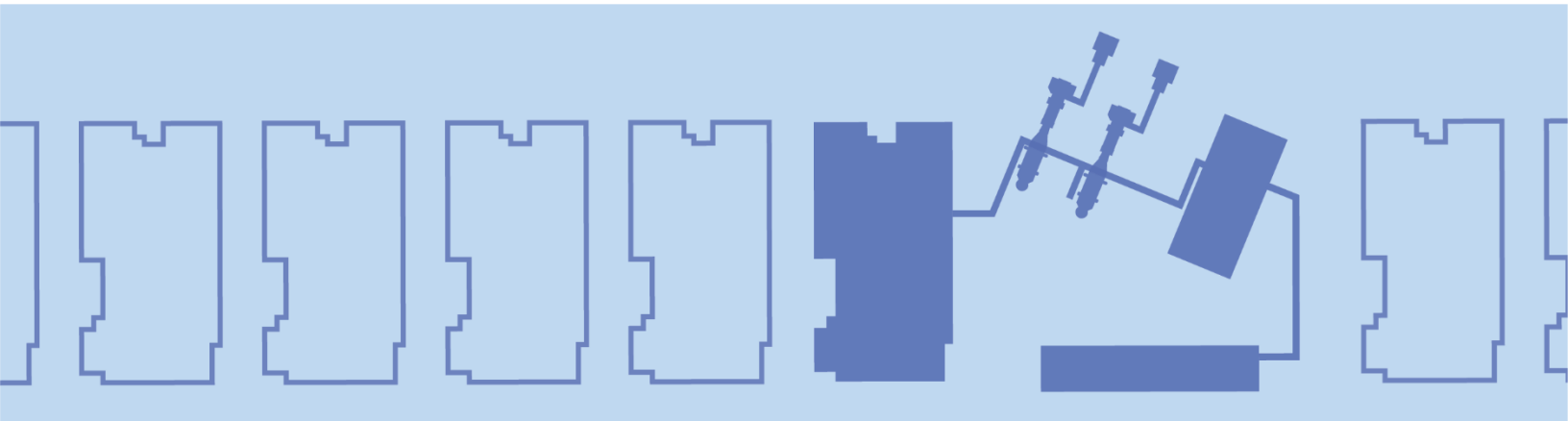
1. As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of 10 CFR 50.36(c)(2)(ii) apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

Blue Energy Global Inc.

BE-POV-LCNRG-LIC-5009
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Non-Proprietary Information

Chapter 17
Quality Assurance



ACRONYM LIST

Acronym	Explanation
BWR	Boiling Water Reactor
CDF	Core Damage Frequency
D-in-D	Defense-in-Depth
D-RAP	Design Reliability Assurance Program
DBA	Design Basis Accident
DEC	Design Extension Condition
DSA	Deterministic Safety Analysis
FSAR	Final Safety Analysis Report
FV	Fussell-Vesely
GVH	GE Vernova Hitachi Nuclear Energy
IAEA	International Atomic Energy Agency
LRF	Large Release Frequency
OPEX	Operating Experience
PSA	Probabilistic Safety Assessment
QA	Quality Assurance
QC	Quality Control
RAW	Risk Achievement Worth
SA	Severe Accident
SAA	Severe Accident Analysis
SC1	Safety Class 1
SSCs	Structures, Systems, and Components

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None.

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None.

17.0 QUALITY ASSURANCE

17.1 Quality Assurance During Design and Construction Phases

The Quality Assurance (QA) Program for site design, construction, and operation activities controlled by Blue Energy Global Inc. (Blue Energy) is described in Section 17.5. Blue Energy owns responsibility for design activities related to BECPV and has delegated the responsibility for BWRX-300 design activities to GE Vernova Hitachi Nuclear Energy (GVH). GVH performs BWRX-300 design activities in accordance with the GVH quality assurance program. Blue Energy considers GVH a vendor under the oversight of the Blue Energy's QA program description in Section 17.5.

17.2 Quality Assurance Program During the Operations Phase

[Section Reserved] The QA program during the operations phase is to be described in Section 17.5.

17.3 Quality Assurance Program Description

[**Reserved**] This section will be addressed in CPA Part 2. For relevant QAP information supporting CPA Part 1 and the associated LWA, see Enclosure 6 (submitted alongside this application) and our current QAPD Topical Report (Reference 17.3-1).

17.3.1 References

17.3-1 BE-QAPD-001-A, “Quality Assurance Program Description (QAPD),” Revision 2, Blue Energy Global Inc., December 2025, ADAMS Accession No. ML25340A001.

17.4 Reliability Assurance Program

[**Reserved**] This section will be addressed in CPA Part 2. For relevant information supporting CPA Part 1 and the associated LWA, see Enclosure 6 (submitted alongside this application) and our current QAPD Topical Report (Reference 17.4-1).

17.4.1 References

17.4-1 BE-QAPD-001-A, “Quality Assurance Program Description (QAPD),” Revision 2, Blue Energy Global Inc., December 2025, ADAMS Accession No. ML25340A001.

17.5 Quality Assurance Program Description – Design Certification, Early Site Permit, and New License Applicants

[**Reserved**] This section will be addressed in CPA Part 2. For relevant QAP information supporting CPA Part 1 and the associated LWA, see Enclosure X (submitted alongside this application) and our current QAPD Topical Report (Reference 17.5-1).

17.5.1 References

17.5-1 BE-QAPD-001-A, “Quality Assurance Program Description (QAPD),” Revision 2, Blue Energy Global Inc., December 2025, ADAMS Accession No. ML25340A001.

17.6 Maintenance Rule

The program for monitoring the effectiveness of maintenance required by 10 CFR 50.65 is described in the operating license application.

Section 17.6 is not applicable to a CP application.