ENCLOSURE 1

Clinch River Nuclear Site
Construction Permit Application
Preliminary Safety Analysis Report
CPA Enclosure 3 (Public Version)

Security-Related Information - Withheld Under 10 CFR 2.390

TENNESSEE VALLEY AUTHORITY

Clinch River Nuclear Site Preliminary Safety Analysis Report

Construction Permit Application – Enclosure 3

Revision 1

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LIST OF ACRONYMS AND ABBREVIATIONS

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
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
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
CPA	Construction Permit Application
CPR	Critical Power Ratio
CR	Control Room
CRD	Control Rod Drive
CRN	Clinch River Nuclear
CRN-1	Clinch River Nuclear Unit 1
CWS	Circulating Water System
DBA	Design Basis Accident
DC	Direct Current
DEC	Design Extension Condition
DP-SC	Diaphragm Plate Steel-Plate Composite
DRG	Design Review Guide
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
FSAR	Final Safety Analysis Report
FY	Government Fiscal Year

LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
GDC	General Design Criteria
GNF	Global Nuclear Fuel
GT	Gamma Thermometer
GVH	GE Vernova Hitachi Nuclear Energy
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
I&C	Instrumentation and Control
ICS	Isolation Condenser System
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronics Engineers
ISG	Interim Staff Guidance
ISI	Inservice Inspection
IST	Inservice Testing
LOCA	Loss-of-Coolant Accident
LOCV	Loss of Condenser Vacuum
LPZ	Low Population Zone
LWR	Light Water Reactor
MCPR	Minimum Critical Power Ratio
MCR	Main Control Room
MSIV	Main Steam Isolation Valve
N/A	Not Applicable
NBS	Nuclear Boiler System
NEI	Nuclear Energy Institute
NPP	Nuclear Power Plant
NUREG	Nuclear Regulatory Report
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

LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
RB	Reactor Building
RCIC	Reactor Core Isolation Cooling
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RG	Regulatory Guide
RIPB	Risk-Informed Performance-Based
RIV	Reactor Isolation Valve
RPV	Reactor Pressure Vessel
RTNSS	Regulatory Treatment of Non-Safety Systems
RWST	Refueling Water Storage Tank
SAFDL	Specified Acceptable Fuel Design Limits
SBO	Station Blackout
SC	Steel-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
TAF	Top of Active Fuel
TBS	Turbine Bypass System
TMI	Three Mile Island
TS	Technical Specifications
TSC	Technical Support Center
TVA	Tennessee Valley Authority
UHS	Ultimate Heat Sink
USCS	United States Customary System
USNRC	U.S. Nuclear Regulatory Commission

LIST OF EFFECTIVE SECTIONS

CHAPTER 1 INTRODUCTION AND GENERAL PLANT DESCRIPTION

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CHAPTER 1 INTRODUCTION AND GENERAL PLANT DESCRIPTION

1.1 Introduction

This Preliminary Safety Analysis Report (PSAR) supports the TVA Construction Permit Application (CPA) for a single unit, Small Modular Reactor (SMR) plant at the Clinch River Nuclear (CRN) Site. The plant is designated as Clinch River Nuclear Unit 1 (CRN-1). 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.

The CRN Site is located in the City of Oak Ridge, Tennessee, and is the site of the former Clinch River Breeder Reactor Project. The CRN Site is comprised of approximately 935 acres, which are adjacent to the Clinch River arm of the Watts Bar Reservoir. The CRN Site is described in the "Clinch River Nuclear Site Early Site Permit (ESP) Application Part 2, Site Safety Analysis Report," (Reference 1.1-1). Portions of Part 2 of the ESP Application, (i.e., the Site Safety Analysis Report (SSAR)), are incorporated by reference in this PSAR. As summarized in the "Final Safety Evaluation Report for the Early Site Permit Application for the CRN Site," (Reference 1.1-2), SMRs having design characteristics consistent with the ESP Application and meeting the terms and conditions of the Final Safety Evaluation Report (FSAR) can be safely located at the CRN Site.

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.

The projected completion date is Government Fiscal Year (FY) 2032. Anticipated commercial operation is FY 2033.

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 CPA, design information is preliminary and is sufficient to assess and demonstrate the safety of the design. The PSAR will 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

Nuclear Regulatory Report (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 U.S. Nuclear Regulatory Commission (USNRC) safety evaluation. A review of the facility against the SRP revision in effect as of January 1, 2025, was performed. Subsection 1.9.2 summarizes this review with an explanation of conformance with applicable SRP acceptance criteria.

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 1.1-3), 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 1.1-4), 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. If a table used in the SSAR is revised for use in the PSAR, the original SSAR table number is appended with an "R;" for example, "SSAR Table 2.0-2" is revised and it is labeled in this PSAR as "Table 2.0-2R." 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. If a figure from the SSAR is reproduced in the PSAR, it is identified accordingly. If a figure from the SSAR is revised for use in the PSAR, the original SSAR figure number is appended with an "R;" for example, "SSAR Figure 1.2-1" is revised and it is labeled in the PSAR as "Figure 1.2-1R." 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:

Chapter 1: Introduction and General Plant Description

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.

Chapter 2: Site Characteristics

Chapter 2 provides site characteristics and siting parameters for the CRN Site.

Chapter 3: Design of Structures, Systems, and Components

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, 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 CRN-1 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 CRN-1 to ensure conformance with regulatory requirements and the design bases specified in the CPA.

1.1.2 References

1.1-1 “Clinch River Nuclear Site Early Site Permit Application Part 2, Site Safety Analysis Report,” Tennessee Valley Authority, Revision 2, January 2019.

1.1-2 “Final Safety Evaluation Report for the Early Site Permit Application for the Clinch River Nuclear Site,” U.S. Nuclear Regulatory Commission, June 2019 (ML19162A157).

1.1-3 “Design Review Guide (DRG): Instrumentation and Controls for Non-Light-Water Reactor (Non-LWR) Reviews,” U.S. Nuclear Regulatory Commission, February 2021.

1.1-4 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 CRN Site and CRN-1 follows.

1.2.1 Principal Site Characteristics

The CRN Site is defined in the SSAR and Chapter 2. Portions of the SSAR (Reference 1.1-1) are incorporated by reference in this PSAR as itemized in Section 1.8. The CRN Site location is shown in Figure 1.2-1R.

1.2.2 Plant Description

The CRN-1 plant, a BWRX-300 design, is a water-cooled SMR with core heat removal by natural circulation. The Balance of Plant 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 (RIV): 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

General arrangement drawings illustrating major structures are provided in Figure 1.2-3 through Figure 1.2-15. Table 1.2-2 provides the size of Power Block structures.

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

Table 1.2-2 Size of Power Block Structures

	Turbine Building	Radwaste Building	Control Building	Service Building
Length:	65.9m	40.4m	30.6m	30.6m
Width:	66.9m	23.4m	33.8m	28.8m
Height:	36.0m	20m	20m	20m

Notes:

1. Values are approximate.
2. Height does not include the Plant Vent Stack or stairs.
3. Equipment structures around Turbine Building, Radwaste Building, and Control Building are not included in the Power Block footprint.

Outer Diameter of Reactor Building	37.0m
Inner Diameter of Reactor Building	35.2m
Outer Diameter of Containment	19.9m
Inner Diameter of Containment	17.5m
Height Above Level 0m	32.7m

Note:

1. Values are approximate.

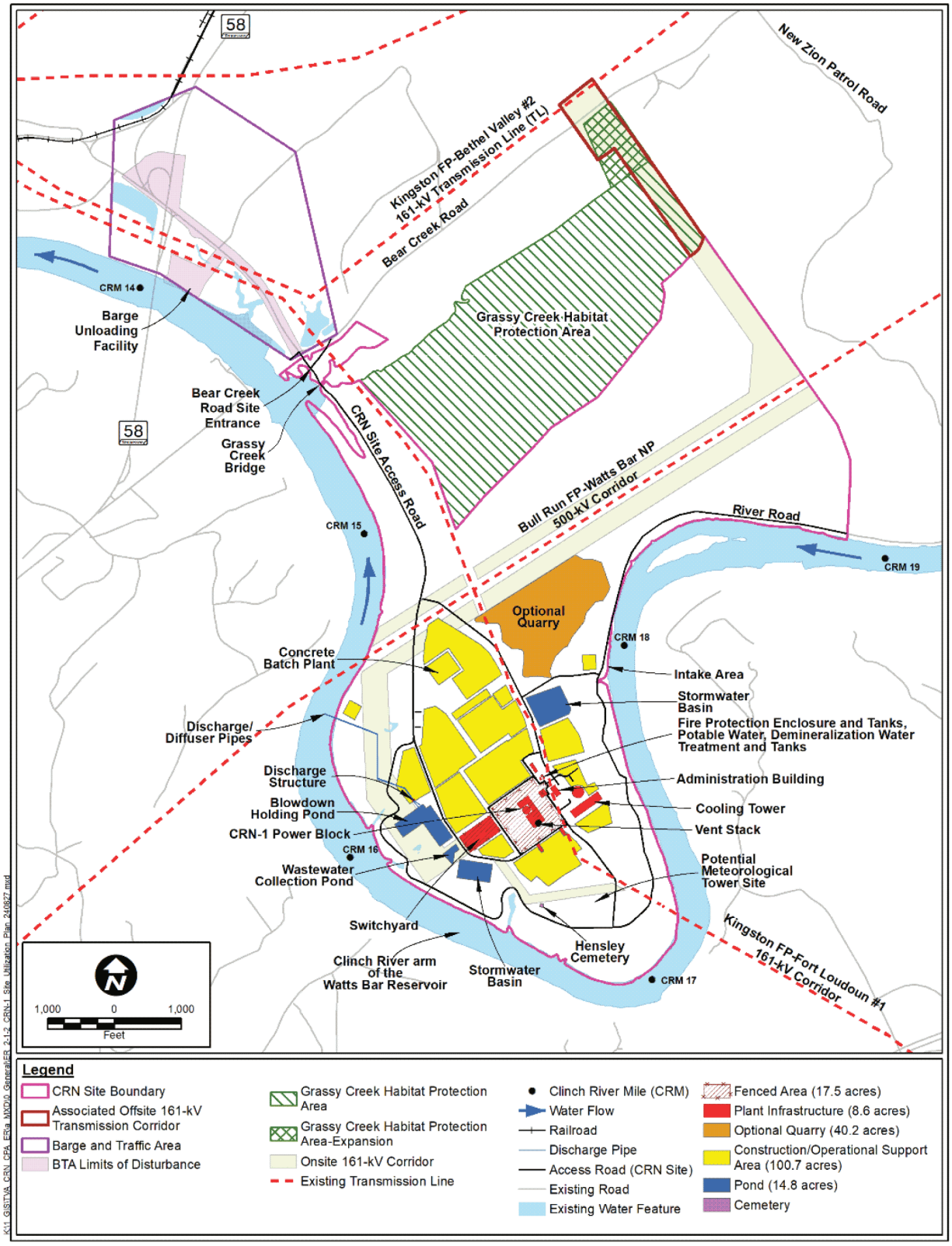
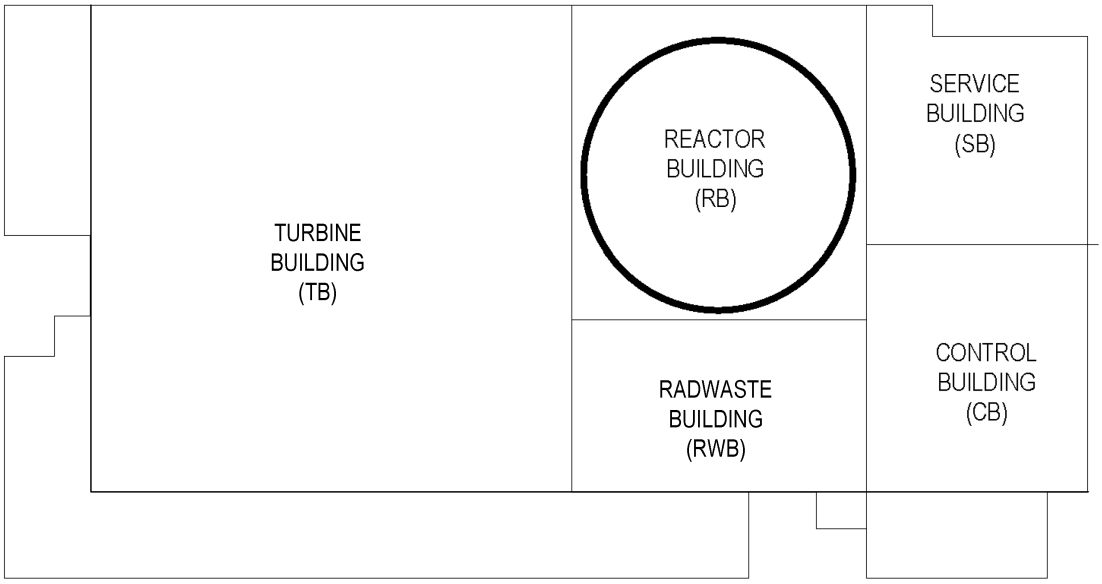


Figure 1.2-1R Clinch River Site



Note:

The Power Block structure consists of the Reactor Building, Turbine Building, Radwaste Building, Control Building, and the Service Building. Equipment structures around the Turbine Building, Radwaste Building, and Control Building are not included in the Power Block footprint.

Figure 1.2-2 BWRX-300 Power Block General Arrangement Key Plan

Figure 1.2-3 BWRX-300 Elevation View

Figure 1.2-4 BWRX-300 Plan View, Elevation 0m

Figure 1.2-5 BWRX-300 Plan View, Elevation 7.5m

Figure 1.2-6 BWRX-300 Plan View, Elevation 15.0m

Figure 1.2-7 BWRX-300 Plan View, Elevation 20.0m

Figure 1.2-8 Reactor Building Plan View, Elevation -33.4m

Figure 1.2-9 Reactor Building Plan View, Elevation -29.0m

Figure 1.2-10 Reactor Building Plan View, Elevation -21.0m

Figure 1.2-11 Reactor Building Plan View, Elevation -14.5m

Figure 1.2-12 Reactor Building Plan View, Elevation -8.5m

Figure 1.2-13 Reactor Building Plan View, Elevation 0m

Figure 1.2-14 Reactor Building Plan View, Elevation 7.5m

Figure 1.2-15 Reactor Building Plan View, Elevation 15m

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. 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

TVA is the holder of an Early Site Permit (ESP) at the CRN Site; the ESP has been approved as indicated in the Final Safety Evaluation Report for the CRN Site (ML19162A157). TVA is the applicant for the construction permit at the CRN Site.

TVA is the United States' largest public power provider. It was established by Congress in 1933 to, among other things, improve navigation on the Tennessee River, reduce the damage from destructive floodwaters within the Tennessee River system and downstream on the lower Ohio and Mississippi Rivers, further the economic development of TVA's service area, and sell the electricity generated at the facilities TVA operates. TVA's service territory includes most of Tennessee and parts of Alabama, Georgia, Kentucky, Mississippi, North Carolina, and Virginia.

1.4.2 Principal Contractors and Participants

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

TVA 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 CRN-1. Operating experience from design, construction, and operation of earlier GVH BWRs is applied in the design, construction, and operation of the BWRX-300.

The contractor responsible for the construction of the project will be Bechtel North American Power Corporation (Bechtel). Bechtel is responsible for procurement, construction, and support of commissioning activities. Additional design engineering for selected site-specific portions of the plant will be performed by Sargent and Lundy, LLC (S&L). These contractors were selected based on experience and proven technical capability in nuclear construction projects or projects of similar scope and complexity.

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 Bechtel North American Power Corporation

Bechtel is one of the foremost architect-engineering firms in the United States, with the design and construction of 150 nuclear power projects in 25 countries to its credit. Bechtel has a history of supporting the nuclear power industry, beginning with the construction in 1950 of the EBR-1 reactor. Since then, Bechtel has engineered and constructed more than 60,000 MWe of nuclear power capacity worldwide. Bechtel is the nation’s largest power contractor and is headquartered in San Francisco, California.

1.4.2.3 Sargent and Lundy, LLC

Since 1891, S&L has been a global leader in engineering and design for the power industry. In North America, S&L is currently providing services to over 90 operating nuclear units. Additionally, S&L has been providing international support for the power industry since 1917 with project experience spanning over 90 countries on five continents. S&L is headquartered in Chicago, Illinois.

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.

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.

Incorporation of the SSAR is addressed in Section 1.8.

1.6.1 Topical Reports

The following Topical Reports are 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.
- NNP-TR-001-NP-A, “Quality Assurance Program Description for TVA New Nuclear,” (Reference 1.6-3), describes the quality program for the site-selection, design, construction, and operation of CRN-1.

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, Revision 1, June 1999.

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

1.6-3 NNP-TR-001-NP-A, “Quality Assurance Program Description for TVA New Nuclear,” Tennessee Valley Authority, Revision 2, September 15, 2025 (ML25258A031).

1.7 Drawings and Other Detailed Information

This 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 Simplified Mechanical Diagrams
(Sheet 1 of 2)

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
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

Table 1.7-2 Simplified Mechanical Diagrams
(Sheet 2 of 2)

Title	PSAR Figure
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

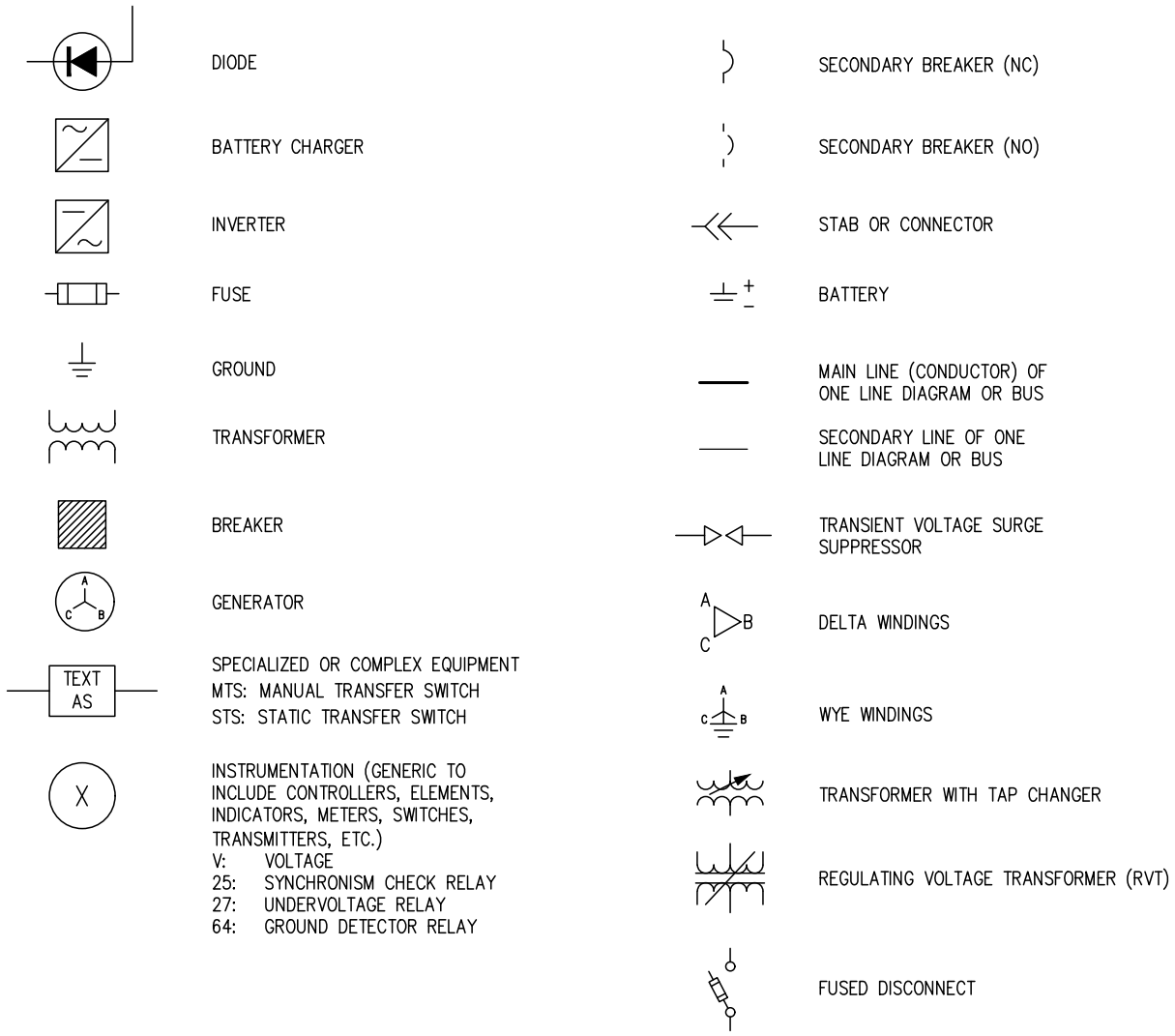
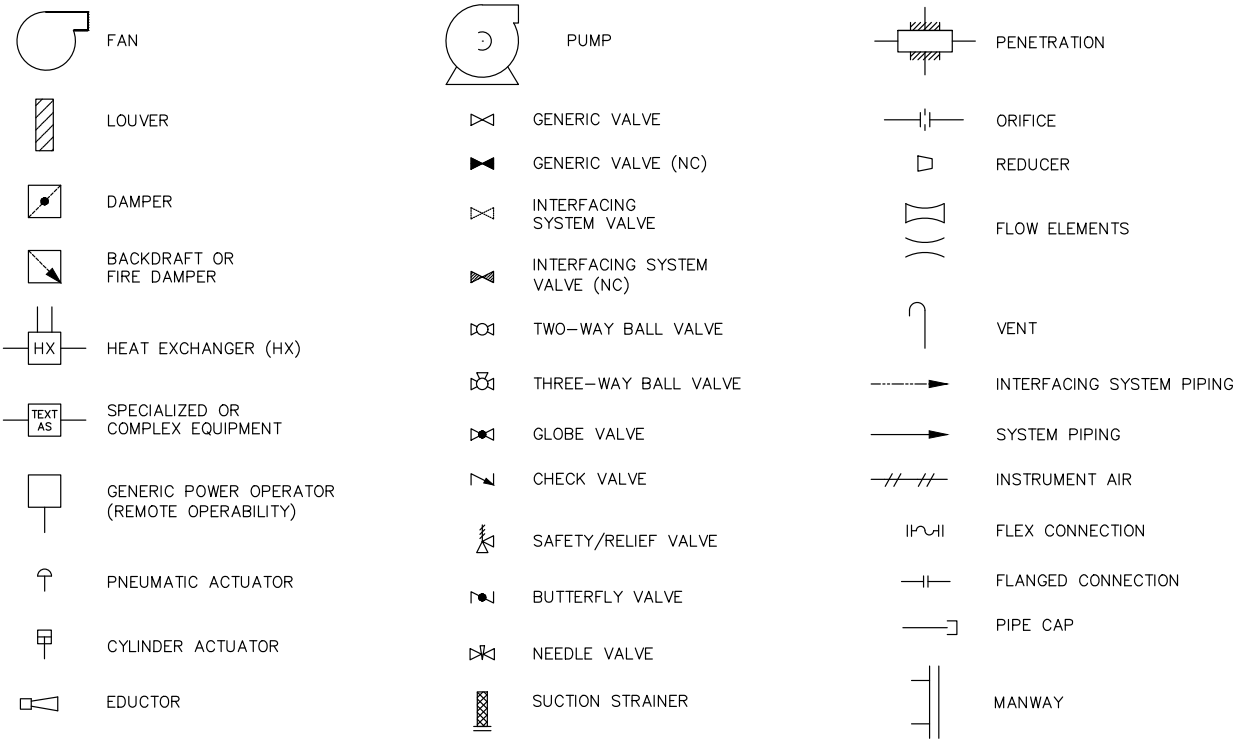


Figure 1.7-1 Simplified Electrical Diagram Symbol Guide



Note:

Additional HVAC symbols will be added for Figures 9.4-1 thru 9.4-6 as detail design progresses.

Figure 1.7-2 Simplified Mechanical Diagram Symbol Guide

1.8 Incorporation by Reference of the Site Safety Analysis Report

As stated in Section 1.1, portions of the “Clinch River Nuclear Site Early Site Permit Application Part 2, Site Safety Analysis Report,” (SSAR) (Reference 1.8-1), are incorporated by reference in this PSAR. Table 1.8-1 lists SSAR sections, the applicable PSAR section to which each SSAR applies, and the SSAR information that is incorporated by reference into this PSAR.

1.8.1 References

1.8-1 “Clinch River Nuclear Site Early Site Permit Application Part 2, Site Safety Analysis Report,” Tennessee Valley Authority, Revision 2, January 2019.

**Table 1.8-1 Site Safety Analysis Report Information Incorporated by Reference into Preliminary Safety Analysis Report
(Sheet 1 of 3)**

SSAR Section	Title	SSAR Information Incorporated by Reference into PSAR	PSAR Section
Ch 1	Introduction and General Description of the Plant		
1.1	Introduction	None, SSAR Section 1.1 provides general information related to the ESP proceeding and is not applicable to any particular PSAR section.	Not Applicable (N/A)
1.2	General Site Description	None, this SSAR section is not Incorporated by Reference into the PSAR.	N/A
1.3	Comparison with Other Facilities	None, this SSAR section is not Incorporated by Reference into the PSAR.	N/A
1.4	Identification of Agents and Contractors	None, this SSAR section is not Incorporated by Reference into the PSAR.	N/A
1.5	Requirements for Additional Technical Information	None, this SSAR section is not Incorporated by Reference into the PSAR.	N/A
1.6	Material Referenced	None, this SSAR section is not Incorporated by Reference into the PSAR.	N/A
1.7	Drawings and Other Detailed Information	None, this SSAR section is not Incorporated by Reference into the PSAR.	N/A
1.8	Interfaces with Standard Design	None, this SSAR section is not Incorporated by Reference into the PSAR.	N/A
1.9	Conformance with Regulatory Criteria	None, this SSAR section is not Incorporated by Reference into the PSAR.	N/A
1.10	Impact of Construction of New NPP Units on Operating Units at Multi-Unit Sites	None, this SSAR section is not Incorporated by Reference into the PSAR.	N/A
1.11	Overview of Reactor Types	None, this SSAR section is not Incorporated by Reference into the PSAR.	N/A

Table 1.8-1 Site Safety Analysis Report Information Incorporated by Reference into Preliminary Safety Analysis Report (Sheet 2 of 3)

SSAR Section	Title	SSAR Information Incorporated by Reference into PSAR	PSAR Section
Ch 2	Site Characteristics		
2.0	Plant Parameter Envelope	This SSAR Section is Incorporated by Reference into the PSAR with supplements.	2.0
2.1	Geography and Demography	This SSAR Section is Incorporated by Reference into the PSAR with supplements.	2.1
2.2	Nearby Industrial, Transportation, and Military Facilities	This SSAR Section is Incorporated by Reference into the PSAR with supplements.	2.2
2.3	Meteorology	This SSAR Section is Incorporated by Reference into the PSAR with supplements.	2.3
2.4	Hydrologic Engineering	This SSAR Section is Incorporated by Reference into the PSAR with supplements.	2.4
2.5	Geology, Seismology, and Geotechnical Engineering	This SSAR Section is Incorporated by Reference into the PSAR with supplements.	2.5
Ch 3	Design of Structures, Systems, Components, and Equipment		
3.5.1.6	Aircraft Hazards	None, this SSAR section is not Incorporated by Reference into the PSAR.	N/A
Ch 11	Radioactive Waste Management		
11.2.3	Liquid Radioactive Releases	None, this SSAR section is not Incorporated by Reference into the PSAR.	N/A
11.3.3	Gaseous Radioactive Releases	None, this SSAR section is not Incorporated by Reference into the PSAR.	N/A
Ch 13	Conduct of Operations		
13.3	Emergency Preparedness	This SSAR section is Incorporated by Reference into the PSAR with supplements.	13.3
13.6	Physical Security	This SSAR section is Incorporated by Reference into the PSAR with supplements.	13.6

**Table 1.8-1 Site Safety Analysis Report Information Incorporated by Reference into Preliminary Safety Analysis Report
(Sheet 3 of 3)**

SSAR Section	Title	SSAR Information Incorporated by Reference into PSAR	PSAR Section
Ch 15	Accident Analysis		
Ch 15	Accident Analysis	None, this SSAR section is not Incorporated by Reference into the PSAR.	N/A
Ch 17	Quality Assurance		
17.5	Quality Assurance Program Description-Design Certification, Early Site Permit and New License Applicants	None, this SSAR section is not Incorporated by Reference into the PSAR.	N/A

1.9 Regulatory Conformance

This section addresses compliance with regulatory requirements 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

With the exception of Table 1.9-7, Table 1.9-1 through Table 1.9-19 summarize conformance with SRP Acceptance Criteria provided in NUREG-0800. The tables list each SRP, revision, title, SRP Acceptance Criteria, topic, conformance, and PSAR section. SRP Acceptance Criteria are based on latest revision at the time of this submittal; draft revisions are not addressed. The PSAR Section column identifies where each criterion is addressed, if applicable. If the criterion is not applicable, the PSAR Section column includes N/A with an explanation provided in the Conformance column. The Conformance column summarizes alternatives to an SRP Acceptance Criteria. Branch Technical Positions (BTP) are included in the table and dispositioned in the manner of SRP Acceptance Criteria.

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 summarizes conformance with RGs. The table lists each RG, revision, title, conformance, alternatives, and PSAR section. The table is based on the RG revision as of January 2025. The table does not include RG drafts, those released for trial use, or those that have been withdrawn. The PSAR Section column identifies where the RG is addressed, if applicable. If the RG is not applicable, the PSAR Section column includes N/A with an explanation provided in the Alternatives column. The Alternatives column also identifies partial conformance with an explanation; a blank indicates that there are no alternatives.

The conformance evaluation was performed on the following Regulatory Guide Divisions:

- Power Reactors (Division 1)
- Environmental and Siting (Division 4)
- Materials and Plant Protection (Division 5)
- Occupational Health (Division 8)

The following Regulatory Guide Divisions do not apply to the SAR of a power reactor and are not included in Table 1.9-20:

- Research and Test Reactors (Division 2)
- Fuels and Materials Facilities (Division 3)
- Products (Division 6)
- Transportation (Division 7)
- Antitrust and Financial Review (Division 9)
- General (Division 10)

1.9.4 Compliance with Three Mile Island Requirements

10 CFR 50.34(f) describes Three Mile Island (TMI) additional requirements for light water reactor construction permit applicants. The TMI requirements specify information to demonstrate that 10 CFR 50.34(f)(1) and 10 CFR 50.34(f)(2) required actions are satisfactorily completed, should be available as part of the FSAR, or by the Operating License stage, respectively. In certain cases, TVA has elected to provide TMI compliance information in the PSAR to support related information in other PSAR chapters.

In addition, 10 CFR 50.34(f) requirements specify that applicants for design certifications and combined licenses shall demonstrate compliance with the technically relevant portions of the requirements in paragraphs (f)(1) through (f)(3), except for paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v). Section 4.1.1 of NEDC-33910P-A, “BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection,” (Reference 1.9-2), identifies and provides a basis for the TMI containment requirements that are not technically relevant. The USNRC acknowledged in the Safety Evaluation that certain TMI requirements were not technically relevant subject to confirmation for a final design. An exemption is not considered necessary because 10 CFR 50.34(f) was intended for traditional LWRs, in lieu of LWRs and non-LWRs with simplified designs and passive safety systems.

TMI requirements that are not technically relevant to the BWRX-300 design are identified in Table 1.9-21.

Table 1.9-21 presents compliance with TMI Requirements as prescribed by 10 CFR 50.34(f). The table provides the regulation subsection, description of the requirement, compliance status, comments, and PSAR section. The PSAR Section column identifies where the regulation is addressed, if applicable. If the regulation is not applicable, the PSAR Section column includes N/A with an explanation provided in the Comments column. The Comments column also identifies partial compliance with an explanation. No entry in the Comments column indicates that the design complies to the TMI Requirement without exception.

1.9.5 Conformance with Advanced and Evolutionary Light Water Reactor Design Issues

Conformance with applicable 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.

1.9-2 NEDC-33910P-A, “BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection,” GE-Hitachi Nuclear Energy Americas, LLC, Revision 2, June 2021.

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

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
1.0	2	Introduction and Interfaces	II.1	General requirements	N/A – No specific acceptance criteria	N/A
1.0	2	Introduction and Interfaces	II.2	Regulatory Considerations	FSAR	FSAR
1.0	2	Introduction and Interfaces	II.3	Performance of New Safety Features	FSAR	FSAR

Table 1.9-2 Conformance with NUREG-0800 (Chapter 2 Site Characteristics and Site Parameters)
(Sheet 1 of 15)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
2.0	1	Site Characteristics and Site Parameters	1	Reference to other sections for specific acceptance criteria	N/A – No specific acceptance criteria	N/A
2.0	1	Site Characteristics and Site Parameters	2	COL Application referencing an ESP but not a Design Certification (DC)	N/A – 10 CFR Part 50 CPA	N/A
2.0	1	Site Characteristics and Site Parameters	3	COL Application referencing a DC but not an ESP	N/A – 10 CFR Part 50 CPA	N/A
2.0	1	Site Characteristics and Site Parameters	4	COL Application referencing both an ESP and a DC	N/A – 10 CFR Part 50 CPA	N/A
2.0	1	Site Characteristics and Site Parameters	5	COL Application referencing neither an ESP nor a DC	N/A – 10 CFR Part 50 CPA	N/A
2.1.1	3	Site Location and Description	1	Specification of Location	Conforms	2.1.1
2.1.1	3	Site Location and Description	2	Site Area Map	Conforms	2.1.1
2.1.2	3	Exclusion Area Authority and Control	1	Establishment of Authority	Conforms	2.1.2
2.1.2	3	Exclusion Area Authority and Control	2	Exclusion or Removal of Personnel and Property	Conforms	2.1.2
2.1.2	3	Exclusion Area Authority and Control	3	Proposed and Permitted Activities	Conforms	2.1.2
2.1.3	3	Population Distribution	1.A	Population Data (Latest census and projections)	Conforms	2.1.3
2.1.3	3	Population Distribution	1.B	Population Data (methodology and sources)	Conforms	2.1.3
2.1.3	3	Population Distribution	1.C	Population Data (transient populations)	Conforms	2.1.3
2.1.3	3	Population Distribution	2	Exclusion Area	Conforms	2.1.3

Table 1.9-2 Conformance with NUREG-0800 (Chapter 2 Site Characteristics and Site Parameters)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
2.1.3	3	Population Distribution	3	Low-Population Zone	Conforms	2.1.3
2.1.3	3	Population Distribution	4	Nearest Population Center Boundary	Conforms	2.1.3
2.1.3	3	Population Distribution	5	Population Density	Conforms	2.1.3
2.2.1 - 2.2.2	3	Identification of Potential Hazards in Site Vicinity	1	Data describing locations and distances from the plant of nearby industrial, military, and transportation facilities	Conforms	2.2.1 2.2.2
2.2.1 - 2.2.2	3	Identification of Potential Hazards in Site Vicinity	2	Descriptions of the nature and extent of activities conducted at the site and in its vicinity, including possible hazards	Conforms	2.2.1 2.2.2
2.2.1 - 2.2.2	3	Identification of Potential Hazards in Site Vicinity	3	Statistical data with respect to hazardous materials	Conforms	2.2.1 2.2.2
2.2.3	3	Evaluation of Potential Accidents	1	Event Probability	Conforms	2.2.3
2.2.3	3	Evaluation of Potential Accidents	2	Design-Basis Events	Conforms	2.2.3
2.3.1	3	Regional Climatology	1	Description of the general climate of the region	Conforms	2.3.1
2.3.1	3	Regional Climatology	2	Data on severe weather phenomena	Conforms	2.3.1
2.3.1	3	Regional Climatology	3	Tornado parameters	Conforms	2.3.1
2.3.1	3	Regional Climatology	4	100-year return period 3-second gust wind speed	Conforms	2.3.1
2.3.1	3	Regional Climatology	5	Maximum evaporation and drift loss of water and minimum water cooling	Conforms	2.3.1

Table 1.9-2 Conformance with NUREG-0800 (Chapter 2 Site Characteristics and Site Parameters)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
2.3.1	3	Regional Climatology	6	Winter precipitation loads	Conforms	2.3.1
2.3.1	3	Regional Climatology	7	Ambient temperature and humidity	Conforms	2.3.1
2.3.1	3	Regional Climatology	8	High air pollution potential	Conforms	2.3.1
2.3.1	3	Regional Climatology	9	Other meteorological and air quality conditions	Conforms	2.3.1
2.3.2	3	Local Meteorology	1	Local summaries of meteorological data based on onsite measurements	Conforms	2.3.2
2.3.2	3	Local Meteorology	2	Topographical description of the site and environs out to a distance of 50 miles (80 km)	Conforms	2.3.2
2.3.2	3	Local Meteorology	3	Influence of the plant and its facilities on the local meteorological and air quality conditions	Conforms	2.3.2
2.3.2	3	Local Meteorology	4	Local site airflow including wind roses and annual joint frequency distributions of wind speed and wind direction by atmospheric stability	Conforms	2.3.2
2.3.3	3	Onsite Meteorological Measurements Programs	1	Pre-operational monitoring program	Conforms	2.3.3
2.3.3	3	Onsite Meteorological Measurements Programs	2	Joint frequency distributions of wind speed and wind direction by atmospheric stability class	Conforms	2.3.3

Table 1.9-2 Conformance with NUREG-0800 (Chapter 2 Site Characteristics and Site Parameters)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
2.3.3	3	Onsite Meteorological Measurements Programs	3	Deviations from RG 1.23	Conforms	Table 1.9-20
2.3.4	3	Short-Term Dispersion Estimates for Accident Releases	1	Atmospheric dispersion models used to calculate χ/Q values for accidental releases	Conforms	2.3.4
2.3.4	3	Short-Term Dispersion Estimates for Accident Releases	2	Meteorological data used for the evaluation which represent annual cycles of hourly values of wind direction, wind speed, and atmospheric stability	Conforms	2.3.4
2.3.4	3	Short-Term Dispersion Estimates for Accident Releases	3	Atmospheric diffusion parameters as a function of distance, topography, and atmospheric conditions	Conforms	2.3.4
2.3.4	3	Short-Term Dispersion Estimates for Accident Releases	4	Hourly cumulative frequency distributions of χ/Q values from the effluent release point(s) to the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ)	Conforms	2.3.4
2.3.4	3	Short-Term Dispersion Estimates for Accident Releases	5	Atmospheric dispersion factors used for the assessment of consequences	Conforms	2.3.4
2.3.4	3	Short-Term Dispersion Estimates for Accident Releases	6	For CR habitability analysis, a site plan drawn to scale showing true North and potential atmospheric accident release pathways	Conforms	2.3.4

Table 1.9-2 Conformance with NUREG-0800 (Chapter 2 Site Characteristics and Site Parameters)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
2.3.5	3	Long-Term Atmospheric Dispersion Estimates for Routine Releases	1	Atmospheric dispersion and deposition models used calculate annual average concentrations due to routine releases	Conforms	2.3.5
2.3.5	3	Long-Term Atmospheric Dispersion Estimates for Routine Releases	2	Atmospheric diffusion parameters for estimating the consequences of routine releases from the site boundary to 50 miles (80 km)	Conforms	2.3.5
2.3.5	3	Long-Term Atmospheric Dispersion Estimates for Routine Releases	3	Meteorological data summaries (onsite and regional) used as input to the dispersion and deposition models	Conforms	2.3.5
2.3.5	3	Long-Term Atmospheric Dispersion Estimates for Routine Releases	4	Points of routine release of radioactive material to the atmosphere, including the characteristics (e.g., location, release mode) of each release point.	Conforms	2.3.5
2.3.5	3	Long-Term Atmospheric Dispersion Estimates for Routine Releases	5	Location of potential receptors of interest (e.g., nearest vegetable garden, nearest resident, nearest milk animal, and nearest meat cow in each 22½ degree direction sector within 5 miles (8 km)	Conforms	2.3.5

Table 1.9-2 Conformance with NUREG-0800 (Chapter 2 Site Characteristics and Site Parameters)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
2.3.5	3	Long-Term Atmospheric Dispersion Estimates for Routine Releases	6.A	χ/Q and D/Q for assessment of routine airborne radiological releases and receptor locations	Conforms	2.3.5
2.3.5	3	Long-Term Atmospheric Dispersion Estimates for Routine Releases	6.B	χ/Q and D/Q values for assessment of routine airborne radiological releases for 16 radial sectors to a distance of 50 miles (80 km)	Conforms	2.3.5
2.4.1	3	Hydrologic Description	1	Interface of the Plant with the Hydrosphere	Conforms	2.4.1
2.4.1	3	Hydrologic Description	2	Hydrological Causal Mechanisms	Conforms	2.4.1
2.4.1	3	Hydrologic Description	3	Surface and Ground Water Uses	Conforms	2.4.1
2.4.1	3	Hydrologic Description	4	Data	Conforms	2.4.1
2.4.1	3	Hydrologic Description	5	Alternate Conceptual Models	Conforms	2.4.1
2.4.1	3	Hydrologic Description	6	Consideration of Other Site-Related Evaluation Criteria	Conforms	2.4.1
2.4.2	4	Floods	1	Local Flooding on the Site and Drainage Design	Conforms	2.4.2
2.4.2	4	Floods	2	Stream Flooding	Conforms	2.4.2
2.4.2	4	Floods	3	Surges	N/A – not credible for the CRN Site	N/A
2.4.2	4	Floods	4	Seiches	N/A – not credible for the CRN Site	N/A
2.4.2	4	Floods	5	Tsunami	N/A – not credible for the CRN Site	N/A

Table 1.9-2 Conformance with NUREG-0800 (Chapter 2 Site Characteristics and Site Parameters)
(Sheet 7 of 15)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
2.4.2	4	Floods	6	Seismically Induced Dam Failures (or Breaches)	Conforms	2.4.2
2.4.2	4	Floods	7	Flooding Caused by Landslides	N/A – not credible for the CRN Site	N/A
2.4.2	4	Floods	8	Effects of Ice Formation in Water Bodies	N/A – not credible for the CRN Site	N/A
2.4.2	4	Floods	9	Combined Events Criteria	Conforms	2.4.2
2.4.2	4	Floods	10	Consideration of Other Site-Related Evaluation Criteria	Conforms	2.4.2
2.4.3	4	Probable Maximum Flood on Streams and Rivers	1	Design Bases for Flooding in Streams and Rivers	Conforms	2.4.3
2.4.3	4	Probable Maximum Flood on Streams and Rivers	2	Design Bases for Site Drainage	Conforms	2.4.3
2.4.3	4	Probable Maximum Flood on Streams and Rivers	3	Consideration of Other Site-Related Evaluation Criteria	Conforms	2.4.3
2.4.4	3	Potential Dam Failures	1	Flood waves from severe breaching of an upstream dam	Conforms	2.4.4
2.4.4	3	Potential Dam Failures	2	Domino-type or cascading dam failures	Conforms	2.4.4
2.4.4	3	Potential Dam Failures	3	Dynamic effects on structures	Conforms	2.4.4
2.4.4	3	Potential Dam Failures	4	Loss of water supply due to failure of a downstream dam	Conforms	2.4.4
2.4.4	3	Potential Dam Failures	5	Effects of sediment deposition and erosion	Conforms	2.4.4
2.4.4	3	Potential Dam Failures	6	Failure of onsite water control or storage structures	Conforms	2.4.4

Table 1.9-2 Conformance with NUREG-0800 (Chapter 2 Site Characteristics and Site Parameters)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
2.4.4	3	Potential Dam Failures	7	Consideration of other site-related evaluation criteria	Conforms	2.4.4
2.4.5	3	Probable Maximum Surge and Seiche Flooding	1	Probable maximum hurricane	Conforms	2.4.5
2.4.5	3	Probable Maximum Surge and Seiche Flooding	2	Probable maximum wind storm	Conforms	2.4.5
2.4.5	3	Probable Maximum Surge and Seiche Flooding	3	Seiche and resonance	Conforms	2.4.5
2.4.5	3	Probable Maximum Surge and Seiche Flooding	4	Wave runup	Conforms	2.4.5
2.4.5	3	Probable Maximum Surge and Seiche Flooding	5	Effects of sediment erosion and deposition	Conforms	2.4.5
2.4.5	3	Probable Maximum Surge and Seiche Flooding	6	Consideration of other site-related evaluation criteria	Conforms	2.4.5
2.4.6	3	Probable Maximum Tsunami Hazards	1	Historical tsunami data	N/A – not credible for the CRN Site	N/A
2.4.6	3	Probable Maximum Tsunami Hazards	2	Probable maximum tsunami	N/A – not credible for the CRN Site	N/A
2.4.6	3	Probable Maximum Tsunami Hazards	3	Tsunami propagation models	N/A – not credible for the CRN Site	N/A
2.4.6	3	Probable Maximum Tsunami Hazards	4	Wave runup, inundation, and drawdown	N/A – not credible for the CRN Site	N/A
2.4.6	3	Probable Maximum Tsunami Hazards	5	Hydrostatic and hydrodynamic forces	N/A – not credible for the CRN Site	N/A
2.4.6	3	Probable Maximum Tsunami Hazards	6	Debris and waterborne projectiles	N/A – not credible for the CRN Site	N/A

Table 1.9-2 Conformance with NUREG-0800 (Chapter 2 Site Characteristics and Site Parameters)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
2.4.6	3	Probable Maximum Tsunami Hazards	7	Effects of sediment erosion and deposition	N/A – not credible for the CRN Site	N/A
2.4.6	3	Probable Maximum Tsunami Hazards	8	Consideration of other site-related evaluation criteria	N/A – not credible for the CRN Site	N/A
2.4.7	3	Ice Effects	1	Historical ice accumulation	N/A – No SC1 SSCs subject to ice-induced forces or blockages from sheet or frazil ice.	N/A
2.4.7	3	Ice Effects	2	High and low water levels	N/A – No SC1 SSCs subject to ice-induced forces or blockages from sheet or frazil ice.	N/A
2.4.7	3	Ice Effects	3	Ice sheet formation	N/A – No SC1 SSCs subject to ice-induced forces or blockages from sheet or frazil ice.	N/A
2.4.7	3	Ice Effects	4	Ice-induced forces and blockages	N/A – No SC1 SSCs subject to ice-induced forces or blockages from sheet or frazil ice.	N/A
2.4.7	3	Ice Effects	5	Consideration of other site-related evaluation criteria	N/A – No SC1 SSCs subject to ice-induced forces or blockages from sheet or frazil ice.	N/A
2.4.8	3	Cooling Water Canals and Reservoirs	1	Hydraulic design bases for protection of structures	N/A – CRN Site does not include cooling water canals or reservoirs.	N/A

Table 1.9-2 Conformance with NUREG-0800 (Chapter 2 Site Characteristics and Site Parameters)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
2.4.8	3	Cooling Water Canals and Reservoirs	2	Hydraulic design bases of canals	N/A – CRN Site does not include cooling water canals or reservoirs.	N/A
2.4.8	3	Cooling Water Canals and Reservoirs	3	Hydraulic design bases of reservoirs	N/A – CRN Site does not include cooling water canals or reservoirs.	N/A
2.4.8	3	Cooling Water Canals and Reservoirs	4	Consideration of other site-related evaluation criteria	N/A – CRN Site does not include cooling water canals or reservoirs.	N/A
2.4.9	3	Channel Diversions	1	Historical channel diversions	Conforms	2.4.9
2.4.9	3	Channel Diversions	2	Regional topographic evidence	Conforms	2.4.9
2.4.9	3	Channel Diversions	3	Ice causes	Conforms	2.4.9
2.4.9	3	Channel Diversions	4	Flooding of site due to channel diversions	Conforms	2.4.9
2.4.9	3	Channel Diversions	5	Human-induced causes of channel diversion	Conforms	2.4.9
2.4.9	3	Channel Diversions	6	Alternate water sources	Conforms	2.4.9
2.4.9	3	Channel Diversions	7	Consideration of other site-related evaluation criteria	Conforms	2.4.9
2.4.10	3	Flooding Protection Requirements	1	Safety-related facilities exposed to flooding	Conforms	2.4.10
2.4.10	3	Flooding Protection Requirements	2	Type of flood protection	Conforms	2.4.10
2.4.10	3	Flooding Protection Requirements	3	Emergency procedures	Conforms	2.4.10
2.4.10	3	Flooding Protection Requirements	4	Consideration of other site-related evaluation criteria	Conforms	2.4.10

Table 1.9-2 Conformance with NUREG-0800 (Chapter 2 Site Characteristics and Site Parameters)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
2.4.11	3	Low Water Considerations	1	Low water from drought	Conforms	2.4.11
2.4.11	3	Low Water Considerations	2	Low water from other phenomena	Conforms	2.4.11
2.4.11	3	Low Water Considerations	3	Effect of low water on safety-related water supply	Conforms	2.4.11
2.4.11	3	Low Water Considerations	4	Water use limits	Conforms	2.4.11
2.4.11	3	Low Water Considerations	5	Consideration of other site-related evaluation criteria	Conforms	2.4.11
2.4.12	3	Groundwater	1	Local and regional groundwater characteristics and use	Conforms	2.4.12
2.4.12	3	Groundwater	2	Effects on plant foundations and other safety-related structures, systems, and components	Conforms	2.4.12
2.4.12	3	Groundwater	3	Reliability of groundwater resources and systems used for safety-related purposes	Conforms	2.4.12
2.4.12	3	Groundwater	4	Reliability of dewatering systems	Conforms	2.4.12
2.4.12	3	Groundwater	5	Consideration of other site-related evaluation criteria	Conforms	2.4.12
2.4.13	3	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	1	Alternate conceptual models	Conforms	2.4.13
2.4.13	3	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	2	Pathways	Conforms	2.4.13

Table 1.9-2 Conformance with NUREG-0800 (Chapter 2 Site Characteristics and Site Parameters)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
2.4.13	3	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	3	Characteristics that affect transport	Conforms	2.4.13
2.4.13	3	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	4	Consideration of other site-related evaluation criteria	Conforms	2.4.13
2.4.13	3	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	5	BTP 11-6	Conforms	2.4.13
2.4.14	3	Technical Specifications and Emergency Operation Requirements	1	Bases for emergency actions	Conforms	2.4.14
2.4.14	3	Technical Specifications and Emergency Operation Requirements	2	Available response time	Conforms	2.4.14
2.4.14	3	Technical Specifications and Emergency Operation Requirements	3	TS	Conforms	2.4.14
2.4.14	3	Technical Specifications and Emergency Operation Requirements	4	Consideration of other site-related evaluation criteria	Conforms	2.4.14
2.4.14	3	Technical Specifications and Emergency Operation Requirements	5	10 CFR 50, Appendix A, GDC 44, as it relates to providing an ultimate heat sink for normal operating and accident conditions.	Conforms	2.4.14

Table 1.9-2 Conformance with NUREG-0800 (Chapter 2 Site Characteristics and Site Parameters)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
2.5.1	5	Geologic Characterization Information	1	Regional geology	Conforms	2.5.1
2.5.1	5	Geologic Characterization Information	2	Local geology	Conforms	2.5.1
2.5.2	5	Vibratory Ground Motion	1	Seismicity	Conforms	2.5.2
2.5.2	5	Vibratory Ground Motion	2	Geologic and tectonic characteristics of site and region	Conforms	2.5.2
2.5.2	5	Vibratory Ground Motion	3	Correlation of earthquake activity with seismic sources	Conforms	2.5.2
2.5.2	5	Vibratory Ground Motion	4	Probabilistic seismic hazard analysis and controlling earthquakes	Conforms	2.5.2
2.5.2	5	Vibratory Ground Motion	5	Seismic Wave Transmission Characteristics of the Site	Conforms	2.5.2
2.5.2	5	Vibratory Ground Motion	6	Ground Motion Response Spectra	Conforms	2.5.2
2.5.3	6	Surface Deformation	1	Geologic, seismic, and geophysical investigations	Conforms	2.5.3
2.5.3	6	Surface Deformation	2	Geologic evidence for surface deformation	Conforms	2.5.3
2.5.3	6	Surface Deformation	3	Timing of deformation	Conforms	2.5.3
2.5.3	6	Surface Deformation	4	Correlation of earthquakes with tectonic features	Conforms	2.5.3
2.5.3	6	Surface Deformation	5	Relationship of geologic features in the site vicinity to regional geologic features	Conforms	2.5.3

Table 1.9-2 Conformance with NUREG-0800 (Chapter 2 Site Characteristics and Site Parameters)
(Sheet 14 of 15)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
2.5.3	6	Surface Deformation	6	Potential for surface deformation at the site	Conforms	2.5.3
2.5.4	5	Stability of Subsurface Materials and Foundations	1	Geologic features	Conforms	2.5.4
2.5.4	5	Stability of Subsurface Materials and Foundations	2	Properties of subsurface materials	Conforms	2.5.4
2.5.4	5	Stability of Subsurface Materials and Foundations	3	Foundation interfaces	Conforms	2.5.4
2.5.4	5	Stability of Subsurface Materials and Foundations	4	Geophysical surveys	Conforms	2.5.4
2.5.4	5	Stability of Subsurface Materials and Foundations	5	Excavation and backfill	Conforms	2.5.4
2.5.4	5	Stability of Subsurface Materials and Foundations	6	Ground water conditions	Conforms	2.5.4
2.5.4	5	Stability of Subsurface Materials and Foundations	7	Response of soil and rock to dynamic loading	Conforms	2.5.4
2.5.4	5	Stability of Subsurface Materials and Foundations	8	Liquefaction potential	Conforms	2.5.4
2.5.4	5	Stability of Subsurface Materials and Foundations	9	Earthquake design basis	Conforms	2.5.4
2.5.4	5	Stability of Subsurface Materials and Foundations	10	Static and dynamic stability	Conforms	2.5.4
2.5.4	5	Stability of Subsurface Materials and Foundations	11	Design criteria	Conforms	2.5.4
2.5.4	5	Stability of Subsurface Materials and Foundations	12	Techniques to improve subsurface conditions	Conforms	2.5.4
2.5.5	5	Stability of Slopes	1	Slope characteristics	Conforms	2.5.5

Table 1.9-2 Conformance with NUREG-0800 (Chapter 2 Site Characteristics and Site Parameters)
(Sheet 15 of 15)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
2.5.5	5	Stability of Slopes	2	Design criteria and analyses	Conforms	2.5.5
2.5.5	5	Stability of Slopes	3	Boring logs	Conforms	2.5.5
2.5.5	5	Stability of Slopes	4	Compacted fill	Conforms	2.5.5

Table 1.9-3 Conformance with NUREG-0800 (Chapter 3 Design of Structures, Components, Equipment, and Systems) (Sheet 1 of 21)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
3.2.1	3	Seismic Classification	1	RG 1.29 guidance regarding seismic design classification	Alternate Approach	Table 1.9-20
3.2.1	3	Seismic Classification	2	RG 1.151 guidance regarding seismic design provisions and classification of safety-related instrument sensing lines.	Conforms	Table 1.9-20
3.2.1	3	Seismic Classification	3	RG 1.143 guidance regarding seismic design of radioactive waste management SSCs to meet the requirements of GDC 2 and GDC 60	Conforms	Table 1.9-20
3.2.1	3	Seismic Classification	4	RG 1.189 guidance regarding design provisions of fire protection SSCs, including seismic design to meet the requirements of GDC 2	Alternate Approach	Table 1.9-20
3.2.2	3	System Quality Group Classification	1	RG 1.26 guidance regarding quality standards for Quality Group B, C and D water and steam-containing components important to safety.	Conforms	Table 1.9-20
3.3.1	3	Wind Loadings	1	Most severe wind that has been historically reported for the site and surrounding area	Conforms	3.3.1

Table 1.9-3 Conformance with NUREG-0800 (Chapter 3 Design of Structures, Components, Equipment, and Systems) (Sheet 2 of 21)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
3.3.1	3	Wind Loadings	2	Acceptance criteria for the design wind speed, recurrence interval, speed variation with height, gust factors, and their bases	Alternate Approach	3.3.1
3.3.1	3	Wind Loadings	3.A	Transformation of wind speed into equivalent pressure on structures or parts; velocity pressure	Alternate Approach	3.3.1
3.3.1	3	Wind Loadings	3.B	Transformation of wind speed into equivalent pressure on structures or parts; upwind exposure category	Alternate Approach	3.3.1
3.3.1	3	Wind Loadings	3.C	Transformation of wind speed into equivalent pressure on structures or parts; design wind loads	Alternate Approach	3.3.1
3.3.2	3	Tornado Loadings	1	Most severe wind that has been historically reported for the site and surrounding area	Conforms	3.3.2
3.3.2	3	Tornado Loadings	2	Acceptance criteria for the design wind speed, recurrence interval, speed variation with height, gust factors, and their bases	Conforms	3.3.2

Table 1.9-3 Conformance with NUREG-0800 (Chapter 3 Design of Structures, Components, Equipment, and Systems) (Sheet 3 of 21)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
3.3.2	3	Tornado Loadings	3.A	Transformation of wind speed into equivalent pressure on structures or parts; characteristics and effects	Conforms	3.3.2
3.3.2	3	Tornado Loadings	3.B	Transformation of wind speed into equivalent pressure on structures or parts; wind effects	Alternate Approach	3.3.2
3.3.2	3	Tornado Loadings	3.C	Transformation of wind speed into equivalent pressure on structures or parts; atmospheric pressure changed effects	Conforms	3.3.2
3.3.2	3	Tornado Loadings	3.D	Transformation of wind speed into equivalent pressure on structures or parts; missile impact effects	Conforms	3.3.2
3.3.2	3	Tornado Loadings	3.E	Transformation of wind speed into equivalent pressure on structures or parts; combined tornado effects	Conforms	3.3.2
3.3.2	3	Tornado Loadings	4.A	Postulated failure of structures or components not designed for wind loads or missiles does not damage safety-related SSCs	Conforms	3.3.2

Table 1.9-3 Conformance with NUREG-0800 (Chapter 3 Design of Structures, Components, Equipment, and Systems) (Sheet 4 of 21)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
3.3.2	3	Tornado Loadings	4.B	Capability of SSCs affected by structures or components not designed for wind loads: safety-related structures resist effects of damage to structures or components not designed for wind loads or missiles	Conforms	3.3.2
3.4.1	3	Internal Flood Protection for Onsite Equipment Failures	1	Compliance with GDC 2	Conforms	3.4.1
3.4.1	3	Internal Flood Protection for Onsite Equipment Failures	2	Compliance with GDC 4	Conforms	3.4.1
3.4.2	3	Analysis Procedures	1	Use of most severe flood and groundwater effects with margin	Conforms	3.4.2
3.4.2	3	Analysis Procedures	2	Consideration of hydrostatic pressure	Conforms	3.4.2
3.4.2	3	Analysis Procedures	3	Consideration of dynamic loads	N/A – flood level is not above proposed plant grade.	N/A
3.5.1.1	3	Internally Generated Missiles (Outside Containment)	1	Statistical significance	Conforms	3.5.1
3.5.1.1	3	Internally Generated Missiles (Outside Containment)	2	Missile protection	Conforms	3.5.1
3.5.1.2	3	Internally Generated Missiles (Inside Containment)	1	Statistical significance	Conforms	3.5.1
3.5.1.2	3	Internally Generated Missiles (Inside Containment)	2	Missile protection	Conforms	3.5.1

Table 1.9-3 Conformance with NUREG-0800 (Chapter 3 Design of Structures, Components, Equipment, and Systems) (Sheet 5 of 21)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
3.5.1.3	3	Turbine Missiles	1	Probability of unacceptable damage	Conforms	3.5.1
3.5.1.3	3	Turbine Missiles	2	Operating Experience (OPEX)	FSAR	FSAR
3.5.1.3	3	Turbine Missiles	3	Periodic testing and inspection	FSAR	FSAR
3.5.1.3	3	Turbine Missiles	4	Criteria for turbine manufacturers that have USNRC -approved reports	FSAR	FSAR
3.5.1.3	3	Turbine Missiles	5	Criteria for turbine manufacturers that do not have USNRC -approved reports	FSAR	FSAR
3.5.1.3	3	Turbine Missiles	6	Acceptance criteria for barriers used for missile protection	FSAR	FSAR
3.5.1.4	4	Missiles Generated by Tornadoes and Extreme Winds	1	Conformance with RG 1.76	Conforms	Table 1.9-20
3.5.1.4	4	Missiles Generated by Tornadoes and Extreme Winds	2	Conformance with RG 1.221	Conforms	Table 1.9-20
3.5.1.4	4	Missiles Generated by Tornadoes and Extreme Winds	3	Method for identifying design-basis missiles generated by natural phenomena	Conforms	3.5.1
3.5.1.5	4	Site Proximity Missiles (Except Aircraft)	1	Probability of radiological consequences exceeding 10 CFR 100	Conforms	3.5.1
3.5.1.5	4	Site Proximity Missiles (Except Aircraft)	2	Compliance with GDC 4 and 10 CFR 100	N/A – the first criterion is met.	N/A

Table 1.9-3 Conformance with NUREG-0800 (Chapter 3 Design of Structures, Components, Equipment, and Systems) (Sheet 6 of 21)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
3.5.1.6	4	Aircraft Hazards	1	Probability of aircraft accidents resulting in radiological exposure > 10 CFR 100 guidelines is < 1E-7/yr	Alternate Approach	3.5.1
3.5.1.6	4	Aircraft Hazards	2	Detailed review of aircraft hazards	Alternate Approach	3.5.1
3.5.2	3	Structures, Systems, and Components to be Protected from Externally-Generated Missiles	-	SSCs required to safely shut down the reactor and maintain it in a safe condition. Conformance with: RG 1.13 RG 1.27 RG 1.115 RG 1.117	 Alternate Approach Not applicable Alternate Approach Conforms	Table 1.9-20
3.5.3	3	Barrier Design Procedures	1.A	Local damage prediction: concrete	Conforms	3.5.3
3.5.3	3	Barrier Design Procedures	1.B	Local damage prediction: steel	Conforms	3.5.3
3.5.3	3	Barrier Design Procedures	1.C	Local damage prediction: composite	Conforms	3.5.3
3.5.3	3	Barrier Design Procedures	2	Overall damage prediction	Conforms	3.5.3
3.6.1	3	Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	1	Separation of high and moderate energy fluid systems from essential systems and components per BTP 3-3	Conforms	3.6.1 3.6.2

Table 1.9-3 Conformance with NUREG-0800 (Chapter 3 Design of Structures, Components, Equipment, and Systems) (Sheet 7 of 21)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
3.6.1	3	Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	2	Enclosure of high and moderate energy fluid systems per BTP 3-3	Conforms	3.6.1 3.6.2
3.6.1	3	Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	3	Situations where neither physical separation nor protective enclosure is practical	Conforms	3.6.1 3.6.2
3.6.1	3	Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	4	Design features per BTP 3-3	Conforms	3.6.1 3.6.2
3.6.1	3	Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	5	Effect of postulated failures on shutdown capability per BTP 3-3	Conforms	3.6.1 3.6.2
3.6.2	3	Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping	1	Postulated pipe rupture locations inside containment per BTP 3-4	Conforms	3.6.1 3.6.2
3.6.2	3	Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping	2	Postulated pipe rupture locations outside containment per BTP 3-4	Conforms	3.6.1 3.6.2
3.6.2	3	Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping	3	Methods of analysis	Conforms	3.6.2

Table 1.9-3 Conformance with NUREG-0800 (Chapter 3 Design of Structures, Components, Equipment, and Systems) (Sheet 8 of 21)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
3.6.3	1	Leak-Before-Break Evaluation Procedures	N/A	Leak-Before-Break Evaluation Procedures	N/A – Leak-before-Break is not a design strategy being used on BWRX-300 design.	N/A
3.7.1	4	Seismic Design Parameters	1	Design ground motion	Conforms	3.7.1
3.7.1	4	Seismic Design Parameters	2	Percentage of critical damping values per RG 1.61	Conforms	Table 1.9-20
3.7.1	4	Seismic Design Parameters	3	Supporting media for Seismic Category I structures	Conforms	3.7.1
3.7.1	4	Seismic Design Parameters	4	Review Considerations for DC and COL Applications	N/A – 10 CFR Part 50 CPA	N/A
3.7.2	4	Seismic System Analysis	1	Seismic analysis methods	Conforms	3.7.2
3.7.2	4	Seismic System Analysis	2	Natural frequencies and responses	Conforms	3.7.2
3.7.2	4	Seismic System Analysis	3	Procedures for analytical modeling	Conforms	3.7.2
3.7.2	4	Seismic System Analysis	4	Soil-Structure Interaction (SSI)	Conforms	3.7.2
3.7.2	4	Seismic System Analysis	5	Development of In-Structure Response Spectra	Conforms	3.7.2
3.7.2	4	Seismic System Analysis	6	Three components of design ground motion per RG 1.92	Conforms	Table 1.9-20
3.7.2	4	Seismic System Analysis	7	Combination of modal responses per RG 1.92	Conforms	Table 1.9-20
3.7.2	4	Seismic System Analysis	8	Interaction of Non-Seismic Category I structures with Seismic Category I SSCs	Conforms	3.7.2

Table 1.9-3 Conformance with NUREG-0800 (Chapter 3 Design of Structures, Components, Equipment, and Systems) (Sheet 9 of 21)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
3.7.2	4	Seismic System Analysis	9	Effects of parameter variations on floor response spectra	Conforms	3.7.2
3.7.2	4	Seismic System Analysis	10	Use of equivalent vertical static factors	Conforms	3.7.2
3.7.2	4	Seismic System Analysis	11	Methods to account for torsional effects	Conforms	3.7.2
3.7.2	4	Seismic System Analysis	12	Comparison of responses	N/A	N/A
3.7.2	4	Seismic System Analysis	13	Analysis procedures for damping	Conforms	3.7.2
3.7.2	4	Seismic System Analysis	14	Determination of overturning moments, sliding forces, structure-to-soil pressures, and frictional forces Seismic Category I structures	Conforms	3.7.2
3.7.3	4	Seismic Subsystem Analysis	1	Seismic analysis methods	Conforms	3.7.3
3.7.3	4	Seismic Subsystem Analysis	2	Earthquake cycles	Conforms	3.7.3
3.7.3	4	Seismic Subsystem Analysis	3	Procedures for analytical modeling	Conforms	3.7.3
3.7.3	4	Seismic Subsystem Analysis	4	Basis for selection of frequencies	Conforms	3.7.3
3.7.3	4	Seismic Subsystem Analysis	5	Analysis procedure for damping	Conforms	3.7.3
3.7.3	4	Seismic Subsystem Analysis	6	Three components of design ground motion	Conforms	3.7.3
3.7.3	4	Seismic Subsystem Analysis	7	Combination of model responses	Conforms	3.7.3
3.7.3	4	Seismic Subsystem Analysis	8	Interaction of Non-Seismic Category I subsystems with Seismic Category I SSCs	Conforms	3.7.3

Table 1.9-3 Conformance with NUREG-0800 (Chapter 3 Design of Structures, Components, Equipment, and Systems) (Sheet 10 of 21)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
3.7.3	4	Seismic Subsystem Analysis	9	Multiply supported equipment and components with distinct inputs	Conforms	3.7.3
3.7.3	4	Seismic Subsystem Analysis	10	Vertical static factors	Conforms	3.7.3
3.7.3	4	Seismic Subsystem Analysis	11	Torsional effects of eccentric masses	Conforms	3.7.3
3.7.3	4	Seismic Subsystem Analysis	12	Seismic Category I buried piping, conduits, and tunnels	N/A – BWRX-300 design does not include Seismic Category I buried pipes, conduits, or tunnels.	N/A
3.7.3	4	Seismic Subsystem Analysis	13	Methods for seismic analysis of Seismic Category I concrete dams	N/A – BWRX-300 design does not include nor require the presence of a dam.	N/A
3.7.3	4	Seismic Subsystem Analysis	14	Methos of seismic analysis of above-ground tanks	N/A – BWRX-300 design does not include Seismic Category I above-ground tanks.	N/A
3.7.4	3	Seismic Instrumentation	1	Conformance with RG 1.12	Conforms	Table 1.9-20
3.7.4	3	Seismic Instrumentation	2	Conformance with RG 1.166	Conforms	Table 1.9-20
3.7.4	3	Seismic Instrumentation	3	Compliance with 10 CFR 20.1101	Conforms	3.7.4
3.8.1	4	Concrete containment	1	Containment description	Conforms	3.8.1
3.8.1	4	Concrete containment	2	Applicable codes, standards, and specifications	Alternate Approach	3.8.1
3.8.1	4	Concrete containment	3	Loads and loading combinations	Conforms	3.8.1
3.8.1	4	Concrete containment	4	Design and analysis procedures	Conforms	3.8.1

Table 1.9-3 Conformance with NUREG-0800 (Chapter 3 Design of Structures, Components, Equipment, and Systems) (Sheet 11 of 21)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
3.8.1	4	Concrete containment	5	Structural acceptance criteria	Alternate Approach	3.8.1
3.8.1	4	Concrete containment	6	Materials, quality control, and special construction techniques	Alternate Approach	3.8.1
3.8.1	4	Concrete containment	7	Testing and inservice surveillance requirements	Alternate Approach	3.8.1
3.8.2	3	Steel containment	1	Containment description	Conforms	3.8.2
3.8.2	34	Steel containment	2	Applicable codes, standards, and specifications	Conforms	3.8.2
3.8.2	3	Steel containment	3	Loads and loading combinations	Conforms	3.8.2
3.8.2	3	Steel containment	4	Design and analysis procedures	Conforms	3.8.2
3.8.2	3	Steel containment	5	Structural acceptance criteria	Conforms	3.8.2
3.8.2	3	Steel containment	6	Materials, quality control, and special construction techniques	Conforms	3.8.2
3.8.2	3	Steel containment	7	Testing and inservice surveillance requirements	Conforms	3.8.2
3.8.3	4	Concrete and Steel Internal Structures of Steel or Concrete Containments	1	Internal structures description	Conforms	3.8.3
3.8.3	4	Concrete and Steel Internal Structures of Steel or Concrete Containments	2	Applicable codes, standards, and specifications	Alternate Approach	3.8.3
3.8.3	4	Concrete and Steel Internal Structures of Steel or Concrete Containments	3	Loads and load combinations	Conforms	3.8.3
3.8.3	4	Concrete and Steel Internal Structures of Steel or Concrete Containments	4	Design and analysis procedures	Conforms	3.8.3

Table 1.9-3 Conformance with NUREG-0800 (Chapter 3 Design of Structures, Components, Equipment, and Systems) (Sheet 12 of 21)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
3.8.3	4	Concrete and Steel Internal Structures of Steel or Concrete Containments	5	Structural acceptance criteria	Alternate Approach	3.8.3
3.8.3	4	Concrete and Steel Internal Structures of Steel or Concrete Containments	6	Materials, quality control, and special construction techniques	Alternate Approach	3.8.3
3.8.3	4	Concrete and Steel Internal Structures of Steel or Concrete Containments	7	Testing and inservice surveillance requirements	Alternate Approach	3.8.3
3.8.4	4	Other Seismic Category I Structures	1	Structures description	Conforms	3.8.4
3.8.4	4	Other Seismic Category I Structures	2	Applicable codes, standards, and specifications	Alternate Approach	3.8.4
3.8.4	4	Other Seismic Category I Structures	3	Loads and load combinations	Conforms	3.8.4
3.8.4	4	Other Seismic Category I Structures	4	Design and analysis procedures	Conforms	3.8.4
3.8.4	4	Other Seismic Category I Structures	5	Structural acceptance criteria	Alternate Approach	3.8.4
3.8.4	4	Other Seismic Category I Structures	6	Materials, quality control, and special construction techniques	Alternate Approach	3.8.4
3.8.4	4	Other Seismic Category I Structures	7	Testing and inservice surveillance requirements	Alternate Approach	3.8.4
3.8.4	4	Other Seismic Category I Structures	8	Masonry walls	N/A – BWRX-300 design does not include masonry walls.	N/A
3.8.5	4	Foundations	1	Foundation description	Conforms	3.8.5

Table 1.9-3 Conformance with NUREG-0800 (Chapter 3 Design of Structures, Components, Equipment, and Systems) (Sheet 13 of 21)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
3.8.5	4	Foundations	2	Applicable codes, standards, and specifications	Alternate Approach	3.8.5
3.8.5	4	Foundations	3	Loads and load combinations	Conforms	3.8.5
3.8.5	4	Foundations	4	Design and analysis procedures	Conforms	3.8.5
3.8.5	4	Foundations	5	Structural acceptance criteria	Alternate Approach	3.8.5
3.8.5	4	Foundations	6	Materials, quality control, and special construction techniques	Alternate Approach	3.8.5
3.8.5	4	Foundations	7	Testing and inservice surveillance requirements	Alternate Approach	3.8.5
3.9.1	4	Special Topics for Mechanical Components	1	Transients used in the design and fatigue analysis of Code Class I components and support structures	Conforms	3.9.1
3.9.1	4	Special Topics for Mechanical Components	2	Computer programs used in dynamic and static analyses	Conforms	3.9.1
3.9.1	4	Special Topics for Mechanical Components	3	Experimental stress analysis	Conforms	3.9.1
3.9.1	4	Special Topics for Mechanical Components	4	Methods of analysis regarding Service Level D limits	Conforms	3.9.1
3.9.2	4	Dynamic Testing and Analysis of Structures, Systems, and Components	1	Compliance with GDCs 1, 2, 4, 14, and 15 during startup testing for thermal expansion, vibration, and dynamic effects	Conforms	3.9.2
3.9.2	4	Dynamic Testing and Analysis of Structures, Systems, and Components	2	Compliance with GDC 2	Conforms	3.9.2

Table 1.9-3 Conformance with NUREG-0800 (Chapter 3 Design of Structures, Components, Equipment, and Systems) (Sheet 14 of 21)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
3.9.2	4	Dynamic Testing and Analysis of Structures, Systems, and Components	3	Compliance with GDC 1 and 4: prediction of reactor internals vibration	Conforms	3.9.2
3.9.2	4	Dynamic Testing and Analysis of Structures, Systems, and Components	4	Compliance with GDCs 1 and 4: Use SRP listed preoperational and initial startup vibration and stress test program in addition to RG 1.20 guidance	Conforms	3.9.2
3.9.2	4	Dynamic Testing and Analysis of Structures, Systems, and Components	5	Compliance with GDCs 2, 4, 14, and 15: dynamic system analyses specified in RG 1.20	Conforms	3.9.2
3.9.2	4	Dynamic Testing and Analysis of Structures, Systems, and Components	6	Compliance with GDC 1: tests and analyses of reactor internals	Conforms	3.9.2
3.9.3	3	ASME Code Class 1, 2, and 3 Components, and Component Supports, and Core Support Structures	1	Loading combinations, operating transients, stress limits	Conforms	3.9.3
3.9.3	3	ASME Code Class 1, 2, and 3 Components, and Component Supports, and Core Support Structures	2	Design and installation of pressure relief devices	Conforms	3.9.3
3.9.3	3	ASME Code Class 1, 2, and 3 Components, and Component Supports, and Core Support Structures	3	Component supports	Conforms	3.9.3

Table 1.9-3 Conformance with NUREG-0800 (Chapter 3 Design of Structures, Components, Equipment, and Systems) (Sheet 15 of 21)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
3.9.4	4	CRD Systems	1	Construction codes and standards	Conforms	3.9.4
3.9.4	4	CRD Systems	2	Load combinations and stress limits	Conforms	3.9.4
3.9.4	4	CRD Systems	3	Operability assurance program	FSAR	FSAR
3.9.5	4	Reactor Pressure Vessel Internals	1	Loads, loading combinations, and limits	Conforms	3.9.5
3.9.5	4	Reactor Pressure Vessel Internals	2	Design and construction of core support structures	Conforms	3.9.5
3.9.5	4	Reactor Pressure Vessel Internals	3	Design criteria, loading combinations, and analyses for internals other than core support structures	Conforms	3.9.5
3.9.5	4	Reactor Pressure Vessel Internals	4	Deformation limits for reactor internals	Conforms	3.9.5
3.9.5	4	Reactor Pressure Vessel Internals	5	Asymmetric blowdown loads from pipe ruptures	Conforms	3.9.2.5
3.9.6	4	Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	1	Design and qualification of pumps, valves, and dynamic restraints	Conforms	3.9.6
3.9.6	4	Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	2	Inservice testing program	Conforms	3.9.6

Table 1.9-3 Conformance with NUREG-0800 (Chapter 3 Design of Structures, Components, Equipment, and Systems) (Sheet 16 of 21)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
3.9.6	4	Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	3	Inservice testing program for pumps	Conforms	3.9.6
3.9.6	4	Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	4	Inservice testing program for valves	Conforms	3.9.6
3.9.6	4	Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	5	Inservice testing program for dynamic restraints	Conforms	3.9.6
3.9.6	4	Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	6	Relief requests and proposed alternatives	Conforms	3.9.6
3.9.6	4	Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	7	Inspections, Tests, Analyses, and Acceptance Criteria	N/A – 10 CFR Part 50 CPA	N/A
3.9.6	4	Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	8	COL Action Items	N/A – 10 CFR Part 50 CPA	N/A
3.9.6	4	Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	9	Operational programs	FSAR	FSAR

Table 1.9-3 Conformance with NUREG-0800 (Chapter 3 Design of Structures, Components, Equipment, and Systems) (Sheet 17 of 21)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
3.9.6	4	Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	10	Tier 2* information	N/A – 10 CFR Part 50 CPA	N/A
3.9.7	0	Risk-Informed Inservice Testing	N/A	Modifications to Inservice Testing (IST) programs	N/A – 10 CFR Part 50 CPA	N/A
3.9.8	0	Risk-Informed Inservice Inspection (ISI) of Piping	N/A	Changes to IST program for piping	N/A – 10 CFR Part 50 CPA	N/A
3.10	4	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	1	Conformance with RG 1.100	Conforms	3.10
3.10	4	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	2	Testing of instrumentation described in RG 1.97	Alternate Approach	Table 1.9-20
3.10	4	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	3	Experienced-based qualification	Conforms	3.10.1
3.10	4	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	4	Requirements for equipment qualification	Conforms	3.10.1
3.10	4	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	5	Compliance with GDC 14 with regard to RCPB abnormal leakage	Conforms	3.10
3.10	4	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	6	Documentation of qualification program	Conforms	3.10.1

Table 1.9-3 Conformance with NUREG-0800 (Chapter 3 Design of Structures, Components, Equipment, and Systems) (Sheet 18 of 21)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
3.11	3	Environmental Qualification of Mechanical and Electrical Equipment	1	Conformance with RG 1.89	Conforms	Table 1.9-20
3.11	3	Environmental Qualification of Mechanical and Electrical Equipment	2	Conformance with IEEE Std 323	Alternate Approach	3.11
3.11	3	Environmental Qualification of Mechanical and Electrical Equipment	3	Conformance with RG 1.40	Conforms	Table 1.9-20
3.11	3	Environmental Qualification of Mechanical and Electrical Equipment	4	Conformance with RG 1.63	Alternate Approach	Table 1.9-20
3.11	3	Environmental Qualification of Mechanical and Electrical Equipment	5	Conformance with RG 1.73	Conforms	Table 1.9-20
3.11	3	Environmental Qualification of Mechanical and Electrical Equipment	6	Conformance with RG 1.89	Conforms	Table 1.9-20
3.11	3	Environmental Qualification of Mechanical and Electrical Equipment	7	Conformance with RG 1.97	Alternate Approach	Table 1.9-20
3.11	3	Environmental Qualification of Mechanical and Electrical Equipment	8	Qualification of cables and field splices; conformance with RG 1.89	Conforms	Table 1.9-20
3.11	3	Environmental Qualification of Mechanical and Electrical Equipment	9	Conformance with RG 1.156	Conforms	Table 1.9-20

Table 1.9-3 Conformance with NUREG-0800 (Chapter 3 Design of Structures, Components, Equipment, and Systems) (Sheet 19 of 21)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
3.11	3	Environmental Qualification of Mechanical and Electrical Equipment	10	Conformance with RG 1.158; RG 1.89	Alternate Approach	Table 1.9-20
3.11	3	Environmental Qualification of Mechanical and Electrical Equipment	11	Conformance with RG 1.180, RG 1.89	Alternate Approach	Table 1.9-20
3.11	3	Environmental Qualification of Mechanical and Electrical Equipment	12	Conformance with RG 1.183	Alternate Approach	Table 1.9-20
3.11	3	Environmental Qualification of Mechanical and Electrical Equipment	13	Effects of chemical exposure	Conforms	3.11.1
3.11	3	Environmental Qualification of Mechanical and Electrical Equipment	14	Process and records for materials, parts, and equipment suitability	Conforms	3.11.2
3.11	3	Environmental Qualification of Mechanical and Electrical Equipment	15	Equipment in a mild environment	Conforms	3.11.1
3.11	3	Environmental Qualification of Mechanical and Electrical Equipment	16	Description of the Environmental Qualification program and milestones	FSAR	FSAR
3.12	1	ASME Code Class 1, 2, and 3 Piping Systems, Piping Components, and Their Associated Supports	A	Analysis methods	Conforms	3.12.1

Table 1.9-3 Conformance with NUREG-0800 (Chapter 3 Design of Structures, Components, Equipment, and Systems) (Sheet 20 of 21)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
3.12	1	ASME Code Class 1, 2, and 3 Piping Systems, Piping Components, and Their Associated Supports	B	Modeling techniques	Conforms	3.12.2
3.12	1	ASME Code Class 1, 2, and 3 Piping Systems, Piping Components, and Their Associated Supports	C	Stress analysis criteria	Conforms	3.12.3
3.12	1	ASME Code Class 1, 2, and 3 Piping Systems, Piping Components, and Their Associated Supports	D	Design of supports	Conforms	3.12.4
3.13	0	Threaded Fasteners – ASME Code Class 1, 2, and 3	1	Design aspects	Conforms	3.13.1
3.13	0	Threaded Fasteners – ASME Code Class 1, 2, and 3	2	Preservice and ISI requirements	Conforms	3.13.2
BTP 3-1	2	Classification of Main Steam Components Other than the Reactor Coolant Pressure Boundary for BWR Plant		Conformance with Table A-1 of BTP 3-1	N/A – BWRX-300 design does not include a main steam isolation valve leakage control system.	N/A
BTP 3-2	2	Classification of BWR/6 Main Steam and Feedwater Components Other than the Reactor Coolant Pressure Boundary	N/A	Main steam and feedwater system components of BWR/6 plants	N/A – applies to BWR/6	N/A

Table 1.9-3 Conformance with NUREG-0800 (Chapter 3 Design of Structures, Components, Equipment, and Systems)
(Sheet 21 of 21)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
BTP 3-3	3	Protection Against Postulated Piping Failures in Fluid Systems Outside Containment		Failures of fluid system piping outside containment affecting safe shutdown equipment	Conforms	3.6
BTP 3-4	3	Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment	1	High energy fluid systems piping	Conforms	3.6
BTP 3-4	3	Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment	2	Moderate energy fluid systems piping	Conforms	3.6
BTP 3-4	3	Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment	3	Type of breaks and leakage cracks	Conforms	3.6

Table 1.9-4 Conformance with NUREG-0800 (Chapter 4 Reactor)
(Sheet 1 of 4)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
4.2	3	Fuel System Design	1.A	Design Basis: Fuel system damage	Conforms	4.2
4.2	3	Fuel System Design	1.B	Design Basis: Fuel rod failure	Conforms	4.2.1
4.2	3	Fuel System Design	1.C	Design Basis: Fuel coolability	Conforms	4.2.4 4.2.3
4.2	3	Fuel System Design	2	Description and design drawings	Conforms	4.2.1
4.2	3	Fuel System Design	3.A	Design evaluation: OPEX	Conforms	4.2.1
4.2	3	Fuel System Design	3.B	Design evaluation: prototype testing - Fuel	N/A – GNF2 is currently licensed.	N/A
				Design evaluation: prototype testing - Control Rod	FSAR	N/A
4.2	3	Fuel System Design	3.C	Design evaluation: analytical predictions - Fuel	Conforms	4.2.1
				Design evaluation: analytical predictions - Control Rod	FSAR	N/A
4.2	3	Fuel System Design	4.A	Testing, inspection, and surveillance plans / testing and inspection of new fuel.	Conforms	4.2.5
4.2	3	Fuel System Design	4.B	Testing, inspection, and surveillance plans / online fuel system monitoring	FSAR	N/A
4.2	3	Fuel System Design	4.C	Testing, inspection, and surveillance plans/post-irradiation surveillance	FSAR	N/A
4.3	3	Nuclear Design	1	Design limits for power densities and power distributions	Conforms	4.3.4

Table 1.9-4 Conformance with NUREG-0800 (Chapter 4 Reactor)
(Sheet 2 of 4)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
4.3	3	Nuclear Design	2	Reactivity coefficients	Conforms	4.3.4
4.3	3	Nuclear Design	3.A	Control rod worth and reactivity insertion rate	Conforms	4.3.1
4.3	3	Nuclear Design	3.B	Equipment, operating limits, and procedures	Conforms	4.3.4
4.3	3	Nuclear Design	4	Analytical methods and data	Conforms	4.3.3
4.4	2	Thermal and Hydraulic Design	1	Uncertainties	Conforms	4.4.1
4.4	2	Thermal and Hydraulic Design	2	Departure from Nucleate Boiling Ratio, Critical Power Ratio (CPR) limits; Critical Heat Flux (CHF) correlation	Conforms	4.4.1
4.4	2	Thermal and Hydraulic Design	3	Core oscillations and thermal-hydraulic instabilities as described in SRP Section 15.9	Conforms	4.3.6 4A.6 4A.7
4.4	2	Thermal and Hydraulic Design	4	Methods for calculating single-phase and two-phase fluid flow in the reactor vessel and other components	Conforms	4.4.2
4.4	2	Thermal and Hydraulic Design	5	TS, including safety limits and limiting safety settings	Conforms	4.4.2 4.4.4
4.4	2	Thermal and Hydraulic Design	6	Preoperational and initial startup test programs	Conforms	4.6.5
4.4	2	Thermal and Hydraulic Design	7	Loose parts monitoring system	N/A – BWRX-300 does not include a loose parts monitoring system.	N/A

Table 1.9-4 Conformance with NUREG-0800 (Chapter 4 Reactor)
(Sheet 3 of 4)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
4.4	2	Thermal and Hydraulic Design	8	Accounting for crud in the CHF calculations, pressure drop throughout the RCS; process monitoring	Conforms	4.4.2
4.4	2	Thermal and Hydraulic Design	9	Instrumentation for indications of inadequate core cooling.	FSAR	FSAR
4.4	2	Thermal and Hydraulic Design	10	Analyses of Anticipated Transient Without Scram (ATWS) event on fuel design limits consistent with SRP Sections 15.8 and 15.9	Conforms	4.2.4 4A.6 4A.7
4.5.1	3	CRD Structural Materials	1	Materials specifications	Conforms	4.5.1
4.5.1	3	CRD Structural Materials	2	Austenitic stainless-steel components	Conforms	4.5.1
4.5.1	3	CRD Structural Materials	3	Other materials	Conforms	4.5.1
4.5.1	3	CRD Structural Materials	4	Cleaning and cleanliness control	Conforms	4.5.1
4.5.2	3	Reactor Internal and Core Support Structure Materials	1	Materials	Conforms	4.5.3
4.5.2	3	Reactor Internal and Core Support Structure Materials	2	Controls on welding	Conforms	4.5.3
4.5.2	3	Reactor Internal and Core Support Structure Materials	3	Nondestructive examinations	Conforms	4.5.3
4.5.2	3	Reactor Internal and Core Support Structure Materials	4	Austenitic stainless steel	Conforms	4.5.3
4.5.2	3	Reactor Internal and Core Support Structure Materials	5	Other materials	Conforms	4.5.3

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

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
4.6	2	Functional Design of Control Rod Drive System	1	CRD System functionality compliance with GDC 4	Conforms	4.6.4
4.6	2	Functional Design of Control Rod Drive System	2	Effect of CRD System failure compliance with GDC 23	Conforms	4.6.4
4.6	2	Functional Design of Control Rod Drive System	3	Single CRD System DS malfunction compliance with GDC 25	Conforms	4.6.4
4.6	2	Functional Design of Control Rod Drive System	4	CRD System control and reliability compliance with GDC 26	Conforms with PDC 26	4.6.4
4.6	2	Functional Design of Control Rod Drive System	5	Combined CRD System and ECCS capability compliance with GDC 27	Conforms	4.6.4
4.6	2	Functional Design of Control Rod Drive System	6	CRD System design should preclude RPV components or fuel damage per GDC 28	Conforms with PDC 28	4.6.4
4.6	2	Functional Design of Control Rod Drive System	7	Probability of CRD System function during AOOs per GDC 29	Conforms with PDC 29	4.6.4
4.6	2	Functional Design of Control Rod Drive System	8	Alternate rod injection system and redundant scram header exhaust valves per 10 CFR 50.62I(3)	Conforms	4.6.3
BTP 4-1	3	Westinghouse Constant Axial Offset Control	B	Operating scheme for Westinghouse designs	N/A – applies to Westinghouse designs.	N/A

**Table 1.9-5 Conformance with NUREG-0800 (Chapter 5 Reactor Coolant System and Connected Systems)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
5.2.1.1	4	Compliance with the Codes and Standards Rule, 10 CFR 50.55a	1	Quality standards for compliance with GDC 1	Conforms	5.2.1
5.2.1.1	4	Compliance with the Codes and Standards Rule, 10 CFR 50.55a	2	Design, fabrication, testing of RCPB components in accordance with 10 CFR 50.55a	Conforms	5.2.1
5.2.1.1	4	Compliance with the Codes and Standards Rule, 10 CFR 50.55a	3	Necessary and sufficient ITAAC in accordance with 10 CFR 52.47	N/A – 10 CFR Part 50 CPA	N/A
5.2.1.1	4	Compliance with the Codes and Standards Rule, 10 CFR 50.55a	4	Per 10 CFR 52.79, programs to ensure systems and components meet ASME requirements in accordance with 10 CFR 50.55a.	N/A – 10 CFR Part 50 CPA	N/A
5.2.1.2	4	Applicable Code Cases	1	Quality standards for compliance with GDC 1	Conforms	5.2.1
5.2.1.2	4	Applicable Code Cases	2	Design, fabrication, testing of RCPB components in accordance with 10 CFR 50.55a	Conforms	5.2.1
5.2.1.2	4	Applicable Code Cases	3	Necessary and sufficient ITAAC in accordance with 10 CFR 52.47	N/A – 10 CFR Part 50 CPA	N/A
5.2.1.2	4	Applicable Code Cases	4	Per 10 CFR 52.79, programs to ensure systems and components meet ASME requirements in accordance with 10 CFR 50.55a	N/A – 10 CFR Part 50 CPA	N/A
5.2.2	3	Overpressure Protection	1	Material specifications	Conforms	5.2.3

Table 1.9-5 Conformance with NUREG-0800 (Chapter 5 Reactor Coolant System and Connected Systems)
(Sheet 2 of 10)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
5.2.2	3	Overpressure Protection	2	Design requirements for BWRs operating at power	Conforms	6.3
5.2.2	3	Overpressure Protection	3	Design requirements for Pressurized Water Reactors (PWRs) operating at power	N/A – BWR design	N/A
5.2.2	3	Overpressure Protection	4	Design requirements for PWRs operating at low power (startup, shutdown)	N/A – BWR design	N/A
5.2.2	3	Overpressure Protection	5	Testing and inspections	Conforms	6.3
5.2.2	3	Overpressure Protection	6	Technical Specifications	FSAR	FSAR
5.2.2	3	Overpressure Protection	7	TMI action plan requirements (relief valves and safety valves).	N/A – BWRX-300 design does not include relief and safety valves in the ICS.	N/A
5.2.3	3	Reactor Coolant Pressure Boundary Materials	1	Material specifications	Conforms	5.2.3
5.2.3	3	Reactor Coolant Pressure Boundary Materials	2	Compatibility of materials with reactor coolant	Conforms	5.2.3
5.2.3	3	Reactor Coolant Pressure Boundary Materials	3	Fabrication and processing of ferritic materials	Conforms	5.2.3
5.2.3	3	Reactor Coolant Pressure Boundary Materials	4	Fabrication and processing of austenitic stainless steel	Conforms	5.2.3
5.2.4	2	Reactor Coolant Pressure Boundary Inservice Inspection and Testing	1	System boundary subject to inspection	Conforms	5.2.4
5.2.4	2	Reactor Coolant Pressure Boundary Inservice Inspection and Testing	2	Accessibility	FSAR	FSAR

Table 1.9-5 Conformance with NUREG-0800 (Chapter 5 Reactor Coolant System and Connected Systems)
(Sheet 3 of 10)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
5.2.4	2	Reactor Coolant Pressure Boundary Inservice Inspection and Testing	3	Examination categories and methods	FSAR	FSAR
5.2.4	2	Reactor Coolant Pressure Boundary Inservice Inspection and Testing	4	Inspection intervals	FSAR	FSAR
5.2.4	2	Reactor Coolant Pressure Boundary Inservice Inspection and Testing	5	Evaluation of examination results	FSAR	FSAR
5.2.4	2	Reactor Coolant Pressure Boundary Inservice Inspection and Testing	6	System pressure tests	FSAR	FSAR
5.2.4	2	Reactor Coolant Pressure Boundary Inservice Inspection and Testing	7	Code exemptions	FSAR	FSAR
5.2.4	2	Reactor Coolant Pressure Boundary Inservice Inspection and Testing	8	Code cases	FSAR	FSAR
5.2.4	2	Reactor Coolant Pressure Boundary Inservice Inspection and Testing	9	Augmented ISI to protect against postulated piping failures	FSAR	FSAR
5.2.4	2	Reactor Coolant Pressure Boundary Inservice Inspection and Testing	10	Other inspection programs	FSAR	FSAR
5.2.4	2	Reactor Coolant Pressure Boundary Inservice Inspection and Testing	11	Operational programs	FSAR	FSAR

Table 1.9-5 Conformance with NUREG-0800 (Chapter 5 Reactor Coolant System and Connected Systems)
(Sheet 4 of 10)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
5.2.5	2	Reactor Coolant Pressure Boundary Leakage Detection	1	For GDC 2, acceptance is based on the guidelines of RG 1.29, Positions C.1	Alternate Approach	Table 1.9-20
5.2.5	2	Reactor Coolant Pressure Boundary Leakage Detection	2	For GDC 30, acceptance is based on meeting the guidelines of RG 1.45.	Alternate Approach	Table 1.9-20
5.3.1	2	Reactor Vessel Materials	1	Materials	Conforms	5.3.1
5.3.1	2	Reactor Vessel Materials	2	Special processes used for manufacture and fabrication of components	Conforms	5.3.1
5.3.1	2	Reactor Vessel Materials	3	Special methods for nondestructive examination	Conforms	5.3.1
5.3.1	2	Reactor Vessel Materials	4	Special controls and processes for ferritic steels and austenitic stainless steels	Conforms	5.3.1
5.3.1	2	Reactor Vessel Materials	5	Fracture toughness	Conforms	5.3.1
5.3.1	2	Reactor Vessel Materials	6	Material surveillance	FSAR	5.3.1
5.3.1	2	Reactor Vessel Materials	7	Reactor vessel fasteners	Conforms	5.3.1
5.3.2	2	Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock	1	P-T limits: regulations, codes, basis documents, and requirements	Conforms	5.3.2
5.3.2	2	Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock	2	Upper shelf energy: regulations, codes, basis documents, and requirements	Conforms	5.3.2
5.3.2	2	Pressure-Temperature Limits, Upper Shelf Energy, and Pressurized Thermal Shock	3	Pressurized thermal shock: regulations, codes, basis documents, and requirements	N/A – BWR design	N/A

Table 1.9-5 Conformance with NUREG-0800 (Chapter 5 Reactor Coolant System and Connected Systems)
(Sheet 5 of 10)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
5.3.3	2	Reactor Vessel Integrity	1	Design (compliance with GDCs)	Conforms	5.3.3
5.3.3	2	Reactor Vessel Integrity	2	Materials of construction	Conforms	5.3.3
5.3.3	2	Reactor Vessel Integrity	3	Fabrication methods	Conforms	5.3.3
5.3.3	2	Reactor Vessel Integrity	4	Inspection requirements	Conforms	5.3.3
5.3.3	2	Reactor Vessel Integrity	5	Shipment and installation	Conforms	5.3.3
5.3.3	2	Reactor Vessel Integrity	6	Operating conditions	Conforms	5.3.3
5.3.3	2	Reactor Vessel Integrity	7	Inservice surveillance	Conforms	5.3.3
5.3.3	2	Reactor Vessel Integrity	8	Operational programs	FSAR	FSAR
5.4	2	Reactor Coolant System Component and Subsystem Design	N/A	Reactor Coolant System Component and Subsystem Design	N/A – SRP cites other SRP sections for acceptance criteria.	N/A
5.4.1.1	3	Pump Flywheel Integrity	N/A	N/A	N/A – BWR design	N/A
5.4.2.1	4	Steam Generator Materials and Design	N/A	N/A	N/A – BWR design	N/A
5.4.2.2	2	Steam Generator Program	N/A	N/A	N/A – BWR design	N/A
5.4.6	4	Reactor Core Isolation Cooling System (BWR)	N/A	N/A	N/A – The BWRX-300 is a passive design and only credits the Isolation Condenser System (ICS). The BWRX-300 design does not include a RCIC system.	N/A
5.4.7	5	Residual Heat Removal System	1	Requirement specified in BTP 5-4	Conforms	5.4.7 6.3
5.4.7	5	Residual Heat Removal System	2	Water hammer prevention; conformance with GDC 4	Conforms	5.4.7

Table 1.9-5 Conformance with NUREG-0800 (Chapter 5 Reactor Coolant System and Connected Systems)
(Sheet 6 of 10)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
5.4.7	5	Residual Heat Removal System	3	System interfaces; conformance with GDC 5	Conforms	5.4.7
5.4.7	5	Residual Heat Removal System	4	Engineered safety feature system requirements	Alternate Approach BWRX-300 residual heat removal function is performed by the ICS.	6.3
5.4.8	3	Reactor Water Cleanup System (BWR)	1	Water purity in normal operation and AOOs	Alternate Approach BWRX-300 reactor water cleanup, filtration, and ion removal is performed by the CFD.	10.4.6
5.4.8	3	Reactor Water Cleanup System (BWR)	2	System isolation, low flow condition, resin transfer, operational issues	Alternate Approach BWRX-300 reactor water cleanup, filtration, and ion removal is performed by the CFD.	10.4.6
5.4.8	3	Reactor Water Cleanup System (BWR)	3	Classification, seismic design	Conforms	5.4.8
5.4.8	3	Reactor Water Cleanup System (BWR)	4	Monitoring	Alternate Approach BWRX-300 reactor water cleanup, filtration, and ion removal is performed by the CFD.	10.4.6
5.4.11	4	Pressurizer Relief Tank	N/A	N/A	N/A - BWR design	N/A

Table 1.9-5 Conformance with NUREG-0800 (Chapter 5 Reactor Coolant System and Connected Systems)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
5.4.12	1	Reactor Coolant System High Point Vents	1	Use of vents during and following accidents	Alternate Approach Removal of noncondensible gases during and following accidents is performed by the ICS.	6.3
5.4.12	1	Reactor Coolant System High Point Vents	2	Venting of gases that inhibit core cooling	Alternate Approach Removal of noncondensible gases during and following accidents is performed by the ICS.	6.3
5.4.12	1	Reactor Coolant System High Point Vents	3	Isolation of the vent path	Conforms	5.4.13
5.4.12	1	Reactor Coolant System High Point Vents	4	Minimization of inadvertent actuation	Conforms	5.4.13
5.4.12	1	Reactor Coolant System High Point Vents	5	Reactor coolant pressure boundary requirements	Conforms	5.4.13
5.4.12	1	Reactor Coolant System High Point Vents	6	Vent sizing in relation to LOCA definition	Conforms	5.4.13
5.4.12	1	Reactor Coolant System High Point Vents	7	Vent discharge	Conforms	5.4.13
5.4.12	1	Reactor Coolant System High Point Vents	8	Vent operation from CR	Conforms	5.4.13
5.4.12	1	Reactor Coolant System High Point Vents	9	CR displays and controls	Conforms	5.4.13

Table 1.9-5 Conformance with NUREG-0800 (Chapter 5 Reactor Coolant System and Connected Systems)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
5.4.12	1	Reactor Coolant System High Point Vents	10	Testing	Conforms	5.4.13
5.4.12	1	Reactor Coolant System High Point Vents	11	Seismic and Environmental Qualification	Conforms	5.4.13
5.4.12	1	Reactor Coolant System High Point Vents	12	Dynamic loads	Conforms	5.4.13
5.4.12	1	Reactor Coolant System High Point Vents	13	Procedures	FSAR	FSAR
5.4.12	1	Reactor Coolant System High Point Vents	14	Quality assurance requirements	Conforms	5.4.13
5.4.13	0	Isolation Condenser System (BWR)	1	Compliance with GDC 2	Conforms	6.3
5.4.13	0	Isolation Condenser System (BWR)	2	Compliance with GDC 4; conformance with RG 1.44	Conforms	6.3
5.4.13	0	Isolation Condenser System (BWR)	3	Compliance with GDC 5	Conforms	6.3
5.4.13	0	Isolation Condenser System (BWR)	4	Compliance with GDC 17	Alternate Approach	3.1.2.8
5.4.13	0	Isolation Condenser System (BWR)	5	Compliance with GDC 33	Conforms	6.3
5.4.13	0	Isolation Condenser System (BWR)	6	Compliance with GDC 34	Conforms	6.3
5.4.13	0	Isolation Condenser System (BWR)	7	Compliance with GDC 35	Conforms	6.3
5.4.13	0	Isolation Condenser System (BWR)	8	Compliance with GDC 36	Conforms	6.3

Table 1.9-5 Conformance with NUREG-0800 (Chapter 5 Reactor Coolant System and Connected Systems)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
5.4.13	0	Isolation Condenser System (BWR)	9	Compliance with GDC 37	Conforms	6.3
5.4.13	0	Isolation Condenser System (BWR)	10	Compliance with GDC 54	Conforms	6.2
5.4.13	0	Isolation Condenser System (BWR)	11	Compliance with 10 CFR 50.46	Conforms	6.3
5.4.13	0	Isolation Condenser System (BWR)	12	Compliance with 10 CFR 50.63	Conforms	6.2 6.3
BTP 5-1	3	Monitoring of Secondary Side Water Chemistry in Pressurized Water Reactor Steam Generators	N/A	N/A	N/A – BWR design	N/A
BTP 5-2	3	Overpressure Protection of Pressurized Water Reactors While Operating at Low Temperatures	N/A	N/A	N/A – BWR design	N/A
BTP 5-3	2	Fracture Toughness Requirements	1	Fracture Toughness Requirements	N/A – Criterion applies to plants with construction permits prior to August 15, 1973.	N/A
BTP 5-3	2	Fracture Toughness Requirements	2	Operating limitations for fracture toughness	FSAR	FSAR
BTP 5-3	2	Fracture Toughness Requirements	3	Inservice surveillance of fracture toughness	FSAR	FSAR
BTP 5-3	2	Fracture Toughness Requirements	4	Pressurized thermal shock (PWR only)	N/A – BWR design	N/A

Table 1.9-5 Conformance with NUREG-0800 (Chapter 5 Reactor Coolant System and Connected Systems)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
BTP 5-4	4	Design Requirements of the Residual Heat Removal System	1	Functional requirements	Alternate Approach Shutdown Cooling system (SDC) leak detection and isolation is a Safety Category 1 function. SDC residual heat removal is a Safety Category 3 function. ICS provides Safety Category 1 residual heat removal. Alternative related to SC1 residual heat removal capable of taking plant to cold shutdown.	5.4.7.3 6.3
BTP 5-4	4	Design Requirements of the Residual Heat Removal System	2	Isolation requirement	Conforms	5.4.7.3
BTP 5-4	4	Design Requirements of the Residual Heat Removal System	3	Pressure relief requirements	Conforms	5.4.7.3
BTP 5-4	4	Design Requirements of the Residual Heat Removal System	4	Pump protection requirements	Conforms	5.4.7.3
BTP 5-4	4	Design Requirements of the Residual Heat Removal System	5	Test requirements	N/A – SDC design pressure is equal to RCS design pressure.	N/A
BTP 5-4	4	Design Requirements of the Residual Heat Removal System	6	Operational procedures	Conforms	5.4.7
BTP 5-4	4	Design Requirements of the Residual Heat Removal System	7	Auxiliary feedwater supply	N/A – BWR design	N/A

Table 1.9-6 Conformance with NUREG-0800 (Chapter 6 Engineered Safety Features)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
6.1.1	2	Engineered Safety Features Materials	1	Materials and fabrication	Conforms	5.2 6.1
6.1.1	2	Engineered Safety Features Materials	2	Composition and compatibility of ESF fluids	Conforms	6.1
6.1.1	2	Engineered Safety Features Materials	3	Components and systems cleaning	N/A – RG 1.37 has been withdrawn.	N/A
6.1.1	2	Engineered Safety Features Materials	4	Thermal insulation	Conforms	5.2 6.1
6.1.2	3	Protective Coating Systems (Paints)-Organic Materials	1	Coating system inside containment	Conforms	3.8 6.1
6.2.1	3	Containment Functional Design	N/A	Containment Functional Design	N/A – SRP cites other SRP sections for acceptance criteria.	N/A
6.2.1.1.A	3	PWR Dry Containments, Including Subatmospheric Containments	1	Compliance with GDC 16 and GDC 50 regarding design pressure and margin	Conforms	6.2.1
6.2.1.1.A	3	PWR Dry Containments, Including Subatmospheric Containments	2	Compliance with GDC 38 regarding rapid reduction of containment pressure	Conforms	6.2.1
6.2.1.1.A	3	PWR Dry Containments, Including Subatmospheric Containments	3	Compliance with GDC 38 regarding rapid reduction of containment pressure (Subatmospheric containments)	Conforms	6.2.1

Table 1.9-6 Conformance with NUREG-0800 (Chapter 6 Engineered Safety Features)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
6.2.1.1.A	3	PWR Dry Containments, Including Subatmospheric Containments	4	Compliance with GDC 38 and GDC 50 regarding containment heat removal and design margin (LOCA analysis)	Conforms	6.2.1
6.2.1.1.A	3	PWR Dry Containments, Including Subatmospheric Containments	5	Compliance with GDC 38 and GDC 50 regarding containment heat removal and design margin (secondary system pipe ruptures)	Conforms	6.2.2
6.2.1.1.A	3	PWR Dry Containments, Including Subatmospheric Containments	6	Compliance with GDC 38 and GDC 50 regarding containment heat removal and design margin (external pressure)	Conforms	6.2.1
6.2.1.1.A	3	PWR Dry Containments, Including Subatmospheric Containments	7	Compliance with GDC 13, GDC 64, and 10 CFR 50.34(f)(2)(xvii) regarding post-accident monitoring	Conforms	6.2.1 Table 1.9-21
6.2.1.1.A	3	PWR Dry Containments, Including Subatmospheric Containments	8	Compliance with 10 CFR 50.46 Appendix K.I.D.2 regarding containment pressure for ECCS capability during reflood and spray cooling	N/A – This topic not applicable to the BWRX-300 design.	N/A
6.2.1.1.A	3	PWR Dry Containments, Including Subatmospheric Containments	9	Compliance with GDC 4 regarding internal structures and component capability to withstand differential pressure loadings	Conforms	6.2.1

Table 1.9-6 Conformance with NUREG-0800 (Chapter 6 Engineered Safety Features)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
6.2.1.1.A	3	PWR Dry Containments, Including Subatmospheric Containments	10	Compliance with 10 CFR 50.34(f)(3)(v)(A)(1) regarding 100% fuel clad metal-water reaction	FSAR	FSAR
6.2.1.1.B	2	Ice Condenser Containments	N/A	Ice Condenser Containments	N/A – BWRX-300 design does not include an ice condenser containment.	N/A
6.2.1.1.C	7	Pressure-Suppression Type BWR Containments	N/A	Pressure-Suppression Type BWR Containments	N/A – BWRX-300 design does not include a pressure-suppression type containment.	N/A
6.2.1.2	3	Subcompartment Analysis	1	Nodalization	BWRX-300 containment design does not have large high energy lines routed through sub	N/A
6.2.1.2	3	Subcompartment Analysis	2	Initial thermodynamic conditions	BWRX-300 containment design does not have large high energy lines routed through sub	N/A
6.2.1.2	3	Subcompartment Analysis	3	Vent flow path: distribution of released mass and energy	BWRX-300 containment design does not have large high energy lines routed through sub	N/A
6.2.1.2	3	Subcompartment Analysis	4	Design pressure	BWRX-300 containment design does not have large high energy lines routed through sub	N/A

Table 1.9-6 Conformance with NUREG-0800 (Chapter 6 Engineered Safety Features)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
6.2.1.3	3	Mass and Energy Release Analysis for Postulated LOCAs	1	General Design Criterion 50	Conforms	6.2.1.3 15.5
6.2.1.3	3	Mass and Energy Release Analysis for Postulated LOCAs	2	Appendix K to 10 CFR Part 50	N/A – The ECCS evaluation model does not use the alternatives provided in 10 CFR 50 Appendix K.	6.3.3.1
6.2.1.3	3	Mass and Energy Release Analysis for Postulated LOCAs	3	10 CFR 52.47(b)(1) regarding ITAAC	N/A – 10 CFR Part 50 CPA	N/A
6.2.1.3	3	Mass and Energy Release Analysis for Postulated LOCAs	4	10 CFR 52.80(a) regarding ITAAC	N/A – 10 CFR Part 50 CPA	N/A
6.2.1.4	2	Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures	1	Source of energy	N/A – BWR design	N/A
6.2.1.4	2	Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures	2	Mass and energy release rate	N/A – BWR design	N/A
6.2.1.4	2	Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures	3	Single failure analysis	N/A – BWR design	N/A
6.2.1.5	3	Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies	1	Containment pressure models meeting 10 CFR 50.46(a)(1(i)	N/A – BWR design	N/A

Table 1.9-6 Conformance with NUREG-0800 (Chapter 6 Engineered Safety Features)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
6.2.1.5	3	Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies	2	Conservatisms in LOCA analyses to meet 10 CFR 50.46(a)(1)(ii)	N/A – BWR design	N/A
6.2.2	5	Containment Heat Removal Systems	1	Containment heat removal system redundancy and power source consistent with ESF design	Conforms	6.2.2
6.2.2	5	Containment Heat Removal Systems	2	Capability of containment spray system to perform safety function	N/A – BWRX-300 design does not include a containment spray system.	N/A
6.2.2	5	Containment Heat Removal Systems	3	Analysis of containment spray system to perform safety function	N/A – BWRX-300 design does not include a containment spray system.	N/A
6.2.2	5	Containment Heat Removal Systems	4	Containment fan cooler performance capability	N/A – BWRX-300 design does not include containment fan coolers as an ESF system.	N/A
6.2.2	5	Containment Heat Removal Systems	5	Containment heat removal capability per GDC 38	Conforms	6.2.2
6.2.2	5	Containment Heat Removal Systems	6	Containment heat removal capability (long term water source)	N/A – BWRX-300 design does not include a containment spray system and does not require an external water source for ECCS function.	N/A

Table 1.9-6 Conformance with NUREG-0800 (Chapter 6 Engineered Safety Features)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
6.2.2	5	Containment Heat Removal Systems	7	Inspection and testing of containment heat removal system	Conforms	6.2.2
6.2.2	5	Containment Heat Removal Systems	8	Instrumentation for monitoring containment heat removal system performance	Conforms	6.2.2
6.2.3	3	Secondary Containment Functional Design	N/A	Functional capability of secondary containment	N/A – BWRX-300 design does not include a secondary containment	N/A
6.2.4	3	Containment Isolation System	1	Isolation criteria for instrument lines	Conforms	6.2.4
6.2.4	3	Containment Isolation System	2	Isolation and detection of leakage outside of containment from ESF systems and related systems	Conforms	6.2.1 6.2.2 6.2.4 6.3.5 9.1.3 9.2.8
6.2.4	3	Containment Isolation System	3	Leakage detection of remote-manual isolation valves in systems needed for safe shutdown	Conforms	6.2.4

Table 1.9-6 Conformance with NUREG-0800 (Chapter 6 Engineered Safety Features)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
6.2.4	3	Containment Isolation System	4	Containment isolation valve configuration ESF, safe shutdown lines	Conforms	5.4.3 5.4.5 5.4.8 6.2.1 6.2.2 6.2.4 6.8 9.2.3 9.2.6 9.2.7 9.3.1 9.3.3 9.3.6 9.3.10 11.2
6.2.4	3	Containment Isolation System	5	Containment isolation valve configuration and protection for ESF systems	Conforms	6.2.4
6.2.4	3	Containment Isolation System	6	Sealed-closed barriers	Conforms	6.2.4
6.2.4	3	Containment Isolation System	7	Relief valves	N/A – BWRX-300 design does not include relief valves as isolation valves.	N/A
6.2.4	3	Containment Isolation System	8	System classification per 10 CFR 50.34(f)(2)(xiv)	Conforms	Table 1.9-21

Table 1.9-6 Conformance with NUREG-0800 (Chapter 6 Engineered Safety Features)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
6.2.4	3	Containment Isolation System	9	Isolation valves outside containment	Conforms	6.2.4
6.2.4	3	Containment Isolation System	10	Automatic isolation valve position on loss of actuating power	Conforms	6.2.4
6.2.4	3	Containment Isolation System	11	Containment isolation pressure setpoint	Conforms	7.1.3
6.2.4	3	Containment Isolation System	12	Diversity of containment isolation parameters	Conforms	7.3
6.2.4	3	Containment Isolation System	13	Containment isolation using radiation monitors on system lines open to environs	FSAR	FSAR
6.2.4	3	Containment Isolation System	14	Isolation performance capability reflects safety importance	Conforms	6.2.4 15.5
6.2.4	3	Containment Isolation System	15	Closed system as isolation barrier	Conforms	6.2.4
6.2.4	3	Containment Isolation System	16	Reliability and performance of isolation barriers reflects safety importance	Conforms	6.2.4
6.2.4	3	Containment Isolation System	17	Isolation of remote-manual valves from Main Control Room (MCR)	Conforms	6.2.4 6.4
6.2.4	3	Containment Isolation System	18	Operability and leakage rate testing	Conforms	6.2.4
6.2.4	3	Containment Isolation System	19	Reopening isolation valves	Conforms	6.2.4
6.2.4	3	Containment Isolation System	20	Containment purging per 10 CFR 50.34(f)(2)(xv)	Conforms	Table 1.9-21

Table 1.9-6 Conformance with NUREG-0800 (Chapter 6 Engineered Safety Features)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
6.2.4	3	Containment Isolation System	21	Guidance for meeting Station Blackout (SBO) rule, 10 CFR 50.63(a)(2)	Conforms	6.2.4
6.2.4	3	Containment Isolation System	22	Determination of fuel failure for radiological calculations	N/A – The ECCS evaluation model does not use the alternatives provided in 10 CFR Part 50 Appendix K.	N/A
6.2.5	3	Combustible Gas Control in Containment	1	Hydrogen generation and control	Conforms	6.2 3.8
6.2.5	3	Combustible Gas Control in Containment	2	Capability of non-inerted containments with regard to hydrogen generation	N/A – BWRX-300 containment is inerted.	N/A
6.2.5	3	Combustible Gas Control in Containment	3	Equipment survivability per 10 CFR 50.44(c)(3)	FSAR	FSAR
6.2.5	3	Combustible Gas Control in Containment	4	Mixed atmosphere	Conforms	6.2.5 9.3.6
6.2.5	3	Combustible Gas Control in Containment	5	RG 1.7, Rev 3, Section C.1	Conforms	Table 1.9-20
6.2.5	3	Combustible Gas Control in Containment	6	Performance tests and monitoring instrumentation to conform with GDC 41	Conforms	9.3.6
6.2.5	3	Combustible Gas Control in Containment	7	Inservice inspection, operability testing, leak rate testing to conform with GDC 41, 42, 43	Conforms	9.3.6

Table 1.9-6 Conformance with NUREG-0800 (Chapter 6 Engineered Safety Features)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
6.2.5	3	Combustible Gas Control in Containment	8	Structural integrity with 100% clad-coolant reaction with combustible gas burning	FSAR	FSAR
6.2.5	3	Combustible Gas Control in Containment	9	Preliminary combustible gas control system design	Conforms	6.2.1 6.2.5 9.3.6
6.2.6	3	Containment Leakage Testing	All	Leakage rate testing, operational programs	Conforms	3.8.1 6.2.1 6.2.4 6.2.6
6.2.7	1	Fracture Prevention of Containment Pressure Boundary	1	Ferritic containment boundary materials	Conforms	3.8 6.2.7
6.2.7	1	Fracture Prevention of Containment Pressure Boundary	2	Ferritic containment boundary materials that have not been fracture toughness tested	N/A – This topic is not applicable as current fracture toughness requirements apply.	N/A
6.3	3	Emergency Core Cooling System	1	Performance criteria associated with 10 CFR 50.46	Conforms	6.3.3 15.3
6.3	3	Emergency Core Cooling System	2	Requirements of GDC 35 and GDC 17	Alternate Approach	3.1 6.3.3
6.3	3	Emergency Core Cooling System	3	Inservice inspection	Conforms	6.3.4
6.3	3	Emergency Core Cooling System	4	Combine reactivity control system capability to meet GDC 27	Conforms	3.1 6.3.3

Table 1.9-6 Conformance with NUREG-0800 (Chapter 6 Engineered Safety Features)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
6.3	3	Emergency Core Cooling System	5	RG 1.1	N/A – RG 1.1 has been withdrawn.	N/A
6.3	3	Emergency Core Cooling System	6	Water hammer prevention per GDC 4	Conforms	6.3.3
6.3	3	Emergency Core Cooling System	7	Design of non safety-related portions to meet GDC 2	FSAR	FSAR
6.3	3	Emergency Core Cooling System	8	Interfaces between ECCS and shared systems	N/A – Does not interface with Service Water or other systems.	N/A
6.3	3	Emergency Core Cooling System	9	TMI Action Plan (NUREG-0737, NUREG-0718) requirements	N/A – High Pressure Coolant Injection, Reactor Core Isolation Cooling are not part of design.	N/A
6.3	3	Emergency Core Cooling System	10	ECCS outage times	FSAR	FSAR
6.4	3	Control Room Habitability	1	CR emergency zone	Alternate Approach	6.4 9.4.1 9.4.6
6.4	3	Control Room Habitability	2	Ventilation system criteria	Alternate Approach	6.4 9.4.1 9.4.6
6.4	3	Control Room Habitability	3	Pressurization systems	Alternate Approach	6.4 9.4.1 9.4.6

Table 1.9-6 Conformance with NUREG-0800 (Chapter 6 Engineered Safety Features)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
6.4	3	Control Room Habitability	4	Emergency standby atmospheric filtration systems conformance with RG 1.52	N/A – BWRX-300 design does not utilize an engineered safety feature ventilation system.	N/A
6.4	3	Control Room Habitability	5	Relative location of source and CR	Alternate Approach	6.4 9.4 15.5
6.4	3	Control Room Habitability	6	Radiation hazards	Alternate Approach	6.4 15.5
6.4	3	Control Room Habitability	7	Toxic gas hazards	Alternate Approach	6.4 9.4
6.5.1	4	Engineered Safety Feature Atmosphere Cleanup Systems	II	Design, testing, and maintenance of ESF system	N/A – BWRX-300 design does not require an ESF containment atmospheric cleanup system. The CIS maintains an inert containment atmosphere.	N/A
6.5.2	4	Containment Spray as a Fission Product Cleanup System	N/A	Design requirements for fission product removal	N/A – BWRX-300 design does not include a containment spray system.	N/A
6.5.3	3	Fission Product Control Systems and Structures	1	Primary containment leak rate, containment isolation, potential doses	FSAR	FSAR
6.5.3	3	Fission Product Control Systems	2	Secondary containment	N/A – BWRX-300 design does not include a secondary containment.	N/A

Table 1.9-6 Conformance with NUREG-0800 (Chapter 6 Engineered Safety Features)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
6.5.3	3	Fission Product Control Systems and Structures	4	Other fission product control systems	N/A – BWRX-300 analysis does not credit other systems, filtration, or adsorption units.	N/A
6.5.4	3	Ice Condenser as a Fission Product Cleanup System	N/A	Fission product removal effectiveness	N/A – BWRX-300 design does not include an ice condenser system.	N/A
6.5.5	1	Pressure Suppression Pool as a Fission Product Cleanup System	N/A	Fission product retention in suppression pool	N/A – BWRX-300 design does not incorporate a suppression pool.	N/A
6.6	2	Inservice Inspection and Testing of Class 2 and 3 Components	1	Components subject to inspection	Conforms	6.6.1 3A
6.6	2	Inservice Inspection and Testing of Class 2 and 3 Components	2	Accessibility for inspection	Conforms	6.6.2
6.6	2	Inservice Inspection and Testing of Class 2 and 3 Components	3	Examination categories and methods	Conforms	6.6.3
6.6	2	Inservice Inspection and Testing of Class 2 and 3 Components	4	Inspection intervals	Conforms	6.6.4
6.6	2	Inservice Inspection and Testing of Class 2 and 3 Components	5	Evaluation of examination results	Conforms	6.6.6
6.6	2	Inservice Inspection and Testing of Class 2 and 3 Components	6	System pressure tests	Conforms	6.6.8
6.6	2	Inservice Inspection and Testing of Class 2 and 3 Components	7	Augmented ISI to protect against piping failures	Conforms	6.6.9
6.6	2	Inservice Inspection and Testing of Class 2 and 3 Components	8	Code exemptions	Conforms	6.6.7

Table 1.9-6 Conformance with NUREG-0800 (Chapter 6 Engineered Safety Features)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
6.6	2	Inservice Inspection and Testing of Class 2 and 3 Components	9	Evaluation of relief requests per 10 CFR 50.55a	Conforms	6.6
6.6	2	Inservice Inspection and Testing of Class 2 and 3 Components	10	Exemptions from Code examination requirements permitted by Code cases	Conforms	6.6.7
6.6	2	Inservice Inspection and Testing of Class 2 and 3 Components	11	Operational programs	FSAR	FSAR
6.7	2	Main Steam Isolation Valve Control System (BWR)	N/A	Main Steam Line valve leakage after LOCA	N/A – BWRX-300 design does not include a main steam isolation valve control system.	N/A
BTP 6-1	0	pH for Emergency Coolant water for Pressurized Water Reactors	N/A	pH of PWR containment sprays	N/A – BWR design	N/A
BTP 6-2	3	Minimum Containment Pressure Model for PWR ECCS Performance Evaluation	N/A	Containment pressure to evaluate PWR ECCS performance	N/A – BWR design	N/A
BTP 6-3	3	Determination of Bypass Leakage Paths in Dual Containment Plants	N/A	Bypass leakage from primary containment to environment	N/A – BWRX-300 design does not incorporate a dual containment.	N/A
BTP 6-4	3	Containment Purging During Normal Plant Operation	N/A	Containment purging during normal operation	N/A – BWRX-300 containment pressure is maintained by the CIS with temperature maintained by the CCS.	N/A

Table 1.9-6 Conformance with NUREG-0800 (Chapter 6 Engineered Safety Features)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
BTP 6-5	3	Currently the Responsibility of Reactor Systems Piping from the Refueling Water Storage Tank (RWST) (or BWST) and Containment Sump(s) to the Safety Injection Pumps	N/A	PWR use of RWST (or BWST) and pumps to provide ECCS injection	N/A – BWRX-300 design does not include ECCS injection pumps.	N/A

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

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)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
8.1	4	Electric Power-Introduction	II	Reference to other sections for specific acceptance criteria	Conforms	8.1
8.2	5	Offsite Power System	1	Withstanding effects of natural phenomena per GDC 2 requirements	N/A – No Safety Category 1 functions rely on offsite AC power. Therefore, the Offsite Power System is not required to meet the requirements of GDC 2.	N/A
8.2	5	Offsite Power System	2	Protection against dynamic effects, including missiles generated from equipment failure per GDC 4 requirements	N/A – No Safety Category 1 functions rely on offsite AC power. Therefore, the Offsite Power System is not required to meet the requirements of GDC 4.	N/A
8.2	5	Offsite Power System	3	Sharing of SSCs at multi-unit stations per GDC 5 requirements	N/A – CRN-1 application is for a single unit.	N/A
8.2	5	Offsite Power System	4	Capability, capacity, availability of preferred power source per GDC 17 requirements	N/A – Offsite power is not required for accident mitigation.	N/A
8.2	5	Offsite Power System	5	Inspection and testing per GDC 18 requirements	N/A – No Safety Category 1 functions rely on offsite AC power. Therefore, the Offsite Power System is not required to meet the requirements of GDC 18.	N/A

Table 1.9-8 Conformance with NUREG-0800 (Chapter 8 Electric Power)
(Sheet 2 of 7)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
8.2	5	Offsite Power System	6	Operation per GDCs 34, 38, 41, and 44 requirements	N/A – Offsite power system not required for accident mitigation.	N/A
8.2	5	Offsite Power System	7	Alternate source per 10 CFR 50.63 requirements	N/A – BWRX-300 design is a passive design with a 72-hour coping capability.	N/A
8.2	5	Offsite Power System	8	Maintenance per 10 CFR 50.65 requirements	FSAR	FSAR
8.3.1	4	AC Power Systems (Onsite)	1	Withstanding natural phenomena per GDC 2 requirements	N/A – No offsite or standby diesel generator AC power is required for accident mitigation.	N/A
8.3.1	4	AC Power Systems (Onsite)	2	Protection against environmental conditions, including missiles generated from equipment failure per GDC 4 requirements	N/A – No offsite or standby diesel generator AC power is required for accident mitigation.	N/A
8.3.1	4	AC Power Systems (Onsite)	3	Sharing of SSCs at multi-unit stations per GDC 5 requirements	N/A – CRN-1 application is for a single unit.	N/A
8.3.1	4	AC Power Systems (Onsite)	4	Capability, capacity, availability of preferred power source per GDC 17 requirements	Alternate Approach	8.1.3
8.3.1	4	AC Power Systems (Onsite)	5	Inspection and testing per GDC 18 requirements	N/A – No offsite or standby diesel generator AC power is required for accident mitigation.	N/A

Table 1.9-8 Conformance with NUREG-0800 (Chapter 8 Electric Power)
(Sheet 3 of 7)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
8.3.1	4	AC Power Systems (Onsite)	6	Design requirements per GDCs 34, 35, 38, 41, and 44 requirements	N/A – No offsite or standby diesel generator AC power is required for accident mitigation.	N/A
8.3.1	4	AC Power Systems (Onsite)	7	Design of containment electrical penetrations per GDC 50	Conforms	8.1.3
8.3.1	4	AC Power Systems (Onsite)	8	Redundancy and reliability of diesel generators per 10 CFR 50.63 requirements	N/A – BWRX-300 design does not include SC1 Emergency Diesel Generators (EDG).	N/A
8.3.1	4	AC Power Systems (Onsite)	9	Maintenance per 10 CFR 50.65 requirements	FSAR	FSAR
8.3.1	4	AC Power Systems (Onsite)	10	Guidance for complying with 10 CFR 50.55a(h): IEEE Std. 603-1991	N/A – No offsite or standby diesel generator AC power is required for accident mitigation	N/A
8.3.2	4	DC Power Systems (Onsite)	1	Independence between redundant DC power systems per RG 1.6	Conforms	Table 1.9-20
8.3.2	4	DC Power Systems (Onsite)	2	Design, operation, testing per RG 1.32	Conforms (Note - testing requirements addressed in FSAR)	Table 1.9-20
8.3.2	4	DC Power Systems (Onsite)	3	Physical independences per RG 1.75	Alternate Approach	Table 1.9-20
8.3.2	4	DC Power Systems (Onsite)	4	Sharing of SSCs, multi-unit sites, per RG 1.81	N/A – CRN-1 application is for a single unit.	N/A

Table 1.9-8 Conformance with NUREG-0800 (Chapter 8 Electric Power)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
8.3.2	4	DC Power Systems (Onsite)	5	Installation of vented lead-acid storage batteries per RG 1.128:	Conforms	Table 1.9-20
8.3.2	4	DC Power Systems (Onsite)	6	Maintenance of vented lead-acid storage batteries per RG 1.129	Conforms	Table 1.9-20
8.3.2	4	DC Power Systems (Onsite)	7	Periodic testing per RG 1.118	Alternate Approach	Table 1.9-20
8.3.2	4	DC Power Systems (Onsite)	8	I&C design, qualification, testing per RG 1.153	Alternate Approach	Table 1.9-20
8.3.2	4	DC Power Systems (Onsite)	9	Single failure criterion per RG 1.53	Alternate Approach	Table 1.9-20
8.3.2	4	DC Power Systems (Onsite)	10	Electric penetration assembly capability with withstand LOCA per RG 1.63	Alternate Approach	Table 1.9-20
8.3.2	4	DC Power Systems (Onsite)	11	SBO capability per RG 1.155	Conforms	Table 1.9-20
8.3.2	4	DC Power Systems (Onsite)	12	Maintenance per RG 1.160	FSAR	FSAR
8.3.2	4	DC Power Systems (Onsite)	13	Managing risk during maintenance per RG 1.182	N/A – RG 1.182 has been withdrawn.	N/A
8.4	1	Station Blackout	1	Compliance with 10 CFR 50.63 per RG 1.155	Conforms	8.1.3 8.4
8.4	1	Station Blackout	2	Alternate AC and Regulatory Treatment of Non-Safety Systems guidelines per SECY-90-016, SECY-94-084	N/A – AC power is not required for SBO mitigation.	N/A
8.4	1	Station Blackout	3	Reliability goals for emergency AC power sources per RG 1.9	N/A – No offsite or standby diesel generator AC power is required for accident mitigation.	N/A

Table 1.9-8 Conformance with NUREG-0800 (Chapter 8 Electric Power)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
8.4	1	Station Blackout	3	Reliability goals for emergency AC power sources per RG 1.155	Conforms	Table 1.9-20
8.4	1	Station Blackout	4	Maintenance of onsite emergency AC power source per RG 1.160	FSAR	FSAR
8.4	1	Station Blackout	5	Managing risk when performing maintenance per RG 1.182	N/A – RG 1.182 has been withdrawn.	N/A
Appendix 8-A	1	General Agenda, Station Site Visits	N/A	USNRC site visits as part of the OL review	FSAR	FSAR
8-A	2	Branch Technical Position (PSB)	N/A	Branch Technical Position (PSB)	N/A – See BTPs below	N/A
Appendix 8-B	0	General Agenda, Station Site Visits	N/A	USNRC site visits as part of the OL review	FSAR	FSAR
BTP 8-1	3	Requirements on Motor-Operated Valves in the ECCS Accumulator Lines	N/A	Requirements on Motor-Operated Valves in the ECCS Accumulator Lines	N/A – BWRX-300 design does not use ECCS accumulators.	N/A
BTP 8-2	3	Use of Diesel-Generator Sets for Peaking	N/A	Use of Diesel-Generator Sets for Peaking	N/A – Standby Diesel Generators are not used for peaking.	N/A
BTP 8-3	3	Stability of Offsite Power Systems	BTP.1	Power systems with supporting grid interfaces	FSAR	FSAR
BTP 8-3	3	Stability of Offsite Power Systems	BTP.2	Isolated power systems	FSAR	FSAR
BTP 8-4	3	Application of the Single Failure Criterion to Manually Controlled Electrically Operated Valves	N/A	Application of the Single Failure Criterion to Manually Controlled Electrically Operated Valves	N/A – BWRX-300 design does not include SC1 Motor Operated Valves.	N/A

Table 1.9-8 Conformance with NUREG-0800 (Chapter 8 Electric Power)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
BTP 8-5	3	Supplemental Guidance for Bypass and Inoperable Status Indication for Engineered Safety Features Systems	BTP.1	Bypass indication	Conforms	8.1.3 8.3.1.13 8.3.2.1
BTP 8-5	3	Supplemental Guidance for Bypass and Inoperable Status Indication for Engineered Safety Features Systems	BTP.2	Protective function bypass of shared system	Conforms	8.1.3
BTP 8-5	3	Supplemental Guidance for Bypass and Inoperable Status Indication for Engineered Safety Features Systems	BTP.3	Cancelling erroneous bypass indications	Conforms	8.1.3
BTP 8-5	3	Supplemental Guidance for Bypass and Inoperable Status Indication for Engineered Safety Features Systems	BTP.4	Use of indication for safety systems; immediate operator action	Conforms	8.1.3
BTP 8-5	3	Supplemental Guidance for Bypass and Inoperable Status Indication for Engineered Safety Features Systems	BTP.5	Adverse effects of indication on safety systems	Conforms	8.1.3
BTP 8-5	3	Supplemental Guidance for Bypass and Inoperable Status Indication for Engineered Safety Features Systems	BTP.6	Verification of indicating and annunciating function	Conforms	8.1.3

Table 1.9-8 Conformance with NUREG-0800 (Chapter 8 Electric Power)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
BTP 8-6	3	Adequacy of Station Electric Distribution System Voltages	N/A	Adequacy of Station Electric Distribution System Voltages	N/A – BWRX-300 offsite power system does not supply power to any SC1 equipment loads and does not support Safety Category 1 functions.	N/A
BTP 8-7	3	Criteria for Alarms and Indications Associated with Diesel-Generator Unit Bypassed and Inoperable Status	N/A	Criteria for Alarms and Indications Associated with Diesel-Generator Unit Bypassed and Inoperable Status	N/A – BWRX-300 design does not include SC1 diesel generators.	N/A
BTP 8-8	0	Onsite (Emergency Diesel Generators) and Offsite Power Sources Allowed Outage Time Extensions	N/A	Onsite (Emergency Diesel Generators) and Offsite Power Sources Allowed Outage Time Extensions	N/A – BWRX-300 design does not include SC1 diesel generators.	N/A
BTP 8-9	0	Open Phase Conditions in Electric Power System	BTP.1	Open phase conditions	Conforms	8.1.3 8.2.2
BTP 8-9	0	Open Phase Conditions in Electric Power System	BTP.2	Open phase design criteria for plants with active safety features	N/A – BWRX-300 design includes passive safety features.	N/A
BTP 8-9	0	Open Phase Conditions in Electric Power System	BTP.3	Open phase design criteria for plants with passive safety features	Conforms	8.1.3 8.2.2

Table 1.9-9 Conformance with NUREG-0800 (Chapter 9 Auxiliary Systems)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
9.1.1	3	Criticality Safety of Fresh and Spent Fuel Storage and Handling	1	Compliance with GDC 62	Conforms	9.1.A
9.1.2	4	New and Spent Fuel Storage	1	Compliance with regard to GDC 2	Conforms	9.1.2
9.1.2	4	New and Spent Fuel Storage	2	Compliance with regard to GDC 4	Conforms	9.1.2
9.1.2	4	New and Spent Fuel Storage	3	Compliance with regard to GDC 5	N/A – CRN-1 application is for a single unit.	N/A
9.1.2	4	New and Spent Fuel Storage	4	Compliance with regard to GDC 61	Conforms	9.1.2
9.1.2	4	New and Spent Fuel Storage	5	Compliance with regard to GDC 63	Conforms	9.1.2
9.1.2	4	New and Spent Fuel Storage	6	Compliance with regard to 10 CFR 20.1101(b)	Conforms	9.1.2
9.1.2	4	New and Spent Fuel Storage	7	Compliance with 10 CFR 50.68 or 10 CFR 70.24	Conforms	9.1.A
9.1.3	2	Spent Fuel Pool Cooling and Cleanup System	1	Compliance with GDC 2	Conforms	9.1.3
9.1.3	2	Spent Fuel Pool Cooling and Cleanup System	2	Compliance with GDC 4	Conforms	9.1.3
9.1.3	2	Spent Fuel Pool Cooling and Cleanup System	3	Compliance with GDC 5	N/A – CRN-1 application is for a single unit.	N/A
9.1.3	2	Spent Fuel Pool Cooling and Cleanup System	4	Compliance with GDC 61	Conforms	9.1.3
9.1.3	2	Spent Fuel Pool Cooling and Cleanup System	5	Compliance with GDC 63	Conforms	9.1.3

Table 1.9-9 Conformance with NUREG-0800 (Chapter 9 Auxiliary Systems)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
9.1.3	2	Spent Fuel Pool Cooling and Cleanup System	6	Compliance with regard to 10 CFR 20.1101(b)	Conforms	9.1.3 12.3
9.1.3	2	Spent Fuel Pool Cooling and Cleanup System	7	Compliance with 10 CFR 52.47(b)(1)	N/A – 10 CFR Part 50 CPA	N/A
9.1.3	2	Spent Fuel Pool Cooling and Cleanup System	8	Compliance with 10 CFR 52.80(a)	N/A – 10 CFR Part 50 CPA	N/A
9.1.4	4	Light Load Handling System and Related Refueling Operations	1	Compliance with GDC 2	Conforms	9.1.4
9.1.4	4	Light Load Handling System and Related Refueling Operations	2	Compliance with GDC 5	N/A – CRN-1 application is for a single unit.	N/A
9.1.4	4	Light Load Handling System and Related Refueling Operations	3	Compliance with GDC 61	Conforms	9.1.4
9.1.4	4	Light Load Handling System and Related Refueling Operations	4	Compliance with GDC 62	Conforms	9.1.4
9.1.5	1	Overhead Heavy Load Handling Systems	1	Compliance with GDC 1	Conforms	9.1.5
9.1.5	1	Overhead Heavy Load Handling Systems	2	Compliance with GDC 2	Conforms	9.1.5
9.1.5	1	Overhead Heavy Load Handling Systems	3	Compliance with GDC 4	Conforms	9.1.5
9.1.5	1	Overhead Heavy Load Handling Systems	4	Compliance with GDC 5	N/A – CRN-1 application is for a single unit.	N/A
9.2.1	5	Station Service Water System	1	Compliance with GDC 2	N/A – No SC1 SSCs affected by this system.	N/A
9.2.1	5	Station Service Water System	2	Compliance with GDC 4	N/A – No SC1 SSCs affected by this system.	N/A

Table 1.9-9 Conformance with NUREG-0800 (Chapter 9 Auxiliary Systems)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
9.2.1	5	Station Service Water System	3	Compliance with GDC 5	N/A – CRN-1 application is for a single unit.	N/A
9.2.1	5	Station Service Water System	4	Compliance with GDC 44	N/A – SW not required to provide cooling to any SC1 components and is not credited as the ultimate heat sink in the Chapter 15 safety analyses.	N/A
9.2.1	5	Station Service Water System	5	Compliance with GDC 45	N/A – SW not required to provide cooling to any SC1 components, no specific inspection requirements are implemented for this GDC.	N/A
9.2.1	5	Station Service Water System	6	Compliance with GDC 46	N/A – SW not required to provide cooling to any SC1 components, so no specific testing requirements are implemented for this GDC.	N/A
9.2.2	4	Reactor Auxiliary Cooling Water Systems	1	Compliance with GDC 2	N/A – Equipment in PCW or cooled by the PCW not required to mitigate DBAs.	N/A
9.2.2	4	Reactor Auxiliary Cooling Water Systems	2	Compliance with GDC 4	N/A – Equipment in PCW or cooled by the PCW not required to mitigate DBAs.	N/A
9.2.2	4	Reactor Auxiliary Cooling Water Systems	3	Compliance with GDC 5	N/A – CRN-1 application is for a single unit.	N/A

Table 1.9-9 Conformance with NUREG-0800 (Chapter 9 Auxiliary Systems)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
9.2.2	4	Reactor Auxiliary Cooling Water Systems	4	Compliance with GDC 44	N/A – PCW not credited for heat removal during DBAs.	N/A
9.2.2	4	Reactor Auxiliary Cooling Water Systems	5	Compliance with GDC 45	N/A – PCW not required to provide cooling during DBAs.	N/A
9.2.2	4	Reactor Auxiliary Cooling Water Systems	6	Compliance with GDC 46	N/A – PCW not required to provide cooling during DBAs.	N/A
9.2.4	3	Potable and Sanitary Water Systems	1	Compliance with GDC 60	Conforms	9.2.4
9.2.5	3	Ultimate Heat Sink	1	Compliance with GDC 2	Conforms	9.2.5
9.2.5	3	Ultimate Heat Sink	2	Compliance with GDC 5	N/A – CRN-1 application is for a single unit.	N/A
9.2.5	3	Ultimate Heat Sink	3	Compliance with GDC 44	Conforms	9.2.5
9.2.5	3	Ultimate Heat Sink	4	Compliance with GDC 45	Conforms	9.2.5
9.2.5	3	Ultimate Heat Sink	5	Compliance with GDC 46	Conforms	9.2.5
9.2.6	3	Condensate Storage Facilities	1	Compliance with GDC 2	N/A – Condensate Storage and Transfer Subsystem not required to operate during or after an accident.	N/A
9.2.6	3	Condensate Storage Facilities	2	Compliance with GDC 5	N/A – CRN-1 application is for a single unit.	N/A
9.2.6	3	Condensate Storage Facilities	3	Compliance with GDC 44	N/A – Condensate Storage and Transfer Subsystem not required to operate during or after an accident.	N/A

Table 1.9-9 Conformance with NUREG-0800 (Chapter 9 Auxiliary Systems)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
9.2.6	3	Condensate Storage Facilities	4	Compliance with GDC 45	N/A – Condensate Storage and Transfer Subsystem not required to operate during or after an accident.	N/A
9.2.6	3	Condensate Storage Facilities	5	Compliance with GDC 46	N/A – Condensate Storage and Transfer Subsystem not required to operate during or after an accident.	N/A
9.2.6	3	Condensate Storage Facilities	6	Compliance with GDC 60	Conforms	9.2.6
9.2.6	3	Condensate Storage Facilities	7	Compliance with 10 CFR 50.63	N/A – Condensate Storage Facilities do not support functions needed for SBO mitigation.	N/A
9.2.7	0	Chilled Water	1	Compliance with GDC 1	N/A – Equipment in CWE or cooled by the CWE not required to mitigate DBAs.	N/A
9.2.7	0	Chilled Water	2	Compliance with GDC 2	N/A – Equipment in CWE or cooled by the CWE not required to mitigate DBAs.	N/A
9.2.7	0	Chilled Water	3	Compliance with GDC 4	N/A – Equipment in CWE or cooled by the CWE not required to mitigate DBAs.	N/A
9.2.7	0	Chilled Water	4	Compliance with GDC 5	N/A – CRN-1 application is for a single unit.	N/A
9.2.7	0	Chilled Water	5	Compliance with GDC 44	N/A – Equipment in CWE or cooled by the CWE not required to mitigate DBAs.	N/A

Table 1.9-9 Conformance with NUREG-0800 (Chapter 9 Auxiliary Systems)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
9.2.7	0	Chilled Water	6	Compliance with GDC 45	N/A – No SC1 SSCs cooled by CWE, and no specific inspection requirements are established for this GDC.	N/A
9.2.7	0	Chilled Water	7	Compliance with GDC 46	N/A – No SC1 SSCs cooled by CWE, and no specific inspection requirements are established for this GDC.	N/A
9.2.7	0	Chilled Water	8	Compliance with 10 CFR 20.1406	Conforms	9.2.7
9.3.1	2	Compressed Air System	1	Compliance with GDC 1	Conforms	9.3.1
9.3.1	2	Compressed Air System	2	Compliance with GDC 2	Conforms	9.3.1
9.3.1	2	Compressed Air System	3	Compliance with GDC 5	N/A – CRN-1 application is for a single unit.	N/A
9.3.1	2	Compressed Air System	4	Compliance with 10 CFR 50.63	N/A – Plant Pneumatics System (PPS) not required to achieve safe shutdown.	N/A
9.3.2	3	Process and Post-Accident Sampling Systems	1	Conformance with applicable GDCs itemized by SSC	Conforms	9.3.2
9.3.2	3	Process and Post-Accident Sampling Systems	2	Analyses and frequencies included in Technical Specifications	FSAR	FSAR
9.3.2	3	Process and Post-Accident Sampling Systems	3	Functional design guidelines	Conforms	9.3.2

Table 1.9-9 Conformance with NUREG-0800 (Chapter 9 Auxiliary Systems)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
9.3.2	3	Process and Post-Accident Sampling Systems	4	GDC 1 and 2 compliance	Conforms	9.3.2
9.3.3	3	Equipment and Floor Drainage System	1	Compliance with GDC 2	Conforms	9.3.3
9.3.3	3	Equipment and Floor Drainage System	2	Compliance with GDC 4	Conforms	9.3.3
9.3.3	3	Equipment and Floor Drainage System	3	Compliance with GDC 60	Conforms	9.3.3
9.3.4	3	Chemical and Volume Control System (PWR) (Including Boron Recovery System)	N/A	Chemical and Volume Control System (PWR)	N/A - BWR design	N/A
9.3.5	3	Standby Liquid Control System (BWR)	N/A	Standby Liquid Control System (SLCS) criteria	N/A – SLCS is not included in the BWRX-300 design.	N/A
9.4.1	3	Control Room Area Ventilation System	1	Compliance with GDC 2	Conforms	9.4.1 9.4.6
9.4.1	3	Control Room Area Ventilation System	2	Compliance with GDC 4	Conforms	9.4.1 9.4.6
9.4.1	3	Control Room Area Ventilation System	3	Compliance with GDC 5	N/A – CRN-1 application is for a single unit.	N/A
9.4.1	3	Control Room Area Ventilation System	4	Compliance with GDC 19	Alternate Approach	9.4.1 9.4.6
9.4.1	3	Control Room Area Ventilation System	5	Compliance with GDC 60	Conforms	9.4.1 9.4.6
9.4.1	3	Control Room Area Ventilation System	6	Compliance with 10 CFR 50.63	Conforms	9.4.1 9.4.6

Table 1.9-9 Conformance with NUREG-0800 (Chapter 9 Auxiliary Systems)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
9.4.2	3	Spent Fuel Pool Area Ventilation System	1	Compliance with GDC 2	Conforms	9.4.6
9.4.2	3	Spent Fuel Pool Area Ventilation System	2	Compliance with GDC 5	N/A – CRN-1 application is for a single unit.	N/A
9.4.2	3	Spent Fuel Pool Area Ventilation System	3	Compliance with GDC 60	Conforms	9.4.6
9.4.2	3	Spent Fuel Pool Area Ventilation System	4	Compliance with GDC 61	Conforms	9.4.6
9.4.3	3	Auxiliary and Radwaste Area Ventilation System	1	Compliance with GDC 2	Conforms	9.4.3
9.4.3	3	Auxiliary and Radwaste Area Ventilation System	2	Compliance with GDC 5	N/A – CRN-1 application is for a single unit.	N/A
9.4.3	3	Auxiliary and Radwaste Area Ventilation System	3	Compliance with GDC 60	Conforms	9.4.3
9.4.4	3	Turbine Area Ventilation System	1	Compliance with GDC 2	Conforms	9.4.4
9.4.4	3	Turbine Area Ventilation System	2	Compliance with GDC 5	N/A – CRN-1 application is for a single unit.	N/A
9.4.4	3	Turbine Area Ventilation System	3	Compliance with GDC 60	Conforms	9.4.4
9.4.5	3	Engineered Safety Feature Ventilation System	1	Compliance with GDC 2	N/A – Not an engineered safety feature.	N/A
9.4.5	3	Engineered Safety Feature Ventilation System	2	Compliance with GDC 4	N/A – Not an engineered safety feature.	N/A
9.4.5	3	Engineered Safety Feature Ventilation System	3	Compliance with GDC 5	N/A – CRN-1 application is for a single unit.	N/A
9.4.5	3	Engineered Safety Feature Ventilation System	4	Compliance with GDC 17	N/A – Not an engineered safety feature.	N/A

Table 1.9-9 Conformance with NUREG-0800 (Chapter 9 Auxiliary Systems)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
9.4.5	3	Engineered Safety Feature Ventilation System	5	Compliance with GDC 60	N/A – Not an engineered safety feature.	N/A
9.4.5	3	Engineered Safety Feature Ventilation System	6	Compliance with 10 CFR 50.63	N/A – Not an engineered safety feature.	N/A
9.5.1.1	0	Fire Protection Program	1	Use of Probability Risk Assessment (PRA) to support changes in the fire protection licensing basis	N/A – 10 CFR Part 50 CPA	N/A
9.5.1.1	0	Fire Protection Program	2	Fire Protection Program considerations for license renewal	N/A – 10 CFR Part 50 CPA	N/A
9.5.1.1	0	Fire Protection Program	3	Staff positions and guidelines for Fire Protection Program	Alternate Approach	9.5.1.9
9.5.1.1	0	Fire Protection Program	4	Fire Protection Program under license termination	N/A – 10 CFR Part 50 CPA	N/A
9.5.1.1	0	Fire Protection Program	5	Fire Protection Program submitted under 10 CFR Part 52	N/A – 10 CFR Part 50 CPA	N/A
9.5.1.1	0	Fire Protection Program	6	Fire protection criteria for evolutionary and passive designs	Conforms	9.5.1
9.5.1.1	0	Fire Protection Program	7	Operational Fire Protection Program	FSAR	FSAR
9.5.1.2	0	Risk-Informed, Performance-Based Fire Protection Program	N/A	Risk-Informed Performance-Based (RIPB) Fire Protection Program	N/A – 10 CFR Part 50 CPA	N/A
9.5.2	3	Communications Systems	1	Compliance with Appendix E to 10 CFR 50, Part IV.E(9)	Conforms	9.5.2

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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
9.5.2	3	Communications Systems	2	Compliance with 10 CFR 50.34(f)(2)(xxv)	FSAR	FSAR
9.5.2	3	Communications Systems	3	Compliance with 10 CFR 50.47(a)(8)	Conforms	9.5.2 13.3
9.5.2	3	Communications Systems	4	Compliance with 10 CFR 50.55a	Conforms	9.5.2
9.5.2	3	Communications Systems	5	Compliance with GDC 1	Conforms	3.1
9.5.2	3	Communications Systems	6	Compliance with GDC 2	Conforms	3.1
9.5.2	3	Communications Systems	7	Compliance with GDC 3	Conforms	3.1
9.5.2	3	Communications Systems	8	Compliance with GDC 4	Conforms	3.1
9.5.2	3	Communications Systems	9	Compliance with GDC 19	Alternate Approach	9.5.2
9.5.2	3	Communications Systems	10	Compliance with 10 CFR 73.45(e)(2)(iii)	FSAR	FSAR
9.5.2	3	Communications Systems	11	Compliance with 10 CFR 73.45(g)(4)(i)	FSAR	FSAR
9.5.2	3	Communications Systems	12	Compliance with 10 CFR 73.46(f)	FSAR	FSAR
9.5.2	3	Communications Systems	13	Compliance with 10 CFR 73.55(e)	FSAR	FSAR
9.5.2	3	Communications Systems	14	Compliance with 10 CFR 73.55(f)	FSAR	FSAR
9.5.3	3	Lighting Systems	1	Operating experience with normal and emergency lighting systems	Conforms	9.5.3
9.5.3	3	Lighting Systems	2	Integrated design of normal lighting system	Conforms	9.5.3

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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
9.5.3	3	Lighting Systems	3	Integrated design of emergency lighting system	Conforms	9.5.3
9.5.3	3	Lighting Systems	4	Lighting levels	Conforms	9.5.3
9.5.4	3	Emergency Diesel Engine Fuel Oil Storage and Transfer System	1	Compliance with GDC 2	N/A – BWRX-300 design does not include SC1 EDGs.	N/A
9.5.4	3	Emergency Diesel Engine Fuel Oil Storage and Transfer System	2	Compliance with GDC 4	N/A – BWRX-300 design does not include SC1 EDGs.	N/A
9.5.4	3	Emergency Diesel Engine Fuel Oil Storage and Transfer System	3	Compliance with GDC 5	N/A – BWRX-300 design does not include SC1 EDGs.	N/A
9.5.4	3	Emergency Diesel Engine Fuel Oil Storage and Transfer System	4	Compliance with GDC 17	N/A – BWRX-300 design does not include SC1 EDGs.	N/A
9.5.5	3	Emergency Diesel Engine Cooling Water System	1	Compliance with GDC 2	N/A – BWRX-300 design does not include SC1 EDGs.	N/A
9.5.5	3	Emergency Diesel Engine Cooling Water System	2	Compliance with GDC 4	N/A – BWRX-300 design does not include SC1 EDGs.	N/A
9.5.5	3	Emergency Diesel Engine Cooling Water System	3	Compliance with GDC 5	N/A – BWRX-300 design does not include SC1 EDGs.	N/A
9.5.5	3	Emergency Diesel Engine Cooling Water System	4	Compliance with GDC 17	N/A – BWRX-300 design does not include SC1 EDGs.	N/A

Table 1.9-9 Conformance with NUREG-0800 (Chapter 9 Auxiliary Systems)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
9.5.5	3	Emergency Diesel Engine Cooling Water System	5	Compliance with GDC 44	N/A – BWRX-300 design does not include SC1 EDGs.	N/A
9.5.5	3	Emergency Diesel Engine Cooling Water System	6	Compliance with GDC 45	N/A – BWRX-300 design does not include SC1 EDGs.	N/A
9.5.5	3	Emergency Diesel Engine Cooling Water System	7	Compliance with GDC 46	N/A – BWRX-300 design does not include SC1 EDGs.	N/A
9.5.6	3	Emergency Diesel Engine Starting System	1	Compliance with GDC 2	N/A – BWRX-300 design does not include SC1 EDGs.	N/A
9.5.6	3	Emergency Diesel Engine Starting System	2	Compliance with GDC 4	N/A – BWRX-300 design does not include SC1 EDGs.	N/A
9.5.6	3	Emergency Diesel Engine Starting System	3	Compliance with GDC 5	N/A – BWRX-300 design does not include SC1 EDGs.	N/A
9.5.6	3	Emergency Diesel Engine Starting System	4	Compliance with GDC 17	N/A – BWRX-300 design does not include SC1 EDGs.	N/A
9.5.7	3	Emergency Diesel Engine Lubrication System	1	Compliance with GDC 2	N/A – BWRX-300 design does not include SC1 EDGs.	N/A
9.5.7	3	Emergency Diesel Engine Lubrication System	2	Compliance with GDC 4	N/A – BWRX-300 design does not include SC1 EDGs.	N/A

Table 1.9-9 Conformance with NUREG-0800 (Chapter 9 Auxiliary Systems)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
9.5.7	3	Emergency Diesel Engine Lubrication System	3	Compliance with GDC 5	N/A – BWRX-300 design does not include SC1 EDGs.	N/A
9.5.7	3	Emergency Diesel Engine Lubrication System	4	Compliance with GDC 17	N/A – BWRX-300 design does not include SC1 EDGs.	N/A
9.5.8	3	Emergency Diesel Engine Combustion Air Intake and Exhaust System	1	Compliance with GDC 2	N/A – BWRX-300 design does not include SC1 EDGs.	N/A
9.5.8	3	Emergency Diesel Engine Combustion Air Intake and Exhaust System	2	Compliance with GDC 4	N/A – BWRX-300 design does not include SC1 EDGs.	N/A
9.5.8	3	Emergency Diesel Engine Combustion Air Intake and Exhaust System	3	Compliance with GDC 5	N/A – BWRX-300 design does not include SC1 EDGs.	N/A
9.5.8	3	Emergency Diesel Engine Combustion Air Intake and Exhaust System	4	Compliance with GDC 17	N/A – BWRX-300 design does not include SC1 EDGs.	N/A

Table 1.9-10 Conformance with NUREG-0800 (Chapter 10 Steam and Power Conversion System)
(Sheet 1 of 4)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
10.2	3	Turbine Generator	1	Compliance with GDC 4	Conforms	10.2.4
10.2	3	Turbine Generator	2	Inservice inspection program for main steam and reheat valves	Conforms	10.2.5
10.2	3	Turbine Generator	3	Connection joints between low pressure turbine exhaust and main condenser	Conforms	10.2.4
10.2.3	2	Turbine Rotor Integrity	1	Materials selection	FSAR	FSAR
10.2.3	2	Turbine Rotor Integrity	2	Fracture toughness	FSAR	FSAR
10.2.3	2	Turbine Rotor Integrity	3	Pre-service inspection	FSAR	FSAR
10.2.3	2	Turbine Rotor Integrity	4	Turbine rotor design	FSAR	FSAR
10.2.3	2	Turbine Rotor Integrity	5	Inservice inspection	FSAR	FSAR
10.3	4	Main Steam Supply System	1	Compliance with GDC 2	Conforms	10.3.3
10.3	4	Main Steam Supply System	2	Compliance with GDC 4	Conforms	10.3.3
10.3	4	Main Steam Supply System	3	Compliance with GDC 5	N/A – CRN-1 application is for a single unit.	N/A

Table 1.9-10 Conformance with NUREG-0800 (Chapter 10 Steam and Power Conversion System)
(Sheet 2 of 4)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
10.3	4	Main Steam Supply System	4	Compliance with GDC 34: A)BTP 5-4, as it relates to design requirements for residual heat removal B)NUREG-0138 Issue #1, as it relates to credit being taken for all valves downstream of the Main Steam Isolation Valves (MSIV) to limit blowdown of a second steam generator if a steamline were to break upstream of the MSIV	A)Alternate Approach B)N/A for BWR design	6.3
10.3	4	Main Steam Supply System	5	Compliance with 10 CFR 50.63	Conforms	8.4
10.3	4	Main Steam Supply System	6	Conformance with RG 1.29	Alternate Approach	Table 1.9-20
10.3	4	Main Steam Supply System	7	Conformance with RG 1.117	Conforms	Table 1.9-20
10.3	4	Main Steam Supply System	8	Conformance with SECY 93-087	N/A – BWRX-300 design does not include a MSIV leakage control system and does not credit main steam line fission product holdup and retention in the analysis of DBA radiological consequences.	N/A
10.3.6	3	Steam and Feedwater System Materials	1	Materials selection and fabrication of Class 2 and 3 components	Conforms	5.2.3

Table 1.9-10 Conformance with NUREG-0800 (Chapter 10 Steam and Power Conversion System)
(Sheet 3 of 4)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
10.3.6	3	Steam and Feedwater System Materials	2	Fracture toughness of Class 2 and 3 components	Conforms	5.2.3
10.4.1	3	Main Condensers	1	Compliance with GDC 60	Conforms	10.4.1.3
10.4.2	3	Main Condenser Evacuation System	1	Compliance with GDC 60	Alternate Approach	10.4.1.3
10.4.3	3	Turbine Gland Sealing System	I	Compliance with GDC 60	Conforms	10.4.3.3
10.4.4	3	Turbine Bypass System	1	Piping failures	Conforms	10.4.4.3
10.4.4	3	Turbine Bypass System	2	Residual heat removal	Conforms	10.4.4.3
10.4.4	3	Turbine Bypass System	3	MSIV alternate leakage path	N/A – BWRX-300 design does not include a MSIV leakage control system and does not credit Turbine Bypass System (TBS) fission product holdup and plateout in the analysis of DBA radiological consequences.	N/A
10.4.5	3	Circulating Water System	1	Compliance with GDC 4	Conforms	10.4.5.3
10.4.6	3	Condensate Cleanup System	1	Compliance with GDC 14 for BWRs	Conforms	10.4.6.3
10.4.6	3	Condensate Cleanup System	2	Compliance with GDC for PWRs	N/A – BWR design	N/A
10.4.7	4	Condensate and Feedwater System	1	Seismic events	Conforms	10.4.7.3
10.4.7	4	Condensate and Feedwater System	2	Fluid instabilities	Conforms	10.4.7.3

Table 1.9-10 Conformance with NUREG-0800 (Chapter 10 Steam and Power Conversion System)
(Sheet 4 of 4)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
10.4.7	4	Condensate and Feedwater System	3	Sharing of structures, systems, and components	N/A – CRN-1 application is for a single unit.	N/A
10.4.7	4	Condensate and Feedwater System	4	Compliance with GDC 44	Alternate Approach	6.3 9.2
10.4.7	4	Condensate and Feedwater System	5	Inspection	Conforms	10.4.7.4
10.4.7	4	Condensate and Feedwater System	6	Testing	Conforms	10.4.7.4
10.4.7	4	Condensate and Feedwater System	7	Flow Accelerated Corrosion	Conforms	10.4.7.2
10.4.7	4	Condensate and Feedwater System	8	Feedwater nozzle design	Conforms	10.4.7.3
10.4.8	3	Steam Generator Blowdown System	N/A	-	N/A – BWR design	N/A
10.4.9	3	Auxiliary Feedwater System (PWR)	N/A	-	N/A – BWR design	N/A
BTP 10-1	3	Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants	N/A	-	N/A – BWR design	N/A
BTP 10-2	4	Design Guidelines for Avoiding Water Hammers in Steam Generators	N/A	-	N/A – BWR design	N/A

Table 1.9-11 Conformance with NUREG-0800 (Chapter 11 Radioactive Waste Management)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
11.1	4	Coolant Source Terms	1	Consideration of liquid, gaseous effluents from normal and AOO source	Conforms	11.1 11.2 11.3
11.1	4	Coolant Source Terms	2	Method for developing waste stream characteristics (e.g., volumes, concentrations, release rates)	Conforms	11.2 11.3
11.1	4	Coolant Source Terms	3	Decontamination Factors to reduce gaseous effluents	Conforms	11.2
11.1	4	Coolant Source Terms	4	Decontamination Factors to reduce liquid effluents	Conforms	11.3
11.1	4	Coolant Source Terms	5	Radwaste management system design augmentations for cost-benefit analysis	FSAR	FSAR
11.1	4	Coolant Source Terms	6	Effluent concentrations at boundary of unrestricted area compliance with 10 CFR 20	Conforms	11.2 11.3
11.1	4	Coolant Source Terms	7	Coolant and steam source terms	Conforms	11.1 11.2 11.3
11.1	4	Coolant Source Terms	8	Coolant source terms information compliance with 10 CFR 50.34	Conforms	11.1 11.2 11.3
11.1	4	Coolant Source Terms	9	Design basis reactor coolant and steam source terms	Conforms	11.1

Table 1.9-11 Conformance with NUREG-0800 (Chapter 11 Radioactive Waste Management)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
11.1	4	Coolant Source Terms	10	Differences with NUREG-0016, NUREG-0017, or ANSI/ANS 18.1-1999	Conforms	11.1 11.2 11.3
11.2	5	Liquid Waste Management System	1	Compliance with 10 CFR Part 50 Appendix I	Conforms	11.2
11.2	5	Liquid Waste Management System	2	Processing capacity during unavailability of major processing equipment	Conforms	11.2
11.2	5	Liquid Waste Management System	3	Seismic design of structures, the safety and quality group equipment, and provisions to prevent and collect spills from storage tanks	Conforms	11.2 12.1 12.3
11.2	5	Liquid Waste Management System	4	Isolation valves, radiation detectors for systems with automatic control features	Conforms	11.2 11.3 11.5
11.2	5	Liquid Waste Management System	5	Exhaust ventilation systems	Conforms	9.4 11.3
11.2	5	Liquid Waste Management System	6	Hypothetical maximum exposed person:	Conforms	11.2
11.3	4	Gaseous Waste Management System	1	GWM capability with regard to 10 CFR Part 50, Appendix I, Sections II.B and II.C	Conforms	11.3
11.3	4	Gaseous Waste Management System	1	GWM capability with regard to 10 CFR Part 50, Appendix I, Section II.D	FSAR	FSAR

Table 1.9-11 Conformance with NUREG-0800 (Chapter 11 Radioactive Waste Management)
(Sheet 3 of 9)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
11.3	4	Gaseous Waste Management System	2	GWM capability with unavailability of major processing equipment	Conforms	11.3
11.3	4	Gaseous Waste Management System	3	Seismic design	Conforms	11.3
11.3	4	Gaseous Waste Management System	4	Compliance with 10 CFR 20.1406	Conforms	12.1 12.3
11.3	4	Gaseous Waste Management System	5	Conformance with RG 1.52	N/A – BWRX-300 design does not utilize an engineered safety feature ventilation system.	N/A
11.3	4	Gaseous Waste Management System	5	Conformance with RG 1.140, RG 1.143	Conforms	Table 1.9-20
11.3	4	Gaseous Waste Management System	6	Isolation valves, radiation detectors for systems with automatic control features	Conforms	11.3, 11.5
11.3	4	Gaseous Waste Management System	7	Evaluation of gaseous radwaste system	Conforms	11.3
11.3	4	Gaseous Waste Management System	8	Placement of sample probes in stacks and ductwork	FSAR	FSAR
11.3	4	Gaseous Waste Management System	9	Design to withstand potential hydrogen explosion	Conforms	11.3
11.3	4	Gaseous Waste Management System	10	Hypothetical maximum exposed person	Conforms	11.3
11.3	4	Gaseous Waste Management System	11	Summary of relevant RGs, Interim Staff Guidance (ISG), and BTPs	Conforms	11.3

Table 1.9-11 Conformance with NUREG-0800 (Chapter 11 Radioactive Waste Management)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
11.4	4	Solid Waste Management System	1	Design parameters	Conforms	11.4
11.4	4	Solid Waste Management System	2	Processing equipment sizing	Conforms	11.4
11.4	4	Solid Waste Management System	3	Plant-specific process control program	FSAR	FSAR
11.4	4	Solid Waste Management System	4	Wet wastes	FSAR	FSAR
11.4	4	Solid Waste Management System	5	Implementation of Solid Waste Management system design objectives and criteria	FSAR	FSAR
11.4	4	Solid Waste Management System	6	Packaging and shipping of radioactive wastes	FSAR	FSAR
11.4	4	Solid Waste Management System	7	Onsite waste storage facilities	Conforms	11.4
11.4	4	Solid Waste Management System	8	Seismic design	Conforms	11.4
11.4	4	Solid Waste Management System	9	Leakage reduction	Conforms	11.4
11.4	4	Solid Waste Management System	10	Compliance with 10 CFR 20.1406	Conforms	12.1 12.3
11.4	4	Solid Waste Management System	11	Long term onsite storage	FSAR	FSAR
11.4	4	Solid Waste Management System	12	Compliance with 10 CFR 61.55, 10 CFR 61.56 regarding Class A, B, C waste	FSAR	FSAR

Table 1.9-11 Conformance with NUREG-0800 (Chapter 11 Radioactive Waste Management)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
11.4	4	Solid Waste Management System	13	Compliance with 10 CFR 61.55, 10 CFR 61.56 regarding greater than Class C waste	FSAR	FSAR
11.4	4	Solid Waste Management System	14	Compliance with 10 CFR 20.2007 regarding mixed wastes	Conforms	11.4
11.4	4	Solid Waste Management System	15	Compliance with 10 CFR 20 regarding Solid Waste Management System interface with plant systems	Conforms	11.2 11.3 11.4
11.4	4	Solid Waste Management System	16	Isolation valves, radiation detectors for systems with automatic control features	N/A – SWM has no automatic features that involve release to environment.	N/A
11.4	4	Solid Waste Management System	17	Exhaust ventilation system design	Conforms	9.4
11.4	4	Solid Waste Management System	18	Seismic design and quality group classification	Conforms	11.4
11.4	4	Solid Waste Management System	19	Operational programs; relevant RGs, ISG, and BTPs	FSAR	FSAR
11.5	6	Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	1	Instrumentation, monitoring, sampling, and analysis of effluent release paths	Conforms	11.5
11.5	6	Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	2	Instrumentation, monitoring, sampling, and analysis of waste process systems	Conforms	11.5

Table 1.9-11 Conformance with NUREG-0800 (Chapter 11 Radioactive Waste Management)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
11.5	6	Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	3	Provisions for Administrative and procedural controls; design of systems	Conforms	11.5
11.5	6	Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	3	Provisions for Administrative and procedural controls; administrative controls and procedures	FSAR	FSAR
11.5	6	Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	4	Instrumentation, monitoring, and sample analysis for gaseous effluents in event of a postulated accident; design of systems	Conforms	11.5
11.5	6	Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	4	Instrumentation, monitoring, and sample analysis for gaseous effluents in event of a postulated accident; administrative controls and procedures	FSAR	FSAR
11.5	6	Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	5	Instrumentation, monitoring, and sample analysis for liquid effluents in event of a postulated accident; design of systems	Conforms	11.5
11.5	6	Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	5	Instrumentation, monitoring, and sample analysis for liquid effluents in event of a postulated accident; administrative controls and procedures	FSAR	FSAR

Table 1.9-11 Conformance with NUREG-0800 (Chapter 11 Radioactive Waste Management)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
11.5	6	Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	6	Compliance with 10 CFR 50.34(f)	FSAR	FSAR
11.5	6	Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	7	Operational programs	FSAR	FSAR
11.5	6	Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	8	Design of leakage detections systems	N/A – BWR design	N/A
BTP 11-3	4	Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants	1	Processing requirements	Conforms	11.4.3
BTP 11-3	4	Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants	2	Stabilization, dewatering	Conforms	11.4.3
BTP 11-3	4	Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants	3	Waste storage	Conforms	11.4.3

Table 1.9-11 Conformance with NUREG-0800 (Chapter 11 Radioactive Waste Management)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
BTP 11-3	4	Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants	4	Portable solid waste systems	Conforms	11.4.3
BTP 11-3	4	Design Guidance for Solid Radioactive Waste Management Systems Installed in Light-Water-Cooled Nuclear Power Reactor Plants	5	Additional design features	Conforms	11.4.3
BTP 11-5	4	Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure	1	Leak or failure analysis	Conforms	11.3.3
BTP 11-5	4	Postulated Radioactive Releases Due to a Waste Gas System Leak or Failure	2	Staff method of analysis	Conforms	11.3.3
BTP 11-6	4	Postulated Radioactive Releases Due to Liquid-containing Tank Failures	1	Failure mechanism and radioactivity releases	Conforms	11.2
BTP 11-6	4	Postulated Radioactive Releases Due to Liquid-containing Tank Failures	2	Mitigating design features	Conforms	11.2
BTP 11-6	4	Postulated Radioactive Releases Due to Liquid-containing Tank Failures	3	Radioactive source term	Conforms	11.2

Table 1.9-11 Conformance with NUREG-0800 (Chapter 11 Radioactive Waste Management)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
BTP 11-6	4	Postulated Radioactive Releases Due to Liquid-containing Tank Failures	4	Calculations of transport capabilities in groundwater or surface water	Conforms	11.2.3
BTP 11-6	4	Postulated Radioactive Releases Due to Liquid-containing Tank Failures	5	Exposure scenarios and acceptance criteria	Conforms	11.2.3
BTP 11-6	4	Postulated Radioactive Releases Due to Liquid-containing Tank Failures	6	Dose acceptance criteria	Conforms	11.2.3
BTP 11-6	4	Postulated Radioactive Releases Due to Liquid-containing Tank Failures	7	Specifications on tank waste radioactivity concentration limits	Conforms	11.2.3

Table 1.9-12 Conformance with NUREG-0800 (Chapter 12 Radiation Protection)
(Sheet 1 of 2)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
12.1	4	Assuring that Occupational Radiation Exposures Are as Low as Is Reasonably Achievable	1	Policy considerations	Conforms	12.1 12.5
12.1	4	Assuring that Occupational Radiation Exposures Are as Low as Is Reasonably Achievable	2	Design considerations	Conforms	12.1.2
12.1	4	Assuring that Occupational Radiation Exposures Are as Low as Is Reasonably Achievable	3	Operational considerations	FSAR	FSAR
12.1	4	Assuring that Occupational Radiation Exposures Are as Low as Is Reasonably Achievable	4	Radiation protection considerations	FSAR	FSAR
12.2	4	Radiation Sources	1	Conformance with RG 1.3	N/A – RG 1.3 has been withdrawn.	N/A
12.2	4	Radiation Sources	2	Conformance with RG 1.4	N/A – RG 1.4 has been withdrawn.	N/A
12.2	4	Radiation Sources	3	Conformance with RG 1.183	Alternative Approach	Table 1.9-20
12.2	4	Radiation Sources	4	Conformance with RG 1.7	Conforms	Table 1.9-20
12.2	4	Radiation Sources	5	Conformance with RG 1.112	Conforms	Table 1.9-20
12.2	4	Radiation Sources	6	Conformance with NUREG-0737, II.B.2	Alternative Approach	15.5
12.2	4	Radiation Sources	7	Conformance with ANSI/ANS Standard 18.1	Conforms	11.1 12.2
12.2	4	Radiation Sources	8	Conformance with RG 1.89	Conforms	Table 1.9-20
12.2	4	Radiation Sources	9	Conformance with RG 4.21	Conforms	Table 1.9-20

Table 1.9-12 Conformance with NUREG-0800 (Chapter 12 Radiation Protection)
(Sheet 2 of 2)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
12.3-12.4	5	Radiation Protection Design Features	1	Facility design features	Conforms	12.3
12.3-12.4	5	Radiation Protection Design Features	2	Shielding	Conforms	12.3.2
12.3-12.4	5	Radiation Protection Design Features	3	Ventilation	Conforms	9.4 12.2.2 12.3.3
12.3-12.4	5	Radiation Protection Design Features	4	Area radiation and airborne radioactivity monitoring systems	Conforms	6.8 11.5 12.3.4
12.3-12.4	5	Radiation Protection Design Features	5	Dose assessment	Conforms	12.3.1 12.4
12.3-12.4	5	Radiation Protection Design Features	6	Minimization of contamination	Conforms	12.1 12.3
12.5	5	Operational Radiation Protection Program	1	Organization	FSAR	FSAR
12.5	5	Operational Radiation Protection Program	2	Equipment, instrumentation, and facilities	FSAR	FSAR
12.5	5	Operational Radiation Protection Program	3	Procedures	FSAR	FSAR
12.5	5	Operational Radiation Protection Program	4	Operational programs	FSAR	FSAR

Table 1.9-13 Conformance with NUREG-0800 (Chapter 13 Conduct of Operations)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
13.1.1	6	Management and Technical Support Organization	III.a	Organizational functions and responsibilities	Conforms	13.1.1
13.1.1	6	Management and Technical Support Organization	III.b	Corporate officer identification and commitment	Conforms	13.1.1
13.1.1	6	Management and Technical Support Organization	III.c	Design and construction responsibilities	Conforms	13.1.1
13.1.1	6	Management and Technical Support Organization	III.d	Consideration of human factors	Conforms	13.1.1
13.1.1	6	Management and Technical Support Organization	III.e	Communications, control, and lines of authority	Conforms	13.1.1
13.1.1	6	Management and Technical Support Organization	III.f	Personnel experience and qualification per RG 1.8	Alternate Approach	Table 1.9-20
13.1.1	6	Management and Technical Support Organization	III.g	Availability of complete and accurate information per 10 CFR 50.34(f)(3)(vii)	Conforms	Table 1.9-21
13.1.2 - 13.1.3	7	Operating Organization	III.a	Conformance with RG 1.33	Conforms	Table 1.9-20
13.1.2 - 13.1.3	7	Operating Organization	III.b	Organizational functions and responsibilities	Conforms	13.1.2
13.1.2 - 13.1.3	7	Operating Organization	III.c	Conformance with RG 1.8	Conforms	Table 1.9-20
13.1.2 - 13.1.3	7	Operating Organization	III.c	Conformance with RG 1.189	Alternate Approach	Table 1.9-20
13.1.2 - 13.1.3	7	Operating Organization	III.d	Onsite shift operating crews per 10 CFR 50.54(i), (j), (k), (l), (m) and NUREG-0737 (Items I.A.1.1, I.A.1,3)	FSAR	FSAR

Table 1.9-13 Conformance with NUREG-0800 (Chapter 13 Conduct of Operations)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
13.2.1	4	Reactor Operator Requalification Program; Reactor Operator Training	III.a	Conformance with RG 1.8	Alternate Approach	Table 1.9-20
13.2.1	4	Reactor Operator Requalification Program; Reactor Operator Training	III.b	Reactor operator training program	FSAR	FSAR
13.2.1	4	Reactor Operator Requalification Program; Reactor Operator Training	III.c	Conformance with RG 1.149	FSAR	FSAR
13.2.1	4	Reactor Operator Requalification Program; Reactor Operator Training	III.d	Licensed operator training using systems approach per 10 CFR 55.4	FSAR	FSAR
13.2.1	4	Reactor Operator Requalification Program; Reactor Operator Training	III.e	Establishment of licensed operator training by 18 months prior to fuel load	Conforms	13.4
13.2.1	4	Reactor Operator Requalification Program; Reactor Operator Training	III.f	Reactor operator requalification programs compliance with 10 CFR 55.59	FSAR	FSAR
13.2.1	4	Reactor Operator Requalification Program; Reactor Operator Training	III.g	Periodic evaluation of licensed operator training program	FSAR	FSAR
13.2.2	4	Non-Licensed Plant Staff Training	III.a	Conformance with RG 1.8	Conforms	Table 1.9-20
13.2.2	4	Non-Licensed Plant Staff Training	III.b	Non- licensed plant staff training program	FSAR	FSAR

Table 1.9-13 Conformance with NUREG-0800 (Chapter 13 Conduct of Operations)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
13.2.2	4	Non-Licensed Plant Staff Training	III.c	Establishment of non- licensed operator training by 18 months prior to fuel load	Conforms	13.4
13.2.2	4	Non-Licensed Plant Staff Training	III.d	Initial training and retraining per 10 CFR 55.4	FSAR	FSAR
13.2.2	4	Non-Licensed Plant Staff Training	III.e	Periodic evaluation of non-licensed plant staff training programs	FSAR	FSAR
13.3	3	Emergency Planning	1	Compliance with 10 CFR 50.47(b); 10 CFR Part 50 Appendix E	Addressed to the extent required to conform with Section II of 10 CFR Part 50 Appendix E.	13.3 13A
13.3	3	Emergency Planning	2	Compliance with 10 CFR 50.47(d)	N/A – 10 CFR Part 50 CPA	N/A
13.3	3	Emergency Planning	3	Compliance with 10 CFR 50.47(b)(4)	Addressed to the extent required to conform with Section II of 10 CFR Part 50 Appendix E.	13.3 13A
13.3	3	Emergency Planning	4	Compliance with 10 CFR 50.47(b)(8), (b)(9); 10 CFR Part 50 Appendix E (Section IV.E.2)	Addressed to the extent required to conform with Section II of 10 CFR Part 50 Appendix E.	13.3 13A
13.3	3	Emergency Planning	5	Compliance with 10 CFR 50.47(b)(6), (b)(8), (b)(9); 10 CFR Part 50 Appendix E (Section IV.E) per NUREG-0737 Supplement 1	Addressed to the extent required to conform with Section II of 10 CFR Part 50 Appendix E.	13.3 13A

Table 1.9-13 Conformance with NUREG-0800 (Chapter 13 Conduct of Operations)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
13.3	3	Emergency Planning	6	Compliance with 10 CFR 50.47(b)(5), (b)(6) per NUREG-0654/FEMA-REP-1, Rev 1 (Appendix 3)	Addressed to the extent required to conform with Section II of 10 CFR Part 50 Appendix E.	13.3 13A
13.3	3	Emergency Planning	7	Conformance with NUREG-0654/FEMA-REP-1, Rev 1 (Supplement 3)	Addressed to the extent required to conform with Section II of 10 CFR Part 50 Appendix E.	13.3 13A
13.3	3	Emergency Planning	8	Compliance with 10 CFR 50.47 per RG 1.101, Rev 2	Addressed to the extent required to conform with Section II of 10 CFR Part 50 Appendix E.	13.3 13A
13.3	3	Emergency Planning	9	State, tribal, and local government planning and preparedness	Addressed to the extent required to conform with Section II of 10 CFR Part 50 Appendix E.	13.3 13A
13.3	3	Emergency Planning	10	Sizing of EPZ per 10 CFR 50.33(g), 10 CFR 50.47(c)(2), 10 CFR Part 50 Appendix E, Section I	Addressed to the extent required to conform with Section II of 10 CFR Part 50 Appendix E.	13.3 13A
13.3	3	Emergency Planning	11	Evacuation time estimates	Addressed to the extent required to conform with Section II of 10 CFR Part 50 Appendix E.	13.3 13A
13.3	3	Emergency Planning	12	Emergency response data system	Addressed to the extent required to conform with Section II of 10 CFR Part 50 Appendix E.	13.3 13A

Table 1.9-13 Conformance with NUREG-0800 (Chapter 13 Conduct of Operations)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
13.3	3	Emergency Planning	13	Finding regarding emergency planning and preparedness prior to fuel loading; low power testing and training	N/A – 10 CFR Part 50 CPA	N/A
13.3	3	Emergency Planning	14	Emergency preparedness with non-participation of State and/or local governments	Addressed to the extent required to conform with Section II of 10 CFR Part 50 Appendix E.	13.3 13A
13.3	3	Emergency Planning	15	Acceptance criteria for ESP applications per 10 CFR 52.17(b)(1)	N/A – 10 CFR Part 50 CPA	N/A
13.3	3	Emergency Planning	16	Evacuation time analysis for ESP application	Addressed to the extent required to conform with Section II of 10 CFR Part 50 Appendix E	13.3 13A
13.3	3	Emergency Planning	17	Unique site characteristics	Addressed to the extent required to conform with Section II of 10 CFR Part 50 Appendix E.	13.3 13A
13.3	3	Emergency Planning	18	Documentation of letters of agreement and other arrangements	Addressed to the extent required to conform with Section II of 10 CFR Part 50 Appendix E.	13.3 13A
13.3	3	Emergency Planning	19	Conformance with NUREG-0654/FEMA-REP-1, Rev 1 (Supplement 2) with regard to ESP	N/A – 10 CFR Part 50 CPA	N/A

Table 1.9-13 Conformance with NUREG-0800 (Chapter 13 Conduct of Operations)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
13.3	3	Emergency Planning	20	Certifications from government agencies	N/A – 10 CFR Part 50 CPA	N/A
13.3	3	Emergency Planning	21	Compliance with 10 CFR 52.17(b)(3) regarding ITAAC inclusion in an ESP application	N/A – 10 CFR Part 50 CPA	N/A
13.3	3	Emergency Planning	22	Compliance with 10 CFR 52.47(b)(1) regarding ITAAC inclusion in a design certification application	N/A – 10 CFR Part 50 CPA	N/A
13.3	3	Emergency Planning	23	Compliance with 10 CFR 52.80(a) regarding ITAAC inclusion in a combined license application	N/A – 10 CFR Part 50 CPA	N/A
13.3	3	Emergency Planning	24	Generic emergency planning ITAAC	N/A – 10 CFR Part 50 CPA	N/A
13.3	3	Emergency Planning	25	Technical Support Center (TSC), Operation Support Center (OSC), near-site EOF	Addressed to the extent required to conform with Section II of 10 CFR Part 50 Appendix E	13.3 13A
13.3	3	Emergency Planning	26	Safety Parameter Display System (SPDS) in TSC and EOF	N/A – 10 CFR Part 50 CPA	N/A
13.3	3	Emergency Planning	27	Capability to promptly obtain and analyze RCS samples per 10 CFR 50.34(f)(2)(viii)	N/A – 10 CFR Part 50 CPA	N/A
13.3	3	Emergency Planning	28	Measure and record containment parameters per 10 CFR 50.34(f)(2)(xvii)	Complies	Table 1.9-21

Table 1.9-13 Conformance with NUREG-0800 (Chapter 13 Conduct of Operations)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
13.3	3	Emergency Planning	29	Notification of USNRC Operations Center per 10 CFR 50.72(a)(3), (c)(3); activation of emergency response data system per 10 CFR 73.71(a)	Addressed to the extent required to conform with Section II of 10 CFR Part 50 Appendix E.	13.3 13A
13.3	3	Emergency Planning	30	Generic communications and Commission orders	Addressed to the extent required to conform with Section II of 10 CFR Part 50 Appendix E.	13.3 13A
13.3	3	Emergency Planning	31	Operational programs	N/A – 10 CFR Part 50 CPA	N/A
13.4	3	Operational Programs	Various	Operational programs and milestones	Conforms	13.4
13.5.1	0	Administrative Procedures	II.A	General administrative procedures	Conforms	13.5.1
13.5.1	0	Administrative Procedures	II.B	Initial test program procedures	Conforms	13.5.1
13.5.1.1	2	Administrative Procedures-General	III.a	Conformance with RG 1.33	Conforms	Table 1.9-20
13.5.1.1	2	Administrative Procedures-General	III.b	Conformance with NUREG-0694, Items I.A.1.2, I.C.3	FSAR	FSAR
13.5.1.1	2	Administrative Procedures-General	III.c	Conformance with NUREG-0694, Item I.C.2	FSAR	FSAR
13.5.1.1	2	Administrative Procedures-General	III.d	Conformance with RG 1.114 with regard to CR personnel	FSAR	FSAR
13.5.1.1	2	Administrative Procedures-General	III.e	Conformance with RG 1.114 with regard to surveillance area	FSAR	FSAR
13.5.1.1	2	Administrative Procedures-General	III.f	Conformance with NUREG-0694, Item I.C.4	FSAR	FSAR

Table 1.9-13 Conformance with NUREG-0800 (Chapter 13 Conduct of Operations)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
13.5.1.1	2	Administrative Procedures-General	III.g	Administrative procedures regarding feedback on operation, design, and construction	Conforms	13.5.1
13.5.1.1	2	Administrative Procedures-General	III.h	Administrative controls regarding crane operation	Conforms	9.1 13.5.1
13.5.1.1	2	Administrative Procedures-General	III.i	Vendor information regarding safety-related components	Conforms	13.5.1
13.5.2.1	2	Operating and Emergency Operating Procedures (EOP)	II.1	Operating procedure schedule	FSAR	FSAR
13.5.2.1	2	Operating and EOPs	II.2	CR and plant procedures	FSAR	FSAR
13.6.1	2	Physical Security-Combined License and Operating Reactors	1	Conformance with RG 1.70	Alternate Approach	Table 1.9-20
13.6.1	2	Physical Security-Combined License and Operating Reactors	2	Conformance with RG 1.206	N/A – 10 CFR Part 50 CPA	N/A
13.6.1	2	Physical Security-Combined License and Operating Reactors	3	Conformance with RG 5.7	FSAR	FSAR
13.6.1	2	Physical Security-Combined License and Operating Reactors	4	Conformance with RG 5.12	FSAR	FSAR
13.6.1	2	Physical Security-Combined License and Operating Reactors	5	Conformance with RG 5.44	FSAR	FSAR
13.6.1	2	Physical Security-Combined License and Operating Reactors	6	Conformance with RG 5.54	FSAR	FSAR
13.6.1	2	Physical Security-Combined License and Operating Reactors	7	Conformance with RG 5.65	N/A – RG 5.65 has been withdrawn.	N/A

Table 1.9-13 Conformance with NUREG-0800 (Chapter 13 Conduct of Operations)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
13.6.1	2	Physical Security-Combined License and Operating Reactors	8	Conformance with RG 5.68	N/A – RG 5.68 has been withdrawn.	N/A
13.6.1	2	Physical Security-Combined License and Operating Reactors	9	Conformance with RG 5.69	FSAR	FSAR
13.6.1	2	Physical Security-Combined License and Operating Reactors	10	Conformance with RG 5.71	FSAR	FSAR
13.6.1	2	Physical Security-Combined License and Operating Reactors	11	Conformance with RG 5.74	FSAR	FSAR
13.6.1	2	Physical Security-Combined License and Operating Reactors	12	Conformance with RG 5.75	FSAR	FSAR
13.6.1	2	Physical Security-Combined License and Operating Reactors	13	Conformance with RG 5.76	FSAR	FSAR
13.6.1	2	Physical Security-Combined License and Operating Reactors	14	Conformance with RG 5.77	FSAR	FSAR
13.6.1	2	Physical Security-Combined License and Operating Reactors	15	Conformance with RG 5.81	FSAR	FSAR
13.6.1	2	Physical Security-Combined License and Operating Reactors	16	Regulatory Issue Summary 2003-06 regarding security and vital area protection	FSAR	FSAR
13.6.1	2	Physical Security-Combined License and Operating Reactors	17	Compliance with policy statement on severe accidents (ML003711521)	FSAR	FSAR
13.6.1	2	Physical Security-Combined License and Operating Reactors	18	Conformance with NUREG-1959	FSAR	FSAR
13.6.1	2	Physical Security-Combined License and Operating Reactors	19	Conformance with NUREG-1964	FSAR	FSAR

Table 1.9-13 Conformance with NUREG-0800 (Chapter 13 Conduct of Operations)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
13.6.1	2	Physical Security-Combined License and Operating Reactors	20	Conformance with NUREG/CR-4250	FSAR	FSAR
13.6.1	2	Physical Security-Combined License and Operating Reactors	21	Conformance with NUREG/CR-5899	FSAR	FSAR
13.6.1	2	Physical Security-Combined License and Operating Reactors	22	Conformance with NUREG/CR-6190	FSAR	FSAR
13.6.1	2	Physical Security-Combined License and Operating Reactors	23	Conformance with NUREG/CR-7145	FSAR	FSAR
13.6.1	2	Physical Security-Combined License and Operating Reactors	24	Conformance with cited Sandia documents	FSAR	FSAR
13.6.1	2	Physical Security-Combined License and Operating Reactors	25	Conformance with UFC 3-340-02 (Structures to Resist the Effects of Accidental Explosions, With Change 2)	FSAR	FSAR
13.6.1	2	Physical Security-Combined License and Operating Reactors	26	Conformance with NIJ Standard 0108.01 (Ballistic Resistant Protective Materials)	FSAR	FSAR
13.6.1	2	Physical Security-Combined License and Operating Reactors	27	Conformance with UL 752 (Standard for Bullet-Resisting Equipment)	FSAR	FSAR
13.6.2	2	Physical Security-Review of Physical Security System Designs-Standard Design Certification and Operator Reactor Licensing Applications	N/A	Physical Security	N/A – 10 CFR Part 50 CPA	N/A

Table 1.9-13 Conformance with NUREG-0800 (Chapter 13 Conduct of Operations)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
13.6.3	2	Physical Security-Operational Program-ESP and Reactor Siting Criteria	Various	Potential to develop site security plans to meet regulatory requirements, as applicable to construction permit application	Conforms	13.6
13.6.4	0	Access Authorization Operational Program	Various	Conformance with NEI 03-12	FSAR	FSAR
13.6.6	0	Cyber Security Plan	II.1	Compliance with 10 CFR 73.54 regarding protection against cyber attacks	FSAR	FSAR
13.6.6	0	Cyber Security Plan	II.2	Compliance with 10 CFR 73.55(a)(1), 10 CFR 73.55(b)(8) regarding a cyber security plan	FSAR	FSAR
13.6.6	0	Cyber Security Plan	II.3	Compliance with 10 CFR 73.55(m)(2) regarding audit	FSAR	FSAR
13.6.6	0	Cyber Security Plan	II.4	Compliance with 10 CFR 73 (Appendix G) regarding recording and reporting safeguards events	FSAR	FSAR
13.6.6	0	Cyber Security Plan	II.5	Compliance with 10 CFR 73.58(b) regarding adverse effects due to plant changes	FSAR	FSAR
13.7.1	0	Fitness for Duty-Operational Program	1	Compliance with 10 CFR 26, Subpart A	FSAR	FSAR
13.7.1	0	Fitness for Duty-Operational Program	2	Compliance with 10 CFR 26, Subpart B	FSAR	FSAR
13.7.1	0	Fitness for Duty-Operational Program	3	Compliance with 10 CFR 26, Subpart C	FSAR	FSAR

Table 1.9-13 Conformance with NUREG-0800 (Chapter 13 Conduct of Operations)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
13.7.1	0	Fitness for Duty-Operational Program	4	Compliance with 10 CFR 26, Subpart D	FSAR	FSAR
13.7.1	0	Fitness for Duty-Operational Program	5	Compliance with 10 CFR 26, Subpart E	FSAR	FSAR
13.7.1	0	Fitness for Duty-Operational Program	6	Compliance with 10 CFR 26, Subpart F	FSAR	FSAR
13.7.1	0	Fitness for Duty-Operational Program	7	Compliance with 10 CFR 26, Subpart G	FSAR	FSAR
13.7.1	0	Fitness for Duty-Operational Program	8	Compliance with 10 CFR 26, Subpart H	FSAR	FSAR
13.7.1	0	Fitness for Duty-Operational Program	9	Compliance with 10 CFR 26, Subpart I	FSAR	FSAR
13.7.1	0	Fitness for Duty-Operational Program	10	Compliance with 10 CFR 26, Subpart N	FSAR	FSAR
13.7.2	0	Fitness for Duty-Construction	1	Compliance with 10 CFR 26.401	Conforms	13.7

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

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
14.2	3	Initial Plant Test Program-Design Certification and New License Applicants	2	Test program conformance with RGs	FSAR	FSAR
14.2	3	Initial Plant Test Program-Design Certification and New License Applicants	3	Initial test program administrative procedures	FSAR	FSAR
14.2	3	Initial Plant Test Program-Design Certification and New License Applicants	4	Initial startup tests	FSAR	FSAR
14.2	3	Initial Plant Test Program-Design Certification and New License Applicants	5	Individual test descriptions	FSAR	FSAR
14.2	3	Initial Plant Test Program-Design Certification and New License Applicants	6	Initial test program acceptance criteria	FSAR	FSAR
14.2.1	0	Generic Guidelines for Extended Power Uprate Testing Programs	N/A	Power ascension testing programs for extended power uprate	N/A – 10 CFR Part 50 CPA	N/A
14.3	0	Inspections, Tests, Analyses, and Acceptance Criteria	N/A	ITAAC for design certification and combined license applicants	N/A – 10 CFR Part 50 CPA	N/A

Table 1.9-15 Conformance with NUREG-0800 (Chapter 15 Transient and Accident Analysis)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
15.0	3	Introduction – Transient and Accident Analysis	I.2.A	Analysis acceptance criteria for AOOs	Conforms	15.3
15.0	3	Introduction – Transient and Accident Analysis	I.2.B	Analysis acceptance criteria for postulated accidents	Alternate Approach	15.3
15.0	3	Introduction – Transient and Accident Analysis	I.3	Plant Characteristics Considered in the Safety Evaluation	Alternate Approach	15.5
15.0	3	Introduction – Transient and Accident Analysis	I.4	Assumed Protection and Safety Systems Actions	Alternate Approach	7.3.1 15.5
15.0	3	Introduction – Transient and Accident Analysis	I.5	Evaluation of Individual Initiating Events	Conforms	15.5 15.7
15.0	3	Introduction – Transient and Accident Analysis	I.6	Event Evaluation	Alternate Approach	15.0 15.5 15.7
15.0.1	0	Radiological Consequence Analyses Using Alternative Source Terms	II	Dose criteria specified in SRP 15.0.1 (Table 1) using methodology reflecting regulatory positions in RG 1.183	Alternative Approach	Table 1.9-20
15.0.2	0	Review of Transient and Accident Analysis Methods	II.1	Evaluation model	Conforms	15.2 15.5
15.0.2	0	Review of Transient and Accident Analysis Methods	II.2	Accident scenario identification process	Conforms	15.1 15.2
15.0.2	0	Review of Transient and Accident Analysis Methods	II.3	Code assessment	Conforms	15.5.1.2
15.0.2	0	Review of Transient and Accident Analysis Methods	II.4	Uncertainty analysis	Conforms	15.5.1.1 15.6

Table 1.9-15 Conformance with NUREG-0800 (Chapter 15 Transient and Accident Analysis)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
15.0.2	0	Review of Transient and Accident Analysis Methods	II.5	Quality assurance plan	Conforms	17.1
15.0.3	0	Design Basis Accident Radiological Consequences of Analyses for Advanced Light Water Reactors	II.1	Criteria for offsite radiological consequences of postulated design basis accidents defined in SRP 15.0.3 (Table 1)	Conforms	15.3 15.5.8 15.5.9.1 15.7
15.0.3	0	Design Basis Accident Radiological Consequences of Analyses for Advanced Light Water Reactors	II.2	CR radiological habitability	Conforms	15.3 15.5.8 15.5.9.1 15.7
15.0.3	0	Design Basis Accident Radiological Consequences of Analyses for Advanced Light Water Reactors	II.2	TSC radiological habitability	FSAR	FSAR
15.1.1 - 15.1.4	2	Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	II.1 (Basic Objectives)	Most limiting increase in heat removal event	Conforms	15.2.4 15.5.3

Table 1.9-15 Conformance with NUREG-0800 (Chapter 15 Transient and Accident Analysis)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
15.1.1 - 15.1.4	2	Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	II.2 (Basic Objectives)	Criteria for fuel damage and system pressure met	Conforms	15.3 15.5.3 15.7
15.1.1 - 15.1.4	2	Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	II.1 (Specific Criteria)	Reactor coolant and main steam pressure maintained below 110 % design values	Conforms	15.3 15.7
15.1.1 - 15.1.4	2	Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	II.2 (Specific Criteria)	Fuel cladding integrity in terms of critical power ratio and Minimum Critical Power Ratio (MCPR) limit	Conforms	15.3 15.7
15.1.1 - 15.1.4	2	Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	II.3 (Specific Criteria)	Moderate frequency incident should not progress to more serious condition without independent failures	Conforms	15.2 15.5.3 15.7

Table 1.9-15 Conformance with NUREG-0800 (Chapter 15 Transient and Accident Analysis)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
15.1.1 - 15.1.4	2	Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	II.4 (Specific Criteria)	Use of RG 1.105 with regard to setpoint accuracy	Alternate Approach	Table 1.9-20
15.1.1 - 15.1.4	2	Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	II.5 (Specific Criteria)	Most limiting single failure analysis conforms with RG 1.53	Alternate Approach	Table 1.9-20
15.1.1 - 15.1.4	2	Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	II.1 (Analytical Parameters)	Analytical model includes 2 % to account for power measurement uncertainties	Alternate Approach	15.2 15.5.3
15.1.1 - 15.1.4	2	Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	II.2 (Analytical Parameters)	Conservative scram characteristics are assumed	Conforms	15.2 15.3 15.5.3

Table 1.9-15 Conformance with NUREG-0800 (Chapter 15 Transient and Accident Analysis)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
15.1.1 - 15.1.4	2	Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	II.3 (Analytical Parameters)	Burnup based on most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution	Conforms	15.2 15.3 15.5.3
15.1.1 - 15.1.4	2	Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	II.4 (Analytical Parameters)	Mitigating system actuation based on setpoint accuracy consistent with RG 1.105	Alternate Approach	Table 1.9-20
15.1.5	3	Steam System Piping Failures Inside and Outside Containment (PWR)	N/A	-	N/A – BWR design	N/A
15.1.5.A	2	Radiological Consequences of Main Steam Line Failures Outside Containment of a PWR	N/A	-	N/A – BWR design	N/A
15.2.1 - 15.2.5	2	Loss of External Load; Turbine Trip; Loss of Condenser Vacuum (LOCV); Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)	II.1.A	Most limiting moderate-frequency event	Conforms	15.2.4 15.5.3.2

Table 1.9-15 Conformance with NUREG-0800 (Chapter 15 Transient and Accident Analysis)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
15.2.1 - 15.2.5	2	Loss of External Load; Turbine Trip; LOCV; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)	II.1.B	Evaluation of most limiting event for fuel damage and system pressure	Conforms	15.2.4 15.5.3.2 15.7
15.2.1 - 15.2.5	2	Loss of External Load; Turbine Trip; LOCV; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)	II.1.C	Use of RG 1.105 with regard to protection system setpoints	Alternate Approach	Table 1.9-20
15.2.1 - 15.2.5	2	Loss of External Load; Turbine Trip; LOCV; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)	II.1.D	Consideration of single failures, operator error and nonsafety-related systems consistent with RG 1.206	N/A - 10 CFR Part 50 CPA	N/A
15.2.1 - 15.2.5	2	Loss of External Load; Turbine Trip; LOCV; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)	II.2.A	Maintaining reactor coolant and main steam pressures below 110 % of design values	Conforms	15.3 15.5.3.2 15.7
15.2.1 - 15.2.5	2	Loss of External Load; Turbine Trip; LOCV; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)	II.2.B	Fuel cladding integrity in terms of minimum critical power ratio limit and applicable Specified Acceptable Fuel Design Limits (SAFDL)	Conforms	15.3 15.5.3.2 15.7

Table 1.9-15 Conformance with NUREG-0800 (Chapter 15 Transient and Accident Analysis)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
15.2.1 - 15.2.5	2	Loss of External Load; Turbine Trip; LOCV; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)	II.2.C	Moderate frequency incident should not progress to more serious condition without independent failures	Conforms	15.3 15.5.3.2 15.7
15.2.1 - 15.2.5	2	Loss of External Load; Turbine Trip; LOCV; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)	II.2.D	Use of RG 1.105 with regard to setpoint accuracy	Alternate Approach	Table 1.9-20
15.2.1 - 15.2.5	2	Loss of External Load; Turbine Trip; LOCV; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)	II.2.E	Most limiting single failure analysis conforms with RG 1.53	Alternate Approach	Table 1.9-20
15.2.1 - 15.2.5	2	Loss of External Load; Turbine Trip; LOCV; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)	II.2.F	Evaluation of nonsafety-related system performance	Conforms	15.2 15.5.3.2 15.7
15.2.1 - 15.2.5	2	Loss of External Load; Turbine Trip; LOCV; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)	II.3.A	Analytical model includes 2 % to account for power measurement uncertainties	Alternate Approach	15.5.3.2

Table 1.9-15 Conformance with NUREG-0800 (Chapter 15 Transient and Accident Analysis)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
15.2.1 - 15.2.5	2	Loss of External Load; Turbine Trip; LOCV; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)	II.3.B	Conservative scram characteristics are assumed	Conforms	15.2 15.5.3.2
15.2.1 - 15.2.5	2	Loss of External Load; Turbine Trip; LOCV; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)	II.3.C	Burnup based on most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution	Conforms	15.2 15.5.3.2
15.2.1 - 15.2.5	2	Loss of External Load; Turbine Trip; LOCV; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed)	II.3.D	Mitigating system actuation based on setpoint accuracy consistent with RG 1.105	Alternate Approach	Table 1.9-20
15.2.6	2	Loss of Nonemergency AC Power to the Station Auxiliaries	II.1	Maintaining reactor coolant and main steam pressures below 110 % of design values	Conforms	15.3 15.5.3.2 15.7
15.2.6	2	Loss of Nonemergency AC Power to the Station Auxiliaries	II.2	Fuel cladding integrity in terms of minimum critical power ratio limit and applicable SAFDL	Conforms	15.3 15.5.3.2 15.7
15.2.6	2	Loss of Nonemergency AC Power to the Station Auxiliaries	II.3	Moderate frequency incident should not progress to more serious condition without independent failures	Conforms	15.2 15.3 15.5.3.2 15.7

Table 1.9-15 Conformance with NUREG-0800 (Chapter 15 Transient and Accident Analysis)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
15.2.6	2	Loss of Nonemergency AC Power to the Station Auxiliaries	II.4	Use of RG 1.105 with regard to setpoint accuracy	Alternate Approach	Table 1.9-20
15.2.6	2	Loss of Nonemergency AC Power to the Station Auxiliaries	II.5	Most limiting single failure analysis conforms with RG 1.53	Alternate Approach	Table 1.9-20
15.2.7	2	Loss of Normal Feedwater Flow	II.1.A (Basic Objectives)	Criteria for fuel damage and system pressure met	Conforms	15.5.3.3
15.2.7	2	Loss of Normal Feedwater Flow	II.1.B (Basic Objectives)	Capacity for long term decay heat removal	Conforms	15.5.3.3
15.2.7	2	Loss of Normal Feedwater Flow	II.1.C (Basic Objectives)	Use of RG 1.105 with regard to setpoint accuracy	Alternate approach	Table 1.9-20
15.2.7	2	Loss of Normal Feedwater Flow	II.1.D (Basic Objectives)	Consideration of single failures, operator error and nonsafety-related systems consistent with RG 1.206	N/A - 10 CFR Part 50 CPA	N/A
15.2.7	2	Loss of Normal Feedwater Flow	II.2.A (Specific Criteria)	Reactor coolant and main steam pressure maintained below 110 % design values	Conforms	15.3 15.5.3.3 15.7
15.2.7	2	Loss of Normal Feedwater Flow	II.2.B (Specific Criteria)	Fuel cladding integrity in terms of critical power ratio and MCPR limit	Conforms	15.3 15.5.3.3 15.7

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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
15.2.7	2	Loss of Normal Feedwater Flow	II.2.C (Specific Criteria)	Moderate frequency incident should not progress to more serious condition without independent failures	Conforms	15.3 15.5.3.3 15.7
15.2.7	2	Loss of Normal Feedwater Flow	II.2.D (Specific Criteria)	Use of RG 1.105 with regard to setpoint accuracy	Alternate Approach	Table 1.9-20
15.2.7	2	Loss of Normal Feedwater Flow	II.2.E (Specific Criteria)	Most limiting single failure analysis conforms with RG 1.53	Alternate Approach	Table 1.9-20
15.2.7	2	Loss of Normal Feedwater Flow	II.2.F (Specific Criteria)	Consideration of nonsafety-related systems and single failures	Conforms	15.5.3.3
15.2.7	2	Loss of Normal Feedwater Flow	II.3.A (Analytical Parameters)	Analytical model includes 2 % to account for power measurement uncertainties	Alternate Approach	15.2 15.5.3.3
15.2.7	2	Loss of Normal Feedwater Flow	II.3.B (Analytical Parameters)	Conservative scram characteristics are assumed	Conforms	15.2 15.5.3.3
15.2.7	2	Loss of Normal Feedwater Flow	II.3.C (Analytical Parameters)	Burnup based on most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution	Conforms	15.2 15.3 15.5.3.3

Table 1.9-15 Conformance with NUREG-0800 (Chapter 15 Transient and Accident Analysis)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
15.2.7	2	Loss of Normal Feedwater Flow	II.3.D (Analytical Parameters)	Mitigating system actuation based on setpoint accuracy consistent with RG 1.105	Alternate Approach	Table 1.9-20
15.2.8	2	Feedwater System Pipe Breaks Inside and Outside Containment (PWR)	N/A	Feedwater System Pipe Breaks Inside and Outside Containment	N/A – BWR design	N/A
15.3.1 - 15.3.2	2	Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions	N/A	Loss of Forced Reactor Coolant Flow	N/A – BWRX-300 design does not include recirculation pumps or forced circulation.	N/A
15.3.3 - 15.3.4	3	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	N/A	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	N/A – BWRX-300 design does not include recirculation pumps or forced circulation.	N/A
15.4.1	3	Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition	II.1.A	Conformance with thermal margin limits specified in SRP 4.4	FSAR	FSAR
15.4.1	3	Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition	II.1.B	Fuel centerline temperature limits	N/A – BWR design	N/A
15.4.1	3	Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition	II.1.C	Uniform cladding strain as specified in SRP 4.2 does not exceed 1 %	FSAR	FSAR

Table 1.9-15 Conformance with NUREG-0800 (Chapter 15 Transient and Accident Analysis)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
15.4.2	3	Uncontrolled Control Rod Assembly Withdrawal at Power	II.1.A	Conformance with thermal margin limits specified in SRP 4.4	Alternate Approach-ATLM initiates a control rod block, keeping plant conditions within normal operation conditions, so AOO analysis is not required	15.2.4.3
15.4.2	3	Uncontrolled Control Rod Assembly Withdrawal at Power	II.1.B	Fuel centerline temperature limits	N/A – BWR design	N/A
15.4.2	3	Uncontrolled Control Rod Assembly Withdrawal at Power	II.1.C	Uniform cladding strain as specified in SRP 4.2 does not exceed 1%	Alternate Approach-ATLM initiates a control rod block, keeping plant conditions within normal operation conditions, so AOO analysis is not required	15.2.4.3
15.4.3	3	Control Rod Misoperation (System Malfunction or Operator Error)	II.1.A	Thermal margin limits	N/A - BWR design	N/A
15.4.3	3	Control Rod Misoperation (System Malfunction or Operator Error)	II.1.B	Fuel centerline temperatures and melting point	N/A - BWR design	N/A
15.4.3	3	Control Rod Misoperation (System Malfunction or Operator Error)	II.1.C	Uniform cladding strain does not exceed 1 %	N/A - BWR design	N/A

Table 1.9-15 Conformance with NUREG-0800 (Chapter 15 Transient and Accident Analysis)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
15.4.4 - 15.4.5	2	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate	II.A (Specific Criteria)	Reactor coolant and main steam pressure maintained below 110 % design values	Conforms	15.5.3.4 15.7
15.4.4 - 15.4.5	2	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate	II.B (Specific Criteria)	Fuel cladding integrity in terms of critical power ratio and MCPR limit	Conforms	15.3 15.5.3.4 15.7
15.4.4 - 15.4.5	2	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate	II.C (Specific Criteria)	Moderate frequency incident should not progress to more serious condition without independent failures	Conforms	15.2 15.5.3.4 15.7
15.4.4 - 15.4.5	2	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate	II.D (Specific Criteria)	Use of RG 1.105 with regard to setpoint accuracy	Alternate Approach	Table 1.9-20

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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
15.4.4 - 15.4.5	2	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate	II.E (Specific Criteria)	Most limiting single failure analysis conforms with RG 1.53	Alternate Approach	Table 1.9-20
15.4.4 - 15.4.5	2	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate	II.F (Specific Criteria)	Evaluation of nonsafety-related system performance	Conforms	15.1 15.2 15.5.3.4
15.4.4 - 15.4.5	2	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate	II.1 (Analytical Parameters)	Analytical model includes 2 % to account for power measurement uncertainties	Alternate Approach	15.5.3.4
15.4.4 - 15.4.5	2	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate	II.2 (Analytical Parameters)	Conservative scram characteristics are assumed	Conforms	15.5.3.4.1

Table 1.9-15 Conformance with NUREG-0800 (Chapter 15 Transient and Accident Analysis)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
15.4.4 - 15.4.5	2	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate	II.3 (Analytical Parameters)	Burnup based on most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution	Conforms	15.5.3.4.1
15.4.4 - 15.4.5	2	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate	II.4 (Analytical Parameters)	Mitigating system actuation based on setpoint accuracy consistent with RG 1.105	Alternate Approach	Table 1.9-20
15.4.6	2	Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (PWR)	N/A	Inadvertent Decrease in Boron Concentration in the Reactor Coolant System	N/A – BWR design	N/A
15.4.7	2	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	II.1	Use of reactor instrumentation to identify fuel loading errors after fueling operations	FSAR	FSAR
15.4.7	2	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	II.2	Offsite consequences of fuel rod failure	FSAR	FSAR
15.4.8	3	Spectrum of Rod Ejection Accidents (PWR)	N/A	Spectrum of Rod Ejection Accidents	N/A – BWR design	N/A
15.4.8.A	1	Radiological Consequences of a Control Rod Ejection Accident (PWR)	N/A	Radiological Consequences of a Control Rod Ejection Accident	N/A – BWR design	N/A

Table 1.9-15 Conformance with NUREG-0800 (Chapter 15 Transient and Accident Analysis)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
15.4.9	3	Spectrum of Rod Drop Accidents (BWR)	II.1	Availability of instrumentation	CRDA is excluded by Exemption Request	15.5.5.1
15.4.9	3	Spectrum of Rod Drop Accidents (BWR)	II.2	Effects of postulated accidents do not result in RCPB that exceeds local yielding or impair core cooling	CRDA is excluded by Exemption Request	15.5.5.1
15.4.9	3	Spectrum of Rod Drop Accidents (BWR)	II.3	Calculating radiological consequences per RG 1.183 and RG 1.195	CRDA is excluded by Exemption Request	N/A
15.4.9.A	2	Radiological Consequences of Control Rod Drop Accident (BWR)	II	Compliance with 10 CFR 100, conformance with RG 1.77	N/A - CRDA is excluded by Exemption Request	N/A
15.5.1 - 15.5.2	2	Inadvertent Operation of Emergency Core Cooling System and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	II.1 (Specific Criteria)	Reactor coolant and main steam pressure maintained below 110 % design values	Conforms	15.3 15.5.3.4 15.7
15.5.1 - 15.5.2	2	Inadvertent Operation of Emergency Core Cooling System and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	II.2 (Specific Criteria)	Fuel cladding integrity in terms of critical power ratio and MCPR limit	Conforms	15.3 15.5.3.4 15.7

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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
15.5.1 - 15.5.2	2	Inadvertent Operation of Emergency Core Cooling System and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	II.3 (Specific Criteria)	Abnormal operating occurrence should not progress to more serious condition without independent failures	Conforms	15.2 15.5.3.4
15.5.1 - 15.5.2	2	Inadvertent Operation of Emergency Core Cooling System and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	II.1 (Analytical Parameters)	Analytical model includes 2 % to account for power measurement uncertainties	Alternate Approach	15.5.3.4 15.7
15.5.1 - 15.5.2	2	Inadvertent Operation of Emergency Core Cooling System and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	II.2 (Analytical Parameters)	Conservative scram characteristics are assumed	Conforms	15.5.3.4 15.7
15.5.1 - 15.5.2	2	Inadvertent Operation of Emergency Core Cooling System and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	II.3 (Analytical Parameters)	Burnup based on most limiting combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution	Conforms	15.5.3.4 15.7
15.6.1	2	Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve	N/A	Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve	N/A – BWRX-300 design does not require a pressure relief valve or use a suppression pool.	N/A

Table 1.9-15 Conformance with NUREG-0800 (Chapter 15 Transient and Accident Analysis)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
15.6.2	2	Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	Various	Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	N/A – SRP 15.0.3 applies to new license applications.	N/A
15.6.3	2	Radiological Consequences of Steam Generator Tube Failure	N/A	Radiological Consequences of Steam Generator Tube Failure	N/A – BWRX-300 design does not include steam generators.	N/A
15.6.4	2	Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)	Various	Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)	N/A – SRP 15.0.3 applies to new license applications.	N/A
15.6.5	3	Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	II.1.A	Calculated maximum fuel temperature does not exceed 1200 C (2200 °F)	Conforms	15.3 15.5.4.5 15.7
15.6.5	3	Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	II.1.B	Total local cladding oxidation does not exceed 17%	Conforms	15.3 15.5.4.5 15.7
15.6.5	3	Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	II.1.C	Total hydrogen generation does not exceed 1% of hypothetical amount	Conforms	15.5.4.5

Table 1.9-15 Conformance with NUREG-0800 (Chapter 15 Transient and Accident Analysis)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
15.6.5	3	Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	II.1.D	Core geometry remains coolable	Conforms	15.5.4.5
15.6.5	3	Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	II.1.E	Adequate core cooling for an extended period	Conforms	15.5.4.5
15.6.5	3	Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	II.2	Radiological consequences within 10 CFR 100, 10 CFR 50.67	Conforms	15.5.4.5 15.7
15.6.5	3	Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	II.3	TMI Action Plan requirements	Conforms	15.5.4.5
15.6.5.A	1	Radiological Consequences of a Design Basis Loss-of-Coolant Accident Including Containment Leakage Contribution	Various		N/A – BWRX-300 design because SRP 15.0.3 applies to new license applications.	N/A

Table 1.9-15 Conformance with NUREG-0800 (Chapter 15 Transient and Accident Analysis)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
15.6.5.B	1	Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage from Engineered Safety Feature Components Outside Containment	Various		N/A – BWRX-300 design because SRP 15.0.3 applies to new license applications.	N/A
15.6.5.D	1	Radiological Consequences of a Design Basis Loss-of-Coolant Accident: Leakage from Main Steam Isolation Valve Leakage Control System (BWR)	Various		N/A – BWRX-300 design because SRP 15.0.3 applies to new license applications.	N/A
15.7.3	2	Postulated Radioactive Releases Due to Liquid-Containing Tank Failures	Various		Conforms	11.2
15.7.4	1	Radiological Consequences of Fuel Handling Accidents	Various		N/A – BWRX-300 design because SRP 15.0.3 applies to new license applications.	N/A
15.7.5	2	Spent Fuel Cask Drop Accidents	Various		N/A – BWRX-300 design because SRP 15.0.3 applies to new license applications.	N/A
15.8	2	Anticipated Transients Without Scram	II.1.A		N/A – BWRX-300 design does not include recirculation pumps.	N/A

Table 1.9-15 Conformance with NUREG-0800 (Chapter 15 Transient and Accident Analysis)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
15.8	2	Anticipated Transients Without Scram	II.1.B	An independent and diverse Automatic Rod Insertion from the reactor trip system sensor output to the final actuation device.	Conforms	7.3.2 15.6.1.4.2 15.6.3.2.2
15.8	2	Anticipated Transients Without Scram	II.1.C	Standby Liquid Control System is not applicable to BWRX-300, which does not require a Standby Liquid Control System	Conforms	15.6.1.4.2 15.6.3.2.2
15.8	2	Anticipated Transients Without Scram	II.1.D	Standby Liquid Control System initiation: the Standby Liquid Control System is not applicable to BWRX-300, which does not require a Standby Liquid Control System	Conforms	15.6.1.4.2 15.6.3.2.2
15.8	2	Anticipated Transients Without Scram	II.1.E	ASME Service Level C limits	Conforms	15.6.1.4.2 15.6.3.2.2
15.8	2	Anticipated Transients Without Scram	II.1.F	ATWS/Stability mitigating actions	Conforms	15.6.1.4.2 15.6.3.2.2
15.8	2	Anticipated Transients Without Scram	N/A	-	N/A – BWR design	N/A
15.8	2	Anticipated Transients Without Scram	II.3.A	Diverse scram system consideration for evolutionary plants	Conforms	7.3
15.8	2	Anticipated Transients Without Scram	II.3.B	Alternative to recirculation pump trip for evolutionary plants	N/A – BWR design	N/A
15.8	2	Anticipated Transients Without Scram	II.3.C	ATWS failure probability	Conforms	15.6.1.4.2 15.6.3.2.2

Table 1.9-15 Conformance with NUREG-0800 (Chapter 15 Transient and Accident Analysis)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
15.8	2	Anticipated Transients Without Scram	II.3.C.i	Coolable geometry	Conforms	15.6.1.4.2 15.6.3.2.2
15.8	2	Anticipated Transients Without Scram	II.3.C.ii	Reactor coolant pressure boundary integrity	Conforms	15.6.1.4.2 15.6.3.2.2
15.8	2	Anticipated Transients Without Scram	II.3.C.iii	Containment integrity	Conforms	15.6.1.4.2 15.6.3.2.2
15.9	0	Boiling Water Reactor Stability	II.1	Design to be free of undamped oscillations and thermal-hydraulic instabilities	Conforms	15.5.2 4A
15.9	0	Boiling Water Reactor Stability	II.2	Detection and suppression of oscillations	Conforms	15.5.2 4A
15.9	0	Boiling Water Reactor Stability	II.3	Methodologies provided in NEDO-31960 and its Supplement 1	Conforms	15.5.2 4A
15.9	0	Boiling Water Reactor Stability	II.4	Stability criteria	Conforms	15.5.2 4A
15.9	0	Boiling Water Reactor Stability	II.5	Operating exclusion area	Conforms	15.5.2 4A
15.9	0	Boiling Water Reactor Stability	II.6	Reactor scram before SAFDL violation	Conforms	15.5.2 4A
15.9	0	Boiling Water Reactor Stability	II.7	Stability modeling	Conforms	15.5.2 4A
15.9	0	Boiling Water Reactor Stability	II.8	Backup options; Technical Specification requirements	Conforms	15.5.2 4A

Table 1.9-15 Conformance with NUREG-0800 (Chapter 15 Transient and Accident Analysis)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
15.9	0	Boiling Water Reactor Stability	II.9	New stability solutions	Conforms	15.5.2 4A
15.9	0	Boiling Water Reactor Stability	II.10	Stability related instrumentation functionality	Conforms	15.5.2 4A
15.9	0	Boiling Water Reactor Stability	II.11	Instability modes other than density wave instability	Conforms	15.5.2 4A

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

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
16.0	3	Technical Specifications	II	Compliance with 10 CFR 50.34	Conforms	16
16.1	1	Risk-informed Decision Making: Technical Specifications	II.1	Traditional Engineering Guidelines – Modification of individual plant's technical specifications.	N/A, CRN-1 will not use Risk-informed Decision Making	N/A
16.1	1	Risk-informed Decision Making: Technical Specifications	II.2	Probabilistic Guidelines – Modification of individual plant's technical specifications.	N/A, CRN-1 will not use Risk-informed Decision Making	N/A

Table 1.9-17 Conformance with NUREG-0800 (Chapter 17 Quality Assurance)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
17.1	2	Quality Assurance During the Design and Construction Phases	Various	10 CFR Part 50 Appendix B is applicable to the BWRX-300 design. Conforms to RG 1.28, Revision 5.	Conforms	17.1
17.2	2	Quality Assurance During the Operations Phase	Various	10 CFR Part 50 Appendix B is applicable to the BWRX-300 design. Details of conformance to be provided in the OL application.	FSAR	FSAR
17.3	0	Quality Assurance Program Description	Various	See Section 17.5.	Conforms	17.3 17.5
17.4	1	Reliability Assurance Program (RAP)	II.A.A.1	Description	Note 1	17.4
17.4	1	Reliability Assurance Program (RAP)	II.A.A.2	Programmatic controls	Note 1	17.4
17.4	1	Reliability Assurance Program (RAP)	II.A.A.3	Methodology for identifying SSCs within scope	Note 1	17.4
17.4	1	Reliability Assurance Program (RAP)	II.A.A.4	Expert panel	Note 1	17.4
17.4	1	Reliability Assurance Program (RAP)	II.A.A.5	Listing of SSCs within scope	FSAR	FSAR
17.4	1	Reliability Assurance Program (RAP)	II.A.A.6	Determination of dominant failure modes	FSAR	FSAR
17.4	1	Reliability Assurance Program (RAP)	II.A.A.7	Quality assurance controls	FSAR	FSAR
17.4	1	Reliability Assurance Program (RAP)	II.A.A.8	ITAAC	N/A – 10 CFR Part 50 CPA	N/A

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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
17.4	1	Reliability Assurance Program (RAP)	II.A.A.9	Combined License items	N/A – 10 CFR Part 50 CPA	N/A
17.4	1	Reliability Assurance Program (RAP)	II.B.B.1	Plant-specific reliability assurance program	FSAR	FSAR
17.4	1	Reliability Assurance Program (RAP)	II.B.B.2	Quality assurance associated with design and construction activities	FSAR	FSAR
17.4	1	Reliability Assurance Program (RAP)	II.B.B.3	Integration of RAP into operational programs	FSAR	FSAR
17.4	1	Reliability Assurance Program (RAP)	II.B.B.4	ITAAC	N/A – 10 CFR Part 50 CPA	N/A
17.5	1	Quality Assurance Program Description – Design Certification, Early Site Permit and New License Applicants	II.A	Organization	Conforms	17.5 13
17.5	1	Quality Assurance Program Description – Design Certification, Early Site Permit and New License Applicants	II.B	Quality assurance program	Conforms	17.5
17.5	1	Quality Assurance Program Description – Design Certification, Early Site Permit and New License Applicants	II.C	Design control and verification	Conforms	17.5
17.5	1	Quality Assurance Program Description – Design Certification, Early Site Permit and New License Applicants	II.D	Procurement document control	Conforms	17.5

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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
17.5	1	Quality Assurance Program Description – Design Certification, Early Site Permit and New License Applicants	II.E	Instructions, procedures, and drawings	Conforms	17.5
17.5	1	Quality Assurance Program Description – Design Certification, Early Site Permit and New License Applicants	II.F	Document control	Conforms	17.5
17.5	1	Quality Assurance Program Description – Design Certification, Early Site Permit and New License Applicants	II.G	Control of purchased material, equipment, and services	Conforms	17.5
17.5	1	Quality Assurance Program Description – Design Certification, Early Site Permit and New License Applicants	II.H	Identification and control of materials, parts, and components	Conforms	17.5
17.5	1	Quality Assurance Program Description – Design Certification, Early Site Permit and New License Applicants	II.I	Control of special processes	Conforms	17.5
17.5	1	Quality Assurance Program Description – Design Certification, Early Site Permit and New License Applicants	II.J	Inspection	Conforms	17.5

Table 1.9-17 Conformance with NUREG-0800 (Chapter 17 Quality Assurance)
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SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
17.5	1	Quality Assurance Program Description – Design Certification, Early Site Permit and New License Applicants	II.K	Test control	Conforms	17.5
17.5	1	Quality Assurance Program Description – Design Certification, Early Site Permit and New License Applicants	II.L	Control of measuring and test equipment	Conforms	17.5
17.5	1	Quality Assurance Program Description – Design Certification, Early Site Permit and New License Applicants	II.M	Handling, storage, and shipping	Conforms	17.5
17.5	1	Quality Assurance Program Description – Design Certification, Early Site Permit and New License Applicants	II.N	Inspection, test, and operating status	Conforms	17.5
17.5	1	Quality Assurance Program Description – Design Certification, Early Site Permit and New License Applicants	II.O	Nonconforming materials, parts, or components	Conforms	17.5
17.5	1	Quality Assurance Program Description – Design Certification, Early Site Permit and New License Applicants	II.P	Corrective action	Conforms	17.5

Table 1.9-17 Conformance with NUREG-0800 (Chapter 17 Quality Assurance)
(Sheet 5 of 6)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
17.5	1	Quality Assurance Program Description – Design Certification, Early Site Permit and New License Applicants	II.Q	Quality assurance records	Conforms	17.5
17.5	1	Quality Assurance Program Description – Design Certification, Early Site Permit and New License Applicants	II.R	Audits	Conforms	17.5
17.5	1	Quality Assurance Program Description – Design Certification, Early Site Permit and New License Applicants	II.S	Training and qualification criteria	Conforms	17.5
17.5	1	Quality Assurance Program Description – Design Certification, Early Site Permit and New License Applicants	II.T	Training and qualification – inspection and test	Conforms	17.5
17.5	1	Quality Assurance Program Description – Design Certification, Early Site Permit and New License Applicants	II.U	Nonsafety-related SSC quality controls	Conforms	17.5
17.5	1	Quality Assurance Program Description – Design Certification, Early Site Permit and New License Applicants	II.V	Quality assurance program commitments	Conforms	17.5
17.6	2	Maintenance Rule	1	Conformance with RG 1.160 (NUMARC 93-01 endorsement)	FSAR	FSAR

Table 1.9-17 Conformance with NUREG-0800 (Chapter 17 Quality Assurance)
(Sheet 6 of 6)

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
17.6	2	Maintenance Rule	2	Categorization of SSCs within RAP scope	FSAR	FSAR
17.6	2	Maintenance Rule	3	Operational programs and implementation milestones	FSAR	FSAR

Note 1: Regulatory Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, Rev. 3, does not require a description of the Reliability Assurance Program in the PSAR. 10 CFR 50.34, Contents of applications; technical information, does not require a description of the Reliability Assurance Program. There is no specific requirement or associated guidance for inclusion of a RAP in a 10 CFR Part 50 construction permit. However, since TVA intends to comply with SRP 17.4 for FSAR, information pertinent to how the program is being established and implemented is being included in the PSAR. Programmatic information, including PRA risk-significant criteria, are included so that USNRC staff understand how the D-RAP is being developed. Additional information, including the list of D-RAP SSCs, will be included in the FSAR.

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

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
18.0	3	Human Factors Engineering			FSAR	FSAR

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

SRP Section	Rev	Title	Criterion	Topic	Conformance	PSAR Section
19.0	3	Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors			FSAR	FSAR
19.1	3	Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities			FSAR	FSAR
19.2		Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance			FSAR	FSAR
19.3	0	Regulatory Treatment of Non-Safety Systems (RTNSS) for Passive Advanced Light Water Reactors			FSAR	FSAR
19.4	0	Strategies and Guidance to Address Loss-of-Large Areas of the Plant Due to Explosions and Fires			FSAR	FSAR
19.5	0	Adequacy of Design features and functional capabilities identified and described for withstanding Aircraft Impacts			FSAR	FSAR

Table 1.9-20 Conformance with Regulatory Guides
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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
	Power Reactors (Division 1)				
1.6	Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems	0	Conforms		8.1.3 8.3.2
1.7	Control of Combustible Gas Concentrations in Containment	3	Conforms		6.2.1 6.8.3 9.3.6
1.8	Qualification and Training of Personnel for Nuclear Power Plants	4	Conforms		12.5.3 13.1 17.5 12.1.1
			Alternative Approach	Licensed Operations Training will be based on NUREG 1021. Non-Licensed Training will based on RG 1.8.	13.2
1.9	Application and Testing of Safety-Related Diesel Generators in NPPs	4	N/A	BWRX-300 design does not include SC1 EDGs.	N/A
1.11	Instrument Lines Penetrating the Primary Reactor Containment	1	Conforms		3.1.5 6.2.4
1.12	NPP Instrumentation or Earthquakes	3	Conforms		3.7.4
1.13	Spent Fuel Storage Facility Design Basis	2	Conforms	C.1, C.2, C.3	3.5
		2	Alternate Approach	Fuel storage and handling facilities are designed to comply with GDC 61.	9.1.2
1.14	Reactor Coolant Pump Flywheel Integrity	1	N/A	BWRX-300 design does not include reactor coolant pumps.	N/A

Table 1.9-20 Conformance with Regulatory Guides
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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
1.20	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing	4	Conforms		3.9.2 14.1.3
1.21	Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste	3	Conforms		9.3.2 11.5.3
1.22	Periodic Testing of Protection System Actuation Functions	0	Alternate Approach	Using IEC 60671:2007 and IEC 62385:2007	Table 7.1-3
1.23	Meteorological Monitoring Programs for Nuclear Power Plants	1	Conforms		2.3.2 2.3.3 2.3.4 2.3.5 15.5.9
1.24	Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure	0	N/A	BWR design	N/A

Table 1.9-20 Conformance with Regulatory Guides
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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
1.26	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants	6	Conforms		3.2.4 3.8.4 4.5.2 4.5.4 5.2.1 5.4.8 6.3.3 6.6.1 17.5
1.27	Ultimate Heat Sink for Nuclear Power Plants	3	Conforms	Position C.4	2.3.1
			N/A	As indicated in RG 1.27, Rev. 3, the guidance provided therein does not apply for those designs.	N/A
1.28	Quality Assurance Program Criteria (Design and Construction)	5	Conforms		3.1.1 3.8.1 3.8.2 3.8.3 3.8.4 3.8.5 3.13.1 4.5.1 6.1.4 17.1 17.5

Table 1.9-20 Conformance with Regulatory Guides
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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
1.29	Seismic Design Classification for Nuclear Power Plants	6	Alternate Approach	C.1 Seismic categorization of components is based on the importance of functions performed. Safety Class 1 (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. See Table 3A-1 for the resulting seismic categorization.	3.2.3 5.2.5 5.4.7 6.3.3 9.1.2 9.1.4 9.1.5 17.5
1.30	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment	1	FSAR		FSAR
1.31	Control of Ferrite Content in Stainless Steel Weld Metal	4	Conforms		4.5.3 4.5.4 5.2.3 5.3.1 6.1.1 6.1.4
1.32	Criteria for Power Systems for Nuclear Power Plants	4	Conforms (Note - testing requirements addressed in FSAR)		8.1.3 14.1.3

Table 1.9-20 Conformance with Regulatory Guides
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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
1.33	Quality Assurance Program Requirements (Operation)	3	Conforms	Conformance of Section 17.5 will apply to the FSAR	5.4.7 13.1 13.5 17.5
1.34	Control of Electroslag Weld Properties	1	N/A	The design does not use electroslag welding except for RPV cladding.	5.2.3 5.3.1
1.35.1	Determining Prestressing Forces for Inspection of Prestressed Concrete Containments	0	N/A	BWRX-300 design does not include prestressed concrete.	N/A
1.36	Nonmetallic Thermal Insulation for Austenitic Stainless Steel	1	Conforms		5.2.3 6.1.4
1.40	Qualification of Continuous Duty Safety-Related Motors for Nuclear Power Plants	1	Conforms		3.11.2
1.43	Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components	1	N/A	The requirements of RG 1.43 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).	5.2.3 5.3.1

Table 1.9-20 Conformance with Regulatory Guides
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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
1.44	Control of the Processing and Use of Stainless Steel	1	Conforms		4.5.1 4.5.3 4.5.4 5.2.3 5.3.1 6.1.1 6.1.4
1.45	Guidance on Monitoring and Responding to Reactor Coolant System Leakage	1	Alternate Approach	An acceptable alternative to RG 1.45 General Positions 1.1 and 1.2 is to monitor only unidentified leakage without leakage separation.	5.2.5 6.3.3
			Conforms		9.3.3
1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems	1	Alternate Approach	Using IEEE Std 603-2018	6.3.3 Table 7.1-3 7.3 8.1.3 8.3.2
1.50	Control of Preheat Temperature for Welding of Low-Alloy Steel	1	Conforms		5.2.3 5.3.1 6.1.4
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	4	N/A	The BWRX-300 does not utilize an engineered safety feature ventilation system.	N/A

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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
1.53	Application of the Single-Failure Criterion to Safety Systems	2	Alternate Approach	Using IEEE Std 379-2014 and IEEE Std 603-2018	6.3.3 Table 7.1-3 8.1.3
1.54	Service Level I, II, III, and In-Scope License Renewal Protective Coatings Applied to Nuclear Power Plants	3	Conforms		3.8.1 3.8.2 6.1.2 6.1.4 17.5
1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components	2	Alternate Approach	The design conforms to RG 1.57 for Class MC components. The design limits, load combinations and leak tightness of the Class MC components backed by concrete are per the alternative design approach discussed in Sections 4.0 and 6.0 of NEDC-33926P-A.	3.8.2
			Conforms		3.13.1
1.59	Design Basis Floods for Nuclear Power Plants	2	Conforms	The site grade is above the maximum flood height (the site is considered to be dry).	2.4.0 2.4.2 2.4.3
1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants	2	Alternate Approach	Site-specific vertical Ground Motion Response Spectra (GMRS) was developed using the guidance in RG 1.208.	2.5.2 3.7.1

Table 1.9-20 Conformance with Regulatory Guides
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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
1.61	Damping Values for Seismic Design of Nuclear Power Plants	2	Conforms		3.7.1 3.7.2 3.7.3 3C
1.62	Manual Initiation of Protective Actions	1	Alternate Approach	Using IEEE Std 603-2018	Table 7.1-3
1.63	Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants	3	Alternate Approach	Using IEEE Std 317-2013	3.11.2 8.1.3 8.3.1
1.65	Materials and Inspections for Reactor Vessel Closure Studs	1	Conforms		3.13.1 5.3.1
1.68	Initial Test Programs for Water-Cooled Nuclear Power Plants	4	Conforms		14.1.1 14.1.3
1.68.1	Initial Test Program of Condensate and Feedwater Systems for Light-Water Reactors	2	FSAR		FSAR
1.68.2	Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants	2	FSAR		FSAR
1.68.3	Preoperational Testing of Instrument and Control Air Systems	1	FSAR		FSAR
1.69	Concrete Radiation Shields and Generic Shield Testing for Nuclear Power Plants	1	Conforms		3.8.1 3.8.3 12.3.2

Table 1.9-20 Conformance with Regulatory Guides
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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
1.70	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)	3	Conforms		1.1.1
1.71	Welder Qualification for Areas of Limited Accessibility	1	Conforms		5.2.3
1.72	Spray Pond Piping Made from Fiberglass-Reinforced Thermosetting Resin	2	N/A	BWRX-300 design does not have a spray pond.	N/A
1.73	Qualification Tests for Safety-Related Actuators in Nuclear Power Plants	2	Conforms		3.9.3 3.11.2
1.75	Criteria for Independence of Electrical Safety Systems	3	Alternate Approach	Using IEC 60709:2018 for independence and separation and IEC 62808:2018 for isolation qualification	Table 7.1-3 7.3
			Alternate Approach	Using IEEE Std 384-2018	3.10.3 3.11.2 8.1.3 8.3.1 8.3.2
1.76	Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants	1	Conforms		2.3.1 3.3 3.5 15.6.1

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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
1.78	Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release	2	Alternate Approach	Events that may result in impairment of the MCR and CB HVS are not evaluated concurrently with a postulated hazardous chemical release from a mobile source. Alternate approach to C3.2, C4.1 (detection system Seismic Category) and C4.2 (electrical support system Safety Category).	2.2.1 2.2.2 2.2.3 2.3.1 9.4.1 9.4.6
1.79	Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors	2	N/A	BWR design	N/A
1.79.1	Initial Test Program of Emergency Core Cooling Systems for New Boiling-Water Reactors	0	FSAR		FSAR
1.81	Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants	1	N/A	N/A – CRN-1 application is for a single unit.	N/A
1.82	Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident	5	N/A	BWRX-300 ECCS design does not recirculate water from the Containment sump.	N/A

Table 1.9-20 Conformance with Regulatory Guides
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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
1.84	Design, Fabrication, and Materials Code Case Acceptability, ASME Section III	40	Conforms		3.13 4.5.1 4.5.3 4.5.4 5.2.1 5.2.3 6.1.4
1.87	Guidance for Construction of Class 1 Components in Elevated-Temperature Reactors	2	N/A	Applicable to non-LWR designs	N/A
1.89	Equipment Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants	2	Conforms		3.11 Table 7.1-3 8.1.3
1.90	Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons	2	N/A	BWRX-300 design does not include prestressed concrete.	N/A
1.91	Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants	3	Conforms		2.2.1 2.2.3
1.92	Combining Modal Responses and Spatial Components in Seismic Response Analysis	3	Conforms		3.7.2 3.7.3 3.12.1 3.12.3 5.4.13
1.93	Availability of Electric Power Sources	1	FSAR		FSAR

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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
1.96	Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants	1	N/A	The BWRX-300 does not include a main steam isolation valve leakage control system.	N/A
1.97	Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants	5	Alternate Approach	Using IEC 63147:2017/IEEE 497-2016	3.11.1 6.8 Table 7.1-3 11.5.3 12.3.4
1.98	Assumptions Used for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling Water Reactor	0	FSAR		FSAR
1.99	Radiation Embrittlement of Reactor Vessel Materials	2	Conforms		5.3.1 5.3.2 5.3.3
1.100	Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants	4	Alternate Approach	Using IEC/IEEE 60980-344:2020	Table 7.1-3 7.3.1

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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
1.100	Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants	4	Conforms	The BWRX-300 Standard Design complies with ASME QME-1-2023, including exceptions and clarifications in RG 1.100. In cases where this standard is used to implement RG 1.100, ASME QME-1-2017 will be utilized. The BWRX-300 Standard Design complies with IEC/IEEE 60980-344-2020. However, in cases where this standard is used to implement RG 1.100, IEEE 344-2013 will be utilized.	3.10 5.4.13 6.3.3
1.101	Emergency Response Planning and Preparedness for Nuclear Power Reactors	6	Conforms		13.3 13A
1.102	Flood Protection for Nuclear Power Plants	1	N/A	The CRN Site is a “dry” site.	N/A
1.105	Setpoints for Safety-Related Instrumentation	4	Alternate Approach	Using IEC 61888:2002	Table 7.1-3
1.106	Thermal Overload Protection for Electric Motors on Motor-Operated Valves	2	N/A	BWRX-300 design does not include SC1 electric motors or motor-operated valves.	N/A
1.107	Qualification for Cement Grouting for Prestressing Tendons in Containment Structures	2	N/A	BWRX-300 design does not include prestressed concrete.	N/A
1.109	Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I	1	Conforms		2.4.13

Table 1.9-20 Conformance with Regulatory Guides
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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
1.110	Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors	1	FSAR		FSAR
1.111	Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors	1	Conforms		2.3.4 2.3.5
1.112	Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Nuclear Power Reactors	1	Conforms		11.2.4 11.3.4
1.113	Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I	1	FSAR		FSAR
1.114	Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit	3	FSAR		FSAR
1.115	Protection Against Turbine Missiles	2	Alternate Approach	The RG 1.115 hazard rate is applicable to an 1800 rpm turbine and will be modified for a 3600 rpm turbine speed.	3.5 10.2.4
1.117	Protection Against Extreme Wind Events and Missiles for Nuclear Power Plants	2	Conforms		3.5 3.8.4 9.1.2

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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
1.118	Periodic Testing of Electric Power and Protection Systems	3	Alternate Approach	Using IEC 60671:2007 and IEC 62385:2007	Table 7.1-3
			Alternate Approach	Using IEEE 338-2022	8.1.3 8.3.2
1.121	Bases for Plugging Degraded PWR Steam Generator Tubes	0	N/A	BWR design	N/A
1.122	Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components	1	Conforms		3.7.2 3.7.3
1.124	Service Limits and Loading Combinations for Class 1 Linear-Type Supports	3	FSAR		FSAR
1.125	Physical Models for Design and Operation of Hydraulic Structures and Systems for NPPs	2	N/A	CRN Site does not include cooling water channels or reservoirs.	N/A
1.126	An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification	2	FSAR		FSAR
1.127	Inspection of Water-Control Structures Associated with Nuclear Power Plants	2	N/A	CRN Site does not include Seismic Category I water control structures.	N/A
1.128	Installation Design and Installation of Vented Lead-Acid Storage Batteries for Nuclear Power Plants	3	Conforms		8.1.3 8.3.2
1.129	Maintenance, Testing, and Replacement of Vented Lead-Acid Storage Batteries for Nuclear Power Plants	4	Conforms		8.1.3 8.3.2
1.130	Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Supports	3	FSAR		FSAR

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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
1.132	Geologic and Geotechnical Site Characterization Investigations for Nuclear Power Plants	3	Conforms		2.5.4
1.133	Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors	1	N/A	The BWRX-300 does not include a loose parts monitoring system.	N/A
1.134	Medical Assessment of Licensed Operators or Applicants for Operator Licenses at Nuclear Power Plants	4	FSAR		FSAR
1.136	Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments	4	Conforms		3.8.1 3.8.5 6.1.4
1.137	Fuel-Oil Systems for Emergency Power Supplies	2	N/A	Standby Diesel Generator Fuel Oil Storage and Transfer System is Non-Safety Class (SCN).	N/A
1.138	Laboratory Investigations of Soils and Rocks for Engineering Analysis and Design of Nuclear Power Plants	3	Alternate Approach	ASTM D7012-10 was used for testing related to unconfined compression, as ASTM D2938 was withdrawn and replaced by ASTM D7012.	2.5.4
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	3	Alternate Approach	The CB HVS follows the guidance of RG 1.140 Criteria C.2 and C.3.	9.4.1 14.1.3
1.141	Containment Isolation Provisions for Fluid Systems	1	Conforms		6.2.4

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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
1.142	Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)	3	Conforms		3.5.3 3.8.4
1.143	Design Guidance for Radioactive Waste Management Structures, Systems, and Components Installed in Light-Water-Cooled Nuclear Power Plants	2	Conforms		3.1.1 3.2.3 3.3 3.5 3.7 3.8.4 Table 3.12-4 9.4.1 9.4.3 9.4.4 11.2 11.3 11.4
1.145	Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants	1	Conforms		2.3.4 15.5.1
1.147	Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1	21	Conforms		5.2.1 6.6
1.149	Nuclear Power Plant Simulation Facilities for Use in Operator Training, License Examinations, and Applicant Experience Requirements	4	FSAR		FSAR

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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
1.151	Instrument Sensing Lines	2	Conforms		3.9.1
1.152	Criteria for Use of Computers in Safety Systems of Nuclear Power Plants	4	Alternate Approach	Using IEC 61513:2011, IEC 60880:2006, IEC 62566:2012, and IEC 62138:2018 for system and software development Using IEC 62645:2019 and IEC 62859:2016 for secure development and operating environment Using IEC 61500:2018 for communication independence	Table 7.1-3 7.3.3 Table 7.4-1
1.153	Criteria for Safety Systems	1	Alternate Approach	Using IEEE Std 603-2018 or IEEE 603-1991 when specified by regulation or regulatory guidance	7.1.3 7.3 7.4.5 8.1.3 8.3.2
1.155	Station Blackout	0	Conforms		6.2.4 6.3.3 8.1.3
1.156	Qualification of Connection Assemblies for Nuclear Power Plants	2	Conforms		3.10.3 3.11.2
1.157	Best-Estimate Calculations of Emergency Core Cooling System Performance	0	Alternate Approach	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.	6.3.3

Table 1.9-20 Conformance with Regulatory Guides
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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
1.158	Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants	1	Alternate Approach	Using IEEE 535-2022	3.11.2 8.3.2
1.159	Assuring the Availability of Funds for Decommissioning Nuclear Reactors	2	N/A	N/A to a PSAR	N/A
1.160	Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	4	FSAR		FSAR
1.161	Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 Ft-Lb	0	N/A	Option provided by this RG is not used.	N/A
1.162	Format and Content of Report for Thermal Annealing of Reactor Pressure Vessels	0	N/A	N/A to a PSAR	N/A
1.163	Performance-Based Containment Leak-Test Program	1	Conforms		6.2.6
1.164	Dedication of Commercial-Grade Items for Use in Nuclear Power Plants	1	Conforms		17.5
1.166	Pre-Earthquake Planning, Shutdown and Restart of a Nuclear Power Plant Following an Earthquake	1	Conforms		3.7.4
1.168	Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	2	Alternate Approach	Using IEC 61513:2011, IEC 60880:2006, IEC 62566:2012, and IEC 62138:2018 for system and software development Using IEC 62645:2019 and IEC 62859:2016 for secure development and operating environment Using IEC 61500:2018 for communication independence	Table 7.1-3 7.3.3 Table 7.4-1

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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
1.169	Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	1	Alternate Approach	Using IEC 61513:2011, IEC 60880:2006, IEC 62566:2012, and IEC 62138:2018 for system and software development Using IEC 62645:2019 and IEC 62859:2016 for secure development and operating environment Using IEC 61500:2018 for communication independence	Table 7.1-3 7.3.3 Table 7.4-1
1.170	Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	1	Alternate Approach	Using IEC 61513:2011, IEC 60880:2006, IEC 62566:2012, and IEC 62138:2018 for system and software development Using IEC 62645:2019 and IEC 62859:2016 for secure development and operating environment Using IEC 61500:2018 for communication independence	Table 7.1-3 7.3.3 Table 7.4-1
1.171	Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	1	Alternate Approach	Using IEC 61513:2011, IEC 60880:2006, IEC 62566:2012, and IEC 62138:2018 for system and software development Using IEC 62645:2019 and IEC 62859:2016 for secure development and operating environment Using IEC 61500:2018 for communication independence	Table 7.1-3 7.3.3 Table 7.4-1

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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
1.172	Software Requirements Specifications for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	1	Alternate Approach	Using IEC 61513:2011, IEC 60880:2006, IEC 62566:2012, and IEC 62138:2018 for system and software development Using IEC 62645:2019 and IEC 62859:2016 for secure development and operating environment Using IEC 61500:2018 for communication independence	Table 7.1-3 7.3.3 Table 7.4-1
1.173	Developing Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	1	Alternate Approach	Using IEC 61513:2011, IEC 60880:2006, IEC 62566:2012, and IEC 62138:2018 for system and software development Using IEC 62645:2019 and IEC 62859:2016 for secure development and operating environment Using IEC 61500:2018 for communication independence	Table 7.1-3 7.3.3 Table 7.4-1
1.174	An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis	3	FSAR		FSAR
1.175	Plant-Specific, Risk-Informed Decision making: Inservice Testing	1	FSAR		FSAR
1.177	Plant-Specific, Risk-Informed Decision making: Technical Specifications	2	FSAR		FSAR

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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
1.178	An Approach for Plant-Specific Risk-Informed Decision making for Inservice Inspection of Piping	2	FSAR		FSAR
1.179	Standard Format and Content of License Termination Plans for Nuclear Power Reactors	2	FSAR		FSAR
1.180	Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems	2	Alternate Approach	Using IEC Standards	3.11 Table 7.1-3 7.3.2 7.3.3 8.3.2
1.181	Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e)	0	FSAR		FSAR
1.183	Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors	1	Alternate Approach	The BWRX-300 safety analysis shows that core damage does not occur during any design basis event. Thus, RG 1.183 guidance is partially applicable. Using RG 1.183 Revision 0 for DBA and Revision 1 for maximum hypothetical accident (MHA) in Appendix 15C.	6.4.3 15.0 15.5.8 15.5.9 15C
1.184	Decommissioning of Nuclear Power Reactors	1	FSAR		FSAR
1.185	Standard Format and Content for Post-Shutdown Decommissioning Activities Report	1	FSAR		FSAR

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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
1.186	Guidance and Examples for Identifying 10 CFR 50.2 Design Bases	0	FSAR		FSAR
1.187	Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments	3	FSAR		FSAR
1.188	Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses	2	FSAR		FSAR
1.189	Fire Protection for Nuclear Power Plants	5	Alternate Approach	Using Revision 4 of RG 1.189 Exception to Position 3.2.1(j) regarding seismic design of the firewater supply to the RB standpipe	3.1.2 9.5.1 9A 17.5
1.190	Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence	0	Conforms		4.1.6
1.191	Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown	1	FSAR		FSAR
1.192	Operation and Maintenance Code Case Acceptability, ASME OM Code	5	Conforms		3.9.6 5.2.1 6.6
1.193	ASME Code Cases Not Approved for Use	8	Conforms		5.2.1
1.194	Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants	0	Conforms		15.5.1 15C

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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
1.195	Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors	0	N/A	N/A - Applies to 10 CFR 52 applications	N/A
1.196	Control Room Habitability at Light-Water Nuclear Power Reactors	1	Conforms		6.4.3
1.197	Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors	0	FSAR		FSAR
1.198	Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites	0	Conforms		2.5.4
1.199	Anchoring Components and Structural Supports in Concrete	1	Conforms		3.8.4
1.200	Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities	3	Conforms		15.6.1
1.202	Standard Format and Content of Decommissioning Cost Estimates for Nuclear Power Reactors	0	FSAR		FSAR
1.203	Transient and Accident Analysis Methods	0	Conforms		6.3.3 15.5.1
1.204	Guidelines for Lightning Protection of Nuclear Power Plants	1	Conforms		8.1.3 8.3.1 8.3.2

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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
1.205	Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants	2	N/A	10 CFR Part 50 CPA	N/A
1.206	Applications for Nuclear Power Plants	1	N/A	10 CFR Part 50 CPA	N/A
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	1	Conforms		3.8.2 3.9.3 3.12.3
1.208	A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion	0	Alternate Approach	Using ASCE/SEI 43-19	2.5 3.7 3.8
1.209	Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants	0	Alternate Approach	Using IEC/IEEE 60780-323:2016	Table 7.1-3
1.210	Qualification of Safety-Related Battery Chargers and Inverters for Nuclear Power Plants	1	Alternate Approach	Using IEEE 650-2006	3.11.2
1.211	Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants	0	Alternate Approach	Using IEEE 383-2023	8.1.3
1.212	Sizing of Large Lead-Acid Storage Batteries	2	Conforms		8.3.2
1.213	Qualification of Safety-Related Motor Control Centers for Nuclear Power Plants	0	Conforms		3.11.2
1.214	Response Strategies for Potential Aircraft Threats (OUO-SRI)	1	Conforms		3N

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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
1.215	Guidance for ITAAC Closure Under 10 CFR Part 52	2	N/A	10 CFR Part 50 CPA	N/A
1.216	Containment Structural Integrity Evaluation for Internal Pressure Loadings Above Design-Basis Pressure	0	Conforms		3.8.1 3.8.2
1.217	Guidance for the Assessment of Beyond-Design-Basis Aircraft Impacts	0	Conforms		3N
1.218	Condition-Monitoring Techniques for Electric Cables Used in Nuclear Power Plants	0	FSAR		FSAR
1.219	Guidance on Making Changes to Emergency Plans for Nuclear Power Reactors	1	FSAR		FSAR
1.221	Design-Basis Hurricane and Hurricane Missiles for Nuclear Power Plants	0	Conforms		2.3 3.3 3.5
1.226	Flexible Mitigation Strategies for Beyond-Design-Basis Events	0	FSAR		FSAR
1.227	Wide-Range Spent Fuel Pool Level Instrumentation	0	Alternate Approach	Using IEC 63147/IEEE 497:2017 for parameter display	Table 7.1-3
			Conforms		9.1.2 9.1.3
1.230	Regulatory Guidance on the Alternate Pressurized Thermal Shock Rule	0	N/A	BWR design	N/A

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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
1.231	Acceptance of Commercial-Grade Design and Analysis Computer Programs Used in Safety-Related Applications for Nuclear Power Plants	0	Conforms		17.5
1.232	Developing Principal Design Criteria for Non-Light Water Reactors	0	N/A	RG is used to inform PDCs	3.1
1.233	Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light Water Reactors	0	N/A	BWR design	N/A
1.234	Evaluating Deviations and Reporting Defects and Noncompliance Under 10 CFR Part 21	1	Conforms		17.5
1.236	Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drive Accidents	0	N/A	N/A - CRDA is excluded by Exemption Request	N/A
1.237	Guidance for Changes During Construction for New Nuclear Power Plants Being Constructed Under a Combined License Referencing a Certified Design Under 10 CFR Part 52	0	N/A	10 CFR Part 50 CPA	N/A
1.239	Licensee Actions to Address Nonconservative Technical Specifications	0	FSAR		FSAR
1.240	Fresh and Spent Fuel Pool Criticality Analysis	0	Conforms		9.1.A

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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
1.242	Performance-Based Emergency Preparedness for Small Modular Reactors, Non-Light-Water Reactors, and Non-Power Production or Utilization Facilities	0	FSAR		FSAR
1.243	Safety-Related Steel Structures and Steel-Plate Composite Walls for other than Reactor Vessels and Containments	0	Conforms		3.5 3.8.3 3.8.4 3.8.5 3H
1.244	Control of Heavy Loads at Nuclear Facilities	0	Conforms		9.1.5
1.245	Preparing Probabilistic Fracture Mechanics Submittals	0	FSAR		FSAR
1.246	Acceptability of ASME Code, Section XI, Division 2, Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants, for Non-Light Water Reactors	0	N/A	BWR design	N/A
1.248	Guide for Assessing, Monitoring, and Mitigating Aging Effects on Electrical Equipment Used in Production and Utilization Facilities	0	FSAR		FSAR
1.249	Use of ARCON Methodology For Calculation Of Accident-Related Offsite Atmospheric Dispersion Factors	0	Conforms		15C
1.250	Dedication of Commercial-Grade Digital Instrumentation and Control Items for Use in Nuclear Power Plants	0	FSAR		FSAR

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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
1.253	Guidance for a Technology-Inclusive Content of Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors	0	N/A	BWR design	N/A
	Environmental and Siting (Division 4)				
4.1	Radiological Environmental Monitoring for Nuclear Power Plants	2	FSAR		FSAR
4.2	Preparation of Environmental Reports for Nuclear Power Stations	3	Conforms	See CPA ER	N/A
4.2S1	Supplement 1 to RG 4.2, Preparation of Environmental Reports for Nuclear Power Plants License Renewal Applications	2	N/A	10 CFR Part 50 CPA	N/A
4.7	General Site Suitability Criteria for Nuclear Power Stations	4	Conforms		2.1.3 2.2.1 2.2.3 2.3.4 2.3.5
4.7	General Site Suitability Criteria for Nuclear Power Stations	3	N/A		N/A
4.9	Preparation of Environmental Reports for Commercial Uranium Enrichment Facilities	1	N/A	10 CFR Part 50 CPA	N/A
4.11	Terrestrial Environmental Studies for Nuclear Power Stations	2	Conforms	See CPA ER	N/A

Table 1.9-20 Conformance with Regulatory Guides
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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
4.13	Performance, Testing, and Procedural Specifications for Thermoluminescence Dosimetry: Environmental Applications	2	FSAR		FSAR
4.14	Radiological Effluent and Environmental Monitoring at Uranium Mills	1	N/A	10 CFR Part 50 CPA	N/A
4.15	Quality Assurance for Radiological Monitoring Programs (Inception through Normal Operations to License Termination) – Effluent Streams and the Environment	2	FSAR		FSAR
4.16	Monitoring and Reporting Radioactive Materials in Liquid and Gaseous Effluents from Nuclear Fuel Cycle Facilities	2	N/A	10 CFR Part 50 CPA	N/A
4.17	Standard Format and Content of Site Characterization Plans for High-Level-Waste Geologic Repositories	1	N/A	10 CFR Part 50 CPA	N/A
4.18	Standard Format and Content of Environmental Reports for Near-Surface Disposal of Radioactive Waste	0	N/A	10 CFR Part 50 CPA	N/A
4.19	Guidance for Selecting Sites for Near-Surface Disposal of Low-Level Radioactive Waste	0	N/A	10 CFR Part 50 CPA	N/A
4.20	Constraint on Releases of Airborne Radioactive Materials to the Environment for Licensees other than Power Reactors	1	N/A	10 CFR Part 50 CPA	N/A

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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
4.21	Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning	0	Conforms		2.4.12 12.1 12.3 11.4.3
4.22	Decommissioning Planning During Operations	0	FSAR		FSAR
4.24	Aquatic Environmental Studies for Nuclear Power Stations	0	Conforms	See CPA ER	N/A
4.25	Assessment of Abnormal Radionuclide Discharges in Ground Water to the Unrestricted Area at Nuclear Power Plant Sites	0	FSAR		FSAR
4.26	Volcanic Hazards Assessment for Proposed Nuclear Power Reactor Sites	0	N/A	No volcanoes near CRN Site.	N/A
	Materials and Plant Protection (Division 5)				
5.4	Standard Analytical Methods for the Measurement of Uranium Tetrafluoride (UF4) and Uranium Hexafluoride (UF6)	0	N/A	10 CFR Part 50 CPA	N/A
5.5	Standard Methods for Chemical, Mass Spectrometric, and Spectrochemical Analysis of Nuclear-Grade Uranium Dioxide Powders and Pellets	0	N/A	10 CFR Part 50 CPA	N/A
5.7	Entry/Exit Control for Protected Areas, Vital Areas, and Material Access Areas	1	FSAR		FSAR

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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
5.8	Design Considerations for Minimizing Residual Holdup of Special Nuclear Material in Drying and Fluidized Bed Operations	1	N/A	10 CFR Part 50 CPA	N/A
5.9	Guidelines for Germanium Spectroscopy Systems for Measurement of Special Nuclear Material	2	N/A	10 CFR Part 50 CPA	N/A
5.11	Nondestructive Assay of Special Nuclear Material Contained in Scrap and Waste	1	N/A	10 CFR Part 50 CPA	N/A
5.12	General Use of Locks in the Protection and Control of Facilities and Special Nuclear Materials, Classified Matter, and Safeguards Information (SGI)	1	FSAR		FSAR
5.20	Training, Equipping, and Qualifying of Guards and Watchmen	0	FSAR		FSAR
5.21	Nondestructive Uranium-235 Enrichment Assay by Gamma Ray Spectrometry	1	N/A	10 CFR Part 50 CPA	N/A
5.23	In Situ Assay of Plutonium Residual Holdup	1	N/A	10 CFR Part 50 CPA	N/A
5.25	Design Considerations for Minimizing Residual Holdup of Special Nuclear Material in Equipment for Wet Process Operations	0	N/A	10 CFR Part 50 CPA	N/A
5.26	Selection of Material Balance Areas and Item Control Areas	1	N/A	10 CFR Part 50 CPA	N/A
5.27	Special Nuclear Material Doorway Monitors	1	N/A	10 CFR Part 50 CPA	N/A
5.29	Nuclear Material Control Systems for Nuclear Power Plants	2	FSAR		FSAR

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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
5.31	Specially Designed Vehicle with Armed Guards for Road Shipment of Special Nuclear Material	1	N/A	10 CFR Part 50 CPA	N/A
5.34	Nondestructive Assay for Plutonium in Scrap Material by Spontaneous Fission Detection	1	N/A	10 CFR Part 50 CPA	N/A
5.37	In Situ Assay of Enriched Uranium Residual Holdup	1	N/A	10 CFR Part 50 CPA	N/A
5.38	Nondestructive Assay of High-Enrichment Uranium Fuel Plates by Gamma Ray Spectrometry	1	N/A	10 CFR Part 50 CPA	N/A
5.39	General Methods for the Analysis of Uranyl Nitrate Solutions for Assay, Isotopic Distribution, and Impurity Determinations	0	N/A	10 CFR Part 50 CPA	N/A
5.41	Shipping, Receiving, and Internal Transfer of Special Nuclear Material at Fuel Cycle Facilities	0	N/A	10 CFR Part 50 CPA	N/A
5.42	Design Considerations for Minimizing Residual Holdup of Special Nuclear Material in Equipment for Dry Process Operations	0	N/A	10 CFR Part 50 CPA	N/A
5.44	Perimeter Intrusion Alarm Systems	3	FSAR		FSAR
5.48	Design Considerations-Systems for Measuring the Mass of Liquids	0	FSAR		FSAR
5.51	Management Review of Nuclear Material Control and Accounting Systems	1	N/A	10 CFR Part 50 CPA	N/A

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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
5.52	Standard Format and Content of a Licensee Physical Protection Plan for Strategic Special Nuclear Material at Fixed Sites (Other than Nuclear Power Plants)	3	N/A	10 CFR Part 50 CPA	N/A
5.53	Qualification, Calibration, and Error Estimation Methods for Nondestructive Assay	1	N/A	10 CFR Part 50 CPA	N/A
5.54	Standard Format and Content of Safeguards Contingency Plans for Nuclear Power Plants	1	FSAR		FSAR
5.55	Standard Format and Content of Safeguards Contingency Plans for Fuel Cycle Facilities	0	N/A	10 CFR Part 50 CPA	N/A
5.58	Considerations for Establishing Traceability of Special Nuclear Material Accounting Measurements	1	FSAR		FSAR
5.59	Standard Format and Content for a Licensee Physical Security Plan for the Protection of Special Nuclear Material of Moderate or Low Strategic Significance	1	FSAR		FSAR
5.60	Standard Format and Content of a Licensee Physical Protection Plan for Strategic Special Nuclear Material in Transit	0	FSAR		FSAR
5.62	Reporting of Safeguards Events	3	Conforms		13.6
5.63	Physical Protection for Transient Shipments	0	FSAR		FSAR
5.66	Access Authorization Program for Nuclear Power Plants	2	FSAR		FSAR

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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
5.69	Guidance for the Application of Radiological Sabotage Design- Basis Threat in the Design, Development, and Implementation of a Physical Security Program that Meets 10 CFR 73.55 Requirements	1	FSAR		FSAR
5.70	Guidance for the Application of the Theft and Diversion Design- Basis Treat in the Design Development, and Implementation of a Physical Security Program that Meets CFR 73.45 and 73.461	1	FSAR		FSAR
5.71	Cyber Security Programs for Nuclear Facilities	1	Alternate Approach	Using NEI 08-09 along with IEC 62645:2019 and IEC 62859:2016 for secure development and operating environment	Table 7.1-3 7.4.4
5.73	Fatigue Management for Nuclear Power Plant Personnel	0	FSAR		FSAR
5.74	Managing the Safety/Security Interface	1	FSAR		FSAR
5.75	Training and Qualification of Security Personnel at Nuclear Power Reactor Facilities	1	FSAR		FSAR
5.76	Physical Protection Programs at Nuclear Power Reactors	1	FSAR		FSAR
5.77	Insider Mitigation Program	1	FSAR		FSAR
5.78	Physical Protection of Mixed Oxide Fuels in Nuclear Power Plants	0	N/A	10 CFR Part 50 CPA	N/A
5.79	Protection of Safeguards Information	0	FSAR		FSAR

Table 1.9-20 Conformance with Regulatory Guides
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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
5.80	Pressure-Sensitive and Tamper-Indicating Device Seals for Material Control and Accounting of Special Nuclear Material	0	FSAR		FSAR
5.81	Target Set Identification and Development for Nuclear Power Reactors	1	FSAR		FSAR
5.83	Cyber Security Event Notifications	0	FSAR		FSAR
5.84	Fitness-For-Duty Programs at New Nuclear Reactor Construction Sites	0	Conforms		13.7
5.86	Applying for Enhanced Weapons Authority, Applying for Preemption Authority, and Performing Firearms Background Checks Under 10 CFR Part 73	1	FSAR		FSAR
5.87	Suspicious Activity Reporting Under 10 CFR Part 73	1	FSAR		FSAR
5.88	Physical Inventories and Material Balances at Fuel Cycle Facilities	0	N/A	10 CFR Part 50 CPA	N/A
5.89	Fitness-For-Duty Programs for Commercial Power Reactor and Category I Special Nuclear Material Licensees	0	FSAR		FSAR
	Occupational Health (Division 8)				
8.2	Administrative Practices in Radiation Surveys and Monitoring	1	Conforms		6.8 12.3.4 12.5.3
8.4	Personnel Monitoring Device – Direct-Reading Pocket Dosimeters	1	Conforms		12.5.3

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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
8.7	Instructions for Recording and Reporting Occupational Radiation Dose Data	4	Conforms		12.5.3
8.8	Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable	3	Conforms		3.7.4 6.8 12.1.1 12.1.2 12.3.4 12.5.3
8.9	Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program	1	Conforms		12.5.3
8.10	Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable	2	Conforms		12.1.1 12.5.3
8.11	Applications of Bioassay for Uranium	1	FSAR		FSAR
8.13	Instruction Concerning Prenatal Radiation Exposure	3	Conforms		12.5.3
8.15	Acceptable Programs for Respiratory Protection	1	Conforms		12.5.3
8.18	Information Relevant to Ensuring that Radiation Exposures at Medical Institutions Will Be as Low as is Reasonably Achievable	2	N/A	10 CFR Part 50 CPA	N/A
8.19	Occupational Radiation Dose Assessment in Light Water Reactor Power Plants - Design Stage Man-Rem Estimates	1	Conforms		12.4
8.20	Applications of Bioassay for Radioiodine	2	FSAR		FSAR
8.22	Bioassay at Uranium Mills	2	N/A	10 CFR Part 50 CPA	N/A

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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
8.24	Health Physics Surveys During Enriched Uranium-235 Processing and Fuel Fabrication	2	N/A	10 CFR Part 50 CPA	N/A
8.25	Air Sampling in the Workplace	1	Conforms		12.3.4
8.26	Applications of Bioassay for Fission and Activation Products	0	FSAR		FSAR
8.27	Radiation Protection Training for Personnel at Light-Water Cooled Nuclear Power Plants	0	Conforms		12.5.3
8.28	Audible-Alarm Dosimeters	0	Conforms		12.5.3
8.29	Instruction Concerning Risks from Occupational Radiation Exposure	1	Conforms		12.5.3
8.30	Health Physics Surveys in Uranium Recovery Facilities	1	N/A	10 CFR Part 50 CPA	N/A
8.31	Information Relevant to Ensuring that Occupational Radiation Exposures at Uranium Recovery Facilities Will Be as Low as Is Reasonably Achievable	1	N/A	10 CFR Part 50 CPA	N/A
8.32	Criteria for Establishing a Tritium Bioassay Program	0	FSAR		FSAR
8.34	Monitoring Criteria and Methods to Calculate Occupational Radiation Doses	1	Conforms		12.5.3
8.35	Planned Special Exposure	1	Conforms		12.5.3
8.36	Radiation Dose to the Embryo/Fetus	0	Conforms		12.5.3
8.37	ALARA Levels for Effluents from Materials Facilities	0	N/A	10 CFR Part 50 CPA	N/A
8.38	Control of Access to High and Very High Radiation Areas of Nuclear Power Plants	1	Conforms		12.5.3

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Reg Guide	Title	Rev	Conformance	Alternatives	PSAR Section
8.39	Release of Patients Administered Radioactive Materials	1	N/A	10 CFR Part 50 CPA	N/A
8.40	Methods for Measuring Effective Dose Equivalent from External Exposure	0	FSAR		FSAR

Table 1.9-21 Compliance with Three Mile Island Requirements (10 CFR 50.34(f))
(Sheet 1 of 16)

Regulation	Description	Compliance	Comments	PSAR Section
50.34(f)(1)(i)	Perform a plant/site-specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant (II.B.8)	FSAR		FSAR
50.34(f)(1)(ii)	Perform an evaluation of the proposed Auxiliary Feedwater System (applicable to PWRs only) (II.E.1.1)	N/A	The regulation is only applicable to PWR designs.	N/A
50.34(f)(1)(iii)	Perform an evaluation of the potential for and impact of reactor coolant pump seal damage following small-break LOCA with loss of offsite power. If damage cannot be precluded, provide an analysis of the limiting small-break loss-of-coolant accident with subsequent reactor coolant pump seal damage. (II.K.2.16 and II.K.3.25)	N/A	Not technically relevant to the BWRX-300. The BWRX-300 design does not have reactor cooling pumps.	N/A
50.34(f)(1)(iv)	Perform an analysis of the probability of a small-break LOCA caused by a stuck-open Power Operated Relief Valve (PORV). If this probability is a significant contributor to the probability of small-break LOCAs from all causes, provide a description and evaluation of the effect on small-break LOCA probability of an automatic PORV isolation system that would operate when the reactor coolant system pressure falls after the PORV has opened. (Applicable to PWR's only). (II.K.3.2)	N/A	The regulation is only applicable to PWR designs.	N/A

Table 1.9-21 Compliance with Three Mile Island Requirements (10 CFR 50.34(f))
(Sheet 2 of 16)

Regulation	Description	Compliance	Comments	PSAR Section
50.34(f)(1)(v)	Perform an evaluation of the safety effectiveness of providing for separation of high pressure coolant injection and Reactor Core Isolation Cooling (RCIC) system initiation levels so that the RCIC system initiates at a higher water level than the high pressure coolant injection system, and of providing that both systems restart on low water level. (For plants with high pressure core spray systems in lieu of high pressure coolant injection systems, substitute the words, high pressure core spray for high pressure coolant injection and HPCS for HPCI) (Applicable to BWR's only). (II.K.3.13)	N/A	Not technically relevant to the BWRX-300. The BWRX-300 is a passive design and only credits the Isolation Condenser System (ICS). The BWRX-300 design does not include a RCIC system.	N/A
50.34(f)(1)(vi)	Perform a study to identify practicable system modifications that would reduce challenges and failures of relief valves, without compromising the performance of the valves or other systems. (Applicable to BWR's only). (II.K.3.16)	N/A	Not technically relevant to the BWRX-300. The design features of the BWRX-300 RCPB include the use of the I&C System and ICS for overpressure protection, and relief and safety valves are not required and are not included in the design to meet the overpressure protection acceptance criteria.	N/A

Table 1.9-21 Compliance with Three Mile Island Requirements (10 CFR 50.34(f))
(Sheet 3 of 16)

Regulation	Description	Compliance	Comments	PSAR Section
50.34(f)(1)(vii)	Perform a feasibility and risk assessment study to determine the optimum Automatic Depressurization System (ADS) design modifications that would eliminate the need for manual activation to ensure adequate core cooling. (Applicable to BWR's only). (II.K.3.18)	N/A	Not technically relevant to the BWRX-300. Automatic actuation of RPV isolation along with automatic actuation of the ICS performs the function to mitigate the effects of a LOCA.	N/A
50.34(f)(1)(viii)	Perform a study of the effect on all core-cooling modes under accident conditions of designing the core spray and low pressure coolant injection systems to ensure that the systems will automatically restart on loss of water level, after having been manually stopped, if an initiation signal is still present. (Applicable to BWR's only). (II.K.3.21)	N/A	Not technically relevant to the BWRX-300.The BWRX-300 is a passive design and only credits the ICS.	N/A
50.34(f)(1)(ix)	Perform a study to determine the need for additional space cooling to ensure reliable long-term operation of the RCIC and high-pressure coolant injection systems, following a complete loss of offsite power to the plant for at least two (2) hours. (For plants with high pressure core spray systems in lieu of high pressure coolant injection systems, substitute the words, high pressure core spray for high pressure coolant injection and HPCS for HPCI) (Applicable to BWR's only). (II.K.3.24)	N/A	Not technically relevant to the BWRX-300. The BWRX-300 does not include RCIC and High Pressure Coolant Injection systems, because automatic actuation of RPV isolation along with automatic actuation of the ICS are sufficient to mitigate the effects of a LOCA.	N/A

Table 1.9-21 Compliance with Three Mile Island Requirements (10 CFR 50.34(f))
(Sheet 4 of 16)

Regulation	Description	Compliance	Comments	PSAR Section
50.34(f)(1)(x)	Perform a study to ensure that the Automatic Depressurization System, valves, accumulators, and associated equipment and instrumentation will be capable of performing their intended functions during and following an accident situation, taking no credit for non-safety related equipment or instrumentation, and accounting for normal expected air (or nitrogen) leakage through valves. (Applicable to BWR's only). (II.K.3.28)	N/A	Not technically relevant to the BWRX-300. The BWRX-300 does not include an ADS because automatic actuation of RPV isolation along with automatic actuation of the ICS are sufficient without the need to blowdown the RCS to a suppression pool either automatically or manually to mitigate the effects of a LOCA.	N/A
50.34(f)(1)(xi)	Provide an evaluation of depressurization methods, other than by full actuation of the automatic depressurization system, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown. (Applicable to BWR's only) (II.K.3.45)	N/A	Not technically relevant to the BWRX-300. The BWRX-300 is a passive design and only credits the ICS.	N/A
50.34(f)(1)(xii)	Perform an evaluation of alternative hydrogen control systems that would satisfy the requirements of paragraph (f)(2)(ix) of this section. As a minimum include consideration of a hydrogen ignition and post-accident inerting system	Complies		6.3.3
50.34(f)(2)(i)	Provide simulator capability that correctly models the control room and includes the capability to simulate small-break LOCAs. (Applicable to construction permit applicants only) (I.A.4.2.)	FSAR		FSAR

Table 1.9-21 Compliance with Three Mile Island Requirements (10 CFR 50.34(f))
(Sheet 5 of 16)

Regulation	Description	Compliance	Comments	PSAR Section
50.34(f)(2)(ii)	Establish a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, reliability analyses, human factors engineering, crisis management, operator training, and coordination with Institute of Nuclear Power Operations (INPO) and other industry efforts. (Applicable to construction permit applicants only) (I.C.9)	FSAR		FSAR
50.34(f)(2)(iii)	Provide, for Commission review, a control room design that reflects state-of-the-art human factor principles prior to committing to fabrication or revision of fabricated control room panels and layouts. (I.D.1)	FSAR		FSAR
50.34(f)(2)(iv)	Provide a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded. (I.D.2)	Complies		7.1.3
50.34(f)(2)(v)	Provide for automatic indication of the bypassed and operable status of safety systems. (I.D.3)	Complies		7.1.3 8.1.3

Table 1.9-21 Compliance with Three Mile Island Requirements (10 CFR 50.34(f))
(Sheet 6 of 16)

Regulation	Description	Compliance	Comments	PSAR Section
50.34(f)(2)(vi)	Provide the capability of high point venting of noncondensible gases from the reactor coolant system, and other systems that may be required to maintain adequate core cooling. Systems to achieve this capability shall be capable of being operated from the CR and their operation shall not lead to an unacceptable increase in the probability of loss-of-coolant accident or an unacceptable challenge to containment integrity. (II.B.1)	N/A	Not technically relevant to the BWRX-300. The intent of this requirement is met by the configuration of the ICS piping and purge capability, which controls potential accumulation of non-condensable gases that could affect ECCS function.	N/A
50.34(f)(2)(vii)	Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain accident source term radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment. (II.B.2)	FSAR		FSAR
50.34(f)(2)(viii)	Provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain accident source term radioactive materials without radiation exposures to any individual exceeding 5 rems to the whole body or 50 rems to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, radioiodines and cesiums, and nonvolatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations. (II.B.3)	FSAR		FSAR

Table 1.9-21 Compliance with Three Mile Island Requirements (10 CFR 50.34(f))
(Sheet 7 of 16)

Regulation	Description	Compliance	Comments	PSAR Section
50.34(f)(2)(ix)	Provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100% fuel-clad metal water reaction. Preliminary design information on the tentatively preferred system option of those being evaluated in paragraph (f)(1)(xii) of this section is sufficient at the construction permit stage.	Complies		6.2.1
50.34(f)(2)(x)	Provide a test program and associated model development and conduct tests to qualify reactor coolant system relief and safety valves and, for PWR's, PORV block valves, for all fluid conditions expected under operating conditions, transients, and accidents. Consideration of Anticipated Transients Without Scram (ATWS) conditions shall be included in the test program. Actual testing under ATWS conditions need not be carried out until subsequent phases of the test program are developed. (II.D.1)	N/A	Not technically relevant to the BWRX-300. The design features of the BWRX-300 RCPB include the use of the I&C System and ICS for overpressure protection, and relief and safety valves are not required and are not included in the design to meet the overpressure protection acceptance criteria.	N/A
50.34(f)(2)(xi)	Provide direct indication of relief and safety valve position (open or closed) in the control room. (II.D.3)	N/A	Not technically relevant to the BWRX-300. 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 issue is not applicable to the closed system ICS used for overpressure protection.	N/A

Table 1.9-21 Compliance with Three Mile Island Requirements (10 CFR 50.34(f))
(Sheet 8 of 16)

Regulation	Description	Compliance	Comments	PSAR Section
50.34(f)(2)(xii)	Provide automatic and manual auxiliary feedwater (AFW) system initiation and provide auxiliary feedwater system flow indication in the control room. (Applicable to PWR's only) (II.E.1.2)	N/A	The regulation is only applicable to PWR designs.	N/A
50.34(f)(2)(xiii)	Provide pressurizer heater power supply and associated motive and control power interfaces sufficient to establish and maintain natural circulation in hot standby conditions with only onsite power available. (Applicable to PWR's only) (II.E.3.1)	N/A	The regulation is only applicable to PWR designs.	N/A
50.34(f)(2)(xiv)	Provide containment isolation systems (II.E.4.2)	Complies		6.2.4 7.1.3
50.34(f)(2)(xv)	Provide a capability for containment purging/venting designed to minimize the purging time consistent with ALARA principles for occupational exposure. Provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions. (II.E.4.4)	Complies		6.2.1 6.2.4
50.34(f)(2)(xvi)	Establish a design criterion for the allowable number of actuation cycles of the emergency core cooling system and Reactor Protection System consistent with the expected occurrence rates of severe overcooling events (considering both anticipated transients and accidents). (Applicable to B&W designs only). (II.E.5.1)	N/A	The regulation is only applicable to B&W designs.	N/A

Table 1.9-21 Compliance with Three Mile Island Requirements (10 CFR 50.34(f))
(Sheet 9 of 16)

Regulation	Description	Compliance	Comments	PSAR Section
50.34(f)(2)(xvii)	Provide instrumentation to measure, record, and readout in the control room: (A) containment pressure, (B) containment water level, (C) containment hydrogen concentration, (D) containment radiation intensity (high level), and (E) noble gas effluents at all potential, accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples. (II.F.1)	Complies		6.2.1.3 6.8 7.1.3 7.3.1 12.3.4
50.34(f)(2)(xviii)	Provide instruments that provide in the control room an unambiguous indication of inadequate core cooling, such as primary coolant saturation meters in PWR's, and a suitable combination of signals from indicators of coolant level in the reactor vessel and in-core thermocouples in PWR's and BWR's. (II.F.2)	FSAR	The design of the instrumentation to show compliance will be discussed in the FSAR.	FSAR
50.34(f)(2)(xix)	Provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage. (II.F.3)	Complies		6.8 7.1.3 7.3.1

Table 1.9-21 Compliance with Three Mile Island Requirements (10 CFR 50.34(f))
(Sheet 10 of 16)

Regulation	Description	Compliance	Comments	PSAR Section
50.34(f)(2)(xx)	Provide power supplies for pressurizer relief valves, block valves, and level indicators such that: (A) Level indicators are powered from vital buses; (B) motive and control power connections to the emergency power sources are through devices qualified in accordance with requirements applicable to systems important to safety and (C) electric power is provided from emergency power sources. (Applicable to PWR's only). (II.G.1)	N/A	The regulation is only applicable to PWR designs.	N/A
50.34(f)(2)(xxi)	Design auxiliary heat removal systems such that necessary automatic and manual actions can be taken to ensure proper functioning when the main feedwater system is not operable. (Applicable to BWR's only). (II.K.1.22)	Complies		6.3.3
50.34(f)(2)(xxii)	Perform a failure modes and effects analysis of the integrated control system to include consideration of failures and effects of input and output signals to the ICS. (Applicable to B&W-designed plants only). (II.K.2.9)	N/A	The regulation is only applicable to B&W designs.	N/A
50.34(f)(2)(xxiii)	Provide, as part of the reactor protection system, an anticipatory reactor trip that would be actuated on loss of main feedwater and on turbine trip. (Applicable to B&W-designed plants only). (II.K.2.10)	N/A	The regulation is only applicable to B&W designs.	N/A
50.34(f)(2)(xxiv)	Provide the capability to record reactor vessel water level in one location on recorders that meet normal post-accident recording requirements. (Applicable to BWR's only). (II.K.3.23)	Complies		7.1.3 7.3.1

Table 1.9-21 Compliance with Three Mile Island Requirements (10 CFR 50.34(f))
(Sheet 11 of 16)

Regulation	Description	Compliance	Comments	PSAR Section
50.34(f)(2)(xxv)	Provide an onsite Technical Support Center, an onsite Operational Support Center, and, for construction permit applications only, a nearsite Emergency Operations Facility. (III.A.1.2).	FSAR		FSAR
50.34(f)(2)(xxvi)	Provide for leakage control and detection in the design of systems outside containment that contain (or might contain) accident source term radioactive materials following an accident. Applicants shall submit a leakage control program, including an initial test program, a schedule for re-testing these systems, and the actions to be taken for minimizing leakage from such systems. The goal is to minimize potential exposures to workers and public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency. (III.D.1.1)	Complies		5.4.7 6.2.1 6.3.3
50.34(f)(2)(xxvii)	Provide for monitoring of in-plant radiation and airborne radioactivity as appropriate for a broad range of routine and accident conditions. (III.D.3.3)	Complies		12.3.4
50.34(f)(2)(xxviii)	Evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in an accident source term release and make necessary design provisions to preclude such problems. (III.D.3.4)	Complies		6.4.3

Table 1.9-21 Compliance with Three Mile Island Requirements (10 CFR 50.34(f))
(Sheet 12 of 16)

Regulation	Description	Compliance	Comments	PSAR Section
50.34(f)(3)(i)	Provide administrative procedures for evaluating operating, design, and construction experience and for ensuring that applicable important industry experiences will be provided in a timely manner to those designing and constructing the plant. (I.C.5)	FSAR		FSAR
50.34(f)(3)(ii)	Ensure that the quality assurance (QA) list required by Criterion II, app. B, 10 CFR part 50 includes all structures, systems, and components important to safety. (I.F.1)	Complies		13.2 13.4 17.5
50.34(f)(3)(iii)	Establish a quality assurance (QA) program based on consideration of: (A) Ensuring independence of the organization performing checking functions from the organization responsible for performing the functions; (B) performing quality assurance/quality control functions at construction sites to the maximum feasible extent; (C) including QA personnel in the documented review of and concurrence in quality related procedures associated with design, construction and installation; (D) establishing criteria for determining QA programmatic requirements; (E) establishing qualification requirements for QA and QC personnel; (F) sizing the QA staff commensurate with its duties and responsibilities; (G) establishing procedures for maintenance of as-built documentation; and (H) providing a QA role in design and analysis activities. (I.F.2)	Complies		13.2 13.4 17.5

Table 1.9-21 Compliance with Three Mile Island Requirements (10 CFR 50.34(f))
(Sheet 13 of 16)

Regulation	Description	Compliance	Comments	PSAR Section
50.34(f)(3)(iv)	Provide one or more dedicated containment penetrations, equivalent in size to a single 3-foot diameter opening, in order not to preclude future installation of systems to prevent containment failure, such as a filtered vented containment system. (II.B.8)	FSAR		FSAR

Table 1.9-21 Compliance with Three Mile Island Requirements (10 CFR 50.34(f))
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Regulation	Description	Compliance	Comments	PSAR Section
50.34(f)(3)(v)(A)(1)	Containment integrity will be maintained (i.e., for steel containments by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, subarticle NE-3220, Service Level C Limits, except that evaluation of instability is not required, considering pressure and dead load alone. For concrete containments by meeting the requirements of the ASME Boiler Pressure Vessel Code, Section III, Division 2 subarticle CC-3720, Factored Load Category, considering pressure and dead load alone) during an accident that releases hydrogen generated from 100% fuel clad metal-water reaction accompanied by either hydrogen burning or the added pressure from post-accident inerting assuming carbon dioxide is the inerting agent. As a minimum, the specific code requirements set forth above appropriate for each type of containment will be met for a combination of dead load and an internal pressure of 45 psig. Modest deviations from these criteria will be considered by the staff, if good cause is shown by an applicant. Systems necessary to ensure containment integrity shall also be demonstrated to perform their function under these conditions.	FSAR		FSAR

Table 1.9-21 Compliance with Three Mile Island Requirements (10 CFR 50.34(f))
(Sheet 15 of 16)

Regulation	Description	Compliance	Comments	PSAR Section
50.34(f)(3)(v)(B)(1)	(B)(1) Containment structure loadings produced by an inadvertent full actuation of a post-accident inerting hydrogen control system (assuming carbon dioxide), but not including seismic or design basis accident loadings will not produce stresses in steel containments in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 1, subarticle NE3220, Service Level A Limits, except that evaluation of instability is not required (for concrete containments the loadings specified above will not produce strains in the containment liner in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 2, subarticle CC-3720, Service Load Category.	FSAR		FSAR
50.34(f)(3)(v)(B)(2)	(2) The containment has the capability to safely withstand pressure tests at 1.10 and 1.15 times (for steel and concrete containments, respectively) the pressure calculated to result from carbon dioxide inerting.	N/A	Not technically relevant to the BWRX-300. The BWRX-300 design does not inert with carbon dioxide.	N/A
50.34(f)(3)(vi)	For plant designs with external hydrogen recombiners, provide redundant dedicated containment penetrations so that, assuming a single failure, the recombiner systems can be connected to the containment atmosphere. (II.E.4.1)	N/A	Not technically relevant to the BWRX-300. The BWRX-300 does not include external recombiners.	N/A

Table 1.9-21 Compliance with Three Mile Island Requirements (10 CFR 50.34(f))
(Sheet 16 of 16)

Regulation	Description	Compliance	Comments	PSAR Section
50.34(f)(3)(vii)	Provide a description of the management plan for design and construction activities, to include: (A) The organizational and management structure singularly responsible for direction of design and construction of the proposed plant; (B) technical resources director by the applicant; (C) details of the interaction of design and construction within the applicant's organization and the manner by which the applicant will ensure close integration of the architect engineer and the nuclear steam supply vendor; (D) proposed procedures for handling the transition to option; (E) the degree of top level management oversight and technical control to be exercised by the applicant during design and construction, including the preparation and implementation of procedures necessary to guide the effort. (II.J.3.1)	Complies		1.4

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

CRN-1 is not co-located at the CRN 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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LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
ALWR	Advanced light water reactor
ASCE/SEI	American Society of Civil Engineers/Structural Engineering Institute
CCSM4	Community Climate System Model, version 4
CFR	Code of Federal Regulations
CRN	Clinch River Nuclear
EPA	U.S. Environmental Protection Agency
EPRI	Electric Power Research Institute
GVH	GE Vernova Hitachi Nuclear Energy
HMR	Hydro-Meteorological Report
IEEE	Institute of Electrical and Electronics Engineers
ISG	Interim Staff Guidance
LCD	local climatological data
LMDCT	linear mechanical draft cooling tower
MCR	Main Control Room
NAAQS	National Ambient Air Quality Standards
NCDC	National Climatological Data Center
NOAA	National Oceanic and Atmospheric Administration
NUREG	U.S. Nuclear Regulatory Commission Regulation
NWS	National Weather Service
PM2.5	particulate matter with aerodynamic diameter less than or equal to 2.5 microns
PM10	particulate matter with aerodynamic diameter less than or equal to 10 microns
SRP	Standard Review Plan
TDS	total dissolved solids
TVA	Tennessee Valley Authority
USNRC	U.S. Nuclear Regulatory Commission

LIST OF EFFECTIVE SECTIONS

CHAPTER 2 SITE CHARACTERISTICS AND SITE PARAMETERS

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CHAPTER 2 SITE CHARACTERISTICS AND SITE PARAMETERS

Chapter 2 of the referenced Clinch River Nuclear Site Early Site Permit Application Part 2, Site Safety Analysis Report (ESPA SSAR) is incorporated by reference with the following supplements.

Replace the first paragraph with the following:

Chapter 2 of the PSAR describes the characteristics of the Clinch River Nuclear (CRN) Site, including site related parameters for the CRN-1 Plant and the geological, seismological, hydrological, and meteorological characteristics of the site and vicinity. The CRN Site location, characteristics, and site-related design parameters, as described in the following sections, are provided in sufficient detail to support a safety assessment:

Section 2.0: Plant Parameter Envelope Evaluation

Section 2.1: Geography and Demography

Section 2.2: Nearby Industrial, Transportation, and Military Facilities

Section 2.3: Meteorology

Section 2.4: Hydrologic Engineering

Section 2.5: Geology, Seismology, and Geotechnical Engineering

2.0 Plant Parameter Envelope Evaluation

Replace ESPA SSAR Section 2.0 with the following:

TVA selected the GE Vernova Hitachi Nuclear Energy (GVH) BWRX-300 small modular reactor (SMR) for deployment at the CRN Site. The site suitability of the CRN Site is established in the ESPA. Evaluation of the BWRX-300 design parameters as compared to the CRN Site ESPA is provided in this section. In support of this safety assessment, Table 2.0-1R provides a comparison of site-related design parameters for which the BWRX-300 SMR is designed to the site characteristics specific of CRN-1. The first four columns of Table 2.0-1R are a compilation of site characteristics of the ESPA SSAR. The last two columns of Table 2.0-1R contain information for the corresponding BWRX-300 SMR site parameter. Table 2.0-2R provides a comparison of site-related design parameters for which the BWRX-300 SMR is designed to the site related design specific for CRN-1 in support of this safety assessment. The first four columns of Table 2.0-2R are a compilation of site related design parameters of the ESPA SSAR. The last two columns of Table 2.0-2R contain information for the corresponding BWRX-300 SMR site parameter. Table 2.0-3R through Table 2.0-6R provide a comparison of releases for the BWRX-300 SMR from: LOCA Design Basis Accident, Annual Normal Gaseous Radioactive Release, Accidental Liquid Radwaste Release and Annual Normal Liquid Radioactive Release, respectively.

Add new Section 2.0.1:

2.0.1 Justification of Exceedances

As shown in Table 2.0-1R and Table 2.0-2R, the Site Characteristics and the Site Related Design Parameters for CRN-1 are bounded by those developed for the ESPA Plant Parameter Envelope (PPE) except for the single unit thermal megawatts value of 800 MWt, the population center value of 4.8 miles, the maximum ground water value of 816.1 feet, and the minimum site grade 821 feet. The BWRX-300 SMR design has an 870 MWt nominal output. The closest population center to the CRN Site is 4.3 miles and the CRN Site grade and the maximum ground water level are 814.5 feet. These exceedances for a single unit are acceptable for the CRN Site because as shown in the PPE comparison tables provided in this section, all other CRN Early Site Permit, ESP-006 (Reference 2.0-1) PPE performance requirements (radiological release, structural design, etc.) are met for both the site and a single unit.

Justifications for the CRN-1 exceedances for the remaining release data are provided below:

LOCA: Design basis accidents for the GVH BWRX-300 SMR are evaluated in Chapter 15. The Feedwater Line Break Accident (FWLB) is the most limiting DBA. Releases for several radionuclides for the FWLB accident exceed the release in the LOCA DBA evaluated in the ESPA SSAR as shown in Table 2.0-3R. However, the dose consequence of the postulated FWLB accident (Table 15.7-6) is below applicable acceptance criteria and therefore the exceedances shown in Table 2.0-3R are acceptable.

Annual Normal Gaseous Radioactive Release: Releases for several radionuclides for Annual Normal Gaseous Radioactive Release exceed the release evaluated in the ESPA SSAR as shown in Table 2.0-4R. As described in Table 2.0-4R footnotes, the exceedances have been evaluated and show that CRN-1 calculated site boundary concentration limits and offsite receptor annual average doses are bounded by values presented in ESPA SSAR Table 11.2-7, and Tables 11.3-3 through 11.3-5; and remain within regulatory limits.

Accidental Liquid Radwaste Release: Releases for several radionuclides for the Accidental Liquid Radwaste Release accident exceed the release evaluated in the ESPA SSAR as shown in Table 2.0-5R. However, the dose consequence of the postulated Accidental Liquid Radwaste Release (Section 2.4.13.8) is less than the 10 CFR 20.1301 limit of 100 mrem TEDE and therefore the exceedances shown in Table 2.0-5R are acceptable.

Annual Normal Liquid Radioactive Release: Releases for several radionuclides for Annual Normal Liquid Radioactive Release exceed the release evaluated in the ESPA SSAR as shown in Table 2.0-6R. The specific radionuclides that exceed the ESPA PPE are minor dose contributors with respect to the considered dominant dose contributing radionuclides. The dominant dose contributing radionuclides (e.g., I-131, Cs-137, and H-3) calculated for CRN-1 are exceeded by many orders of magnitude by the corresponding ESPA PPE values. Therefore, the normal operation dose consequence from liquid effluents to members of the public from CRN-1 is expected to be bounded by ESPA PPE.

2.0.2 References

2.0-1 “Clinch River Nuclear Site Early Site Permit, ESP-006,” U.S. Nuclear Regulatory Commission, December 2019 (ML19352D868) and Final Safety Evaluation Report, June 2019 (ML19162A157).

Replace Table 2.0-1 with the following:

Table 2.0-1R Site Characteristics
(Sheet 1 of 8)

Characteristic/ Parameter	PPE Site-Specific Value	Description	ESPA SSAR Section	CRN-1 Site-Related Design Parameter	Comments
Geography and Demography					
EAB	Clinch River Property Boundary	The area surrounding the reactors, in which the reactor licensee has the authority to determine all activities, including exclusion or removal of personnel and property from the area.	2.1.1	Clinch River Property Boundary	Design parameter equals Site-Specific Value.
LPZ	1 mi from CRN Site center point	The area immediately surrounding the exclusion area which contains residents, the total number and density of which are such that there is a reasonable probability that appropriate protective measures could be taken in their behalf in the event of a serious accident.	2.1.3.4	1 mi from CRN Site center point	Design parameter equals Site-Specific Value.
Population Center Distance	4.8 mi (southeast)	The distance from the site center point to the nearest boundary of a densely populated center containing more than about 25,000 residents.	2.1.3.5	4.3 mi (southeast)	Smaller than ESPA PPE.
Meteorology and Hydrology					
Winter Precipitation					
Normal Winter Precipitation Event	21.9 psf	The maximum ground-level weight (lb/ft^2) of the 1) 100-year return snowpack (snow cover), 2) historical snowpack (snow cover), 3) 100-year-return 2-day snowfall event, or 4) historical maximum 2-day snowfall event.	2.3.1.3.6.2	52.2 psf	Design parameter bounds Site-Specific Value.
Extreme Frozen Winter Precipitation Event	21.9 psf	The maximum ground-level weight (lb/ft^2) of the 1) 100-year return 2-day snowfall event or 2) historical maximum 2-day snowfall event.	2.3.1.3.6.2	104 psf	Design parameter bounds Site-Specific Value.

Table 2.0-1R Site Characteristics
(Sheet 2 of 8)

Characteristic/ Parameter	PPE Site-Specific Value	Description	ESPA SSAR Section	CRN-1 Site-Related Design Parameter	Comments
Extreme Liquid Winter Precipitation Event (48-hour Probable Maximum Winter Precipitation)	23.5 in	The extreme liquid winter precipitation event is defined as the theoretically greatest ground-level depth of precipitation (in inches of water) for a 48-hour period that is physically possible over a 25.9 square kilometer (10 square mile) area at a particular geographical location during those months with the historically highest snowpacks.	2.3.1.3.6.2	N/A	Not relevant to BWRX-300 design.
Potential for Frazil Ice in Ultimate Heat Sink Water Storage Facility	N/A	Potential for accumulated ice formation in the Ultimate Heat Sink Water Storage Facility in a turbulent flow condition.	2.4.7	N/A	Same as ESPA PPE.
Maximum Rainfall Rate	18.8 in/hr 6 in/5 minutes	Probable Maximum Precipitation for 1-hour and for 5-minute durations at the site estimated from Hydro-Meteorological Report HMR-52.	2.3.1.3.3	19.4 in/hr 6.2 in/5 minutes.	Design parameter bounds Site-Specific Value.
Maximum Flood (or Tsunami)	[ ]	Predicted maximum flood level (including wave run-up) from external events, not including local Probable Maximum Precipitation.	2.4.2, 2.4.3, and 2.4.10	[ ]	The Site-Specific maximum flood level is below the Plant design maximum flood level.
Maximum Ground Water	816.1 ft NAVD88	Maximum groundwater level under deep foundation structures in power block area.	2.4.12	At plant grade (814.5 ft NAVD88)	Maximum ground water will be at plant grade.
Basic Wind Speed	96.3 mph for a 3-second gust	Wind velocity at 33 ft above ground for Exposure Category C associated with a 100-year return period in the site area.	2.3.1.3.2	160 mph	Design parameter bounds Site-Specific Value.
Historical Maximum Wind Speed	87 mph for a 3-second gust 73 mph fastest mile	Wind velocity at 33 ft above ground associated with the most severe hurricane wind that has been historically observed in the site region.	2.3.1.3.2	N/A	Not relevant to BWRX-300 design.
Design-Basis Hurricane Windspeed	130 mph for a 3-second gust	The resulting windspeed for nominal 3-second peak-gust values at a height of 33 ft in flat open terrain.	2.3.1.3.5	200 mph	Design parameter bounds Site-Specific Value.

Table 2.0-1R Site Characteristics
(Sheet 3 of 8)

Characteristic/ Parameter	PPE Site-Specific Value	Description	ESPA SSAR Section	CRN-1 Site-Related Design Parameter	Comments
Tornado					
Maximum Pressure Drop	1.2 psi	Decrease in ambient pressure from normal atmospheric pressure at the site due to passage of a tornado having a probability of occurrence of 10^{-7} per year.	2.3.1.3.4	1.2 psi	Design parameter equals Site-Specific Value.
Maximum Rotational Speed	184 mph	Rotation component of maximum wind speed at the site due to passage of a tornado having a probability of occurrence of 10^{-7} per year.	2.3.1.3.4	184 mph	Design parameter equals Site-Specific Value.
Maximum Translational Speed	46 mph	Translation component of maximum wind speed at the site due to the movement across the ground of a tornado having a probability of occurrence of 10^{-7} per year.	2.3.1.3.4	46 mph	Design parameter equals Site-Specific Value.
Maximum Wind Speed	230 mph	Sum of the maximum rotational and translational wind speed components at the site due to passage of a tornado having a probability of occurrence of 10^{-7}per year.	2.3.1.3.4	230 mph	Design parameter equals Site-Specific Value.
Radius of Maximum Rotational Speed	150 ft	Distance from the center of the tornado at which the maximum rotational wind speed occurs at site due to passage of a tornado having a probability of occurrence of 10^{-7} per year.	2.3.1.3.4	150 ft	Design parameter equals Site-Specific Value.
Rate of Pressure Drop	0.5 psi/s	Maximum rate of pressure drop at site due to passage of a tornado having a probability of occurrence of 10^{-7} per year.	2.3.1.3.4	0.5 psi/s	Design parameter equals Site-Specific Value.

Table 2.0-1R Site Characteristics
(Sheet 4 of 8)

Characteristic/ Parameter	PPE Site-Specific Value	Description	ESPA SSAR Section	CRN-1 Site-Related Design Parameter	Comments
Site Characteristic Ambient Air Temperatures		Site characteristic wet bulb and dry bulb temperatures associated with the listed exceedance values and the 100-year return period.	2.3.1.4		
Maximum Dry Bulb Temperature with Maximum Wet Bulb Temperature		The maximum dry-bulb temperature that has existed at the site for 2 hours or more combined with the maximum wet-bulb temperature that exists in that population of dry-bulb temperatures			
95% Annual Exceedance	30°F DB			N/A	Not relevant to BWRX-300 design.
5% Annual Exceedance	85°F DB 71.8 °F Coincident WB			N/A	Not relevant to BWRX-300 design.
2% Annual Exceedance	90°F DB 73.7°F Coincident WB			96°F DB 79.0°F mean coincident WB	Design parameter bounds Site-Specific Value.
1% Annual Exceedance	92°F DB 74.2°F Coincident WB			100°F DB 79°F mean coincident WB	Design parameter bounds Site-Specific Value.
0.4% Annual Exceedance	95°F DB 74.9°F Coincident WB			N/A	Not relevant to BWRX-300 design.
0% Annual Exceedance	105°F DB 74.6°F Coincident WB			117°F DB 80°F mean coincident WB	Design parameter bounds Site-Specific Value.
100-Year Return Period	107°F DB 73.1°F Coincident WB			N/A	Not relevant to BWRX-300 design.

Table 2.0-1R Site Characteristics
(Sheet 5 of 8)

Characteristic/ Parameter	PPE Site-Specific Value	Description	ESPA SSAR Section	CRN-1 Site-Related Design Parameter	Comments
Maximum Non-Coincident Wet Bulb Temperature		The maximum historic wet-bulb temperature recorded for 2 or more hours.			
2% Annual Exceedance	75.7°F			81.0°F	Design parameter bounds Site-Specific Value.
1% Annual Exceedance	76.7°F			82°F	Design parameter bounds Site-Specific Value.
0.4% Annual Exceedance	77.6°F			N/A	Not relevant to BWRX-300 design.
0% Annual Exceedance	81.7°F			88°F	Design parameter bounds Site-Specific Value.
100-Year Return Period	83.6°F			N/A	Not relevant to BWRX-300 design.
Minimum Dry Bulb Temperature					
2% Annual Exceedance	25°F			-10°F	Design parameter bounds Site-Specific Value.
1% Annual Exceedance	21°F			-11.2°F	Design parameter bounds Site-Specific Value.
0.4% Annual Exceedance	16°F			N/A	Not relevant to BWRX-300 design.
0% Annual Exceedance	-9°F			-40°F	Design parameter bounds Site-Specific Value.
100-Year Return Period	-9.9°F			N/A	Not relevant to BWRX-300 design.

Table 2.0-1R Site Characteristics
(Sheet 6 of 8)

Characteristic/ Parameter	PPE Site-Specific Value	Description	ESPA SSAR Section	CRN-1 Site-Related Design Parameter	Comments
Atmospheric Dispersion(X/Q) (Accident)[f]		Atmospheric dispersion coefficients used in the design safety analyses to estimate dose consequences of accident airborne releases.			
0-2 hr @ EAB	4.96×10^{-3} s/m^3		2.3.4	4.96×10^{-3} s/m^3	Design parameter equals Site-Specific Value.
0-8 hr @ LPZ	3.10×10^{-4} s/m^3			3.10×10^{-4} s/m^3	Design parameter equals Site-Specific Value.
8-24 hr @ LPZ	2.26×10^{-4} s/m^3			2.26×10^{-4} s/m^3	Design parameter equals Site-Specific Value.
1-4 day @ LPZ	1.14×10^{-4} s/m^3			1.14×10^{-4} s/m^3	Design parameter equals Site-Specific Value.
4-30 day @ LPZ	4.30×10^{-5} s/m^3			4.30×10^{-5} s/m^3	Design parameter equals Site-Specific Value.
Atmospheric Dispersion(X/Q) (Annual Average)[f]	Refer to ESPA SSAR Table 2.3.5-10	Atmospheric dispersion coefficient used in the safety analysis for the dose consequences of normal airborne releases.	2.3.5	See values in ESPA SSAR Table 2.3.5-10	Design parameter equals Site-Specific Value.
Gaseous Releases					
Dose Consequences					
Normal	10 CFR 20, App. B 10 CFR 50, App. I	Estimated design radiological dose consequences due to gaseous releases from normal operation of the plant.	11.3.3	10 CFR 20 10 CFR 50 App I.	These are the dose related limits for normal release effluents.
Post-Accident	10 CFR 52.17(a)(1)(ix)	Estimated design radiological dose consequences due to gaseous releases from postulated accidents.	15	10 CFR 50.34(a)(1)(ii)(D)	These are the accident dose limits.
Minimum Distance from Release Point to EAB	1100 ft	Minimum lateral distance from the effluent release boundary to the EAB	2.1.1.2 and 2.3.4	1100 ft	The CRN-1 EAB is the site boundary. No impact of larger EAB.

Table 2.0-1R Site Characteristics
(Sheet 7 of 8)

Characteristic/ Parameter	PPE Site-Specific Value	Description	ESPA SSAR Section	CRN-1 Site-Related Design Parameter	Comments
Liquid Releases					
Dose Consequences					
Normal	10 CFR 20, App. B 10 CFR 50, App. I	Estimated design radiological dose consequences due to liquid releases from normal operation of the plant.	11.2.3	10 CFR 50 App. I 10 CFR 20 App. B	These are the dose related limits for normal release effluents.
Post-Accident	10 CFR 20, App. B DC/COL-ISG-013	Estimated design radiological dose consequences due to liquid releases from postulated accidents.	2.4.13	NUREG-0800, BTP 11-6	These are the accident dose limits.
Geology, Seismology, and Geotechnical Engineering					
Ground Motion Response Spectra	Refer to ESPA SSAR Figure 2.5.2-78	The design response spectra used to establish a plant's seismic design	2.5.2	Refer to Figure 2.5.2-95R	PSAR curve meets the design requirements for the BWRX-300.
Capable Tectonic Structures or Sources	None	The assumption made in a plant design about the presence of capable faults or earthquake sources in the vicinity of the plant site (e.g., no fault displacement potential within the investigative area).		No fault displacement within the site area, no permanent ground deformation per 10 CFR 100.23.	The site condition meets the design requirements for the BWRX-300.
Soil and Rock Properties					
Liquefaction	None	Liquefaction potential at the site.	2.5.4	None	The site condition meets the design requirements for the BWRX-300.
Minimum Bearing Capacity	110 ksf (static)	Allowable load-bearing capacity of layer supporting plant structures.	2.5.4	Total maximum bearing demand (static and dynamic) 33.4 ksf	ESPA PPE is bounding.
Minimum Shear Wave Velocity	4650 ft/s	Propagation velocity of shear waves through foundation materials.	2.5.4	4650 ft/s	Design parameter equals Site-Specific Value.
Dynamic Bearing Capacity	110 ksf	Capacity of the foundation soil/rock to resist loads imposed by the structures in the event of an earthquake.	2.5.4	Total maximum bearing demand (static and dynamic) 33.4 ksf	ESPA PPE is bounding.

Table 2.0-1R Site Characteristics
(Sheet 8 of 8)

Characteristic/ Parameter	PPE Site-Specific Value	Description	ESPA SSAR Section	CRN-1 Site-Related Design Parameter	Comments
Minimum Soil Angle of Internal Friction	36°	Minimum value of the internal friction angle of foundation soils, fill soils, or excavation slopes that would provide a safe design of the plant through soil structure interaction analyses including sliding along the base.	2.5.4	30°	ESPA PPE is bounding.

Notes:
°F = degrees Fahrenheit
DB = dry bulb
EAB = Exclusion Area Boundary
ft = feet
hr = hour(s)
in. = inch(s)
ksf = thousand pounds per square foot
lb = pound(s)
LPZ = Low Population Zone
mi = mile(s)
mph = miles per hour
NAVD88 = North American Vertical Datum of 1988
NGVD29 = National Geodetic Vertical Datum of 1929
psf = pounds per square foot
psi = pound-force per square inch
s = second(s)

Replace Table 2.0-2 with the following:

Table 2.0-2R Site-Related Design Parameters
(Sheet 1 of 2)

Characteristic/ Parameter	PPE Value	Description	ESPA SSAR Section	CRN-1 Site-Related Design Parameter	Comments
Structure Height	160 ft	The height from finished grade to the top of the tallest power block structure, excluding stacks and cooling towers.	3.5.1.6	<103 ft	Smaller than ESPA PPE.
Structure Foundation Embedment	138 ft	The depth from finished grade to the bottom of the basemat for the most deeply embedded power block structure.	2.4.12	<120 ft	Smaller than ESPA PPE.
Plant Megawatts Thermal	800 MWt (805 MWt including reactor coolant pump) (2420 MWt)	The maximum thermal power generated by one unit and the maximum thermal power for the site.	1.2.2	870 MWt	Smaller than ESPA PPE for site total; larger than ESPA PPE for a single unit.
Minimum Site Grade	821 ft NAVD88	Minimum finished ground elevation in the power block area.	2.4.1	814.5 ft NAVD88	Smaller than ESPA PPE.
Condenser/Heat Exchanger Design Duty	(5593 MBTU/hr)	Design value for the waste heat rejected to the circulating water system across the condensers.	-	~570 MW (1646 MBTU/hr)	Smaller than ESPA PPE.
Gaseous Releases					
Source Term (Accident)	Refer to ESPA SSAR Table 2.0-3	Bounding design basis accident atmospheric release by post-accident interval.	15.2	Table 2.0-3R	ESPA PPE is bounding.
Source Term (Normal)	Refer to ESPA SSAR Table 2.0-4	Annual activity, by radionuclide, contained in routine plant airborne effluent streams.	11.3.3	Table 2.0-4R	ESPA PPE is bounding.
Release Point Elevation					
Accident	Ground level	The elevation above finished grade of the release point for releases due to an accident.	2.3.4	Ground level	Same as ESPA PPE.
Normal	Ground level	The elevation above finished grade of the release point for normal effluent releases.	2.3.5	Ground level release is assumed for normal release	The CRN-1 source term is bounded by the ESPA source term.

Table 2.0-2R Site-Related Design Parameters
(Sheet 2 of 2)

Characteristic/ Parameter	PPE Value	Description	ESPA SSAR Section	CRN-1 Site-Related Design Parameter	Comments
Liquid Releases					
Source Term (Accident)	Refer to ESPA SSAR Table 2.0-5	The assumed activity, by radionuclide, contained in accidental liquid radwaste release.	2.4.13	Table 2.0-5R	ESPA PPE is bounding.
Source Term (Normal)	Refer to ESPA SSAR Table 2.0-6	Annual activity, by radionuclide, contained in routine plant liquid effluent streams.	11.2.3	Table 2.0-6R	ESPA PPE is bounding.

Notes:
ft = feet
hr = hour(s)
MBTU = millions British thermal units
MW = megawatt(s)
MWt = megawatt thermal
NAVD88 = North American Vertical Datum of 1988

Rename and Replace Table 2.0-3 with the following:

Table 2.0-3R CRN-1 Projected LOCA Bounding Design Basis Accident Radioactive Release
(Sheet 1 of 3)

Radionuclide	LOCA ESPA PPE Worst 2 hr (Ci)	CRN-1 Release - FWLB Pre-Incident Iodine Spike (Ci)[a]
Ag-110m		**5.97E-06**
Am-241	1.56E-05	
Ba-139	4.14E+00	
Ba-137m		**6.05E-02**
Ba-140	1.33E+01	6.12E-01
Ce-141	3.19E+01	2.95E-02
Ce-143	2.81E-01	
Ce-144	2.65E-01	4.39E-03
Cm-242	3.16E-03	
Cm-244	1.84E-04	
Co-58		**1.37E-03**
Co-60		**2.66E-03**
Cr-51		**5.76E-03**
Cs-134	2.35E+01	3.96E-02
Cs-136	6.70E+00	3.09E-02
Cs-137	1.80E+01	6.05E-02
Cs-138	1.14E+01	2.95E+00
Cu-64		**2.81E-02**
Fe-55		**5.97E-03**
Fe-59		**1.58E-03**
H-3		**2.41E-01**
I-130	2.12E+00	
I-131	1.34E+02	9.39E+00
I-132	9.61E+01	9.39E+01
I-133	2.59E+02	6.93E+01
I-134	4.98E+01	**2.74E+02**
I-135	2.06E+02	1.37E+02
Kr-83m		**8.03E-02**
Kr-85	3.01E+01	5.50E-04
Kr-85m	3.51E+02	2.25E-03
Kr-87	2.66E+02	1.55E-02
Kr-88	7.48E+02	8.45E-03
Kr-89		**2.88E+00**

Table 2.0-3R CRN-1 Projected LOCA Bounding Design Basis Accident Radioactive Release
(Sheet 2 of 3)

Radionuclide	LOCA ESPA PPE Worst 2 hr (Ci)	CRN-1 Release - FWLB Pre-Incident Iodine Spike (Ci)[(a)]
La-140	1.32E-01	**6.12E-01**
La-142	4.13E-02	
Mn-54		**2.88E-03**
Mn-56		**1.08E-02**
Mo-99	1.71E+00	5.69E-01
Na-24		**5.90E-03**
Nb-95	1.34E-01	1.22E-01
Nd-147	4.89E-02	
Ni-63		5.97E-06
Np-239	3.16E+00	4.68E-01
P-32		2.37E-04
PR-143	1.17E-01	
Pr-144		4.39E-03
Pu-238	5.94E-04	
Pu-239	7.10E-05	
Pu-240	1.08E-04	
Pu-241	2.64E-04	
Rb-86	2.06E-01	
Rb-89		**2.73E+00**
Rh-103m		**2.95E-02**
Rh-105	8.37E-01	
Rh-106		**4.39E-03**
Ru-103	1.40E+00	2.95E-02
Ru-105	6.27E-01	
Ru-106	4.75E-01	4.39E-03
Sb-127	1.59E+00	
Sb-129	3.38E+00	
Sr-89	7.79E+00	4.39E-03
Sr-90	9.52E-01	2.23E-04
Sr-91	8.01E+00	2.66E+00
Sr-92	5.48E+00	**5.76E+00**
Tc-99m	1.16E+00	5.69E-01
Te-127	1.34E+00	
Te-127m	2.74E-01	
Te-129	1.14E+00	

Table 2.0-3R CRN-1 Projected LOCA Bounding Design Basis Accident Radioactive Release
(Sheet 3 of 3)

Radionuclide	LOCA ESPA PPE Worst 2 hr (Ci)	CRN-1 Release - FWLB Pre-Incident Iodine Spike (Ci)[a]
Te-129m	9.09E-01	5.83E-02
Te-131m	3.32E+00	7.92E-02
Te-132	2.59E+01	1.37E-02
W-187		**1.37E-03**
Xe-131m	1.92E+01	4.51E-04
Xe-133	3.82E+03	4.09E-03
Xe-133m	1.17E+00	2.68E-04
Xe-135	8.53E+02	4.09E-02
Xe-135m	2.08E+00	7.33E-02
Xe-137		**1.36E-01**
Xe-138	5.81E+00	3.80E-01
Y-90	9.55E-03	2.23E-04
Y-91	1.01E-01	5.83E-02
Y-92	6.37E-02	**1.80E+00**
Y-93	9.72E-02	**1.94E-01**
Zn-65		**1.22E-03**
Zr-95	1.32E-01	1.22E-01
Zr-97	1.17E-01	
Total	7.02E+03	6.07E+02

(a)**Bold** text indicates that CRN-1 value exceeds PPE value. The specific radionuclides that exceed ESP PPE are considered to be minor dose contributors with respect to the considered dominant dose contributing radionuclides. The dominant dose contributing radionuclides (e.g., I-131, Cs-137, and H-3) exceed CRN-1 by many orders of magnitude.

Replace Table 2.0-4 with the following:

Table 2.0-4R CRN-1 Projected Annual Normal Gaseous Radioactive Release (Ci/yr)
(Sheet 1 of 3)

Radionuclide	ESPA PPE Total Site Release	CRN-1 Release[a]
Ag-110m	2.14E-03	6.5E-07
Ar-41	5.44E+02	8.8E-03
Ba-140	1.67E-02	1.9E-04
Br-84	1.28E-05	
C-14	1.00E+01	**1.1E+01[c]**
Ce-141	5.68E-03	1.2E-05
Ce-143	1.16E-07	
Ce-144	1.17E-05	2.0E-06
Co-57	1.10E-04	
Co-58	6.90E-02	1.4E-04
Co-60	2.64E-02	3.0E-04
Cr-51	2.17E-02	5.3E-04
Cs-134	6.90E-03	1.8E-05
Cs-136	3.68E-04	9.9E-06
Cs-137	3.26E-02	2.8E-05
Cs-138	1.05E-04	3.0E-06
Cu-64	6.18E-03	1.6E-04
Fe-55	4.01E-03	6.6E-04
Fe-59	9.55E-04	1.6E-04
H-3	1.01E+03	2.6E+01
I-129	8.02E-11	
I-131	2.31E-01	1.4E-02
I-132	1.35E+00	8.3E-02
I-133	1.05E+00	6.5E-02
I-134	2.33E+00	2.4E-01
I-135	1.49E+00	1.3E-01
Kr-83m	1.28E-02	**8.9E+00[d]**
Kr-85	7.20E+02	6.0E+01
Kr-85m	3.39E+02	9.3E-01
Kr-87	3.27E+01	3.2E+00
Kr-88	1.45E+02	3.2E+00
Kr-89	5.00E-07	**3.3E+02[b]**
La-140	1.12E-03	4.4E-05
Mn-54	5.22E-03	3.1E-04

Table 2.0-4R CRN-1 Projected Annual Normal Gaseous Radioactive Release (Ci/yr)
(Sheet 2 of 3)

Radionuclide	ESPA PPE Total Site Release	CRN-1 Release[a]
Mn-56	2.17E-03	1.2E-05
Mo-99	3.68E-02	7.0E-05
Na-24	2.50E-03	3.9E-05
Nb-95	7.50E-03	4.8E-05
Ni-63	1.46E-02	6.6E-07
Np-239	7.35E-03	4.8E-05
P-32	5.68E-04	1.9E-05
Pr-144	1.17E-05	2.3E-09
Rb-88	9.80E-06	
Rb-89	2.67E-05	1.2E-06
Rh-103m	1.48E-08	**5.0E-08[e]**
Rh-106	4.57E-11	**2.8E-07[f]**
Ru-103	2.17E-03	1.2E-05
Ru-106	2.34E-04	2.0E-06
Sb-124	1.12E-04	1.3E-08
Sb-125	3.77E-05	
Sr-89	9.00E-03	2.2E-06
Sr-90	3.60E-03	9.8E-08
Sr-91	6.18E-04	4.6E-05
Sr-92	4.84E-04	2.8E-05
Tc-99m	1.83E-04	6.1E-06
Te-129m	1.35E-04	2.3E-05
Te-131m	4.68E-05	4.1E-06
Te-132	7.13E-05	1.8E-06
W-187	1.17E-04	1.4E-05
Xe-131m	1.67E+03	1.0E+00
Xe-133	2.24E+03	4.3E+01
Xe-133m	1.05E+02	3.1E-02
Xe-135	2.82E+02	3.4E+01
Xe-135m	1.28E+01	**3.2E+01[d]**
Xe-137	3.00E+00	**4.5E+01[b]**
Xe-138	1.14E+01	**6.6E+01[d]**
Y-90	2.84E-05	2.4E-08
Y-91	1.49E-04	2.4E-05
Y-92	3.84E-04	1.1E-05
Y-93	6.86E-04	3.5E-06

**Table 2.0-4R CRN-1 Projected Annual Normal Gaseous Radioactive Release (Ci/yr)
(Sheet 3 of 3)**

Radionuclide	ESPA PPE Total Site Release	CRN-1 Release[(a)]
Zn-65	6.86E-03	1.3E-04
Zr-95	3.00E-03	5.1E-05
Total (without H-3)	6.12E+03	6.4E+02
H-3	1.01E+03	2.6E+01
Total Gaseous Radionuclide Release Activity	7.13E+03	6.6E+02

(a) **Bold** text indicates that CRN-1 value exceeds PPE value.

(b) BWRX-300 short-lived radionuclide exceedances (Kr-89 and Xe-137) are shown to be acceptable because of significant decay during atmospheric transport between the source and offsite receptors.

(c) Based on population dose results presented in ESPA SSAR Table 11.2-9, higher BWRX-300 C-14 radionuclide releases are offset by lower H-3 emissions.

(d) Based on Gaseous Radioactive Effluent Concentrations at the Site Exclusion Area Boundary presented in ESPA SSAR Table 11.3-3 for noble gases, higher BWRX-300 Kr-83m, Xe-135m, and Xe-138 radionuclide releases are offset by lower Ar-41 noble gas emission.

(e) Based on Gaseous Radioactive Effluent Concentrations at the Site Exclusion Area Boundary presented in ESPA SSAR Table 11.3-3, higher BWRX-300 Rh-103m releases are offset by lower Co-60 particulate radionuclide release source term.

(f) Rh-106 increases are negligible contributors to dose because of decay during atmospheric transport to the site boundary

Replace Table 2.0-5 with the following:

Table 2.0-5R CRN-1 Projected Accidental Liquid Radwaste Release Source Term (Ci)
(Sheet 1 of 3)

Radionuclide	ESPA PPE Release	CRN-1 Release[(a)]
Ag-110m	4.31E+02	5.18E-04
Ba-137m	0.00E+00[2]	**5.25E+00**
Ba-140	2.50E+05	5.31E+01
C-14	1.37E-01	0.00E+00
Ce-141	2.36E+05	2.56E+00
Ce-144	2.02E+05	3.81E-01
Co-58	1.21E+03	1.19E-01
Co-60	2.59E+02	2.31E-01
Cr-51	8.04E+03	5.00E-01
Cs-134	3.01E+04	3.44E+00
Cs-136	1.00E+04	2.69E+00
Cs-137	2.45E+04	5.25E+00
Cs-138	2.71E+05	2.56E+02
Cu-64	5.25E-01	**2.44E+00**
Fe-55	2.99E+03	5.18E-01
Fe-59	1.93E+02	1.37E-01
H-3	1.25E+02	1.75E+01
I-129	5.89E-03	0.00E+00
I-130	2.42E+03	0.00E+00
I-131	1.42E+05	1.12E+01
I-132	2.07E+05	1.12E+02
I-133	2.92E+05	8.12E+01
I-134	3.28E+05	3.31E+02
I-135	2.78E+05	1.62E+02
Kr-83m	0.00E+00	0.00E+00
Kr-85	0.00E+00	0.00E+00
Kr-85m	0.00E+00	0.00E+00
Kr-87	0.00E+00	0.00E+00
Kr-88	0.00E+00	0.00E+00
Kr-89	0.00E+00	0.00E+00
La-140	2.58E+05	5.31E+01
Mn-54	7.49E+02	2.50E-01
Mn-56	2.26E+04	9.37E-01
Mo-99	2.65E+05	4.93E+01

**Table 2.0-5R CRN-1 Projected Accidental Liquid Radwaste Release Source Term (Ci)
(Sheet 2 of 3)**

Radionuclide	ESPA PPE Release	CRN-1 Release[(a)]
Na-24	7.68E+01	5.12E-01
Nb-95	5.00E+03	1.06E+01
Ni-63	9.63E+01	5.18E-04
Np-239	2.72E+06	4.06E+01
P-32	1.95E+01	2.06E-02
P-32	1.95E+01	2.06E-02
Pr-143	2.15E+05	0.00E+00
Pr-144	0.00E+00	**3.81E-01**
Rb-89	1.30E+05	2.37E+02
Rh-103m	2.24E+05	2.56E+00
Rh-106	9.16E+04	3.81E-01
Ru-103	2.24E+05	2.56E+00
Ru-106	8.63E+04	3.81E-01
Sb-129	1.18E+02	0.00E+00
Sr-89	1.34E+05	3.81E-01
Sr-90	1.87E+04	1.94E-02
Sr-91	1.71E+05	2.31E+02
Sr-92	1.84E+05	5.00E+02
Tc-99m	2.35E+05	4.93E+01
Te-129m	7.21E+03	5.06E+00
Te-131m	2.73E+04	6.87E+00
Te-132	2.02E+05	1.19E+00
W-187	0.00E+00	**1.19E-01**
Xe-131m	0.00E+00	0.00E+00
Xe-133	0.00E+00	0.00E+00
Xe-133m	0.00E+00	0.00E+00
Xe-135	0.00E+00	0.00E+00
Xe-135m	0.00E+00	0.00E+00
Xe-137	0.00E+00	0.00E+00
Y-90	1.94E+04	1.94E-02
Y-91	1.76E+05	5.06E+00
Y-92	1.86E+05	1.56E+02
Y-93	2.10E+05	1.69E+01
Zn-65	8.50E-05	**1.06E-01**
Zr-95	5.00E+03	1.06E+01

**Table 2.0-5R CRN-1 Projected Accidental Liquid Radwaste Release Source Term (Ci)
(Sheet 3 of 3)**

Radionuclide	ESPA PPE Release	CRN-1 Release[(a)]
Total	8.11E+06	2.43E+03

(a) **Bold** text indicates that CRN-1 value exceeds PPE value. The dose consequence of the postulated Accidental Liquid Radwaste Release (Section 2.4.13.8) is less than the 10 CFR 20.1301 limit of 100 mrem TEDE and therefore the exceedances shown in Table 2.0-5R are acceptable.

Replace Table 2.0-6 with the following:

Table 2.0-6R CRN-1 Projected Annual Normal Liquid Radioactive Release (Ci/yr)
(Sheet 1 of 3)

Radionuclide	ESPA PPE Total Site Release	CRN-1 Release[(a)]
Ag-110	3.48E-08	
Ag-110m	2.66E-02	7.00E-05
Am-241	1.85E-10	
Ba-137m	2.07E-03	
Ba-139	6.16E-08	**2.20E-04**
Ba-140	4.80E-02	4.90E-03
Br-82	7.48E-06	
Br-83	1.41E-05	**6.70E-04**
Br-84	1.01E-03	
Br-85	9.68E-09	
C-14	9.83E-03	
Ce-141	1.58E-04	**3.30E-04**
Ce-143	3.25E-04	1.60E-04
Ce-144	2.99E-03	2.30E-04
Cm-242	3.78E-08	
Cm-244	1.76E-09	
Co-58	5.51E-02	3.40E-03
Co-60	8.21E-03	6.60E-03
Cr-51	1.28E-01	1.20E-02
Cs-134	3.44E-02	2.30E-03
Cs-136	1.17E-02	1.30E-03
Cs-137	4.24E-02	3.50E-03
Cs-138	1.42E-02	
Cu-64	6.72E-03	**7.40E-03**
Fe-55	4.87E-02	1.40E-02
Fe-59	1.19E-02	3.50E-03
H-3	8.85E+02	
I-129	5.04E-09	
I-130	1.85E-05	
I-131	1.66E-01	1.10E-03
I-132	1.32E-01	1.50E-04
I-133	2.76E-01	1.90E-03
I-134	3.91E-02	
I-135	1.64E-01	8.60E-04

Table 2.0-6R CRN-1 Projected Annual Normal Liquid Radioactive Release (Ci/yr)
(Sheet 2 of 3)

Radionuclide	ESPA PPE Total Site Release	CRN-1 Release[(a)]
La-140	4.27E-03	
La-141	8.80E-08	
La-142	1.19E-08	**1.80E-04**
Mn-54	6.53E-02	6.50E-03
Mn-56	1.09E-03	2.00E-04
Mo-99	4.52E-02	2.80E-03
Na-24	8.40E-03	1.90E-03
Nb-95	1.07E-03	**1.20E-03**
Nb-98		**1.00E-05**
Nd-147	1.07E-06	**4.00E-05**
Ni-63	1.84E-01	1.00E-04
Ni-65		**4.00E-05**
Np-239	2.99E-02	2.10E-03
P-32	3.03E-04	**4.70E-04**
Pr-143	6.93E-05	**5.90E-04**
Pr-144	1.69E-03	
Pu-238	2.64E-09	
Pu-239	3.39E-10	
Pu-240	4.27E-10	
Pu-241	1.28E-07	
Rb-86	7.48E-05	
Rb-88	1.49E-02	
Rb-89	6.18E-04	
Rh-103m	4.37E-06	
Rh-105	4.27E-07	
Rh-106	3.74E-07	
Ru-103	2.63E-03	2.70E-04
Ru-105	7.04E-08	**7.90E-04**
Ru-106	3.92E-02	4.80E-04
Sb-124	2.29E-04	
Sb-125	7.92E-09	
Sb-127	4.40E-08	
Sb-129	1.76E-08	
Sr-89	1.67E-04	5.00E-05
Sr-90	1.43E-05	
Sr-91	6.67E-04	**2.00E-03**

Table 2.0-6R CRN-1 Projected Annual Normal Liquid Radioactive Release (Ci/yr)
(Sheet 3 of 3)

Radionuclide	ESPA PPE Total Site Release	CRN-1 Release[(a)]
Sr-92	2.36E-04	**4.80E-04**
Tc-99	1.76E-08	
Tc-99m	2.27E-02	2.80E-03
Te-127	1.28E-05	
Te-127m	5.72E-06	
Te-129	1.65E-04	
Te-129m	6.90E-02	5.00E-04
Te-131	4.05E-05	
Te-131m	1.98E-03	2.10E-04
Te-132	1.32E-01	7.00E-05
Te-134	1.06E-06	
W-187	6.30E-04	**7.40E-04**
Y-90	1.86E-06	
Y-91	1.25E-04	**6.70E-04**
Y-91m	2.67E-05	
Y-92	9.01E-04	**1.50E-03**
Y-93	7.25E-04	1.60E-04
Zn-65	2.11E-02	2.70E-03
Zn-69m		**4.20E-03**
Zr-95	2.20E-03	1.10E-03
Zr-97	4.40E-07	**2.00E-05**
Total Liquid Radionuclide Release Activity	8.87E+02	9.95E-02

(a)**Bold** text indicates that BWRX-300 value exceeds PPE value. The dominant dose contributing radionuclides (e.g., I-131, Cs-137, and H-3) calculated for CRN-1 are exceeded by many orders of magnitude by the corresponding ESPA PPE values.

2.1 Geography and Demography

Section 2.1 of the referenced ESPA SSAR is incorporated by reference with the following supplements.

2.1.1 Site Location and Description

2.1.1.1 Specification of Location

Replace Reference 2.1-10 with Reference 2.1-13

Replace Reference 2.1-12 with Reference 2.1-14

Replace Reference 2.1-16 with Reference 2.1-18

Replace the fourth paragraph with the following:

The CRN Site is approximately 935 acres (ac) and is located on a peninsula formed by a meander in the Clinch River arm of the Watts Bar Reservoir. The approximately 1200-ac CRN Property is bounded on the east, south, and west by the Clinch River arm of the Watts Bar Reservoir, and on the north by the U.S. Department of Energy's Oak Ridge Reservation and Wildlife Management Area. The site is situated on the historical Clinch River Breeder Reactor Project (CRBRP) Site. At the time of the project's cancellation in 1983, preliminary site work was essentially completed for the CRBRP, including all necessary sediment ponds, construction shops, concrete batch plants, the nuclear island excavation, and a foundation for a ringer crane. Approximately three million cubic yards of earth and rock were removed during this excavation. Upon project termination, the main site area of approximately 182 ac was remediated in accordance with the Site Remediation Plan. This included partial backfilling of the nuclear island excavation. The finished elevation for the remediated nuclear island excavation area was set at approximately 814.5 ft.

2.1.2 Exclusion Area Authority and Control

2.1.2.1 Site Area Map

Replace Reference 2.1-13 with Reference 2.1-15

2.1.2.2 Control of Activities Unrelated to Plant Operation

Replace Reference 2.1-13 with Reference 2.1-15

Replace Reference 2.14 with Reference 2.1-16

Replace Reference 2.15 with Reference 2.1-17

2.1.3 Population Distribution

*This section is replaced in its entirety with the following. TVA has updated the 2010 census data in the ESPA with data collected in 2020 (*Reference 2.1-9*). While the numbers have changed slightly, the trends remain the same.*

Population distributions calculated for the CRN Site were derived by an approach that accounts for multiple units at the CRN Site. This approach results in “analytical Exclusion Area Boundaries (EABs)” encompassed by an ellipse with a short axis of 0.326 mi (0.524 kilometers [km]) from the site center point and a long axis of 0.535 mi (0.864 km) from the center point. To maintain consistency with the ESPA Environmental Report (ER) site center point, a 0.535-mi radius was used to establish a buffer for the demographic components of the CPA ER.

Thus, population distributions calculated for the Clinch River Nuclear (CRN) Site were calculated at 0.535 mi (0.864 km) from the site center point. Thus, distance measurements in relation to the CRN Site in this report are made from the buffered site center point.

The population surrounding the site, up to a 50-mi (80 km) radius, was estimated based upon the most recent 2020 U.S. Census Bureau (USCB) decennial census data. The population estimates were distributed into fifteen concentric bands at 0–0.3 mi (0.5 km), 0.3–1 mi (1.6 km), 1–2 mi (3.2 km), 2–3 mi (4.83 km), 3–4 mi (6.44 km), 4–5 mi (8.05 km), 5–6 mi (9.66 km), 6–7 mi (11.3 km), 7–8 mi (12.9 km), 8–9 mi (14.5 km), 9–10 mi (16.1 km), 10–20 mi (32.2 km), 20–30 mi (48.3 km), 30–40 mi (64.4 km), and 40–50 mi (80.5 km) from the buffered site center point. These bands are subdivided into 16 directional sectors, each centered on one of the 16 compass directions and consisting of 22.5 degrees. Population sectors out to 10 mi (16 km) are shown in Figure 2.1-6R, and population sectors out to 50 mi (80 km) are shown in Figure 2.1-7R.

Population projections were obtained from the University of Tennessee’s Boyd Center for Business and Economic Research, the University of Louisville’s Kentucky State Data Center, and the North Carolina Office of State Budget & Management. Projections from each of these sources were available by county; projections for counties in Tennessee extended through 2070 while the population projections for counties in Kentucky and North Carolina were available through 2050. The counties that were used for the population projections are listed in Table 2.1-1. The percentage change in the population of each county in the 50-mi (80 km) radius from 2020 to 2025 was calculated, and the resultant growth rate was used in conjunction with the 2020 census data to produce a growth ratio. Calculations of population growth rates were continued in 5-year increments through the end of the available dataset for each jurisdiction (2050 or 2070). For counties located in Kentucky and North Carolina, the projected growth rate at the end of the available dataset (from 2045 to 2050) was used to calculate subsequent projections through 2070. For counties lacking a positive projected growth rate for the entirety of the licensing period, the population of that county was conservatively held constant (zero percent increase) for the remaining projections through 2070.

Census population counts (provided at the block level) were sorted into each of the directional sectors. Because census block boundaries may be split between sector boundaries, the proportion of each block within a sector was determined. Accordingly, the percentage of the

census block population allocated to each sector corresponded to the percentage of the census block land area in the sector. Finally, the population of each sector was calculated by summing the population of each block or portion of a block located within that sector. The years selected for the projection period represent the 2020 census, and every 10-years thereafter until the anticipated end of normal plant license period (2070).

2.1.3.1 Population Within 10 Miles

Permanent population is projected every 10-years thereafter 2020 until the anticipated end of normal plant license period (2070). The projected permanent and transient population for each sector out to 10 mi (16 km), for the years 2030, 2040, 2050, 2060, and 2070 is shown on Table 2.1-2R. Population for all sectors in the 10-mi (16-km) radius for each projected year is shown in the “Totals” field of Table 2.1-2R. The project population estimated for the year 2020 for the 10-mi (16-km) radius of the buffered site center point is shown in Figure 2.1-6R.

A notable special population source within 10 mi (16 km) of the site is the Oak Ridge National Laboratory (ORNL). ORNL employs a total of 5,772 staff and the Y-12 National Security Complex which employs 7,241 staff (Reference 2.1-1). While these employees were counted by the USCB at their residences, a portion of this workforce, based on commuter data trends, were transient within the 10-mi radius.

2.1.3.2 Population Between 10 and 50 Miles

Figure 2.1-7R shows the estimated regional permanent population for the year 2020 within 50 mi (80 km) of the buffered site center point. The region between 10 and 50 mi (16 and 80 km) includes all or part of the counties listed in Table 2.1-1. The distances defining the sectors are 10 mi (16 km), 20 mi (40 km), 30 mi (48 km), 40 mi (64 km), and 50 mi (80 km).

Knoxville is the largest city within 50 mi, with a 2020 population of 599,356. Smaller cities within the region include Sevierville (23,305), Athens (15,935), Crossville (16,377), La Follette (19,504), Cleveland (71,855), Morristown (60,492), Oneida (4,482), Strawberry Plains (2,739), Dayton (19,937), Fairfield Glade (7,783), Sweetwater (5,374), Harriman-Kingston-Rockwood (21,814), Madisonville (5056), Etowah (3870) and Norris (3,537) (Reference 2.1-10). Many other smaller towns and cities are distributed within the region.

The projected permanent population for each sector in the 10-50 mi (16-80 km) region for the years 2030, 2040, 2050, 2060, and 2070 is shown Table 2.1-3R. Permanent population is projected every 10-years thereafter until the anticipated end of normal plant license period (2070). The total number of people in the 10-50 mi (16-80 km) region for each projected year is shown in the “Totals” field of Table 2.1-3R.

2.1.3.3 Transient Population

Per Regulatory Guide (RG) 4.7, transients are defined as people (other than those just passing through the area) who work, reside part time, or engage in recreational activities but are not permanent residents of the area (Reference 2.1-7). Transient populations are additionally

defined by NUREG-0800, as workers, occupants of schools, hospitals, recreational facilities, and others who do not permanently reside in the area (Reference 2.1-6).

Analysis of transient populations within the 10-mi (16 km) radius included an assessment of major population generators such as employers, tourists and recreational areas, educational facilities, hospitals, national and state parks, and recreation areas. Population attending local public venues are not counted as transients as this is likely to result in over-estimates of transient populations due to the high contribution of attendance at these venues by permanent residents within the 0- to 10-mi (0- to 16-km) area.

Though relatively rural in nature, the region surrounding the site has numerous facilities and recreation areas that contribute to the transient population. Transient data was gathered from the East Tennessee Economic Development Agency, National Center for Education Statistics, Tennessee Department of Human Services, Tennessee Department of Health, Early Site Permit Application (ESPA) Demographic Calculation (Reference 2.1-2), and Google Earth search within the region. This method for collecting transient data provided a more accurate assessment of people visiting the area and a much more precise location of transient contributors. The identified transient facilities that contribute to the transient population in the 0- to 10-mi region are shown in Table 2.1-4R.

RG 1.70 requires that seasonal and daily variations in population and population distribution resulting from land uses such as recreational or industrial should be generally described within the 10-mi and 50-mi (16-km and 80-km) radius (Reference 2.1-5). There are several national and state parks that contribute to the regional transient population (10-50 mi [16-80 km] radius), these parks and visitor counts are detailed in Table 2.1-6R.

2.1.3.3.1 Transient Population Within 10 Miles (16 km)

Table 2.1-4R lists the daily visitors for all transient facilities identified within a 10-mi (16 km) area of the buffered site center point. There were 84 individual transient facilities identified within 10 mi (16 km) of the site that contribute approximately 12,700 daily visitors to the transient population (Table 2.1-5R). The largest transient facilities within 10 mi (16 km) of the site include, the Y-12 National Security Complex, an employer contributed 3,548 transients, Oak Ridge National Laboratory, an employer, contributed 2,831 transients, and Mayor Bob Leonard Park, at full capacity contributed 570 transients (Table 2.1-5R).

2.1.3.3.2 Total Permanent and Transient Populations within 10 Miles (16 km)

Table 2.1-2R summarizes the total permanent and transient population by sector and year of interest within the 0- to 10-mi (0- to 16- km) radius. The estimated permanent and transient population for 2020 in the 10-mi (16 km) radius was approximately 98,127 (Table 2.1-2R). Using calculated county growth rates, the estimated total population for 2070 is 119,055 (Table 2.1-2R).

2.1.3.3.3 Transient Population Between 10-50 Miles (16-80 km)

Transient populations were identified in the 10- to 50-mi (16- to 80- km) radius, in accordance with RG definitions of transient and included a review of employment in major economic centers

and recreation areas. The Knoxville, Tennessee metro area falls within 10 and 50 mi of the CRN Site. The Knoxville, Tennessee economic center includes the counties of Morgan, Roane, Loudon, Blount, Knox, Anderson, Campbell, Union, and Grainger, all of which fall within the 50 mi (80 km) radius. Recreational contributors to the regional total transient population include state and national parks. Table 2.1-6R lists the state and national parks located within the region. Annual visitation reflects data obtained from 2017-2021 depending on the park. The Great Smoky Mountains National Park had the largest transient population, with a recorded annual visitor count of 14,161,548 persons (Reference 2.1-4). There are 18 Tennessee state parks, national parks, and national forests within the 50-mi (80 km) region. The state park with the largest daily visitor estimate is the Hiwassee Ocoee State Park with approximately 1,714 daily visitors. The remaining state parks have daily visitation that ranges from 556 to 54 persons. The remaining national parks within the region are the Big South Fork National River and Recreation Area, Obed Wild and Scenic River, and Manhattan Project National Historic Park. The national forests within the region are Cherokee National Forest, Daniel Boone National Forest, and Nantahala-Pisgah National Forest (Reference 2.1-4). The peak daily visitor estimates for state and national parks within the region that contribute to the total transient population are listed in Table 2.1-6R.

2.1.3.4 Low-Population Zone

For the CRN Site, the low-population zone (LPZ) is defined as a 1 mi (1.6 km) radius from the site center point. Figure 2.1-8R illustrates the LPZ and total permanent and transient population by sector.

According to the USCB 2020 data, there are 112 people living within the LPZ (Table 2.1-8). The distribution of permanent and transient populations within the LPZ in the 16 cardinal directions is provided in Table 2.1-8. There are no transient facilities within the LPZ, however transient facilities within 5 mi (8 km) of the site center point are identified on Figure 2.1-9R (Reference 2.1-8 and Reference 2.1-3).

2.1.3.5 Population Center

Population centers, as defined by 10 Code of Federal Regulations (CFR) 100.3, are densely populated clusters containing more than 25,000 people. To identify population centers within the site region, the USCB census-delineated urban areas were used (Reference 2.1-11). Rather than being based on political boundaries, census-defined urban areas are based largely on population density (Reference 2.1-11). Table 2.1-8 details and Figure 2.1-10 illustrates the urban areas and population centers identified within the 50-mi (80-km) radius. Of the urban areas located within the region, only three met the requirements to be classified as population centers in 2020: Knoxville (599,356 persons), Cleveland (71,855 persons), and Morristown (60,492 persons) Tennessee. The Knoxville urban area is a combination of several smaller cities, including Alcoa, Clinton, Coalfield, Eagleton Village, Farragut, Friendsville, Halls, John Sevier, Karns, Knoxville, Lenoir City, Loudon, Louisville, Maryville, Mascot, Oak Ridge, Oliver Springs, Powell, Rockford, Sevierville, Seymour, Tellico Village, and Wildwood, Tennessee. The Cleveland urban area is a combination of seven smaller cities. The Morristown urban area is a combination of five smaller cities (Reference 2.1-11). The closest urban area to the CRN Site is the

Harriman-Kingston-Rockwood, Tennessee area, located approximately 3.4 mi (5.5 km) west of the CRN Site. RG 4.7 stipulates that the nearest population center must be at least one-and-one-third times the distance from the reactor to the LPZ (Reference 2.1-7). The distance between the site center point to the nearest boundaries of the closest population center are greater than one and one-third times the distance from the site center point to the boundary of the LPZ.

Using the county population projection ratios, the populations of one additional urban area is anticipated to meet the population center criteria by 2070: Sevierville, Tennessee. At the anticipated end of normal plant license period, Knoxville remains the largest population center to the site with an estimated 2070 population of 843,955. However, Harriman—Kingston—Rockwood remains the closest urban area to the site.

The City of Oak Ridge, with a 2020 population of 31,402, is the closest city to the CRN Site that exceeds 25,000 people (Reference 2.1-9). The CRN Site is located within the southern extent of the City of Oak Ridge, with the city's territory primarily extending to the northeast of the CRN Site. The densely populated portions of the City of Oak Ridge are located in these northeast portions. Therefore, densely populated portions of the City of Oak Ridge are located beyond the distance required by RG 4.7. This is further supported by the U.S. Census Bureau, which has delineated the densely populated portions of the City of Oak Ridge as part of the greater Knoxville urban area (Reference 2.1-11).

The transient population is not considered in these calculations because 10 CFR 100.3 defines a population center as "the distance from the reactor to the nearest boundary of a densely populated center containing more than about 25,000 residents." Transient populations by nature are not considered to be part of the resident population.

2.1.3.6 Population Density

RG 4.7 recommends that at the time of the projected initial site approval and within about 5 years thereafter, the population density, including weighted transient population, averaged over any radial distance out to 20 mi, should not exceed 500 persons per square mile (Reference 2.1-7). Additional guidance from RG 1.70 states that the cumulative resident population out to 30 mi should not exceed 500 people per square mile for the estimated initial year of operation (2030) or 1,000 people per square mile at the anticipated end of normal plant license period (2070) (Reference 2.1-5). For the purpose of evaluating projected population densities per the above mentioned RGs, population densities were calculated to a distance of 30-mi (48 km). Transient population within the 0- to 30-mi (0- to 48-km) area were weighted according to the fraction of the time the transient populations were expected to be in the area. The weighted transient population was projected out to 2070 and combined with the residential population resulting in a cumulative resident and weighted transient population. Population densities were derived by taking the cumulative total projected population at a distance divided by the area at that distance for the estimated year of initial operation (2030), five years after (2035), and the anticipated end of normal plant license period (2070). The 2030, 2035, and 2070 cumulative total projected population at a distance of 30-mi is approximately 798,304, 822,782, and 1,000,456, respectively.

The relationship between cumulative population totals and USNRC siting criteria is illustrated in Figure 2.1-11. This blue line on the figure represents the number of people within that distance equivalent to a population density of either 500 or 1000 per mi^2. Cumulative projected populations at those distances are represented by the green line. By comparing the curves, it is evident that cumulative projected populations within 30-mi (80 km) of the CRN Site does not exceed guidance criteria.

2.1.4 References

Replace ESPA SSAR Section 2.1.4 with the following:

2.1-1 East Tennessee Economic Development Agency (ETEDA). Largest Employers. Website: https://www.eteda.org/data-library/major-employers/largest-employers/, accessed January 2023.

2.1-2 ENERCON. Permanent, Transient, and Total Population Projection in Sectors. TVA Clinch River Project Early Site Permit Application. Calculation TVA029-CALC-001. Prepared for Tennessee Valley Authority (TVA), 2013.

2.1-3 IES NCES. Data & Tools. Available at https://nces.ed.gov/datatools/, accessed January 2023.

2.1-4 National Park Service (NPS). Welcome to Visitor Use Statistics. Website: https://irma.nps.gov/Stats/, accessed January 2023.

2.1-5 USNRC. Regulatory Guide 1.70, Revision 3. Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition) part 1. Office of Standards Development. Washington, DC. November 1978.

2.1-6 USNRC. NUREG-0800, Revision 3. U.S. Nuclear Regulatory Commission Standard Review Plan. March 2007.

2.1-7 USNRC, Regulatory Guide 4.7, Revision 3. General Site Suitability Criteria for Nuclear Power Stations, . Washington, DC. March 2014.

2.1-8 Tennessee Department of Health (TDOH). Find Health Care Facilities. Available at https://dhlrapps.health.tn.gov/FacilityListings, accessed January 2023.

2.1-9 U.S. Census Bureau (USCB). “Data.” Available at https://data.census.gov/all, accessed January 2023.

2.1-10 USCB. “Urban Area Criteria for the 2020 Census- Final Criteria,” Federal Register, Vol. 87, No. 57, March 24, 2022.

2.1-11 USCB. “2020 Census Urban and Rural Classification and Urban Area Criteria.” Available at https://www.census.gov/programs-surveys/geography/guidance/geo-areas/urban-rural.html, accessed January 2023

2.1-12 WSP. Clinch River Nuclear Facility Construction Permit Application. Demographic Calculation. Rev. 0. Prepared for Tennessee Valley Authority. August-2023.

2.1-13 U.S. Census Bureau, “American FactFinder.” Available at http://factfinder2.census.gov/faces/nav/jsf/pages/community_facts.xhtml, accessed December 6, 2013.

2.1-14 U.S. Department of Agriculture, “USDA Natural Resources Conservation Service Geospatial Data Gateway.” Available at http://datagateway.nrcs.usda.gov/, accessed January 14, 2015.

2.1-15 U.S. Department of Transportation, Bureau of Transportation Statistics, “National Transportation Atlas Databases 2013,” 2013.

2.1-16 DOE, “Environmental Assessment/Transfer of Land and Facilities within the East Tennessee Technology Park and Surrounding Area, Oak Ridge, Tennessee,” DOE/EA-1640, October 2011.

2.1-17 Science Applications International Corporation, Baseline Environmental Analysis Report for the K-1251 Barge Facility at the Tennessee Technology Park, Oak Ridge Tennessee, BJC/OR-2885, December 2007.

2.1-18 National Geospatial-Intelligence Agency (NGA) and Office of the Deputy Under Secretary of Defense Installations and Environment (ODUSD (I&E)), “U.S. Military Installations, Ranges, and Training Areas as of May 24, 2012.” Available at http://www.acq.osd.mil/ie/bei/disdi/NGA_MAP_SEND_final_12_28_12_smaller.pdf, accessed April 14, 2014.

Rename and Replace Table 2.1-2 with the following:

Table 2.1-2R Projected Resident and Transient Population Distribution for Each Sector within 0-10 Miles, 2020-2070 (Sheet 1 of 6)

Directional Sector/Year	0-0.3 (mile)	0.3-1 (mile)	1-2 (mile)	2-3 (mile)	3-4 (mile)	4-5 (mile)	5-6 (mile)	6-7 (mile)	7-8 (mile)	8-9 (mile)	9-10 (mile)	Total in 0-10 (mile)
NORTH												
2020	0	0	0	0	18	215	444	377	479	693	903	3,129
2030	0	0	0	0	18	215	443	376	478	696	912	3,139
2040	0	0	0	0	18	210	433	368	468	686	903	3,086
2050	0	0	0	0	17	205	423	359	457	674	892	3,029
2060	0	0	0	0	17	201	414	352	448	665	883	2,980
2070	0	0	0	0	17	197	407	346	440	656	876	2,939
NNE												
2020	0	0	0	0	0	13	781	1,420	2,112	2,430	4,351	11,108
2030	0	0	0	0	0	13	780	1,418	2,150	2,509	4,497	11,367
2040	0	0	0	0	0	13	763	1,387	2,151	2,550	4,578	11,442
2050	0	0	0	0	0	13	745	1,355	2,150	2,590	4,656	11,510
2060	0	0	0	0	0	13	730	1,328	2,153	2,631	4,736	11,590
2070	0	0	0	0	0	12	717	1,305	2,161	2,676	4,822	11,693
NE												
2020	2	0	0	22	273	287	655	299	424	2,310	4,542	8,813
2030	2	0	0	22	272	286	661	310	439	2,390	4,699	9,081
2040	2	0	0	22	266	280	655	315	447	2,436	4,789	9,213
2050	2	0	0	21	260	273	648	321	455	2,480	4,877	9,339
2060	2	0	0	21	255	268	643	327	464	2,526	4,966	9,471
2070	2	0	0	21	250	263	640	333	473	2,574	5,062	9,617

Table 2.1-2R Projected Resident and Transient Population Distribution for Each Sector within 0-10 Miles, 2020-2070 (Sheet 2 of 6)

Directional Sector/Year	0-0.3 (mile)	0.3-1 (mile)	1-2 (mile)	2-3 (mile)	3-4 (mile)	4-5 (mile)	5-6 (mile)	6-7 (mile)	7-8 (mile)	8-9 (mile)	9-10 (mile)	Total in 0-10 (mile)
ENE												
2020	5	9	4	7	328	432	246	213	379	704	800	3,126
2030	5	9	4	7	327	431	250	225	414	767	871	3,310
2040	4	9	3	6	320	422	250	234	445	825	937	3,458
2050	4	9	3	6	313	412	250	244	477	886	1,006	3,611
2060	4	9	3	6	306	404	251	254	511	948	1,077	3,774
2070	4	9	3	6	301	396	252	264	547	1,015	1,152	3,949
EAST												
2020	4	13	79	44	52	340	86	338	467	3,542	4,347	9,312
2030	4	13	79	47	59	382	96	369	512	3,861	4,737	10,160
2040	4	12	77	49	64	414	104	398	552	4,153	5,095	10,923
2050	4	12	75	52	69	448	113	427	594	4,458	5,469	11,720
2060	4	12	74	55	75	487	122	458	638	4,774	5,856	12,556
2070	4	12	73	58	82	531	133	492	685	5,107	6,265	13,440
ESE												
2020	3	14	68	328	160	478	435	1,217	1,609	2,286	3,415	10,012
2030	3	14	68	336	180	537	489	1,367	1,807	2,564	3,790	11,157
2040	3	13	66	338	195	582	530	1,483	1,960	2,780	4,098	12,050
2050	3	13	65	339	211	629	573	1,602	2,118	3,003	4,416	12,972
2060	3	13	63	343	230	685	624	1,744	2,305	3,265	4,776	14,050
2070	3	13	62	348	250	746	679	1,899	2,511	3,554	5,166	15,233

Table 2.1-2R Projected Resident and Transient Population Distribution for Each Sector within 0-10 Miles, 2020-2070 (Sheet 3 of 6)

Directional Sector/Year	0-0.3 (mile)	0.3-1 (mile)	1-2 (mile)	2-3 (mile)	3-4 (mile)	4-5 (mile)	5-6 (mile)	6-7 (mile)	7-8 (mile)	8-9 (mile)	9-10 (mile)	Total in 0-10 (mile)
SE												
2020	1	21	58	269	172	641	968	1,762	4,474	6,321	1,410	16,098
2030	1	21	57	273	187	720	1,088	1,980	5,027	7,103	1,585	18,042
2040	1	21	56	272	198	781	1,180	2,147	5,452	7,703	1,719	19,530
2050	1	20	55	270	209	844	1,274	2,320	5,891	8,323	1,857	21,064
2060	1	20	54	270	222	918	1,387	2,525	6,411	9,058	2,021	22,887
2070	1	20	53	271	237	1,001	1,511	2,751	6,984	9,868	2,201	24,898
SSE												
2020	2	229	63	106	271	273	304	177	493	1,376	988	4,282
2030	2	229	62	106	272	306	342	199	554	1,546	1,110	4,728
2040	2	224	61	103	267	332	371	216	600	1,677	1,204	5,058
2050	2	219	60	101	262	358	401	233	649	1,812	1,300	5,397
2060	2	214	58	99	258	390	436	254	706	1,972	1,415	5,805
2070	2	210	57	97	256	424	475	277	769	2,148	1,542	6,258
SOUTH												
2020	6	20	50	80	189	184	144	319	417	495	985	2,891
2030	6	20	50	80	189	191	151	327	432	556	1,107	3,110
2040	5	20	49	78	185	194	155	329	437	604	1,200	3,256
2050	5	19	48	77	181	197	158	330	443	652	1,297	3,407
2060	5	19	47	75	177	202	163	333	452	710	1,411	3,593
2070	5	19	46	74	174	207	169	338	463	773	1,537	3,805

Table 2.1-2R Projected Resident and Transient Population Distribution for Each Sector within 0-10 Miles, 2020-2070 (Sheet 4 of 6)

Directional Sector/Year	0-0.3 (mile)	0.3-1 (mile)	1-2 (mile)	2-3 (mile)	3-4 (mile)	4-5 (mile)	5-6 (mile)	6-7 (mile)	7-8 (mile)	8-9 (mile)	9-10 (mile)	Total in 0-10 (mile)
SSW												
2020	6	23	35	69	138	176	150	167	224	293	418	1,700
2030	6	23	35	69	138	176	150	167	226	300	428	1,717
2040	6	22	34	68	135	172	147	163	222	301	429	1,699
2050	6	22	34	66	131	168	143	159	219	302	430	1,680
2060	6	21	33	65	129	164	140	156	216	305	433	1,668
2070	6	21	32	64	127	161	138	153	214	309	439	1,663
SW												
2020	3	21	51	126	184	231	199	193	167	136	240	1,551
2030	3	21	50	126	184	231	198	192	167	136	240	1,549
2040	3	21	49	123	180	226	194	188	163	133	235	1,515
2050	3	20	48	120	176	221	190	184	159	130	229	1,480
2060	3	20	47	118	172	216	186	180	156	127	225	1,449
2070	3	19	46	115	169	212	182	177	153	125	221	1,424
WSW												
2020	1	31	76	380	407	391	1,040	2,287	1,903	513	651	7,680
2030	1	31	76	380	406	391	1,038	2,283	1,899	512	650	7,667
2040	1	31	74	371	397	382	1,016	2,233	1,858	501	636	7,500
2050	1	30	72	363	388	373	992	2,181	1,815	489	621	7,325
2060	1	29	71	355	380	366	971	2,136	1,777	479	608	7,174
2070	1	29	70	349	373	359	954	2,099	1,746	470	597	7,047

Table 2.1-2R Projected Resident and Transient Population Distribution for Each Sector within 0-10 Miles, 2020-2070 (Sheet 5 of 6)

Directional Sector/Year	0-0.3 (mile)	0.3-1 (mile)	1-2 (mile)	2-3 (mile)	3-4 (mile)	4-5 (mile)	5-6 (mile)	6-7 (mile)	7-8 (mile)	8-9 (mile)	9-10 (mile)	Total in 0-10 (mile)
WEST												
2020	4	61	148	140	577	538	339	786	1,299	1,199	2,301	7,390
2030	4	61	148	140	576	537	339	784	1,297	1,197	2,297	7,378
2040	4	60	144	137	563	525	331	767	1,268	1,171	2,248	7,218
2050	3	58	141	133	550	513	323	749	1,239	1,143	2,195	7,049
2060	3	57	138	131	539	502	317	734	1,213	1,120	2,150	6,904
2070	3	56	136	128	529	493	311	721	1,192	1,100	2,112	6,782
WNW												
2020	10	63	204	392	170	204	195	365	563	1,459	3,251	6,876
2030	10	63	204	391	170	204	194	364	562	1,456	3,246	6,864
2040	10	62	200	382	166	199	190	356	550	1,425	3,176	6,716
2050	10	60	195	373	162	195	186	348	537	1,391	3,101	6,558
2060	10	59	191	366	159	191	182	341	526	1,363	3,038	6,423
2070	9	58	188	359	156	187	179	335	516	1,339	2,984	6,310
NW												
2020	8	39	67	66	36	175	172	272	538	407	103	1,882
2030	8	38	67	65	36	175	172	272	538	408	105	1,884
2040	8	38	65	64	35	171	168	266	527	401	105	1,848
2050	8	37	64	63	34	167	164	260	515	394	105	1,810
2060	8	36	63	61	33	164	161	254	505	387	105	1,777
2070	7	35	61	60	33	161	158	250	496	382	105	1,749

Table 2.1-2R Projected Resident and Transient Population Distribution for Each Sector within 0-10 Miles, 2020-2070 (Sheet 6 of 6)

Directional Sector/Year	0-0.3 (mile)	0.3-1 (mile)	1-2 (mile)	2-3 (mile)	3-4 (mile)	4-5 (mile)	5-6 (mile)	6-7 (mile)	7-8 (mile)	8-9 (mile)	9-10 (mile)	Total in 0-10 (mile)
NNW												
2020	0	1	25	33	12	258	334	286	179	378	773	2,279
2030	0	1	25	33	12	258	334	285	181	388	793	2,310
2040	0	1	24	32	12	252	327	279	180	390	798	2,295
2050	0	1	24	31	12	246	319	273	179	391	800	2,274
2060	0	1	23	31	12	241	312	267	177	392	803	2,259
2070	0	1	23	30	11	237	307	262	176	394	805	2,247
TOTALS												
2020	56	546	927	2,061	2,987	4,837	6,492	10,476	15,727	24,542	29,477	98,127
2030	56	545	925	2,075	3,026	5,053	6,724	10,918	16,682	26,389	31,068	103,462
2040	55	533	905	2,047	3,002	5,157	6,813	11,130	17,283	27,735	32,148	106,807
2050	53	520	884	2,016	2,975	5,263	6,902	11,345	17,896	29,117	33,251	110,224
2060	52	510	866	1,995	2,964	5,411	7,039	11,642	18,658	30,720	34,503	114,359
2070	51	501	850	1,981	2,965	5,590	7,212	12,000	19,528	32,489	35,887	119,055

Source: Reference 2.1-12

Note: Population estimates are rounded to the nearest whole number, accounting for potential discrepancies between the total value and the sum of the individual distance bands.

Rename and Replace Table 2.1-3 with the following:

Table 2.1-3R Projected Resident Population Distribution for Each Sector within 10-50 Miles, 2020-70
(Sheet 1 of 4)

Directional Sector/Year	10-20 (mile)	20-30 (mile)	30-40 (mile)	40-50 (mile)	Total in 10-50 (mile)
NORTH					
2020	1,215	464	5,127	6,964	13,769
2030	1,239	462	5,123	6,901	13,725
2040	1,241	453	5,042	6,736	13,472
2050	1,241	441	4,928	6,535	13,146
2060	1,243	429	4,802	6,374	12,849
2070	1,247	417	4,676	6,212	12,552
NNE					
2020	4,693	10,875	24,138	8,906	48,611
2030	4,856	10,957	23,545	8,707	48,064
2040	4,949	10,862	22,524	8,349	46,684
2050	5,039	10,746	21,413	7,960	45,159
2060	5,131	10,642	20,352	7,588	43,712
2070	5,230	10,557	19,340	7,233	42,359
NE					
2020	34,935	21,076	11,540	10,846	78,397
2030	36,156	22,092	11,813	10,993	81,055
2040	36,856	22,832	12,135	11,164	82,987
2050	37,540	23,584	12,464	11,312	84,900
2060	38,230	24,354	12,799	11,459	86,842
2070	38,974	25,173	13,148	11,605	88,901
ENE					
2020	77,240	175,222	46,898	21,346	320,705
2030	84,011	190,950	50,926	22,660	348,547
2040	90,190	205,387	54,648	23,505	373,730
2050	96,638	220,459	58,531	24,207	399,835
2060	103,318	236,080	62,555	24,983	426,935
2070	110,357	252,533	66,791	25,831	455,512

Table 2.1-3R Projected Resident Population Distribution for Each Sector within 10-50 Miles, 2020-70
(Sheet 2 of 4)

Directional Sector/Year	10-20 (mile)	20-30 (mile)	30-40 (mile)	40-50 (mile)	Total in 10-50 (mile)
EAST					
2020	91,783	68,555	40,216	53,578	254,132
2030	100,024	74,736	44,615	60,060	279,436
2040	107,577	80,235	48,342	65,516	301,671
2050	115,457	85,876	52,060	70,940	324,332
2060	123,623	91,720	56,018	76,805	348,166
2070	132,225	97,864	60,194	83,018	373,301
ESE					
2020	20,588	80,704	7,818	4,137	113,248
2030	22,525	88,130	8,544	4,647	123,847
2040	24,126	93,778	9,099	5,078	132,080
2050	25,715	99,011	9,614	5,508	139,848
2060	27,390	104,421	10,147	5,972	147,931
2070	29,156	110,042	10,702	6,464	156,363
SE					
2020	8,112	13,823	240	2,941	25,115
2030	9,012	15,115	258	2,883	27,269
2040	9,702	16,097	272	2,876	28,946
2050	10,389	17,014	284	2,875	30,563
2060	11,173	17,975	297	2,877	32,321
2070	12,023	18,979	311	2,878	34,191
SSE					
2020	14,432	7,305	1,664	660	24,061
2030	16,111	7,734	1,755	698	26,299
2040	17,380	7,965	1,801	736	27,882
2050	18,679	8,156	1,838	773	29,447
2060	20,218	8,373	1,880	812	31,282
2070	21,913	8,602	1,923	854	33,292
SOUTH					
2020	10,569	16,606	12,675	2,964	42,814
2030	11,515	17,515	13,285	3,111	45,426
2040	12,165	17,974	13,617	3,188	46,944
2050	12,802	18,340	13,902	3,248	48,291
2060	13,551	18,754	14,233	3,316	49,854
2070	14,370	19,192	14,599	3,391	51,551

Table 2.1-3R Projected Resident Population Distribution for Each Sector within 10-50 Miles, 2020-70
(Sheet 3 of 4)

Directional Sector/Year	10-20 (mile)	20-30 (mile)	30-40 (mile)	40-50 (mile)	Total in 10-50 (mile)
SSW					
2020	5,513	10,697	30,468	15,386	62,065
2030	5,870	11,173	31,688	16,249	64,981
2040	6,083	11,449	32,437	16,907	66,877
2050	6,282	11,696	33,135	17,546	68,659
2060	6,520	11,989	33,973	18,266	70,748
2070	6,783	12,320	34,946	19,074	73,123
SW					
2020	2,679	5,404	10,624	24,014	42,721
2030	2,676	5,657	11,159	25,313	44,805
2040	2,620	5,774	11,475	26,237	46,106
2050	2,559	5,843	11,726	27,096	47,224
2060	2,508	5,912	11,988	28,011	48,419
2070	2,464	5,978	12,253	28,970	49,665
WSW					
2020	4,606	5,981	4,295	9,943	24,824
2030	4,598	6,290	4,561	10,561	26,010
2040	4,498	6,505	4,721	10,831	26,556
2050	4,393	6,709	4,848	10,920	26,869
2060	4,303	6,932	4,979	10,944	27,157
2070	4,227	7,169	5,113	10,935	27,444
WEST					
2020	10,188	3,493	30,085	7,362	51,127
2030	10,214	3,838	33,054	8,081	55,186
2040	10,025	4,009	34,528	8,446	57,008
2050	9,818	4,132	35,589	8,711	58,251
2060	9,648	4,295	36,998	9,056	59,997
2070	9,509	4,472	38,523	9,427	61,931
WNW					
2020	2,570	8,416	7,424	9,484	27,895
2030	2,630	9,217	8,094	10,168	30,110
2040	2,637	9,612	8,414	10,508	31,171
2050	2,636	9,895	8,640	10,753	31,924
2060	2,639	10,272	8,944	11,054	32,909
2070	2,642	10,679	9,274	11,372	33,967

Table 2.1-3R Projected Resident Population Distribution for Each Sector within 10-50 Miles, 2020-70
(Sheet 4 of 4)

Directional Sector/Year	10-20 (mile)	20-30 (mile)	30-40 (mile)	40-50 (mile)	Total in 10-50 (mile)
NW					
2020	5,261	3,313	2,969	11,004	22,548
2030	5,400	3,401	3,040	11,234	23,075
2040	5,430	3,420	3,039	11,159	23,048
2050	5,444	3,429	3,026	11,017	22,915
2060	5,464	3,442	3,019	10,913	22,837
2070	5,483	3,453	3,012	10,818	22,767
NNW					
2020	4,533	722	6,417	5,990	17,662
2030	4,652	738	6,439	6,012	17,841
2040	4,678	739	6,360	5,935	17,712
2050	4,690	737	6,239	5,819	17,485
2060	4,708	736	6,105	5,691	17,240
2070	4,724	735	5,968	5,563	16,989
TOTALS					
2020	298,917	432,658	242,597	195,523	1,169,695
2030	321,490	468,006	257,900	208,278	1,255,674
2040	340,158	497,089	268,456	217,171	1,322,873
2050	359,323	526,068	278,236	225,220	1,388,847
2060	379,666	556,325	289,089	234,121	1,459,200
2070	401,324	588,164	300,773	243,644	1,533,906

Source: Reference 2.1-12

Note: Population estimates are rounded to the nearest whole number, accounting for potential discrepancies between the total value and the sum of the individual distance bands.

Rename and Replace Table 2.1-4 with the following:

Table 2.1-4R Transient Populations within 10 Miles of the CRN Site for All Transient Facilities (Sheet 1 of 5)

Transient Facility Type	Facility Name [4]	Direction	Distance Band (mi)[1]	County	Transient Population[2]		
					Unweighted[3]	Weighted[3]	Weight Factor[5]
Parks/Recreation	Soaring Eagle	SSE	0.3-1	Roane	207	207	1
Parks/ Recreation	East Tennessee Technology Park Overlook	NNW	1-2	Roane	25	7	0.27
Major Employer	Crete Carrier Corporation	ESE	2-3	Roane	221	60	0.27
Major Employer	Michael Dunn Center	WSW	2-3	Roane	192	52	0.27
Major Employer	Oak Ridge National Laboratory	ENE	2-3	Roane	7	2	0.27
Major Employer	Oak Ridge National Laboratory	NE	2-3	Roane	22	6	0.27
Parks/Recreation	K-25 History Center	NNW	2-3	Roane	32	9	0.27
Parks/Recreation	Crosseyed Cricket	SE	2-3	Roane	86	86	1
Major Employer	Oak Ridge National Laboratory	ENE	3-4	Roane	328	89	0.27
Major Employer	Oak Ridge National Laboratory	NE	3-4	Roane	273	74	0.27
Lodging[3]	Motel 6 Kingston, Tennessee	WSW	3-4	Roane	75	75	1
Parks/Recreation	Four Season Campground	WSW	3-4	Roane	92	92	1
Parks/Recreation	Melton Hill Dam Recreation Area	E	4-5	Loudon	180	49	0.27
Parks/Recreation	Melton Hill Dam Campground	E	4-5	Loudon	112	112	1
Major Employer	Oak Ridge National Laboratory	ENE	4-5	Roane	432	117	0.27
Major Employer	Oak Ridge National Laboratory	NE	4-5	Roane	287	77	0.27
Daycare	Kiddie Kingdom Daycare	SE	4-5	Loudon	12	3	0.27
Major Employer	Oak Ridge National Laboratory	ENE	5-6	Anderson	242	65	0.27
School	North Middle School	ESE	5-6	Loudon	21	6	0.27
Major Employer	Oak Ridge National Laboratory	NE	5-6	Roane	655	117	0.27
Daycare	Clubhouse Central	SE	5-6	Loudon	9	2	0.27

Table 2.1-4R Transient Populations within 10 Miles of the CRN Site for All Transient Facilities (Sheet 2 of 5)

Transient Facility Type	Facility Name [4]	Direction	Distance Band (mi)[1]	County	Transient Population[2]		
					Unweighted[3]	Weighted[3]	Weight Factor[5]
School	Eatons Elementary School	SE	5-6	Loudon	22	6	0.27
Assisted Living Facility	Jamestowne Assisted Living	WSW	5-6	Roane	11	3	0.27
School	Kingston Elementary School	WSW	5-6	Roane	23	6	0.27
Major Employer	Oak Ridge National Laboratory	ENE	6-7	Anderson	131	35	0.27
Major Employer	Oak Ridge National Laboratory	NE	6-7	Anderson	297	80	0.27
Assisted Living Facility	Patriot Hill Assisted Living	NNE	6-7	Roane	4	1	0.27
Lodging[3]	Comfort Inn	SE	6-7	Loudon	80	80	1
Lodging[3]	Days Inn by Wyndham Lenoir City	SE	6-7	Loudon	80	80	1
Lodging[3]	Econo Lodge	SE	6-7	Loudon	75	75	1
Lodging[3]	Holiday Inn Express & Suites Lenoir City (Knoxville Area)	SE	6-7	Loudon	80	80	1
Lodging[3]	King's Inn Motel	SE	6-7	Loudon	75	75	1
Lodging[3]	Fairfield Inn & Suites by Marriott Knoxville Lenoir City/I-75	SE	6-7	Loudon	80	80	1
Lodging[3]	Super 8 by Wyndham Kingston	W	6-7	Roane	75	75	1
School	Cherokee Middle School	WSW	6-7	Roane	16	4	0.27
Daycare	Kiddie Korner Daycare and Preschool	WSW	6-7	Roane	30	8	0.27
Major Employer	Oak Ridge National Laboratory	NE	7-8	Anderson	118	32	0.27
School	Dyllis Springs Elementary School	NNE	7-8	Roane	12	3	0.27
Assisted Living Facility	Morning Pointe of Lenoir City	SE	7-8	Loudon	11	3	0.27
Assisted Living Facility	The Lantern at Morning Pointe Alzheimer's Center	SE	7-8	Loudon	8	2	0.27
Daycare	Head Start Harrison Hills	SE	7-8	Loudon	10	3	0.27

Table 2.1-4R Transient Populations within 10 Miles of the CRN Site for All Transient Facilities (Sheet 3 of 5)

Transient Facility Type	Facility Name [4]	Direction	Distance Band (mi)[1]	County	Transient Population[2]		
					Unweighted[3]	Weighted[3]	Weight Factor[5]
Hospital	Fort Loudon Medical Center (Employees)	SE	7-8	Loudon	224	60	0.27
Hospital	Fort Loudon Medical Center (Patients)	SE	7-8	Loudon	71	71	1
Lodging	Hampton Inn Lenoir City	SE	7-8	Loudon	75	75	1
School	Lenoir City High School	SE	7-8	Loudon	30	8	0.27
School	Lenoir City Intermediate School	SE	7-8	Loudon	20	5	0.27
School	Loudon County Technology Center	SE	7-8	Loudon	2	1	0.27
Major Employer	Y-12 National Security Complex	NE	7-8	Anderson	273	74	0.27
Parks/Recreation	Kingston Fall Fest (Kingston County Fair)	W	7-8	Roane	3	1	0.27
Parks/Recreation	Kingston City Park	W	7-8	Roane	77	21	0.27
School	Roane County High School	W	7-8	Roane	13	3	0.27
Assisted Living Facility	Sycamore Trace	WSW	7-8	Roane	8	2	0.27
Parks/Recreation	Fort Southwest Point	WSW	7-8	Roane	285	77	0.27
Parks/Recreation	Mayor Bob Leonard Park	E	8-9	Knox	570	154	0.27
Major Employer	Kimberly-Clark	SSE	8-9	Loudon	443	120	0.27
Major Employer	Malibu Boats	SSE	8-9	Loudon	343	93	0.27
School	Linden Elementary School	NNE	8-9	Anderson	16	4	0.27
Major Employer	Oak Ridge National Laboratory	NE	8-9	Anderson	39	10	0.27
Assisted Living Facility	River Oaks Place - Lenoir City	SE	8-9	Loudon	6	2	0.27
School	Highland Park Elementary School	SE	8-9	Loudon	14	4	0.27
Daycare	Jo's Kids	SE	8-9	Loudon	5	1	0.27
Parks/Recreation	Lee Russell Recreation Complex	SE	8-9	Loudon	217	59	0.27

Table 2.1-4R Transient Populations within 10 Miles of the CRN Site for All Transient Facilities (Sheet 4 of 5)

Transient Facility Type	Facility Name [4]	Direction	Distance Band (mi)[1]	County	Transient Population[2]		
					Unweighted[3]	Weighted[3]	Weight Factor[5]
Daycare	Kiddie Korner of Lenoir City	SE	8-9	Loudon	5	1	0.27
Daycare	LCES Leaps and Bounds	SE	8-9	Loudon	6	2	0.27
School	Lenoir City Elementary School	SE	8-9	Loudon	21	6	0.27
Major Employer	Y-12 National Security Complex	NE	8-9	Anderson	1,721	465	0.27
Parks/Recreation	Riley Creek Campground	WSW	8-9	Roane	82	82	1
Daycare	Primrose School of Farragut	E	9-10	Knox	25	7	0.27
Daycare	Trinity Learning Center	E	9-10	Knox	17	5	0.27
Major Employer	Hubbell Lenoir City	SE	9-10	Loudon	172	46	0.27
Major Employer	Monterey Mushroom	S	9-10	Loudon	281	76	0.27
Major Employer	Total Transportation of Mississippi	S	9-10	Loudon	149	40	0.27
Daycare	Blossom Center for Childhood Excellence	NE	9-10	Anderson	10	3	0.27
Daycare	Heritage Preschool	NE	9-10	Anderson	12	3	0.27
Lodging[3]	Days Inn by Wyndham Oak Ridge Knoxville	NE	9-10	Anderson	80	80	1
Lodging[3]	Double Tree by Hilton Hotel Oak Ridge - Knoxville	NE	9-10	Anderson	80	80	1
Lodging[3]	Hampton Inn Oak Ridge Knoxville	NE	9-10	Anderson	80	80	1
Lodging[3]	Holiday Inn Express & Suites Knoxville West - Oak Ridge	NE	9-10	Anderson	80	80	1
Parks/Recreation	Oak Ridge Baseball Complex	NE	9-10	Anderson	293	79	0.27
Daycare	La Petite Academy of Oak Ridge	NE	9-10	Anderson	13	4	0.27
Daycare	Oak Ridge Schools Preschool	NE	9-10	Anderson	29	8	0.27
School	Robertsville Middle School	NE	9-10	Anderson	25	7	0.27
Daycare	Brookside Learning Center	NNE	9-10	Anderson	7	2	0.27

Table 2.1-4R Transient Populations within 10 Miles of the CRN Site for All Transient Facilities (Sheet 5 of 5)

Transient Facility Type	Facility Name [4]	Direction	Distance Band (mi)[1]	County	Transient Population[2]		
					Unweighted[3]	Weighted[3]	Weight Factor[5]
School	Norwood Elementary School	NNE	9-10	Anderson	9	2	0.27
School	Norwood Middle School	NNE	9-10	Anderson	7	2	0.27
School	Willow Brook Elementary School	NNE	9-10	Anderson	16	4	0.27
School	Coalfield High School	NNW	9-10	Morgan	16	4	0.27
Daycare	Young at Heart Learning Academy II	SE	9-10	Loudon	12	3	0.27
School	Bowers Elementary School	W	9-10	Roane	18	5	0.27
Hospital	Roane Medical Center (Employees)	W	9-10	Roane	196	53	0.27
Hospital	Roane Medical Center (Patients)	W	9-10	Roane	82	82	1
School	Midtown Elementary School	W	9-10	Roane	10	3	0.27
School	Harriman High School	WNW	9-10	Roane	11	3	0.27
School	Harriman Middle School	WNW	9-10	Roane	10	3	0.27
Parks/Recreation	Hooray for Harriman	WNW	9-10	Roane	14	4	0.27
Parks/Recreation	May Day	WNW	9-10	Roane	8	2	0.27
Daycare	Trenton Street Baptist Church Daycare	WNW	9-10	Roane	12	3	0.27
Major Employer	Y-12 National Security Complex	NE	9-10	Anderson	1,554	420	0.27

Source: Reference 2.1-12

Notes:

1. Distance band from buffered site center point.
2. Estimated transient populations are assumed to be representative of 2020 population.
3. Population values rounded to the nearest whole number.
4. Weight Factor details the ratio of hours spent within the 0–10-mile area based on 24 hours in a day.
5. Facilities listed more than once cover multiple directional sectors based divided within the 16 compass points.

Rename and Replace Table 2.1-5 with the following:

Table 2.1-5R Transient Populations within 10 Miles of the CRN Site by Transient Facility Type

Transient Facility[3]	Count	Unweighted Transients[1,2]	Weighted Transient[1,2]
Assisted Living	6	47	13
Daycares	16	214	58
Hospitals	2	573	267
Lodging	13	1,012	1,012
Major Employers	9	8,177	2,208
Parks and Recreation	17	2,347	1,056
Schools	21	329	89
Grand Total	84	12,700	4,702

Source: Reference 2.1-12

Notes:

1. Estimated transient populations are assumed to be representative of 2020 population.
2. Population values rounded to the nearest whole number.

Rename and Replace Table 2.1-6 with the following:

Table 2.1-6R Major Public Recreation Areas within 10-50 Miles of the CRN Site

Recreation Area	State	Daily Visitation	Annual Visitation[1]
Big Ridge State Park	TN	556	203,000
Big South Fork National River and Recreation	TN	2,287	834,724
Cherokee National Forest	TN	7,877	2,875,000
Cove Lake State Park	TN	292	106,401
Cumberland Mountain State Park	TN	368	134,442
Cumberland Trail State Park	TN	74	27,108
Daniel Boone National Forest	KY	5,142	1,877,000
Fort Loudon State Historic Park	TN	105	38,273
Frozen Head State Park	TN	125	45,611
Great Smoky Mountains National Park	TN	38,799	14,161,548
Hiwassee Ocoee State Park	TN	1,714	625,751
Indian Mountain State Park	TN	229	83,726
Manhattan Project National Historic Park[3]	TN	71	26,063
Nantahala-Pisgah National Forest (Nantahala National Forest)[2]	SC	19,995	7,298,000
Norris Dam State Park	TN	54	19,829
Obed National and Scenic River	TN	679	247,826
Seven Island State Birding Park	TN	99	36,039

Source: Reference 2.1-12

Notes:

1. Annual visitation for multiple source years ranging from 2017-2021
2. Annual visitation value is a combination of Nantahala National Forest and Pisgah National Forest
3. Portions of the Manhattan Project National Historic Park fall within the 0-10 mile radius

Rename and Replace Table 2.1-7 with the following:

Table 2.1-7R 2020 Resident and Transient Population Distribution within the Low Population Zone

Direction	Combined Resident and Transient Population (0-1 mile)
North	0
NNE	0
NE	2
ENE	7
East	7
ESE	6
SE	5
SSE	5
South	10
SSW	11
SW	5
WSW	2
West	13
WNW	22
NW	16
NNW	0
Total	112

Source: Reference 2.1-12

Note: Population counts are rounded to the nearest whole number, accounting for a potential discrepancy between the total value and the sum of the individual distance bands.

Add new Table 2.1-8:

Table 2.1-8 Urban Areas and Population Centers within 50 Miles of the CRN Site

Urban Area	2020 Resident Population	2070 Projected Resident Population	Distance from CRN-1 Site (miles)
Athens, TN	15,935	18,277	26.5
Cleveland, TN*	71,855	101,194	44.7
Crossville, TN	16,377	20,970	33.7
Dayton, TN	9,937	12,128	40.8
Etowah, TN	3,870	4,439	35.0
Fairfield Glade, TN	7,783	9,966	28.0
Harriman--Kingston--Rockwood, TN	21,814	20,017	3.4
Knoxville, TN*	599,356	843,955	4.3
La Follette, TN	19,504	16,407	22.8
Madisonville, TN	5,056	5,844	23.1
Morristown, TN*	60,492	76,581	48.9
Norris, TN	3,537	3,942	23.3
Oneida, TN	4,482	4,126	35.4
Sevierville, TN**	23,305	36,504	42.4
Strawberry Plains, TN	2,739	3,616	39.0
Sweetwater, TN	5,374	6,208	17.9
Urban Area Totals	871,416	1,184,174	

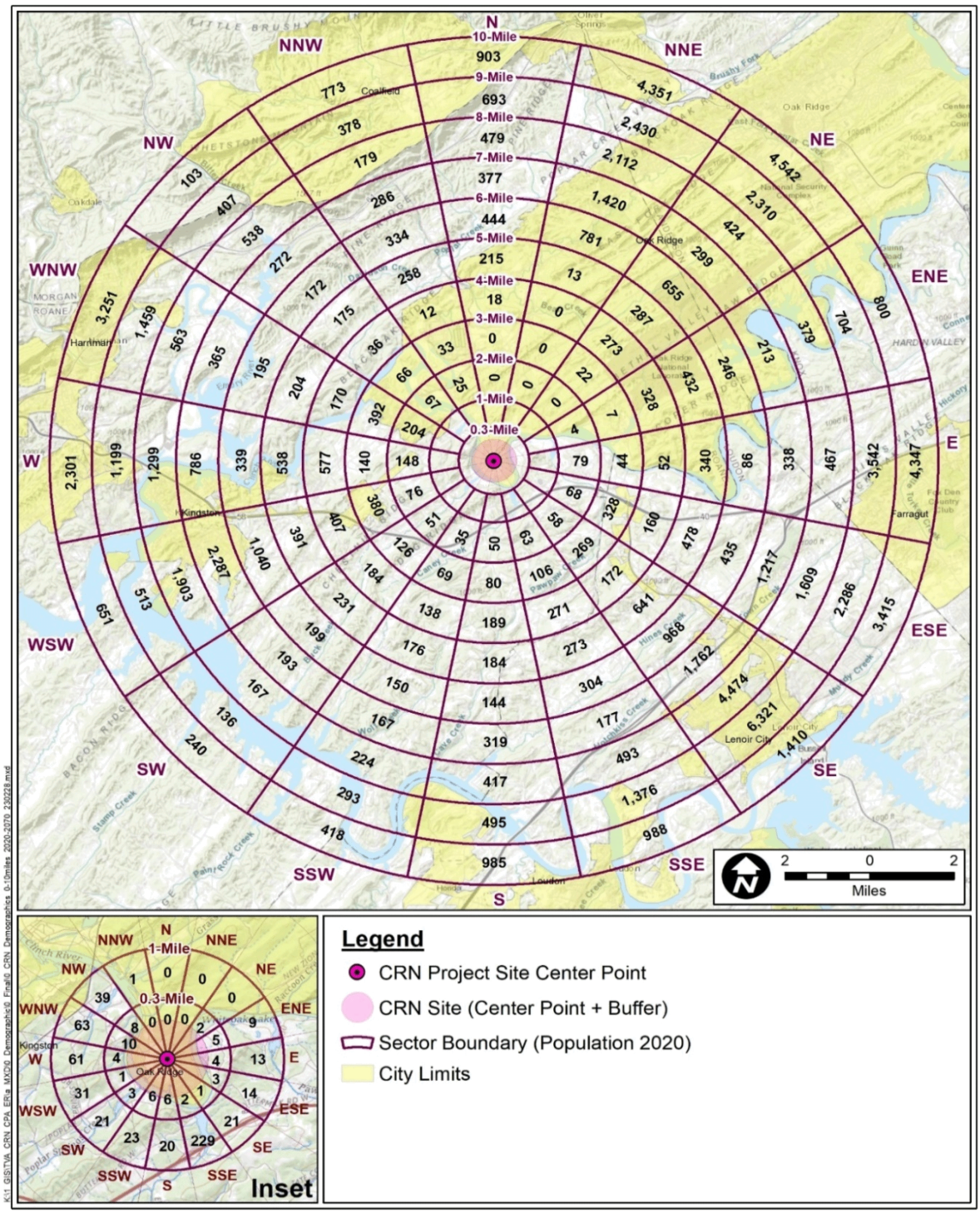
Source: Reference 2.1-12

Notes:

* Urban Areas meeting the requirements to be considered a Population Center.

** Urban Area meeting the requirements to be considered a Population Center in 2070.

Replace Figures 2.1-6 through 2.1-9 with the following: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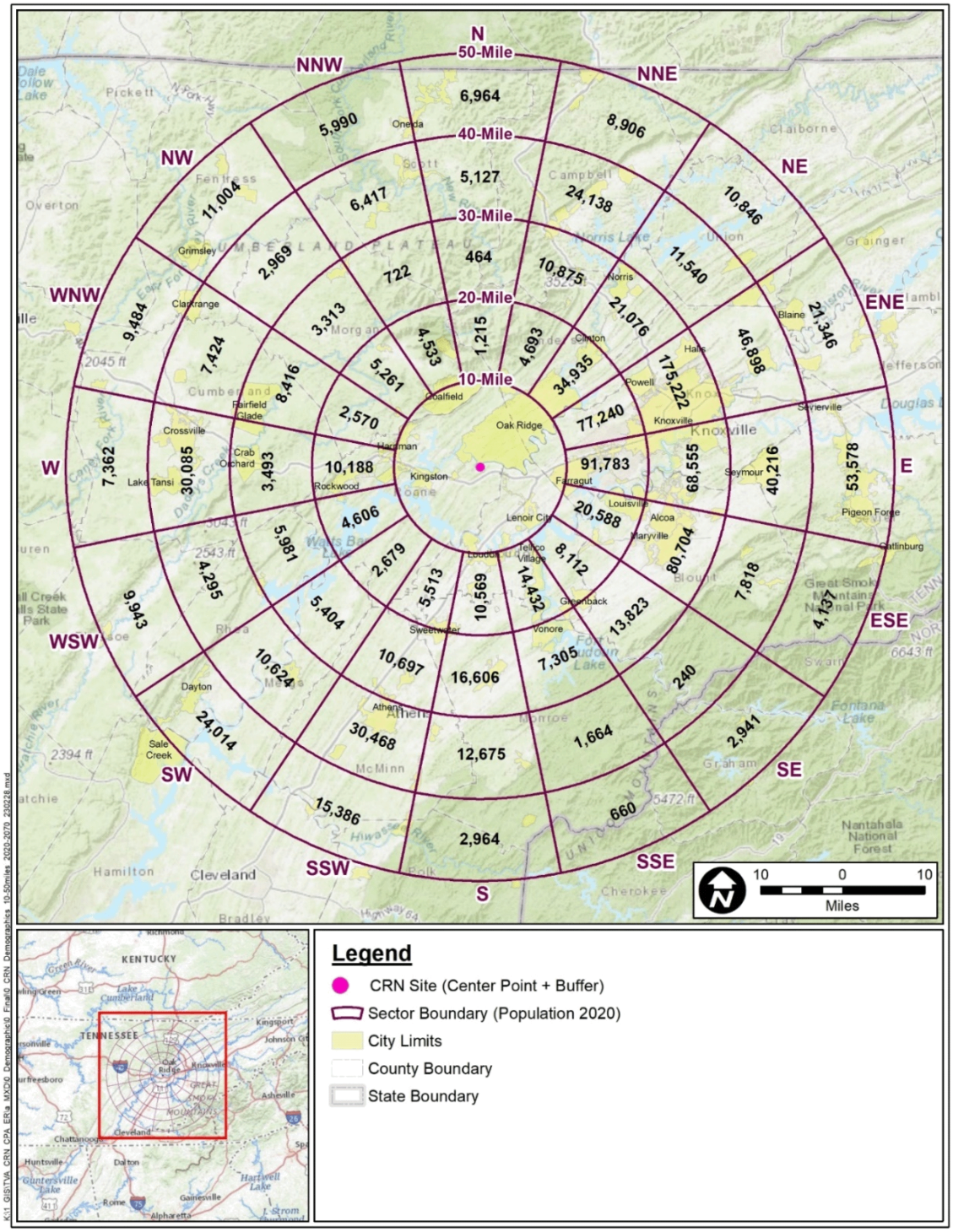

Figure 2.1-7R Resident Population Distribution within 10-50 Miles, 2020 Population

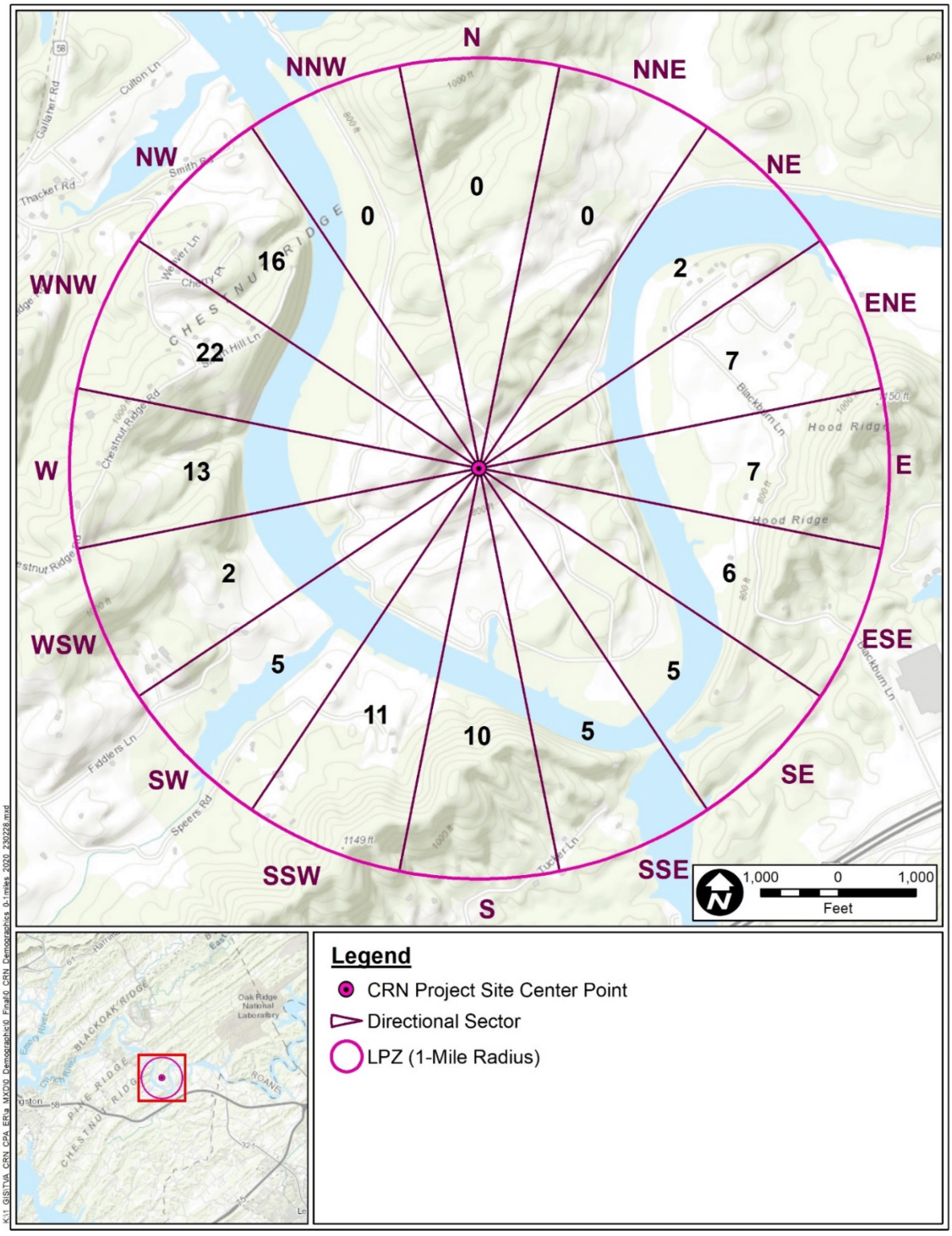

Figure 2.1-8R 2020 Resident and Transient Population Distribution within the Low Population Zone

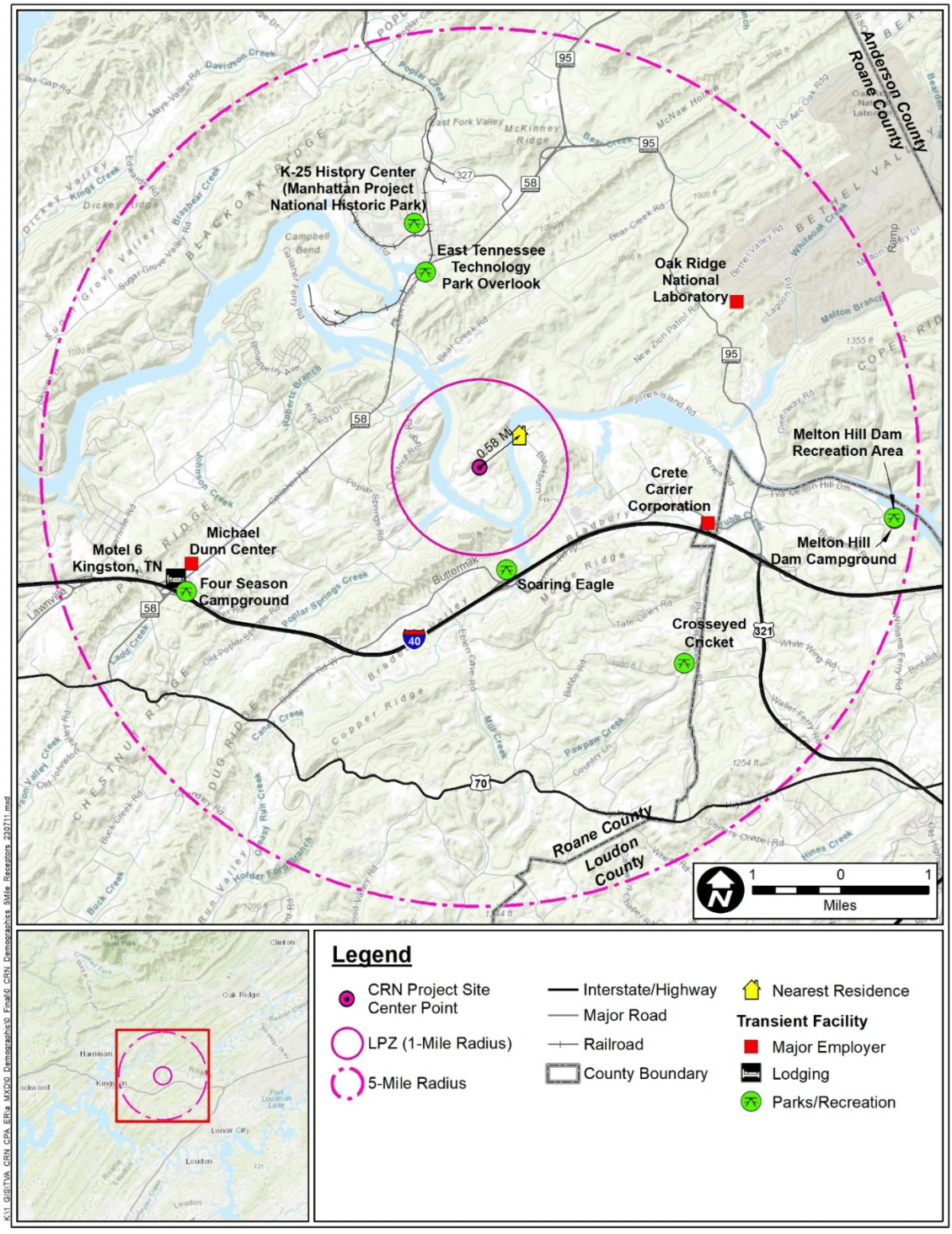

Figure 2.1-9R Transient Facilities and Nearest Residence within 5 Miles of the CRN Project Site Center Point

Add new Figures 2.1-10 and 2.1-11:

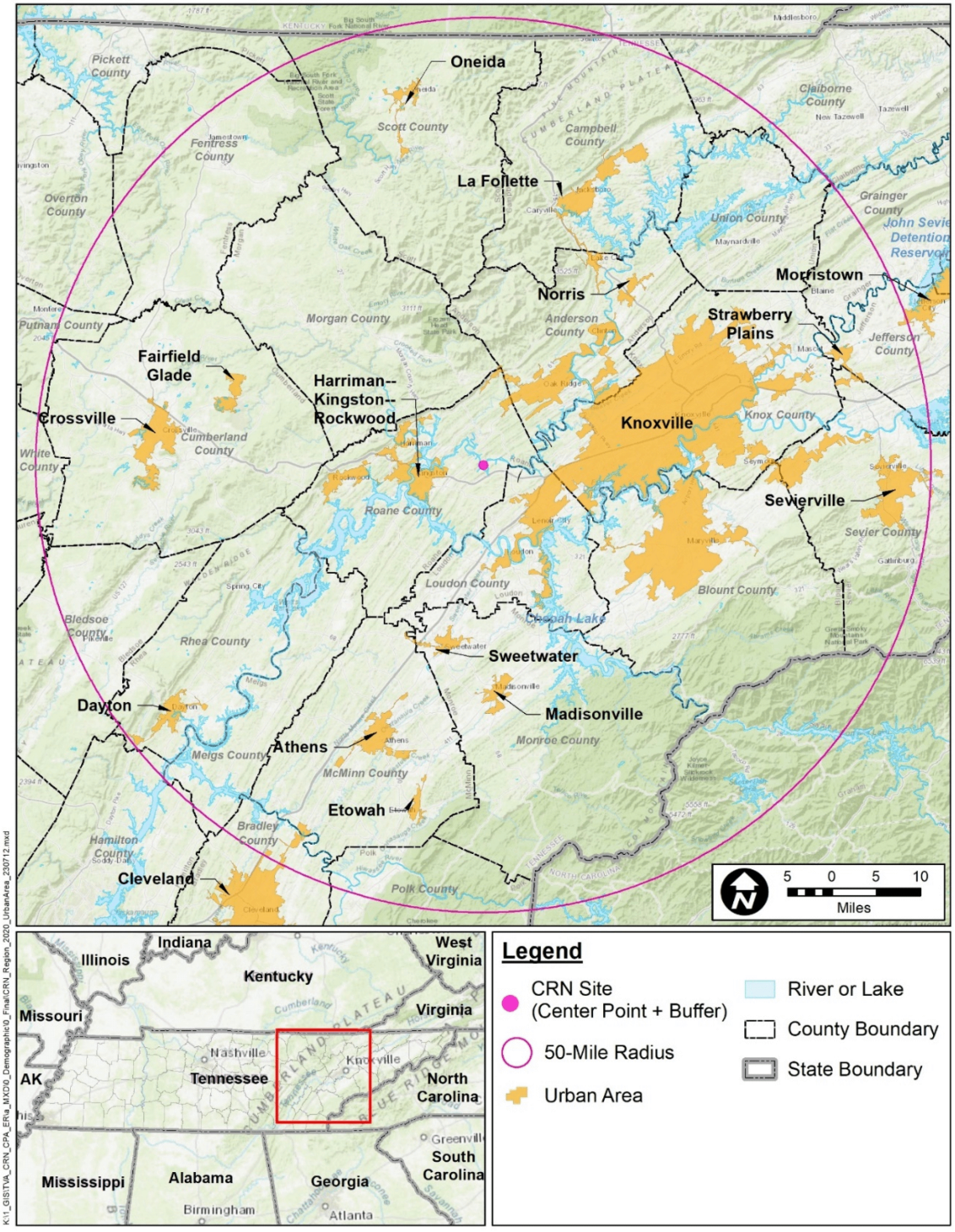

Figure 2.1-10 Urban Areas within 50 Miles of the CRN Site

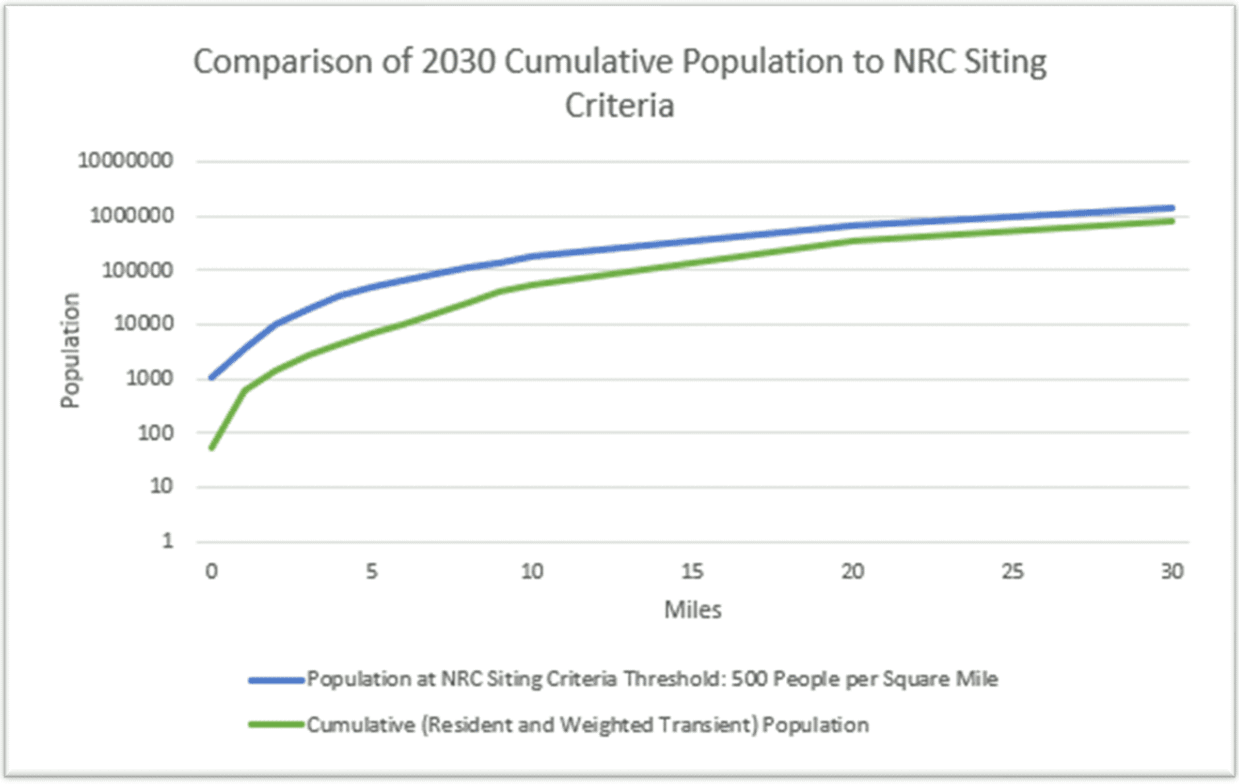

Figure 2.1-11 (Sheet 1 of 3) Comparison of 2030 Cumulative Population to USNRC Siting Criteria

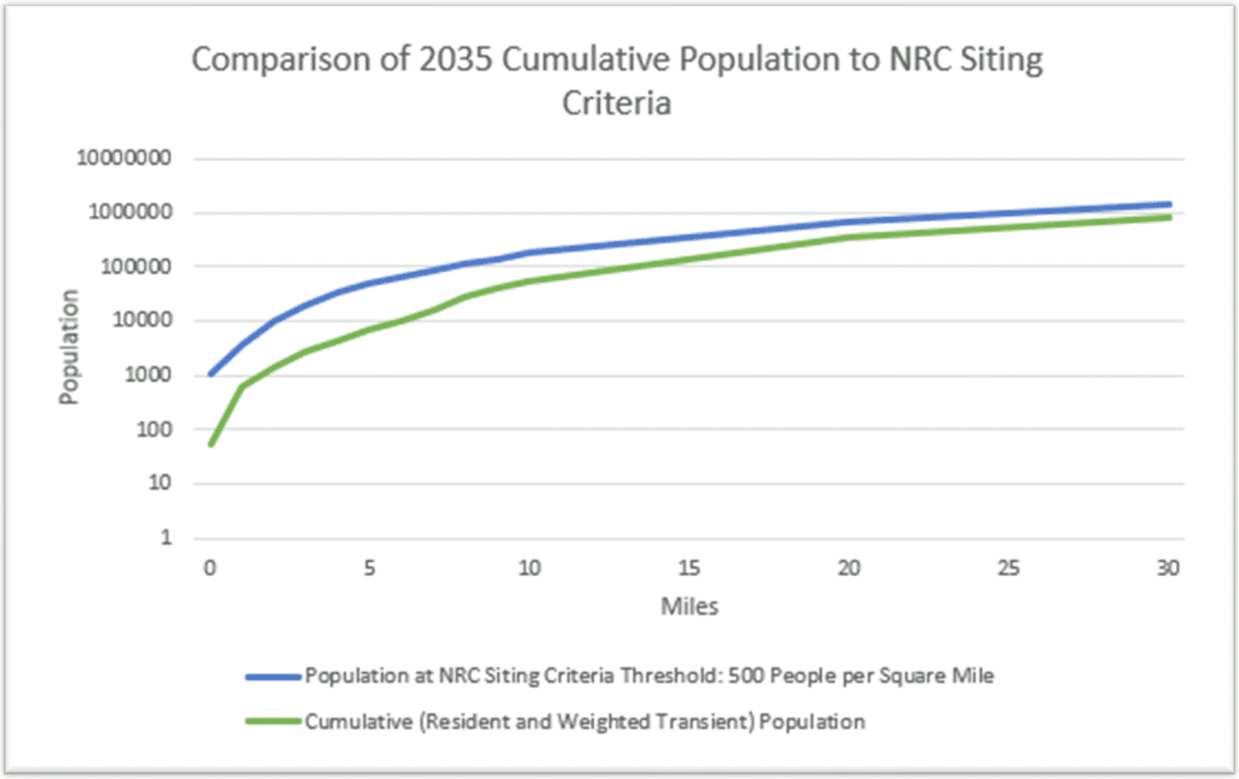

Figure 2.1-11 (Sheet 2 of 3) Comparison of 2035 Cumulative Population to USNRC Siting Criteria

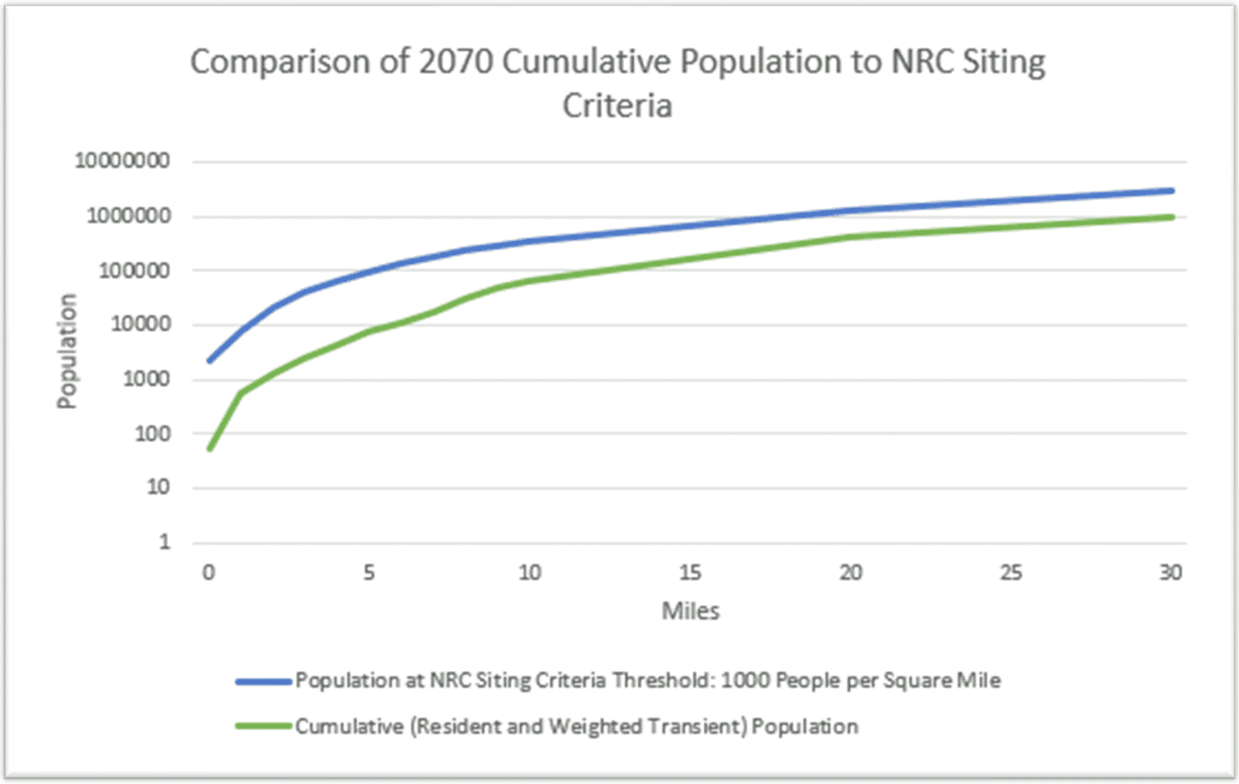

Figure 2.1-11 (Sheet 3 of 3) Comparison of 2070 Cumulative Population to USNRC Siting Criteria

2.2 Nearby Industrial, Transportation and Military Facilities

Section 2.2 of the referenced ESPA SSAR is incorporated by reference with the following supplements.

Replace the first sentence in ESPA SSAR Section 2.2 with the following:

The purpose of this section is to establish whether the effects of potential accidents in the CRN Site vicinity from present and projected industrial, transportation, and military installations and operations should be used as design-basis events for plant design parameters related to the selected accidents. Facilities and activities within the 5-mile (mi) vicinity of the Clinch River Nuclear (CRN) Site were considered to meet the guidance in NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition.*

2.2.1 Locations and Routes

Delete the first bullet of the first paragraph of ESPA SSAR Section 2.2.1.

Replace the second paragraph to the end of ESPA SSAR Section 2.2.1 with the following:

An investigation of the identified potential external hazard facilities and operations (Figure 2.2-1R) within the 5-mi vicinity of the CRN Site identified the following significant industrial facilities for assessment (Reference 2.2-11).

- Diversified Scientific Services
- Duracap Asphalt Paving
- East Tennessee Technology Park
- Energy Solutions Services – Bear Creek Facility
- Harrison Construction – Gallaher Concrete Plant
- Knoxville PDC - Volkswagen Group of America
- NAC Philotechnics – Horizon Center
- Oak Ridge National Laboratory – Battelle
- Oak Ridge National Laboratory – UCOR
- TVA Melton Hill Hydro Plant
- Weigel's #58

An evaluation of the identified major transportation routes within the 5-mi vicinity of the CRN Site (Figure 2.2-2) identified one navigable waterway, one major highway, four major roads, one minor rail line, and two natural gas pipelines for assessment (Reference 2.2-1, Reference 2.2-2 and Reference 2.2-3):

- Clinch River arm of Watts Bar Reservoir (Reference 2.2-20 through Reference 2.2-22 and Figure 2.2-2)
- Interstate 40 (I-40) (Reference 2.2-27 and Figure 2.2-2)

- Tennessee State Highway 1 (TN 1)/US11-70, and Tennessee State Highways 58 (TN 58), 95 (TN 95), and 327 (TN 327) (Reference 2.2-27)
- Heritage Railroad Corporation Railway (Reference 2.2-1 and Reference 2.2-3)
- Two active natural gas transmission pipelines: East Tennessee Natural Gas Pipeline 1 (East) and Pipeline 2 (North) (Reference 2.2-5)

An evaluation of nearby industrial facilities and transportation routes beyond the 5-mi vicinity of the CRN Site was undertaken. In going beyond 10 mi of the CRN Site, particular attention was paid to identifying whether any facilities contain highly toxic, highly volatile chemicals with Risk Management Program (RMP) calculated endpoint distances of at least 25 mi (Reference 2.2-6). The following industrial facilities located beyond the 5-mi vicinity of the site were identified for assessment (Reference 2.2-11 and Reference 2.2-28):

- Diversified Scientific Services
- Duracap Asphalt Paving
- East Tennessee Technology Park
- Energy Solutions Services – Bear Creek Facility
- Harrison Construction – Gallaher Concrete Plant
- Knoxville PDC – Volkswagen Group of America
- NAC Philotechnics – Horizon Center
- Oak Ridge National Laboratory – Battelle
- Oak Ridge National Laboratory – UCOR
- TVA Melton Hill Hydro Plant
- Weigel's #58

The evaluation of transport routes beyond the 5-mi vicinity of the CRN Site identified both rail transport and roads. Rail transport within 5 to 10 mi of the site includes two major and one minor rail line. At closest approach, the two major rail lines operated by Norfolk Southern are 7 mi or more from the CRN Site. In addition to being farther than 5 mi, these rail lines are within complex intervening terrain and therefore accident scenarios on those lines are not evaluated further. The minor rail line, operated by EnergySolutions Heritage Railroad Corporation, is an extension of the EnergySolutions Heritage Railroad Corporation Railway identified within the 5-mi vicinity of the CRN Site. This rail spur is used by Carbon Rivers to transport materials for recycling wind turbine blades (Reference 2.2-25). Wind turbine blades do not pose a threat to the CRN Site due to their physical properties. Solids have a vapor pressure sufficiently low such that the formation of a vapor cloud is not likely. That is, the air dispersion hazard of the material is not a likely exposure route. Nor is the solid material considered explosive. Therefore, this railway is not considered further in this evaluation.

Roads identified beyond the 5-mi vicinity of the CRN Site included I-75 and other major roads. I-75 and other major roads located between 5 and 10 mi of the CRN Site are not considered further given that I-40 was identified within the 5-mi vicinity of the site, where plausible chemicals may be transported, and would, therefore, bound an analysis of roads located farther away.

There are no identified roads, railways or navigable waterways at distances greater than 10 mi that are significant potential hazards.

In summary, the evaluation identified several industrial facilities, one major highway, four major roads, and two natural gas pipelines that are significant enough to be considered for further review (Figure 2.2-1R and Figure 2.2-2). These include:

Industrial Facilities

- Facilities are identified in Table 2.2-1R

Transport Routes

- Clinch River arm of Watts Bar Reservoir
- I-40
- TN 1/US11-70, TN 58, TN 95, and TN 327
- Energy Solutions Railway
- East Tennessee Natural Gas Pipeline 1 (East) and Pipeline 2 (North)

Airports and Airways

Figure 2.2-3R illustrates the following identified airports and airway routes within 10 mi of the CRN Site (Reference 2.2-7 and Reference 2.2-8):

- Big T
- Wolf Creek
- Cox Farm
- Will A Hildreth Farm
- Meadowlake
- Oak Ridge Airport (Under Development)
- Federal Airways V16, J46 and Q65

The Oliver Springs and Fergusons Flying Circus airports are within 10 to 15 mi of the CRN Site (Reference 2.2-7).

2.2.2 Descriptions

Replace ESPA SSAR Section 2.2.2 with the following:

Descriptions of the industrial, transportation, and military facilities located in the vicinity of the CRN Site and identified in Section 2.2.1 are provided in the subsequent subsections in accordance with SRP 2.2.1-2.2.2.

2.2.2.1 Descriptions of Facilities

Replace ESPA SSAR Section 2.2.2.1 with the following:

The facilities identified for further review are listed in Table 2.2-1R. Table 2.2-1R provides a concise description of each facility, including its primary function and major products.

2.2.2.2 Description of Products and Materials

Replace ESPA SSAR Section 2.2.2.2 with the following:

A more detailed description of offsite chemicals associated with each of the above facilities is provided in the following subsections. This description includes information about the products and materials regularly manufactured, stored, used or transported in the CRN Site vicinity. In keeping with the guidance of RG 1.78, chemicals stored or situated at distances greater than the 5-mi vicinity of the proposed CRN Site are not considered unless they are determined to potentially have a significant impact on the proposed CRN Site. The U.S. Environmental Protection Agency's (EPA) Envirofacts/Enviromapper database was queried for facilities and sent to the Military Department of Tennessee in a request for the Superfund Amendments and Reauthorization Act (SARA) Title III, Tier II reports for the queried facilities (References 2.2-10 and 2.2-11).

In addition to the request submitted to the Military Department of Tennessee, the RMP database was queried to obtain reports for facilities within a 25-mi radius of the CRN Site (Reference 2.2-28). The facilities identified herein were determined to have a possible significant impact on the CRN Site; therefore, further analysis of facilities other than these is not required.

2.2.2.2.1 Offsite Chemicals

Replace ESPA SSAR Section 2.2.2.2.1 with the following:

The chemicals stored at the offsite facilities identified in Subsection 2.2.2.1 are detailed in Table 2.2-2R and Table 2.2-5R. Table 2.2-2R identifies the offsite facility in which each chemical is used and the maximum stored quantity reported. Table 2.2-5R provides the disposition for all the offsite chemicals identified for evaluation. The offsite chemicals with the potential to be toxic, flammable, and/or explosive were evaluated for possible effects on the safe operation or shutdown of the CRN Site.

2.2.2.2.2 Industrial Facilities

Replace the first paragraph of ESPA SSAR Section 2.2.2.2.2 with the following:

ORNL (Battelle)

ORNL is located in Oak Ridge, TN, approximately 4.3 mi northeast of the CRN Site power block area. ORNL conducts research and development relating to national energy and security issues and employs approximately 6000 employees (Reference 2.2-17).

Delete paragraphs 2 and 3 from ESPA SSAR Section 2.2.2.2.2.

TVA Kingston Fossil Plant

Replace first sentence of the 4th paragraph of ESPA SSAR Section 2.2.2.2.2 with the following:

TVA – Kingston Fossil Plant is located in Kingston, TN, approximately 7.7 mi west of the CRN Site power block area.

Delete paragraphs 5 through 7 from ESPA SSAR Section 2.2.2.2.2.

Oak Ridge Water Treatment Plant

Replace the first sentence of the 8th paragraph of ESPA SSAR Section 2.2.2.2.2 with the following:

The Oak Ridge WTP is located in Oak Ridge, TN, approximately 10 mi northeast of the CRN Site power block area.

Delete paragraphs 9 through 15 of ESPA SSAR Section 2.2.2.2.2

Delete paragraphs 17 through 19 of ESPA SSAR Section 2.2.2.2.2

Insert the following after the last paragraph of ESPA SSAR Section 2.2.2.2.2:

A. E. Staley Manufacturing / Tate & Lyle Manufacturing

A.E. Staley Manufacturing and Tate & Lyle Manufacturing are the same company and operate a facility located in Loudon, TN, approximately 11 miles south of the CRN Site. This facility develops ingredients for food products and beverages (Reference 2.2-54). The facility uses sulfur dioxide as part of the processes.

City of Maryville Wastewater Treatment Plant

The City of Maryville Wastewater Treatment Plant (Reference 2.2-55) is located approximately 24 miles east of the CRN Site. The facility uses chlorine as a disinfectant.

City of Oak Ridge Municipal Pool

The City of Oak Ridge Municipal Pool is located approximately 10.8 miles northeast of the CRN Site. The municipal pool is provided for recreational purposes (Reference 2.2-56). The pool uses chlorine as a disinfectant.

Custom Foods of America

Custom Foods of America is located approximately 23.6 miles east of the CRN Site. This facility develops food products for restaurants and convenience stores (Reference 2.2-57). The facility uses anhydrous ammonia as part of the processes.

Diversified Scientific Services

Diversified Scientific Services is located approximately 3.4 miles west of the CRN Site. This facility provides radioactive waste treatment services (Reference 2.2-58). The facility uses sodium hydroxide as part of the processes.

Duracap Asphalt Paving

Duracap Asphalt Paving is located approximately 2.3 miles southeast of the CRN Site. This facility provides construction materials for asphalt paving, grading, drainage and site utilities (Reference 2.2-59).

East Tennessee Mechanical Contractors Water Treatment Plant

East Tennessee Mechanical Contractors WTP (Reference 2.2-60) is located approximately 10.5 miles northeast of the CRN Site. The company treats water in support of the work processes. The facility uses chlorine as a disinfectant as part of water treatment.

East Tennessee Technology Park

East Tennessee Technology Park (Reference 2.2-61 and Reference 2.2-62) is currently undergoing a clean-up / remediation effort. The technology park is located approximately 3 miles north of the CRN Site. The facility uses gasoline during the clean-up / remediation efforts.

Energy Solutions Services – Bear Creek Facility

The Energy Solutions Services – Bear Creek Facility is located on Bear Creek Road, approximately 1.6 miles north of the CRN Site. The facility processes radioactive waste (Reference 2.2-63). The facility uses nitric acid and propane as part of the processes.

First Utility District of Knox County Water Treatment and Wastewater Treatment Facilities

The First Utility District of Knox County operates a water treatment facility (Reference 2.2-64) and a wastewater treatment facility (Reference 2.2-65). The water treatment facility is located approximately 14.2 miles east of the CRN Site. The wastewater treatment facility is located approximately 13.4 miles east of the CRN Site. Both facilities use chlorine as part of the treatment processes.

Food Lion – Delhaize Distribution Center

The Food Lion – Delhaize Distribution Center is located approximately 20 miles northeast of the CRN Site. The distribution center provides food to the grocery stores in the vicinity (Reference 2.2-66). The distribution center uses anhydrous ammonia.

Fourth Creek Wastewater Treatment Plant

Knoxville has two wastewater treatment plants (Reference 2.2-67). The Fourth Creek wastewater treatment plant is located approximately 22 miles east of the CRN Site. The facility uses chlorine as part of the treatment process.

Harrison Construction – Gallaher Concrete Plant

Harrison Construction operates the Gallaher Concrete Plant (Reference 2.2-68). The Gallaher Concrete Plant is located approximately 3 miles west of the CRN Site. This facility provides construction materials, including aggregate and ready mix concrete.

Knoxville PDC - Volkswagen Group of America

Volkswagen Group of America is a distribution service (Reference 2.2-82). The facility is located approximately 1 mile southeast of the CRN Site. Electric cars at the facility contain lead and sulfuric acid.

Linde Inc.

Linde Inc. is a local gas and equipment supplier (Reference 2.2-69). The facility is located approximately 11 miles south of the CRN Site. Linde Inc. stores anhydrous ammonia and carbon dioxide.

NAC Philotechnics – Horizon Center

NAC Philotechnics operates the Horizon Center. The Horizon Center is located approximately 4.9 miles north of the CRN Site. This facility provides radioactive and mixed waste treatment services (Reference 2.2-70).

ORNL - UCOR

UCOR is involved with deactivation and demolition work at ORNL (Reference 2.2-71). The facility is located approximately 4.9 miles north of the CRN Site. Nitric acid is used as part of the processes.

Primary Products Ingredients Americas, LLC

Primary Products Ingredients is located in Loudon, TN, approximately 11 miles south of the CRN Site. This facility develops ingredients for food products and beverages (Reference 2.2-72). The facility uses anhydrous ammonia as part of the processes.

Processed Food Corporation

Processed Food Corporation is located approximately 23.6 miles east of the CRN Site. The facility develops food products for restaurants and convenience stores (Reference 2.2-73). The facility uses anhydrous ammonia as part of the processes.

Tellico Area Service System Water Treatment Plant

The Tellico Area Service System WTP (Reference 2.2-74) is located approximately 21.8 miles south of the CRN Site. The facility uses chlorine as part of the treatment process.

Tennessee Valley Authority - Melton Hill Hydro Plant

TVA operates the Melton Hill Hydro Plant (Reference 2.2-75). The Melton Hill Hydro Plant is located approximately 4.7 miles east-southeast of the CRN Site. The facility uses chlorine as part of the treatment process.

U.S. Foodservice - Knoxville

U.S. Foodservice provides food distribution to restaurants (Reference 2.2-76). The facility is located approximately 23.3 miles southeast of the CRN Site. The facility uses anhydrous ammonia.

Weigel's #58

Weigel's is a convenience store that dispenses fuel and other commodities (Reference 2.2-77). Store #58 is located approximately 3.7 miles west of the CRN Site. Weigel's stores gasoline, diesel and propane.

West Knox Utility District

West Knox Utility District operates a water treatment facility (Reference 2.2-78) and a wastewater treatment facility (Reference 2.2-79). The water treatment facility is located approximately 9.1 miles northeast of the CRN Site. The wastewater treatment facility is located approximately 14.4 miles northeast of the CRN Site. Both facilities use chlorine as part of the treatment processes.

Yale Security Inc.

Yale Security manufacturers commercial locks and hardware at a facility in Lenoir City (Reference 2.2-80), located approximately 9.5 miles southeast of the CRN Site. The facility uses chlorine as part of the manufacturing processes.

As provided in RG 1.78, chemicals stored or situated at distances greater than 5 mi from the CRN Site do not need to be considered because if a release occurs at such a distance, atmospheric dispersion will dilute and disperse the incoming plume to such a degree that either toxic limits will never be reached or there would be sufficient time for the control room operators to take appropriate action. Although several of the facilities are located greater than 5 mi from the

CRN Site, some of these were identified for further analysis because of their use of anhydrous ammonia, chlorine or sulfur dioxide. According to 40 CFR 68, anhydrous ammonia, chlorine and sulfur dioxide are highly toxic/highly volatile chemicals with greater toxic endpoints and requires an offsite consequence analysis with a worst-case release scenario for RMPs under the Clean Air Act.

The chemicals stored at these facilities identified for possible analysis are presented in Table 2.2-2R. The disposition of hazards associated with these chemicals is summarized in Table 2.2-5R and the subsequent analysis of these chemicals is addressed in Subsection 2.2.3 For facilities located more than 5 miles from the CRN Site, Table 2.2-5R only list those chemicals identified as very hazardous substances per 40 CFR 68.130; such as ammonia, chlorine and sulphur dioxide.

The transportation route used for shipping and receiving supplies at the facilities may vary, and unless a material is prohibited on a route, there are no restrictions that would prevent the driver from taking another route. The I-40 corridor is identified as the most significant and closest highway to the CRN Site. Therefore, it is conservatively assumed that the chemicals stored at these facilities are transported along the I-40 corridor, in close proximity to the CRN Site, and are included in the evaluation of potential accidents associated with nearby transport routes.

2.2.2.3 Description of Pipelines

Insert the following after the first paragraph of ESPA SSAR Section 2.2.2.3

Subsequent to the SSAR, Spectra Energy was acquired by Enbridge Inc. and operates under this name. (Reference 2.2-5)

2.2.2.4 Description of Waterways

Replace the 2nd paragraph of ESPA SSAR Section 2.2.2.4 with the following:

Significant waterborne transport in the CRN Site vicinity is only possible on the Clinch River arm of the Watts Bar Reservoir. Annual waterborne commerce data compiled by the United States Army Corps of Engineers' Waterborne Commerce Statistics Center, for the period of 2015 through 2021, indicate that there was inconsequential shipping on the river, that is there was no transport of hazardous materials (e.g., chemicals and related products, petroleum, ordinance, etc.) that could pose a threat to operations at the CRN Site (Reference 2.2-20 through Reference 2.2-22). As a result of this finding, waterborne shipping did not warrant further consideration in determining bounding accident scenarios involving transport of hazardous materials in the CRN Site vicinity.

2.2.2.5 Description of Highways

Replace the 3rd sentence of the first paragraph of ESPA SSAR Section 2.2.2.5 with the following:

According to the Tennessee Department of Transportation (TDOT) Traffic Count Database System (TCDS) the average daily traffic for I-40 near the CRN Site was 52,212 vehicles in 2022. (Reference 2.2-2).

Replace the 3rd sentence of the second paragraph of ESPA SSAR Section 2.2.2.5 with the following:

Average daily traffic data for TN 58 and TN 95 was obtained from the TDOT TCDS (Reference 2.2-2). For 2022, the average daily traffic reported for TN 58 was 10,812 vehicles. For 2022, the average daily traffic for TN 95 was 6382 vehicles.

Replace the 4th sentence of the 2nd paragraph of ESPA SSAR Section 2.2.2.5 with the following:

While several industrial facilities are located at the East Tennessee Technology Park (ETTP) off Bear Creek Road, as detailed in Section 2.2.2.2, a search of the SARA Title III, Tier II reports and EPA's Envirofacts/Enviromapper database revealed that one facility stores hazardous materials that could potentially impact the CRN Site (i.e., Energy Solutions Services Bear Creek Facility) (Reference 2.2-11). Bear Creek Valley is about 1.5 mi from the CRN Property on the side of Chestnut Ridge that bounds the property to the north (Reference 2.2-1).

Delete the 4th and 5th sentences in the 3rd paragraph of ESPA SSAR Section 2.2.2.5.

Replace the 4th sentence of the 4th paragraph of ESPA SSAR Section 2.2.2.5 with the following:

Information obtained regarding offsite facilities took into account SRP 2.2.1-2.2.2 guidance to consider all identified facilities and activities within 5 mi of the nuclear plant and include facilities and activities at greater distances depending on their significance.

Replace Table 2.2-3 with Table 2.2-3R in the 4th paragraph of ESPA SSAR Section 2.2.2.5.

2.2.2.6 Description of Railroads

Replace from the 3rd sentence to the end of the 2nd paragraph in ESPA SSAR Section 2.2.2.6 with the following:

Materials transported on this rail line consist of shipments of wind turbine blades (Reference 2.2-25) which do not pose a threat to the CRN Site due to their physical properties. As presented in Section 2.2.2.1, these materials do not pose a significant threat to the site due to the physical properties of the materials; therefore, accidents from the transport of hazardous materials in the vicinity of the CRN Site by rail are not considered further.

2.2.2.7 Description of Airports, Aircraft and Airway Hazards

Section 2.2.2.7 is reformatted to add separate sub-sections for the different types of aircraft and airway hazard.

Replace ESPA SSAR Section 2.2.2.7 with the following:

NUREG-0800 establishes that the risks as the result of aircraft hazards should be sufficiently low, in that each requires that aircraft accidents that could lead to radiological consequences in excess of the exposure guidelines of 10 CFR 50.34(a)(1) with a probability of occurrence greater than an order of magnitude of 10^{-7} per year should be considered in the design of the plant.

Further, SRP 3.5.1.6 provides that 10 CFR 100.20, 10 CFR 100.21, and 10 CFR 52.17 requirements are met if the probability of aircraft accidents resulting in radiological consequences greater than the 10 CFR 100 exposure guidelines is less than an order of magnitude of 10^{-7} per year. Additionally, these guidelines provide that the probability is considered to be less than an order of magnitude of 10^{-7} per year by inspection if the distances from the plant meet all of the criteria listed below:

1. The plant-to-airport distance, D, is between 5 and 10 statute miles, and the projected annual number of operations is less than 500 D2, or the plant-to-airport distance, D, is greater than 10 statute miles, and the projected annual number of operations is less than 1000 D2;

2. The plant is at least 5 statute miles from the nearest edge of military training routes, including low-level training routes, except for those associated with usage greater than 1000 flights per year, or where activities (such as practice bombing) may create an unusual stress situation;

3. The plant is at least 2 statute miles beyond the nearest edge of a Federal airway, holding pattern, or approach pattern.

Therefore, utilizing the proximity criteria presented above, a screening analysis was first performed to establish whether the probability of aircraft accidents for the proposed CRN Site is considered to be less than an order of magnitude of 10^{-7} per year by inspection.

2.2.2.7.1 Airports

Five small privately-owned airports (Big T, Wolf Creek, Cox Farm, Will A Hildreth Farm, and Meadowlake) are located between 5 and 10 statute miles of the CRN Site (Figure 2.2-3R). These airports have no Federal Aviation Administration (FAA) Terminal Area Forecast (TAF) data available due to their size and low number of operations; however, their projected number of operations, based on available data, is less than the significance factor (i.e., the allowable annual number of operations) called for by criterion 1 (Table 2.2-7R).

Two small privately-owned airports, Oliver Springs and Fergusons Flying Circus, are within 10 to 15 statute miles of the CRN Site (Figure 2.2-3R). No FAA TAF data is available due to their size and low number of operations. Based on available data, the projected number of operations for each airport is less than the significance factor (i.e., the allowable annual number of

operations) by criterion 1, as specified in NUREG-0800 (Table 2.2-7R). Airports located at distances greater than 15 statute miles were also evaluated to confirm that each would meet the significance factor specified in criterion 1 (Table 2.2-7R).

The results, summarized in Table 2.2-7R, indicate that the proximity screening criterion 1 is met for each evaluated existing airport; therefore, no nearby existing airports need further evaluation.

A new Oak Ridge Airport (under development) is located approximately 2.25 miles north of the CRN Site. The Oak Ridge Airport is planned for general aviation anticipated to begin operation in 2026. The projected total number of operations at year 20 is 15,860 with 29 based aircraft (Reference 2.2-50). Due to the proximity to the CRN Site, the proposed Oak Ridge Airport does not meet proximity screen criterion 1.

2.2.2.7.2 Military Training Routes

The probability of aircraft accidents at the CRN Site was evaluated to determine if the CRN Site met proximity screening criterion 2, defined previously. The centerline of the closest military training route, IR2, is located approximately 19.2 miles to the WNW (Figure 2.2-3R). The Department of Defense Flight Information Publication, AP/1B, Area Planning Military Training Routes North and South America, also provides the width of each military training route segment. This publication denotes the width of IR2 as 5 nautical miles on each side of the centerline for the entire route. IR-type military training routes must be conducted on Instrument Flight Rules (IFR) flight plans regardless of weather conditions. The Department of Defense Flight Information publication also states that pilots must enter and exit the route via published entry and exit points or published alternate entry and exit points, and must have a specific air traffic control clearance prior to entering or exiting the route. Once on the route, the pilot must remain within the published route corridor (width and altitude). (Reference 2.2-33) The CRN Site is located about 19.2 statute miles from the centerline of this training route or approximately 13.4 statute miles from the edge of the training route.

The closest military operation area (MOA) is the Snowbird MOA located approximately 36 miles from the CRN Site (Figure 2.2-3R). The primary users of the Snowbird MOA were Air National Guard units, which have since been relocated or converted from fighter aircraft to other missions. Relatively high terrain for the eastern part of the country (Snowbird overlies the Smoky Mountains) and altitude allocated to accommodate civilian overflights severely limit the area's flexibility and utility for military operations (Reference 2.2-9).

Given this separation distance between the CRN Site and the nearest military training route (greater than 5 mi from the nearest edge of a military training route), along with the distance to the nearest MOA, criterion 2 is met.

2.2.2.7.3 Federal Airways

The probability of aircraft accidents at the CRN Site was also evaluated to determine if the CRN Site met proximity screening criterion 3, defined previously. There are two Federal airways, one victor (V) and one jet (J) route (V16 and J46, respectively) whose nearest edge lies within 2 statute miles of the CRN Site (Figure 2.2-3R and References 2.2-30 through 2.2-32). Thus, due

to the proximity of Federal airways V16 and J46, the CRN Site does not meet proximity screening criterion 3. Federal airway Q65 is within the 10 mile radius of the CRN Site. Although it is screened out by the proximity screening criteria, it was added as a conservatism to the detailed evaluation due to its edge being within 5 miles of the CRN Site, but it does not have a large impact.

2.2.2.7.4 Aircraft Hazards

The CRN Site does not meet acceptance criteria 1 and 3. Therefore, as required by SRP 3.5.1.6, a detailed review of aircraft hazards was required to determine the accident probability rate.

As discussed, aircraft accidents that could lead to radiological consequences in excess of the exposure guidelines of 10 CFR 100 with a probability of occurrence greater than an order of magnitude of 10^{-7} per year should be considered in the design of the plant. However, a conservative calculation showing that the probability of occurrence of doses in excess of the exposure guidelines, of about an order of magnitude 10^{-6} per year, is acceptable, in accordance with SRP 2.2.3, if, it can be shown with rigorous analysis, using realistic assumptions and reasonable arguments, that the estimated probability can be shown to be lower.

NUREG-0800 gives the following equation for estimating the probability per year of an aircraft crashing into the plant from nearby airports and heli-ports.

$$P_A = \sum_{i=1}^{L} \sum_{j=1}^{M} C_j * N_{ij} * A_{eff_j} \qquad \text{Equation 2.2-1}$$

where:

P_A = probability of an aircraft crashing into the plant

M = number of different types of aircraft using the airport

L = number of flight trajectories affecting the site

C_j = probability per square mile of a crash per aircraft movement, per aircraft movement of the j^{th} aircraft

N_{ij} = number of movements per year of the j^{th} aircraft along the i^{th} flight path

A_j = effective area of the plant in square mi for the j^{th} aircraft

NUREG-0800 gives the following equation for estimating the probability per year of an aircraft crashing into the plant when Federal airways pass through the vicinity of the site:

$$P_{FA} = C \times N \times A_{eff} / W \qquad \text{Equation 2.2-2}$$

where,

P_{FA} = the probability of an aircraft crashing into the plant

C = the in-flight crash rate per mi for aircraft using the airway

N = the number of flights per year along the airway

A_{eff} = the effective area of the plant in square mi

W = the width of the airway (plus twice the distance from the airway edge to the site when the site is outside the airway) in mi.

The airways included in this part of the analysis include the new Oak Ridge Airport traffic pattern, and airways J46, V16 and Q65.

To calculate the effective area (A) in the above equation for the CRN Site bounding building, the methodology provided in DOE Standard, DOE-STD-3014-96, (Reference 2.2-34), was used as shown below. For the BWRX-300, two separate effective areas are determined. One effective area is the Reactor Building; which is the only SC1 structure. The second effective area includes the Reactor Building, Radwaste Building and the Control Building.

$$A_{eff} = A_f(effective\ fly-in\ area) + A_s(effective\ skid\ area) \quad \text{Equation 2.2-3}$$

$$A_f = (WS+R)*H\cot\Phi + \frac{2*L*W_f*WS}{R} + L \quad \text{Equation 2.2-4}$$

$$A_S = (WS+R)*S \quad \text{Equation 2.2-5}$$

where,

WS = the aircraft wingspan

R = the length of the diagonal of the facility

H = the height of the facility

L = the length of the facility

W_f = the width of the facility

S = the aircraft skid distance

cotΦ = the mean of the cotangent of the aircraft impact angle

The input values WS, cotΦ, and S are provided in DOE-STD-3014-96 (Reference 2.2-34), and the input values R, H, W_f, and L are facility-specific.

Based on the effective area of the Reactor Building, the probability of an aircraft crashing due to the new Oak Ridge Airport into the Reactor Building is 7.45×10^{-6}. Based on the combined effective area of the Reactor Building, Radwaste Building and the Control Building the probability of an aircraft crashing due to the new Oak Ridge Airport is 1.30×10^{-5}.

Based on the effective area of the Reactor Building, the probability of an aircraft in nearby airways crashing into the Reactor Building is 2.21×10^{-5}. Based on the combined effective area of the Reactor Building, Radwaste Building and the Control Building the probability of an aircraft in nearby airways crashing at some point of this combination is 3.83×10^{-5}.

The total hazard probabilities are determined by adding the probability due to the new Oak Ridge Airport and due to nearby airways. The total hazard probability for the Reactor Building is 2.96×10^{-5} per year and for the combined Reactor Building, Radwaste Building and Control Building is 5.13×10^{-5} per year.

2.2.2.8 Projections of Industrial Growth

Replace ESPA SSAR Section 2.2.2.8 with the following:

DOE ETTP has begun a major environmental site cleanup with the long-term goal of converting the ETTP into a private industrial park called Heritage Center Industrial Park. The cleanup activities are currently being conducted and as cleanup is completed, DOE transfers ownership of the uncontaminated buildings to the Community Reuse Organization of East Tennessee, who in turn leases this property for immediate private industrial use. Many of the buildings will be slated for potential reuse and the remediated land will be available for new construction (Reference 2.2-26).

The City of Oak Ridge has selected a site for a new general aviation airport (Reference 2.2-50). The location is approximately 2.25 miles from the CRN Site. Potential hazards associated with this airport are evaluated in Section 2.2.2.7.

Other nearby facilities that are currently proposed or under construction are the Kairos Power Hermes Facility, Kairos Power Atlas Fuel Fabrication Facility, TRISO-X Fuel Facility, and Coqui Pharmaceutical.

The Coqui Pharmaceutical site is excluded from the discussion in the following sections as there is currently not enough information available for an analysis. However, the radiological effects from the radiopharmaceutical production facility at the Coqui Pharmaceutical site would be within regulatory offsite dose limits for routine operations and accidents. Additionally, the operations are expected to be similar to the SHINE Medical Technologies which received a construction permit from the USNRC in 2016 for a radioisotope production facility located in Janesville, Wisconsin. SHINE Medical Technologies demonstrated in its PSAR that releases of onsite chemicals were not a hazard to personnel in the facility control room. As such, chemicals stored onsite at the Coqui Pharmaceutical site would similarly not be expected to have an impact on nearby facilities, including the CRN Site.

The location and quantities of chemicals that may be stored nearby at the proposed Kairos Power Hermes Facility, the proposed Kairos Power Atlas Fuel Fabrication Facility, the proposed TRISO-X Fuel Facility, the proposed Coqui Pharmaceutical Facility, and the proposed regional airport have not yet been determined. The effects of explosions, flammable vapor clouds, and toxic chemicals for chemical storage at these nearby facilities will be reviewed in the Operating License Application and accounted for in the final design to the extent information regarding hazards from those nearby facilities are available.

2.2.3 Evaluation of Potential Accidents

Replace the second sentence of ESPA SSAR 2.2.3 with the following:

The potential effects of these accidents on the nuclear plant in terms of design parameters and physical phenomena were evaluated, using the guidance in RG 1.78, RG 1.91, RG 4.7, and SRP 2.2.1-2.2.2.

2.2.3.1 Determination of Potential Accidents

Replace ESPA SSAR Section 2.2.3.1 with the following:

SRP 2.2.1-2.2.2 states that design-basis events on site or in the vicinity of the nuclear plant are defined as those accidents with a probability of occurrence on the order of magnitude of 10^{-7} per year, with potential consequences sufficiently serious to affect the safety of the plant to the extent that 10 CFR 100 guidelines could be exceeded. The following accident categories are considered in selecting design-basis events: explosions, flammable vapor clouds (delayed ignition), toxic chemicals, aircraft hazards, fires, collisions with the intake structure, and liquid spills. The postulated accidents within these categories were analyzed at the following locations:

Nearby Storage Facilities

- Facilities listed in Table 2.2-1R

Nearby Transportation Routes

- East Tennessee Natural Gas Pipelines 1 and 2
- I-40
- Oak Ridge Airport (Under Development)
- Federal Airways V16, J46 and Q65

On-Site Storage Facilities

- Substances stored on-site are identified in Table 2.2-12R.

2.2.3.1.1 Explosions

Renumber Equation 2.2-5 as Equation 2.2-6 in ESPA SSAR Section 2.2.3.1.1.

Renumber Equation 2.2-6 as Equation 2.2-7 in ESPA SSAR Section 2.2.3.1.1.

Renumber Equation 2.2-7 as Equation 2.2-8 in ESPA SSAR Section 2.2.3.1.1.

Replace the 15th and 16th paragraphs of ESPA SSAR Section 2.2.3.1.1 with the following:

The materials stored at nearby facilities (Table 2.2-2R) and potentially transported along I-40 (Table 2.2-3R) and by pipeline (Table 2.2-4) are evaluated to ascertain whether they have the potential to explode. The effects of these explosion events from external sources are summarized in Table 2.2-9R, and are described in the following subsections relative to the release source.

The nearest on-site source of potentially explosive gases is the hydrogen gas storage cylinders. Table 2.2-12R lists the quantity of hydrogen stored. The hydrogen is stored in several cylinders. Each cylinder is assumed to store 12,584 standard cubic feet (scf).

The safe separation distance between the hydrogen storage area and nearest SC1 structure is determined using the methodology in EPRI NP-5283-SR, “Guidelines for Permanent BWR Hydrogen Water Chemistry Installations” (Reference 2.2-24). Consistent with EPRI NP-5283-SR the required separation distance is determined for two different considerations. The first consideration is the required separation distance such that the SC1 structure is not adversely affected by the postulated hydrogen explosion. The second consideration is the required separation distance to air pathways into SC1 structures versus the internal diameter of the high pressure piping.

For the first consideration (vessel explosion), consistent with EPRI NP-5283-SR, it is assumed that the explosion is limited to one cylinder. The required distance is conservatively determined using the British E.S.T.C. method described in Appendix B of EPRI NP-5283-SR. This approach is conservative as it is for a structural wall design less robust than the Reactor Building wall design.

The British E.S.T.C. method uses the following relationship to determine the required separation distance:

$$R = \frac{56 * W^{1/3}}{\left[1 + \left(\frac{7{,}000}{W}\right)^2\right]^{1/6}} \qquad \text{Equation 2.2-9}$$

where:

R = Separation distance in feet

W = TNT equivalent explosive yield in lbs. For gaseous hydrogen storage, W is determined using the relationship from EPRI NP-5283-SR: 1,000 scf hydrogen = 27.1 lbs. TNT

Using the above British E.S.T.C. method for the same conditions (i.e., one cylinder fails) the required separation distance for a cylinder explosion is 142.8 feet.

The required separation distance for the second consideration (gaseous pipe break) determined using EPRI NP-5283-SR is approximately 230 feet.

The hydrogen cylinders will be located more than 230 feet from the Reactor Building and the Control Building. The transportation route to replace hydrogen cylinders will be greater than the minimum separation distance such that a cylinder explosion during transport will not adversely impact the plant.

2.2.3.1.1.1 Nearby Facilities

Replace ESPA SSAR Section 2.2.3.1.1.1 with the following:

The following were identified as facilities of concern with regard to storage of materials with the potential for explosion within the vicinity of the CRN Site (Section 2.2.2.2.2):

- Custom Foods of America, located approximately 23.6 mi (125,000 ft) from the site,
- East Tennessee Technology Park, located approximately 3 mi (16,000 ft) from the site,
- Energy Solutions Services – Bear Creek Facility, located approximately 1.6 mi (8,290 feet) from the site,
- Food Lion – Delhaize Distribution Center, located approximately 20 mi (108,000 ft) from the site,
- Linde Inc., located approximately 11 mi (57,600 ft) from the site,
- ORNL-Battelle, located approximately 4.3 mi (22,900 ft) from the site,
- Primary Products Ingredients, located approximately 11 mi (57,600 ft) from the site,
- Processed Food Corporation, located approximately 23.6 mi (125,000 ft) from the site,
- TVA – Kingston Fossil Plant, located approximately 7.7 mi (40,900 ft) from the site,
- U.S. Foodservice - Knoxville, located approximately 23.3 mi (123,000 ft) from the site,
- Weigel's #58, located approximately 3.7 mi (19,400 ft) from the site.

Each material stored at the identified offsite facilities was evaluated with respect to explosion potential. Each material was then dispositioned based on the identified physical properties of the material and whether a bounding analysis exists. The results of this evaluation are found in Table 2.2-5R (footnotes describe the bases for explosive materials not carried forward for further analysis).

A conservative analysis using the TNT equivalency methods described in Section 2.2.3.1.1 was used to determine safe distances for the identified potentially explosive materials. The results indicate that the safe distances are less than the minimum separation distances from the CRN Site power block area to the storage locations for the identified materials (Table 2.2-9R). Propane

stored at the Energy Solution Services - Bear Creek Facility resulted in the largest safe distance, 2673 ft, which is less than the distance of 8,290 ft to the CRN Power Block.

Therefore, damaging overpressures from an explosion resulting from a complete failure of the total stored quantity for each material evaluated at each of the identified facilities of concern with regard to the storage of materials with the potential for explosion would not adversely affect the operation or shutdown of the CRN unit.

2.2.3.1.1.2 Nearby Transportation Routes/Roadways

Replace Table 2.2-3 with Table 2.2-3R in ESPA SSAR Section 2.2.3.1.1.2.

Replace the second paragraph of ESPA SSAR Section 2.2.3.1.1.2 with the following:

The results of this evaluation are found in Table 2.2-6R (footnotes describe the basis for explosive materials not carried forward for further analysis). The hazardous materials that are identified for further analysis with regard to explosion potential were butane, gasoline, hydrogen, methanol, methyl chloroform, and propane (Table 2.2-6R).

Replace Table 2.2-9 with Table 2.2-9R in the 3rd paragraph of ESPA SSAR Section 2.2.3.1.1.2.

2.2.3.1.1.3 Nearby Transportation Routes/Pipelines

Renumber Equation 2.2-8 to Equation 2.2-10 in ESPA SSAR Section 2.2.3.1.1.3.

Renumber Equation 2.2-9 to Equation 2.2-11 in ESPA SSAR Section 2.2.3.1.1.3.

Replace the 8th paragraph of ESPA SSAR Section 2.2.3.1.1.3 with the following:

Using the flammable mass calculated by the above methodologies, the equivalent mass of TNT was calculated using Equations 2.2-7 and 2.2-8.

Replace Table 2.2-9 with Table 2.2-9R in the 9th paragraph of ESPA SSAR Section 2.2.3.1.1.3.

2.2.3.1.2 Flammable Vapor Clouds (Delayed Ignition and Jet Fire)

Replace the 4th sentence of the 2nd paragraph of ESPA SSAR Section 2.2.3.1.2 with the following:

The materials stored at nearby facilities (Table 2.2-2R) and potentially transported along I-40 (Table 2.2-3R) and by pipeline (Table 2.2-4) were evaluated to ascertain whether they have the potential to form flammable and/or explosive vapor clouds (Table 2.2-5R and 2.2-6R).

Replace the 1st and 2nd sentences of the 5th paragraph in ESPA SSAR Section 2.2.3.1.2 with the following:

Conservative assumptions were used in the ALOHA analyses regarding both meteorological inputs and identified scenarios for both events. Each postulated event was evaluated under a spectrum of meteorological conditions.

Replace the 7th paragraph in ESPA SSAR Section 2.2.3.1.2 with the following:

For some of the chemicals considered herein, the evaluation of possible impacts due to a flammable vapor cloud identified that these were enveloped by analyses using ALOHA. The effects of flammable vapor clouds and vapor cloud explosions from external sources are summarized in Table 2.2-10R and are described in the following subsections relative to the release source.

The nearest on-site source of potentially explosive gases is the hydrogen gas storage cylinders for the Hydrogen Water Chemistry system. Table 2.2-12R lists the maximum quantities of flammable gas (hydrogen) stored at the CRN Site. The safe separation distance between the hydrogen storage area and the nearest SC1 structures is 230 ft. The method used to evaluate the safe separation for hydrogen gas storage is described in Section 2.2.3.1.1.

2.2.3.1.2.1 Nearby Facilities

Replace ESPA SSAR Section 2.2.3.1.2.1 with the following:

The following were identified as facilities of concern with regard to storage of materials with the potential for formation of flammable/explosive vapor clouds within the vicinity of the CRN Site (Subsection 2.2.2.2.2):

- Custom Foods of America, located approximately 23.6 mi (125,000 ft) from the site,
- East Tennessee Technology Park, located approximately 3 mi (16,000 ft) from the site,
- Energy Solutions Services – Bear Creek Facility, located approximately 1.6 mi (8,290 feet) from the site,
- Food Lion – Delhaize Distribution Center, located approximately 20 mi (108,000 ft) from the site,
- Linde Inc., located approximately 11 mi (57,600 ft) from the site,
- ORNL–Battelle, located approximately 4.3 mi (22,900 ft) from the site;
- Primary Products Ingredients, located approximately 11 mi (57,600 ft) from the site,
- Processed Food Corporation, located approximately 23.6 mi (125,000 ft) from the site,
- TVA – Kingston Fossil Plant, located approximately 7.7 mi (40,900 ft) from the site;
- U.S. Foodservice – Knoxville, located approximately 23.3 mi (123,000 ft) from the site, and
- Weigel's #58, located approximately 3.7 mi (19,400 ft) from the site.

Each material stored at the identified offsite facilities was evaluated with respect to its potential for formation of flammable/explosive vapor clouds. Each material was then dispositioned based

on the identified physical properties of the material and whether a bounding analysis exists. The results of this evaluation are found in Table 2.2-5R (for flammable/explosive materials not carried forward for further analysis, footnotes indicating the reason for the disposition are provided in the table). The materials stored at each site identified for further analysis are identified in Table 2.2-5R.

As described in Subsection 2.2.3.1.2, the ALOHA dispersion model was used to determine the distance a vapor cloud could travel to reach the LFL boundary once a vapor cloud has formed from an accidental release of the identified chemical. The results indicated that any plausible vapor cloud that could form and mix sufficiently under stable atmospheric conditions would be below the LFL boundary before reaching the CRN Site power block area (Table 2.2-10R).

A vapor cloud explosion analysis was also completed following the methodology as detailed in Subsection 2.2.3.1.2 in order to obtain safe distances. The results conclude that the safe distances (the minimum distance required for an explosion to have less than a 1 psi peak incident pressure) are less than the shortest distance to the CRN Site power block area and the storage location of these chemicals (Table 2.2-10R).

Therefore, a flammable vapor cloud with the possibility of ignition or explosion formed from the storage of chemicals at offsite facilities will not adversely affect the safe operation or shutdown of units within the CRN Site power block area.

2.2.3.1.2.2 Nearby Transportation Routes/Roadways

Replace Table 2.2-3 with Table 2.2-3R in ESPA SSAR Section 2.2.3.1.2.2.

Replace Table 2.2-6 with Table 2.2-6R in ESPA SSAR Section 2.2.3.1.2.2.

Replace the 2nd paragraph of ESPA SSAR Section 2.2.3.1.2.2 with the following:

The materials identified for further analysis with regard to flammable vapor clouds were acrylonitrile, butane, gasoline, methanol, methyl chloroform, methyltrichlorosilane and propane (Table 2.2-6R). It was conservatively assumed that the maximum quantity of the butane and propane potentially transported on the roadway is 11,500 gallons (gal), which represents the maximum transport quantity in an MC-331 high pressure tank truck (49 CFR 173.315 and Reference 2.2-42). For gasoline, it is conservatively assumed that 8500 gal is potentially transported on I-40. Both quantities are greater than 50,000 lb which is the maximum probable solid cargo highway transport quantity identified in RG 1.91. The transported quantities for the other materials is based on the storage quantity at the associated location.

Replace the 3rd and 4th paragraphs of ESPA SSAR Section 2.2.3.1.2.2 with the following:

An analysis for the identified chemicals was conducted using ALOHA as described in Subsection 2.2.3.1.2. The results indicate that any plausible vapor cloud that could form and mix sufficiently would be below the LFL boundary before reaching the CRN Site power block area. The safe distances are less than the minimum separation distances from the CRN Site power block area to I-40 for each of the identified chemicals (Table 2.2-10R).

A vapor cloud explosion analysis was also completed as detailed in Subsection 2.2.3.1.2 to obtain safe distances. The results indicate that the safe distances, the minimum distances required for an explosion to have less than a 1 psi peak incident pressure are less than the shortest distance to the power block area to I-40 (Table 2.2-10R).

2.2.3.1.2.3 Nearby Transportation Routes/Pipelines

Replace Table 2.2-10 with Table 2.2-10R in ESPA SSAR Section 2.2.3.1.2.3.

2.2.3.1.3 Toxic Chemicals

Replace Table 2.2-2 with Table 2.2-2R in ESPA SSAR Section 2.2.3.1.3.

Replace Table 2.2-3 with Table 2.2-3R in ESPA SSAR Section 2.2.3.1.3.

Replace the 2nd paragraph in ESPA SSAR Section 2.2.3.1.3 with the following:

Each identified chemical is evaluated based upon the chemical's properties, quantities, and distance in relation to the power block area without consideration of plant design factors, such as control room ventilation. Chemicals that lead to concentrations above the Immediately Dangerous to Life or Health (IDLH) limit at the power block area boundary will be identified.

Replace the 1st and 2nd sentences of the 3rd paragraph ESPA SSAR Section 2.2.3.1.3 with the following:

Chemicals were initially screened using the methodology in RG 1.78 to determine which warranted more detailed analyses. In selected cases, the ALOHA air dispersion model was used to predict the chemical concentrations within a toxic or asphyxiating vapor cloud as it disperses downwind for facilities and sources.

Replace the 1st sentence of the 5th paragraph ESPA SSAR Section 2.2.3.1.3 with the following:

Each postulated toxicity/asphyxiation event was evaluated under a spectrum of meteorological conditions to determine the worst-case meteorological condition.

Replace the last paragraph of ESPA SSAR Section 2.2.3.1.3 with the following:

The effects of toxic chemical releases from nearby facilities and transportation routes are summarized in Table 2.2-11R and are described in the following subsections relative to the release sources.

The effects of toxic chemical releases from onsite chemical storage are described in Subsection 2.2.3.1.3.4.

2.2.3.1.3.1 Nearby Facilities

Replace ESPA SSAR Section 2.2.3.1.3.1 with the following:

The following were identified as facilities of concern with regard to storage of materials with the potential for release of a toxic vapor cloud within the vicinity of the CRN Site (see Subsection 2.2.2.2.2):

- A.E. Staley / Tate & Lyle Manufacturing, located approximately 11 mi (58,000 ft) from the site,
- City of Maryville Wastewater Treatment Plant, located approximately 24 mi (127,700 ft) from the site,
- City of Oak Ridge Municipal Pool, located approximately 10.8 mi (57,000 ft) from the site,
- Custom Foods of America, located approximately 23.6 mi (125,000 ft) from the site,
- Diversified Scientific Services, located approximately 3.4 mi (17,800 ft) from the site,
- Duracap Asphalt Paving, located approximately 2.3 m (12,000 ft) from the site,
- East Tennessee Mechanical Contractors WTP, located approximately 10.5 mi (55,200 ft) from the site,
- East Tennessee Technology Park, located approximately 3 mi (16,000 ft) from the site,
- Energy Solutions Services – Bear Creek Facility, located approximately 1.6 mi (8,290 feet) from the site,
- First Utility District of Knox County WTP, located approximately 14.2 mi (75,100 ft) from the site,
- First Utility District of Knox County WWTP, located approximately 13.4 mi (70,700 ft) from the site
- Food Lion – Delhaize Distribution Center, located approximately 20 mi (108,000 ft) from the site,
- Fourth Creek Wastewater Treatment Plant, located approximately 22 mi (116,700 ft) from the site,
- Hallsdale Powell Utility District Melton Hill WTP, located approximately 18.2 mi (96,000 ft) from the site,
- Harrison Construction – Gallaher Concrete Plant, located approximately 3 mi (16,000 ft) from the site,
- Linde, Inc., located approximately 11 mi (57,600 ft) from the site,
- NAC Philotechnics – Horizon Center, located approximately 4.9 mi (26,000 ft) from the site,
- Oak Ridge WTP, located approximately 10.5 mi (55,000 ft) from the site,
- ORNL–Battelle, located approximately 4.3 mi (22,900 ft) from the site,
- ORNL-UCOR, located approximately 4.9 miles north of the CRN Site,
- Primary Products Ingredients, located approximately 11 mi (57,600 ft) from the site,
- Processed Food Corporation, located approximately 23.6 mi (125,000 ft) from the site,
- Tellico Area Service System Water Treatment Plant, located approximately 21.8 mi (115,000 ft) from the site,

- TVA – Kingston Fossil Plant, located approximately 7.7 mi (40,900 ft) from the site,
- TVA – Melton Hill Hydro Plant, located approximately 4.7 mi (24,500 ft) from the site,
- U.S. Foodservice – Knoxville, located approximately 23.3 mi (123,000 ft) from the site,
- Weigel's #58, located approximately 3.7 mi (19,400 ft) from the site.
- West Knox Utility District WTP, located approximately 9.1 (48,000 ft) from the site,
- West Knox Utility District WWTP, located approximately 14.4 mi (75,900 ft) from the site, and
- Yale Security Inc., located approximately 9.5 miles (50,000 ft) from the site.

Each material stored at the identified offsite facilities was evaluated with respect to its potential for formation of a toxic vapor cloud. Each material was then dispositioned based on the identified physical properties of the material and whether a bounding analysis exists. The results of this evaluation are found in Table 2.2-5R (for toxic materials not carried forward for further analysis, footnotes indicating the reason for the disposition are provided in the table). Materials identified for further analysis are identified in Table 2.2-5R.

Table 2.2-11R shows the chemicals from Table 2.2-5R that were carried over for further evaluation. For each chemical, Table 2.2-11R shows the quantity analyzed, the associated IDLH limit, the distance from the source to the CRN Power Block in feet, and the distance from the source to where the concentration is less than the IDLH limit. In some cases, where the RG 1.78 simplified calculation methodology was used, Table 2.2-11R identifies that the distance from the source to the IDLH limit is less than the distance from the source to the CRN Site.

As described in Subsection 2.2.3.1.3, some of the identified chemicals were analyzed using the ALOHA dispersion model to determine whether the formed vapor cloud would reach the CRN Site power block area with concentrations greater than the determined toxicity limit.

For those chemicals that are gases under the atmospheric release conditions for the given scenario the distances to the IDLH/asphyxiating or other determined toxicity limits were calculated following a 10-min release from the largest storage. For those chemicals that are liquids under the atmospheric release conditions for the given scenario the release scenario in each of these analyses includes the total loss of the largest vessel, resulting in an unconfined 1-cm thick puddle.

In the case of each of the atmospheric gases analyzed, the distances to the IDLH/asphyxiating or other determined toxicity limit were calculated. The results indicate that any plausible toxic vapor cloud that could form would be below the IDLH or other identified toxicity limit before reaching the CRN Site power block area (Table 2.2-11R).

Therefore, the formation of a toxic vapor cloud following an accidental release of the analyzed hazardous materials stored on site will not adversely affect the safe operation or shutdown of units within the CRN Site power block area (Table 2.2-11R).

2.2.3.1.3.2 Nearby Transportation Routes/Roadways

Replace ESPA SSAR Section 2.2.3.1.3.2 with the following:

The nearest approach from I-40 to the CRN Site power block area is approximately 1.1mi (5800 ft). Table 2.2-3R details the hazardous materials potentially transported on I-40. The materials identified for further analysis with regard to formation of toxic vapor clouds were acrylonitrile, anhydrous ammonia, argon, butane, carbon dioxide, chlorine, chlorine dioxide, gasoline, helium, HFC-134A, hydrochloric acid, hydrofluoric acid, hydrogen chloride, methanol, methyl chloroform, methyltrichlorosilane, nitric acid, nitric oxide, nitrogen, nitrogen dioxide, propane, and sulfur dioxide (Table 2.2-6R). It is conservatively assumed that the maximum quantity of the anhydrous ammonia and butane potentially transported on the roadway is 11,500 gal, which is the maximum transport quantity in an MC-331 high pressure tank truck (49 CFR 173.315 and Reference 2.2-42). For chlorine, a maximum transport quantity of 22 T is assumed (Reference 2.2-18). For hydrochloric acid and nitric acid, it was conservatively assumed that the maximum quantity potentially transported on the roadway is 6000 gal, which is the maximum transport quantity in an MC-312/DOT412 Corrosive Tanker (Reference 2.2-42).

For gasoline, it was conservatively assumed that 8500 gal is potentially transported on I-40. For argon, carbon dioxide, helium, HFC-134A, methanol, methyl chloroform, propane and sulfur dioxide a transported quantity of 50,000 lbs is assumed which is the maximum probable cargo highway transport quantity identified in RG 1.91. The transported weights of acrylonitrile, chlorine dioxide, hydrofluoric acid, hydrogen chloride, methyltrichlorosilane, nitric oxide, and nitrogen dioxide are based on the quantity stored at the associated location.

An analysis for the identified chemicals was conducted using the simplified approach in RG 1.78, and, where necessary, using ALOHA as described in Subsection 2.2.3.1.3. The results indicated that, except for anhydrous ammonia, chlorine, hydrochloric acid and nitric acid, the distances to the identified toxicity limit for any plausible toxic vapor cloud that could form following an accidental release at the closest approach from the transportation route (I-40) are less than the minimum separation distances from the CRN Site power block area to I-40 (Table 2.2-11R).

A release of anhydrous ammonia results in a distance of 13,728 ft to the toxicity endpoint and a release of chlorine results in a distance of 23,760 ft to the toxicity endpoint. Both determined distances to the toxic endpoints are greater than the distance of 5800 ft to the CRN Site power block area. Hydrochloric acid and nitric acid were not able to be screened using the methodology in RG 1.78, Appendix A.

Add new Section 2.2.3.1.3.4:

2.2.3.1.3.4 On-Site Storage

The identified onsite chemicals are listed in Table 2.2-12R; which, for each chemical, lists the on-site location and the quantity stored. Each chemical is dispositioned relative to control room habitability, flammability, or explosive concern. The disposition for each chemical is summarized

in Table 2.2-13R. Table 2.2-13R shows that the majority of the chemicals are not toxic, flammable, or explosive.

Hydrogen is analyzed for explosion concern as described in Subsection 2.2.3.1.1. Cryogenic nitrogen can pose a potential asphyxiation concern for the control room operators. This potential was evaluated using the HABIT computer code and determined that the atmosphere in the Control Room would not be reduced to an oxygen deficient environment. For chemicals with IDLH values listed in Table 2.2-13R, the effects of toxic vapors or gases and their potential for incapacitating the control room operators were evaluated and determined that these were not a hazard to the operators in the Control Room.

Therefore, the formation of a toxic vapor cloud following an accidental release of hazardous materials stored on site will not adversely affect the safe operation or shutdown of the unit (Table 2.2-13R).

This evaluation is based on the chemicals identified at this time. The effects of explosions, flammable vapor clouds, and toxic chemicals from onsite chemical storage will be reviewed in the Operating License Application and accounted for in the final design.

2.2.3.1.4 Fires

Replace ESPA SSAR Section 2.2.3.1.4 with the following:

As discussed in Subsection 2.2.3.1.3.4, there are no concerns with fire from chemicals stored on site. Onsite fuel storage facilities are designed in accordance with applicable fire codes, and plant safety is not jeopardized by fires or smoke in these areas. A description of the fire protection system is presented in Subsection 9.5.1.

Fire and smoke from accidents at nearby homes, industrial facilities, transportation routes, or from area forest or brush fires, do not jeopardize the safe operation of the unit due to the separation distance of potential fires from the unit. A description of the Control Building Heating, Ventilation, and Cooling System is presented in Subsection 9.4.1.

External accidents were considered in the vicinity of the CRN Site that could lead to high heat fluxes or smoke, and nonflammable gas or chemical-bearing clouds from the release of materials as a consequence of fires. Fires from nearby facilities and fires from transportation accidents and pipelines are evaluated as events that could lead to high heat fluxes or to the formation of such clouds.

Those chemicals stored at nearby facilities and transported by roadway on I-40, are evaluated in Subsection 2.2.3.1.2 for potential effects of accidental releases leading to a delayed ignition of any formed vapor cloud. For each of the stored or transported materials evaluated, the results indicate that any formed vapor cloud will dissipate below the LFL before reaching the CRN Site power block area boundary. Therefore, it is not expected that there would be any hazardous effects to units at the CRN Site from fires or heat fluxes associated with the storage of chemicals at nearby facilities or from the transport of chemicals on I-40.

Furthermore, a heat flux analysis for a pipeline break indicates that there would be no effect on the safe operation or shutdown of units within the CRN Site power block area. ALOHA is used to determine the limiting heat flux from a jet fire due to a natural gas pipeline break. The results of the jet fire analysis conclude that the safe distance (i.e., the distance to 5 kW/m^2) for the 6-in. pipeline (Pipeline 1) of 312 ft is less than the separation distance, 5800 ft, between the CRN Site power block area and the pipeline break. The safe distance for the 22-in. pipeline (Pipeline 2) of 1203 ft is less than the separation distance, 15,800 ft.

Therefore, it is not expected that there would be any hazardous effects to units at the CRN Site from fires or heat fluxes associated with the storage of chemicals at nearby facilities, the transport of chemicals on I-40, or from the East Tennessee Natural Gas Pipelines evaluated.

2.2.3.2 Effects of Design-Basis Events

Replace ESPA SSAR Section 2.2.3.2 with the following:

Evaluations were performed of potential hazards nearby and on the CRN Site. Except where noted, these evaluations concluded that potential accidents involving explosions, flammable vapor clouds, collisions with intake structures, and liquid spills do not pose a threat to the CRN Site.

2.2.4 References

Replace ESPA SSAR Section 2.2.4 with the following:

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2.2-2 Tennessee Department of Transportation (TDOT), Traffic Count Database System (TCDS), Available at https://tdot.public.ms2soft.com/tcds/tsearch.asp?loc=Tdot&mod=TCDS, accessed August 30, 2023.

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2.2-5 Enbridge Inc., Available at https://www.enbridge.com/enbridge-and-spectra, accessed August 29, 2023.

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2.2-18 The Chlorine Institute, Inc., Pamphlet 5, Bulk Storage of Liquid Chlorine, 7th ed., October 2005.

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2.2-23 Not Used.

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2.2-25 Federal Aviation Administration, Environmental Assessment (Draft), Proposed Oak Ridge Airport, Oak Ridge, Roane County, Tennessee, dated April 17, 2023.

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Replace Table 2.2-1 with the following:

Table 2.2-1R Description of Facilities - Products and Materials
(Sheet 1 of 3)

Facility	Concise Description	Primary Function	Major Products
A. E. Staley Manufacturing Company-Loudon / Tate & Lyle, Loudon [(b)]	Food ingredient development	Develop ingredients for food and beverages	Food ingredients
City of Maryville Wastewater Treatment Plant	Wastewater treatment plant	Wastewater treatment	Clean water Residual biosolids product that can be used for soil amendment
City of Oak Ridge Municipal Pool	Municipal swimming pool	Public access to outdoor swimming	Recreation
Custom Foods of America	Products for the food industry	Food products for restaurants and convenience stores	Ready to eat food products
Diversified Scientific Services, Inc.	Radioactive waste treatment	Operate waste treatment facilities	Waste treatment
Duracap Asphalt Paving	Construction services	Asphalt paving, grading and drainage, site utilities	Asphalt paving
East Tennessee Mechanical Contractors (ETMC) WTP	Construction company	Facilities management, General construction, Mechanical design and construction	Construction management
East Tennessee Technology ParkEast Tennessee Technology Park - UCOR LLC-UCOR Rollup©	Managing hazardous wastes	Managing wastes and site remediation	Site cleanup for eventual site transition to public use
Energy Solution Services, Bear Creek Facility	Radioactive waste processing facility	Process radioactive waste	Processed waste materials
First Utility District of Knox County WTP	Water treatment plant	Water treatment and distribution	Drinking water supply
First Utility District of Knox County WWTP	Wastewater treatment plant	Wastewater treatment	Clean water
Food Lion - Delhaize DC #6	Food retail	Food retail to the general public	Food

Table 2.2-1R Description of Facilities - Products and Materials
(Sheet 2 of 3)

Facility	Concise Description	Primary Function	Major Products
Fourth Creek Wastewater Treatment Plant	Wastewater treatment plant	Wastewater treatment	Clean water
Hallsdale Powell Utility District - Melton Hill WTP	Water treatment plant	Water treatment and distribution	Drinking water supply
Harrison Construction, Gallaher Concrete Plant	Construction materials	Provide aggregates, ready mix concrete, etc.	Aggregates, Ready Mix Concrete
Knoxville PDC – Volkswagen Group of America	Car Distribution Service	Car distribution	Car distribution for retail
Linde, Inc.	Gas and equipment supplier	Provide industrial and specialty gases	Industrial and specialty gases. Provide associated supply systems
NAC Philotechnics – Horizon Center	Radioactive and mixed waste processing facility	Process radioactive waste	Processed waste materials
Oak Ridge National Laboratory	Federal government science and energy laboratory	Federal sponsored unique research and development	Government research and development laboratory
Oak Ridge National Laboratory - UCOR LLC - UCOR Rollup	Oak Ridge National Laboratory Cleanup	Deactivation and demolishing	Site remediation
Oak Ridge Water Treatment Plant	Water treatment plant	Water treatment and distribution	Drinking water supply
Primary Products Ingredients Americas, LLC	Food ingredient development	Develop ingredients for food and beverages	Corn syrup, crystalline fructose, fuel ethanol
Processed Food Corporation	Products for the food industry	Food products for restaurants and convenience stores	Ready to eat food products
Tellico Area Service System WTP	Water treatment plant	Water treatment and distribution	Drinking water supply
TVA – Kingston Fossil Plant	Coal-fired power plant	Power generation	Electrical power
TVA – Melton Hill Hydro Plant	Hydroelectric power plant	Power generation	Electrical power

Table 2.2-1R Description of Facilities - Products and Materials
(Sheet 3 of 3)

Facility	Concise Description	Primary Function	Major Products
U.S. Foodservice - Knoxville	Restaurant supplier	Food distribution to restaurants	Food storage and distribution
Weigel's #58	Convenience Store	Dispense fuel and other commodities	Gasoline, propane, diesel
West Knox Utility District WTP	Water treatment plant	Water treatment and distribution	Drinking water supply
West Knox Utility District WWTP	Wastewater treatment plant	Wastewater treatment	Clean water
Yale Security Inc.	Manufacture commercial locks and hardware	Manufacturing, including metalworking, electroplating, polishing, etc.	Commercial locks and hardware

a) Reference 2.2-11, Reference 2.2-14, Reference 2.2-15, Reference 2.2-16, Reference 2.2-17, Reference 2.2-28, Reference 2.2-54, Reference 2.2-55, Reference 2.2-56, Reference 2.2-57, Reference 2.2-58, Reference 2.2-59, Reference 2.2-60, Reference 2.2-61, Reference 2.2-62, Reference 2.2-63, Reference 2.2-64, Reference 2.2-65, Reference 2.2-66, Reference 2.2-67, Reference 2.2-68, Reference 2.2-69, Reference 2.2-70, Reference 2.2-71, Reference 2.2-72, Reference 2.2-73, Reference 2.2-74, Reference 2.2-75, Reference 2.2-76, Reference 2.2-77, Reference 2.2-78, Reference 2.2-79, Reference 2.2-80, Reference 2.2-82.

b) A.E. Staley Manufacturing and Tate & Lyle are the same facility.

c) East Tennessee Technology Park and East Tennessee Technology Park - UCOR LLC- UCOR Rollup are grouped together as similar chemicals are used and the proximity to the CRN Site is the same.

Notes:

U.S. Department of Energy Oak Ridge National Laboratory (ORNL)

UT-Battelle LLC (Battelle)

URS Corporation (URS)

United Cleanup Oak Ridge (UCOR)

Tennessee Valley Authority (TVA)

Water Treatment Plant (WTP)

Wastewater Treatment Plant (WWTP)

Replace Table 2.2-2 with the following:

Table 2.2-2R Nearby Facilities Chemical Storage
(Sheet 1 of 13)

Chemical	Facility/Location	Quantity (lbs)[(a)] (unless otherwise noted)	Toxicity Limit (IDHL) [(b)]
A.E. Staley Manufacturing / Tate & Lyle			
Sulphur Dioxide (Anhydrous)	A. E. Staley Manufacturing Company-Loudon/ Tate & Lyle	144,000	100 ppm
City Of Maryville Wastewater Treatment			
Chlorine	City Of Maryville Wastewater Treatment Plant	12,000	29mg/m^3 (10 ppm)
City of Oak Ridge Municipal Pool			
Chlorine	City of Oak Ridge Municipal Pool	4,000	29mg/m^3 (10 ppm)
Custom Foods of America			
Anhydrous Ammonia	Custom Foods Of America	28,600	210 mg/m^3 (300 ppm)
Diversified Scientific Services			
Diesel Oil #2	Diversified Scientific Services, Inc.	14,520: 7,260 in largest container	Not Available
Diesel Oil #2	Diversified Scientific Services, Inc.	99,000	Not Available
Nonylphenol Ethoxylates	Diversified Scientific Services, Inc.	2,910: 485 in largest container	Not Available
Sodium Hydroxide (20% - 40%)	Diversified Scientific Services, Inc.	10,000: 3,350 in largest container	10 mg/m^3
Triethanolamine Mixture	Diversified Scientific Services, Inc.	4,000: 490 in largest container	Not Available
Duracap Asphalt Paving			
Asphalt Mix	Duracap Asphalt Paving Co., Inc	9,999,999, maximum in largest containers 4,000,000	Not Available
Diesel Fuel	Duracap Asphalt Paving Co., Inc	15,000, maximum in largest container 6,970	Not Available

Table 2.2-2R Nearby Facilities Chemical Storage
(Sheet 2 of 13)

Chemical	Facility/Location	Quantity (lbs)[a] (unless otherwise noted)	Toxicity Limit (IDHL) [b]
Limestone	Duracap Asphalt Paving Co., Inc	11,000,000	Not Available
Liquid Asphalt (CAS 8052424)	Duracap Asphalt Paving Co., Inc	499,999 maximum in largest container 215,000	Not Available
ULTRATACK (CAS 8052424)	Duracap Asphalt Paving Co., Inc	79,000	Not Available
East Tennessee Mechanical Contractors			
Chlorine	East Tennessee Mechanical Contractors (ETMC) WTP	12,000	29mg/m^3 (10 ppm)
East Tennessee Technology Park East Tennessee Technology Park – UCOR LLC			
CETCO Quik-Solid Granular	East Tennessee Technology Park	24,999	Not Available
Chemical Specialties Ultrpoles	East Tennessee Technology Park	49,999	5 mg /m^3
Diesel Fuel #2	East Tennessee Technology Park	49,999	Not Available
Flextura FGM Erosion Control Agent	East Tennessee Technology Park	24,999	Not Available
Gasoline	East Tennessee Technology Park	24,999	750 ppm
Lead	East Tennessee Technology Park	2,499	100 mg/m^3
Oils, Misc. Lubricating	East Tennessee Technology Park	74,999	Not Available
Silica, Crystalline Quartz	East Tennessee Technology Park	24,999	3000 mg/m^3
Sodium Polyacrylate (CAS 9003047)	East Tennessee Technology Park	24,999	Not Available
Sulfuric Acid	East Tennessee Technology Park	4,999	15 mg/m^3
Energy Solution Services, Bear Creek Facility			
Caustic Soda (CAS 1310732)	Energy Solution Services, Bear Creek Facility	64,000	10 mg/m3

Table 2.2-2R Nearby Facilities Chemical Storage
(Sheet 3 of 13)

Chemical	Facility/Location	Quantity (lbs)[a] (unless otherwise noted)	Toxicity Limit (IDHL) [b]
Diesel Fuel #2	Energy Solution Services, Bear Creek Facility	27,824: maximum in largest container 7,150	Not Available
Lead	Energy Solution Services, Bear Creek Facility	200,000	100 mg/m^3
Magnesium Oxide (CAS 1309484)	Energy Solution Services, Bear Creek Facility	28,000: maximum in largest container 3,000	750 mg/m^3
Nitric Acid	Energy Solution Services, Bear Creek Facility	645	25 ppm
Oxygen (CAS 7782447)	Energy Solution Services, Bear Creek Facility	86,890:, maximum in largest container 85,500	Not Available
Propane	Energy Solution Services, Bear Creek Facility	25,000:, maximum in largest container 20,000	2100 ppm
First Utility District of Knox County Water Treatment Plant and Wastewater Treatment Plant			
Chlorine	First Utility District Of Knox County WTP	22,000	29mg/m^3 (10 ppm)
Chlorine	First Utility District Of Knox County WWTP	20,000	29mg/m^3 (10 ppm)
Food Lion			
Anhydrous Ammonia	Food Lion - Delhaize DC #6	16,000	210 mg/m^3 (300 ppm)
Fourth Creek Wastewater Treatment Plant			
Chlorine	Fourth Creek WWTP	16,000	29mg/m^3 (10 ppm)
Hallsdale Powell Utility District – Melton Hill Water Treatment Plant			
Aluminum Chlorohydrate	Hallsdale Powell Utility District - Melton Hill WTP	24,820	Not Available

Table 2.2-2R Nearby Facilities Chemical Storage
(Sheet 4 of 13)

Chemical	Facility/Location	Quantity (lbs)[a] (unless otherwise noted)	Toxicity Limit (IDHL) [b]
Carbon, Activated	Hallsdale Powell Utility District - Melton Hill WTP	30,000: maximum in largest container 1,000	Not Available
Caustic Soda	Hallsdale Powell Utility District - Melton Hill WTP	8,422	10 mg/m^3
Chlorine	Hallsdale Powell Utility District - Melton Hill WTP	22,000	29mg/m^3 (10 ppm)
Chlorine Dioxide Solution-(500-5000 mg/l)	Hallsdale Powell Utility District - Melton Hill WTP	24,131: maximum in largest container 6,033.	5 ppm
Hydrofluorosilicic Acid	Hallsdale Powell Utility District - Melton Hill WTP	11,159	Not Available
Sodium Hypochlorite	Hallsdale Powell Utility District - Melton Hill WTP	18,465	10 ppm as Cl
Sulfuric Acid	Hallsdale Powell Utility District - Melton Hill WTP	15,800: maximum in largest container 7,900	15 mg/m^3
WSU 137 - Blended Phosphate	Hallsdale Powell Utility District - Melton Hill WTP	55,797	Not Available
Harrison Construction – Gallaher Concrete Plant			
#2 Grease	Gallaher Concrete Plant	999	Not Available
15W 40 Motor Oil	Gallaher Concrete Plant	999	2500 mg/m^3
Calcium Chloride	Gallaher Concrete Plant	4,999	Not Available
Fly Ash	Gallaher Concrete Plant	9,999	Not Available
Hydraulic Oil	Gallaher Concrete Plant	999	2500 mg/m^3
Portland Cement (CAS 65997151)	Gallaher Concrete Plant	9,999	5000 mg/m^3
Knoxville PDC – Volkswagen Distributor			
Lead	Knoxville PDC - Volkswagen Group of America	49,999: maximum in largest container 1,890	100 mg/m^3

Table 2.2-2R Nearby Facilities Chemical Storage
(Sheet 5 of 13)

Chemical	Facility/Location	Quantity (lbs)[a] (unless otherwise noted)	Toxicity Limit (IDHL) [b]
Sulfuric Acid	Knoxville PDC – Volkswagen Group of America	4,999: maximum in largest container 200	15 mg/m^3
Linde, Inc.			
Anhydrous Ammonia	Linde, Inc., Loudon, TN	9,640	210 mg/m^3 (300 ppm)
Carbon Dioxide	Linde, Inc., Loudon, TN	1,800,000	40,000 ppm
Hydrocat	Linde, Inc., Loudon, TN	120,000	Not Available
NAC Philotechnics – Horizon Center			
Lead	Horizon Center	24,999, max amount in largest container 2,500	100 mg/m^3
Oak Ridge National Laboratory			
Acrylonitrile	Oak Ridge National Laboratory	12,039	149 mg/m^3
Ammonia	Oak Ridge National Laboratory	1,655	210 mg/m^3 (300 ppm)
Carbon Dioxide	Oak Ridge National Laboratory	303,887	40,000 ppm
Carbon Disulfide, ACS Reagent, >=99.9%	Oak Ridge National Laboratory	41	500 ppm
Chlorine	Oak Ridge National Laboratory	52	10 ppm
Chloroform	Oak Ridge National Laboratory	906	500 ppm
Cyclohexylamine (CAS 108918)	Oak Ridge National Laboratory	156	Not Available
Diesel Fuel	Oak Ridge National Laboratory	1,926,045	Not Available
Ethylenediamine (CAS 107153)	Oak Ridge National Laboratory	35	1000 ppm
Formaldehyde Solution (CAS 50000)	Oak Ridge National Laboratory	97	20 ppm
Fuel Oil, [No. 2]	Oak Ridge National Laboratory	27,666	Not Available

Table 2.2-2R Nearby Facilities Chemical Storage
(Sheet 6 of 13)

Chemical	Facility/Location	Quantity (lbs)[a] (unless otherwise noted)	Toxicity Limit (IDHL) [b]
Gasoline	Oak Ridge National Laboratory	44,948	750 ppm
Helium (CAS 7440597)	Oak Ridge National Laboratory	32,223	Asphyxiant
Hydrofluoric Acid (Anhydrous Gas >/=48.3%) (CAS 7664393)	Oak Ridge National Laboratory	348	30 ppm
Hydrogen Chloride, Anhydrous (CAS 7647010)	Oak Ridge National Laboratory	66	50 ppm
Hydrogen Peroxide (Conc.>52%) (CAS 7722841)	Oak Ridge National Laboratory	545	75 ppm
Hydroquinone (CAS 123319)	Oak Ridge National Laboratory	29	50 mg/m^3
Lead	Oak Ridge National Laboratory	103,870	100 mg/m^3
Mercury	Oak Ridge National Laboratory	46,535	10 mg/m^3
Methyltrichlorosilane (CAS 75796)	Oak Ridge National Laboratory	140	0.5 ppm
Nitric Acid, <=100%	Oak Ridge National Laboratory	28,826	25 ppm
Nitric Oxide (CAS 10102439)	Oak Ridge National Laboratory	60	100 ppm
Nitrobenzene (CAS 98953)	Oak Ridge National Laboratory	31	200 ppm
Nitrogen Dioxide (CAS 10102440)	Oak Ridge National Laboratory	262	13 ppm
Peracetic Acid (CAS 79210)	Oak Ridge National Laboratory	104	Not Available
Phenol Solution (CAS 108952)	Oak Ridge National Laboratory	182	250 ppm
Sulfuric Acid	Oak Ridge National Laboratory	343,247	15 mg/m^3
Vanadium(V) Oxide	Oak Ridge National Laboratory	35	35 mg/m^3

Table 2.2-2R Nearby Facilities Chemical Storage
(Sheet 7 of 13)

Chemical	Facility/Location	Quantity (lbs)[a] (unless otherwise noted)	Toxicity Limit (IDHL) [b]
Vinyl Acetate (CAS 108054)	Oak Ridge National Laboratory	1,589	Not Available
Oak Ridge National Laboratory – UCOR			
Diesel Fuel	Oak Ridge National Laboratory - UCOR LLC - UCOR Rollup	24,999	Not Available
Ferric Sulfate (CAS 10028225)	Oak Ridge National Laboratory - UCOR LLC - UCOR Rollup	9,999	Not Available
Lead	Oak Ridge National Laboratory - UCOR LLC - UCOR Rollup	49,999	100 mg/m^3
Nitric Acid	Oak Ridge National Laboratory - UCOR LLC - UCOR Rollup	24,999	25 ppm
Oils, Misc. Lubricating	Oak Ridge National Laboratory - UCOR LLC - UCOR Rollup	24,999	Not Available
Silica, Crystalline Quartz	Oak Ridge National Laboratory - UCOR LLC - UCOR Rollup	49,999	3000 mg/m^3
Sodium Hydroxide Solutions	Oak Ridge National Laboratory - UCOR LLC - UCOR Rollup	499,999	10 mg/m^3
Sodium Polyacrylate	Oak Ridge National Laboratory - UCOR LLC - UCOR Rollup	24,999	Not Available
Sulfuric Acid	Oak Ridge National Laboratory - UCOR LLC - UCOR Rollup	74,999	15 mg/m^3
Oak Ridge Water Treatment Plant			
Chlorine	Oak Ridge Water Treatment Plant	16,000	29mg/m^3 (10 ppm)
Primary Products Ingredients			
1,3 -Propanediol CAS 504632	Primary Products Ingredients Americas, LLC	2,925,000	Not Available

Table 2.2-2R Nearby Facilities Chemical Storage
(Sheet 8 of 13)

Chemical	Facility/Location	Quantity (lbs)[(a)] (unless otherwise noted)	Toxicity Limit (IDHL) [(b)]
3D Trasar 3DT294 CAS 7664939	Primary Products Ingredients Americas, LLC	28,000	30 ppm
Actisorb S 7 RS CAS 1344281	Primary Products Ingredients Americas, LLC	39,500	Not Available
Alkabuild, NO. 393	Primary Products Ingredients Americas, LLC	54,400	Not Available
Anhydrous Ammonia	Primary Products Ingredients Americas, LLC	190,539	210 mg/m^3 (300 ppm)
Corn Gluten Feed	Primary Products Ingredients Americas, LLC	9,300,000: maximum in largest container 3,200,000	Not Available
Corn Gluten Meal	Primary Products Ingredients Americas, LLC	5,100,000: 3,400,000 maximum in largest container	Not Available
Crystalline Fructose	Primary Products Ingredients Americas, LLC	270,738: 167,134 maximum in largest container	Not Available
DCI-11 Plus If1	Primary Products Ingredients Americas, LLC	45,810	Not Available
Diatomaceous Earth	Primary Products Ingredients Americas, LLC	554,000	Not Available
Dipotassium Phosphate	Primary Products Ingredients Americas, LLC	67,500	Not Available
Ethyl Alcohol	Primary Products Ingredients Americas, LLC	8,572,893, 985,500 maximum in largest container	Not Available
Extenda Go	Primary Products Ingredients Americas, LLC	112,264: 54,527 maximum in largest container	Not Available
Fermentation ByProduct	Primary Products Ingredients Americas, LLC	168,000	Not Available
Gemsweet IGI-VHF	Primary Products Ingredients Americas, LLC	9,900	Not Available

Table 2.2-2R Nearby Facilities Chemical Storage
(Sheet 9 of 13)

Chemical	Facility/Location	Quantity (lbs)[a] (unless otherwise noted)	Toxicity Limit (IDHL) [b]
Glucoamylase (glucan 1,4-alpha-glucosidase)	Primary Products Ingredients Americas, LLC	123,513: 64,144 maximum in largest container	Not Available
Hydrochloric Acid	Primary Products Ingredients Americas, LLC	3,272,800: 3,020,000 maximum in largest container	50 ppm
Hydrogen	Primary Products Ingredients Americas, LLC	2,000	Asphyxiant
Kochkleen Enzyme	Primary Products Ingredients Americas, LLC	54.400: 2,560 maximum in largest container	Not Available
Lead Acid Batteries	Primary Products Ingredients Americas, LLC	349	Not Available
Liquefied Natural Gas (Cryogenic)	Primary Products Ingredients Americas, LLC	500,000: 50,000 maximum in largest container	Not Available
Liquezyme	Primary Products Ingredients Americas, LLC	130,203, 65,245 maximum in largest container	Not Available
Lphera	Primary Products Ingredients Americas, LLC	33,642	Not Available
Magnesium Bisulfite	Primary Products Ingredients Americas, LLC	122,132, 62,776 maximum in largest container	Not Available
Magnesium Hydroxide	Primary Products Ingredients Americas, LLC	270,000: 108,000 maximum in largest container	Not Available
Magnesium Sulfate	Primary Products Ingredients Americas, LLC	60,000	Not Available
Maltodextrin	Primary Products Ingredients Americas, LLC	810,000: 180,000 maximum in largest container	Not Available
Molten Sulfur	Primary Products Ingredients Americas, LLC	8,680	Not Available
Nalco 19 PULV	Primary Products Ingredients Americas, LLC	4,800	Not Available

Table 2.2-2R Nearby Facilities Chemical Storage
(Sheet 10 of 13)

Chemical	Facility/Location	Quantity (lbs)[a] (unless otherwise noted)	Toxicity Limit (IDHL) [b]
Natural Gasoline (Ethane)	Primary Products Ingredients Americas, LLC	440,000: 240,000 maximum in largest container	750 ppm
Nitrogen	Primary Products Ingredients Americas, LLC	113,626: 63,900 maximum in largest container	Asphyxiant
Phosphoric Acid	Primary Products Ingredients Americas, LLC	48,000	1000 ppm
Rock Salt	Primary Products Ingredients Americas, LLC	42,000	Not Available
Sodium Carbonate, Anhydrous	Primary Products Ingredients Americas, LLC	1,889,000	Not Available
Sodium Hydroxide, [Liquid]	Primary Products Ingredients Americas, LLC	2,097,353: maximum in largest container 893,200	10 mg/m^3
Sodium Hypochlorite	Primary Products Ingredients Americas, LLC	120,871: maximum in largest container 65,280	10 ppm as Cl
Spotleak 1009	Primary Products Ingredients Americas, LLC	8,800	Not Available
Sulfuric Acid	Primary Products Ingredients Americas, LLC	151,330, maximum in largest container 97,302	15 mg/m^3
Tri-Act 1820	Primary Products Ingredients Americas, LLC	8,800	Not Available
Unleaded Gasoline	Primary Products Ingredients Americas, LLC	8,000	750 ppm
Whole Grain-Combustible Dust	Primary Products Ingredients Americas, LLC	10,672	Not Available
Processed Food Corporation			
Anhydrous Ammonia	Processed Food Corporation	19,000	210 mg/m^3 (300 ppm)

Table 2.2-2R Nearby Facilities Chemical Storage
(Sheet 11 of 13)

Chemical	Facility/Location	Quantity (lbs)[a] (unless otherwise noted)	Toxicity Limit (IDHL) [b]
Tellico Area Service System Water Treatment Plant			
Chlorine	Tellico Area Service System WTP	4,000	29mg/m^3 (10 ppm)
Tennessee Valley Authority (TVA) – Kingston Fossil Plant			
Acetic Acid	TVA-Kingston Fossil Plant	20,000	50 ppm
Anhydrous Ammonia	TVA-Kingston Fossil Plant	999,999, maximum in largest container 160,000	210 mg/m^3 (300 ppm)
Biocide Nalco H550 and BIOMATE-MBC2881	TVA-Kingston Fossil Plant	2,600	Not Available
Bottom Ash	TVA-Kingston Fossil Plant	1,000,000,000	Not Available
Calcium Bromide Mercontrol	TVA-Kingston Fossil Plant	74,999	Not Available
Carbon Dioxide CO_2	TVA-Kingston Fossil Plant	74,999	40,000 ppm
Caustic Soda 50% and 25%	TVA-Kingston Fossil Plant	9,999	10 mg/m^3
Coal	TVA-Kingston Fossil Plant	1,000,000,000	80 mg/m^3 for coal dust
Defoamer	TVA-Kingston Fossil Plant	99	Not Available
Fly Ash	TVA-Kingston Fossil Plant	1,000,000,000	Not Available
Fuel Oil (Diesel)	TVA-Kingston Fossil Plant	9,999,999	Not Available
Gasoline	TVA-Kingston Fossil Plant	99,999	750 ppm
Gypsum	TVA-Kingston Fossil Plant	99,999,999	Not Available
Hydrogen	TVA-Kingston Fossil Plant	99,999	Asphyxiant
Lead	TVA-Kingston Fossil Plant	99,999	100 mg/m^3
Limestone	TVA-Kingston Fossil Plant	50,000,000	Not Available
Lubricating Oil	TVA-Kingston Fossil Plant	999,999	Not Available
Mineral Oil	TVA-Kingston Fossil Plant	9,999,999	2500 mg/m
Poly Aluminum Chloride	TVA-Kingston Fossil Plant	13,000	Not Available
Rock Salt Halite	TVA-Kingston Fossil Plant	10,000	Not Available
Sodium Sulphide Mercontrol	TVA-Kingston Fossil Plant	999,999	Not Available
Sulfuric Acid	TVA-Kingston Fossil Plant	9,999	15 mg/m^3
Trisodium Phosphate	TVA-Kingston Fossil Plant	4,999	Not Available
Tennessee Valley Authority (TVA) – Melton Hill Hydro Plant			
Lead	Melton Hill Hydro Plant	8,580	100 mg/m^3

Table 2.2-2R Nearby Facilities Chemical Storage
(Sheet 12 of 13)

Chemical	Facility/Location	Quantity (lbs)[(a)] (unless otherwise noted)	Toxicity Limit (IDHL) [(b)]
Sulfuric Acid (Aqueous 30%)	Melton Hill Hydro Plant	3,822	15 mg/m^3
U.S. Foodservice – Knoxville			
Anhydrous Ammonia	U.S. Foodservice - Knoxville	17,000	210 mg/m^3 (300 ppm)
Weigel's			
Diesel Fuel #2	Weigel's #58	363,125	Not Available
Gasoline	Weigel's #58	160,225	750 ppm
Propane	Weigel's #58	372	2100 ppm
Urea/DEF	Weigel's #58	74,999	Not Available
West Knox Utility District Water Treatment Plant and Wastewater Treatment Plant			
Chlorine	West Knox Utility District WTP	12,000	29mg/m^3 (10 ppm)
Chlorine	West Knox Utility District WWTP	6,000	29mg/m^3 (10 ppm)
Yale Security			
Chlorine	Yale Security Inc.	8,000	29mg/m^3 (10 ppm)

Table 2.2-2R Nearby Facilities Chemical Storage
(Sheet 13 of 13)

Chemical	Facility/Location	Quantity (lbs)[a] (unless otherwise noted)	Toxicity Limit (IDHL) [b]

a) Where a capacity number was obtained from the Superfund Amendments and Reauthorization Act (SARA) Title III, Tier II report, or the RMP report, the upper range number is shown and was used in the analysis.

b) Immediately Dangerous to Life or Health. “Not Available” indicates that there has not been a toxicity limit established for this chemical.

c) Gasoline does not have an identified IDLH. The Threshold Limit Value-Short Term Exposure Limit TLV-STEL is 500 ppm; the Threshold Limit Value-Time-weighted Average (TLV-TWA) is 300 ppm; and the Protective Action Criteria (PAC) PAC-2 guideline is 1000 ppm for gasoline. For the analyses, n-Heptane is used as a surrogate and has an IDLH of 750 ppm. This selection is conservative given the PAC-2 guideline most closely correlates with the definition of IDLH.

Notes:

URS Corporation (URS)

Parts per million (ppm)

Milligram per cubic meter (mg/m^3)

Immediately Dangerous to Life or Health (IDLH)

Water Treatment Plant (WTP)

Wastewater Treatment Plant (WWTP)

Time-Weighted Average (TWA)

Sources: Reference 2.2-4, Reference 2.2-11, Reference 2.2-13 and Reference 2.2-28

Replace Table 2.2-3 with the following:

Table 2.2-3R Hazardous Materials Potentially Transported Along I-40 in the CRN Site Vicinity
(Sheet 1 of 2)

Chemical	Quantity (lbs)[a]	Toxicity Limit (IDHL)
Acrylonitrile	12,039[h]	85 ppm
Anhydrous Ammonia	11,500 gal[i]	300 ppm
Argon	50,000[c]	Asphyxiant
Butane	11,500 gal[i]	Asphyxiant
Carbon Dioxide	50,000[c]	40,000 ppm
Chlorine	44,000[d]	10 ppm
Chlorine Dioxide Solution (500 - 5000 mg/l)	24,131[h]	5 ppm
Chloroform	906[h]	500 ppm
Gasoline	50,000[c][f]	750 ppm
Helium	50,000[c]	Asphyxiant
HFC-134A	50,000[c]	Asphyxiant
Hydrochloric Acid	6,000 gal[g]	50 ppm
Hydrofluoric Acid	499[h]	30 ppm
Hydrogen	15,032.84 ft^3/tube[e]	Asphyxiant[b]
Hydrogen Chloride, Anhydrous	66[h]	50 ppm
Liquid Nitrogen	8,910[h]	Asphyxiant
Methanol	50,000[c]	6,000 ppm
Methyl Chloroform	50,000[c]	700 ppm
Methyltrichlorosilane	140[h]	0.5 ppm
Nitric Acid	6,000 gal[g]	25 ppm
Nitric Oxide	60[h]	100 ppm
Nitrogen	50,000[c]	Asphyxiant
Nitrogen Dioxide	262[h]	13 ppm
Propane	50,000[c]	2100 ppm
Sulfur Dioxide	50,000[c]	100 ppm

Table 2.2-3R Hazardous Materials Potentially Transported Along I-40 in the CRN Site Vicinity
(Sheet 2 of 2)

Chemical	Quantity (lbs)[(a)]	Toxicity Limit (IDHL)

a) Quantity is in lbs unless otherwise specified.

b) Gaseous hydrogen is not considered as a toxicity concern due to the extreme buoyancy. That is, hydrogen gas would rise extremely rapidly and not cause a travelling vapor cloud.

c) Per RG 1.91, the maximum probable cargo for a single highway truck is 50,000 lb and used for the quantity transported unless a more appropriate value could be determined.

d) Chlorine gas quantity determined from The Chlorine Institute Bulk Storage of Liquid Chlorine (Reference 2.2-18).

e) Hydrogen gas quantity determined from Weldship Corporation super jumbo tube product specifications (the largest size tube available) (Reference 2.2-29).

f) The maximum highway cargo capacity, 50,000 lb provided in RG 1.91 was converted to gal for gasoline.

g) The maximum capacity of MC-312/DOT412 corrosive tanker is 6000 gal.

h) Transport quantity is assumed to be the same as the site storage quantity.

i) The maximum capacity of MC-331 high pressure tank truck is 11,500 gal per 49 CFR 137.315.

Notes:

Parts per million (ppm)

Immediately Dangerous to Life or Health (IDLH)

Time-weighted Average (TWA)

Sources: Reference 2.2-4, Reference 2.2-13, Reference 2.2-18, Reference 2.2-28 and Reference 2.2-29

Update Footnote in Table 2.2-4 to identify that the gas lines are now owned by Enbridge instead of Spectra. Reference 2.2-5 is still applicable.

Replace Table 2.2-5 with the following:

Table 2.2-5R Offsite Chemical Storage - Disposition
(Sheet 1 of 15)

Chemical	Quantity (lbs)	Toxicity Limit (IDLH)	Vapor Pressure	Flammable/ Explosive? Yes or No	Distance to CRN (miles)	Disposition
A.E. Staley Manufacturing / Tate & Lyle						
Sulphur Dioxide (Anhydrous)	144,000	100 ppm	2432 mm Hg at 68°F	No	11.05	Toxicity Analysis[a]
City of Maryville Wastewater Treatment						
Chlorine	12,000	10 ppm	7600 mm Hg @ 86°F (6.8 atm)	No	24.18	Toxicity Analysis[a]
City of Oak Ridge Municipal Pool						
Chlorine	4,000	10 ppm	7600 mm Hg @ 86°F (6.8 atm)	No	10.8	Toxicity Analysis[a]
Custom Foods of America						
Anhydrous Ammonia	28,600	210 mg/m^3 (300 ppm)	400 mm Hg @ 49.72°F	Yes 16% - 25%	23.65	Toxicity Analysis[a] Flammability and Explosion Analysis
Diversified Scientific Services						
Diesel Oil #2	14,520: 7,260 in largest container	Not Available	2.17 mm Hg @ 70°F	Yes 1.3% - 6%	3.38	No further analysis required[b]
Diesel Oil #2	99,000	Not Available	2.17 mm Hg @ 70°F	Yes 1.3% - 6%	3.38	No further analysis required[b]

Table 2.2-5R Offsite Chemical Storage - Disposition
(Sheet 2 of 15)

Chemical	Quantity (lbs)	Toxicity Limit (IDLH)	Vapor Pressure	Flammable/ Explosive? Yes or No	Distance to CRN (miles)	Disposition
Nonylphenol Ethoxylates	2,910: 485 amount in largest container	Not Available	Not pertinent	No	3.38	No further analysis required(h)
Sodium Hydroxide (20% - 40%)	10,000: 3,350 in largest container	10 mg/m3	Solid - in a solution	No	3.38	No further analysis required(c)
Triethanolamine Mixture	4,000: 490 in largest container	Not Available	Less than 0.01 mm Hg @ 68°F 10 mm Hg @ 401°F	No	3.38	No further analysis required(b)
Duracap Asphalt Paving						
Asphalt Mix	9,999,999, maximum in largest containers 4,000,000	Not Available	Data unavailable	Yes	2.28	No further analysis required(k)
Diesel Fuel	15,000, maximum in largest container 6,970	Not Available	2.17 mm Hg @ 70°F	Yes 1.3% - 6%	2.28	No further analysis required(b)
Limestone	11,000,000	Not Available	0 mm Hg	No	2.28	No further analysis required(g)

Table 2.2-5R Offsite Chemical Storage - Disposition
(Sheet 3 of 15)

Chemical	Quantity (lbs)	Toxicity Limit (IDLH)	Vapor Pressure	Flammable/ Explosive? Yes or No	Distance to CRN (miles)	Disposition
Liquid Asphalt (CAS 8052424)	499,999 maximum in largest container 215,000	Not Available	0.018 psi @ 210°F	Yes	2.28	No further analysis required[(o)]
ULTRATACK (CAS 8052424)	79,000	Not Available	Data unavailable	Yes	2.28	No further analysis required[(s)]
East Tennessee Mechanical Contractors						
Chlorine	12,000	29mg/m^3 (10 ppm)	7600 mm Hg @ 86°F (6.8 atm)	No	10.45	Toxicity Analysis[(a)]
East Tennessee Technology Park						
CETCO Quik-Solid Granular	24,999	Not Available	< 10 mm Hg	No	3.02	No further analysis required[(b)(l)]
Chemical Specialties Ultrapoles	49,999	5 mg /m^3 As, 250 mg/m^3 Cr	1 mm Hg @ 2941°F - Cr	No	3.02	No further analysis required[(b)(m)]
Diesel Fuel #2	49,999	Not Available	2.17 mm Hg @ 70°F	Yes 1.3% - 6%	3.02	No further analysis required[(b)]
Flextura FGM Erosion Control Agent	24,999	Not Available	0 mm Hg	Yes	3.02	No further analysis required[(b)(n)]
Gasoline	24,999	750 ppm	382.58 mm Hg	Yes	3.02	Toxicity Analysis Flammability and Explosion Analysis
Lead	2,499	100 mg/m^3	1.77 mm Hg	No	3.02	No further analysis required[(b)(f)]

Table 2.2-5R Offsite Chemical Storage - Disposition
(Sheet 4 of 15)

Chemical	Quantity (lbs)	Toxicity Limit (IDLH)	Vapor Pressure	Flammable/ Explosive? Yes or No	Distance to CRN (miles)	Disposition
Oils, Misc. Lubricating	74,999	Not Available	2.17 mm Hg @ 70“ F 0.042 psi	Yes	3.02	No further analysis required[(b)]
Silica, Crystalline Quartz	24,999	3000 mg/m^3	0 mm Hg	No	3.02	No further analysis required[(b)(r)]
Sodium Polyacrylate (CAS 9003047)	24,999	Not Available	< 10 mm Hg	No	3.02	No further analysis required[(b)]
Sulfuric Acid	4,999	15 mg/m^3	1 mm Hg @ 294.8°F	No	3.02	No further analysis required[(b)]
Energy Solution Services, Bear Creek Facility						
Caustic Soda (CAS 1310732)	64,000	10 mg/m^3	0 mm Hg	No	1.57	No further analysis required[(b)]
Diesel Fuel #2	27,824: maximum in largest container 7,150	Not Available	2.17 mm Hg @ 70°F	Yes 1.3% - 6%	1.57	No further analysis required[(b)]
Lead	200,000	100 mg/m^3	1.77 mm Hg	No	1.57	No further analysis required[(b)(f)]
Magnesium Oxide (CAS 1309484)	28,000: maximum in largest container 3,000	750 mg/m^3	0 mm Hg	No	1.57	No further analysis required[(b)]
Nitric Acid	645	64.5 mg/m^3 (25 ppm)	48 mm Hg	No	1.57	Toxicity Analysis

Table 2.2-5R Offsite Chemical Storage - Disposition
(Sheet 5 of 15)

Chemical	Quantity (lbs)	Toxicity Limit (IDLH)	Vapor Pressure	Flammable/ Explosive? Yes or No	Distance to CRN (miles)	Disposition
Oxygen (CAS 7782447)	86,890:, maximum in largest container 85,500	Not Available	Data unavailable	Yes	1.57	No further analysis required[(p)]
Propane	25,000:, maximum n largest container 20,000	2100 ppm	9823 mm Hg	Yes	1.57	Toxicity Analysis Flammability and Explosion Analysis
Sulfuric Acid	774: maximum in largest container 10	15 mg/m^3	1 mm Hg @ 294.8°F	No	1.57	No further analysis required[(b)]
First Utility District of Knox County Water Treatment Plant and Wastewater Treatment Plant						
Chlorine (WTP)	22,000	10 ppm	7600 mm Hg @ 86°F (6.8 atm)	No	14.23	Toxicity Analysis[(a)]
Chlorine (WWTP)	20,000	10 ppm	7600 mm Hg @ 86°F (6.8 atm)	No	13.4	Toxicity Analysis[(a)]
Food Lion – Delhaize DC #6						
Anhydrous Ammonia	16,000	300 ppm	400 mm Hg @ 49.72°F	Yes 16% - 25%	20.5	Toxicity Analysis[(a)] Flammability and Explosion Analysis
Fourth Creek Wastewater Treatment Plant						
Chlorine	16,000	10 ppm	7600 mm Hg @ 86°F (6.8 atm)	No	22.11	Toxicity Analysis[(a)]

Table 2.2-5R Offsite Chemical Storage - Disposition
(Sheet 6 of 15)

Chemical	Quantity (lbs)	Toxicity Limit (IDLH)	Vapor Pressure	Flammable/ Explosive? Yes or No	Distance to CRN (miles)	Disposition
Hallsdale Powell Utility District Melton Hill Water Treatment Plant						
Chlorine	22,000	10 ppm	7600 mm Hg @ 86°F (6.8 atm)	No	18.2	Toxicity Analysis[(a)]
Harrison Construction – Gallaher Concrete Plant						
#2 Grease	999	Not Available	2.17 mm Hg @ 70° F 0.042 psi	Yes	3.06	No further analysis required[(b)(i)]
15W 40 Motor Oil	999	2500 mg/m^3	2.17 mm Hg @ 70° F 0.042 psi	Yes	3.06	No further analysis required[(b)]
Calcium Chloride	4,999	Not Available	Solid in solution	No	3.06	No further analysis required[(d)]
Fly Ash	9,999	Not Available	Data unavailable	No	3.06	No further analysis required[(e)]
Hydraulic Oil	999	2500 mg/m^3	2.17 mm Hg @ 70° F 0.042 psi	Yes	3.06	No further analysis required[(b)]
Portland Cement (CAS 65997151)	9,999	5000 mg/m^3	0 mm Hg	No	3.06	No further analysis required[(b)]
Knoxville PDC – Volkswagen Group of America						
Lead	49,999: maximum in largest container 1,890	100 mg/m^3	1.77 mm Hg	No	1.02	No further analysis required[(b)]

Table 2.2-5R Offsite Chemical Storage - Disposition
(Sheet 7 of 15)

Chemical	Quantity (lbs)	Toxicity Limit (IDLH)	Vapor Pressure	Flammable/ Explosive? Yes or No	Distance to CRN (miles)	Disposition
Sulfuric Acid	4,999: maximum in largest container 200	15 mg/m^3	1 mm Hg @ 294.8°F	No	1.02	No further analysis required[(b)]
Linde Inc.						
Anhydrous Ammonia	9,640	300 ppm	400 mm Hg @ 49.72°F	Yes 16% - 25%	10.91	Toxicity Analysis[(a)] Flammability and Explosion Analysis
NAC Philotechnics – Horizon Center						
Lead	24,999, maximum in largest container 2,500	100 mg/m^3	1.77 mm Hg	No	4.94	No further analysis required[(b)(f)]
Oak Ridge National Laboratory						
Acrylonitrile	12,039	149 mg/m^3	100 mm Hg at 73.4°F	Yes 3% - 7%	4.34	Toxicity Analysis Flammability and Explosion Analysis
Ammonia	1,655	300 ppm	400 mm Hg @ 49.72°F	Yes 16% - 25%	4.34	Toxicity Analysis[(j)] Flammability and Explosion Analysis
Carbon Dioxide	303,887	40,000 ppm	56.5 atm @ 68°F	No	4.34	Toxicity Analysis

Table 2.2-5R Offsite Chemical Storage - Disposition
(Sheet 8 of 15)

Chemical	Quantity (lbs)	Toxicity Limit (IDLH)	Vapor Pressure	Flammable/ Explosive? Yes or No	Distance to CRN (miles)	Disposition
Carbon Disulfide, ACS Reagent, >=99.9%	41	500 ppm	360 mm Hg at 77°F	Yes 1% - 50%	4.34	Toxicity Analysis Flammability and Explosion Analysis
Chlorine	52	10 ppm	7600 mm Hg @ 86°F (6.8 atm)	No	4.34	Toxicity Analysis
Chloroform	906	500 ppm	160 mm Hg	No	4.34	Toxicity Analysis
Cyclohexylamine (CAS 108918)	156	Not Available	11 mm Hg	Yes 1.5% - 9.4 %	4.34	No further analysis required[(b)]
Diesel Fuel	1,926,045	Not Available	2.17 mm Hg @ 70°F	Yes 1.3% - 6%	4.34	No further analysis required[(b)]
Ethylenediamine (CAS 107153)	35	1000 ppm	10.7 mm Hg at 68°F	Yes 4.2% - 14.4%	4.34	No further analysis required[(b)]
Formaldehyde Solution (Cas 50000)	97	20 ppm	10 mm Hg at -126.4°F	Yes	4.34	No further analysis required[(b)]
Fuel Oil, [no. 2]	27,666	Not Available	2.17 mm Hg @ 70°F	Yes 1.3% - 6%	4.34	No further analysis required[(b)]
Gasoline	44,948	750 ppm	382.58 mm Hg	Yes	4.34	Toxicity Analysis Flammability and Explosion Analysis
Helium (CAS 7440597)	32,223	Asphyxiant	Data unavailable	No	4.34	Toxicity Analysis
Hydrofluoric Acid (Anhydrous Gas >/=48.3%) (CAS 7664393)	348	30 ppm	400 mm Hg at 36.5°F	No	4.34	Toxicity Analysis

Table 2.2-5R Offsite Chemical Storage - Disposition
(Sheet 9 of 15)

Chemical	Quantity (lbs)	Toxicity Limit (IDLH)	Vapor Pressure	Flammable/ Explosive? Yes or No	Distance to CRN (miles)	Disposition
Hydrogen Chloride, Anhydrous (CAS 7647010)	66	50 ppm	32452 mm Hg at 70°F ; 760 mm Hg at -120.6°F	No	4.34	Toxicity Analysis
Hydrogen Peroxide (CONC.>52%) (CAS 7722841)	545	75 ppm	1 mm Hg at 59.54°F	No	4.34	No further analysis required(b)
Hydroquinone (CAS 123319)	29	50 mg/m^3	4 mm Hg at 302°F	No	4.34	No further analysis required(b)
Lead	103,870	100 mg/m^3	1.77 mm Hg	No	4.34	No further analysis required(b)(f)
Mercury	46,535	10 mg/m^3	0.0012 mm Hg	No	4.34	No further analysis required(b)
Methyltrichlorosilane (CAS 75796)	140	0.5 ppm	167 mm Hg at 25° C	Yes 7.6% - 20%	4.34	Toxicity Analysis Flammability and Explosion Analysis
Nitric Acid, <=100%	28,826	25 ppm	48 mm Hg	No	4.34	Toxicity Analysis
Nitric Oxide (CAS 10102439)	60	100 ppm	26000 mm Hg at 68°F	No	4.34	Toxicity Analysis
Nitrobenzene (CAS 98953)	31	200 ppm	1 mm Hg at 111.92°F	Yes	4.34	No further analysis required(b)
Nitrogen Dioxide (CAS 10102440)	262	13 ppm	720 mm Hg at 68°F	No	4.34	Toxicity Analysis
Peracetic Acid (CAS 79210)	104	Not Available	14.5 mm Hg	Yes	4.34	No further analysis required(q)

Table 2.2-5R Offsite Chemical Storage - Disposition
(Sheet 10 of 15)

Chemical	Quantity (lbs)	Toxicity Limit (IDLH)	Vapor Pressure	Flammable/ Explosive? Yes or No	Distance to CRN (miles)	Disposition
Phenol Solution (CAS 108952)	182	250 ppm	0.2 mm Hg at 68°F ; 0.35 mm Hg at 77°F	No	4.34	No further analysis required[(b)]
Sulfuric Acid	343,247	15 mg/m^3	1 mm Hg @ 294.8°F	No	4.34	No further analysis required[(b)]
Vanadium(V) Oxide	35	35 mg/m^3	0 at 68F	No	4.34	No further analysis required[(b)]
Vinyl Acetate (Cas 108054)	1,589	Not Available	83 to 140 mm Hg at 68 to 86 °F	Yes 2.6% - 13.4%	4.34	No further analysis required[(t)]
Oak Ridge National Laboratory – UCOR						
Diesel Fuel	24,999	Not Available	2.17 mm Hg @ 70°F	Yes 1.3% - 6%	4.87	No further analysis required[(b)]
Ferric Sulfate (CAS 10028225)	9,999	Not Available	Data unavailable	No	4.87	No further analysis required[(v)]
Lead	49,999	100 mg/m^3	1.77 mm Hg	No	4.87	No further analysis required[(b)(f)]
Nitric Acid	24,999	25 ppm	48 mm Hg	No	4.87	Toxicity Analysis
Oils, Misc. Lubricating	24,999	Not Available	2.17 mm Hg @ 70“ F 0.042 psi	Yes	4.87	No further analysis required[(b)]
Silica, Crystalline Quartz	49,999	3000 mg/m^3	0 mm Hg	No	4.87	No further analysis required[(b)(r)]
Sodium Hydroxide Solutions	499,999	10 mg/m^3	Solid in solution	No	4.87	No further analysis required[(b)(c)]
Sodium Polyacrylate	24,999	Not Available	< 10 mm Hg	No	4.87	No further analysis required[(b)]

Table 2.2-5R Offsite Chemical Storage - Disposition
(Sheet 11 of 15)

Chemical	Quantity (lbs)	Toxicity Limit (IDLH)	Vapor Pressure	Flammable/ Explosive? Yes or No	Distance to CRN (miles)	Disposition
Sulfuric Acid	74,999	15 mg/m^3	1 mm Hg @ 294.8°F	No	4.87	No further analysis required[(b)]
Oak Ridge Water Treatment Plant						
Chlorine	16,000	10 ppm	7600 mm Hg @ 86°F (6.8 atm)	No	10.47	Toxicity Analysis[(a)]
Primary Products Ingredients						
Anhydrous Ammonia	190,539	210 mg/m^3 (300 ppm)	400 mm Hg @ 49.72°F	Yes 16% - 25%	10.97	Toxicity Analysis[(a)] Flammability and Explosion Analysis
Processed Food Corporation						
Anhydrous Ammonia	19,000	210 mg/m^3 (300 ppm)	400 mm Hg @ 49.72°F	Yes 16% - 25%	23.65	Toxicity Analysis[(a)] Flammability and Explosion Analysis
Tellico Area Service System Water Treatment Plant						
Chlorine	4,000	10 ppm	7600 mm Hg @ 86°F (6.8 atm)	No	21.8	Toxicity Analysis[(a)]
Tennessee Valley Authority (TVA) – Kingston Fossil Plant						
Anhydrous Ammonia	999,999, maximum in largest container 160,000	210 mg/m^3 (300 ppm)	400 mm Hg @ 49.72°F	Yes 16% - 25%	7.74	Toxicity Analysis[(a)] Flammability and Explosion Analysis
Tennessee Valley Authority (TVA) – Melton Hill Hydro Plant						
Lead	8,580	100 mg/m^3	1.77 mm Hg	No	4.66	No further analysis required[(b)(f)]

Table 2.2-5R Offsite Chemical Storage - Disposition
(Sheet 12 of 15)

Chemical	Quantity (lbs)	Toxicity Limit (IDLH)	Vapor Pressure	Flammable/ Explosive? Yes or No	Distance to CRN (miles)	Disposition
Sulfuric Acid (Aqueous 30%)	3,822	15 mg/m^3	1 mm Hg @ 294.8°F	No	4.66	No further analysis required[(b)]
U.S. Foodservice – Knoxville						
Anhydrous Ammonia	17,000	210 mg/m^3 (300 ppm)	400 mm Hg @ 49.72°F	Yes 16% - 25%	23.32	Toxicity Analysis[(a)] Flammability and Explosion Analysis
Weigel's #58						
Diesel Fuel #2	363,125	Not Available	2.17 mm Hg @ 70°F	Yes 1.3% - 6%	3.67	No further analysis required[(b)]
Gasoline	160,225	750 ppm	382.58 mm Hg	Yes	3.67	Toxicity Analysis Flammability and Explosion Analysis
Propane	372	2100 ppm	9823 mm Hg	Yes	3.67	Toxicity Analysis Flammability and Explosion Analysis
Urea/DEF	74,999	Not Available	Data unavailable	No	3.67	No further analysis required[(b)(u)]
West Knox Utility District Water Treatment Plant and Wastewater Treatment Plant						
Chlorine (WTP)	12,000	10 ppm	7600 mm Hg @ 86°F (6.8 atm)	No	9.1	Toxicity Analysis[(a)]
Chlorine (WWTP)	6,000	10 ppm	7600 mm Hg @ 86°F (6.8 atm)	No	14.37	Toxicity Analysis[(a)]

Table 2.2-5R Offsite Chemical Storage - Disposition
(Sheet 13 of 15)

Chemical	Quantity (lbs)	Toxicity Limit (IDLH)	Vapor Pressure	Flammable/ Explosive? Yes or No	Distance to CRN (miles)	Disposition
Yale Security						
Chlorine	8,000	10 ppm	7600 mm Hg @ 86°F (6.8 atm)	No	9.5	Toxicity Analysis[a]

Notes:

a) Per RG 1.78, chemicals stored or situated at distances greater than 5 miles from the plant need not be considered because atmospheric dispersion will dilute and disperse the incoming plume such that toxic limits will never be reached. The exceptions are those chemicals considered very hazardous substances per 40 CFR 68.130, chemicals such as ammonia, chlorine, and sulfur dioxide. These very hazardous substances require an offsite consequence analysis for Risk Management Programs required under the Clean Air Act consisting of a worst-case release scenario and/or alternative release scenarios with a toxic endpoint evaluated at facilities greater than 5 miles from Clinch River (as defined in 40 CFR 68.25).

b) According to RG 1.78, if a substance's vapor pressure is below 10 torr (approximately 10 mm Hg); the vapor pressure is sufficiently low such that the formation of a vapor cloud is not a likely event. Therefore, an air dispersion hazard is not a likely exposure route.

c) Sodium hydroxide solution is a dark, thick liquid that is a severe irritant and is toxic and corrosive. It will not burn under typical fire conditions and is normally stable but it unstable at elevated temperatures and pressure.

d) Calcium Chloride is a white to off-white solid in solution and does not burn. No IDLH, LEL, UEL, AEGLs, or ERPGs are established and the protective action criteria (PAC-1) is 3 mg/m3, PAC-2 is 33 mg/m3 and PAC-3 is 200 mg/m3. An air dispersion hazard is not a likely exposure route.

e) Fly Ash is a noncombustible solid and the vapor pressure is not applicable. It is stable under most conditions, thus, an air dispersion hazard is not a likely route of exposure.

f) Lead, in its pure form, is a noncombustible solid and, therefore, has a very low vapor pressure. No toxicity limit in air is established for this chemical. Therefore, an air dispersion hazard is not a likely route of exposure.

g) Limestone is a noncombustible solid and has a very low vapor pressure. No toxicity limit is established for this chemical; therefore an air dispersion hazard is not a likely route of exposure.

Table 2.2-5R Offsite Chemical Storage - Disposition
(Sheet 14 of 15)

Chemical	Quantity (lbs)	Toxicity Limit (IDLH)	Vapor Pressure	Flammable/ Explosive? Yes or No	Distance to CRN (miles)	Disposition

h) Nonylphenol Ethoxylates (NPE) is a colorless liquid or white solid with a mild odor. It does not react rapidly with air or water and no toxicity limit is established for this chemical, therefore an air dispersion hazard is not a likely route of exposure.

i) Considered to be equivalent to Oils for hazard analysis.

j) The facility does not specify if this substance is in an aqueous solution or is anhydrous. Anhydrous is assumed.

k) 95% limestone (CAS 1317653) and 5% liquid asphalt (CAS 8052424). Data is for liquid asphalt. Not listed in RG 1.78.

l) An advanced super adsorbent media for containing aqueous matter in critical wastes. The specialty adsorbent is capable of adsorbing as much as 350 times (by weight) of water.

m) 3% arsenic acid CAS 7778394, 4% chromium CAS 7440473, 2% Copper oxide, 91% sour water CAS 7732185.

n) Green dyed wood fibers, biodegradable fibers, minerals and a proprietary binder mixture.

o) Similar to asphalt cement.

p) Oxygen is a colorless, odorless and tasteless gas. It will support life. It is noncombustible, but will actively support the burning of combustible materials. Pure oxygen is nonflammable.

q) Peracetic acid is a colorless liquid with a strong, pungent acrid odor. Used as a bactericide and fungicide, especially in food processing; as a reagent in making caprolactam and glycerol; as an oxidant for preparing epoxy compounds; as a bleaching agent; a sterilizing agent. No toxicity limit is established for this chemical; thus, an air dispersion hazard is not a likely route of exposure

r) Silica, or silicon dioxide (SiO_2) naturally occurs in both crystalline and amorphous forms.

s) ULTRACK is an asphalt emulsion that yields a fast setting, non-tack coat. It may also be used in prime coat applications. The product is made primarily of asphalt (CAS 8052-42-4) and water. No toxicity limit is established for this chemical; thus, an air dispersion hazard is not a likely route of exposure.

t) Vinyl Acetate is clear colorless liquid which is slightly soluble in water. Used to make adhesives, paints, and plastics. No toxicity limit is established for this chemical; thus, an air dispersion hazard is not a likely route of exposure

u) Urea is solid, odorless, white crystals or pellets that are noncombustible. No toxicity limit is established for this chemical; thus, an air dispersion hazard is not a likely route of exposure.

Table 2.2-5R Offsite Chemical Storage - Disposition
(Sheet 15 of 15)

Chemical	Quantity (lbs)	Toxicity Limit (IDLH)	Vapor Pressure	Flammable/ Explosive? Yes or No	Distance to CRN (miles)	Disposition

v) Ferric sulfate is a yellow crystalline solid or a grayish-white powder. It is used for water purification, and as a soil conditioner. No toxicity limit is established for this chemical; thus, an air dispersion hazard is not a likely route of exposure.

Notes:

Immediately Dangerous to Life or Health (IDLH)

Parts per million (ppm)

Millimeters of mercury (mm Hg)

Time-weighted Average (TWA)

Pascal (Pa)

Kelvin (K)

Atmosphere (atm)

Water Treatment Plant (WTP)

Wastewater Treatment Plant (WWTP)

Replace Table 2.2-6 with the following:

Table 2.2-6R Disposition of Hazardous Materials Potentially Transported on Interstate I-40
(Sheet 1 of 3)

Chemical	Toxicity Limit (IDHL)	Vapor Pressure	Flammability	Explosion Hazard	Disposition
Acrylonitrile	85 ppm	100 mm Hg at 73.4°F	3 - 17%	Vapor may explode	Toxicity Analysis[(b)]Explosion Analysis[(b)]
Anhydrous Ammonia	300 ppm	400 mm Hg @ -49.72°F	16 - 25%	Vapor may explode [(e)]	Toxicity Analysis[(a)]
Argon	Asphyxiant	1,044,630 Pa @ 117.3K	Not Flammable	None Listed	Toxicity Analysis[(b)]
Butane	Asphyxiant	760 mm Hg @ 31.1°F	1.6-8.4%	Vapor may explode	Toxicity Analysis[(b)]Flammability Analysis / Explosion Analysis[(b)]
Carbon Dioxide	40,000 ppm	56.5 atm @ 68°F	Not Flammable	None Listed	Toxicity Analysis[(b)]
Chlorine	10 ppm	7600 mm Hg @ 86°F (6.8 atm)	Not Flammable	None Listed	Toxicity Analysis[(c)]
Chlorine Dioxide Solution (500 - 5000 mg/l)	5 ppm	> 1 atm	Not Flammable	None Listed	Toxicity Analysis[(b)]
Chloroform	500 ppm	160 mm Hg	Not Flammable	None Listed	Toxicity Analysis[(b)]
Gasoline	750 ppm	382.58 mm Hg	1.4 - 7.6%	Vapor may explode	Toxicity Analysis[(b)]Flammability Analysis / Explosion Analysis [(b)]
Helium	Asphyxiant	Data unavailable	Not Flammable	None Listed	Toxicity Analysis[(b)]
HFC-134A	Asphyxiant	4990 mm Hg at 25°C	Not Flammable	None Listed	Toxicity Analysis[(b)]

Table 2.2-6R Disposition of Hazardous Materials Potentially Transported on Interstate I-40
(Sheet 2 of 3)

Chemical	Toxicity Limit (IDHL)	Vapor Pressure	Flammability	Explosion Hazard	Disposition
Hydrochloric Acid	50 ppm	413.6 mm Hg	Not Flammable	None Listed	Toxicity Analysis[a]
Hydrofluoric Acid	30 ppm	400 mm Hg at 36.5°F	Not Flammable	None Listed	Toxicity Analysis[b]
Hydrogen	Asphyxiant	1.231 psi @ 434°F- 4 - 75%	Vapor may explode		Explosion Analysis[b][d]
Hydrogen Chloride, Anhydrous	50 ppm	32452 mm Hg at 70°F ; 760 mm Hg at -120.6°F	Not Flammable	None Listed	Toxicity Analysis[b]
Liquid Nitrogen	Asphyxiant	1.931 psi @ -344°F	Not Flammable	None Listed	Toxicity Analysis[b]
Methanol	6,000 ppm	100 mm Hg at 70.2°F	6 - 36.5%	Vapor may explode	Toxicity Analysis[b]Flammability / Explosive Analysis [b]
Methyl Chloroform	700 ppm	100 mm Hg at 68°F	7.5 - 12.5%	Vapor may explode	Toxicity Analysis[b] Flammability / Explosive Analysis [b]
Methyltrichlorosilane	0.5 ppm	167 mm Hg at 25	7.6 - 20%	Vapor may explode	Toxicity Analysis [b]Flammability / Explosive Analysis [b]
Nitric Acid	25 ppm	48 mm Hg	Not Flammable	None Listed	Toxicity Analysis [a]
Nitric Oxide	100 ppm	26000 mm Hg at 68°F	Not Flammable	None Listed	Toxicity Analysis [b]
Nitrogen	Asphyxiant	1.931 psi @ -344°F	Not Flammable	None Listed	Toxicity Analysis [b]
Nitrogen Dioxide	13 ppm	720 mm Hg at 68°F	Not Flammable	None Listed	Toxicity Analysis [b]

Table 2.2-6R Disposition of Hazardous Materials Potentially Transported on Interstate I-40
(Sheet 3 of 3)

Chemical	Toxicity Limit (IDHL)	Vapor Pressure	Flammability	Explosion Hazard	Disposition
Propane	2100 ppm	9823 mm Hg	2.1 - 9.5%	Vapor may explode	Toxicity Analysis[(b)] Flammability / Explosive Analysis[(b)]
Sulfur Dioxide	100 ppm	2432 mm Hg	Not Flammable	None Listed	Toxicity Analysis[(b)]

a) Single or multiple storage locations; maximum evaluated quantity greater than 50,000 lbs; maximum evaluated safe distance is greater than 1 mi for toxicity and less than 1 mi for flammable/explosive evaluations.

b) Single or multiple storage locations (For single storage location, assumed transport quantity is equivalent to maximum storage quantity. For multiple storage locations, assumed transport quantity is either based on RG 1.91 or the maximum tanker size); maximum evaluated safe distance is less than 1 mi for each accident category evaluated.

c) Multiple storage location; maximum evaluated quantity is less than 50,000 lbs; and maximum evaluated safe distance is greater than 1 mi.

d) Due to the buoyant nature of gaseous hydrogen such as in tube trailers, gaseous hydrogen is not considered to present an outdoor vapor cloud explosion potential or toxic threat to the control room.

e) Studies have shown that an ammonia-air mixture does not ignite at less than 1562°F. Conditions favorable for ignition are seldom encountered during normal operations due to high ignition temperature required.

Notes:

Immediately Dangerous to Life or Health (IDLH)

Millimeters of mercury (mm Hg)

Pascal (Pa)

Kelvin (K)

Atmosphere (atm)

Lower Explosive Limit (LEL)

Time-weighted Average (TWA)

Replace Table 2.2-7 with the following:

Table 2.2-7R Aircraft Operations - Significance Factors
(Sheet 1 of 2)

Airport	Projected Number of Operations per Year [a]	Distance from Plant, D (statue mile (mi))	Significance Factor [b]
Oak Ridge Airport (Under development)	15,860 [k]	2.25	NA
Big T (80TN)	<30 [d]	5 [c]	12,500
Wolf Creek (2TN7)	<50 [e]	5 [c]	12,500
Cox Farm (TN71)	<10 [f]	5 [c]	12,500
Will A Hildreth Farm (TN74)	<10 [g]	5 [c]	12,500
Meadowlake Airport (12TN)	11,000 [h]	9.76 [c]	47,629
Oliver Springs (TN08)	9,125 [i]	10 [l]	100,000
Fergusons Flying Circus (TN09)	<10 [j]	10 [l]	100,000
Knoxville McGhee Tyson Airport (TYS)	182,118	22.3	497,290
Knoxville Downtown Island Airport (DKX)	94,499	28.8	829,440
Chattanooga Lovell Field (CHA)	130,439	74.9	5,610,010
Lexington Blue Grass Airport (LEX)	181,363	136.0	18,496,000
Louisville International Standiford Field Airport (SDF)	343,793	178.0	31,684,000
Charlotte Douglas International Airport (CLT)	979,285	199.1	39,649,969
Atlanta International Airport (ATL)	1,437,036	212.0	44,944,000
James M. Cox Dayton International Airport (DAY)	41,927	277.6	77,061,760

Table 2.2-7R Aircraft Operations - Significance Factors
(Sheet 2 of 2)

Airport	Projected Number of Operations per Year [a]	Distance from Plant, D (statue mile (mi))	Significance Factor [b]

a) The values for airports with available FAA Terminal Area Forecasts (TAF) data (Reference 2.2-35) operations are projected 2050 APO values (where the projected number of operations is not derived from FAA TAF data, the data is obtained from Reference 2.2-7).

b) $500D^2$ movements per year for sites within 5 to 10 miles and $1000D^2$ movements per year for sites outside 10 miles where D is the distance from the CRN Site to the airport.

c) Conservatively, except for Meadowlake, for airports located between 5 and 10 miles of the CRN Site the significance factor was based on the distance of 5 miles.

d) Private airport with four single engine plane based. Permission is required to land.

e) Private airport with two single engine planes based.

f) Private airport with one single engine planes based. Permission is required to land.

g) Private airport with four single engine planes based. Permission is required to land.

h) Private airport with thirty four single engine planes and 3 multi-engine planes based. 11,000 operations reported for 12 months ending 12/31/2022.

i) Private airport with twenty two single engine planes based. 7,118 general aviation and 2,007 general aviation itinerant operations reported.

j) Private airport with nine single engine planes based. No operations reported.

k) Oak Ridge Airport operation values are from Reference 2.2-50, the projected number of flights is for the year 2040.

l) Conservatively, for airports located between 10 and 15 miles of the CRN Site the significance factor was based on the distance of 10 miles.

Replace Table 2.2-9 with the following:

Table 2.2-9R Design Basis Events - Explosions
(Sheet 1 of 3)

Source	Chemical Evaluated	Quantity Analyzed	Heat of Combustion (Btu/lb)	Distance to CRN Power Block (ft)	Safe Distance to Explosion to have less than 1 psi of Peak Incident Pressure (ft)
Nearby Offsite Facilities					
Custom Foods Of America	Anhydrous Ammonia	28,600 lb	7,992	124,872	(c)
East Tennessee Technology Park	Gasoline	24,999 lb	18,720	15,998	(e)
Energy Solution Services, Bear Creek Facility	Propane	20,000 lb	19,900	8,290	2,673
Food Lion - Delhaize DC #6	Anhydrous Ammonia	16,000 lb	7,992	108,240	(c)
Linde Inc.	Anhydrous Ammonia	9,640 lb	7,992	57,605	(c)
Oak Ridge National Laboratory	Acrylonitrile	12,039 lb	14,265	22,915	1,031
	Anhydrous Ammonia	1,655 lb	7,992	22,915	(c)
	Carbon Disulfide, ACS Reagent, >=99.9%	41 lb	6,280	22,915	155
	Gasoline	44,948 lb	18,720	22,915	(e)
	Methyltrichlorosilane	140 lb	3,000	22,915	272
Primary Products Ingredients Americas, LLC	Anhydrous Ammonia	190,539 lb	7,992	57,922	(c)(d)
Processed Food Corporation	Anhydrous Ammonia	19,000 lb	7,992	124,872	(c)
TVA-Kingston Fossil Plant	Anhydrous Ammonia	160,000 lb	7,992	40,867	(c)
U.S. Foodservice – Knoxville	Anhydrous Ammonia	17,000 lb	7,992	123,130	(c)
Weigel's #58	Gasoline	160,225 lb	18,720	19,378	2,444
	Propane	372 lb	19,900	19,378	(f)

Table 2.2-9R Design Basis Events - Explosions
(Sheet 2 of 3)

Source	Chemical Evaluated	Quantity Analyzed	Heat of Combustion (Btu/lb)	Distance to CRN Power Block (ft)	Safe Distance to Explosion to have less than 1 psi of Peak Incident Pressure (ft)
Nearby Transportation Routes/Roadways					
I-40	Acrylonitrile	12,039 lb	14,265	5800	1,031
	Butane	11,500 gal	19,512		3,864
	Gasoline	8,500 gal	18,720		273
	Hydrogen	15,032 ft^3/tube (a)	50,080		520
	Methanol	50,000 lb	8,600		2,742
	Methyl Chloroform	50,000 lb	4,700		2,241
	Methyltrichlorosilane	140 lb	3,000		272
	Propane	50,000 lb	19,900		3,627
Nearby Transport Routes/Pipelines					
East Tennessee Natural Pipeline 1 (6-in)	Natural gas (as methane)	1,960 lb(b)	21,517	5,800	1,250
East Tennessee Natural Pipeline 2 (22-in)	Natural gas (as methane)	26,400 lb(b)	21,517	15,800	2,970

Table 2.2-9R Design Basis Events - Explosions
(Sheet 3 of 3)

Source	Chemical Evaluated	Quantity Analyzed	Heat of Combustion (Btu/lb)	Distance to CRN Power Block (ft)	Safe Distance to Explosion to have less than 1 psi of Peak Incident Pressure (ft)

a) Transport quantity for a super jumbo tube (Reference 2.2-29).

b) Quantity of natural gas released over 5 seconds after a postulated pipe rupture.

c) Enveloped by explosion analyses of 170,000 lbs of Anhydrous Ammonia; which determined a distance of 270.9 feet to a pressure of 1 psi. Analysis of 9,999 lbs of Anhydrous Ammonia determined a distance of 105.3 feet to a pressure of 1 psi.

d) The quantity of Anhydrous Ammonia stored at Primary Products Ingredients is approximately 10% greater than that analyzed(c). The distance from Primary Products Ingredients to the CRN Site (57,922 feet) is much farther than the distance to a pressure of 1 psi (270.9 feet).

e) Enveloped by gasoline explosion at Weigel's #58.

f) Enveloped by propane explosion at Energy Solution Services, Bear Creek Facility.

Notes:

British Thermal Unit (Btu)

Tennessee Valley Authority (TVA)

U.S. Department of Energy Oak Ridge National Laboratory-Battelle (ORNL-Battelle)

Replace Table 2.2-10 with the following:

Table 2.2-10R Design Basis Events - Flammable Vapor Clouds (Delayed Ignition) Deflagration and Detonation and Jet Fire (Sheet 1 of 3)

Source	Chemical Evaluated	Quantity Analyzed[(h)]	Distance to CRN Power Block (ft)	Distance to LFL (ft)	Safe Distance for Vapor Cloud Explosion (VCE) to have less than 1 psi of Peak Incident Pressure (ft)	Jet Fire – Distance to 5 kW/m^2 (ft)
Nearby Offsite Facilities						
Custom Foods Of America	Anhydrous Ammonia	28,600 lb	124,872	(a)	(a)	
East Tennessee Technology Park	Gasoline	24,999 lb	15,946	(j)	(j)	
Energy Solution Services - Bear Creek Facility	Propane	20,000 lb	8,290	(i)	(i)	
Food Lion - Delhaize DC #6	Anhydrous Ammonia	16,000 lb	108,240	(a)	(a)	
Linde Inc.	Anhydrous Ammonia	9,640 lb	57,605	(a)	(a)	
Oak Ridge National Laboratory	Acrylonitrile	12,039 lb	22,915	(i)	(i)	
	Anhydrous Ammonia	1,655 lb	22,915	(a)	(a)	
	Gasoline	44,948 lb	22,915	(j)	(j)	
Primary Products Ingredients Americas, LLC	Anhydrous Ammonia	190,539 lb	57,922	(a)(c)	(a)(c)	
Processed Food Corporation	Anhydrous Ammonia	19,000 lb	124,872	(a)	(a)	
TVA-Kingston Fossil Plant	Anhydrous Ammonia	160,000 lb	40,867	(a)	(a)	
U.S. Foodservice - Knoxville	Anhydrous Ammonia	17,000 lb	123,130	(a)	(a)	
Weigel's #58	Gasoline	160,225	19,378	(k)	(k)	
	Propane	372 lb	19,378	(i)	(i)	

Table 2.2-10R Design Basis Events - Flammable Vapor Clouds (Delayed Ignition) Deflagration and Detonation and Jet Fire (Sheet 2 of 3)

Source	Chemical Evaluated	Quantity Analyzed[h]	Distance to CRN Power Block (ft)	Distance to LFL (ft)	Safe Distance for Vapor Cloud Explosion (VCE) to have less than 1 psi of Peak Incident Pressure (ft)	Jet Fire – Distance to 5 kW/m^2 (ft)
Nearby Transportation Routes/Roadways						
I-40	Acrylonitrile	12,039 lb	5800	(i)	(i)	
	Butane	11,500 gal		1,827[b]	3,864[b]	
	Gasoline	8,500 gal		132[b]	618[d]	
	Methanol	50,000 lb		(i)	(i)	
	Propane	50,000 lb		(i)	(i)	
Nearby Transport Routes/Pipelines						
East Tennessee Natural Pipeline 1 (6-in)	Natural gas (as methane)	666,312 lb[f] 683,023 lb[g]	5,800	477[e]	1,575[b]	312[b]
East Tennessee Natural Pipeline 2 (22-in)		9,624,751 lb[f] 9,866,045 lb[g]	15,800	1,401[e]	4,572[b]	1,203[b]

a) Enveloped by other analysis of Anhydrous Ammonia release scenarios. Analyses of 170,000 lbs of Anhydrous Ammonia determined a distance of 924 feet to LFL (with F stability class and two m/s) and 2601 feet to a pressure of 1 psi from a VCE (with F stability class and one m/s). Analysis of 9,999 lbs of Anhydrous Ammonia determined a distance of 345 feet to LFL and 918 feet to a pressure of 1 psi from a VCE (with F stability class and one m/s).

b) Worst-case scenario meteorological condition is F stability class at one m/s.

c) The quantity of Anhydrous Ammonia stored at Primary Products Ingredients is approximately 10% greater than that analyzed(a). The distance from Primary Products Ingredients (57,922 feet) is much farther than the distance to LFL (924 feet) and the distance to a pressure of 1 psi from a VCE (2601 feet).

d) Worst-case scenario meteorological condition is B stability class at 1.5 m/s.

e) Worst-case scenario meteorological condition is D stability class at 5.5 m/s.

Table 2.2-10R Design Basis Events - Flammable Vapor Clouds (Delayed Ignition) Deflagration and Detonation and Jet Fire (Sheet 3 of 3)

Source	Chemical Evaluated	Quantity Analyzed[(h)]	Distance to CRN Power Block (ft)	Distance to LFL (ft)	Safe Distance for Vapor Cloud Explosion (VCE) to have less than 1 psi of Peak Incident Pressure (ft)	Jet Fire – Distance to 5 kW/m^2 (ft)

f) Quantity of natural gas (as methane) released from a break in the natural gas pipeline under worst case meteorological condition for deflagration scenario; assumed pipeline length of 200 times the diameter; pressure 734.7 pounds per square inch atmosphere (psia); complete break; and connected to infinite source (Subsection 2.2.3.1.2.3).

g) Quantity of natural gas (as methane) released from a break in the natural gas pipeline under worst case meteorological condition for detonation scenario; assumed pipeline length of 200 times the diameter; pressure 734.7 psia; complete break; and connected to infinite source. (Subsection 2.2.3.1.2.3)

h) Where a capacity number was obtained from the Superfund Amendments and Reauthorization Act (SARA) Title III, Tier II report, the upper range number is shown and was used in the analysis.

i) Enveloped by analysis of transportation release of 11,500 gal (57,367 lb) of Butane based on less mass and similar (or smaller) heat of combustion. Thus, the distance to LFL and distance to pressure of 1 psi from a VCE will be less than the distance to the CRN Power Block.

j) Enveloped by analysis of transportation gasoline accident.

k) The gasoline is stored underground in three tanks at Weigel's #58 convenience store (Plus 89 Ethanol, Premium 93 Ethanol, and Regular 87 Ethanol). The quantity considered is the total stored quantity and not the quantity in each tank. Weigels uses underground gasoline tanks for customer delivery at the convenience store (i.e., a gas station). The underground tanks are not pressurized and do not present a vapor cloud explosion concern.

Notes:

Lower Flammability Limit (LFL)

Tennessee Valley Authority (TVA)

U.S. Department of Energy Oak Ridge National Laboratory-Battelle (ORNL-Battelle)

Vapor Cloud Explosion (VCE)

Replace Table 2.2-11 with the following:

Table 2.2-11R Design-Basis Events - Toxic Vapor Clouds
(Sheet 1 of 6)

Source	Chemical	Quantity Analyzed[(n)]	IDHL Limit	Distance to CRN Power Block (ft)	Distance to IDLH (ft)
Nearby Offsite Facilities					
A. E. Staley Manufacturing Company / Tate & Lyle - Loudon	Sulphur Dioxide (Anhydrous)	144,000 lb	100 ppm	58,344	(d)
City Of Maryville Wastewater Treatment Plant	Chlorine	12,000 lb	10 ppm	127,670	(f)
City Of Oak Ridge Municipal Pool	Chlorine	4,000 lb	10 ppm	57,024	(f)
Custom Foods Of America	Anhydrous Ammonia	28,600 lb	300 ppm	124,872	(g)
East Tennessee Mechanical Contractors (ETMC) WTP	Chlorine	12,000 lb	10 ppm	55,176	(f)
East Tennessee Technology Park	Gasoline	24,999 lb	750 ppm[(e)]	15,998	(d)
Energy Solution Services, Bear Creek Facility	Nitric Acid	645 lb	25 ppm	8,290	(o)
	Propane	20,000 lb	2100 ppm	8,290	(d)
First Utility District Of Knox County WWTP	Chlorine	22,000 lb	10 ppm	75,134	(f)
First Utility District Of Knox County WTP	Chlorine	20,000 lb	10 ppm	70,752	(f)
Food Lion - Delhaize DC #6	Anhydrous Ammonia	16,000 lb	300 ppm	108,240	(g)
Fourth Creek Wastewater Treatment Plant	Chlorine	16,000 lb	10 ppm	116,741	(f)
Hallsdale Powell Utility District - Melton Hill WTP	Chlorine	22,000	10 ppm	96,096	(f)
Linde Inc.	Anhydrous Ammonia	9,640 lb	300 ppm	57,605	(g)

Table 2.2-11R Design-Basis Events - Toxic Vapor Clouds
(Sheet 2 of 6)

Source	Chemical	Quantity Analyzed[n]	IDHL Limit	Distance to CRN Power Block (ft)	Distance to IDLH (ft)
Oak Ridge National Laboratory	Acrylonitrile	12,039 lb	85 ppm	22,915	(d)
	Ammonia	1,655 lb	300 ppm	22,915	(g)
	Carbon Dioxide	303,887 lb	40,000 ppm	22,915	(d)
	Carbon Disulfide, ACS Reagent	41 lb	500 ppm	22,915	(d)
	Chlorine	52 lb	10 ppm	22,915	(f)
	Chloroform	906 lb	500 ppm	22,915	(d)
	Gasoline	44,948 lb	750 ppm[e]	22,915	(d)
	Helium	32,223 lb	71,400[b]	22,915	(d)
	Hydrofluoric Acid (Anhydrous Gas)	348 lb	30 ppm	22,915	(d)
	Hydrogen Chloride, Anhydrous	66 lb	50 ppm	22,915	(d)
	Methyltrichlorosilane	140 lb	0.5 ppm	22,915	(d)
	Nitric Acid	28,826 lb	25 ppm	22,915	(o)
	Nitric Oxide	60 lb	100 ppm	22,915	(d)
	Nitrogen Dioxide	262 lb	13 ppm	22,915	(d)
Oak Ridge National Laboratory - UCOR LLC - UCOR Rollup	Nitric Acid	24,999 lb	25 ppm	25,714	(o)
Oak Ridge Water Treatment Plant	Chlorine	16,000 lb	10 ppm	55,282	(f)
Primary Products Ingredients Americas, LLC	Anhydrous Ammonia	190,539 lb	300 ppm	57,922	(i)
Processed Food Corporation (Custom Foods)	Anhydrous Ammonia	19,000 lb	300 ppm	124,872	(g)
Tellico Area Service System WTP	Chlorine	4,000 lb	10 ppm	115,104	(f)

Table 2.2-11R Design-Basis Events - Toxic Vapor Clouds
(Sheet 3 of 6)

Source	Chemical	Quantity Analyzed[n]	IDHL Limit	Distance to CRN Power Block (ft)	Distance to IDLH (ft)
TVA – Kingston Fossil Plant	Anhydrous Ammonia	160,000 lb	300 ppm	40,867	(g)
U.S. Foodservice – Knoxville	Anhydrous Ammonia	17,000 lb	300 ppm	123,130	(g)
Weigel's #58	Gasoline	160,225 lb	750 ppm[e]	19,378	(d)
	Propane	372 lb	2100 ppm	19,378	(d)
West Knox Utility District WTP	Chlorine	12,000 lb	10 ppm	48,048	(f)
West Knox Utility District WWTP	Chlorine	6,000 lb	10 ppm	75,874	(f)
Yale Security Inc.	Chlorine	8,000 lb	10 ppm	50,160	(f)
Nearby Transportation Routes/Roadways					
I-40	Acrylonitrile	12,039 lb	85 ppm	5800	(d)
	Anhydrous Ammonia	11,500 gal	300 ppm		13,728[a][j]
	Argon	50,000 lb	71,400[b]		(d)
	Butane	11,500 gal	71,400[b]		849[a]
	Carbon Dioxide	50,000 lb	40,000 ppm		(d)
	Chlorine Dioxide	24,131 lb	10 ppm		(d)
	Chlorine	44,000 lb	5 ppm		23,760[c][k]
	Chloroform	906 lb	500 ppm		(d)
	Gasoline	8,500 gal	750 ppm[e]		687[a][l]
	Helium	50,000 lb	71,400[b]		(d)
	HFC-134A	50,000 lb	71,400[b]		(d)
	Hydrochloric Acid	6,000 gal	50 ppm		(p)
	Hydrofluoric Acid	499 lb	30 ppm		(d)
	Hydrogen Chloride - Anhydrous	66 lb	50 ppm		(d)

Table 2.2-11R Design-Basis Events - Toxic Vapor Clouds
(Sheet 4 of 6)

Source	Chemical	Quantity Analyzed[n]	IDHL Limit	Distance to CRN Power Block (ft)	Distance to IDLH (ft)
I-40 (continued)	Liquid Nitrogen	8,910 lb	71,400[b]		(r)
	Methanol	50,000 lb	6,000 ppm		(d)
	Methyl Chloroform	50,000 lb	700 ppm		(d)
	Methyltrichlorosilane	140 lb	0.5 ppm		(d)
	Nitric Acid	6,000 gal	25 ppm		(q)
	Nitric Oxide	60 lb	100 ppm		(d)
	Nitrogen Dioxide	262 lb	13 ppm		(d)
	Nitrogen	50,000 lb	71,400[b]		(d)
	Propane	50,000 lb	2100 ppm		(d)
	Sulfur Dioxide (Anhydrous)	50,000 lb	100 ppm		(d)
Nearby Transport Routes/Pipelines					
East Tennessee Pipeline 1 (6-in)	Natural Gas (as methane)	666,312 lb[m]	71,400 ppm[b]	5,800	282[h]
East Tennessee Pipeline 2 (22-in)		9,624,751 lb[m]		15,800	846[h]

a) Worst-case scenario meteorological condition is F stability class at one meter per second (m/s).

b) Asphyxiation Limit.

c) Worst-case scenario meteorological condition is F stability class at three m/s.

d) Chemical screened in accordance with RG 1.78, Appendix A, and determined that the quantity stored/transported is less than that which would result in exceeding the IDLH at the CRN Power Block.

e) Gasoline does not have an identified IDLH. The Threshold Limit Value-Short Term Exposure Limit (TLV-STEL) is 500 ppm; the Threshold Limit Value-Time-weighted Average (TLV–TWA) is 300 ppm; and the Protective Action Criteria (PAC) PAC-2 guideline is 1000 ppm for gasoline. For the analyses, n-Heptane is used as a surrogate and has an IDLH of 750 ppm. This selection is conservative given the PAC-2 guideline most closely correlates with the definition of IDLH.

Table 2.2-11R Design-Basis Events - Toxic Vapor Clouds
(Sheet 5 of 6)

Source	Chemical	Quantity Analyzed[(n)]	IDHL Limit	Distance to CRN Power Block (ft)	Distance to IDLH (ft)

f) Enveloped by toxic hazard analyses of Chlorine release. Analysis of 24,999 lbs of Chlorine release determined a distance of 20,064 feet to the IDLH. Analysis of 10,000 lbs of Chlorine release determined a distance of 15,312 feet to the IDLH.

g) Enveloped by toxic hazard analyses of Anhydrous Ammonia release. Analysis of 170,000 lbs of Anhydrous Ammonia release determined a distance of 21,648 feet to the IDLH. Analysis of 9,999 lbs of Anhydrous Ammonia release determined a distance of 6,336 feet to the IDLH. Worst-case scenario meteorological condition is F stability class at one meter per second (m/s).

h) Worst-case scenario meteorological condition is D stability class at 5.5 m/s.

i) The quantity of Anhydrous Ammonia stored at Primary Products Ingredients is approximately 10% greater than that analyzed. The distance from Primary Products Ingredients (57,922 feet) is much farther than the distance to the IDLH (21,648 feet).

j) The toxicity limit at the CRN Site power block area is exceeded.

k) The toxicity limit at the CRN Site power block area is exceeded.

l) The distance to the TLV-STEL is 843 feet and the distance to the TLV-TWA is 1089 feet.

m) Quantity of natural gas (as methane) released from a break in the natural gas pipeline under worst case meteorological condition for the toxic vapor cloud scenario; assumed pipeline length of 200 times the diameter; pressure 734.7 pounds per square inch atmosphere (psia); complete break; and connected to infinite source. (Subsection 2.2.3.1.2.3)

n) Where a capacity number was obtained from the Superfund Amendments and Reauthorization Act (SARA) Title III, Tier II report, the upper range number is shown and was used in the analysis.

o) Enveloped by toxic hazard analyses of Nitric Acid release. Analysis of 499,999 lbs of Nitric Acid release determined a distance of 15,312 feet to the IDLH with worst-case scenario meteorological condition of D stability class at 5.5 m/s. Analysis of 74,826 lbs of Nitric Acid release determined a distance of 5,280 feet to the IDLH with worst-case scenario meteorological condition of F stability class at three m/s.

p) The toxicity limit at the CRN Site power block area is exceeded.

Table 2.2-11R Design-Basis Events - Toxic Vapor Clouds
(Sheet 6 of 6)

Source	Chemical	Quantity Analyzed[(n)]	IDHL Limit	Distance to CRN Power Block (ft)	Distance to IDLH (ft)

q) The toxicity limit at the CRN Site power block area is exceeded.

r) Enveloped by hazard analysis of release of 9,999 lb of liquid nitrogen, which showed a limiting distance to the asphyxiation level at 489 feet.

Notes:

Immediately Dangerous to Life and Health (IDLH)

Parts per million (ppm)

Ton (T)

Tennessee Valley Authority (TVA)

Water Treatment Plant (WTP)

Wastewater Treatment Plant (WWTP)

U.S. Department of Energy Oak Ridge National Laboratory-Battelle (ORNL-Battelle)

U.S. Department of Energy Oak Ridge National Laboratory-URS (ORNL-URS)

Replace Table 2.2-12 with the following:

Table 2.2-12R On-Site Chemicals
(Sheet 1 of 2)

Chemical/Material (Formula/Trade/State)	Location	Quantity
Nitrogen	Cryogenic Storage Tank – south of Power Block outside of Protected Area	43.1 m^3 liquid nitrogen
Hydrogen	South of Power Block outside of Protected Area	Gaseous Cylinder Bank; 356.1 scm (12,584.2 scf) per tube.
Standby Diesel Generator (SDG) Fuel Oil (FO)	Above ground storage tank located east of Protected Area Day tanks are located in the associated SDG Room.	30,000 gallon Storage Tank. Day Tanks (one per EDG) (volume not specified at this time)
Diesel Driven Fire Pump Fuel Oil (FO)	Day tanks are located in the Fire Pump Enclosure east of the Protected Area	Day Tank (volume not specified at this time)
Turbine Lubricating Oil (LO)	Above ground storage tank located east of Protected Area	Storage Tank (volume not specified at this time).
Sodium Hypochlorite (10 to 15% Solution)	Water Treatment Area (near Cooling Tower east of the Protected Area)	9,000 gallon tank
Sodium Bisulfite for CWS Discharge (40% Solution)	Water Treatment Area (near Cooling Tower east of the Protected Area)	1,000 gallon tank
Sulfuric Acid for CWS	Adjacent to Cooling Tower, east of the Protected Area	7,000 gallon tank
Sodium Bisulfite for Demineralized Water Production (30% Solution)	Adjacent to Demineralized Water Treatment Area, east of the Protected Area	25 gallon tank
ChemTreat CL1355 for CWS	Water Treatment Area (near Cooling Tower east of the Protected Area)	2,000 gallon tank
ChemTreat CL1355 for Demineralized Water Production	Adjacent to Demineralized Water Treatment Area, east of the Protected Area	15 gallon tank

Table 2.2-12R On-Site Chemicals
(Sheet 2 of 2)

Chemical/Material (Formula/Trade/State)	Location	Quantity
ChemTreat CL5633 for Demineralized Water System Cleaning	Adjacent to Demineralized Water Treatment Area, east of the Protected Area	20 gallon tank/tote
Sodium Hydroxide (25% solution) for Demineralized Water System Cleaning	Adjacent to Demineralized Water Treatment Area, east of the Protected Area	10 gallon tank/tote
Sodium Hypochlorite for Demineralized Water System Cleaning	Adjacent to Demineralized Water Treatment Area, east of the Protected Area	10 gallon tank/tote
Gasoline	Vehicle Maintenance Garage	50 gallon containers / vehicle gasoline tank
Propylene Glycol	P25, Chilled Water System, throughout the power block	~37,000 liters (~10,000 gallons)
R-134a (Tetrafluoroethane)	Radwaste Building Roof	Four Chillers with 250 kg (551 lbs) per Chiller
$Na_2Pt(OH)^6$ (Noble Metal Solution)	Reactor Building	Approximately 38 liters (10 gallons) of 1% noble metal solution is utilized over a 2-week time frame per year.
Depleted Zinc Oxide (DZO)	Turbine Building	Volume – Not Specified
R410a (used in mini-split AC units)	Turbine Building Control Building	Volume – Not Specified

Replace Table 2.2-13 with the following:

Table 2.2-13R On-Site Chemical Evaluation
(Sheet 1 of 2)

Chemical/Material (Formula/Trade/State)	Toxicity Limit (IDLH)	Flammable/ Explosive?	Disposition
Cryogenic Nitrogen	Asphyxiant	No / No	Toxic analysis (asphyxiation) performed and found to not impact Main Control Room (MCR) habitability.
Hydrogen	Asphyxiant	Yes / Yes	Toxic evaluation (asphyxiation) performed and found to not impact MCR habitability. Safe separation distance due to an explosion is provided in Subsection 2.2.3.1.1
Standby Diesel Generator (SDG) Fuel Oil (FO)	None established	Yes (Varies) / No	No further analysis required (1)
Diesel Driven Fire Pump Fuel Oil (FO)	None established	Yes (Varies) / No	No further analysis required (1)
Turbine Lubricating Oil (LO)	None established	Yes (Varies) / No	No further analysis required (1)
Sodium Hypochlorite (10 to 15% Solution)	10 ppm (as Chlorine)	No / No	Toxic evaluation performed and found to not impact MCR habitability.
Sodium Bisulfite for CWS Discharge (40% Solution)	None established	No / No	No further analysis required.
Sulfuric Acid for CWS	15 mg/m^3	No / No	Toxic evaluation performed and found to have no impact to MCR habitability.
Sodium Bisulfite for Demineralized Water Production (30% Solution)	None established	No / No	No further analysis required.
ChemTreat CL1355 for CWS	None established	No / No	No further analysis required.
ChemTreat CL1355 for Demineralized Water Production	None established	No / No	No further analysis required.
ChemTreat CL5633 for Demineralized Water System Cleaning	None established	No / No	No further analysis required.

Table 2.2-13R On-Site Chemical Evaluation
(Sheet 2 of 2)

Chemical/Material (Formula/Trade/State)	Toxicity Limit (IDLH)	Flammable/ Explosive?	Disposition
Sodium Hydroxide (25% solution) for Demineralized Water System Cleaning	10 mg/m^3	No / No	Toxic evaluation performed and found to not impact MCR habitability.
Sodium Hypochlorite for Demineralized Water System Cleaning	10 ppm (as Chlorine)	No / No	Toxic evaluation performed and found to not impact MCR habitability.
Gasoline	750 ppm	Yes / Yes	On-site gasoline storage tank location has not been determined. Final location will provide adequate separation distance per RG 1.91.
Propylene Glycol	None established	No / No	No further analysis required.
R-134a (Tetrafluoroethane)	Asphyxiant	No / No	Toxic evaluation (asphyxiation) performed and found to not impact MCR habitability.
Na_2Pt (OH)6(Noble Metal Solution)	None established	No / No	No further analysis required.
Depleted Zinc Oxide (DZO)	None established	No / No	No further analysis required.
R410a (used in mini-split AC units)	None established; asphyxiant	No / No	Toxic evaluation (asphyxiation) performed and found to not impact MCR habitability.

1) A fluid with an extremely low vapor pressure will not explode per NFPA 422 (Reference 2.2-81) which states that the vapor space in tanks storing low vapor pressure liquids is normally too lean to burn. The vapor pressure of diesel fuel is low enough such that the vapor concentration above the liquid (0.36%) is significantly lower than the LFL (1.3%). As a result the air-gas mixture is expected to be too lean to ignite and/or explode. Similarly, kerosene grade fuel ordinarily has a low tendency to vaporize, and in a closed tank, the fuel vapor and air mixture can be too lean to burn.

Replace Figure 2.2-1 with the following:

Figure 2.2-1R Identified Industrial Facilities Nearby the Clinch River Nuclear Site (5 mile radius)

Replace Figure 2.2-3 with the following:

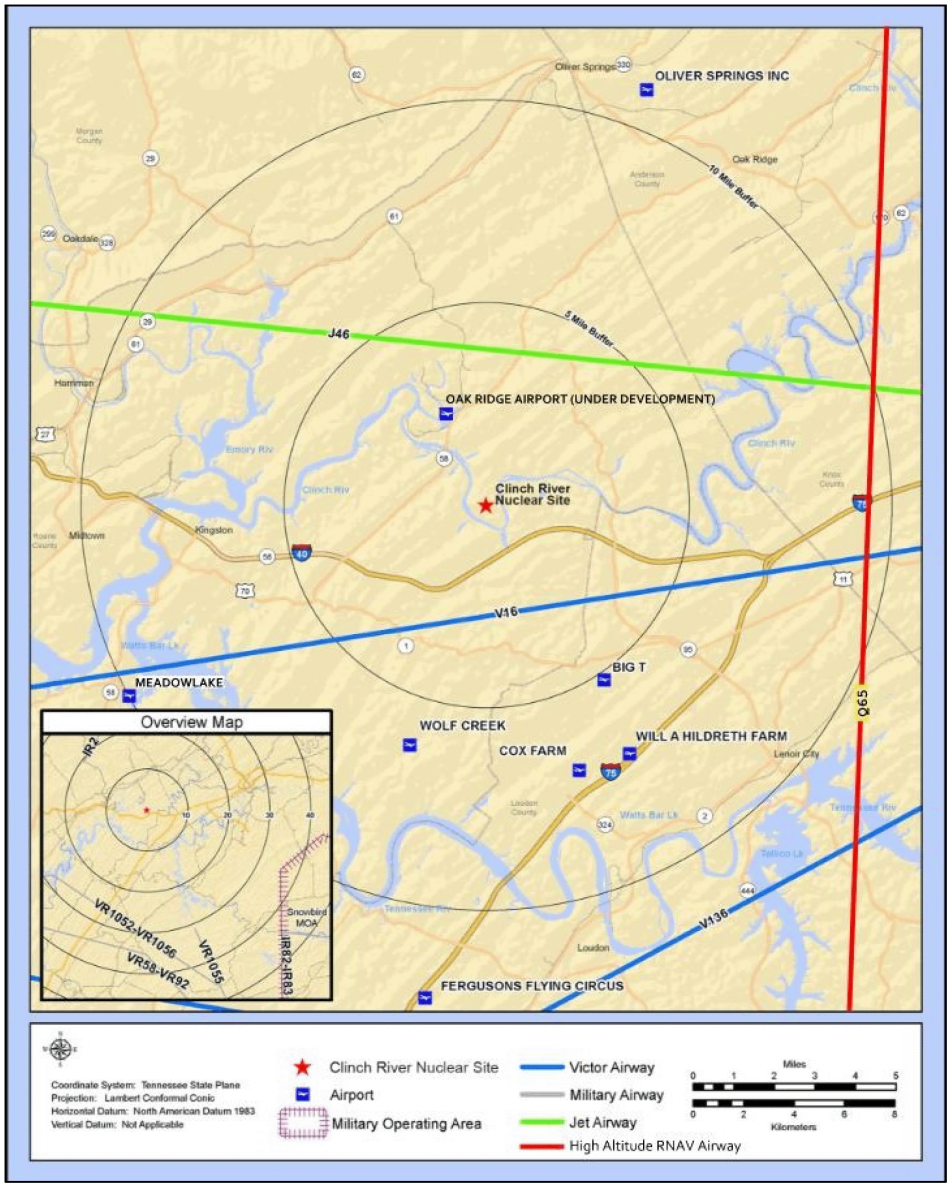

Figure 2.2-3R Location of Airports and Airways

2.3 Meteorology

Section 2.3 of the referenced ESPA SSAR is incorporated by reference with the following supplements.

2.3.1 Regional Climatology

2.3.1.1 General Climate

Replace from the fourth sentence of the second paragraph to the end of the fourth paragraph with the following:

Wind speeds are low, with a mean annual wind speed of 2.7 miles per hour (mph) at Oak Ridge (Reference 2.3.1-1). During winter when the jet stream moves southward, the Cumberland Mountains also serve to retard or moderate cold outbreaks by blocking dense, cold polar continental air masses. In summer, the topography in the region may enhance thunderstorm activity due to orographic lift (References 2.3.1-1 and 2.3.1-2).

Area temperatures measured in Oak Ridge indicate warm summers and mild winters. In January, the normal daily maximum temperature is 48.1°F with a normal daily minimum temperature 30.0°F based on 30 years of data. In July, the normal daily maximum temperature is 89.1°F, while the normal daily minimum temperature is 69.0°F based on 30 years of data (Reference 2.3.1-1). Relative humidity in the region averaged 73 percent based on a 30-year period of record from the Knoxville Local Climatological Data (LCD) (Reference 2.3.1-2).

Precipitation averages about 56 inches (in.) annually (Reference 2.3.1-1). Late winter (January– March) is usually the wettest season, with more than 15 in., while late summer-early autumn (August–October) is the driest season, with less than 11 in. Droughts are uncommon in this region of the United States.

Replace Reference 2.3.1-27 with Reference 2.3.1-4 in the last paragraph in ESPA SSAR Section 2.3.1.1.

2.3.1.2 Regional Meteorological Conditions for Design and Operating Basis

Replace Table 2.0-1 with Table 2.0-1R in ESPA SSAR Section 2.3.1.2.

2.3.1.3 Severe Weather

Replace the third sentence of the first paragraph in ESPA SSAR Section 2.3.1.3 with the following:

Most data are taken from the National Climatological Data Center (NCDC) Storm Events Database (Reference 2.3.1-4) that covers the 74-year period of 1950–2023, but even longer data periods are used for some phenomena to try to capture the occurrence of rare events.

2.3.1.3.1 Thunderstorms, Hail and Lightning

Replace ESPA SSAR Section 2.3.1.3.1 with the following:

Thunderstorms

Thunderstorms are common in the region based on a 30-year period of record (1991–2020) range of 43 days with thunderstorms at Tri-Cities to 57 days with thunderstorms at Chattanooga (References 2.3.1-2, 2.3.1-5, 2.3.1-6, 2.3.1-7 and 2.3.1-39). The greatest frequency of thunderstorms is during the summer with a range of 31-36 days during May–August. This is characteristic of a diurnal afternoon thunderstorm pattern due to solar heating.

Hail

In Roane County, hail events were reported 36 days from 1950 through December 31, 2022, or approximately 0.50 days per year, (Reference 2.3.1-4). Hail was also reported in adjacent Loudon and Knox counties 50 and 94 times, or 0.69 and 1.29 days per year, respectively, during this time period (Reference 2.3.1-4).

Lightning

The CRN Site averages 13 cloud-to-ground lightning flashes per square mile (2.6 km^2) annually (Reference 2.3.1-9).

2.3.1.3.2 Extreme Winds

Replace the second paragraph in ESPA SSAR Section 2.3.1.3.2 with the following:

Estimated extreme winds are based on climatological data from Oak Ridge and Knoxville, Tennessee (References 2.3.1-1,2.3.1-2, 2.3.1-8, 2.3.1-10, 2.3.1-11, 2.3.1-12, and 2.3.1-13) and hourly observations from TVA Sequoyah and Watts Bar Nuclear Plants (References 2.3.1-14 and 2.3.1-15).

Replace Table 2.3.1-1 with Table 2.3.1-1R in the third paragraph in ESPA SSAR Section 2.3.1.3.2.

Replace Reference 2.3.1-3 with Reference 2.3.1-10 in the fourth paragraph in ESPA SSAR Section 2.3.1.3.2.

Replace Reference 2.3.1-30 with Reference 2.3.1-16 in the fifth paragraph in ESPA SSAR Section 2.3.1.3.2.

Replace the sixth paragraph in ESPA SSAR Section 2.3.1.3.2 with the following:

For comparison, hourly average (scalar) wind speeds at the nominal 33-foot level are available from the CRN Site during 2011–2013, the Watts Bar Nuclear Plant (30 miles southwest from the CRN Site) during 1973–2018, and the Sequoyah Nuclear Plant during 1971–2018. The

maximum hourly average wind speed values of 15.1 mph for CRN Site, 30 mph for Watts Bar Nuclear Plant, and 29.8 mph for Sequoyah Nuclear Plant correspond to 3-second gust wind speeds of 23 mph, 45 mph, and 45 mph, respectively.

Replace Reference 2.3.1-15 with Reference 2.3.1-17 in the seventh paragraph in ESPA SSAR Section 2.3.1.3.2.

2.3.1.3.3 Precipitation Extremes

Replace the first sentence of the first paragraph in ESPA SSAR Section 2.3.1.3.3 with the following:

Historical precipitation for the CRN Site meteorological vicinity obtained from several surrounding National Weather Service (NWS) and TVA sites (References 2.3.1-1, 2.3.1-2, 2.3.1-5, 2.3.1-6, 2.3.1-7, 2.3.1-14 and 2.3.1-15) is summarized in Table 2.3.1-2R.

Replace Reference 2.3.1-34 with Reference 2.3.1-18 in last sentence of the first paragraph in ESPA SSAR Section 2.3.1.3.3.

Replace the first sentence of the second paragraph in ESPA SSAR Section 2.3.1.3.3 with the following:

The maximum estimated annual precipitation is in the range of 46–55 in.

Replace Table 2.3.1-2 with Table 2.3.1-2R in the second sentence of the second paragraph in ESPA SSAR Section 2.3.1.3.3.

Replace Reference 2.3.1-36 with Reference 2.3.1-19 in the third sentence of the second paragraph in ESPA SSAR Section 2.3.1.3.3.

Replace Reference 2.3.1-36 with Reference 2.3.1-19 in the fourth sentence of the second paragraph in ESPA SSAR Section 2.3.1.3.3.

2.3.1.3.4 Tornadoes

Replace the first paragraph in ESPA SSAR Section 2.3.1.3.4 with the following:

The probability of a tornado occurring at the CRN Site is low based on records from the NWS Morristown Tornado Database (Reference 2.3.1-21) and the NCDC Storm Events Database (Reference 2.3.1-4). During the 72-year period 1950-2022, six tornadoes were reported within 10 miles of the CRN Site (Table 2.3.1-3R). Only one of these had an intensity greater than F0/EF0.

Replace Reference 2.3.1-15 with Reference 2.3.1-17 in the third paragraph in ESPA SSAR Section 2.3.1.3.4.

Replace Reference 2.3.1-17 with Reference 2.3.1-22 in the parameter definition section of Equation 2.3.1-1 in ESPA SSAR Section 2.3.1.3.4.

Replace Table 2.0-1 with Table 2.0-1R in the last paragraph of ESPA SSAR Section 2.3.1.3.4.

2.3.1.3.5 Hurricanes

Replace the third paragraph in ESPA SSAR Section 2.3.1.3.5 with the following:

Review of the NCDC Storm Events Database for the period of January 1, 1950, through December 31, 2022, shows that there was only one tropical storm on September 16, 2004, near Roane County that caused minimal damage. This storm was associated with Hurricane Ivan.

2.3.1.3.6 Winter Storm Events

Replace Reference 2.3.1-5 with Reference 2.3.1-2 in the first sentence of the first paragraph in ESPA SSAR Section 2.3.1.3.6.

Replace Table 2.3.1-2 with Table 2.3.1-2R in the second sentence of the first paragraph in ESPA SSAR Section 2.3.1.3.6.

Replace Reference 2.3.1-2 with Reference 2.3.1-5 in the second sentence of the first paragraph in ESPA SSAR Section 2.3.1.3.6.

Replace Reference 2.3.1-18 with Reference 2.3.1-23 at the end of the second paragraph in ESPA SSAR Section 2.3.1.3.6.

2.3.1.3.6.1 Ice Storms

Replace Reference 2.3.1-19 with Reference 2.3.1-24 in first paragraph in ESPA SSAR Section 2.3.1.3.6.1.

Replace Reference 2.3.1-30 with Reference 2.3.1-16 in the third paragraph in ESPA SSAR Section 2.3.1.3.6.1.

2.3.1.3.6.2 Normal and Extreme Winter Precipitation Events

Replace the third paragraph in ESPA SSAR Section 2.3.1.3.6.2 with the following:

The maximum reported snow depth at Chattanooga, TN (Reference 2.3.1-5), the highest snow depth at a nearby NWS station, was used to estimate the weight of the maximum historic snowpack at the CRN Site. The greatest snow depth reported during the 84-year period of record (1938–2022) for Chattanooga, was 20 in. in March 1993. Interim Staff Guidance (ISG) on Assessment of Normal and Extreme Winter Precipitation Loads on the Roofs of Seismic Category I Structures (ISG-7) (Reference 2.3.1-25), provides an algorithm (below) for converting historical maximum snowpack depth to a ground snow load.

Replace the fourth paragraph in ESPA SSAR Section 2.3.1.3.6.2 with the following:

Using the 20-inch snow depth for Chattanooga gives a snow load of 16.4 psf for the maximum historical snowpack.

Replace Reference 2.3.1-39 with Reference 2.3.1-26 in the fifth paragraph in ESPA SSAR Section 2.3.1.3.6.2.

Replace Reference 2.3.1-32 with Reference 2.3.1-27 in the eighth paragraph in ESPA SSAR Section 2.3.1.3.6.2.

2.3.1.4 Design Basis Dry- and Wet-Bulb Temperatures

Replace Reference 2.3.1-33 with Reference 2.3.1-28 in the second paragraph in ESPA SSAR Section 2.3.1.4.

Replace Table 2.0-1 with Table 2.0-1R in the last paragraph of ESPA SSAR Section 2.3.1.4.

2.3.1.5 Meteorological Data for Evaluating Ultimate Heat Sink

Replace the first sentence of the first paragraph in ESPA SSAR Section 2.3.1.5 with the following:

The GVH BWRX-300 design being evaluated does not rely on an external water source as its ultimate heat sink.

2.3.1.6 Climate Changes

Replace the first paragraph in ESPA SSAR Section 2.3.1.6 with the following:

While climatic conditions change over time, such changes are cyclical in nature on various time and spatial scales. The timing, magnitude, relative contributions to, and implications of these changes are generally more speculative, even for specific areas or locations. Further, the most extreme projected changes are for time scales much longer than any facility will be operated at CRN. Projected changes are generally small compared to natural variation. General predictions of global or U.S. climatic changes expected during the period of reactor operation are uncertain and are only applicable on a macroclimatic scale. Since the maximum data span available was used in the severe weather analysis, accurate severe weather phenomena have been provided based on best-available historic data. Projections of future severe weather conditions at the CRN Site are highly uncertain at best, based on current understanding and modeling of global climate change. Predictions provided by the USGS (Reference 2.3.1-29) vary considerably, yet 100% of models agree on temperature increasing. For example, the Community Climate System Model, version 4 (CCSM4) model gives a summer maximum temperature increase from approximately 88ºF to 95ºF with a standard deviation of approximately 3ºF over the period of 2025 through 2075. The HadGEM2-ES365 model gives a summer maximum temperature

increase from approximately 88ºF to 101ºF with a standard deviation of approximately 4ºF over the period of 2025 through 2075. TVA's major planning processes, consistent with its Climate Change Adaptation Plan, identify opportunities as well as climate change risks with the potential to substantially impair, obstruct or prevent the success of Agency mission activities, both in the near term and particularly in the long term, using the best available science and information.

Replace Table 2.3.1-4 with Table 2.3.1-4R in the second paragraph in ESPA SSAR Section 2.3.1.6.

Replace the last sentence of the second paragraph in ESPA SSAR Section 2.3.1.6 with the following:

Based on the consistency of these records over the period of 1921 to 2020, significant climate changes are not expected during the licensing period.

2.3.1.7.1 Background Air Quality

Replace the last sentence of the second paragraph in ESPA SSAR Section 2.3.1.7.1 with the following:

Ambient air quality measurements in the vicinity of the CRN Site in 2013 and 2022 are shown in Table 2.3.1-5R (Reference 2.3.1-32). As of 2017, all neighboring counties have been redesignated to maintenance (Table 2.3.1-6).

Replace References 2.3.1-22 through 2.3.1-25 with References 2.3.1-32 through 2.3.1-35 in the first sentence of the second paragraph in ESPA SSAR Section 2.3.1.7.1.

Replace Reference 2.3.1-23 with Reference 2.3.1-33 in the second sentence of the second paragraph in ESPA SSAR Section 2.3.1.7.1.

Replace Reference 2.3.1-24 with Reference 2.3.1-34 in the third sentence of the second paragraph in ESPA SSAR Section 2.3.1.7.1.

Replace Reference 2.3.1-25 with Reference 2.3.1-35 in the first sentence of the fourth paragraph in ESPA SSAR Section 2.3.1.7.1.

Replace Reference 2.3.1-22 with Reference 2.3.1-32 in the second sentence of the fourth paragraph in ESPA SSAR Section 2.3.1.7.1.

Replace References 2.3.1-35 with Reference 2.3.1-36 in the second sentence of the fourth paragraph in ESPA SSAR Section 2.3.1.7.1.

Replace Reference 2.3.1-26 with Reference 2.3.1-37 in the last sentence of the last paragraph in ESPA SSAR Section 2.3.1.7.1.

2.3.1.7.2 Projected Air Quality

Insert the following after the last sentence in ESPA SSAR Section 2.3.1.7.2:

If there is construction of an air contaminant source on the CRN Site, a Non-Title V Construction Permit and then Non-Title V Operating Permit will be required from the Tennessee Division of Air Pollution Control.

2.3.1.8 References

Replace ESPA SSAR Section 2.3.1.8 with the following:

2.3.1-1 2022 Annual Local Climatological Data (LCD) for Oak Ridge, TN; National Climatic Data Center.

2.3.1-2 2022 Annual Local Climatological Data (LCD) for Knoxville, TN; National Climatic Data Center.

2.3.1-3 Tennessee Department of Environment and Conservation, Air Pollution Control Division, “Ambient Air Monitoring Plan 2012,” June 15, 2012.

2.3.1-4 NCDC Storm Events Database (http://www.ncdc.noaa.gov/stormevents), accessed. August 23, 2023.

2.3.1-5 Chattanooga, TN daily average precipitation data for years 1928 through 2022; National Climatic Data Center.

2.3.1-6 2022 Annual Local Climatological Data (LCD) for Nashville, TN; National Climatic Data Center.

2.3.1-7 2022 Annual Local Climatological Data (LCD) for Tri-Cities, TN; National Climatic Data Center.

2.3.1-8 1995 Annual Local Climatological Data (LCD) for Knoxville, TN; National Climatic Data Center.

2.3.1-9 S. A. Changnon, “Thunderstorms Across the Nation”, 2001, Figure 29.

2.3.1-10 1974 Annual Local Climatological Data (LCD) for Knoxville, TN; National Climatic Data Center.

2.3.1-11 1959 Annual Local Climatological Data (LCD) for Oak Ridge, TN; National Climatic Data Center.

2.3.1-12 1975 Annual Local Climatological Data (LCD) for Oak Ridge, TN; National Climatic Data Center.

2.3.1-13 1998 Annual Local Climatological Data (LCD) for Oak Ridge, TN; National Climatic Data Center.

2.3.1-14 Climatological Summary for Sequoyah Nuclear Plant (1971-2018), TVA.

2.3.1-15 Climatological Summary for Watts Bar Nuclear Plant (1973-2018), TVA.

2.3.1-16 American Society of Civil Engineers, ASCE/SEI 7-05, “Minimum Design Loads for Buildings and Other Structures.”

2.3.1-17 Thom, H. C. S., New Distributions of Extreme Winds in the United States, Journal of the Structural Division, Proceedings of the American Society of Civil Engineers, Vol 94, No. ST 7, pp 1787-1801, 1968; Figure 5.

2.3.1-18 The Tennessee Encyclopedia of History and Culture, Version 3.0, Tennessee Historical Society.

2.3.1-19 NOAA Hydrometeorological Report HMR-52, Application of Probable Maximum Precipitation Estimates – United States East of the 105th Meridian, August 1982.

2.3.1-20 Not Used.

2.3.1-21 NWS Morristown Tornado Database (http://innovation.srh.noaa.gov/tors/index.php?cw=mrx), accessed November 13, 2014.

2.3.1-22 NUREG/CR-4461, Revision 2, “Tornado Climatology of the Contiguous United States”.

2.3.1-23 NOAA Manual NOS NGS 1, “Geodetic Bench Marks”, NOAA, September 1978.

2.3.1-24 Paul Tattleman and Irving I. Gringorton,” Estimated Glaze Ice and Wind Loads at the Earth’s Surface for the Contiguous United States,” Air Force Cambridge Research Laboratories, October 1973.

2.3.1-25 Interim Staff Guidance (ISG) on Assessment of Normal and Extreme Winter Precipitation Loads on the Roofs of Seismic Category I Structures (DC/COL-ISG-07), USNRC, 2009.

2.3.1-26 Ludlum, David M., Weather Record Book, United States and Canada, Weatherwise, Inc., 1971, page 73.

2.3.1-27 Hydrometeorological Report HMR-53, NUREG/CR-1486, Seasonal Variation of 10-Square-Mile Probable Maximum Precipitation Estimates, United States East of the 105th Meridian.

2.3.1-28 EPRI Advanced Light Water Reactor (ALWR) Utility Requirements Document, EPRI TR-016780 – Utility Requirements Document, Volume II, Rev 8.

2.3.1-29 U.S. Geological Survey, National Climate Change Viewer, Summary for Roane County, Tennessee, September 7, 2023.

2.3.1-30 Annual Local Climatological Data (LCD) for Knoxville, TN; for year's 1953, 1963, 1973, 1983, 1993, 2003, and 2013, National Climatic Data Center.

2.3.1-31 Annual Local Climatological Data (LCD) for Oak Ridge, TN; for year's 1953, 1963, 1973, 1983, 1993, 2003, and 2013, National Climatic Data Center.

2.3.1-32 Environmental Protection Agency, Air Quality System (AQS) Database. http://www.epa.gov/airquality/airdata.

Link to AQ reports

Air Quality Statistics Report | US EPA (https://www.epa.gov/outdoor-air-quality-data/air-quality-statistics-report) Accessed 9/7/2023

2.3.1-33 Environmental Protection Agency News Release dated 07/13/2015, "EPA Approves Redesignation of Knoxville Area to Attainment for the 2008 8-Hour Ozone Standard" https://www.epa.gov/archive/epa/newsreleases/epa-approves-redesignation-knoxville-area-attainment-2008-8-hour-ozone-standard.html. Accessed 9/15/2023.

2.3.1-34 Federal Register Vol. 79, No. 105, June 2, 2014, page 31566.

2.3.1-35 Federal Register Vol. 64, No. 126, July 1, 1999, page 35714, 40 CFR Part 51, Regional Haze Regulations: Final Rule, July 1, 1999.

2.3.1-36 Federal Register Vol. 70, No. 128, July 6, 2005, page 39104.

2.3.1-37 National Park Service Explore Nature, Explore Air Website. Map of all NPS, FWS, and USFS Class I Areas. http://www.nature.nps.gov/air/maps/classiloc.cfm Accessed 9/11/2023.

2.3.1-38 TIA-222-G, "Structural Standards for Antenna Support Structures and Antennas", 2009.

2.3.1-39 2022 Annual Local Climatological Data (LCD) for Chattanooga, TN; National Climatic Data Center.

Replace Table 2.3.1-1 with the following:

Table 2.3.1-1R Conversion of Oak Ridge and Knoxville Wind Speeds to Consistent Units

Station	Time Period	Observed Maximum (mph)		Estimated Maximum (mph)	
		Fastest Mile	3-second Gust	Fastest Mile	3-second Gust
Oak Ridge	1985–1998(a)	NR	51 (Nov 1995)	40	51
	2000–2022(b)	NR	53 (Feb 2009)	42	53
Knoxville	1942–1974(c)	73 (Jul 1961)	NR	73	87
	1985–1995(d)	NR	86 (Aug 1995)	72	86
	1996–2022(e)	NR	76 (Apr 1996)	63	76

Station	Time Period	Maximum Hourly Average (mph)	Adjustment to 3-second gust (mph)
TVA Sequoyah[(f)](10-m level)	1971–2018	29.8 (Dec 18, 1972)	45
TVA Watts Bar[(g)](10-m level)	1973–2018	30.0 (Mar 25, 1975)	45
CRN Site (10-m level)	April 21, 2011–July 9, 2014	15.1 (Feb 19, 2013)	23

(a) Reference 2.3.1-13

(b) Reference 2.3.1-1

(c) Reference 2.3.1-10

(d) Reference 2.3.1-8

(e) Reference 2.3.1-2

(f) Reference 2.3.1-14

(g) Reference 2.3.1-15

Notes:

NR = Not Recorded

Conversions are based on TIA-222-G, “Structural Standards for Antenna Support Structures and Antennas” (Reference 2.3.1-38), Annex L.

Replace Table 2.3.1-2 with the following:

Table 2.3.1-2R Precipitation for Stations around Clinch River Nuclear Site

Station	Period of Record (years)	Normal Annual Rainfall (in.)	Max 24-hour Rainfall (in.)	Max Monthly Rainfall (in.)	Normal Annual Snowfall (in.)	Max 24-hour Snowfall (in.)	Max Monthly Snowfall (in.)
Oak Ridge NWS Station	30[a]	50.91			11.1		
	66[b]		7.48 (Aug 1960)	19.27 (Jul 1967)			
	52[b]					12.0 (Mar 1960)	21.0 (Mar 1960)
Knoxville NWS Station[b]	30	51.93			6.5		
	81		5.98 (Sep 2011)	13.08 (Feb 2019)			
	78					18.2 (Nov 1952)	23.3 (Feb 1960)
Chattanooga NWS Station[b]	30	55.00			3.9		
	74		9.50 (Sep 2011)	16.32 (Mar 1980)			
	76					20.0 (Mar 1993)	20.0 (Mar 1993)
Nashville NWS Station[b]	30	50.51			4.7		
	83		9.09 (May 2010)	16.43 (May 2010)			
	75					10.2 (Dec 1963)	18.9 (Feb 1979)
TVA Sequoyah[c]		30	45.79				
		45		8.04 (Sep 2004)	13.34 (Mar 1980)		
TVA Watts Bar[d]		30	45.70				
		46		8.43 (Sep 2011)	12.84 (Jul 2015)		

(a) Reference 2.3.1-13

(b) Reference 2.3.1-1, 2.3.1-2, 2.3.1-5, and 2.3.1-6

(c) Reference 2.3.1-14

(d) Reference 2.3.1-15

Replace Table 2.3.1-3 with the following:

Table 2.3.1-3R Tornadoes Reported within 10 Miles of Clinch River Nuclear Site (1950–2022)

Date	Counties	Magnitude	Length (miles)	Width (yards)	Closest Distance to CRN Site
10/1/1977	Roane	F0 (40–72 mph)	0.2	100	9 miles
2/21/1993	Roane, Loudon, Blount	F3 (158–206 mph)	30	100	4 miles
5/18/1995	Morgan	F0 (40–72 mph)	0.5	23	10 miles
4/27/2011	Knox	EF0 (65–85 mph)	1	50	10 miles
6/22/2011	Roane	EF0 (65–85 mph)	0.6	20	9 miles
6/10/2014	Roane	EF0 (65–85 mph)	0.5	100	9 miles

Sources:

Reference 2.3.1-21

Reference 2.3.1-4

Replace Table 2.3.1-4 with the following:

Table 2.3.1-4R Normal Temperature and Precipitation

Period	Normal Daily Maximum Temperature (°F)	Normal Dry-Bulb Temperature (°F)	Normal Daily Minimum Temperature (°F)	Normal Annual Precipitation (in.)
Knoxville NWS Station[a]				
1921–1950	70.2	59.3	48.4	45.51
1931–1960	70.4	59.6	48.8	45.85
1941–1970	69.8	59.7	49.5	46.18
1951–1980	69.0	58.9	48.7	47.29
1961–1990	68.9	57.6	46.3	47.14
1971–2000	68.3	58.9	48.4	48.22
1981–2010	69.5	59.1	48.8	47.86
1991–2020	70.0	59.5	49.2	51.93
Oak Ridge NWS Station[b]				
1921–1950	70.2	57.9	45.5	53.72
1931–1960	69.8	58.5	47.2	54.71
1941–1970	68.6	57.8	47.0	52.60
1951–1980	68.4	57.5	46.5	54.76
1961–1990	68.0	56.6	45.1	53.77
1971–2000	68.9	57.6	46.2	55.05
1981–2010	69.4	58.0	48.3	50.91
1991–2020	70.4	59.8	49.3	55.53

(a) 1953, 1963, 1973, 1983, 1993, 2003, and 2013 Annual Local Climatological Data (LCD) for Knoxville, TN; National Climatic Data Center. (Reference 2.3.1-30)

(b) 1953, 1963, 1973, 1983, 1993, 2003, and 2013 Annual Local Climatological Data (LCD) for Oak Ridge, TN; National Climatic Data Center. (Reference 2.3.1-31)

Replace Table 2.3.1-5 with the following:

Table 2.3.1-5R Ambient Air Quality Concentrations in the Vicinity of Clinch River Nuclear Site in 2013 (2022)

Pollutant	Roane County	Anderson County	Blount County	Knox County	Loudon County	NAAQS
4^{th} Max 8-hr Ozone (ppm)	ND (ND)	0.060 (0.059)	0.064 (0.068)	0.061 (0.060)	0.061 (0.058)	0.070
2^{nd} Max 24-hr PM_{10} ($\mu g/m^3$)	ND (ND)	ND (ND)	ND (ND)	27 (37)	ND (ND)	150
98^{th} Percentile 24-hr $PM_{2.5}$ ($\mu g/m^3$)	18 (17)	ND (ND)	17 (15)	20 (17)	17 (13)	35
Annual Mean $PM_{2.5}$ ($\mu g/m^3$)	8.6 (7.1)	ND (ND)	8.6 (7.0)	10.2 (8.7)	9.1 (6.3)	15
99^{th} Percentile1-hr SO_2 (ppb)	ND (ND)	5 (ND)	4 (ND)	ND (ND)	ND (ND)	75

Notes:

ND indicates No Data.

Parentheses indicate values for 2022

ppm = parts per million

$\mu g/m^3$ = micrograms per cubic meters

ppb = parts per billion

NAAQS, PM_{10} PM_2.

Add new Table 2.3.1-6:

Table 2.3.1-6 Nonattainment/Maintenance Status for Each County by Year for All Criteria Pollutants

County	NAAQS	Nonattainment (years)	Redesignation to Maintenance (years)
Anderson	8-Hour Ozone (1997)[1]	2004-2010	3/8/2011
	8-Hour Ozone (2008)	2012-2014	8/12/2015
	$PM_{2.5}$ (1997)[2]	2005-2016	8/29/2017
	$PM_{2.5}$ (2006)	2009-2016	9/27/2017
Blount	8-Hour Ozone (1997)[1]	2004-2010	3/8/2011
	8-Hour Ozone (2008)	2012-2014	8/12/2015
	$PM_{2.5}$ (1997)[2]	2005-2016	8/29/2017
	$PM_{2.5}$ (2006)	2009-2016	9/27/2017
Knox	1-Hour Ozone (1979)[1]	1992	10/27/1993
	8-Hour Ozone (1997)[1]	2004-2010	3/8/2011
	8-Hour Ozone (2008)	2012-2014	8/12/2015
	$PM_{2.5}$ (1997)[2]	2005-2016	8/29/2017
	$PM_{2.5}$ (2006)	2009-2016	9/27/2017
Loudon	8-Hour Ozone (1997)[1]	2004-2010	3/8/2011
	$PM_{2.5}$ (1997)[2]	2005-2016	8/29/2017
	$PM_{2.5}$ (2006)	2009-2016	9/27/2017
Roane	$PM_{2.5}$ (1997)[2]	2002-2016	8/29/2017
	$PM_{2.5}$ (2006)	2009-2016	9/27/2017

2.3.2 Local Meteorology

Replace the first sentence of the third paragraph in ESPA SSAR Section 2.3.2 with the following:

The CRN Site is located at an elevation of approximately 814.5 feet above mean sea-level on a peninsula formed by the Clinch River arm of the Watts Bar Reservoir.

Replace the last sentence of the third paragraph in ESPA SSAR Section 2.3.2 with the following:

Figures 2.3.2-70 through 2.3.2-77 show the elevation profiles within 50 miles of the site in each of the compass directions.

2.3.2.1.1 Winds

Replace the third and fourth sentences of the second paragraph in ESPA SSAR Section 2.3.2.1 with the following:

Wind roses for Chattanooga, based on ten years of data (2012--2022), are presented in Figures 2.3.2-29R through 2.3.2-41R. Wind roses for Oak Ridge, based on ten years of data (2012–2022), are presented in Figures 2.3.2-42R through 2.3.2-54R and wind roses for Knoxville based on 2012-2022 are presented in Figures 2.3.2-55R through 2.3.2-67R.

Replace Figure 2.3.2-55 with Figure 2.3.2-68 in the second sentence of the fifth paragraph in ESPA SSAR Section 2.3.2.1.1.

2.3.2.1.2 Air Temperature

Replace Tables 2.3.2-5 and 2.3.2-6 with Tables 2.3.2-5R and 2.3.2-6R in ESPA SSAR Section 2.3.2.1.2.

2.3.2.1.3 Atmospheric Moisture

Replace Table 2.3.2-7 with Table 2.3.2-7R in ESPA SSAR Section 2.3.2.1.3.

2.3.2.1.4 Precipitation

Replace the second sentence of the fourth paragraph in ESPA SSAR Section 2.3.2.1.4 with the following:

This data was taken from NOAA Atlas 14 Point Precipitation Frequency Estimates Data Server (Reference 2.3.2-4) and NOAA Atlas 14 Precipitation-Frequency Atlas of the United States (Reference 2.3.2-5).

Replace Table 2.3.2-9 with Table 2.3.2-9R in ESPA SSAR Section 2.3.2.1.4.

Replace Table 2.3.2-10 with Table 2.3.2-10R in ESPA SSAR Section 2.3.2.1.4.

Replace Table 2.3.2-11 with Table 2.3.2-11R in ESPA SSAR Section 2.3.2.1.4.

Replace Table 2.3.2-16 with Table 2.3.2-16R in ESPA SSAR Section 2.3.2.1.4.

Replace Reference 2.3.2-4 with Reference 2.3.2-6 in the sixth paragraph in ESPA SSAR Section 2.3.2.1.4.

2.3.2.1.5 Fog

Replace Table 2.3.2-13 with Table 2.3.2-13R in ESPA SSAR Section 2.3.2.1.5.

2.3.2.1.6 Atmospheric Stability

Replace Reference 2.3.2-5 with Reference 2.3.2-7 in the third paragraph in ESPA SSAR Section 2.3.2.1.6.

2.3.2.2 Potential Influence of the Plant and Its Facilities on Local Meteorology

Replace the second paragraph in ESPA SSAR Section 2.3.2.2 (Addresses ESP COL Action Item 2.3-1):

The specific cooling tower design has not been determined for the CRN Site. Subsequent analysis performed since the CRN ESPA has shown that the cooling tower plume characteristics meet the NUREG 1555 guidelines. The updated evaluations of cooling tower impacts using the BWRX-300 layout on the CRN Site, (assuming linear, back-to-back cell, mechanical draft cooling towers (LMDCT)), confirmed that no salt deposition rates in the highest category are expected. The nearest offsite vegetation receptor (opposite side of the river) is approximately 900 feet (274 meters). The limiting salt deposition occurs during the winter. At 200 meters, the highest deposition rate is 354.52 kg/km^2/month in the ENE direction. Therefore, no significant impact to local vegetation is expected and the assumed cooling tower is suitable for use at the CRN Site.

The minimum distance to the switchyard is approximately 1184 feet (360 meters) and the minimum distance to the transformers is approximately 721 feet (220 meters). Thus, the distance to the transformers is the limiting distance for salt deposition. The maximum salt deposition rate is in the winter with a value of 203.09 kg/km^2/month at 200 meters in the WSW direction. Therefore, impact to the switchyard and the plant transformers from cooling tower salt deposition is expected to be minimal. Water deposition, annual salt and Total Dissolved Solids (TDS) deposition, plume shadowing, plume length frequency, and plume height frequency were also determined for the LMDCTs. Results for these parameters are illustrated in Figures 2.3.2-78R through 2.3.2-82R.

2.3.2.4 References

Replace ESPA SSAR Section 2.3.2.4 with the following:

2.3.2-1 National Climatic Data Center, “2022 Annual Local Climatological Data for Knoxville, Tennessee,” 2023.

2.3.2-2 National Climatic Data Center, “2022 Annual Local Climatological Data (LCD) for Oak Ridge,” Tennessee, 2023.

2.3.2-3 National Climatic Data Center, “2011–2013 Monthly Local Climatological Data (LCD) for Oak Ridge, Tennessee,” 2014.

2.3.2-4 NOAA ATLAS 14 Point Precipitation Frequency Estimates: TN, Precipitation Frequency Data Server. https://hdsc.nws.noaa.gov/pfds/pfds_map_cont.html?bkmrk=tn, Accessed: September 12, 2023

2.3.2-5 NOAA Atlas 14 Precipitation-Frequency Atlas of the United States, Vol 2 Version 3.0, U.S. Department of Commerce, 2006.

2.3.2-6 National Climatic Data Center, “1998 Annual Local Climatological Data (LCD) for Oak Ridge, Tennessee,” 1999.

2.3.2-7 Holzworth, George. C., “Mixing Heights, Wind Speeds, And Potential for Urban Air Pollution Throughout the Contiguous United States,” Environmental Protection Agency, January 1972.

2.3.2-8 EPRI CS-3403-CCM, “User’s Manual: Cooling-Tower-Plume Prediction Code,” September 1987.

2.3.2-9 NUREG-1555, Rev. 0, “Standard Review Plans for Environmental Reviews for Nuclear Power Plants,” October 1999.

2.3.2-10 IEEE Standard C57.19.100-1995, “IEEE Guide for Application of Power Apparatus Bushings.”

Replace Table 2.3.2-5 with the following:

Table 2.3.2-5R Air Temperatures for Knoxville, Tennessee

	Normal Daily Maximum	Normal Dry Bulb	Normal Daily Minimum	Extreme Daily Maximum	Extreme Daily Minimum
Period of Record[a] (yr)	**30[a]**	**30[a]**	**30[a]**	**81[d]**	**81[d]**
January	48.2	39.1	30	77	-24[c]
February	52.8	42.9	33.1	83	-8
March	61.4	50.6	39.8	86	1
April	71.2	59.6	48	92	22
May	78.9	67.9	56.9	96	32
June	85.7	75.3	64.9	105[b]	43
July	88.4	78.5	68.7	105[b]	49
August	87.8	77.6	67.5	102	49
September	82.5	71.8	61.1	103	36
October	71.7	60.3	48.9	96	25
November	60	49.0	38.1	85	5
December	51	41.9	32.8	80	-6
Annual	70	59.5	49.2	105[b]	-24[c]

(a) 1991-2020

(b) June 2012 and July 2012

(c) January 1985

(d) Reference 2.3.2-1

Notes:

Air Temperature (°F) from 2022 Annual Knoxville Local Climatological Data.

Replace Table 2.3.2-6 with the following:

Table 2.3.2-6R Air Temperatures for Oak Ridge, Tennessee

	Normal Daily Maximum	Normal Dry Bulb	Normal Daily Minimum	Extreme Daily Maximum	Extreme Daily Minimum
Period of Record[a] (yr)	**30[a]**	**30[a]**	**30[a]**	**75**	**75**
January	48.1	39.1	30	76	-17[c]
February	53	43	33	81	-13
March	62.2	51.1	40	86	1
April	72.1	60	47.9	92	20
May	79.7	68.3	56.9	95	30
June	86.1	75.6	65.2	105[b]	39
July	89.1	79	69	105[b]	49
August	88.4	78.1	67.7	103	50
September	82.8	72	61.2	102	33
October	72.1	60.6	49	98	21
November	60.1	49.2	38.3	85	0
December	50.8	42	33.2	78	-7
Annual	70.4	59.8	49.3	105[b]	-17[c]

(a)1991-2020

(b)July 1952 and June 2012

(c)January 1985

Notes:

Air Temperature (°F) from 2022 Annual Oak Ridge Local Climatological Data.

Replace Table 2.3.2-7 with the following:

Table 2.3.2-7R Humidity Values for Knoxville and Oak Ridge, Tennessee

Knoxville, TN	Mean Dry Bulb Temperature[(a)]	Mean Dewpoint Temperature[(a)]	Mean Relative Humidity (%)[(a)]	Mean Absolute Humidity (g/m^3)
January	39.2	31.8	74	4.71
February	41	34.2	70	4.71
March	50	40	66	6.20
April	58.6	48.7	65	8.20
May	64.5	58.7	73	11.25
June	74.3	66	75	15.87
July	78.2	69.4	75	17.98
August	77.1	68.5	76	17.56
September	71	62.5	75	14.29
October	60.3	52	75	10.05
November	48.3	41.6	74	6.55
December	41.3	35.2	75	5.15

Oak Ridge, TN	Mean Dry Bulb Temperature[(b)]	Mean Dewpoint Temperature	Mean Relative Humidity (%)[(c)]	Mean Absolute Humidity (g/m^3)
January	37.1	32.9	69	4.18
February	40.7	35.6	66	4.37
March	49.7	42.4	57	5.95
April	58.6	50.7	61	7.97
May	66.8	59.9	71	11.81
June	74.4	66.8	65	14.65
July	77.8	70.5	78	17.71
August	77	69.6	77	16.81
September	70.7	63.7	75	14.34
October	59.3	53.4	69	9.46
November	48.3	42.5	68	6.08
December	40.7	36.7	77	5.12

(a) Reference 2.3.2-1

(b) Reference 2.3.2-2

(c) Mean Relative humidity for Oak Ridge from 2022 Annual Oak Ridge Local Climatological Data.

Notes:

Temperatures and Dewpoints (°F) from 30 year Normals (1991-2020) Knoxville and Oak Ridge Local Climatological Data.

Replace Table 2.3.2-9 with the following:

Table 2.3.2-9R Historical Precipitation Data for Oak Ridge, Tennessee

	Normal Monthly	Maximum Monthly	Minimum Monthly	Maximum in 24 hours	Days with Precipitation (≥ 0.01 inch)	Days with Thunderstorms[a]
Period of Record (yr)	**30[b]**	**66[f]**	**66[f]**	**66[f]**	**30[b]**	**17**
January	5.07	13.27	0.93	4.25	10.9	0.7
February	5.59	12.78	0.84	5.18	10.1	1.7
March	5.12	12.24	2.13	4.74	11.2	2.5
April	5.2	14.03	0.88	6.24	10.4	4.0
May	4.07	10.70	0.80	4.41	11.9	7.0
June	4.55	11.14	0.53	3.70	10.8	7.6
July	5.54	19.27[c]	1.23	4.91	13.0	10.4
August	3.46	10.46	0.54	7.48[e]	8.9	8.7
September	3.83	10.14	0.13	6.54	8.4	3.3
October	3.12	8.02	Trace[d]	2.66	8.3	1.3
November	4.62	12.22	1.14	5.29	9.3	1.1
December	5.36	12.64	0.67	5.12	11.3	0.8
Annual	55.53	19.27[c]	Trace[d]	7.48[e]	124.5	49.1

(a) From 1998 Annual Oak Ridge Local Climatological Data

(b) 1991-2020

(c) July 1967

(d) October 1963

(e) August 1960

(f) Reference 2.3.2-2

Notes:

Precipitation (inches) from 2022 Annual Oak Ridge Local Climatological Data

Replace Table 2.3.2-10 with the following:

Table 2.3.2-10R Precipitation (Inches) for Oak Ridge During 2011-2013 From Oak Ridge Monthly Local Climatological Data
(Sheet 1 of 2)

Year	Month	Monthly Observed	
2011	January	3.99	
	February	5.70	
	March	6.65	
	April	9.13	
	May	2.14	
	June	7.30	
	July	4.80	
	August	0.91	
	September	10.14	
	October	4.59	
	November	10.89	Annual
	December	5.02	71.26
2012	January	6.52	
	February	3.76	
	March	5.59	
	April	3.10	
	May	2.84	
	June	1.40	
	July	5.84	
	August	2.89	
	September	7.17	
	October	1.66	
	November	1.14	Annual
	December	6.58	48.49

Table 2.3.2-10R Precipitation (Inches) for Oak Ridge During 2011-2013 From Oak Ridge Monthly Local Climatological Data
(Sheet 2 of 2)

Year	Month	Monthly Observed	
2013	January	10.51	
	February	2.32	
	March	5.72	
	April	6.37	
	May	5.33	
	June	7.92	
	July	8.04	
	August	4.61	
	September	3.38	
	October	0.72	
	November	4.43	Annual
	December	8.04	67.39
Normal Annual Precipitation 55.53 in. based on 30 years of data (1991-2020)			

Notes:

Shading represents the 26-month period of data collection at the Clinch River Nuclear Site.

Replace Table 2.3.2-11 with the following:

Table 2.3.2-11R Historical Snowfall for Knoxville and Oak Ridge, Tennessee

Knoxville, TN	Normal Monthly	Maximum Monthly	Maximum in 24 hours	Maximum Snow Depth (inches)	Normal Number of Days with Snowfall ≥ 1.0 inch
Period of Record (yr)	**30[a]**	**78**	**78**	**62**	**30[a]**
January	2.7	15.1	12.0	10	0.7
February	1.6	23.3[b]	17.5	15[d]	0.6
March	0.9	20.2	14.1	15[d]	0.2
April	0.5	10.7	10.7	7	0.0
May-October	0.0	Trace	Trace	0	0.0
November	0.0	18.2	18.2[c]	10	0.2
December	0.8	12.2	8.9	6	0.3
Annual	6.5	23.3[b]	18.2[c]	15[d]	1.7

Oak Ridge, TN	Normal Monthly	Maximum Monthly	Maximum in 24 hours	Maximum Snow Depth (inches)	Normal Number of Days with Snowfall ≥ 1.0 inch
Period of Record (yr)	**30[e]**	**52**	**52**	**34**	**30[e]**
January	4.0	9.6	8.3	8	1.4
February	3.8	17.2	11.3	6	1.3
March	0.8	21.0[f]	12.0[f]	3	0.2
April	0.2	5.9	5.4	3	0.1
May-October	0.0	Trace	Trace	0	0.0
November	0.1	6.5	6.5	1	0.0
December	2.2	14.8	10.8	10[g]	0.6
Annual	11.1	21.0[f]	12.0[f]	10[g]	3.6

(a) 1991–2020

(b) February 1960

(c) November 1952

(d) February 1960 and March 1993

(e) 1961–1990

(f) March 1960

(g) December 1963

Notes:

Snowfall (inches) from 2022 Annual Knoxville and 1998 Annual Oak Ridge Local Climatological Data.

Replace Table 2.3.2-13 with the following:

Table 2.3.2-13R Fog Occurrence for Knoxville and Oak Ridge, Tennessee

	Number of Days with Heavy Fog (visibility ≤ 1/4 mile)	
	Knoxville, TN	**Oak Ridge, TN**
Period of Record (yr)	59	23
January	2.2	2
February	1.6	1.5
March	1.4	1.3
April	1.2	2
May	2	4.3
June	1.6	3.9
July	1.8	4.9
August	2.9	4.8
September	3.2	7.2
October	3.6	7.2
November	2.4	4.9
December	2.1	3.9
Annual	26	47.9

Notes:

Days with heavy fog from 2022 Annual Oak Ridge and Knoxville Local Climatological Data.

Replace Table 2.3.2-16 with the following:

Table 2.3.2-16R Point Precipitation (Inches) by Recurrence Interval for Region

	Recurrence Intervals (yr)						
Duration	**1**	**2**	**5**	**10**	**25**	**50**	**100**
5 minutes	0.3	0.4	0.5	0.5	0.6	0.7	0.8
10 minutes	0.5	0.6	0.7	0.8	1.0	1.1	1.2
15 minutes	0.7	0.8	0.9	1.1	1.2	1.4	1.5
30 minutes	0.9	1.1	1.3	1.6	1.8	2.1	2.3
1 hour	1.1	1.4	1.7	2.0	2.5	2.8	3.2
2 hours	1.4	1.6	2.0	2.4	2.9	3.3	3.8
3 hours	1.5	1.8	2.2	2.5	3.1	3.5	4.0
6 hours	1.8	2.2	2.6	3.1	3.7	4.2	4.7
12 hours	2.3	2.7	3.3	3.8	4.5	5.1	5.7
24 hours	2.8	3.3	4.1	4.6	5.5	6.1	6.8
2 days	3.4	4.1	5.0	5.7	6.7	7.5	8.3
4 days	3.6	4.4	5.3	6.0	7.0	7.8	8.7
7 days	3.9	4.7	5.6	6.4	7.4	8.2	9.0
10 days	4.8	5.7	6.8	7.7	8.8	9.7	10.6

Notes:

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[1] Precipitation frequency (PF) estimates in this table are based on frequency analysis of partial duration series (PDS) (Reference 2.3.2-4)

Replace Figure 2.3.2-29 through Figure 2.3.2-54 with the following:

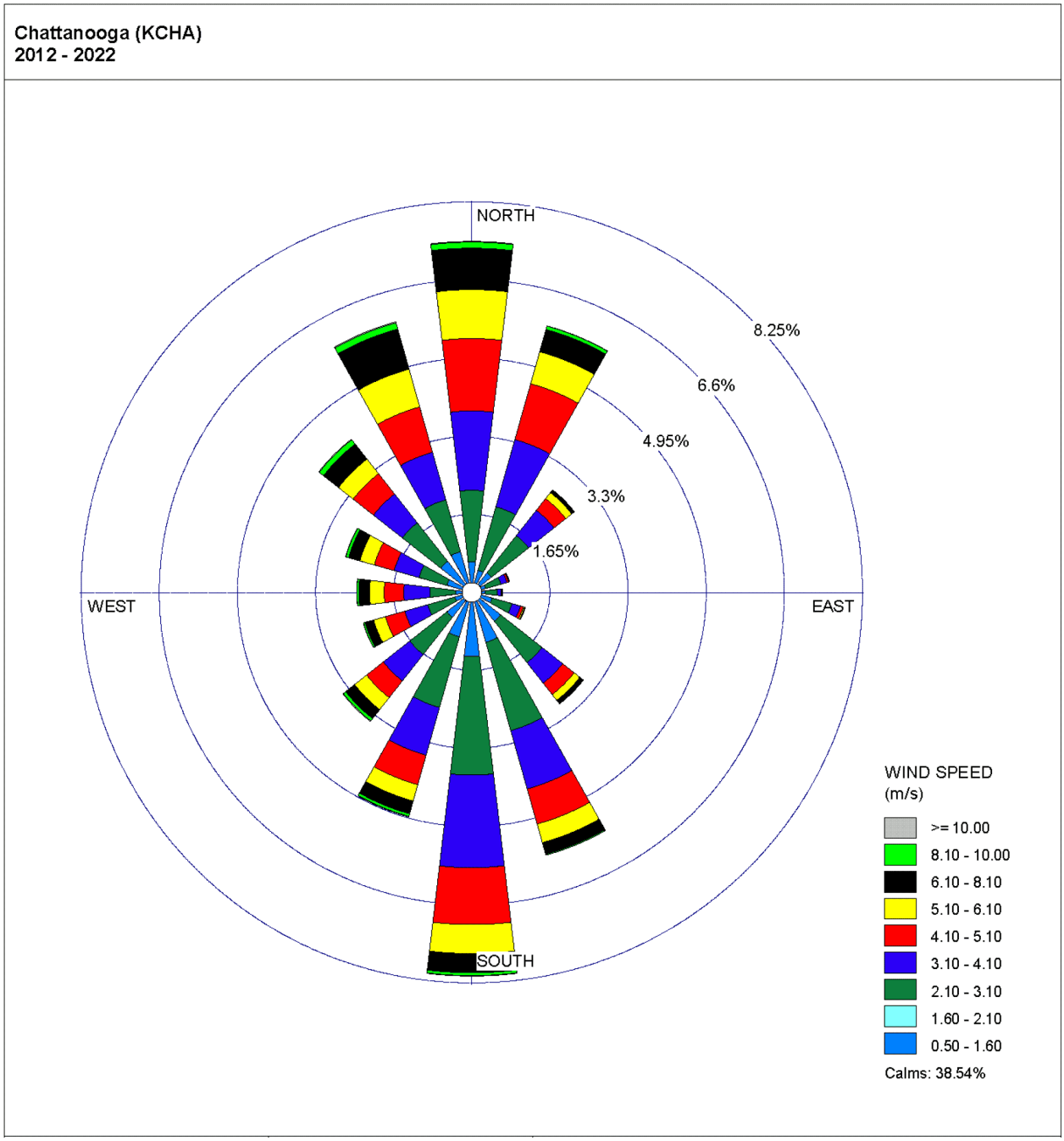

Figure 2.3.2-29R Wind Rose Chattanooga 10 Years (2012–2022) All Data

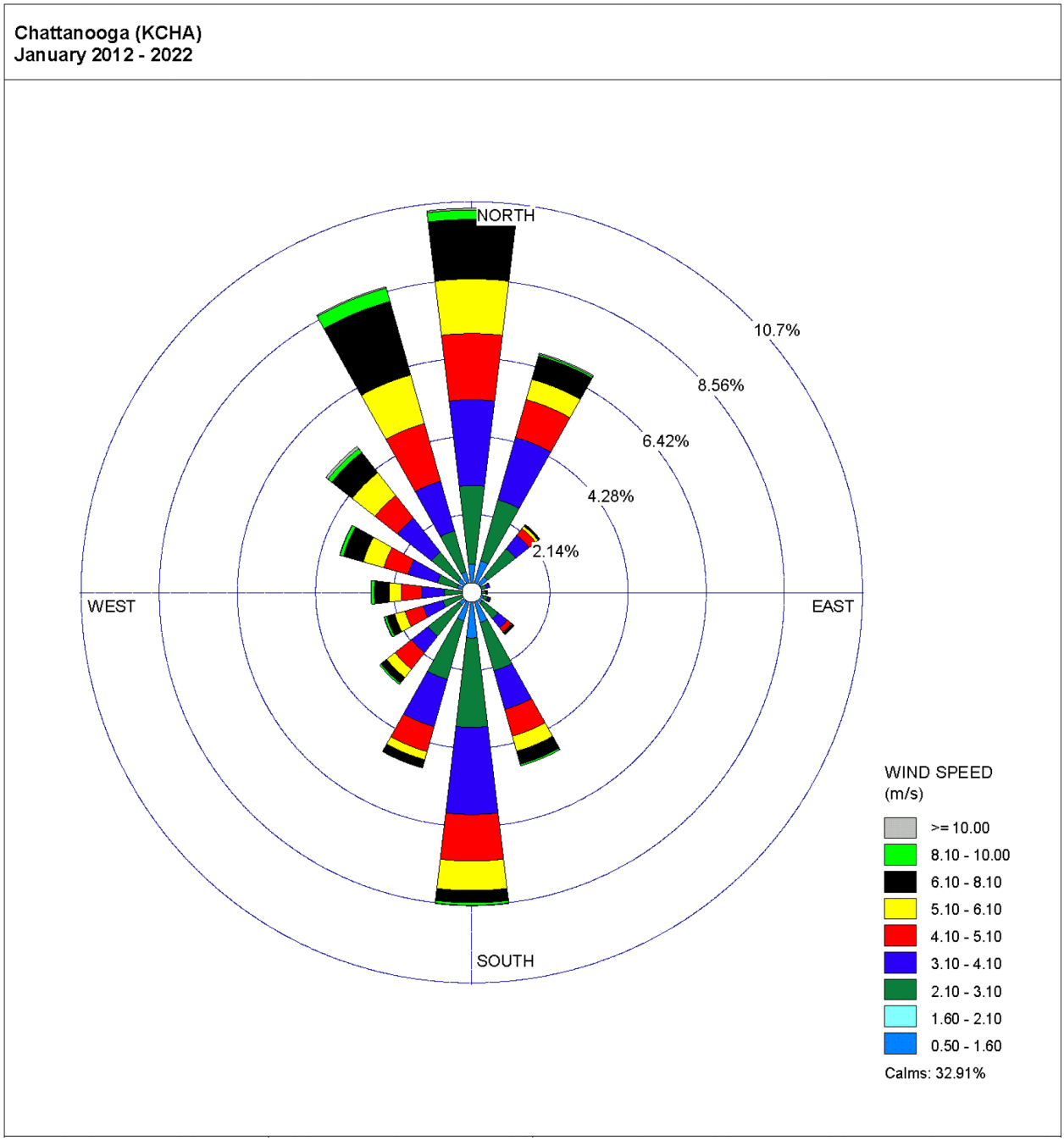

Figure 2.3.2-30R Wind Rose Chattanooga 10 Years (2012–2022) January

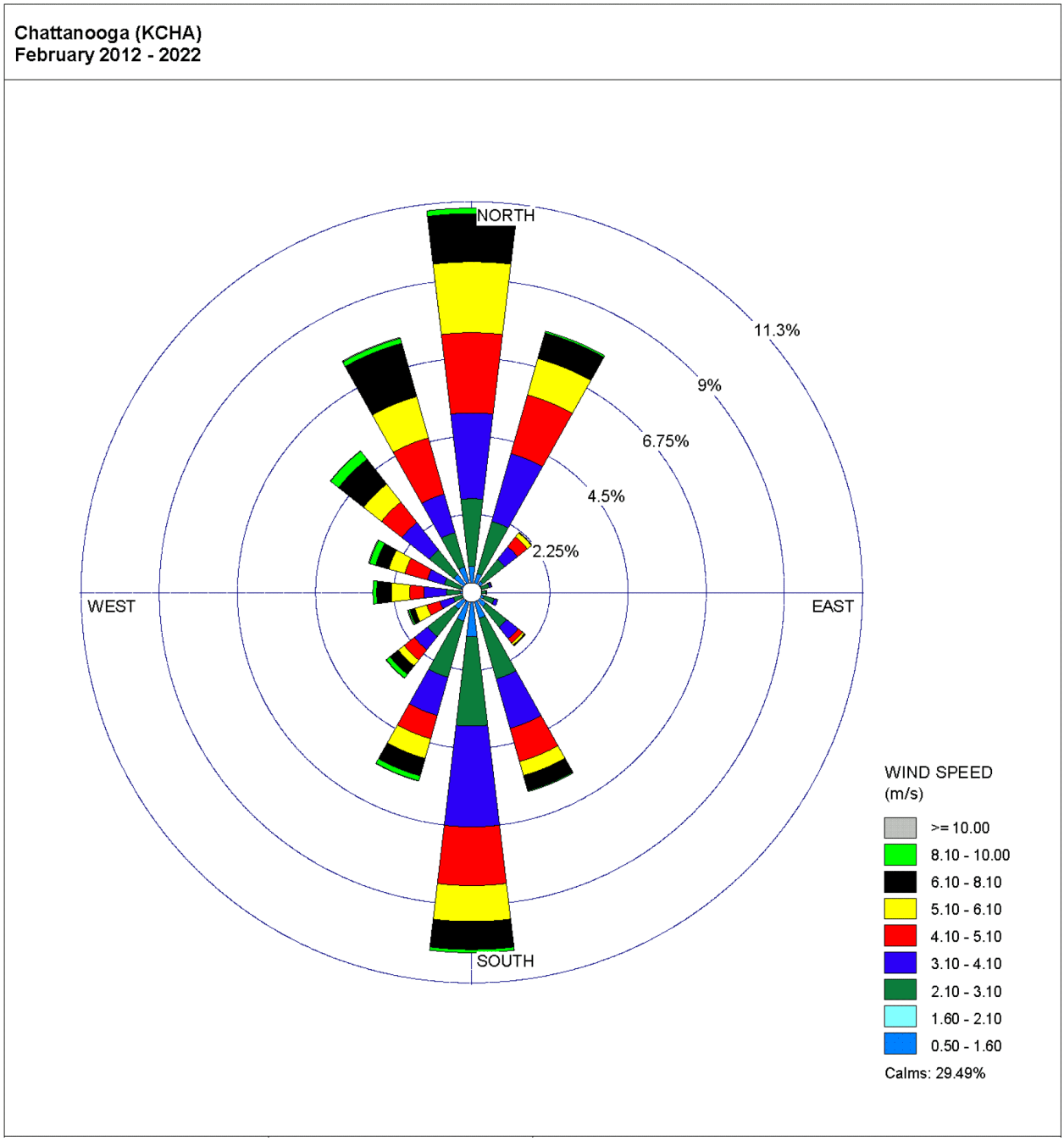

Figure 2.3.2-31R Wind Rose Chattanooga 10 Years (2012–2022) February

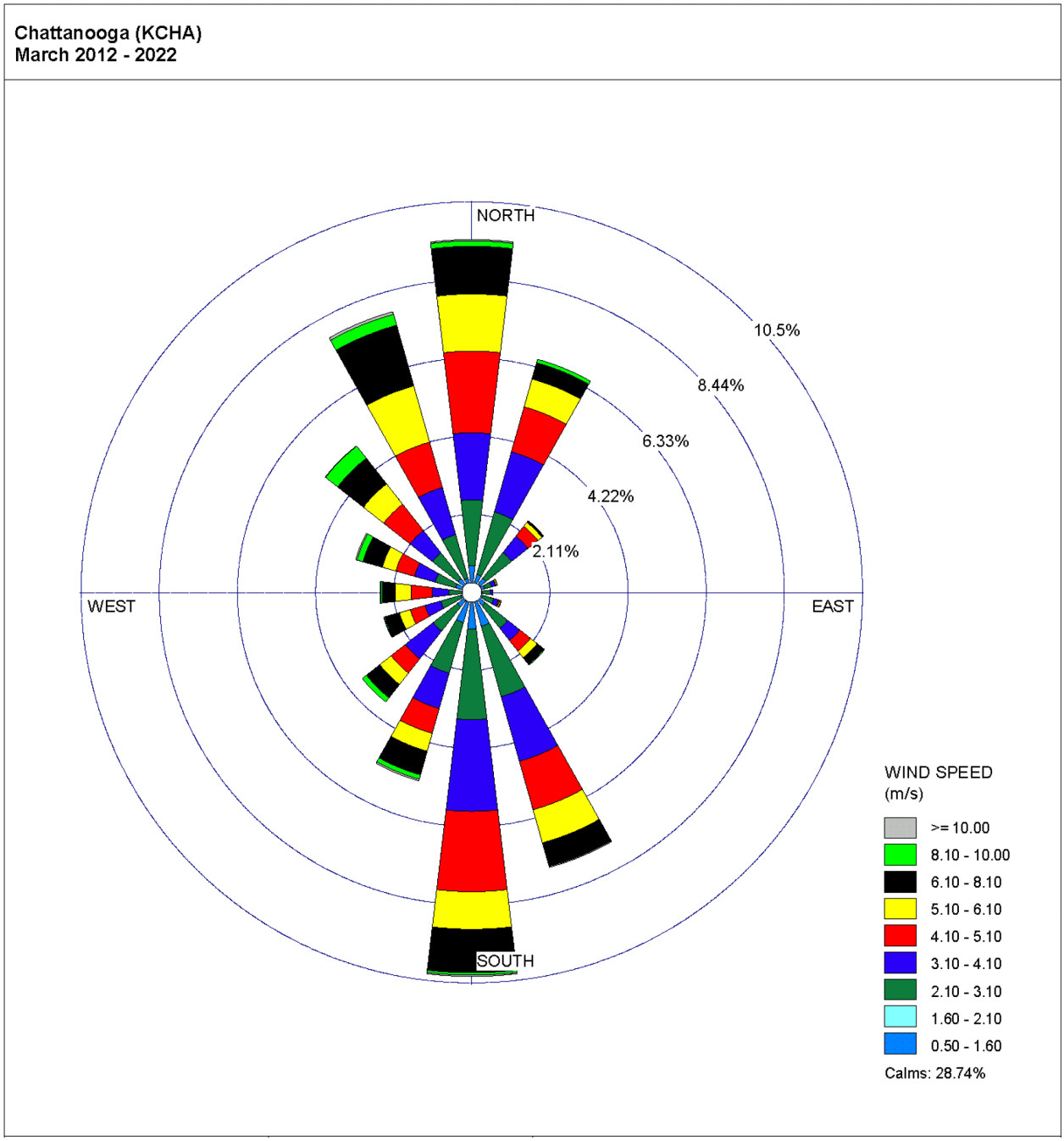

Figure 2.3.2-32R Wind Rose Chattanooga 10 Years (2012–2022) March

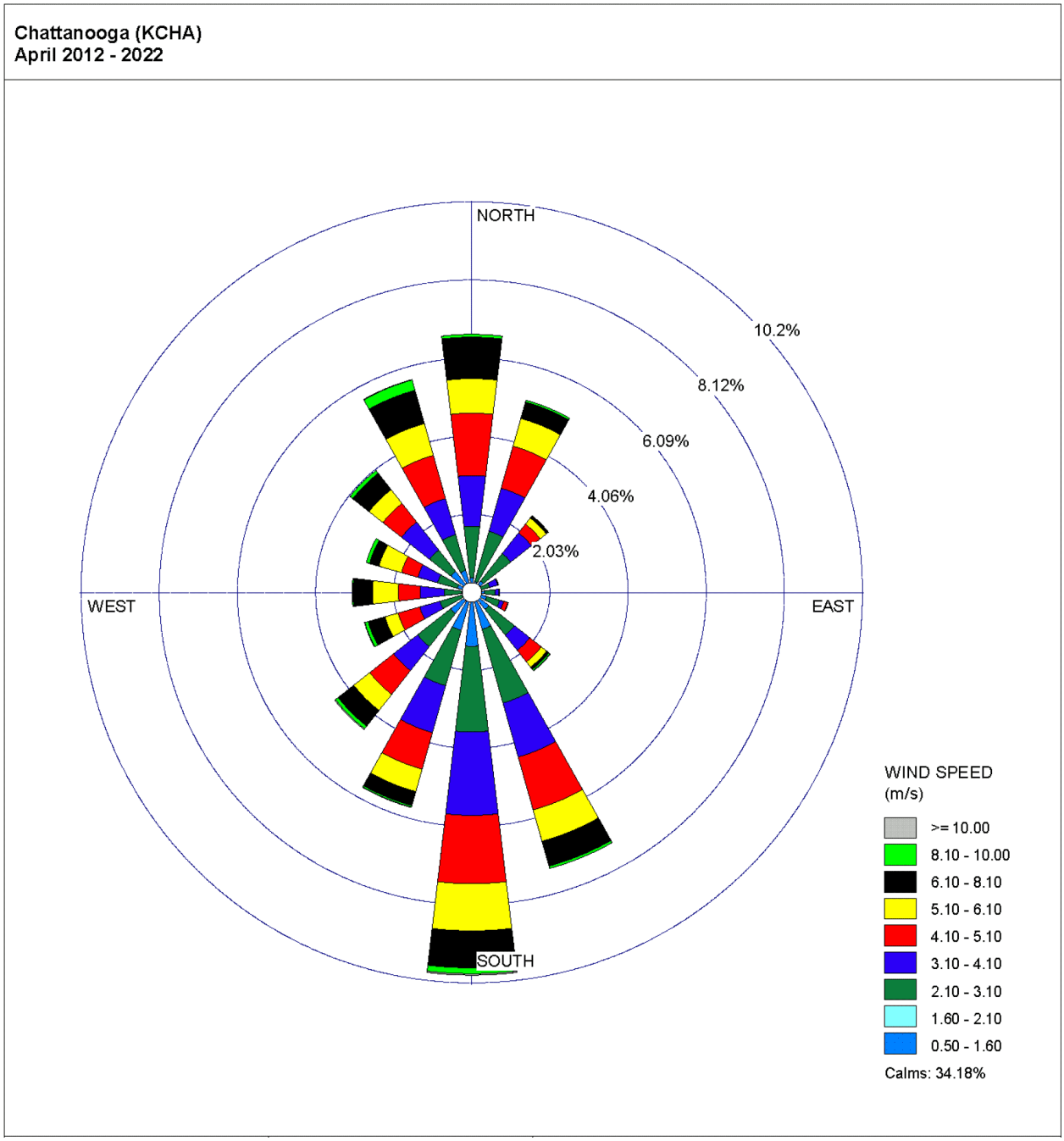

Figure 2.3.2-33R Wind Rose Chattanooga 10 Years (2012–2022) April

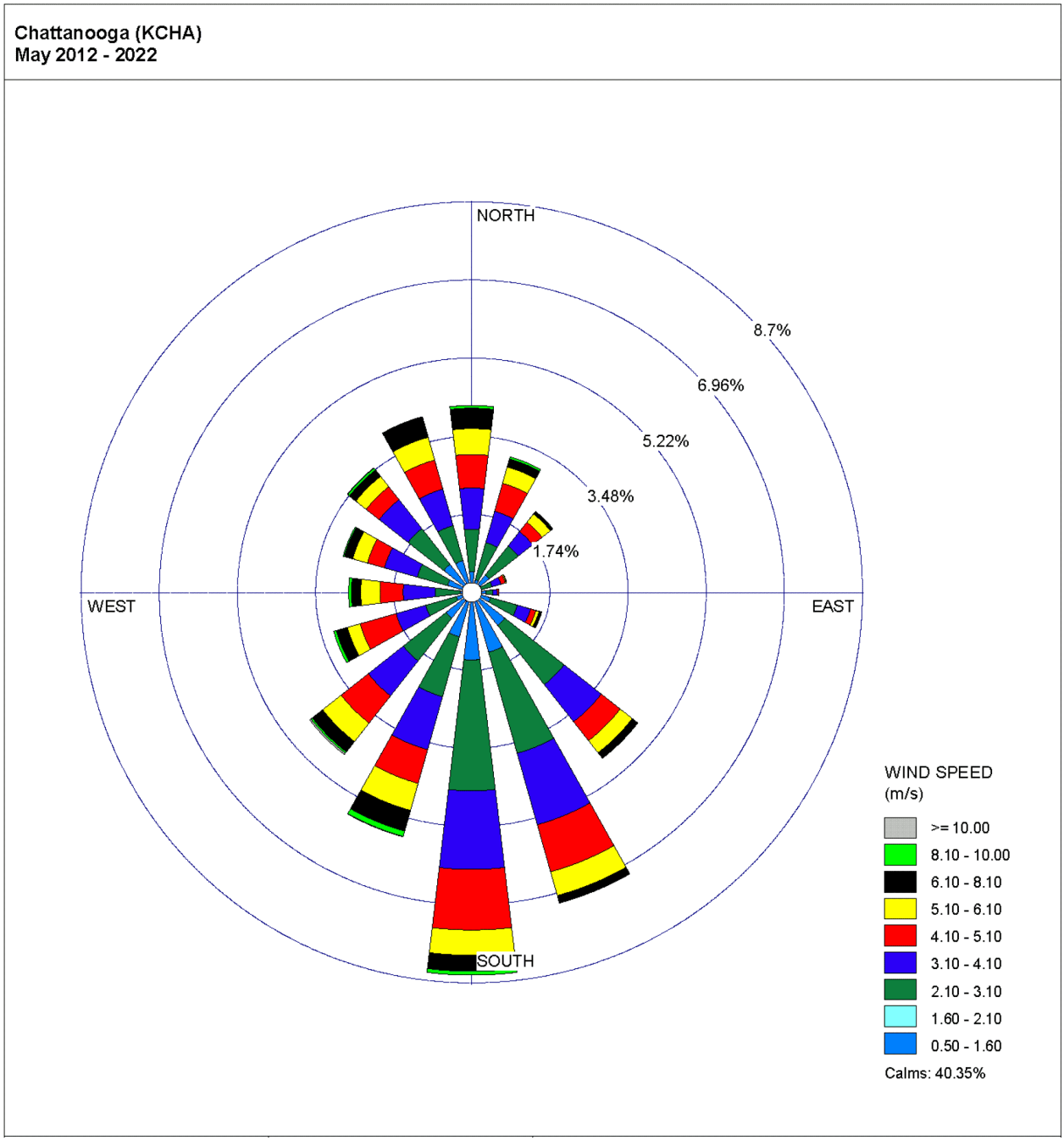

Figure 2.3.2-34R Wind Rose Chattanooga 10 Years (2012–2022) May

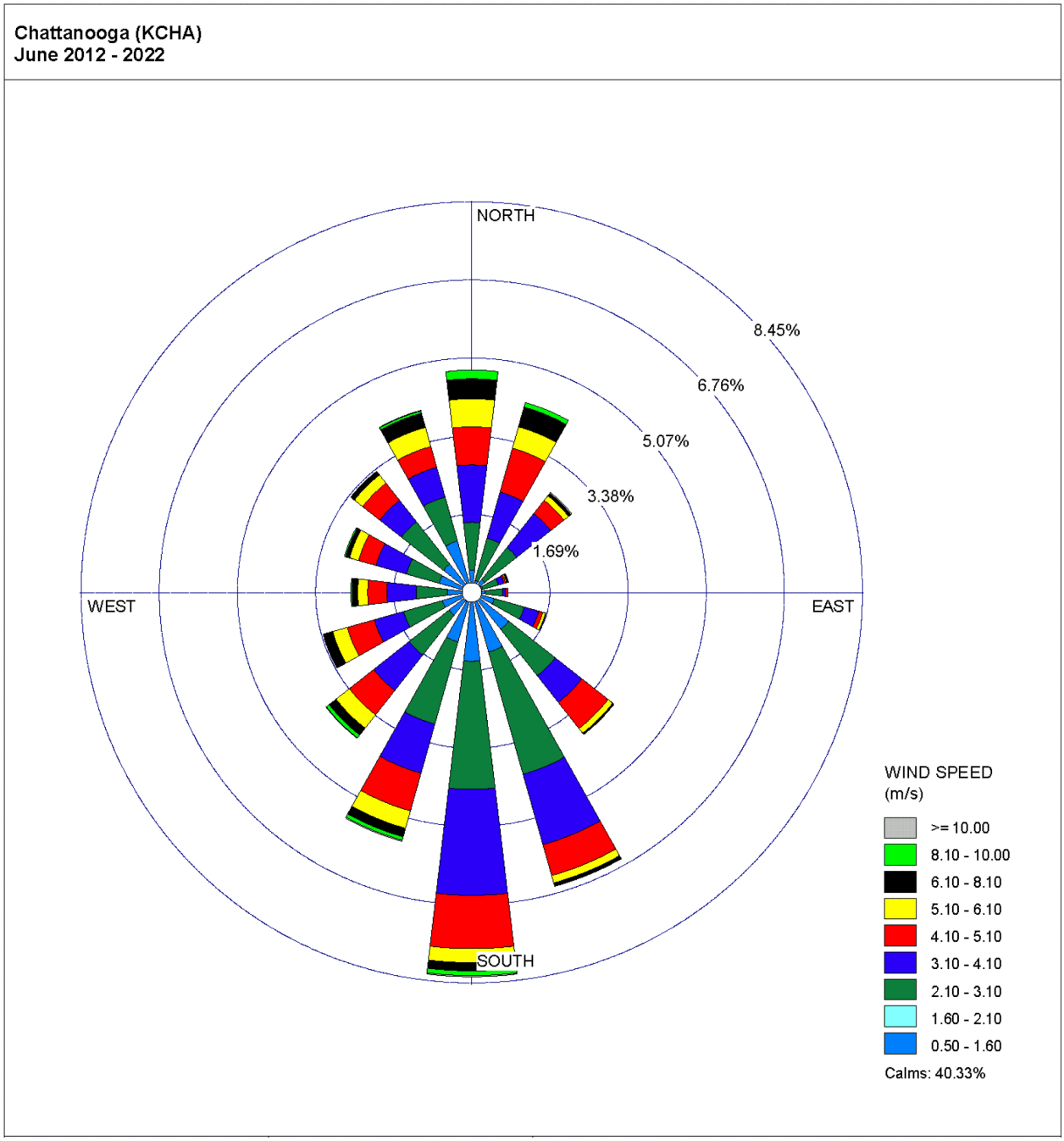

Figure 2.3.2-35R Wind Rose Chattanooga 10 Years (2011–2022) June

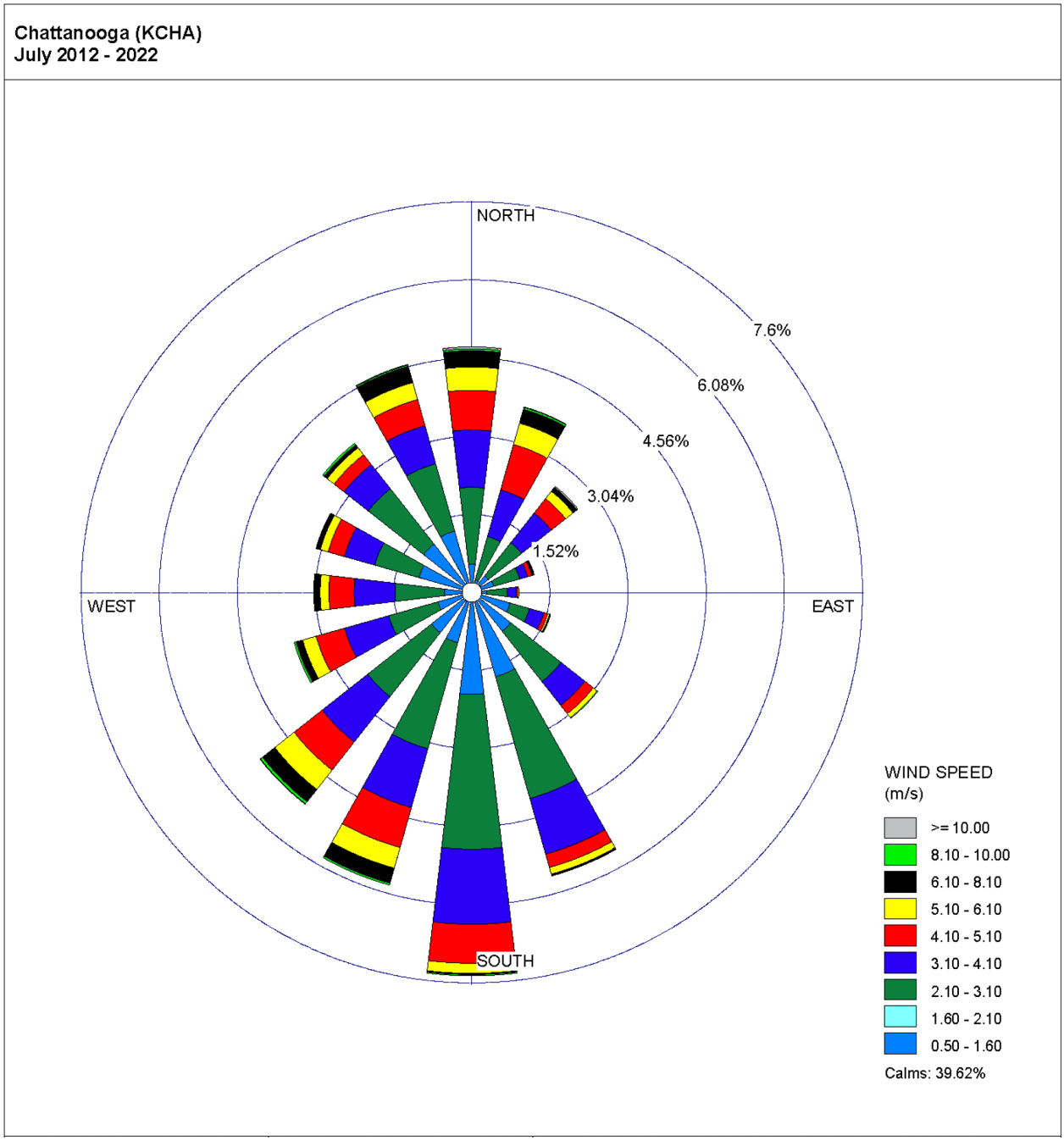

Figure 2.3.2-36R Wind Rose Chattanooga 10 Years (2012–2022) July

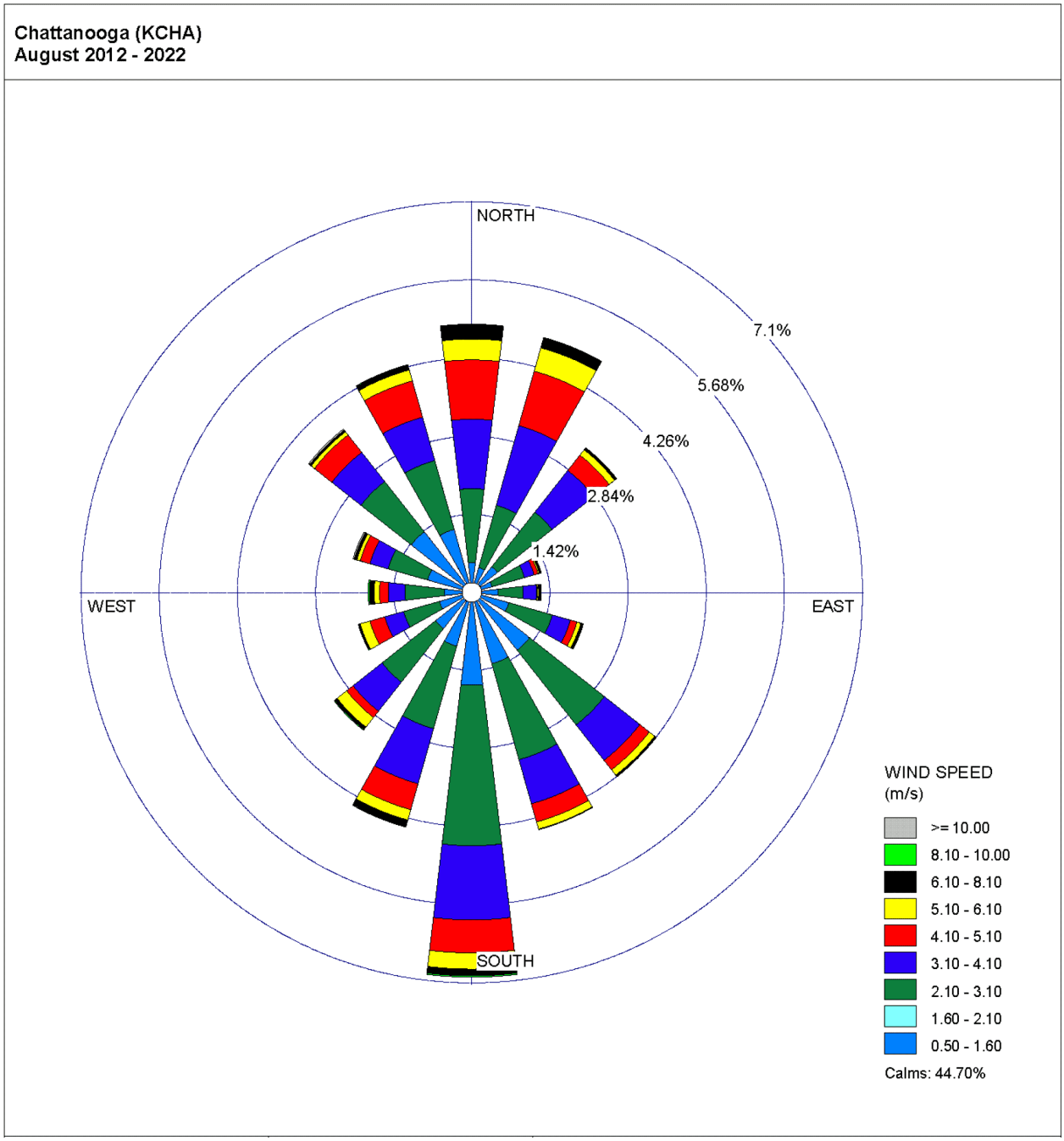

Figure 2.3.2-37R Wind Rose Chattanooga 10 Years (2012–2022) August

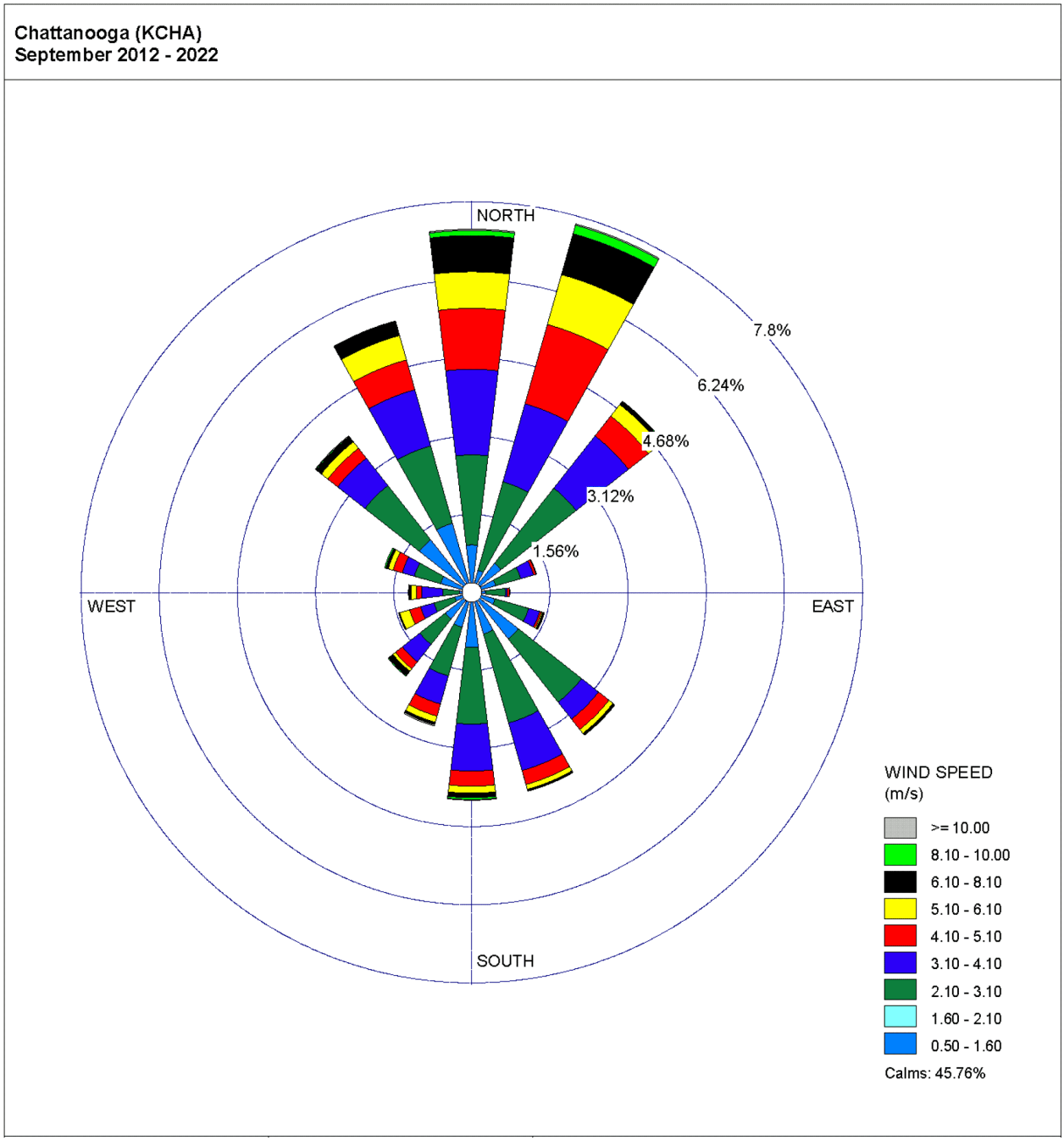

Figure 2.3.2-38R Wind Rose Chattanooga 10 Years (2012–2022) September

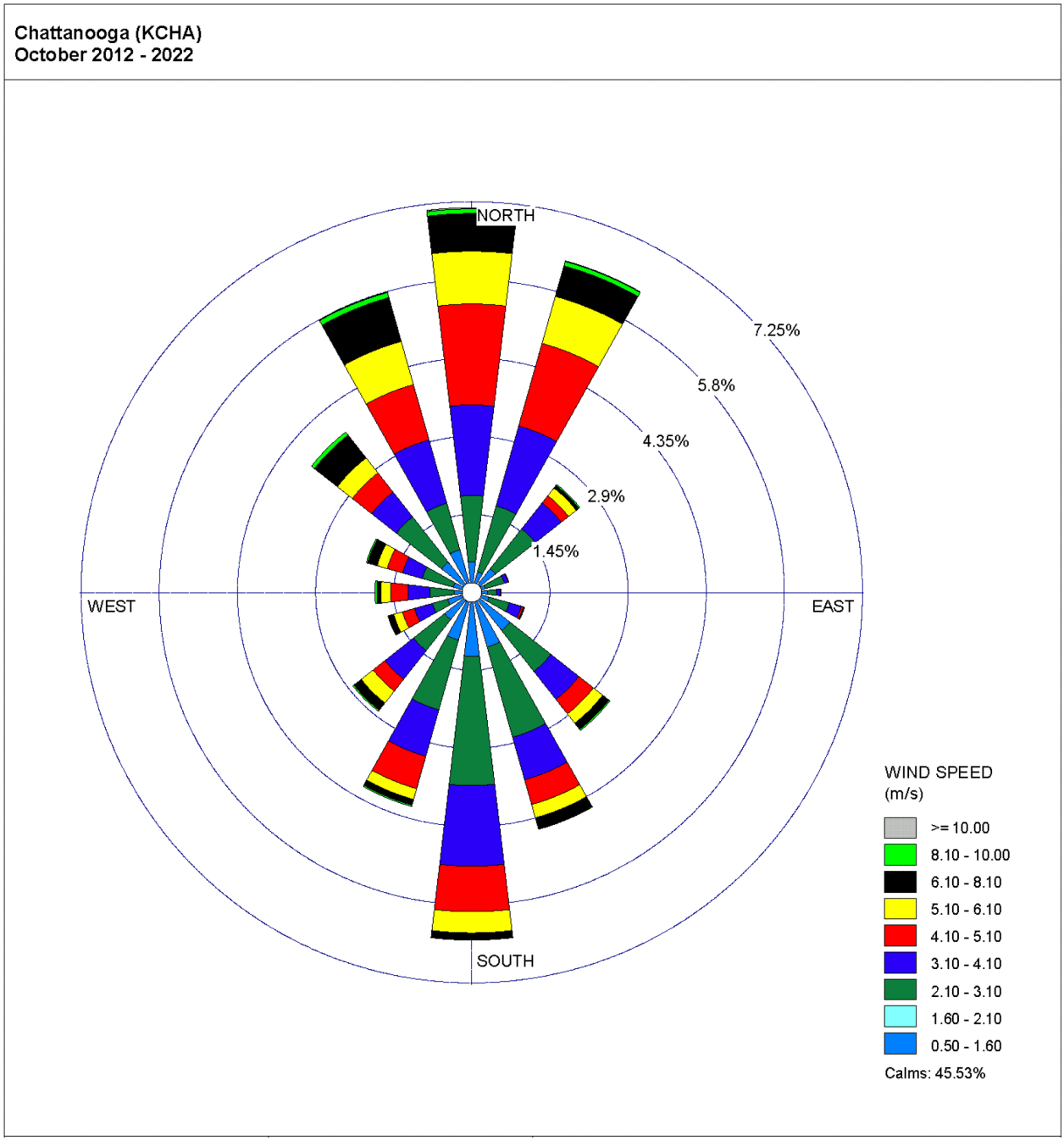

Figure 2.3.2-39R Wind Rose Chattanooga 10 Years (2012–2022) October

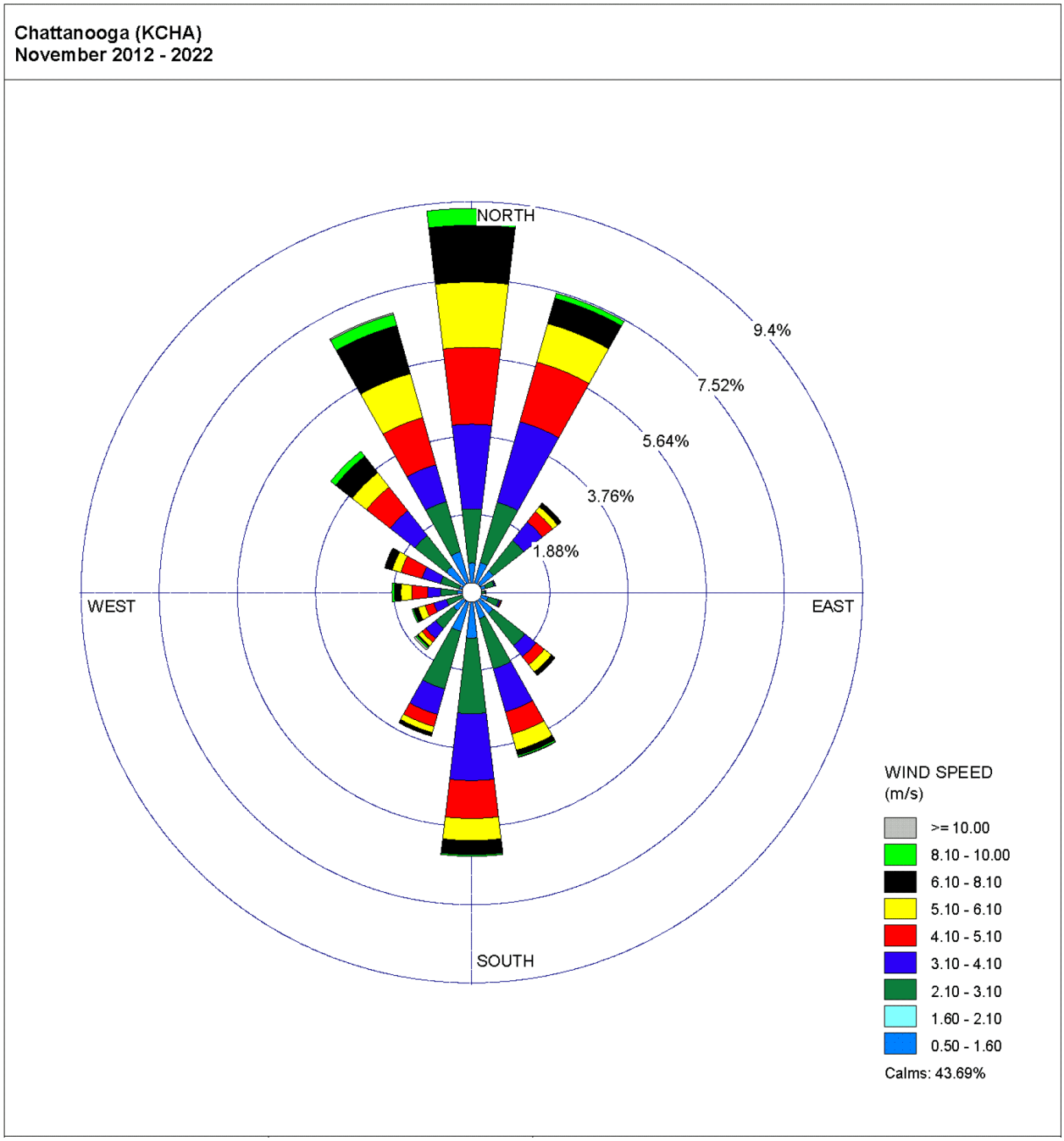

Figure 2.3.2-40R Wind Rose Chattanooga 10 Years (2012–2022) November

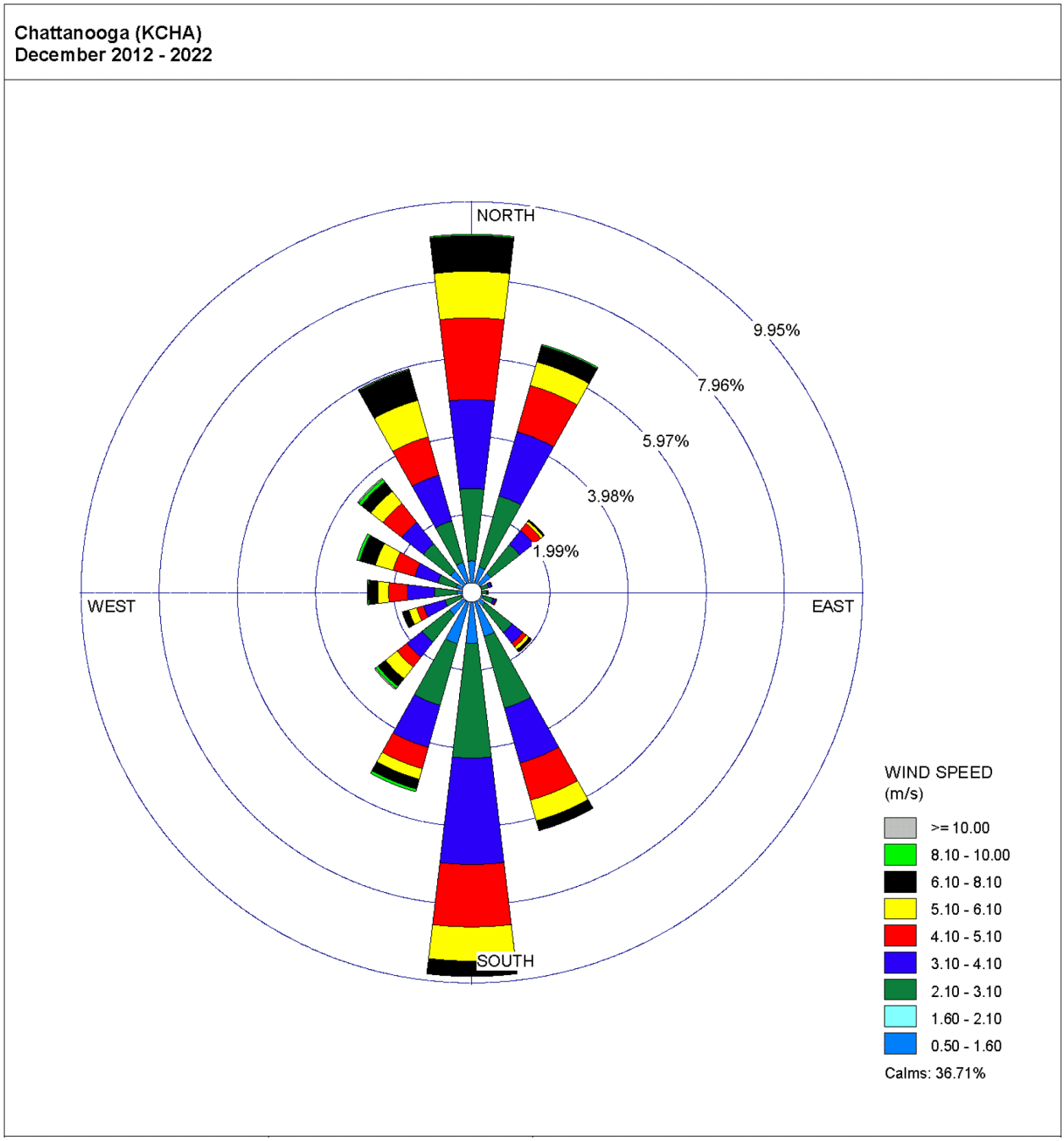

Figure 2.3.2-41R Wind Rose Chattanooga 10 Years (2012–2022) December

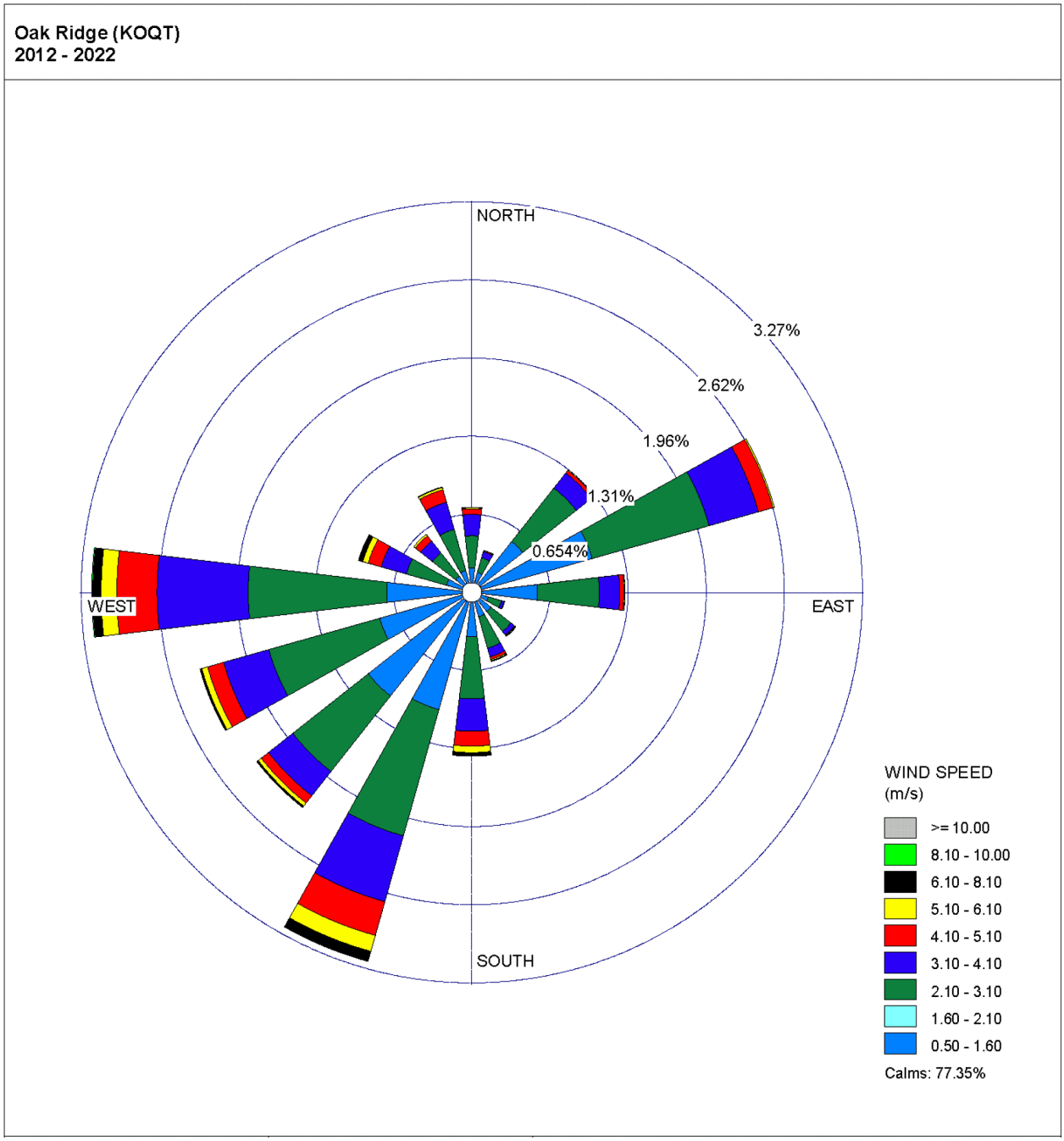

Figure 2.3.2-42R Wind Rose Oak Ridge 10 Years (2012–2022) All Data

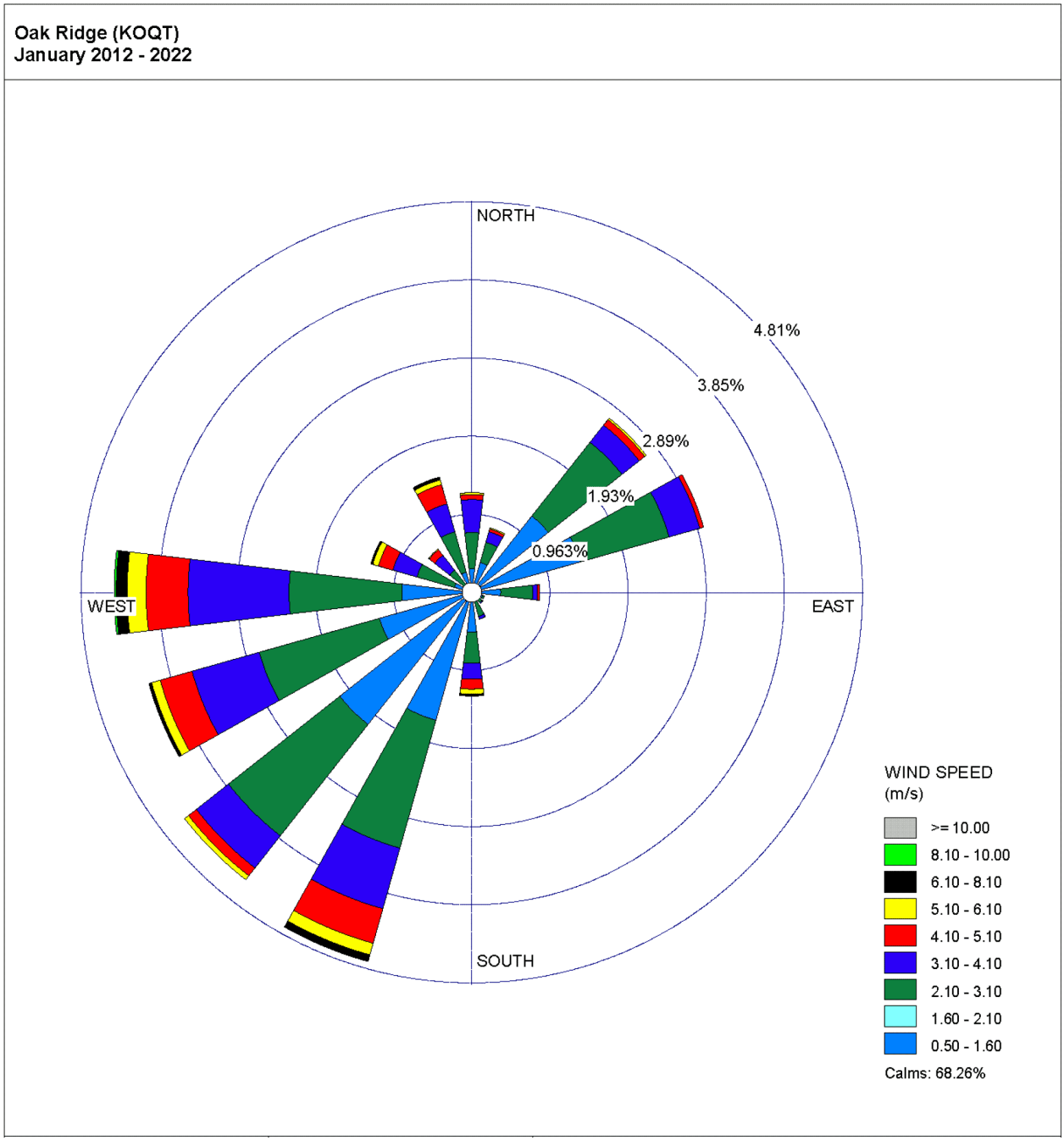

Figure 2.3.2-43R Wind Rose Oak Ridge 10 Years (2012–2022) January

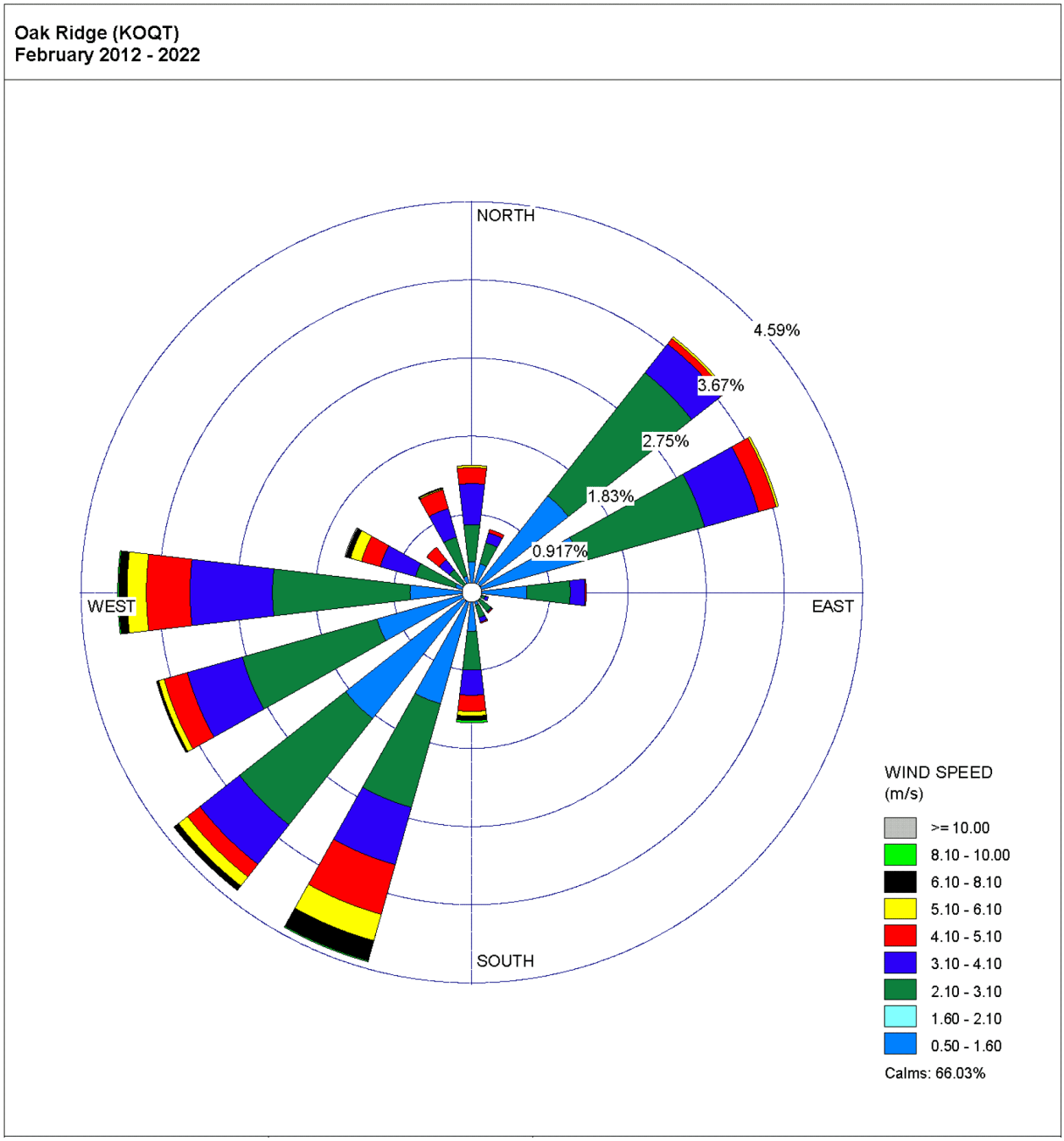

Figure 2.3.2-44R Wind Rose Oak Ridge 10 Years (2012–2022) February

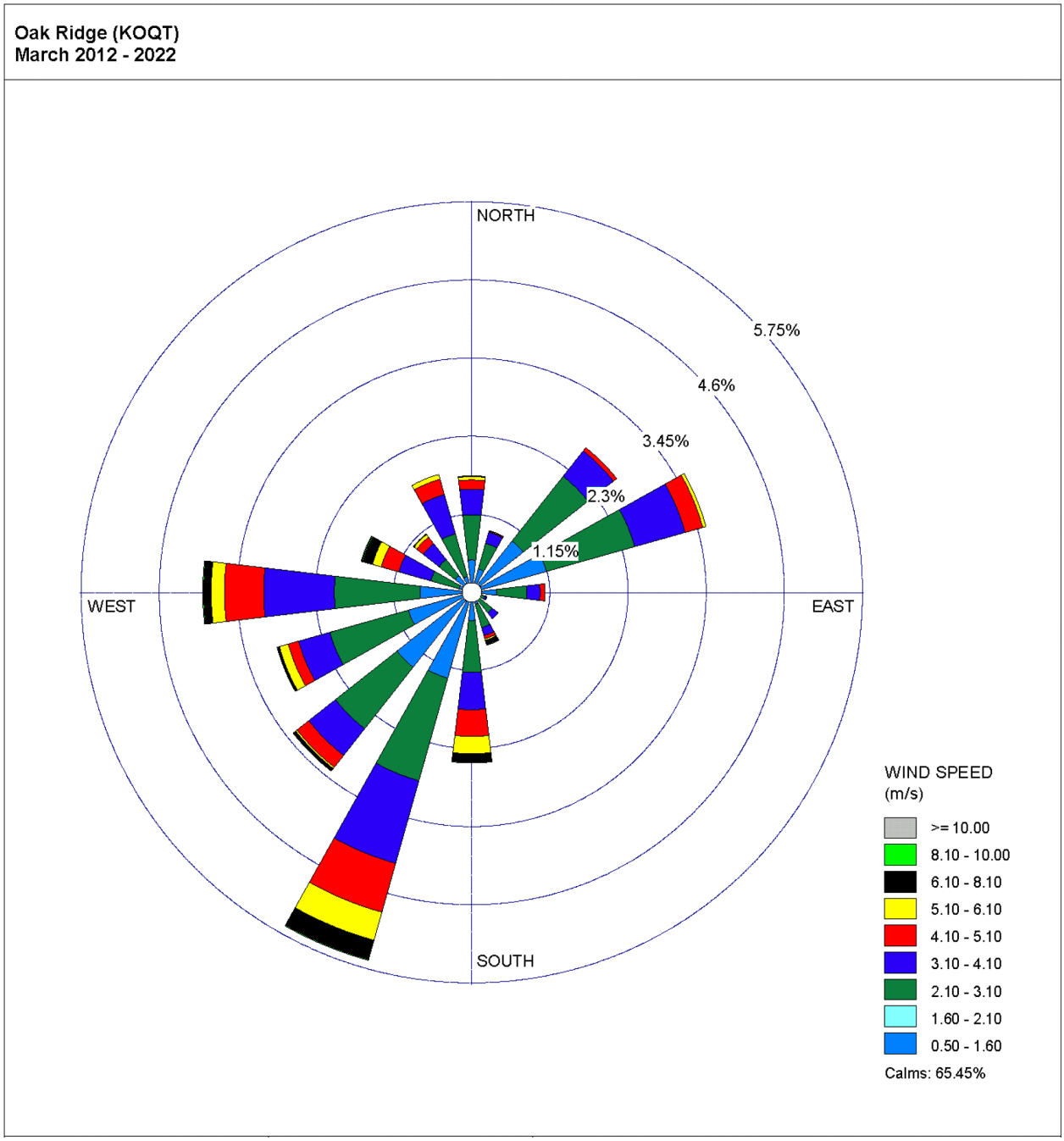

Figure 2.3.2-45R Wind Rose Oak Ridge 10 Years (2012–2022) March

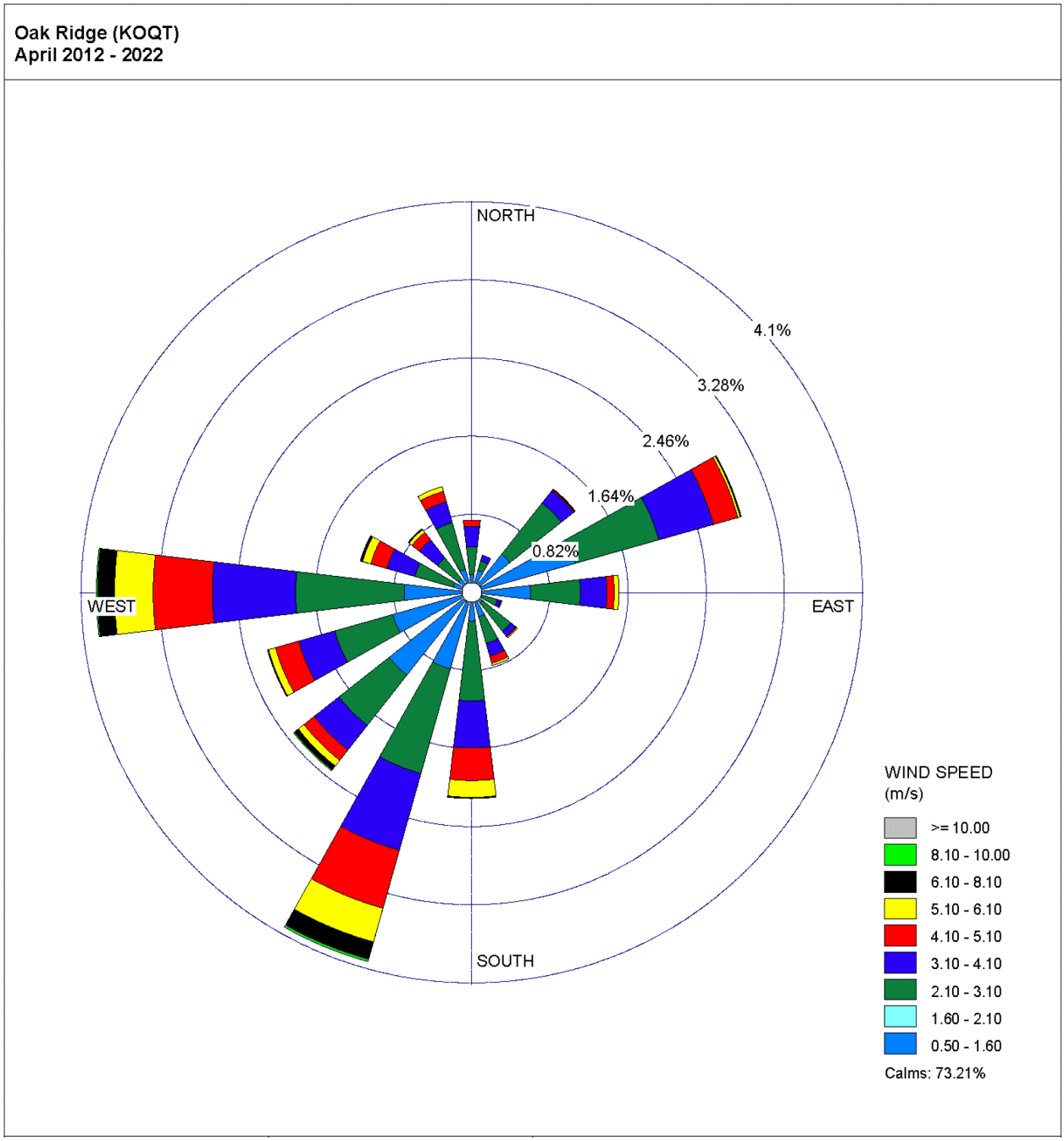

Figure 2.3.2-46R Wind Rose Oak Ridge 10 Years (2012–2022) April

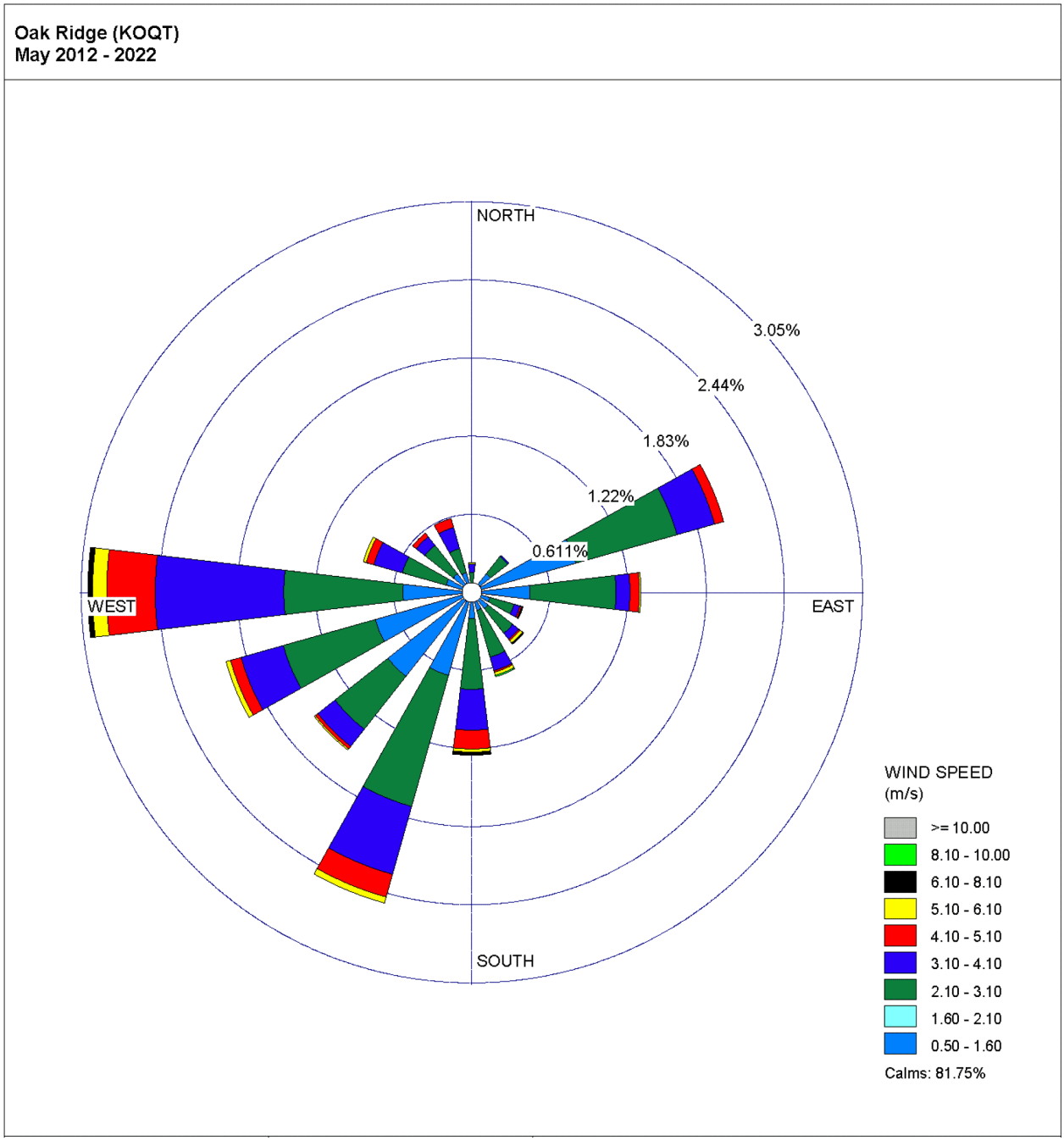

Figure 2.3.2-47R Wind Rose Oak Ridge 10 Years (2012–2022) May

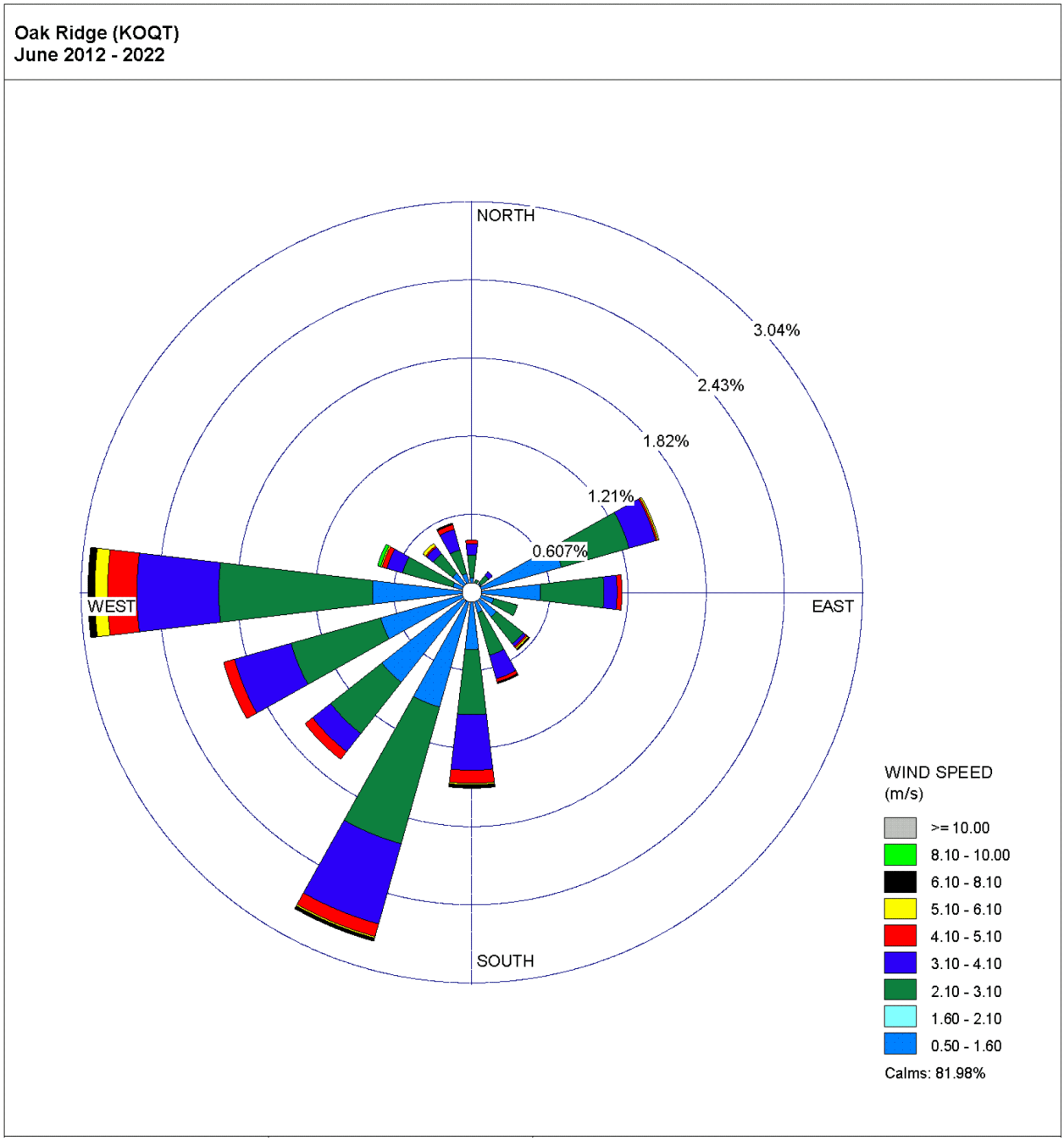

Figure 2.3.2-48R Wind Rose Oak Ridge 10 Years (2012–2022) June

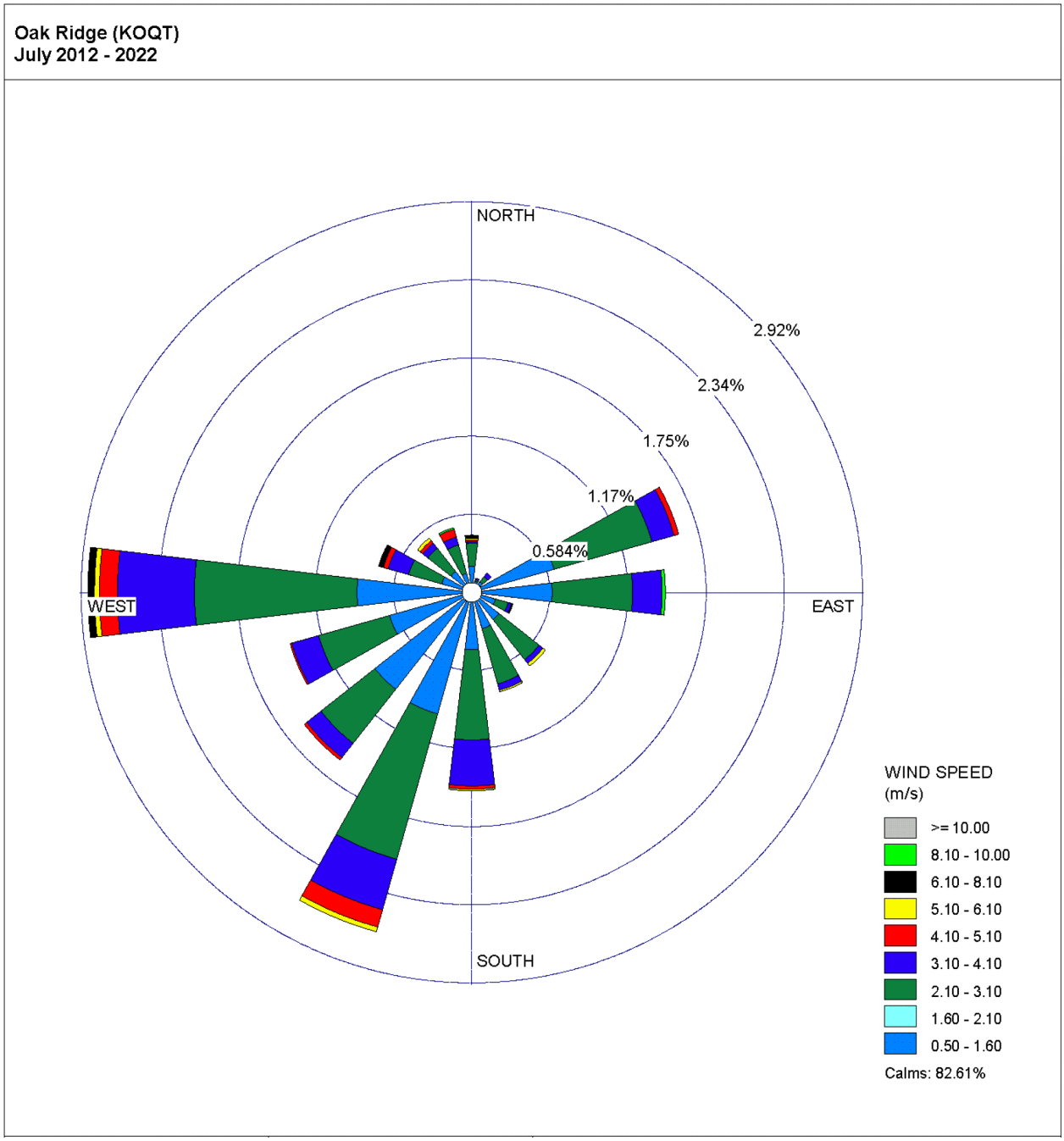

Figure 2.3.2-49R Wind Rose Oak Ridge 10 Years (2012–2022) July

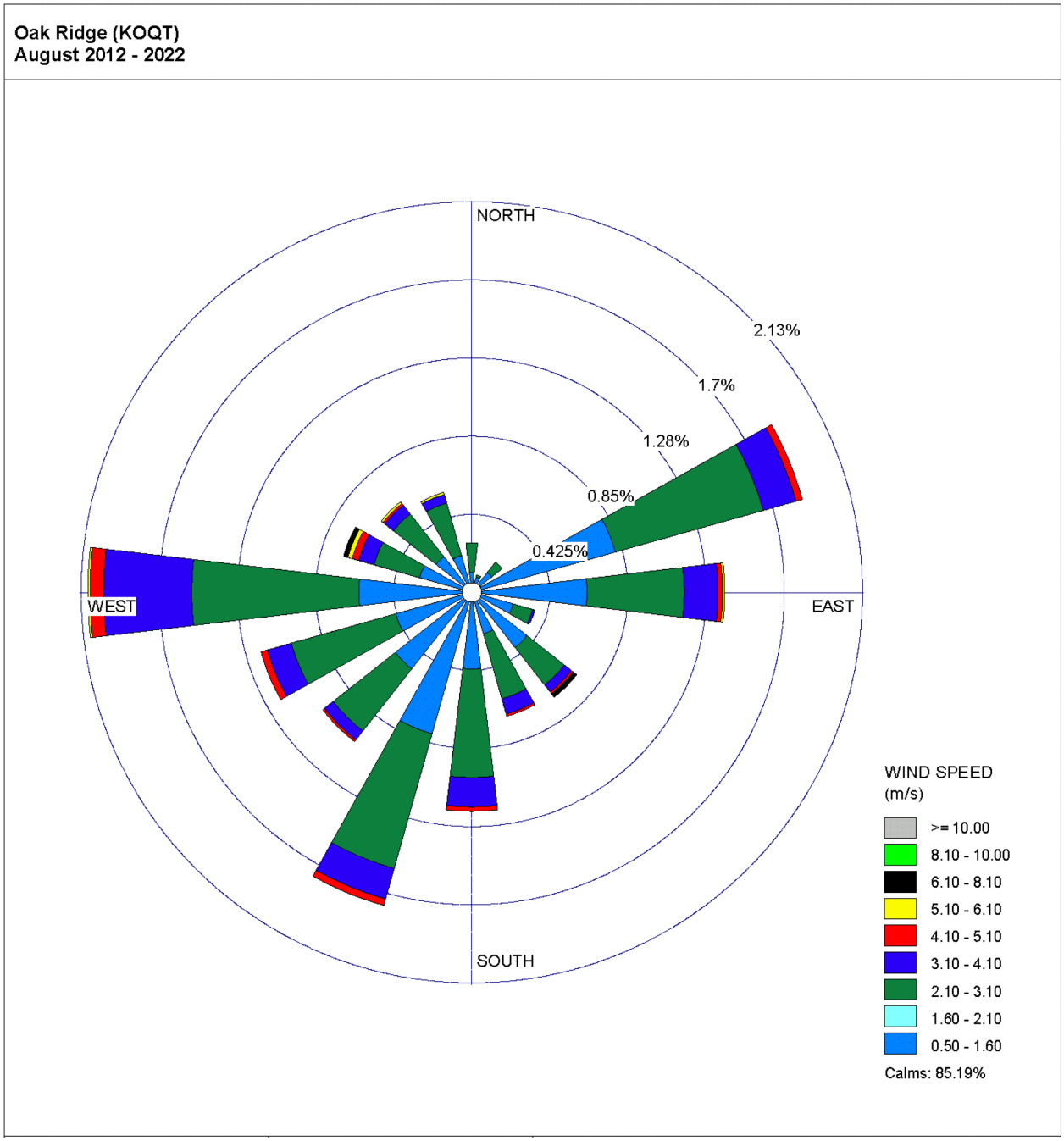

Figure 2.3.2-50R Wind Rose Oak Ridge 10 Years (2012–2022) August

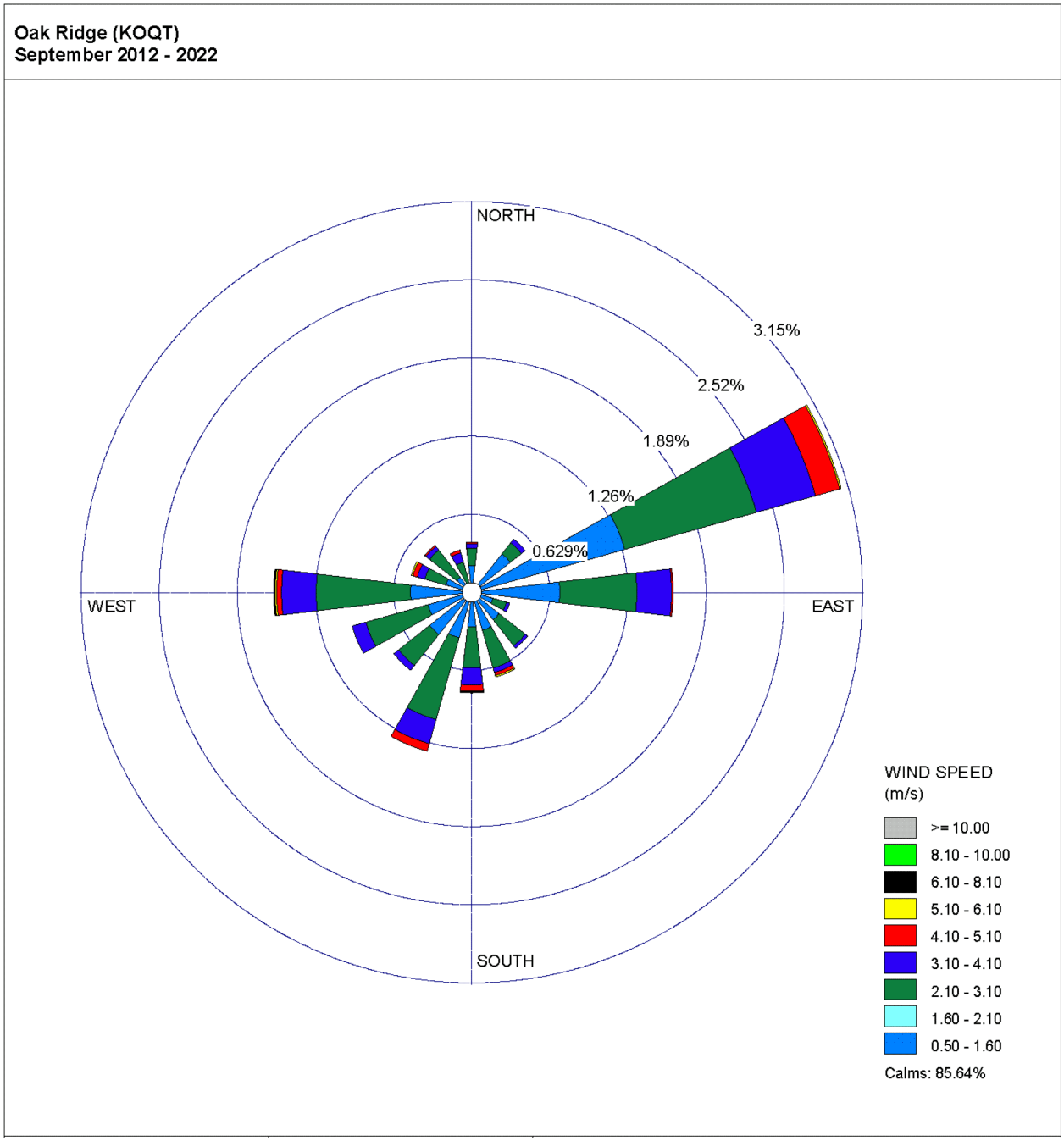

Figure 2.3.2-51R Wind Rose Oak Ridge 10 Years (2012–2022) September

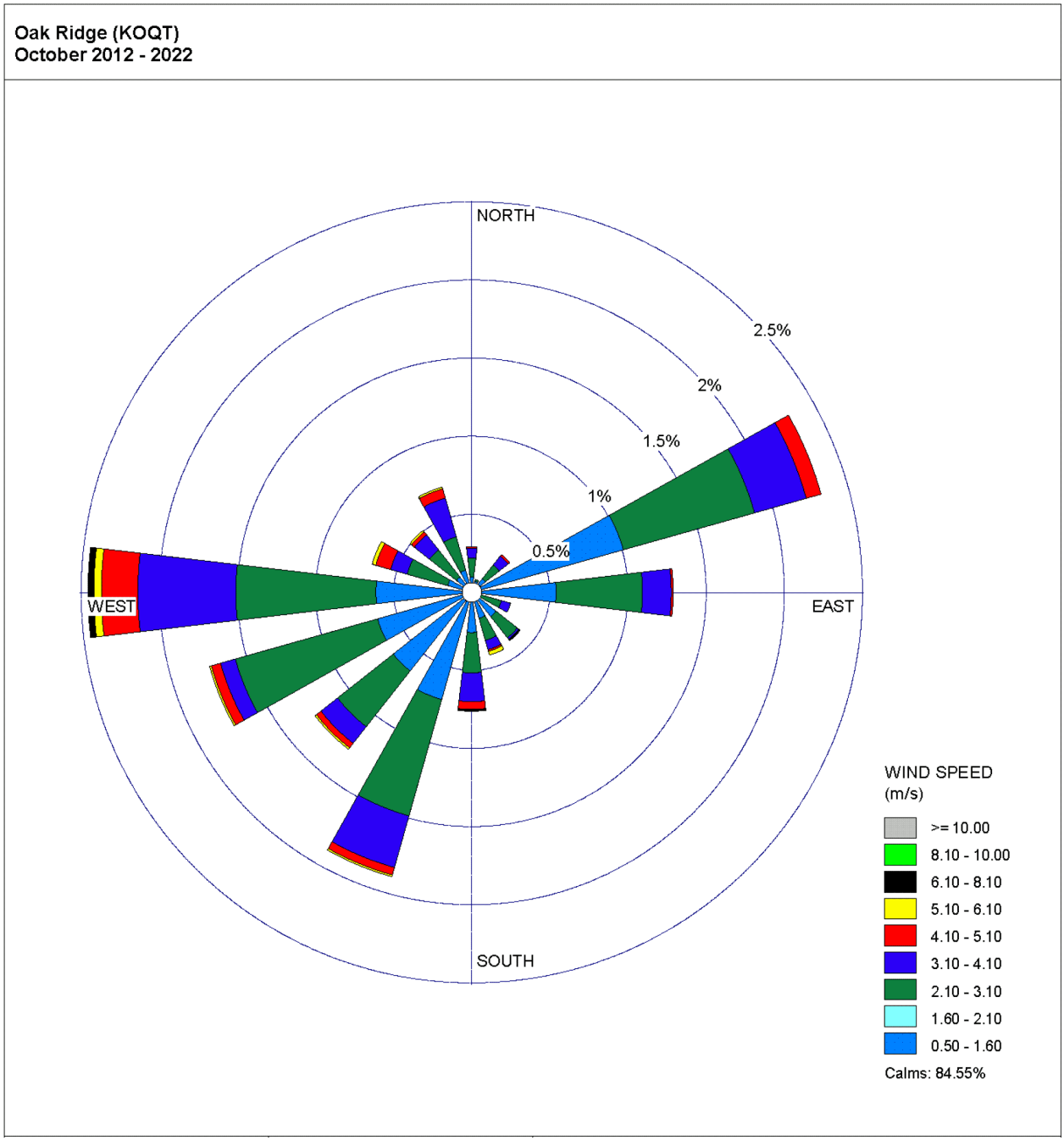

Figure 2.3.2-52R Wind Rose Oak Ridge 10 Years (2012–2022) October

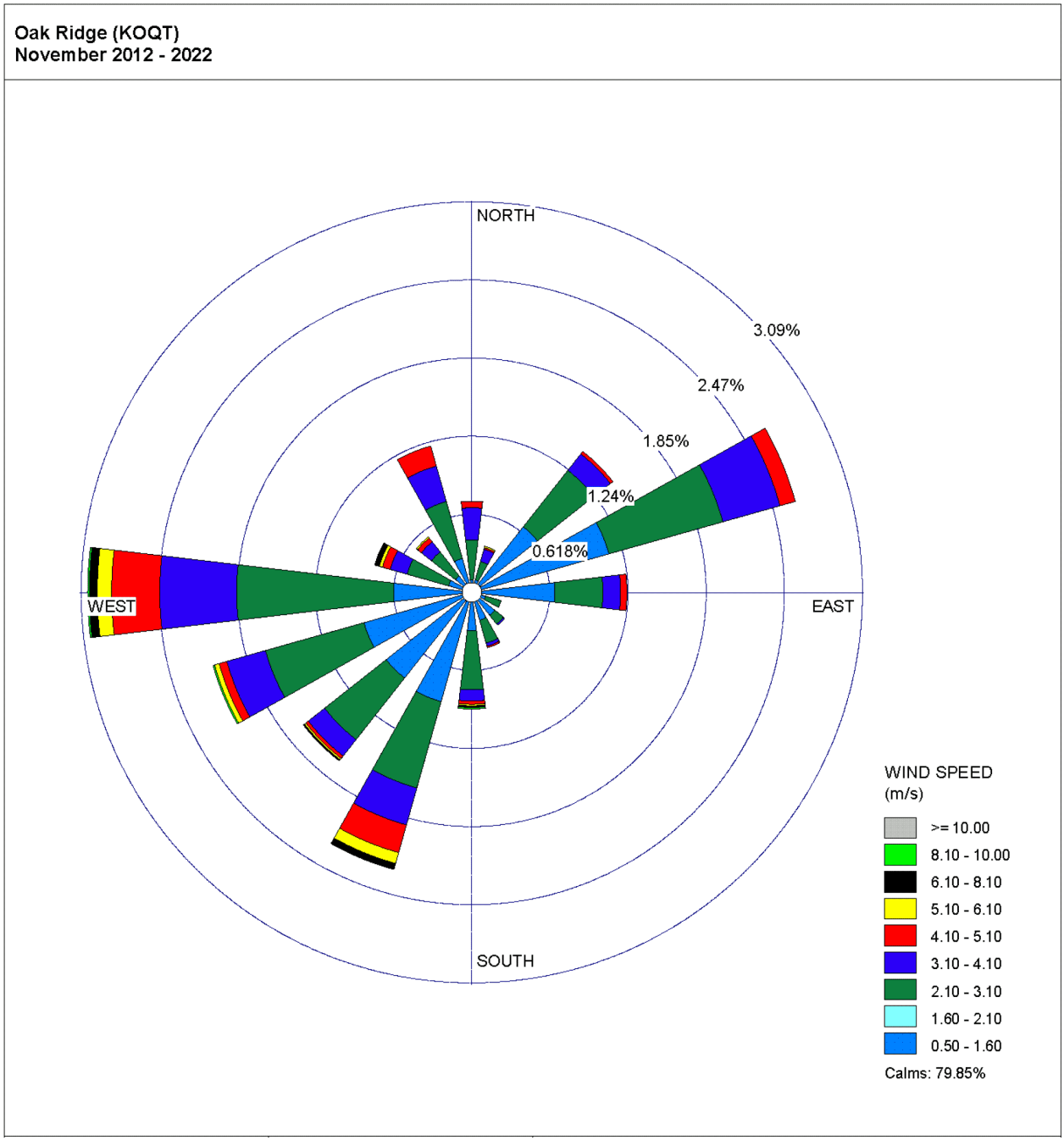

Figure 2.3.2-53R Wind Rose Oak Ridge 10 Years (2012–2022) November

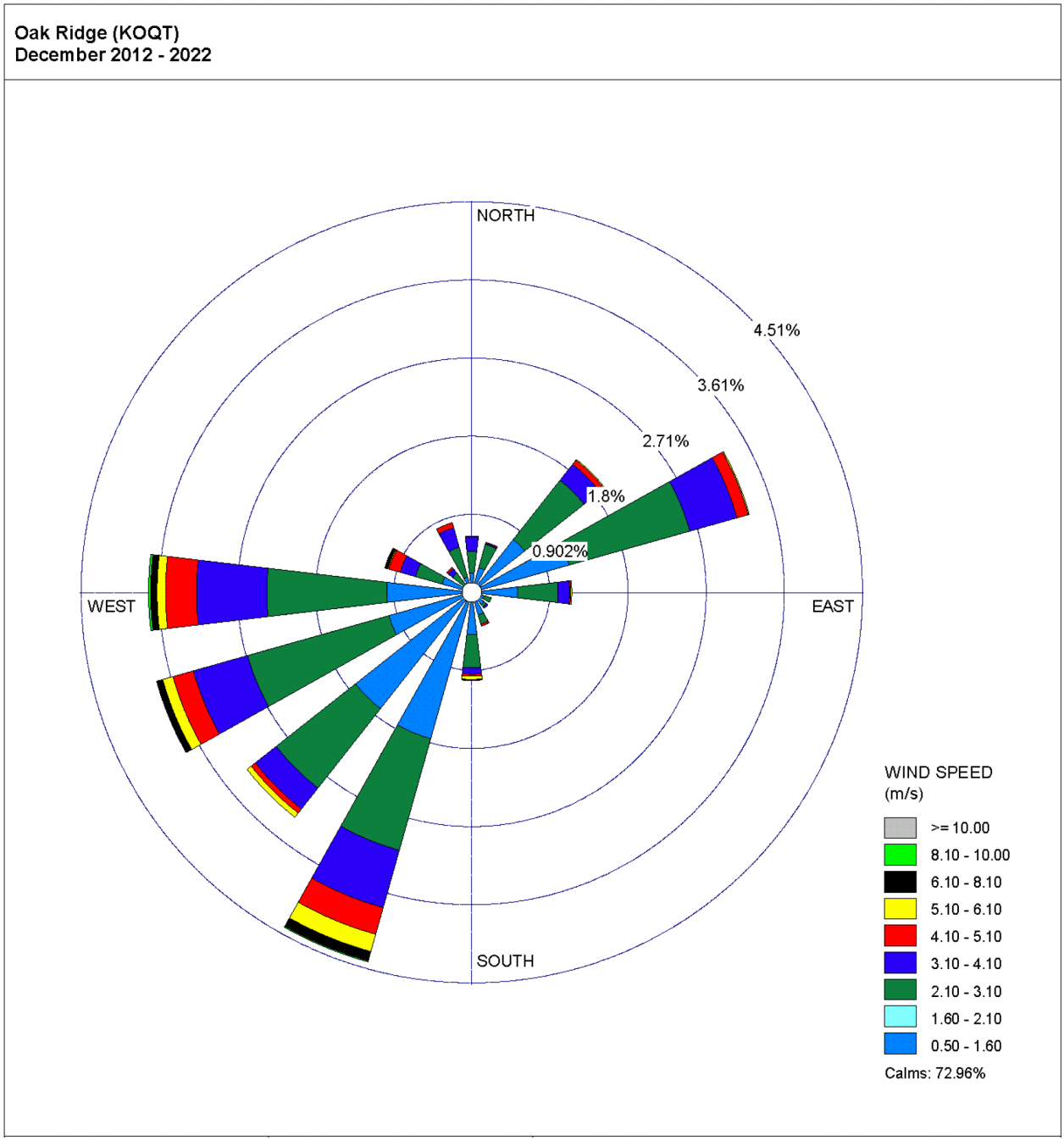

Figure 2.3.2-54R Wind Rose Oak Ridge 10 Years (2012–2022) December

Renumber ESPA SSAR Figure 2.3.2-55 through Figure 2.3.2-64 to Figure 2.3.2-68 through Figure 2.3.2-77.

Add new Figure 2.3.2-55R through Figure 2.3.2-67R with the following:

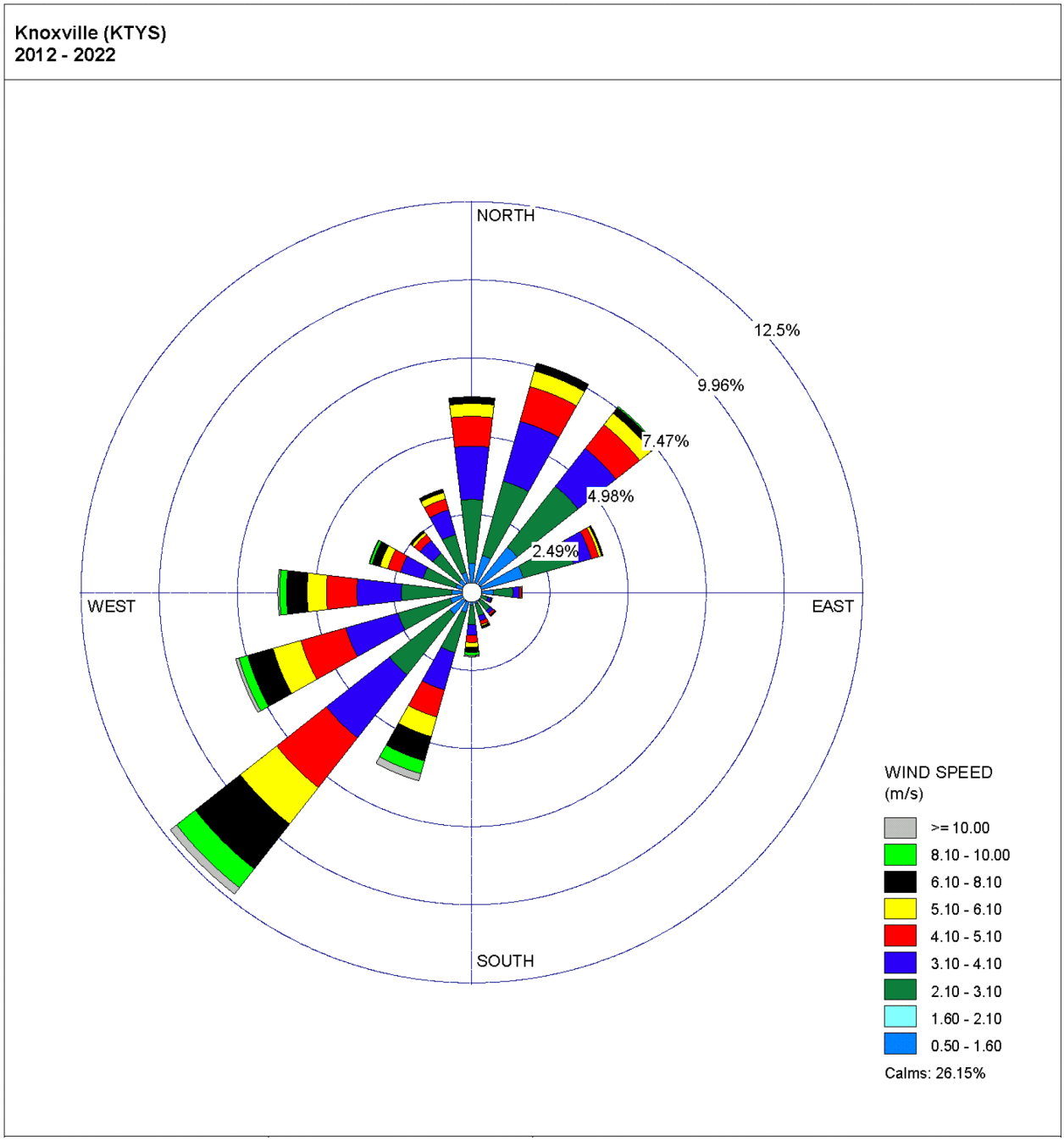

Figure 2.3.2-55R Wind Rose Knoxville 10 Years (2012–2022) All years

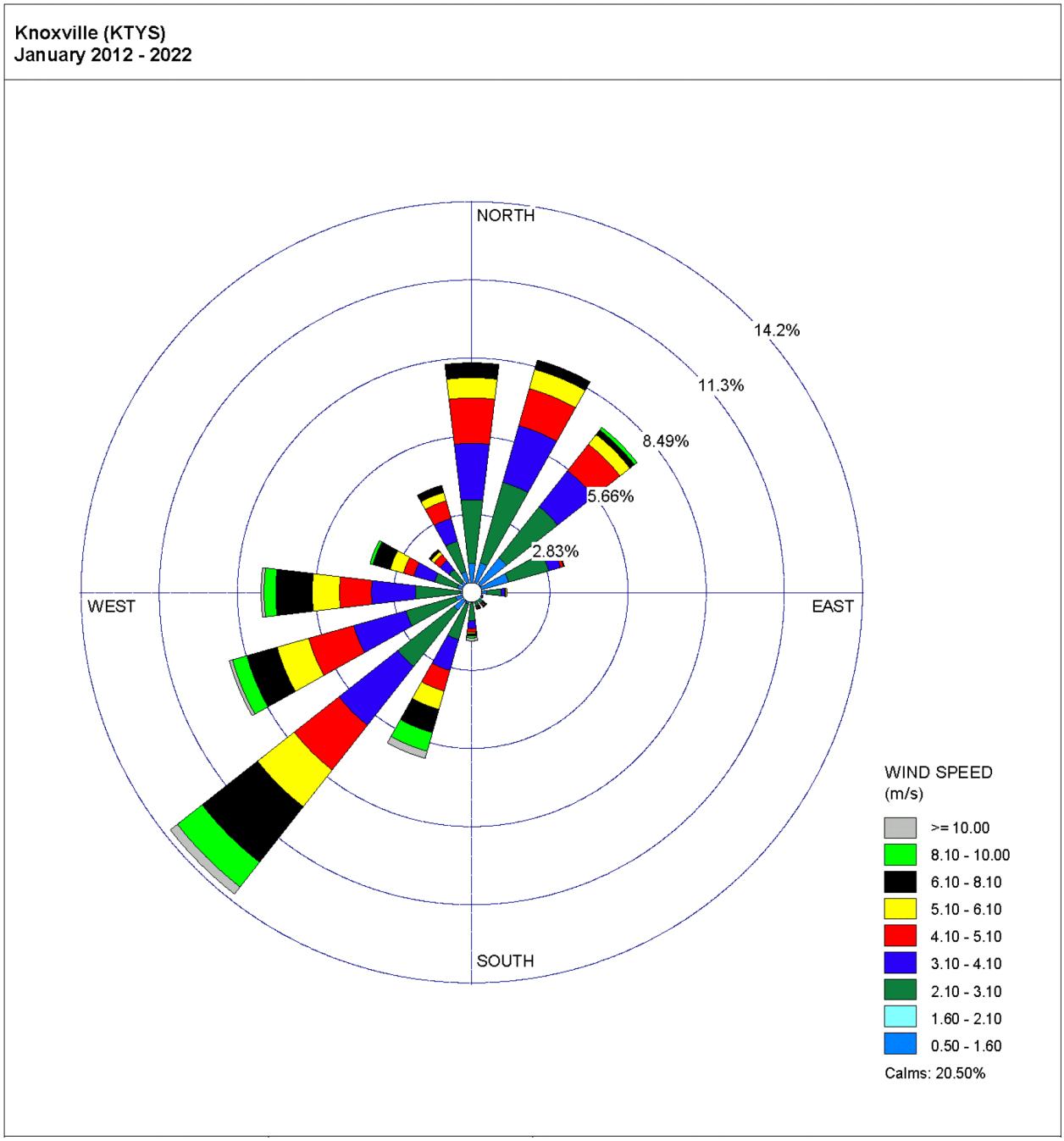

Figure 2.3.2-56R Wind Rose Knoxville 10 Years (2012–2022) January

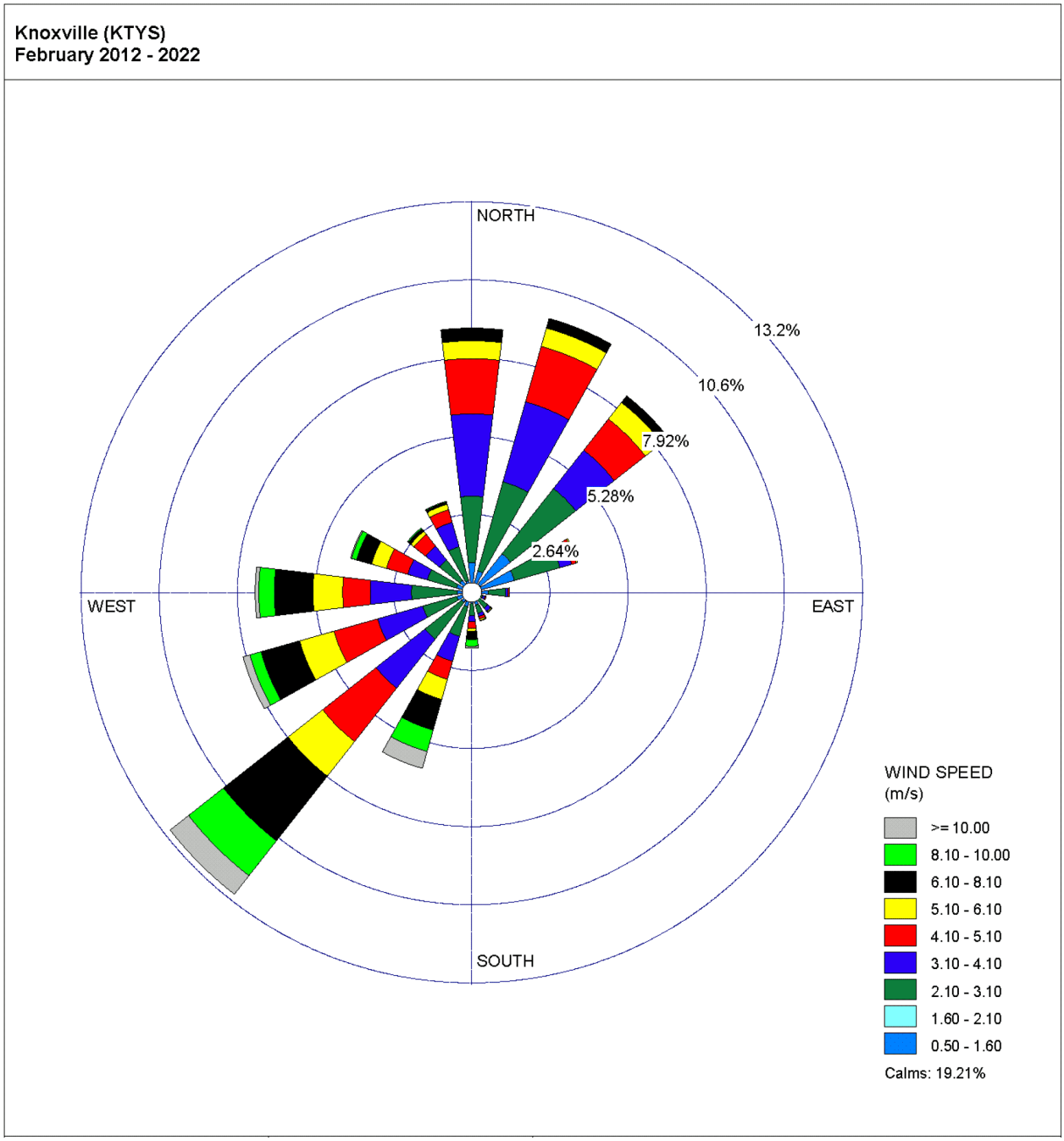

Figure 2.3.2-57R Wind Rose Knoxville 10 Years (2012–2022) February

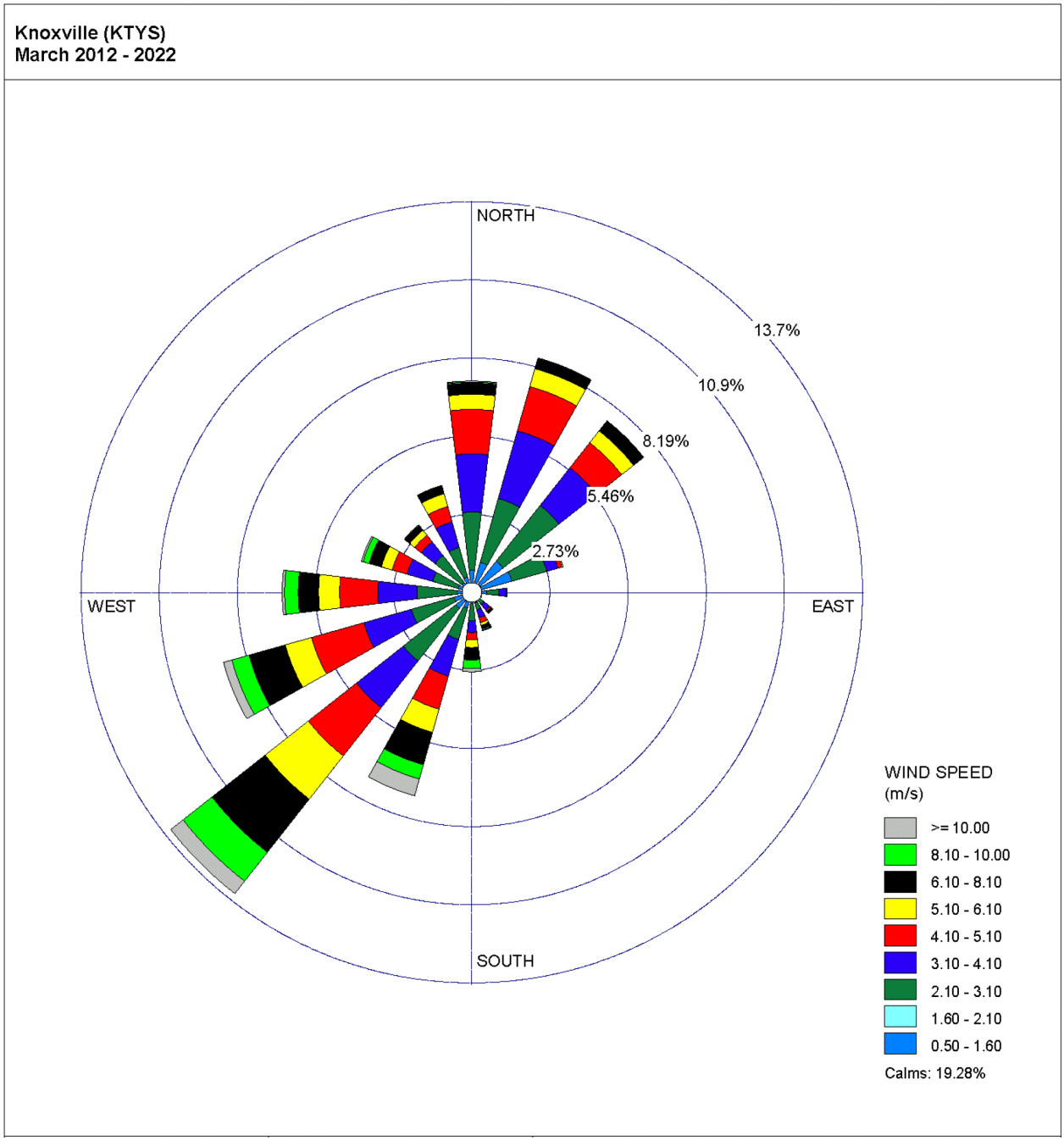

Figure 2.3.2-58R Wind Rose Knoxville 10 Years (2012–2022) March

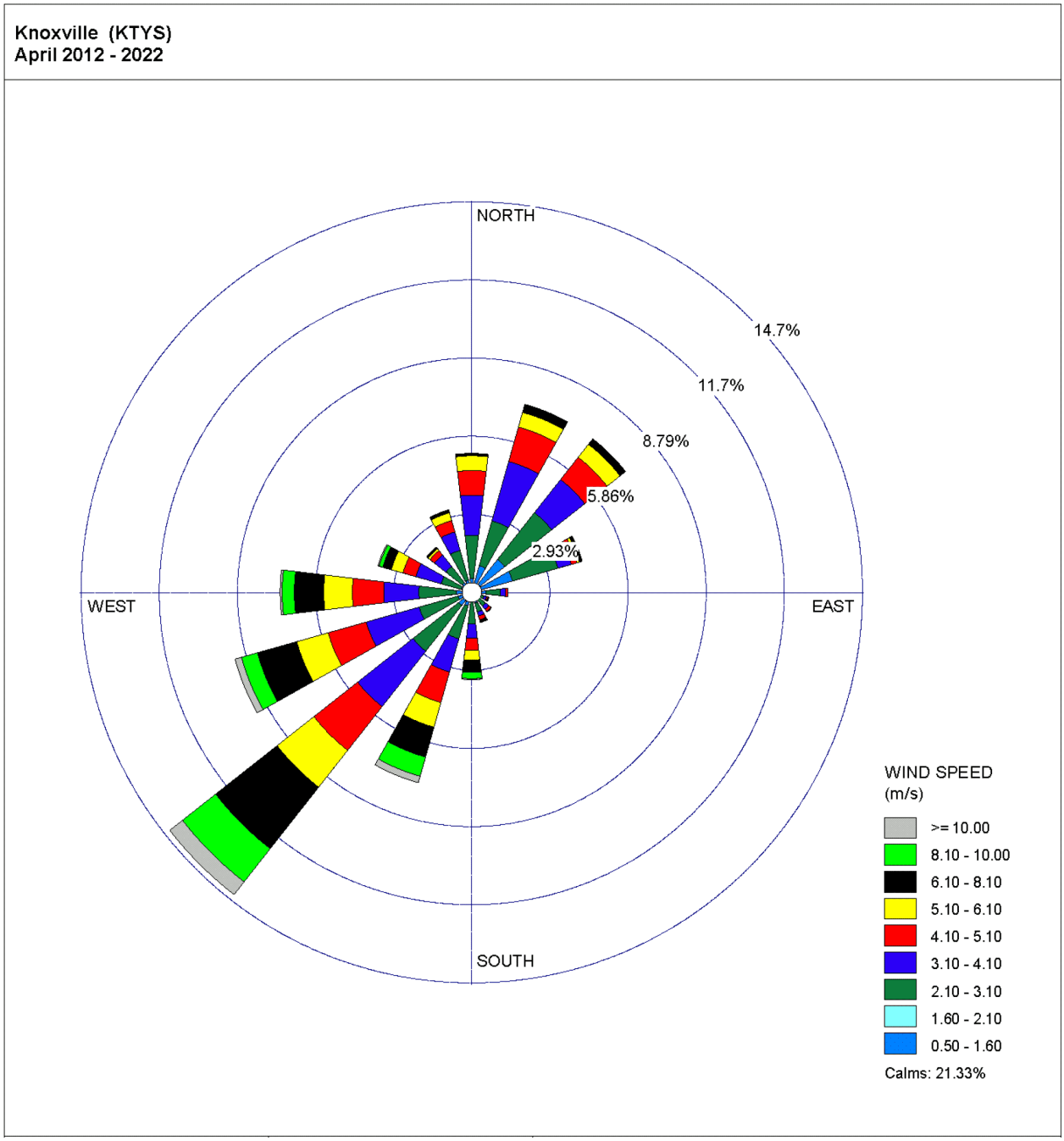

Figure 2.3.2-59R Wind Rose Knoxville 10 Years (2012–2022) April

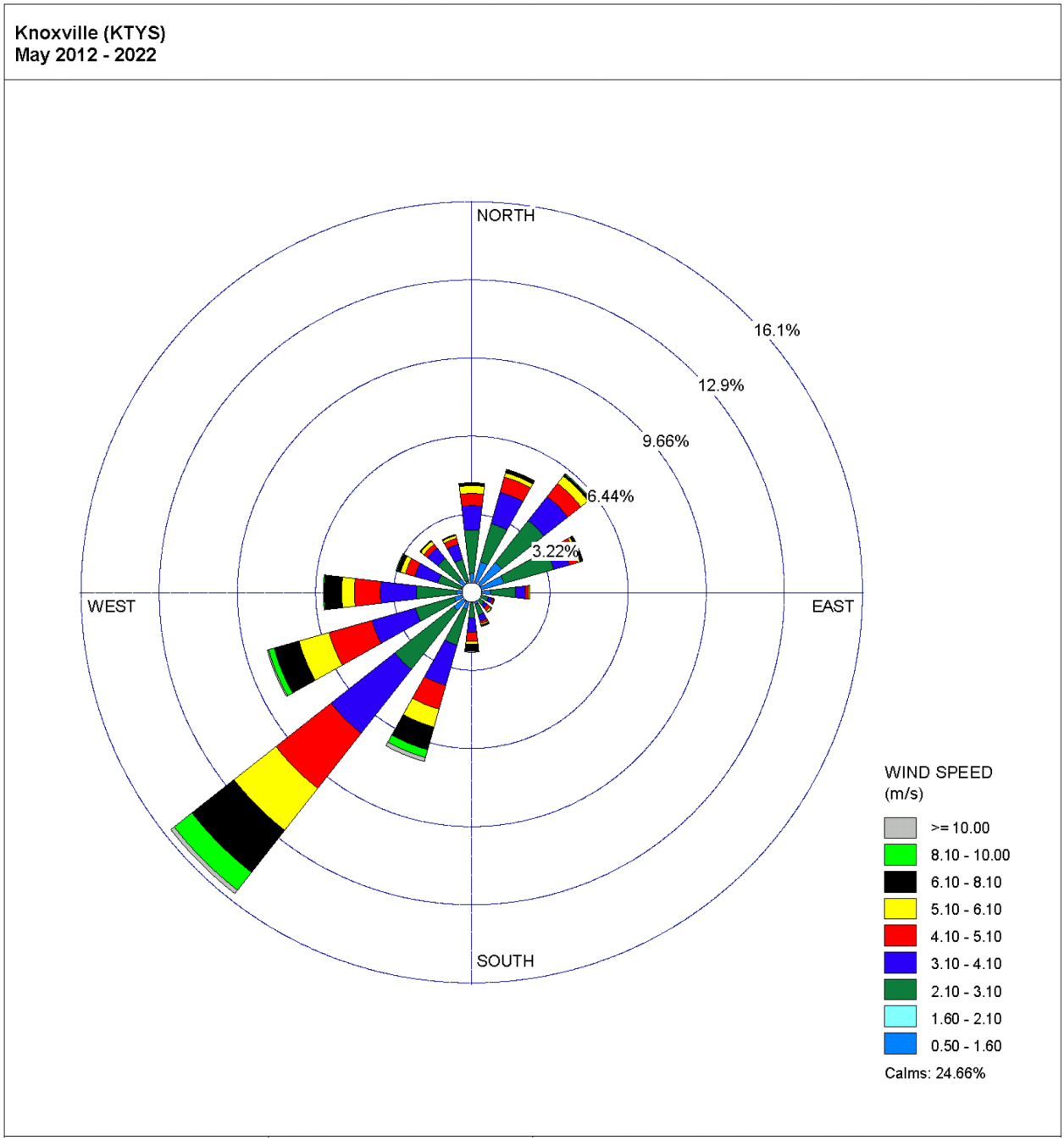

Figure 2.3.2-60R Wind Rose Knoxville 10 Years (2012–2022) May

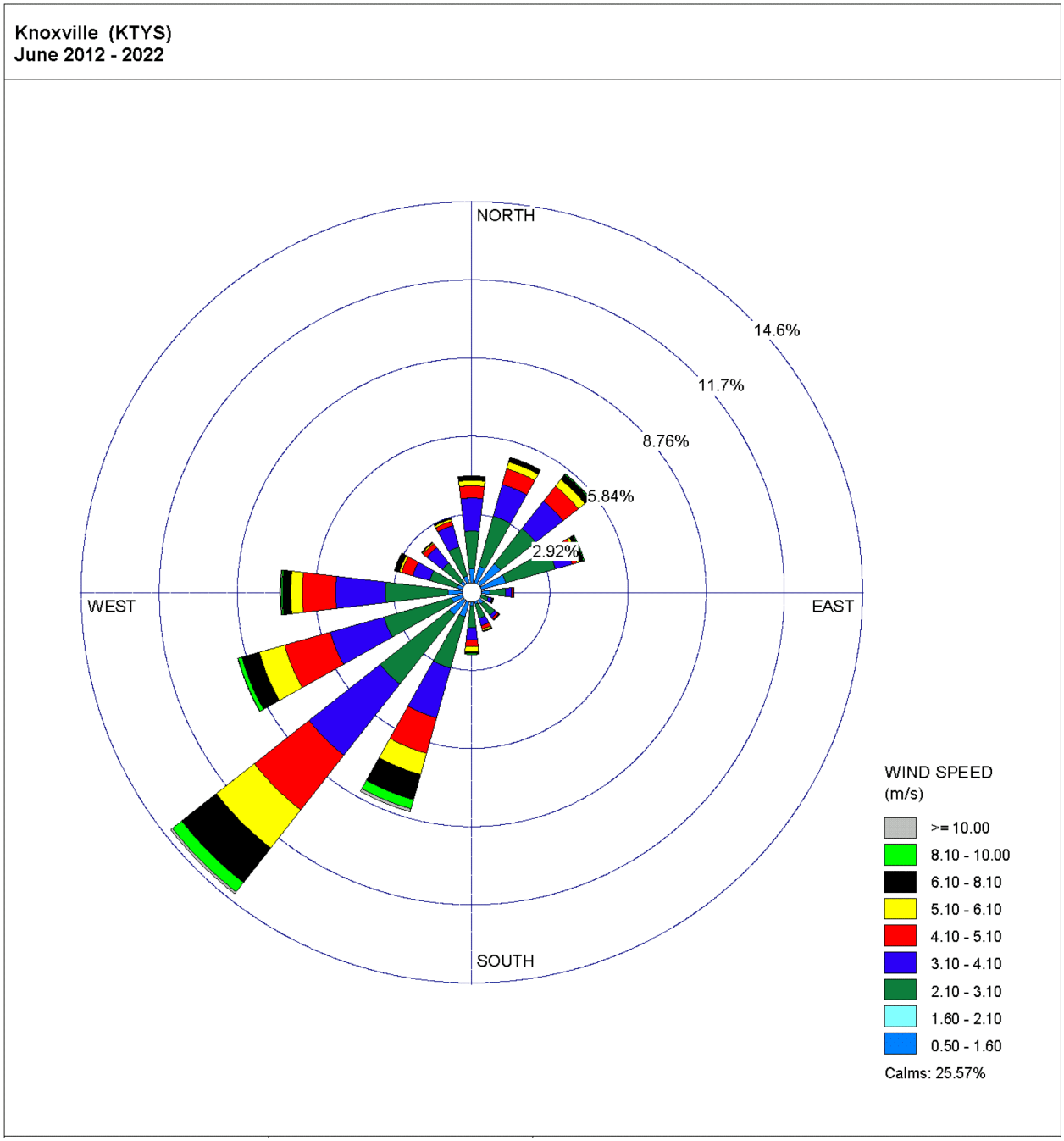

Figure 2.3.2-61R Wind Rose Knoxville 10 Years (2012–2022) June

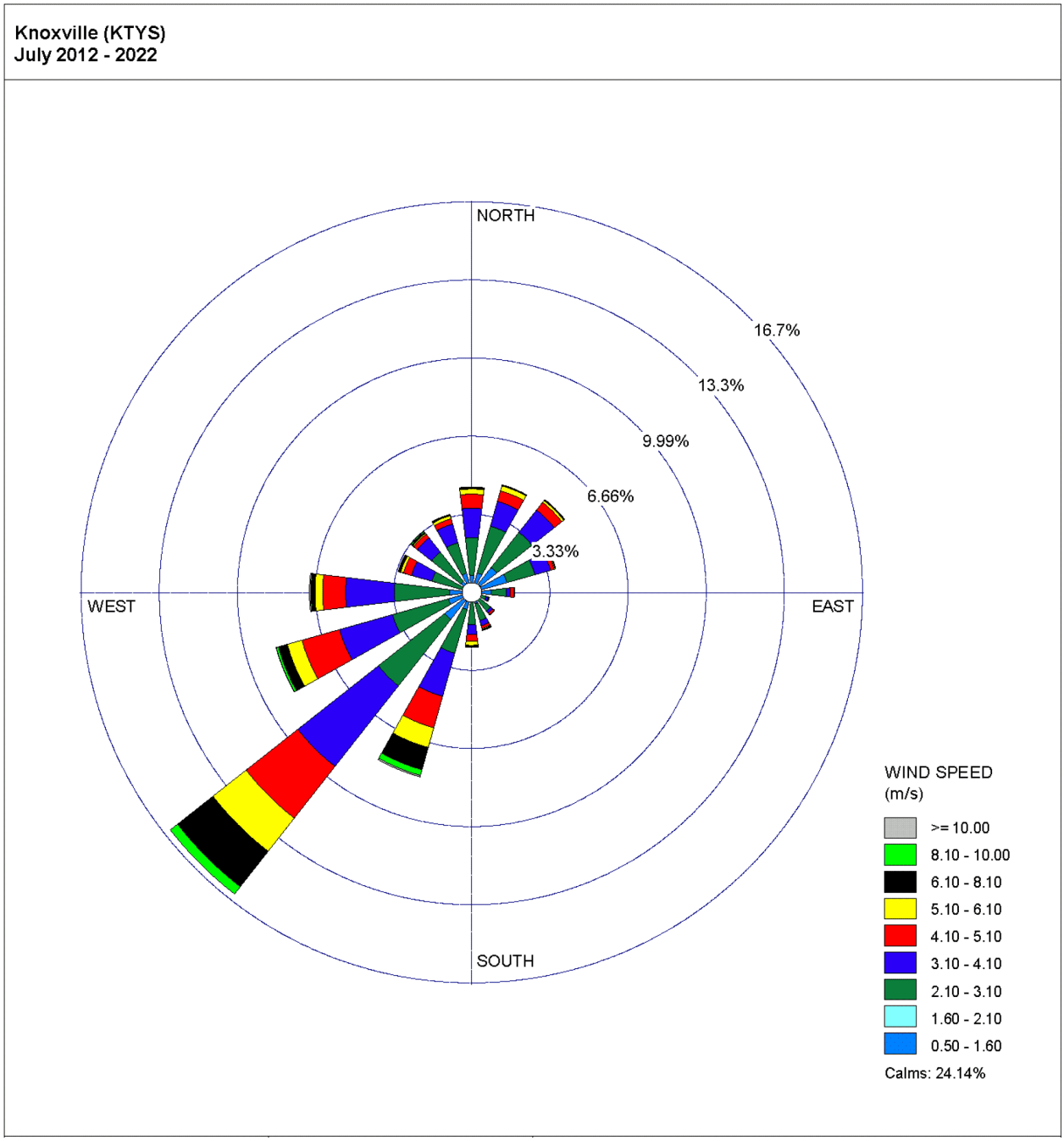

Figure 2.3.2-62R Wind Rose Knoxville 10 Years (2012–2022) July

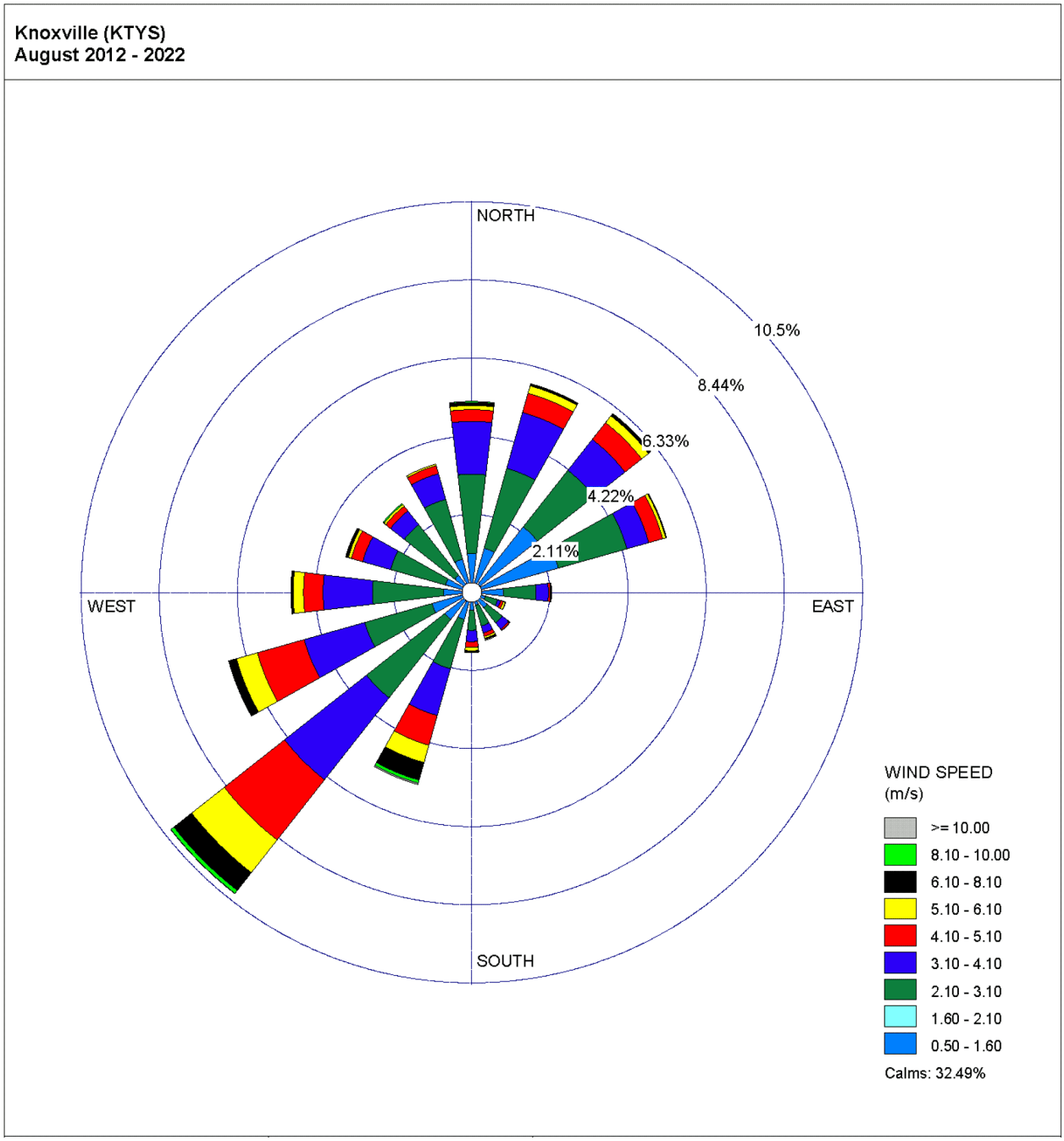

Figure 2.3.2-63R Wind Rose Knoxville 10 Years (2012–2022) August

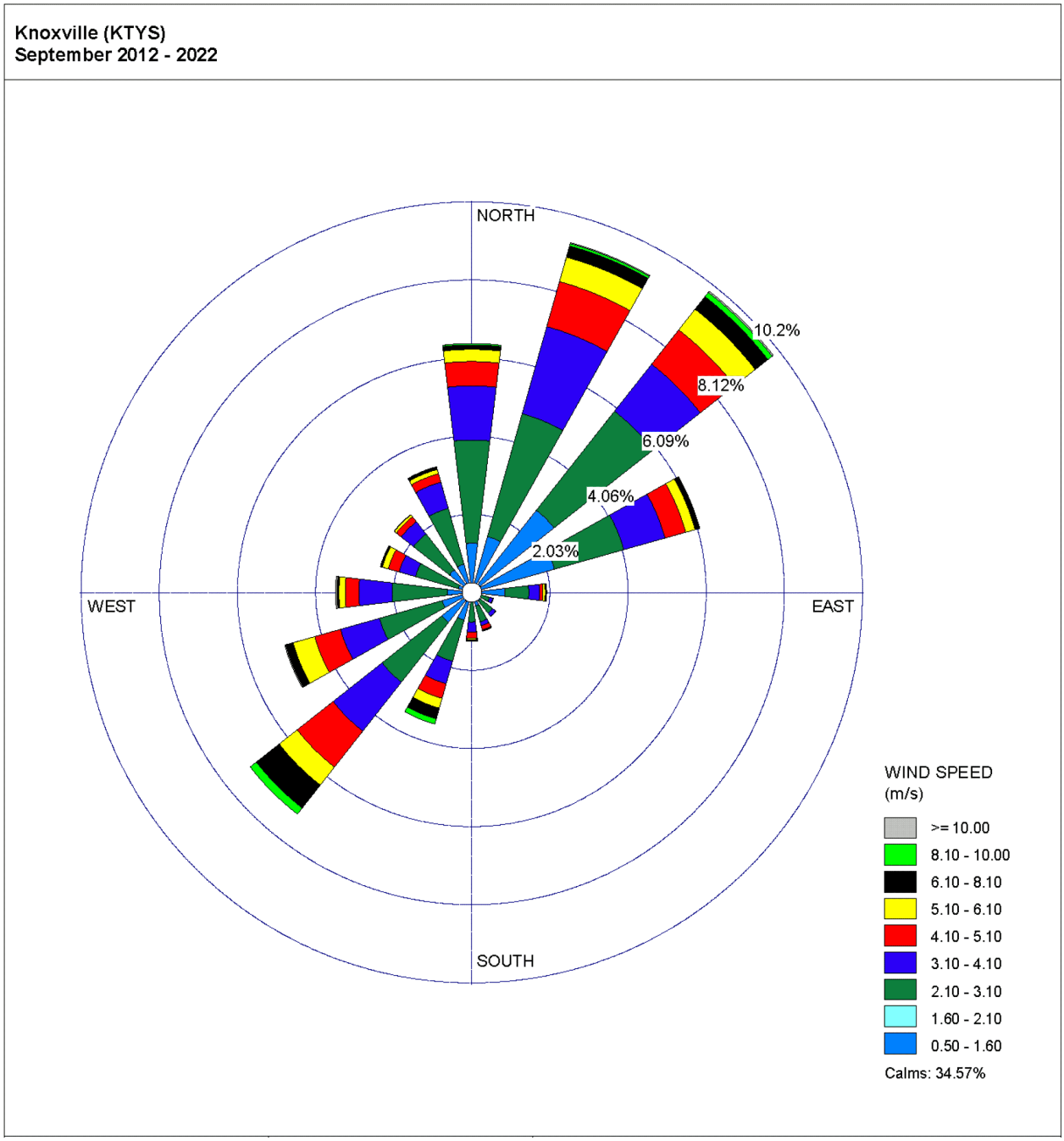

Figure 2.3.2-64R Wind Rose Knoxville 10 Years (2012–2022) September

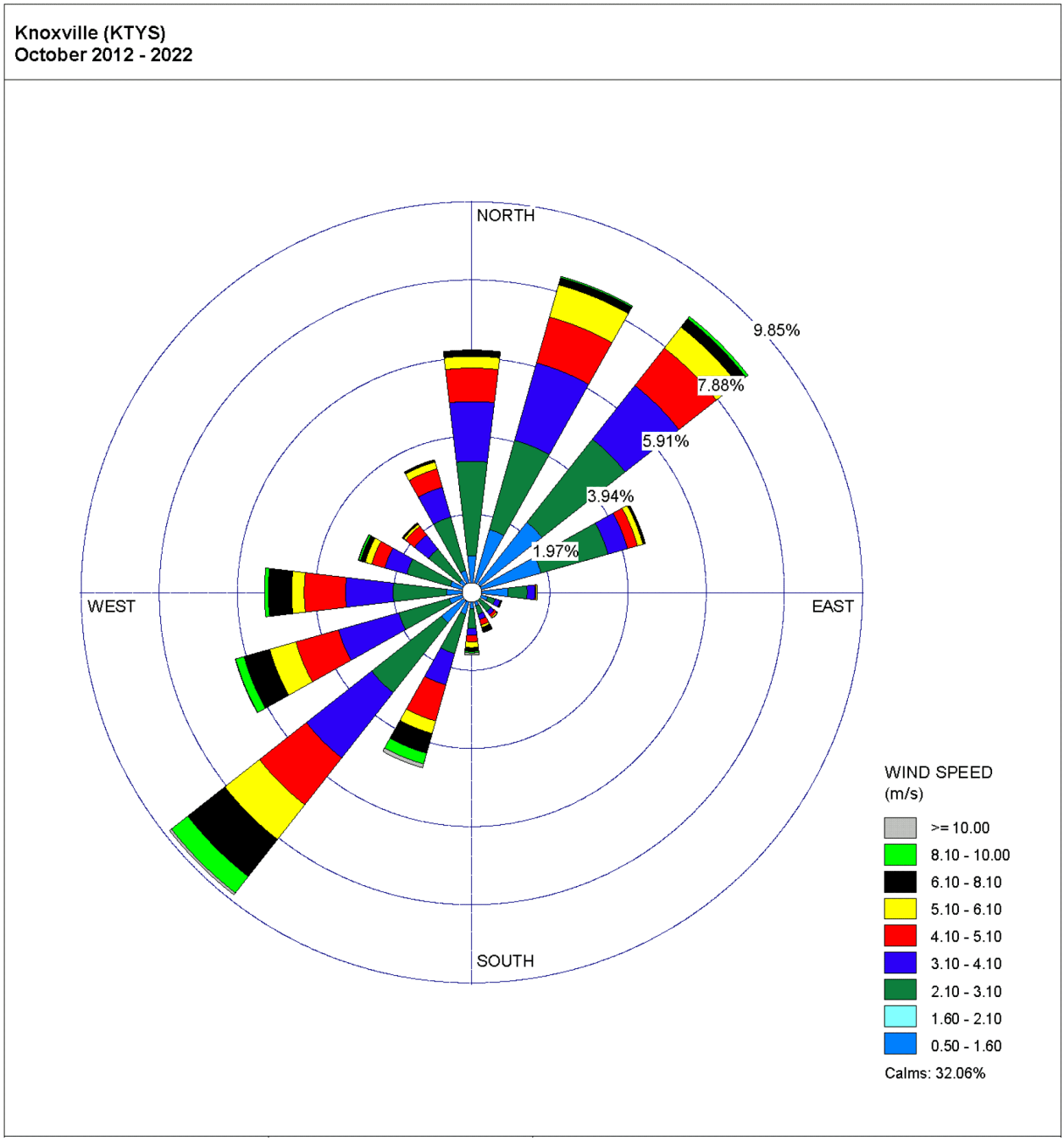

Figure 2.3.2-65R Wind Rose Knoxville 10 Years (2012–2022) October

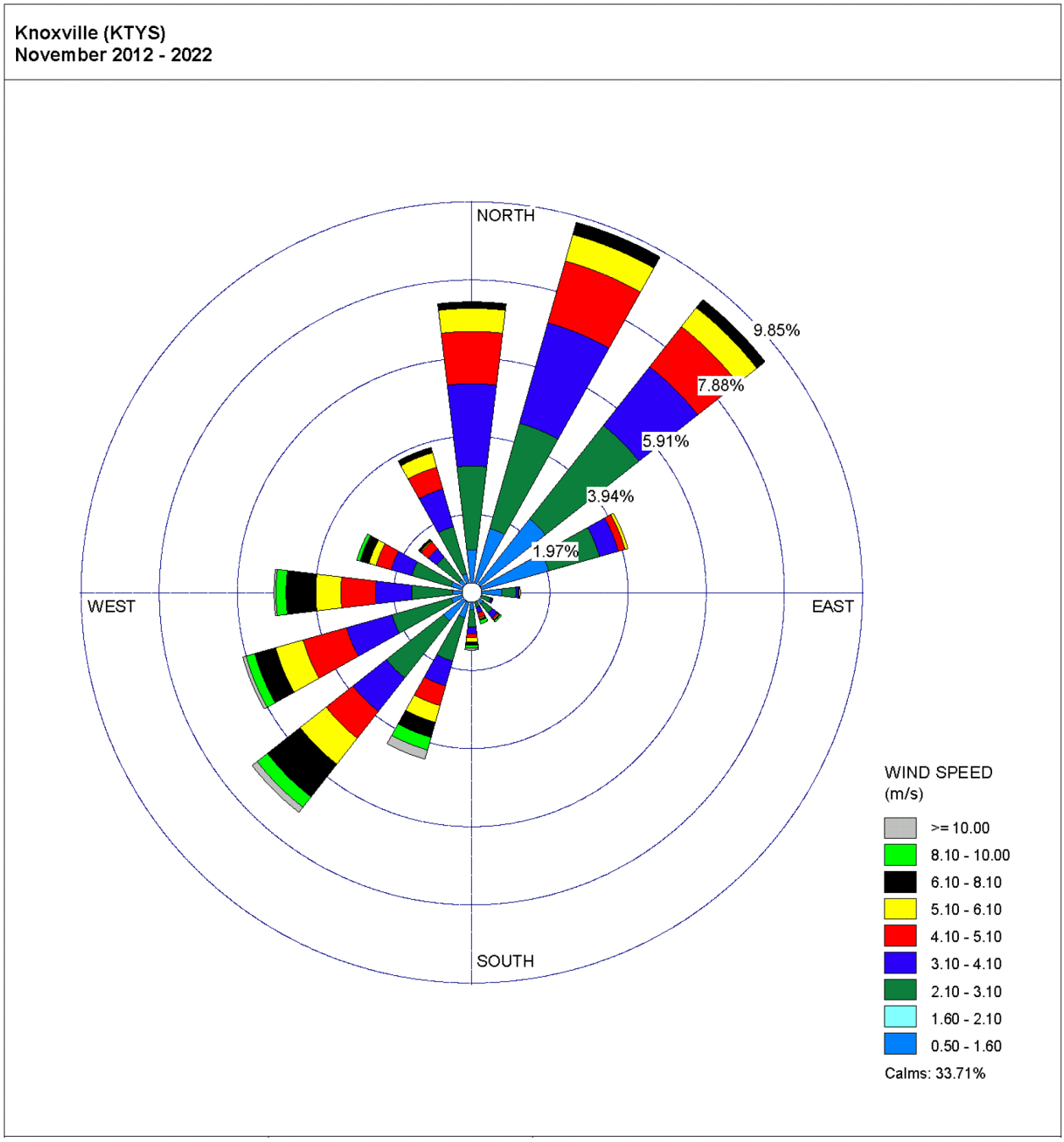

Figure 2.3.2-66R Wind Rose Knoxville 10 Years (2012–2022) November

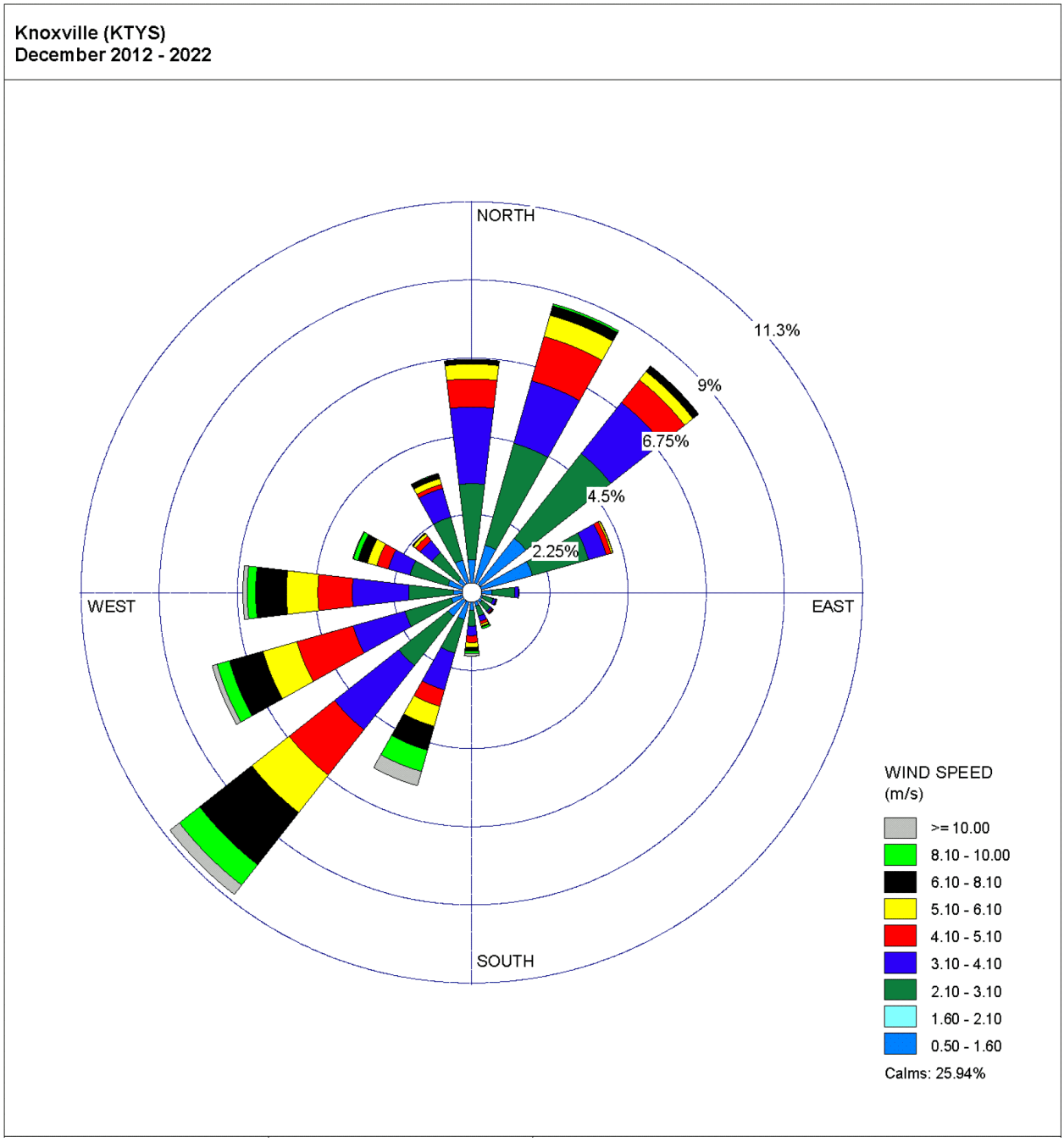

Figure 2.3.2-67R Wind Rose Knoxville 10 Years (2012–2022) December

Replace Figures 2.3.2-65 through 2.3.2-70 with Figure 2.3.2-78R through Figure 2.3.2-82R:

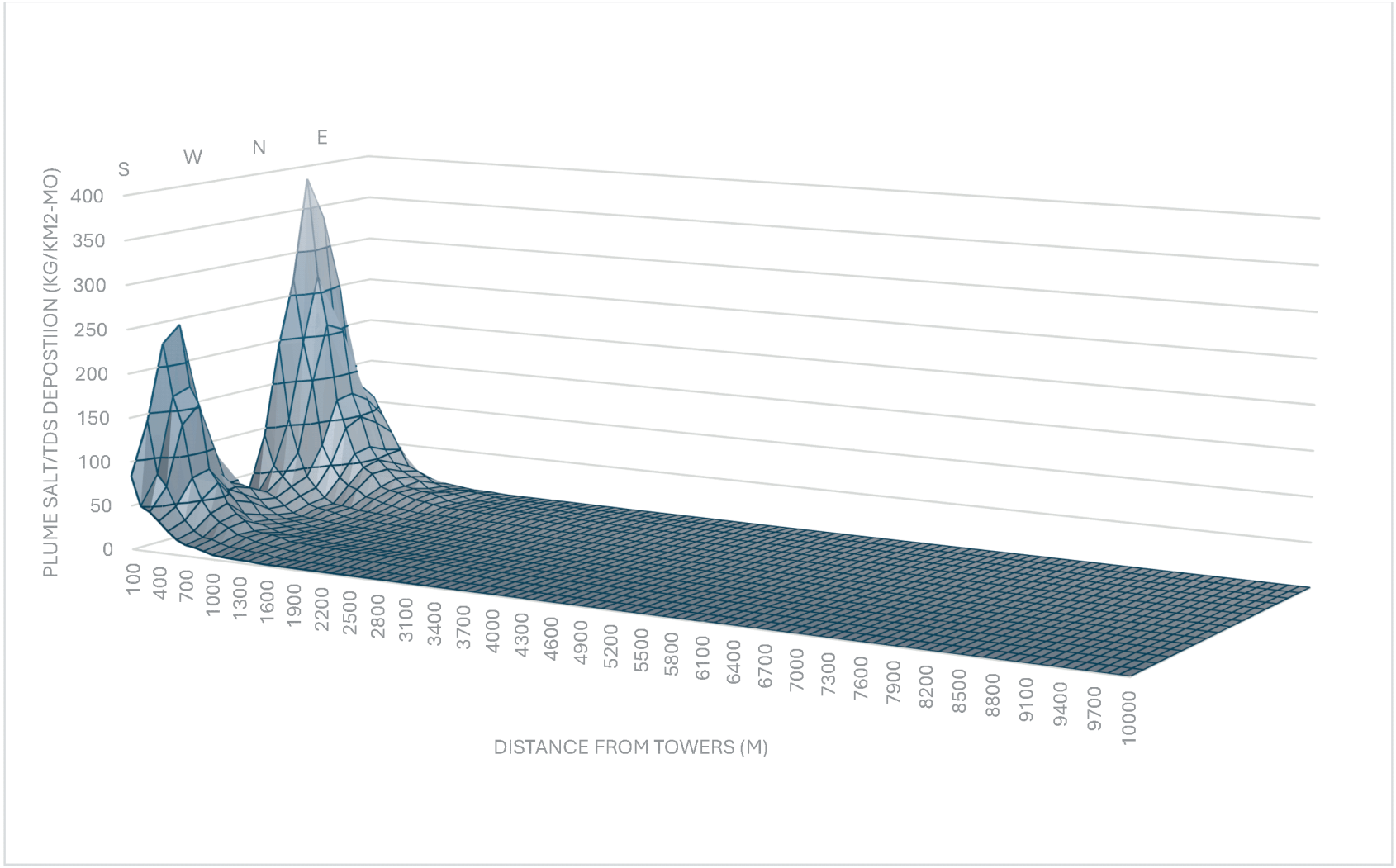

Figure 2.3.2-78R Annual Salt/TDS Deposition – Single Tower (CRN Tower) in kg/km^2/month

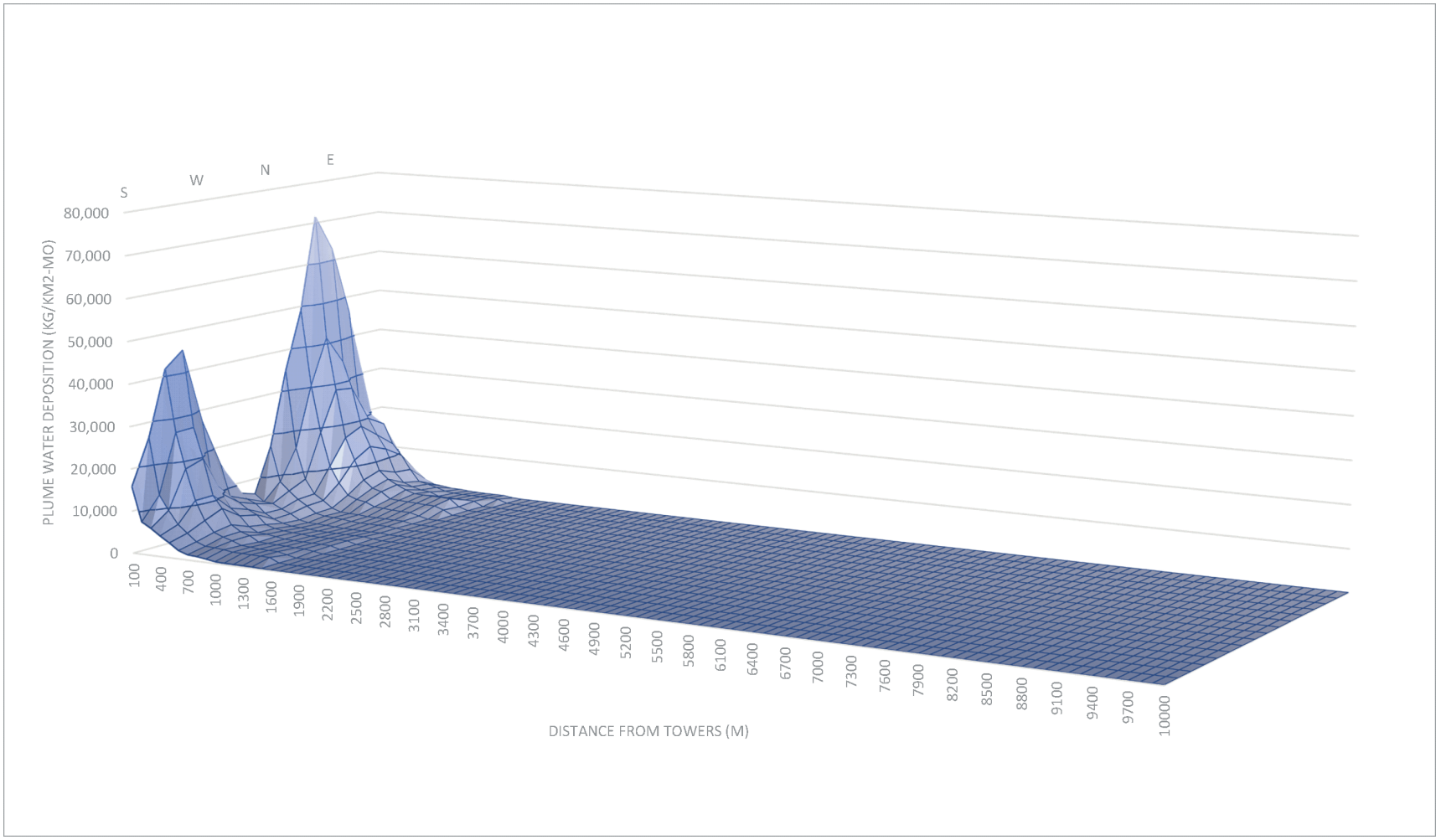

Figure 2.3.2-79R Annual Water Deposition – Single Tower (CRN Tower) in kg/km^2/month

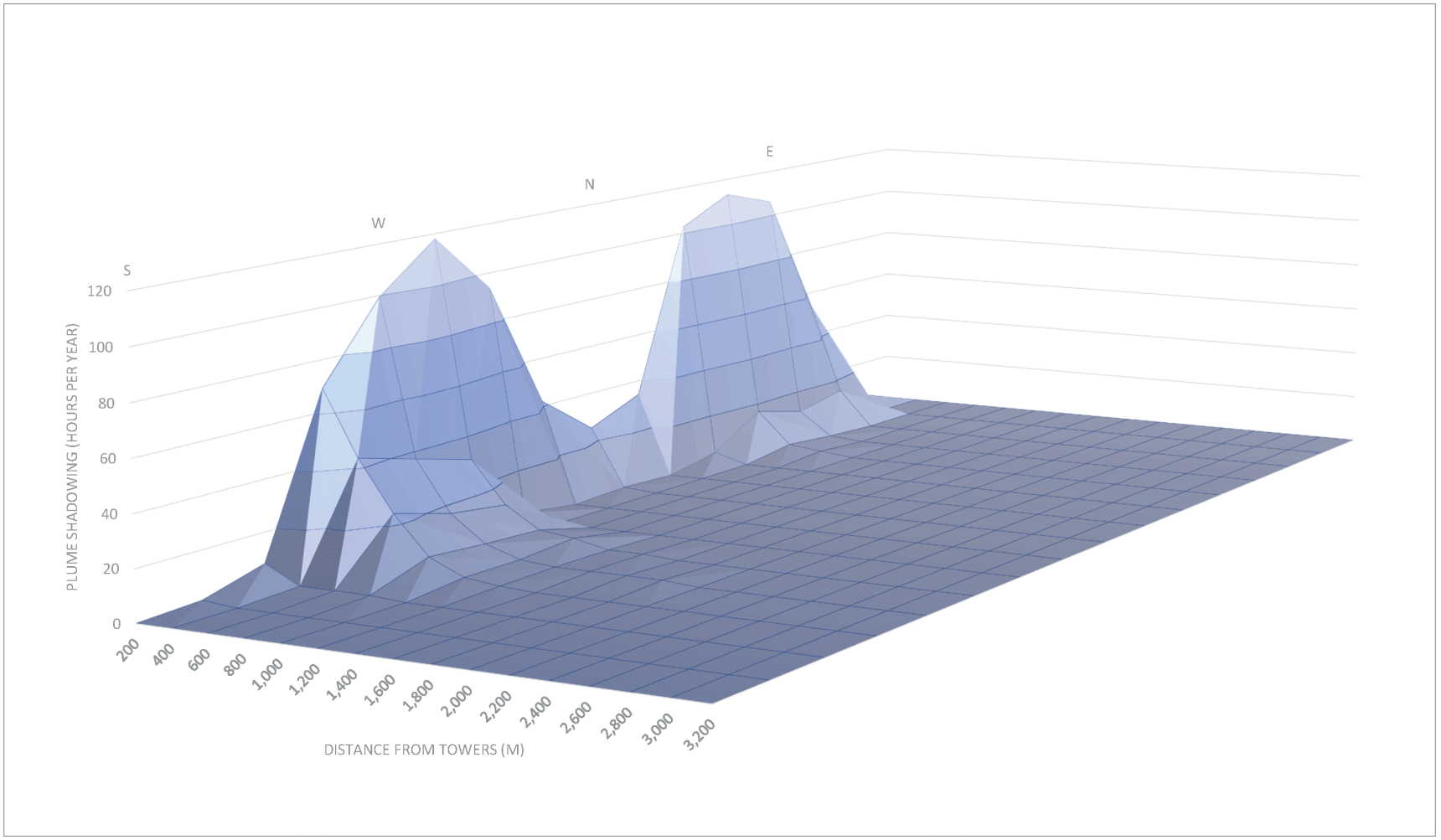

Figure 2.3.2-80R Annual Hours of Plume Shadowing – Single Tower (CRN Tower)

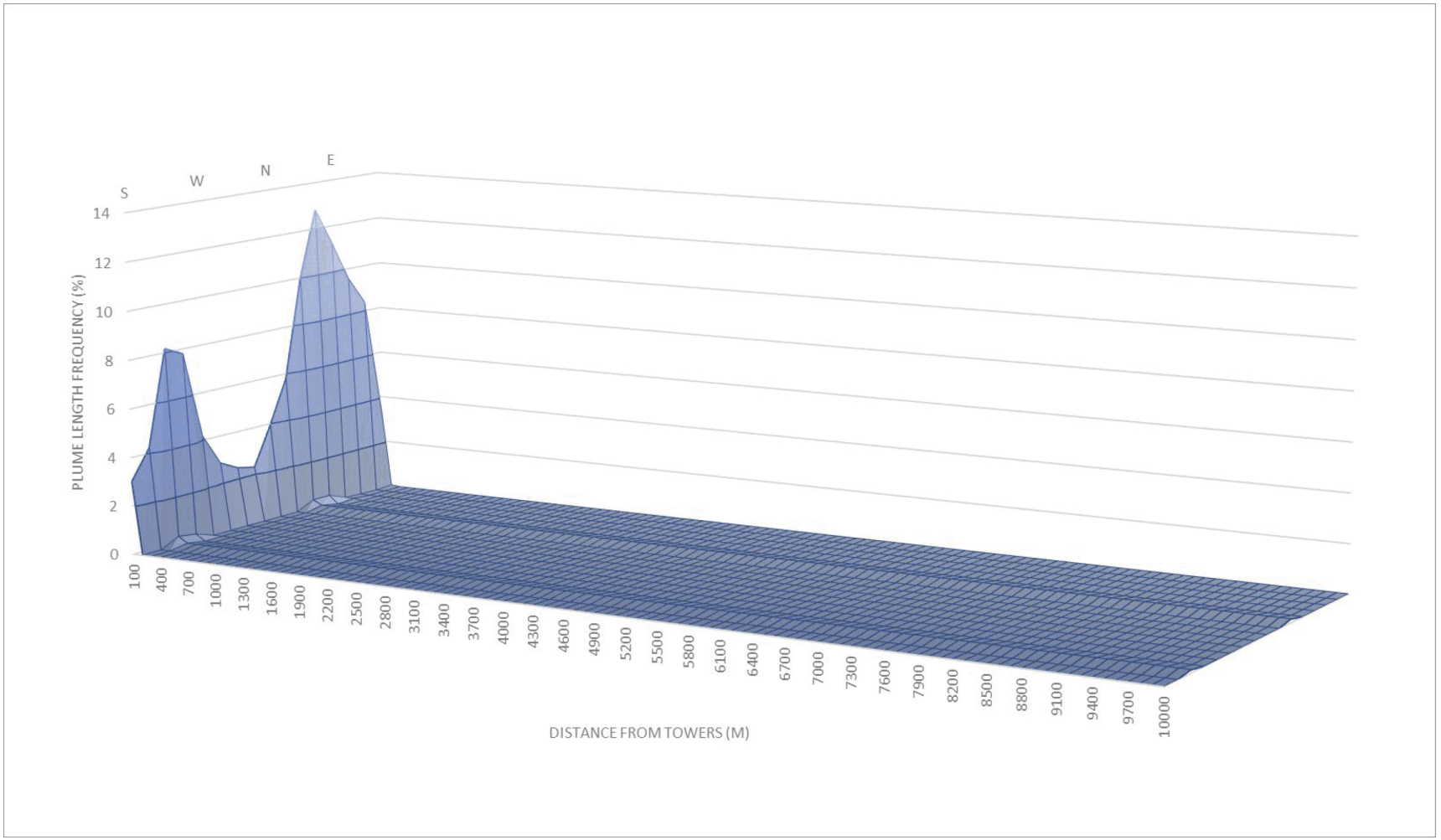

Figure 2.3.2-81R Annual Plume Length Frequency – Single Tower (CRN Tower)

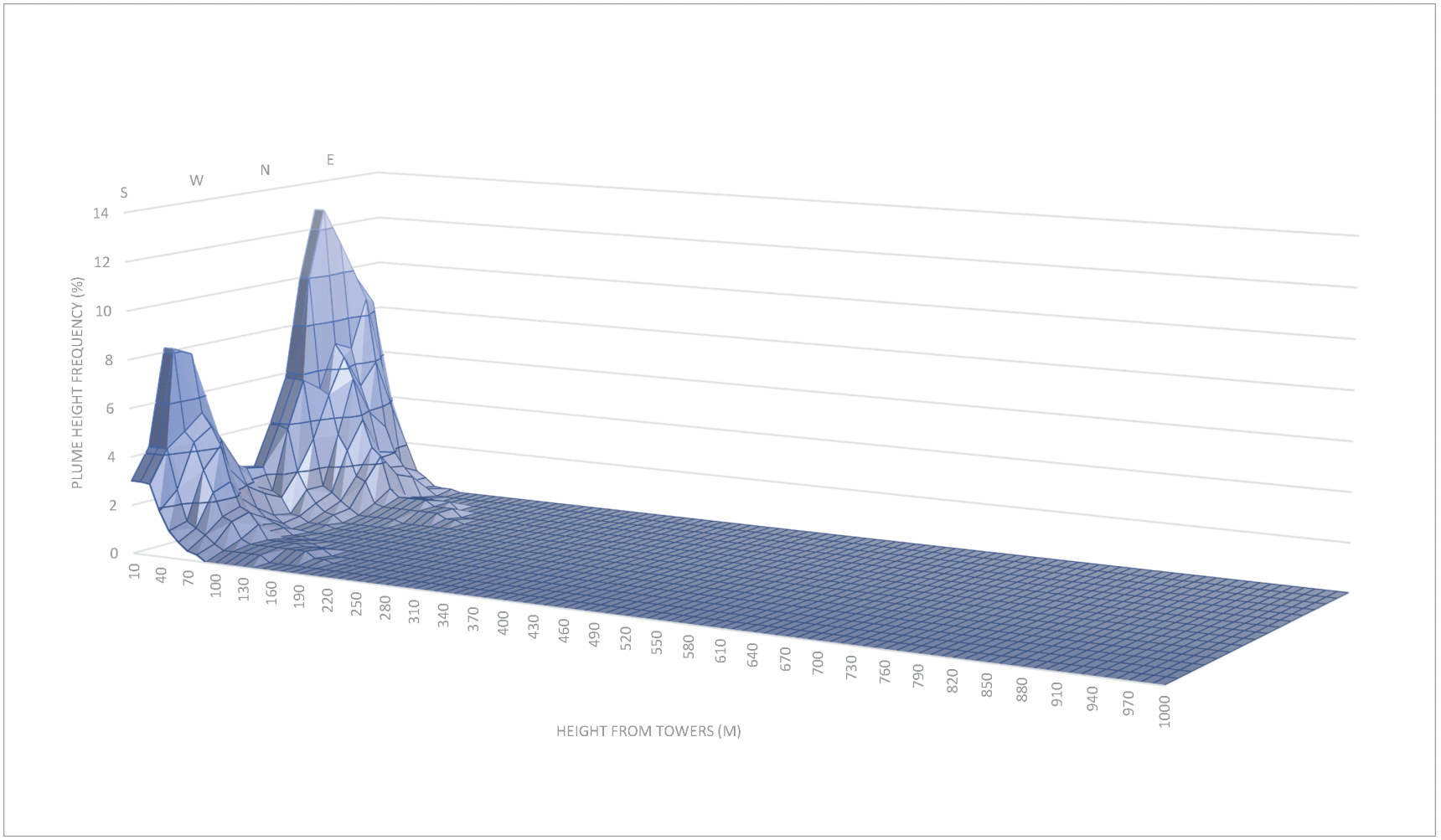

Figure 2.3.2-82R Annual Plume Height Frequency – Single Tower (CRN Tower)

2.3.3 Onsite Meteorological Measurements Program

Insert the following after the first paragraph in ESPA SSAR Section 2.3.3 (Addresses ESP COL Action Items 2.3-2, 2.3-3 and 2.3-4):

Subsequent to USNRC issuance of the early site permit for the CRN Site, TVA selected the GVH BWRX-300 reactor technology for evaluation at the CRN Site. The three years of existing onsite meteorological data for the CRN Site used in the ESPA SSAR provide conservative and representative data for use in the construction permit. The existing onsite meteorological data as presented in ESPA SSAR Section 2.3.3 is used for subsequent evaluation of the BWRX-300 design in the construction permit application.

A new meteorological tower will be installed to collect data during the CRN Site operational phase. This tower, and the associated instrumentation, will be designed to meet the requirements of RG 1.23, Meteorological Monitoring Programs for Nuclear Power Plants. Meteorological data will be collected, consistent with the guidance of ANSI/ANS-3.11, Determining Meteorological Information at Nuclear Facilities, and retained for the life of the facility once a facility becomes operational at the CRN Site.

2.3.4 Short-Term (Accident) Diffusion Estimates

2.3.4.2 Calculation Methodology and Assumptions

Replace 1st sentence of the 2nd paragraph with the following:

SRP 2.3.3 recommends that the Applicant should provide meteorological data from at least two consecutive annual cycles, including the most recent one-year period for calculating the short-term and long-term atmospheric dispersion estimates.

2.3.5 Long-Term (Routine) Diffusion Estimates

2.3.5.2 Calculation Methodology and Assumptions

Replace 1st sentence of the 1st paragraph with the following:

SRP 2.3.3 recommends that the Applicant should provide meteorological data from at least two consecutive annual cycles, including the most recent one-year period for calculating the short-term and long-term atmospheric dispersion estimates.

2.4 Hydrologic Engineering

Section 2.4 of the referenced ESPA SSAR is incorporated by reference with the following supplements.

2.4.0 Hydrology Review

Replace the first and second sentences of the second paragraph of ESPA SSAR Section 2.4.0 with the following:

The CRN Site is located on the north bank of the Clinch River arm of Watts Bar Reservoir between Clinch River Mile (CRM) 19 to CRM 14.5 (Reference 2.4.1-1) with preliminary plant grade at elevation 814.5 feet (ft), referenced to the North American Vertical Datum of 1988 (NAVD88). A single BWRX-300 SMR developed by GVH is planned for the CRN Site and included as Unit 1 in the Construction Permit Application. Final grade elevation will be established in consideration of requirements to provide flood protection for SC1 systems, structures and components (SSCs) in accordance with RG 1.59, Position 1.

2.4.1 Hydrologic Description

2.4.1.1 Site and Facilities

Replace the third sentence of the first paragraph of ESPA SSAR Section 2.4.1.1 with the following:

[

]

Delete third sentence of the second paragraph of ESPA SSAR Section 2.4.1.1.

Replace the fifth sentence of the third paragraph of ESPA SSAR Section 2.4.1.1 with the following:

The river would not supply any SC1 systems.

2.4.1.2.2 Groundwater

Replace the first sentence of the first paragraph of ESPA SSAR Section 2.4.1.2.2 with the following:

Subsection 2.4.12 contains a detailed description of the hydrogeologic conditions at and in the vicinity of the CRN Site and includes the regional and local groundwater resources that could be affected by the construction and operation of BWRX-300 SMR at the CRN Site.

2.4.2 Floods

2.4.2.2 Flood Design Considerations

Replace the last sentence of the Flooding from Landslides section of ESPA SSAR Section 2.4.2.2 with the following (Addresses COL Action Item 2.4-2):

[

]

2.4.2.3 Effects of Local Intense Precipitation

Replace ESPA SSAR Section 2.4.2.3 with the following (Addresses COL Action Item 2.4-1):

[

]

2.4.2.3.1 Precipitation Distribution

Replace ESPA SSAR Section 2.4.2.3.1 with the following:

The LIP depths for the plant were determined from guidance presented in Hydro-Meteorological Report No. 56 and No. 52 (HMR-56 and HMR-52) (References 2.4.2-2 and 2.4.2-1, respectively). The guidelines set forth in HMR-56 and HMR-52 were followed to form the rainfall hyetograph of the 1-hour (hr), 1-sq mi PMP for CRN. PMP rainfall values in HMR-56 were chosen based on the terrain distribution roughness of the site, as shown in Figure 68 of Reference 2.4.2-2. The CRN Site falls in the rough zones in Figure 68.

The base LIP hyetograph for the CRN Site is determined by adjusting the “rough terrain” precipitation depths in Table 6 of HMR-56. The moisture reduction index factor of 95.6 percent from HMR-56, Figure 20, representative of the CRN Site location was omitted, and rainfall is therefore conservative.

A rainfall depth-duration curve was created using depths from HMR-56 for 5-minute increments up to 1-hour duration. Times of concentration were determined for the respective site sub-watersheds and corresponding rainfall depths from the depth-duration curve were applied as steady state. The cumulative rainfall values are summarized in Table 2.4.2-2R. Additional analysis will be performed at OLA.

2.4.2.3.2 Runoff Model

Replace ESPA SSAR Section 2.4.2.3.2 with the following:

Runoff was assumed at 100% of rainfall with constant peak flow using a Rational Method approach. The site layout and facilities at the CRN Site have not been finalized. Thus, the final runoff model has not been developed. This task would be performed as part of detailed engineering included with the OLA.

2.4.2.4 Site Drainage System

Replace ESPA SSAR Section 2.4.2.4 with the following:

The location and design of stormwater conveyance facilities have not been determined. These tasks would be performed as part of detailed engineering at OLA. Stormwater would be carried away from the plant area by a stormwater collection system comprising appropriate combinations of swales, open channels, or a subsurface system of catch basins, pipes and culverts. Runoff would be routed to the Watts Bar Reservoir. However, the finished grade for the site would slope away from buildings and be designed such that the peak discharges from the LIP would not flood SC1 facilities if the subsurface drainage system were plugged.

Replace Table 2.4.2-2 with the following:

Table 2.4.2-2R Clinch River Nuclear Site Local Intense Precipitation Values—1-hr, 1-sq mi Probable Maximum Precipitation

Sub-Watershed	Time of Concentration (minutes)	Model Rainfall Depth (inches)	Rainfall Intensity (inches/hour)
A	14.48	7.92	32.79
B	17.04	8.83	31.10
C	16.38	8.60	31.51
D	8.20	5.19	37.97
E	13.79	7.65	33.29
F	14.48	7.91	32.79
G	11.33	6.65	35.22
H	15.34	8.23	32.19

2.4.3 Probable Maximum Flood (PMF) on Streams and Rivers

2.4.3.7 Coincident Wind Wave Activity

Replace the fifth sentence of the second paragraph of ESPA SSAR Section 2.4.3.7 with the following:

The effective fetch found for the CRN Site from available GIS terrain data was 4.25 mi, based on the site grade elevation of 814.5 ft, and results in the critical site fetch length (Figure 2.4.3-16).

Replace the seventh sentence of the second paragraph of ESPA SSAR Section 2.4.3.7 with the following:

CRN Site grade is 814.5 ft NAVD88 (814.9 ft NGVD29), 8.9 ft higher than the maximum calculated water surface elevation with wind wave height.

2.4.4 Potential Dam Failures

2.4.4.4 Water Level at CRN Site

Replace the fourth sentence of the last paragraph of ESPA SSAR Section 2.4.4.5 with the following:

There are no impacts to SC1 SSCs at the CRN Site.

2.4.4.5 Coincident Wind Wave

Replace the fifth sentence of ESPA SSAR Section 2.4.4.5 with the following:

The effective fetch found for the CRN Site from available GIS terrain data was 4.25 mi, based on the site grade elevation of 814.5 ft NAVD88, which results in the critical site fetch length (Figure 2.4.3-16).

Replace the eighth sentence of ESPA SSAR Section 2.4.4.5 with the following:

CRN Site grade is 814.5 ft NAVD88 (814.9 ft NGVD29), 8.9 ft higher than the maximum calculated water surface elevation with wind wave height.

2.4.4.6 Erosions and Deposition Effects

Replace the last sentence of ESPA SSAR Section 2.4.4.6 with the following:

This less than two foot rise in Watts Bar Reservoir is far short of the nearly 18.5 ft of elevation rise needed based on the bounding PMF results to reach preliminary CRN plant grade.

2.4.4.7 Erosions and Deposition Effects

Replace the last sentence of ESPA SSAR Section 2.4.4.7 with the following:

This less than two foot rise in Watts Bar Reservoir is far short of the nearly 18.5 ft of elevation rise needed based on the bounding PMF results to reach preliminary CRN plant grade.

2.4.5 Probable Maximum Surge and Seiche Flooding

Replace the second sentence of the first paragraph of ESPA SSAR Section 2.4.5 with the following:

[

]

2.4.6 Probable Maximum Tsunami Hazards

Replace the second sentence of the first paragraph of ESPA SSAR Section 2.4.6 with the following:

In addition, the site finish grade elevation is at 814.5 ft NAVD88 (814.9 ft NGVD29).

Replace the second sentence of the second paragraph of ESPA SSAR Section 2.4.6 with the following:

As summarized in Section 2.5.3, there are no capable faults within the site region, and there is negligible potential for tectonic fault rupture within the CRN Site vicinity.

Replace the third sentence of the last paragraph of ESPA SSAR Section 2.4.6 with the following:

This results in a water depth of about 41 ft at the normal maximum operating pool level. Because the CRN Site is located approximately 50 river miles upstream of the Watts Bar Dam and 73.5 ft above the reservoir normal maximum operating pool level, any landslide-generated tsunami in the reservoir, with water depth of about 41 ft near the dam, is unlikely to affect the safe functioning of the CRN Site.

2.4.7 Ice Effects

Replace the first paragraph of ESPA SSAR Section 2.4.7 with the following:

The potential impact of ice effects on the CRN Site is evaluated in this section. Because the site is located in a region of continental, humid subtropical climate, extreme ice conditions with potential adverse effects on SC1 SSCs are not typically present. Additionally, the BWRX-300 SMR does not rely on an external, SC1 water source and therefore will not be affected by any potential loss of water supply due to ice conditions in the region.

2.4.7.2 High and Low Water Levels

Replace the first paragraph of ESPA SSAR Section 2.4.7.2 with the following:

[

]

[

]

Replace the last paragraph of ESPA SSAR Section 2.4.7.2 with the following:

Low water conditions resulting from ice formation or accumulations that may cause interruption of stream flow in Clinch River arm of the Watts Bar Reservoir will not impact the site, as BWRX-300 SMR does not rely on an external water source for an SC1 water supply.

2.4.7.3 Ice Sheet Formation

Replace the first paragraph of ESPA SSAR Section 2.4.7.3 with the following:

The BWRX-300 SMR does not require external water sources for SC1 SSCs and does not rely on any open storage facilities or impoundments to supply or maintain the inventory for SC1 water use. Consequently, a reduction of stream flow in the river or storage volume in an impoundment due to ice sheet formation will not introduce any adverse impacts to SC1 SSCs.

Replace the first sentence of the second paragraph of ESPA SSAR Section 2.4.7.3 with the following:

The BWRX-300 SMR uses cooling towers to dissipate the excess heat load from the main steam condensers.

Replace the second sentence of the third paragraph of ESPA SSAR Section 2.4.7.3 with the following:

The ice layer thickness was estimated to evaluate the impact of ice formation on the cooling tower basins and intake structure.

2.4.7.4 Ice-Induced Forces and Blockages

Replace ESPA SSAR Section 2.4.7.4 with the following:

The BWRX-300 SMR does not include SC1 SSCs that are subject to ice-induced forces or blockages from either sheet ice or frazil ice.

2.4.7.7 Conclusion

Replace ESPA SSAR Section 2.4.7.7 with the following:

Historical and postulated ice conditions applicable to the CRN Site are evaluated to determine the impact on the safety of the plant. Due to the relatively mild climatic condition at the CRN Site, and the elevated design plant grade above natural drainages, in combination with the BWRX-300 SMR plant design that does not rely on external water sources for SC1 water use, it is concluded that ice effects will not cause flooding or water availability concerns for the SC1 SSCs.

2.4.8 Cooling Water Canals and Reservoirs

Replace ESPA SSAR Section 2.4.8 with the following:

The BWRX-300 SMR plant design does not rely on the Clinch River arm of the Watts Bar Reservoir for an SC1 water supply, and the CRN Site does not include cooling water canals or reservoirs. The CRN Site uses the Clinch River arm of the Watts Bar Reservoir as the source of makeup water for normal plant cooling.

2.4.9 Channel Diversions

Replace ESPA SSAR Section 2.4.9 with the following:

This section describes the effects of channel diversions as potential hazards to the BWRX-300 SMR. Based on the review of hydrologic, hydraulic, climatic, topographic and geologic evidence and anthropogenic impacts near the CRN Site, as described below, channel diversions are not expected to cause flooding hazards at the site. Additionally, the BWRX-300 SMR does not require SC1 water supply from the Clinch River arm of the Watts Bar Reservoir. Consequently, the safety of the plant will not be compromised due to a loss of water supply even in the highly unlikely event of a channel diversion near the site.

2.4.9.6 Alternative Water Sources

Replace the first paragraph of ESPA SSAR Section 2.4.9.6 with the following:

The BWRX-300 SMR does not rely on the Clinch River arm of the Watts Bar Reservoir for SC1 water use. However, it is the source for the cooling tower makeup water for the normal heat sink with the water intake located on the river. The makeup water intake, while unlikely to be affected by channel diversions under normal operation conditions, could be subject to impacts from shoreline erosion. As described previously, the shoreline erosion processes would occur at a localized scale and at a rate that could be mitigated. In the unlikely event that a channel diversion occurs (due to, for example, dam break and slope failure) interrupting the makeup water supply, the plant can still be safely shutdown. Plant water use is further described in Subsection 2.4.11.

2.4.9.8 Conclusions

Replace the second sentence of the first paragraph of ESPA SSAR Section 2.4.9.8 with the following:

Moreover, in the unlikely event that channel diversions occur, there will be no impact to SC1 SSCs at the CRN Site.

2.4.10 Flooding Protection Requirements

Replace the first sentence of the first paragraph of ESPA SSAR Section 2.4.10 with the following:

[

]

2.4.11 Low Water Considerations

Replace the first and second sentences of the first paragraph of ESPA SSAR Section 2.4.11 with the following:

The Clinch River Nuclear (CRN) Site does not rely on the Clinch River arm of the Watts Bar Reservoir to support SC1 SSCs, including the ultimate heat sink that provides emergency cooling water. However, the Clinch River arm of the Watts Bar Reservoir would be used as a water source for non-SC1 SSCs, primarily the normal cooling system and water processing plant that feeds the demineralized water users and liquid radwaste treatment system.

Replace the first sentence of the second paragraph of ESPA SSAR Section 2.4.11 with the following:

The CRN Site is located on the Clinch River arm of the Watts Bar Reservoir between River Mile 14.5 and River Mile 19, and the surface water demand on the reservoir would be supplied using a pump intake structure located at Clinch River Mile 17.9, approximately 5.2 river miles downstream from Melton Hill Dam.

2.4.11.1.1 Flow Release from Melton Hill Dam

Replace the first two sentences of the first paragraph of the Reservoir Storage section of ESPA SSAR Section 2.4.11.1.1 with the following:

To evaluate the surface water availability for CRN Site water demand, the depletion rate of the reservoir storage below the winter/normal minimum operating pool level was estimated using the stage-storage rating curve of Watts Bar Reservoir in conjunction with conservative assumptions on the expected water demands, reservoir inflows and outflows.

Replace the fifth sentence of the second paragraph of ESPA SSAR Section 2.4.11.1.1 with the following:

In addition, the average weekly discharge from Melton Hill Reservoir over the long term is about 4800 cfs, with a maximum weekly discharge of about 25,450 cfs, much higher than the projected water demand (consumptive surface water use) of 28.5 cfs from the CRN Site.

Replace the first sentence of the third paragraph of the Reservoir Storage section of ESPA SSAR Section 2.4.11.1.1 with the following:

It can be concluded from the above assessment that the amount of storage volume available in Watts Bar Reservoir, even during the worst drought condition when hypothetically the only inflow to the Reservoir is the minimum release from Melton Hill Dam, can conservatively sustain the continuous withdrawal of surface water to meet the water demands of the CRN Site for extended periods of time, in addition to the continuous supply to other water users.

2.4.11.2 Low Water Resulting from Surges, Seiches, or Tsunamis

Replace the first and second sentences of the first paragraph of ESPA SSAR Section 2.4.11.2 with the following:

There is no safety impact on the CRN Site due to low water conditions generated by any mechanism because the plant does not rely on the Clinch River arm of the Watts Bar Reservoir for its SC1 water use. The non-SC1 water supply is also not expected to be affected normally by low water conditions resulting from surges, seiches, tsunamis or other external events.

Replace the second sentence of the second paragraph of ESPA SSAR Section 2.4.11.2 with the following:

However, no SC1 SSCs at the CRN Site would be impacted.

Replace the third paragraph of ESPA SSAR Section 2.4.11.2 with the following:

Similarly, the unlikely failures of the Watts Bar Dam and Melton Hill Dam would potentially create a low water condition at the CRN Site, but there would be no SC1 impacts.

2.4.11.4 Future Controls

Replace the first and second sentences of the first paragraph of ESPA SSAR Section 2.4.11.4 with the following:

The CRN Site does not rely upon the Clinch River arm of the Watts Bar Reservoir for SC1 water use and therefore there would not be any safety impact as a result of future uses or controls in the Reservoir. For normal water use, the storage capacity of Watts Bar Reservoir is adequate to sustain the continuous supply of surface water to meet the needs of the CRN Site through a drought event equivalent to the drought of record as discussed in Subsection 2.4.11.3.

2.4.11.5 Heat Sink Dependability Requirement

Delete the second sentence of the first paragraph of ESPA SSAR Section 2.4.11.5.

2.4.12 Groundwater

Replace the first sentence of the second paragraph of ESPA SSAR Section 2.4.12 with the following:

Regional and local groundwater resources that could be affected by the construction and operation of the BWRX-300 SMR are described below.

2.4.12.1.1 Physiography and Geomorphology

Replace the last sentence of the third paragraph of ESPA SSAR Section 2.4.12.1.1 with the following:

Plant grade is at approximately 814.5 ft, placing the BWRX-300 SMR about 74.5 ft above the water level of the river.

Replace Figure 2.4.12-4 with Figure 2.4.12-4R in the first and last sentence of the eighth paragraph of ESPA SSAR Section 2.4.12.1.1.

Replace Figure 2.4.12-5 with Figure 2.4.12-5R in the second sentence of the eighth paragraph of ESPA SSAR Section 2.4.12.1.1.

2.4.12.1.4.1 CRBRP Investigation

Delete the fourth sentence of the first paragraph of ESPA SSAR Section 2.4.12.1.4.1.

Insert the following after the first paragraph in ESPA SSAR Section 2.4.12.1.4.1:

TVA identified no abandoned wells in the area adjacent to the CRBRP excavation while performing the well drilling associated with the 2013 CRN Site subsurface investigation and based on the information on well locations for the CRBRP, it is likely that the wells were destroyed and/or removed when the excavation and subsequent site redress for the CRBRP was performed.

Replace the first sentence of the fifth paragraph of ESPA SSAR Section 2.4.12.1.4.1 with the following:

Water level measurements on the CRN Site indicated as much as a 25 ft fluctuation in water levels.

Delete the sixth paragraph of ESPA SSAR Section 2.4.12.1.4.1

2.4.12.1.4.2 CRN Site Investigation

Replace the fourth sentence of the second paragraph of ESPA SSAR Section 2.4.12.1.4.2 with the following:

Due to the naturally occurring hydrocarbon in OW-422L, this well cluster (OW-422 U, L and D) was not developed and since then fully grouted.

Replace Figure 2.4.12-4 with Figure 2.4.12-4R in the fifth sentence of the first paragraph of ESPA SSAR Section 2.4.12.1.4.2.

2.4.12.2.1.3 Projected Groundwater Use

Replace the first sentence of the second paragraph of ESPA SSAR Section 2.4.12.2.1.3 with the following:

The BWRX-300 SMR does not require groundwater as a source for cooling water, potable water, or other plant needs.

2.4.12.2.1.4 CRN Site Groundwater Use Summary

Replace the last paragraph of ESPA SSAR Section 2.4.12.2.1.4 with the following:

The BWRX-300 SMR does not rely on groundwater for any part of its operating supply. Thus, there is no groundwater demand due to the BWRX-300 SMR. There are no active groundwater current or projected users for the CRN Site.

2.4.12.2.2 Groundwater Flow Directions

Replace the fourth sentence of the third paragraph of ESPA SSAR Section 2.4.12.2.2 with the following:

Figure 2.4.12-19R through Figure 2.4.12-28R present the potentiometric surface maps.

Replace the eleventh sentence of the third paragraph of ESPA SSAR Section 2.4.12.2.2 with the following:

This is derived based on a shortest distance of 1200 ft from the reactor locations to the edge of the Clinch River arm of the Watts Bar Reservoir; lowest stage of the reservoir at 735 ft NAVD88 (during the monitoring period); and the maximum water levels at OW-101U and OW-202U of 798.99 and 800.30 ft NAVD88.

2.4.12.2.4.1 Hydrogeological Properties

Replace Figure 2.4.12-4 with Figure 2.4.12-4R in second sentence of the first paragraph of the Aquifer Pumping Test subsection.

2.4.12.3.2 Advective Transport

Replace the last paragraph of ESPA SSAR Section 2.4.12.3.2 with the following:

Table 2.4.12-17R presents a summary of these parameters and the linear velocity and travel times determined from these parameters. Using the representative parameter values, a travel time of 308 days is determined (Table 2.4.12-17R).

2.4.12.5.1 Groundwater Flow Model

Replace Figure 2.4.12-40 with Figure 2.4.12-40R in the second sentence of the second paragraph of ESPA SSAR Section 2.4.12.5.1.

Replace the fifth paragraph of ESPA SSAR Section 2.4.12.5.1 with the following:

Post-construction modeling is performed to estimate the response of groundwater levels to the inclusion of the following BWRX-300 specific structures in the model: radwaste building, reactor

building, and turbine building. A single BWRX-300 SMR developed by GVH is planned for the CRN Site and included as Unit 1 in the Construction Permit Application. The location of the Unit 1 reactor building, radwaste building, and turbine building coincide with the Profile C cross-sectional model. The embedment foundation depth of the reactor building is 129.5 feet below grade. The grade elevation at the power block area at the reactor building was assigned a value of 814.5 ft NAVD88. A uniform recharge of 8.76 in./yr was assigned in the post-construction models except at the power block area and part of the turbine area (which are comprised of paved areas and buildings). A construction fill hydraulic conductivity of 1.5 ft/day was assigned in the post-construction model. Modeling results indicate a maximum piezometric head underneath the reactor building of 814.5 ft NAVD88, which is the same as the power block plant site grade elevation. The maximum piezometric head underneath the turbine building foundation is estimated to be 814.2 ft NAVD88, which is 0.3 ft below the plant site grade elevation. The estimated maximum piezometric head of 814.5 ft NAVD88 meets the BWRX-300 generic groundwater heads (Reference 2.4.12-51) conservatively selected to be at plant grade.

2.4.12.5.2 Hydrostatic Loading

Replace Figure 2.4.12-41 with Figure 2.4.12-41R in the last paragraph of ESPA SSAR Section 2.4.12.5.2.

2.4.12.7 References

Add the following Reference to ESPA SSAR Section 2.4.12.7:

2.4.12-51 NEDO-33914-A, BWRX-300 Advanced Civil Construction and Design Approach, GE Hitachi Nuclear Energy, Revision 2, June 2022.

Replace Table 2.4.12-19 with Table 2.4.12-19R in ESPA SSAR Table 2.4.12-8.

Replace Table 2.4.12-17 with the following:

Table 2.4.12-17R Groundwater Linear Velocity and Travel Time

Property	Representative Value	Source
Hydraulic Conductivity (ft/d)	2.6	Maximum calculated value as documented in Table 2.4.12-12 (observation well OW-423L)
Horizontal Hydraulic Gradient (ft/ft)	0.07	Mean value as presented in Table 2.4.12-8
Effective Porosity	0.0467	Mean value determined in Table 2.4.12-13, using the Immersion test method results which the referenced author identified as the test method that yields results that most accurately approaches the true effective porosity value.
Distance to Receptor (ft)	1200	Shortest distance from edge of power block area to Clinch River arm of the Watts Bar Reservoir (Figure 2.4.13-1R)

Calculated Values	
Linear Velocity (ft/d)	3.90
Travel Time (days)	308
Travel Time (years)	0.84

Replace Figure 2.4.12-4 with the following:

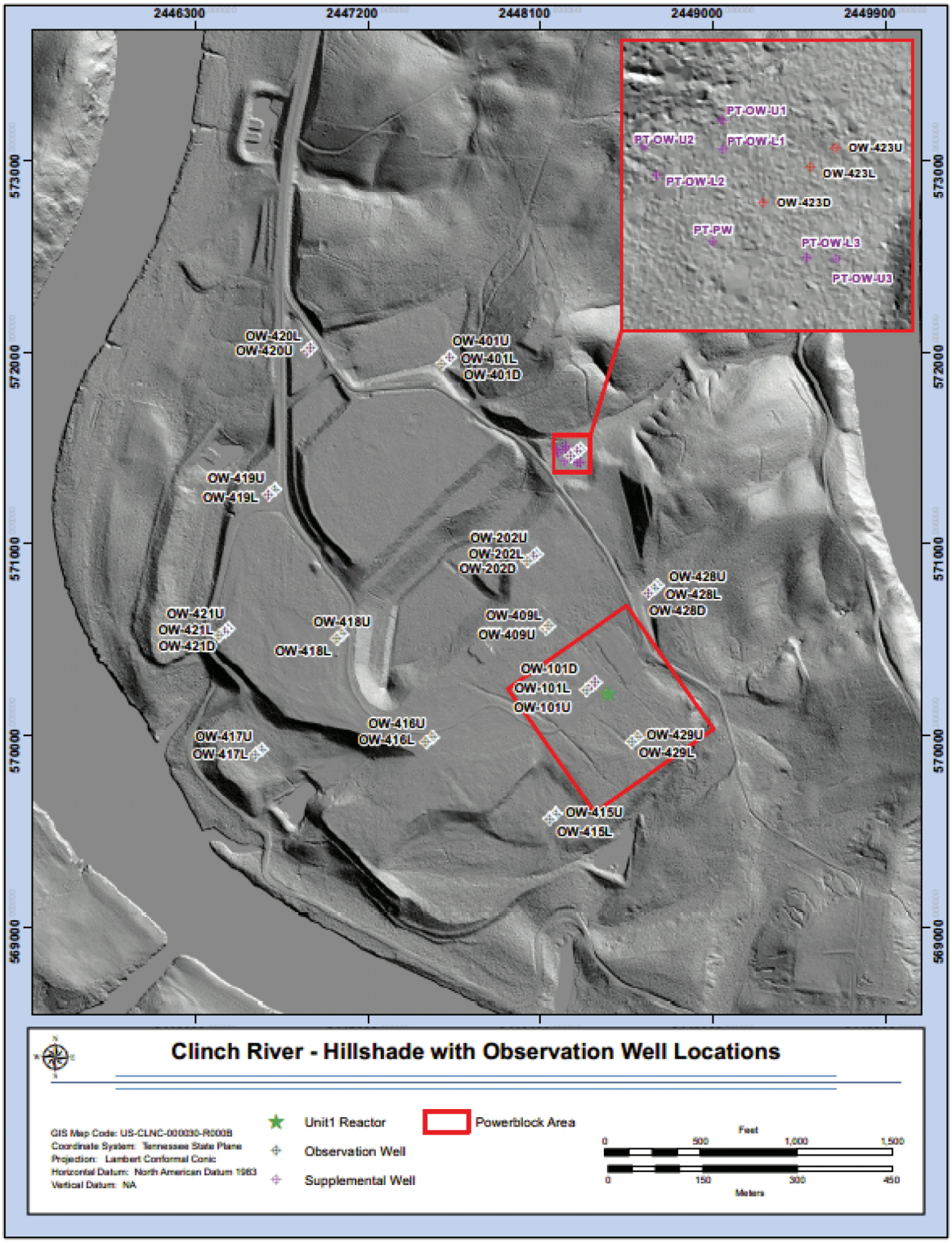

Figure 2.4.12-4R Current Site Topography and Observation Well Locations

Replace Figure 2.4.12-5 with the following:

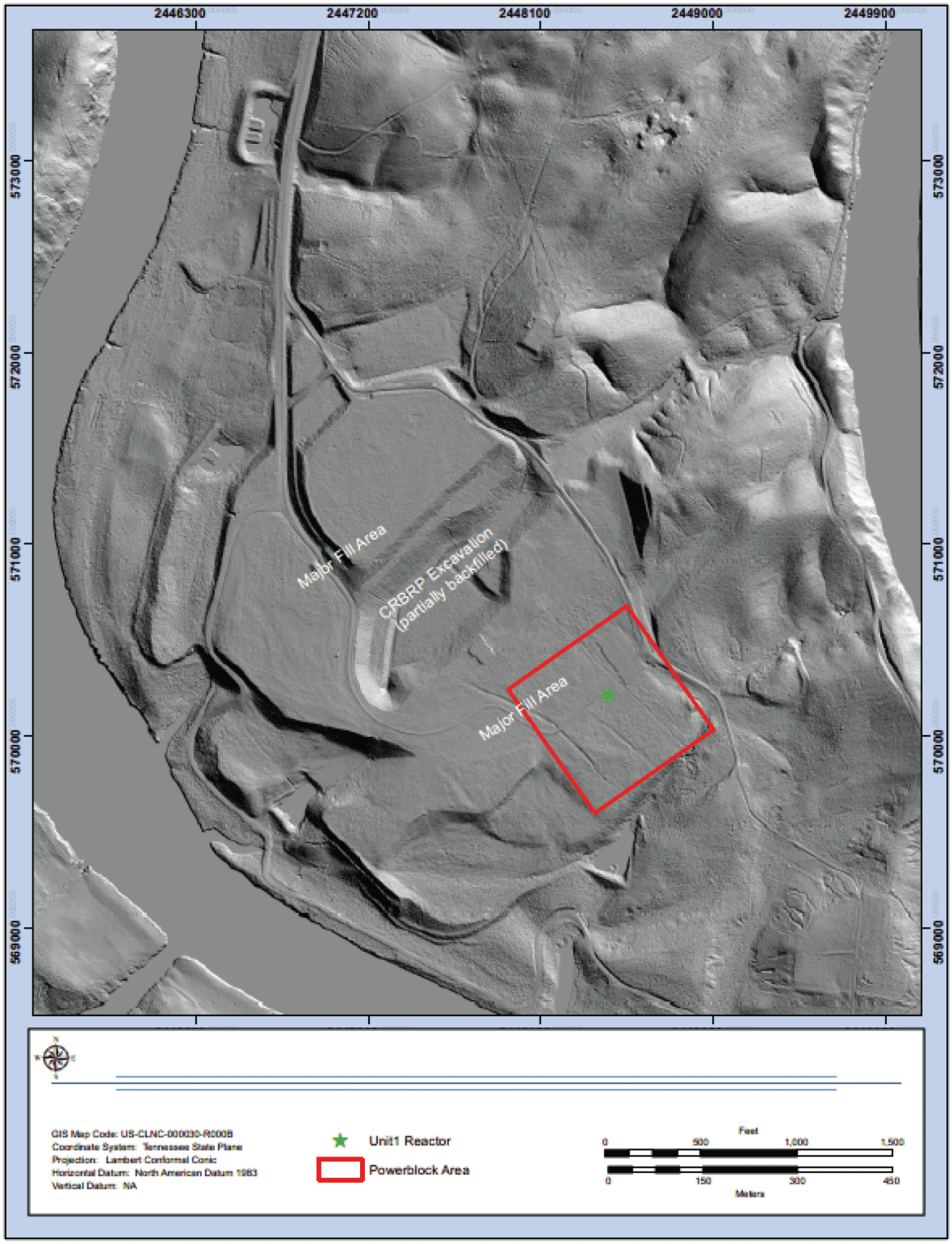

Figure 2.4.12-5R Clinch River Breeder Reactor Project Fill and Excavation Areas

Replace Figure 2.4.12-19 with the following:

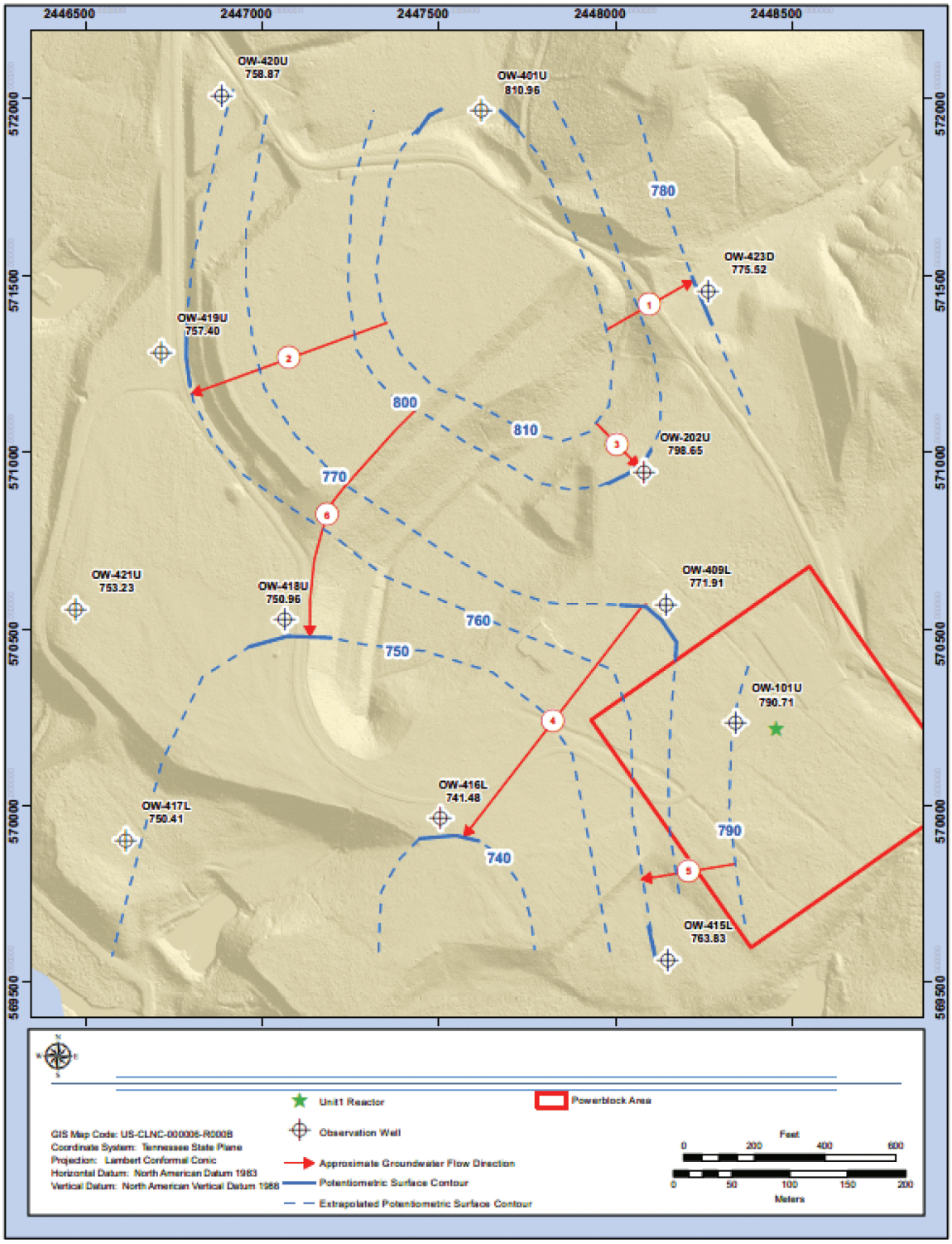

Figure 2.4.12-19R Potentiometric Surface Map for September 24, 2013

Replace Figure 2.4.12-20 with the following:

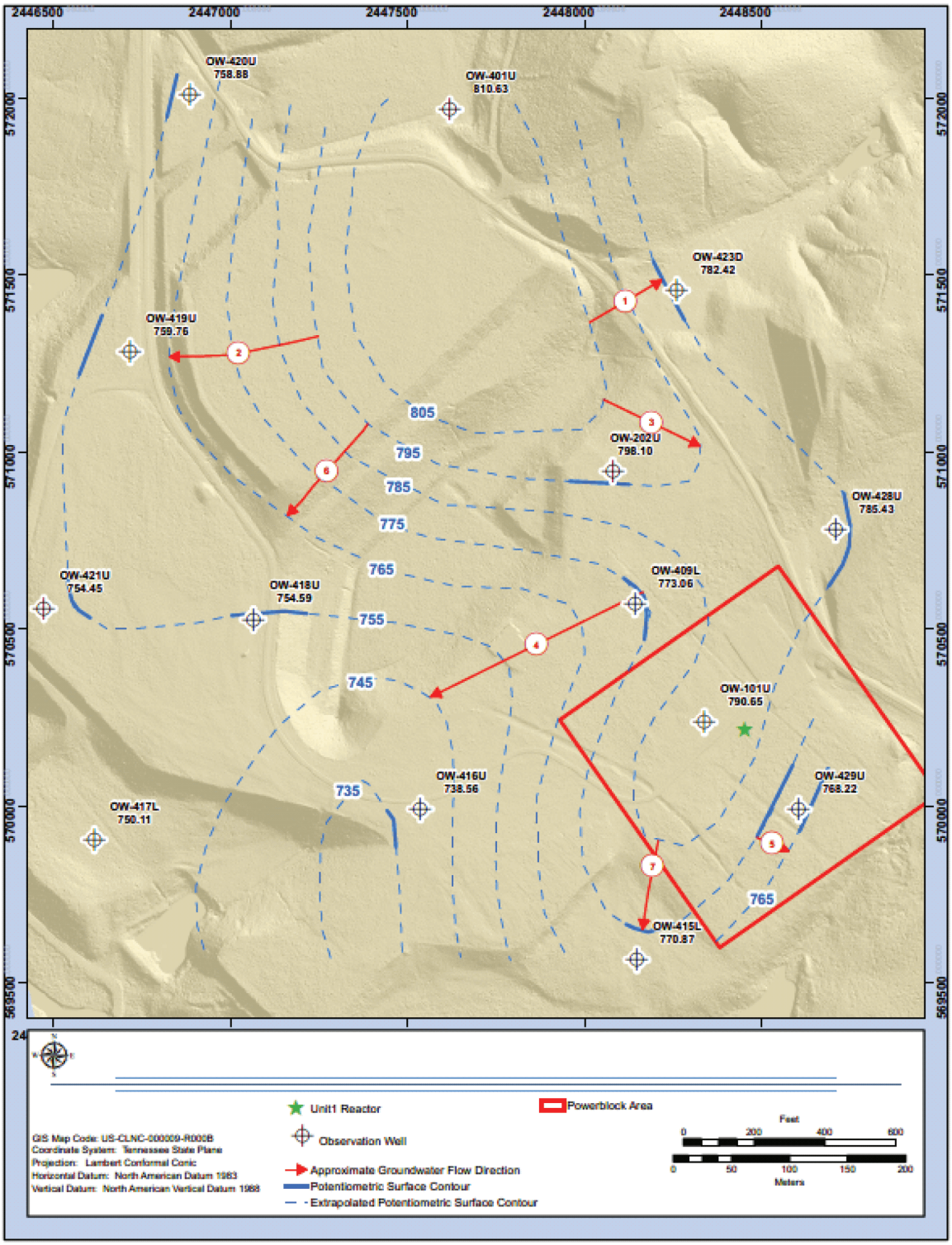

Figure 2.4.12-20R Potentiometric Surface Map for December 20, 2013

Replace Figure 2.4.12-21 with the following:

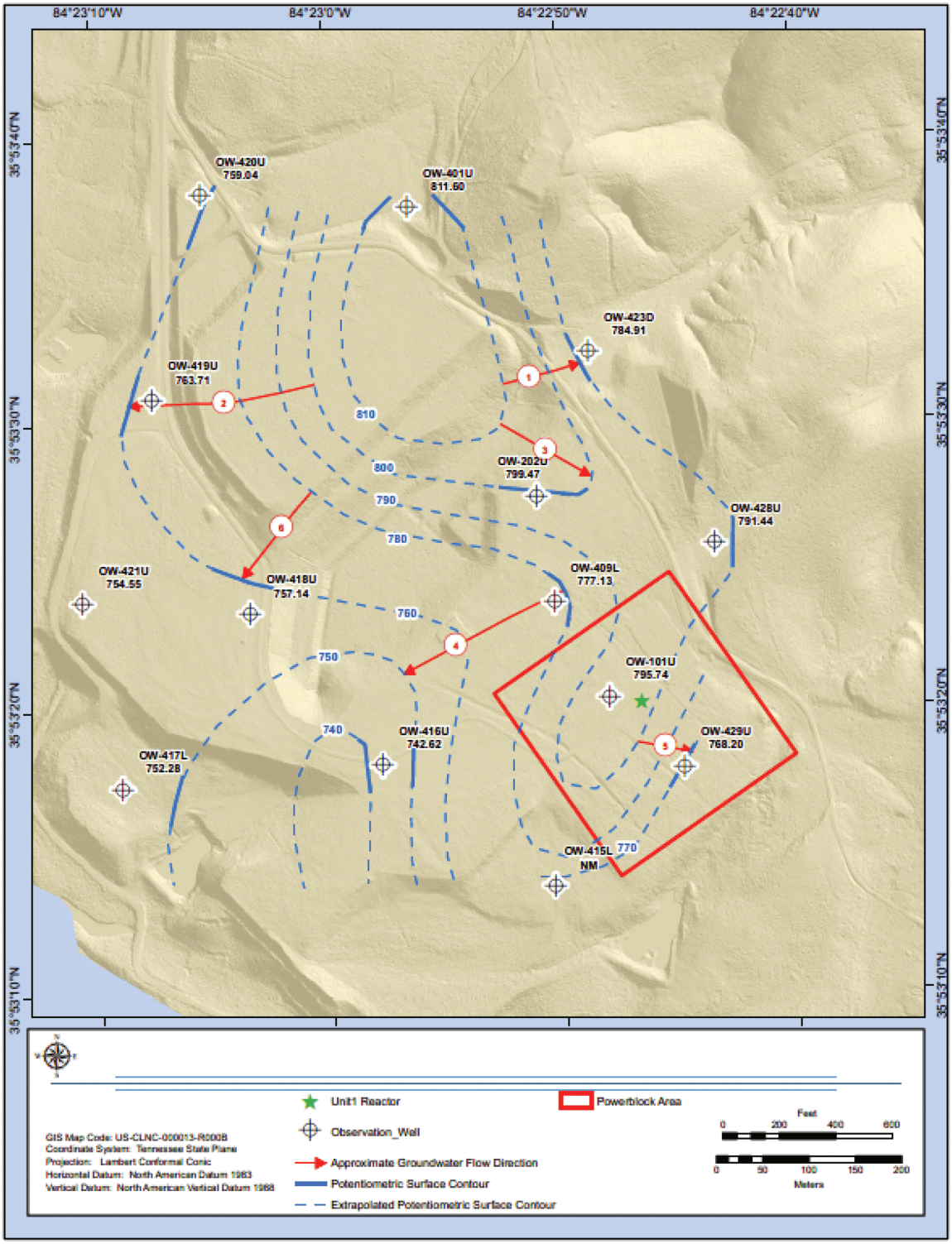

Figure 2.4.12-21R Potentiometric Surface Map for January 13, 2014

Replace Figure 2.4.12-22 with the following:

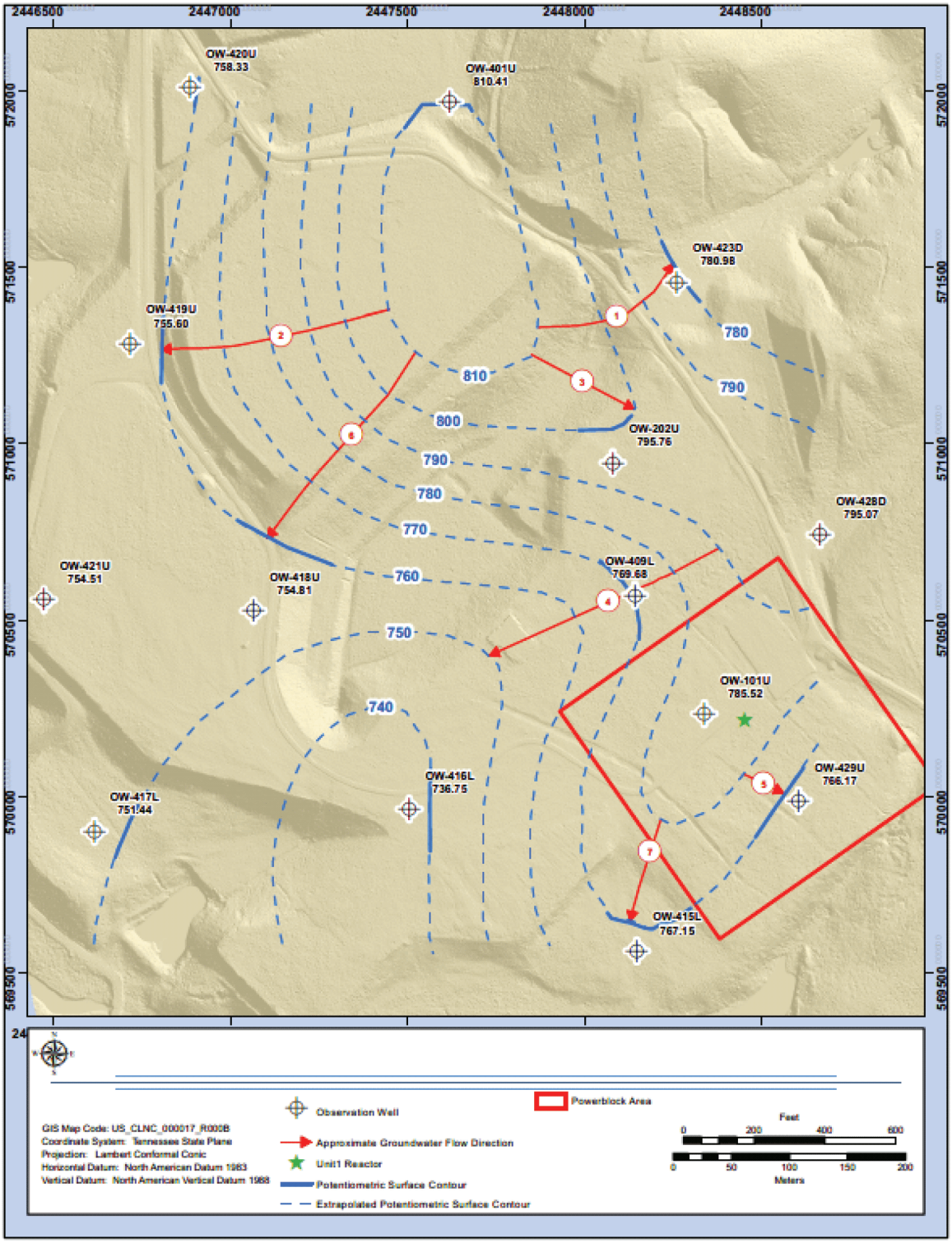

Figure 2.4.12-22R Potentiometric Surface Map for March 16, 2014

Replace Figure 2.4.12-23 with the following:

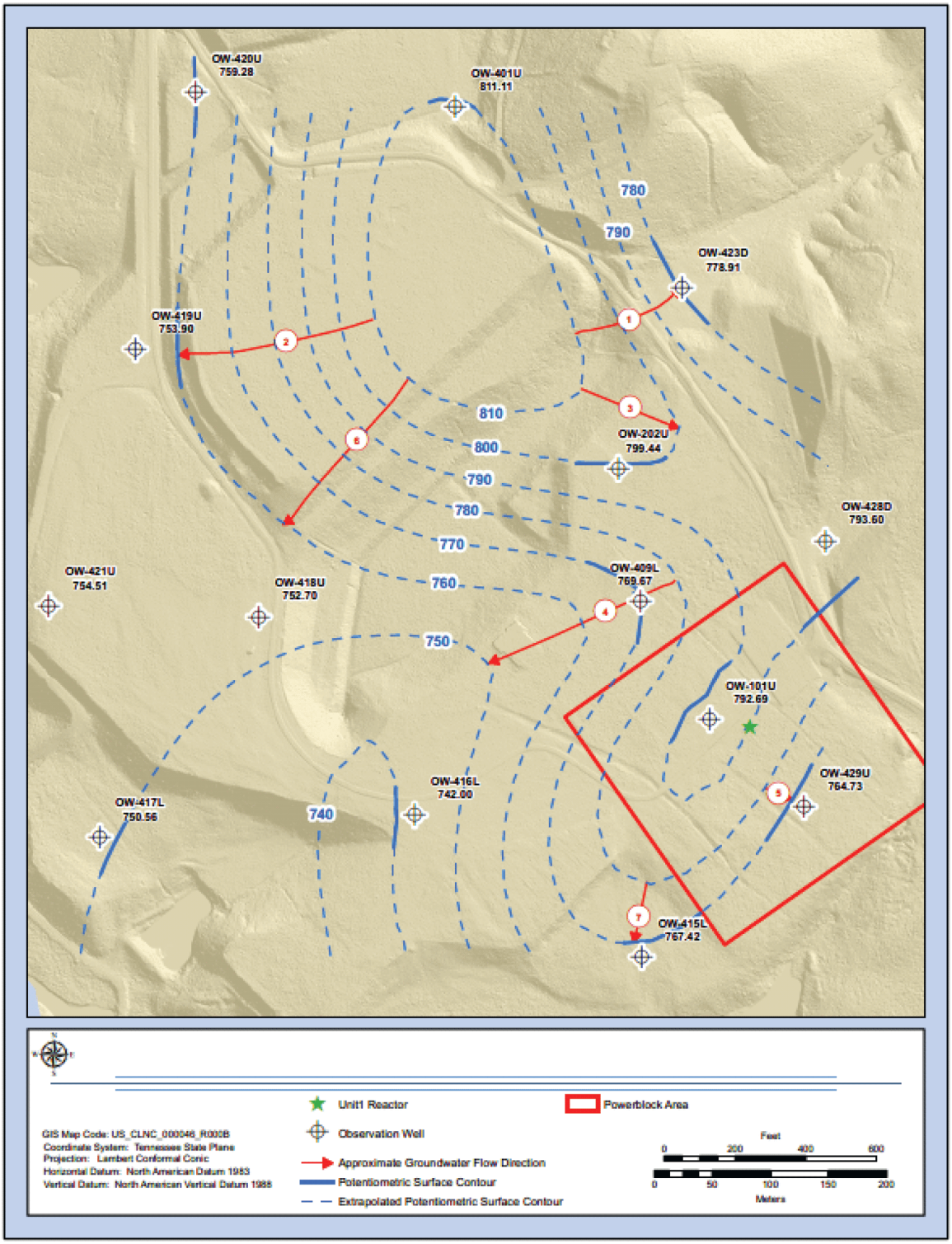

Figure 2.4.12-23R Potentiometric Surface Map for May 15, 2014

Replace Figure 2.4.12-24 with the following:

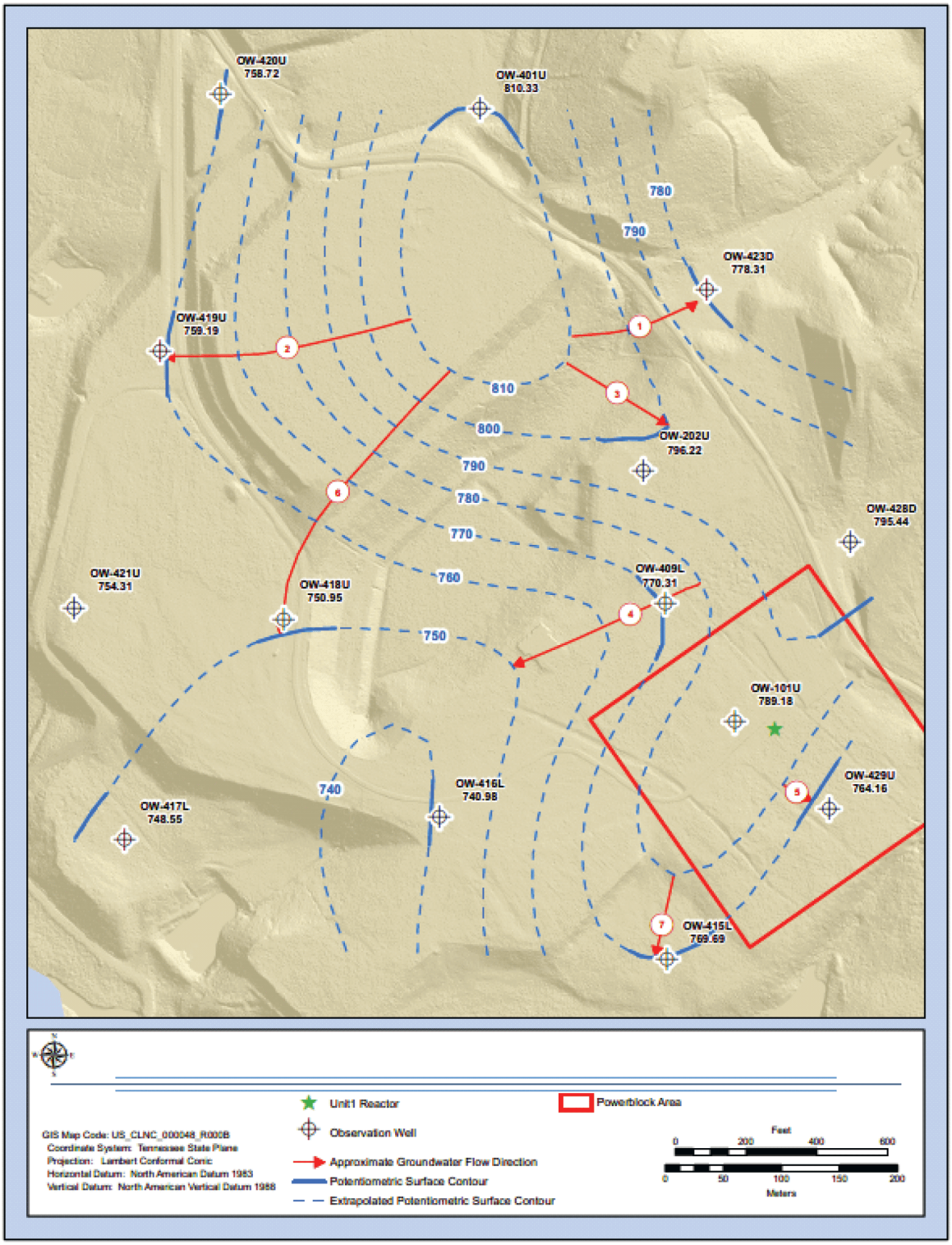

Figure 2.4.12-24R Potentiometric Surface Map for August 18, 2014

Replace Figure 2.4.12-25 with the following:

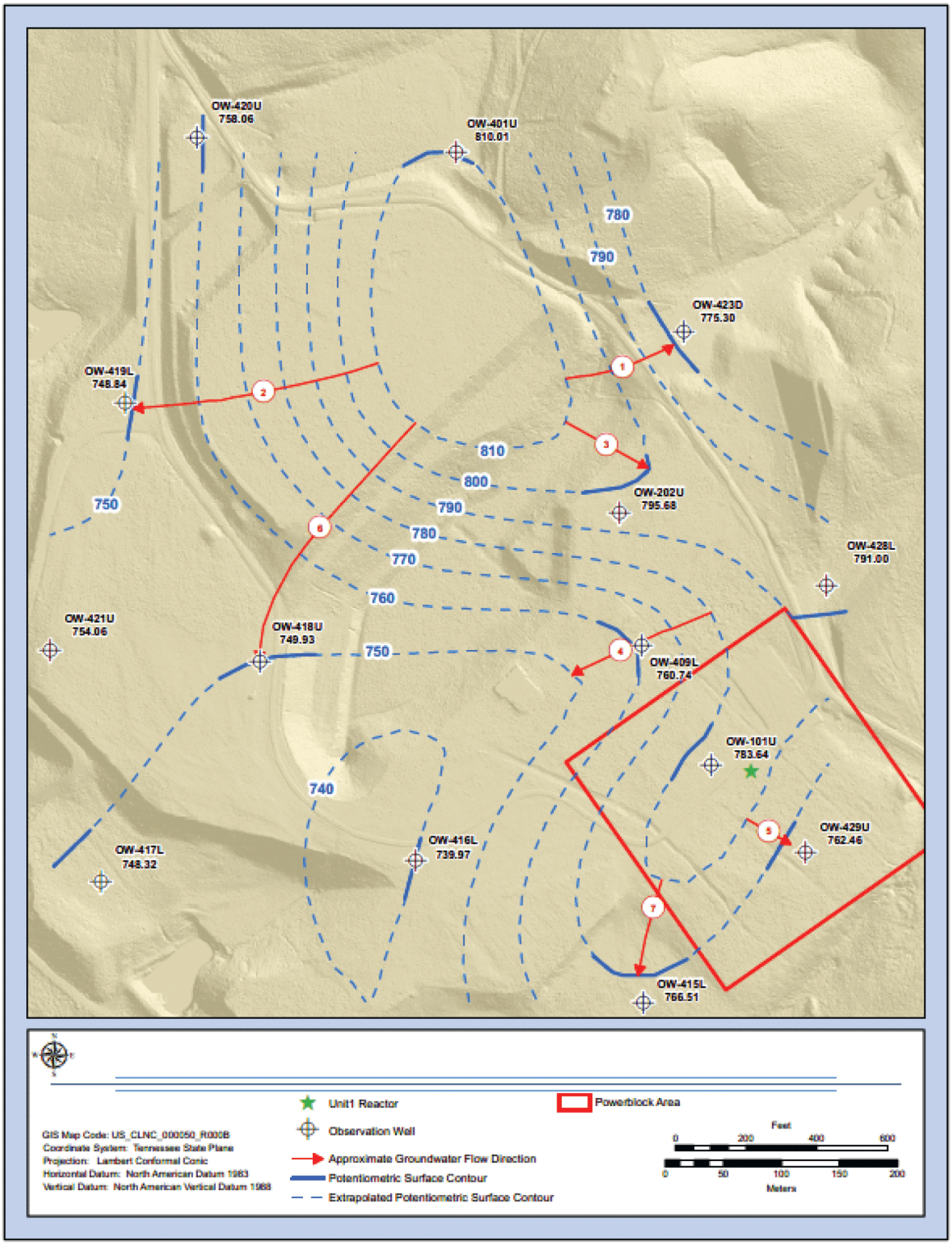

Figure 2.4.12-25R Potentiometric Surface Map for November 4, 2014

Replace Figure 2.4.12-26 with the following:

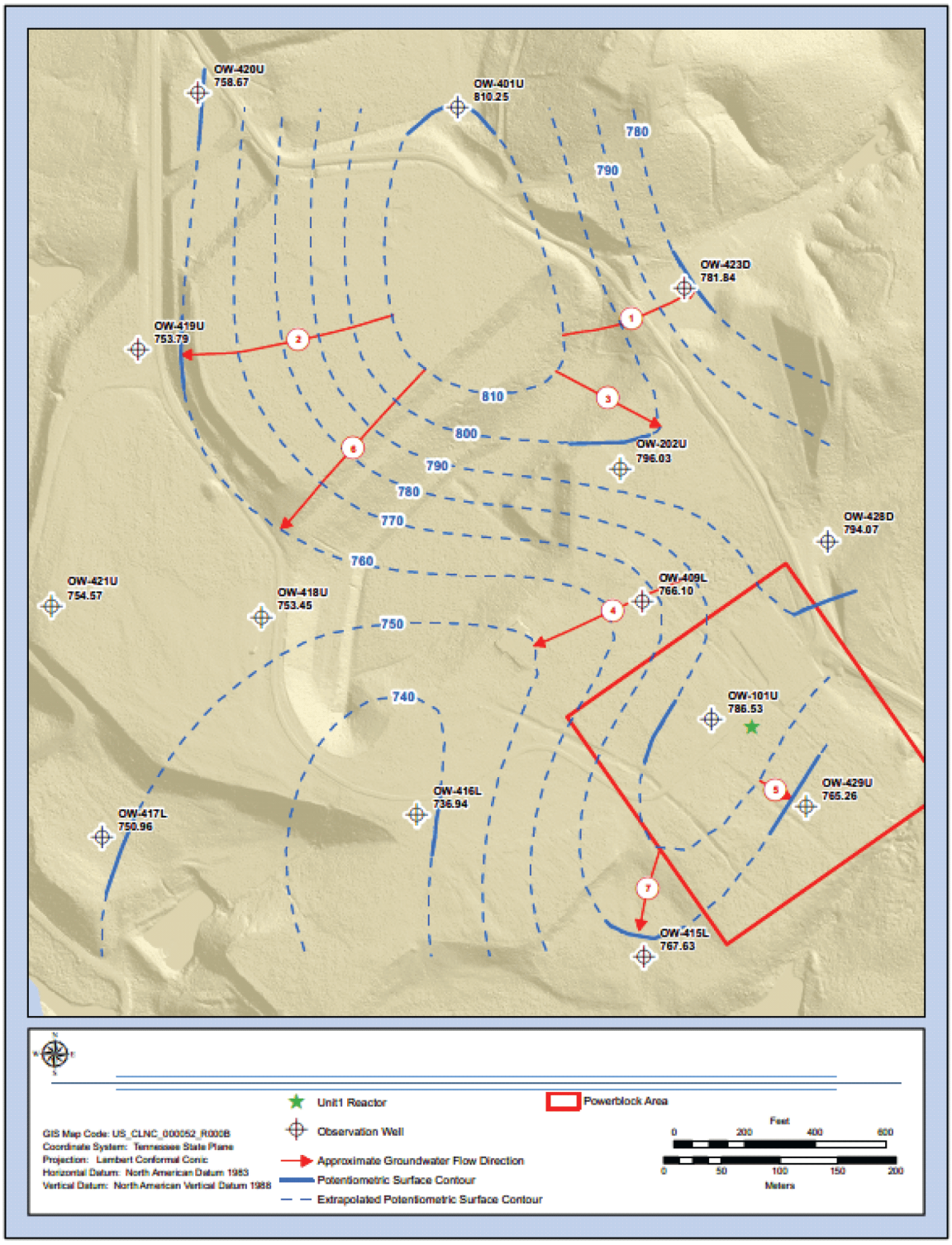

Figure 2.4.12-26R Potentiometric Surface Map for February 12, 2015

Replace Figure 2.4.12-27 with the following:

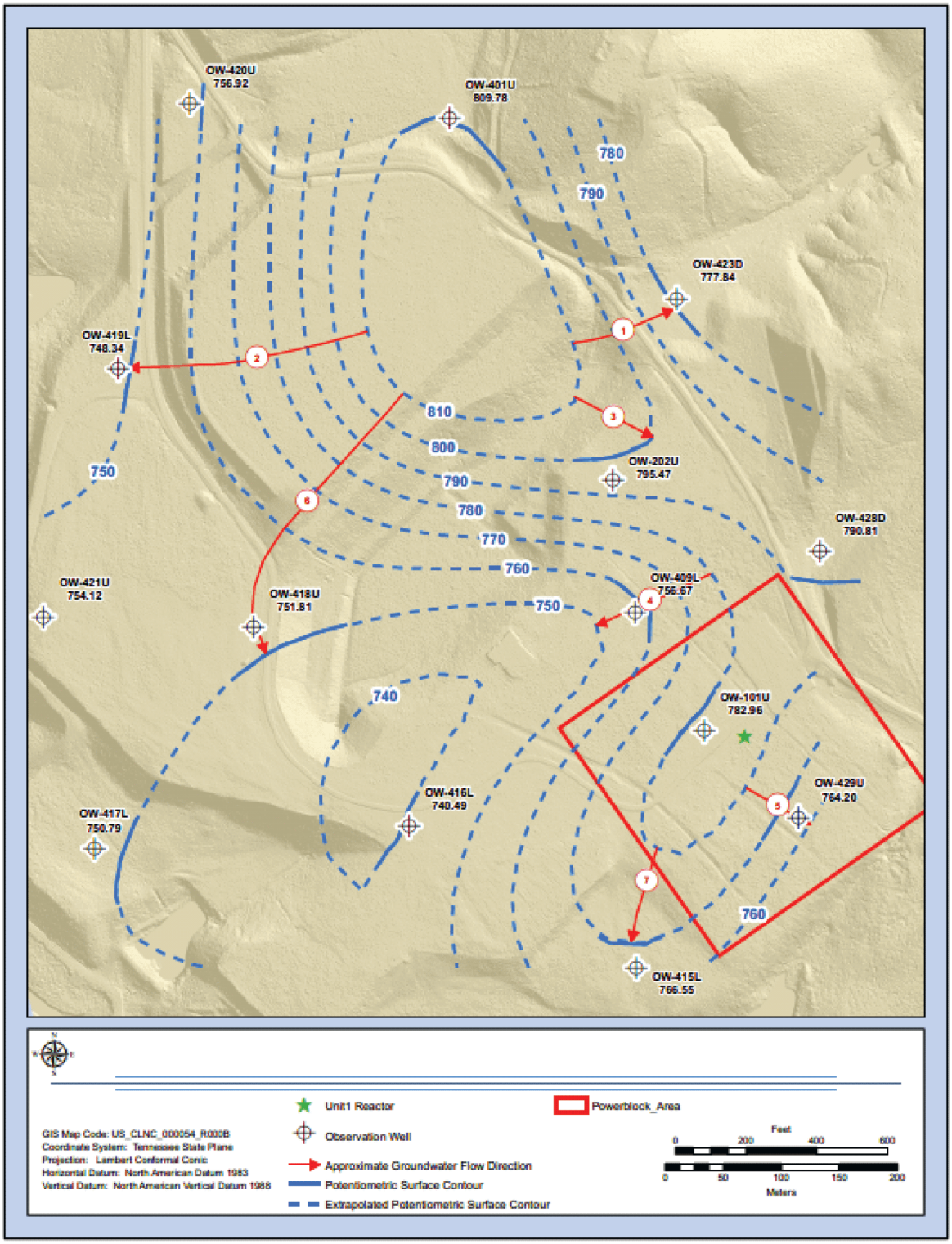

Figure 2.4.12-27R Potentiometric Surface Map for May 19, 2015

Replace Figure 2.4.12-28 with the following:

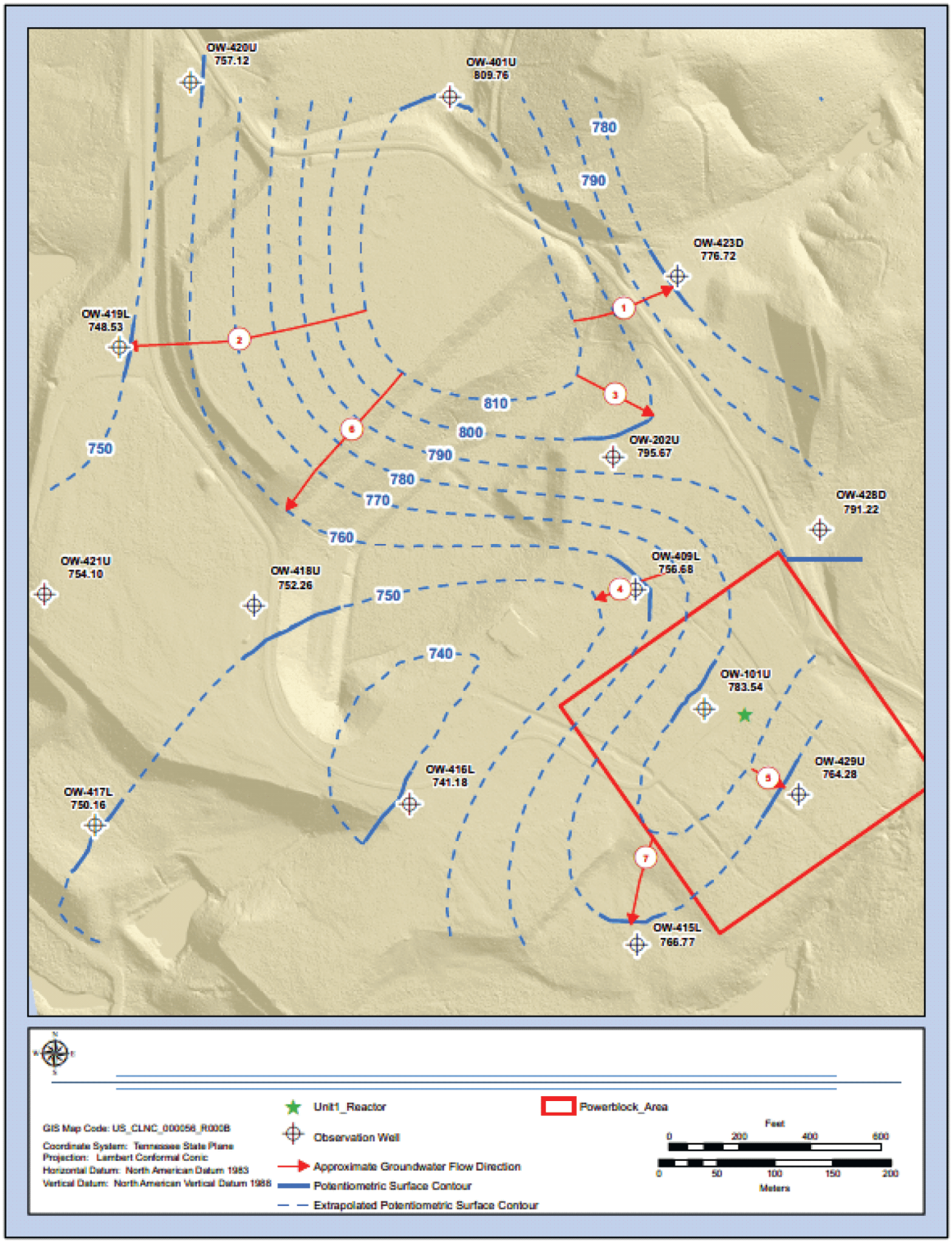

Figure 2.4.12-28R Potentiometric Surface Map for August 10, 2015

Replace Figure 2.4.12-40 with the following:

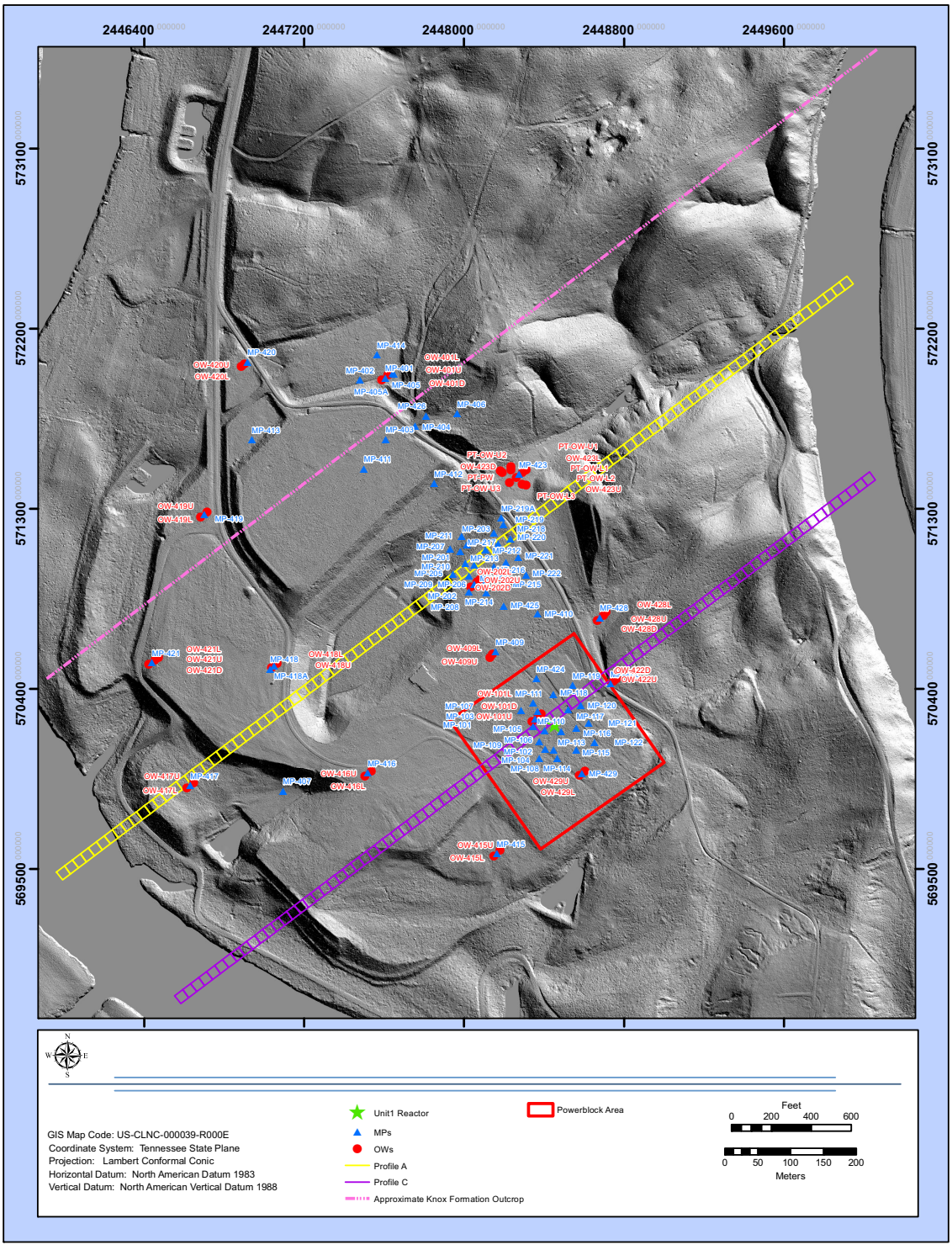

Figure 2.4.12-40R Plan View of the Profile Model with Grids

Replace Figure 2.4.12-41 with the following:

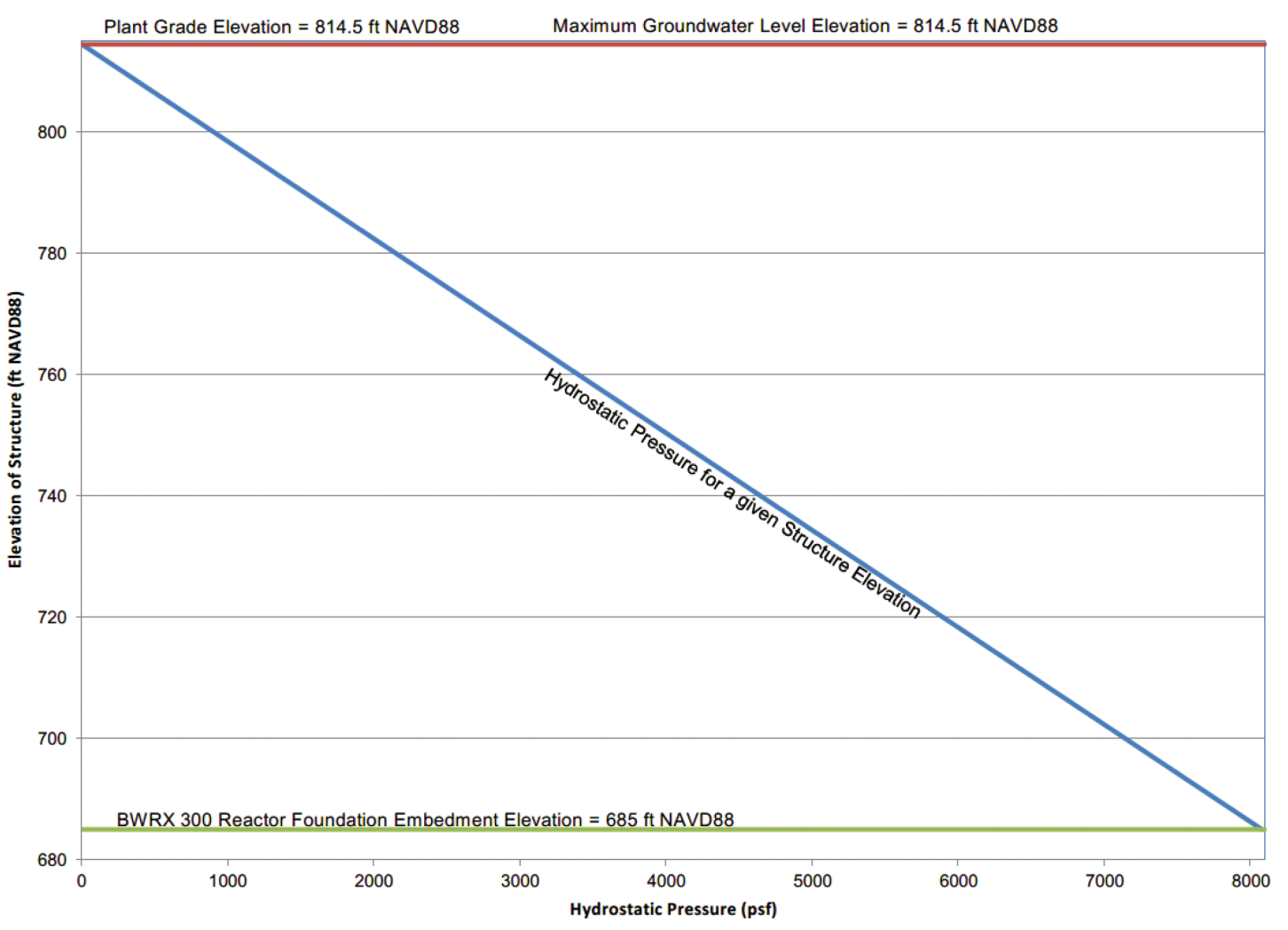

Figure 2.4.12-41R Maximum Hydrostatic Pressure

Appendix 2.4.12B Aquifer Pumping Test Results

2.4.12B Aquifer Pumping Test Results

2.4.12B.1.1 Purpose

Replace Figure 2.4.12B-1 with Figure 2.4.12B-1R in the first paragraph of ESPA SSAR Section 2.4.12B.1.1.

2.4.12B.1.2 Hydrogeologic Setting

Replace the fourth sentence of the second paragraph of ESPA SSAR Section 2.4.12B.1.2 with the following:

Measurements from the 2013 CRN Site subsurface investigation indicate a primary discontinuity set oriented N60°E 59°NW and a secondary set oriented N60°E 38°SE.

2.4.12B.2.1 Observation Wells

Replace Figure 2.4.12B-1 with Figure 2.4.12B-1R of ESPA SSAR Section 2.4.12B.2.1.

2.4.12B.4. Results and Discussion

Replace CRN Site Investigation with 2013 CRN Site Investigation in the third paragraph of ESPA SSAR Section 2.4.12B.2.1.

Replace Figure 2.4.12B-1 with the following:

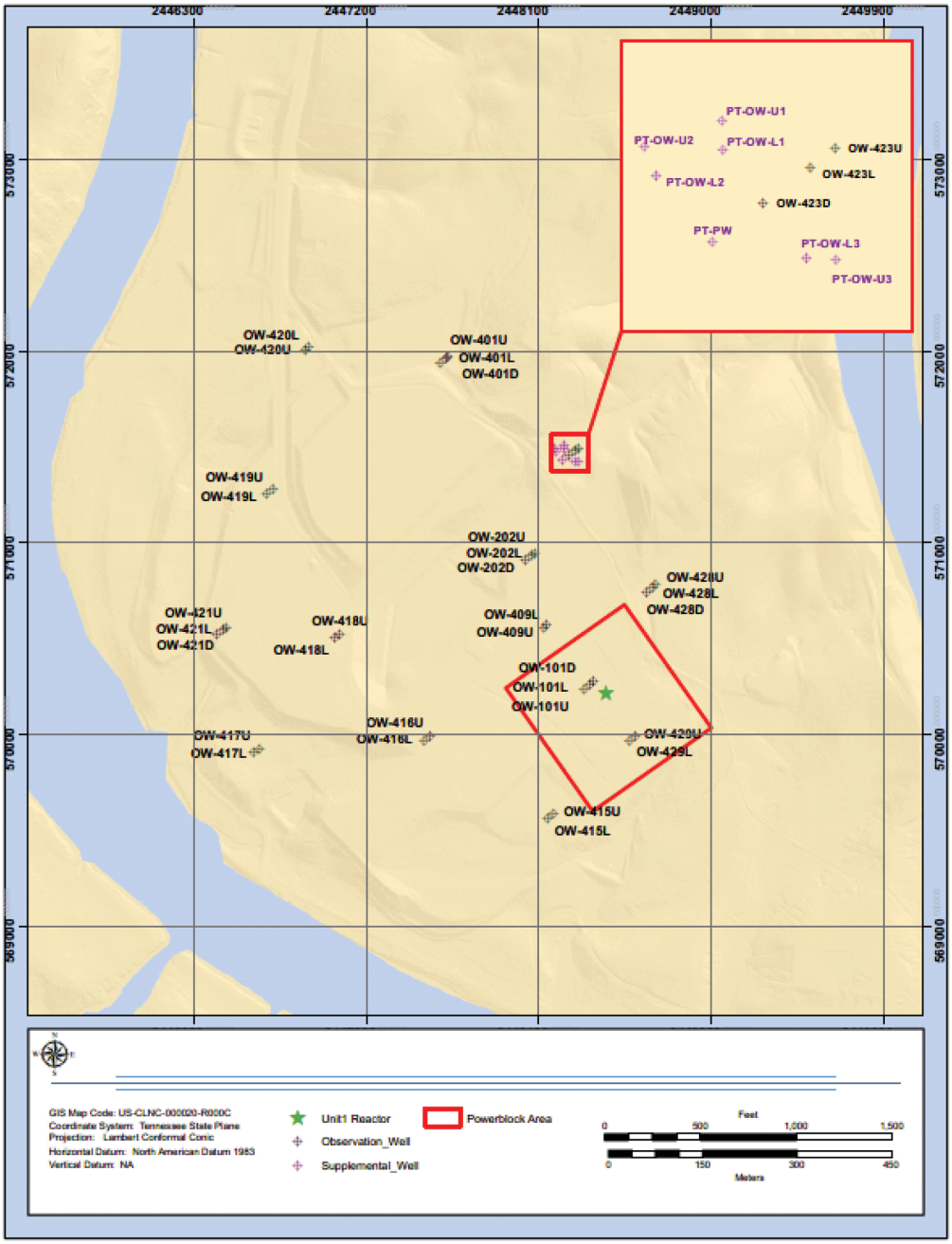

Figure 2.4.12B-1R Location of the Site Observation Wells

Appendix 2.4.12C Groundwater Flow Model for the Clinch River Nuclear Site

2.4.12C Groundwater Flow Model for the Clinch River Nuclear Site

2.4.12C.1 Objective and Scope

Replace the first paragraph of ESPA SSAR Section 2.4.12C.1 with the following:

The CRN Site groundwater flow model for the CRN Site in Roane County, Tennessee is prepared to evaluate groundwater levels during the operation of the BWRX-300 SMR constructed at the site. The primary objective of the groundwater model is to assess the post-construction maximum groundwater levels for BWRX-300 SMR power block area.

Replace the third bullet of the second paragraph of ESPA SSAR Section 2.4.12C.1 with the following:

- Developing a pre-construction site model to simulate maximum water level at and around the BWRX-300 SMR power block area and compare against the observed maximum water level

Replace the fifth bullet of the second paragraph of ESPA SSAR Section 2.4.12C.1 with the following:

- Performing predictive post-construction simulations to determine maximum groundwater levels at and around the BWRX-300 SMR power block area

2.4.12C.2.2 Regional Hydrogeologic Setting

Replace the last sentence of the third paragraph of ESPA SSAR Section 2.4.12C.2.2 with the following:

This land disturbance affected the stormflow zone at the CRN Site, and it is likely to have been replaced by the fill materials encountered in the more recently drilled CRN geologic borings in 2013.

2.4.12C.3 Site-Specific Information

Replace the first paragraph of ESPA SSAR Section 2.4.12C.3 with the following:

CRN Site-specific information was obtained primarily from the 2013 CRN Site subsurface investigation program conducted between July 2013 and March 2014, documented in Reference 2.4.12C-1. Previous site CRBRP subsurface investigation data (Reference 2.4.12C-24) were also reviewed; the geologic and hydrogeologic data were similar to that of the 2013 CRN Site subsurface investigation. The 2013 CRN Site subsurface investigation indicates that the lithologic units at the CRN Site are generally composed of man-made fill materials, residual soil, weathered bedrock, fractured bedrock, and competent bedrock. The thickness of the fill materials varies spatially at the site from less than 1 ft to approximately 50 ft

(Reference 2.4.12C-1). Thickness of residual soil varies from 0 to approximately 50 ft. Weathered bedrock units also vary spatially from 0 to approximately 21 ft in thickness.

Replace Figure 2.4.12-4 with Figure 2.4.12-4R in the fourth and sixth sentences of the third paragraph of ESPA SSAR Section 2.4.12C.3.

2.4.12C.3.1 Groundwater Level Measurements

Replace the first paragraph of ESPA SSAR Section 2.4.12C.3.1 with the following:

Groundwater level measurements were taken from 34 groundwater observation wells of which six locations consisted of three-well clusters, and eight locations consisted of two-well clusters. The locations of the wells are shown in Figure 2.4.12C-4R. The footprint of the BWRX-300 power block area is shown in Figure 2.4.12C-4R.

2.4.12C.3.2 Groundwater Level and Hydraulic Gradient Analysis

Replace Figure 2.4.12C-5 with Figure 2.4.12C-5R in the first bullet of the first paragraph of ESPA SSAR Section 2.4.12C.3.2.

Replace the third bullet of the first paragraph of ESPA SSAR Section 2.4.12C.3.2 with the following:

- The maximum water level elevation measured during the monitoring period (September 24, 2013 to March 16, 2014) in the general area of the BWRX-300 SMR power block area was 800.3 ft NAVD88 at observation well OW-202U.

2.4.12C.3.3 Packer Test Analysis

Replace Figure 2.4.12C-6 with Figure 2.4.12C-6R in the first sentence of the first paragraph of ESPA SSAR Section 2.4.12C.3.3.

2.4.12C.3.4 Slug Test Analysis

Replace Figure 2.4.12C-4 with Figure 2.4.12C-4R in the first sentence of the first paragraph of ESPA SSAR Section 2.4.12C.3.4.

2.4.12C.3.5 Aquifer Pumping Test Analysis

Replace the first paragraph of ESPA SSAR Section 2.4.12C.3.5 with the following:

An aquifer pumping test was conducted in the vicinity of OW-423 U/L/D well series and included seven supplemental wells (Figure 2.4.12C-4R). Transmissivity, storage coefficient, and hydraulic conductivity values were estimated from the drawdown and recovery plots of the pumping test observation wells using the Hantush-Jacob solution for a leaky aquifer with partial penetration and no aquitard storage.

2.4.12C.3.6 Hydrogeologic Data - Literature Information

Replace the first sentence of the fourth paragraph of ESPA SSAR Section 2.4.12C.3.6 with the following:

CRBRP packer testing, which utilized larger test intervals than that performed for the 2013 CRN Site subsurface investigation, derived hydraulic conductivity values similar to those determined at the ORR.

Replace the first sentence of the fifth paragraph of ESPA SSAR Section 2.4.12C.3.6 with the following:

CRBRP on-site observation water level measurements indicated up to 25 ft fluctuation in water levels.

2.4.12C.3.7 Recharge Data – Literature Information

Replace the first sentence of the second bullet of the third paragraph of ESPA SSAR Section 2.4.12C.3.7 with the following:

- Bailey (Reference 2.4.12C-2) investigated the distribution of recharge in a cross-sectional, finite-difference model of the waste burial grounds at the ORR. Based on this model, most of the recharge occurs on the ridges of Pine Ridge (approximately two miles north of the BWRX-300 SMR power block area) and Chestnut Ridge (approximately one mile north of the BWRX-300 SMR power block area).

2.4.12C.4 Conceptual Site Model and Assumptions

Replace the second sentence in #1 of ESPA SSAR Section 2.4.12C.4 with the following:

The fill materials identified from the 2013 CRN Site subsurface investigation (Reference 2.4.12C-1) are likely to be from the excavation work undertaken for the defunct CRBRP (Reference 2.4.12C-24).

Replace the third sentence in #4 of ESPA SSAR Section 2.4.12C.4 with the following:

The 2013 CRN Site subsurface investigation indicates a primary discontinuity set oriented at a strike of N60°E and dip of 59°NW and a secondary set oriented N60°E and dip of 38°SE.

Replace #5 in the fourth paragraph of ESPA SSAR Section 2.4.12C.4 with the following:

5. The fill, residual soil, and weathered bedrock units (regolith) along with the shallower depths of the competent bedrock units act as a porous medium (Figure 2.4.12C-12R (Sheet 1 of 11) and Figures 2.4.12C-12 (Sheets 2 to 11)). The shear zone depicted on Figures 2.4.12C-12 (Sheets 6 to 10) represents a laterally continuous interval of calcite-filled and cemented fractures in the lower Eidson. As these fractures are completely healed, they do not represent a zone of secondary porosity that would be conducive to preferential groundwater flow. Webster and Bradley (Reference 2.4.12C-37) state that groundwater flow in the regolith is characteristic of both porous media as well as fracture controlled flow at the nearby ORNL Site.

2.4.12C.5 Numerical Model

Replace the first two paragraphs of ESPA SSAR Section 2.4.12C.5 with the following:

Figure 2.4.12C-13R presents a site layout of Profile A and Profile C and the power block area of Unit 1 of the GVH BWRX-300. Geotechnical borings (referred to as MPs) are concentrated within the power block area and centered near borings MP-101 and MP-202 (Figure 2.4.12C-13R). Based on geology observed in the geotechnical borings, most of the footprint of the power block area lies within the Chickamauga Group of rocks (interbedded siltstone and limestone), whereas the Knox Group (comprised mostly of dolomite) outcrops in the northern extremities of the power block area as indicated on Figure 2.4.12C-13R. Within the power block area, the Knox is significantly deeper. For example, at the MP-202 and MP-101 locations, it is encountered at a depth of approximately 450 and 1000 ft bgs, respectively, while at MP-423 and MP-426 the Knox is first encountered at a depth of approximately 30 to 50 ft bgs.

Two, two-dimensional, vertical profile, groundwater models were developed along the strike of the bedding planes (approximately N52°E) for the CRN Site. Figure 2.4.12C-13R shows the plan view of the profile models. Profile A intersects and lies close to the MP-202 series of geotechnical borings while Profile C (Profile B is not used) intersects the MP-101 series of geotechnical borings and Unit 1 of the BWRX-300 power block area (Figure 2.4.12C-13R). Both profiles fall along a three-cluster groundwater monitoring well series (OW-202 U/L/D and OW-101 U/L/D). The geologic formations present along the two profiles are different; Profile A mostly intersects the Fleanor Shale Member of the Lincolnshire Formation, while Profile C mostly intersects the Benbolt Formation. The hydraulic conductivity distribution along these lines, determined from the slug and packer tests, is also different (Table 2.4.12C-6). The combination of a concentration of geotechnical borings in two different clusters and the difference in hydraulic conductivity values led to the need to create two profile models to adequately represent the area.

Replace the eighth sentence of the third paragraph of ESPA SSAR Section 2.4.12C.5 with the following:

The profile models are used to provide conservative estimates of the maximum groundwater heads below SC1 structures.

Replace the last paragraph of ESPA SSAR Section 2.4.12C.5 with the following:

The groundwater model was developed using the pre- and post-processor groundwater modeling software, Groundwater Vistas, Version 8.19, Build 4. The numerical code used to simulate the profile models is MODFLOW-SURFACT, Version 3.0, as implemented in Groundwater Vistas (Reference 2.4.12C-11).

2.4.12C.5.2 Model Layers

Replace the first sentence of the first paragraph of ESPA SSAR Section 2.4.12C.5.2 with the following:

Model layers thicknesses for the vertical profiles (Profiles A and C) were based on the 2013 CRN Site subsurface investigation geotechnical boring logs (Reference 2.4.12C-1).

Delete the last sentence of the first paragraph of ESPA SSAR Section 2.4.12C.5.2.

2.4.12C.6.2 Groundwater Budget

Replace the seventh sentence of the first paragraph of ESPA SSAR Section 2.4.12C.6.2 with the following:

During the 2013 CRN Site subsurface field investigation, seepage was observed in some outcrops in the low lying CRBRP disturbed area.

2.4.12C.6.3.2 Variable Recharge Rates

Replace the fifth sentence of the second paragraph of ESPA SSAR Section 2.4.12C.6.3.2 with the following:

The delineation of the recharge zone is arbitrary; however, the distribution was also based on site-specific observations during the 2013 CRN Site subsurface investigation at the CRN Site.

2.4.12C.7 Post-Construction Modeling

Replace ESPA SSAR Section 2.4.12C.7 with the following:

Post-construction modeling is performed to estimate the response of groundwater levels to the inclusion of the following BWRX-300 specific structures in the model: radwaste building, reactor building, and turbine building. A single BWRX-300 SMR developed by GVH is planned for the

CRN Site and included as Unit 1 in a Construction Permit Application (Figure 2.4.12C-23R (Sheet 1 of 2)). The excavation depth of the reactor building was set to 129.5 feet below grade (Figure 2.4.12C-23R (Sheet 2 of 2)). There will be some backfill with lean concrete between the excavation depth and the bottom of the basemat. Post-construction model was only done for Profile C because it coincides with the location of the Power Block structures (Figure 2.4.12C-24R) and thus post-construction simulations for Profile A were not performed because the Construction Permit Application is being submitted for only one unit.

2.4.12C.7.1 Modifications to Pre-Construction Models

Replace ESPA SSAR Section 2.4.12C.7.1 with the following:

The structure of the model along with hydraulic conductivity distributions for subsurface layers remained the same as the pre-construction model; however, some nominal changes were made to the Profile C model to include the representative structures and placement of construction fill to raise the site grade (Figure 2.4.12C-26R). The following describes the changes between the pre- and post-construction models:

- The site grade elevation within the power block, which includes the radwaste, reactor, control and turbine buildings, was set to an elevation of 814.5 ft NAVD88 (Figure 2.4.12C-23R (Sheet 1 of 2), Figure 2.4.12C-23R (Sheet 2 of 2) and Figure 2.4.12C-24R). The site grade outside of the power block was adjusted to reflect the elevations shown in Figure 2.4.12C-23R (Sheet 1 of 2).
- Construction fill was included in areas where the existing site grade was raised to the design site grade. The hydraulic conductivity was calculated using the empirical formula of Reference 2.4.12C-8, which requires the D_{10} and D_{60} grain size values. These grain-size values were obtained from the Tennessee Department of Transportation (TDOT) specification for Type A, Grading D aggregate. Figure 2.4.12C-27R shows the grain-size distribution (lower bound, upper bound, average) for TDOT Type A, Grading D aggregate. Table 2.4.12C-11R provides the calculated hydraulic conductivity values based on the upper, lower, and average D_{10} and D_{60} for the specified grain-size distributions. The “C2” values for D_{10} and D_{60} shown in Figure 2.4.12C-27R resulted in the lowest calculated hydraulic conductivity value of 1.5 ft/day (5.3x10-4 cm/s). To conservatively estimate maximum groundwater heads in the power block area, a construction fill hydraulic conductivity of 1.5 ft/day was assigned in the post-construction model.
- The site grade in the power block area corresponds to an elevation of 814.5 ft NAVD88. Impermeable structures embedded below the water table (e.g., reactor building) are represented as inactive (no flow) cells in the Profile C model (Table 2.4.12C-26R). Those buildings with shallow embedment depths do not have any appreciable impact on groundwater flow and are not represented.
- A model layer was added below the bottom of the reactor building (layer 8) to allow the model to calculate the maximum head imposed at the base of the foundation. The hydraulic conductivity for this layer was assigned the same value as the pre-construction model layer 7. Figure 2.4.12C-26R show hydraulic conductivity distributions for the layers in the post-construction model cross-section for Profile C.

- The width of the power block area in the cross-sectional model approximates the width of the power block in the site layout drawing (Figure 2.4.12C-24R).
- The bottom of the reactor building was set to an elevation of 685 ft NAVD88, which is 129.5 ft below the grade elevation of 814.5 ft NAVD88.
- The bottom of the turbine building foundation was set to an elevation of 800 ft NAVD88, which is 14.5 ft below the grade elevation of 814.5 ft NAVD88. Existing fill, soil, and weathered rock beneath the turbine building is replaced by construction fill. Although the Profile C model does not exactly pass through the turbine building (as shown in Figure 2.4.12C-25R), nevertheless the turbine building was conservatively represented in the Profile C post-construction model.

2.4.12C.7.2 Results

Replace ESPA SSAR Section 2.4.12C.7.2 with the following:

The objective of the groundwater modeling as outlined in Subsection 2.4.12C.1 is to determine post-construction maximum groundwater head in the power block area.

The maximum piezometric head underneath the Unit 1, BWRX-300 reactor building is 814.5 ft NAVD88, which is the same as the power block plant site grade elevation (Table 2.4.12C-12R and Figures 2.4.12C-28R and 2.4.12C-29R). Figure 2.4.12C-29R depicts groundwater flow away from the power block.

NEDO-33914-A, “BWRX-300 Advanced Civil Construction and Design Approach,” provides a generic conservative groundwater level located at plant grade (Reference 2.4.12C-39). This groundwater level is used in stability calculations to account for the buoyancy force.

The maximum piezometric head underneath the turbine building foundation is 814.2 ft NAVD88, which is 0.3 ft below the plant site grade elevation (Table 2.4.12C-12R and Figures 2.4.12C-28R and 2.4.12C-29R).

Sensitivity analysis was conducted by varying the construction fill hydraulic conductivity to 50.7 ft/day (as shown in Table 2.4.12C-11R for curve C1) but the piezometric head underneath the Unit 1, BWRX-300 reactor building remains unchanged, i.e., to the site grade of 814.5 ft NAVD88. Piezometric head underneath the turbine building was 811.5 ft NAVD88, i.e., decreased by 2.7 ft from the simulated head of 814.2 ft NAVD88 with a construction fill hydraulic conductivity of 1.5 ft/day.

2.4.12C.8 Conclusions

Replace ESPA SSAR Section 2.4.12C.8 with the following:

The following are concluded from the groundwater modeling results:

- Pre-construction groundwater model simulated heads closely matched the observed maximum groundwater heads at observation wells near the proposed location of the BWRX-300 SMR for this modeling exercise.
- Groundwater heads are sensitive to recharge in the groundwater model, similar to observed water level response due to precipitation during the subsurface investigations at the CRN Site. Higher recharge translates to higher water levels. A uniform recharge of 8.76 in./yr in the pre-construction model results in water levels close to the land surface at the power block area. This provides a conservative estimate with respect to the maximum water levels under predictive model simulations.
- Simulated piezometric head underneath the Unit 1 BWRX-300 reactor building foundation embedment is 814.5 ft NAVD88, which is at the plant grade. The piezometric head underneath the turbine building embedment foundation is 814.2 ft NAVD88.
- The calculated piezometric head under post-construction simulation underneath the Unit 1 BWRX-300 reactor building foundation reaches the plant grade elevation of 814.5 ft NAVD88 and meets the BWRX-300 generic groundwater heads as described in NEDO-33914-A, "BWRX-300 Advanced Civil Construction and Design Approach."

2.4.12C.9 References

Add the following Reference to ESPA SSAR Section 2.4.12C.9:

2.4.12C-39 NEDO-33914-A, BWRX-300 Advanced Civil Construction and Design Approach, GE Hitachi Nuclear Energy, Revision 2, June 2022.

Rename and Replace Table 2.4.12C-11 with the following:

Table 2.4.12C-11R Calculated Construction fill Hydraulic Conductivity Value from TDOT Type A Grading D Aggregate Sieve Analysis

Sieve Number	% Passing			
	Diameter (mm)	Lower	Upper	Average
2.5”	63			
2.0”	50			
1.5”	37.5	100	100	100
1”	25	85	100	92.5
3/4”	19	60	95	77.5
3/8”	9.5	50	80	65
No. 4	4.75	40	65	52.5
No. 16	1.18	20	40	30
No. 100	0.15	9.0	18	13.5
Curve	**D_{10} (m)**	**D_{60} (m)**	**K (m/s)**	**K (ft/d)**
Lower	0.0002	0.019	1.3E-04	36.7
Upper	0.00004	0.0037	7.2E-06	2.0
Average	0.00005	0.0072	9.7E-06	2.7
C1	0.0002	0.0037	1.8E-04	50.7
C2	0.00004	0.019	5.2E-06	1.5

Rename and Replace Table 2.4.12C-12 with the following:

Table 2.4.12C-12R Simulated Piezometric Head Below the Turbine Building and Reactor Building Foundation Embedment

Location	Piezometric Head Below BWRX-300 Reactor Building Excavation (ft, NAVD88)	Piezometric Head Below Turbine Building Excavation (ft, NAVD88)
A	-	814.0
B	-	814.2
C	814.5	-
D	814.5	-
E	814.5	-

Note: Refer to Figure 2.4.12C-28R for the location

Replace Figure 2.4.12C-4 with the following:

Figure 2.4.12C-4R Location of the Site Observation Wells

Replace Figure 2.4.12C-5 with the following:

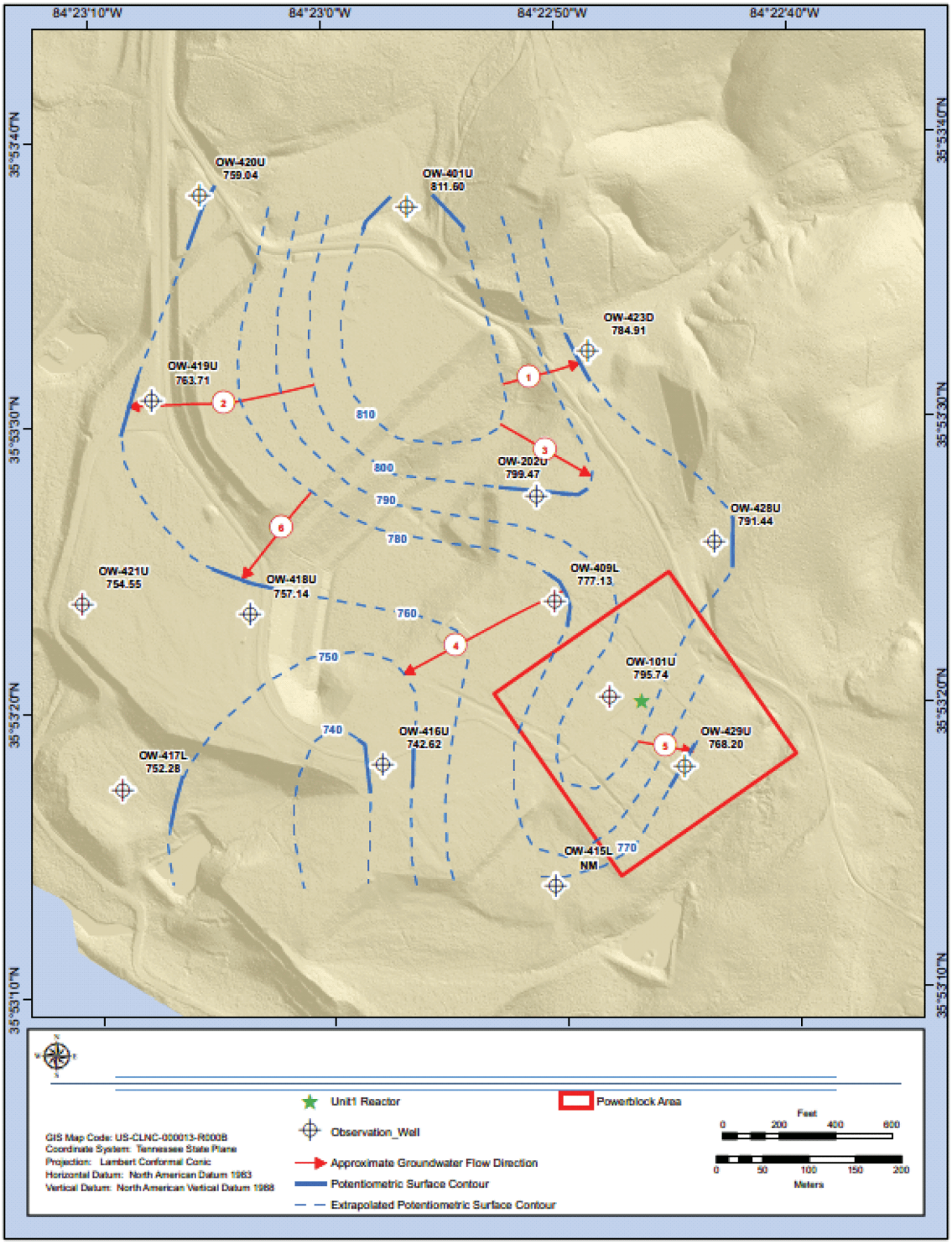

Figure 2.4.12C-5R Maximum Potentiometric Surface Map and Flow Direction for January 13, 2014

Replace Figure 2.4.12C-6 with the following:

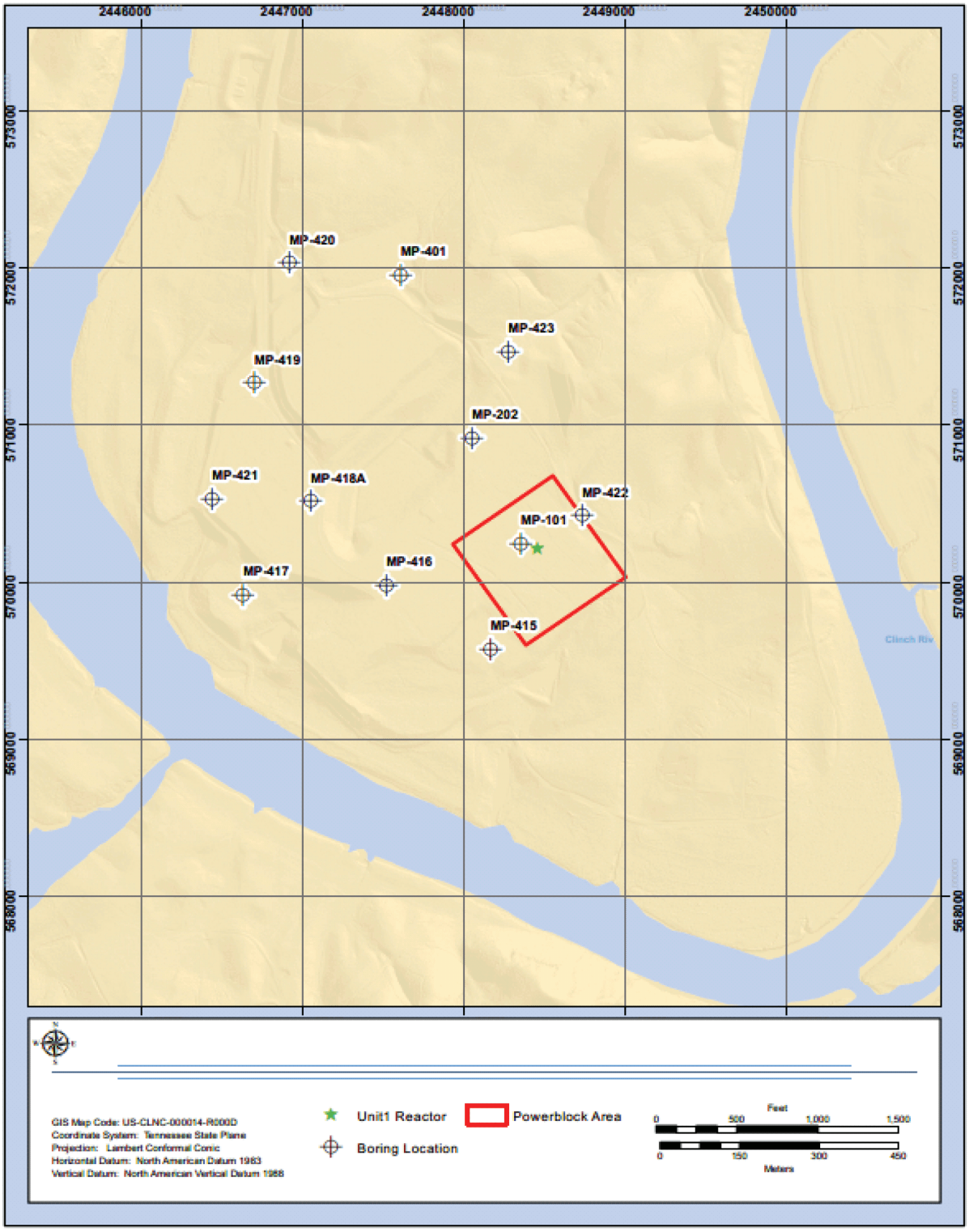

Figure 2.4.12C-6R Packer Test Locations

Replace Figure 2.4.12C-12 (Sheet 1 of 11) with the following:

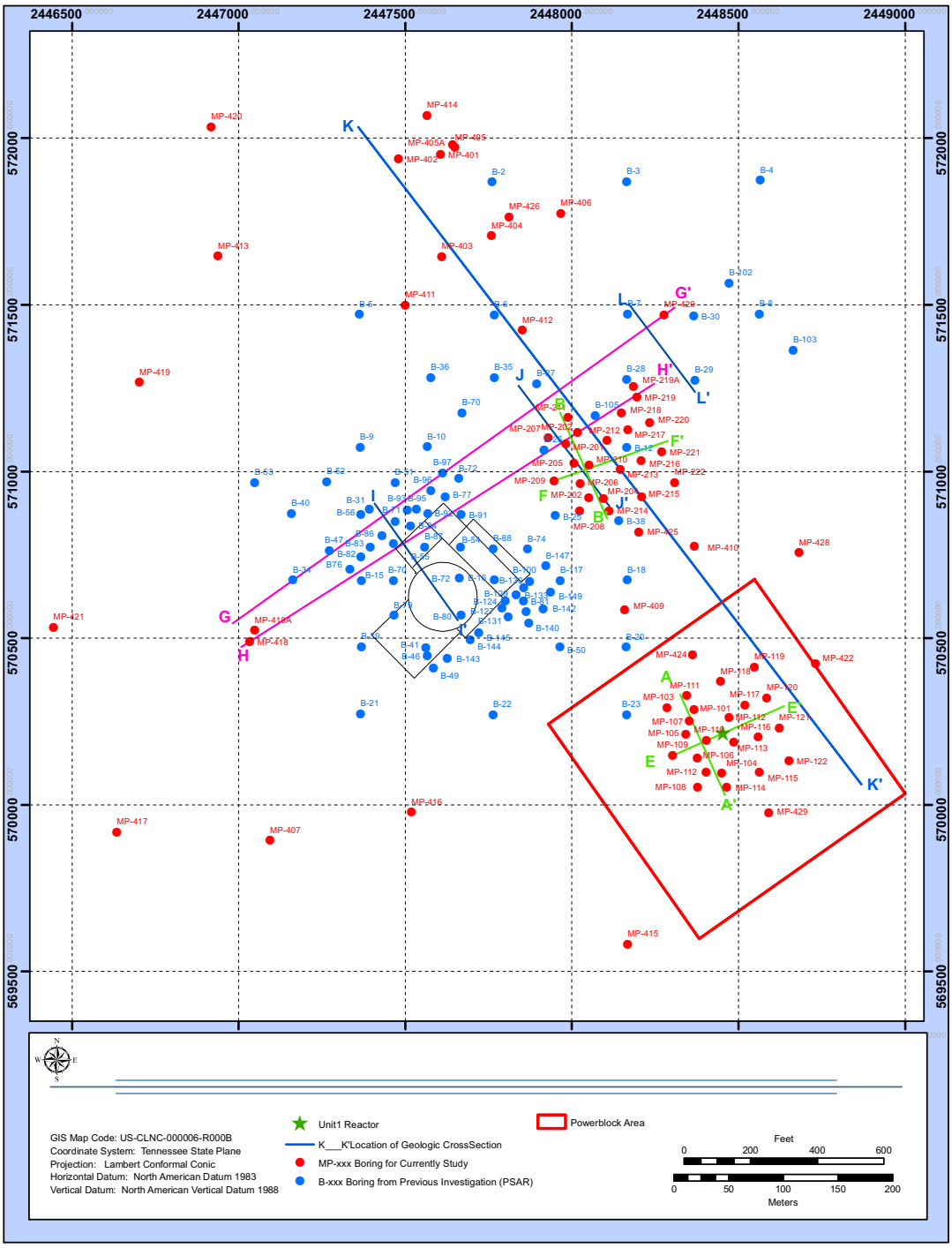

Figure 2.4.12C-12R (Sheet 1 of 11) Geologic Cross-Section Location Plan

Note: The black circular and rectangular area within the PPE Extent is the old Clinch River Breeder Reactor Project footprint which is a terminated project.

Rename and Replace Figure 2.4.12C-13 with the following:

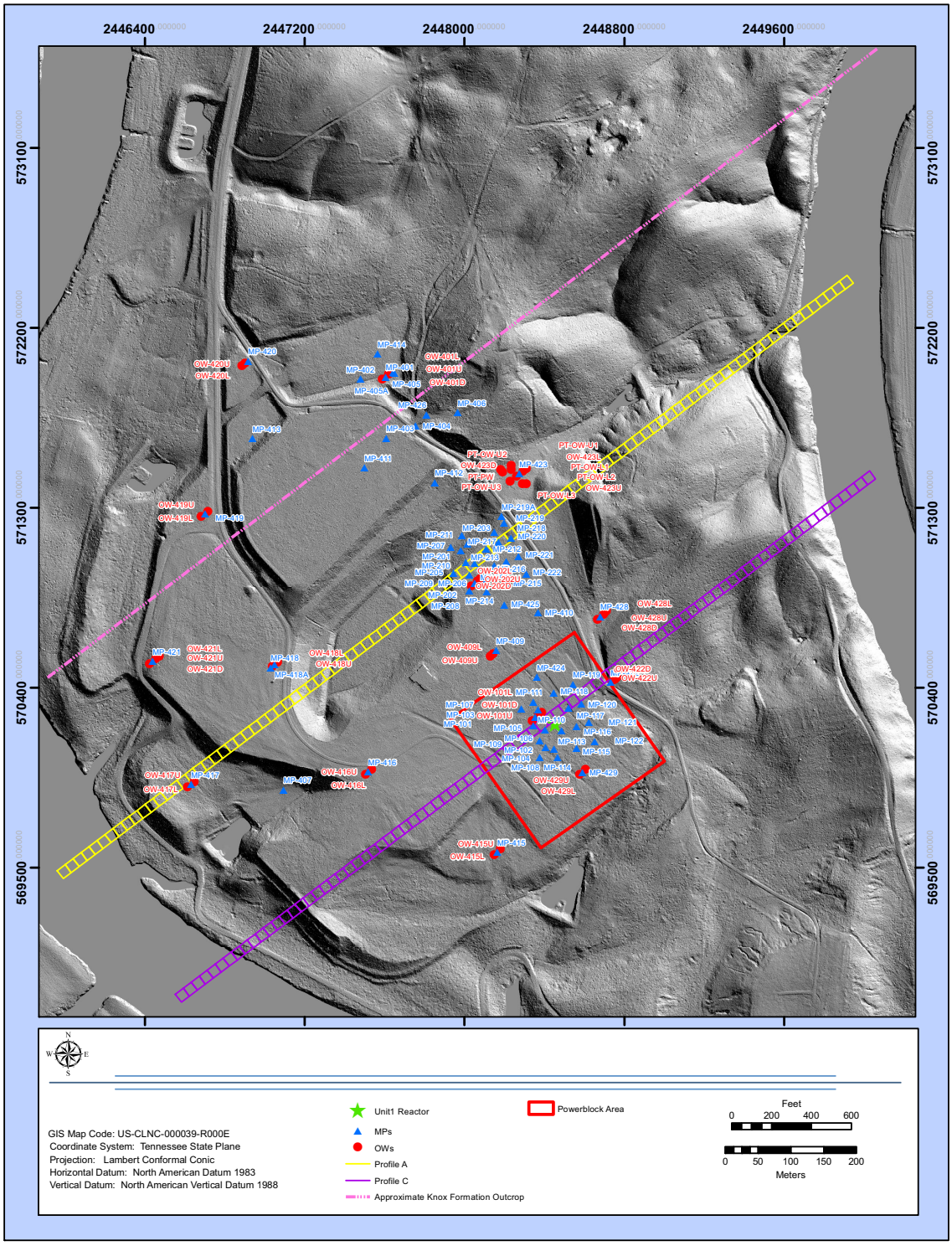

Figure 2.4.12C-13R Plan View of the Profile Models with Grids

Replace Figure 2.4.12C-23 with the following:

Figure 2.4.12C-23R (Sheet 1 of 2) Site Plan View of Overall Unit 1 Layout

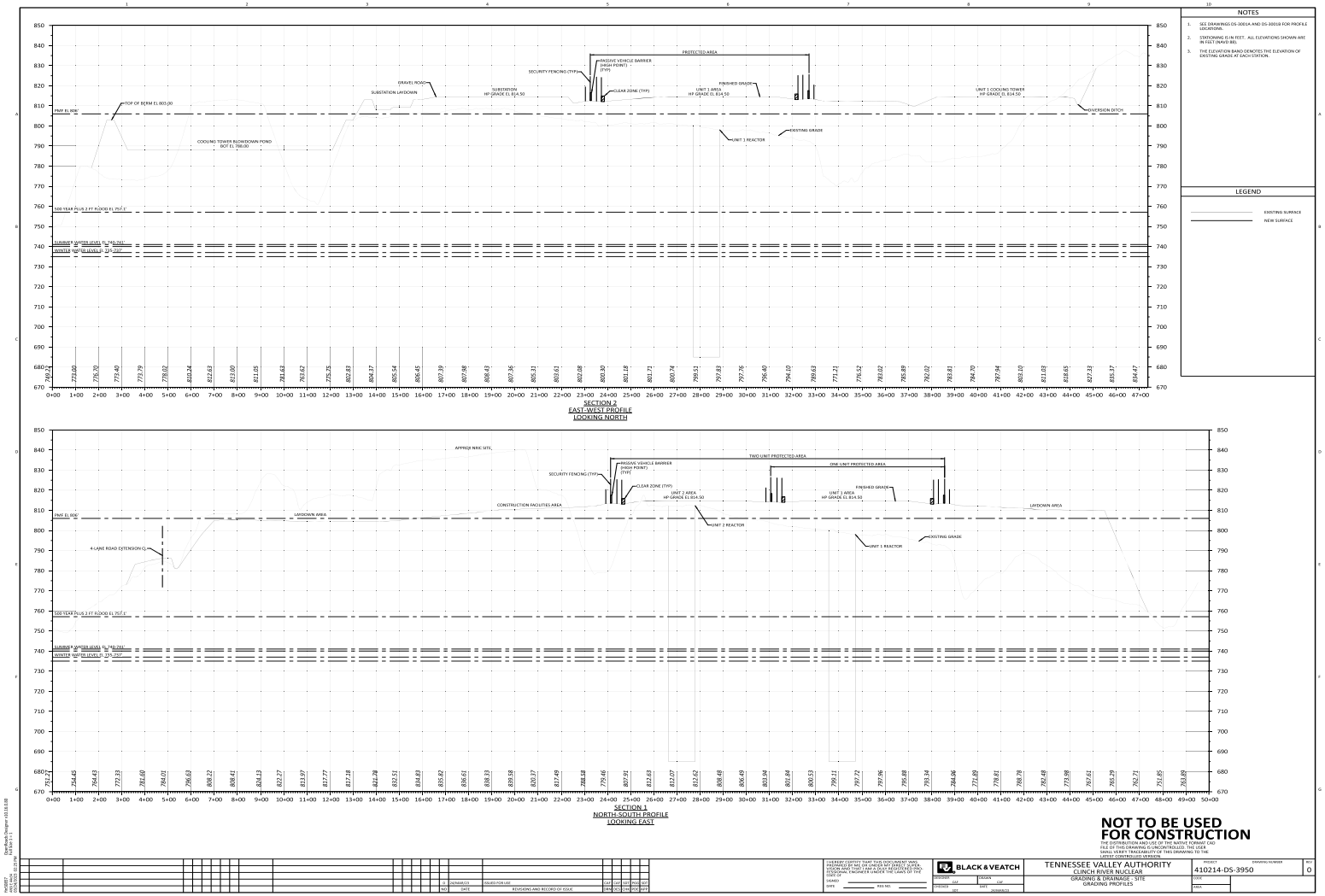

Figure 2.4.12C-23R (Sheet 2 of 2) Profile Along Section 2

Rename and Replace Figure 2.4.12C-24 with the following:

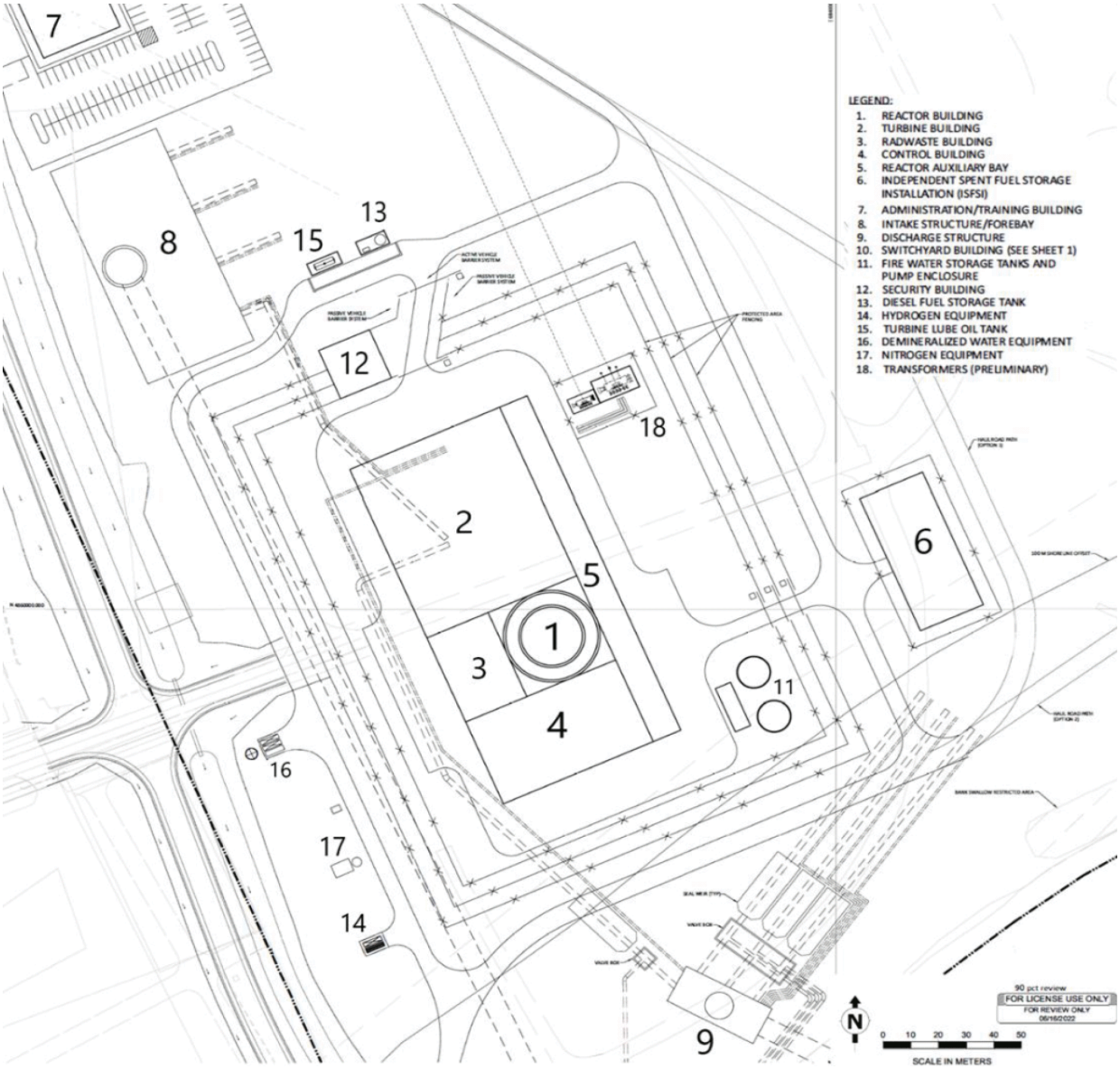

Figure 2.4.12C-24R Plan View Showing Locations of Buildings Supporting Unit 1 Reactor
Note: Does not reflect updated layout at the power block but the difference is minimal to have any significant impact to groundwater heads at the power block.

Rename and Replace Figure 2.4.12C-25 with the following:

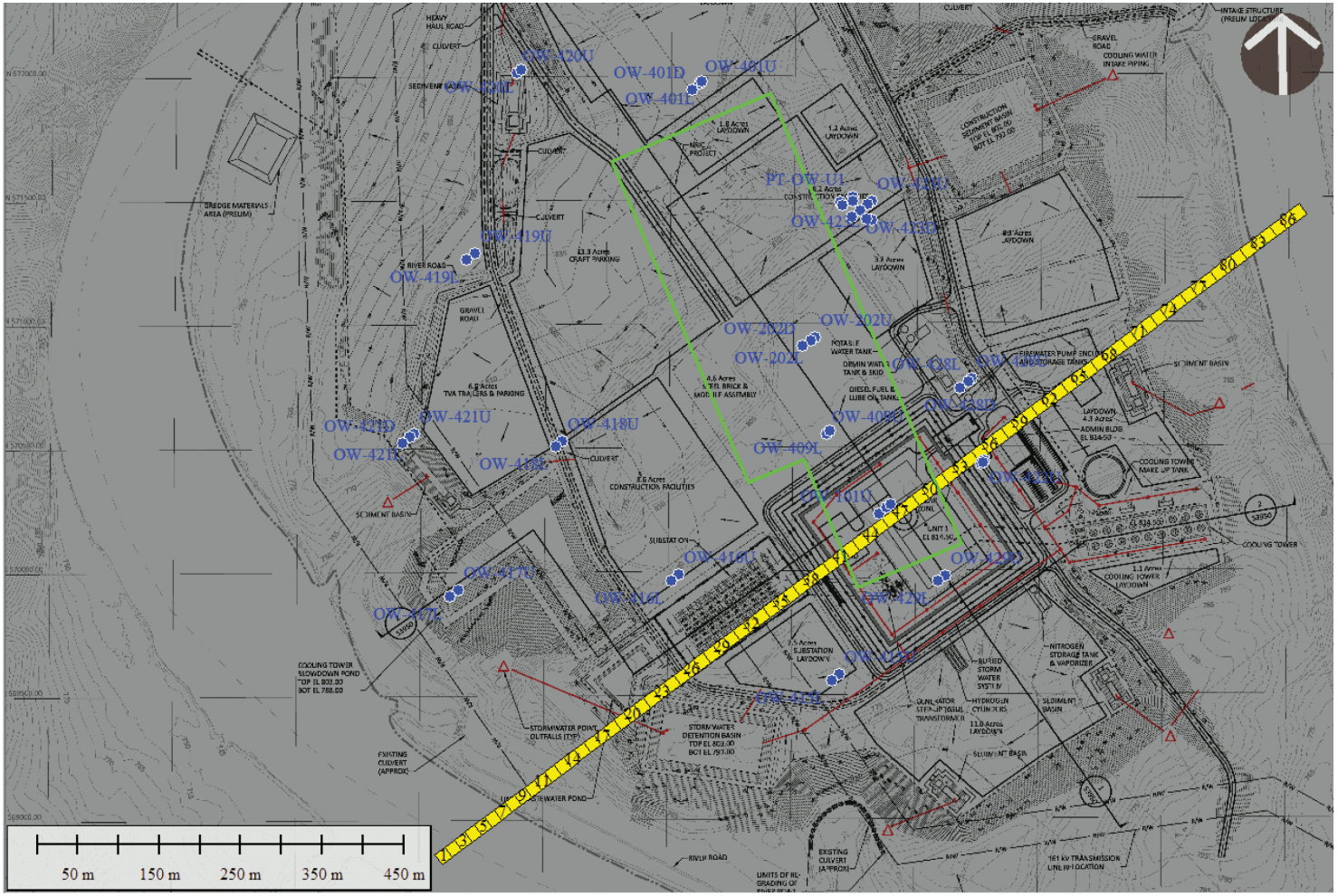

Figure 2.4.12C-25R Plan View Showing Locations of Grids Along Profile C

Note: Profile C is shown in yellow with grid numbers. The rectangular polygon shown in green is the power block area based on generic SMR layout from ESPA. The background layout shows the site plan for the BWRX-300 deployment at the CRN Site.

Rename and Replace Figure 2.4.12C-26 with the following:

Profile C Post-Construction Model

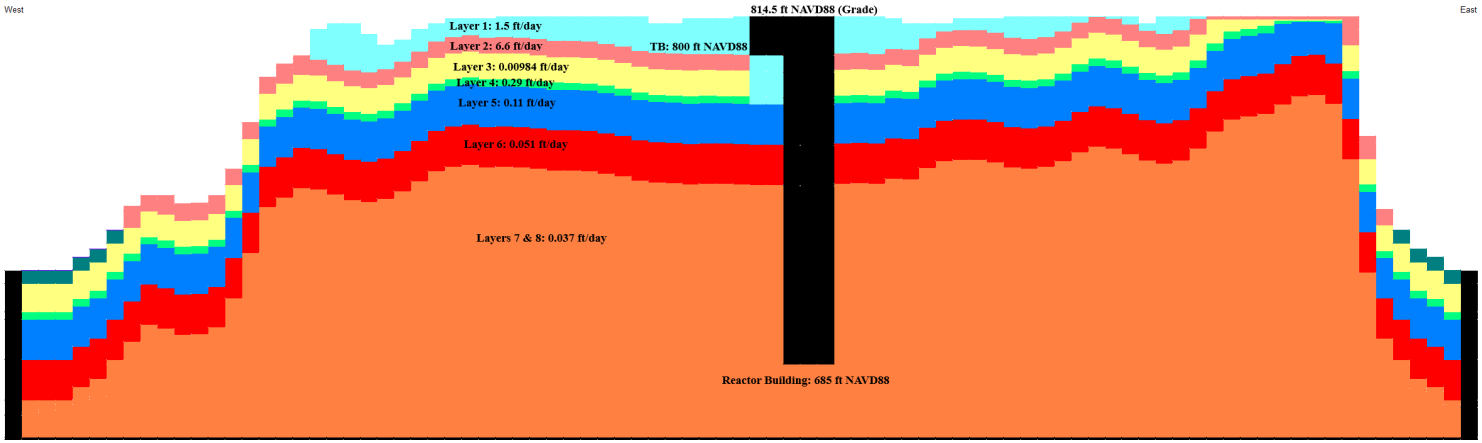

Figure 2.4.12C-26R Hydraulic Conductivity Distribution in Profile C Post-Construction Model

Note: TB is Turbine Building

Replace Figure 2.4.12C-27 with the following:

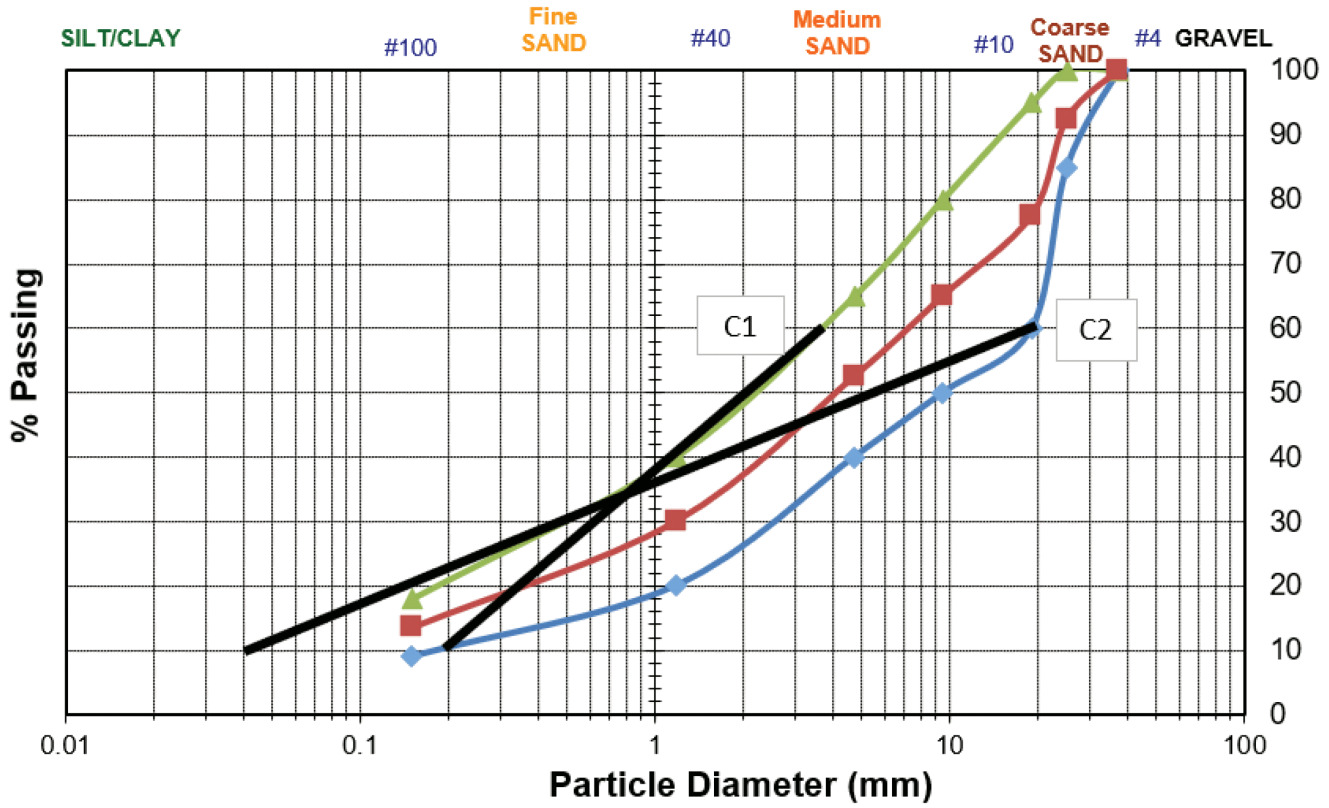

Figure 2.4.12C-27R Particle Diameter Versus Percent by Weight Passing Through Sieves for TDOT Type A, Grading D Aggregates

Note: blue diamonds and blue curve represents “lower curve” in Table 2.4.12C-11R; red squares and red curve represents “average curve” in Table 2.4.12C-11R; green triangles and green curve represents “upper curve” in Table 2.4.12C-11R; C1 is D_{60} particle diameter at “upper curve” and a straight line to D_{10} at “lower curve”; C2 is D_{60} particle diameter at “lower curve” and a straight line to D_{10} at approximate value of “upper curve”

Rename and Replace Figure 2.4.12C-28 with the following:

Profile C

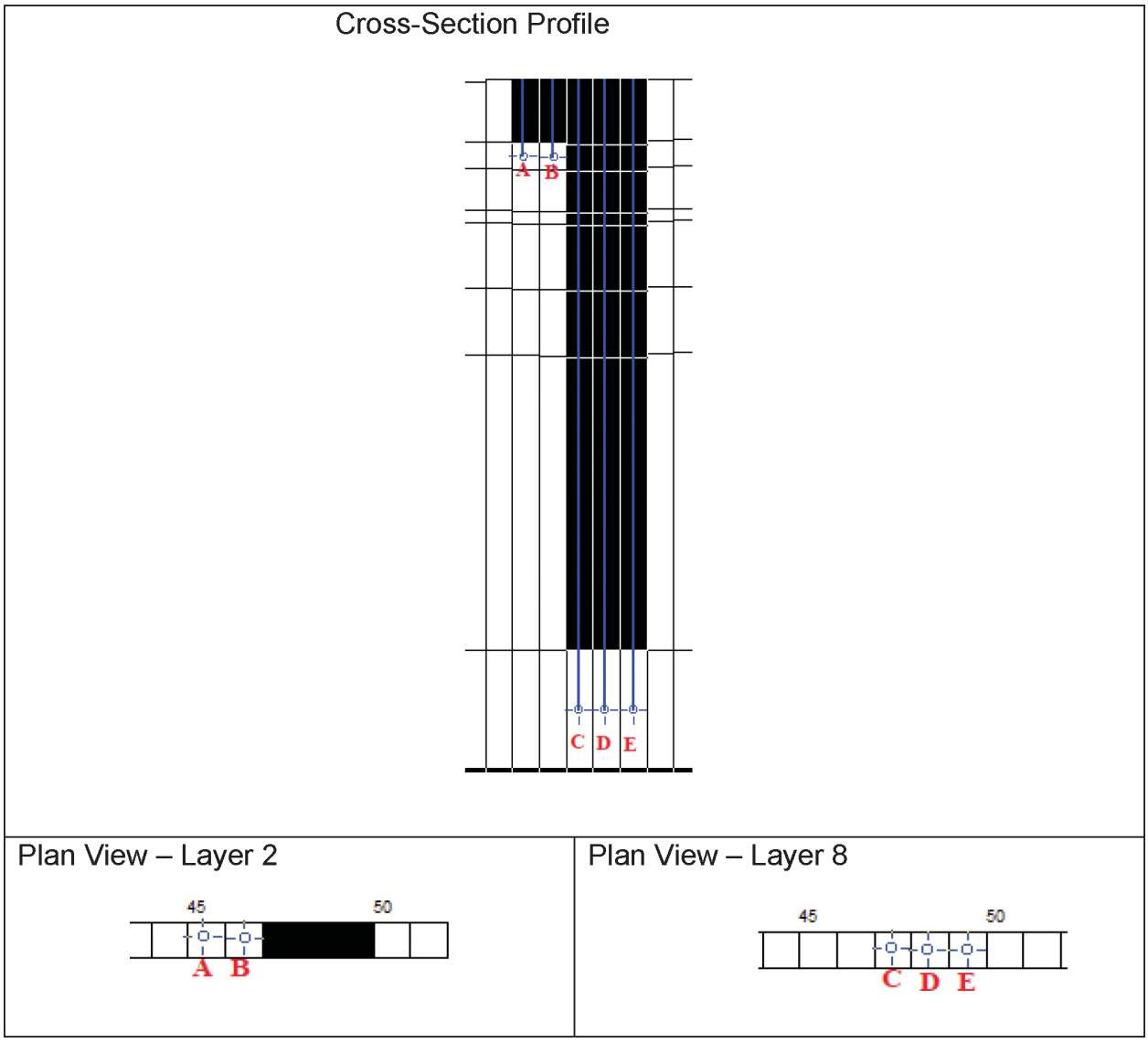

Figure 2.4.12C-28R Locations of Simulated Piezometric Heads in Profile C
Post-Construction Model

Note: The numbers outside the grid represents column numbers. The circle represents location of the simulated groundwater level. Letters besides the circle are the names of the simulated groundwater level locations.

Rename and Replace Figure 2.4.12C-29 with the following:

Profile C Post-Construction Model

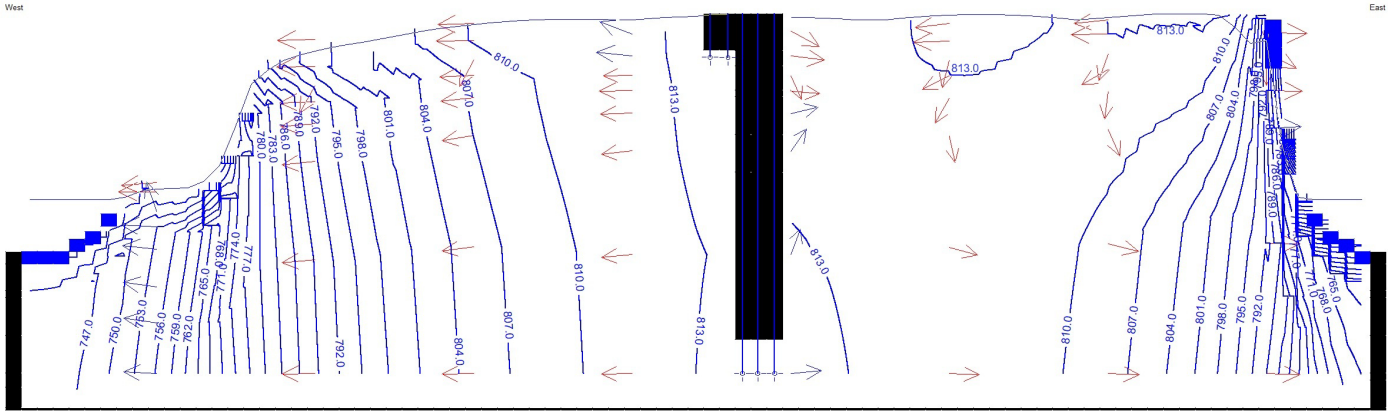

Figure 2.4.12C-29R Water Table – Construction Fill Hydraulic Conductivity of 5.3×10^{-4} cm/sec in Profile C Post-Construction Model

Note: red arrow indicates downward flow and blue arrow indicates upward flow; blue lines with numbers indicates equipotential lines; deep blue blocks represent constant head of Clinch River; and black blocks represents no flow cells.

Delete ESPA SSAR Figures 2.4.12C-30 through 2.4.12C-32.

2.4.13 Accidental Releases of Liquid Effluents to Ground and Surface Waters

Replace the first paragraph of ESPA SSAR Section 2.4.13 with the following:

The purpose of this section is to provide a conservative analysis of a postulated, accidental release of radioactive liquid effluents to the environment at the CRN Site (Figure 2.4.13-1R), in accordance with the guidance provided in Branch Technical Position (BTP) 11-6, Postulated Radioactive Releases Due to Liquid-Containing Tank Failures. The accident scenario is presented. The conceptual model used to evaluate radionuclide transport is discussed, along with potential exposure pathways for water users. The radionuclide transport model is described, and exposure concentrations are calculated. The radionuclide concentrations to which a water user might be exposed, and the associated dose are compared against regulatory limits.

Rename and Replace ESPA SSAR Section 2.4.13.1 with the following:

2.4.13.1 Postulated Accidental Scenario

It is postulated that a Liquid Waste Management System (LWMS) tank outside of containment ruptures with its contents released to the environment. The BWRX-300 LWMS tank having the volume and inventory that would result in the maximum dose to the public was determined to be the Refueling Water Storage Tank (RWST), which contains unprocessed reactor coolant and has a capacity of 165,000 gallons and total activity of 2.43E+03 Curie (Ci). Table 2.4.13-1R lists the radionuclides and activities expected to be present in the RWST. It is assumed that 80 percent of the tank volume is transferred instantaneously to the groundwater at a point within the power block area and no credit is taken for the time that the radionuclides may take (and the associated radioactive decay and other attenuation mechanisms acting during unsaturated zone transport) to travel from the RWST to the saturated zone. Analyzing a release of 80 percent of the tank volume is based on BTP 11-6, which states the following: “The radionuclide inventory for the tank and its components assumed to fail should be based on a conservative estimate of 80 percent capacity of that tank and its components.”

BTP 11-6 indicates that an applicant may take credit for mitigating design features provided they demonstrate that “such features are durable and passive, and that the receiving system has the storage capacity to hold the expected volume of liquid wastes.” In cases where the mitigating design features of tanks can be demonstrated, a consequence analysis in the context of SRP 11.2 and BTP 11-6 may not be required. However, BTP 11-6 further states “the presence of mitigating design features does not change the guidance in SRP 2.4.12 that relates to demonstrating the adequacy of the site’s hydrogeologic properties via a conservative analysis that uses combined literature data and site-specific parameters characterizing transport mechanisms, such as aquifer materials, hydraulic conductivity, and porosity, etc.” Therefore, the analysis presented in this section is required, independent of any acceptable mitigating design features.

Rename and Replace ESPA SSAR Section 2.4.13.2 with the following

2.4.13.2 Exposure Scenarios

BTP 11-6 requires the evaluation of the following exposure scenarios:

A. Direct water use - This scenario assumes that members of the public would consume drinking water withdrawn, near or at the point of entry in an unrestricted area. The sources of water include a groundwater well, or from a surface water body or river.

B. Indirect water use - This scenario assumes the use of water in indirect human consumption. Such scenarios may include 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 water used as an ingredient in food products or in food processing.

C. Combined use of water - This scenario assumes both direct and indirect water use. In such an instance, the scenarios would be modeled separately, and the resulting doses would be summed and compared to the acceptance criteria.

For the consequence analysis, BTP 11-6 defines the point of entry in unrestricted areas as a location beyond the site boundary where the applicant has no administrative controls that could be used to restrict the use of surface water or groundwater resources. When considering surface water resources, the selected point of entry in unrestricted areas should be identified as that location in a surface water body or stream affected only by near-field dilution near the point of entry. BTP 11-6 defines “a member of the public” to be any individual who is not receiving an occupational dose. For the CRN Site, the point of entry in an unrestricted area is assumed to be the right bank (looking downstream) of the Clinch River arm of the Watts Bar Reservoir (hereafter referred to as the Reservoir) at which point the surface water could be used by a member of the public.

Downstream of the CRN Site, the nearest surface water intake is for the former Oak Ridge Bear Creek Road Plant, which used the Clinch River as a source of industrial water and is located downstream of the CRN Site as shown on Table 2.4.1-1 and Figure 2.4.1-1 (Location No. 45). The Oak Ridge Bear Creek Road Plant was also known as the West End Water Treatment Plant (WTP) when the City of Oak Ridge assumed control of the facility. The plant ceased water production on September 30, 2014. Subsequently, this plant has been completely decommissioned and there are no plans to use this facility in the future because the City of Oak Ridge is constructing a new water treatment facility upstream of the CRN Site.

Add new Section 2.4.13.3:

2.4.13.3 Dose Acceptance Criteria

Dose acceptance criteria are based on doses to a member of the public. BTP 11-6 provides the following dose acceptance criteria:

1. Radioactive releases associated with the postulated failure of a tank and components should not do the following:

    a. Result in radioactive material concentrations leading to a dose in excess of 100 millirem (mrem) 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, or

    b. Result in a total dose in excess of 100 mrem 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 dose acceptance criteria.

2. Complying with the dose acceptance criteria does not relieve the applicant from complying with the Table 2, Column 2 effluent concentration limits (ECLs) and the unity rule of Appendix B to 10 CFR Part 20.

Based on the above, dose acceptance and regulatory compliance requires satisfactory demonstration of the following: (1) compliance with the dose standards described above, and (2) compliance with the ECLs and unity rule of 10 CFR 20, Appendix B.

Renumber ESPA SSAR Section 2.4.13.3 to Section 2.4.13.4.

2.4.13.4 Primary Conceptual Model

Renumber Figure 2.4.13-2 in the first sentence of the first paragraph to Figure 2.4.13-2R.

Replace the second paragraph of ESPA SSAR Section 2.4.13.4 with the following:

Following release, it is assumed that radionuclides travel through the construction fill material and the shallow, pervasively fractured bedrock before reaching the Reservoir (Figure 2.4.13-2R). The shortest distance between the BWRX-300 power block area and right bank of the Reservoir is approximately 1200 ft (Figure 2.4.13-1R). While the actual distance between the RWST and the Clinch River is longer, the 1200-ft distance provides a conservative estimate, as less travel time allows for less decay of radionuclides.

Replace the reference to Subsection 2.4.13.5 in the third paragraph with Subsection 2.4.13.6.

Renumber ESPA SSAR Section 2.4.13.4 to Section 2.4.13.5.

2.4.13.5 Alternate Conceptual Model

Renumber Figure 2.4.13-3 in the first paragraph to Figure 2.4.13-3R.

Renumber ESPA SSAR Section 2.4.13.5 to Section 2.4.13.6.

2.4.13.6 Radionuclide Transport Analysis

Replace the first paragraph of ESPA SSAR Section 2.4.13.6 with the following:

A radionuclide transport analysis has been conducted to estimate the radionuclide concentrations that might result in exposure to existing and future water users based on an instantaneous release of the radioactive liquid from the BWRX-300 RWST. Analysis of liquid effluent release commences with a screening model, using demonstratively conservative assumptions and coefficients. Radionuclide concentrations resulting from the screening analysis are then compared to the ECLs identified in 10 CFR 20, Appendix B, Table 2, Column 2, to determine acceptability. Further analysis, using more realistic modeling techniques, is conducted for the radionuclides identified in the screening analysis.

Renumber ESPA SSAR Section 2.4.13.5.1 to Section 2.4.13.6.1.

2.4.13.6.1 Radionuclide Transport in Groundwater and Surface Water

Replace the first paragraph of ESPA SSAR Section 2.4.13.6.1 with the following:

The effects of transport in groundwater and dilution in the Reservoir are modeled using an analytical approach outlined in Reference 2.4.13-1 for quantifying solute flux through groundwater following an instantaneous release and the resulting dilution in the intercepting surface water body. The flux in this case is given by Equation 4.41 of Reference 2.4.13-1. Multiplying this equation by V_T/Q gives the following expression for the dilution factor:

Renumber Reference 2.4.13-21 in the second sentence of the last paragraph to Reference 2.4.13-2.

Renumber ESPA SSAR Section 2.4.13.5.2 to Section 2.4.13.6.2 and Revise the title to Estimation of Source Concentrations (C_0).

2.4.13.6.2 Estimation of Source Concentrations (C_0)

Replace ESPA SSAR Section 2.4.13.6.2 with the following:

Application of dilution factors obtained from Equation 2.4.13-1 requires a source concentration (C_0) for each radionuclide of concern. This is obtained by dividing the radionuclide activity in the

RWST by the tank volume. However, some radionuclides produce daughter products that are important dosimetrically but are not accounted for in the RWST inventory listed in Table 2.4.13-1R. Peak activity was determined by simulating decay of the source term inventory at various time steps ranging from 0 to 50 years for relevant radionuclides to capture the maximum activity of daughter products (Table 2.4.13-1R). The maximum activity for each radionuclide was selected and divided by the release volume to arrive at C_0 values for these radionuclides (Table 2.4.13-2R).

This approach effectively assumes that the contents of an RWST are allowed to sit idle for 50 years after the plant has ceased operating and then accidentally released to the groundwater; however, it is more likely that the RWST will be emptied when the plant shuts down. The approach overestimates the total activity released, as parent radionuclide activities are not decreased while daughter product activities are increased. Although Nb-93m did not reach peak activity within 50 years, it is likely that the RWST contents will be emptied within this period. In addition, the parent radionuclide, Zr-93, is very long lived (half-life over 1 million years) and, therefore, significant concentrations of Nb-93m, which is relatively short-lived with a half-life of about 16 years, will not accumulate.

Renumber ESPA SSAR Section 2.4.13.5.3 to Section 2.4.13.6.3.

2.4.13.6.3 Input Parameters

Renumber ESPA SSAR Section 2.4.13.5.3.1 to Section 2.4.13.6.3.1.

2.4.13.6.3.1 Tank Volume and Volume Released (VT)

Replace the first paragraph of ESPA SSAR Section 2.4.13.6.3.1 with the following:

The RWST volume is 165,000 gallons as described in Subsection 2.4.13.1. It is assumed that 80 percent of the tank volume is transferred instantaneously to the groundwater at a point within the power block area. Thus, 132,000 gallons (17,647 cubic feet) is used as input for V_T in Equation 2.4.13-1.

Renumber ESPA SSAR Section 2.4.13.5.3.2 to Section 2.4.13.6.3.2.

2.4.13.6.3.2 Groundwater Pore Velocity (U) and Groundwater Travel Time

Replace ESPA SSAR Section 2.4.13.6.3.2 with the following:

The magnitude of groundwater pore velocity, U, is computed using the relation (Equation 4.8 of Reference 2.4.13-1):

$$U = \frac{K}{n_e}\frac{dh}{dx} \qquad \text{Equation 2.4.13-2}$$

---

where:

K = hydraulic conductivity (L/T)

n_e = effective porosity (unitless)

$\frac{dh}{dx}$ = hydraulic gradient (L/L)

A hydraulic conductivity of 2.6 feet per day (ft/d) is used based on the results of the onsite aquifer pumping test analysis as discussed in Subsection 2.4.12.2.4.1. This is the computed hydraulic conductivity for the observation well oriented along the predominant groundwater flow path (parallel to strike).

The effective porosity was assigned a value of 0.0467 based on testing performed at the Oak Ridge Reservation (References 2.4.13-3 and 2.4.13-4) as discussed in Subsection 2.4.12.2.4.1. The selected value is the average of 90 tests conducted using immersion-saturation porosimetry.

Based on site water level measurements, as discussed in Subsection 2.4.12.2.2, the horizontal hydraulic gradient at the CRN Site ranges from 0.02 to 0.12. The mean, 0.07, is used as the representative horizontal hydraulic gradient.

The resulting groundwater pore velocity (U), computed using Equation 2.4.13-2, is about 3.9 ft/d. The distance from the power block area to the Reservoir is approximately 1200 ft. Thus, the groundwater travel time from the edge of the power block area to the Reservoir is estimated to be about 308 days (1200 ft / 3.9 ft/d).

Renumber ESPA SSAR Section 2.4.13.5.3.3 to Section 2.4.13.6.3.3.

2.4.13.6.3.3 River Flow Rate (Q)

Renumber Reference 2.4.13-3 in the third sentence of the third paragraph to Reference 2.4.13-1.

Renumber ESPA SSAR Section 2.4.13.5.3.4 to Section 2.4.13.6.3.4.

2.4.13.6.3.4 Aquifer Bulk Density (ρb)

Renumber Reference 2.4.13-6 in the second sentence of the first paragraph to Reference 2.4.13-5.

Renumber ESPA SSAR Section 2.4.13.5.3.5 to Section 2.4.13.6.3.5.

2.4.13.6.3.5 Aquifer Longitudinal Dispersivity (α_L)

Renumber Reference 2.4.13-7 in the first paragraph to Reference 2.4.13-6.

Replace the second paragraph of ESPA SSAR Section 2.4.13.6.3.5 with the following:

The length scale, x, in the above equation is in meters and α_L is also in meters. The length scale used in the above equation is 365.8 meters (m) (1200 ft) for the transport distance from the edge of power block area to the edge of the Reservoir. The estimated α_L is 8.05 m (26.4 ft).

Renumber ESPA SSAR Section 2.4.13.5.3.6 to Section 2.4.13.6.3.6.

2.4.13.6.3.6 Radioactive Half-Life and Decay Constant

Replace ESPA SSAR Section 2.4.13.6.3.6 with the following:

Radioactive half-lives for the radionuclides under consideration were obtained from Reference 2.4.13-6. Radionuclide half-lives were used to calculate the decay constant for each radionuclide, which are given in Table 2.4.13-2R.

Renumber ESPA SSAR Section 2.4.13.5.3.7 to Section 2.4.13.6.3.7.

2.4.13.6.3.7 Distribution Coefficients (Kd)

Replace ESPA SSAR Section 2.4.13.6.3.7 with the following:

Distribution coefficients, K_d, were applied to selected radionuclides based on laboratory testing carried out using samples from the CRN Site as shown on Table 2.4.13-4. K_d is used in the computation of the retardation coefficient, R_d, using the relation:

$$R_d = 1 + \frac{\rho_b K_d}{n_e} \qquad \text{Equation 2.4.13-4}$$

where ρ_b is the bulk density and n_e is the effective porosity of the aquifer.

Non-zero K_d values were used for elements for which site-specific laboratory measurements were available. The geometric mean of the laboratory-derived K_d values was assigned as the representative value for each element. K_d test result summary statistics and analysis notes are presented in Table 2.4.13-4.

Renumber ESPA SSAR Section 2.4.13.6 to Section 2.4.13.7.

2.4.13.7 Radionuclide Concentration in the Reservoir

Replace the first two paragraphs of ESPA SSAR Section 2.4.13.7 with the following:

Minimum dilution factors and associated maximum concentrations in the Reservoir are provided in Table 2.4.13-5R. For screening purposes, dilution factors were initially calculated without considering the effects of sorption (using Equation 2.4.13-1 with R_d = 1). The resulting

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concentrations in the Reservoir were compared against the ECLs. It was found that none of the radionuclides exceeded their respective ECLs. Summing of the C/ECL ratios gives a value of 4.34E-02, which is less than one and compliant with the unity rule of 10 CFR Part 20, Appendix B.

Further analysis accounting for sorption of the radioisotopes listed in Subsection 2.4.13.6.3.7 (Table 2.4.13-4) was performed. This resulted in further reductions in the estimated concentrations in the Reservoir for isotopes with non-zero K_d values (Table 2.4.13-5R). When accounting for sorption and retardation for those radionuclides where site-specific K_d values were determined, summing the C/ECL ratios yields a smaller value of 2.53E-04, demonstrating compliance with the unity rule by a large margin. These concentrations were carried forward for dose evaluation, as described in the Subsection 2.4.13.8.

Renumber Table 2.4.13-5 in the third paragraph to Table 2.4.13-5R.

Replace sixth paragraph of ESPA SSAR Section 2.4.13.7 with the following:

The distribution coefficient for radionuclides, for which no site-specific distribution coefficients are available (e.g., Y-91, Nb-95), is assumed to be zero, even though non-zero values are reported in literature. A larger retardation factor (associated with non-zero distribution coefficients) would result in longer radionuclide travel time, more radioactive decay, and lower concentrations in the Reservoir. In the case of Y, its chemistry is similar to a lanthanide and is often associated in the environment with lanthanides such as cerium (Ce) (Reference 2.4.13-7). The site-specific geometric mean K_d value for Ce is 54 milliliters per gram (mL/g) and thus the likely K_d value for Y is also 54 mL/g; however, the K_d value for Y has been conservatively assigned a value of zero. In addition, Reference 2.4.13-8 reports distribution coefficient values for Nb ranging from 80 to 100 mL/g, based on batch and column experiments conducted with crushed rock, whereas its K_d value has been conservatively assigned a value of zero. If the preceding K_d values were used for Y-91 and Nb-95 in the analysis, the calculated exposure concentrations for these radionuclides in the Reservoir would be essentially zero, and the dose to the receptor would be reduced.

Renumber ESPA SSAR Section 2.4.13.7 to Section 2.4.13.8.

2.4.13.8 Dose Evaluation

Renumber Table 2.4.13-5 in the first sentence of the first paragraph to Table 2.4.13-5R.

Renumber Reference 2.4.13-12 in the third sentence of the first paragraph to Reference 2.4.13-10.

Replace the third paragraph to the end of ESPA SSAR Section 2.4.13.8 with the following (Addresses COL Action Item 2.4-4):

The major inputs and assumptions are as follows:

- No dilution is credited beyond the concentrations shown in Table 2.4.13-5R

- Transit time to dose receptors is assumed to be zero
- Irrigation rate is assumed to be 1 inch per week, which bounds the actual rate near the plant of 0.24 inches per week (Reference 2.4.13-10)
- Consumption and usage rates are the default values for the maximally exposed individual from RG 1.109, Table E-5, while assuming that the time spent on boating and swimming is each the same as that for shoreline activities
- Exposure duration is assumed to be 1 year
- TEDE dose conversion factors for ingestion are obtained from Federal Guidance Report 11 (Reference 2.4.13-11)
- TEDE dose conversion factors for ground deposition and immersion are obtained from Federal Guidance Report 12 (Reference 2.4.13-12)

LADTAP II requires the source nuclide activities to be input in the units of Curies per year (Ci/y). The concentrations in Table 2.4.13-5R are converted into release rates (Ci/y) by multiplying by an arbitrary flow rate of 1 cfs, which is then entered as the liquid effluent discharge rate. LADTAP II divides the isotopic activity by the discharge rate to obtain the initial concentrations, causing the flow rate to cancel out. With the transit time to the dose receptor input as zero, LADTAP II calculates concentrations at the receptor that are identical to those in Table 2.4.13-5R. The resulting total dose from all exposure pathways is 0.62 mrem TEDE to an adult, the age group receiving the maximum dose. This dose is within the 10 CFR 20.1301 limit of 100 mrem TEDE.

Renumber ESPA SSAR Section 2.4.13.8 to Section 2.4.13.9.

2.4.13.9 References

Replace ESPA SSAR Section 2.4.13.9 with the following:

2.4.13-1 NUREG/CR-3332, “Radiological Assessment: A Textbook on Environmental Dose Analysis,” pp. 4-1–4-53, September 1983.

2.4.13-2 Taylor, S.W. and H. Guha, “Time and Magnitude of Peak Concentration of Reactive Groundwater Contaminants Discharged to a River,” Groundwater, Volume 55, No. 1, January-February 2017, pp. 63–72.

2.4.13-3 Oak Ridge National Laboratory Environmental Services Division, ORNL/GWPO-021, “Effective Porosity and Pore-Throat Sizes of Conasauga Group Mudrock: Application, Test and Evaluation of Petrophysical Techniques,” April 1996.

2.4.13-4 Oak Ridge National Laboratory Environmental Services Division, ORNL/GWPO-026, “Effective Porosity and Density of Carbonate Rocks (Maynardville Limestone and Copper Ridge Dolomite) within Bear Creek Valley on the Oak Ridge Reservation Based on Modern Petrophysical Techniques,” February 1997.Terzaghi, K., and R.B. Peck, Soil Mechanics in Engineering Practice, Wiley, 2d ed., 1967.

2.4.13-5 Xu, M., and Y. Eckstein, Use of Weighted Least-Squares Method in Evaluation of the Relationship Between Dispersivity and Field Scale, Groundwater, Vol. 33, Issue 6, pp. 905–908, 1995.

2.4.13-6 International Commission on Radiation Protection, ICRP Publication 107, Annals ICRP, Vol. 38, Issue 3, Nuclear Decay Data for Dosimetric Calculations, 2008.

2.4.13-7 Strenge, D.L., and S.R. Peterson, “Chemical data bases for the Multimedia Environmental Pollutant Assessment System (MEPAS): Ver. 1,” Prepared by Pacific Northwest National Laboratory, December 1989.

2.4.13-8 Anderson, K., Torstenfelt, B., and Rydberg, J., “Leakage of Niobium-94 from an Underground Rock Repository,” Department of Nuclear Chemistry, Chalmers University of Technology, Göteborg, Sweden, 1979. Available at http://www.iaea.org/inis/collection/NCLCollectionStore/_Public/11/561/11561931. pdf

2.4.13-9 NUREG/CR-4013, LADTAP II – Technical Reference and User Guide, April 1986.

2.4.13-10 TVA (River Operations and Renewables), “Water Use in the Tennessee Valley for 2010 and Projected Use in 2035,” July 2012.

2.4.13-11 Environmental Protection Agency, EPA 520/1-88-020, “Federal Guidance Report No. 11, Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” 1988.

2.4.13-12 Environmental Protection Agency, EPA-402-R-93-081, “Federal Guidance Report No. 12, External Exposure to Radionuclides in Air, Water, and Soil,” 1993.

2.4.13-13 Nehls, G.J., and G.G. Akland, “Procedures for Handling Aerometric Data,” Journal of the Air Pollution Control Association, Vol. 23, pp.180-184, 1973.

2.4.13-14 Moore, G.K., “Quantification of Ground-Water Flow in Fractured Rock, Oak Ridge, Tennessee,” Groundwater, Vol. 35, No. 3, May–June 1997.

Replace Table 2.4.13-1 with the following:

Table 2.4.13-1R Liquid Release Source Terms vs Decay Time
(Sheet 1 of 3)

Isotope	Activity (Ci) vs Decay Time														Peak Activity (Ci)	Time of Peak Activity
	0 hr	1 hr	24 hr 1 d	100 hr	168 hr 1 wk	720 hr 1 mo	8760 hr 1 yr	26,280 h (3 yr)	43,800 hr (5 yr)	87,600 hr (10 yr)	131,400 hr (15 yr)	219,000 hr (25 yr)	306,600 hr (35 yr)	438,000 hr (50 yr)		
H-3	1.75E+01	1.75E+01	1.75E+01	1.75E+01	1.75E+01	1.74E+01	1.65E+01	1.48E+01	1.32E+01	9.98E+00	7.54E+00	4.30E+00	2.45E+00	1.06E+00	1.75E+01	0 h
Na-24	5.12E-01	4.89E-01	1.69E-01	5.09E-03	2.21E-04	1.95E-15	0	0	0	0	0	0	0	0	5.12E-01	0 h
P-32	2.06E-02	2.06E-02	1.96E-02	1.68E-02	1.47E-02	4.81E-03	4.19E-10	1.73E-25	0	0	0	0	0	0	2.06E-02	0 h
Cr-51	5.00E-01	5.00E-01	4.88E-01	4.51E-01	4.20E-01	2.36E-01	5.40E-05	6.31E-13	7.37E-21	0	0	0	0	0	5.00E-01	0 h
Mn-54	2.50E-01	2.50E-01	2.49E-01	2.48E-01	2.46E-01	2.34E-01	1.11E-01	2.20E-02	4.35E-03	7.58E-05	1.32E-06	4.00E-10	1.21E-13	6.39E-19	2.50E-01	0 h
Mn-56	9.37E-01	7.16E-01	1.49E-03	2.03E-12	2.37E-20	0	0	0	0	0	0	0	0	0	9.37E-01	0 h
Fe-55	5.18E-01	5.18E-01	5.18E-01	5.17E-01	5.15E-01	5.07E-01	4.00E-01	2.39E-01	1.42E-01	3.90E-02	1.07E-02	8.07E-04	6.08E-05	1.26E-06	5.18E-01	0 h
Fe-59	1.37E-01	1.37E-01	1.35E-01	1.28E-01	1.23E-01	8.59E-02	4.69E-04	5.51E-09	6.46E-14	3.05E-26	0	0	0	0	1.37E-01	0 h
Co-58	1.19E-01	1.19E-01	1.18E-01	1.14E-01	1.11E-01	8.87E-02	3.34E-03	2.63E-06	2.07E-09	3.62E-17	6.30E-25	0	0	0	1.19E-01	0 h
Co-60	2.31E-01	2.31E-01	2.31E-01	2.31E-01	2.30E-01	2.29E-01	2.03E-01	1.56E-01	1.20E-01	6.21E-02	3.22E-02	8.64E-03	2.32E-03	3.23E-04	2.31E-01	0 h
Ni-63	5.18E-04	5.18E-04	5.18E-04	5.18E-04	5.18E-04	5.18E-04	5.14E-04	5.07E-04	5.00E-04	4.83E-04	4.67E-04	4.36E-04	4.07E-04	3.67E-04	5.18E-04	0 h
Cu-64	2.44E+00	2.31E+00	6.59E-01	1.04E-02	2.54E-04	2.10E-17	0	0	0	0	0	0	0	0	2.44E+00	0 h
Zn-65	1.06E-01	1.06E-01	1.06E-01	1.05E-01	1.04E-01	9.73E-02	3.76E-02	4.73E-03	5.95E-04	3.34E-06	1.87E-08	5.90E-13	1.86E-17	3.28E-24	1.06E-01	0 h
Rb-89	2.37E+02	1.52E+01	5.85E-27	0	0	0	0	0	0	0	0	0	0	0	2.37E+02	0 h
Sr-89	3.81E-01	4.27E-01	4.25E-01	4.06E-01	3.91E-01	2.85E-01	2.89E-03	1.30E-07	5.84E-12	7.91E-23	0	0	0	0	4.27E-01	1 h
Sr-90	1.94E-02	1.94E-02	1.94E-02	1.94E-02	1.94E-02	1.94E-02	1.89E-02	1.81E-02	1.72E-02	1.53E-02	1.35E-02	1.06E-02	8.36E-03	5.83E-03	1.94E-02	0 h
Sr-91	2.31E+02	2.15E+02	4.00E+01	1.54E-01	1.07E-03	3.17E-21	0	0	0	0	0	0	0	0	2.31E+02	0 h
Sr-92	5.00E+02	3.87E+02	1.08E+00	3.92E-09	1.10E-16	0	0	0	0	0	0	0	0	0	5.00E+02	0 h
Y-90	1.94E-02	1.94E-02	1.94E-02	1.94E-02	1.94E-02	1.94E-02	1.90E-02	1.81E-02	1.72E-02	1.53E-02	1.36E-02	1.07E-02	8.40E-03	5.87E-03	1.94E-02	0 h
Y-91m	0	7.16E+01	2.51E+01	1.01E-01	6.92E-04	4.50E-21	0	0	0	0	0	0	0	0	7.16E+01	1 h
Y-91	5.06E+00	5.13E+00	6.27E+00	6.31E+00	6.10E+00	4.65E+00	8.79E-02	1.54E-05	2.71E-09	1.11E-18	4.53E-28	0	0	0	6.31E+00	100 h
Y-92	1.56E+02	2.06E+02	1.28E+01	5.68E-06	9.41E-12	0	0	0	0	0	0	0	0	0	2.06E+02	1 h
Y-93	1.69E+01	1.58E+01	3.36E+00	2.02E-02	2.08E-04	1.54E-20	0	0	0	0	0	0	0	0	1.69E+01	0 h
Zr-93	0	8.45E-10	1.04E-08	1.30E-08	1.30E-08	1.30E-08	1.30E-08	1.30E-08	1.30E-08	1.30E-08	1.30E-08	1.30E-08	1.30E-08	1.30E-08	1.30E-08	1 wk
Zr-95	1.06E+01	1.06E+01	1.05E+01	1.01E+01	9.83E+00	7.66E+00	2.03E-01	7.46E-05	2.74E-08	7.09E-17	1.83E-25	0	0	0	1.06E+01	0 h
Nb-93m	0	2.27E-15	8.43E-13	6.06E-12	1.09E-11	5.05E-11	6.12E-10	1.75E-09	2.77E-09	4.93E-09	6.59E-09	8.89E-09	1.03E-08	1.14E-08	1.14E-08	50 y
Nb-95m	0	7.60E-04	1.66E-02	5.12E-02	6.74E-02	7.27E-02	1.99E-03	7.91E-07	2.91E-10	9.73E-19	2.52E-27	0	0	0	7.27E-02	1 mo
Nb-95	1.06E+01	1.06E+01	1.06E+01	1.06E+01	1.05E+01	9.84E+00	4.40E-01	1.67E-04	6.14E-08	1.72E-16	4.45E-25	0	0	0	1.06E+01	0 h
Mo-99	4.93E+01	4.88E+01	3.83E+01	1.73E+01	8.45E+00	2.57E-02	0	0	0	0	0	0	0	0	4.93E+01	0 h
Tc-99m	4.93E+01	4.86E+01	3.68E+01	1.65E+01	8.09E+00	2.58E-02	0	0	0	0	0	0	0	0	4.93E+01	0 h
Tc-99	0	2.05E-08	4.27E-07	1.23E-06	1.57E-06	1.89E-06	1.89E-06	1.89E-06	1.89E-06	1.89E-06	1.89E-06	1.89E-06	1.89E-06	1.89E-06	1.89E-06	1 y
Ru-103	2.56E+00	2.56E+00	2.52E+00	2.38E+00	2.26E+00	1.51E+00	4.14E-03	1.08E-08	2.83E-14	3.13E-28	0	0	0	0	2.56E+00	0 h
Ru-106	3.81E-01	3.81E-01	3.80E-01	3.78E-01	3.76E-01	3.60E-01	1.91E-01	4.80E-02	1.21E-02	3.83E-04	1.21E-05	1.22E-08	1.22E-11	3.90E-16	3.81E-01	0 h

Table 2.4.13-1R Liquid Release Source Terms vs Decay Time
(Sheet 2 of 3)

Isotope	Activity (Ci) vs Decay Time														Peak Activity (Ci)	Time of Peak Activity
	0 hr	1 hr	24 hr 1 d	100 hr	168 hr 1 wk	720 hr 1 mo	8760 hr 1 yr	26,280 h (3 yr)	43,800 hr (5 yr)	87,600 hr (10 yr)	131,400 hr (15 yr)	219,000 hr (25 yr)	306,600 hr (35 yr)	438,000 hr (50 yr)		
Rh-103m	2.56E+00	2.56E+00	2.51E+00	2.38E+00	2.26E+00	1.52E+00	4.66E-03	1.41E-08	3.70E-14	6.36E-28	0	0	0	0	2.56E+00	0 h
Rh-106	3.81E-01	3.81E-01	3.80E-01	3.78E-01	3.76E-01	3.60E-01	1.94E-01	4.94E-02	1.24E-02	4.10E-04	1.30E-05	1.40E-08	1.41E-11	4.83E-16	3.81E-01	0 h
Ag-110m	5.18E-04	5.18E-04	5.17E-04	5.12E-04	5.08E-04	4.77E-04	1.90E-04	2.56E-05	3.44E-06	2.29E-08	1.52E-10	6.70E-15	2.96E-19	8.67E-26	5.18E-04	0 h
Ag-110	0	7.77E-06	7.75E-06	7.68E-06	7.62E-06	7.16E-06	2.90E-06	3.99E-07	5.37E-08	3.80E-10	2.52E-12	1.24E-16	5.46E-21	1.78E-27	7.77E-06	1 h
Te-129m	5.06E+00	5.06E+00	4.96E+00	4.64E+00	4.38E+00	2.72E+00	2.67E-03	7.45E-10	2.08E-16	0	0	0	0	0	5.06E+00	0 h
Te-129	0	1.44E+00	3.13E+00	2.93E+00	2.76E+00	1.73E+00	1.94E-03	6.44E-10	1.80E-16	0	0	0	0	0	3.13E+00	1 d
Te-131m	6.87E+00	6.71E+00	3.95E+00	6.82E-01	1.42E-01	4.11E-07	0	0	0	0	0	0	0	0	6.87E+00	0 h
Te-131	0	1.21E+00	8.83E-01	1.56E-01	3.22E-02	1.18E-07	0	0	0	0	0	0	0	0	1.21E+00	1 h
Te-132	1.19E+00	1.18E+00	9.62E-01	4.91E-01	2.69E-01	2.01E-03	0	0	0	0	0	0	0	0	1.19E+00	0 h
I-129	0	3.97E-12	3.55E-10	1.52E-09	2.49E-09	8.61E-09	1.86E-08	1.87E-08	1.87E-08	1.87E-08	1.87E-08	1.87E-08	1.87E-08	1.87E-08	1.87E-08	3 y
I-131	1.12E+01	1.12E+01	1.07E+01	8.58E+00	6.79E+00	9.39E-01	2.71E-13	0	0	0	0	0	0	0	1.12E+01	0 h
I-132	1.12E+02	8.30E+01	1.07E+00	5.06E-01	2.77E-01	2.23E-03	0	0	0	0	0	0	0	0	1.12E+02	0 h
I-133	8.12E+01	7.86E+01	3.66E+01	2.95E+00	3.09E-01	3.47E-09	0	0	0	0	0	0	0	0	8.12E+01	0 h
I-134	3.31E+02	1.50E+02	1.84E-06	0	0	0	0	0	0	0	0	0	0	0	3.31E+02	0 h
I-135	1.62E+02	1.46E+02	1.31E+01	4.54E-03	3.64E-06	2.69E-31	0	0	0	0	0	0	0	0	1.62E+02	0 h
Xe-131m	0	3.02E-04	6.90E-03	2.38E-02	3.30E-02	2.80E-02	1.41E-10	3.03E-29	0	0	0	0	0	0	3.30E-02	1 wk
Xe-133m	0	3.03E-02	4.32E-01	3.60E-01	1.64E-01	1.17E-04	0	0	0	0	0	0	0	0	4.32E-01	1 d
Xe-133	0	4.26E-01	6.70E+00	8.66E+00	6.36E+00	3.14E-01	1.95E-20	0	0	0	0	0	0	0	8.66E+00	100 h
Xe-135m	0	2.17E+01	2.15E+00	8.28E-04	6.53E-07	1.64E-31	0	0	0	0	0	0	0	0	2.17E+01	1 h
Xe-135	0	1.07E+01	3.45E+01	2.02E-01	1.20E-03	6.76E-22	0	0	0	0	0	0	0	0	3.45E+01	1 d
Cs-134	3.44E+00	3.44E+00	3.44E+00	3.43E+00	3.42E+00	3.35E+00	2.46E+00	1.26E+00	6.41E-01	1.19E-01	2.22E-02	7.70E-04	2.67E-05	1.72E-07	3.44E+00	0 h
Cs-135	0	1.44E-10	2.59E-08	4.14E-08	4.14E-08	4.14E-08	4.14E-08	4.14E-08	4.14E-08	4.14E-08	4.14E-08	4.14E-08	4.14E-08	4.14E-08	4.14E-08	1 wk
Cs-136	2.69E+00	2.68E+00	2.55E+00	2.15E+00	1.85E+00	5.44E-01	9.54E-09	1.20E-25	0	0	0	0	0	0	2.69E+00	0 h
Cs-137	5.25E+00	5.25E+00	5.25E+00	5.25E+00	5.25E+00	5.24E+00	5.13E+00	4.90E+00	4.68E+00	4.17E+00	3.72E+00	2.96E+00	2.35E+00	1.67E+00	5.25E+00	0 h
Cs-138	2.56E+02	7.04E+01	8.89E-12	0	0	0	0	0	0	0	0	0	0	0	2.56E+02	0 h
Ba-136m	0	4.03E-01	3.83E-01	3.24E-01	2.79E-01	8.36E-02	2.09E-09	4.32E-26	0	0	0	0	0	0	4.03E-01	1 h
Ba-137m	5.25E+00	4.97E+00	4.97E+00	4.97E+00	4.96E+00	4.96E+00	4.86E+00	4.64E+00	4.43E+00	3.96E+00	3.53E+00	2.81E+00	2.23E+00	1.59E+00	5.25E+00	0 h
Ba-140	5.31E+01	5.30E+01	5.03E+01	4.24E+01	3.63E+01	1.05E+01	1.37E-07	9.01E-25	0	0	0	0	0	0	5.31E+01	0 h
La-140	5.31E+01	5.31E+01	5.26E+01	4.73E+01	4.14E+01	1.21E+01	2.01E-07	2.20E-24	0	0	0	0	0	0	5.31E+01	0 h
Ce-141	2.56E+00	2.56E+00	2.51E+00	2.34E+00	2.21E+00	1.35E+00	1.08E-03	1.92E-10	3.43E-17	0	0	0	0	0	2.56E+00	0 h
Ce-144	3.81E-01	3.81E-01	3.80E-01	3.77E-01	3.75E-01	3.54E-01	1.57E-01	2.65E-02	4.47E-03	5.25E-05	6.16E-07	8.49E-11	1.17E-14	1.89E-20	3.81E-01	0 h
Pr-144m	0	5.70E-03	5.70E-03	5.66E-03	5.62E-03	5.32E-03	2.39E-03	4.11E-04	6.95E-05	8.62E-07	1.01E-08	1.53E-12	2.11E-16	3.75E-22	5.70E-03	1 d
Pr-144	3.81E-01	3.81E-01	3.80E-01	3.77E-01	3.75E-01	3.55E-01	1.59E-01	2.74E-02	4.64E-03	5.74E-05	6.74E-07	1.02E-10	1.40E-14	2.50E-20	3.81E-01	0 h
Nd-144	0	0	0	1.03E-18	2.05E-18	9.94E-18	8.34E-17	1.32E-16	1.40E-16	1.41E-16	1.42E-16	1.42E-16	1.42E-16	1.42E-16	1.42E-16	15 y
W-187	1.19E-01	1.16E-01	5.93E-02	6.51E-03	9.03E-04	9.77E-11	0	0	0	0	0	0	0	0	1.19E-01	0 h

Table 2.4.13-1R Liquid Release Source Terms vs Decay Time
(Sheet 3 of 3)

Isotope	Activity (Ci) vs Decay Time														Peak Activity (Ci)	Time of Peak Activity
	0 hr	1 hr	24 hr 1 d	100 hr	168 hr 1 wk	720 hr 1 mo	8760 hr 1 yr	26,280 h (3 yr)	43,800 hr (5 yr)	87,600 hr (10 yr)	131,400 hr (15 yr)	219,000 hr (25 yr)	306,600 hr (35 yr)	438,000 hr (50 yr)		
Re-187	0	2.13E-16	3.74E-15	7.04E-15	7.39E-15	7.45E-15	7.45E-15	7.45E-15	7.45E-15	7.45E-15	7.45E-15	7.45E-15	7.45E-15	7.45E-15	7.45E-15	1 mo
U-235m	0	4.90E-05	2.65E-06	7.77E-06	9.51E-06	1.08E-05	1.08E-05	1.08E-05	1.08E-05	1.08E-05	1.08E-05	1.08E-05	1.08E-05	1.08E-05	4.90E-05	1 h
Np-239	4.06E+01	4.01E+01	3.02E+01	1.19E+01	5.13E+00	5.75E-03	0	0	0	0	0	0	0	0	4.06E+01	0 h
Pu-239	0	1.32E-07	2.77E-06	7.66E-06	9.45E-06	1.08E-05	1.08E-05	1.08E-05	1.08E-05	1.08E-05	1.08E-05	1.08E-05	1.08E-05	1.08E-05	1.08E-05	1 mo
Total	2.43E+03	1.82E+03	4.83E+02	2.37E+02	1.91E+02	8.97E+01	3.12E+01	2.62E+01	2.33E+01	1.84E+01	1.49E+01	1.01E+01	7.06E+00	4.32E+00	2.62E+03	

Replace Table 2.4.13-2 with the following:

Table 2.4.13-2R Radionuclide Release Concentrations (C_0)
(Sheet 1 of 2)

Parent	Progeny	Half-Life	Source Activity (Ci)	Source Concentration ($\mu Ci/cm^3$)
H-3		12.35 y	1.75E+01	2.80E-02
Na-24		15 h	5.12E-01	8.20E-04
P-32		14.29 d	2.06E-02	3.30E-05
Cr-51		27.704 d	5.00E-01	8.00E-04
Mn-54		312.12 d	2.50E-01	4.00E-04
Fe-55		2.737 y	5.18E-01	8.29E-04
Mn-56		2.5785 h	9.37E-01	1.50E-03
Co-58		70.8 d	1.19E-01	1.91E-04
Fe-59		44.529 d	1.37E-01	2.19E-04
Co-60		5.2713 y	2.31E-01	3.70E-04
Ni-63		100.1 y	5.18E-04	8.29E-07
Cu-64		12.700 h	2.44E+00	3.91E-03
Zn-65		244.06 d	1.06E-01	1.70E-04
Rb-89		15.15 m	2.37E+02	3.79E-01
	Sr-89	50.53 d	4.27E-01	6.84E-04
Sr-90		28.79 y	1.94E-02	3.11E-05
	Y-90	64.1 h	1.94E-02	3.11E-05
Sr-91		9.63 h	2.31E+02	3.70E-01
	Y-91m	49.71 m	7.16E+01	1.15E-01
	Y-91	58.51 d	6.31E+00	1.01E-02
Sr-92		2.66 h	5.00E+02	8.00E-01
	Y-92	3.54 h	2.06E+02	3.30E-01
Y-93		10.18 h	1.69E+01	2.71E-02
	Zr-93	1.53E+06 y	1.30E-08	2.08E-11
	Nb-93m	16.13 y	1.14E-08	1.83E-11
Zr-95		64.032 d	1.06E+01	1.70E-02
	Nb-95m	3.61 d	7.27E-02	1.16E-04
	Nb-95	34.991 d	1.06E+01	1.70E-02
Mo-99		65.94 h	4.93E+01	7.89E-02
	Tc-99m	6.015 h	4.93E+01	7.89E-02
	Tc-99	2.11E+05 y	1.89E-06	3.03E-09
Ru-103		39.26 d	2.56E+00	4.10E-03
	Rh-103m	56.114 m	2.56E+00	4.10E-03

Table 2.4.13-2R Radionuclide Release Concentrations (C_0)
(Sheet 2 of 2)

Parent	Progeny	Half-Life	Source Activity (Ci)	Source Concentration ($\mu Ci/cm^3$)
Ru-106		373.59 d	3.81E-01	6.10E-04
	Rh-106	29.8 s	3.81E-01	6.10E-04
Ag-110m		249.76 d	5.18E-04	8.29E-07
	Ag-110	24.6 s	7.77E-06	1.24E-08
Te-129m		33.6 d	5.06E+00	8.10E-03
	Te-129	69.6 m	3.13E+00	5.01E-03
	I-129	1.57E+07 y	1.87E-08	2.99E-11
Te-131m		30 h	6.87E+00	1.10E-02
	Te-131	25 m	1.21E+00	1.94E-03
	I-131	8.0207 d	1.12E+01	1.79E-02
Te-132		3.204 d	1.19E+00	1.91E-03
	I-132	2.295 h	1.12E+02	1.79E-01
I-133		20.8 h	8.12E+01	1.30E-01
	Xe-133m	2.19 d	4.32E-01	6.92E-04
	Xe-133	5.243 d	8.66E+00	1.39E-02
Cs-134		2.0648 y	3.44E+00	5.51E-03
I-134		52.5 m	3.31E+02	5.30E-01
I-135		6.57 h	1.62E+02	2.59E-01
	Xe-135m	15.29 m	2.17E+01	3.47E-02
	Xe-135	9.14 h	3.45E+01	5.52E-02
Cs-136		13.16 d	2.69E+00	4.31E-03
Cs-137		30.1671 y	5.25E+00	8.40E-03
	Ba-137m	2.552 m	5.25E+00	8.40E-03
Cs-138		33.41 m	2.56E+02	4.10E-01
Ba-140		12.752 d	5.31E+01	8.50E-02
	La-140	1.6781 d	5.31E+01	8.50E-02
Ce-141		32.508 d	2.56E+00	4.10E-03
Ce-144		284.91 d	3.81E-01	6.10E-04
	Pr-144m	7.2 m	5.70E-03	9.13E-06
	Pr-144	17.28 m	3.81E-01	6.10E-04
W-187		23.72 h	1.19E-01	1.91E-04
	Re-187	4.12E+10 y	7.45E-15	1.19E-17
Np-239		2.3565 d	4.06E+01	6.50E-02
	Pu-239	2.41E+04 y	1.08E-05	1.73E-08

Ci = Curies; y = years; d = days, m = minutes; s = seconds

Table 2.4.13-3 Computation of Aquifer Bulk Density

Renumber Reference 2.4.13-5 in the Notes to Reference 2.4.13-4.

Table 2.4.13-4 Distribution Coefficient (K_d) Test Results and Summary Statistics

Renumber Reference 2.4.13-16 in the Notes to Reference 2.4.13-14.

Replace Table 2.4.13-5 with the following:

Table 2.4.13-5R Transport/Dilution Analysis Parameters and Results
(Sheet 1 of 3)

Source Term Characteristics								Dilution - No Sorption			Sorption Parameters		Dilution - With Sorption		
Parent	Progeny	Half-life[1]	Units	Decay Rate[2] (d^{-1})	Activity[3] (Ci)	Activity Concentration[4] ($\mu Ci/cm^3$)	ECL[5] ($\mu Ci/cm^3$)	Minimum Dilution Factor[6]	Reservoir Concentration[7] (mCi/cm^3)	C/ECL[8]	K_d[9] (cm^3/g)	R[10]	Minimum Dilution Factor[6]	Reservoir Concentration[7] ($\mu Ci/cm^3$)	C/ECL[8]
H-3		12.35	y	1.54E-04	1.75E+01	2.80E-02	1.00E-03	3.25E+05	8.63E-08	8.63E-05	0	1.0	3.25E+05	8.63E-08	8.63E-05
Na-24		15	h	1.11E+00	5.12E-01	8.20E-04	5.00E-05	5.34E+49	1.54E-53	0	0	1.0	5.34E+49	1.54E-53	0
P-32		14.29	d	4.85E-02	2.06E-02	3.30E-05	9.00E-06	2.80E+10	1.18E-15	0	0	1.0	2.80E+10	1.18E-15	0
Cr-51		27.704	d	2.50E-02	5.00E-01	8.00E-04	5.00E-04	1.96E+08	4.09E-12	0	0.6	19.0	1.01E+33	7.96E-37	0
Mn-54		312.12	d	2.22E-03	2.50E-01	4.00E-04	3.00E-05	5.91E+05	6.77E-10	2.26E-05	793.9	23801.0	*	*	0
Fe-55		2.737	y	6.93E-04	5.18E-01	8.29E-04	1.00E-04	3.80E+05	2.18E-09	2.18E-05	0.6	19.0	2.14E+08	3.88E-12	0
Mn-56		2.5785	h	6.45E+00	9.37E-01	1.50E-03	7.00E-05	1.73E+125	8.66E-129	0	793.9	23801.0	*	*	0
Co-58		70.8	d	9.79E-03	1.19E-01	1.91E-04	2.00E-05	4.70E+06	4.05E-11	2.02E-06	345.5	10358.6	*	*	0
Fe-59		44.529	d	1.56E-02	1.37E-01	2.19E-04	1.00E-05	2.06E+07	1.07E-11	1.07E-06	0.6	19.0	9.67E+25	2.27E-30	0
Co-60		5.2713	y	3.60E-04	2.31E-01	3.70E-04	3.00E-06	3.45E+05	1.07E-09	3.57E-04	345.5	10358.6	1.21E+98	3.07E-102	0
Ni-63		100.1	y	1.90E-05	5.18E-04	8.29E-07	1.00E-04	3.12E+05	2.66E-12	0	84.9	2546.2	6.80E+13	1.22E-20	0
Cu-64		12.700	h	1.31E+00	2.44E+00	3.91E-03	2.00E-04	2.04E+54	1.92E-57	0	0	1.0	2.04E+54	1.92E-57	0
Zn-65		244.06	d	2.84E-03	1.06E-01	1.70E-04	5.00E-06	7.05E+05	2.41E-10	4.81E-05	405.4	12154.3	*	*	0
Rb-89		15.15	m	6.59E+01	2.37E+02	3.79E-01	9.00E-04	*	*	0	0	1.0	*	*	0
	Sr-89	50.53	d	1.37E-02	4.27E-01	6.84E-04	8.00E-06	1.29E+07	5.28E-11	6.60E-06	10.2	306.8	4.64E+102	1.47E-106	0
Sr-90		28.79	y	6.59E-05	1.94E-02	3.11E-05	5.00E-07	3.16E+05	9.82E-11	1.96E-04	10.2	306.8	1.96E+10	1.59E-15	0
	Y-90	64.1	h	2.60E-01	1.94E-02	3.11E-05	7.00E-06	1.22E+23	2.54E-28	0	0	1.0	1.22E+23	2.54E-28	0
Sr-91		9.63	h	1.73E+00	2.31E+02	3.70E-01	2.00E-05	6.98E+62	5.30E-64	0	10.2	306.8	*	*	0
	Y-91m	49.71	m	2.01E+01	7.16E+01	1.15E-01	2.00E-03	3.27E+225	3.50E-227	0	0	1.0	3.27E+225	3.50E-227	0
	Y-91	58.51	d	1.18E-02	6.31E+00	1.01E-02	8.00E-06	8.04E+06	1.26E-09	1.57E-04	0	1.0	8.04E+06	1.26E-09	1.57E-04
Sr-92		2.66	h	6.25E+00	5.00E+02	8.00E-01	4.00E-05	2.28E+123	3.51E-124	0	10.2	306.8	*	*	0
	Y-92	3.54	h	4.70E+00	2.06E+02	3.30E-01	4.00E-05	1.86E+106	1.77E-107	0	0	1.0	1.86E+106	1.77E-107	0
Y-93		10.18	h	1.63E+00	1.69E+01	2.71E-02	2.00E-05	1.17E+61	2.32E-63	0	0	1.0	1.17E+61	2.32E-63	0
	Zr-93	1.53E+06	y	1.24E-09	1.30E-08	2.08E-11	4.00E-05	3.10E+05	6.71E-17	0	3.2	96.9	3.01E+07	6.92E-19	0
	Nb-93m	16.13	y	1.18E-04	1.14E-08	1.83E-11	2.00E-04	3.21E+05	5.68E-17	0	3.2	96.9	1.46E+11	1.25E-22	0
Zr-95		64.032	d	1.08E-02	1.06E+01	1.70E-02	2.00E-05	6.17E+06	2.75E-09	1.38E-04	3.2	96.9	1.89E+50	8.98E-53	0
	Nb-95m	3.61	d	1.92E-01	7.27E-02	1.16E-04	3.00E-05	5.82E+19	2.00E-24	0	0	1.0	5.82E+19	2.00E-24	0
	Nb-95	34.991	d	1.98E-02	1.06E+01	1.70E-02	3.00E-05	5.79E+07	2.93E-10	9.77E-06	0	1.0	5.79E+07	2.93E-10	9.77E-06
Mo-99		65.94	h	2.52E-01	4.93E+01	7.89E-02	2.00E-05	5.66E+22	1.39E-24	0	0	1.0	5.66E+22	1.39E-24	0
	Tc-99m	6.015	h	2.77E+00	4.93E+01	7.89E-02	1.00E-03	3.43E+80	2.30E-82	0	0	1.0	3.43E+80	2.30E-82	0
	Tc-99	2.11E+05	y	8.99E-09	1.89E-06	3.03E-09	6.00E-05	3.10E+05	9.75E-15	0	0	1.0	3.10E+05	9.75E-15	0
Ru-103		39.26	d	1.77E-02	2.56E+00	4.10E-03	3.00E-05	3.44E+07	1.19E-10	3.97E-06	8.6	258.8	9.49E+106	4.32E-110	0

Table 2.4.13-5R Transport/Dilution Analysis Parameters and Results
(Sheet 2 of 3)

Source Term Characteristics								Dilution - No Sorption			Sorption Parameters		Dilution - With Sorption		
Parent	Progeny	Half-life[1]	Units	Decay Rate[2] (d^{-1})	Activity[3] (Ci)	Activity Concentration[4] ($\mu Ci/cm^3$)	ECL[5] ($\mu Ci/cm^3$)	Minimum Dilution Factor[6]	Reservoir Concentration[7] (mCi/cm^3)	C/ECL[8]	K_d[9] (cm^3/g)	R[10]	Minimum Dilution Factor[6]	Reservoir Concentration[7] ($\mu Ci/cm^3$)	C/ECL[8]
	Rh-103m	56.114	m	1.78E+01	2.56E+00	4.10E-03	6.00E-03	9.44E+210	4.34E-214	0	0	1.0	9.44E+210	4.34E-214	0
Ru-106		373.59	d	1.86E-03	3.81E-01	6.10E-04	3.00E-06	5.32E+05	1.15E-09	3.82E-04	8.6	258.8	2.08E+34	2.93E-38	0
	Rh-106	29.8	s	2.01E+03	3.81E-01	6.10E-04	NA	*	*	NA	0	1.0	*	*	NA
Ag-110m		249.76	d	2.78E-03	5.18E-04	8.29E-07	6.00E-06	6.92E+05	1.20E-12	0	149.5	4482.8	1.27E+179	6.54E-186	0
	Ag-110	24.6	s	2.43E+03	7.77E-06	1.24E-08	NA	*	*	NA	149.5	4482.8	*	*	NA
Te-129m		33.6	d	2.06E-02	5.06E+00	8.10E-03	7.00E-06	7.03E+07	1.15E-10	1.65E-05	20.9	627.6	7.46E+181	1.09E-184	0
	Te-129	69.6	m	1.43E+01	3.13E+00	5.01E-03	4.00E-04	1.28E+189	3.92E-192	0	20.9	627.6	*	*	0
	I-129	1.57E+07	y	1.21E-10	1.87E-08	2.99E-11	2.00E-07	3.10E+05	9.65E-17	0	0	1.0	3.10E+05	9.65E-17	0
Te-131m		30	h	5.55E-01	6.87E+00	1.10E-02	8.00E-06	2.73E+34	4.03E-37	0	20.9	627.6	*	*	0
	Te-131	25	m	3.99E+01	1.21E+00	1.94E-03	8.00E-05	*	*	0	20.9	627.6	*	*	0
	I-131	8.0207	d	8.64E-02	1.12E+01	1.79E-02	1.00E-06	2.42E+13	7.40E-16	0	0	1.0	2.42E+13	7.40E-16	0
Te-132		3.204	d	2.16E-01	1.19E+00	1.91E-03	9.00E-06	1.03E+21	1.84E-24	0	20.9	627.6	*	*	0
	I-132	2.295	h	7.25E+00	1.12E+02	1.79E-01	1.00E-04	6.67E+132	2.69E-134	0	0	1.0	6.67E+132	2.69E-134	0
I-133		20.8	h	8.00E-01	8.12E+01	1.30E-01	7.00E-06	6.84E+41	1.90E-43	0	0	1.0	6.84E+41	1.90E-43	0
	Xe-133m	2.19	d	3.17E-01	4.32E-01	6.92E-04	NA	4.08E+25	1.70E-29	NA	0	1.0	4.08E+25	1.70E-29	NA
	Xe-133	5.243	d	1.32E-01	8.66E+00	1.39E-02	NA	2.46E+16	5.63E-19	NA	0	1.0	2.46E+16	5.63E-19	NA
Cs-134		2.0648	y	9.19E-04	3.44E+00	5.51E-03	9.00E-07	4.06E+05	1.36E-08	1.51E-02	845.2	25338.9	1.95E+246	2.83E-249	0
I-134		52.5	m	1.90E+01	3.31E+02	5.30E-01	4.00E-04	5.55E+218	9.55E-220	0	0	1.0	5.55E+218	9.55E-220	0
I-135		6.57	h	2.53E+00	1.62E+02	2.59E-01	3.00E-05	6.74E+76	3.85E-78	0	0	1.0	6.74E+76	3.85E-78	0
	Xe-135m	15.29	m	6.53E+01	2.17E+01	3.47E-02	NA	*	*	NA	0	1.0	*	*	NA
	Xe-135	9.14	h	1.82E+00	3.45E+01	5.52E-02	NA	4.04E+64	1.37E-66	NA	0	1.0	4.04E+64	1.37E-66	NA
Cs-136		13.16	d	5.27E-02	2.69E+00	4.31E-03	6.00E-06	6.27E+10	6.87E-14	0	845.2	25338.9	*	*	0
Cs-137		30.1671	y	6.29E-05	5.25E+00	8.40E-03	1.00E-06	3.16E+05	2.66E-08	2.66E-02	845.2	25338.9	4.34E+64	1.93E-67	0
	Ba-137m	2.552	m	3.91E+02	5.25E+00	8.40E-03	NA	*	*	NA	0	1.0	*	*	NA
Cs-138		33.41	m	2.99E+01	2.56E+02	4.10E-01	4.00E-04	4.89E+274	8.38E-276	0	845.2	25338.9	*	*	0
Ba-140		12.752	d	5.44E-02	5.31E+01	8.50E-02	8.00E-06	8.64E+10	9.84E-13	0	0	1.0	8.64E+10	9.84E-13	0
	La-140	1.6781	d	4.13E-01	5.31E+01	8.50E-02	9.00E-06	2.91E+29	2.92E-31	0	0	1.0	2.91E+29	2.92E-31	0
Ce-141		32.508	d	2.13E-02	2.56E+00	4.10E-03	3.00E-05	8.29E+07	4.94E-11	1.65E-06	54	1619.8	*	*	0
Ce-144		284.91	d	2.43E-03	3.81E-01	6.10E-04	3.00E-06	6.28E+05	9.71E-10	3.24E-04	54	1619.8	1.05E+100	5.81E-104	0
	Pr-144m	7.2	m	1.39E+02	5.70E-03	9.13E-06	NA	*	*	NA	0	1.0	*	*	NA
	Pr-144	17.28	m	5.78E+01	3.81E-01	6.10E-04	6.00E-04	*	*	0	0	1.0	*	*	0
W-187		23.72	h	7.01E-01	1.19E-01	1.91E-04	3.00E-05	1.03E+39	1.86E-43	0	0	1.0	1.03E+39	1.86E-43	0
	Re-187	4.12E+10	y	4.61E-14	7.45E-15	1.19E-17	8.00E-03	3.10E+05	3.84E-23	0	0	1.0	3.10E+05	3.84E-23	0
Np-239		2.3565	d	2.94E-01	4.06E+01	6.50E-02	2.00E-05	4.44E+24	1.46E-26	0	0	1.0	4.44E+24	1.46E-26	0

Table 2.4.13-5R Transport/Dilution Analysis Parameters and Results
(Sheet 3 of 3)

Source Term Characteristics								Dilution - No Sorption			Sorption Parameters		Dilution - With Sorption		
Parent	Progeny	Half-life[1]	Units	Decay Rate[2] (d^{-1})	Activity[3] (Ci)	Activity Concentration[4] ($\mu Ci/cm^3$)	ECL[5] ($\mu Ci/cm^3$)	Minimum Dilution Factor[6]	Reservoir Concentration[7] (mCi/cm^3)	C/ECL[8]	K_d[9] (cm^3/g)	R[10]	Minimum Dilution Factor[6]	Reservoir Concentration[7] ($\mu Ci/cm^3$)	C/ECL[8]
	Pu-239	2.41E+04	y	7.87E-08	1.08E-05	1.73E-08	2.00E-08	3.10E+05	5.57E-14	2.79E-06	16.7	501.6	1.57E+08	1.10E-16	0
										4.34E-02					2.53E-04

[1] Reference 2.4.13-6; [2] ln(2)/Half-Life; [3] Table 2.4.13-2R; [4] Table 2.4.13-2R; [5] 10 CFR Part 20, Appendix B, Table 2, Column 2; [6] Equation 2.4.13-1; [7] Activity Concentration/Dilution Factor; [8] Reservoir Concentration/ECL; [9] Table 2.4.13-4; [10] Equation 2.4.13-4

NA = Not available (no published ECL)

* Indicates dilution factor approaching infinity and reservoir concentration approaching zero

C/ECL values less than 1.0E-06 are reported as zero

Replace Figure 2.4.13-1 with the following:

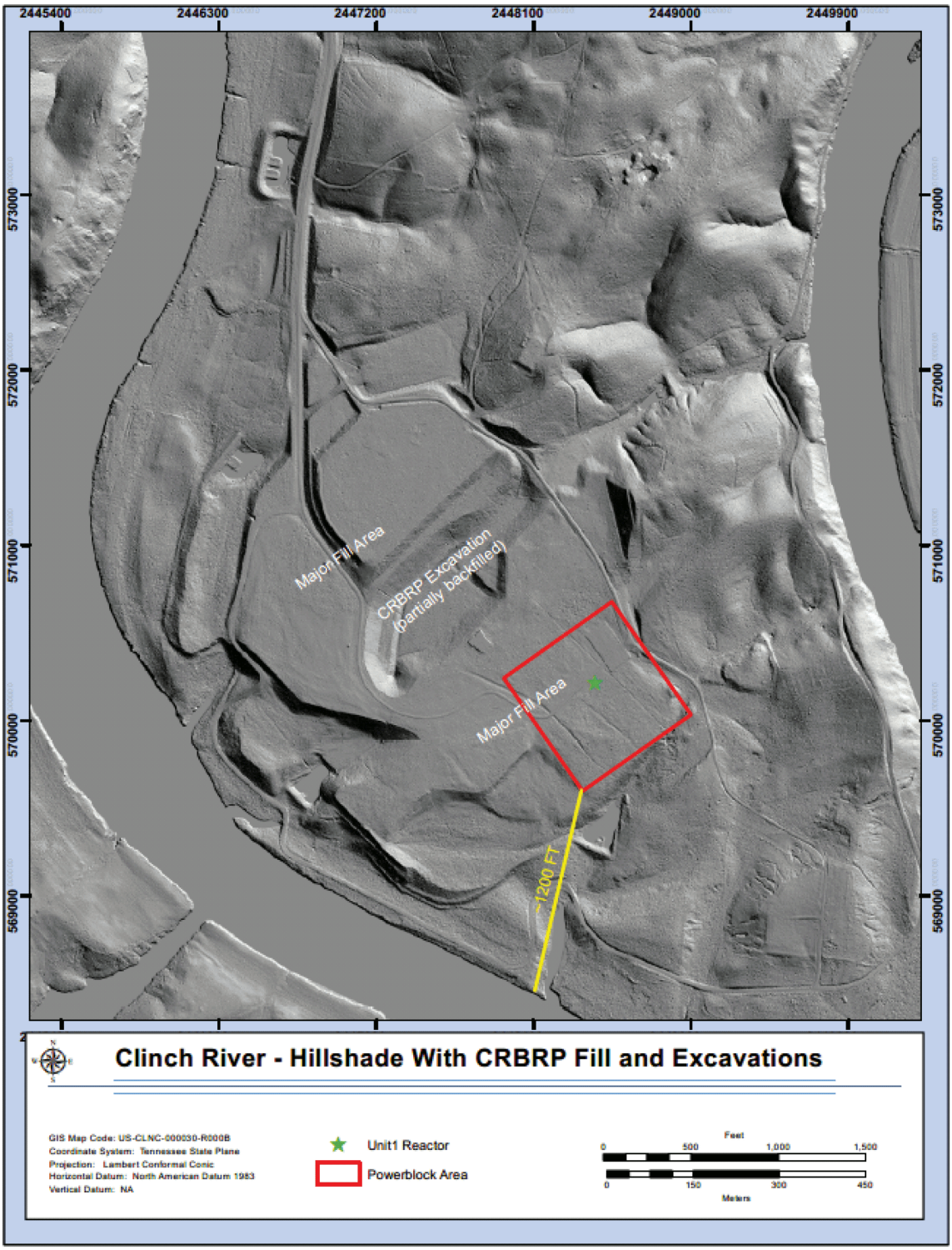


Figure 2.4.13-1R Minimum Distance from the BWRX-300 Power Block Area to the Clinch River Arm of the Watts Bar Reservoir

Replace Figure 2.4.13-2 with the following:

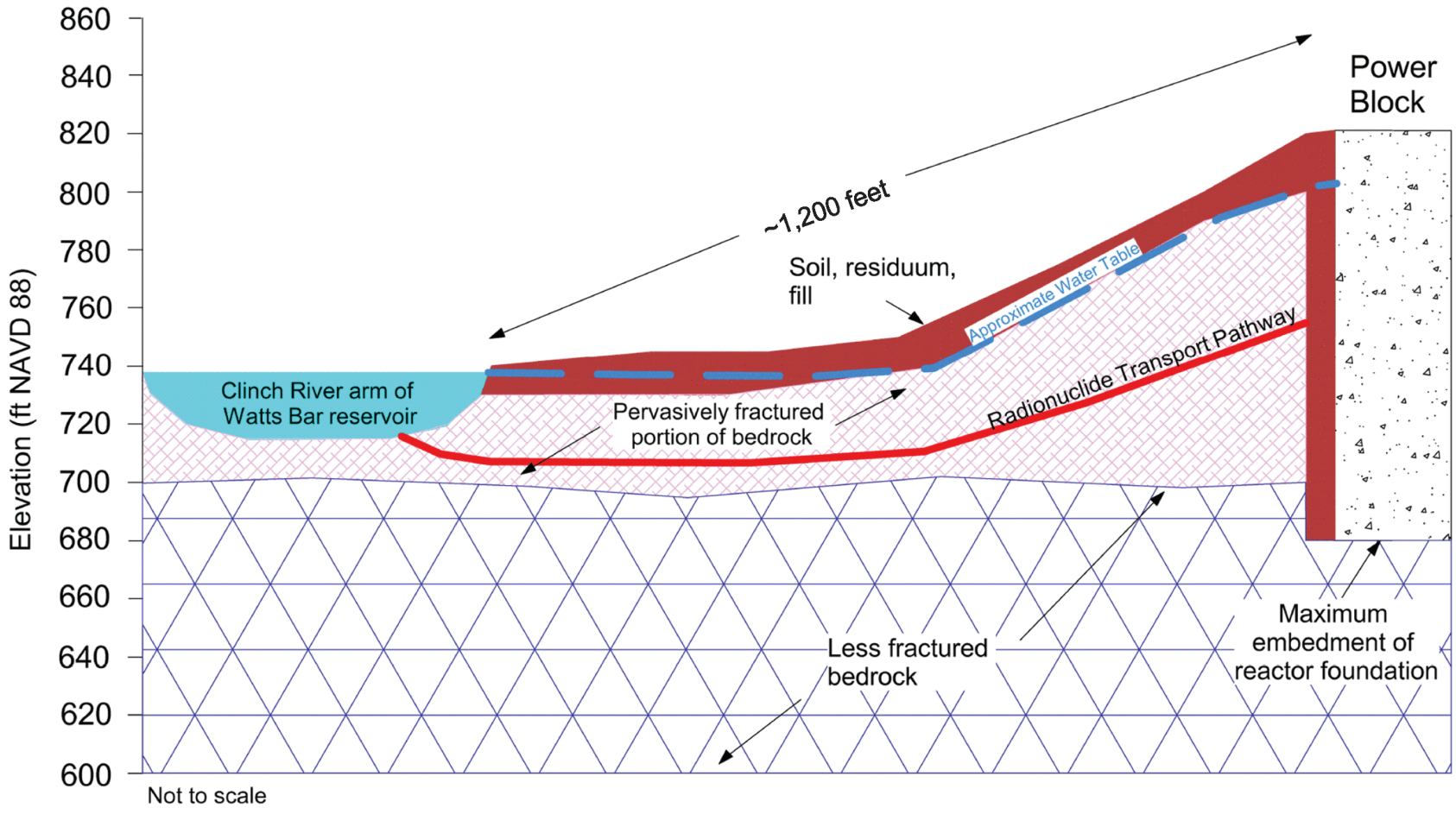


Figure 2.4.13-2R Conceptual Model for Radionuclide Transport

Replace Figure 2.4.13-3 with the following:

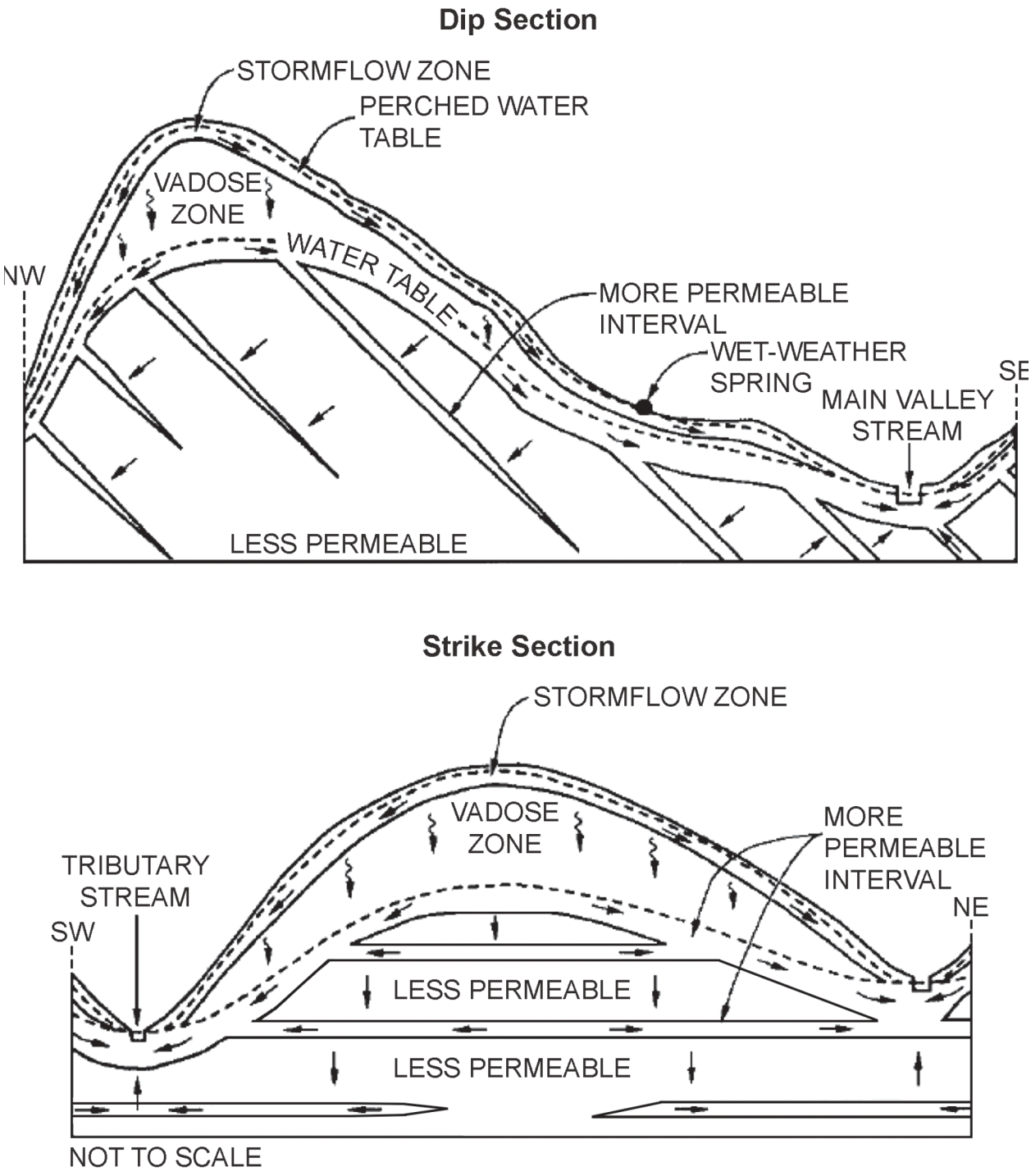


Figure 2.4.13-3R Alternate Conceptual Model for Radionuclide Transport

Source: Adapted from Reference 2.4.13-14

2.4.14 Technical Specifications and Emergency Operation Requirements

Replace the first two sentences of the first paragraph of ESPA SSAR Section 2.4.14 with the following:

The grade elevation of the CRN Site is 814.5 ft NAVD88, and the maximum flood level, as indicated in Subsection 2.4.2.2, is [ ]. The maximum groundwater elevation, as indicated in Subsection 2.4.12, is 814.5 ft NAVD88.

Replace the last sentence of the first paragraph of ESPA SSAR Section 2.4.14 with the following:

Accordingly, there are no requirements for emergency protective measures designed to minimize the impact of hydrology-related events on SC1 facilities, and none are incorporated into the Technical Specifications or emergency operating procedures.

Replace the last paragraph of ESPA SSAR Section 2.4.14 with the following:

The BWRX-300 plant design does not require use of an SC1 source of cooling water from the Clinch River arm of the Watts Bar Reservoir, and thus related Technical Specifications or emergency operation requirements are not necessary.

2.5 Geology, Seismology, and Geotechnical Engineering

Section 2.5 of the referenced ESPA SSAR is incorporated by reference with the following supplements.

2.5.1 Geologic Characterization Information

Replace ESPA SSAR Section 2.5.1 with the following:

The geological and seismological information presented in this subsection was developed from a review of previous reports for the proposed Clinch River Breeder Reactor, published geologic literature, and interpretations of data obtained as part of the surface and subsurface field investigations for the 2013 Early Site Permit Application, Site Safety Analysis Report, herein referred to as the 2013 Clinch River Nuclear (CRN) Small Modular Reactor (SMR) investigation and additional site characterization activities in 2023 focused on the installation of the GVH BWRX-300 SMR to supplement the previous investigations. The investigation data is compiled in a Field Data Report (Reference 2.5.1-341) and the results of analysis of the data are summarized within a Field Data Analysis Report (Reference 2.5.1-340). The results of this investigation and testing campaign conclude that no new geological information was identified at the specific location of Unit 1 at the CRN Site, and therefore the conclusions previously presented in ESPA SSAR Section 2.5.1 regarding geotechnical suitability have not changed.

This subsection demonstrates compliance with the requirements of 10 CFR 100.23(c). Information on the geological and seismological characteristics of the CRN Site region (200-mile (mi) radius), site vicinity (25-mi radius), site area (5-mi radius), and site location (0.6-mi radius) is presented in this subsection. Subsection 2.5.1.1 describes the geologic and tectonic characteristics of the site region and for stratigraphy, the site vicinity. Subsection 2.5.1.2 describes the geologic and tectonic characteristics of the site vicinity, site area and site location. The geological and seismological information was developed in accordance with RG 1.208. SRP 2.5.1, provides guidance for the development of Subsection 2.5.1.

2.5.1.1 Regional Geology (within the 200-Mile Regional Radius)

2.5.1.1.3.1 Valley and Ridge Province

Renumber Table 2.5.1-1 to Table 2.5.1-1R in the twenty-first paragraph of EPSA SSAR Section 2.5.1.1.3.1.

Renumber Figure 2.5.1-30 to Figure 2.5.1-30R in the twenty-fifth paragraph of EPSA SSAR Section 2.5.1.1.3.1.

2.5.1.1.4.3.1 Current Stress Regime in the Eastern United States

Replace (Figure 2.5.1-21) with (Figure 2.5.1-21 (Sheet 1 of 2) and Figure 2.5.1-21R (Sheet 2 of 2)) in the first, second and fourth paragraphs of EPSA SSAR Section 2.5.1.1.4.3.1.

Renumber Figure 2.5.1-21 (Sheet 2 of 2) to Figure 2.5.1-21R (Sheet 2 of 2) in the third and fifth paragraphs of EPSA SSAR Section 2.5.1.1.4.3.1.

Renumber Figure 2.5.1-21 (Sheet 2 of 2) to Figure 2.5.1-21R (Sheet 2 of 2) in the fourth paragraph of the Potential Driving Mechanism of Stresses in the Eastern United States section of EPSA SSAR Section 2.5.1.1.4.3.1.

2.5.1.1.4.3.2 Distribution of Seismicity: The Eastern Tennessee Seismic Zone

Insert the following after the last paragraph of EPSA SSAR Section 2.5.1.1.4.3.2:

Hatcher et al. (Reference 2.5.1-199) describe possible evidence of multiple paleoearthquakes from paleoseismic trenches and geologic mapping (faults, fluidization structures, bleached or clay-filled fractures, and sand dikes), suggesting the most-recent event may have occurred sometime after approximately 73 ka. Warrell et al. (Reference 2.5.1-200) provides additional information on possible paleoseismic features at Douglas Reservoir. Description of these features is presented in Subsection 2.5.3.2.6.

A conference presentation by Cox et al. (Reference 2.5.1-322) describe evidence for surface faulting with ≥6 ft (2 m) of throw exposed in an alluvial terrace along the Little Tennessee River near Vonore, Tennessee. The age of the terrace is not well constrained, and Cox et al. (Reference 2.5.1-322) do not report evidence for multiple events at this site. In another abstract, Glasbrenner et al. (Reference 2.5.1-323) describe geologic mapping and paleoseismic trenching along the Tellico Reservoir about 22 mi (34 km) southeast of the CRN Site. Based on limited geologic field observations and an apparent seismicity alignment, Cox et al. (Reference 2.5.1-324) and Warrell et al. (Reference 2.5.1-325) propose an active, 70 mi (110-km)-long “Quaternary faulting corridor” (referred to as the Dandridge-Vonore Fault zone; DVFZ) in the ETSZ. They suggest that this Fault corridor may be responsible for at least some of the paleoearthquakes interpreted from exposures near Dandridge, Alcoa, and Vonore, Tennessee. Additional details regarding the proposed DVFZ are provided in a recent field trip guidebook (Reference 2.5.1-326); however, no peer-reviewed publication of these interpretations is available.

Other studies of the ETSZ that post-date development of the CEUS-SSC model have examined the deep crustal structure (Reference 2.5.1-327; Reference 2.5.1-328; Reference 2.5.1-329) and the shallow crustal structure (Reference 2.5.1-330; Reference 2.5.1-331) of the ETSZ. Carpenter et al. (Reference 2.5.1-332) suggests that the **M** 4.2 Perry County, Kentucky, earthquake is related to the ETSZ, thus extending the zone farther north than previously recognized. Yang and Salem (Reference 2.5.1-333) used Permanent Scatterers Synthetic Aperture Radar Interferometry (PSInSAR) data to suggest that land surface deformation may be spatially associated with areas of higher seismicity rate in eastern Tennessee, but caution that many other processes besides earthquake activity can contribute to surface deformation.

A PSHA performed for the nearby Department of Energy Y-12 National Security Complex in Oak Ridge in 2021 (Reference 2.5.1-334) evaluated this new information on the ETSZ to see if it warranted changes to the CEUS-SSC model. This study developed an additional hazard

sensitivity analysis to evaluate the postulated Quaternary faulting corridor associated with the Dandridge-Vonore Fault zone.

The results of DVFZ hazard sensitivity indicate that the DVFZ, as characterized, does not significantly increase hazards at the Y-12 site (Reference 2.5.1-334). The studies further concluded that results from the ongoing paleoseismic studies in the ETSZ are preliminary and do not provide adequate evidence for defining the ages and repeat times of earthquakes and therefore do not provide adequate evidence for defining an RLME seismic source for the ETSZ, and thus no revisions to the CEUS-SSC model are required to account for new data from the ETSZ (Reference 2.5.1-334) (Subsection 2.5.2).

2.5.1.2 Local Geology

Replace EPSA SSAR Section 2.5.1.2 with the following:

The following subsection presents a summary of geologic conditions of the 25-mi (40 km) radius site vicinity, 5-mi (8 km) radius site area, and the 0.6-mi (1 km) radius site location. Site physiography, geomorphology, geologic history, stratigraphy, structural geology, non-seismic geologic hazards, and engineering geology are discussed. The information presented for Clinch River Nuclear Site is based on a review of Clinch River Breeder Reactor Project (CRBRP) reports and documents, Oak Ridge National Lab (ORNL) reports, review of published and unpublished geologic literature, and the results of geotechnical and geologic field investigations conducted in 2013 at the CRN Site for the ESPA and the supplemental site characterization activities for the GVH BWRX-300 SMR in 2023. Geologic investigations, including field reconnaissance, karst mapping, river-terrace mapping, and geomorphic analyses were complemented with high-resolution LiDAR digital elevation data acquired during the 2013 CRN SMR investigation. LiDAR coverage encompasses the 5-mi (8 km) radius site area, with pixel resolution of 0.5 ft (15 centimeter (cm)) over a total of 168 sq mi (435 sq km). (Figure 2.5.1-23). Geologic field reconnaissance way points are located on Figure 2.5.1-25.

2.5.1.2.1.2.4 Karst

Replace ESPA SSAR Section 2.5.1.2.1.2.4 with the following:

Karst has developed in eastern Tennessee, within the 25-mi radius site vicinity and the 5-mi radius site area. Karst development in both the site vicinity and site area is described in Subsection 2.5.1.2.5, *Local Geologic Hazards*. Subsection 2.5.1.2.5, emphasizes detailed investigations in the site area, including previous studies within the ORR, and 2013 CRN SMR subsurface studies conducted for the Clinch River Nuclear Small Modular Reactor Project.

2.5.1.2.3 Local Stratigraphy and Lithology

Replace EPSA SSAR Section 2.5.1.2.3 with the following:

The following summary of the stratigraphy and lithology within the CRN Site (Figure 2.5.1-29) is largely based on the description of the geologic units presented in Hatcher et al.

(Reference 2.5.1-9) and Lee and Ketelle (Reference 2.5.1-110), and from site-specific data generated as part of the 2013 CRN SMR site subsurface investigation (Reference 2.5.1-214) and the 2023 supplemental site characterization activities for the GVH BWRX-300 SMR (Reference 2.5.1-341).

2.5.1.2.3.1 Stratigraphic Nomenclature Approach

Replace the third sentence of the first paragraph of the EPSA SSAR Section 2.5.1.2.3.1 with the following:

Lee and Ketelle (Reference 2.5.1-110) completed a study at the ORNL (X-10 Facility) that provides an analog to the CRN Site as it is located directly along strike of the section investigated in 2013 for the CRN SMR Project and 2023 for the BWRX-300 SMR, approximately 4.5-mi (7.2 km) to the northeast.

Replace the third sentence of the second paragraph of the EPSA SSAR Section 2.5.1.2.3.1 with the following:

The 2013 CRN SMR investigation adopted the Hatcher et al.(Reference 2.5.1-9) stratigraphic nomenclature. This nomenclature was carried forward for the 2023 BWRX-300 SMR investigation.

Renumber Table 2.5.1-1 to Table 2.5.1-1R in the third paragraph of EPSA SSAR Section 2.5.1.2.3.1.

Replace the fourth sentence of the third paragraph of the EPSA SSAR Section 2.5.1.2.3.1 with the following:

In general, unit boundaries correlate to distinctive changes in the overall gamma and/or conductivity data observed in the borings logged in 2013 for the CRN SMR Project and 2023 for the BWRX-300 SMR.

2.5.1.2.3.2 Local Stratigraphy

Renumber Figure 2.5.1-30 to Figure 2.5.1-30R in the first paragraph of EPSA SSAR Section 2.5.1.2.3.2.

Replace the last two sentences of the first paragraph of the EPSA SSAR Section 2.5.1.2.3.2 with the following:

Rocks of the Rome Formation do not outcrop at the site, but they were encountered by two 2013 CRN SMR subsurface investigation boreholes. The 2013 CRN SMR subsurface investigation borehole locations and the location of the geologic cross-section depicted in Figure 2.5.1-30R are shown on Figure 2.5.1-31R.

Replace the first sentence of the second paragraph of the EPSA SSAR Section 2.5.1.2.3.2 with the following:

A total of 82 geotechnical boreholes (76 in rock) were drilled on the CRN Site as part of the 2013 CRN SMR investigation (Reference 2.5.1-214) and 27 geotechnical boreholes (16 in rock) were drilled on the CRN Site as part of the 2023 BWRX-300 SMR subsurface investigation (Reference 2.5.1-341).

Replace the third sentence of the second paragraph of the EPSA SSAR Section 2.5.1.2.3.2 with the following:

The Witten Formation, which stratigraphically lies between the older Bowen Formation and the younger Moccasin Formation, was not encountered during the 2013 CRN SMR subsurface investigation drilling program but was encountered in the 2023 BWRX-300 SMR subsurface investigation.

Replace Table 2.5.1-2 with Table 2.5.1-2 (Sheets 1-3 of 5) and Table 2.5.1-2R (Sheets 4-5 of 5) in the second paragraph of EPSA SSAR Section 2.5.1.2.3.2.

Renumber Table 2.5.1-3 with Table 2.5.1-3R in the second paragraph of EPSA SSAR Section 2.5.1.2.3.2.

2.5.1.2.3.3 Local Lithology

Replace the first paragraph of EPSA SSAR Section 2.5.1.2.3.3 with the following:

From oldest to youngest, the following descriptions of the bedrock units encountered at the CRN Site are based on an assessment of the 2013 CRN SMR subsurface investigation boring logs, geologic mapping, and previous investigations (Reference 2.5.1-214). Descriptions of the bedrock units encountered in the 2023 BWRX-300 SMR subsurface investigation are discussed in Reference 2.5.1-340. Lithologic descriptions of the CRN Site stratigraphic units are supplemented with laboratory analytical data, downhole geophysical data, and petrographic data generated during the 2013 CRN SMR and 2023 BWRX-300 SMR subsurface investigations, as available (Reference 2.5.1-214; Reference 2.5.1-341). Definitions of lithologic adjectives and modifiers used in the 2013 descriptions below are provided in Table 2.5.1-4.

Renumber Table 2.5.1-1 to Table 2.5.1-1R in the first paragraph of the Knox Group – Newala Formation – Kingsport Formation/Mascot Dolomite subsection of EPSA SSAR Section 2.5.1.2.3.3.

Renumber Table 2.5.1-3 to Table 2.5.1-3R in the first paragraph of the Chickamauga Group – Blackford Formation subsection of EPSA SSAR Section 2.5.1.2.3.3.

Renumber Table 2.5.1-3 to Table 2.5.1-3R in the first paragraph of the Chickamauga Group – Eidson Member of the Lincolnshire Formation subsection of EPSA SSAR Section 2.5.1.2.3.3.

Renumber Table 2.5.1-3 to Table 2.5.1-3R in the first paragraph of the Chickamauga Group – Fleanor Member of the Lincolnshire Formation subsection of EPSA SSAR Section 2.5.1.2.3.3.

Insert the following after the last paragraph of the Chickamauga Group – Fleanor Member of the Lincolnshire Formation subsection of ESPA SSAR Section 2.5.1.2.3.3.

Descriptions of the Fleanor Member from the 2023 BWRX-300 SMR subsurface investigation are presented in Reference 2.5.1-340.

Renumber Table 2.5.1-3 to Table 2.5.1-3R in the first paragraph of the Chickamauga Group – Rockdell Formation subsection of EPSA SSAR Section 2.5.1.2.3.3.

Insert the following after the last paragraph of the Chickamauga Group – Rockdell Formation subsection of ESPA SSAR Section 2.5.1.2.3.3.

Descriptions of the Rockdell Formation (Unit C and D members) from the 2023 BWRX-300 SMR subsurface investigation are presented in Reference 2.5.1-340.

Renumber Table 2.5.1-3 to Table 2.5.1-3R in the first paragraph of the Chickamauga Group –Benbolt Formation subsection of EPSA SSAR Section 2.5.1.2.3.3.

Insert the following after the last paragraph of the Chickamauga Group – Benbolt Formation subsection of ESPA SSAR Section 2.5.1.2.3.3.

Descriptions of the Benbolt Formation from the 2023 BWRX-300 SMR subsurface investigation are presented in Reference 2.5.1-340.

Renumber Table 2.5.1-1 to Table 2.5.1-1R in the second paragraph of the Chickamauga Group – Bowen Formation subsection of EPSA SSAR Section 2.5.1.2.3.3.

Insert the following after the last paragraph of the Chickamauga Group – Bowen Formation subsection of ESPA SSAR Section 2.5.1.2.3.3.

Descriptions of the Bowen Formation from the 2023 BWRX-300 SMR subsurface investigation are presented in Reference 2.5.1-340.

Replace the first paragraph of the Chickamauga Group – Witten Formation subsection of ESPA SSAR Section 2.5.1.2.3.3 with the following:

The Witten Formation was not encountered during the 2013 CRN SMR subsurface investigation drilling program but was encountered in the 2023 BWRX-300 SMR drilling program in U1-B15 (Figure 2.5.1-30R). Descriptions of the Witten Formation from the 2023 investigation are presented in Reference 2.5.1-340. Alternatively, lithologic descriptions from Lee and Ketelle (Reference 2.5.1-110) and Hatcher et al. (Reference 2.5.1-9) are summarized here.

2.5.1.2.3.4 Karst Evaluation

Replace the first sentence of the first paragraph of ESPA SSAR Section 2.5.1.2.3.4 with the following:

A review of the cavity data from the 2013 CRN SMR and CRBRP Site drilling programs reveal several trends illustrated in Figures 2.5.1-75 through 2.5.1-77.

Replace the third and fourth sentences of the first paragraph of ESPA SSAR Section 2.5.1.2.3.4 with the following:

For this analysis, karst cavity data were partitioned into three elevation intervals. Intervals were as follows: (1) above the 2013 CRN SMR proposed plant grade of elevation 821 ft NAVD88; (2) between elevations 821 ft NAVD88 and 740 ft NAVD88, the shallowest embedment depth considered for the 2013 CRN SMR plant and also the Watts Bar Reservoir pool elevation; and (3) lower than elevation 740 ft NAVD88.

Replace the first sentence of the second paragraph of ESPA SSAR Section 2.5.1.2.3.4 with the following:

The cavities that occur below the current Watts Bar Reservoir elevation of 740 ft NAVD88, which is the current Watts Bar Reservoir elevation as well as the shallowest embedment depth considered in the 2013 CRN SMR investigation, reflect dominantly phreatic development below the water table.

Replace the fifth sentence of the third paragraph of ESPA SSAR Section 2.5.1.2.3.4 with the following:

Additionally, carbonate contents were determined from rock core samples during the 2013 CRN SMR subsurface investigation (Figure 2.5.1-49).

Replace the fourth paragraph of ESPA SSAR Section 2.5.1.2.3.4 with the following:

The spatial distribution of cavities is consistent with the trends discussed above. A map of boreholes indicating the presence and elevation interval of cavities is presented in Figure 2.5.1-79. Several boreholes within the Rockdell Formation in the south-center of the previously planned CRN SMR power block area exhibit cavities in the middle and lower elevation intervals. The footprint of the BWRX-300 lies within the footprint of Location A of the previously planned CRN SMR. The boreholes and cavities occur along strike with bedding. However, elevations of individual cavities within this cluster do not appear to correlate directly. Boreholes B-144 and B-145, spaced approximately 33 ft apart, have cavities at elevation 781 ft NAVD88, although connectivity between cavities is uncertain.

Contrary to previous drilling investigations at the CRN Site in which many dissolution features larger than 0.1 ft were described, only two (2) of such cavity size were described in cored samples recovered and described in the 2023 BWRX-300 SMR investigation. One cavity was described in U1-B16 at a depth of 29.4 ft in the rocks of the Witten Formation, while the other

cavity was interpreted as a 0.9 ft maximum size based on core loss in U1-B05 between depths of 145.4 and 150.4 ft, within rocks of the Rockdell Formation (Reference 2.5.1-340). This fact highlights the scarcity of karst in the units below the Unit 1 location.

2.5.1.2.3.5 Unconsolidated Soils/Fill/Terraces

Replace the first paragraph of ESPA SSAR Section 2.5.1.2.3.5 with the following:

Residual soils and backfill encountered in the 2013 CRN SMR subsurface investigation boreholes are discussed in Subsection 2.5.4, while those encountered in the 2023 BWRX-300 SMR subsurface investigation boreholes are discussed in the Reference 2.5.1-340.

2.5.1.2.4.2.1 Geophysical Data

Replace the first sentence of the first paragraph of ESPA SSAR Section 2.5.1.2.4.2.1 with the following:

Seismic reflection and seismic refraction surveys were conducted at the CRN Site to support the 2013 CRN SMR subsurface investigation (Reference 2.5.2-214).

Insert the following after the first paragraph of ESPA SSAR Section 2.5.1.2.4.2.1:

Surface Analysis of Seismic Waves (SASW) was conducted at the CRN Site to support the 2023 BWRX-300 SMR subsurface investigation (Figure 2.5.1-93) (Reference 2.5.1-341). The primary objectives of the SASW survey were to evaluate the shear wave velocity profile of the subsurface at CRN Site. The shear wave velocity profiles from the SASW survey are discussed in Reference 2.5.1-340.

2.5.1.2.4.2.1.1 Seismic Reflection Data

Replace the first sentence of the second paragraph of ESPA SSAR Section 2.5.1.2.4.2.1.1 with the following:

Two seismic reflection surveys, SRL-1 and SRL-2, were conducted in 2013 at the CRN Site (Reference 2.5.2-214) (also see Subsection 2.5.4.4.2.1).

Replace the last sentence of the third paragraph of ESPA SSAR Section 2.5.1.2.4.2.1.1 with the following:

However, wavelength depends on velocities above and below a specific reflector, so a reliable estimate of vertical resolution that covers the entire time sections of both lines is not possible with the 2013 CRN SMR investigation data.

Renumber Figure 2.5.1-30 to Figure 2.5.1-30R in the seventh paragraph of EPSA SSAR Section 2.5.1.2.4.2.1.1.

Add new Section 2.5.1.2.4.2.1.3:

2.5.1.2.4.2.1.3 Surface Analysis of Seismic Waves

Five SASW lines were acquired at the CRN Site during the 2023 BWRX-300 SMR subsurface investigation (Figure 2.5.1-93) (Reference 2.5.1-341). SASW field testing involves generating surface waves at one location on the exposed material surface and simultaneously measuring the motions perpendicular to the surface created by the passage of surface waves between multiple pairs of surface receivers. Analysis of the SASW survey is discussed in Reference 2.5.1-340.

2.5.1.2.4.2.3 Copper Creek Fault

Renumber Figure 2.5.1-30 to Figure 2.5.1-30R in the second paragraph of EPSA SSAR Section 2.5.1.2.4.2.3.

2.5.1.2.4.3.2 Bedding

Renumber Figure 2.5.1-30 to Figure 2.5.1-30R in the second paragraph of EPSA SSAR Section 2.5.1.2.4.3.2.

Insert the following after the last paragraph of ESPA SSAR Section 2.5.1.2.4.3.2:

The 2023 BWRX-300 SMR subsurface investigation ATV data reveals incredibly consistent bedding measurements (n=855) between all boreholes and stratigraphic units encountered. The strike is predominantly between 50- and 70-degrees azimuth and a mean of 61.3 degrees, while the dips predominantly range between 140- and 160-degrees azimuth with a mean of 151.3 degrees. The median orientation of all of the bedding measurements is 060/33, which is highly concordant with the 2013 mapping and ATV data, all having mean dips of 33 degrees. Geologic cross-sections using subsurface contacts observed in boreholes (Figure 2.5.1-30R) also indicate bedding dips are consistent across the site (Reference 2.5.1-340).

2.5.1.2.4.3.3 Fractures

Replace the second sentence of the fifth paragraph of ESPA SSAR Section 2.5.1.2.4.3.3 with the following:

In the 2013 CRN SMR investigation, fracture surfaces were defined as planar, discontinuous or irregular, while fracture thickness is described as hairline or open, based on whether there was observed material loss along the joint surface.

Insert the following after the last paragraph of ESPA SSAR Section 2.5.1.2.4.3.3:

The 2023 supplementary investigation ATV data are broadly consistent with that of the 2013 data, though with a much smaller dataset due to the lesser number of ATV data collected. The

strike of the 2023 fractures ranges predominantly between 220- and 250-degrees azimuth with a mean of 234.4 degrees, while the dips predominantly range between 310- and 340-degrees azimuth with a mean of 324.4 degrees. The two dominant sets in the 2023 fracture data are consistent with the dominant sets in the 2013 dataset in that both are oriented parallel to strike. While the most dominant set is most consistent (mean orientation of 234/56), the subordinate set dips SE, close to bedding orientation, but has overall a steeper average dip and slightly rotated strike (mean orientation of 075/54) than the secondary set in the 2013 data. Greater details of the 2023 BWRX-300 SMR subsurface investigation fracture sets and comparison to early investigation data are discussed in Reference 2.5.1-340.

2.5.1.2.4.3.4 Shear-Fracture Zones

Replace the first and second sentences of the third paragraph of ESPA SSAR Section 2.5.1.2.4.3.4 with the following:

Shear-fracture zones that include the attributes described above are identified in boreholes in the Lincolnshire (Eidson Member), Rockdell, and Benbolt Formations in the 2013 CRN SMR subsurface investigation (see Subsection 2.5.1.2.6) (Reference 2.5.2-214). The Lincolnshire Formation (Eidson Member) and the Unit A Limestone defined in the CRBRP PSAR are equivalent (see Table 2.5.1-1R).

Replace the first sentence of the fifth paragraph of ESPA SSAR Section 2.5.1.2.4.3.4 with the following:

As noted above, additional shear-fracture zone features were also observed up-section in boreholes during the 2013 CRN SMR investigation further to the southeast within the Benbolt and Rockdell Formations (Figure 2.5.1-65).

Insert the following after the fifth paragraph of ESPA SSAR Section 2.5.1.2.4.3.4:

Similar healed shear-fracture zones were identified during the 2023 field investigation located and characterized in only 8 of the 16 rock core boreholes drilled. These healed shear-fractures were identified in depth intervals corresponding to the Rockdell Formation (25 intervals out of 28 described intervals), the Benbolt Formation (2 intervals) and one occurrence in the recently first-time drilled Witten Formation. The presence of these healed shear-fractures is not observed to be laterally continuous zones through multiple borings with shear-fracture zone features in the Benbolt only occurring in the upper strata of the formation, while those identified in the Rockdell Formation appear to comprise different localized zones all throughout the formation thickness. The only occurrence in the Witten Formation was described in the bottom third of the formation. These features as identified in the 2023 subsurface investigation are discussed in more detail in Reference 2.5.1-340.

Replace the fifth sentence of the sixth paragraph of ESPA SSAR Section 2.5.1.2.4.3.4 with the following:

In other words, the “western shear zone” of Drakulich (Reference 2.5.1-303) is not observed to correspond to the Eidson Member “shear zone” identified in the CRBRP PSAR

(References 2.5.1-100 and 2.5.1-238) or the shear-fracture zone in the 2013 CRN SMR subsurface investigation.

Replace the seventh sentence of the seventh paragraph of ESPA SSAR Section 2.5.1.2.4.3.4 with the following:

Similar features to the bed-normal and bed-parallel veins are common throughout borehole logs of the 2013 CRN SMR subsurface investigation (Reference 2.5.2-214), which describe calcite healed tensional fractures oriented orthogonal to and parallel to bedding.

Replace the eighth paragraph of ESPA SSAR Section 2.5.1.2.4.3.4 with the following:

Lemiszki (Reference 2.5.2-215) observed and briefly described local “shear fractures” that are most common in, but not exclusive to the K-25 site area of the Oak Ridge National Laboratory. The shear fractures are described as single, discrete fractures to wide zones of en echelon fractures and tension gashes with left- and right-lateral shears based on offset of chert marker beds and mineral-filling geometries. That study describes the shears as “commonly perpendicular to bedding,” which is not consistent with descriptions of the “shear zone” observed in the CRBRP (Reference 2.5.1-100 and 2.5.1-238), the bed-parallel veins described by Foreman and Dunne (Reference 2.5.1-237), or the shear-fracture zones as described in the 2013 CRN SMR subsurface study (Reference 2.5.2-214). However, the “shear fractures” described by Lemiszki (Reference 2.5.2-215) could be related to the bed-normal veins observed by Foreman and Dunne (Reference 2.5.1-237) and similar features observed in the 2013 CRN SMR subsurface study (Reference 2.5.2-214).

Replace the fifth sentence of the tenth paragraph of ESPA SSAR Section 2.5.1.2.4.3.4 with the following:

These subvertical stylolites are referred to as tectonic stylolites. The stylolites observed in the 2013 CRN SMR subsurface investigation are generally described as bedding joints that dip between 20 to 40° conformable with bedding (Reference 2.5.2-214).

Insert the following after the last paragraph of ESPA SSAR Section 2.5.1.2.4.3.4:

Based on the determined age of the shear fracture zone features during Alleghenian shortening, they are not considered an active hazard to the CRN Site. It is furthermore suggested that the previously established term “shear-fracture” for these features is a misnomer within the site, as they are not traditional tectonic shear zones. Regardless, they are factors in the foundation rock mass assessment for bearing capacity and further characterization of such features, if identified, shall be performed via geologic mapping during excavation for the power block area.

2.5.1.2.4.4 Characterization of Alleghenian Foreland Fold-Thrust Structures

Replace the first sentence of the fourth paragraph of ESPA SSAR Section 2.5.1.2.4.4 with the following:

As discussed in Subsection 2.5.1.2.4.3.4, zones of localized deformation termed shear-fracture zones were encountered in borings related to the CRBRP investigation (References 2.5.1-100, 2.5.1-238), the 2013 CRN SMR subsurface investigation (Reference 2.5.1-214) and the 2023 BWRX-300 SMR subsurface investigation (Reference 2.5.1-340).

2.5.1.2.5.1.2 Karst Processes and Features at the Clinch River Nuclear Site

Replace the first sentence of the first paragraph of the Karst-Related Surface Features at the Site subsection of ESPA SSAR Section 2.5.1.2.5.1.2 with the following:

Karst-related surface features at the site include large funnel-shaped and dish-shaped sinkholes, and small holes in the ground. Surface features were identified by both the CRBRP and 2013 CRN SMR investigations.

Replace the last two sentences of the first paragraph of the Karst Features Observed in Boreholes subsection of ESPA SSAR Section 2.5.1.2.5.1.2 with the following:

Due to grading of the site during the 1980s, the irregular bedrock surface is best expressed in the CRBRP soil borings and in the 2013 CRN SMR soil borings outside the graded areas. In a few of these 2013 CRN SMR soil borings, blow counts are observed to decrease with depth consistent with dissolution of the rockhead and loss of the deepest horizons of the soil into slots and cavities in the rock.

Replace the second paragraph of the Karst Features Observed in Boreholes subsection of ESPA SSAR Section 2.5.1.2.5.1.2 with the following:

Cavities, both open and clay-filled, were encountered in rock core borings drilled at the site. A total of 240 cavities, defined as being at least 0.1 ft (3 cm) in height in the core, were logged in rock core borings, including borings from the CRBRP drilling program (Reference 2.5.1-100), the 2013 CRN SMR drilling program (Table 2.5.1-11 (Sheets 1-6 of 7) and Table 2.5.1-11R (Sheet 7 of 7) and the 2023 BWRX-300 SMR drilling program. However, only 2 of those cavities were identified during the 2023 program. Of the 196 rock core borings drilled at the site (104 for the CRBRP, 76 for the 2013 CRN SMR investigation and 16 for the 2023 BWRX-300 SMR investigation), 77 borings, or 39 percent, encountered one or more cavities (Table 2.5.1-11 (Sheets 1-6 of 7) and Table 2.5.1-11R (Sheet 7 of 7)). This number should be considered a minimum as not all possible solution features were logged as cavities. The Clinch River Nuclear Site CRBRP borings encountered clay intervals up to 8 ft (2.4 m) thick in five borings (Reference 2.5.1-100). The clay was described as similar to the overburden residual soils and interpreted to represent in situ weathering. Alternatively, these intervals could represent cavities that have been filled with soil washed in from above. Weathered and fractured zones were encountered in some 2013 CRN SMR and 2023 BWRX-300 SMR borings where core losses of

1 to 3 ft (0.3 to 1 m) took place. Dissolution is likely to have played a role in the weathering of these zones, as the lithologies are calcareous.

Replace Table 2.5.1-11 with Table 2.5.1-11 (Sheets 1-6 of 7) and Table 2.5.1-11R (Sheet 7 of 7) in the third paragraph of the Karst Features Observed in Boreholes section of EPSA SSAR Section 2.5.1.2.5.1.2.

2.5.1.2.5.1.3 Potential Karst Hazard at the CRN Site

Replace Table 2.5.1-11 with Table 2.5.1-11 (Sheets 1-6 of 7) and Table 2.5.1-11R (Sheet 7 of 7) in the first paragraph of EPSA SSAR Section 2.5.1.2.5.1.3.

Replace the fifth sentence of the first paragraph of ESPA SSAR Section 2.5.1.2.5.1.3 with the following:

A complete understanding of the extent and spatial distribution of these cavities or other potential karst cavities contains some uncertainty; however, the significant amount of data collected at the CRN Site during the 2013 CRN SMR and 2023 BWRX-300 SMR investigations, as well as during past investigations and excavations, provides a comprehensive understanding of the karst setting within approximately 300 ft (91.4 m) of the ground surface.

2.5.1.2.5.2 Other Local Geologic Hazards

Replace the first sentence of the fifth paragraph of ESPA SSAR Section 2.5.1.2.5.2 with the following:

Underground mining and petroleum and gas extraction have not taken place at the site, however naturally occurring hydrocarbon fluids have been identified in the subsurface at the site (Subsection 2.5.1.2.5.2.1).

Add new Section 2.5.1.2.5.2.1:

2.5.1.2.5.2.1 Subsurface Petroleum Feature

Groundwater observation wells OW-422U, OW-422L, and OW-422D were installed on the southeast portion of the CRN Property as part of the 2013 geotechnical exploration, with total approximate termination depths of 44 feet below ground surface (bgs), 181 feet bgs, and 330 feet bgs, respectively (Reference 2.5.1-339). During the development of groundwater observation well OW-422L, the presence of petroleum was detected within the well, exceeding two feet in thickness. The groundwater and light non-aqueous phase liquids (LNAPL) from the well were sampled and analyzed by TVA, Department of Energy (DOE), and the TDEC personnel with the analytical results being reported to TDEC in Nashville, Tennessee (Reference 2.5.1-339). Due to this discovery, alternate borings were drilled to replace OW-422U, OW-422D, and OW-422L for geotechnical considerations.

The initial laboratory data from well OW-422L reportedly identified gasoline- and diesel-range organic compounds (GRO and DRO) including benzene, ethylbenzene, toluene, 1,2,4 trimethylbenzene, and xylenes, as well as other volatile organic compounds (VOCs) including acetone, carbon disulfide, chloroform, and isopropyl benzene (Reference 2.5.1-339). Subsequent aquifer pump tests were performed for all three wells in 2014 which concluded that groundwater elevation measurements collected indicated a “direct and instantaneous” response to precipitation received by the site with respect to changes in groundwater elevations (Reference 2.5.1-339).

Later testing of the groundwater and petroleum product in all three wells was performed in 2022 for total petroleum hydrocarbons (TPH) using Gas Chromatography-Flame Ionization Detection (GC-FID) and for chemical analysis of VOCs to provide current chemical laboratory data for each well for comparison to historical VOC laboratory results. The data was further evaluated by comparing the sample chromatogram in conjunction with a chromatogram of a normal alkane series generated with the same chromatographic conditions (Reference 2.5.1-339). An independent review of the supplemental chemical data was then performed to determine the source (refined or naturally occurring crude oil) of the petroleum present in OW-422L.

The results of the analysis indicate that the OW-422L (LNAPL and water) samples did not exhibit enrichment of specific analytes such as naphthalenes and mono-aromatics (benzene, toluene, ethylbenzene, and xylene (BTEX) in quantities consistent with those in refined petroleum product, nor did the samples exhibit olefins constituents in quantities or contain additive compounds consistent with refined petroleum products (Reference 2.5.1-339). Further comparison revealed the absence of 2-methyl anthracene (2-MA) typically produced during thermal or catalytic cracking (refining) of crude oils. The comparison between the LNAPL and the groundwater collected from well OW-422L revealed similar analytes (n-paraffins and alkylcyclohexanes) indicating the petroleum is consistent with immature to medium maturity crude oil feedstock (Reference 2.5.1-339). Laboratory analytical results for groundwater from observation wells OW-422U and OW-422D did not report concentrations of VOCs above laboratory reported detection levels.

These results indicate that the petroleum product in well OW-422L is a naturally occurring hydrocarbon product. Furthermore, since the other two wells in the cluster do not seem to have any similar product, it can be speculated that the product is likely specific to the screened in interval in well OW-422L which is in the lower portion of the Benbolt Formation. The occurrence of this naturally occurring petroleum product at the location near the base of the Benbolt formation could only be speculated upon. It could potentially be due to this stratigraphic interval being a reservoir with petroleum product that migrated from greater depths, or it could potentially be that this interval is the source as well as the reservoir.

There are two main petroleum source systems within Tennessee, which are the Knox/Middle Ordovician System and the Devonian/Mississippian System (Reference 2.5.1-335). Considering the site area is underlain by a repeated section of Cambrian to Middle Ordovician strata, the Devonian/Mississippian source can be ruled out. Within the Knox/Middle Ordovician petroleum system are three main hydrocarbon reservoirs, which are the Cambro-Ordovician Knox Group, the Nashville Group, and the Stones River Group (Reference 2.5.1-335).

The post-Knox unconformity is a known oil reservoir that has been explored and exploited throughout the Appalachian Basin (Reference 2.5.1-336). This contact occurs at the site at the boundary of the Newala and Blackford Formations that crop out approximately 1,500 ft to the northwest of the site center. The Knox unconformity occurs at an interpreted depth of approximately 1,000 ft below the site, or an elevation of -300 ft msl. A review of oil well permits was completed within Roane County and surrounding counties (Reference 2.5.1-338). There are no oil wells within the site area. Within the Valley and Ridge physiographic province, there is one Knox oil well in Reigs County to the Southwest of the site, one in Loudon County to the southeast of the site, and three within Anderson County northeast of the site.

The middle Ordovician Nashville Group and Stones River Group are geologically equivalent to the Trenton Group and Black River Group (Reference 2.5.1-335) that have been explored significantly in the northern Appalachian Basin (Reference 2.5.1-337). Research completed by Hatcher et al. (Reference 2.5.1-9) indicate that the Nashville Group and Stones River Group are part of the Chickamauga Supergroup, which contains the Benbolt Formation where OWL-422 was screened. Analysis of oil well permit data (Reference 2.5.1-338) reveals that there are no Chickamauga Group oil wells within the site area, or within the Valley and Ridge province portions of Anderson, Loudon, Meigs, or McMinn Counties.

The 2023 supplemental site investigation for the BWRX-300 technology contained provisions to further investigate the potential presence of the petroleum product within samples collected from this stratigraphic depth. However, the OW-422 well cluster was grouted shut and closed in early 2023, prior to the supplemental site investigation, ruling out observation of any potential petroleum product activity in OW-422L. The 2023 supplemental site investigation for the BWRX-300 employed a process for screening for potential naturally occurring crude in the subsurface.

The borings were screened at specified intervals during the drilling and rock core samples were collected and tested for VOCs using a flame ionization detector (FID) as well as for methane, hydrogen sulfide, and oxygen levels using a 4-gas meter. The screening was conducted at the top of the drill tool string and in the work area of the drill crew and recorded by the Rig Geologist/Geotechnical Engineer. Rig personnel also made a point to observe the return water for sheen and coloring that may be indicative of the presence of crude oil. If FID screening levels exceed 350 parts per million (ppm); and/or methane exceeds the upper explosive limit (UEL) of 17% or hydrogen sulfide exceeds 44%; and/or oxygen is below 19%; work was stopped at the drill depth where the readings were recorded, and appropriate actions were taken to address the issue.

Petroleum was visually observed within the recovered core samples at several borings (Reference 2.5.1-340). Table 2.5.1-20 summarizes the borings in which petroleum was observed. The petroleum observed in the rock core was screened using the FID and the depth interval and screening values were documented on the boring log. Additional readings were collected at the top of the drill string at selected depths and were recorded in the boring log.

The only instance of elevated FID readings exceeding the 350 ppm threshold set for the project was when naturally occurring petroleum (crude) was encountered. This was in boring U1-B02 within the rock core at a depth of approximately 320.0 feet below ground surface (elevation

480.3 ft). The FID reading collected from the top of the drill string when the boring reached 323.0 feet was 60.9 parts per million (ppm). When the boring had been drilled to a depth of 343.0 feet an FID reading was collected from the top of the drill string which indicated a concentration of 436.0 ppm.

Additional readings were collected over time from the drill string and within the work area. The 4-gas meter did not detect any exceedances of the UEL or reduction of oxygen within the work area. The FID readings reduced to safe levels over time, and drilling resumed at the boring. The FID readings in Boring U1-B02 from 343.0 feet to the termination of the boring at 445.1 feet ranged from 10.7 to 255.0 ppm. Boring U1-B02 was the only boring to detect levels of VOCs which exceeded the site safety threshold values.

The other instances of visually observed crude within the recovered core samples (Table 2.5.1-20) did not result in elevated FID readings exceeding the 350 ppm threshold. During the performance of the subsurface investigation there were no detections at any of the borings of concentrations which exceeded the UEL for methane (17%), and/or hydrogen sulfide (44%), and/or oxygen below 19%.

Except for a heavy oil sample observed in a fracture in U1-B15 (115.5 to 115.9 ft-BGS) all observed petroleum identified within the recovered core samples (Table 2.5.1-20) show similar characteristics to the one recovered and described in the OW-422 well cluster. All presence of petroleum in boreholes U1-B01, U1-B02, U1-B03 and U1-B05 was identified in rocks of the Rockdell Formation, or in the base of the Benbolt Formation right in the contact or near the contact with the underlying Rockdell Formation. Petroleum described in U1-B08 was identified in rocks of the Benbolt Formation. The two intervals where petroleum was identified in U1-B15 were described in rocks of the Witten Formation.

Based on the depth of the potential reservoir layer below the ground surface in OW-422L (screened between El 643-623), the stratigraphic interval exists below the bottom of the proposed foundation elevation along strike from the well. However, the potential exists for migration of the petroleum product, either stratigraphically up dip into the excavation or along fractures or karst features, to some other part of the excavation.

The potential presence of the natural occurring crude is a mitigatable issue for construction or operation of the SMR as the foundation concrete that will be placed along the excavation surface of the reactor vessel should be expected to form a barrier preventing any potential amount of natural crude from entering into the foundation of the proposed SMR during operation.

The possibility of natural crude is merely a condition that should be planned and accounted for during foundation excavation and construction of the plant. A contingency plan should be formed for the possibility of petroleum seepage during excavation that establishes protocols if the presence is identified. Monitoring the fluids transferred through the dewatering system for the presence of any hydrocarbon contamination would be a practical detection method. Ultimately, consultation with the excavation and blasting subcontractors and all involved parties will need to take place to determine a consensus remedy and path forward if contamination is detected.

2.5.1.2.6.1 Summary of Subsurface Conditions

Insert the following after the third sentence of the first paragraph of ESPA SSAR Section 2.5.1.2.6.1:

A supplemental site investigation in 2023 was performed for the BWRX-300 SMR technology. Soil boring and rock coring logs are presented in Appendix III to Reference 2.5.1-341.

Replace the fourth sentence of the first paragraph of ESPA SSAR Section 2.5.1.2.6.1 with the following:

The locations of the borings are shown on the boring location plan on Figure 2.5.1-31R.

Replace the second sentence of the second paragraph of ESPA SSAR Section 2.5.1.2.6.1 with the following:

The stratigraphic units underlying the BWRX-300 power block area, from lower to upper units are predominantly the Lincolnshire Formation (Fleanor Member), and Rockdell and Benbolt Formations belonging to the Chickamauga Group (Appendix B.1 Reference 2.5.1-214 and Appendix III to Reference 2.5.1-341).

Renumber Figure 2.5.1-30 to Figure 2.5.1-30R in the second paragraph of EPSA SSAR Section 2.5.1.2.6.1.

Replace the second sentence of the third paragraph of ESPA SSAR Section 2.5.1.2.6.1 with the following:

Information on the orientation of the bedding planes and fractures is from downhole ATV logging performed at the site in 27 borings during the 2013 CRN SMR subsurface investigation and 8 borings during the 2023 BWRX-300 SMR subsurface investigation.

Replace the fourth paragraph of ESPA SSAR Section 2.5.1.2.6.1 with the following:

The footprint of the power block area of the currently planned BWRX-300 SMR technology is shown on Figure 2.5.1-31R. The final grade of the currently planned BWRX-300 SMR technology is expected to be 814.5 ft (248 m) NAVD88 and the foundation level within the power block area is expected to be approximately 115 ft (35.2 m) below the planned final grade (El. 699.5 ft (213 m) NAVD88). The supplemental site-specific subsurface investigation and testing program for the 2023 BWRX-300 technology) was conducted to supplement the CPA for the selected GVH BWRX-300 technology. This additional testing and exploration program was done to ensure that the site-specific safety requirements were met, and potential hazards were identified and addressed before construction begins. The testing also provides additional data to supplement the existing information and datasets to inform any necessary modifications to the design or operational procedures. The 2023 BWRX-300 SMR investigation dataset is presented in Reference 2.5.1-341 with analysis of the data presented in Reference 2.5.1-340 which describes how the existing data is bounding to the previous datasets.

2.5.1.2.6.2 Rock Mass Characterization

Renumber Figure 2.5.1-31 to Figure 2.5.1-31R in the third paragraph of EPSA SSAR Section 2.5.1.2.6.2.

Insert the following after first paragraph of the Fleanor Member subsection of ESPA SSAR Section 2.5.1.2.6.2:

Parallel GSI calculations were performed on the 2023 BWRX-300 SMR investigation boring log fracture dataset for all rock formations encountered in the 2023 investigation, using essentially the same methods as the 2013 investigation calculation. Generally, the GSI ratings for all formations at the CRN Site are between 60 and 85, which correspond with very block to blocky rock mass structure. These structures are described in Reference 2.5.1-265 as interlocked, partially disturbed to undisturbed rock masses, made up of cubical to angular blocks formed by 3 or more joint sets. These structures are of the highest rated rock structures for stability. When comparing the 2013 and 2023 investigations, GSI calculations produced for the major formations encountered have almost identical results for the typical GSI range. The details of the 2023 calculation are presented in Reference 2.5.1-340.

Replace the fourth paragraph of Newala Formation subsection of ESPA SSAR Section 2.5.1.2.6.2 with the following:

The GSI classifications of the stratigraphic units are based on a select number of 100-, 200- and 400-series borings drilled at the CRN Site.

2.5.1.2.6.3 Fracture Zones

Replace the first sentence of the first paragraph of ESPA SSAR Section 2.5.1.2.6.3 with the following:

Borings drilled during the 2013 CRN SMR and 2023 BWRX-300 SMR investigations (Appendix B.1 Reference 2.5.1-214; Appendix III Reference 2.5.1-341) indicate the presence of weathered or fracture zones within the stratigraphic units.

Replace the last sentence of the first paragraph of ESPA SSAR Section 2.5.1.2.6.3 with the following:

The 2013 CRN SMR investigation data indicates that few bedding fractures have weathering or weakening below the power block foundation level.

Replace the fourth paragraph of ESPA SSAR Section 2.5.1.2.6.3 with the following:

Characterization of the weathered or fracture zones is based on the 100- and 200-series borings drilled at the CRN Site during the 2013 CRN SMR investigation. Further evaluation of these zones may be required, when the excavation method has been finalized.

2.5.1.2.6.4 Deformational Zones

Replace the first sentence of the first paragraph of ESPA SSAR Section 2.5.1.2.6.4 with the following:

Borings drilled at during the 2013 CRN SMR and 2023 BWRX-300 SMR investigations encountered what is termed shear-fracture zones, which represent localized zones of strain accommodated via pressure solution with limited cataclasis (Subsection 2.5.1.2.4.3.4).

Insert the following after the second paragraph of ESPA SSAR Section 2.5.1.2.6.4:

A discussion of the shear-fracture zones (greater than or equal to approximately 0.9 ft (27 cm) thick (apparent thickness along boring axis)) encountered in the borings drilled in 2023 at the CRN Site are contained in Reference 2.5.1-340.

Replace the fifth paragraph of ESPA SSAR Section 2.5.1.2.6.4 with the following:

Characterization of the shear-fracture zones is based on the 100-, 200-, and 400-series borings drilled at the CRN Site during the 2013 CRN SMR investigation and the 2023 BWRX-300 SMR investigation. Based on the determined age of the shear fracture zone features during Alleghenian shortening, they are not considered an active hazard to the CRN Site.

The term “shear-fracture” for these features is a misnomer within the site, as they are not traditional tectonic shear zones. During excavation for the power block area, detailed geologic mapping of the foundations for the SC1 structures shall be performed to provide further characterization of such features, if identified.

2.5.1.2.6.5 Karst Features

Insert the following after the fourth paragraph of ESPA SSAR Section 2.5.1.2.6.5:

Contrary to previous drilling investigations at the CRN Site in which many dissolution features larger than 0.1 ft were described, only two (2) of such cavity size were described in cored rock Clinch River Nuclear Site samples recovered and described in the 2023 BWRX-300 SMR investigation. One cavity was described in U1-B16 at a depth of 29.4 ft in the rocks of the Witten Formation, while the other cavity was interpreted as a 0.9 ft maximum size based on core loss in U1-B05 between depths of 145.4 and 150.4 ft, within rocks of the Rockdell Formation (Reference 2.5.1-340).

2.5.1.2.6.8 Effects of Human Activities

Renumber Table 2.5.1-18 to Table 2.5.1-18R in the second paragraph of ESPA SSAR Section 2.5.1.2.6.8.

Replace the second sentence of the second paragraph of ESPA SSAR Section 2.5.1.2.6.8 with the following:

Accessing the Water Resources Permits Dataviewer for oil and gas wells reveals four permit applications for Roane County, TN, dating from June 1981 to March 2023, as listed in Table 2.5.1-18R.

Replace Figure 2.5.1-61 with Figure 2.5.1-61R (Sheet 1 of 2) and Figure 2.5.1-61 (Sheet 2 of 2) in the third paragraph of ESPA SSAR Section 2.5.1.2.6.8.

Replace the fourth paragraph of ESPA SSAR Section 2.5.1.2.6.8 with the following:

Construction materials mined or quarried in Tennessee include dimension stone (sandstone and marble), crushed stone, limestone and clay and sand and gravel (Reference 2.5.1-271). Accessing the TDEC Water Resources Permits Mapviewer (Reference 2.5.1-273), Figure 2.5.1-61R (Sheet 1 of 2) and Figure 2.5.1-61 (Sheet 2 of 2) show the water resource permit applications that currently exist within approximately 10 miles of the CRN Site. As shown on this figure several of these water permits are for mining purposes, however, none of these are within 5 mi (8 km) of the CRN Site. The closest quarry to the site is approximately 8 mi (13 km) northwest and is the Roane County Quarry. The quarry produces crushed and broken limestone. The Dixie Lee quarry is located approximately 8 miles to the east but is currently being reclaimed.

2.5.1.2.6.10 Unforeseen Geologic Conditions *(Addresses ESP Permit Condition #3 and ESP COL Action Item 2.5-3)*

Replace the first sentence of the first paragraph of ESPA SSAR Section 2.5.1.2.6.10 with the following:

Future excavations for SC1 structures would be geologically mapped.

Replace the first sentence of the second paragraph of ESPA SSAR Section 2.5.1.2.6.10 with the following:

Detailed geologic mapping of excavation walls during construction will permit TVA to document the characteristics of dissolution features in the near-surface carbonate rock units and verify whether cavities decrease in size and abundance with depth, as predicted by the 2013 CRN SMR subsurface investigation and geological mapping.

Replace the third paragraph of ESPA SSAR Section 2.5.1.2.6.10 with the following:

The presence of cavities at and below the level of the foundations within the power block area may affect the bearing capacity of the foundation rock mass and groundwater flow. A mitigation plan, including detailed geologic mapping of the excavation floor, a potential grouting program and geophysical surveys to address possible cavities at and below the foundation levels of SC1 structures will be developed based on the excavation method chosen, and executed during construction. The geophysical surveys will be designed to detect cavities below the foundation elevation that could adversely affect foundation performance and are dependent on the excavation method chosen for the BWRX-300 SMR at the CRN Site. Consistent with the ESPA cavity evaluation, two spherical karst features (cavities) were considered to examine the stability of the BWRX-300 RB deeply embedded vertical shaft and surrounding structures using first generation layout of the plant. The details are provided in Subsection 2.5.4. Additional drilling may be required to characterize geophysical anomalies. It is anticipated that the USNRC staff will be notified when the excavation(s) are open for inspection.

2.5.1.2.8 Tsunami and Seiche Hazards

Replace the first sentence of the first paragraph of ESPA SSAR Section 2.5.1.2.5 with the following:

As indicated in Subsection 2.4.6, the CRN Site is located more than 300 mi from the nearest seacoast. In addition, the plant finish grade elevation is at 814.5 ft NAVD88, which is much higher than the sea level at the coast.

2.5.1.2.9 Relational Analysis

Renumber Table 2.5.1-1 to Table 2.5.1-1R in the second paragraph of the General section of EPSA SSAR Section 2.5.1.2.9.

Renumber Table 2.5.1-1 to Table 2.5.1-1R in the first paragraph of the Local Stratigraphy section of EPSA SSAR Section 2.5.1.2.9.

Replace the second paragraph of the Local Stratigraphy subsection of ESPA SSAR Section 2.5.1.2.9 with the following:

As shown on Figure 2.5.1-30R, the BWRX-300 SMR Site power block area has a foundation excavation elevation of approximately 685 ft (209 m) NAVD88. The stratigraphic units that occur at this elevation in the BWRX-300 SMR Unit 1 power block area include the Benbolt Formation and the Rockdell Formation, with the Fleanor member at depth. Descriptions of these stratigraphic units are also provided in Subsection 2.5.1.2.3.3.

Renumber Figure 2.5.1-30 to Figure 2.5.1-30R in the Local Structural Geology subsection of ESPA SSAR Section 2.5.1.2.9.

Renumber Table 2.5.1-1 to Table 2.5.1-1R in the Local Structural Geology section of EPSA SSAR Section 2.5.1.2.9.

Replace the seventh sentence of the second paragraph of the Local Structural Geology subsection of ESPA SSAR Section 2.5.1.2.9 with the following:

The similarities between the CRBRP Site, the 2013 CRN SMR Site and the 2023 BWRX-300 SMR site are shown on Figures 2.5.1-30R, 2.5.1-37, 2.5.1-80, and 2.5.1-81.

Replace the sixth sentence of the second paragraph of the Dissolution and Karst subsection of ESPA SSAR Section 2.5.1.2.9 with the following:

The Rockdell underlies the northwestern edge of the BWRX-300 Unit 1 power block area, while the Witten Formation crops out south of the power block area and dips to the southeast.

Replace the fifth paragraph of the Dissolution and Karst subsection of ESPA SSAR Section 2.5.1.2.9 with the following:

Figures 2.5.1-30R, 2.5.1-51, and 2.5.1-81 are geologic cross sections drawn parallel to dip. Figure 2.5.1-81 indicates the extent of the CRBRP Site excavation. Figure 2.5.1-30R is based on borings drilled in support of the 2013 CRN SMR and 2023 BWRX-300 SMR subsurface investigations and indicates the extent of the Unit 1 power block area. Figure 2.5.1-51 incorporates data from both the CRBRP and the 2013 CRN SMR subsurface investigations. This figure shows the locations of cavities based on the combined project data. Figure 2.5.1-81 shows that the excavation for the CRBRP was planned primarily in the Unit A Upper Siltstone (current Fleanor member) and to a lesser extent in the Unit B Limestone (current lower Rockdell Formation). The stratigraphy and structure of the CRBRP Site is similar to Location B within the 2013 CRN SMR power block area whereas the BWRX-300 SMR Unit 1 location is similar to Location A (Figures 2.5.1-30R and 2.5.1-81). These figures show that the stratigraphic units in the CRBRP excavation and in CRN SMR location B are identical. The differences in stratigraphic nomenclature are discussed above and shown in Table 2.5.1-1R. Except for the Mascot Formation, the karst depression densities and area ratios for the other stratigraphic units within the 2013 CRN SMR power block area are all less than those in the stratigraphic units noted above as occurring to the northwest and southeast of the 2013 CRN SMR power block area.

Replace the second sentence of the first paragraph of the Voids/Cavities Encountered at and Below the Depth of Foundation subsection of ESPA SSAR Section 2.5.1.2.9 with the following:

The CRBRP Site investigation began in 1972, and site drilling was completed in 1980 (References 2.5.1-100 and 2.5.1-303), whereas the 2013 CRN SMR subsurface investigation was conducted in 2013 (Reference 2.5.1-214) and the 2023 BWRX-300 SMR subsurface investigation in 2023 (Reference 2.5.1-341).

Replace the Geotechnical Site Investigations subsection of ESPA SSAR Section 2.5.1.2.9 with the following:

The boundary of the 2013 CRN SMR subsurface investigation and 2023 BWRX-300 SMR subsurface investigation overlap to a large extent with that of the 1972-1980 CRBRP Site

geotechnical site investigation (Figure 2.5.1-74). A total of 104 borings were drilled during the CRBRP Site investigation (Reference 2.5.1-100), 74 were drilled for the 2013 CRN SMR subsurface investigation (Reference 2.5.1-214) and 27 were drilled for the 2023 BWRX-300 SMR subsurface investigations (Reference 2.5.1-341). All 3 site investigations included widely distributed borings to fully characterize the site stratigraphy and concentrated most borings in the areas of SC1 facilities. The northern cluster of 2013 CRN SMR borings, which corresponds to Location B, is centered northeast of the greatest concentration of borings related to the CRBRP Site investigation and occurs within the same geologic unit (Figure 2.5.1-74). The southern cluster of 2013 CRN SMR borings, which corresponds to proposed Location A and the 2023 BWRX-300 SMR Unit 1 location, is centered within the Benbolt formation, a slightly younger geologic unit with similar lithologic characteristics. These three concentrations of borings fall within the Chickamauga Group, a middle Ordovician sequence of limestones, silty and cherty limestones, and calcareous siltstones.

Although the CRBRP utilized a different stratigraphic nomenclature than the 2013 CRN SMR and 2023 BWRX-300 SMR investigations, the unit designations are directly correlative. The positions of formation contacts correlate well between 2013 CRN SMR, 2023 BWRX-300 SMR and CRBRP borings, which demonstrates consistency in topographic survey and geologic logging. The programs began rock coring at the top of rock and cored continuously to boring termination.

Cavities encountered in the site investigations were logged to the nearest tenth of a foot, with bottom and top elevations of the cavity recorded. A total of 216 cavities were logged during the CRBRP Site investigation, whereas the 2013 CRN SMR investigation logged a total of 23 cavities while the 2023 BWRX-300 SMR subsurface investigation logged only 2 cavities. The fewer number of cavities encountered in the 2013 CRN SMR and 2023 BWRX-300 SMR borings are consistent with removal of the cavity-rich near-surface strata during the deep CRBRP excavation and associated grading activities prior to 2013 (Figure 2.5.1-74). The CRBRP and 2013 CRN SMR combined cavities dataset is illustrated on the distribution of cavities cross section (Figure 2.5.1-51) and on the plot of cavity size versus elevation (Figure 2.5.1-52). Additionally, the distribution of cavity size vs. elevation of the 2013 CRN SMR Site data is consistent with that of the CRBRP Site data. Based on the quality and compatibility of the boring investigations, the combined dataset was used for CRN Site analysis of cavities in boreholes.

Replace the Conclusions subsection of ESPA SSAR Section 2.5.1.2.9 with the following:

The aforementioned relational analysis provides a comparison of the CRN SMR Site with the CRBRP Site with respect to geologic formation, rock type, geologic structure and occurrence and character of karst and voids/cavities encountered at and below the depth of foundations. The geologic units mapped in the CRBRP Site excavation (Fleanor member and Rockdell Formation) are the same as those occurring in Location B of the 2013 CRN SMR power block area. However, interpretation of the subsurface data with respect to the BWRX-300 power block area reveal that the bulk of the excavation will be within the Benbolt Formation. Except for the Mascot Formation, the karst depression densities and area ratios for the other stratigraphic units within the power block area are all less than those in the stratigraphic units noted above as occurring to the northwest and southeast of the power block area; indicating that the power block area carbonates appear to have similar dissolution characteristics to the Rockdell Formation.

2.5.1.3 References

Replace the following References in ESPA SSAR Section 2.5.1.3:

2.5.1-272 TDEC, Oil and Gas Well Permits, Website: tdeconline.tn.gov, accessed March 27, 2023, at 03:41 PM.

2.5.1-304 Lundstern, J.E., and M.D. Zoback, “Mulicale variations of the Crustal Stress Field Throughout North America,” Nature Communications 11, 1951, 2020.

Add the following References in ESPA SSAR Section 2.5.1.3:

2.5.1-322 Cox, R.T., E. Gamble, R.D. Hatcher Jr., J.C. Glasbrenner, and R. Counts, “Surface faulting and possible kinematics of the East Tennessee Seismic Zone,” Poster and abstract from the 2014 Geological Society of America Meeting in Vancouver, British Columbia, Canada, paper no. 319–10, 2014.

2.5.1-323 Glasbrenner, J.C., R.D. Hatcher, R.T., Cox, and E. Gamble, “Possible evidence for Quaternary deformation in the east Tennessee seismic zone,” Geological Society of America Abstracts with Programs, 46:593, 2014.

2.5.1-324 Cox, R.T., R.D. Hatcher Jr., R. Counts, E. Gamble, J.C. Glasbrenner, and K. Warrell, “A Seismicity/Quaternary Faulting Corridor in the Eastern Tennessee Seismic Zone: An Update,” Presentation and abstract from the 2017 Geological Society of America Southeastern Section Meeting, Abstracts with Programs, 49, 2017.

2.5.1-325 Warrell, K.F., R.T. Cox, R.D. Hatcher, J.D. Vaughn, and R. Counts, “Paleoseismic evidence of multiple Mw ≥ 6 earthquakes in the Eastern Tennessee Seismic Zone during the Late Quaternary,” Bulletin of the Seismological Society of America, 107, 2017.

2.5.1-326 Cox, R.T., R.D. Hatcher, R. Counts, E. Gamble, J. Glasbrenner, and K.F. Warrell, “Quaternary Faulting Along the Dandridge-Vonore Fault Zone in the Eastern Tennessee Seismic Zone,” Geological Society of America, Field Guide 50, p. 14, 2018.

2.5.1-327 Powell, C.A., M.W. Withers, R.T. Cox, G. Vlahovic, and P. Arroucau, “Crustal velocity structure associated with the eastern Tennessee seismic zone: V_P and V_S images based upon local earthquake tomography,” Journal of Geophysical Research-Solid Earth, 119:464–489, 2014.

2.5.1-328 Graw, J.H., C.A. Powell, and C.A. Langston, “Crustal and upper mantle velocity structure in the vicinity of the eastern Tennessee seismic zone based upon radial P wave transfer functions,” Journal of Geophysical Research–Solid Earth, 120: 243–258, 2015.

2.5.1-329 Powell, C.A., and W.A. Thomas, “Grenville basement structure associated with the Eastern Tennessee Seismic Zone, southeastern USA,” Geology, 44:39–42, 2016.

2.5.1-330 Brandmayr, E., and G. Vlahovic, “The upper crust of the eastern Tennessee seismic zone: insights from potential fields inversions,” Tectonophysics, 685:1–7, 2016.

2.5.1-331 Brandmayr, E., A.P. Kuponiyi, P. Arrocau, and G. Vlahovic, “Group velocity tomography of the upper crust in the eastern Tennessee seismic zone from ambient noise data,” Tectonophysics, 688:148–156, 2016.

2.5.1-332 Carpenter, N.S., E.W. Woolery, and Z. Wang, “The Mw 4.2 Perry County, Kentucky, earthquake of 10 November 2012: evidence of the Eastern Tennessee Seismic Zone in Southeastern Kentucky,” Seismological Research Letters, 85:931–939, 2014.

2.5.1-333 Yang, Z., and A. Salem, “Examination of land surface deformation in the Eastern Tennessee Seismic Zone (ETSZ) using persistent scatters interferometric synthetic aperture radar (PSInSAR),” Journal of Geography, Environment, and Earth Science International, 6:1–18, 2016.

2.5.1-334 Lettis Consultants International, Inc. (LCI), “Department of Energy Y-12 National Security Complex Probabilistic Seismic Hazard Analysis (PSHA) Update and Development of Seismic Design Response Spectra,” Project No. 1863.000, DAC SYAREA-F-0004 000 00, p. 611, 2021.

2.5.1-335 Hatcher, R.D., Final Report for the Project “Geologic Controls of Hydrocarbon Occurrence in the Southern Appalachian Basin in Eastern Tennessee, Southwestern Virginia, Eastern Kentucky, and Southern West Virginia”, United States: N., p. 39, 2006.

2.5.1-336 Baranoski, M.R., R.A. Riley, and M.E. Wolfe, “Play ЄOk; Cambrian-Ordovician Knox Group unconformity play, in Roen, J.B. and Walker, B.J., eds., The atlas of major Appalachian gas plays,” West Virginia Geological and Economic Survey, 181–187, 1996.

2.5.1-337 Wickstrom, L.H., “Play MOf; Middle Ordovician fractured carbonates,” in Roen, J.B., and Walker, B.J., eds., The atlas of major Appalachian gas plays: West Virginia Geological and Economic Survey, 172–176, 1996.

2.5.1-338 TDEC, Oil and Gas Well Permits, Oil and Gas Wells (tdeconline.tn.gov), accessed June 13, 2023, at 11:06 AM.

2.5.1-339 S&ME, “Limited Groundwater Monitoring and Petroleum Forensic Evaluation Report, TVA-Clinch River Site, Oak Ridge, Tennessee,” S&ME Project No. 22440161, April 18, 2023.

2.5.1-340 RIZZO, “Supplemental Geotechnical Field Investigation, Data Analysis Report, Clinch River Nuclear Site,” RIZZO International Inc., Pittsburgh, Pennsylvania, Revision 0, December 20, 2024.

2.5.1-341 S&ME, “Data Report, CRN Supplemental Geotechnical Investigation, Oak Ridge, Tennessee,” Document Control No. CRN-23-0140-A, Revision C, March 25, 2024.

Replace Table 2.5.1-1 with the following:

Table 2.5.1-1R Stratigraphic Units Encountered at the Clinch River Nuclear Site

CRBR PSAR Reference 2.5.1-100		**Lee & Ketelle (1988) Reference 2.5.1-110 Stockdale (1951) Reference 2.5.1-109**		**Hatcher et. al (1992) Reference 2.5.1-9 (2013 CRN SMR Investigation)**		
Undifferentiated Chickamauga		Chickamauga Group	H	Chickamauga Group	Moccasin Formation	
			G		Witten Formation	
			F		Bowen Formation	
			E		Benbolt Formation	
Chickamauga Group	Unit B		C/D		Rockdell Formation	
	Unit A USS		B		Lincolnshire Formation	Fleanor Member
	Unit A LS		A2			Eidson Member
	Unit A LSS		A1		Blackford Formation	
Knox Group		Knox Group		Knox Group (Newala Fm)		Mascot Dolomite
						Kingsport Formation

Replace Table 2.5.1-2 (Sheet 4 of 4) with the following:

Table 2.5.1-2R Clinch River Nuclear Stratigraphic Boundaries (Sheet 4 of 5)

GEOTECHNICAL BORINGS STRATIGRAPHIC SUMMARY TABLE CLINCH RIVER SMR PROJECT (AMEC NO. 6468-13-1072)								Rome Formation		CHICKAMAUGA GROUP																		KNOX GROUP	
										Moccasin Formation		Witten Formation		Bowen Formation		Benbolt Formation		Rockdell Formation				Lincolnshire Formation				Blackford Formation		Five Oaks Formation			
																				Fleanor Member		Eidson Member							
Boring Number	As-Drilled Coordinates/Elevations/Depths				Fill	Residual Soil	Weathered Rock			Unit H		Unit G		Unit F		Unit E		Unit D		Unit C		Unit B		Unit A						Newala Formation	
	Northing (US ft)	Easting (US ft)	Ground Surface Elevation (ft)	Total Drilled Depth (ft)	Depth (ft)	Depth (ft)	Depth (ft)	Start Depth (ft)	End Depth (ft)	Start Depth (ft)	End Depth (ft)	Start Depth (ft)	End Depth (ft)	Start Depth (ft)	End Depth (ft)	Start Depth (ft)	End Depth (ft)	Start Depth (ft)	End Depth (ft)	Start Depth (ft)	End Depth (ft)	Start Depth (ft)	End Depth (ft)	Start Depth (ft)	End Depth (ft)	Start Depth (ft)	End Depth (ft)	Start Depth (ft)	End Depth (ft)	Start Depth (ft)	End Depth (ft)
MP-417	569915.4	2446630.3	772.7	320.0	16.5	37.0	43.0															43.0	148.2	148.2	252.8	252.8	320.1 (TD)				
MP-418	570500.3	2447030.2	811.6	87.8	27.0	43.0	46.8																	46.8	87.8 (TD)						
MP-418A	570514.7	2447049.6	811.1	320.0	27.0	78.0	78.6																	78.6	118.1	118.1	305.6	305.6	320.6 (TD)		
MP-419	571269.8	2446700.6	799.6	321.1	14.0	51.4	51.8																							51.8	321.1 (TD)
MP-420	572033.0	2446918.3	803.1	319.4	2.0	19.2																								19.2	319.4 (TD)
MP-421	570532.3	2446439.6	803.6	320.3	22.0	48.9	50.2																			22.0*	42.0*	42.0*	119.5	119.5	320.3 (TD)
MP-422	570423.7	2448732.0	799.9	320.0	2.0	9.0	10.7									10.7	191.0	191.0	320.0 (TD)												
MP-423	571470.3	2448276.4	799.0	319.3	2.0	34.0	37.3																	37.3	131.6	131.6	276.7			276.7	319.3 (TD)
MP-424(e)	570450.2	2448361.2	800.6	273.2	24.0	36.5												36.5	187.1	187.1	273.2 (TD)										
MP-425(f)	570814.6	2448199.5	811.9	272.9	3.8		6.2													6.2	81.8	81.8	272.9 (TD)								
MP-426(g)	571764.5	2447811.0	842.2	272.3	2.5	19.0	21.9																					21.9	49.1	49.1	272.3 (TD)
MP-428	570755.5	2448681.6	803.8	250.7		23.0	30.8											30.8	201.7	201.7	250.7 (TD)										
MP-429	569975.5	2448591.1	796.0	199.2	4.5	27.0	31.0							31.0	37.6	37.6	199.2 (TD)														
U1-B01	570217.5	2448452.0	799.0	495.7	4.2	16.2	26.5+									16.2	187.8	187.8	381.2	381.2	473.5	473.5	495.7 (TD)								
U1-B02	570264.0	2448418.5	800.3	445.1	18.3	18.4	24.1									24.1	135.3	135.3	340.1	340.1	432.0	432.0	445.1 (TD)								
U1-B03	570167.6	2448485.0	797.4	399.6	12.0	17.0	24.6+									18.7	210.0	210.0	399.6 (TD)												
U1-B04	570182.7	2448405.3	798.9	200.5	7.5	16.0	23.3									23.3	184.8	184.8	200.5 (TD)												
U1-B05	570405.5	2448430.7	800.6	270.4	29.3		33.7									33.7	82.8	82.8	269.7	269.7	270.4 (TD)										

Table 2.5.1-2R Clinch River Nuclear Stratigraphic Boundaries (Sheet 5 of 5)

GEOTECHNICAL BORINGS STRATIGRAPHIC SUMMARY TABLE CLINCH RIVER SMR PROJECT (AMEC NO. 6468-13-1072)								Rome Formation		CHICKAMAUGA GROUP																		KNOX GROUP	
										Moccasin Formation		Witten Formation		Bowen Formation		Benbolt Formation		Rockdell Formation				Lincolnshire Formation				Blackford Formation		Five Oaks Formation		
																				Fleanor Member		Eidson Member							
Boring Number	As-Drilled Coordinates/Elevations/Depths				Fill	Residual Soil	Weathered Rock			Unit H		Unit G		Unit F		Unit E		Unit D		Unit C		Unit B		Unit A						Newala Formation	
	Northing (US ft)	Easting (US ft)	Ground Surface Elevation (ft)	Total Drilled Depth (ft)	Depth (ft)	Depth (ft)	Depth (ft)	Start Depth (ft)	End Depth (ft)	Start Depth (ft)	End Depth (ft)	Start Depth (ft)	End Depth (ft)	Start Depth (ft)	End Depth (ft)	Start Depth (ft)	End Depth (ft)	Start Depth (ft)	End Depth (ft)	Start Depth (ft)	End Depth (ft)	Start Depth (ft)	End Depth (ft)	Start Depth (ft)	End Depth (ft)	Start Depth (ft)	End Depth (ft)	Start Depth (ft)	End Depth (ft)	Start Depth (ft)	End Depth (ft)
U1-B08	570064.9	2448257.9	800.0	201.5	8.0	13.5	21.5+									13.7	159.5	159.5	201.5 (TD)												
U1-B09	570009.1	2448539.9	797.0	201.1	12.5		21.2+									15.2	113.9	113.9	201.1 (TD)												
U1-B10	570028.1	2448635.1	796.2	269.5	13.0	21.9	31.8+							21.9	31.8	31.8	269.5 (TD)														
U1-B11	570076.1	2448703.7	795.6	269.8	3.0	24.8	39.8+									33.7	269.8 (TD)														
U1-B12	570108.0	2448543.8	797.3	401	10.0	16.1	26.5+									16.1	255.0	255.0	401.0 (TD)												
U1-B13	569800.6	2447661.1	808.9	61.1	2.5		3.8									3.8	61.1 (TD)														
U1-B14	569913.6	2447818.7	806.6	61.1	9.6		16.1+									9.6	61.1 (TD)														
U1-B15	570119.7	2449154.3	782.7	201.1	7.1	21.1						21.1	128.2	128.2	168.2	168.2	201.1 (TD)														
U1-B16	570144.9	2449405.4	786.6	59.8	19.8	22.6	24.8+					24.8	59.8 (TD)																		

(a) Angle Boring, Inclination= 28 degrees, Azimuth= N47°E
(b) Angle Boring, Inclination= 28 degrees, Azimuth= N37°W
(c) Angle Boring, Inclination= 27 degrees, Azimuth= N47°E
(d) Angle Boring, Inclination= 28 degrees, Azimuth= N34°W
(e) Angle Boring, Inclination= 25 degrees (± 5 degrees), Azimuth= N38°W (± 5 degrees)
(f) Angle Boring, Inclination= 29 degrees, Azimuth= N49°E
(g) Angle Boring, Inclination= 28 degrees, Azimuth= N40°W

Notes:

* = Inferred contact in residual soils (Blackford/Five Oaks Formations) MP-421

\+ = Weathered rock depth designation with this marking is based on observation of core and not just sampling method. Therefore, thickness of weathered rock shown may extend below top of rock depths and contain the upper portion of bedrock unit.

(TD) = Total Depth Drilled For Boring

Blank cells indicate formations not encountered for a specific boring.

Horizontal/Vertical Datum: Tennessee State Plane Coordinates (East Zone) NAD83/NAVD88

True unit thickness calculated as follows: Assuming a uniform dip of 33 degrees, true thickness = cos 33 x observed thickness

Depths shown for angle borings are lengths below ground surface measured along the inclination (from vertical) and azimuth of the borings as stated below and are not corrected to represent true vertical depth. Source: Reference 2.5.1-214 and Reference 2.5.1-340

Replace Table 2.5.1-3 with the following:

Table 2.5.1-3R Average Thickness and Variability of Each Stratigraphic Unit

Rock Unit Thicknesses Across Site		
Strata	**Average Thickness (ft)**	
	True	**Vertical**
Rome Formation[a]	-	-
Moccasin Formation[a]	-	-
Witten Formation[b]	-	-
Bowen Formation[a,c]	33.5	40
Benbolt Formation	277	330
Rockdell Formation	242	289
Fleanor Member	216	257
Eidson Member	86	102
Blackford Formation	213	254
Newala Formation[a]	-	-

(a) Unit not fully penetrated during the 2013 CRN SMR subsurface investigation drilling program

(b) Unit only (but not fully) encountered during the 2023 BWRX-300 SMR subsurface investigation drilling program

(c) Unit only fully encountered during the 2023 BWRX-300 SMR subsurface investigation drilling program

Note: The 2023 BWRX-300 SMR subsurface investigation drilling program only encountered the Witten Formation through Fleanor Member

Renumber Table 2.5.1-1 to Table 2.5.1-1R in the Notes of Table 2.5.1-10.

Replace Table 2.5.1-11 (Sheet 7 of 7) with the following:

Table 2.5.1-11R (Sheet 7 of 7) Cavities in Boreholes Drilled at Clinch River Nuclear Site, 1973–1978 and 2013

Borehole	CRBRP Strat Unit	SMR Strat Unit	Cavity Top Elev'n (ft)	Cavity Bottom Elev'n (ft)	Cavity Average Elev'n (ft)	Cavity Height (ft)	Water Table Elev'n (ft)	Top of Rock Elev'n (ft)	Bottom of Hole Elev'n (ft)
147	B	Rockdell	850.1	850	850.05	0.1	805	852.1	759.3
147	B	Rockdell	832.7	832.6	832.65	0.1			
147	B	Rockdell	811.8	811.7	811.75	0.1			
149	B	Rockdell	819.1	803.8	811.45	15.3	803	825	759.2
MP-102	NA	Benbolt	790.1	789.9	790	0.2	791	792.1	447.3
MP-102	NA	Benbolt	788.1	787.9	788	1.2			
MP-102	NA	Benbolt	785.1	783.6	784.35	1.5			
MP-104	NA	Benbolt	769.7	769.1	769.4	0.6	809-788	785.6	620.5
MP-406	NA	Blackford	800.8	799.7	800.25	1.1	792	839.8	653.8
MP-410	NA	Rockdell D	797.2	796.2	796.7	1	796	801.3	608.4
MP-410	NA	Rockdell D	795.1	793.4	794.25	1.7			
MP-410	NA	Rockdell D	793.1	791.7	792.4	1.4			
MP-410	NA	Rockdell D	788.2	778.1	783.15	10.1			
MP-412	NA	Blackford	808.1	807.4	807.75	0.7	783	809.1	502.7
MP-418	NA	Eidson	762.2	751.4	756.8	10.8	760	764.8	723.8
MP-418	NA	Eidson	750.2	748.1	749.15	2.1			
MP-418	NA	Eidson	746.4	743.9	745.15	2.5			
MP-418	NA	Eidson	741.9	741.1	741.5	0.8			
MP-418	NA	Eidson	740.1	730.6	735.35	9.5			
MP-420	NA	Ok/Oma	762.5	756.2	759.35	6.3	744-767	782.4	483.7
MP-423	NA	Eidson	756	754.2	755.1	1.8	755	761.7	479.7
MP-424	NA	Rockdell D	687.7	683.5	685.6	4.2	742	767.5	553.0
MP-424	NA	Rockdell D	679.9	675.6	677.7	4.3			
MP-424	NA	Rockdell D	675.3	674.1	674.7	1.3			
MP-424	NA	Rockdell D	665.7	661.8	663.7	3.9			
MP-424	NA	Rockdell D	661.2	660.5	660.8	0.7			
MP-428	NA	Rockdell D	768.6	764	766.3	4.6	775	773	553.1
U1-B05	NA	Rockdell D	655.2	650.2	652.7	5.0	ND	766.9	530.2
U1-B16	NA	Witten	757.2	757.1	757.15	0.1	ND	761.8	726.8

Table 2.5.1-11R (Sheet 7 of 7) Cavities in Boreholes Drilled at Clinch River Nuclear Site, 1973–1978 and 2013

Borehole	CRBRP Strat Unit	SMR Strat Unit	Cavity Top Elev'n (ft)	Cavity Bottom Elev'n (ft)	Cavity Average Elev'n (ft)	Cavity Height (ft)	Water Table Elev'n (ft)	Top of Rock Elev'n (ft)	Bottom of Hole Elev'n (ft)

Notes:

ND = no data

NA = not applicable

Ok/Oma = Ordovician Knox Group/Ordovician Mascot Undif Chk = Undifferentiated Chickamauga

CRBRP borings were drilled from 1973 to 1978 and reported in Reference 2.5.1-100.

Boreholes MP-102 through MP-428 were drilled in 2013 for the CRN SMR Project and reported in Reference 2.5.2-214. Boreholes U1-B05 and U1-B16 were drilled in 2023 for the BWRX-300 SMR Investigation and reported in Reference 2.5.1-341.

Refer to Table 2.5.1-1R for more details.

Replace Table 2.5.1-18 with the following:

Table 2.5.1-18R List of Oil and Gas Well Permits in Roane County, Tennessee

County	EFO Name	Permit Date	Permit No.	API No.	Operator Name	Well Name and No.	Latitude	Longitude	Elevation	Formation and Total Depth
Roane	Knoxville	Jul 26, 2006	0011096	145-20142	Miller Energy Resources, Inc.	Edwards-Gann #1	35.962944	-84.432972	885	Knox Group
Roane	Knoxville	Jun 04, 1981	0004034	145-20001	United American Energy Inc.	Roane Co Industrial Park RCIP #1	35.875222	-84.645028	848.8	Knox Group
Roane	Knoxville	Apr 13, 2005	0010574	145-20002	Enrema, LLC.	Eula Butler Etal #1	35.979111	-84.40775	855.7	Knox Group
Roane	Knoxville	Nov 01, 2005	0010766	145-20141	Enrema, LLC.	Edwards-Fowler Unit #1	35.959694	-84.428611	815	Knox Group

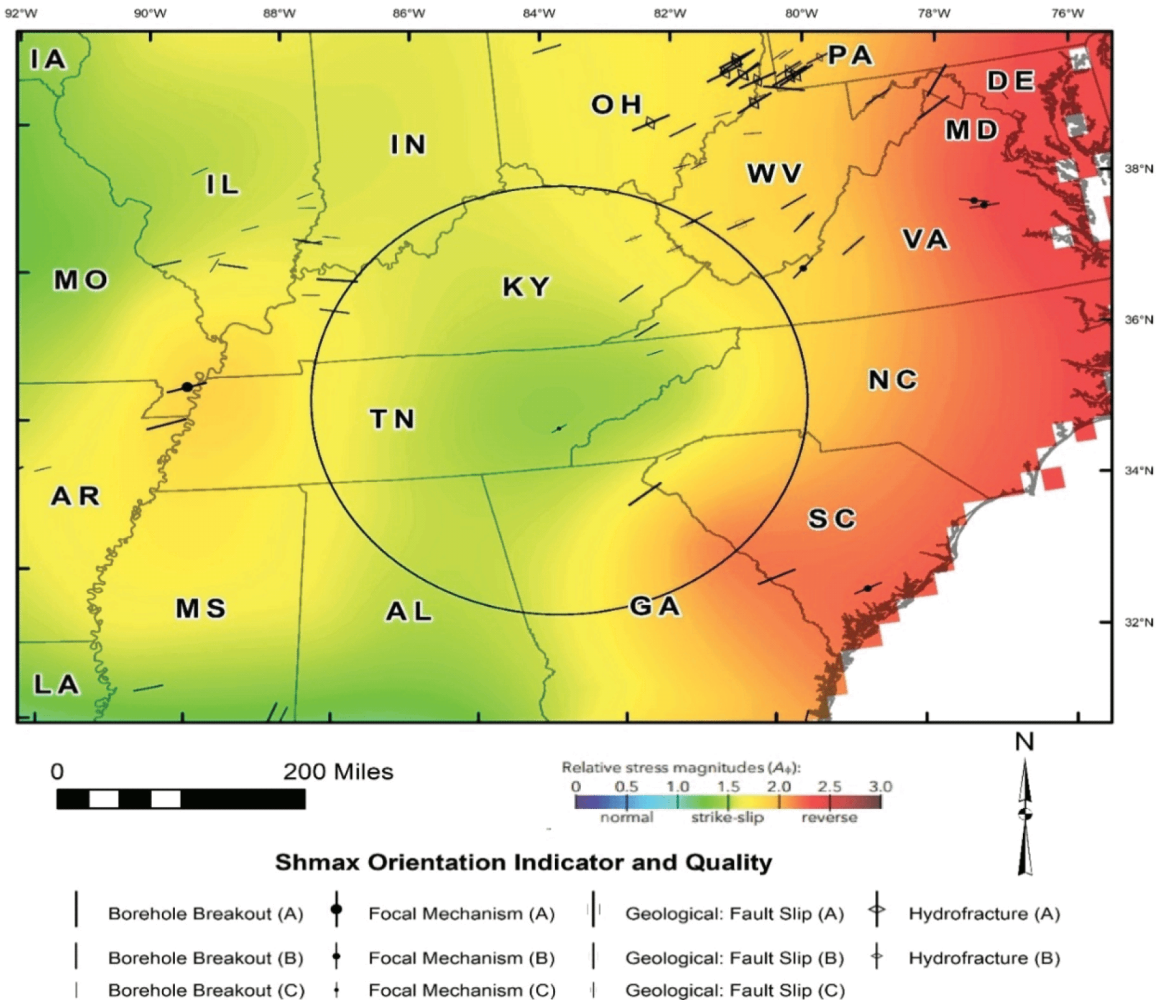
Notes:
Search based upon county field containing “Roane”
Last data refresh on March 27, 2023.
Source: Reference 2.5.1-272.

Add new Table 2.5.1-20:

Table 2.5.1-20 Petroleum Observations During Supplementary Site Investigation

Boring ID	Depth Petroleum Observed (ft-BGS)	FID Value (ppm)
U1-B01	343.1 to 345.6, 364.0 to 366.2	6.0 to 11.2
U1-B02	324.7 to 340.0	14.8 to 436.0
U1-B03	198.8 to 199.6, 386.3 to 394.6	0.0 to 3.1
U1-B05	79.3 to 80.2, 233.3 to 237.5, 249.6 and 256.3	0.7 to 2.7
U1-B08	70.2	0.0
U1-B15	71.2 and 115.0 to 119.0	0.0 to 3.1

Replace Figure 2.5.1-21 (Sheet 2 of 2) with the following:



Notes:

Modified from Reference 2.5.1-304.

SH - maximum horizontal stress.

Source: Reference 2.5.1-304.

Figure 2.5.1-21R (Sheet 2 of 2) Regional Stress Map from Hurd and Zoback

Replace Figure 2.5.1-30 through Figure 2.5.1-31 with the following:

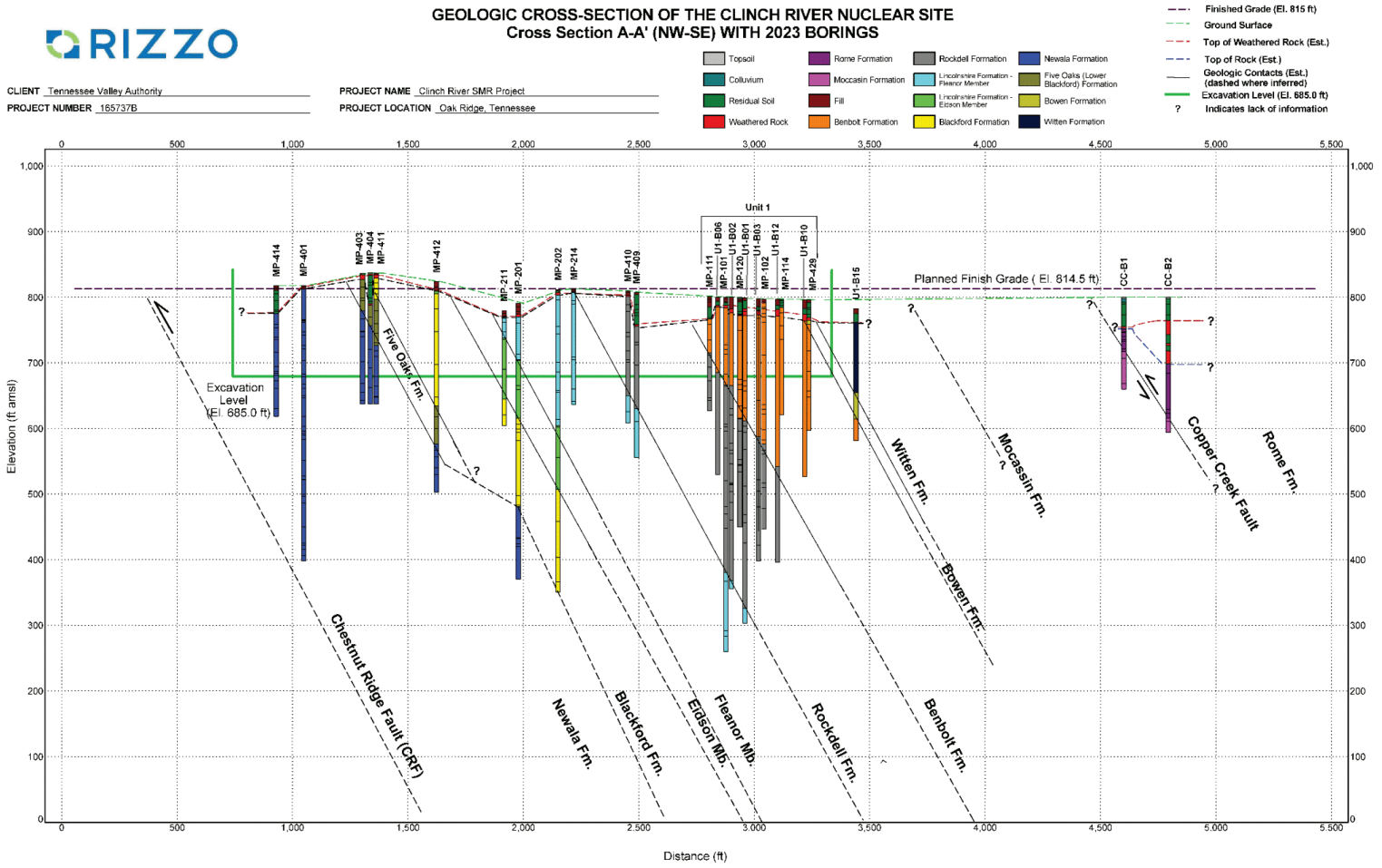


Figure 2.5.1-30R Geologic Cross-Section A-A' of the Clinch River Nuclear Site Depicting the Subsurface Geology and Planned Excavation for the Unit 1 BWRX-300 SMR

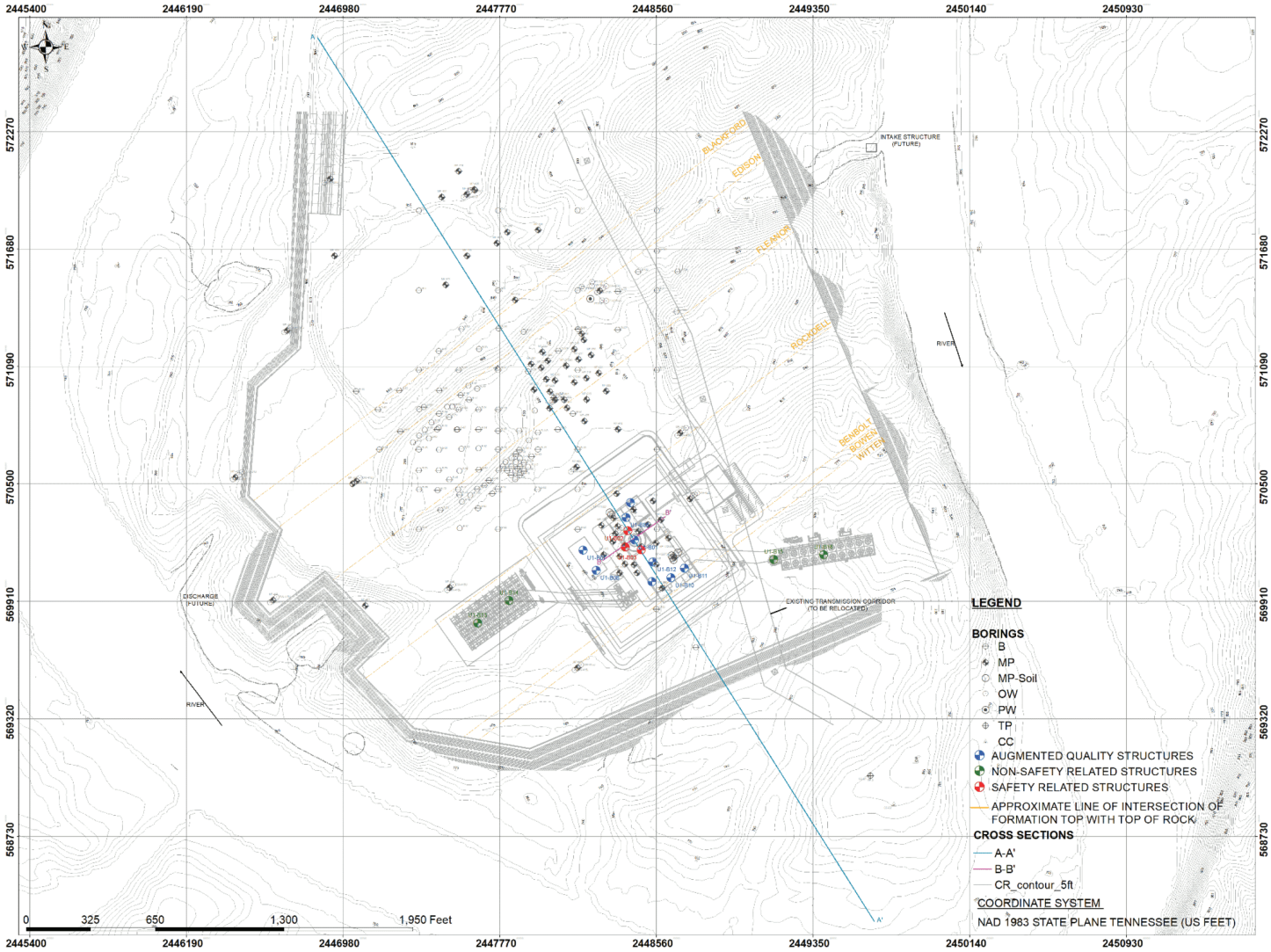
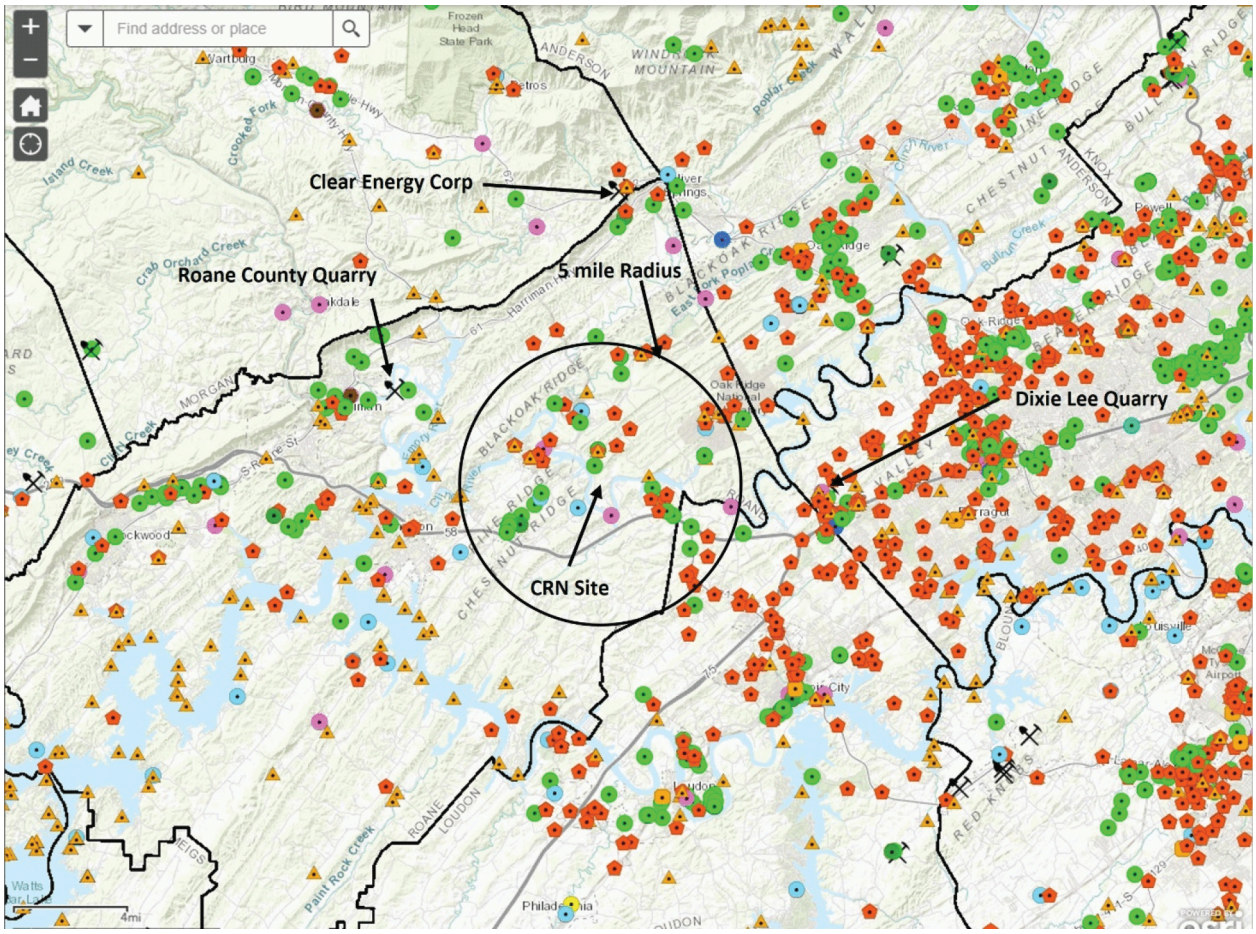


Figure 2.5.1-31R Subsurface Investigation Borehole and Geologic Cross-Section A-A' Locations with Plant Layout

Replace Figure 2.5.1-61 (Sheet 1 of 2) with the following:



Notes:

Source: Reference 2.5.1-273

The symbols shown on this map represent water resource permits for various applications such as construction, water treatment plants, individual wells, hydrostatic purposes etc. See the legend to this figure on the next sheet.

Mine locations are shown as ⚒ and include the following on the map:

Roane County Quarry: Crushed and Broken Limestone

Dixie Lee Quarry: Crushed and Broken Limestone; currently being reclaimed by Vulcan Construction

Oak Ridge Quarry: Crushed and Broken Limestone

Clear Energy Corporation (formerly Dalco of Tennessee LLC.): Bituminous Coal and Lignite, Surface

Figure 2.5.1-61R (Sheet 1 of 2) Map Showing the Water Resource Permit Application Sites Within Approximately 10 Miles of the Clinch River Nuclear Site

Renumber Figure 2.5.1-30 to Figure 2.5.1-30R in the Note of Figure 2.5.1-81.

Add new Figure 2.5.1-93:

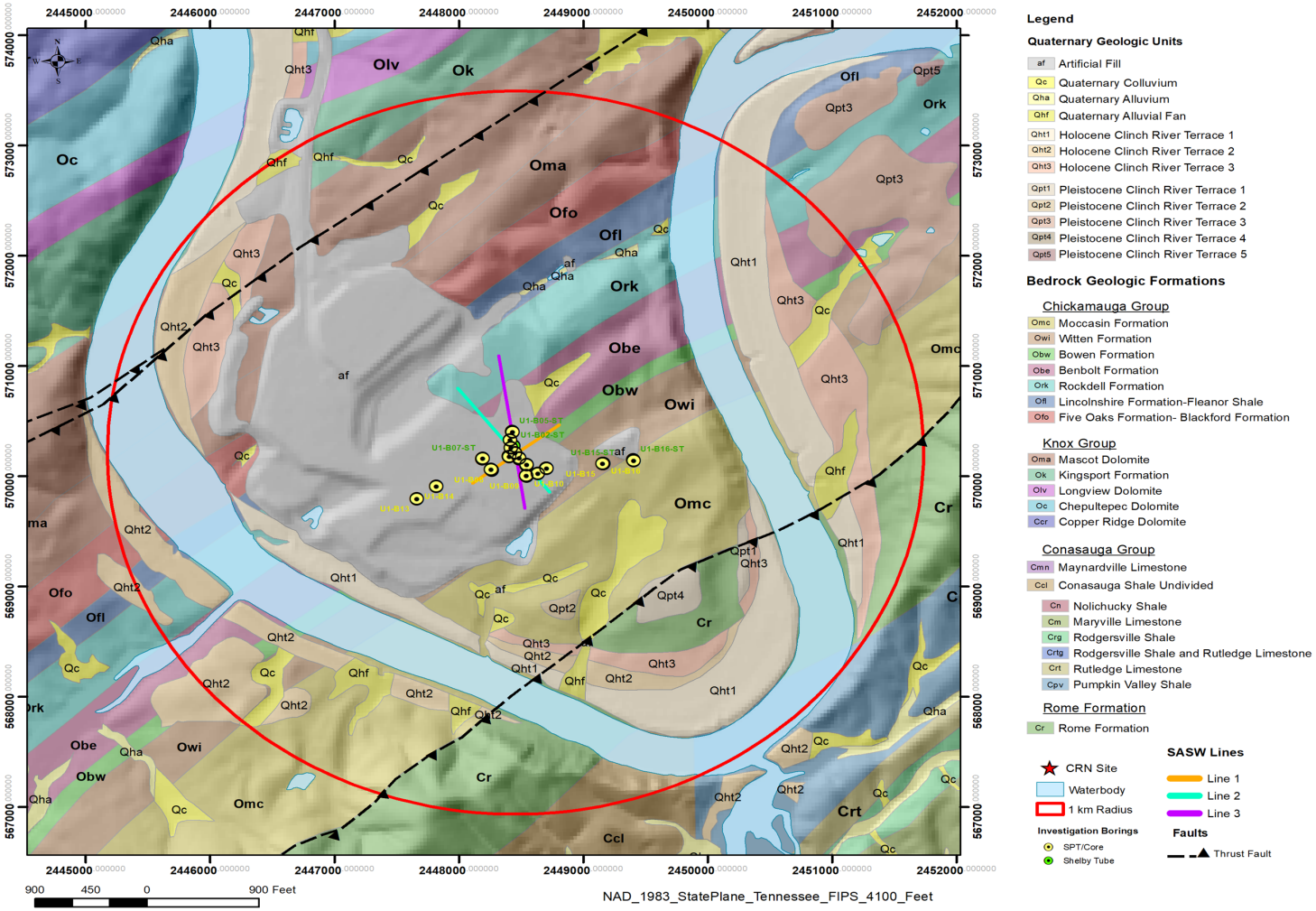


Figure 2.5.1-93 Site Location Geologic Map Showing 2023 Borings and SASW Lines

2.5.2 Vibratory Ground Motion

Replace the first paragraph of ESPA SSAR Section 2.5.2 with the following:

This subsection provides a detailed description of vibratory ground motion assessments that were carried out for the Clinch River Nuclear (CRN) Site. This subsection presents the development of the safe shutdown earthquake (SSE) ground motion response spectra (GMRS) that satisfies the requirements of 10 CFR 100.23, where the GMRS for the power block at the CRN Site was developed by considering a plant parameter envelope (PPE) at Locations A and B (Figure 2.5.4-11). The SSE represents the design earthquake ground motion at the site and is the vibratory ground motion for which certain systems, structures, and components are designed to remain functional. The GMRS is the site-specific ground motion response spectra characterized by horizontal and vertical response spectra determined as free-field motions on the ground surface or as free-field outcrop motions on the uppermost in-situ competent material using performance-based procedures.

In this subsection, the vibratory ground motion is updated to incorporate the requirements for the selected technology and layout at the location of Unit 1 of the CRN Site. Consistent with NEDO-33914-A (Reference 2.5.2-134), three sets of performance-based site-specific response spectra are defined between the foundation level and ground surface elevations, namely, the foundation input, performance-based intermediate and surface response spectra. The foundation input response spectra (FIRS) return the amplitude and frequency content of the design ground motion at the foundation level. The performance-based intermediate response spectra (PBIRS) and surface response spectra (PBSRS) provide the amplitude and frequency content of the design ground motion at the intermediate level and ground surface, respectively. The PBIRS is defined as the outcrop free-field motion at a selected elevation corresponding to significant shear-wave velocity (V_S) contrasts in the subgrade profile. The response spectra were developed with consideration of the guidance provided in SRP 2.5.2, as well as the appropriate sections of Regulatory Guides referenced within that SRP section.

Replace the last sentence of the last paragraph of ESPA SSAR Section 2.5.2 with the following:

Subsection 2.5.2.5.8 describes the development of the horizontal and vertical response spectra (i.e., FIRS, PBIRS, and PBSRS) for Unit 1 at the CRN Site.

2.5.2.1 Seismicity

Replace the last sentence of the third paragraph and the fourth paragraph of ESPA SSAR Section 2.5.2.1 with the following:

For the CRN Site ESPA, the seismicity catalog was one of the potentially significant data sets requiring an update. While the CEUS SSC Report was published in 2012, its seismicity characterization was based on data only through the end of 2008, therefore, a review of more recent seismicity was warranted and was undertaken as part of the CRN Site ESPA.

To evaluate the potential significance of any re-interpretation of past earthquakes and to consider the impact of more recent seismicity, the CEUS SSC earthquake catalog was reviewed and updated for the CEUS for the time period from the beginning of 2009 through mid-September 2013 as part of the CRN Site ESPA. Further review and update of the CEUS SSC earthquake catalog was performed as part of a PSHA for the Y-12 National Security Complex in Oak Ridge, Tennessee (Reference 2.5.2-150) for the time period through 2018. The catalog for the Y-12 PSHA also included catalog revisions documented in EPRI Technical Report 3002005288 (Reference 2.5.2-212).

2.5.2.1.1 Regional Earthquake Catalog Developed for the CEUS SSC Study

Renumber Figure 2.5.2-17 to Figure 2.5.2-17R in EPSA SSAR Section 2.5.2.1.1.

2.5.2.1.2.2.2 Development of a Uniform Moment Magnitude Earthquake Catalog

Replace the first sentence of the first paragraph of ESPA SSAR Section 2.5.2.1.1.2.2 with the following:

An important goal of the development of the CEUS SSC earthquake catalog was to provide an earthquake catalog that could be used to develop unbiased estimates of the recurrence of earthquakes as a function of moment magnitude, as defined by Hanks and Kanamori (Reference 2.5.2-21), consistent with modern GMPEs for the CEUS.

Replace the fifth sentence of the fifth paragraph of ESPA SSAR Section 2.5.2.1.1.2.2 with the following:

If one or more reported magnitudes are directly measured moment magnitudes, then the application of Equations 2.5.2-2 and 2.5.2-3 is done only for the reported moment magnitudes (first adjusted per Equation 2.5.2-1), and other non-moment reported magnitudes (albeit, previously converted to moment magnitudes) are ignored.

2.5.2.1.3 Updated CEUS SSC Earthquake Catalog

Renumber Table 2.5.2-2 to Table 2.5.2-2R in EPSA SSAR Section 2.5.2.1.2.

Insert the following after the last paragraph of ESPA SSAR Section 2.5.2.1.2:

For the Y-12 PSHA, the CEUS earthquake catalog was updated using the CEUS SSC methodology and including events through 2018. The updated catalog for the Y-12 PSHA also includes catalog revisions documented in EPRI Technical Report 3002005288.

Comparison of the smoothed seismicity rates computed for the Y-12 project using the penalized-likelihood approach with post-Y-12 earthquake counts from the ORNL earthquake catalog indicates no significant changes in seismicity rates. As such, the gridded seismicity rates computed for the Y-12 project should be used for the Mmax and seismotectonic zones in the ORNL PSHA.

2.5.2.1.4 Induced Earthquakes

Replace the second sentence of the first paragraph of ESPA SSAR Section 2.5.2.1.3 with the following:

Since about 2009 there has been a notable increase in the number of small to moderate magnitude earthquakes in several regions of the CEUS, including parts of Arkansas, Oklahoma, Kansas, and Texas, largely attributed to human activities involving subsurface fluid injection—including one in November 2011 with a magnitude of 5.6 (Petersen et al.) (Reference 2.5.2-5) and one in September 2016 with a magnitude of 5.8 (Reference 2.5.2-213).

Replace the third sentence of the third paragraph of ESPA SSAR Section 2.5.2.1.3 with the following:

Nevertheless, none of the 17 sites investigated by the USGS (Reference 2.5.2-5) are within 200 mi of the CRN Site, therefore, there are no currently recognized locations within the CRN Site region (200 mi) at which induced seismicity has been noted.

Insert the following after the last paragraph of ESPA SSAR Section 2.5.2.1.3:

The updated earthquake catalog for the Y-12 PSHA includes catalog revisions documented in EPRI Technical Report 3002005288. The EPRI Technical Report 3002005288 earthquake catalog includes updates for reservoir-induced seismicity in the southeastern United States and for some earthquakes related to the large Charleston, South Carolina earthquake in 1886.

2.5.2.1.5 Significant Site Region Earthquakes

Replace the first paragraph of ESPA SSAR Section 2.5.2.1.4:

This subsection presents available information on cataloged earthquakes of E[**M**] 5.0 or greater occurring within 320 km (200 mi) from the CRN Site.

Insert the following after the last paragraph of ESPA SSAR Section 2.5.2.1.4:

August 9, 2020 MW 5.1

Sparta, North Carolina

Distance to CRN Site = 302 km (188 mi)

The August 9, 2020, **M** 5.1 Sparta earthquake located ~ 3 km (~ 2 mi) south of Sparta, North Carolina (Reference 2.5.3-82), approximately 302 km (188 mi) ENE of the CRN Site. The Sparta earthquake was the largest in North Carolina in nearly 100 years and the strongest in the eastern United States since the 2011 **M** 5.8 (E[**M**] 5.71) Mineral, Virginia earthquake. The earthquake sequence included eight foreshocks (**M** 1.8-2.6) during the previous 24 hours before the mainshock and at least 300 aftershocks over the next six months (up to MW 2.9) (Reference 2.5.3-82). The mainshock had a maximum intensity of VI-VII in Sparta and generated

peak ground accelerations of at least 0.2 g. According to the USGS (Reference 2.5.3-86) an intensity of II was recorded at the CRN Site area.

The mainshock hypocenter and focal mechanism have been estimated independently by several groups (Figure 2.5.3-23), namely the Center for Earthquake Research and Information at the University of Memphis (CERI) (Reference 2.5.3-87), the Saint Louis University Earthquake Center (Reference 2.5.3-88) and the United States Geological Survey (Reference 2.5.3-86). While the first two entities calculated the depth of the hypocenter at 0.6 mi (1 km) or less, the USGS hypocenter is now established to be 2.5 mi (4.1 km) (Reference 2.5.3-86) and is the preferred location (Reference 2.5.3-82). Although all three calculations resulted in a NW-SE to WNW-ESE striking plane with dips between approximately 50-60º SW for the focal mechanism, the preferred USGS (Reference 2.5.3-86) orientation is 176, 48º W for the nodal plane (Reference 2.5.3-82). These orientations are all oblique to the regional NE-SW structural trend.

The Sparta earthquake produced the first documented seismic surface rupture in the eastern United States allowing identification of the Little River Fault (Reference 2.5.3-82). Although the Little River Fault produced strong ground shaking and surface rupture, the geomorphology and focal mechanism analysis of fault seismicity is suggestive of an underlying structural control (Reference 2.5.2-223; Reference 2.5.2-225) and evidence for repeated fault movements within the Quaternary have been documented (Reference 2.5.2-224; Reference 2.5.2-225), it is not considered a hazard to the CRN Site. This is primarily because the Little River Fault only has ~2.5 mi (4 km) of documented surface rupture, which is less than the 4.3 mi (7 km) length criteria needed to be considered a seismic hazard by the U.S. Geological Survey National Seismic Hazard Model (NSHM) for the conterminous United States. For this reason, the Little River Fault is not included as a fault section in the updated 2023 National Seismic Hazard Model database (Reference 2.5.2-223).

Furthermore, the Sparta earthquake is not within the ETSZ (Figure 2.5.2-22R) and is not considered related to the same tectonics as the ETSZ, as it was a shallow event (~0.6 mi (1 km) depth) and all events in the ETSZ have deep sources (4.3-15.5 mi (7-25 km) depth) (Reference 2.5.2-226). The Sparta earthquake and Little River Fault are discussed in more detail in Subsection 2.5.3.7.1.

The Sparta earthquake occurred within the CRN Site Region and caused surface rupture along the Little River Fault (Reference 2.5.3-82), indicating that it is likely a capable tectonic source. The Little River Fault is however outside of the ETSZ, and the Sparta earthquake was determined to be shallow (<3 mi (5 km)) (Reference 2.5.3-87) indicating that it does not fit into the characteristics of the ETSZ. The Little River Fault is therefore not considered a ground motion or faulting hazard to the CRN Site. Future earthquakes associated with it are adequately represented by the CEUS SSC distributed seismicity source zones in which it is located.

2.5.2.2 Geologic and Tectonic Characteristics of the Site and Region

Replace the second sentence of the first paragraph of ESPA SSAR Section 2.5.2.1.1.2.2 with the following:

As described in Subsection 2.5.1, a comprehensive review of available geological, seismological, and geophysical data has been performed for the CRN Site region, as well as for portions of seismic sources that extend beyond the CRN Site region.

2.5.2.2.1 Overview of CEUS SSC

Renumber Figure 2.5.2-17 to Figure 2.5.2-17R in EPSA SSAR Section 2.5.2.2.1.

2.5.2.2.2 CEUS SSC Mmax Zones Included in the Clinch River Nuclear Site PSHA

Renumber Figure 2.5.2-18 to Figure 2.5.2-18R in EPSA SSAR Section 2.5.2.2.2.

2.5.2.2.2.1 MESE (Wide and Narrow)

Renumber Figure 2.5.2-18 to Figure 2.5.2-18R in EPSA SSAR Section 2.5.2.2.2.1.

2.5.2.2.2.2 NMESE (Wide and Narrow)

Replace the eighth sentence of the first paragraph of ESPA SSAR Section 2.5.2.2.2.2 with the following:

All Mmax zones, or portions thereof, within 640 km (400 mi) from the CRN Site (double the CRN Site region) are included in the hazard calculation (see Figure 2.5.2-18R).

2.5.2.2.3.1 ECC-AM

Renumber Figure 2.5.2-19 to Figure 2.5.2-19R in EPSA SSAR Section 2.5.2.2.3.1.

Renumber Figure 2.5.2-20 to Figure 2.5.2-20R in EPSA SSAR Section 2.5.2.2.3.1.

2.5.2.2.3.2 Extended Continental Crust-Gulf Coast Zone

Renumber Figure 2.5.2-20 to Figure 2.5.2-20R in EPSA SSAR Section 2.5.2.2.3.2.

2.5.2.2.3.3 PEZ (Wide and Narrow)

Renumber Figure 2.5.2-20 to Figure 2.5.2-20R in EPSA SSAR Section 2.5.2.2.3.3.

2.5.2.2.3.4 Midcontinent Craton (A, B, C, and D)

Renumber Figure 2.5.2-19 to Figure 2.5.2-19R in EPSA SSAR Section 2.5.2.2.3.4.

Replace the third sentence of the second paragraph of ESPA SSAR Section 2.5.2.2.3.4 with the following:

Although only MidC-A and MidC-B intersect the 320-km (200-mi) CRN Site region, all four model alternatives (A, B, C, and D) are included in the hazard calculations.

2.5.2.2.3.5 Illinois Basin Extended Basement

Replace the second sentence of the second paragraph of ESPA SSAR Section 2.5.2.2.3.5 with the following:

The extent of these basement structures, however, is poorly constrained. Only a small portion of the IBEB zone lies within the CRN Site region (Figure 2.5.2-12).

2.5.2.2.4 CEUS SSC RLME Sources Included in the CRN Site PSHA

Renumber Figure 2.5.2-17 to Figure 2.5.2-17R in EPSA SSAR Section 2.5.2.2.4.

Renumber Figure 2.5.2-22 to Figure 2.5.2-22R in EPSA SSAR Section 2.5.2.2.4.

2.5.2.2.4.1 New Madrid Fault System

Renumber Figure 2.5.2-17 to Figure 2.5.2-17R in EPSA SSAR Section 2.5.2.2.4.1.

Renumber Figure 2.5.2-22 to Figure 2.5.2-22R in EPSA SSAR Section 2.5.2.2.4.1.

Renumber Figure 2.5.2-24 to Figure 2.5.2-24R in EPSA SSAR Section 2.5.2.2.4.1.

Renumber Figure 2.5.2-25 to Figure 2.5.2-25R in EPSA SSAR Section 2.5.2.2.4.1.

2.5.2.2.4.2 Charleston

Renumber Figure 2.5.2-17 to Figure 2.5.2-17R in EPSA SSAR Section 2.5.2.2.4.2.

Renumber Figure 2.5.2-22 to Figure 2.5.2-22R in EPSA SSAR Section 2.5.2.2.4.2.

Renumber Figure 2.5.2-23 to Figure 2.5.2-23R in EPSA SSAR Section 2.5.2.2.4.2.

2.5.2.2.5 Post-CEUS SSC Studies

Replace the second sentence of the first paragraph of ESPA SSAR Section 2.5.2.2.5 with the following:

The Technical Integration (TI) Team reviewed scientific literature published since 2012 (and earlier), contacted experts who have developed data and/or interpretations of seismic sources in the CRN Site region, reviewed an updated seismicity catalog developed for this project, and performed site-specific studies, as needed, to assess the quality of data and uncertainty associated with recently published studies.

Replace the fourth and fifth sentences of the second paragraph of ESPA SSAR Section 2.5.2.2.5 with the following:

Updating the catalog added 157 events of E(**M**) > 2.2, of which 9 events are E(**M**) > 2.9, to the CRN Site region (Table 2.5.2-1). The largest event added to the catalog is the August 23, 2011 **M** 5.8 Mineral, Virginia earthquake (E(**M**) 5.7), which occurred beyond the CRN Site region but which lies within areal source zones of the CEUS SSC model that extend into the CRN Site region.

Insert the following after the last paragraph of ESPA SSAR Section 2.5.2.2.5:

Subsequent SSHAC Level 2 studies performed for other sites in the region (i.e., Savannah River Site (SRS), the Y-12 National Security Complex (Y-12), and Oak Ridge National Laboratory (ORNL)) indicate the need to update other elements of the CEUS-SSC model for use at the CRN Site. These updates are needed for the New Madrid Seismic Zone (Subsection 2.5.2.2.5.3) and the Charleston Seismic Zone (Subsection 2.5.2.2.5.4). Evaluations of new data and investigations from the Eastern Tennessee Seismic Zone (ETSZ) indicate that no changes to the CEUS-SSC model are required for the characterization of the ETSZ (Subsection 2.5.2.2.5.1).

Also, as mentioned in Sections 2.5.2.1.3 and 2.5.2.1.4, for the Y-12 PSHA, the earthquake catalog was updated using the CEUS SSC methodology to include events through 2018. The update also included revisions related reservoir-induced and Charleston-era earthquakes as documented in Reference 2.5.2-212.

Rename and Replace ESPA SSAR Section 2.5.2.2.5.1 with the following:

2.5.2.2.5.1 Investigations of the Eastern Tennessee Seismic Zone

The ETSZ is an approximately 300- to 400-km-long and 50-km-wide, northeasterly trending, elongate band of seismicity in the Valley and Ridge and western Blue Ridge physiographic provinces (Reference 2.5.2-128; Reference 2.5.2-173) (Figure 2.5.2-26). The ETSZ underlies eastern Tennessee, and parts of North Carolina, Georgia, and Alabama (e.g., Reference 2.5.2-130; Reference 2.5.2-133; Reference 2.5.2-128; Reference 2.5.2-90). After the New Madrid seismic zone, the ETSZ has the second highest rate of small (**M** < 5) earthquakes in

the eastern U.S. but what localizes modern strain release in eastern Tennessee is still an open question (Reference 2.5.2-135; Reference 2.5.2-141).

In the CEUS-SSC model, the ETSZ was not modeled as a distinct seismic source, due to the lack of evidence for repeated, large-magnitude earthquakes or other distinguishing characteristics. Instead, the CEUS-SSC model uses the spatial smoothing of seismicity to maintain the locally high seismicity rates in the ETSZ, and assigns a broad distribution on magnitude, including very large-magnitude earthquakes, for those distributed seismicity sources that host the ETSZ. Thus, the CEUS-SSC model allows for relatively frequent and large-magnitude earthquakes in the eastern Tennessee region.

Studies of the ETSZ that post-date development of the CEUS-SSC model have examined the deep crustal structure (Reference 2.5.2-148; Reference 2.5.2-149; Reference 2.5.2-155) and the shallow crustal structure (Reference 2.5.2-158 Reference 2.5.2-157). Reference 2.5.2-159) suggest that the 2012 **M** 4.2 Perry County, Kentucky, earthquake is related to the ETSZ, thus extending the zone farther north than previously recognized. Yang and Salem (Reference 2.5.2-160) used Permanent Scatterers Synthetic Aperture Radar Interferometry (PSInSAR) data to suggest that land surface deformation may be spatially associated with areas of higher seismicity rate in eastern Tennessee, but caution that many other processes besides earthquake activity can contribute to surface deformation. Gallen and Thigpen (Reference 2.5.2-162) present a model in which lithological focusing of transient erosion in the Upper Tennessee drainage basin has augmented the local stress field through redistribution of surface loads, allowing preexisting basement structures to slip in the regional stress field.

Several paleoseismic studies in the ETSZ post-date the CEUS-SSC. Hatcher et al. (Reference 2.5.2-88) interpreted potential paleoseismic features near Douglas Reservoir as evidence for at least two **M** ≥ 6.5 earthquakes in the ETSZ in the last approximately 73-112 ka. Related and other studies are published as a Geological Society of America (GSA) field trip guidebook (Reference 2.5.2-163), student theses (e.g., Reference 2.5.2-89; Reference 2.5.2-168), numerous abstracts (e.g., Reference 2.5.2-214; Reference 2.5.2-215; Reference 2.5.2-216; Reference 2.5.2-217; Reference 2.5.2-172), and elsewhere.

A recent paper by Cox et al. (Reference 2.5.2-173) summarizes and synthesizes the relevant paleoseismic and geomorphic information, so much of the discussion that follows relies on this recent synthesis paper. Building on previous studies, Cox et al. (Reference 2.5.2-173) proposed an approximately 80- to 110-km-long, northeast-trending zone of active faulting in the ETSZ that extends from approximately the Douglas Reservoir near Dandridge, Tennessee on the northeast to at least Vonore, Tennessee on the southwest. They named this the Dandridge-Vonore fault corridor. Along this trend they interpret Quaternary-active reverse, strike-slip, and normal faults from trench excavations and natural exposures. At the Dandridge site, a thrust fault exposed in French Broad River alluvium previously identified by Hatcher et al. (Reference 2.5.2-88) and Warrell et al. (Reference 2.5.2-217) is interpreted as recording two earthquakes in the last 40 ka. Approximately 50 km southwest near Alcoa, Tennessee, Cox et al. (Reference 2.5.2-173) interpret evidence for two earthquakes between 15 and 10 ka from a riverbank exposure of a thrust fault that cuts Little River alluvium. Approximately 30 km farther southwest near Vonore, Tennessee, they interpret thrust and normal faults on the north and south sides of the Little

Tennessee River valley. They interpret the thrust fault displaces Quaternary alluvium ≥ 2 m over bedrock, and the alluvium is normal faulted > 2 m. Based on these observations, Cox et al. (Reference 2.5.2-173) interpret at least two **M** ≥ 6.5 earthquakes in the Dandridge-Vonore fault corridor in the last 40 ka. If this interpretation is correct, it would suggest the need for an RLME source to model future earthquakes in the Dandridge-Vonore fault corridor.

A recent paper by USGS and USNRC geologists documents their evaluation of evidence for and against tectonic surface deformation in the ETSZ (Reference 2.5.2-174). They conclude that evidence for tectonic surface deformation remains elusive in the ETSZ. Thompson Jobe et al. (Reference 2.5.2-174) visited Cox et al.'s (Reference 2.5.2-173) Alcoa paleoseismic site on the Little River, where Cox et al. (Reference 2.5.2-173) interpreted two late Pleistocene earthquakes based on fault relationships exposed along the riverbank. Thompson Jobe et al.'s (Reference 2.5.2-174) preferred interpretation of this same site is that it records a series of cut and fill channels with some mass wasting and bank-collapse features that were deposited on top of a bedrock strath surface. In other words, Thompson Jobe et al. (Reference 2.5.2-174) interpret that the exposure is likely the result of fluvial processes, as opposed to young faulting.

The results of the ongoing paleoseismic studies in the ETSZ are preliminary and do not provide adequate evidence for defining the ages and repeat times of earthquakes and therefore do not provide adequate evidence for defining an RLME seismic source for the ETSZ. This conclusion is consistent with the 2023 version of the National Seismic Hazard Model (NSHM23; (Reference 2.5.2-175; Reference 2.5.2-176), which also excludes the ETSZ as a discrete seismic source. Additional rationale for excluding the ETSZ as a source in the SSC for the CRN Site is provided by hazard sensitivity tests, as described in Subsection 2.5.2.2.6.1.3.

2.5.2.2.5.2 Investigations of the 2011 Mineral, Virginia Earthquake

Insert the following after the last paragraph of ESPA SSAR Section 2.5.2.2.5.2:

Also, the potential source does not exhibit evidence of two or more earthquakes with **M** ≥ 6.5.

Add new Section 2.5.2.2.5.3:

2.5.2.2.5.3 Investigations of the New Madrid Seismic Zone

The New Madrid Seismic Zone (NMSZ) is an area of elevated seismicity rate in the northern Mississippi embayment in parts of Missouri, Arkansas, Tennessee, Mississippi, and Illinois. The New Madrid Fault System (NMFS) is the source of the 1811–1812 New Madrid earthquake sequence, which includes three of the largest earthquakes to have occurred within the North American plate in historical times (Reference 2.5.2-178).

In the CEUS-SSC model, the NMSZ is modeled by five RLME sources: the New Madrid Fault System (NMFS), Eastern Rift Margin-North (ERM-N), Eastern Rift Margin-South (ERM-S), Marianna, and Commerce Fault Zone (Commerce) sources (Figure 2.5.2-22R). The magnitudes and rates for these RLMEs are based in part on paleoseismic and paleoliquefaction records. In the CEUS-SSC, the NMFS RLME is modeled by fault sources, the ERM-N, ERM-S, and

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Commerce RLMEs are modeled as narrow, elongated “fault-like” source zones, and the Marianna RLME is modeled as a roughly square-shaped source zone.

Since publication of the CEUS-SSC model in 2012, several studies provide relevant new information on the NMSZ seismic sources, and this information was used to update the NMSZ RLME sources for the CRN Site. Much of that new information is summarized in the documentation of the NSHM23 (Reference 2.5.2-223; Reference 2.5.2-175; Reference 2.5.2-176). Subsection 2.5.2.2.5.3.1 summarizes new information on the NMFS RLME. Subsection 2.5.2.2.5.3.2 summarizes new information on the other four RLMEs in the NMSZ (ERM-N, ERM-S, Commerce, and Marianna). Subsection 2.5.2.2.5.3.3 summarizes changes to the NMSZ RLMEs based on this new information.

2.5.2.2.5.3.1 New Information on the New Madrid Fault System (NMFS) RLME

This subsection describes new data that post-date the CEUS-SSC for the NMFS RLME, and is organized into discussions of source geometry, magnitude, and recurrence.

Geometry: In the CEUS-SSC, the fault sources that compose the NMFS RLME include the New Madrid South fault, the New Madrid North fault, and the Reelfoot thrust (Figure 2.5.2-22R). Uncertainties in the locations, extents, and rupture combinations of these faults are included in a complex logic tree in the CEUS-SSC for the New Madrid Fault System RLME (Figure 2.5.2-22R).

For the NMFS, the NSHM23 model updates the names of some and geometries of all the constituent faults, as shown in Figure 2.5.2-24R. For example, the Blytheville Arch (BA) fault name is changed to the Axial Fault (AF), and it is extended approximately 35 km farther southwest based on seismic reflection data in Reference 2.5.2-179. Similarly, the short and long depictions of the Reelfoot Thrust fault source (RFT-S and RFT-L, respectively) are renamed the New Madrid West (NMW) and Reelfoot faults, with minor changes to their geometries. These new names and, with one exception, geometries are adopted for use in this project. Consistent with NSHM23, the southeastern extent of the Reelfoot fault has been moved to the southwest to align with new observations of deformed late Pleistocene terraces (Reference 2.5.2-180). One additional change was made to the to the geometry of the Reelfoot fault source for use at the CRN Site. In the NSHM23 model, the NMW and Reelfoot faults are discontinuous and do not connect with one another. The USGS PSHA software allows fault ruptures to “jump” across fault discontinuities, but the PSHA software used for this project does not. For this reason, the northwest end of the Reelfoot fault source was modified so that it directly connects with the NMW source to allow for combined ruptures on these two faults (Figure 2.5.2-24R). Given the distance to the CRN Site, this minor change in source geometry for the Reelfoot fault is assumed to be insignificant to hazard.

Magnitude: In the CEUS-SSC model, the NMFS RLME is characterized with a broad Mmax distribution that ranges from **M** 6.7–7.9 (Table 2.5.2-11). Since publication of the CEUS-SSC model, several studies provide new information on earthquake magnitudes for the NMFS. For example, Boyd and Cramer (Reference 2.5.2-182) derived a new macroseismic intensity prediction equation for the central and eastern U.S. to estimate magnitudes of the 1811–1812 New Madrid earthquakes. Their mean estimate ranges are near the lower range of estimates from previous studies and include **M** 6.9–7.1 for the 16 December 1811 earthquake, **M** 6.8–7.1

for the 23 January 1812 earthquake, and **M** 7.3–7.6 for the 7 February 1812 earthquake. Boyd and Cramer (Reference 2.5.2-182) caution that there is significant uncertainty associated with these estimates, and that one-sigma uncertainties could be as large as 0.3-0.4 magnitude units.

Cramer and Boyd (Reference 2.5.2-183) evaluated intensity data at large distances to estimate magnitudes for the historical New Madrid earthquakes that are slightly greater than the estimates from Boyd and Cramer (Reference 2.5.2-182). Cramer and Boyd's (Reference 2.5.2-183) magnitude estimates for the three 1811-1812 mainshocks are: **M** 7.5 for the 16 December 1811 earthquake, **M** 7.3 for the 23 January 1812 earthquake, and **M** 7.7 for the 7 February 1812 earthquake. For each event, Cramer and Boyd (Reference 2.5.2-183) estimate an uncertainty of 0.3 magnitude units at the 95% confidence level.

Chung and Rogers (Reference 2.5.2-184) reevaluated intensity data from southeast Missouri to calculate a best-estimate magnitude for the 7 February 1812 earthquake of **M** 7.7. This estimate is the same as that from Cramer and Boyd (Reference 2.5.2-183) for the same event.

Ramirez-Guzman et al. (Reference 2.5.2-185) performed a suite of numerical simulations based on the 1811–1812 earthquakes, and these simulations included 3-D geologic structures and rupture directivity effects out to distances of about 500 km. They conclude that basin response and rupture directivity effects have significant impacts on patterns and levels of estimated intensities, which lead to additional uncertainty not previously considered in magnitude estimates for the 1811–1812 earthquake sequence based only on historical reports. As such, the Ramirez-Guzman et al. (Reference 2.5.2-185) study suggests that a broad range in magnitude, such as is currently used in the CEUS-SSC model, is appropriate for the NMFS RLME.

The magnitude distributions for the NMFS from the CEUS-SSC (**M** 6.7–7.9) and the NSHM23 (**M** 6.8–8.0) are in close agreement, with the weighted mean magnitude from the CEUS-SSC slightly higher.

The new magnitude data described above fall within the range of values currently included in the CEUS-SSC model for the NMFS RLME. Thus, no revisions to the CEUS-SSC are required to account for these new magnitude data.

Recurrence: Several new studies provide recurrence information for the NMFS. For example, Page and Hough (Reference 2.5.2-186) examined historical and instrumental seismicity in the New Madrid region to determine whether present-day activity is predominantly aftershocks of the 1811–1812 sequence. They conclude that the ongoing seismic activity is not entirely explained as aftershocks and that distributed (background) seismicity is driven by ongoing strain accumulation, albeit at low rates.

Craig and Calais (Reference 2.5.2-187) evaluated approximately 15 years of continuous GPS data in the New Madrid region to conclude that motions are consistently within the 95% confidence limit of zero deformation. Their data place an upper bound on strain accrual of 0.2 mm/yr for faults in the New Madrid seismic zone. Craig and Calais (Reference 2.5.2-187) argue that the low strain accrual rate is inconsistent with the 500–900-year repeat time of paleoearthquakes in mid- to late Holocene time, and that their results are consistent with temporal clustering of earthquake behavior on the New Madrid fault system.

Boyd et al. (Reference 2.5.2-188) modeled Global Navigation Satellite System (GNSS) data from 2000–2014 across the New Madrid region to evaluate between several deformation mechanisms. They indicate that many of the recent earthquakes appear to be part of a long-lived aftershock sequence related to the 1811–1812 earthquakes, and that present strain accrual rates are very low to zero. However, they caution that their results are consistent with temporal clustering of large earthquakes and that seismic hazard likely remains significant.

New paleoseismic event ages in the NMFS are provided in recent studies by Tuttle et al. (Reference 2.5.2-220) and Gold et al. (Reference 2.5.2-189). Tuttle et al. (Reference 2.5.2-220) suggest a newly recognized liquefaction event at approximately A.D. 0 and revised timing of an 1811–1812-like compound event at approximately 1050 B.C. In the CEUS-SSC model for the NMFS “All In” branch, earthquake recurrence is computed based on the most-recent three events only (i.e., A.D. 1811–1812, 1450, and 900) because the older portion of the paleoliquefaction record was judged to be incomplete. For the CRN PSHA, it was concluded that the older portion of the record remains incomplete, the results of Tuttle et al. (Reference 2.5.2-220) notwithstanding. In another recent study, Gold et al. (Reference 2.5.2-189) provide event ages for sackungen features in the NMSZ and they observe that the timing of these sackungen-forming events corresponds to previously documented earthquakes in the region.

As such, based on evaluations of new studies by Tuttle et al. (Reference 2.5.2-220), Gold et al. (Reference 2.5.2-189), and others, no changes were made to the recurrence data of large-magnitude earthquakes in the NMFS RLME, which was based on the record of the three most-recent clusters of events that comprise the “complete” portion of the paleoseismic record.

The CEUS-SSC includes two alternative models for large-earthquake recurrence of the NMFS RLME source: (1) the Poisson (time independent) model; and (2) the Brownian Passage Time or “renewal” (time-dependent) model. The Poisson model assumes that the average time between events is known, but the exact timing of events is independent of the time since the last and previous events. A renewal model for the occurrence probability of earthquakes is appropriate when the data suggest a more periodic recurrence behavior than is implied by the Poisson model. Renewal models typically are used where a fault is subjected to quasi steady-state loading, such as may occur in a plate boundary setting. In the CEUS-SSC, much greater weight was given to the Poisson model (0.75 weight) than the renewal model (0.25 weight) when on the “All In” branch for the NMFS (Figure 2.5.2-25R).

Faults in the NMFS RLME are reasonably well understood, and the recurrence of large-magnitude earthquakes appears quasi-periodic, at least when in a cluster. The strain loading mechanism for the NMFS remains unclear, but the strain release rates over the last several earthquake cycles are comparable to some plate boundary settings. For these reasons, the CEUS-SSC model weights were revised to give equal weights to the Poisson and renewal models of earthquake recurrence for the NMFS when on the “All In” branch of the NMFS logic tree. Given the apparent quasi-periodic behavior of the NMFS over the last few earthquake cycles and the relatively fast loading rate implied by the high rate of earthquake occurrence, this increased weight to the renewal model is justified.

2.5.2.2.5.3.2 New Information on the ERM-N, ERM-S, Commerce, and Marianna RLMEs

This subsection summarizes new information that post-dates CEUS-SSC on the ERM-N, ERM-S, Commerce, and Marianna RLME sources.

ERM-N and ERM-S: In the CEUS-SSC, the ERM-N and ERM-S RLME sources are modeled as narrow, elongated “fault-like” source zones. These northeast-trending zones are “leaky” at both ends such that ruptures that nucleate within the zones are allowed to extend partially outside the zones. In the NSHM23, the USGS has collapsed these polygonal source zones into fault sources with single traces, given the relatively good definition of the fault at the surface and at depth (Figure 2.5.2-24R) (Reference 2.5.2-223; Reference 2.5.2-176). In the NSHM23, the southern extent of the ERM-N source is extended 12 km to the south. Magnitudes and rates for the ERM-N and ERM-S sources are based on historical and paleoliquefaction records, as opposed to fault dimensions, so no changes are needed for magnitudes and rates. In the CEUS-SSC, the names of the alternative southern extensions of ERM-S are Crittenden County (CC) and River Picks (RP); in the NSHM23, the RP name is changed to Meeman-Shelby. These changes to geometry and names are adopted for the CRN Site.

Commerce: In the CEUS-SSC, the Commerce RLME source is modeled as a narrow, elongated “fault-like” source zone. This northeast-trending zone is “leaky” at both ends such that ruptures that nucleate within the zone are allowed to extend partially outside the zone. In the NSHM23, the USGS has collapsed this polygonal source zone to a fault source with a single 75-km-long trace (Figure 2.5.2-24R), given the relatively good definition of the fault at the surface and at depth (Reference 2.5.2-223; Reference 2.5.2-176). This change to geometry is adopted for the CRN Site.

Marianna: The Marianna RLME source is based mainly on paleoliquefaction evidence for repeated, large-magnitude earthquakes. In the Marianna region, Odum et al. (Reference 2.5.2-195) collected new seismic reflection data across the Daytona Beach lineament, a northwest-trending, >10-km-long tonal lineament seen in aerial photographs. From these data they interpret a fault zone that offsets Tertiary and possibly Quaternary units. Tuttle et al. (Reference 2.5.2-197) document the identification and dating of large sand blows along the Daytona Beach lineament, including three Holocene events and at least two late Pleistocene events. Tuttle et al. (Reference 2.5.2-197) speculate that the paleoliquefaction may have formed by large earthquakes on the fault that underlies the Daytona Beach lineament.

In their documentation of earthquake geology inputs for the NSHM23, Thompson Jobe et al. (Reference 2.5.2-223) conclude that it is unclear whether the Daytona Beach lineament produced any of the documented paleoliquefaction and that considerable uncertainty remains regarding the locations, extents, and activities of other potentially active faults in the Marianna RLME source zone (Figure 2.5.2-24R). For these reasons, no updates were made to the NSHM23 characterization of the Marianna source (Reference 2.5.2-176). The new data from the Marianna area are interesting, but currently no changes to the CEUS-SSC model of the Marianna RLME source are required for the CRN Site.

2.5.2.2.5.3.3 Summary of Changes to CEUS-SSC

This section describes changes to the CEUS-SSC models for the five RLME sources in the NMSZ that are documented in Reference 2.5.2-265.

NMFS RLME: Two changes have been implemented. The first is to adopt the changes to fault source names and geometries based on the NSHM23, as shown in Figure 2.5.2-24R. The second is to assign equal weight to the Poisson and renewal models of time dependence for the "All In" branch of the NMFS RLME.

ERM-N, ERM-S, and Commerce RLMEs: In the NSHM23, the geometries of the ERM-N, ERM-S, and Commerce RLME sources are changed from narrow, elongated source zones to fault sources (Figure 2.5.2-28R). For use at the CRN Site, the revised fault source geometries are adopted from NSHM23 for the ERM-N, ERM-S, and Commerce RLME sources. Besides their geometries, no other changes are made to these three RLME sources.

Marianna RLME: In the NSHM23, there are no changes to the CEUS-SSC characterization of the Marianna RLME. We are not aware of any significant new data for the Marianna RLME. As such, the Marianna RLME source is adopted as-is from the CEUS-SSC for use at the CRN Site.

Add new Section 2.5.2.2.5.4:

2.5.2.2.5.4 Investigations of the Charleston Seismic Zone

The Charleston seismic zone is an area of elevated seismicity rate near Charleston, South Carolina. The CEUS-SSC characterization of the Charleston RLME source was recently updated as part of an augmented SSHAC Level 2 study performed for the Savannah River Site (SRS) by Lettis Consultants International, Inc. (Reference 2.5.2-198). This subsection summarizes the changes, and the rationale for those changes, to the Charleston RLME that were used at the SRS and are adopted for use at the CRN Site. The SRS augmented SSHAC Level 2 report provides more complete documentation (Reference 2.5.2-198). Data that post-date the SRS augmented SSHAC Level 2 study are summarized in Subsection 2.5.2.2.5.4.1.

Magnitude: In the CEUS-SSC model, the magnitude distribution for future Charleston earthquakes is based on magnitude estimates for the 1886 and prehistoric Charleston earthquakes. This distribution ranges from **M** 6.7–7.5, with a weighted mean of **M** 7.1 (Table 2.5.2-10). Since publication of the CEUS-SSC, several studies have provided additional information on magnitudes of the 1886 and prehistoric 1886-type Charleston earthquakes. For example, Boyd and Cramer (Reference 2.5.2-182) derived a new macroseismic intensity prediction equation for the CEUS to estimate a magnitude of **M** 6.9 ± 0.3 for the 1886 earthquake. Similarly, Cramer and Boyd (Reference 2.5.2-183) estimated a magnitude of **M** 7.0 ± 0.3 for the 1886 event.

Gheibi and Gassman (Reference 2.5.2-199) analyzed geotechnical data from the Hollywood, South Carolina paleoliquefaction site to estimate a minimum earthquake magnitude range of **M** 5.6–6.5 for prehistoric earthquakes that triggered sand blows in the Charleston seismic source zone. More recently, Gheibi et al. (Reference 2.5.2-200) analyzed geotechnical data from several

paleoliquefaction sites in the Charleston area to estimate minimum earthquake magnitudes. Based on data from the Hollywood, Sampit, and Gapway sites, Gheibi et al. (Reference 2.5.2-200) estimated a minimum earthquake magnitude range of **M** 5.5–7.2 for prehistoric earthquakes in the Charleston seismic source zone. Based on data from the Fort Dorchester site, Gheibi et al. (Reference 2.5.2-201) estimated a minimum earthquake magnitude range of **M** 5.0–6.3, and Gheibi et al. (Reference 2.5.2-200) estimated a minimum earthquake magnitude range of **M** 5.1–5.7.

Taken together, the new data on the magnitude of the 1886 earthquake and on the minimum magnitudes of liquefaction-inducing prehistoric earthquakes are consistent with the magnitude distribution defined in the CEUS-SSC for the Charleston RLME source. As such, no revisions to the CEUS-SSC are required to account for the new magnitude information.

Fault Geometry and Style: The CEUS-SSC defines three alternative geometries for the Charleston RLME source, based on the distribution of paleoliquefaction features, instrumental seismicity, and subsurface faults interpreted from seismic reflection and other data (Figure 2.5.2-22R). The “Local” source alternative was designed to include the 1886 meizoseismal area, concentrated microseismicity in the Middleton Place-Summerville seismic zone, the preponderance of 1886 liquefaction features, and numerous postulated faults. In the CEUS-SSC model, the Local source alternative is characterized by:

- A relatively high weight of 0.5
- Future earthquake ruptures that are oriented northeast, parallel to the long axis of the zone and the regional structural grain of the Appalachian orogen
- Future ruptures that are modeled as occurring on vertical strike-slip faults
- All boundaries of the Charleston Local source are strict, such that ruptures are not allowed to extend beyond the zone boundaries

The “Narrow” source alternative was defined to contain postulated north-northeast-striking structures such as the Woodstock fault, the southern segment of the East Coast fault system, and Chapman and Beale’s (Reference 2.5.2-83) “fault C.” In the CEUS-SSC model, the Narrow source alternative is characterized by:

- A moderate weight of 0.3
- Future earthquake ruptures that are oriented north-northeast, parallel to the long axis of the zone
- Future ruptures that are modeled as occurring on vertical strike-slip faults
- The northeast and southwest boundaries of the Narrow source are leaky, such that ruptures that initiate within the zone are allowed to extend beyond the zone boundaries in those directions, but by no more than 50% of the rupture length beyond the boundary. The northwest and southeast boundaries of the Charleston Narrow source are strict, such that ruptures are not allowed to extend beyond the zone boundaries to the northwest and southeast.

The “Regional” source alternative is a relatively large area that envelops both the Local and Narrow alternatives, covers most of coastal South Carolina, and extends offshore to include the Helena Banks fault zone. In the CEUS-SSC model, the Regional source alternative is characterized by:

- A relatively low weight of 0.2
- Future ruptures that are oriented either: (1) parallel to the long axis of the source (northeast) with 0.8 weight; or (2) parallel to the short axis of the source (northwest) with 0.2 weight
- Future ruptures that are modeled as occurring on vertical strike-slip faults
- All boundaries of the Charleston Regional source are strict, such that ruptures are not allowed to extend beyond the zone boundaries

Several studies that post-date the CEUS-SSC model provide new information on possible fault locations, geometries, and styles for the Charleston RLME source. Pratt et al. (Reference 2.5.2-202) interpreted reprocessed seismic reflection data in the epicentral area of the 1886 earthquake and identified generally steeply dipping faults that cut Cretaceous- and Cenozoic-age Coastal Plain strata. Pratt et al. (Reference 2.5.2-202) describe two main faults in the Charleston epicentral area, both of which are interpreted as striking northeast. The first was previously identified by Chapman and Beale (Reference 2.5.2-83) and was incorporated into development of the CEUS-SSC model. The second is located approximately 15 km northwest of Charleston where it crosses the Ashley River. Based on the steep fault dips and lack of detectable surface scarps for the 1886 event, Pratt et al. (Reference 2.5.2-202) suggest a component of strike-slip for faulting but acknowledge that some modern focal mechanisms indicate thrust motion. Pratt et al.’s (Reference 2.5.2-202) northeast-striking, steeply dipping faults are consistent with the characterization of the location, strike, and dip of the Local and Narrow source zones.

Hardy (Reference 2.5.2-203) and Chapman et al. (Reference 2.5.2-204) describe data collected during the period 2011 to 2012 from an eight-station seismic network located in the Charleston epicentral area. The network located 134 earthquakes with duration magnitudes ranging from MD -1.8 to 2.6, and they observed that the earthquakes occurred in a tabular zone striking approximately N186°E and dipping approximately 43°W. From these data, Chapman et al. (Reference 2.5.2-204) interpret that the modern microseismicity is the lingering aftershock sequence of the 1886 earthquake, and that hypocentral locations and focal mechanisms suggest the 1886 earthquake was produced by reverse slip on a south-striking, moderately west-dipping fault plane. Chapman et al.’s (Reference 2.5.2-204) postulated source of the 1886 earthquake is in the Narrow seismic source, although they suggest the 1886 Charleston seismic source may be a west-dipping reverse fault, as opposed to vertical strike-slip as modeled in the CEUS-SSC.

Liberty (Reference 2.5.2-205) describes ongoing work to identify and characterize active faults in the 1886 epicentral area from seismic reflection data. He identifies evidence for a 4-km-wide zone of faulting and folding in the 1886 epicentral area, evidence for possible Quaternary slip on Talwani and Durá-Gómez’s (Reference 2.5.2-206) Woodstock fault, and faults consistent with a northeast-striking zone of deformation identified by Chapman and Beale (Reference 2.5.2-83). The (Reference 2.5.2-206) and Chapman and Beale (Reference 2.5.2-83) studies were available during development of, and were incorporated into, the CEUS-SSC.

In a presentation for the Charleston Area Earthquake Hazard Mapping Project (CAEHMP), Cramer (Reference 2.5.2-207) describes ongoing work to develop seismic and liquefaction hazard maps for the Charleston area based on three alternative sources for the 1886 earthquake. These alternative sources are Talwani and Dur-Gómez's (Reference 2.5.2-206) Woodstock fault, Chapman et al.'s (Reference 2.5.2-204) west-dipping fault, and a simple vertical strike-slip fault in the Summerville area. Scenario **M** 7.0 PGA hazard maps from these alternative sources were compared with building and chimney destruction locations documented by Bollinger (Reference 2.5.2-229). Cramer (Reference 2.5.2-207) concludes that, although none of the three alternative sources can be eliminated, the Woodstock fault and vertical strike-slip alternatives better explain the 1886 damage observations than Chapman et al.'s (Reference 2.5.2-204) west-dipping fault.

Marple and Hurd (Reference 2.5.2-230) evaluated lidar and aerial photo data in the Summerville area and identified several topographic and tonal lineaments that they suggest are related to active tectonic deformation. These features include a 40-km-long, east-northeast-trending lineament near Woodstock fault, a 17-km-long, east-northeast-trending lineament near Middleton Place, and several east-west-trending scarps.

Shah et al. (Reference 2.5.2-231) combine interpretation of a new aeromagnetic dataset from coastal and central South Carolina with observations from seismic reflection studies and other information to identify possible seismogenic structures in the Charleston seismic zone. They identify evidence for east-southeast-trending lineaments that they interpret as Mesozoic faults in the South Georgia Rift. They note that these features are favorably oriented to be reactivated in the current stress regime, but that they appear too short (<20 km) to produce **M**~7 earthquakes. They also identify northeast-trending lineaments interpreted as inherited Paleozoic structures and note that these features appear long enough to produce **M**~7 earthquakes, but that transpressive slip would be required.

The new information on fault geometry and style indicates that changes to the CEUS-SSC model are required to account for a broader range in fault orientations, dips, and mechanisms. Revisions to the Charleston RLME source for use in the SRS PSHA are described below in Subsection 2.5.2.2.5.4.2. These same revisions are adopted for use in the CRN Site PSHA.

Recurrence: The recurrence intervals of large-magnitude earthquakes in the Charleston RLME source are based on the 1886 earthquake and the ages of large prehistoric earthquakes inferred from liquefaction features preserved in the geologic record. In the CEUS-SSC, when "In cluster", these data indicate a weighted mean recurrence of about 580 years over the past approximately 2,000 years of record, and a weighted mean recurrence of about 1,730 years over the past approximately 5,500 years of record. Greater weight (0.8) is given to the recurrence derived from the shorter period than the longer period (0.2) because of concerns that the longer-period record is more likely to be incomplete (Figure 2.5.2-23R). No new paleoliquefaction age data are available that post-date the CEUS-SSC and that would change the interpreted number or timing of events in the Charleston RLME.

The CEUS-SSC includes two alternative models for large-earthquake recurrence of the Charleston RLME source: (1) the Poisson (time-independent) model; and (2) the Brownian Passage Time or "renewal" (time-dependent) model. The Poisson model assumes that the

average time between events is known, but the exact timing of events is independent of the time since the last and previous events. A renewal model for the occurrence probability of earthquakes is appropriate when the data suggest a more periodic recurrence behavior than is implied by the Poisson model. Renewal models typically are used where a fault is subjected to quasi steady-state loading, such as may occur in a plate boundary setting.

In the CEUS-SSC, much greater weight was given to the Poisson model alternative. For the Narrow source zone, the Poisson model is assigned a weight of 0.9 and the renewal model is assigned a weight of 0.1. For the Regional and Local source zones, full weight (1.0) is given to the Poisson model.

Faults in the Charleston RLME are poorly understood. The fault source of the 1886 earthquake has not been conclusively identified, and it remains unclear whether the same fault or a different fault or faults produced the paleoliquefaction features observed in coastal South Carolina. If one fault is responsible for all the RLME events, then greater weight on the renewal model may be warranted. However, if different faults are responsible for different RLME events, then each fault may have its own “clock”, and the composite of their rupture histories may be better approximated as a Poisson process.

For use in the SRS PSHA, increased weight was assigned to the renewal model of earthquake recurrence, as described below in Subsection 2.5.2.2.5.4.2. These same revisions are adopted for the CRN Site PSHA (Reference 2.5.2-265).

2.5.2.2.5.4.1 New Information that Post-Dates the SRS Augmented SSHAC Level 2 Study

Several new studies on the Charleston seismic zone post-date the SRS augmented SSHAC Level 2 study (Reference 2.5.2-198), but none requires changes to the characterization of the Charleston RLME source for use in the CRN Site PSHA (Reference 2.5.2-265). Below we briefly summarize selected post-SRS Charleston studies.

Pratt et al. (Reference 2.5.2-232) presents reinterpreted seismic reflection profiles, reprocessed older aeromagnetic data, and new ground-penetrating radar data to identify faults that deform Cretaceous and younger Coastal Plain strata in the epicentral area of the 1886 earthquake. Lead author Thomas Pratt shared much of this information with the SRS SSC TI Team by providing a preprint of this manuscript and by participating in the augmented SSHAC Level 2 Workshop for the SRS project. As a result, the key information in the Pratt et al. (Reference 2.5.2-232) paper was evaluated and integrated into the SRS SSC.

Shah et al. (Reference 2.5.2-232) used new, high-resolution aeromagnetic data, previous borehole logs, and reprocessed seismic reflection profiles to image subsurface structures and evaluate recent fault activity in the Charleston seismic zone. Lead author Anji Shah also participated in the augmented SSHAC Level 2 Workshop for the SRS project and shared much of this information with the SRS SSC TI Team.

Bilham and Hough (Reference 2.5.2-234) developed an elastic deformation model for the 1886 earthquake based on the sparse available constraints on coseismic deformation. Their preferred model yields an estimated **M** 7.3 for the 1886 earthquake, which is consistent with, but on the

high side of, the CEUS-SSC range of **M** 6.7–7.5 for the Charleston RLME. Their preferred model includes rupture on a west-dipping reverse fault with 6.5 ± 0.3 m of dextral slip and 2 ± 0.5 m of reverse slip below 450-m depth.

In a follow-on paper, Bilham and Hough (Reference 2.5.2-234) used documented disturbances of three railroad lines in the Charleston area to investigate the 1886 earthquake and to question the commonly held belief that William Wilson's famous photograph of deformed railroad tracks near Rantowles records surface deformation above subsurface dextral slip or as the result of thrust faulting (e.g., (Reference 2.5.2-232). This is consistent with the CEUS-SSC characterization of the Charleston RLME, which allows for considerable uncertainty in the source of the 1886 and prehistoric earthquakes in the area.

Hough and Bilham (Reference 2.5.2-236) investigated archival sources of information on the 1886 earthquake, especially Earle Sloan's detailed geological field observations. Hough and Bilham ((Reference 2.5.2-236) point out that Sloan's observations are an invaluable source of information on the earthquake, but that his reports include some data entry, transcription, and other errors. Hough and Bilham ((Reference 2.5.2-236) also speculate that Sloan may not have been in the Charleston area during the mainshock and the following week, and that he may have relied on some secondhand information in compiling his reports.

2.5.2.2.5.4.2 Summary of Changes to CEUS-SSC

Based on review of data that post-date the LCI (Reference 2.5.2-198) characterization of the Charleston RLME for the SRS, it is concluded that the SRS characterization of the Charleston RLME is appropriate for use at the CRN Site and that no additional changes are needed (Reference 2.5.2-265). Revisions to the CEUS-SSC's Charleston RLME source for use in the CRN PSHA are summarized in Table 2.5.2-16R and Figure 2.5.2-29R and include the following:

- Replace the "Narrow" source geometry with a new "1886 Rupture" source geometry that represents Chapman et al.'s (Reference 2.5.2-204) hypothesized source of the 1886 earthquake. The Poisson and renewal recurrence models are equally weighted in this "fault-like" source.
- Increase weight on the renewal recurrence model from 0.0 to 0.1 for the Local zone. A weight of 1.0 on the Poisson model is retained in the larger Regional zone.
- For all three Charleston source zones, increase the range of permissible rupture strikes, dips, and fault styles. The overall northeast-oriented elongation of the Local and Regional zones is retained, reflecting the northeast-oriented elongation of the isoseismals from the 1886 earthquake and the regional structural grain.
- Zone boundaries for the 1886 Rupture and Local zones are leaky, such that ruptures that initiate within the zone are allowed to extend beyond the zone boundaries.

2.5.2.2.6.1.2 Impact of Updated Seismicity Catalog

Replace the first sentence of the third paragraph of ESPA SSAR Section 2.5.2.2.6.1.2 with the following:

While the updated earthquake catalog found no new earthquakes (post-2008) greater than E[**M**] 5.0 in the CRN Site region, there were a total of four events (including the 2011 Mineral earthquake) exceeding E[**M**] 5.0 east of longitude -105 degrees and within the CEUS SSC Study Region.

2.5.2.2.6.1.3 ETSZ Mmax Sensitivity Studies

Insert the following after the last paragraph of ESPA SSAR Section 2.5.2.2.6.1.3:

An additional sensitivity study was performed as part of the SSHAC Level 2 evaluation for the Y-12 site (Reference 2.5.2-150). This hazard sensitivity analysis was performed based on the postulated Dandridge-Vonore fault zone (DVFZ) within the ETSZ (Subsection 2.5.2.2.5.1). A 110-km-long “Quaternary fault corridor,” the DVFZ was proposed by Reference 2.5.2-216 and Reference 2.5.2-217 based on limited geologic field observations and an apparent seismicity alignment. The input parameters for the sensitivity analysis of DVFZ were taken from Reference 2.5.2-218. The Mmax values and corresponding weights for the DVFZ were 6.0 (0.2), 6.5 (0.5), and 7.0 (0.2). The slip rates and associated weights were 0.001 mm/yr (0.45), 0.01 mm/yr (0.45), and 0.1 mm/yr (0.1). In addition, a probability of activity of 0.1 was assigned to the fault.

The results of the sensitivity analysis indicated that addition of the postulated DVFZ with the range of the input described above, did not significantly increase the hazard at the Y-12 site.

2.5.2.2.6.2 Earthquake Recurrence Rates

Insert the following after the last paragraph of ESPA SSAR Section 2.5.2.2.6.2:

As part of the Y-12 PSHA, a statistical test was performed to determine whether the new data in the Y-12 updated catalog (i.e., main events that occurred between 1/1/2009 and 12/31/2018) require recalculation of recurrence parameters for the Mmax and Seismotectonic distributed seismicity zones. The test was performed for those distributed seismicity zones that affect seismic hazard at the Y-12 site (Reference 2.5.2-150). The statistical analysis was performed to determine whether the rates of **M** ≥ 2.9 and **M** ≥ 3.5 earthquakes in the time period between 2009 and 2018 were consistent with predictions from the CEUS SSC recurrence parameters. The test was performed for a circular area within 320 km of the Y-12 site, which is similar to the CRN Site region. Details of the statistical test were given in Reference 2.5.2-150. The results of the test did not conclusively require recalculation of the recurrence parameters, in part because the **M** 2.9-3.6 range has a low weight in the CEUS SSC recurrence methodology. The recurrence parameters, however, were recalculated using the Y-12 updated catalog for the distributed source zones that make significant contributions to seismic hazard at the Y-12 site (namely, PEZ-N, NMESE-N, STUDY-R, PEZ-W, MESEW, MIDC-A, IBEB, RR, ECC-AM, and MESE-N).

Recalculation of the recurrence parameters for the Y-12 site followed the CEUS SSC methodology.

2.5.2.3 Correlation of Seismicity with Seismic Sources

Renumber Figure 2.5.2-18 to Figure 2.5.2-18R in EPSA SSAR Section 2.5.2.3.

Renumber Figure 2.5.2-19 to Figure 2.5.2-19R in EPSA SSAR Section 2.5.2.3.

Renumber Figure 2.5.2-22 to Figure 2.5.2-22R in EPSA SSAR Section 2.5.2.3.

Renumber Figure 2.5.2-24 to Figure 2.5.2-24R in EPSA SSAR Section 2.5.2.3.

Replace the second paragraph of ESPA SSAR Section 2.5.2.3 with the following:

As described in Subsection 2.5.2.1, the catalog has been updated for the CRN Site to include events through mid-September 2013. Within the CRN Site region (200-mi radius), 185 earthquakes (including dependent and small events down to magnitude E[**M**] 2.2) were added to the catalog. Of these post-2008 earthquakes, only 9 events were E[**M**] 2.9 or larger. The updated seismicity does not directly highlight or define seismogenic structures within the CRN Site region, but generally follows regional patterns of seismicity, such as the ETSZ. The approximately 5-year period of updated seismicity does not provide sufficient events to define new structures or suggest changes to the geometry of background seismotectonic source zones within the CRN Site region. Beyond the CRN Site region, however, the **M** 5.8 2011 Mineral, VA earthquake has highlighted a southeast-dipping fault based on the moment tensor solution and the distribution of aftershock hypocenters (Reference 2.5.2-23).

Replace the third sentence of the fifth paragraph of ESPA SSAR Section 2.5.2.3 with the following:

The discussion of correlation of seismicity with seismic sources presented in this subsection is limited to seismotectonic sources within the CRN Site region and two RLMEs beyond the CRN Site region (NMFS and Charleston) (Subsections 2.5.2.2.3 and 2.5.2.2.4, respectively).

2.5.2.4 Probabilistic Seismic Hazard Analysis and Controlling Earthquakes

2.5.2.4.1 CEUS SSC Model Implementation

Delete the second, third, fourth, and fifth bullets in ESPA SSAR Section 2.5.2.4.1.

2.5.2.4.2 Ground Motion Models

Replace ESPA SSAR Section 2.5.2.4.2 with the following:

The ground motion models developed by the Next Generation Attenuation for Central and Eastern North America project (NGA-East) (Reference 2.5.2-221) were used in the hard rock PSHA for the CRN Site (Reference 2.5.2-265).

As part of the Y-12 PSHA, a SSHAC Level 2 process was used to compile and evaluate new information and data that were potentially relevant to ground motion characterization in the Y-12 site region, which is essentially equivalent to the CRN Site region. The study compiled ground motion data since 2011 for comparison to the NGA-East model, interviewed Resource and Proponent experts to learn of new data, models, and methods, reviewed new or updated GMMs, and compiled and reviewed other information potentially relevant to ground motion characterization in the CRN Site region. Based on their evaluations of the new information, the Y-12 Ground Motion Technical Integration Team concluded that the NGA-East GMM remains applicable for the calculation of basement rock ground motions without modifications or revision (Reference 2.5.2-150).

Crustal GMMs for tectonically active regions like the western U.S. are empirical in nature and derived from strong motion data from such areas as California, Taiwan, Japan, and Italy. In contrast, few usable strong motion records exist for earthquakes in the Central and Eastern North America (CENA). Thus, GMMs for the CENA have been developed, in large part, using seismological-based numerical models. During the past decade, ground motion models for the CENA have been derived using three different approaches: the stochastic method, the Green's function method, and the complex/empirical source method.

The Pacific Earthquake Engineering Research Center (PEER) developed a new ground-motion characterization (GMC) model applicable for use in Central and Eastern North America (CENA, east of approximately 105 degrees west) through the Next Generation Attenuation for Central and Eastern North America project (NGA-East) (Reference 2.5.2-221). The GMC model consisted of two parts for use in a PSHA:

- A set of new GMMs for the median ground motions and their associated weights in the associated logic-trees.
- A set of new models for the aleatory standard deviations of ground motion, and their associated logic trees.

The final results and models of NGA-East were developed under the framework of the Seismic Senior Hazard Analysis Committee (SSHAC) (Reference 2.5.2-42; Reference 2.5.2-44) Level 3.

The NGA-East GMC is to be used in conjunction with the CEUS SSC products (Reference 2.5.2-1). The CEUS-SSC Project was also conducted under the SSHAC Level 3 framework; thus, having the GMC component of the NGA-East under SSHAC Level 3 framework ensured that both projects are of the same high-quality level for the end-uses.

The NGA-East ground-motion database (Reference 2.5.2-181) included the two- and three-component ground-motion recordings from numerous selected events (**M** > 2.5, distances up to 1500 km) recorded in the CENA region since 1988, up to December 2011. The database contains over 29,000 records from 81 earthquake events and 1379 recording stations. The events included “mainshocks,” “aftershocks,” and tectonic events, as well as potentially induced events (PIEs). The NGA-East database constitutes the largest database of processed recorded ground motions in stable continental regions (SCRs).

The NGA-East Project provided the best estimate of the distribution (median and standard deviation) of RotD50 (Reference 2.5.2-237) horizontal ground motions for horizontal peak ground acceleration (PGA), peak ground velocity (PGV), and 5%-damped pseudo-acceleration response spectra (PSA) for oscillator periods ranging from 0.01 to 10 sec on reference “hard-rock” site condition. The reference site conditions were defined by the NGA-East Geotechnical Working Group as corresponding to a V_S of 3000 m/sec and a kappa (κ) of 0.006 sec (Reference 2.5.2-238; Reference 2.5.2-239). The significance of the reference rock definition was that it represents the site condition for which ground motions will be predicted using the GMMs, and it represents the site condition to which site amplification factors are referenced (i.e., site amplification is unity for reference rock).

Final models in NGA-East are applicable to **M** 4.0 to 8.2 range, and for distances up to 1500 km from earthquakes source, using rupture distance as a distance metric. The final median ground-motion logic tree consists of 17 branches, each corresponding to a GMM with a weight assigned.

The NGA-East Project evaluated regions of different geologic and tectonic settings in the CENA. The regions are the Central North America, the Appalachian Province, the Atlantic Coastal Plain, and the Gulf Coast/ Mississippi Embayment Region. Each region is described by a statistically representative crustal seismic velocity-depth structure and Q-factor model. The crustal structural models are for very hard rock conditions and do not include any sediments. The ground motion analyses for different regions demonstrates that there are two distinct attenuation groups:

- Mid-Continent Region: Central North America, Appalachians, and the Atlantic Coastal Plain.
- Gulf Coast Region: Gulf Coast including the Mississippi Embayment.

The Gulf Coast Region was found to clearly belong to a separate attenuation group. This result agrees with previous analyses that have found that the Gulf Coast Region has unique attenuation characteristics. The NGA-East base median GMMs are defined for the Mid-Continent Region. Adjustment models for the Gulf Coast Region are developed in the NGA-East Project. The site is situated within the Gulf Coast Region.

In addition to regionalization, the NGA-East Project developed adjustment models for source depth and hanging-wall effects. The source-depth adjustment accounts for depth dependence of the stress parameter. The hanging-wall adjustment accounts for the increase in ground motions at sites located above dipping earthquake ruptures compared to what would be expected at sites located at the same rupture distance but on the footwall side of the rupture.

In NGA-East GMC, standard deviation models are provided for site-specific analysis (single-station standard deviation) and for general PSHA applications (ergodic standard deviation). PSHA studies typically aim to evaluate the hazard at a single site. This requires characterizing the ground motion (median and variability) at the particular site due to the future occurrence of different earthquakes. Since repeated recordings at the site of interest are usually not available, the ergodic assumption is adopted, whereby the ground-motion variability over different sites and source regions is assumed applicable to an individual site. In such cases, the use of the ergodic assumption is valid as the PSHA is typically concerned with rare events with the likelihood of repeated occurrences being very low. More recently, the availability of repeated recordings at individual stations in ground-motion databases allowed for the estimation of the site-to-site variability and for removing it from the ground-motion aleatory variability. This led to a reduced aleatory variability as observed in previous studies (e.g., Reference 2.5.2-165; Reference 2.5.2-240). The application of the single-station approach, however, requires adjusting the median GMMs from representing average site conditions in the dataset used to develop the models to become site-specific. Site-specific adjustments of the median GMMs can be done using empirical ground-motion recordings when available at or near the site of interest or as part of the more commonly used analytical site response studies. The epistemic uncertainty resulting from the limited knowledge of the site-specific parameters and adjustment factors needs to be properly accounted for as part of the single-station approach.

In addition to the reduction in aleatory variability, the use of single-station sigma has multiple advantages. It allows for proper identification of the various components of ground-motion variability where some of the apparent randomness can be transformed to epistemic uncertainty. With the acquisition of additional data, this epistemic uncertainty can, in theory, be removed or decreased. Moreover, the use of single-station sigma has the advantage of avoiding double counting of the uncertainty in cases where the ergodic sigma is used and a site-specific response study performed. In such cases, the double-counting of the uncertainty is the result of the site-to-site variability being part of the total ergodic sigma, and the uncertainty in the site amplification factors are also captured as part of the site response study.

Nine single-station and ergodic sigma models were developed which represent the discrete approximations of the three continuous distributions of sigma models. To reduce the computational burden in hazard analyses resulting from the use of nine alternative sigma values, the three continuous distributions of the sigma models along with their weights are represented by a composite sigma model. The composite sigma model is then discretized into a three-point distribution. Three discrete points selected at the 5^{th}, 50^{th}, and 95^{th} percentiles. The distributions for sigma show only a small degree of skewness, and the nominal weights of 0.185, 0.63, and 0.185 on the 5^{th}, 50^{th}, and 95^{th} percentiles, respectively, produce values for the mean and standard deviation of σ that adequately represent those computed from the continuous distribution.

The NGA-East Project adopted two alternatives to represent the shape of ground-motion residuals: traditional lognormal distribution with a weight of 0.2 and a mixture of two normal distributions with weight of 0.8. The mixture model consists of an equally weighted mixture of two normal distributions, with one component having 0.8 times the normal sigma and the other component having 1.2 times the normal sigma.

The NGA-East team considered the CENA single-station sigma and total ergodic sigma models are applicable to both Mid-Continent Region and Gulf Coast Region. For this study, the full NGA-East median GMMs logic (17 models with associated weights) are used along with the logic tree given for single-station aleatory standard deviation.

For the Idaho National Laboratory (INL) SSHAC Level 3 project (Reference 2.5.2-260) GMC study and given the strong evidence of heavy tails and the improved performance of the mixture model in capturing the tails of the distribution, the mixture model was adopted with a weight of one. The normal distribution was not retained because it is not supported by statistical evidence. For this project, the mixture model with a weight of one was adopted and the normal distribution is excluded.

Moreover, the INL GMC TI team evaluated the spatial correlation of the ground-motion residuals observed and evaluated in other studies including Jayram and Baker (Reference 2.5.2-261 and Reference 2.5.2-262), Shahi et al. (Reference 2.5.2-263), and GeoPentech (Reference 2.5.2-264). They concluded that the ground-motion residuals are indeed spatially-correlated, and this correlation results in a small bias in the total single-station sigma. To account for the bias a larger weight was given to the high branch of the single-station sigma. INL (2022) study adopted weights of 0.15, 0.55, and 0.3 on low, central, and high branches of the NGA-East single-station sigma. For this project, the same modified weights are applied to the low, central, and high branches of the NGA-East single-station sigma.

Ramos-Sepúlveda *et al*. (Reference 2.5.2-241) investigated whether the NGA-East GMMs for the CEUS reference hard-rock condition of 3000 m/sec and the site amplification models developed by Stewart *et al*. (Reference 2.5.2-242) and Hashash *et al*. (Reference 2.5.2-243), when used in tandem, are compatible with an expanded CENA ground-motion dataset relative to that used in the NGA-East Project. The expanded dataset includes recordings in CENA from November 2011 (end date of NGA-East data curation) to April 2022. While expanded, the range of magnitudes in the database remains limited (**M** ~ 4.0 to 5.8), hence their focus was mainly to assess mean model misfit within the range of the data, rather than scaling relationships that are also important for hazard applications. They evaluated the need for model adjustments to fit the available data, while leaving source and path scaling as-is.

They computed residuals using the 17 NGA-East GMMs and computed mean misfits using three subsets of the CENA data that reflect within-CENA regional variations in ground-motion attributes:

1. Non-TOK / Non-CP: Events and sites within Texas-Oklahoma-Kansas (TOK) and the Coastal Plains (CP) are excluded.

2. Non-TOK: All non-TOK events are considered including CP. No TOK events are considered.

3. Partial TOK: All non-TOK events are considered. Within TOK, random events were selected so that the event density (number of events per area) is consistent with other parts of CENA.

Ramos-Sepúlveda *et al.* (Reference 2.5.2-241) considered the 17 alternative NGA-East GMMs and three data selection criteria to compute the misfit terms. Various weighting combinations

were considered for data selection, including equal weighting and weighting that gives preference to the non-TOK/non-CP data set. Ultimately, the recommended weights were 0.2 for the non-TOK set, 0.2 for the partial TOK set, and 0.6 for the non-TOK/non-CP set. All sites were considered in the development of the model adjustment factors presented in this section. They provide adjustment factors to the NGA-East GMMs and their epistemic uncertainties.

Ramos-Sepúlveda *et al*. (Reference 2.5.2-241) concluded that two factors potentially influence these misfits: (1) differences in the site amplification models used to adjust the data to the reference condition during NGA-East GMM development relative to CENA amplification models developed by Stewart *et al*. (Reference 2.5.2-242) and Hashash *et al*. (Reference 2.5.2-243), and (2) potential bias in simulation-based factors used to adjust ground motions from the hard-rock reference condition to a V_{S30} 760 m/sec condition. They provided adjustment factors and their epistemic uncertainties and discussed implications for applications. These adjustment factors were adopted in this study.

For the 2023 NSHM, the USGS included the Ramos-Sepúlveda *et al*. (Reference 2.5.2-241) adjustments with 0.5 weight (Reference 2.5.2-244). This decision was due to the recency of observations of misfits from CEUS GMMs and the USGS desire for greater understanding of the causes of this effect, as well as broader discussions regarding use of these adjustments. The USGS agreed that these data are predominantly from small magnitudes and that there may be effects that would differ for larger magnitudes, which was part of the motivation for the lower weight. For this project, it was decided to follow the same approach. Hence, for the hazard calculations, the following logic-tree weights were used:

- 0.5: NGA-East GMMs as published (no adjustment)
- 0.5: NGA-East GMMs with adjustment factors given in Ramos-Sepúlveda et al. (Reference 2.5.2-241).

2.5.2.4.3 Probabilistic Seismic Hazard Analysis and Deaggregation

Replace the second sentence of the first paragraph of ESPA SSAR Section 2.5.2.4.3 with the following:

The calculation of rock seismic hazard used the CEUS SSC seismic source characteristics with modifications described in Section 2.5.2.2.4.1, the NGA-East ground motion models (Reference 2.5.2-221) with adjustments (Reference 2.5.2-241) updated seismicity files that include the effect of recent seismicity on background sources and corrected and updated Mmax distributions for some of the background sources (Reference 2.5.2-265).

2.5.2.4.3.1 Seismic Sources

Replace the fourth sentence in the second paragraph of ESPA SSAR Section 2.5.2.4.3.1 with the following:

All background sources are represented with 24 sets of annual activity rates and *b*-value parameters for each source.

Replace the last paragraph of ESPA SSAR Section 2.5.2.4.3.1 with the following:

As described earlier, updates were made to the CEUS SSC Mmax and recurrence parameters for some of the source zones. The updates are as follows.

Updated a- and b- values based on the Y-12 updated earthquake catalog were used for the following sources:

1. Non-Mesozoic and younger extended prior-narrow
2. Mesozoic and younger extended prior-wide
3. Study Region
4. Paleozoic Extended Crust-narrow
5. Paleozoic Extended Crust-wide
6. Mesozoic and younger extended prior-narrow
7. Midcontinent-Craton alternative A
8. Extended Continental Crust-Atlantic Margin
9. Illinois Basin Extended Basement
10. Reelfoot Rift

Corrected Mmax distributions from EPRI were used for the following sources:

1. Paleozoic Extended Crust-narrow
2. Paleozoic Extended Crust-wide
3. Midcontinent-Craton alternative A
4. Midcontinent-Craton alternative B
5. Midcontinent-Craton alternative C
6. Midcontinent-Craton alternative D
7. Extended Continental Crust-Atlantic Margin
8. Illinois Basin Extended Basement

As discussed in Section 2.5.2.2.5.3.3, the New Madrid Fault System RLME was revised.

2.5.2.4.3.2 Seismic Hazard Results

Replace ESPA SSAR Section 2.5.2.4.3.2 with the following:

Seismic hazard results for hard rock for the CRN Site were computed for 23 frequencies and PGA (Reference 2.5.2-265). Mean and fractile rock hazard curves for seven representative frequencies (0.5, 1, 2.5, 5, 10, 25 Hz and PGA) are shown in Figure 2.5.2-30R through Figure 2.5.2-36R. Mean rock hazard curves for all 23 frequencies and PGA are provided in Table 2.5.2-17R. The contributions to rock hazard from the total of all background sources and each individual RLME source (Charleston, Commerce, ERM-N, ERM-S, Marianna, NMFS, and Wabash Valley) for 1 Hz and 10 Hz are plotted in Figure 2.5.2-37R and Figure 2.5.2-38R. The total mean hazard for 1.0 Hz is dominated by the NMFS and the background sources for AFEs of 10^{-3} and 10^{-4} (Figure 2.5.2-37R). The relative contribution to the total hazard from background sources increases with decreasing AFEs, such that at 10^{-5} AFE, the hazard is completely dominated by the background sources. The total mean hazard for 10 Hz is dominated by the background sources (Figure 2.5.2-38R).

Contributions to 1 Hz and 10 Hz hazard by individual background sources only are plotted in Figure 2.5.2-39R and Figure 2.5.2-40R. The dominant background source zones are PEZ-N, Study-R and NMESE-N, which are host zones for the CRN Site.

Sensitivity of the 10 Hz rock hazard to the 17 NGA-East median ground motion models (GMMs) unadjusted and adjusted for bias as described in Section 2.5.2.4.2 is shown on Figure 2.5.2-41R and Figure 2.5.2-42R. Similarly, Figure 2.5.2-43R and Figure 2.5.2-44R show the sensitivities of the 1 Hz rock hazard to the 17 NGA-East GMMs.

Figure 2.5.2-45R shows the mean total rock hazard curves for seven representative frequencies (PGA (200 Hz), 25, 10, 5, 2.5, 1 and 0.5 Hz). The hazard curves all extend to 10^{-7} AFE or lower. Mean total rock hazard curves for all 23 frequencies and PGA are provided in Table 2.5.2-17R.

2.5.2.4.3.3 Deaggregation and Uniform Hazard Response Spectra

Renumber Figure 2.5.2-35 to Figure 2.5.2-31R in EPSA SSAR Section 2.5.2.4.3.3.

Renumber Figure 2.5.2-36 to Figure 2.5.2-34R in EPSA SSAR Section 2.5.2.4.3.3.

Replace Table 2.5.2-17 with Table 2.5.2-18R in the first paragraph of EPSA SSAR Section 2.5.2.4.3.3.

Replace third and fourth paragraphs of ESPA SSAR Section 2.5.2.4.3.3 with the following:

Deaggregation of the hard rock hazard results was performed at several spectral periods (Reference 2.5.2-265). For the CRN Site PSHA, low-frequency (LF) and high-frequency (HF) deaggregation plots for each MAFE are shown in Figure 2.5.2-46R through Figure 2.5.2-51R. The LF plot represents the average deaggregation of the 1 and 2.5 Hz spectral acceleration

hazard; the HF plot represents the average deaggregation of the 5 and 10 Hz spectral acceleration hazard.

For both high and low frequencies, the seismic hazard at the CRN Site is noticeably dominated by the background seismic sources within 100 km or less (Figure 2.5.2-46R through Figure 2.5.2-51R). For the low frequencies, seismic sources located about 400 to 500 km from the site, likely the NMFS RLME, have a significant contribution, as suggested in Figure 2.5.2-37R. The NMFS RLME contributes between 45 and 50 percent at 10^{-4} AFE, between 20 and 45 percent at 10^{-5} AFE and approximately 20 percent at 10^{-6} AFE.

Replace the fourth sentence in the fifth paragraph of ESPA SSAR Section 2.5.2.4.3 with the following:

In addition, the LF hazard contains contributions from distant sources, dominated by the NMFS RLME.

Replace the sixth, seventh and eight paragraphs of ESPA SSAR Section 2.5.2.4.3.3 with the following:

Based on the LF and HF deaggregation shown in Figure 2.5.2-46R through Figure 2.5.2-51R, the mean deaggregated magnitude and distance are calculated. Table 2.5.2-18R provides the controlling earthquake magnitudes and distances for a suite of eleven mean AFEs ranging from 10^{-2} to $2x10^{-7}$. Durations for these 11 events are also provided in Table 2.5.2-18R.These magnitudes and distances are used to develop input control motions for site response analyses at the CRN Site.

The hazard rock hazard results were interpolated to calculate UHRS for a suite of AFE (Reference 2.5.2-265). UHRS for 10^{-4}, 10^{-5} and 10^{-6} are shown on Figure 2.5.2-52R. The UHRS at 10^{-3}, 10^{-4}, 10^{-5} and 10^{-6} AFE are provided in Table 2.5.2-19R.

2.5.2.5 Seismic Wave Transmission Characteristics of the Site

Replace the second sentence of the first paragraph of ESPA SSAR Section 2.5.2.5 with the following:

The site amplification factors are used in determination of the FIRS, PBIRS, and PBSRS for the CRN Site. The seismic wave transmission characteristics of the CRN Site, presented in the CRN Site ESPA (Reference 2.5.2-196), were computed by considering a plant parameter envelope (PPE) at Locations A and B (Figure 2.5.4-11). The computations, as presented in this subsection, have been revisited and updated to incorporate the BWRX-300 SMR.

Replace the second paragraph of ESPA SSAR Section 2.5.2.5 with the following:

In order to examine the potential impacts of non-vertically propagating waves due to the dipping stratigraphy beneath the CRN Site (about 33 degrees) on the site response analysis results, a series of sensitivity analyses using an expanded version of the computer code SASSI (System

for Analysis of Soil Structure Interaction) are developed and presented in the CRN Site ESPA (Reference 2.5.2-196). Performing site-specific sensitivity evaluations on the potential effects of non-vertically propagating seismic waves is also one of the requirements of Section 5.3.3 of NEDO-33914-A (Reference 2.5.2-134). The results of the 2D sensitivity analysis are presented Subsection 2.5.2.6.

Replace the third paragraph of ESPA SSAR Section 2.5.2.5 with the following:

A geologic cross-section at the site that illustrates the depth to Precambrian rock, drawn perpendicular to the strike direction, is shown on Figure 2.5.1-63. The final plant surface grade at the site has been set at Elevation 814.5 ft. The planned bottom of the foundation in the vertical shaft for Reactor Building (RB) is taken at Elevation 699.5 ft, which is at a higher elevation compared to the foundation level considered in the ESPA (i.e., 683 ft). The proposed layout of the BWRX-300 SMR at the CRN Site is located in the vicinity of Location A. In fact, the proposed location of the RB vertical shaft is surrounded by the same borings that were used to develop the subsurface seismic wave velocity profile for Location A in the ESPA. As described in Section 2.5.4.5.1, the base-case V_S profile in the ESPA was developed with data from boreholes that were within or just outside Location A (within 100 ft). Boreholes whose data were used in developing the V_S profile on Figure 2.5.4-14 fall within the footprint of the layout of the BWRX-300 SMR or in its close proximity and approximately envelope the RB. In fact, boring MP-101 borders the Service Building, MP-102 is adjacent to the RB and Turbine Building, MP-111 is located within the footprint of the Service Building. Borehole MP-120 is slightly to the East (considering the plant North) of the Radwaste Building. Moreover, the top of the competent rock at the BWRX-300 SMR location was considered similar to the top of unweathered rock at Location A. It is recognized that the top of competent rock varies across the CRN Site, as shown in Figure 2.5.4-2R. Since the depth of the vertical shaft of the RB is expected to be around 115 ft below the final plant grade elevation, the foundation elevation of the RB foundation (for FIRS) is noticeably below the top of unweathered rock. All elevations cited in this subsection are based on the North American Vertical Datum of 1988 (NAVD88).

Supplemental site-specific field exploration (References 2.5.2-227 and 2.5.2-228) and testing have been conducted to supplement the CPA for the selected GVH BWRX-300 SMR. The general guidance from Section 3.3.1 of NEDO-33914-A (Reference 2.5.2-134) were considered in developing this investigation plan for the GVH BWRX-300 SMR layout. These data were used to update the velocity profile at the CRN Site underlying the BWRX-300 SMR.

The proposed layout of the BWRX-300 SMR at the CRN Site includes surface-founded structures around the vertical shaft of the RB. At this stage of the project, it is considered that these buildings will be constructed on deep foundations (drilled shafts). Considering the existing elevation at the site will be raised to the final plant grade elevation (i.e., Elevation 814.5 ft) using engineered fill, the generalized soil and rock profile at Unit 1 location consists of three strata: soil (including engineered fill on top of the existing fill and residual soil), in-situ weathered rock, and in-situ unweathered rock. The soil and weathered rock layers were not incorporated in the development of the amplification factors in the ESPA for the CRN Site (Reference 2.5.2-196); however, in the present analyses, they are included in the soil/rock column to develop the amplification factors at the final plant grade elevation. It is assumed that the horizontal extent of

the engineered fill, to raise the existing surface elevation at the site to the final plant grade elevation, will be large enough to represent a reasonable 1D profile. Moreover, the potential impact of the deep foundations of the surface-founded structures on the dynamic properties of soil and weathered rock and subsequently amplification factors are examined, as presented in Subsection 2.5.2.5.9, by developing an equivalent uniform 1D soil and rock profile in a set of sensitivity analyses.

Delete the third sentence of the last paragraph of ESPA SSAR Section 2.5.2.5.

2.5.2.5.1 Base-case Site-Specific Geologic Profiles and Uncertainties

Replace ESPA SSAR Section 2.5.2.5.1 with the following:

A V_S base-case profile for the CRN Site was developed for use in the SRA (Reference 2.5.2-265). This profile was extended by attaching the deep profile down to the Precambrian basement. This section describes the methodology for developing the V_S profiles.

Shallow Profile:

Geotechnical and geophysical tests were performed at the CRN Site in both 2013 and 2023 to determine the site profile. The MP series borehole logging was conducted in 2013 (Reference 2.5.2-245) and the U1B series borehole logging in 2024 (Reference 2.5.2-228). The CRN Site ESPA (Reference 2.5.2-196) shows the summary tables for the 2013 data. In the 2023 study, new data are also added to update the summary tables. Table 2.5.2-20R shows the elevation and thickness of the shallow soil layers along with the depth to bedrock. The median of existing ground surface elevation is 799 ft (244 m). The existing fill and residuum layer is about 18 ft (5.5 m) thick with spatial variability. Likewise, the median bedrock depth is at 777 ft (237 m) elevation with a standard deviation of 0.2. The median thickness of gravelly soil layer between the residuum and hard rock is about 4 ft (1.21 m). This gravelly soil layer was referred to as partially weathered rock in previous reports (Reference 2.5.2-196; Reference 2.5.2-246). Table 2.5.2-21R shows the depth to the different layers of soil/rock considering the final ground elevation of 814.5 ft (248 m).

To develop the V_S profiles, Geovision Geophysical Services performed P-S logging (suspension) twice at the CRN Site. Geovision (Reference 2.5.2-247) reported suspension measurements of P-wave (V_P) and V_S at approximately 1.64 ft (0.5 m) vertical intervals to the depth of penetration. At Location A, suspension logging was performed in seven borings: MP-101, MP-102, MP-111, MP-111PS, MP-120, MP-122PSA, and MP-122PSB. Two directional borings (PS-112 and PS-113) that were drilled at a nominal inclination of 28 degrees are excluded from this list. Geovision (Reference 2.5.2-248) report (Appendix IV of Reference 2.5.2-228) shows eight suspension logging test results namely U1B01, U1B02, U1B03, U1B04, U1B06, U1B07, U1B10, and U1B15. U1B15 which is outside of the footprint of the CRN Unit-1 building is not considered in this report. The suspension logging data are smoothed every three data points which is 4.92 ft (1.5 m) intervals of depth and plotted in Figure 2.5.2-53R. The lognormal mean of the suspension logging data for the rock profile was calculated and plotted in the same figure. MP-111PS,

MP-122PSA, and MP-122PSB that show the shallow profile (containing soil) were excluded while calculating the lognormal mean of the rock profile.

The University of Texas at Austin (UT Austin) also recently performed downhole surveys and SASW at the site (Reference 2.5.2-249). Eight downhole surveys were conducted at the same location as the Geovision (Reference 2.5.2-248) suspension logging. SASW was conducted across three different linear surface arrays. Moreover, free-free resonant column testing was conducted to measure the V_S of core samples extracted from the site. All these results are summarized in Stokoe *et al*. (Reference 2.5.2-249) as shown in Figure 2.5.2-54R. Compared to the Geovision data, these data showed higher variability in the top 138 ft (42 m). Comparisons between V_S profiles from the SASW method, downhole seismic testing, and laboratory tests were made. The statistical analyses showed higher variability in the upper layers (COV as high as 0.76) and more uniform properties at depths greater than 138 ft (42 m) (COV less than 0.12). The results indicate a large difference between SASW and downhole velocity measurements (as much as 6,000 ft/sec (1829 m/sec)), where SASW is showing lower values in a range of about 40 to 140 ft depth (12.2 to 42.7 m). The boring logs found limestone in this depth range. Therefore, the low V_S (about 2,000 ft/sec (610 m/sec)) shown in SASW in this depth range is not realistic. Likewise, if the large impedance contrast shown in downhole test results is considered as the transition from soil to hard rock, the depth range for this transition is found to be from about 38 to 105 ft (11.6 to 32 m). However, as shown in Table 2.4.2-20, the boring logs show the depth of transition to hard rock is shallower (22 +/- 8 ft) considering the existing ground surface elevation is 799 ft (244 m). Hence, after an evaluation of all the available data, the Geovision suspension logging data were used to develop the rock V_S profile. The soil V_S profile was adopted from the boring log data shown in Table 2.5.2-20R and Table 2.5.2-21R following the V_S recommendations in the CRN Site ESPA (Reference 2.5.2-196) Table 2.5.4-11 (600 ft/sec (183 m/sec) for existing fill and residuum and 1870 ft/sec (570 m/sec) for gravelly soil). The V_S of new fill is assumed to be 1000 ft/sec (305 m/sec) (Reference 2.5.2-250) considering a compacted granular fill.

Deep Profile:

Input ground motions used in the SRA were calculated for a hard rock site conditions of V_S 3,000 m/sec using the NGA-East GMMs. These models require a uniform hard rock site condition with V_S increasing with depth. Also, the effects of the alternating V_S layers beneath CRN Site needed to be addressed in the SRA. Thus, the shallow V_S profile needed to be extended down to uniform hard rock. Beneath the CRN Site, this coincides with the Precambrian basement at a depth of approximately 11,000 ft (3,352 m).

At depths below the shallow lognormal V_S profile where there are no measured V_S data, the geologic profile shown in Figure 2.5.4-16 was used to extend the profile. An average velocity was assigned to each geological unit down to and including the uppermost unit of the Knox Formation (Newala unit) based on the mean V_S shown in Table 2.5.4-16. Examination of the V_S profile for each unit showed generally constant V_S with depth, thereby a single mean V_S value for each unit was utilized. The lower and upper range base-case profiles will account for the epistemic uncertainty in the mean base-case profile. The mean V_S adopted for the Newala unit was used

for the rest of the Knox unit (rounded up to 11,000 ft/s) due to their similar lithologies. Figure 2.5.4-16 shows the V_S profile down to the Newala unit.

Beneath the Knox unit, there are no data for the deeper units at the CRN Site, which include the Conasauga shale, Rome Formation sandstone, and Pumpkin Valley shale. These units repeat themselves with depth because of the overthrusting (Figure 2.5.4-13). V_S data were available for all three formations from the deep SASW surveys that were performed at WBN2 (Reference 2.5.2-172). At a depth of 500 ft, the Conasauga shale and Pumpkin Valley shale have measured V_S values of 6000 ft/s and 6000 to 7000 ft/s, respectively at WBN2. The Rome sandstone had measured values of 9000 to 10,000 ft/s at a depth of 1500 ft (Reference 2.5.2-172). An unconfined free-free resonant column (URC) test also measured a V_S of almost 9000 ft/s for a sample of Rome Formation (Reference 2.5.2-172, Appendix B). Although these V_S values are not site-specific, they likely are representative of the V_S for these units beneath the CRN Site at depths below the OYO surveys based on regional stratigraphic relationships.

To estimate the V_S of the units below depths of 500 to 1500 ft, a review was performed of the deep V_S profiles available in the literature that went down several kilometers. Based on the review, the generic V_S profile for CEUS hard rock developed by Boore and Joyner (Reference 2.5.2-266) was selected. No region-specific V_S profile information was available. Examination of the central and eastern U.S. V_S database compiled by PE&A support the use of the value of 6000 ft/s at a depth of 500 ft for the Conasauga shale and 9000 ft/s at a depth of 1500 ft for the Rome Formation and the Pumpkin Valley shale (Figure 2.5.4-13). The adjusted Boore and Joyner (Reference 2.5.2-266) generic profiles were then used to estimate the V_S for the deeper units (Figures 2.5.4-18 and 2.5.4-19). Once Precambrian rock was reached, the V_S profiles were truncated at a V_S of 11,000 ft/s.

2.5.2.5.1.1 Epistemic Uncertainties in V_S

Replace ESPA SSAR Section 2.5.2.5.1.1 with the following:

For each best-estimate (BE) base-case profile, lower- and upper-range profiles (LR and UR, respectively) were calculated applying a factor of 1.28 to the best-estimate base-case V_S profiles (11.28 and 11.28, respectively) corresponding to an epistemic σln of 0.2, following the SPID (Reference 2.5.2-209). Below 541 ft (165 m) depth, where measured V_S data are not available, a factor of 1.57 was used corresponding to a σ_{ln} of 0.35 SPID (Reference 2.5.2-209). This calculation of lower- and upper-range profiles is consistent with the previous PSHA done for this site as described in AECOM (Reference 2.5.2-251). The best-estimate, lower-range, and upper-range profiles are weighted 0.4, 0.3, and 0.3, respectively, following the SPID (Reference 2.5.2-209). The upper-range profiles were truncated at a maximum V_S of 11,000 ft/sec (3,352 m/sec), the V_S of the Precambrian basement. The site profile is shown in Figure 2.5.2-55R. The details of the discretized site profiles are provided in Table 2.5.2-22R and Table 2.5.2-23R.

For randomization of V_S, aleatory variability is assigned to each layer in a depth-dependent manner per the SPID (Reference 2.5.2-209). Values range from σ_{ln} of 0.25 at the existing ground surface to 0.15 at and below a depth of 50 feet (15.2 m) and are interpolated above this depth. For the new fill layer, σ_{ln} of 0.15 was used. Realizations of V_S in each layer are restricted to $\pm 2\sigma_{ln}$ and are correlated following recommendations of the SPID (Reference 2.5.2-209).

2.5.2.5.1.2 Unit Weights

Replace ESPA SSAR Section 2.5.2.5.1.2 with the following:

The unit weights were collected from Table 2.5.4-21. Existing fill/residual soil which are mainly clayey/silty soil of usually high plasticity (CH, MH, ML) has a unit weight of 120 pcf. The gravelly soil layer which are mainly clayey/silty gravel (GC, GM) and in some cases poorly graded gravel with sand (GP) has a unit weight of 140 pcf. The Benbolt Formation beneath the site, mostly consisted of limestone and siltstone, was found to have a unit weight of 168 pcf. Finally, the new granular backfill has a unit weight of 135 pcf, which will consist of processed graded aggregate meeting the requirements of Type A aggregate (Reference 2.5.2-252) as mentioned in Section 2.5.4.5.3.

2.5.2.5.2 Estimation of Kappa

Replace ESPA SSAR Section 2.5.2.5.2 with the following:

WSP (Reference 2.5.2-267) developed an empirical equation to estimate kappa using kappa values estimated from earthquake recordings at five dams in the TVA region (not including the CRN Site). The equation incorporates the time-averaged shear wave velocity in the top 100 m (V_{S100}) and the total thickness of shale units within the top 2 km of the crust (T_{Shale}). The model for estimating κ is as follows:

$$\ln(\kappa) = -2.958 - 0.2484 \times \ln(V_{S100}) + 0.1085 \times \ln(\max(T_{Shale}, 0.21)) \qquad \text{Equation 2.5.2-8}$$

where the units of:

V_{S100} is m/s

T_{Shale} is m

κ is seconds

The standard deviation of ln(κ) is 0.36.

For the CRN Site, 460 m of Conasauga shale is present in the top 2 km of the crust as shown in Figure 4-10 of AECOM (Reference 2.5.2-251) report. Therefore, a T_{Shale} value of 460 m is used such that Equation 2.5.2-8 produces a κ value of 0.0173 sec (where V_{S100} = 1213.2 m/s) (BE profile (Table 2.5.2-24R).

Following Appendix B of Reference 2.5.2-209, epistemic uncertainty in site kappa is represented by the 10th and 90th percentiles. Using sigma ln(κ) of 0.36, this results in multiplicative factors of 0.699 and 1.434 for the 10th and 90th percentiles, respectively. Assuming negative correlation between V_S uncertainty and kappa uncertainty, the 10th percentile κ is assigned to the 90th percentile V_S (UR profile) and the 90th percentile κ is assigned to the 10th percentile V_S (LR profile). Table 2.5.2-24R lists the resulting kappa values.

2.5.2.5.3 Dynamic Material Nonlinearity

Replace the last four paragraphs of ESPA SSAR Section 2.5.2.5.3:

The soil column for the site response analysis at the CRN Site consists of three general strata: soil (engineered fill on top of the existing fill and residual soil), in-situ weathered rock, and in-situ unweathered rock (Section 2.5.1.2). In this subsection the selection of modulus reduction and damping ratio curves that represent the nonlinearity of dynamic properties for these strata are discussed.

Resonant column and torsional shear (RCTS) tests were performed for the CRN Site specimens in 2013 to determine appropriate dynamic material properties, shear modulus reduction and damping ratio curves (MRD curves) for CRN Site soil (Reference 2.5.2-253). Two soil specimens were tested (MP111UD and MP122UDA) at two different confining pressure levels. In 2023, RCTS tests were conducted again on soil and rock specimens from CRN Site by Stokoe *et al*. (Reference 2.5.2-254; Reference 2.5.2-255) from UT Austin. Three soil specimens (specimens 1, 2 and 3) were collected from different depths below ground surface and tested at three different confining pressure levels. All the old and new soil specimens show a similar trend of the MRD curves except for specimen 2 which was collected from a shallow depth of 6.5 ft (2 m). The results from this specimen are not considered, as it may have been more disturbed because it was extracted from a shallow depth. The MRD curves are plotted in Figure 2.5.2-56R. The generic EPRI (Reference 2.5.2-136) clay curves are also shown in this Figure 2.5.2-54R for plasticity index (PI) =30 and PI=40. A lognormal mean is calculated which is referred as Clinch River soil. As this curve ends at low strain (0.025%), it is manually extended (shown with dashed black line) to higher strain levels based on the trend of the existing lognormal mean curve and the trend of the EPRI clay curves.

Resonant column (RC) tests were conducted (Reference 2.5.2-255) on ten rock core specimens collected from different depths at CRN Site. Although they were tested at different confining pressure, the curves related to only the highest confining pressure were considered in this report. Stokoe *et al*. (Reference 2.5.2-255) recommended using the results from the highest confining pressure because all of the specimens had micro-cracks, and the increasing confining pressure tended to close the microcracks. Also, excluding the shallowest and deepest specimens, all the other specimens showed similar trends in the MRD curves. These two specimens were excluded from this report because the shallowest specimen may have been more disturbed by sampling/coring and the deepest specimen may be too stiff compared to the RC test apparatus to provide a reliable measurement. The MRD curves from all other specimens are plotted in Figure 2.5.2-57R and a lognormal mean is also calculated (referred as Clinch River rock). Similar

to the Clinch River soil curve (Figure 2.5.2-56R), this curve is also manually extended (shown with dashed black line) to higher strain levels based on the trend of the existing curve and the trend of the EPRI (Reference 2.5.2-136) rock curve. The EPRI rock curve (501-1000 ft) is also shown in Figure 2.5.2-55R.

Three equally weighted models for G/Gmax and damping are used to represent epistemic uncertainty in these properties at the CRN Site (Table 2.5.2-22R and Table 2.5.2-23R) (Reference 2.5.2-265). In the shallow portion of the profiles (uppermost approximately 500 ft (152 m)), three material models (M1, M2, and M3) are used. M1 and M2 are appropriate generic models (Reference 2.5.2-209), while M3 uses the MRD curves from site specimens when available. If test results are not available for a layer (fill and gravelly soil), another generic curve is used for M3. For rock, the EPRI rock (501-1000 ft) modulus reduction and damping model was used for M1, while M2 is linear with damping equal to the low-strain damping of M1, while M3 again uses the curves from test results. Below about 500 ft (152 m), M1, M2, and M3 curves are linear with damping. The depth of 500 ft (152 m) is consistent with the SPID (Reference 2.5.2-209) and is not a strict cutoff. The material property change is made in the first layer that begins below that depth. Table 2.5.2-26R to Table 2.5.2-32R list the MRD curves used at various levels of shear strain (%). The damping ratio for all curves will be capped at 15% in the SRA.

For randomization of dynamic material properties, aleatory variability in G/Gmax and damping is assigned to each material per the SPID (Reference 2.5.2-209). The SPID recommends σ_{ln} of 0.15 for G/Gmax at a strain of 0.03%, and σ_{ln} of 0.30 for damping at a strain of 0.03%, with correlation of -0.5. Realizations of V_S in each layer are restricted to $\pm 2\sigma_{ln}$ (Reference 2.5.2-265).

2.5.2.5.3.1 Linear Analyses

Replace ESPA SSAR Section 2.5.2.5.3.1 Linear Analyses with the following:

Deleted

Replace Section 2.5.2.5.3 with the following:

2.5.2.5.3.2 Total Effective Kappa

The lower-range, best-estimate, and upper-range values for kappa at the CRN Site are estimated as 0.0248, 0.0173, and 0.0121 sec, respectively (Table 2.5.2-24R) (Reference 2.5.2-265). Following the recommendations in Keefer and Bodily (Reference 2.5.2-161) and SPID (Reference 2.5.2-209), the weights for the kappa branches are 0.3, 0.4, and 0.3 for the LR, BE, and UR estimates of kappa, respectively. It is noted that this weight scheme is different from the weights (0.2, 0.6, and 0.2) in the ESPA. To distribute damping throughout the V_S profiles BE, LR, and UR, the total effective kappa was used as a constraint for the sum of damping throughout the profiles. It is important to note the details of how the damping is distributed through the profiles, conditional on the total effective kappa, has a minor effect on the amplification (Reference 2.5.2-164). As such the damping (in terms of Q) may be made either constant with

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depth, increasing with depth due perhaps to increasing confining pressure (e.g. typical damping curves and Reference 2.5.2-167), or proportional to V_S. Lacking a physical basis for preference, proportional to V_S was used (References 2.5.2-168 and 2.5.2-155) as it accommodates confining pressure dependencies in a consistent manner.

Total effective kappa for each profile is determined by summing the contributions of the ground motion model (GMM) and each layer using assigned low-strain dynamic properties. The contribution to total effective kappa of the NGA-East GMM is 0.006 sec (Reference 2.5.2-221; Reference 2.5.2-239) Total effective kappa (in seconds) is therefore computed as

$$\kappa = 0.006 + \sum_{i=1}^{n} \frac{2H_i D_i}{V_{S,i}} \qquad \text{Equation 2.5.2-9}$$

where:

H is the thickness of layer

i and *D* are low-strain damping

In this manner, the contribution of the shallow portion of each profile to total effective kappa is fixed by each model. Damping of the linear material in the deep portion of each profile is then set such that Equation 2.5.2-9 sums to the target kappa value. This is repeated for each combination of profile (best-estimate, lower-range, and upper-range) and material model (M1, M2, and M3). Damping in the deep portion of each profile is on the order of 0.37% or less. Resulting damping values are provided in Table 2.5.2-25R.

2.5.2.5.4 Capturing Site-Specific Geologic Column Properties, Uncertainties, and Correlations

Replace ESPA SSAR Section 2.5.2.5.4 with the following:

Addressing the epistemic uncertainties in the V_S profiles, site kappa, and MRD curves were previously discussed in Section 2.5.2.5.1.1, 2.5.2.5.2.1, and 2.5.2.5.3, respectively.

To provide an objective means of smoothing resonances that tend to be over-predicted by the vertically propagating V_S model in a one-dimensional profile (Reference 2.5.2-136) and, to a far lesser extent, to account for the aleatory variability in dynamic material properties that is expected to occur across a site at the scale of a typical facility, variability in the V_S profiles and other dynamic properties is incorporated. For the CRN Site, random V_S profiles have been developed about the smoothed base-case, lower-range, and upper-range profiles (Reference 2.5.2-265).

A profile randomization scheme has been developed that varies layer velocity, depth to baserock, G/Gmax curves, and damping curves. The randomization of V_S is based on a correlation model

developed from the statistical analysis of about 500 measured V_S velocity profiles (Reference 2.5.2-136; Reference 2.5.2-142). Depth to baserock is varied by plus or minus 20 percent using a uniform distribution. The standard deviations used in the randomization of V_S and degradation properties are generally based on the SPID (Reference 2.5.2-209). These randomization parameters reflect a minimum variability while providing for a realistic variation across footprint dimensions (several hundred feet) resulting in an objective smoothing of narrow resonances while preserving broad resonances yet not developing deep low-velocity zones that introduce artificial damping through wave scattering (scattering kappa).

For the V_S profile and the PBSRS control point, the depth to the top of the gravel layer (at 33.5 ft (10.2 m) below the ground surface) and rock (at 37.5 ft (11.4 m) below the ground surface) were also randomized using a normal distribution. The standard deviation used for this randomization is provided in Section 2.5.2.5.1.1. It should be noted that the depths to the top of the gravel and rock were not randomized for the PBIRS control point.

A total of 60 randomized profiles are generated for each base-case, lower-range, and upper-range profiles. Figure 2.5.2-58R shows the randomized V_S profiles for the best-estimate profile for the PBSRS control point. As mentioned above, the depths to the gravel and rock layers were both randomized. Figure 2.5.2-59R shows the randomized V_S profiles for the best-estimate profile for the PBIRS and FIRS control points. For these cases, the depths to the gravel and rock layers were not randomized. Separate panels are shown for the top portion (left) and for the entire profile (right). Each panel shows the 60 randomized velocities (green), their geometric mean and ± two-sigma range (solid and dashed blue). In addition to the ± two-sigma range provided as input (dashed and dot-dashed blue). In addition to the ± two-sigma limits, the randomized velocities are not allowed to exceed 11,000 m/sec, which is the value assigned to Precambrian basement hard-rock at the base of each profile. A similar limitation is imposed at low velocities to maintain the geometric mean of the velocities close to the target value specified as input. Layer thicknesses are not randomized, as this will not have a significant effect on the mean and variability of amplification when velocities are also randomized.

2.5.2.5.5 Site Response Analysis

Renumber Equation 2.5.2-8 through 2.5.2-11 to 2.5.2-10 through 2.5.2-13 in ESPA SSAR Section 2.5.2.5.5.

2.5.2.5.5.2 RVT-Based Equivalent-Linear Site Response Approach

Replace the fourth sentence of the second paragraph of ESPA SSAR Section 2.5.2.5.5.2 with the following:

A stepwise analysis approach was formalized into a 1D, vertically propagating S-wave code called SHAKE (Reference 2.5.2-129).

Replace the third paragraph to the end of ESPA SSAR Section 2.5.2.5.5.2 with the following:

The computational scheme employed to compute the amplification factors in this study uses an alternative approach employing RVT (Reference 2.5.2-156). In this approach, as embodied in the computer program TRANSAMP, Version 1.2 (Reference 2.5.2-256), the peak values using random vibration theory (RVT), or extreme value statistics are computed.

Inputs to TRANSAMP are as follows:

- Location of input and output motions within the site profile.
- Input (control) motions characterized by earthquake power spectra. In the program, the object motion must be specified by the user as bedrock outcrop motion in the form of Fourier amplitude spectrum.
- A vertical profile consisting of uniform plane-layer layers with specified thickness, seismic velocity, and density as a function of depth.
- Nonlinear dynamic properties of the material at the site, consisting of low strain damping, strain-dependent shear modulus and damping curves for each layer.

Input (control) motions must be calculated for input into the site response analysis that are representative of the earthquake magnitude and distance dominating the hazard at the desired rate of exceedance. The basis for the control motions are the magnitude and distances specified by the hazard deaggregation.

The LF and HF deaggregated magnitudes and distances (Table 2.5.1-18R) are used to develop a representation of the ground motion on baserock that is consistent with the hard rock PSHA for the CRN Site (Reference 2.5.2-265). This representation consists of 22 response spectra and their associated durations, covering a broad range of hazard-significant amplitudes (associated with an AFE range from 10^{-2} to 10^{-7}) and spectral shapes. Eleven of these spectra correspond to HF associated with nearby earthquakes and 11 to LF associated with more distant large-magnitude earthquakes in the RLME zones.

The HF spectrum for a given AFE is constructed as follows:

1. The spectral ordinates at and above 2.5 Hz all correspond to the same AFE from the rock hazard analysis (i.e., they correspond to the HF portion of the UHRS for that AFE).

2. A scenario spectrum is obtained by using the weighted mean of the NGA-East GMM amplitudes associated with the mean magnitude and distance obtained in the high-frequency (5 and 10 Hz) deaggregation (Table 2.5.2-18R).

3. The spectrum from Step 2 is truncated to remove ground motions above 2.5 Hz, then scaled so that it matches the UHRS at 2.5 Hz.

4. The HF spectrum at this AFE consists of the scaled scenario spectrum from Step 3 below 2.5 Hz, and the UHRS from Step 1 at and above 2.5 Hz.

A similar approach is followed for the LF spectra, using the LF portion of the UHRS together with a spectral shape at high frequencies dictated by the GMM and the low-frequency magnitude and distance obtained in the deaggregation (Reference 2.5.2-265). Figure 2.5.2-60R shows all 22 control motions. The control motions are provided in Table 2.5.2-33.

2.5.2.5.5.3 Horizontal Amplification Factors

Replace ESPA SSAR Section 2.5.2.5.5.3 with the following:

Site response analyses (SRA) were performed to calculate the site amplification functions associated with the PBSRS, PBIRS, and FIRS control points (Reference 2.5.2-265). The PBSRS control point is at the ground surface at elevation of 814.5 ft (248 m). The PBIRS control point is at the top of the rock, 37.5 ft below the surface at elevation of 777 ft (235 m). The FIRS control point is 115 ft (35 m) below the surface at elevation of 699.5 ft (204 m).

Note that for the PBSRS control point at ground surface, site amplification calculations were performed consistent with a geologic outcrop analysis for a free surface condition according to ISG-017(Reference 2.5.2-257). For other PBIRS and FIRS control points that are below the ground surface, since the corresponding structures will be modeled and analyzed as embedded structures, the site amplification calculations were performed consistent with a full column outcrop analysis according to ISG-017. Therefore, for control points other than ground surface, the derived PBIRS and FIRS will be a full-column outcrop ground motion which includes the full effect of the soil above the control points. The full-column outcrop includes the effects of down-going waves from soils above the elevation of the outcrop (ISG-017).

Site amplification functions (SAFs) were calculated for each combination of input control motion, randomized profile, and each of the three material models discussed in Section 2.5.2.5.3 for each of the base-case, lower-range, and upper-range profiles and PBSRS, PBIRS, and FIRS control points. Figure 2.5.2-61R through Figure 2.5.2-69R show the amplified response spectra and amplification factors for the 60 randomized profiles associated with base-case, lower-range, and upper-range profiles and for the input control motions associated with AFEs of 10^{-3}, 10^{-4}, 10^{-5}, and 10^{-6}. These figures show the results for material model 1.

SAFs from the three material models were combined with equal weights to develop the compound SAFs for each of the base-case, lower-range, and upper-range profiles.

Figure 2.5.2-70R through Figure 2.5.2-78R show the SAFs for the 11 input control motions inputs and for each of the base-case, lower-range, and upper-range profiles and PBSRS, PBIRS, and FIRS control points. In these figures, the median SAFs are shown on left and the standard deviation of SAFs (in natural log units) are shown in right. These figures illustrate the effect of non-linearity in the response, particularly for the soil sites. The increased damping reduces the SAF at high frequencies. The decreased stiffness reduces the fundamental period and amplitude of the resonant peak.

Figure 2.5.2-79R compares the SAFs for the PBSRS, PBIRS, and FIRS control points for the input control motions associated with AFEs of 10^{-3}, 10^{-4}, and 10^{-5}.

Results of the SRA are post-processed to obtain amplitude-dependent SAFs for use in subsequent calculations (Reference 2.5.2-265). Three sets of SAFs are defined – one each for the base-case, lower-range, and upper-range profiles that characterize epistemic uncertainty in the SRA weighted 0.4, 0.3, and 0.3, respectively. The SAFs were developed separately for PBSRS, PBIRS, and FIRS control points. The SAF and its logarithmic standard deviation depend on both input rock amplitude and spectral frequency. Following the SPID (Reference 2.5.2-209), median amplification factors are not allowed to be lower than 0.5.

Figure 2.5.2-80R shows mean effective strain among all randomized profiles versus depth resulting from HF and LF input control motions associated with AFEs of 10^{-3}, 10^{-4}, and 10^{-5} considering the best-estimate velocity profile and material models 1 (left), 2 (center) and 3 (right) in the top 200 ft. Input control motion AFEs of 10^{-3}, 10^{-4}, and 10^{-5} were selected for these figures since they mainly are used to derive the PBSRS, PBIRS, and FIRS.

It is clear from Figure 2.5.2-80R that higher strains are concentrated in the softer material near the ground surface, so Figure 2.5.2-78R focuses on the top 200 ft (61 m) of the profile. It is generally accepted that equivalent-linear approaches to SRA are accurate for effective strains up to a range of 0.1% to 0.4% (e.g., Reference 2.5.2-258). For the selected input control motions, the strain levels are generally within the 0.1% to 0.4% range with larger values for AFE of 10^{-5} in a limited depth range centered around 30 ft.

2.5.2.5.6 Development of V/H Ratios

Replace ESPA SSAR Section 2.5.2.5.6 with the following:

Vertical design spectra were calculated following the procedure in AECOM (Reference 2.5.2-251), with the exception of using Pezeshk *et al*. (2022) V/H model (Reference 2.5.2-259) in place of NUREG/CR-6728 (Reference 2.5.2-132). This recent model for V/H response spectral ratios was based on a comprehensive regression analysis of the NGA-East (Reference 2.5.2-181) database. The database included recordings from four earthquakes with magnitude greater than **M** 5.5. They are the 1988 **M** 5.9 Saguenay earthquake with 14 recordings; the 1985 **M** 6.8 Nahanni earthquake with three recordings; the 2011 **M** 5.7 Mineral, Virginia, earthquake with 134 recordings and the 2011 **M** 5.7 Sparks earthquake with 201 recordings (Reference 2.5.2-259). The magnitudes used for the V/H ratio range from **M** 5.9 to 6.0 at close distances of 8 to 15 km, associated with the deaggregated magnitude and distances for PGA at AFEs of 1E-4 and 1E-5.

The methodology implemented to develop the vertical ground motions follows closely that used to develop fully probabilistic site-specific horizontal motions (Reference 2.5.2-265). For application to the development of site-specific vertical hazard, the same fully probabilistic approach was used with V/H ratios (median and uncertainty estimates) substituted for horizontal amplification factors. In this case, the distributions of V/H ratios were applied to the site-specific horizontal hazard curves. As with the development of the site-specific horizontal hazard, epistemic, uncertainty in median V/H ratios was incorporated in this case by including σ_{μ}, uncertainty about the median V/H ratio.

Since V/H ratios for site conditions similar to the conditions at CRN Site are not well constrained due to a lack of recordings at design magnitudes and distances, epistemic uncertainty (σ_μ) was accommodated in the Approach 3 analyses. Epistemic uncertainty in V/H ratios was assessed by examining the range in available models for V/H ratios appropriate for firm to hard rock site conditions (Reference 2.5.2-138). The models considered include Bozorgnia and Campbell (Reference 2.5.2-151), Bommer *et al*. (Reference 2.5.2-152), Edwards *et al*. (Reference 2.5.2-153), and Gülerce and Abrahamson (Reference 2.5.2-137), with the range in ratios in the models spanning a factor of 1.5 to 1.8 depending on frequency. The range was taken to reflect $2\sigma_\mu$ estimates with $\sigma_{\mu ln}$ taken as the average at 0.25 and used for epistemic uncertainty in the Pezeshk *et al*. (Reference 2.5.2-259) V/H ratios. In the Approach 3 analyses for the vertical component the source (magnitude), path (distance) and site (V_{S30}) dependent V/H ratios of Pezeshk *et al*. (Reference 2.5.2-259) along with $\sigma_{\mu ln}$ = 0.25 were implemented with the site-specific horizontal hazard curves at the AFEs necessary to compute design spectra.

Additional aleatory variability about the V/H ratio, results in increased aleatory variability for the vertical component compared to the horizontal component, hence some attention is required as the contribution of site-specific aleatory variability to the hazard has already been accommodated in developing the site-specific horizontal component hazard. The site component of aleatory variability was included in developing the hard rock hazard in the variability about the hard rock ground motion prediction models. Additionally, in the Approach 3 analyses to develop the site-specific horizontal hazard, the variability about the median amplification factors, treated as aleatory variability, was included as well. As a result, there is some double-counting of site aleatory variability, which is commonly acceptable practice. However, in the application of Approach 3 in developing the vertical hazard, care must be exercised in contributing additional unnecessary aleatory variability in the vertical component, with the implication that vertical motions reflect higher variability than horizontal motions.

The development of NGA-West2 vertical GMMs (Reference 2.5.2-154) using the same suite of recordings as in the development of the horizontal ground motion models permits a straightforward assessment of the relative degrees of aleatory variability between horizontal and vertical components for this dataset. This comparison indicates that estimates of aleatory variability in the vertical GMMs are comparable to the corresponding horizontal GMMs, suggesting that adequate aleatory variability for the vertical component has been accommodated in developing the horizontal hazard. As a result, only epistemic uncertainty in V/H ratios was incorporated in developing the vertical hazard (Reference 2.5.2-265).

2.5.2.5.7 Site-Specific Horizontal and Vertical UHRS

Replace ESPA SSAR Section 2.5.2.5.7 with the following:

Control point hazard is calculated using as Approach 3 per the SPID (Reference 2.5.2-209). That procedure computes a site-specific soil hazard curve for the spectral acceleration at a selected spectral frequency (or period) given the site-specific hazard curve for the hard rock spectral acceleration at the same oscillator period and site-specific estimates of soil/soft rock response (median and standard deviation of the site amplification functions). Approach 3 is implemented in

a sequential manner, in which the hard rock hazard curve (typically only mean rock hazard) is convolved with the amplitude-dependent mean and standard deviation of site amplification.

In this project, Approach 3 is used to convolve the site amplification functions calculated for base-case, lower-range, and upper-range profiles with the rock hazard curves separately. The computed hazard curves for each profile are combined to calculate the weighted average hazard curves using the weight of 0.4 for base-case and 0.3 for lower-range, and upper-range profiles.

Mean hazard curves at the control points were calculated for the 23 spectral periods used to calculate the rock hazard (Reference 2.5.2-265). The computed hazard curves are given in tabular form in Table 2.5.2-34 through Table 2.5.2-36 for PBSRS, PBIRS, and FIRS control points (CPs). Mean hazard curves at 10 Hz, and 1.0 Hz are shown in Figure 2.5.2-81R through Figure 2.5.2-86R for PBSRS, PBIRS, and FIRS control points.

Mean UHRS for each control point are developed by interpolating the control point hazard curves at AFEs of 10^{-4}, 10^{-5}, and 10^{-6} (Reference 2.5.2-265). Rock and control point (CP) UHRS at AFEs of 10^{-4}, 10^{-5}, and 10^{-6} are shown in Figure 2.5.2-87R through Figure 2.5.2-89R for PBSRS, PBIRS, and FIRS control points and given in tabular form in Table 2.5.2-37.

Analogously, the horizontal site-specific hazard curves themselves were convolved with the suite of median V/H factors discussed in Subsection 2.5.2.5.6 to derive the corresponding vertical site-specific hazard curves (Reference 2.5.2-265). Mean horizontal and vertical UHS at AFEs of 10^{-4}, 10^{-5}, and 10^{-6} are shown in Figure 2.5.2-90R through Figure 2.5.2-92R for PBSRS, PBIRS, and FIRS control points. Vertical UHS at AFEs of 10^{-4}, 10^{-5}, and 10^{-6} are provided in tabular form in Table 2.5.2-38.

2.5.2.5.8 Site-Specific Ground Motion Response Spectra

Replace ESPA SSAR Section 2.5.2.5.8 with the following:

The five-percent damped horizontal site-specific FIRS, PBIRS, and PBSRS are obtained by scaling the site-specific mean UHRS by appropriate design factors (DF) as detailed in RG 1.208. This approach is consistent with the performance-based approach specified in ASCE/SEI 43-19.

According to RG 1.208, the horizontal site-specific, performance-based spectra are obtained by scaling the site-specific mean UHRS by a design factor (DF):

$$Response\ Spectra = UHRS \times DF \qquad \text{Equation 2.5.2-14}$$

where UHRS is the mean 10^{-4} UHRS by incorporating the amplification factors from the SRA and the design factor is computed as below:

$$DF = max\{1.0\ ,\ 0.6(A_R)^{0.8}\} \qquad \text{Equation 2.5.2-15}$$

where A_R is the ground motion slope ratio of spectral acceleration, frequency by frequency, from a seismic hazard curve corresponding to a ten-fold reduction in hazard exceedance frequency and is computed as below:

$$A_R = (mean 10^{-5} UHRS)/(mean 10^{-4} UHRS) \quad \text{Equation 2.5.2-16}$$

This assumes that the hazard curves are approximated by a power law equation (i.e., linear on a log-log plot) in the range of 10^{-4} to 10^{-5}. If A_R is greater than 4.2, then this assumption is not valid. In such a case, it is acceptable to use a value equal to 45 percent of the mean 10^{-5} UHRS.

FIRS, PBIRS, and PBSRS were calculated using the UHRS and Equations 2.5.2-14 to 2.5.2-16 given above. In Subsection 2.5.2.4.3.2, UHRS were calculated at the 23 spectral frequencies used in the PSHA (Reference 2.5.2-265). To capture the characteristics of site amplification at spectral frequencies other than the 23 at which the PSHA is performed, mean amplified spectral shapes are used to interpolate the UHRS to 158 spectral frequencies between 0.1 and 100 Hz. These shapes are available for each control point and profile. It is assumed that the mean amplified shape resulting from the 10^{-xx} input control motion is appropriate for interpolating the 10^{-xx} control point UHRS (where xx is -4 or -5). Furthermore, the shape resulting from the low-frequency (LF) 10^{-xx} input control motion is used to interpolate the UHRS at and below 2.5 Hz, and the shape resulting from the high-frequency (HF) input control motion is used to interpolate the UHRS above 2.5 Hz.

The computed horizontal FIRS, PBIRS, and PBSRS are shown in Figure 2.5.2-93R. The computed horizontal FIRS, PBIRS, and PBSRS are given in tabular form in Table 2.5.2-39.

Figure 2.5.2-94R illustrates the V/H ratios used in the calculations and they are tabulated in Table 2.5.2-40. Figure 2.5.2-95R shows the resulting vertical PBSRS, PBIRS, and FIRS along with the corresponding horizontal spectra (Table 2.5.2-41).

Add new Section 2.5.2.5.9:

2.5.2.5.9 Hazard-Consistent Strain-Compatible Properties

Hazard-consistent strain-compatible properties (HCSCPs) are developed for V_S, V_P, and damping for shear- and compression-waves (D_S and D_P) for hazard levels corresponding to the PBSRS and FIRS consistent with the methodology recommended by the SPID (Reference 2.5.2-209) and NEDO-33914-A (Reference 2.5.2-134, see associated Limitation and Condition (L&C) 8.4). Distributions of strain-compatible property values are obtained from the base-case, lower-range, and upper-range profile site response analyses performed in Section 5 in the same manner as site amplification functions (Reference 2.5.2-265). The distribution of each property, $f(SCP|S_a^R)$, is conditional on input rock spectral acceleration (S_a^R) and assumed to be lognormal. Consistent with the SPID (Reference 2.5.2-209), these distributions are developed for spectral frequencies of 1 Hz and PGA. HCSCPs are derived for each spectral frequency (1 Hz and PGA) and profile depth increment using

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$$HCSCP\left(S_a^S\right) = \iint f\left(SCP \middle| S_a^R\right) f\left(S_a^R \middle| S_a^S\right) d_{SCP} d_{S_a^R} \qquad \text{Equation 2.5.2-17}$$

where:

S_a^S is the hazard-consistent soil spectral acceleration for each spectral frequency (1 Hz and PGA) from the calculations above, and

$f\left(S_a^R \middle| S_a^S\right)$ is the distribution of rock spectral acceleration given soil spectral acceleration.

$f\left(S_a^R \middle| S_a^S\right)$ is the inverse conditional relationship of the site amplification functions described above and can be obtained from those results using probability theory.

The moments of the distribution for each HCSCP yield mean values and standard deviations. HCSCPs are assumed to be lognormally distributed, so medians and log standard deviations are obtained directly from the mean and standard deviation. The distributions obtained for 1 Hz and PGA are each calculated using the soil spectral amplitude corresponding to the appropriate FIRS ordinate. These two distributions are compared and combined according to the SPID (Reference 2.5.2-209) into a single distribution for each property and depth increment. Three soil profiles are derived from the resulting distribution: the best-estimate (BE) profile corresponding to the median value of each property at each depth increment; the lower-bound (LB) profile corresponding to -1 log standard deviation, and the upper-bound (UB) profile corresponding to +1 log standard deviation. The variation in shear modulus (G) represented by the LB and UB profiles is checked to ensure it covers at least BE/(1+COV) to BE*(1+COV) at each depth increment according to ISG-017. A minimum COV value of 0.5, corresponding to a "well-investigated" site per SRP 3.7.2, is used for this project. An upper V_S limit of 11,000 ft/s (3,352 m/sec) is also applied (this limit in some cases overrides the UB value of V_S implied by the minimum range of G just described).

Note that V_P and G are both calculated directly from V_S. Poisson's ratio is needed to obtain V_P and unit weight is needed to obtain G. Poisson's ratios of ratios of 0.48 for the saturated fill and soil, 0.4 for the gravel, and 0.32 for the weathered and unweathered rock were used (Reference 2.5.2-246). For this analysis it is assumed the water table elevation is at the ground surface. Unit weights are given in Section 2.5.2.5.1.2. Also note that V_P is calculated from low-strain values of V_S, while G is obtained from strain-compatible V_S. D_P was set equal to half of D_S for soil-structure interaction analysis.

A minimum value of 5,000 ft/sec (152 m/sec) is used for saturated V_P values calculated from low-strain V_S values.

A final constraint on each set of V_P values is that they do not result in a Poisson's ratio greater than 0.48 when calculated from strain-compatible V_S. This constraint is applied to avoid numerical issues common in SSI software. Some resulting V_P values may therefore fall below 5,000 ft/sec (152 m/sec).

BE, LB, and UB HCSCPs for V_S, V_P, D_S and D_P are shown versus depth in Figure 2.5.2-96R through Figure 2.5.2-99R for FIRS and PBSRS.

For SSI analysis, UB damping should be paired with LB velocities at each depth increment, and vice versa. As required by SPID (Reference 2.5.2-209), HCSCPs are derived using a probabilistic approach which maintain the hard rock annual probability of exceedances. The distribution of strain-compatible properties obtained during development of the SAFs are used to incorporate the variability in the site-specific dynamic material properties.

Add new Section 2.5.2.5.10:

2.5.2.5.10 FIRS After ISG-017 Check

The ISG-017 check was performed which ensures the adequacy of the FIRS as required by ISG-017 (Reference 2.5.2-257) before conducting the soil-structure interaction (SSI) analysis. Amongst the two available options that ISG-017 provides, option one was followed. This method ensures that the performance-based surface response spectra (PBSRS) are enveloped by the FIRS, which were propagated to the surface using the hazard-consistent strain-compatible properties (HCSCP) calculated in Subsection 2.5.2.5.9.

To check the adequacy of the FIRS, first the Fourier amplitude spectra (FAS) is derived from the response spectral values using inverse random vibration theory (RVT). Next, the FIRS-compatible FAS was propagated from the associated elevation toward the surface using the three deterministic hazard-consistent strain-compatible profiles (BE, LB, and UB) generated in Subsection 2.5.2.5.9. Thus, the corresponding surface response spectra (SRS) at the surface were generated. Then the envelope of the SRS from the BE, LB, and UB profiles is checked to see that it enveloped the corresponding PBSRS. This evaluation was performed for horizontal and vertical FIRS/PBSRS (Reference 2.5.2-265). The SRS for the BE, LB, and UB profiles along with their envelope and PBSRS are shown in Figure 2.5.2-100R and Figure 2.5.2-101R.

If the SRS envelope fell below the PBSRS in certain frequency ranges, the FIRS was scaled higher in those frequencies and repeated the convolution until the modified FIRS convolved up to the surface enveloped the PBSRS (Reference 2.5.2-265). The final scale factors for horizontal and vertical FIRS are shown in Figure 2.5.2-102R. Figure 2.5.2-103R and Figure 2.5.2-104R show PBSRS and the SRS for the BE, LB, and UB profiles along with their envelope using the modified FIRS. The updated horizontal and vertical FIRS after ISG-017 check are given in Table 2.5.2-42.

2.5.2.6 2D Sensitivity Analysis

Replace the first paragraph of ESPA SSAR Section 2.5.2.6 with the following:

In the development of the site amplification factors, a 1D equivalent-linear site response approach was used. Potential 2D effects, due to the same geologic unit and associated velocity occurring at different depths across the site as well as broad-band amplification (basin edge) effects resulting from dipping impedance contrasts, were initially addressed by evaluating the epistemic uncertainty in V_S beneath the CRN Site. In the CRN Site ESPA (Reference 2.5.2-196) a series of 2D sensitivity analyses were performed to evaluate the potential 2D effects on earthquake ground motions due to the dipping nature of the underlying stratigraphy beneath the CRN Site (approximately 33 degrees) (Figure 2.5.4-12). The 2D model was developed by considering both Locations A and B in the power block (cited in the ESPA). Since the new proposed layout of Unit 1 with the selected SMR technology at the CRN Site is located in the vicinity of Location A, it is considered that the results of the sensitivity analysis presented in the CRN Site ESPA is applicable to the new proposed layout. In summary, the results of the 2D sensitivity analysis show that the ratio of the 2D to 1D response spectra remains below 1 (Figure 2.5.2-129) for Location A, as specified in the CRN Site ESPA; therefore, the conservative bounding design spectra developed in Subsection 2.5.2.5 should be bounding any potential 2D effects due to the dipping subsurface stratigraphy at the CRN Site. The rest of this subsection describes the development of the 2D sensitivity analysis, and results as presented in the CRN Site ESPA.

Renumber the following in ESPA SSAR Section 2.5.2.6:

Figure 2.5.2-80 to Figure 2.5.2-105

Figure 2.5.2-81 to Figure 2.5.2-106

Figure 2.5.2-82 to Figure 2.5.2-107

Figure 2.5.2-83 to Figure 2.5.2-108

Figure 2.5.2-84 to Figure 2.5.2-109

Figure 2.5.2-85 to Figure 2.5.2-110

Figure 2.5.2-86 to Figure 2.5.2-111

Figure 2.5.2-87 to Figure 2.5.2-112

Figure 2.5.2-88 to Figure 2.5.2-113

Figure 2.5.2-89 to Figure 2.5.2-114

Figure 2.5.2-90 to Figure 2.5.2-115

Figure 2.5.2-91 to Figure 2.5.2-116

Figure 2.5.2-92 to Figure 2.5.2-117

Figure 2.5.2-93 to Figure 2.5.2-118

Figure 2.5.2-94 to Figure 2.5.2-119

Figure 2.5.2-95 to Figure 2.5.2-120

Figure 2.5.2-96 to Figure 2.5.2-121

Figure 2.5.2-97 to Figure 2.5.2-122

Figure 2.5.2-98 to Figure 2.5.2-123

Figure 2.5.2-99 to Figure 2.5.2-124

Figure 2.5.2-100 to Figure 2.5.2-125

Figure 2.5.2-101 to Figure 2.5.2-126

Figure 2.5.2-102 to Figure 2.5.2-127

Figure 2.5.2-103 to Figure 2.5.2-128

Figure 2.5.2-104 to Figure 2.5.2-129

Figure 2.5.2-105 to Figure 2.5.2-130

Figure 2.5.2-106 to Figure 2.5.2-131

Figure 2.5.2-107 to Figure 2.5.2-132

Figure 2.5.2-108 to Figure 2.5.2-133

Figure 2.5.2-109 to Figure 2.5.2-134

Figure 2.5.2-110 to Figure 2.5.2-135

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TID	Coordinated Universal Time (UTC)						Epicentral Location		Depth (km)	ERH (km)	E[M]	N*	sigM	EQNO	FLAG	Dist (km)
	Year	Month	Day	Hour	Minute	Second	Latitude	Longitude								
TMP00672	1817	12	11	0	0	0	35.73	-83.09	0	30	4.53	1.390	0.371	188	0	118
TMP00678	1818	8	14	14	36	0	36.091	-84.131	0	38	3.31	1.872	0.512	190	0	32
TMP00714	1824	8	22	0	0	0	34.828	-87.665	0	38	3.31	1.872	0.512	215	0	320
TMP00723	1827	5	11	0	0	0	36.1	-81.2	0	30	3.11	1.123	0.220	219	0	287
TMP00726	1827	8	7	7	0	0	38.3	-85.8	0	30	3.98	1.877	0.513	222	0	296
TMP00767	1834	11	20	19	40	0	38.65	-83.8	0	38	3.47	1.093	0.193	245	0	311
TMP00824	1843	8	9	0	0	0	35.6	-87.1	0	30	3.58	1.091	0.191	282	0	248
TMP00828	1843	10	30	3	0	0	38.63	-83.78	0	38	2.98	1.877	0.513	284	0	309
TMP00836	1844	11	28	8	0	0	35.95	-83.82	0	30	4.41	1.390	0.371	290	0	51
TMP00988	1854	2	28	0	0	0	37.6	-84	0	30	3.61	1.091	0.191	347	0	193
TMP01100	1860	10	22	5	0	0	34.13	-82.64	0	30	2.99	1.123	0.220	403	0	252
TMP01105	1861	1	3	16	30	0	35.09	-83.36	0	30	4.23	1.107	0.206	405	0	128
TMP01114	1861	8	31	5	0	0	35.91	-82.82	0	30	5.63	1.175	0.260	409	0	141
TMP01174	1868	2	22	11	0	0	36.78	-81.75	0	30	3.05	1.126	0.223	446	0	256
TMP01190	1869	2	20	0	0	0	38.1	-84.5	0	30	3.27	1.371	0.363	456	0	246
TMP01322	1874	3	1	0	0	0	35.929	-86.842	0	34	4.30	1.886	0.515	528	0	222
TMP01340	1875	4	10	0	0	0	35.29	-82.51	0	30	3.35	1.111	0.210	542	0	182
TMP01354	1875	11	1	22	30	0	33.49	-82.9	0	30	4.30	1.052	0.145	547	0	299
TMP01414	1877	10	9	1	0	0	35.3	-82.5	13	38	3.03	1.128	0.224	573	0	182
TMP01420	1877	11	16	7	38	0	35.5	-84	0	30	3.48	1.055	0.149	576	0	55
TMP01431	1878	5	18	0	0	0	35.5	-82.2	0	34	4.63	1.896	0.517	581	0	202
TMP01516	1881	9	18	4	0	0	33.379	-84.788	0	34	3.98	1.877	0.513	612	0	282
TMP01622	1884	8	24	19	45	0	36.07	-83.83	0	30	3.29	1.114	0.212	667	0	53
TMP01663	1885	8	6	9	0	0	36.12	-81.83	0	30	3.15	1.069	0.167	681	0	231
TMP01682	1886	2	5	2	0	0	35.93	-81.52	0	30	3.41	1.059	0.155	689	0	258
TMP01731	1886	9	1	6	0	0	33.91	-82.02	0	38	4.54	1.108	0.207	717	0	308
TMP01739	1886	9	1	9	45	0	34.3	-82.86	0	30	4.17	1.107	0.206	721	0	224
TMP02019	1886	10	22	5	0	0	34.71	-81.66	0	30	4.13	1.107	0.206	971	0	280
TMP02425	1888	9	17	21	30	0	33.42	-83.35	0	30	3.43	1.112	0.211	1287	0	290
TMP02486	1889	9	29	5	0	0	35.03	-84.87	0	30	3.46	1.111	0.210	1332	0	105
TMP02488	1889	10	16	3	0	0	35.26	-82.46	0	34	3.98	1.877	0.513	1334	0	187
TMP02489	1889	10	24	10	0	0	34.72	-82.6	0	30	3.11	1.123	0.220	1335	0	207

Table 2.5.2-2R Earthquakes within 320 km (200 mi) of the Clinch River Nuclear Site from the Updated CEUS SSC Earthquake Catalog for Mainshock (Independent) Earthquakes with Magnitudes E[M] ≥ 2.9 (Sheet 2 of 11)

TID	Coordinated Universal Time (UTC)						Epicentral Location		Depth (km)	ERH (km)	E[M]	N*	sigM	EQNO	FLAG	Dist (km)
	Year	Month	Day	Hour	Minute	Second	Latitude	Longitude								
TMP02511	1890	12	23	11	0	0	35.8	-84	0	38	3.27	1.371	0.363	1348	0	36
TMP02541	1891	11	17	0	0	0	36.728	-86.574	0	38	2.98	1.877	0.513	1371	0	218
TMP02582	1892	12	12	8	4	0	35.35	-85.55	0	38	3.96	1.106	0.205	1387	0	122
TMP02591	1893	1	11	20	0	0	38.3	-85.7	0	38	3.27	1.371	0.363	1389	0	292
TMP02784	1895	8	19	15	0	0	38.5	-83.7	0	38	3.40	1.093	0.193	1461	0	296
TMP02937	1897	9	29	0	0	0	34.941	-82.411	0	34	3.98	1.877	0.513	1530	0	207
TMP02973	1898	6	6	8	0	0	38.25	-84.05	0	34	3.75	1.091	0.191	1546	0	264
TMP02979	1898	6	19	15	0	0	36.07	-83.67	0	30	3.69	1.107	0.206	1549	0	67
TMP02981	1898	7	23	21	30	0	36.75	-82.83	0	30	3.54	1.107	0.206	1550	0	169
TMP03017	1899	10	5	10	15	0	35.556	-83.999	0	34	4.63	1.896	0.517	1570	0	51
TMP03088	1902	5	29	7	30	0	35.1	-85.3	0	20	3.27	1.371	0.363	1600	0	121
TMP03101	1902	10	18	22	0	0	35	-85.3	0	20	3.15	1.115	0.213	1607	0	129
TMP03116	1903	3	15	9	0	0	34.65	-84.8	0	25	3.44	1.371	0.363	1613	0	143
TMP03150	1904	3	5	0	30	0	35.7	-83.5	0	20	3.46	1.055	0.149	1629	0	82
TMP39338	1907	1	26	6	0	0	37.27	-81.223	0	25	2.98	1.877	0.513	1667	0	320
TMP39339	1908	12	27	21	15	0	36.8	-87.5	0	25	3.97	1.107	0.206	1696	0	297
TMP03390	1909	10	8	10	0	0	34.9	-85	0	20	3.13	1.056	0.151	1712	0	124
TMP03419	1910	6	8	23	0	0	34.769	-84.976	0	25	3.31	1.872	0.512	1728	0	136
TMP03440	1911	4	20	22	0	0	35.1	-82.7	0	25	3.21	1.046	0.137	1742	0	176
TMP03469	1913	1	1	18	28	0	34.7	-81.7	0	22	4.54	1.270	0.316	1762	0	277
TMP03474	1913	3	28	21	50	0	36.2	-83.7	0	22	3.65	1.055	0.149	1764	0	70
TMP03476	1913	4	17	16	30	0	35.3	-84.2	0	20	3.59	1.053	0.147	1765	0	68
TMP03487	1913	11	11	14	0	0	38.2	-85.8	0	22	3.61	1.373	0.364	1774	0	286
TMP03495	1914	1	24	3	24	0	35.6	-84.5	0	20	3.43	1.094	0.194	1776	0	34
TMP03504	1914	3	5	20	5	0	33.5	-83.5	0	25	4.43	1.052	0.146	1783	0	278
TMP03543	1915	10	29	5	23	0	35.8	-82.7	0	20	2.94	1.135	0.230	1806	0	152
TMP03555	1916	2	21	22	39	0	35.5	-82.5	0	20	5.13	1.270	0.316	1812	0	175
TMP03574	1916	8	26	19	36	0	36	-81	0	20	3.55	1.058	0.154	1820	0	305
TMP03577	1916	10	18	22	3	40	33.5	-86.2	0	22	4.98	1.270	0.316	1821	0	314
TMP39345	1917	3	27	20	0	0	36.1	-83.5	0	20	3.33	1.083	0.183	1833	0	83
TMP03631	1918	1	16	15	45	0	35.9	-83.9	0	20	3.27	1.100	0.200	1848	0	43
TMP03641	1918	6	22	1	0	0	36.1	-84.1	0	20	3.48	1.054	0.148	1855	0	34
TMP03697	1920	12	24	7	30	0	36	-85	0	20	3.30	1.118	0.216	1896	0	57

Table 2.5.2-2R Earthquakes within 320 km (200 mi) of the Clinch River Nuclear Site from the Updated CEUS SSC Earthquake Catalog for Mainshock (Independent) Earthquakes with Magnitudes E[M] ≥ 2.9 (Sheet 3 of 11)

TID	Coordinated Universal Time (UTC)						Epicentral Location		Depth (km)	ERH (km)	E[M]	N*	sigM	EQNO	FLAG	Dist (km)
	Year	Month	Day	Hour	Minute	Second	Latitude	Longitude								
TMP03723	1921	7	15	0	0	0	36.6	-82.3	0	20	3.93	1.371	0.363	1904	0	203
TMP03740	1921	12	15	13	20	0	35.8	-84.6	0	20	3.30	1.118	0.216	1913	0	22
TMP03808	1924	10	20	8	30	0	35	-82.6	0	20	4.21	1.039	0.126	1954	0	189
TMP03810	1924	11	13	10	30	0	36.6	-82.2	0	20	2.98	1.877	0.513	1956	0	211
TMP03952	1926	7	8	9	50	0	35.9	-82.1	0	22	3.23	1.152	0.243	2019	0	205
TMP04013	1927	6	16	12	0	0	34.7	-86	0	20	3.41	1.061	0.157	2048	0	198
TMP04019	1927	10	8	4	30	0	35.1	-85.3	0	20	3.01	1.122	0.219	2052	0	121
TMP04044	1928	3	7	2	45	0	35.6	-87	0	20	2.99	1.097	0.197	2070	0	239
TMP04091	1928	11	3	4	2	49.8	36.11	-82.83	0	39	4.43	1.029	0.109	2087	0	142
TMP04099	1928	11	20	3	45	0	35.8	-82.3	0	28	3.57	1.058	0.154	2092	0	188
TMP04195	1930	8	30	9	28	0	35.9	-84.4	0	20	3.18	1.097	0.197	2144	0	2
TMP04212	1930	10	16	21	50	0	36	-83.9	0	20	3.11	1.373	0.364	2151	0	45
TMP04256	1931	5	5	12	18	0	33.7	-86.6	0	22	3.86	1.056	0.151	2171	0	317
TMP04338	1933	5	28	15	10	0	38.6	-83.7	0	20	3.22	1.103	0.202	2210	0	307
TMP04404	1935	1	1	8	15	0	35.1	-83.6	0	39	3.59	1.040	0.128	2249	0	113
TMP04495	1936	1	1	8	0	0	35.1	-84	0	25	3.20	1.114	0.212	2274	0	94
TMP04616	1938	3	31	10	10	0	35.6	-83.5	0	20	3.59	1.091	0.191	2324	0	86
TMP04764	1939	5	5	2	45	0	33.7	-85.8	0	20	3.25	1.120	0.218	2364	0	276
TMP04785	1939	6	24	10	27	0	34.7	-86.6	0	20	3.27	1.063	0.160	2370	0	241
TMP04876	1940	10	19	5	54	0	34.7	-85.1	0	20	3.27	1.054	0.148	2417	0	148
TMP04889	1940	12	25	6	50	0	35.9	-82.9	0	20	3.45	1.060	0.156	2425	0	133
TMP04930	1941	9	8	9	45	0	35	-85.3	0	20	3.03	1.068	0.166	2449	0	129
TMP05142	1945	6	14	3	25	0	35	-84.5	0	20	3.57	1.058	0.153	2564	0	100
TMP05238	1947	12	27	19	0	0	35	-85.3	0	20	3.21	1.117	0.215	2624	0	129
TMP05256	1948	2	10	0	4	0	36.4	-84.1	0	20	3.53	1.088	0.188	2637	0	62
TMP05310	1949	9	17	9	30	0	36.7	-83	0	20	3.10	1.123	0.220	2663	0	153
TMP05331	1950	6	19	4	19	0	35.8	-84	0	20	3.62	1.052	0.146	2675	0	36
TMP05414	1952	6	11	20	20	0	36.3	-82.3	0	20	3.27	1.371	0.363	2731	0	192
TMP05488	1953	11	10	15	45	0	36	-83.9	0	20	2.94	1.136	0.231	2783	0	45
TMP05495	1954	1	2	3	25	0	36.6	-83.7	0	28	4.12	1.038	0.125	2790	0	100
TMP05523	1954	1	23	1	0	0	35.3	-84.4	0	20	3.05	1.116	0.214	2796	0	66
TMP05571	1955	1	6	20	30	0	36.6	-82.2	0	20	3.30	1.114	0.212	2826	0	211
TMP05582	1955	1	25	19	34	0	36	-83.9	0	20	3.23	1.103	0.202	2834	0	45

Table 2.5.2-2R Earthquakes within 320 km (200 mi) of the Clinch River Nuclear Site from the Updated CEUS SSC Earthquake Catalog for Mainshock (Independent) Earthquakes with Magnitudes E[M] ≥ 2.9 (Sheet 4 of 11)

TID	Coordinated Universal Time (UTC)						Epicentral Location		Depth (km)	ERH (km)	E[M]	N*	sigM	EQNO	FLAG	Dist (km)
	Year	Month	Day	Hour	Minute	Second	Latitude	Longitude								
TMP05618	1955	9	28	7	1	41.5	36.6	-81.3	0	28	3.48	1.055	0.150	2854	0	287
TMP05669	1956	9	7	13	35	50.8	36.44	-83.79	5	39	3.88	1.056	0.151	2891	0	81
TMP05670	1956	9	7	13	49	29	35.5	-84	0	17	3.88	1.052	0.145	2892	0	55
TMP05704	1957	4	23	9	23	39	33.77	-86.72	5	39	4.00	1.056	0.151	2915	0	318
TMP05710	1957	5	13	14	24	51.1	35.8	-82.14	5	39	3.79	1.057	0.152	2919	0	202
TMP05712	1957	6	23	6	34	16	35.95	-84.1	5	39	3.20	1.124	0.221	2921	0	26
TMP05714	1957	7	2	9	33	1	35.6	-82.7	0	22	3.63	1.040	0.128	2923	0	155
TMP05731	1957	11	24	20	6	17	35	-83.5	0	39	3.63	1.052	0.146	2937	0	127
TMP05782	1958	10	20	6	16	0	34.5	-82.7	0	20	3.31	1.872	0.512	2977	0	217
TMP05817	1959	8	12	18	6	1.4	34.79	-86.56	5	39	3.61	1.058	0.154	3003	0	233
TMP05833	1960	1	3	7	30	0	35.9	-82.1	0	25	3.07	1.125	0.222	3011	0	205
TMP05839	1960	2	9	14	0	6	35.3	-82.5	0	20	2.92	1.153	0.244	3015	0	182
TMP05844	1960	4	15	10	10	10	35.8	-84	0	28	3.37	1.062	0.158	3018	0	36
TMP06076	1963	10	28	22	38	0.3	36.7	-81	0	20	3.48	1.041	0.129	3157	0	316
TMP06093	1964	1	20	13	37	52	35.9	-82.3	0	20	2.91	1.139	0.233	3170	0	187
TMP06102	1964	2	18	9	31	10.4	34.67	-85.39	15	6	3.72	1.045	0.135	3176	0	164
TMP06105	1964	3	7	18	2	58.6	33.72	-82.39	5	39	2.98	1.153	0.244	3177	0	302
TMP06106	1964	3	13	1	20	17.5	33.19	-83.31	1	39	3.66	1.046	0.137	3178	0	316
TMP06271	1965	4	26	15	26	19.7	37.32	-81.6	5	6	3.18	1.153	0.244	3283	0	295
TMP06328	1965	9	9	14	42	20	34.7	-81.2	0	20	3.40	1.073	0.172	3316	0	317
TMP06370	1965	11	8	12	58	1	33.2	-83.2	0	39	2.90	1.074	0.173	3345	0	318
TMP06598	1966	8	24	6	0	0	35.8	-84	0	28	2.96	1.134	0.229	3408	0	36
TMP06816	1967	12	16	12	23	33.4	37.36	-81.6	2	17	3.18	1.153	0.244	3522	0	297
TMP06906	1968	12	11	16	0	0	38.3	-85.8	0	28	3.31	1.872	0.512	3585	0	296
TMP06934	1969	5	9	0	0	0	33.95	-82.58	0	17	2.95	1.074	0.173	3598	0	271
TMP06953	1969	7	13	21	51	9.8	36.12	-83.69	1	6	3.90	1.030	0.111	3610	0	67
TMP07012	1969	12	13	10	19	29.7	35.04	-82.85	6	17	3.46	1.025	0.101	3630	0	168
TMP07051	1970	7	30	8	48	53	36.999	-82.163	7	20	2.98	1.153	0.244	3653	0	234
TMP07053	1970	7	31	0	31	0	37.7	-83.4	0	20	3.18	1.153	0.244	3655	0	219
TMP07060	1970	9	10	1	41	5.2	36.02	-81.42	1	17	3.37	1.025	0.102	3660	0	267
TMP07153	1971	7	13	2	3	0	36	-84	0	20	3.18	1.044	0.134	3713	0	36
TMP07154	1971	7	13	3	3	0	36	-84.3	0	20	3.34	1.111	0.210	3714	0	14
TMP07159	1971	7	13	11	42	26	34.8	-83	0	17	3.63	1.018	0.087	3718	0	174

Table 2.5.2-2R Earthquakes within 320 km (200 mi) of the Clinch River Nuclear Site from the Updated CEUS SSC Earthquake Catalog for Mainshock (Independent) Earthquakes with Magnitudes E[M] ≥ 2.9 (Sheet 5 of 11)

TID	Coordinated Universal Time (UTC)						Epicentral Location		Depth (km)	ERH (km)	E[M]	N*	sigM	EQNO	FLAG	Dist (km)
	Year	Month	Day	Hour	Minute	Second	Latitude	Longitude								
TMP07200	1971	10	22	21	55	0	36	-83	0	20	2.98	1.153	0.244	3745	0	125
TMP07229	1972	1	9	23	24	29	37.4	-81.6	0	20	3.10	1.074	0.173	3760	0	299
TMP07259	1972	5	20	19	39	6	37	-82.2	0	20	3.30	1.079	0.178	3777	0	231
TMP07329	1973	1	7	22	56	6.2	37.4	-87.22	14	17	2.95	1.055	0.150	3810	0	304
TMP07436	1973	11	30	7	48	40.5	35.89	-83.99	12	6	4.01	1.090	0.190	3849	0	35
TMP07514	1974	6	5	0	16	40.2	38.48	-84.75	10	17	3.29	1.066	0.164	3884	0	290
TMP07565	1974	8	2	8	52	11.1	33.91	-82.53	4	6	3.91	1.055	0.150	3906	0	277
TMP07658	1974	11	5	3	0	0	33.73	-82.22	0	28	3.35	1.064	0.161	3937	0	311
TMP07827	1975	4	1	21	9	0	33.2	-83.2	0	8	3.89	1.047	0.139	4012	0	318
TMP07967	1975	8	29	4	22	52.1	33.66	-86.59	4	6	3.89	1.032	0.114	4075	0	320
TMP08078	1975	11	25	15	17	34.8	34.93	-82.9	10	6	3.21	1.062	0.159	4117	0	171
TMP08160	1976	1	19	6	20	39.6	36.87	-83.86	1	6	4.00	1.025	0.102	4138	0	118
TMP08183	1976	2	4	19	53	53	34.97	-84.7	14	6	3.37	1.032	0.115	4148	0	106
TMP08258	1976	4	15	7	3	34.4	37.38	-87.31	4	6	3.15	1.124	0.221	4180	0	309
TMP08330	1976	6	19	5	54	13.4	37.34	-81.6	1	17	3.29	1.029	0.110	4208	0	296
TMP08741	1977	7	27	22	3	20.8	35.42	-84.41	5	2	3.38	1.072	0.171	4352	0	52
TMP09145	1978	5	2	1	46	11.8	34.187	-82.738	10	6	3.13	1.096	0.196	4507	0	241
TMP09261	1978	7	9	7	3	35.6	35.505	-82.798	10	17	3.00	1.203	0.278	4549	0	149
TMP09822	1979	5	4	12	13	8.9	34.33	-81.95	1	8	2.93	1.208	0.281	4690	0	281
TMP09965	1979	7	19	10	27	3.9	35.229	-84.825	10	4	3.08	1.200	0.276	4724	0	84
TMP10004	1979	8	13	5	18	56.8	35.21	-84.36	10	1	3.63	1.071	0.169	4736	0	76
TMP39373	1979	8	13	5	19	25.2	33.9	-82.54	23	6	3.99	1.174	0.259	4737	0	278
TMP10034	1979	8	26	1	31	45	34.916	-82.956	1	22	3.64	1.056	0.151	4744	0	168
TMP10050	1979	9	6	20	38	16.3	35.3	-83.24	10	8	3.31	1.189	0.269	4749	0	122
TMP10054	1979	9	12	6	24	4	35.58	-83.91	12	2	3.20	1.072	0.171	4750	0	55
TMP10103	1979	10	8	8	53	52.8	36.44	-82.08	5	17	3.61	1.180	0.263	4762	0	215
TMP10151	1979	11	9	21	29	59.8	38.49	-82.81	1	6	3.49	1.062	0.158	4775	0	320
TMP10337	1980	3	23	21	38	16.2	37.6	-86.76	9	17	3.17	1.072	0.171	4828	0	285
TMP10453	1980	6	25	18	2	1.6	35.73	-84.03	1	2	3.03	1.124	0.221	4869	0	36
TMP10504	1980	7	27	18	52	21.4	38.19	-83.89	16	6	5.01	1.038	0.125	4888	0	259
TMP10851	1981	2	3	14	26	50	35.312	-84.539	4	1	3.00	1.203	0.278	4976	0	66
TMP10940	1981	4	2	6	32	39.1	35.319	-84.524	5	1	3.00	1.203	0.278	5004	0	65
TMP10960	1981	4	9	7	10	31.2	35.51	-82.05	0	1	3.28	1.046	0.137	5013	0	215

Table 2.5.2-2R Earthquakes within 320 km (200 mi) of the Clinch River Nuclear Site from the Updated CEUS SSC Earthquake Catalog for Mainshock (Independent) Earthquakes with Magnitudes E[M] ≥ 2.9 (Sheet 6 of 11)

TID	Coordinated Universal Time (UTC)						Epicentral Location		Depth (km)	ERH (km)	E[M]	N*	sigM	EQNO	FLAG	Dist (km)
	Year	Month	Day	Hour	Minute	Second	Latitude	Longitude								
TMP11061	1981	6	3	20	54	22.4	36.18	-81.67	1	8	2.91	1.124	0.221	5034	0	246
TMP11310	1981	9	4	17	21	44.5	34.63	-85.17	3	15	3.16	1.195	0.273	5078	0	157
TMP11484	1982	1	2	2	0	26.2	35.18	-86.43	13	1	3.07	1.065	0.162	5129	0	202
TMP11606	1982	1	30	12	39	12.9	35.8	-83.96	20	15	3.00	1.203	0.278	5212	0	39
TMP11787	1982	4	13	13	4	13.3	36.51	-82.04	3	15	3.16	1.195	0.273	5288	0	221
TMP12002	1982	7	16	14	16	2.9	34.32	-81.55	2	17	3.09	1.149	0.241	5354	0	311
TMP12114	1982	9	2	21	52	45.5	34.96	-82.9	3	8	3.16	1.195	0.273	5385	0	169
TMP12122	1982	9	5	10	11	9.4	35.19	-84.51	13	25	2.96	1.072	0.171	5389	0	79
TMP12170	1982	9	24	21	57	42.5	35.68	-84.24	13	25	3.28	1.031	0.113	5398	0	27
TMP39919	1982	9	24	22	19	21.5	35.86	-84.46	10	15	2.93	1.208	0.281	5400	0	8
TMP12312	1982	11	14	10	31	59.1	36.21	-82.88	10	15	2.93	1.208	0.281	5450	0	139
TMP12510	1983	1	31	23	41	1.4	34.302	-82.394	8	2	3.08	1.097	0.197	5545	0	253
TMP12633	1983	3	25	2	47	11.1	35.33	-82.46	12	25	3.21	1.048	0.140	5585	0	184
TMP12796	1983	7	8	19	29	5.9	35.55	-84.15	10	25	3.14	1.048	0.140	5656	0	43
TMP12879	1983	8	17	14	3	15	38.47	-82.77	10	2	3.49	1.039	0.126	5686	0	320
TMP12900	1983	8	28	22	45	7.4	36.68	-83.82	18	1	3.00	1.066	0.163	5691	0	101
TMP13080	1983	11	29	19	30	28.1	36.027	-82.653	7	15	2.93	1.208	0.281	5759	0	156
TMP13291	1984	2	14	20	54	30.9	36.13	-83.74	10	1	3.57	1.029	0.109	5831	0	63
TMP13442	1984	3	17	23	26	11.4	35.83	-84.05	7	1	3.24	1.082	0.181	5870	0	30
TMP13680	1984	8	9	2	42	35.8	34.62	-86.3	8	1	3.29	1.061	0.157	5960	0	224
TMP13717	1984	8	30	16	26	28.4	35.57	-84.34	13	15	3.26	1.043	0.132	5981	0	36
TMP13790	1984	10	9	11	54	26.9	34.75	-85.2	12	1	4.20	1.038	0.125	6018	0	147
TMP13826	1984	10	22	18	58	41.7	36.36	-81.68	11	1	3.18	1.072	0.171	6031	0	248
TMP13872	1984	11	18	20	23	58.1	34.744	-86.6	1	31	2.93	1.208	0.281	6044	0	238
TMP13931	1984	12	17	18	48	28.2	35.415	-84.309	14	15	3.00	1.203	0.278	6059	0	53
TMP14282	1985	7	12	18	20	28.4	35.198	-85.156	3	1	3.16	1.195	0.273	6182	0	104
TMP14583	1985	12	22	0	56	5	35.701	-83.72	13	5	3.38	1.186	0.267	6289	0	63
TMP14623	1986	1	7	1	26	43.3	35.609	-84.762	22	5	3.35	1.091	0.191	6305	0	47
TMP14699	1986	1	27	6	44	26.8	35.926	-83.636	20	5	2.93	1.208	0.281	6336	0	67
TMP14740	1986	2	13	11	35	45.3	34.793	-82.907	5	2	3.32	1.073	0.172	6351	0	181
TMP14850	1986	4	19	7	40	53	35.187	-85.51	27	5	3.16	1.195	0.273	6389	0	129
TMP14908	1986	5	19	23	46	47	35.516	-84.529	11	5	3.00	1.203	0.278	6414	0	44
TMP14989	1986	6	24	19	22	42	35.99	-83.931	24	5	3.08	1.200	0.276	6450	0	42

Table 2.5.2-2R Earthquakes within 320 km (200 mi) of the Clinch River Nuclear Site from the Updated CEUS SSC Earthquake Catalog for Mainshock (Independent) Earthquakes with Magnitudes E[M] ≥ 2.9 (Sheet 7 of 11)

TID	Coordinated Universal Time (UTC)						Epicentral Location		Depth (km)	ERH (km)	E[M]	N*	sigM	EQNO	FLAG	Dist (km)
	Year	Month	Day	Hour	Minute	Second	Latitude	Longitude								
TMP15024	1986	7	11	14	26	14.8	34.937	-84.987	13	22	3.65	1.028	0.108	6460	0	119
TMP15079	1986	8	7	12	36	46	35.506	-84.561	20	5	2.94	1.101	0.201	6481	0	46
TMP15118	1986	8	19	20	51	26	36.291	-85.02	30	5	3.08	1.200	0.276	6494	0	73
TMP15251	1986	10	18	8	31	38.8	34.946	-81.172	23	5	3.09	1.097	0.197	6530	0	309
TMP15373	1986	12	11	14	7	11.5	34.898	-82.88	9	2	3.16	1.195	0.273	6567	0	175
TMP15524	1987	2	22	10	35	26.5	36.34	-84.186	19	5	3.00	1.203	0.278	6613	0	53
TMP15601	1987	3	27	7	29	30.4	35.567	-84.229	19	22	4.03	1.034	0.118	6634	0	38
TMP15813	1987	6	19	1	5	14.1	36.431	-82.619	12	1	2.93	1.208	0.281	6694	0	169
TMP15870	1987	7	11	0	4	29.4	36.103	-83.817	25	25	3.53	1.029	0.110	6718	0	56
TMP16034	1987	9	22	17	23	50.1	35.623	-84.311	19	25	3.24	1.065	0.162	6760	0	30
TMP16085	1987	10	20	22	49	55.9	35.841	-84.444	13	25	2.93	1.168	0.255	6779	0	8
TMP16104	1987	11	4	20	30	45.8	34.699	-85.265	12	5	2.93	1.208	0.281	6788	0	155
TMP16141	1987	11	27	18	58	20.8	36.85	-83.092	13	1	2.98	1.100	0.200	6797	0	157
TMP16201	1987	12	18	23	20	17.4	35.104	-82.97	9	1	2.93	1.208	0.281	6818	0	155
TMP16239	1988	1	9	0	16	55.4	35.274	-84.188	9	5	3.08	1.200	0.276	6835	0	71
TMP16314	1988	2	16	15	26	54.5	36.561	-82.304	5	1	3.32	1.051	0.144	6860	0	201
TMP16691	1988	9	7	2	28	8.6	38.142	-83.834	10	1	4.14	1.035	0.120	6970	0	255
TMP16710	1988	9	18	16	16	1	37.31	-87.21	13	2	2.93	1.208	0.281	6979	0	298
TMP16778	1988	11	1	16	34	16.3	36.266	-83.714	20	5	2.93	1.208	0.281	7001	0	73
TMP17295	1989	7	15	0	8	2.6	38.607	-83.569	10	2	3.21	1.072	0.170	7140	0	311
TMP17298	1989	7	15	18	58	28.8	34.448	-87.339	10	1	2.93	1.082	0.181	7142	0	313
TMP17406	1989	9	7	5	18	6.2	35.493	-84.436	14	5	3.08	1.200	0.276	7177	0	44
TMP17629	1990	1	19	17	36	52.7	34.956	-84.907	11	5	3.00	1.203	0.278	7256	0	114
TMP17636	1990	1	24	18	20	24.4	38.133	-86.434	5	1	3.92	1.049	0.141	7258	0	309
TMP17705	1990	2	17	16	30	32	35.716	-84.23	1	5	2.93	1.208	0.281	7288	0	24
TMP17777	1990	3	21	9	11	39.4	38.142	-85.65	4	1	2.93	1.208	0.281	7311	0	275
TMP17970	1990	7	11	18	41	10.7	35.799	-83.991	17	5	3.08	1.200	0.276	7362	0	36
TMP18002	1990	7	28	11	9	39	34.919	-85.008	14	5	3.08	1.200	0.276	7373	0	122
TMP18041	1990	8	17	21	1	17.9	36.794	-83.34	10	1	3.54	1.036	0.121	7388	0	137
TMP18086	1990	9	8	0	3	57.4	38.061	-83.731	5	2	3.34	1.072	0.170	7410	0	248
TMP18110	1990	9	19	21	56	45	34.539	-85.491	6	5	3.08	1.200	0.276	7422	0	181
TMP18112	1990	9	20	10	36	18.8	34.989	-86.963	4	5	3.00	1.203	0.278	7424	0	255
TMP18383	1991	1	28	11	43	55.7	37.35	-87.32	1	2	3.00	1.203	0.278	7527	0	309

Table 2.5.2-2R Earthquakes within 320 km (200 mi) of the Clinch River Nuclear Site from the Updated CEUS SSC Earthquake Catalog for Mainshock (Independent) Earthquakes with Magnitudes E[M] ≥ 2.9 (Sheet 8 of 11)

TID	Coordinated Universal Time (UTC)						Epicentral Location		Depth (km)	ERH (km)	E[M]	N*	sigM	EQNO	FLAG	Dist (km)
	Year	Month	Day	Hour	Minute	Second	Latitude	Longitude								
TMP18590	1991	5	29	2	47	18.9	35.415	-84.292	22	5	3.08	1.200	0.276	7585	0	53
TMP18720	1991	8	17	17	59	9.2	34.914	-85.483	21	5	2.93	1.208	0.281	7622	0	148
TMP18769	1991	9	24	7	21	6.4	35.711	-84.095	5	25	3.09	1.072	0.170	7636	0	33
TMP18824	1991	10	30	14	54	12.6	34.904	-84.713	8	5	3.23	1.192	0.271	7655	0	114
TMP18951	1992	1	3	4	21	22.2	33.946	-82.465	5	25	3.48	1.070	0.168	7699	0	278
TMP18959	1992	1	7	11	48	40.1	33.835	-81.945	3	2	2.93	1.208	0.281	7703	0	319
TMP19475	1993	1	1	5	8	5.3	35.877	-82.09	3	1	3.16	1.195	0.273	7863	0	206
TMP19499	1993	1	15	2	2	51.8	35.075	-84.974	1	25	3.23	1.051	0.144	7876	0	105
TMP19660	1993	4	23	17	36	53.9	33.376	-82.853	0	4	2.93	1.208	0.281	7941	0	312
TMP20209	1994	4	5	22	21	59	34.961	-85.493	5	25	3.29	1.072	0.170	8121	0	144
TMP20284	1994	5	4	9	12	2.7	34.198	-87.174	5	1	3.15	1.052	0.145	8143	0	316
TMP20313	1994	5	26	4	6	4.5	35.176	-84.5	13	1	2.93	1.208	0.281	8154	0	80
TMP20769	1995	3	18	22	6	21	35.425	-84.922	17	1	3.35	1.144	0.237	8309	0	71
TMP20952	1995	6	26	0	36	17	36.747	-81.452	5	4	3.29	1.045	0.135	8367	0	279
TMP20969	1995	7	5	14	16	44.4	35.366	-84.212	10	3	3.54	1.104	0.203	8374	0	60
TMP20974	1995	7	7	21	1	2.8	36.515	-81.873	11	2	3.03	1.036	0.121	8378	0	235
TMP22251	1997	5	19	19	45	33.6	34.782	-85.444	5	25	2.98	1.072	0.171	8639	0	157
TMP22378	1997	7	19	17	6	34.3	35.056	-84.808	10	25	3.46	1.071	0.169	8677	0	101
TMP22404	1997	7	30	12	29	23.3	36.436	-83.509	5	25	3.70	1.036	0.122	8684	0	99
TMP22595	1997	10	19	18	39	55.1	35.286	-84.753	15	22	3.15	1.165	0.253	8734	0	75
TMP23124	1998	6	17	8	0	23.4	35.926	-84.405	10	25	3.38	1.048	0.140	8863	0	5
TMP23127	1998	6	18	5	9	59.5	34.714	-85.194	3	5	2.93	1.208	0.281	8865	0	150
TMP23201	1998	7	23	23	11	41.9	34.857	-84.947	7	5	2.93	1.208	0.281	8897	0	126
TMP23570	1999	1	17	18	38	4.7	36.854	-83.691	5	1	2.96	1.072	0.171	8968	0	124
TMP23888	1999	5	13	10	38	59.9	35.092	-87.026	22	25	2.91	1.156	0.246	9014	0	255
TMP24812	2000	3	20	10	0	50.2	34.685	-85.362	4	5	3.00	1.203	0.278	9145	0	161
TMP25320	2000	7	11	14	59	47.5	35.88	-83.81	5	25	3.12	1.082	0.182	9196	0	51
TMP25626	2000	9	14	16	5	22	34.675	-85.296	0	5	2.93	1.208	0.281	9239	0	159
TMP26409	2001	3	7	17	12	25	35.51	-84.81	6	25	3.09	1.072	0.171	9319	0	57
TMP26482	2001	3	21	23	35	35	34.857	-85.439	3	25	3.16	1.146	0.239	9328	0	150
TMP26588	2001	4	13	16	36	20.7	36.53	-83.34	0	1	3.16	1.195	0.273	9340	0	117
TMP26813	2001	5	17	6	43	48.5	36.045	-83.838	0	5	2.93	1.208	0.281	9354	0	52
TMP27286	2001	7	26	5	26	44.7	35.932	-83.575	5	1	3.12	1.072	0.170	9399	0	73

Table 2.5.2-2R Earthquakes within 320 km (200 mi) of the Clinch River Nuclear Site from the Updated CEUS SSC Earthquake Catalog for Mainshock (Independent) Earthquakes with Magnitudes E[M] ≥ 2.9 (Sheet 9 of 11)

TID	Coordinated Universal Time (UTC)						Epicentral Location		Depth (km)	ERH (km)	E[M]	N*	sigM	EQNO	FLAG	Dist (km)
	Year	Month	Day	Hour	Minute	Second	Latitude	Longitude								
TMP28136	2001	12	8	1	8	21.5	34.735	-86.245	5	2	3.68	1.063	0.160	9499	0	212
TMP39403	2003	3	18	6	4	24.2	33.689	-82.888	5	2	3.19	1.124	0.221	9741	0	280
TMP31104	2003	4	29	8	59	39	34.494	-85.629	19	22	4.57	1.024	0.100	9767	0	192
TMP32103	2004	3	13	17	7	58	36.48	-84.03	9	3	2.93	1.208	0.281	9960	0	73
TMP32131	2004	3	21	18	11	34.2	35.502	-82.053	9	5	2.93	1.208	0.281	9973	0	215
TMP33025	2004	9	17	15	21	43.6	36.933	-84.004	1	2	3.60	1.047	0.139	10096	0	121
TMP39406	2004	11	30	23	59	34.2	36.936	-83.893	10	2	3.16	1.195	0.273	10131	0	124
TMP33590	2004	12	23	6	54	21	35.43	-84.2	7	2	3.04	1.150	0.242	10144	0	54
TMP33796	2005	2	8	11	42	53	37.22	-81.93	9	2	2.93	1.208	0.281	10161	0	264
TMP33974	2005	3	18	1	2	16	35.72	-84.16	9	2	2.91	1.072	0.171	10175	0	27
TMP34061	2005	4	5	20	37	42.6	36.147	-83.693	10	3	3.08	1.200	0.276	10185	0	68
TMP34101	2005	4	14	15	38	16	35.47	-84.09	15	1	3.00	1.203	0.278	10190	0	54
TMP34631	2005	8	25	3	9	42	35.88	-82.8	8	1	3.63	1.024	0.100	10268	0	142
TMP34800	2005	10	12	6	27	30	35.51	-84.54	8	1	3.54	1.036	0.122	10287	0	45
TMP35348	2006	3	6	10	28	2	35.895	-82.359	0	4	3.20	1.154	0.245	10376	0	182
TMP35592	2006	5	10	12	17	29	35.53	-84.4	24	25	3.16	1.146	0.239	10410	0	40
TMP35729	2006	6	16	0	57	27.7	35.515	-83.229	4	1	3.17	1.072	0.170	10433	0	112
TMP35930	2006	8	7	8	44	28	34.94	-85.46	12	1	3.08	1.200	0.276	10453	0	144
TMP36410	2006	11	2	17	53	2.11	37.2	-81.92	1	2	3.98	1.153	0.244	10516	0	264
TMP36586	2006	12	18	8	34	27	35.36	-84.35	17	25	3.35	1.144	0.237	10528	0	59
TMP36651	2007	1	3	23	5	45	35.92	-83.95	15	2	2.98	1.100	0.200	10537	0	39
TMP37341	2007	6	19	18	16	27	35.79	-85.36	1	2	3.50	1.077	0.176	10636	0	89
TMP37480	2007	8	4	10	4	46	35.49	-82.09	9	2	3.00	1.073	0.172	10663	0	212
TMP37804	2007	10	23	5	16	12	35.59	-84.1	21	2	3.00	1.203	0.278	10720	0	42
TMP38741	2008	6	23	23	30	20	34.92	-84.84	9	2	3.12	1.082	0.182	10909	0	116
TMP39088	2008	10	31	16	37	34	35.77	-84	7	25	2.97	1.153	0.244	10964	0	37
UPD00016	2009	1	27	11	20	12	36.777	-84.132	25	--	3.31	1.161	0.250	11000	0	101
UPD00075	2009	4	4	20	45	32	33.183	-83.209	7	--	3.06	1.074	0.173	11059	0	320
UPD00113	2009	5	23	1	3	31	35.588	-84.159	6	--	2.93	1.161	0.250	11097	0	39
UPD00206	2009	8	31	14	7	10	35.778	-84.127	14	--	3.38	1.161	0.250	11190	0	26
UPD00520	2010	4	20	9	28	20.04	35.725	-84.001	2	--	2.98	1.148	0.240	11504	0	39
UPD00849	2010	8	30	16	31	3	36.49	-81.724	10	--	2.93	1.161	0.250	11833	0	248
UPD03141	2011	11	9	16	44	34.4	34.773	-84.982	5	--	2.93	1.161	0.250	14125	0	136

Table 2.5.2-2R Earthquakes within 320 km (200 mi) of the Clinch River Nuclear Site from the Updated CEUS SSC Earthquake Catalog for Mainshock (Independent) Earthquakes with Magnitudes E[M] ≥ 2.9 (Sheet 10 of 11)

TID	Coordinated Universal Time (UTC)						Epicentral Location		Depth (km)	ERH (km)	E[M]	N*	sigM	EQNO	FLAG	Dist (km)
	Year	Month	Day	Hour	Minute	Second	Latitude	Longitude								
UPD04173	2012	11	10	17	8	14.29	37.139	-83.054	17	--	4.17	1.012	0.071	15157	0	183
UPD05342	2013	8	25	19	50	40.29	36.165	-81.667	9	--	3.08	1.161	0.250	16326	0	246
UPD06716	2014	2	15	3	23	38.20	33.8166	-82.092	5	--	4.09	1.012	0.071	17700	0	311
UPD07596	2014	4	13	2	34	0.60	35.4486	-84.4665	14	--	3.00	1.161	0.250	18580	0	50
UPD07686	2014	4	21	5	5	7.00	35.555	-84.1578	12	--	3.00	1.161	0.250	18670	0	42
UPD08373	2014	6	8	12	12	4.90	35.5315	-85.0905	14	--	3.00	1.161	0.250	19357	0	76
UPD11454	2014	12	15	6	44	16.60	36.0589	-81.5195	13	--	2.98	1.074	0.173	22438	0	258
UPD13073	2015	3	8	17	35	1.07	36.6697	-84.1178	9	--	3.00	1.161	0.250	24057	0	90
UPD13195	2015	3	14	4	51	28.79	35.4547	-83.2798	0	--	3.00	1.161	0.250	24179	0	111
UPD13451	2015	3	26	3	21	9.97	35.5628	-84.659	17	--	3.00	1.161	0.250	24435	0	44
UPD16108	2015	7	21	3	10	45.37	36.4481667	-83.7421667	14	--	2.94	1.161	0.250	27092	0	84
UPD16631	2015	8	20	2	45	41.33	37.9953333	-84.8406667	34	--	2.98	1.161	0.250	27615	0	238
UPD17419	2015	10	6	2	52	13.09	38.2915	-85.261	4	--	2.96	1.161	0.250	28403	0	278
UPD20629	2016	5	8	9	9	56	36.1145	-83.7825	16	--	2.98	1.078	0.177	31928	0	59
UPD21201	2016	6	17	14	38	15	35.3501667	-84.8731667	18	--	2.91	1.078	0.177	32500	0	75
UPD23034	2016	10	25	2	42	50	35.8158333	-84.0938333	3	--	2.93	1.161	0.250	34333	0	27
UPD23113	2016	10	30	4	1	47	35.619	-84.2315	10	--	2.90	1.078	0.177	34412	0	33
UPD23615	2016	12	13	11	37	31	35.7103333	-82.8548333	6	--	2.91	1.078	0.177	34914	0	139
UPD23950	2017	1	15	16	47	29	36.4586667	-83.0818333	23	--	2.91	1.078	0.177	35249	0	133
UPD24412	2017	3	11	13	56	48	34.5706667	-87.1506667	5	--	2.99	1.078	0.177	35711	0	291
UPD24632	2017	4	6	1	49	13	33.2605	-83.04	11	--	3.00	1.078	0.177	35931	0	317
UPD25135	2017	6	2	14	8	36	34.4156667	-85.5663333	13	--	2.91	1.078	0.177	36434	0	196
UPD25272	2017	6	20	2	6	48	37.2525	-83.3596667	5	--	2.91	1.078	0.177	36571	0	177
UPD25314	2017	6	25	10	0	57	35.7468333	-84.256	21	--	2.91	1.078	0.177	36613	0	19
UPD25512	2017	7	17	12	44	57	36.1256667	-83.7026667	10	--	3.06	1.078	0.177	36811	0	66
UPD25566	2017	7	23	13	22	59	35.725	-84.165	23	--	3.01	1.078	0.177	36865	0	27
UPD25705	2017	8	4	0	40	15	35.945	-83.5611667	10	--	2.93	1.161	0.250	37004	0	74
UPD25961	2017	8	25	20	16	27	35.9343333	-84.0301667	11	--	2.98	1.078	0.177	37260	0	32
UPD26425	2017	10	20	15	38	11	34.7478333	-85.9156667	5	--	3.00	1.078	0.177	37724	0	189
UPD26443	2017	10	22	10	48	16	35.7848333	-84.645	24	--	3.05	1.078	0.177	37742	0	27
UPD26511	2017	10	30	5	39	37	36.8578333	-83.843	3	--	3.01	1.078	0.177	37810	0	118
UPD26765	2017	11	24	1	40	15	37.8671667	-84.7398333	35	--	2.93	1.078	0.177	38064	0	222
UPD26828	2017	11	30	6	16	33	36.3186667	-83.7091667	19	--	2.99	1.078	0.177	38127	0	77

Table 2.5.2-2R Earthquakes within 320 km (200 mi) of the Clinch River Nuclear Site from the Updated CEUS SSC Earthquake Catalog for Mainshock (Independent) Earthquakes with Magnitudes E[M] ≥ 2.9 (Sheet 11 of 11)

TID	Coordinated Universal Time (UTC)						Epicentral Location		Depth (km)	ERH (km)	E[M]	N*	sigM	EQNO	FLAG	Dist (km)
	Year	Month	Day	Hour	Minute	Second	Latitude	Longitude								
UPD26960	2017	12	17	5	8	36	38.6785	-83.8958333	14	--	3.06	1.078	0.177	38259	0	313
UPD27000	2017	12	20	23	55	4	34.657	-86.1143333	8	--	2.99	1.078	0.177	38299	0	209
UPD27036	2017	12	26	17	15	52	35.3401667	-83.867	9	--	3.01	1.078	0.177	38335	0	77
UPD27155	2018	1	7	9	53	20	34.9551667	-85.192	11	--	2.99	1.078	0.177	38454	0	127
UPD27252	2018	1	21	4	37	24	37.7581667	-84.663	11	--	2.91	1.078	0.177	38551	0	209
UPD27265	2018	1	22	12	7	44	36.895	-83.6473333	26	--	3.00	1.078	0.177	38564	0	130
UPD28070	2018	4	25	7	56	17	36.0261667	-83.7891667	8	--	3.01	1.078	0.177	39369	0	55
UPD28102	2018	4	29	6	39	23	34.9481667	-84.853	12	--	2.91	1.078	0.177	39401	0	113
UPD28108	2018	4	29	22	32	8	36.2968333	-83.3895	21	--	3.23	1.161	0.250	39407	0	100
UPD28260	2018	5	22	7	54	6	34.7206667	-85.3491667	10	--	2.92	1.078	0.177	39559	0	157
UPD28277	2018	5	24	20	8	51	35.8633333	-84.2945	16	--	3.01	1.078	0.177	39576	0	8
UPD28369	2018	6	10	11	51	58	35.6063333	-84.7268333	5	--	2.98	1.078	0.177	39668	0	44
UPD28374	2018	6	10	23	56	37	36.2665	-81.101	2	--	2.99	1.078	0.177	39673	0	298
UPD28911	2018	8	29	2	45	6	35.397	-84.424	18	--	2.98	1.078	0.177	40210	0	55
UPD28992	2018	9	9	3	0	4	34.8306667	-85.2381667	10	--	2.93	1.078	0.177	40291	0	141
UPD29021	2018	9	13	6	27	41	33.9026667	-82.2683333	0	--	2.91	1.078	0.177	40320	0	293
UPD29297	2018	10	23	9	48	17	34.624	-85.4253333	10	--	2.91	1.078	0.177	40596	0	170
UPD29393	2018	11	23	14	59	55	34.4363333	-85.0358333	9	--	3.00	1.078	0.177	40692	0	172
UPD29456	2018	12	12	9	14	44	35.6123333	-84.732	8	--	4.36	1.011	0.068	40755	0	44
UPD29471	2018	12	16	4	12	2	36.0383333	-83.6955	20	--	3.16	1.161	0.250	40770	0	64
UPD29496	2018	12	23	20	10	30	35.4911667	-84.1481667	19	--	3.06	1.078	0.177	40795	0	49

Replace Table 2.5.2-16 through Table 2.5.2-32 with the following:

Table 2.5.2-16R Summary of Source Parameters in the Revised Charleston RLME Source

Zone	Boundary Type	Sense of Slip	Rupture Strike	Rupture Dip	Recurrence Model
1886 Rupture (0.3)*	Leaky (1.0)	Reverse (1.0)	S0–15°W uniformly distributed (1.0)	40–50°W uniformly distributed (1.0)	Poisson (0.5) Renewal (0.5)
Local (0.5)	Leaky (1.0)	Reverse (0.5)	N-S (0.6)	30–60° random both directions (1.0)**	Poisson (0.9) Renewal (0.1)
			N45°E (0.2)		
			E-W (0.1)		
			N45°W (0.1)		
		Strike-slip (0.5)	N-S (0.3)	75–90° random both directions (1.0)**	
			N45°E (0.5)		
			E-W (0.1)		
			N45°W (0.1)		
Regional (0.2)	Strict (1.0)	Reverse (0.5)	N-S (0.6)	30–60° random both directions (1.0)**	Poisson (1.0)
			N45°E (0.2)		
			E-W (0.1)		
			N45°W (0.1)		
		Strike-slip (0.5)	N-S (0.3)	75–90° random both directions (1.0)**	
			N45°E (0.5)		
			E-W (0.1)		
			N45°W (0.1)		

Notes:

* See Figure 2.5.2-29R for geographic coordinates of the new “1886 Rupture” zone, which replaces the “Narrow” zone.

** Equally likely dip direction.

Table 2.5.2-17R Total Mean Rock Hazard for 23 Spectral Frequencies and PGA (Sheet 1 of 3)

(a) Spectral Frequencies of 200, 100, 50, 40, 33.3, 25, 20, and 13.3 Hz								
Spectral Acceleration	**Mean Annual Frequency of Exceedance (1/yr)**							
	200 Hz (PGA)	**100 Hz**	**50 Hz**	**40 Hz**	**33.33 Hz**	**25 Hz**	**20 Hz**	**13.33 Hz**
0.0001	1.23E-01	1.23E-01	1.23E-01	1.23E-01	1.24E-01	1.24E-01	1.24E-01	1.25E-01
0.000234	1.12E-01	1.12E-01	1.12E-01	1.13E-01	1.14E-01	1.15E-01	1.16E-01	1.18E-01
0.000549	8.43E-02	8.44E-02	8.56E-02	8.81E-02	8.95E-02	9.25E-02	9.54E-02	9.86E-02
0.001286	5.18E-02	5.19E-02	5.41E-02	5.66E-02	5.82E-02	6.14E-02	6.44E-02	6.80E-02
0.003012	2.86E-02	2.88E-02	3.10E-02	3.28E-02	3.40E-02	3.62E-02	3.81E-02	4.04E-02
0.007057	1.50E-02	1.54E-02	1.74E-02	1.86E-02	1.94E-02	2.08E-02	2.18E-02	2.28E-02
0.016532	7.16E-03	7.59E-03	9.03E-03	9.88E-03	1.04E-02	1.13E-02	1.18E-02	1.20E-02
0.03873	2.85E-03	3.15E-03	4.00E-03	4.47E-03	4.80E-03	5.29E-03	5.54E-03	5.40E-03
0.090733	9.44E-04	1.09E-03	1.49E-03	1.69E-03	1.84E-03	2.06E-03	2.15E-03	1.97E-03
0.21256	2.92E-04	3.49E-04	5.08E-04	5.92E-04	6.45E-04	7.20E-04	7.44E-04	6.52E-04
0.49798	8.14E-05	9.99E-05	1.58E-04	1.90E-04	2.09E-04	2.32E-04	2.37E-04	2.00E-04
1.1666	1.64E-05	2.08E-05	3.72E-05	4.70E-05	5.31E-05	5.98E-05	6.04E-05	4.88E-05
2.7331	1.96E-06	2.60E-06	5.46E-06	7.37E-06	8.68E-06	1.01E-05	1.03E-05	7.87E-06
6.4028	1.30E-07	1.81E-07	4.51E-07	6.54E-07	8.12E-07	1.00E-06	1.02E-06	7.43E-07
15.0	5.06E-09	7.35E-09	2.14E-08	3.32E-08	4.33E-08	5.62E-08	5.81E-08	4.03E-08

Table 2.5.2-17R Total Mean Rock Hazard for 23 Spectral Frequencies and PGA (Sheet 2 of 3)

(b) Spectral Frequencies of 10, 6.7, 5, 4, 3.3, 2.5, 2, and 1.3 Hz								
Spectral Acceleration	**Mean Annual Frequency of Exceedance (1/yr)**							
	10 Hz	**6.67 Hz**	**5 Hz**	**4 Hz**	**3.33 Hz**	**2.5 Hz**	**2 Hz**	**1.33 Hz**
0.0001	1.25E-01	1.25E-01	1.26E-01	1.26E-01	1.26E-01	1.25E-01	1.25E-01	1.24E-01
0.000234	1.20E-01	1.22E-01	1.22E-01	1.23E-01	1.23E-01	1.22E-01	1.20E-01	1.14E-01
0.000549	1.02E-01	1.07E-01	1.09E-01	1.09E-01	1.09E-01	1.06E-01	1.03E-01	8.79E-02
0.001286	7.20E-02	7.74E-02	8.00E-02	7.90E-02	7.92E-02	7.47E-02	6.96E-02	5.35E-02
0.003012	4.26E-02	4.57E-02	4.70E-02	4.57E-02	4.54E-02	4.18E-02	3.79E-02	2.73E-02
0.007057	2.35E-02	2.42E-02	2.41E-02	2.29E-02	2.24E-02	2.01E-02	1.78E-02	1.26E-02
0.016532	1.20E-02	1.17E-02	1.12E-02	1.04E-02	9.88E-03	8.64E-03	7.54E-03	5.35E-03
0.03873	5.22E-03	4.79E-03	4.42E-03	4.04E-03	3.78E-03	3.22E-03	2.74E-03	1.93E-03
0.090733	1.83E-03	1.56E-03	1.38E-03	1.24E-03	1.13E-03	9.16E-04	7.35E-04	4.86E-04
0.21256	5.66E-04	4.32E-04	3.46E-04	2.95E-04	2.52E-04	1.88E-04	1.39E-04	8.15E-05
0.49798	1.63E-04	1.11E-04	7.99E-05	6.27E-05	4.99E-05	3.38E-05	2.30E-05	1.17E-05
1.1666	3.71E-05	2.26E-05	1.47E-05	1.10E-05	8.15E-06	5.00E-06	3.08E-06	1.39E-06
2.7331	5.55E-06	2.94E-06	1.73E-06	1.23E-06	8.56E-07	4.75E-07	2.65E-07	1.09E-07
6.4028	4.85E-07	2.17E-07	1.16E-07	7.95E-08	5.23E-08	2.68E-08	1.38E-08	5.34E-09
15.0	2.44E-08	9.35E-09	4.57E-09	3.04E-09	1.91E-09	9.06E-10	4.31E-10	1.60E-10

Table 2.5.2-17R Total Mean Rock Hazard for 23 Spectral Frequencies and PGA (Sheet 3 of 3)

(c) Spectral Frequencies of 1, 0.67, 0.5, 0.33, 0.25, 0.2, 0.13, and 0.1 Hz								
Spectral Acceleration	**Mean Annual Frequency of Exceedance (1/yr)**							
	1 Hz	**0.67 Hz**	**0.5 Hz**	**0.33 Hz**	**0.25 Hz**	**0.2 Hz**	**0.13 Hz**	**0.1 Hz**
0.0001	1.21E-01	1.09E-01	9.63E-02	6.63E-02	5.07E-02	3.78E-02	2.18E-02	1.47E-02
0.000234	1.04E-01	8.04E-02	6.39E-02	3.82E-02	2.79E-02	2.05E-02	1.22E-02	8.50E-03
0.000549	7.22E-02	4.79E-02	3.58E-02	2.03E-02	1.48E-02	1.10E-02	7.03E-03	5.09E-03
0.001286	4.06E-02	2.50E-02	1.83E-02	1.06E-02	7.90E-03	6.11E-03	4.09E-03	2.94E-03
0.003012	2.00E-02	1.22E-02	9.21E-03	5.63E-03	4.22E-03	3.29E-03	2.18E-03	1.47E-03
0.007057	9.22E-03	5.87E-03	4.55E-03	2.81E-03	2.02E-03	1.53E-03	9.36E-04	5.53E-04
0.016532	3.95E-03	2.54E-03	1.97E-03	1.13E-03	7.41E-04	5.21E-04	2.67E-04	1.24E-04
0.03873	1.38E-03	8.29E-04	6.11E-04	3.03E-04	1.62E-04	1.00E-04	3.91E-05	1.35E-05
0.090733	3.23E-04	1.64E-04	1.07E-04	4.24E-05	1.77E-05	9.76E-06	3.06E-06	8.63E-07
0.21256	4.83E-05	2.05E-05	1.13E-05	3.74E-06	1.29E-06	6.43E-07	1.70E-07	3.99E-08
0.49798	6.14E-06	2.25E-06	1.06E-06	2.86E-07	7.72E-08	3.39E-08	7.18E-09	1.37E-09
1.1666	6.51E-07	2.05E-07	8.19E-08	1.70E-08	3.58E-09	1.37E-09	2.35E-10	3.69E-11
2.7331	4.60E-08	1.24E-08	4.29E-09	7.07E-10	1.21E-10	4.11E-11	5.62E-12	6.71E-13
6.4028	2.06E-09	4.74E-10	1.44E-10	1.87E-11	2.55E-12	7.43E-13	7.35E-14	5.98E-15
15.0	5.58E-11	1.07E-11	2.77E-12	2.61E-13	2.64E-14	6.25E-15	4.11E-16	2.16E-17

Table 2.5.2-18R Mean Magnitude (M), Distance (R), and Durations for Low Frequency and High Frequency Response Spectra for Eleven Mean Annual Frequency of Exceedances

Frequency range	Value	1.00E-02	4.00E-03	2.00E-03	1.00E-03	4.00E-04	1.00E-04	4.00E-05	1.00E-05	4.00E-06	1.00E-06	2.00E-07
HF	M	6.1	6.2	6.2	6.1	6.0	6.0	6.1	6.2	6.3	6.4	6.5
	R (km)	218.4	183.5	151.5	113.3	66.4	25.9	15.8	9.7	8.0	6.5	5.7
	Dur(sec)	21	18	14	11	9.0	6.5	6.5	6.5	6.5	6.5	6.5
LF	M	6.7*	7.1*	7.3*	7.4*	7.5*	7.6*	7.6*	7.7*	7.7*	6.7	6.8
	R (km)	388.4*	408.9*	418.8*	424.3*	426.3*	423.1*	418.2*	408.5*	399.7*	21.8	12.3
	Dur(sec)	42	48	51	54	56	57	57	56	56	9.0	8.6

Notes:

HF = High Frequency = 5 to 10 Hz (per RG 1.208)

LF = Low Frequency = 1 to 2.5 Hz (per RG 1.208)

M = Magnitude

R = Distance

*M and R calculated for R>100km per RG 1.208 because the contribution to hazard for R>100 km is more than 5% of the total hazard.

Table 2.5.2-19R Mean Rock Uniform Hazard Response Spectra for Mean Annual Frequency of Exceedance of 10^{-3}, 10^{-4}, 10^{-5}, and 10^{-6}

Frequency (Hz)	Horizontal Spectral Acceleration (g)			
	10^{-3} AFE	10^{-4}AFE	10^{-5} AFE	10^{-6} AFE
200	0.08680	0.43424	1.42247	3.37509
100	0.09669	0.49762	1.57475	3.71004
50	0.12418	0.65186	2.08952	4.87896
40	0.13904	0.73682	2.37521	5.51523
33.33333	0.14879	0.78789	2.55720	5.94161
25	0.16288	0.84486	2.74693	6.40258
20	0.16772	0.85247	2.75826	6.43931
13.33333	0.15297	0.75642	2.44382	5.75110
10	0.14075	0.65898	2.09984	4.97232
6.666667	0.12178	0.52755	1.63965	3.88705
5	0.11054	0.43728	1.36062	3.24590
4	0.10324	0.38518	1.20986	2.91671
3.333333	0.09722	0.34556	1.05992	2.57730
2.5	0.08549	0.29076	0.85680	2.08790
2	0.07438	0.24847	0.70835	1.72359
1.333333	0.05809	0.19280	0.53061	1.30347
1	0.04684	0.15347	0.40726	0.99113
0.666667	0.03360	0.11116	0.28017	0.66449
0.5	0.02707	0.09293	0.22216	0.50796
0.333333	0.01791	0.06256	0.15055	0.32905
0.25	0.01282	0.04664	0.10927	0.22934
0.2	0.00987	0.03879	0.08994	0.18511
0.133333	0.00660	0.02555	0.06108	0.12608
0.1	0.00421	0.01795	0.04254	0.08668

Table 2.5.2-20R Summary of Subsurface Layer Elevations from Boring Logs

Boring ID	Existing elevation (ft)	Bottom depth elevation (ft)		Bedrock starts at (ft)
		Fill+Residuum (ft)	Gravel (ft)	
MP-101	800.5	787.5	779.9	779.9
MP-102	797.9	793.4	776.3	776.3
MP-103	800.6	783.6	780.4	780.4
MP-104	797.7	786.5	784.9	784.9
MP-105	800.2	790.7	783.0	783.0
MP-106	798.7	793.7	774.5	774.5
MP-107	801.6	762.6	761.6	761.6
MP-108	798.5	790.4	790.4	790.4
MP-109	799.9	794.9	783.6	783.6
MP-110	798.7	784.4	774.7	774.7
MP-111	801.1	766.4	765.1	765.1
MP-114	797.2	785.3	-	785.3
MP-115	796.9	779.9	779.4	779.4
MP-116	797.6	764.9	763.6	763.6
MP-117	800.0	774.0	770.3	770.3
MP-118	799.8	781.2	780.7	780.7
MP-119	802.1	798.1	781.2	781.2
MP-120	800.1	775.1	770.2	770.2
MP-121	797.6	774.3	773.6	773.6
MP-122	796.7	769.7	765.8	765.8
U1-B01	799.0	782.8	-	782.8
U1-B02	800.3	781.9	776.2	776.2
U1-B03	797.4	780.4	778.7	778.7
U1-B04	798.9	782.9	775.6	775.6
U1-B05	800.6	771.3	766.9	766.9
U1-B06	800.4	788.6	785.6	785.6
U1-B07	801.2	774.2	772.6	772.6
U1-B08	800.0	786.5	786.3	786.3
U1-B09	797.0	784.5	781.8	781.8
U1-B10	796.2	774.3	-	774.3
U1-B11	795.6	770.8	761.9	761.9
U1-B12	797.3	781.2	-	781.2
Median	**799**	**781**	**777**	**777**

Table 2.5.2-21R Summary of Subsurface Layer Depths Considering the Final Surface Elevation of 814.5 ft

Boring ID	Bottom depth (ft)		Bedrock starts at (ft)
	Existing Fill+Residuum (ft)	Gravel (ft)	
MP-101	27.0	34.6	34.6
MP-102	21.1	38.2	38.2
MP-103	30.9	34.1	34.1
MP-104	28.0	29.6	29.6
MP-105	23.8	31.5	31.5
MP-106	20.8	40.0	40.0
MP-107	51.9	52.9	52.9
MP-108	24.1	24.1	24.1
MP-109	19.6	30.9	30.9
MP-110	30.1	39.8	39.8
MP-111	48.1	49.4	49.4
MP-114	29.2	-	29.2
MP-115	34.6	35.1	35.1
MP-116	49.6	50.9	50.9
MP-117	40.5	44.2	44.2
MP-118	33.3	33.8	33.8
MP-119	16.4	33.3	33.3
MP-120	39.4	44.3	44.3
MP-121	40.2	40.9	40.9
MP-122	44.8	48.7	48.7
U1-B01	31.7	-	31.7
U1-B02	32.6	38.3	38.3
U1-B03	34.1	35.8	35.8
U1-B04	31.6	38.9	38.9
U1-B05	43.2	47.6	47.6
U1-B06	25.9	28.9	28.9
U1-B07	40.3	41.9	41.9
U1-B08	28.0	28.2	28.2
U1-B09	30.0	32.7	32.7
U1-B10	40.2	-	40.2
U1-B11	43.7	52.6	52.6
U1-B12	33.3	-	33.3
Median	**33.5**	**37.5**	**37.5**
Standard Deviation	**0.3**	**0.2**	**0.2**
Median + Std	**42.5**	**45.4**	**45.4**
Median - Std	**24.3**	**30.6**	**30.6**

**Table 2.5.2-22R Discretized Site Profile for Case 1 with Existing Fill and Residuum
(Sheet 1 of 4)**

Layer	Elevation (ft)	Depth to Unit Top (ft)	Thick-ness (ft)	BE Vs (ft/s)	LR Vs (ft/s)	UR Vs (ft/s)	Unit Weight (pcf)	Aleatory σ_{lnVs}	Epistemic σ_{lnVs}	Geologic Unit	Material Model 1 (0.33)	Material Model 2 (0.33)	Material Model 3 (0.34)
1	814.5	0	15.5	1000	774	1292	135	0.15	0.2	Fill	EPRI Soil (0 - 20ft)	PR Soil (0 - 50ft)	Rollins et al. 2020 Gravel (0-20 ft)
2	799.0	15.5	18	600	464	775	120	0.25	0.2	Existing Fill and Residuum	EPRI Clay (PI=30)	EPRI Clay (PI=40)	Clinch River Soil
3	781.0	33.5	4	1870	1448	2416	140	0.21	0.2	Gravelly Soil	Rollins et al. 2020 Gravel (20-50 ft)	PR Soil (0 - 50ft)	EPRI Rock (0-20ft)
4	777.0	37.5	6	4000	3097	5167	168	0.21	0.2	Limestone	EPRI Rock (501-1000ft)	EPRI Rock (501-1000ft, Linear)	Clinch River Rock
5	771.0	43.5	7.09	6101	4723	7880	168	0.19	0.2				
6	763.9	50.6	3.28	7450	5767	9624	168	0.18	0.2				
7	760.6	53.9	4.92	7703	5963	9951	168	0.17	0.2				
8	755.7	58.8	4.92	7816	6051	10097	168	0.16	0.2				
9	750.8	63.7	4.92	8158	6316	10539	168	0.15	0.2				
10	745.9	68.6	4.93	8027	6214	10369	168	0.15	0.2				
11	740.9	73.6	4.92	7966	6167	10291	168	0.15	0.2				
12	736.0	78.5	4.92	7659	5929	9893	168	0.15	0.2				
13	731.1	83.4	4.92	7698	5959	9943	168	0.15	0.2				
14	726.2	88.3	4.92	8054	6235	10403	168	0.15	0.2				
15	721.3	93.2	4.92	8121	6286	10490	168	0.15	0.2				
16	716.3	98.2	4.92	7885	6104	10186	168	0.15	0.2				
17	711.4	103.1	5.25	8188	6339	10577	168	0.15	0.2				
18	706.2	108.3	4.6	8350	6464	10786	168	0.15	0.2				
19	701.6	112.9	4.92	8127	6291	10498	168	0.15	0.2				
20	696.7	117.9	4.92	8007	6199	10343	168	0.15	0.2				
21	691.7	122.8	4.92	8476	6562	10950	168	0.15	0.2				
22	686.8	127.7	4.92	8541	6612	11000	168	0.15	0.2				
23	681.9	132.6	4.92	8356	6469	10794	168	0.15	0.2				
24	677.0	137.5	4.92	8008	6199	10344	168	0.15	0.2				
25	672.1	142.5	4.92	8117	6284	10485	168	0.15	0.2				
26	667.1	147.4	5.25	8318	6439	10745	168	0.15	0.2				
27	661.9	152.6	4.6	8259	6394	10669	168	0.15	0.2				
28	657.3	157.2	4.92	8285	6414	10702	168	0.15	0.2				
29	652.4	162.1	5.25	8665	6708	11000	168	0.15	0.2				
30	647.1	167.4	4.59	8142	6303	10518	168	0.15	0.2				
31	642.5	172.0	4.92	7849	6076	10139	168	0.15	0.2				
32	637.6	176.9	4.92	7781	6024	10052	168	0.15	0.2				

Table 2.5.2-22R Discretized Site Profile for Case 1 with Existing Fill and Residuum
(Sheet 2 of 4)

Layer	Elevation (ft)	Depth to Unit Top (ft)	Thick-ness (ft)	BE Vs (ft/s)	LR Vs (ft/s)	UR Vs (ft/s)	Unit Weight (pcf)	Aleatory σ_{lnVs}	Epistemic σ_{lnVs}	Geologic Unit	Material Model 1 (0.33)	Material Model 2 (0.33)	Material Model 3 (0.34)
33	632.7	181.8	4.92	8285	6414	10702	168	0.15	0.2	Limestone	EPRI Rock (501-1000ft)	EPRI Rock (501-1000ft, Linear)	Clinch River Rock
34	627.8	186.7	4.93	7972	6172	10298	168	0.15	0.2				
35	622.8	191.7	4.92	8172	6326	10556	168	0.15	0.2				
36	617.9	196.6	4.92	8276	6407	10690	168	0.15	0.2				
37	613.0	201.5	4.92	8259	6394	10669	168	0.15	0.2				
38	608.1	206.4	4.92	8636	6686	11000	168	0.15	0.2				
39	603.2	211.4	4.92	8540	6611	11000	168	0.15	0.2				
40	598.2	216.3	4.92	9013	6977	11000	168	0.15	0.2				
41	593.3	221.2	4.92	9396	7274	11000	168	0.15	0.2				
42	588.4	226.1	4.93	9489	7345	11000	168	0.15	0.2				
43	583.5	231.0	4.92	9290	7191	11000	168	0.15	0.2				
44	578.5	236.0	4.92	9369	7253	11000	168	0.15	0.2				
45	573.6	240.9	4.92	9342	7232	11000	168	0.15	0.2				
46	568.7	245.8	4.92	9283	7186	11000	168	0.15	0.2				
47	563.8	250.7	4.92	9299	7199	11000	168	0.15	0.2				
48	558.9	255.6	4.92	9445	7312	11000	168	0.15	0.2				
49	553.9	260.6	4.92	9107	7050	11000	168	0.15	0.2				
50	549.0	265.5	4.93	9644	7466	11000	168	0.15	0.2				
51	544.1	270.4	4.92	10040	7773	11000	168	0.15	0.2				
52	539.2	275.3	4.92	9562	7402	11000	168	0.15	0.2				
53	534.3	280.3	4.92	9542	7387	11000	168	0.15	0.2				
54	529.3	285.2	4.92	9485	7342	11000	168	0.15	0.2				
55	524.4	290.1	4.92	9310	7207	11000	168	0.15	0.2				
56	519.5	295.0	4.92	9713	7519	11000	168	0.15	0.2				
57	514.6	299.9	4.92	10160	7865	11000	168	0.15	0.2				
58	509.7	304.9	4.93	9740	7540	11000	168	0.15	0.2				
59	504.7	309.8	4.92	9620	7447	11000	168	0.15	0.2				
60	499.8	314.7	4.92	9869	7640	11000	168	0.15	0.2				
61	494.9	319.6	4.92	9901	7665	11000	168	0.15	0.2				
62	490.0	324.5	4.92	10105	7823	11000	168	0.15	0.2				
63	485.0	329.5	4.92	10250	7935	11000	168	0.15	0.2				
64	480.1	334.4	4.92	9760	7556	11000	168	0.15	0.2				
65	475.2	339.3	4.92	10127	7840	11000	168	0.15	0.2				
66	470.3	344.2	4.93	10410	8059	11000	168	0.15	0.2				

Table 2.5.2-22R Discretized Site Profile for Case 1 with Existing Fill and Residuum
(Sheet 3 of 4)

Layer	Elevation (ft)	Depth to Unit Top (ft)	Thick-ness (ft)	BE Vs (ft/s)	LR Vs (ft/s)	UR Vs (ft/s)	Unit Weight (pcf)	Aleatory σ_{lnVs}	Epistemic σ_{lnVs}	Geologic Unit	Material Model 1 (0.33)	Material Model 2 (0.33)	Material Model 3 (0.34)
67	465.4	349.2	4.92	10438	8080	11000	168	0.15	0.2	Limestone	EPRI Rock (501-1000ft)	EPRI Rock (501-1000ft, Linear)	Clinch River Rock
68	460.4	354.1	4.92	9631	7456	11000	168	0.15	0.2				
69	455.5	359.0	4.92	9913	7674	11000	168	0.15	0.2				
70	450.6	363.9	4.92	9961	7711	11000	168	0.15	0.2				
71	445.7	368.8	4.92	10160	7866	11000	168	0.15	0.2				
72	440.8	373.8	4.92	10132	7844	11000	168	0.15	0.2				
73	435.8	378.7	4.92	9731	7533	11000	168	0.15	0.2				
74	430.9	383.6	4.93	9425	7297	11000	168	0.15	0.2				
75	426.0	388.5	4.92	9901	7665	11000	168	0.15	0.2				
76	421.1	393.4	4.92	8944	6924	11000	168	0.15	0.2				
77	416.1	398.4	4.92	8635	6685	11000	168	0.15	0.2				
78	411.2	403.3	4.92	8838	6842	11000	168	0.15	0.2				
79	406.3	408.2	4.92	9400	7277	11000	168	0.15	0.2				
80	401.4	413.1	4.92	9133	7070	11000	168	0.15	0.2				
81	396.5	418.0	4.92	9076	7026	11000	168	0.15	0.2				
82	391.5	423.0	4.93	8797	6810	11000	168	0.15	0.2				
83	386.6	427.9	4.92	9255	7164	11000	168	0.15	0.2				
84	381.7	432.8	4.92	9495	7350	11000	168	0.15	0.2				
85	376.8	437.7	4.92	8991	6960	11000	168	0.15	0.2				
86	371.9	442.7	4.92	8709	6742	11000	168	0.15	0.2				
87	366.9	447.6	4.92	8498	6579	10978	168	0.15	0.2				
88	362.0	452.5	4.92	8453	6544	10920	168	0.15	0.2				
89	357.1	457.4	4.92	8004	6196	10339	168	0.15	0.2				
90	352.2	462.3	4.93	8161	6318	10542	168	0.15	0.2				
91	347.2	467.3	4.92	8537	6609	11000	168	0.15	0.2				
92	342.3	472.2	4.92	8525	6600	11000	168	0.15	0.2				
93	337.4	477.1	4.92	8631	6681	11000	168	0.15	0.2				
94	332.5	482.0	4.92	9023	6985	11000	168	0.15	0.2				
95	327.6	486.9	4.92	8868	6865	11000	168	0.15	0.2				
96	322.6	491.9	4.92	8531	6605	11000	168	0.15	0.2				
97	317.7	496.8	4.92	9147	7081	11000	168	0.15	0.2				

**Table 2.5.2-22R Discretized Site Profile for Case 1 with Existing Fill and Residuum
(Sheet 4 of 4)**

Layer	Elevation (ft)	Depth to Unit Top (ft)	Thick-ness (ft)	BE Vs (ft/s)	LR Vs (ft/s)	UR Vs (ft/s)	Unit Weight (pcf)	Aleatory σ_{lnVs}	Epistemic σ_{lnVs}	Geologic Unit	Material Model 1 (0.33)	Material Model 2 (0.33)	Material Model 3 (0.34)
98	312.8	501.7	4.93	8213	6358	10610	168	0.15	0.2	Limestone	Linear	Linear	Linear
99	307.9	506.6	4.92	8080	6255	10437	168	0.15	0.2				
100	303.0	511.6	4.92	8113	6281	10480	168	0.15	0.2				
101	298.0	516.5	4.92	8233	6374	10635	168	0.15	0.2				
102	293.1	521.4	4.92	8930	6913	11000	168	0.15	0.2				
103	288.2	526.3	4.92	9000	6967	11000	168	0.15	0.2				
104	283.3	531.2	4.92	8063	6242	10416	168	0.15	0.2				
105	278.4	536.2	4.92	7847	6074	10136	168	0.15	0.2				
106	273.4	541.1	3.43	7200	4600	11000	168	0.15	0.35				
107	270.0	544.5	184	7200	4600	11000	168	0.15	0.35	Chickamauga Group			
108	86.0	728.5	102	9000	5750	11000	168	0.15	0.35				
109	-16.0	830.5	254	8200	5239	11000	168	0.15	0.35				
110	-270.0	1084.5	3997	10999	7028	11000	168	0.15	0.35	Newala + Knox group			
111	-4266.8	5081.3	2100	7200	4600	11000	168	0.15	0.35	Conasauga Shale			
112	-6366.7	7181.2	2300	10150	6485	11000	168	0.15	0.35	Rome Fm + Pumpkin Valley Shale			
113	-8666.6	9481.1	700	7333	4685	11000	168	0.15	0.35	Conasauga Shale			
114	-9366.6	10181.1	700	7333	4685	11000	168	0.15	0.35				
115	-10066.5	10881.0	1798	10227	6534	11000	168	0.15	0.35	Rome Fm + Pumpkin Valley Shale			
116	-11864.8	12679.3		11549	11549	11549	168			Basement			

Table 2.5.2-23R Discretized Site Profile for Case 2 with Backfill up to Hard Rock
(Sheet 1 of 4)

Layer	Elevation (ft)	Depth to Unit Top (ft)	Thick-ness (ft)	BE Vs (ft/s)	LR Vs (ft/s)	UR Vs (ft/s)	Unit Weight (pcf)	Aleatory σ_{lnVs}	Epistemic σ_{lnVs}	Geologic Unit	Material Model 1 (0.33)	Material Model 2 (0.33)	Material Model 3 (0.34)
1	814.5	0	15.5	1000	774	1292	135	0.15	0.2	Fill	EPRI Soil (0 - 20ft)	PR Soil (0 - 50ft)	Rollins et al. 2020 Gravel (0-20 ft)
2	799.0	15.5	4.5	1000	774	1292	135	0.15	0.2				
3	794.5	20	17.5	1000	774	1292	135	0.15	0.2		EPRI Soil (21 - 50ft)	PR Soil (0 - 50ft)	Rollins et al. 2020 Gravel (20-50 ft)
4	777.0	37.5	6	4000	3097	5167	168	0.21	0.2	Limestone	EPRI Rock (501-1000 ft)	EPRI Rock (501-1000ft, Linear)	Clinch River Rock
5	771.0	43.5	7.09	6101	4723	7880	168	0.19	0.2				
6	763.9	50.6	3.28	7450	5767	9624	168	0.18	0.2				
7	760.6	53.9	4.92	7703	5963	9951	168	0.17	0.2				
8	755.7	58.8	4.92	7816	6051	10097	168	0.16	0.2				
9	750.8	63.7	4.92	8158	6316	10539	168	0.15	0.2				
10	745.9	68.6	4.93	8027	6214	10369	168	0.15	0.2				
11	740.9	73.6	4.92	7966	6167	10291	168	0.15	0.2				
12	736.0	78.5	4.92	7659	5929	9893	168	0.15	0.2				
13	731.1	83.4	4.92	7698	5959	9943	168	0.15	0.2				
14	726.2	88.3	4.92	8054	6235	10403	168	0.15	0.2				
15	721.3	93.2	4.92	8121	6286	10490	168	0.15	0.2				
16	716.3	98.2	4.92	7885	6104	10186	168	0.15	0.2				
17	711.4	103.1	5.25	8188	6339	10577	168	0.15	0.2				
18	706.2	108.3	4.6	8350	6464	10786	168	0.15	0.2				
19	701.6	112.9	4.92	8127	6291	10498	168	0.15	0.2				
20	696.7	117.9	4.92	8007	6199	10343	168	0.15	0.2				
21	691.7	122.8	4.92	8476	6562	10950	168	0.15	0.2				
22	686.8	127.7	4.92	8541	6612	11000	168	0.15	0.2				
23	681.9	132.6	4.92	8356	6469	10794	168	0.15	0.2				
24	677.0	137.5	4.92	8008	6199	10344	168	0.15	0.2				
25	672.1	142.5	4.92	8117	6284	10485	168	0.15	0.2				
26	667.1	147.4	5.25	8318	6439	10745	168	0.15	0.2				
27	661.9	152.6	4.6	8259	6394	10669	168	0.15	0.2				
28	657.3	157.2	4.92	8285	6414	10702	168	0.15	0.2				
29	652.4	162.1	5.25	8665	6708	11000	168	0.15	0.2				
30	647.1	167.4	4.59	8142	6303	10518	168	0.15	0.2				
31	642.5	172.0	4.92	7849	6076	10139	168	0.15	0.2				
32	637.6	176.9	4.92	7781	6024	10052	168	0.15	0.2				

Table 2.5.2-23R Discretized Site Profile for Case 2 with Backfill up to Hard Rock
(Sheet 2 of 4)

Layer	Elevation (ft)	Depth to Unit Top (ft)	Thick-ness (ft)	BE Vs (ft/s)	LR Vs (ft/s)	UR Vs (ft/s)	Unit Weight (pcf)	Aleatory σ_{lnVs}	Epistemic σ_{lnVs}	Geologic Unit	Material Model 1 (0.33)	Material Model 2 (0.33)	Material Model 3 (0.34)
33	632.7	181.8	4.92	8285	6414	10702	168	0.15	0.2	Limestone	EPRI Rock (501-1000 ft)	EPRI Rock (501-1000ft, Linear)	Clinch River Rock
34	627.8	186.7	4.93	7972	6172	10298	168	0.15	0.2				
35	622.8	191.7	4.92	8172	6326	10556	168	0.15	0.2				
36	617.9	196.6	4.92	8276	6407	10690	168	0.15	0.2				
37	613.0	201.5	4.92	8259	6394	10669	168	0.15	0.2				
38	608.1	206.4	4.92	8636	6686	11000	168	0.15	0.2				
39	603.2	211.4	4.92	8540	6611	11000	168	0.15	0.2				
40	598.2	216.3	4.92	9013	6977	11000	168	0.15	0.2				
41	593.3	221.2	4.92	9396	7274	11000	168	0.15	0.2				
42	588.4	226.1	4.93	9489	7345	11000	168	0.15	0.2				
43	583.5	231.0	4.92	9290	7191	11000	168	0.15	0.2				
44	578.5	236.0	4.92	9369	7253	11000	168	0.15	0.2				
45	573.6	240.9	4.92	9342	7232	11000	168	0.15	0.2				
46	568.7	245.8	4.92	9283	7186	11000	168	0.15	0.2				
47	563.8	250.7	4.92	9299	7199	11000	168	0.15	0.2				
48	558.9	255.6	4.92	9445	7312	11000	168	0.15	0.2				
49	553.9	260.6	4.92	9107	7050	11000	168	0.15	0.2				
50	549.0	265.5	4.93	9644	7466	11000	168	0.15	0.2				
51	544.1	270.4	4.92	10040	7773	11000	168	0.15	0.2				
52	539.2	275.3	4.92	9562	7402	11000	168	0.15	0.2				
53	534.3	280.3	4.92	9542	7387	11000	168	0.15	0.2				
54	529.3	285.2	4.92	9485	7342	11000	168	0.15	0.2				
55	524.4	290.1	4.92	9310	7207	11000	168	0.15	0.2				
56	519.5	295.0	4.92	9713	7519	11000	168	0.15	0.2				
57	514.6	299.9	4.92	10160	7865	11000	168	0.15	0.2				
58	509.7	304.9	4.93	9740	7540	11000	168	0.15	0.2				
59	504.7	309.8	4.92	9620	7447	11000	168	0.15	0.2				
60	499.8	314.7	4.92	9869	7640	11000	168	0.15	0.2				
61	494.9	319.6	4.92	9901	7665	11000	168	0.15	0.2				
62	490.0	324.5	4.92	10105	7823	11000	168	0.15	0.2				
63	485.0	329.5	4.92	10250	7935	11000	168	0.15	0.2				
64	480.1	334.4	4.92	9760	7556	11000	168	0.15	0.2				
65	475.2	339.3	4.92	10127	7840	11000	168	0.15	0.2				
66	470.3	344.2	4.93	10410	8059	11000	168	0.15	0.2				

Table 2.5.2-23R Discretized Site Profile for Case 2 with Backfill up to Hard Rock
(Sheet 3 of 4)

Layer	Elevation (ft)	Depth to Unit Top (ft)	Thick-ness (ft)	BE Vs (ft/s)	LR Vs (ft/s)	UR Vs (ft/s)	Unit Weight (pcf)	Aleatory σ_{lnVs}	Epistemic σ_{lnVs}	Geologic Unit	Material Model 1 (0.33)	Material Model 2 (0.33)	Material Model 3 (0.34)
67	465.4	349.2	4.92	10438	8080	11000	168	0.15	0.2	Limestone	EPRI Rock (501-1000 ft)	EPRI Rock (501-1000ft, Linear)	Clinch River Rock
68	460.4	354.1	4.92	9631	7456	11000	168	0.15	0.2				
69	455.5	359.0	4.92	9913	7674	11000	168	0.15	0.2				
70	450.6	363.9	4.92	9961	7711	11000	168	0.15	0.2				
71	445.7	368.8	4.92	10160	7866	11000	168	0.15	0.2				
72	440.8	373.8	4.92	10132	7844	11000	168	0.15	0.2				
73	435.8	378.7	4.92	9731	7533	11000	168	0.15	0.2				
74	430.9	383.6	4.93	9425	7297	11000	168	0.15	0.2				
75	426.0	388.5	4.92	9901	7665	11000	168	0.15	0.2				
76	421.1	393.4	4.92	8944	6924	11000	168	0.15	0.2				
77	416.1	398.4	4.92	8635	6685	11000	168	0.15	0.2				
78	411.2	403.3	4.92	8838	6842	11000	168	0.15	0.2				
79	406.3	408.2	4.92	9400	7277	11000	168	0.15	0.2				
80	401.4	413.1	4.92	9133	7070	11000	168	0.15	0.2				
81	396.5	418.0	4.92	9076	7026	11000	168	0.15	0.2				
82	391.5	423.0	4.93	8797	6810	11000	168	0.15	0.2				
83	386.6	427.9	4.92	9255	7164	11000	168	0.15	0.2				
84	381.7	432.8	4.92	9495	7350	11000	168	0.15	0.2				
85	376.8	437.7	4.92	8991	6960	11000	168	0.15	0.2				
86	371.9	442.7	4.92	8709	6742	11000	168	0.15	0.2				
87	366.9	447.6	4.92	8498	6579	10978	168	0.15	0.2				
88	362.0	452.5	4.92	8453	6544	10920	168	0.15	0.2				
89	357.1	457.4	4.92	8004	6196	10339	168	0.15	0.2				
90	352.2	462.3	4.93	8161	6318	10542	168	0.15	0.2				
91	347.2	467.3	4.92	8537	6609	11000	168	0.15	0.2				
92	342.3	472.2	4.92	8525	6600	11000	168	0.15	0.2				
93	337.4	477.1	4.92	8631	6681	11000	168	0.15	0.2				
94	332.5	482.0	4.92	9023	6985	11000	168	0.15	0.2				
95	327.6	486.9	4.92	8868	6865	11000	168	0.15	0.2				
96	322.6	491.9	4.92	8531	6605	11000	168	0.15	0.2				
97	317.7	496.8	4.92	9147	7081	11000	168	0.15	0.2				

Table 2.5.2-23R Discretized Site Profile for Case 2 with Backfill up to Hard Rock
(Sheet 4 of 4)

Layer	Elevation (ft)	Depth to Unit Top (ft)	Thick-ness (ft)	BE Vs (ft/s)	LR Vs (ft/s)	UR Vs (ft/s)	Unit Weight (pcf)	Aleatory σ_{lnVs}	Epistemic σ_{lnVs}	Geologic Unit	Material Model 1 (0.33)	Material Model 2 (0.33)	Material Model 3 (0.34)
98	312.8	501.7	4.93	8213	6358	10610	168	0.15	0.2	Limestone	Linear	Linear	Linear
99	307.9	506.6	4.92	8080	6255	10437	168	0.15	0.2				
100	303.0	511.6	4.92	8113	6281	10480	168	0.15	0.2				
101	298.0	516.5	4.92	8233	6374	10635	168	0.15	0.2				
102	293.1	521.4	4.92	8930	6913	11000	168	0.15	0.2				
103	288.2	526.3	4.92	9000	6967	11000	168	0.15	0.2				
104	283.3	531.2	4.92	8063	6242	10416	168	0.15	0.2				
105	278.4	536.2	4.92	7847	6074	10136	168	0.15	0.2				
106	273.4	541.1	3.43	7200	4600	11000	168	0.15	0.35				
107	270.0	544.5	184	7200	4600	11000	168	0.15	0.35	Chickamauga Group			
108	86.0	728.5	102	9000	5750	11000	168	0.15	0.35				
109	-16.0	830.5	254	8200	5239	11000	168	0.15	0.35				
110	-270.0	1084.5	3997	10999	7028	11000	168	0.15	0.35	Newala + Knox group			
111	-4266.8	5081.3	2100	7200	4600	11000	168	0.15	0.35	Conasauga Shale			
112	-6366.7	7181.2	2300	10150	6485	11000	168	0.15	0.35	Rome Fm + Pumpkin Valley Shale			
113	-8666.6	9481.1	700	7333	4685	11000	168	0.15	0.35	Conasauga Shale			
114	-9366.6	10181.1	700	7333	4685	11000	168	0.15	0.35				
115	-10066.5	10881.0	1798	10227	6534	11000	168	0.15	0.35	Rome Fm + Pumpkin Valley Shale			
116	-11864.8	12679.3		11549	11549	11549	168			Basement			

Table 2.5.2-24R Kappa Target Ranges

Kappa (sec)		
UR Vs Profile	**BE Vs Profile**	**LR Vs Profile**
0.0121	0.0173	0.0248

Table 2.5.2-25R Additional Damping in Deep Profile

Damping ratio in deep profile (%)								
UR Vs profile			BE Vs profile			LR Vs profile		
M1	M2	M3	M1	M2	M3	M1	M2	M3
0.15	0.15	0.15	0.30	0.29	0.29	0.35	0.34	0.35

Table 2.5.2-26R Modulus Reduction and Damping Ratio Curves for Clinch River Soil

Shear Strain (%)	Damping Ratio, D (%)	Normalized Shear Modulus, G/Gmax
0.000024	3.16	1.00
0.000047	3.15	1.00
0.000094	3.14	1.00
0.000186	3.13	1.00
0.00038	3.10	1.00
0.00076	3.09	1.00
0.00141	3.15	0.99
0.00278	3.29	0.97
0.00531	3.64	0.93
0.00943	4.41	0.86
0.0155	5.64	0.75
0.0247	7.59	0.62
0.0372	10.19	0.52
0.1	16*	0.32
0.32		0.14
1		0.05

* Damping ratio is capped at 15% for this curve and all the following MRD curves

Table 2.5.2-27R Modulus Reduction and Damping Ratio Curves for Clinch River Rock

Shear Strain (%)	Damping Ratio, D (%)	Normalized Shear Modulus, G/Gmax
0.000003	1.11	1.00
0.000005	1.11	1.00
0.00001	1.11	1.00
0.00002	1.12	1.00
0.000042	1.13	1.00
0.000083	1.14	1.00
0.00017	1.18	0.99
0.00035	1.26	0.99
0.00069	1.40	0.97
0.0013	1.61	0.95
0.0021	1.83	0.93
0.0032	2.11	0.90
0.005	2.54	0.86
0.01	3.6	0.77
0.032	6.1	0.53
0.1	10	0.28
0.32	16	0.12
1	22	0.04

Table 2.5.2-28R Modulus Reduction and Damping Ratio Curves for EPRI Soil 0-20 ft and 21-50 ft

Shear Strain (%)	Damping Ratio, D (%)		Normalized Shear Modulus, G/Gmax	
	0-20 ft	21-50 ft	0-20 ft	21-50 ft
0.0001	1.4	1.2	1	1
0.0003162	1.5	1.2	1	1
0.001	1.8	1.4	0.98	0.99
0.003162	2.8	2.1	0.914	0.946
0.01	5	3.6	0.75	0.82
0.03162	9.3	7	0.509	0.608
0.1	15.3	12.4	0.27	0.36
0.3162	21.9	19.1	0.116	0.165
1	27	24.9	0.04	0.06
3.162	30	28	0.02	0.03

Table 2.5.2-29R Modulus Reduction and Damping Ratio Curves for Peninsular Range (PR) Soil 0-50 ft

Shear Strain (%)	Damping Ratio, D (%)	Normalized Shear Modulus, G/Gmax
0.0001	1.06	1
0.000316	1.06	1
0.001	1.21	1
0.00316	1.66	0.963
0.01	2.88	0.868
0.0316	5.6	0.677
0.1	10.59	0.427
0.316	16.65	0.221
1	23	0.088

Table 2.5.2-30R Modulus Reduction and Damping Ratio Curves for EPRI Clay PI=30 and PI=40

Shear Strain (%)	Damping Ratio, D (%)		Normalized Shear Modulus, G/Gmax	
	PI=30	PI=40	PI=30	PI=40
0.0001	1.43	1.68	1	1
0.000178	1.42	1.67	1	1
0.000316	1.42	1.67	1	1
0.000562	1.51	1.74	1	1
0.001	1.60	1.78	0.997	0.999
0.00178	1.74	1.89	0.990	0.995
0.00316	2.03	2.08	0.968	0.982
0.00562	2.48	2.41	0.931	0.956
0.01	3.10	2.93	0.872	0.912
0.0178	4.11	3.73	0.784	0.841
0.0316	5.64	4.99	0.671	0.746
0.0562	7.74	6.76	0.543	0.629
0.1	10.30	9.03	0.409	0.497
0.178	13.20	11.71	0.288	0.370
0.316	16.09	14.51	0.188	0.259
0.562	18.39	17.01	0.117	0.171
1	18.96	18.19	0.074	0.114

Table 2.5.2-31R Modulus Reduction and Damping Ratio Curves for EPRI Rock 0-20 ft and 501-1000 ft

Shear Strain (%)	Damping Ratio, D (%)		Normalized Shear Modulus, G/Gmax	
	0-20 ft	501-1000 ft	0-20 ft	501-1000 ft
0.0001	2	2	1	1
0.0003162	2.39	2.133	1	1
0.001	3.017	2.24	0.9716	1
0.003162	4.58	2.713	0.8614	0.9822
0.01	9.191	5.208	0.6294	0.8784
0.03162	14.397	8.176	0.383	0.6796
0.1	21.091	13.593	0.1747	0.4097
0.3162	26.578	19.667	0.0714	0.2067
1	30.601	25.909	0.0238	0.0796
3.162	32.53	29.904	0.0084	0.0296

Table 2.5.2-32R Modulus Reduction and Damping Ratio Curves for Rollins et al. (2020) Gravel 0-20 ft and 20-50 ft

Shear Strain (%)	Damping Ratio, D (%)		Normalized Shear Modulus, G/Gmax	
	0-20ft	20-50ft	0-20ft	20-50ft
0.0001	1.1	1.1	0.99	0.993
0.000178	1.1	1.1	0.984	0.989
0.000316	1.13	1.05	0.974	0.983
0.000562	1.4	1.31	0.959	0.972
0.001	1.73	1.62	0.935	0.956
0.00178	2.15	2.02	0.899	0.931
0.00316	2.67	2.5	0.846	0.892
0.00562	3.31	3.1	0.772	0.836
0.01	4.1	3.84	0.676	0.759
0.0178	5.07	4.75	0.563	0.659
0.0316	6.26	5.86	0.443	0.545
0.0562	7.7	7.21	0.329	0.424
0.1	9.41	8.82	0.232	0.312
0.178	11.39	10.67	0.157	0.219
0.316	13.55	12.69	0.103	0.147
0.562	15.77	14.77	0.066	0.096
1	17.84	16.71	0.042	0.062

Add new Tables 2.5.2-33 through 2.5.2-42:

Table 2.5.2-33 High Frequency (HF) and Low Frequency (LF) Input Control Rock Response Spectra for Eleven Mean Annual Frequency of Exceedances
(Sheet 1 of 2)

(a) HF Input Response Spectra											
Freq. (Hz)	Horizontal Spectral Acceleration (g)										
	10^{-2} AFE	$4x10^{-3}$ AFE	$2x10^{-3}$ AFE	10^{-3} AFE	$4x10^{-4}$ AFE	10^{-4} AFE	$4x10^{-5}$ AFE	10^{-5} AFE	$4x10^{-6}$ AFE	10^{-6} AFE	10^{-7} AFE
200	1.13E-02	2.83E-02	5.09E-02	8.68E-02	1.69E-01	4.34E-01	7.26E-01	1.42E+00	2.05E+00	3.38E+00	5.59E+00
100	1.19E-02	3.07E-02	5.57E-02	9.67E-02	1.92E-01	4.98E-01	8.18E-01	1.57E+00	2.29E+00	3.71E+00	6.20E+00
50	1.45E-02	3.87E-02	7.03E-02	1.24E-01	2.53E-01	6.52E-01	1.12E+00	2.09E+00	3.04E+00	4.88E+00	8.04E+00
40	1.63E-02	4.27E-02	7.84E-02	1.39E-01	2.85E-01	7.37E-01	1.26E+00	2.38E+00	3.39E+00	5.52E+00	8.98E+00
33.3333	1.73E-02	4.55E-02	8.41E-02	1.49E-01	3.05E-01	7.88E-01	1.33E+00	2.56E+00	3.61E+00	5.94E+00	9.62E+00
25	1.90E-02	4.99E-02	9.29E-02	1.63E-01	3.31E-01	8.45E-01	1.41E+00	2.75E+00	3.85E+00	6.40E+00	1.03E+01
20	1.99E-02	5.19E-02	9.62E-02	1.68E-01	3.37E-01	8.52E-01	1.42E+00	2.76E+00	3.87E+00	6.44E+00	1.04E+01
13.3333	2.01E-02	4.99E-02	8.97E-02	1.53E-01	3.02E-01	7.56E-01	1.28E+00	2.44E+00	3.49E+00	5.75E+00	9.39E+00
10	1.99E-02	4.81E-02	8.45E-02	1.41E-01	2.69E-01	6.59E-01	1.12E+00	2.10E+00	3.06E+00	4.97E+00	8.24E+00
6.6667	1.92E-02	4.44E-02	7.51E-02	1.22E-01	2.23E-01	5.28E-01	8.61E-01	1.64E+00	2.40E+00	3.89E+00	6.55E+00
5	1.83E-02	4.17E-02	6.91E-02	1.11E-01	1.95E-01	4.37E-01	7.06E-01	1.36E+00	1.96E+00	3.25E+00	5.39E+00
4	1.71E-02	3.90E-02	6.44E-02	1.03E-01	1.77E-01	3.85E-01	6.20E-01	1.21E+00	1.73E+00	2.92E+00	4.81E+00
3.3333	1.63E-02	3.68E-02	6.07E-02	9.72E-02	1.64E-01	3.46E-01	5.53E-01	1.06E+00	1.53E+00	2.58E+00	4.26E+00
2.5	1.43E-02	3.21E-02	5.35E-02	8.55E-02	1.42E-01	2.91E-01	4.58E-01	8.57E-01	1.26E+00	2.09E+00	3.53E+00
2	1.23E-02	2.75E-02	4.55E-02	7.18E-02	1.17E-01	2.38E-01	3.75E-01	7.02E-01	1.04E+00	1.72E+00	2.92E+00
1.3333	8.56E-03	1.92E-02	3.14E-02	4.84E-02	7.73E-02	1.57E-01	2.52E-01	4.78E-01	7.17E-01	1.20E+00	2.07E+00
1	6.12E-03	1.38E-02	2.25E-02	3.41E-02	5.35E-02	1.08E-01	1.74E-01	3.34E-01	5.08E-01	8.63E-01	1.50E+00
0.6667	3.54E-03	8.12E-03	1.31E-02	1.93E-02	2.94E-02	5.93E-02	9.81E-02	1.92E-01	2.99E-01	5.18E-01	9.20E-01
0.5	2.36E-03	5.46E-03	8.77E-03	1.26E-02	1.85E-02	3.68E-02	6.16E-02	1.22E-01	1.93E-01	3.39E-01	6.11E-01
0.3333	1.11E-03	2.61E-03	4.17E-03	5.85E-03	8.44E-03	1.68E-02	2.88E-02	5.84E-02	9.38E-02	1.68E-01	3.10E-01
0.25	6.88E-04	1.61E-03	2.57E-03	3.54E-03	4.97E-03	9.56E-03	1.62E-02	3.29E-02	5.33E-02	9.64E-02	1.79E-01
0.2	4.48E-04	1.06E-03	1.68E-03	2.29E-03	3.17E-03	6.09E-03	1.04E-02	2.13E-02	3.49E-02	6.37E-02	1.19E-01
0.1333	2.07E-04	4.99E-04	7.87E-04	1.04E-03	1.40E-03	2.73E-03	4.82E-03	1.01E-02	1.69E-02	3.14E-02	5.99E-02
0.1	1.18E-04	2.86E-04	4.49E-04	5.86E-04	7.77E-04	1.49E-03	2.64E-03	5.59E-03	9.40E-03	1.77E-02	3.41E-02

Table 2.5.2-33 High Frequency (HF) and Low Frequency (LF) Input Control Rock Response Spectra for Eleven Mean Annual Frequency of Exceedances
(Sheet 2 of 2)

(b) LF Input Response Spectra											
Freq. (Hz)	Horizontal Spectral Acceleration (g)										
	10^{-2} AFE	$4x10^{-3}$ AFE	$2x10^{-3}$ AFE	10^{-3} AFE	$4x10^{-4}$ AFE	10^{-4} AFE	$4x10^{-5}$ AFE	10^{-5} AFE	$4x10^{-6}$ AFE	10^{-6} AFE	10^{-7} AFE
200	7.67E-03	1.78E-02	3.02E-02	4.87E-02	8.18E-02	1.71E-01	2.71E-01	5.18E-01	7.70E-01	2.77E+00	4.84E+00
100	7.86E-03	1.82E-02	3.08E-02	4.98E-02	8.35E-02	1.74E-01	2.76E-01	5.28E-01	7.86E-01	3.23E+00	5.67E+00
50	9.09E-03	2.09E-02	3.54E-02	5.71E-02	9.60E-02	2.01E-01	3.20E-01	6.17E-01	9.23E-01	4.55E+00	8.06E+00
40	9.87E-03	2.26E-02	3.81E-02	6.14E-02	1.03E-01	2.16E-01	3.44E-01	6.66E-01	9.98E-01	5.14E+00	9.13E+00
33.3333	1.03E-02	2.35E-02	3.95E-02	6.36E-02	1.07E-01	2.24E-01	3.57E-01	6.91E-01	1.04E+00	5.45E+00	9.68E+00
25	1.12E-02	2.51E-02	4.21E-02	6.74E-02	1.13E-01	2.37E-01	3.77E-01	7.31E-01	1.10E+00	5.84E+00	1.03E+01
20	1.19E-02	2.63E-02	4.39E-02	7.01E-02	1.17E-01	2.45E-01	3.90E-01	7.55E-01	1.14E+00	5.92E+00	1.03E+01
13.3333	1.30E-02	2.85E-02	4.72E-02	7.53E-02	1.25E-01	2.63E-01	4.18E-01	8.11E-01	1.22E+00	5.41E+00	9.45E+00
10	1.41E-02	3.10E-02	5.12E-02	8.15E-02	1.35E-01	2.82E-01	4.49E-01	8.64E-01	1.30E+00	4.89E+00	8.44E+00
6.6667	1.58E-02	3.47E-02	5.73E-02	9.10E-02	1.51E-01	3.12E-01	4.96E-01	9.45E-01	1.41E+00	4.08E+00	6.99E+00
5	1.60E-02	3.52E-02	5.80E-02	9.23E-02	1.53E-01	3.15E-01	4.99E-01	9.43E-01	1.40E+00	3.34E+00	5.67E+00
4	1.58E-02	3.52E-02	5.84E-02	9.32E-02	1.54E-01	3.18E-01	5.03E-01	9.49E-01	1.41E+00	2.91E+00	4.94E+00
3.3333	1.56E-02	3.47E-02	5.74E-02	9.16E-02	1.52E-01	3.11E-01	4.91E-01	9.22E-01	1.37E+00	2.55E+00	4.30E+00
2.5	1.43E-02	3.21E-02	5.35E-02	8.55E-02	1.42E-01	2.91E-01	4.58E-01	8.57E-01	1.26E+00	2.09E+00	3.53E+00
2	1.25E-02	2.82E-02	4.75E-02	7.44E-02	1.24E-01	2.48E-01	3.83E-01	7.08E-01	1.04E+00	1.72E+00	2.96E+00
1.3333	8.86E-03	2.11E-02	3.76E-02	5.81E-02	9.95E-02	1.93E-01	2.91E-01	5.31E-01	7.65E-01	1.30E+00	2.23E+00
1	6.45E-03	1.63E-02	2.87E-02	4.68E-02	8.01E-02	1.53E-01	2.30E-01	4.07E-01	5.86E-01	9.91E-01	1.70E+00
0.6667	3.81E-03	1.04E-02	1.98E-02	3.36E-02	5.68E-02	1.11E-01	1.62E-01	2.80E-01	3.99E-01	6.64E-01	1.18E+00
0.5	2.72E-03	8.05E-03	1.63E-02	2.71E-02	4.76E-02	9.29E-02	1.32E-01	2.22E-01	3.09E-01	5.08E-01	8.67E-01
0.3333	1.39E-03	4.57E-03	9.70E-03	1.79E-02	3.23E-02	6.26E-02	9.26E-02	1.51E-01	2.08E-01	3.29E-01	5.55E-01
0.25	9.32E-04	3.20E-03	7.12E-03	1.28E-02	2.33E-02	4.66E-02	6.64E-02	1.09E-01	1.47E-01	2.29E-01	3.73E-01
0.2	6.33E-04	2.30E-03	5.23E-03	9.87E-03	1.90E-02	3.88E-02	5.42E-02	8.99E-02	1.20E-01	1.85E-01	2.98E-01
0.1333	3.18E-04	1.32E-03	3.29E-03	6.60E-03	1.26E-02	2.55E-02	3.84E-02	6.11E-02	8.29E-02	1.26E-01	2.03E-01
0.1	1.82E-04	7.97E-04	2.06E-03	4.21E-03	8.48E-03	1.80E-02	2.55E-02	4.25E-02	5.65E-02	8.67E-02	1.36E-01

Table 2.5.2-34 Control Point Hazard Curves for PBSRS

Period (sec)	Annual Frequency of Exceedance (1/yr)														
	Acceleration (g)														
	1.00E-04	2.34E-04	5.49E-04	1.29E-03	3.01E-03	7.06E-03	1.65E-02	3.87E-02	9.07E-02	2.13E-01	4.98E-01	1.17E+00	2.73E+00	6.40E+00	1.50E+01
0.01	1.30E-01	1.18E-01	9.88E-02	6.83E-02	4.02E-02	2.20E-02	1.13E-02	4.49E-03	1.27E-03	2.52E-04	4.28E-05	6.51E-06	6.51E-07	3.07E-08	1.15E-09
0.02	1.28E-01	1.16E-01	9.57E-02	6.57E-02	3.93E-02	2.22E-02	1.19E-02	4.97E-03	1.53E-03	3.43E-04	7.00E-05	1.26E-05	1.64E-06	1.01E-07	3.88E-09
0.025	1.28E-01	1.17E-01	9.67E-02	6.70E-02	4.05E-02	2.32E-02	1.26E-02	5.49E-03	1.77E-03	4.38E-04	9.29E-05	1.74E-05	2.50E-06	1.80E-07	7.04E-09
0.03	1.27E-01	1.17E-01	9.75E-02	6.82E-02	4.16E-02	2.40E-02	1.32E-02	5.89E-03	1.96E-03	5.04E-04	1.12E-04	2.11E-05	3.22E-06	2.57E-07	1.07E-08
0.04	1.27E-01	1.18E-01	9.96E-02	7.09E-02	4.37E-02	2.53E-02	1.40E-02	6.49E-03	2.23E-03	6.03E-04	1.36E-04	2.58E-05	4.22E-06	4.06E-07	2.07E-08
0.05	1.27E-01	1.19E-01	1.02E-01	7.41E-02	4.62E-02	2.67E-02	1.48E-02	6.89E-03	2.40E-03	6.60E-04	1.46E-04	2.76E-05	4.67E-06	4.79E-07	2.62E-08
0.075	1.27E-01	1.20E-01	1.06E-01	7.96E-02	5.08E-02	2.95E-02	1.61E-02	7.52E-03	2.67E-03	7.11E-04	1.55E-04	2.87E-05	4.32E-06	4.47E-07	2.78E-08
0.1	1.28E-01	1.22E-01	1.10E-01	8.66E-02	5.72E-02	3.33E-02	1.78E-02	8.32E-03	3.09E-03	8.89E-04	1.95E-04	3.46E-05	4.73E-06	4.15E-07	2.43E-08
0.15	1.29E-01	1.25E-01	1.18E-01	1.00E-01	7.23E-02	4.40E-02	2.37E-02	1.13E-02	4.35E-03	1.32E-03	2.92E-04	4.75E-05	5.40E-06	3.50E-07	1.18E-08
0.2	1.29E-01	1.26E-01	1.21E-01	1.08E-01	8.36E-02	5.38E-02	2.95E-02	1.42E-02	5.67E-03	1.63E-03	3.33E-04	4.79E-05	4.76E-06	2.96E-07	1.09E-08
0.25	1.29E-01	1.26E-01	1.21E-01	1.08E-01	8.28E-02	5.25E-02	2.83E-02	1.34E-02	5.40E-03	1.64E-03	3.29E-04	4.22E-05	3.48E-06	2.15E-07	9.62E-09
0.3	1.28E-01	1.25E-01	1.20E-01	1.05E-01	7.85E-02	4.87E-02	2.56E-02	1.19E-02	4.74E-03	1.41E-03	2.68E-04	3.24E-05	2.56E-06	1.62E-07	8.30E-09
0.4	1.28E-01	1.23E-01	1.14E-01	9.34E-02	6.47E-02	3.75E-02	1.87E-02	8.25E-03	3.09E-03	8.65E-04	1.71E-04	2.15E-05	1.96E-06	1.65E-07	9.89E-09
0.5	1.28E-01	1.23E-01	1.11E-01	8.75E-02	5.67E-02	3.07E-02	1.44E-02	6.15E-03	2.27E-03	6.36E-04	1.28E-04	1.86E-05	1.96E-06	1.92E-07	1.41E-08
0.75	1.28E-01	1.17E-01	9.62E-02	6.45E-02	3.58E-02	1.74E-02	7.74E-03	3.11E-03	1.04E-03	2.68E-04	5.23E-05	8.82E-06	1.36E-06	1.48E-07	9.34E-09
1	1.30E-01	1.11E-01	8.51E-02	5.29E-02	2.78E-02	1.33E-02	5.89E-03	2.35E-03	7.27E-04	1.65E-04	3.47E-05	6.80E-06	1.12E-06	1.27E-07	8.87E-09
1.5	1.16E-01	8.44E-02	5.26E-02	2.83E-02	1.41E-02	6.74E-03	2.97E-03	1.05E-03	2.63E-04	5.54E-05	1.11E-05	1.84E-06	2.25E-07	1.96E-08	1.11E-09
2	1.01E-01	6.65E-02	3.81E-02	1.98E-02	9.97E-03	4.91E-03	2.13E-03	6.95E-04	1.39E-04	2.53E-05	4.31E-06	5.15E-07	4.46E-08	2.91E-09	1.27E-10
3	6.93E-02	3.99E-02	2.14E-02	1.12E-02	5.89E-03	2.93E-03	1.20E-03	3.35E-04	5.57E-05	6.63E-06	6.55E-07	5.09E-08	3.05E-09	1.18E-10	3.02E-12
4	5.43E-02	2.99E-02	1.59E-02	8.48E-03	4.50E-03	2.17E-03	8.21E-04	1.93E-04	2.52E-05	2.31E-06	1.63E-07	8.73E-09	3.35E-10	8.69E-12	1.28E-13
5	4.18E-02	2.27E-02	1.22E-02	6.71E-03	3.62E-03	1.71E-03	6.13E-04	1.31E-04	1.52E-05	1.24E-06	7.39E-08	3.23E-09	1.06E-10	2.23E-12	2.39E-14
7.5	2.60E-02	1.45E-02	8.26E-03	4.78E-03	2.61E-03	1.20E-03	3.89E-04	7.26E-05	7.53E-06	5.52E-07	2.85E-08	1.08E-09	3.09E-11	5.84E-13	5.69E-15
10	1.75E-02	1.02E-02	5.99E-03	3.48E-03	1.82E-03	7.60E-04	2.09E-04	3.31E-05	2.88E-06	1.96E-07	8.19E-09	2.65E-10	6.53E-12	1.04E-13	8.33E-16

Table 2.5.2-35 Control Point Hazard Curves for PBIRS

Period (sec)	Annual Frequency of Exceedance (1/yr)														
	Acceleration (g)														
	1.00E-04	2.34E-04	5.49E-04	1.29E-03	3.01E-03	7.06E-03	1.65E-02	3.87E-02	9.07E-02	2.13E-01	4.98E-01	1.17E+00	2.73E+00	6.40E+00	1.50E+01
0.01	1.18E-01	9.90E-02	6.85E-02	4.03E-02	2.20E-02	1.14E-02	4.91E-03	1.72E-03	5.33E-04	1.48E-04	3.29E-05	4.72E-06	3.71E-07	1.76E-08	6.74E-10
0.02	1.17E-01	9.79E-02	6.83E-02	4.12E-02	2.34E-02	1.26E-02	5.86E-03	2.28E-03	7.82E-04	2.43E-04	6.19E-05	1.07E-05	1.05E-06	6.05E-08	2.74E-09
0.025	1.18E-01	9.96E-02	7.06E-02	4.32E-02	2.48E-02	1.36E-02	6.58E-03	2.65E-03	9.39E-04	3.04E-04	8.18E-05	1.49E-05	1.61E-06	1.03E-07	5.06E-09
0.03	1.19E-01	1.01E-01	7.29E-02	4.51E-02	2.61E-02	1.45E-02	7.17E-03	2.96E-03	1.06E-03	3.52E-04	9.76E-05	1.89E-05	2.10E-06	1.40E-07	7.27E-09
0.04	1.20E-01	1.05E-01	7.82E-02	4.96E-02	2.89E-02	1.62E-02	8.28E-03	3.52E-03	1.28E-03	4.27E-04	1.22E-04	2.53E-05	3.15E-06	2.13E-07	1.16E-08
0.05	1.21E-01	1.09E-01	8.32E-02	5.38E-02	3.15E-02	1.77E-02	9.10E-03	3.91E-03	1.42E-03	4.67E-04	1.32E-04	2.79E-05	3.63E-06	2.53E-07	1.39E-08
0.075	1.23E-01	1.14E-01	9.19E-02	6.16E-02	3.63E-02	2.02E-02	1.04E-02	4.48E-03	1.58E-03	5.07E-04	1.46E-04	3.19E-05	4.16E-06	3.01E-07	1.57E-08
0.1	1.25E-01	1.18E-01	9.95E-02	6.93E-02	4.10E-02	2.25E-02	1.14E-02	4.85E-03	1.68E-03	5.18E-04	1.43E-04	3.09E-05	4.13E-06	2.87E-07	1.42E-08
0.15	1.25E-01	1.21E-01	1.06E-01	7.73E-02	4.62E-02	2.46E-02	1.19E-02	4.83E-03	1.60E-03	4.34E-04	1.07E-04	2.10E-05	2.46E-06	1.58E-07	6.85E-09
0.2	1.26E-01	1.22E-01	1.08E-01	7.95E-02	4.74E-02	2.45E-02	1.14E-02	4.56E-03	1.41E-03	3.49E-04	7.56E-05	1.31E-05	1.45E-06	9.29E-08	3.71E-09
0.25	1.26E-01	1.22E-01	1.09E-01	8.00E-02	4.73E-02	2.41E-02	1.10E-02	4.34E-03	1.32E-03	3.04E-04	6.13E-05	1.01E-05	1.09E-06	7.09E-08	2.76E-09
0.3	1.26E-01	1.22E-01	1.09E-01	8.00E-02	4.68E-02	2.33E-02	1.04E-02	3.92E-03	1.17E-03	2.57E-04	4.86E-05	7.63E-06	7.89E-07	4.94E-08	1.85E-09
0.4	1.26E-01	1.22E-01	1.06E-01	7.56E-02	4.29E-02	2.08E-02	9.03E-03	3.39E-03	9.73E-04	2.06E-04	3.65E-05	5.38E-06	5.42E-07	3.27E-08	1.16E-09
0.5	1.26E-01	1.21E-01	1.05E-01	7.47E-02	4.24E-02	2.06E-02	8.90E-03	3.36E-03	9.79E-04	1.98E-04	3.37E-05	4.77E-06	4.69E-07	2.90E-08	1.06E-09
0.75	1.26E-01	1.15E-01	9.14E-02	5.79E-02	3.05E-02	1.43E-02	6.20E-03	2.33E-03	6.40E-04	1.20E-04	1.77E-05	2.25E-06	2.05E-07	1.19E-08	4.21E-10
1	1.27E-01	1.09E-01	8.14E-02	4.92E-02	2.54E-02	1.20E-02	5.26E-03	1.99E-03	5.43E-04	9.79E-05	1.33E-05	1.51E-06	1.35E-07	7.57E-09	2.64E-10
1.5	1.14E-01	8.27E-02	5.10E-02	2.72E-02	1.35E-02	6.46E-03	2.81E-03	9.68E-04	2.16E-04	3.12E-05	3.58E-06	3.36E-07	2.52E-08	1.19E-09	3.56E-11
2	9.90E-02	6.52E-02	3.72E-02	1.93E-02	9.70E-03	4.76E-03	2.05E-03	6.51E-04	1.21E-04	1.41E-05	1.39E-06	1.17E-07	6.87E-09	2.67E-10	6.47E-12
3	6.85E-02	3.94E-02	2.11E-02	1.10E-02	5.81E-03	2.90E-03	1.17E-03	3.23E-04	4.83E-05	4.41E-06	3.51E-07	2.21E-08	9.66E-10	2.74E-11	4.35E-13
4	5.34E-02	2.94E-02	1.56E-02	8.34E-03	4.45E-03	2.16E-03	8.06E-04	1.85E-04	2.21E-05	1.72E-06	1.07E-07	5.11E-09	1.81E-10	4.08E-12	4.73E-14
5	4.09E-02	2.22E-02	1.19E-02	6.59E-03	3.57E-03	1.69E-03	6.04E-04	1.27E-04	1.38E-05	1.00E-06	5.51E-08	2.34E-09	7.42E-11	1.48E-12	1.45E-14
7.5	2.55E-02	1.42E-02	8.14E-03	4.73E-03	2.59E-03	1.19E-03	3.85E-04	7.29E-05	7.12E-06	4.85E-07	2.38E-08	8.83E-10	2.48E-11	4.47E-13	4.07E-15
10	1.73E-02	1.00E-02	5.94E-03	3.47E-03	1.82E-03	7.60E-04	2.10E-04	3.23E-05	2.73E-06	1.74E-07	7.70E-09	2.48E-10	6.11E-12	9.74E-14	7.74E-16

Table 2.5.2-36 Control Point Hazard Curves for FIRS

Period (sec)	Annual Frequency of Exceedance (1/yr)														
	Acceleration (g)														
	1.00E-04	2.34E-04	5.49E-04	1.29E-03	3.01E-03	7.06E-03	1.65E-02	3.87E-02	9.07E-02	2.13E-01	4.98E-01	1.17E+00	2.73E+00	6.40E+00	1.50E+01
0.01	1.17E-01	9.63E-02	6.52E-02	3.80E-02	2.07E-02	1.06E-02	4.43E-03	1.55E-03	4.77E-04	1.37E-04	3.06E-05	4.30E-06	3.35E-07	1.57E-08	5.99E-10
0.02	1.16E-01	9.47E-02	6.46E-02	3.85E-02	2.17E-02	1.16E-02	5.34E-03	2.03E-03	6.94E-04	2.20E-04	5.54E-05	9.25E-06	8.80E-07	4.94E-08	2.20E-09
0.025	1.17E-01	9.69E-02	6.72E-02	4.07E-02	2.33E-02	1.27E-02	6.03E-03	2.37E-03	8.23E-04	2.70E-04	7.21E-05	1.30E-05	1.38E-06	8.38E-08	4.03E-09
0.03	1.18E-01	9.87E-02	6.95E-02	4.25E-02	2.45E-02	1.35E-02	6.59E-03	2.67E-03	9.38E-04	3.08E-04	8.37E-05	1.58E-05	1.81E-06	1.17E-07	5.97E-09
0.04	1.19E-01	1.02E-01	7.46E-02	4.66E-02	2.71E-02	1.51E-02	7.61E-03	3.21E-03	1.15E-03	3.84E-04	1.08E-04	2.20E-05	2.74E-06	1.84E-07	1.00E-08
0.05	1.21E-01	1.06E-01	7.98E-02	5.09E-02	2.96E-02	1.65E-02	8.34E-03	3.53E-03	1.27E-03	4.24E-04	1.20E-04	2.51E-05	3.28E-06	2.32E-07	1.29E-08
0.075	1.23E-01	1.12E-01	8.94E-02	5.93E-02	3.48E-02	1.93E-02	9.78E-03	4.15E-03	1.45E-03	4.65E-04	1.34E-04	2.89E-05	3.97E-06	2.86E-07	1.51E-08
0.1	1.24E-01	1.17E-01	9.78E-02	6.75E-02	3.97E-02	2.17E-02	1.09E-02	4.57E-03	1.59E-03	4.82E-04	1.34E-04	2.85E-05	3.92E-06	2.82E-07	1.41E-08
0.15	1.25E-01	1.20E-01	1.04E-01	7.45E-02	4.38E-02	2.31E-02	1.10E-02	4.45E-03	1.44E-03	3.96E-04	1.00E-04	1.98E-05	2.41E-06	1.57E-07	6.86E-09
0.2	1.25E-01	1.21E-01	1.06E-01	7.62E-02	4.44E-02	2.26E-02	1.03E-02	3.96E-03	1.21E-03	3.02E-04	6.79E-05	1.22E-05	1.41E-06	9.23E-08	3.68E-09
0.25	1.25E-01	1.21E-01	1.06E-01	7.58E-02	4.35E-02	2.17E-02	9.69E-03	3.66E-03	1.11E-03	2.58E-04	5.39E-05	9.42E-06	1.06E-06	7.00E-08	2.73E-09
0.3	1.25E-01	1.21E-01	1.06E-01	7.56E-02	4.30E-02	2.10E-02	9.19E-03	3.44E-03	1.01E-03	2.22E-04	4.36E-05	7.21E-06	7.69E-07	4.87E-08	1.82E-09
0.4	1.25E-01	1.21E-01	1.05E-01	7.39E-02	4.14E-02	1.99E-02	8.60E-03	3.24E-03	9.28E-04	1.92E-04	3.43E-05	5.11E-06	5.20E-07	3.11E-08	1.10E-09
0.5	1.26E-01	1.21E-01	1.05E-01	7.44E-02	4.22E-02	2.04E-02	8.82E-03	3.29E-03	9.36E-04	1.90E-04	3.26E-05	4.61E-06	4.51E-07	2.74E-08	9.92E-10
0.75	1.26E-01	1.16E-01	9.15E-02	5.80E-02	3.06E-02	1.44E-02	6.22E-03	2.30E-03	6.18E-04	1.13E-04	1.70E-05	2.16E-06	1.98E-07	1.15E-08	4.09E-10
1	1.27E-01	1.10E-01	8.16E-02	4.94E-02	2.55E-02	1.21E-02	5.26E-03	1.96E-03	5.34E-04	9.47E-05	1.31E-05	1.46E-06	1.28E-07	7.11E-09	2.46E-10
1.5	1.13E-01	8.24E-02	5.06E-02	2.70E-02	1.34E-02	6.43E-03	2.82E-03	9.56E-04	2.15E-04	2.99E-05	3.53E-06	3.27E-07	2.45E-08	1.15E-09	3.40E-11
2	9.84E-02	6.48E-02	3.68E-02	1.90E-02	9.58E-03	4.71E-03	2.04E-03	6.46E-04	1.22E-04	1.44E-05	1.39E-06	1.12E-07	6.40E-09	2.43E-10	5.68E-12
3	6.84E-02	3.94E-02	2.10E-02	1.10E-02	5.81E-03	2.89E-03	1.18E-03	3.23E-04	4.70E-05	4.47E-06	3.53E-07	2.18E-08	9.50E-10	2.68E-11	4.22E-13
4	5.32E-02	2.93E-02	1.56E-02	8.31E-03	4.44E-03	2.15E-03	8.11E-04	1.88E-04	2.23E-05	1.72E-06	1.08E-07	5.18E-09	1.84E-10	4.15E-12	4.83E-14
5	4.10E-02	2.22E-02	1.20E-02	6.61E-03	3.57E-03	1.69E-03	6.06E-04	1.28E-04	1.39E-05	1.00E-06	5.59E-08	2.38E-09	7.57E-11	1.51E-12	1.49E-14
7.5	2.56E-02	1.43E-02	8.15E-03	4.73E-03	2.59E-03	1.18E-03	3.84E-04	7.05E-05	7.14E-06	4.82E-07	2.33E-08	8.63E-10	2.41E-11	4.32E-13	3.89E-15
10	1.72E-02	1.00E-02	5.92E-03	3.45E-03	1.81E-03	7.51E-04	2.10E-04	3.16E-05	2.80E-06	1.77E-07	7.58E-09	2.44E-10	5.97E-12	9.43E-14	7.41E-16

Table 2.5.2-37 Control Point Horizontal UHRS at 10^{-4}, 10^{-5}, and 10^{-6} AFE

Frequency (Hz)	Horizontal Spectral Acceleration (g)								
	PBSRS			PBIRS			FIRS		
	10^{-4} AFE	10^{-5} AFE	10^{-6} AFE	10^{-4} AFE	10^{-5} AFE	10^{-6} AFE	10^{-4} AFE	10^{-5} AFE	10^{-6} AFE
100	0.33139	0.96090	2.33216	0.26537	0.83949	1.96102	0.25387	0.80864	1.89690
50	0.41134	1.28444	3.17956	0.36928	1.19703	2.77175	0.34577	1.12427	2.60984
40	0.47822	1.48739	3.67668	0.43729	1.35991	3.16759	0.40321	1.28876	3.01131
33.33333	0.52661	1.63527	4.05213	0.49011	1.49288	3.45257	0.44337	1.39734	3.28485
25	0.58228	1.82060	4.61220	0.55548	1.70438	3.92596	0.51991	1.61003	3.75447
20	0.60490	1.89672	4.86232	0.57984	1.78981	4.12650	0.54938	1.71408	4.00337
13.33333	0.62131	1.87384	4.73177	0.61519	1.89427	4.33818	0.58545	1.83915	4.27102
10	0.69225	1.98387	4.70764	0.60741	1.88006	4.29806	0.58471	1.82883	4.25092
6.666667	0.82290	2.14684	4.61932	0.51709	1.56694	3.61316	0.49856	1.53792	3.59560
5	0.84445	2.07858	4.40977	0.42627	1.29528	3.06442	0.39939	1.26069	3.04337
4	0.81579	1.90626	4.00044	0.38383	1.17236	2.80762	0.35576	1.13288	2.78670
3.333333	0.74059	1.73004	3.65132	0.34421	1.02988	2.50085	0.32278	0.99956	2.47279
2.5	0.62062	1.53216	3.44559	0.30330	0.88558	2.17723	0.29332	0.86414	2.14255
2	0.55537	1.47432	3.49658	0.29541	0.84552	2.07025	0.28995	0.83264	2.04188
1.333333	0.35532	1.09859	3.07672	0.23042	0.63034	1.55570	0.22459	0.61977	1.53518
1	0.27957	0.95413	2.85695	0.21030	0.55661	1.34962	0.20689	0.55231	1.33143
0.666667	0.15399	0.52276	1.49328	0.12731	0.33245	0.78801	0.12625	0.32883	0.78207
0.5	0.10684	0.33214	0.89438	0.09776	0.24118	0.55828	0.09817	0.24262	0.55692
0.333333	0.06872	0.18037	0.42624	0.06549	0.15888	0.35011	0.06498	0.15883	0.35120
0.25	0.05095	0.12608	0.27822	0.04953	0.11819	0.25112	0.04986	0.11847	0.25105
0.2	0.04306	0.10449	0.22659	0.04247	0.10080	0.21284	0.04253	0.10092	0.21254
0.133333	0.03293	0.08158	0.17514	0.03296	0.08014	0.16898	0.03250	0.08005	0.16877
0.1	0.02324	0.05881	0.12690	0.02315	0.05801	0.12384	0.02306	0.05801	0.12466

Table 2.5.2-38 Control Point Vertical UHRS at 10^{-4}, 10^{-5}, and 10^{-6} AFE

Frequency (Hz)	Vertical Spectral Acceleration (g)								
	PBSRS			PBIRS			FIRS		
	10^{-4} AFE	10^{-5} AFE	10^{-6} AFE	10^{-4} AFE	10^{-5} AFE	10^{-6} AFE	10^{-4} AFE	10^{-5} AFE	10^{-6} AFE
100	0.23464	0.66887	1.62213	0.21915	0.69079	1.64759	0.21662	0.69243	1.66024
50	0.29962	0.93346	2.35339	0.30926	1.00988	2.40647	0.29007	0.94606	2.25730
40	0.35104	1.11196	2.80947	0.36246	1.18158	2.80406	0.34079	1.12655	2.67497
33.33333	0.41398	1.30336	3.28982	0.41054	1.30585	3.07756	0.40226	1.30534	3.12074
25	0.46115	1.44175	3.70504	0.49466	1.52781	3.59302	0.47558	1.50554	3.57096
20	0.44158	1.38360	3.58925	0.46035	1.43970	3.40603	0.46498	1.47805	3.52188
13.33333	0.38365	1.17932	2.99158	0.42606	1.33594	3.12616	0.42247	1.35322	3.21495
10	0.39430	1.15259	2.77565	0.36882	1.18597	2.77485	0.35606	1.15643	2.72207
6.666667	0.43414	1.15425	2.51708	0.29172	0.90149	2.11494	0.30088	0.94974	2.25805
5	0.44816	1.12186	2.40284	0.26131	0.78959	1.90548	0.25639	0.79495	1.94131
4	0.47306	1.12484	2.37239	0.27911	0.83966	2.05548	0.26056	0.80758	2.01902
3.333333	0.46647	1.10800	2.34941	0.27290	0.80936	1.99475	0.25740	0.78214	1.95895
2.5	0.41671	1.05583	2.39469	0.26124	0.76183	1.89109	0.25223	0.74412	1.85468
2	0.36832	1.00654	2.41880	0.25492	0.72977	1.80383	0.27017	0.77858	1.93470
1.333333	0.23981	0.74277	2.06708	0.20063	0.55562	1.37712	0.21169	0.58710	1.45967
1	0.19681	0.66713	1.97323	0.18093	0.49234	1.20046	0.19437	0.52272	1.27508
0.666667	0.11267	0.37880	1.10975	0.11844	0.31474	0.75191	0.11735	0.31206	0.75135
0.5	0.07831	0.24352	0.66145	0.09123	0.22990	0.53490	0.09146	0.23169	0.53653
0.333333	0.05016	0.13370	0.31750	0.05575	0.13961	0.31025	0.05632	0.14091	0.31313
0.25	0.03122	0.07992	0.17785	0.03988	0.09675	0.20715	0.04287	0.10442	0.22448
0.2	0.02278	0.05755	0.12594	0.03019	0.07487	0.15960	0.03020	0.07471	0.15951
0.133333	0.01997	0.05109	0.11171	0.02535	0.06441	0.13827	0.02636	0.06660	0.14301
0.1	0.01715	0.04429	0.09637	0.02124	0.05489	0.11926	0.02122	0.05517	0.12029

Table 2.5.2-39 Horizontal PBSRS, PBIRS, and FIRS
(Sheet 1 of 5)

Frequency (Hz)	Horizontal Ground Motion Response Spectra (g)		
	FIRS*	PBIRS	PBSRS
0.10000	2.89E-02	2.90E-02	2.93E-02
0.10523	3.09E-02	3.10E-02	3.13E-02
0.11074	3.29E-02	3.30E-02	3.34E-02
0.11654	3.50E-02	3.51E-02	3.55E-02
0.12264	3.70E-02	3.71E-02	3.76E-02
0.12905	3.89E-02	3.91E-02	3.96E-02
0.13333	4.01E-02	4.03E-02	4.08E-02
0.13581	4.07E-02	4.09E-02	4.15E-02
0.14292	4.22E-02	4.23E-02	4.30E-02
0.15040	4.35E-02	4.37E-02	4.44E-02
0.15827	4.47E-02	4.49E-02	4.58E-02
0.16655	4.60E-02	4.61E-02	4.71E-02
0.17527	4.73E-02	4.74E-02	4.85E-02
0.18444	4.87E-02	4.87E-02	5.00E-02
0.19409	5.01E-02	5.01E-02	5.16E-02
0.20000	5.09E-02	5.09E-02	5.25E-02
0.20425	5.17E-02	5.16E-02	5.33E-02
0.21494	5.35E-02	5.33E-02	5.54E-02
0.22619	5.54E-02	5.52E-02	5.77E-02
0.23803	5.75E-02	5.73E-02	6.03E-02
0.25000	5.98E-02	5.96E-02	6.31E-02
0.25049	5.99E-02	5.97E-02	6.32E-02
0.26359	6.29E-02	6.27E-02	6.69E-02
0.27739	6.63E-02	6.61E-02	7.12E-02
0.29191	7.00E-02	6.98E-02	7.58E-02
0.30719	7.38E-02	7.36E-02	8.08E-02
0.32326	7.75E-02	7.75E-02	8.60E-02
0.33333	7.97E-02	7.98E-02	8.92E-02
0.34018	8.11E-02	8.14E-02	9.15E-02
0.35798	8.46E-02	8.51E-02	9.72E-02
0.37672	8.82E-02	8.89E-02	1.03E-01
0.39644	9.22E-02	9.28E-02	1.10E-01
0.41718	9.67E-02	9.72E-02	1.18E-01
0.43902	1.02E-01	1.02E-01	1.27E-01
0.46199	1.09E-01	1.09E-01	1.38E-01
0.48617	1.17E-01	1.16E-01	1.51E-01

Table 2.5.2-39 Horizontal PBSRS, PBIRS, and FIRS
(Sheet 2 of 5)

Frequency (Hz)	Horizontal Ground Motion Response Spectra (g)		
	FIRS*	PBIRS	PBSRS
0.50000	1.21E-01	1.21E-01	1.59E-01
0.51162	1.26E-01	1.25E-01	1.66E-01
0.53839	1.34E-01	1.33E-01	1.82E-01
0.56657	1.42E-01	1.42E-01	1.99E-01
0.59622	1.49E-01	1.51E-01	2.14E-01
0.62743	1.55E-01	1.58E-01	2.28E-01
0.66027	1.62E-01	1.64E-01	2.43E-01
0.66667	1.63E-01	1.65E-01	2.46E-01
0.69482	1.67E-01	1.67E-01	2.44E-01
0.73119	1.73E-01	1.73E-01	2.56E-01
0.76945	1.80E-01	1.81E-01	2.70E-01
0.80972	1.88E-01	1.90E-01	2.88E-01
0.85210	1.99E-01	2.03E-01	3.09E-01
0.89670	2.16E-01	2.21E-01	3.39E-01
0.94363	2.41E-01	2.44E-01	3.80E-01
0.99301	2.68E-01	2.70E-01	4.23E-01
1.00000	2.70E-01	2.72E-01	4.26E-01
1.04500	2.86E-01	2.89E-01	4.75E-01
1.09970	2.89E-01	2.93E-01	4.85E-01
1.15720	2.87E-01	2.91E-01	4.86E-01
1.21780	2.87E-01	2.91E-01	4.91E-01
1.28150	2.93E-01	2.98E-01	5.06E-01
1.33333	3.04E-01	3.09E-01	5.26E-01
1.34860	3.07E-01	3.12E-01	5.32E-01
1.41920	3.25E-01	3.30E-01	5.63E-01
1.49350	3.41E-01	3.45E-01	5.89E-01
1.57160	3.49E-01	3.54E-01	6.04E-01
1.65390	3.53E-01	3.61E-01	6.12E-01
1.74040	3.62E-01	3.71E-01	6.24E-01
1.83150	3.77E-01	3.86E-01	6.45E-01
1.92740	3.93E-01	4.01E-01	6.68E-01
2.00000	4.05E-01	4.11E-01	7.28E-01
2.02820	4.08E-01	4.14E-01	7.34E-01
2.13440	4.14E-01	4.21E-01	7.48E-01
2.24610	4.16E-01	4.24E-01	7.57E-01
2.36360	4.15E-01	4.25E-01	7.61E-01

Table 2.5.2-39 Horizontal PBSRS, PBIRS, and FIRS
(Sheet 3 of 5)

Frequency (Hz)	Horizontal Ground Motion Response Spectra (g)		
	FIRS*	PBIRS	PBSRS
2.48740	4.16E-01	4.27E-01	7.65E-01
2.50000	4.18E-01	4.29E-01	7.67E-01
2.61750	4.30E-01	4.41E-01	7.85E-01
2.75450	4.47E-01	4.61E-01	8.20E-01
2.89870	4.66E-01	4.81E-01	8.48E-01
3.05040	4.75E-01	4.90E-01	8.62E-01
3.21000	4.76E-01	4.92E-01	8.69E-01
3.33333	4.78E-01	4.96E-01	8.76E-01
3.37800	4.80E-01	4.98E-01	8.79E-01
3.55480	4.91E-01	5.10E-01	8.96E-01
3.74090	5.08E-01	5.29E-01	9.22E-01
3.93670	5.34E-01	5.57E-01	9.58E-01
4.00000	5.39E-01	5.63E-01	9.65E-01
4.14270	5.51E-01	5.75E-01	9.84E-01
4.35950	5.57E-01	5.80E-01	9.94E-01
4.58770	5.64E-01	5.86E-01	1.00E+00
4.82780	5.84E-01	6.06E-01	1.03E+00
5.00000	6.01E-01	6.22E-01	1.04E+00
5.08040	6.09E-01	6.30E-01	1.05E+00
5.34630	6.27E-01	6.46E-01	1.05E+00
5.62610	6.45E-01	6.64E-01	1.05E+00
5.92060	6.77E-01	6.95E-01	1.06E+00
6.23050	7.08E-01	7.25E-01	1.06E+00
6.55650	7.29E-01	7.46E-01	1.06E+00
6.66667	7.37E-01	7.53E-01	1.06E+00
6.89970	7.51E-01	7.68E-01	1.06E+00
7.26080	7.67E-01	7.85E-01	1.04E+00
7.64080	7.73E-01	7.91E-01	1.02E+00
8.04070	7.94E-01	8.12E-01	1.01E+00
8.46150	8.19E-01	8.38E-01	1.00E+00
8.90430	8.44E-01	8.66E-01	9.93E-01
9.37040	8.66E-01	8.91E-01	9.83E-01
9.86080	8.74E-01	9.00E-01	9.69E-01
10.00000	8.74E-01	9.00E-01	9.64E-01
10.37700	8.73E-01	8.98E-01	9.51E-01
10.92000	8.68E-01	8.92E-01	9.36E-01

Table 2.5.2-39 Horizontal PBSRS, PBIRS, and FIRS
(Sheet 4 of 5)

Frequency (Hz)	Horizontal Ground Motion Response Spectra (g)		
	FIRS*	PBIRS	PBSRS
11.49100	8.63E-01	8.88E-01	9.24E-01
12.09300	8.64E-01	8.92E-01	9.16E-01
12.72600	8.71E-01	9.01E-01	9.09E-01
13.33333	8.78E-01	9.08E-01	9.02E-01
13.39200	8.78E-01	9.08E-01	9.02E-01
14.09300	8.80E-01	9.09E-01	9.00E-01
14.83000	8.75E-01	9.04E-01	8.98E-01
15.60600	8.64E-01	8.93E-01	8.95E-01
16.42300	8.51E-01	8.79E-01	8.91E-01
17.28300	8.40E-01	8.69E-01	8.90E-01
18.18700	8.32E-01	8.64E-01	8.94E-01
19.13900	8.26E-01	8.62E-01	9.01E-01
20.00000	8.19E-01	8.57E-01	9.06E-01
20.14100	8.18E-01	8.57E-01	9.05E-01
21.19500	8.10E-01	8.48E-01	8.98E-01
22.30400	7.98E-01	8.38E-01	8.90E-01
23.47100	7.87E-01	8.30E-01	8.82E-01
24.70000	7.74E-01	8.21E-01	8.73E-01
25.00000	7.71E-01	8.17E-01	8.70E-01
25.99200	7.60E-01	8.08E-01	8.61E-01
27.35300	7.41E-01	7.87E-01	8.45E-01
28.78400	7.18E-01	7.64E-01	8.26E-01
30.29100	6.97E-01	7.44E-01	8.09E-01
31.87600	6.80E-01	7.29E-01	7.94E-01
33.33333	6.66E-01	7.17E-01	7.82E-01
33.54400	6.65E-01	7.15E-01	7.80E-01
35.30000	6.51E-01	6.99E-01	7.61E-01
37.14700	6.36E-01	6.79E-01	7.41E-01
39.09200	6.20E-01	6.59E-01	7.20E-01
40.00000	6.13E-01	6.50E-01	7.11E-01
41.13800	6.01E-01	6.39E-01	6.97E-01
43.29100	5.81E-01	6.18E-01	6.72E-01
45.55600	5.62E-01	5.99E-01	6.50E-01
47.94000	5.45E-01	5.82E-01	6.29E-01
50.00000	5.33E-01	5.68E-01	6.14E-01
50.44900	5.29E-01	5.64E-01	6.11E-01

Table 2.5.2-39 Horizontal PBSRS, PBIRS, and FIRS
(Sheet 5 of 5)

Frequency (Hz)	Horizontal Ground Motion Response Spectra (g)		
	FIRS*	PBIRS	PBSRS
53.09000	5.11E-01	5.43E-01	5.95E-01
55.86800	4.94E-01	5.25E-01	5.81E-01
58.79200	4.79E-01	5.09E-01	5.68E-01
61.86900	4.65E-01	4.94E-01	5.56E-01
65.10700	4.53E-01	4.81E-01	5.45E-01
68.51500	4.42E-01	4.69E-01	5.34E-01
72.10100	4.32E-01	4.58E-01	5.24E-01
75.87400	4.23E-01	4.47E-01	5.14E-01
79.84500	4.15E-01	4.37E-01	5.05E-01
84.02400	4.07E-01	4.27E-01	4.96E-01
88.42100	4.00E-01	4.18E-01	4.87E-01
93.04900	3.93E-01	4.10E-01	4.78E-01
97.91900	3.87E-01	4.03E-01	4.69E-01
100.00000	3.85E-01	4.00E-01	4.66E-01

* The FIRS is before ISG-017 check

Table 2.5.2-40 Median Vertical to Horizontal (V/H) Spectral Ratio
(Sheet 1 of 5)

Frequency (Hz)	Median V/H ratios		
	FIRS	PBIRS	PBSRS
0.1	0.881708	0.86355	0.706005
0.10523	0.863929	0.845016	0.682046
0.11074	0.846488	0.826859	0.658874
0.11654	0.829395	0.809088	0.636484
0.12264	0.812657	0.791709	0.614868
0.12905	0.79628	0.774728	0.594014
0.133333	0.785958	0.764036	0.581022
0.13581	0.781591	0.759852	0.578289
0.14292	0.769611	0.748371	0.570778
0.1504	0.757817	0.737067	0.563367
0.15827	0.746207	0.725935	0.556053
0.16655	0.734777	0.714974	0.548835
0.17527	0.723513	0.70417	0.541706
0.18444	0.712429	0.693537	0.534675
0.19409	0.701515	0.683065	0.527734
0.2	0.695174	0.676979	0.523694
0.20425	0.703764	0.685462	0.531147
0.21494	0.725046	0.706488	0.549672
0.22619	0.746973	0.72816	0.568843
0.23803	0.769565	0.750499	0.588686
0.25	0.791935	0.772627	0.60842
0.25049	0.79228	0.772981	0.608838
0.26359	0.801312	0.782274	0.619835
0.27739	0.810456	0.791688	0.631043
0.29191	0.819703	0.801213	0.642452
0.30719	0.829055	0.810854	0.654067
0.32326	0.838509	0.820604	0.665885
0.333333	0.84425	0.826529	0.673099
0.34018	0.84625	0.82852	0.674982
0.35798	0.851287	0.833535	0.679727
0.37672	0.856357	0.838584	0.684508
0.39644	0.861457	0.843662	0.689322
0.41718	0.866584	0.848769	0.694168
0.43902	0.871746	0.853909	0.69905
0.46199	0.876935	0.859078	0.703964
0.48617	0.882156	0.86428	0.708915

Table 2.5.2-40 Median Vertical to Horizontal (V/H) Spectral Ratio
(Sheet 2 of 5)

Frequency (Hz)	Median V/H ratios		
	FIRS	PBIRS	PBSRS
0.5	0.88504	0.867154	0.711652
0.51162	0.884507	0.866577	0.710747
0.53839	0.883325	0.865298	0.708742
0.56657	0.882144	0.864021	0.706742
0.59622	0.880965	0.862746	0.704749
0.62743	0.879787	0.861472	0.70276
0.66027	0.87861	0.860201	0.700777
0.666667	0.878388	0.85996	0.700403
0.69482	0.877486	0.858774	0.697056
0.73119	0.876374	0.857314	0.69295
0.76945	0.875264	0.855856	0.688869
0.80972	0.874155	0.8544	0.684812
0.8521	0.873047	0.852947	0.680778
0.8967	0.871941	0.851496	0.676768
0.94363	0.870836	0.850048	0.672781
0.99301	0.869733	0.848602	0.668819
1	0.869581	0.848403	0.668276
1.045	0.869887	0.848489	0.666722
1.0997	0.870241	0.848587	0.664925
1.1572	0.870594	0.848686	0.663135
1.2178	0.870949	0.848784	0.661348
1.2815	0.871303	0.848883	0.659567
1.333333	0.871579	0.848959	0.658185
1.3486	0.871172	0.848563	0.657878
1.4192	0.869349	0.846787	0.656504
1.4935	0.867529	0.845016	0.655132
1.5716	0.865716	0.84325	0.653765
1.6539	0.863904	0.841485	0.652399
1.7404	0.862098	0.839726	0.651038
1.8315	0.860294	0.837969	0.649678
1.9274	0.858494	0.836216	0.648321
2	0.857191	0.834948	0.647339
2.0282	0.854249	0.832149	0.645679
2.1344	0.843607	0.822027	0.639666
2.2461	0.833103	0.812034	0.633712
2.3636	0.822734	0.802166	0.627816

Table 2.5.2-40 Median Vertical to Horizontal (V/H) Spectral Ratio
(Sheet 3 of 5)

Frequency (Hz)	Median V/H ratios		
	FIRS	PBIRS	PBSRS
2.4874	0.812482	0.792406	0.621967
2.5	0.811475	0.791447	0.621392
2.6175	0.801327	0.781696	0.614847
2.7545	0.790204	0.771005	0.607659
2.8987	0.779234	0.760459	0.600553
3.0504	0.768419	0.750061	0.593532
3.21	0.757757	0.739807	0.586595
3.333333	0.74997	0.732317	0.581519
3.378	0.745094	0.72762	0.578292
3.5548	0.726697	0.709899	0.566089
3.7409	0.70875	0.692605	0.554141
3.9367	0.69125	0.675736	0.542448
4	0.685867	0.670547	0.538843
4.1427	0.674214	0.659485	0.532542
4.3595	0.657609	0.643715	0.523505
4.5877	0.641409	0.628317	0.514619
4.8278	0.625612	0.613291	0.505886
5	0.614984	0.603176	0.499971
5.0804	0.612533	0.600946	0.499519
5.3463	0.60476	0.593869	0.498074
5.6261	0.597086	0.586875	0.496634
5.9206	0.589508	0.579963	0.495198
6.2305	0.582026	0.573132	0.493766
6.5565	0.574642	0.566384	0.492338
6.666667	0.57225	0.564197	0.491873
6.8997	0.573404	0.565514	0.49454
7.2608	0.575122	0.567476	0.498526
7.6408	0.576845	0.569445	0.502545
8.0407	0.578573	0.57142	0.506596
8.4615	0.580306	0.573402	0.51068
8.9043	0.582044	0.575391	0.514796
9.3704	0.583788	0.577388	0.518947
9.8608	0.585537	0.579391	0.52313
10	0.586019	0.579942	0.524286
10.377	0.596816	0.590625	0.533914
10.92	0.612025	0.605671	0.547476

Table 2.5.2-40 Median Vertical to Horizontal (V/H) Spectral Ratio
(Sheet 4 of 5)

Frequency (Hz)	Median V/H ratios		
	FIRS	PBIRS	PBSRS
11.491	0.62761	0.62109	0.561372
12.093	0.643622	0.636931	0.575647
12.726	0.660029	0.653162	0.590273
13.33333	0.675386	0.668354	0.603963
13.392	0.676536	0.669476	0.604833
14.093	0.690048	0.682652	0.615036
14.83	0.703817	0.696075	0.625402
15.606	0.717869	0.70977	0.635949
16.423	0.732208	0.723741	0.646678
17.283	0.746837	0.73799	0.657592
18.187	0.761742	0.752504	0.668677
19.139	0.776955	0.767314	0.679957
20	0.79032	0.780321	0.689839
20.141	0.792502	0.782438	0.691398
21.195	0.80852	0.797982	0.702822
22.304	0.824859	0.813833	0.714433
23.471	0.841528	0.829998	0.726236
24.7	0.858547	0.846496	0.738243
25	0.862623	0.850446	0.741112
25.992	0.862399	0.850091	0.739666
27.353	0.862105	0.849625	0.737775
28.784	0.861812	0.849159	0.73589
30.291	0.861519	0.848694	0.734008
31.876	0.861225	0.848228	0.732132
33.33333	0.860968	0.847821	0.730492
33.544	0.860527	0.847377	0.730033
35.3	0.856961	0.843792	0.726332
37.147	0.853411	0.840223	0.722651
39.092	0.849873	0.836668	0.718986
40	0.848286	0.835073	0.717343
41.138	0.846351	0.833128	0.715341
43.291	0.842845	0.829604	0.711715
45.556	0.839353	0.826096	0.708109
47.94	0.835876	0.822601	0.704519
50	0.833018	0.81973	0.701573
50.449	0.832593	0.819299	0.701095

Table 2.5.2-40 Median Vertical to Horizontal (V/H) Spectral Ratio
(Sheet 5 of 5)

Frequency (Hz)	Median V/H ratios		
	FIRS	PBIRS	PBSRS
53.09	0.83017	0.81684	0.698374
55.868	0.827756	0.81439	0.695665
58.792	0.825348	0.811947	0.692965
61.869	0.822946	0.809511	0.690276
65.107	0.820552	0.807082	0.687598
68.515	0.818165	0.80466	0.68493
72.101	0.815785	0.802246	0.682272
75.874	0.813412	0.79984	0.679625
79.845	0.811045	0.79744	0.676988
84.024	0.808686	0.795047	0.674361
88.421	0.806333	0.792663	0.671745
93.049	0.803987	0.790284	0.669138
97.919	0.801648	0.787913	0.666542
100	0.800686	0.786938	0.665474

Table 2.5.2-41 Vertical Design Spectra
(Sheet 1 of 5)

Frequency (Hz)	Vertical Spectra (g)		
	FIRS*	PBIRS	PBSRS
0.1	2.73E-02	2.72E-02	2.20E-02
0.10523	2.85E-02	2.83E-02	2.27E-02
0.11074	2.97E-02	2.93E-02	2.34E-02
0.11654	3.08E-02	3.02E-02	2.41E-02
0.12264	3.18E-02	3.10E-02	2.47E-02
0.12905	3.27E-02	3.17E-02	2.52E-02
0.133333	3.32E-02	3.21E-02	2.54E-02
0.13581	3.35E-02	3.24E-02	2.56E-02
0.14292	3.42E-02	3.33E-02	2.62E-02
0.1504	3.47E-02	3.40E-02	2.66E-02
0.15827	3.52E-02	3.46E-02	2.70E-02
0.16655	3.56E-02	3.52E-02	2.73E-02
0.17527	3.61E-02	3.58E-02	2.77E-02
0.18444	3.66E-02	3.64E-02	2.81E-02
0.19409	3.71E-02	3.71E-02	2.85E-02
0.2	3.74E-02	3.75E-02	2.87E-02
0.20425	3.86E-02	3.83E-02	2.95E-02
0.21494	4.16E-02	4.06E-02	3.17E-02
0.22619	4.48E-02	4.30E-02	3.41E-02
0.23803	4.85E-02	4.57E-02	3.68E-02
0.25	5.24E-02	4.86E-02	3.97E-02
0.25049	5.25E-02	4.87E-02	3.99E-02
0.26359	5.52E-02	5.18E-02	4.34E-02
0.27739	5.83E-02	5.53E-02	4.75E-02
0.29191	6.17E-02	5.91E-02	5.20E-02
0.30719	6.51E-02	6.31E-02	5.71E-02
0.32326	6.84E-02	6.72E-02	6.25E-02
0.333333	7.04E-02	6.97E-02	6.59E-02
0.34018	7.19E-02	7.14E-02	6.76E-02
0.35798	7.57E-02	7.54E-02	7.18E-02
0.37672	7.96E-02	7.96E-02	7.63E-02
0.39644	8.40E-02	8.40E-02	8.12E-02
0.41718	8.90E-02	8.89E-02	8.69E-02
0.43902	9.48E-02	9.46E-02	9.35E-02
0.46199	1.02E-01	1.02E-01	1.01E-01
0.48617	1.10E-01	1.10E-01	1.11E-01

Table 2.5.2-41 Vertical Design Spectra
(Sheet 2 of 5)

Frequency (Hz)	Vertical Spectra (g)		
	FIRS*	PBIRS	PBSRS
0.5	1.15E-01	1.15E-01	1.16E-01
0.51162	1.19E-01	1.18E-01	1.21E-01
0.53839	1.28E-01	1.26E-01	1.33E-01
0.56657	1.35E-01	1.35E-01	1.45E-01
0.59622	1.41E-01	1.43E-01	1.56E-01
0.62743	1.47E-01	1.49E-01	1.66E-01
0.66027	1.53E-01	1.54E-01	1.76E-01
0.666667	1.54E-01	1.55E-01	1.78E-01
0.69482	1.57E-01	1.56E-01	1.76E-01
0.73119	1.64E-01	1.61E-01	1.86E-01
0.76945	1.70E-01	1.67E-01	1.96E-01
0.80972	1.78E-01	1.74E-01	2.09E-01
0.8521	1.88E-01	1.84E-01	2.25E-01
0.8967	2.05E-01	1.99E-01	2.47E-01
0.94363	2.29E-01	2.19E-01	2.77E-01
0.99301	2.55E-01	2.39E-01	3.09E-01
1	2.57E-01	2.41E-01	3.12E-01
1.045	2.73E-01	2.57E-01	3.45E-01
1.0997	2.75E-01	2.60E-01	3.47E-01
1.1572	2.72E-01	2.57E-01	3.43E-01
1.2178	2.72E-01	2.57E-01	3.41E-01
1.2815	2.78E-01	2.63E-01	3.46E-01
1.333333	2.87E-01	2.72E-01	3.55E-01
1.3486	2.90E-01	2.75E-01	3.59E-01
1.4192	3.07E-01	2.89E-01	3.81E-01
1.4935	3.22E-01	3.02E-01	3.99E-01
1.5716	3.29E-01	3.09E-01	4.09E-01
1.6539	3.32E-01	3.14E-01	4.14E-01
1.7404	3.40E-01	3.22E-01	4.23E-01
1.8315	3.53E-01	3.34E-01	4.37E-01
1.9274	3.68E-01	3.47E-01	4.53E-01
2	3.78E-01	3.55E-01	4.94E-01
2.0282	3.79E-01	3.58E-01	4.99E-01
2.1344	3.77E-01	3.63E-01	5.09E-01
2.2461	3.72E-01	3.65E-01	5.17E-01
2.3636	3.65E-01	3.66E-01	5.21E-01

Table 2.5.2-41 Vertical Design Spectra
(Sheet 3 of 5)

Frequency (Hz)	Vertical Spectra (g)		
	FIRS*	PBIRS	PBSRS
2.4874	3.59E-01	3.68E-01	5.25E-01
2.5	3.60E-01	3.69E-01	5.26E-01
2.6175	3.65E-01	3.74E-01	5.32E-01
2.7545	3.73E-01	3.85E-01	5.49E-01
2.8987	3.83E-01	3.95E-01	5.60E-01
3.0504	3.84E-01	3.96E-01	5.62E-01
3.21	3.78E-01	3.92E-01	5.60E-01
3.333333	3.76E-01	3.91E-01	5.59E-01
3.378	3.74E-01	3.90E-01	5.57E-01
3.5548	3.73E-01	3.89E-01	5.56E-01
3.7409	3.77E-01	3.93E-01	5.59E-01
3.9367	3.86E-01	4.03E-01	5.67E-01
4	3.86E-01	4.04E-01	5.68E-01
4.1427	3.87E-01	4.02E-01	5.71E-01
4.3595	3.80E-01	3.91E-01	5.65E-01
4.5877	3.74E-01	3.81E-01	5.59E-01
4.8278	3.77E-01	3.80E-01	5.61E-01
5	3.80E-01	3.80E-01	5.60E-01
5.0804	3.85E-01	3.83E-01	5.63E-01
5.3463	3.94E-01	3.88E-01	5.64E-01
5.6261	4.04E-01	3.95E-01	5.63E-01
5.9206	4.21E-01	4.09E-01	5.67E-01
6.2305	4.38E-01	4.22E-01	5.70E-01
6.5565	4.49E-01	4.29E-01	5.69E-01
6.666667	4.53E-01	4.32E-01	5.70E-01
6.8997	4.62E-01	4.44E-01	5.70E-01
7.2608	4.73E-01	4.58E-01	5.69E-01
7.6408	4.78E-01	4.67E-01	5.63E-01
8.0407	4.93E-01	4.85E-01	5.62E-01
8.4615	5.09E-01	5.06E-01	5.62E-01
8.9043	5.27E-01	5.29E-01	5.62E-01
9.3704	5.42E-01	5.50E-01	5.62E-01
9.8608	5.48E-01	5.62E-01	5.59E-01
10	5.48E-01	5.63E-01	5.58E-01
10.377	5.59E-01	5.71E-01	5.56E-01
10.92	5.71E-01	5.79E-01	5.55E-01

Table 2.5.2-41 Vertical Design Spectra
(Sheet 4 of 5)

Frequency (Hz)	Vertical Spectra (g)		
	FIRS*	PBIRS	PBSRS
11.491	5.83E-01	5.88E-01	5.56E-01
12.093	6.01E-01	6.03E-01	5.59E-01
12.726	6.22E-01	6.21E-01	5.63E-01
13.33333	6.43E-01	6.38E-01	5.65E-01
13.392	6.45E-01	6.39E-01	5.66E-01
14.093	6.59E-01	6.51E-01	5.76E-01
14.83	6.68E-01	6.58E-01	5.86E-01
15.606	6.73E-01	6.61E-01	5.95E-01
16.423	6.76E-01	6.62E-01	6.04E-01
17.283	6.81E-01	6.65E-01	6.15E-01
18.187	6.89E-01	6.72E-01	6.29E-01
19.139	6.98E-01	6.82E-01	6.46E-01
20	7.04E-01	6.88E-01	6.61E-01
20.141	7.05E-01	6.90E-01	6.62E-01
21.195	7.10E-01	7.00E-01	6.69E-01
22.304	7.13E-01	7.09E-01	6.76E-01
23.471	7.16E-01	7.20E-01	6.83E-01
24.7	7.18E-01	7.31E-01	6.88E-01
25	7.17E-01	7.32E-01	6.89E-01
25.992	7.08E-01	7.20E-01	6.82E-01
27.353	6.89E-01	6.97E-01	6.70E-01
28.784	6.68E-01	6.73E-01	6.56E-01
30.291	6.48E-01	6.52E-01	6.42E-01
31.876	6.32E-01	6.35E-01	6.30E-01
33.33333	6.19E-01	6.22E-01	6.22E-01
33.544	6.16E-01	6.20E-01	6.18E-01
35.3	5.92E-01	6.04E-01	5.93E-01
37.147	5.68E-01	5.86E-01	5.67E-01
39.092	5.43E-01	5.68E-01	5.41E-01
40	5.32E-01	5.60E-01	5.30E-01
41.138	5.20E-01	5.48E-01	5.17E-01
43.291	4.99E-01	5.28E-01	4.97E-01
45.556	4.79E-01	5.09E-01	4.77E-01
47.94	4.61E-01	4.92E-01	4.60E-01
50	4.48E-01	4.78E-01	4.46E-01
50.449	4.45E-01	4.75E-01	4.44E-01

Table 2.5.2-41 Vertical Design Spectra
(Sheet 5 of 5)

Frequency (Hz)	Vertical Spectra (g)		
	FIRS*	PBIRS	PBSRS
53.09	4.30E-01	4.57E-01	4.31E-01
55.868	4.17E-01	4.41E-01	4.20E-01
58.792	4.05E-01	4.27E-01	4.09E-01
61.869	3.94E-01	4.14E-01	3.99E-01
65.107	3.84E-01	4.02E-01	3.90E-01
68.515	3.75E-01	3.91E-01	3.81E-01
72.101	3.67E-01	3.81E-01	3.73E-01
75.874	3.60E-01	3.72E-01	3.65E-01
79.845	3.53E-01	3.62E-01	3.57E-01
84.024	3.47E-01	3.54E-01	3.50E-01
88.421	3.41E-01	3.46E-01	3.43E-01
93.049	3.36E-01	3.39E-01	3.35E-01
97.919	3.31E-01	3.32E-01	3.28E-01
100	3.29E-01	3.29E-01	3.25E-01

* The FIRS is before ISG-017 check

Table 2.5.2-42 FIRS after ISG-017 Check
(Sheet 1 of 5)

Frequency (Hz)	Spectra (g)	
	Horizontal	Vertical
0.10000	0.0289	0.0273
0.10523	0.0310	0.0285
0.11074	0.0330	0.0297
0.11654	0.0350	0.0308
0.12264	0.0370	0.0318
0.12905	0.0389	0.0327
0.13333	0.0401	0.0332
0.13581	0.0407	0.0335
0.14292	0.0422	0.0342
0.15040	0.0435	0.0347
0.15827	0.0447	0.0352
0.16655	0.0460	0.0356
0.17527	0.0473	0.0361
0.18444	0.0487	0.0366
0.19409	0.0501	0.0371
0.20000	0.0509	0.0374
0.20425	0.0517	0.0386
0.21494	0.0535	0.0416
0.22619	0.0554	0.0448
0.23803	0.0575	0.0485
0.25000	0.0598	0.0524
0.25049	0.0599	0.0525
0.26359	0.0629	0.0552
0.27739	0.0665	0.0583
0.29191	0.0706	0.0617
0.30719	0.0748	0.0651
0.32326	0.0790	0.0684
0.33333	0.0815	0.0704
0.34018	0.0832	0.0719
0.35798	0.0875	0.0757
0.37672	0.0919	0.0796
0.39644	0.0969	0.0840
0.41718	0.1025	0.0890
0.43902	0.1091	0.0948
0.46199	0.1170	0.1018
0.48617	0.1263	0.1126

Table 2.5.2-42 FIRS after ISG-017 Check
(Sheet 2 of 5)

Frequency (Hz)	Spectra (g)	
	Horizontal	Vertical
0.50000	0.1320	0.1192
0.51162	0.1369	0.1244
0.53839	0.1478	0.1358
0.56657	0.1569	0.1464
0.59622	0.1682	0.1565
0.62743	0.1755	0.1666
0.66027	0.1827	0.1770
0.66667	0.1841	0.1790
0.69482	0.1883	0.1847
0.73119	0.1960	0.1948
0.76945	0.2035	0.2049
0.80972	0.2121	0.2164
0.85210	0.2246	0.2322
0.89670	0.2443	0.2559
0.94363	0.2723	0.2890
0.99301	0.3025	0.3254
1.00000	0.3050	0.3287
1.04500	0.3237	0.3528
1.09970	0.3265	0.3542
1.15720	0.3241	0.3500
1.21780	0.3245	0.3489
1.28150	0.3316	0.3549
1.33333	0.3430	0.3658
1.34860	0.3464	0.3690
1.41920	0.3671	0.3892
1.49350	0.3856	0.4069
1.57160	0.3946	0.4144
1.65390	0.3993	0.4172
1.74040	0.4089	0.4253
1.83150	0.4260	0.4409
1.92740	0.4443	0.4576
2.00000	0.4572	0.4692
2.02820	0.4484	0.4703
2.13440	0.4254	0.4665
2.24610	0.4156	0.4586
2.36360	0.4148	0.4477

Table 2.5.2-42 FIRS after ISG-017 Check
(Sheet 3 of 5)

Frequency (Hz)	Spectra (g)	
	Horizontal	Vertical
2.48740	0.4162	0.4395
2.50000	0.4177	0.4401
2.61750	0.4297	0.4448
2.75450	0.4473	0.4478
2.89870	0.4659	0.4509
3.05040	0.4747	0.4443
3.21000	0.4759	0.4224
3.33333	0.4784	0.4083
3.37800	0.4799	0.4030
3.55480	0.4910	0.3874
3.74090	0.5083	0.3768
3.93670	0.5336	0.3855
4.00000	0.5392	0.3864
4.14270	0.5510	0.3873
4.35950	0.5569	0.3805
4.58770	0.5640	0.3745
4.82780	0.5841	0.3769
5.00000	0.6011	0.3804
5.08040	0.6092	0.3849
5.34630	0.6270	0.3941
5.62610	0.6455	0.4037
5.92060	0.6768	0.4212
6.23050	0.7078	0.4381
6.55650	0.7294	0.4491
6.66667	0.7366	0.4528
6.89970	0.7508	0.4623
7.26080	0.7666	0.4733
7.64080	0.7728	0.4784
8.04070	0.7936	0.4925
8.46150	0.8186	0.5094
8.90430	0.8442	0.5267
9.37040	0.8662	0.5418
9.86080	0.8740	0.5481
10.00000	0.8735	0.5482
10.37700	0.8733	0.5591
10.92000	0.8679	0.5712

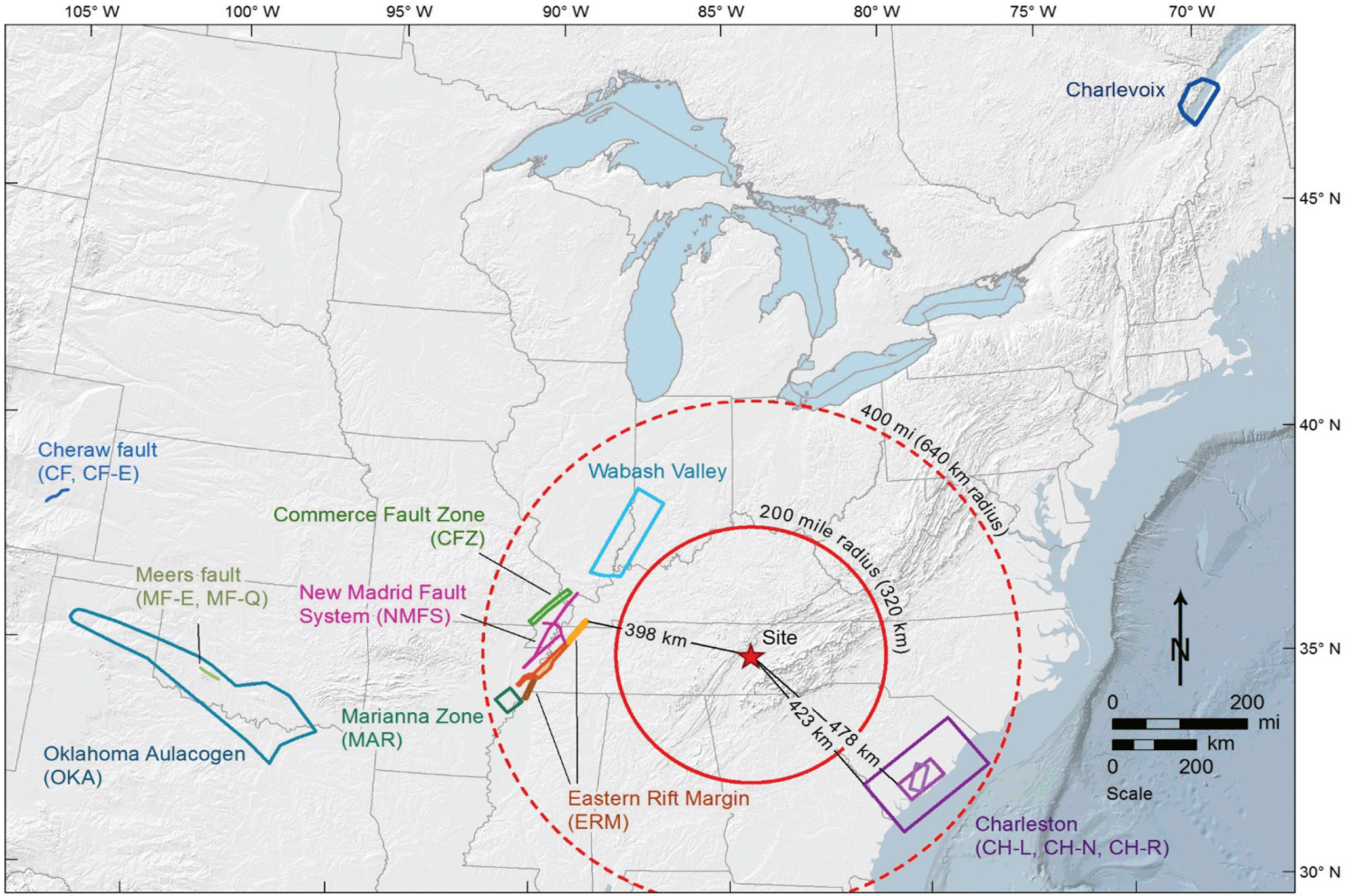
Table 2.5.2-42 FIRS after ISG-017 Check
(Sheet 4 of 5)

Frequency (Hz)	Spectra (g)	
	Horizontal	Vertical
11.49100	0.8625	0.5835
12.09300	0.8637	0.6006
12.72600	0.8708	0.6224
13.33333	0.8777	0.6433
13.39200	0.8785	0.6449
14.09300	0.8801	0.6592
14.83000	0.8745	0.6682
15.60600	0.8637	0.6733
16.42300	0.8506	0.6765
17.28300	0.8397	0.6813
18.18700	0.8323	0.6889
19.13900	0.8261	0.6976
20.00000	0.8191	0.7037
20.14100	0.8185	0.7049
21.19500	0.8096	0.7102
22.30400	0.7984	0.7134
23.47100	0.7866	0.7159
24.70000	0.7744	0.7178
25.00000	0.7706	0.7174
25.99200	0.7602	0.7075
27.35300	0.7405	0.6889
28.78400	0.7179	0.6675
30.29100	0.6972	0.6481
31.87600	0.6801	0.6318
33.33333	0.6664	0.6189
33.54400	0.6650	0.6161
35.30000	0.6514	0.5925
37.14700	0.6360	0.5677
39.09200	0.6201	0.5431
40.00000	0.6129	0.5322
41.13800	0.6013	0.5200
43.29100	0.5809	0.4987
45.55600	0.5621	0.4791
47.94000	0.5454	0.4614
50.00000	0.5328	0.4481
50.44900	0.5293	0.4452

Table 2.5.2-42 FIRS after ISG-017 Check
(Sheet 5 of 5)

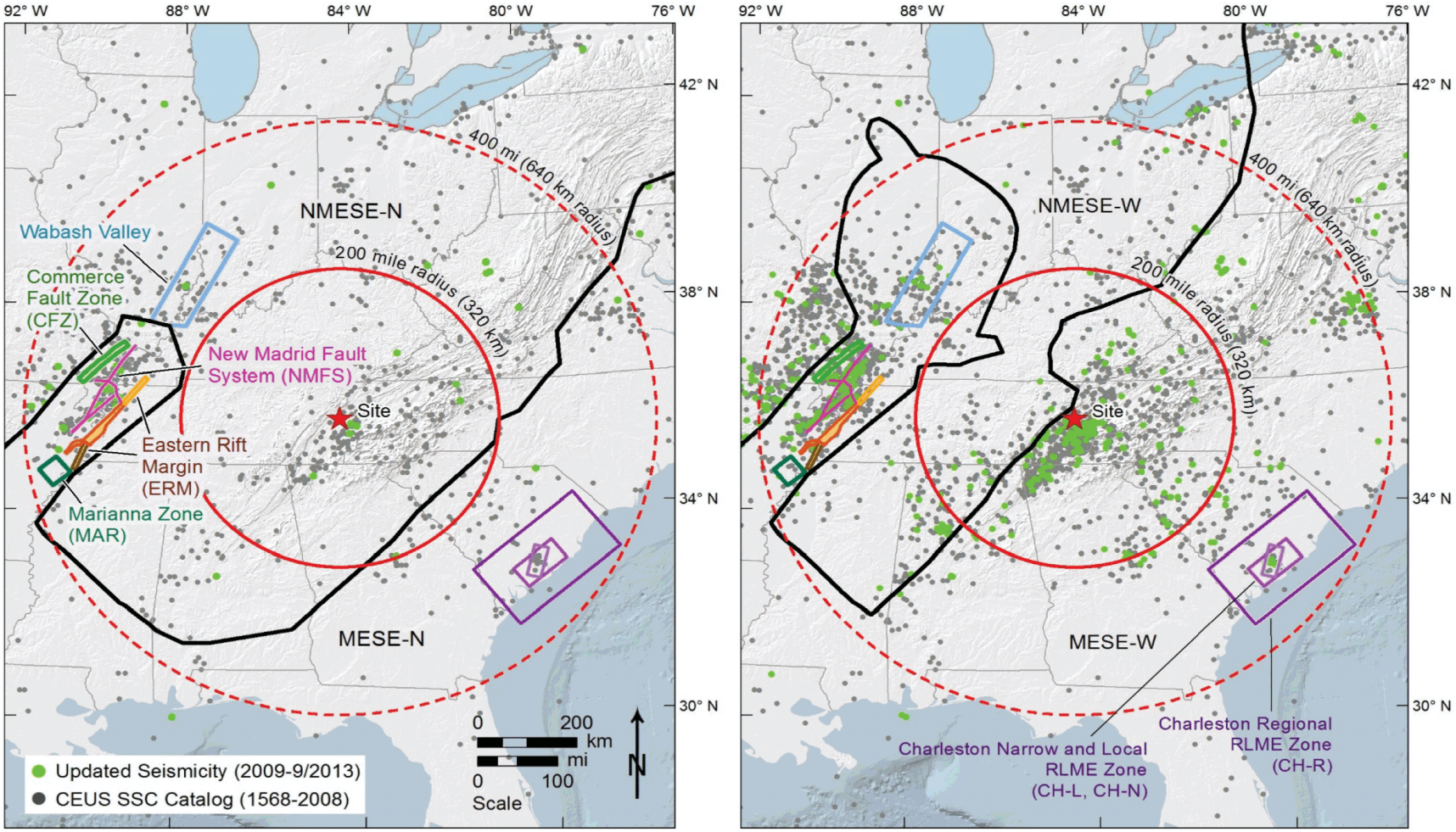
Frequency (Hz)	Spectra (g)	
	Horizontal	Vertical
53.09000	0.5107	0.4302
55.86800	0.4941	0.4168
58.79200	0.4791	0.4047
61.86900	0.4654	0.3936
65.10700	0.4531	0.3837
68.51500	0.4422	0.3750
72.10100	0.4322	0.3670
75.87400	0.4232	0.3598
79.84500	0.4149	0.3532
84.02400	0.4071	0.3469
88.42100	0.3999	0.3412
93.04900	0.3932	0.3359
97.91900	0.3871	0.3311
100.00000	0.3848	0.3293

Replace Figures 2.5.2-17 through 2.5.2-20, 2.5.2-22, 2.5.2-23, 2.5.2-24 and 2.5.2-25 with the following:



Note: Modified after Figure 6.1-1 in Reference 2.5.2-1.

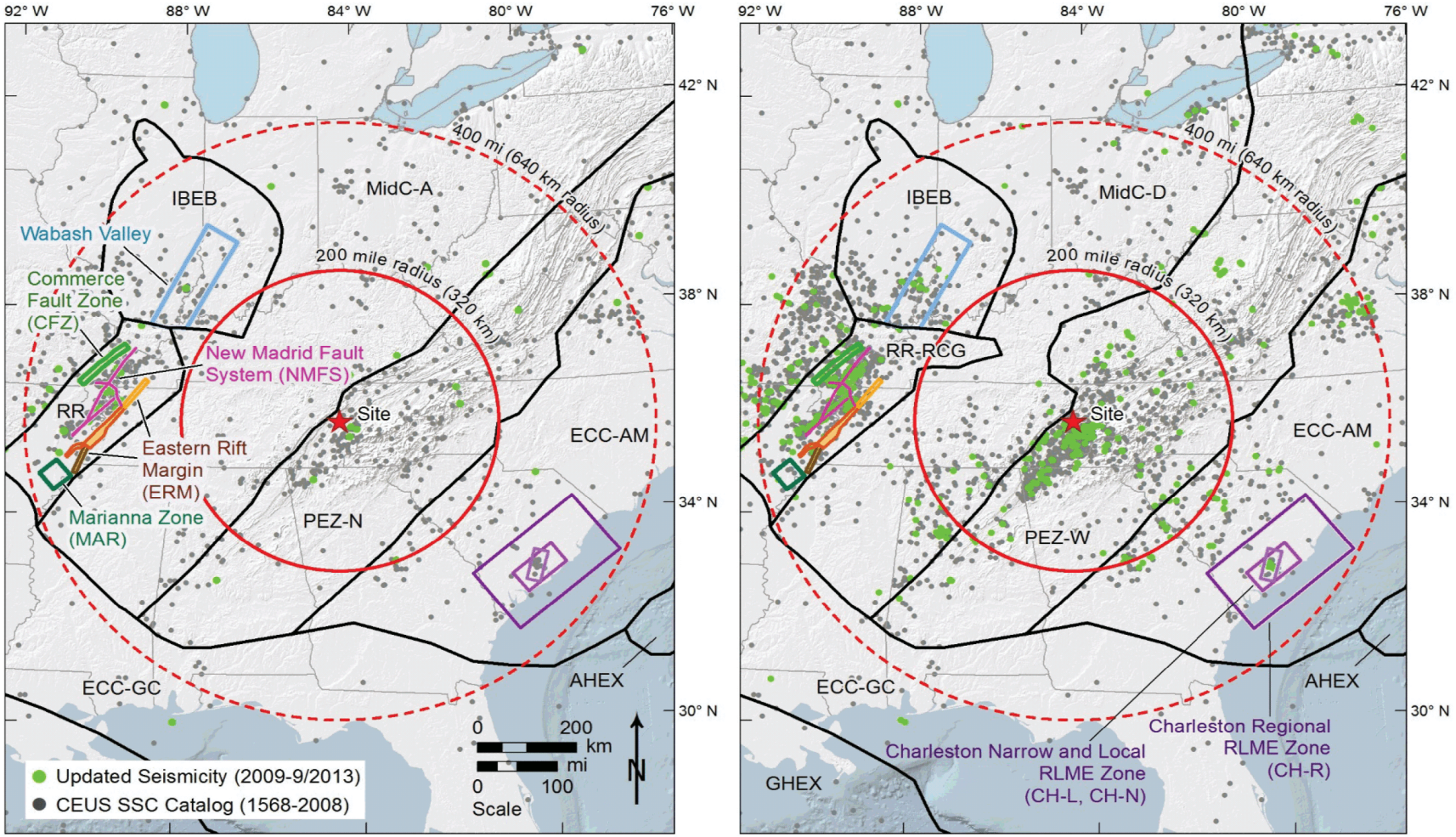
Figure 2.5.2-17R Repeated Large Magnitude Earthquake Source Zones in the CEUS SSC Model



Notes: Modified from Reference 2.5.2-1. Zone acronyms are defined in Table 2.5.2-4.

The Study Region Mmax (maximum magnitude) source zone is not explicitly shown but is taken as a radius of 640 km from the CRN Site. The project earthquake catalog is plotted in both panels with left panel showing independent events E[**M**]>2.9 (used to calculate rates) and right panel shows all events E[**M**]>2.2 – see Subsection 2.5.2.1.

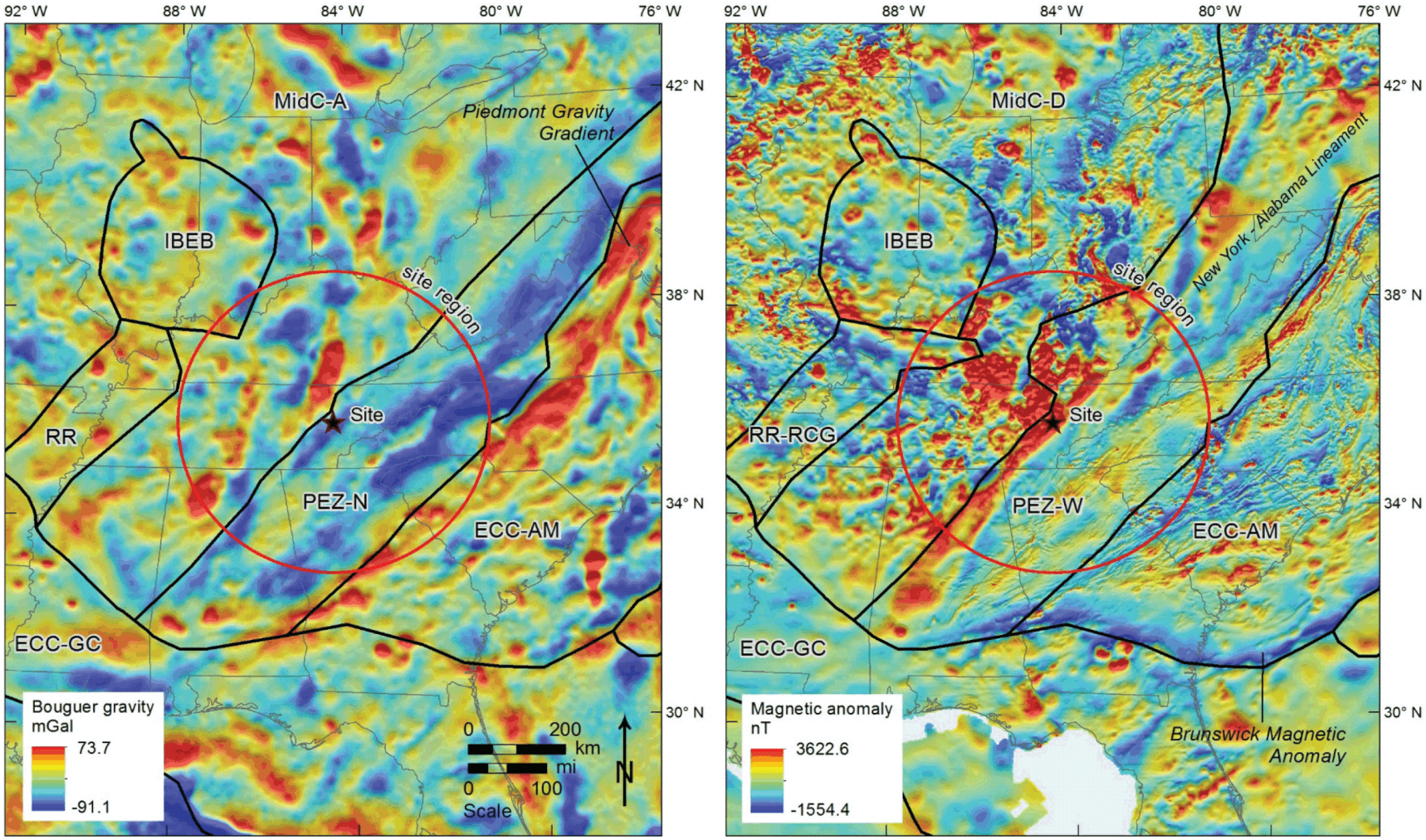
Figure 2.5.2-18R Maximum Magnitude and Repeated Large Magnitude Earthquake Source Zones in the CEUS SSC Model



Notes: Modified from Reference 2.5.2-1. Zone acronyms are defined in Table 2.5.2-4.

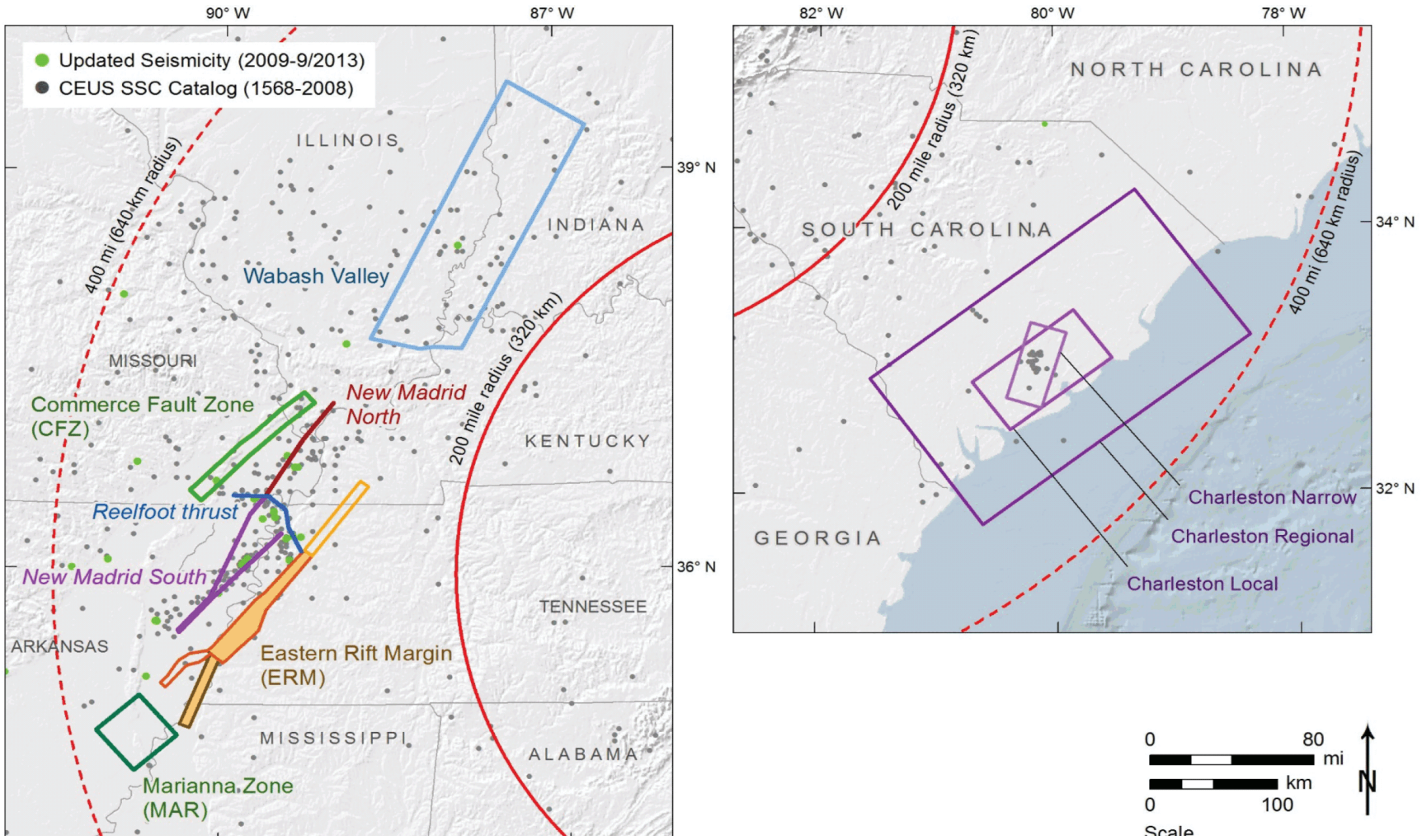
The project earthquake catalog is plotted in both panels with left panel showing independent events E[**M**]>2.9 (used to calculate rates) and right panel shows all events E[**M**]>2.2 – see Subsection 2.5.2.1.

Figure 2.5.2-19R Seismotectonic and Repeated Large Magnitude Earthquake Source Zones in the CEUS SSC Model



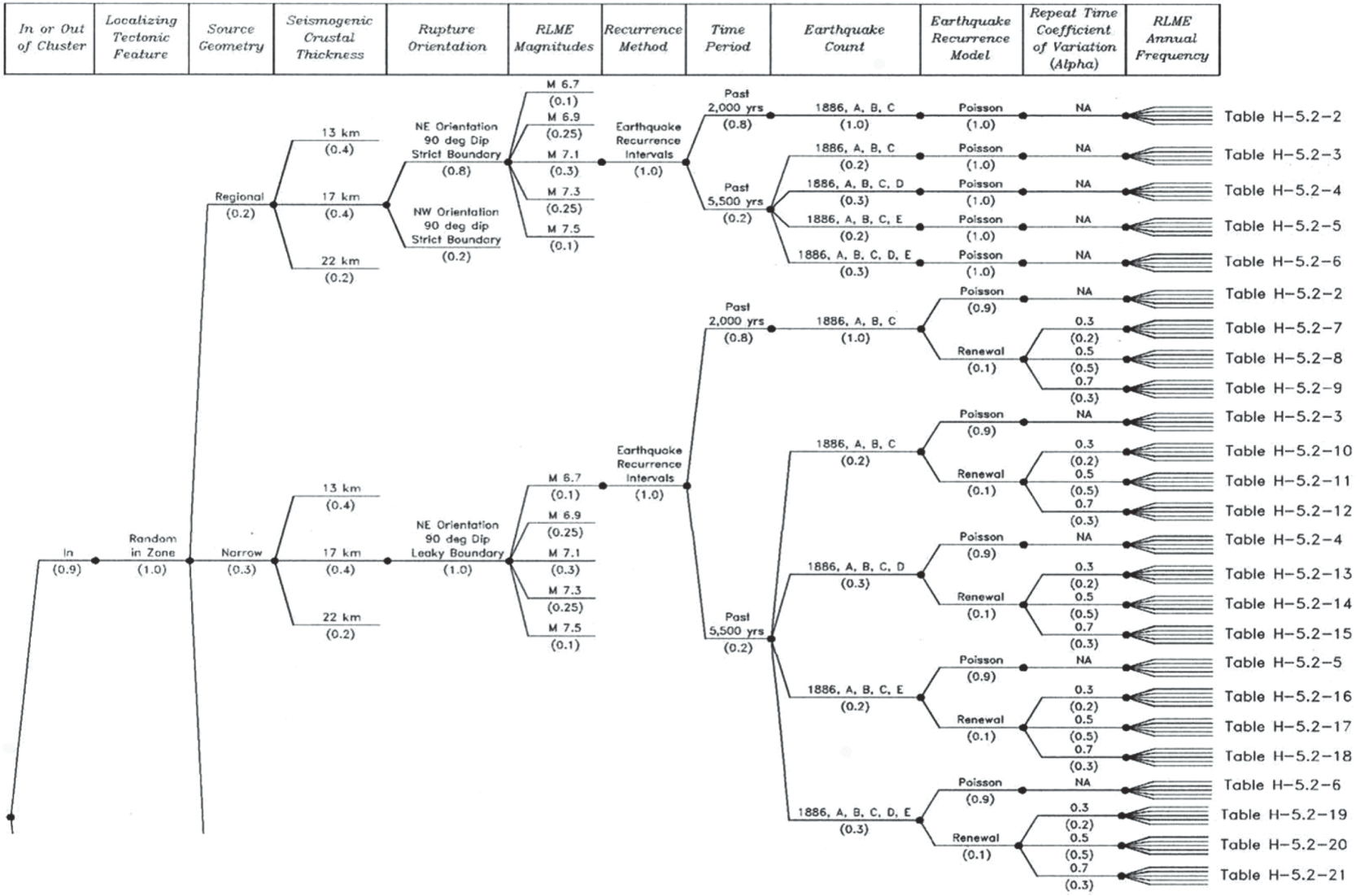
Note: Modified from the GIS Database in Reference 2.5.2-1. Zone acronyms are defined in Table 2.5.2-4.

Figure 2.5.2-20R Seismotectonic Zones with Bouguer Gravity (Left) and Total Intensity Aeromagnetic Anomaly Data (Right)



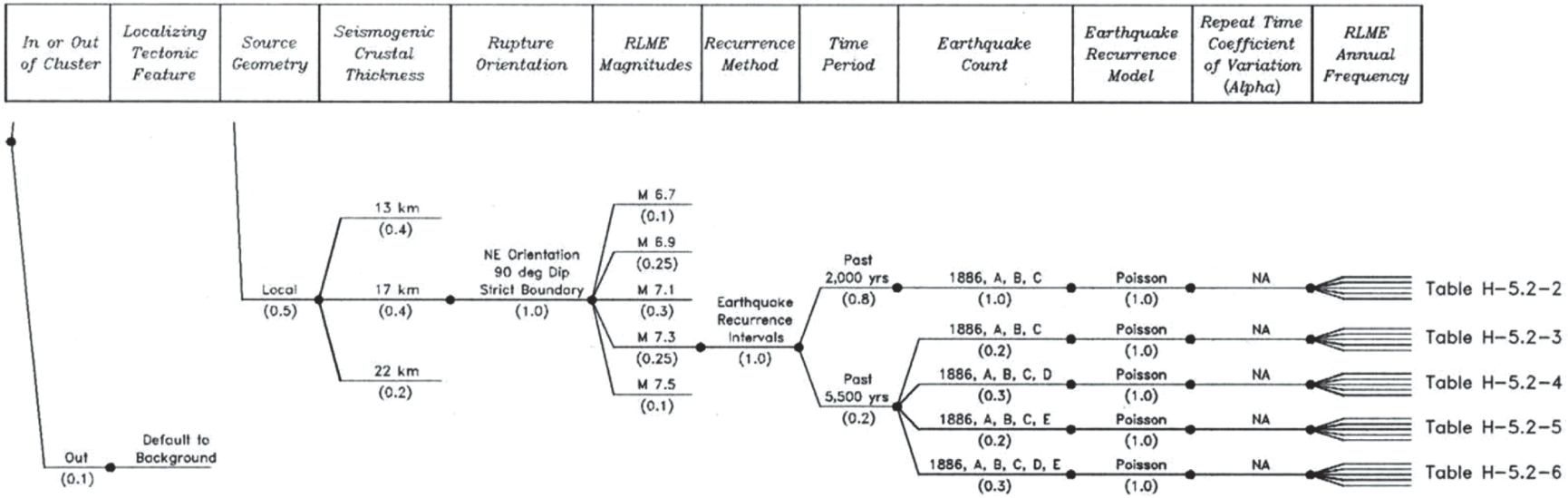
Notes: Modified from Figure 6.1-2b of Reference 2.5.2-1. Both panels show independent earthquakes E[**M**]>2.9 from the project earthquake catalog – see Subsection 2.5.2.1.

Figure 2.5.2-22R Repeated Large Magnitude Earthquake Sources in the CEUS SSC Model within 640 km (400 mi) of the Clinch River Nuclear Site



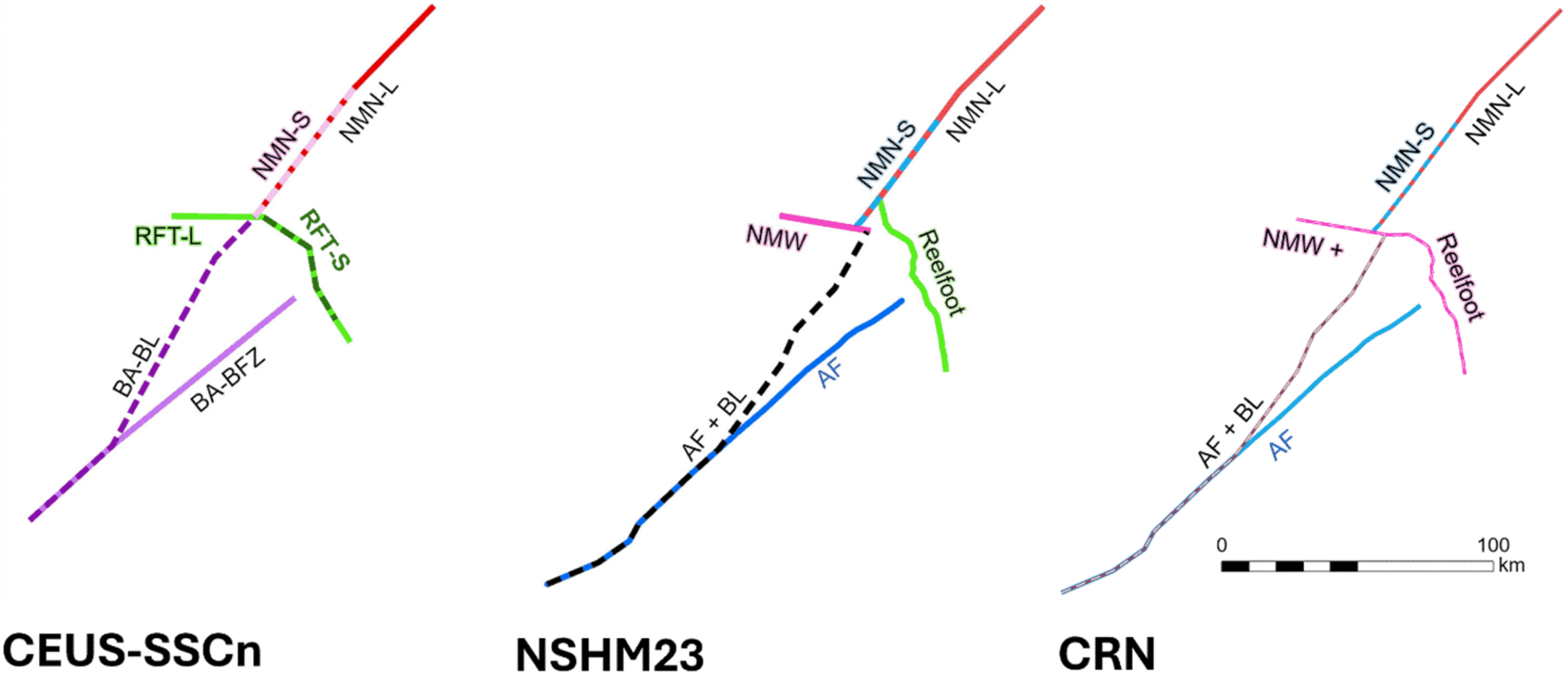
Note: From Figure H-5.2-1(a) in Reference 2.5.2-1.
RLME = Repeated Large Magnitude Earthquake

Figure 2.5.2-23R (Sheet 1 of 2) Charleston Repeated Large Magnitude Earthquake Logic Tree



Note: From Figure H-5.2-1(b) in Reference 2.5.2-1
RLME = Repeated Large Magnitude Earthquake

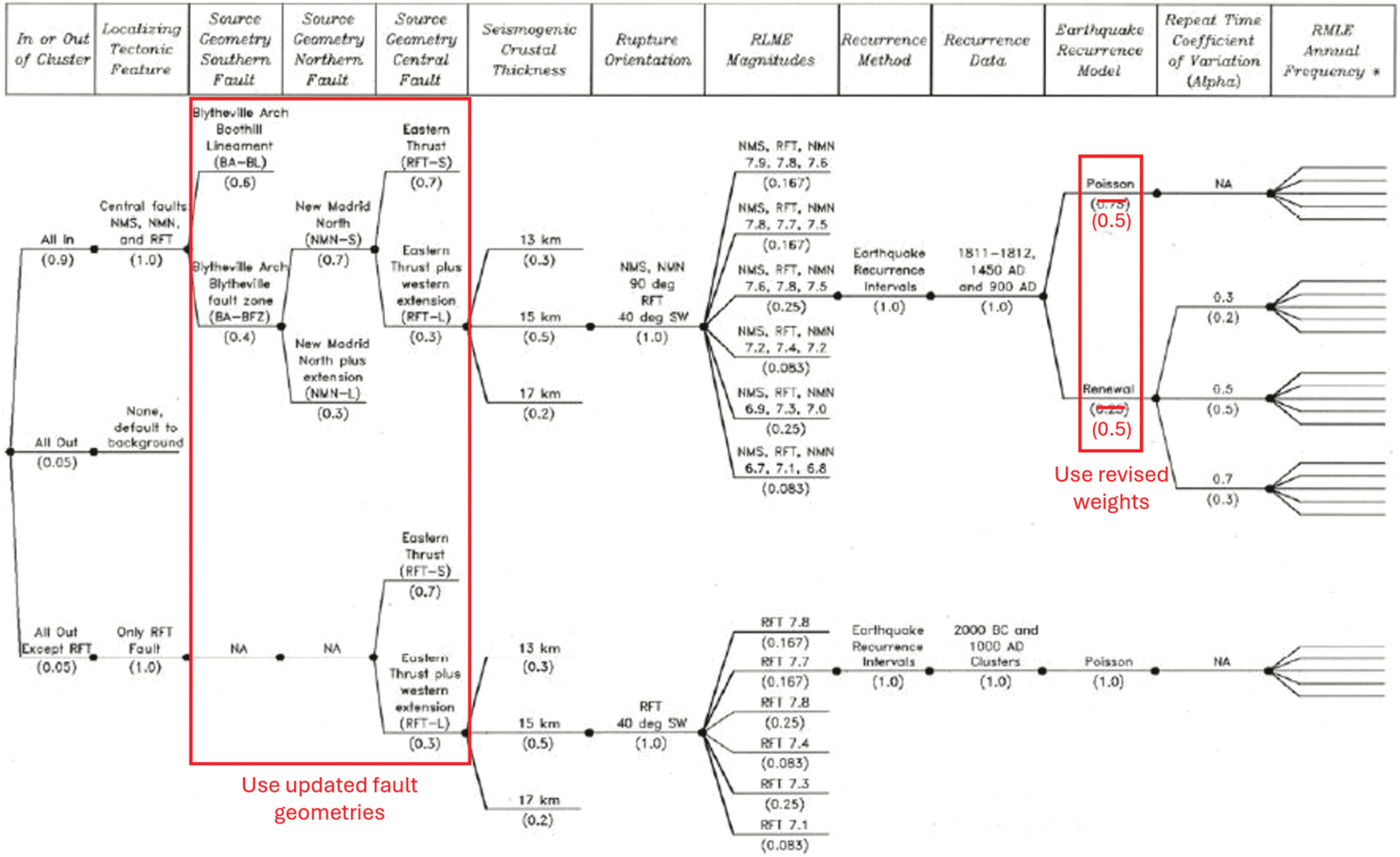
Figure 2.5.2-23R (Sheet 2 of 2) Charleston Repeated Large Magnitude Earthquake Logic Tree



BA = Blytheville arch, BL = Bootheel lineament, BFZ = Blytheville fault zone, RFT-L and RFT-S = Reelfoot thrust-Long and -Short, NMN-L and NMN-S = New Madrid North-Long and -Short, AF = Axial fault, NMW = New Madrid West

Figure 2.5.2-24R Depictions of the NMFS RLME Source from CEUS-SSCn

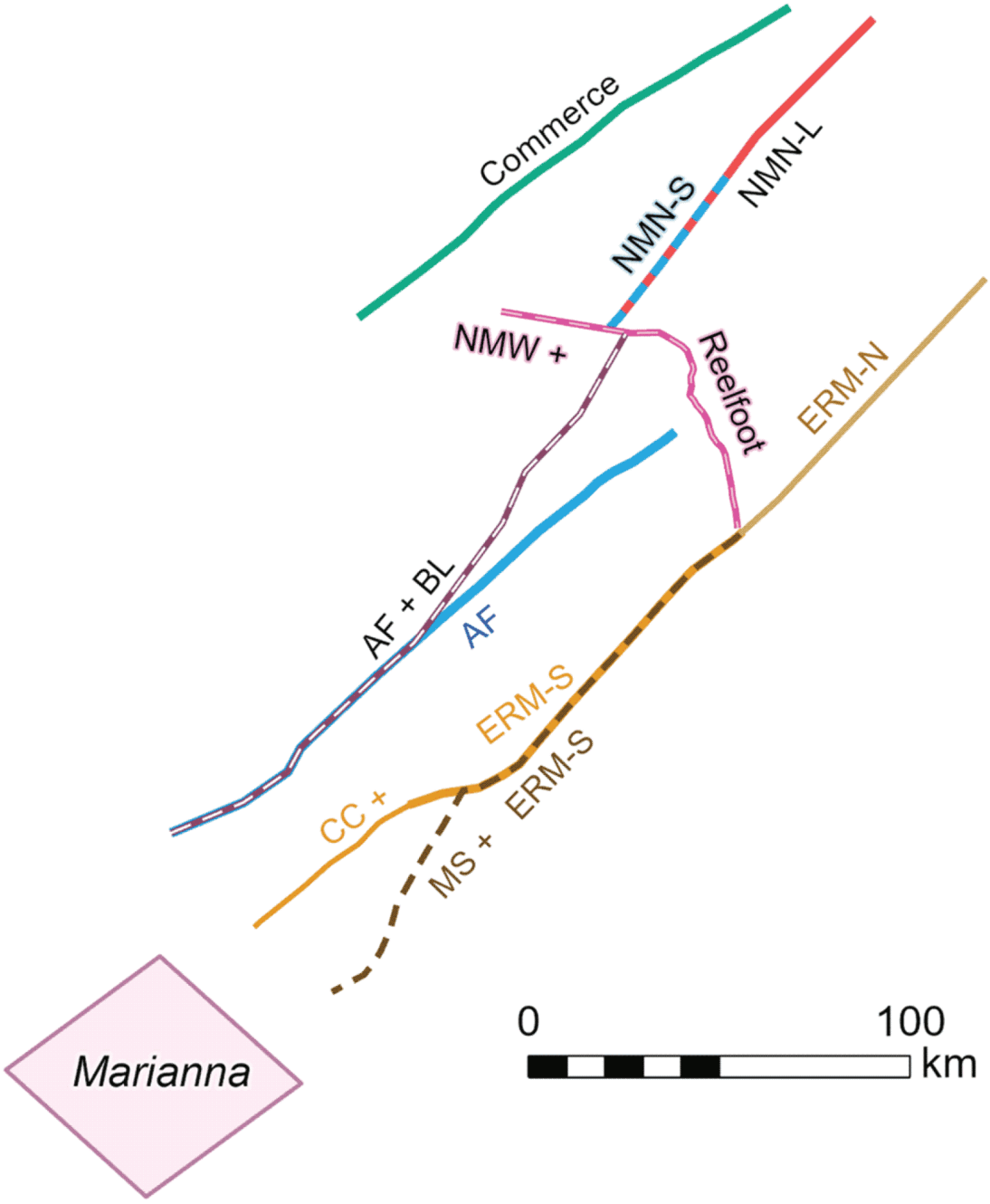
(Reference 2.5.2-1), NSHM23 (Reference 2.5.2-175 and Reference 2.5.2-176), and CRN (Reference 2.5.2-265)



Note: From Figure H-5.5-1 in Reference 2.5.2-1
RLME = Repeated Large Magnitude Earthquake

Figure 2.5.2-25R Logic Tree of New Madrid Fault System Repeated Large Magnitude Earthquake

Add new Figures 2.5.2-28R and 2.5.2-29R:



Modified after Shumway et al., (Reference 2.5.2-176)

AF = Axial fault, BL = Bootheel lineament, BA = Blytheville arch, NMN-L and NMN-S = New Madrid North-Long and -Short, CC = Crittenden County, ERM-S and ERM-N = Eastern Rift Margin-South and -North, MS = Meeman-Shelby, NMW = New Madrid West

Figure 2.5.2-28R Revised Sources in the New Madrid Seismic Zone RLME

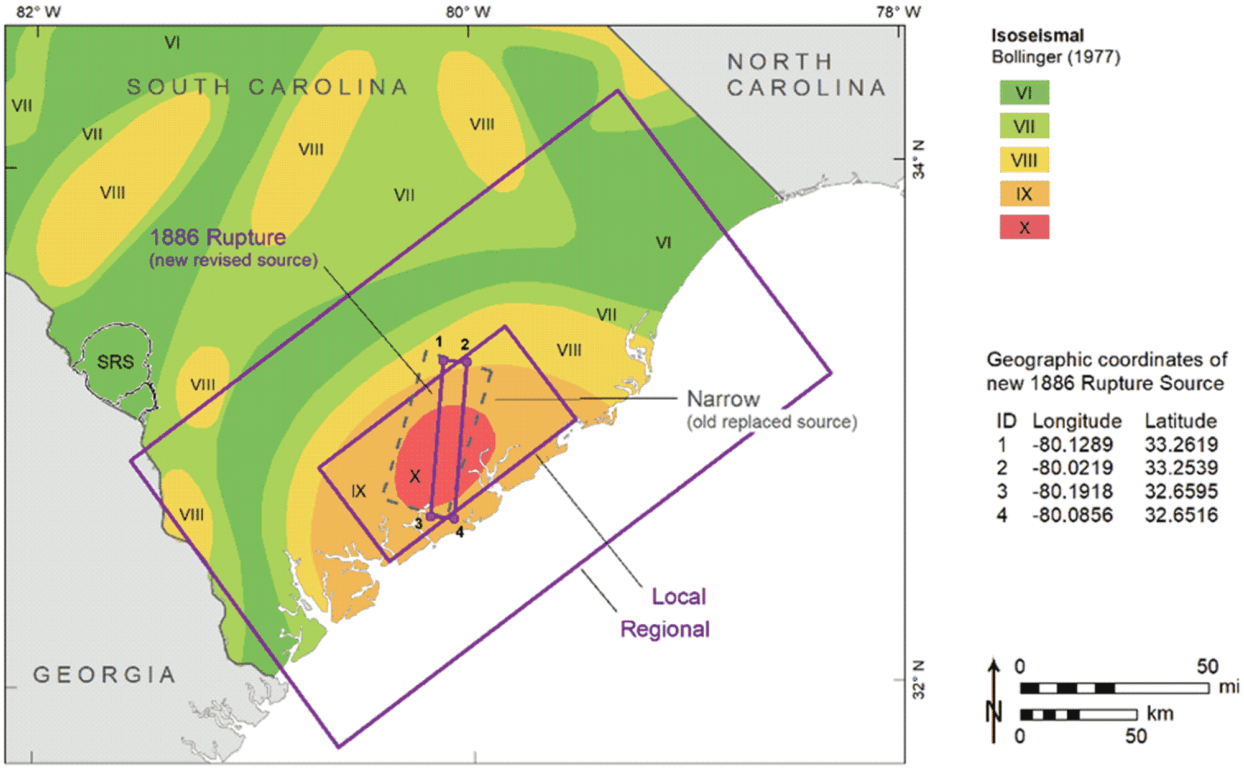


Figure 2.5.2-29R Revised Charleston RLME Source, Showing the New 1886 Rupture Zone that Replaces the Narrow Zone

Replace Figures 2.5.2-28 through 2.5.2-79 with the following:

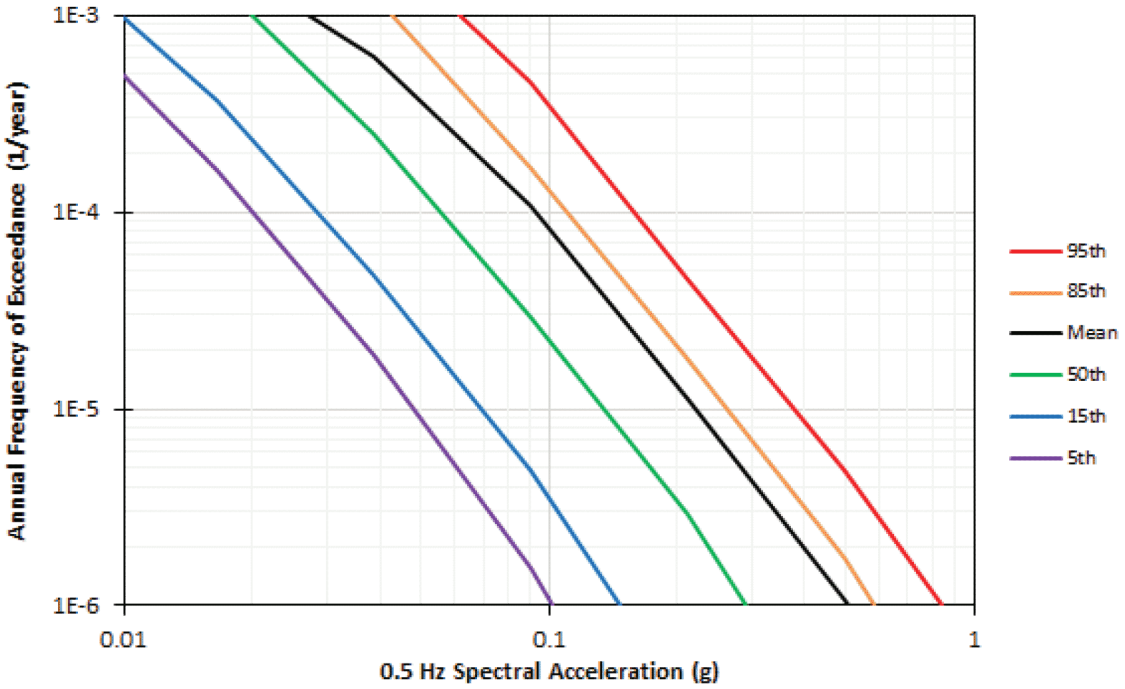


Figure 2.5.2-30R Mean and Fractile Rock Hazard for 0.5 Hz

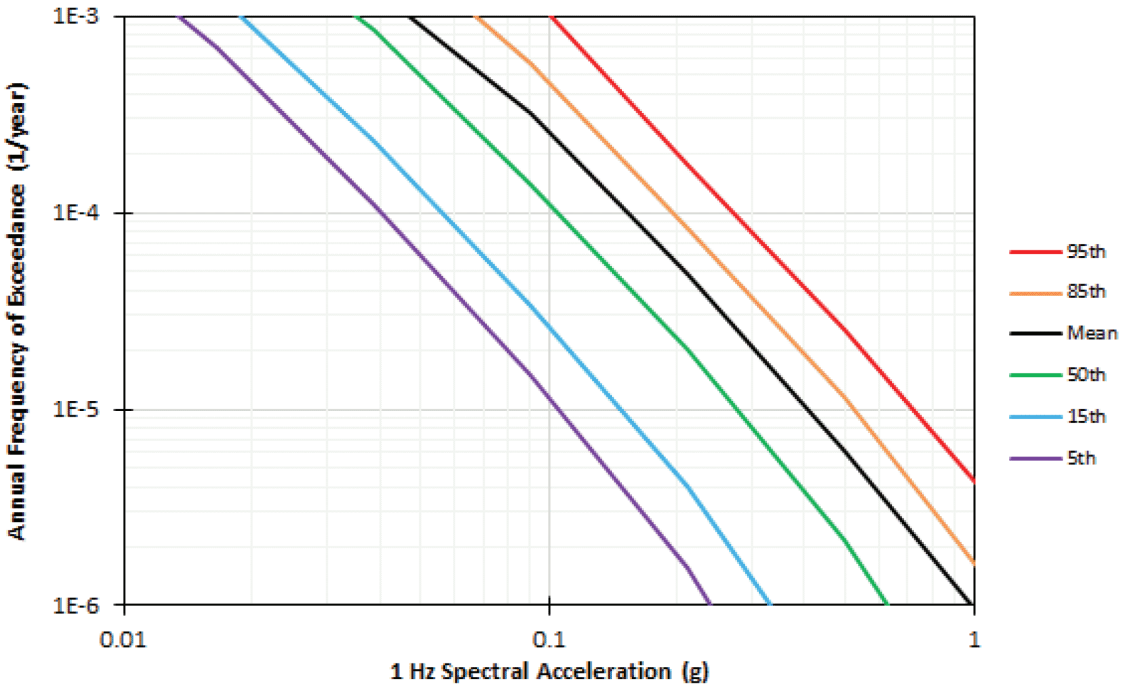


Figure 2.5.2-31R Mean and Fractile Rock Hazard for 1 Hz

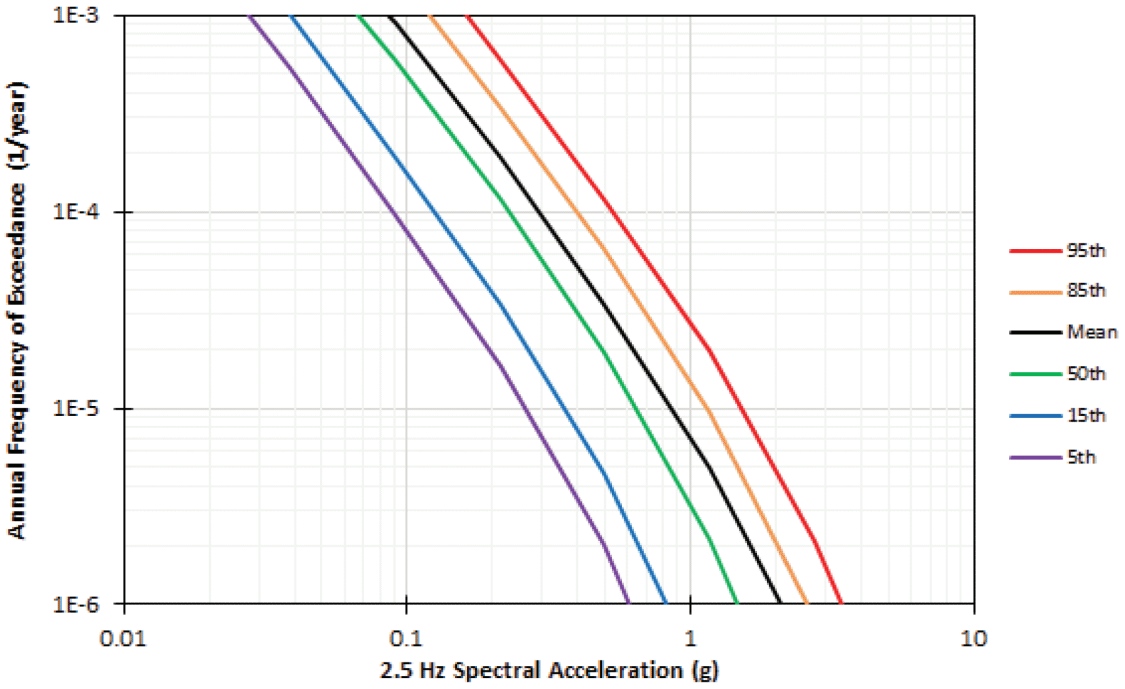


Figure 2.5.2-32R Mean and Fractile Rock Hazard for 2.5 Hz

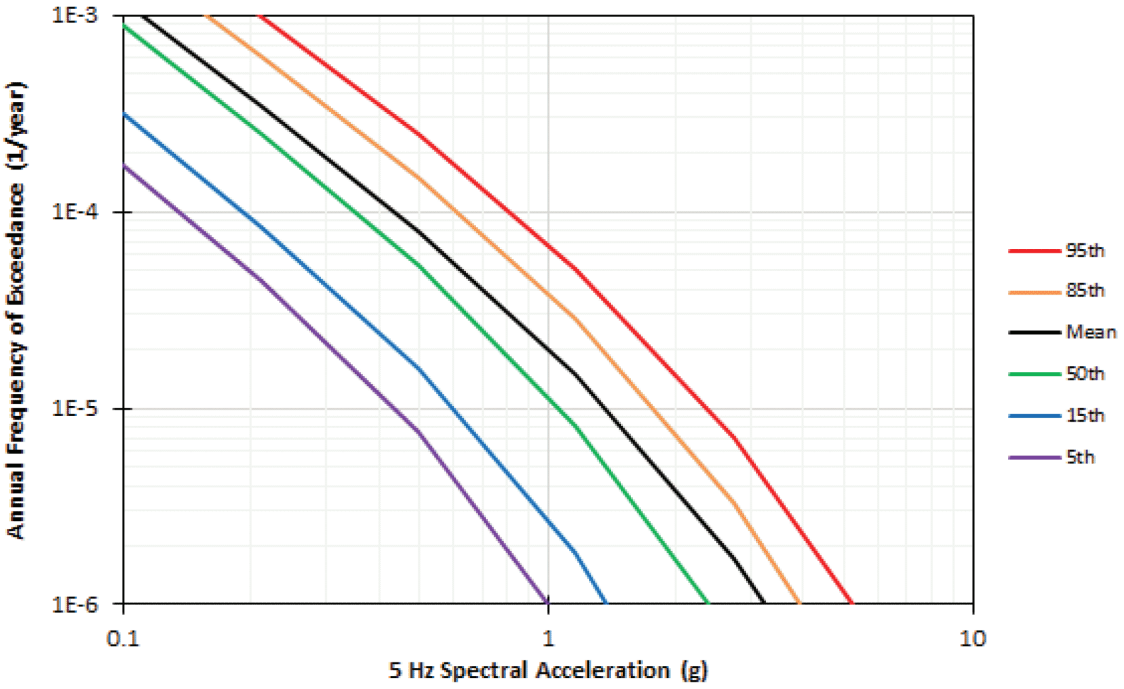


Figure 2.5.2-33R Mean and Fractile Rock Hazard for 5 Hz

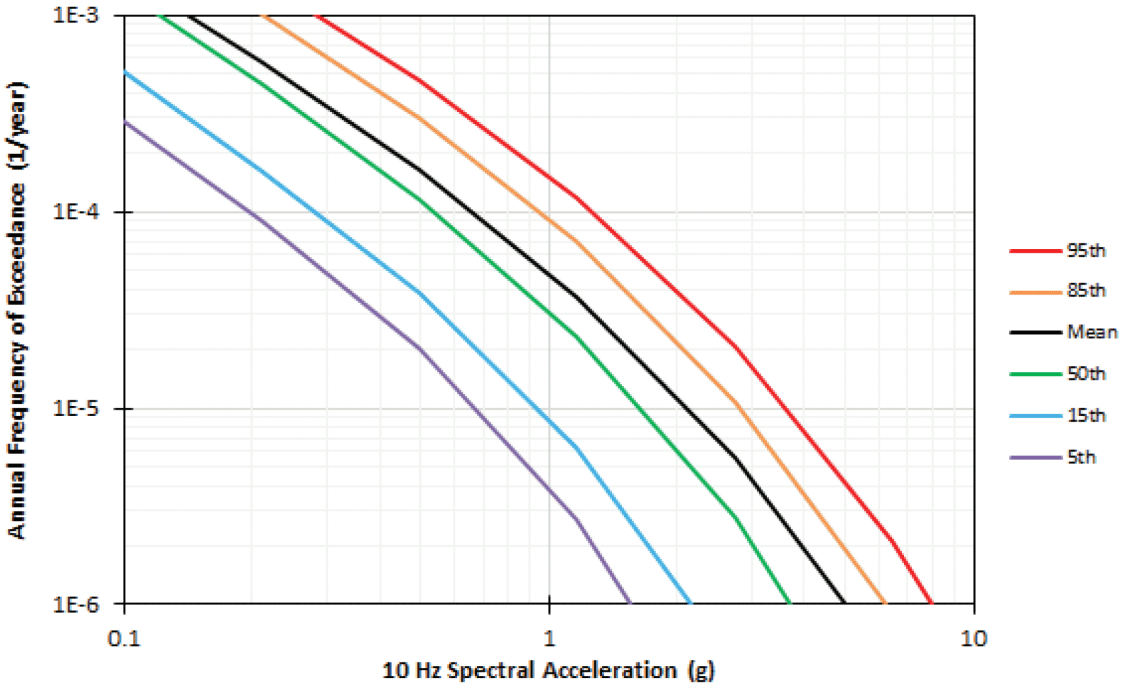


Figure 2.5.2-34R Mean and Fractile Rock for 10 Hz

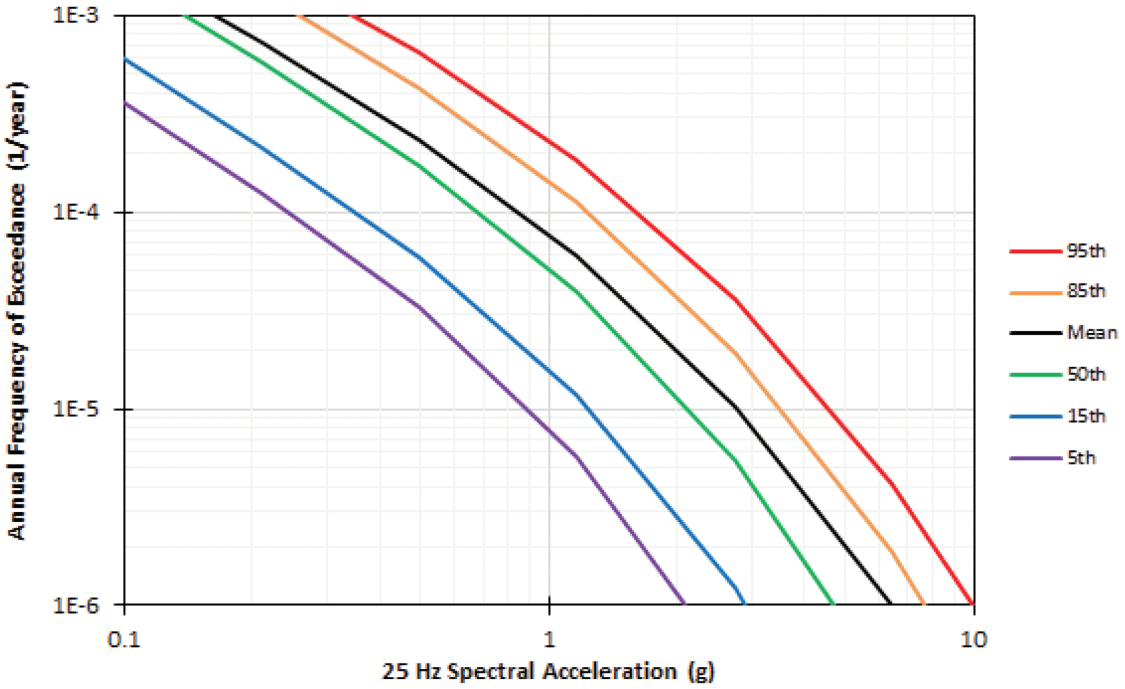


Figure 2.5.2-35R Mean and Fractile Rock for 25 Hz

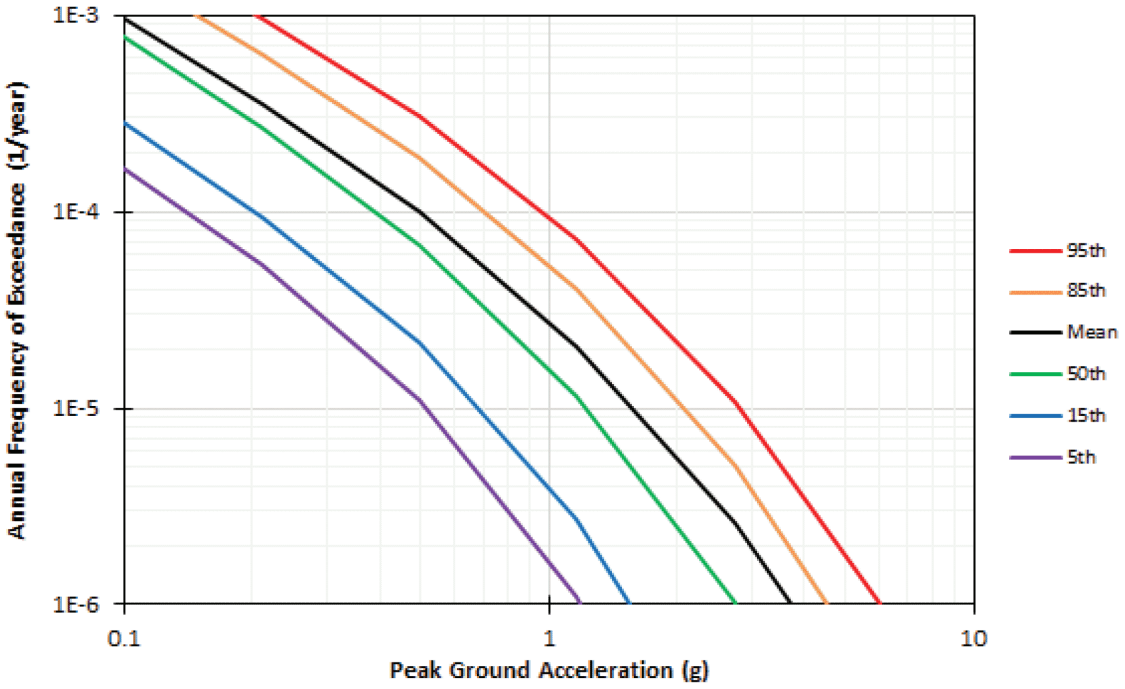


Figure 2.5.2-36R Mean and Fractile Rock Hazard for Peak Ground Acceleration

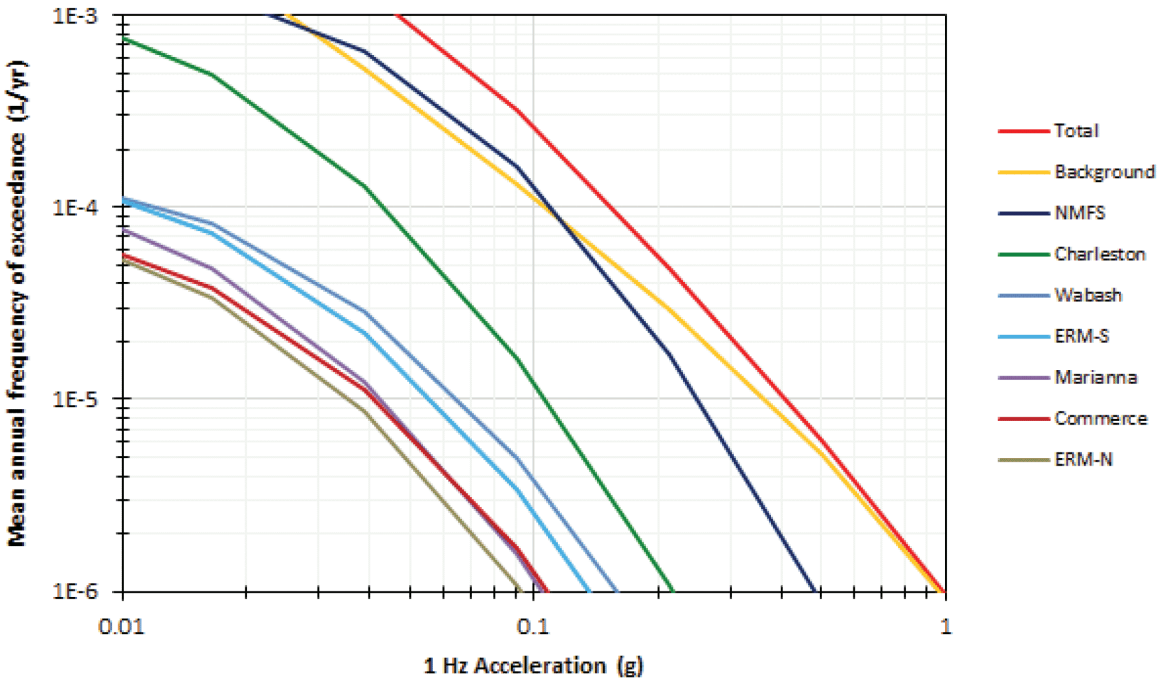


Figure 2.5.2-37R 1Hz Mean Rock Hazard from Background and Repeated Large Magnitude Earthquake Sources

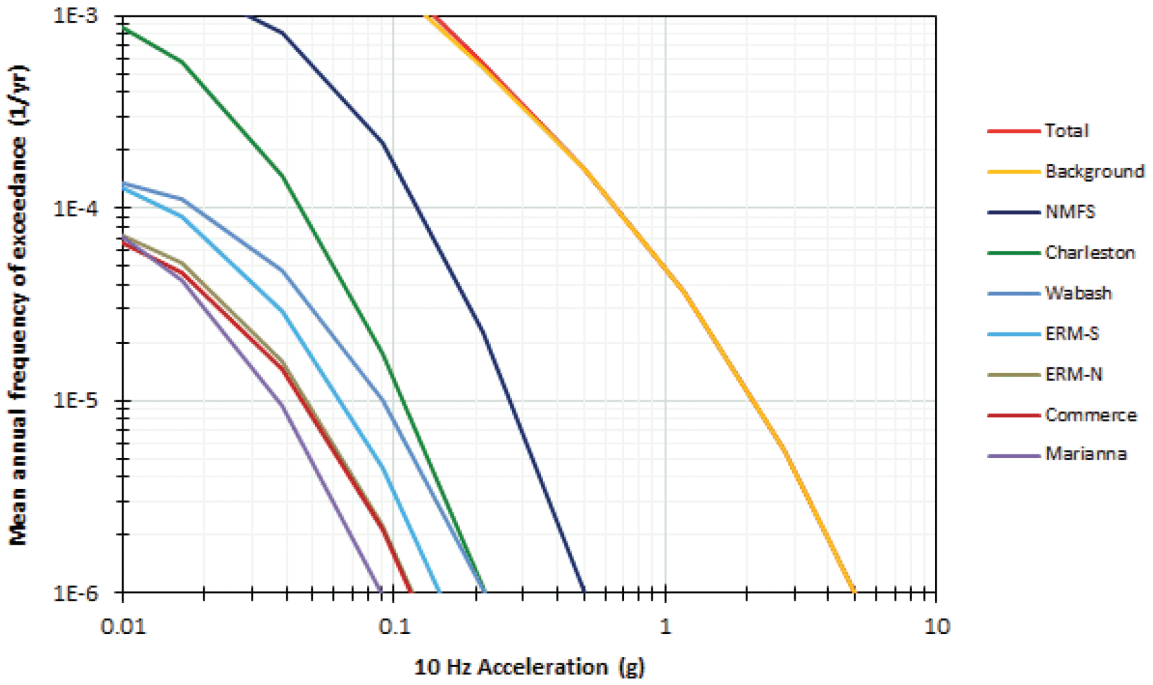


Figure 2.5.2-38R 10 Hz Mean Rock Hazard from Background and Repeated Large Magnitude Earthquake Sources

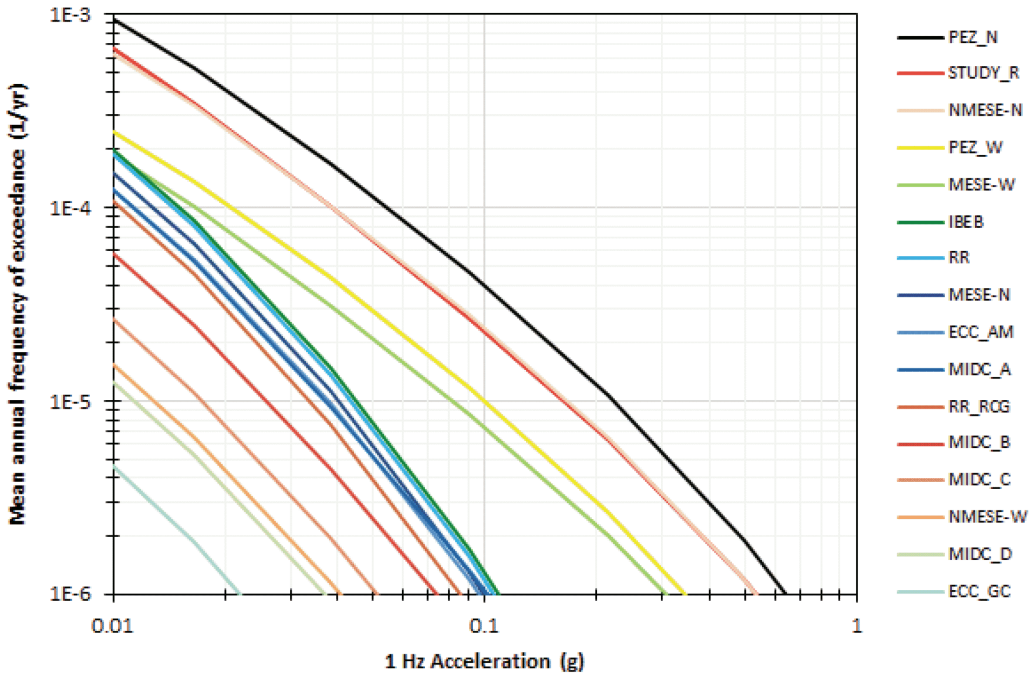


Figure 2.5.2-39R 1 Hz Mean Rock Hazard from Individual Weighted Background Sources

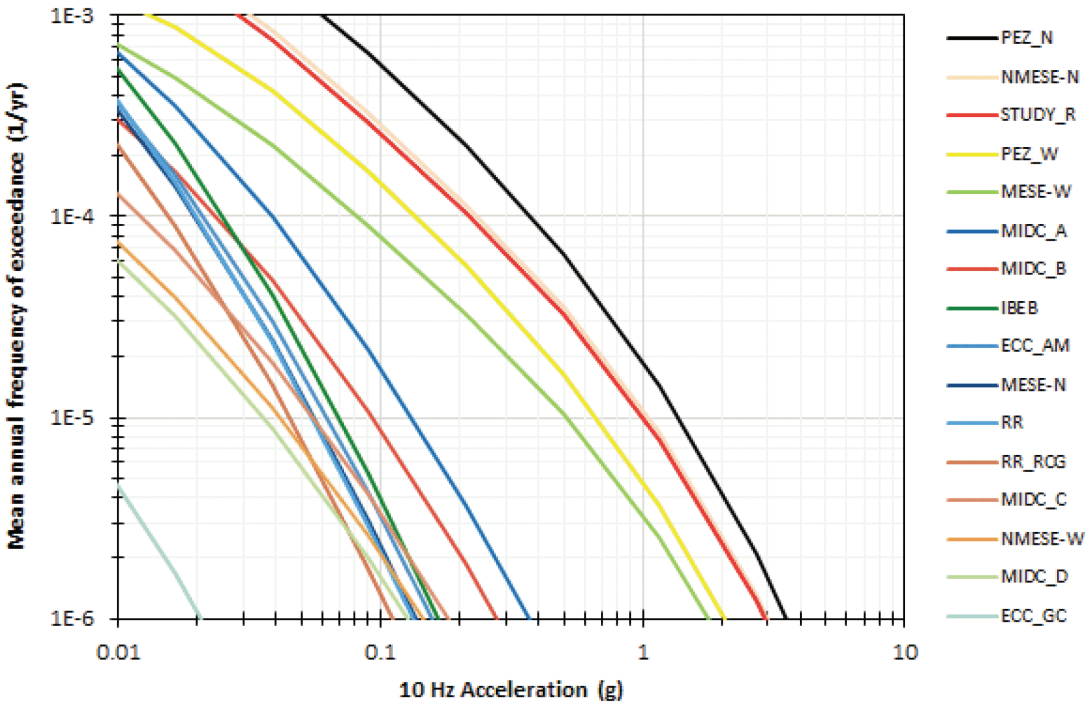


Figure 2.5.2-40R 10 Hz Mean Rock Hazard from Individual Weighted Background Sources

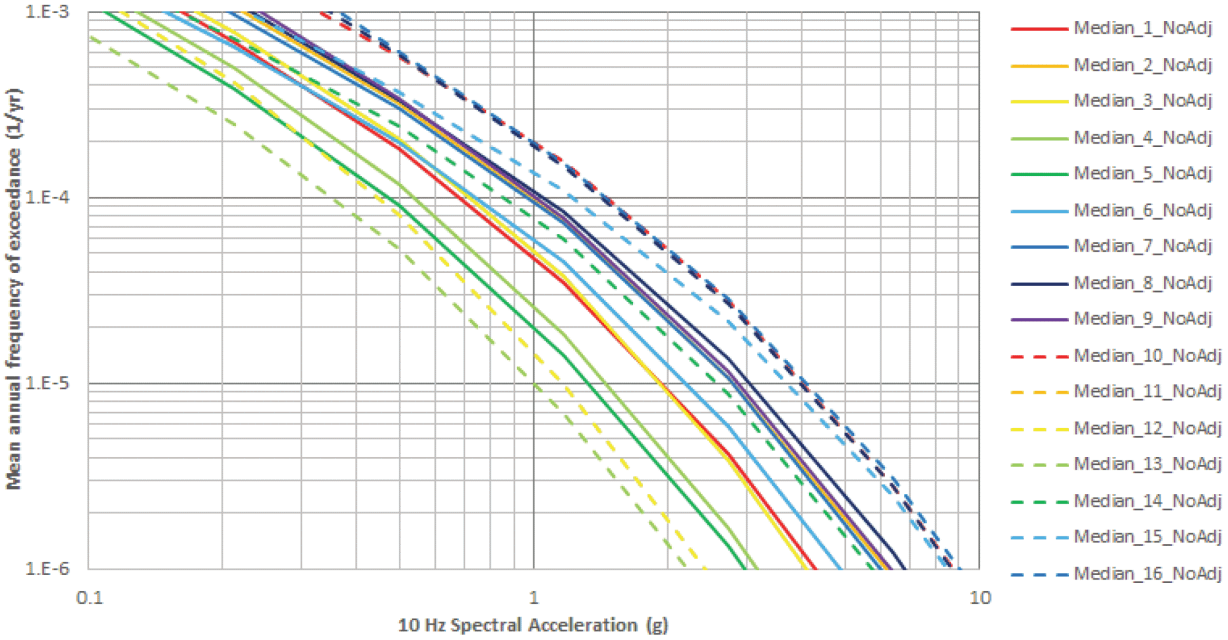


Figure 2.5.2-41R Unweighted Sensitivity to the 17 NGA-East Median Ground Motion Models at 10 Hz

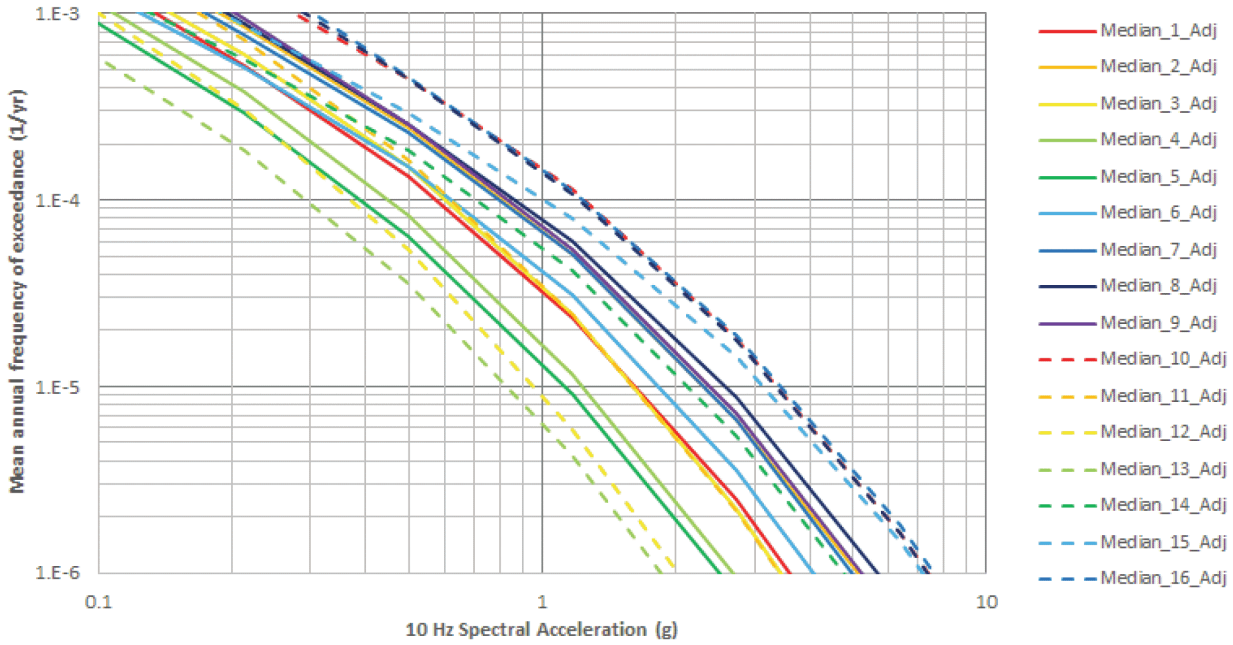


Figure 2.5.2-42R Unweighted Sensitivity to the 17 NGA-East Median Ground Motion Models Adjusted for Bias at 10 Hz

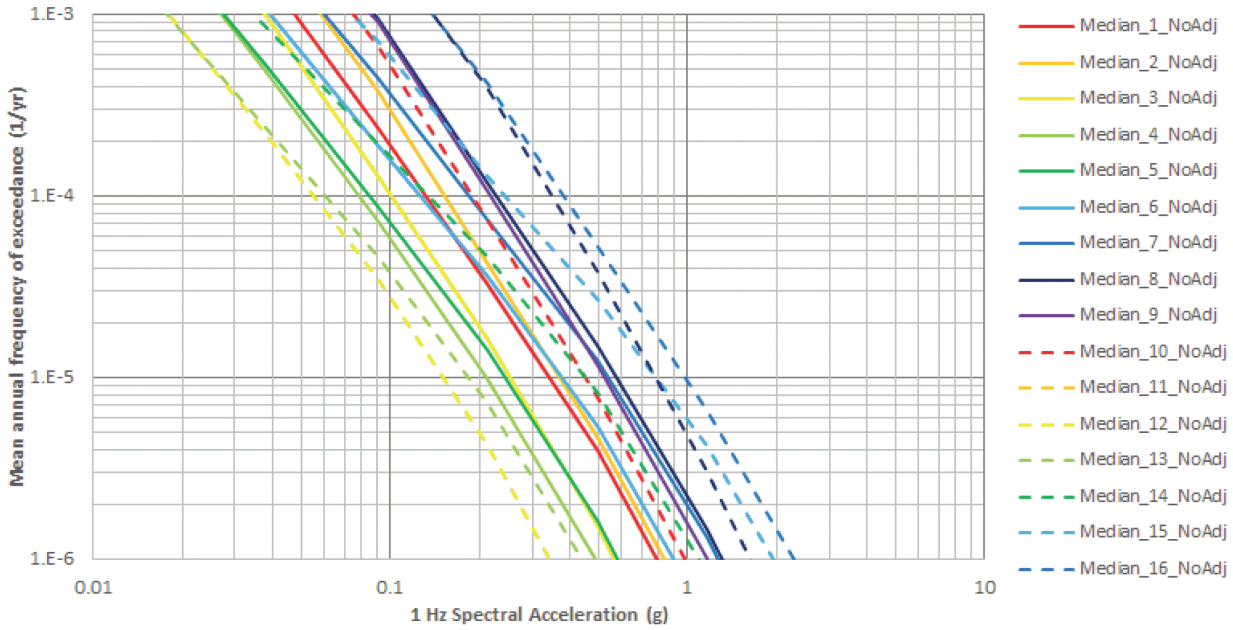


Figure 2.5.2-43R Unweighted Sensitivity to the 17 NGA-East Median Ground Motion Models at 1 Hz

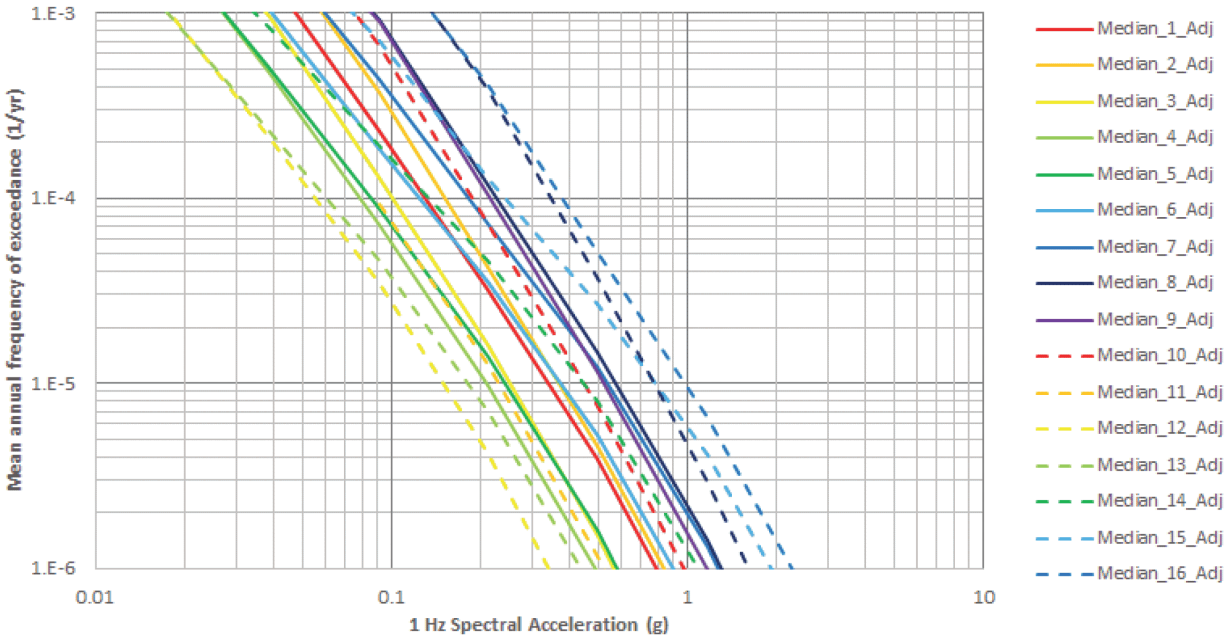


Figure 2.5.2-44R Unweighted Sensitivity to the 17 NGA-East Median Ground Motion Models Adjusted for Bias at 1 Hz

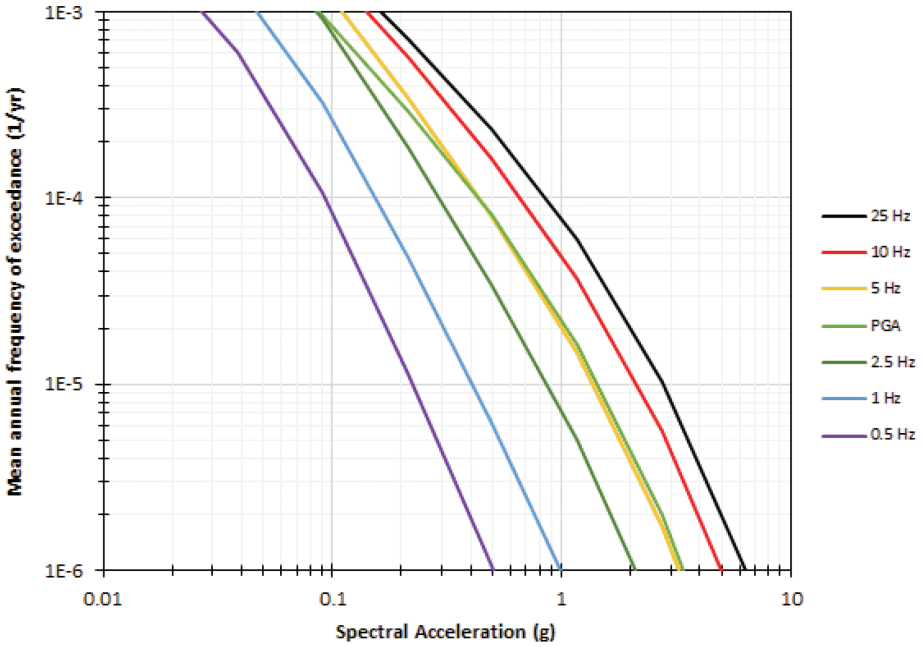


Figure 2.5.2-45R Mean Total Rock Hazard Curves for Seven Representative Frequencies

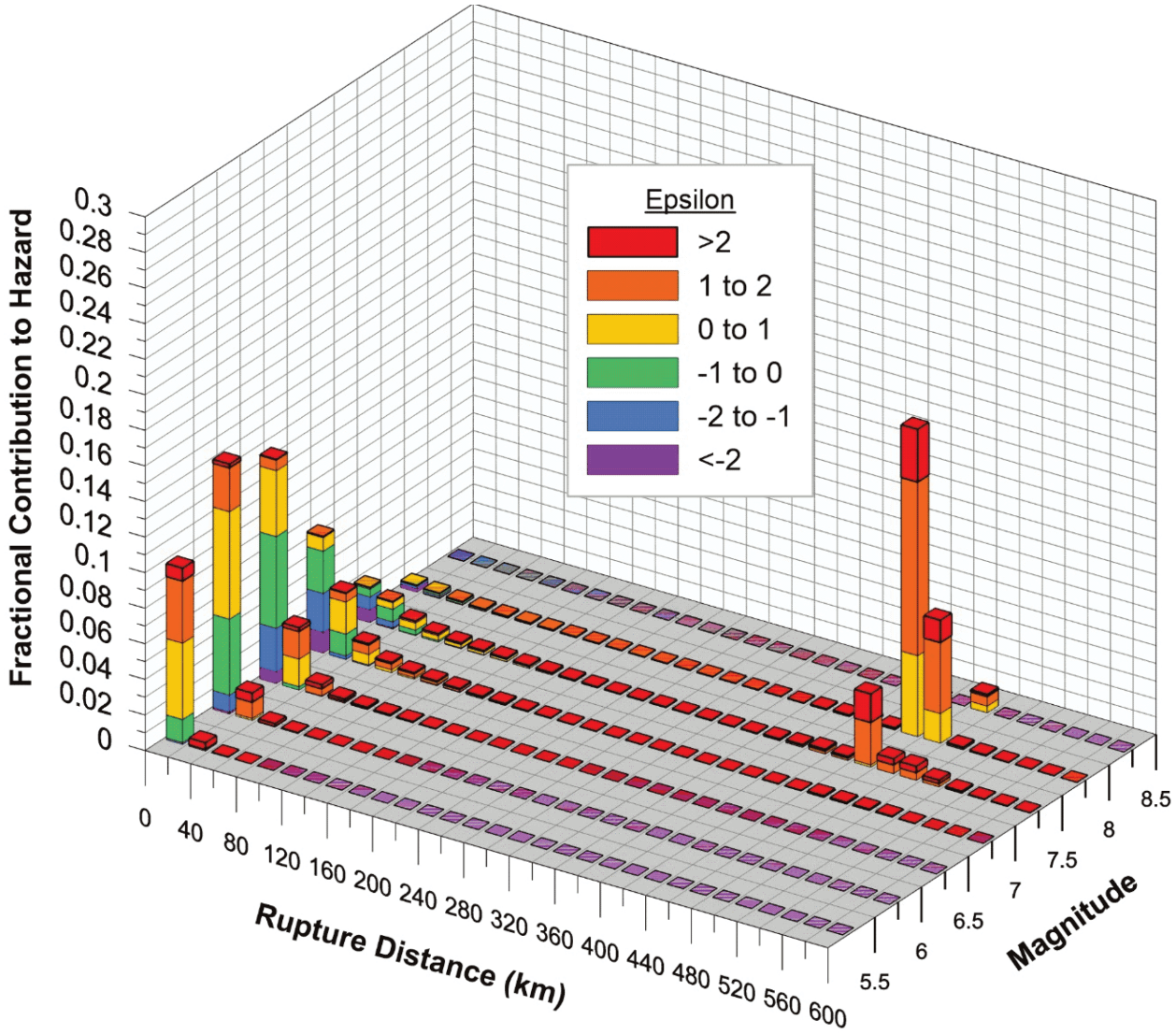


Figure 2.5.2-46R Combined Deaggregation of Mean Rock Hazard for 10^{-4} Magnitude-Distance-Epsilon Deaggregation for 1 and 2.5 Hz (Low Frequency)

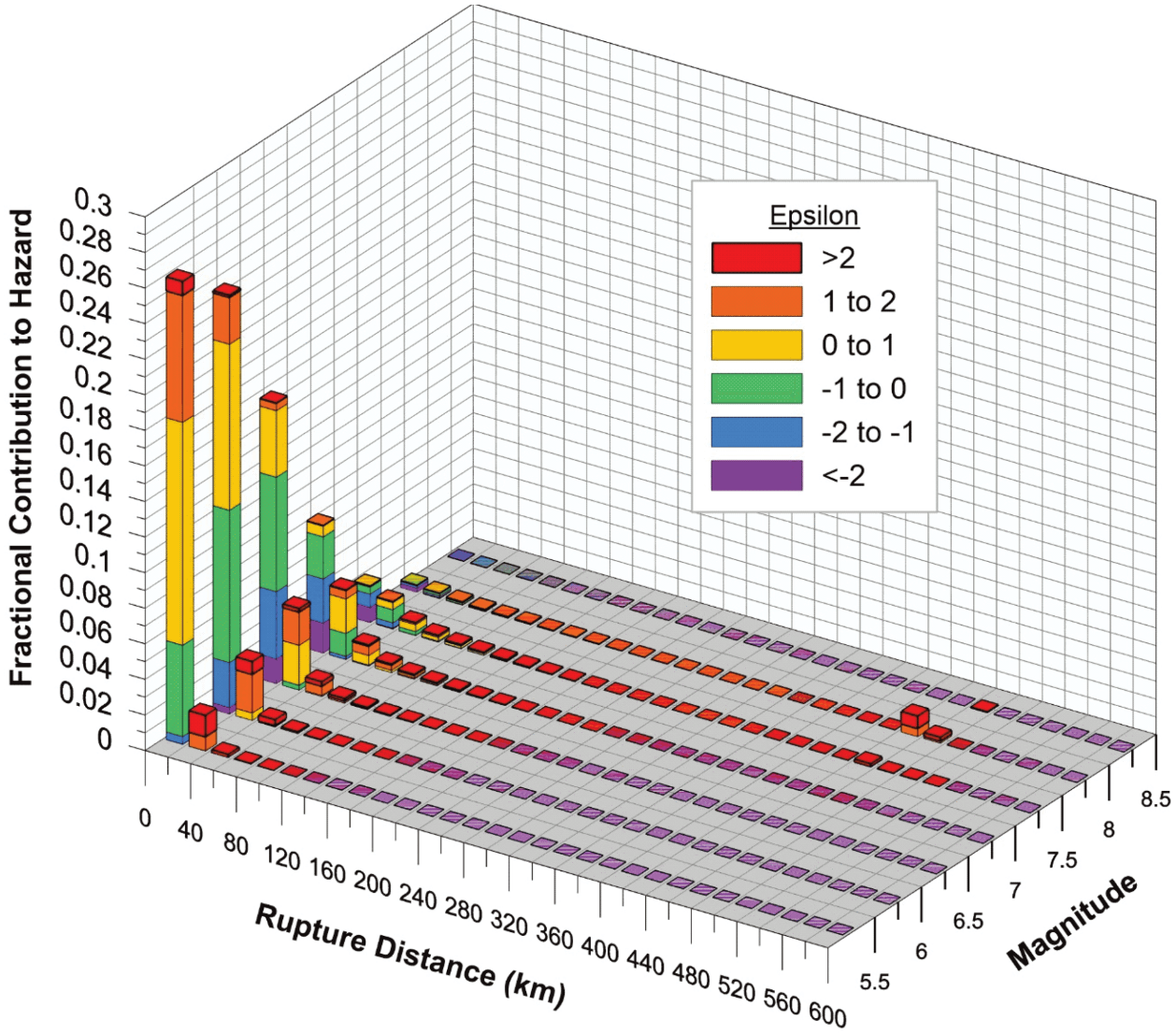


Figure 2.5.2-47R Combined Deaggregation of Mean Rock Hazard for 10^{-4} Magnitude-Distance-Epsilon Deaggregation for 5 and 10 Hz (High Frequency)

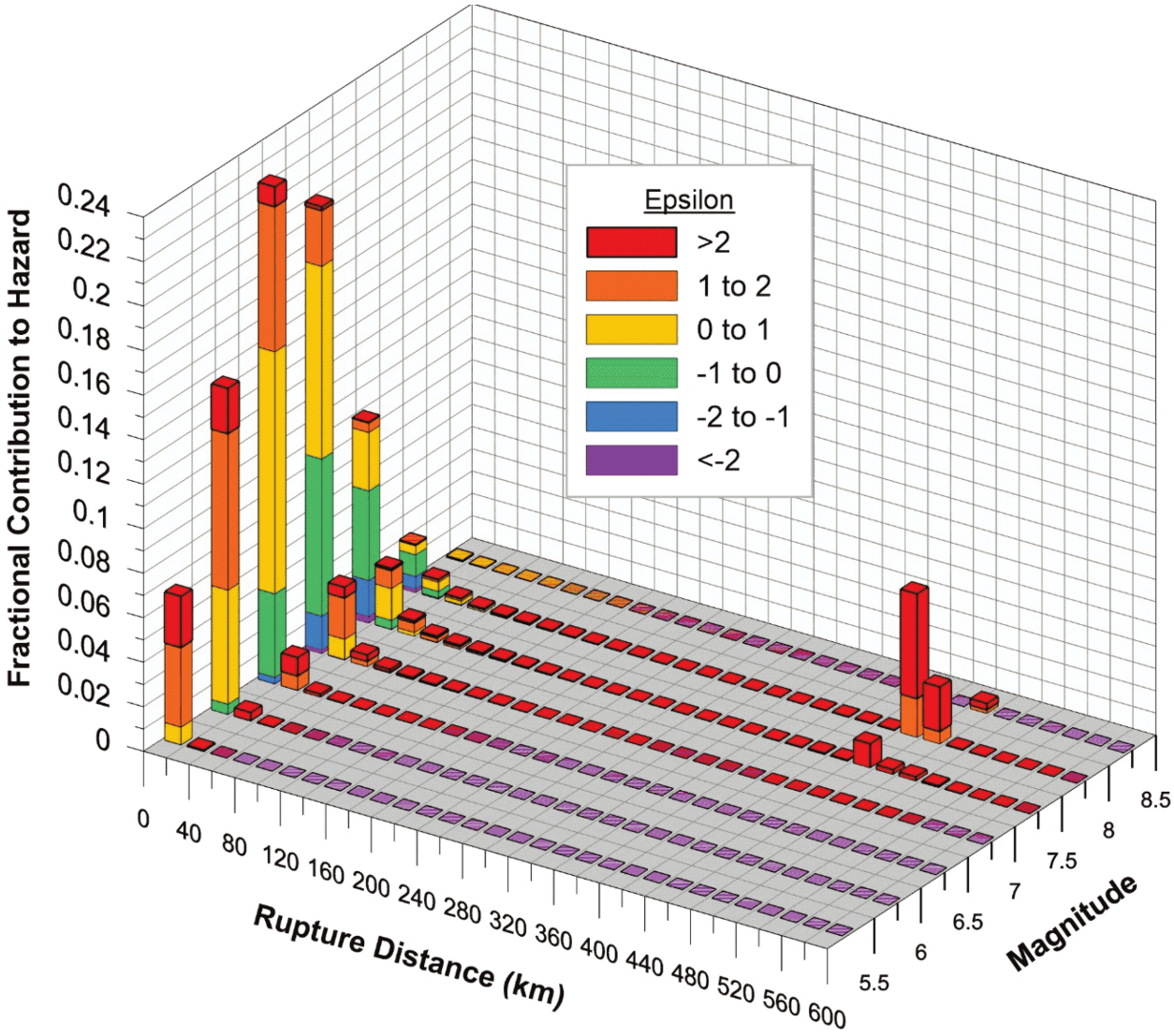


Figure 2.5.2-48R Combined Deaggregation of Mean Rock Hazard for 10^{-5} Magnitude-Distance-Epsilon Deaggregation for 1 and 2.5 Hz (Low Frequency)

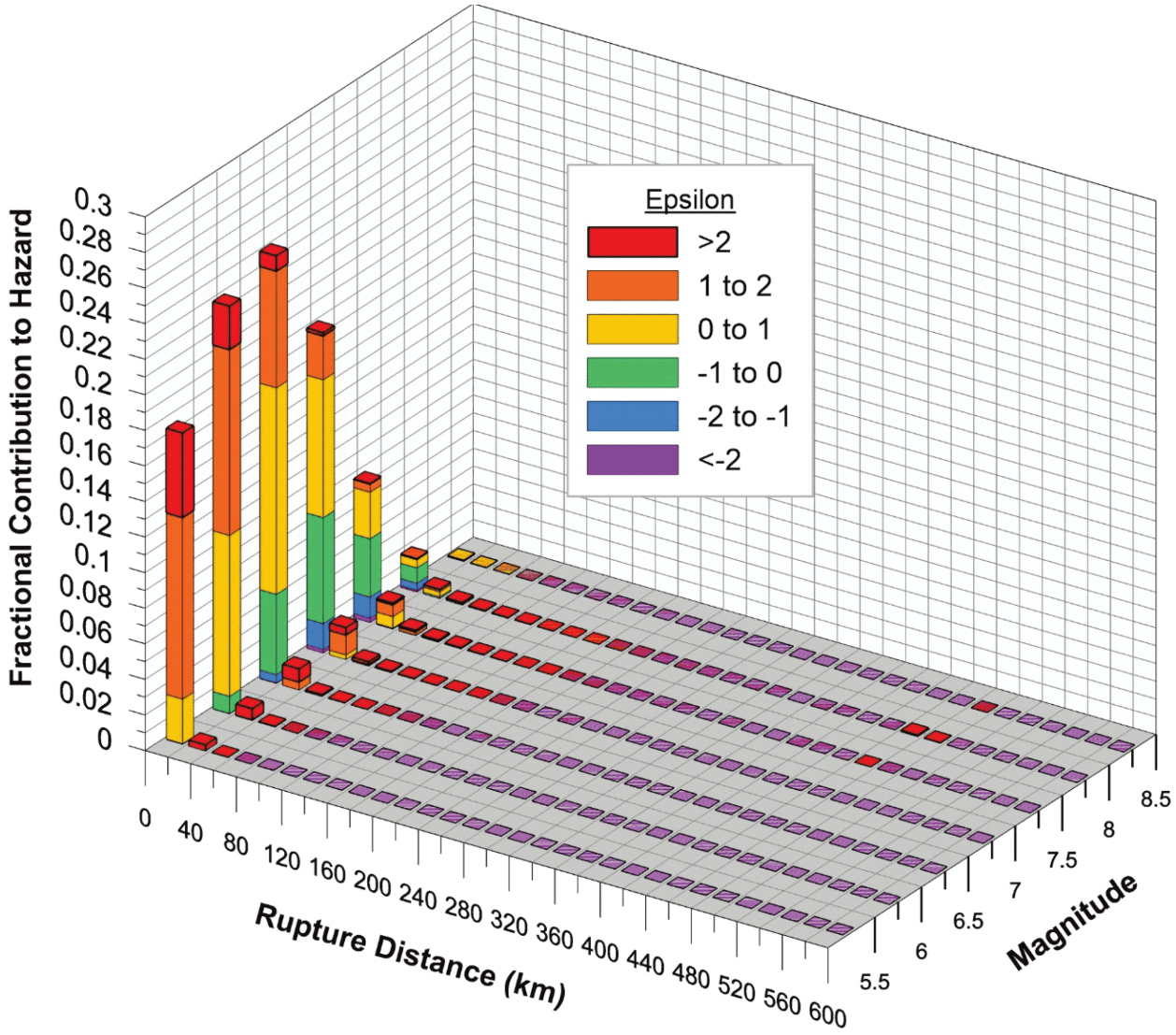


Figure 2.5.2-49R Combined Deaggregation of Mean Rock Hazard for 10^{-5} Magnitude-Distance-Epsilon Deaggregation for 5 and 10 Hz (High Frequency)

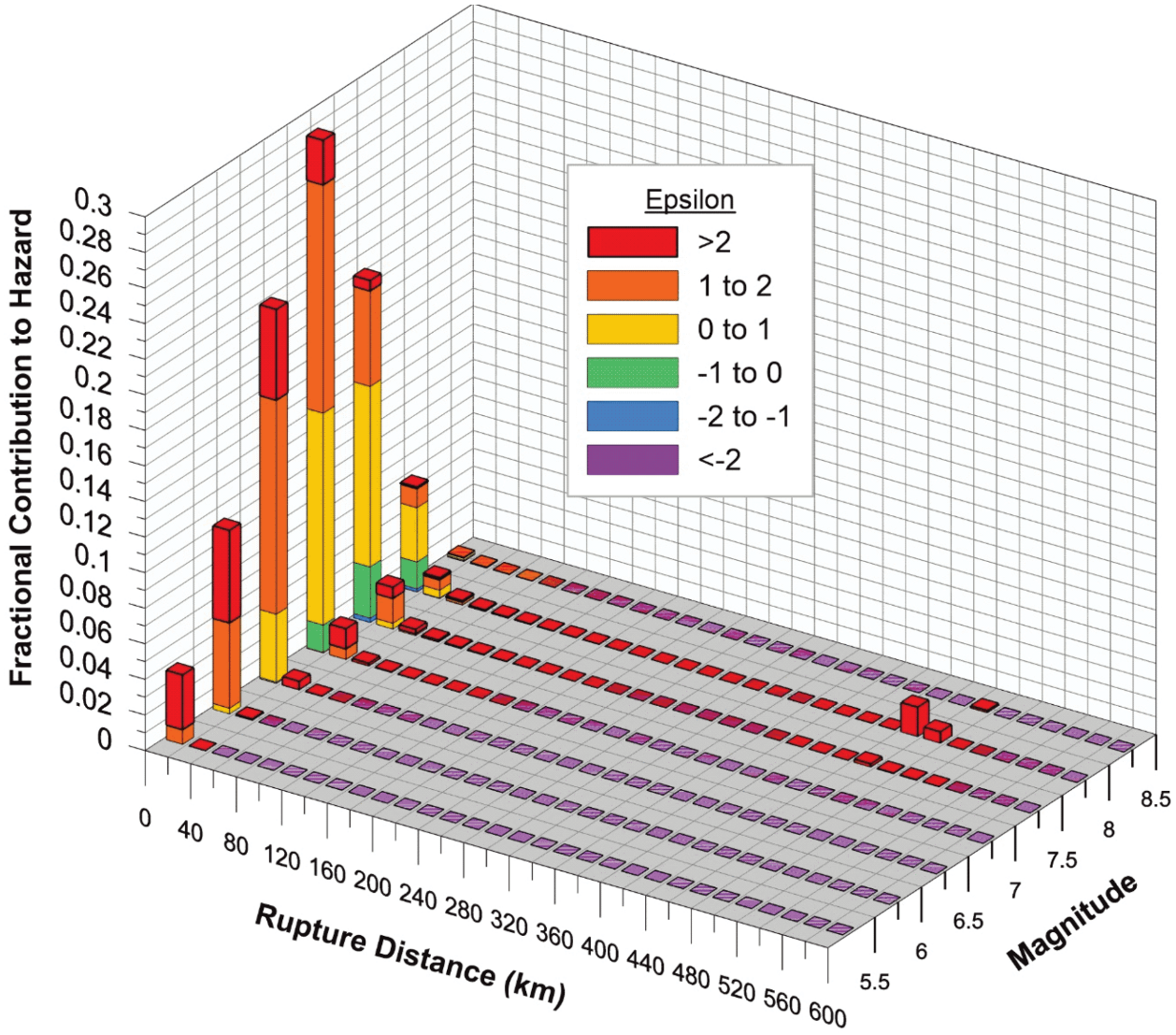


Figure 2.5.2-50R Combined Deaggregation of Mean Rock Hazard for 10^{-6} Magnitude-Distance-Epsilon Deaggregation for 1 and 2.5 Hz (Low Frequency)

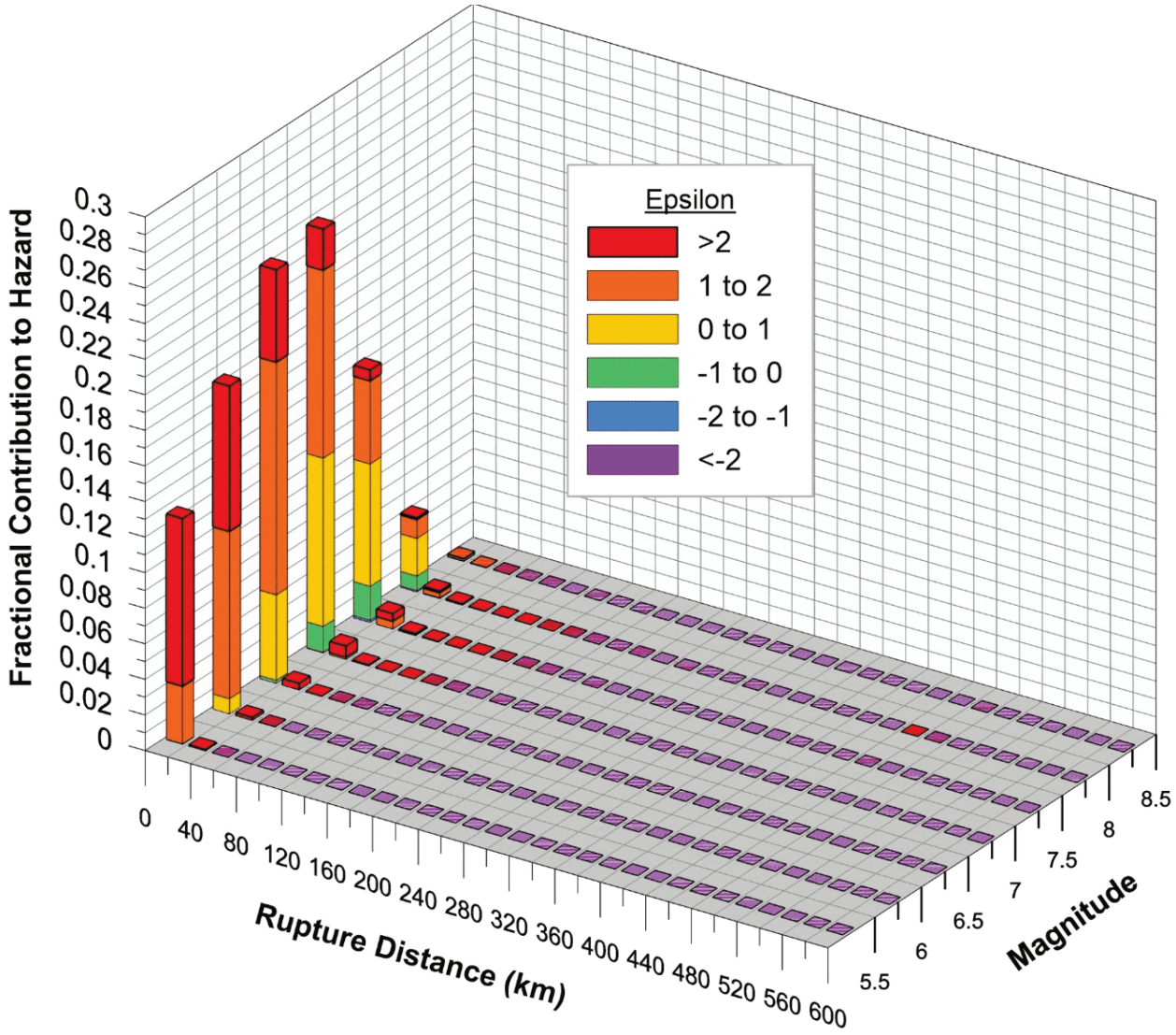


Figure 2.5.2-51R Combined Deaggregation of Mean Rock Hazard for 10^{-6} Magnitude-Distance-Epsilon Deaggregation for 5 and 10 Hz (High Frequency)

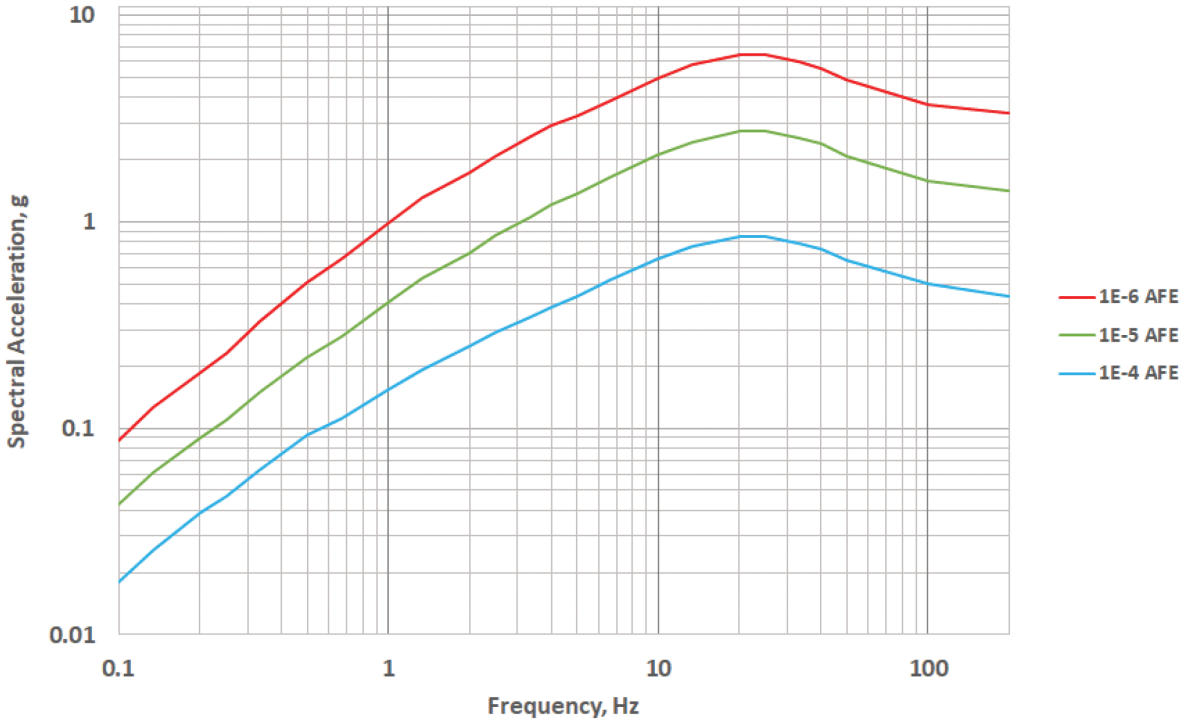


Figure 2.5.2-52R Mean Rock Uniform Hazard Response Spectra for Mean Annual Frequencies of Exceedance of 10^{-4}, 10^{-5}, and 10^{-6}

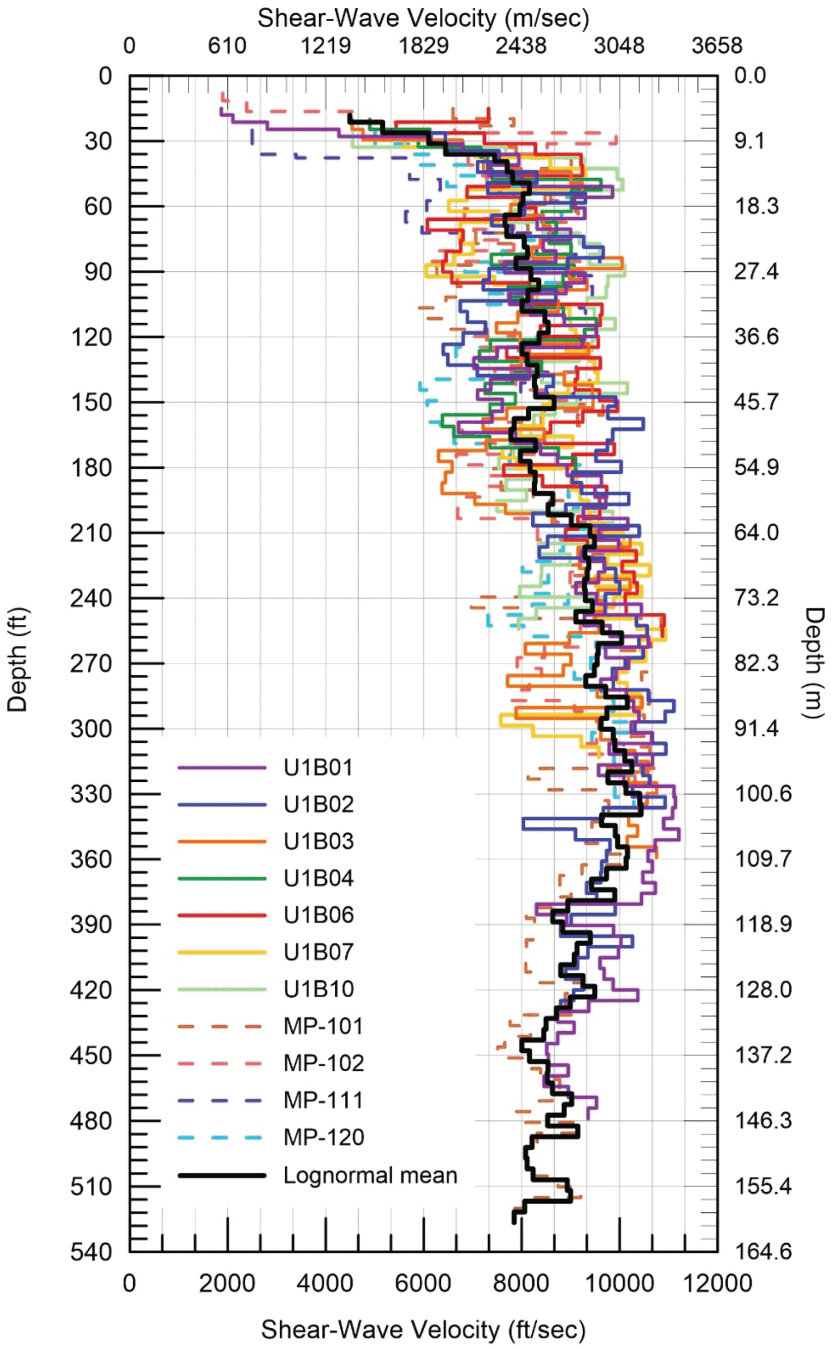


Figure 2.5.2-53R Shear-Wave Velocity Profiles from Suspension Logging

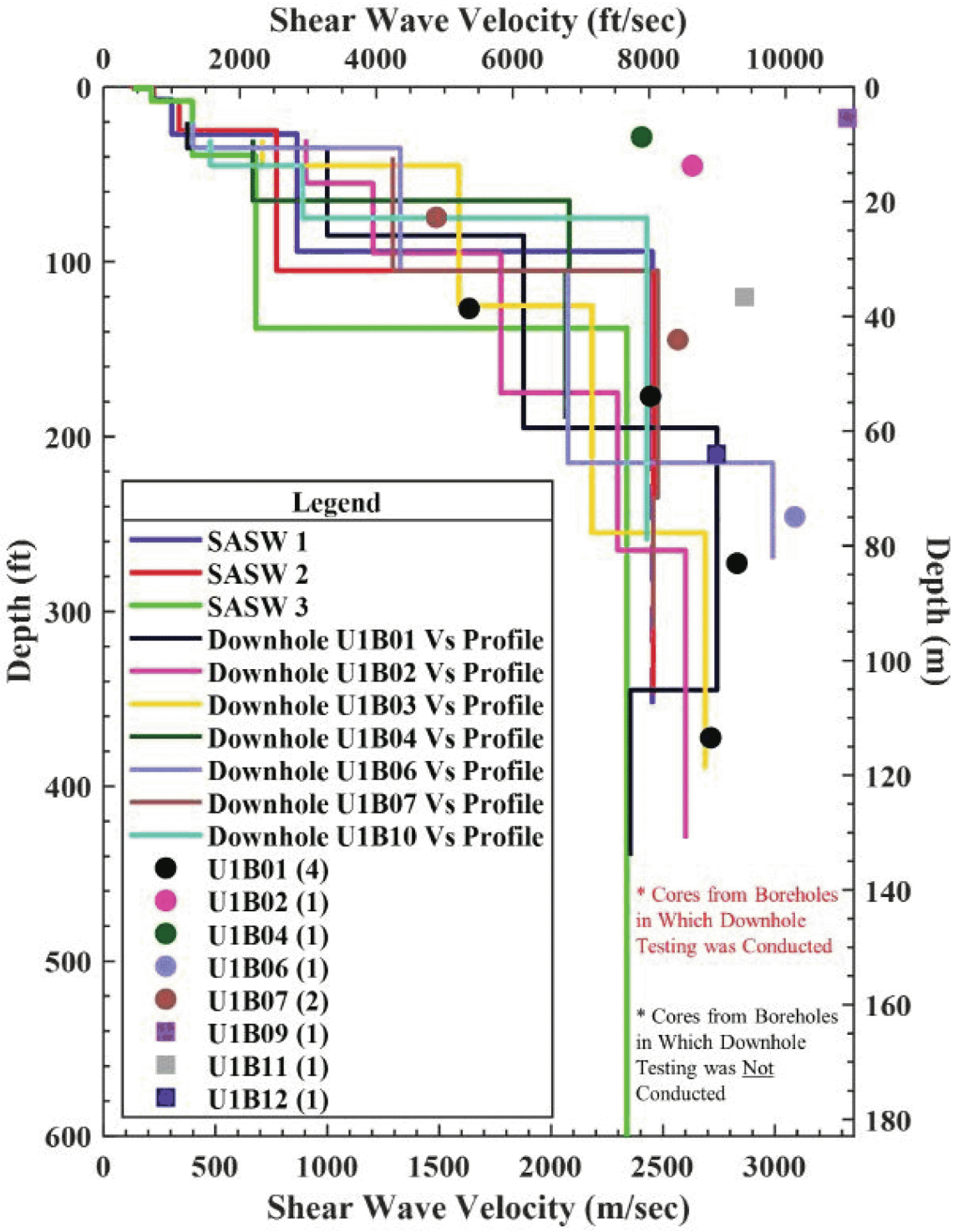


Figure 2.5.2-54R Shear Wave Velocity Profiles from Downhole and SASW
(Reference 2.5.2-249**)**

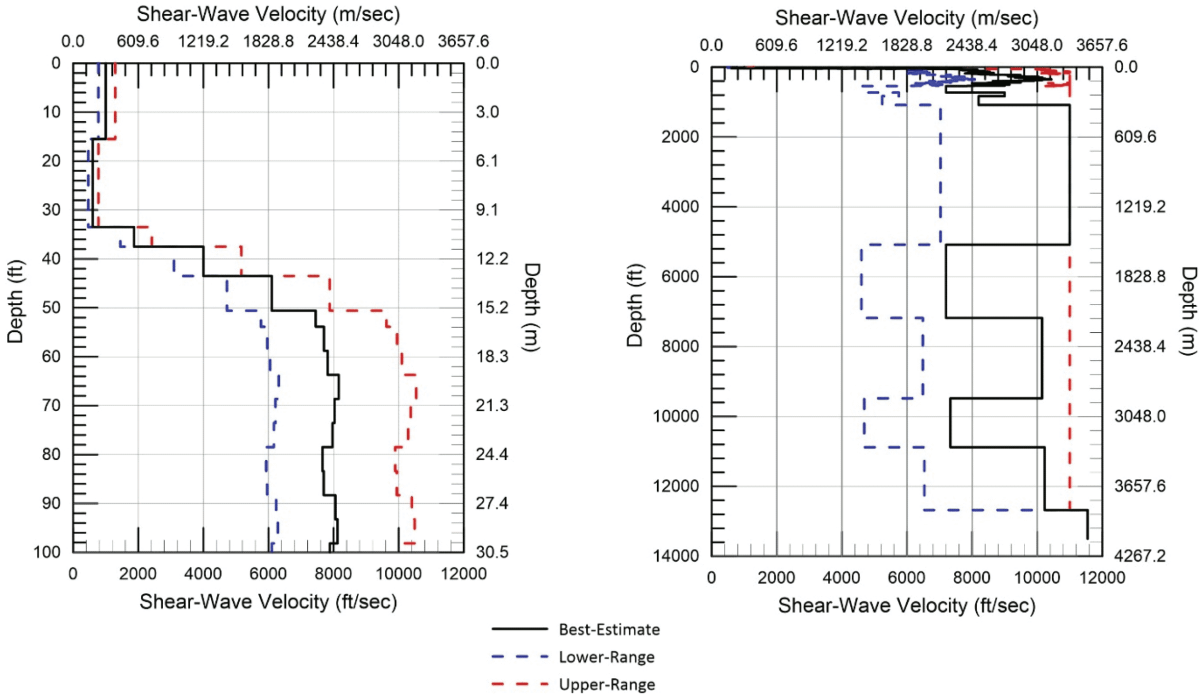


Figure 2.5.2-55R Shallow and Deep Shear Wave Velocity Profiles

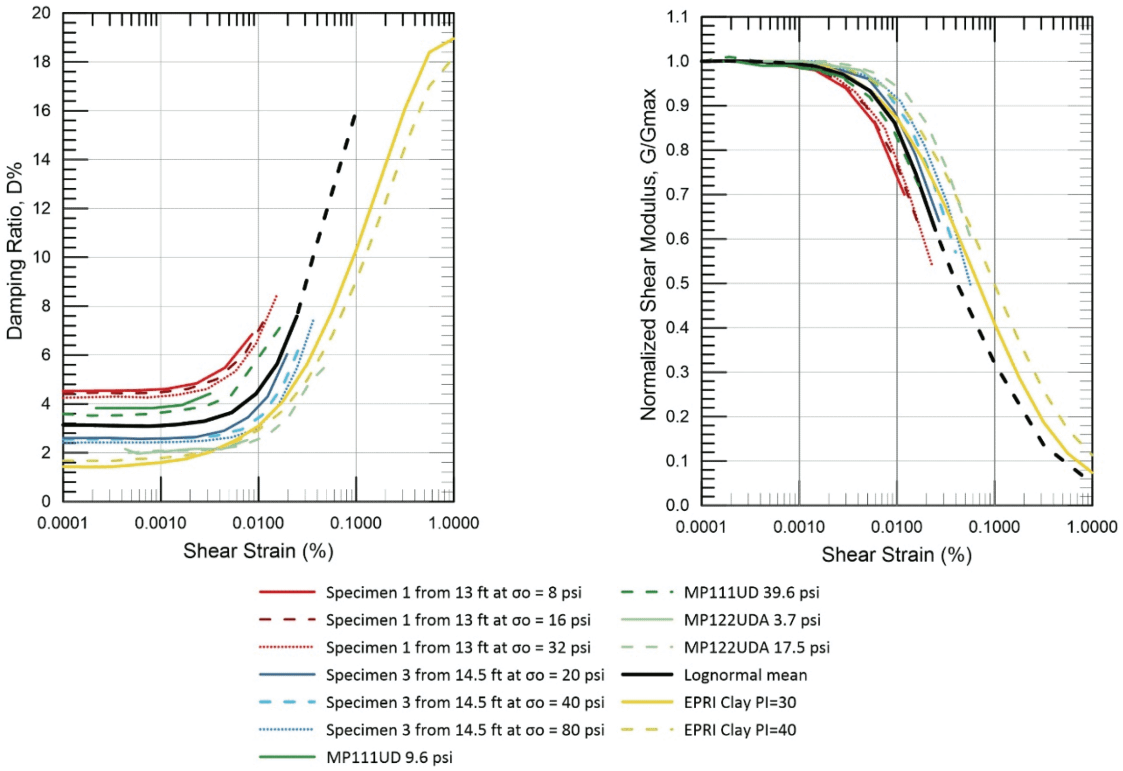


Figure 2.5.2-56R Comparison of Modulus Reduction and Damping Ratio Curves for Clinch River Soil

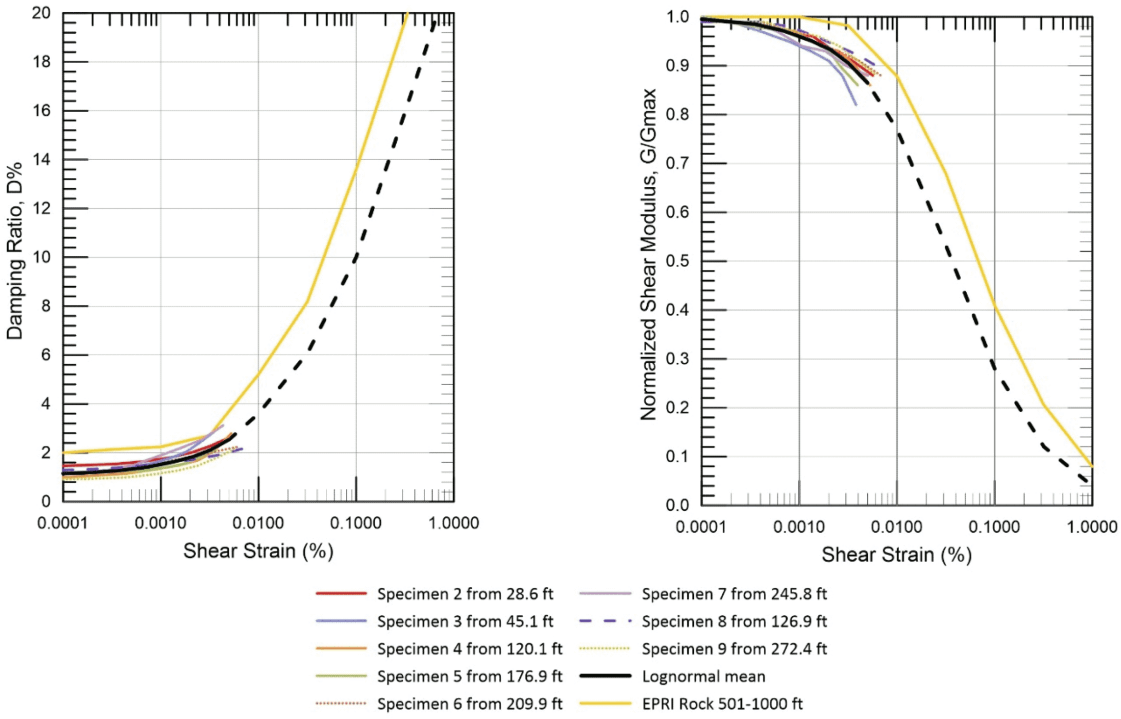


Figure 2.5.2-57R Comparison of Modulus Reduction and Damping Ratio Curves for Clinch River Rock

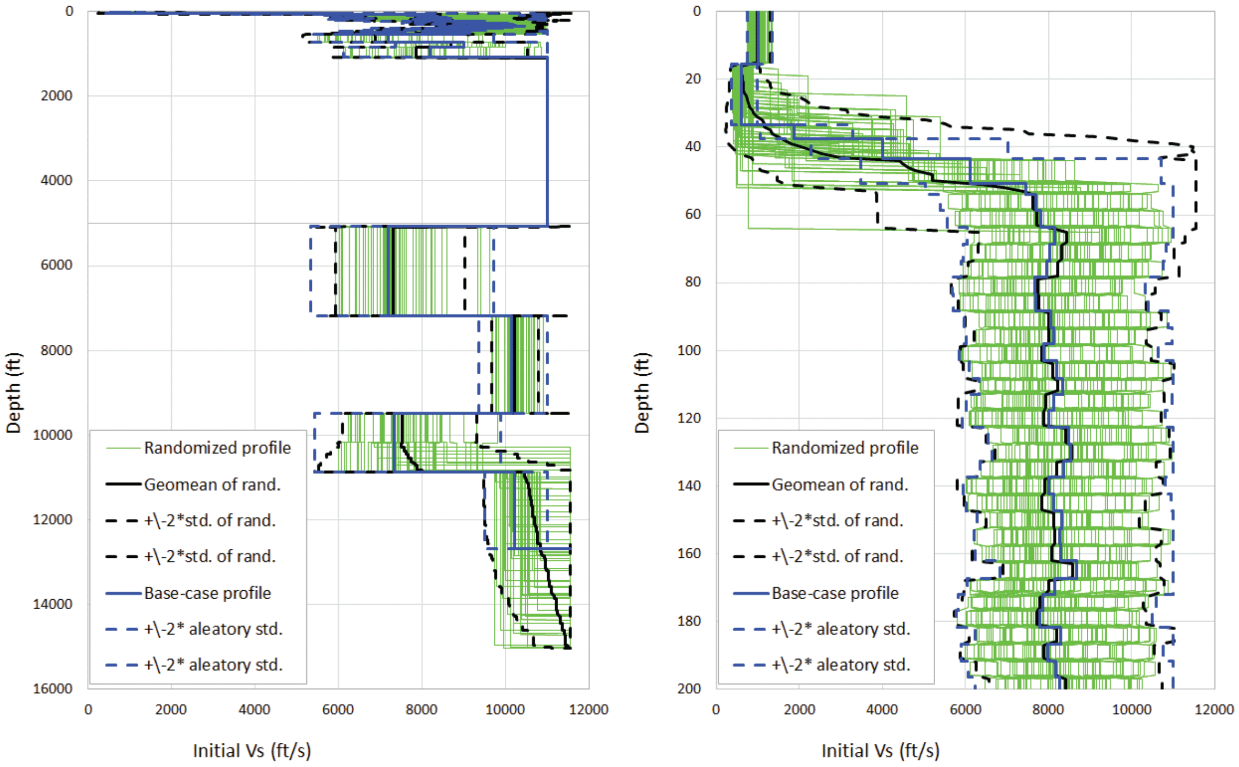


Figure 2.5.2-58R Randomization of Best-Estimate Profile for PBSRS and FIRS

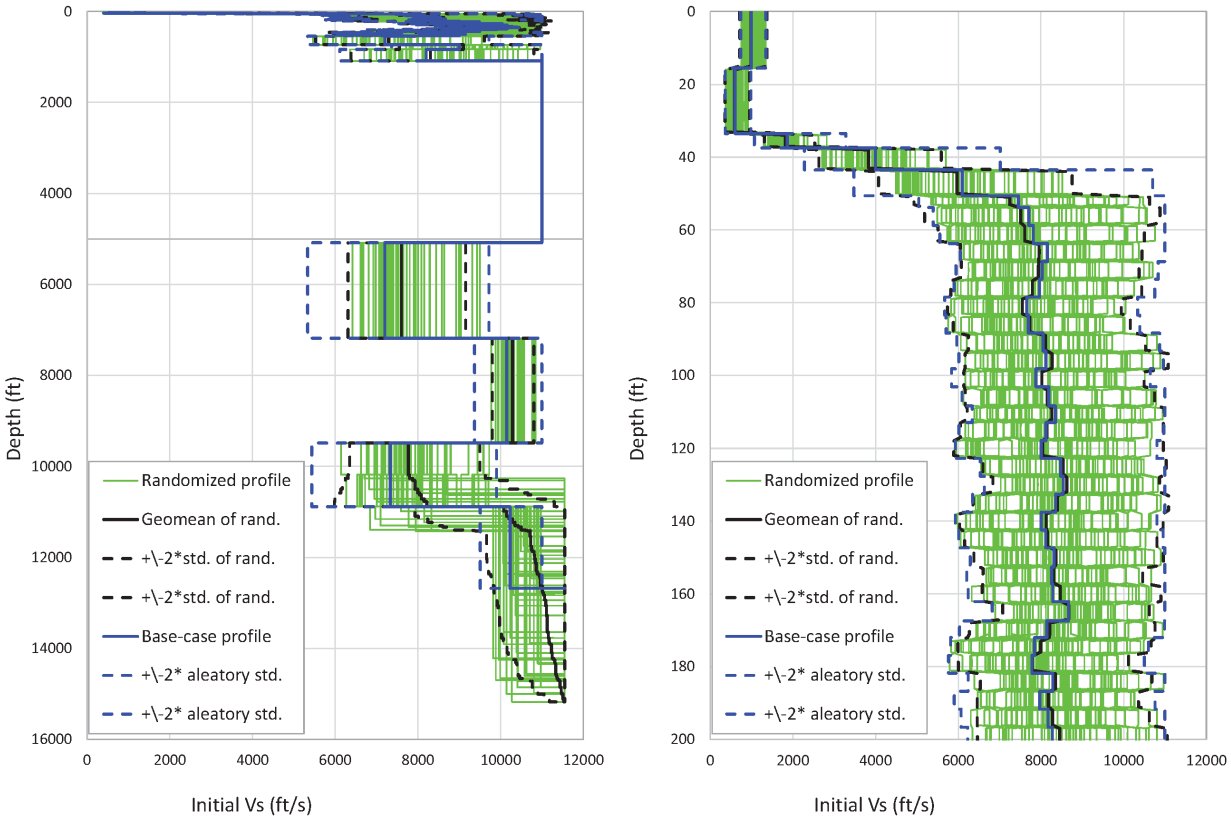


Figure 2.5.2-59R Randomization of Best-Estimate Profile for PBIRS

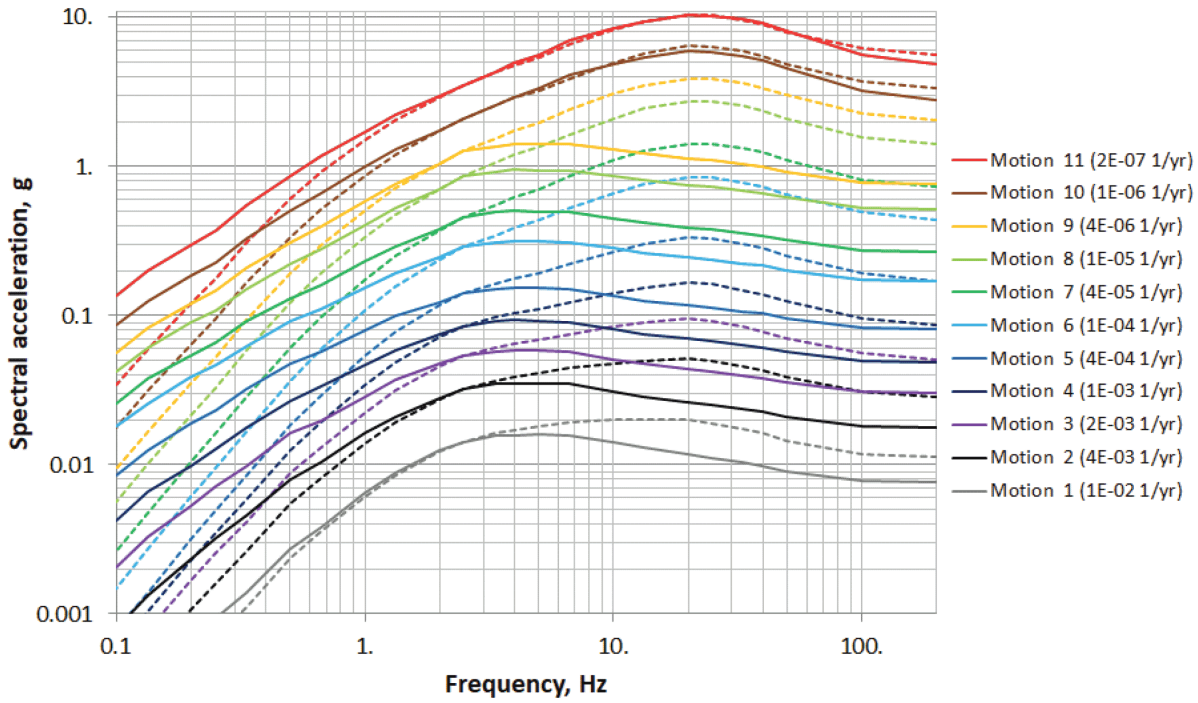


Figure 2.5.2-60R High and Low Frequency Input Response Spectra for Eleven Mean Annual Frequencies of Exceedance

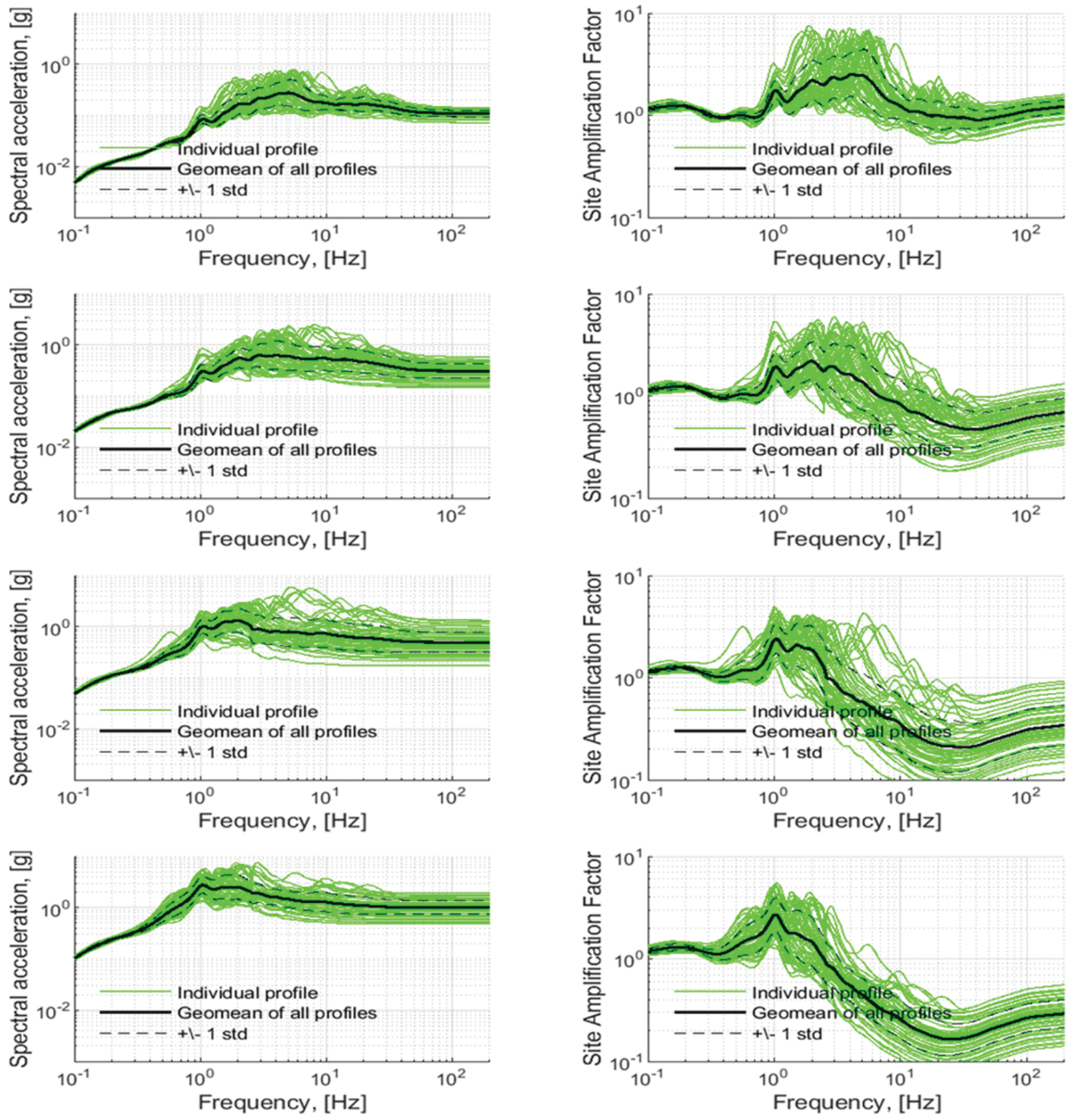


Figure 2.5.2-61R Amplified Response Spectra (Left) and Site Amplification Factors (Right) for the Base-Case Profile and Material Model 1 at AFEs of 1E-3 (First Row), 1E-4 (Second Row), 1E-5 (Third Row), and 1E-6 (Fourth Row) for PBSRS

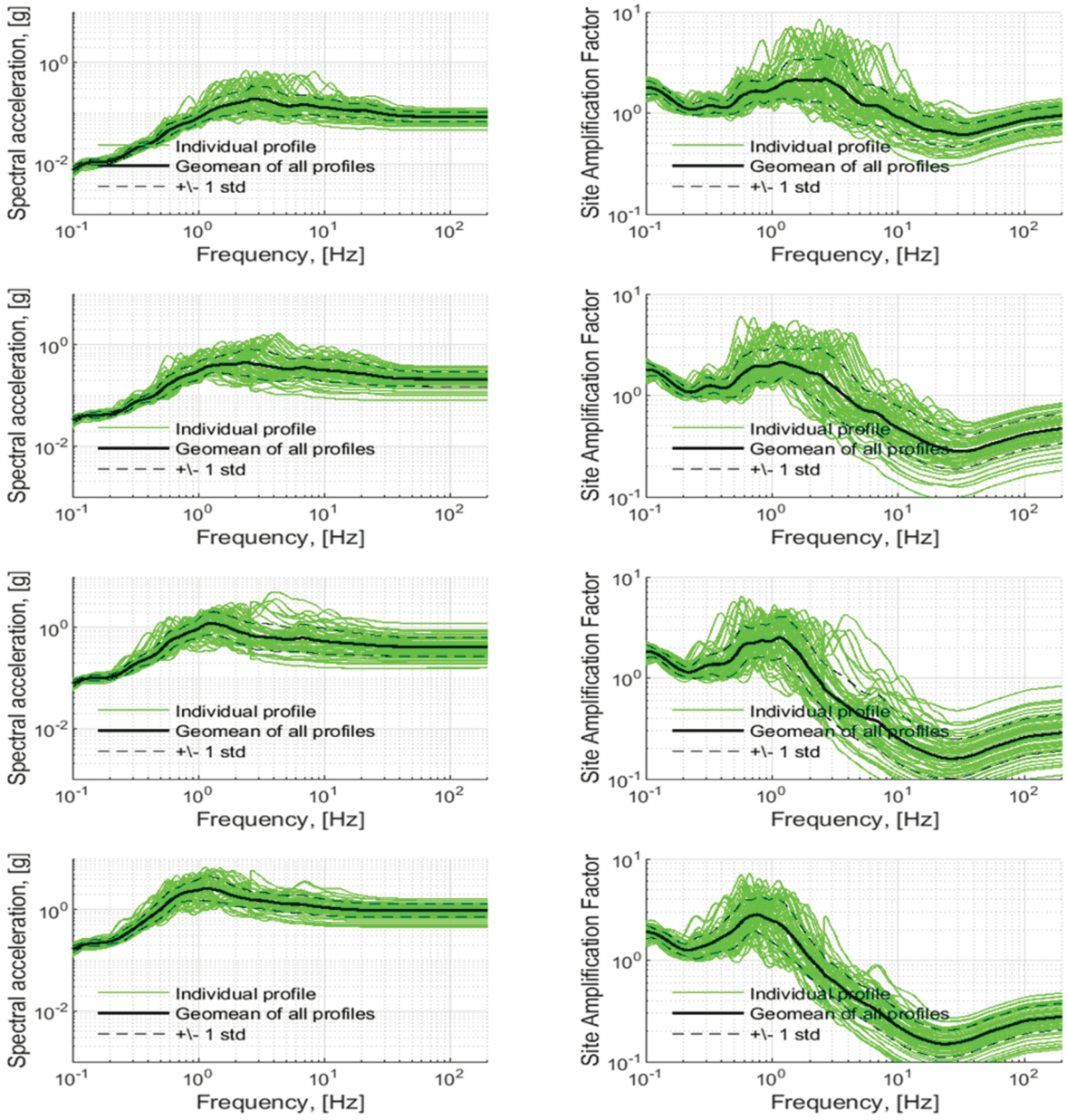


Figure 2.5.2-62R Amplified Response Spectra (Left) and Site Amplification Factors (Right) for the Lower-Range Profile and Material Model 1 at AFEs of 1E-3 (First Row), 1E-4 (Second Row), 1E-5 (Third Row), and 1E-6 (Fourth Row) for PBSRS

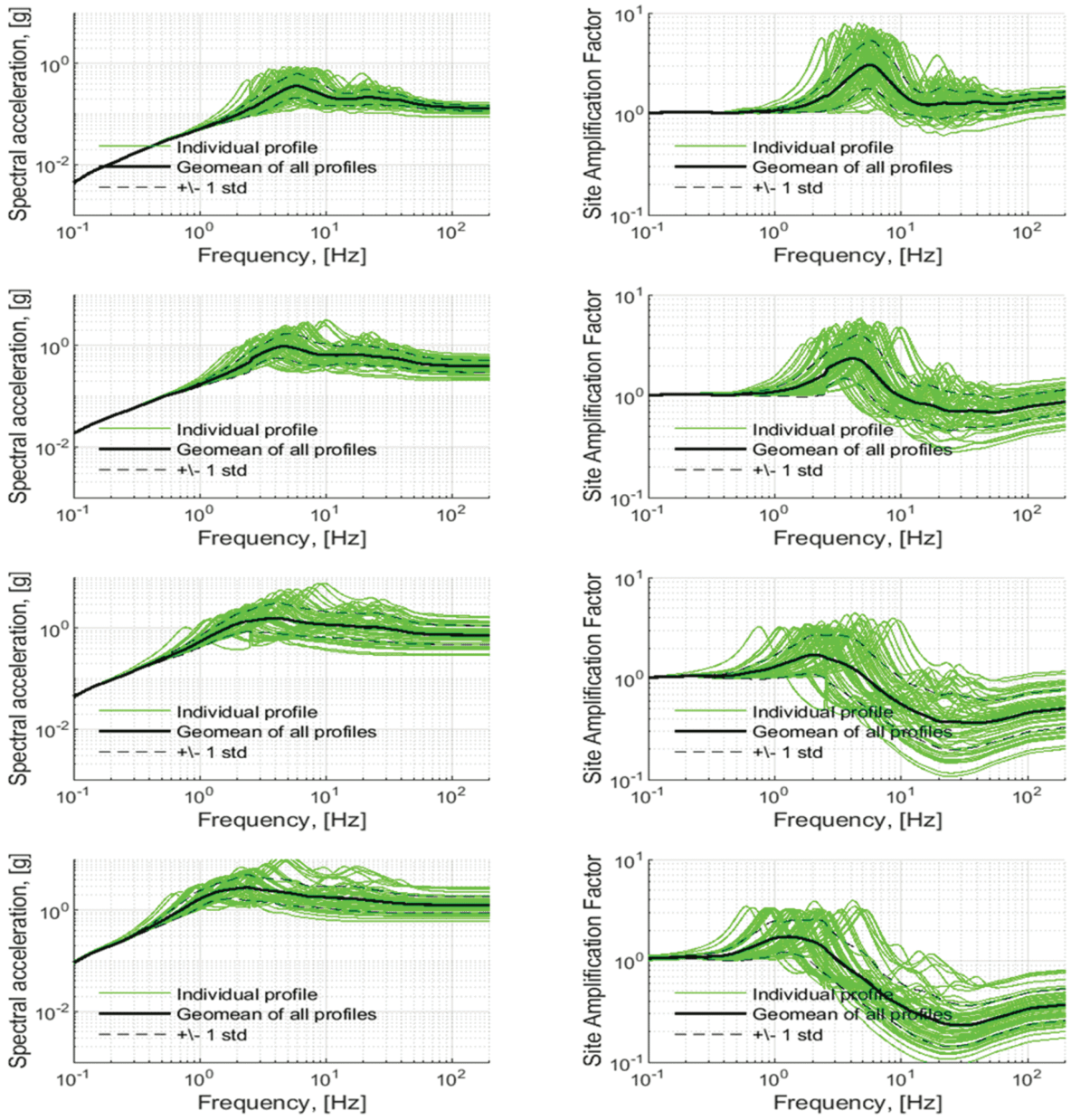


Figure 2.5.2-63R Amplified Response Spectra (Left) and Site Amplification Factors (Right) for the Upper-Range Profile and Material Model 1 at AFEs of 1E-3 (First Row), 1E-4 (Second Row), 1E-5 (Third Row), and 1E-6 (Fourth Row) for PBSRS

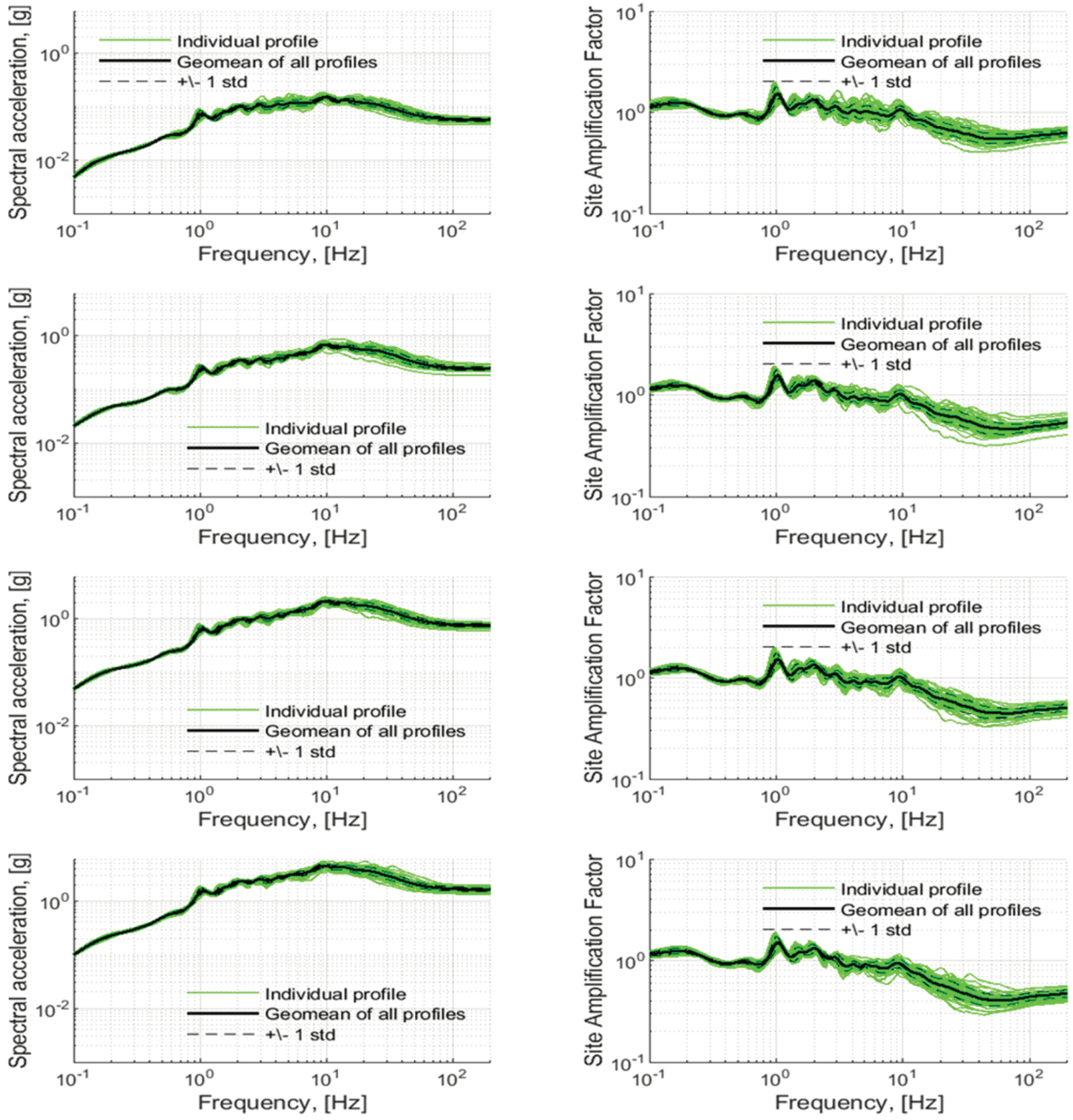


Figure 2.5.2-64R Amplified Response Spectra (Left) and Site Amplification Factors (Right) for the Base-Case Profile and Material Model 1 at AFEs of 1E-3 (First Row), 1E-4 (Second Row), 1E-5 (Third Row), and 1E-6 (Fourth Row) for PBIRS

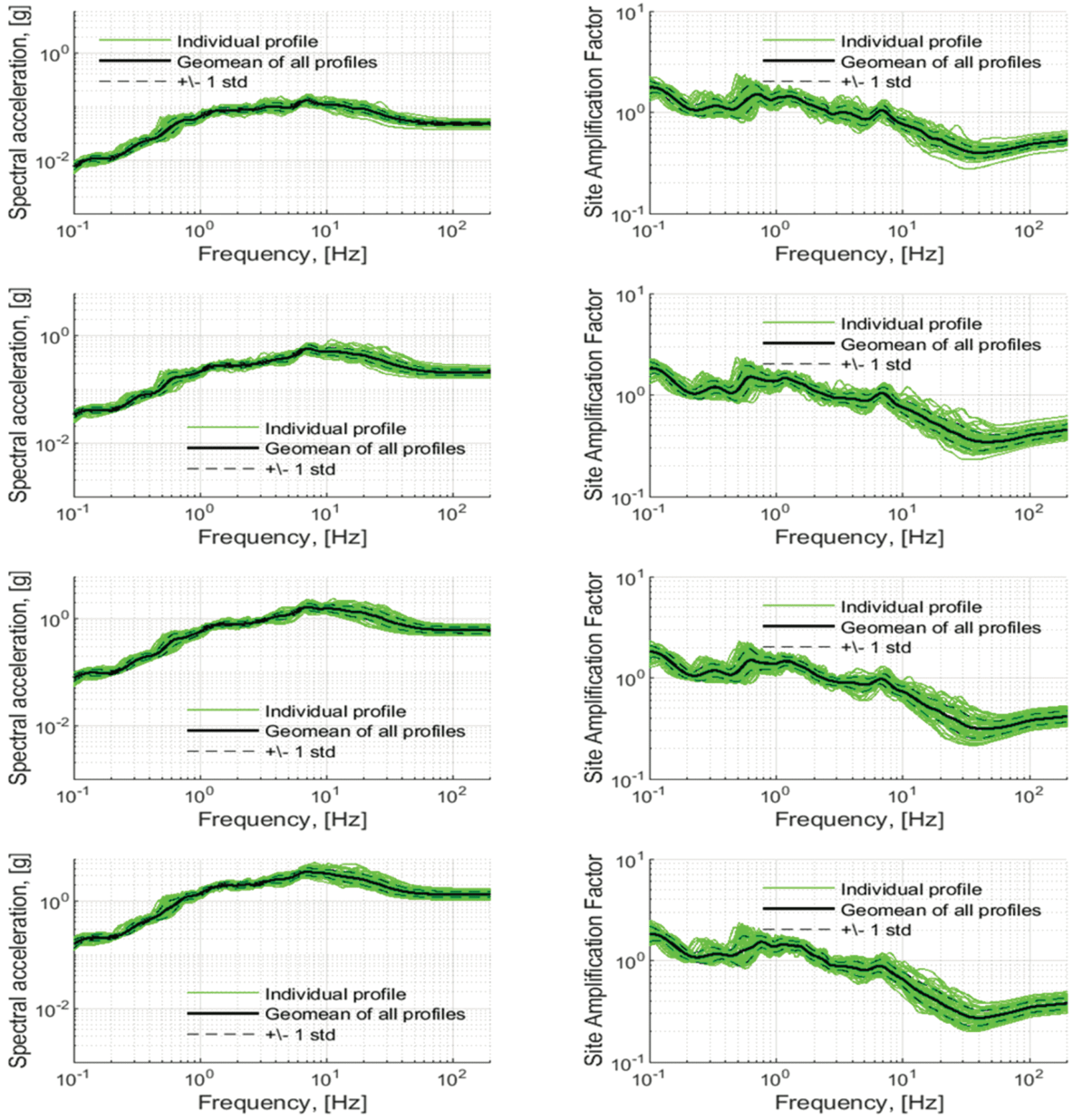


Figure 2.5.2-65R Amplified Response Spectra (Left) and Site Amplification Factors (Right) for the Lower-Range Profile and Material Model 1 at AFEs of 1E-3 (First Row), 1E-4 (Second Row), 1E-5 (Third Row), and 1E-6 (Fourth Row) for PBIRS

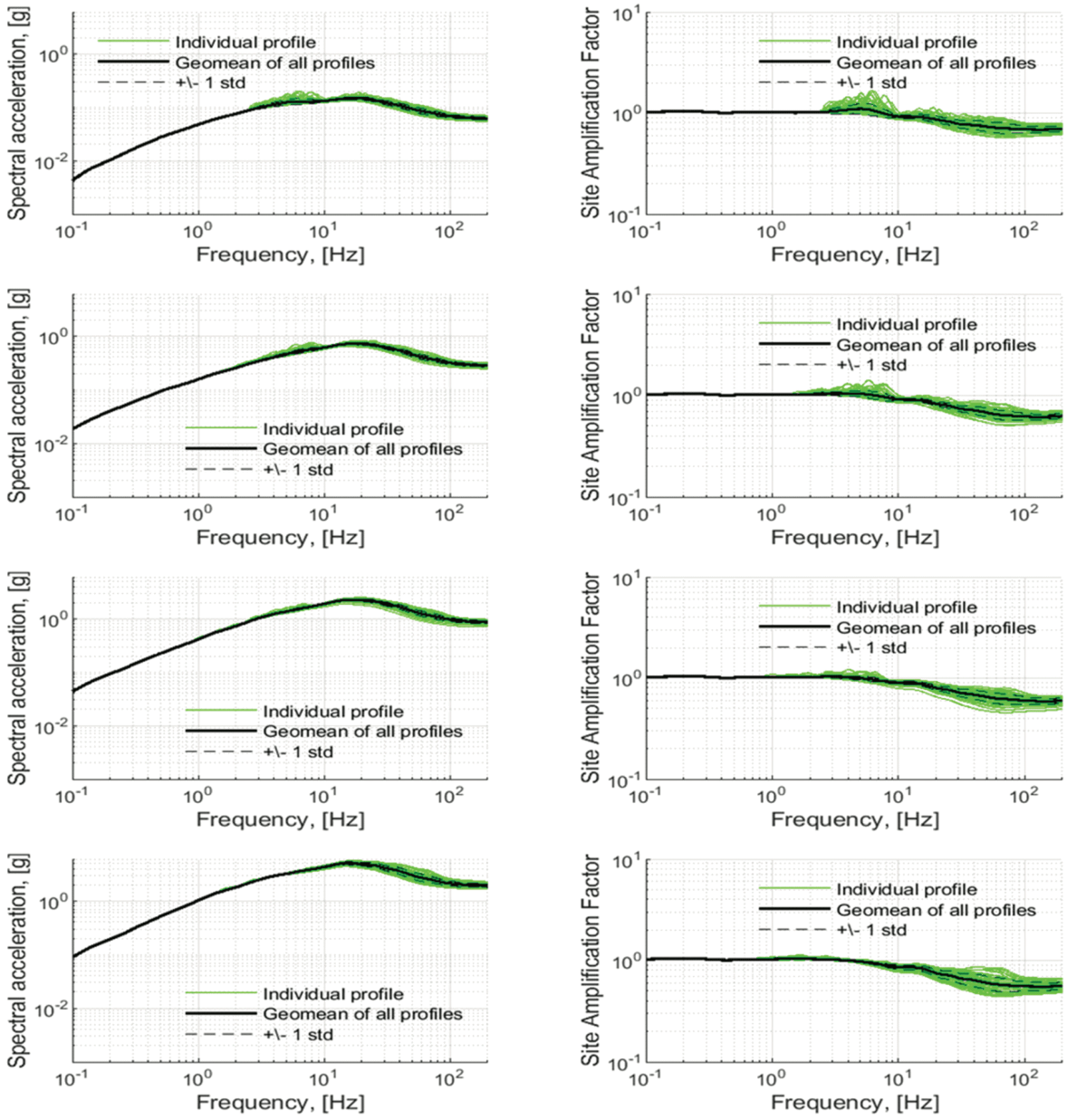


Figure 2.5.2-66R Amplified Response Spectra (Left) and Site Amplification Factors (Right) for the Upper-Range Profile and Material Model 1 at AFEs of 1E-3 (First Row), 1E-4 (Second Row), 1E-5 (Third Row), and 1E-6 (Fourth Row) for PBIRS

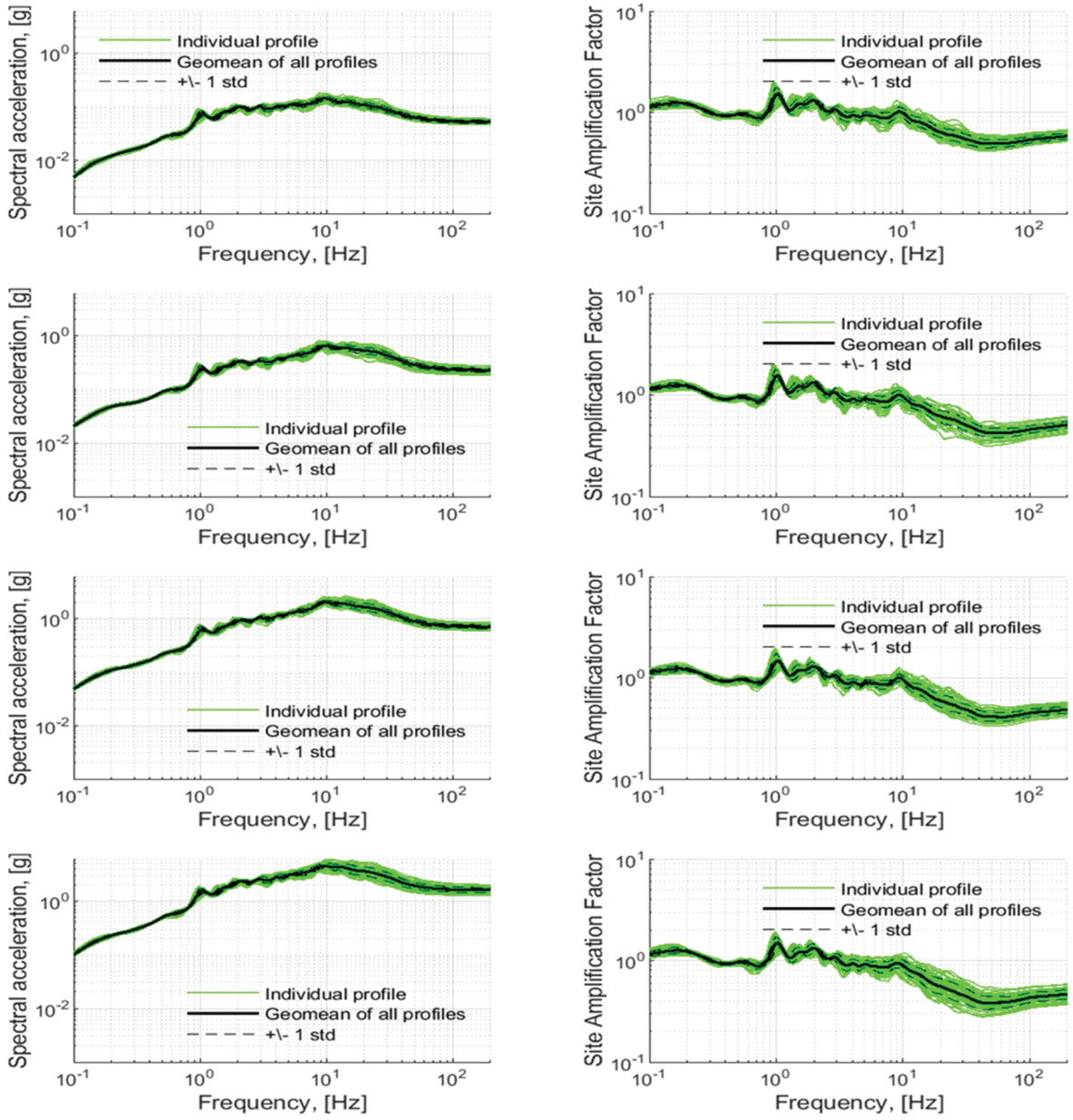


Figure 2.5.2-67R Amplified Response Spectra (Left) and Site Amplification Factors (Right) for the Base-Case Profile and Material Model 1 at AFEs of 1E-3 (First Row), 1E-4 (Second Row), 1E-5 (Third Row), and 1E-6 (Fourth Row) for FIRS

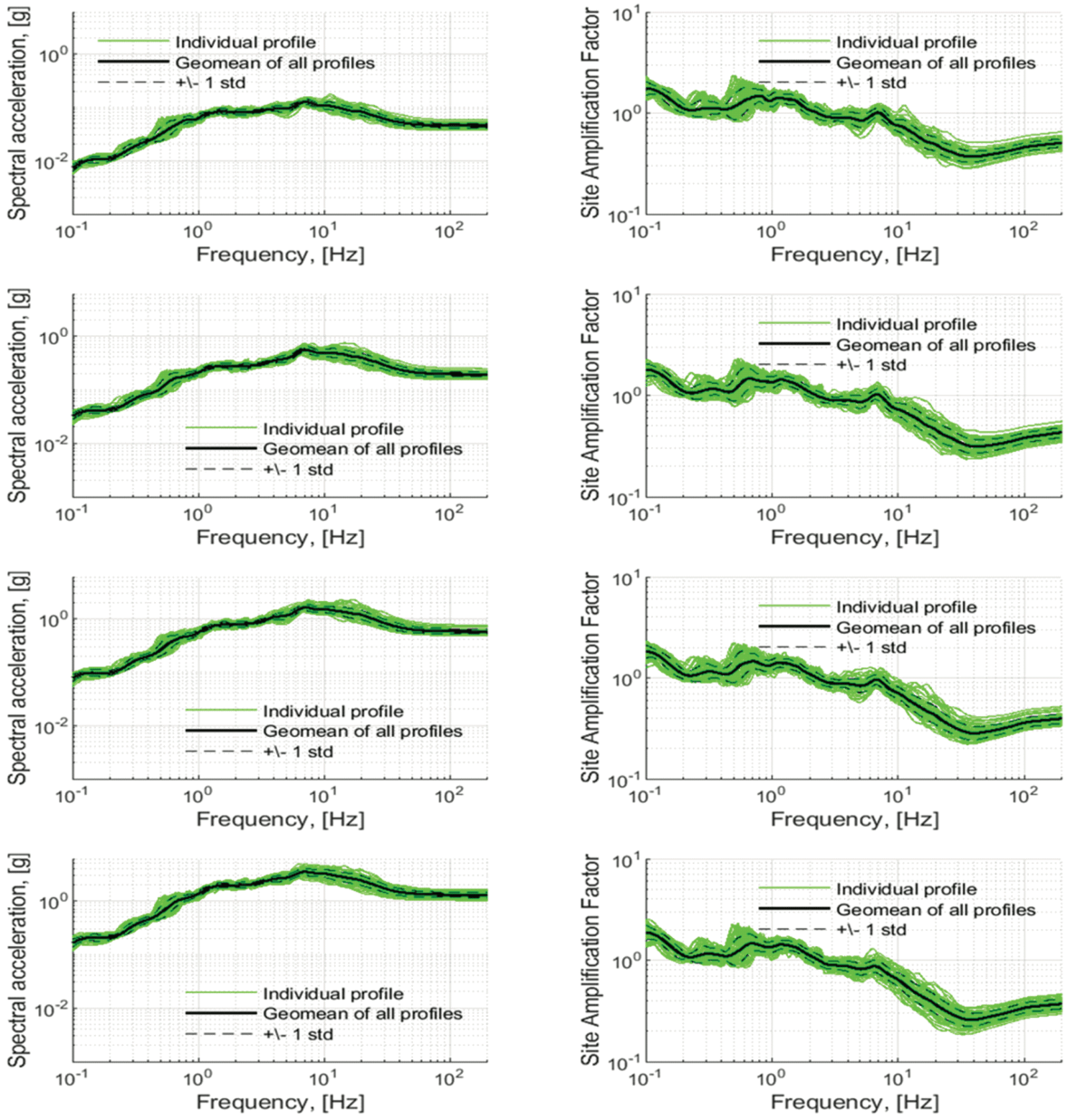


Figure 2.5.2-68R Amplified Response Spectra (Left) and Site Amplification Factors (Right) for the Lower-Range Profile and Material Model 1 at AFEs of 1E-3 (First Row), 1E-4 (Second Row), 1E-5 (Third Row), and 1E-6 (Fourth Row) for FIRS

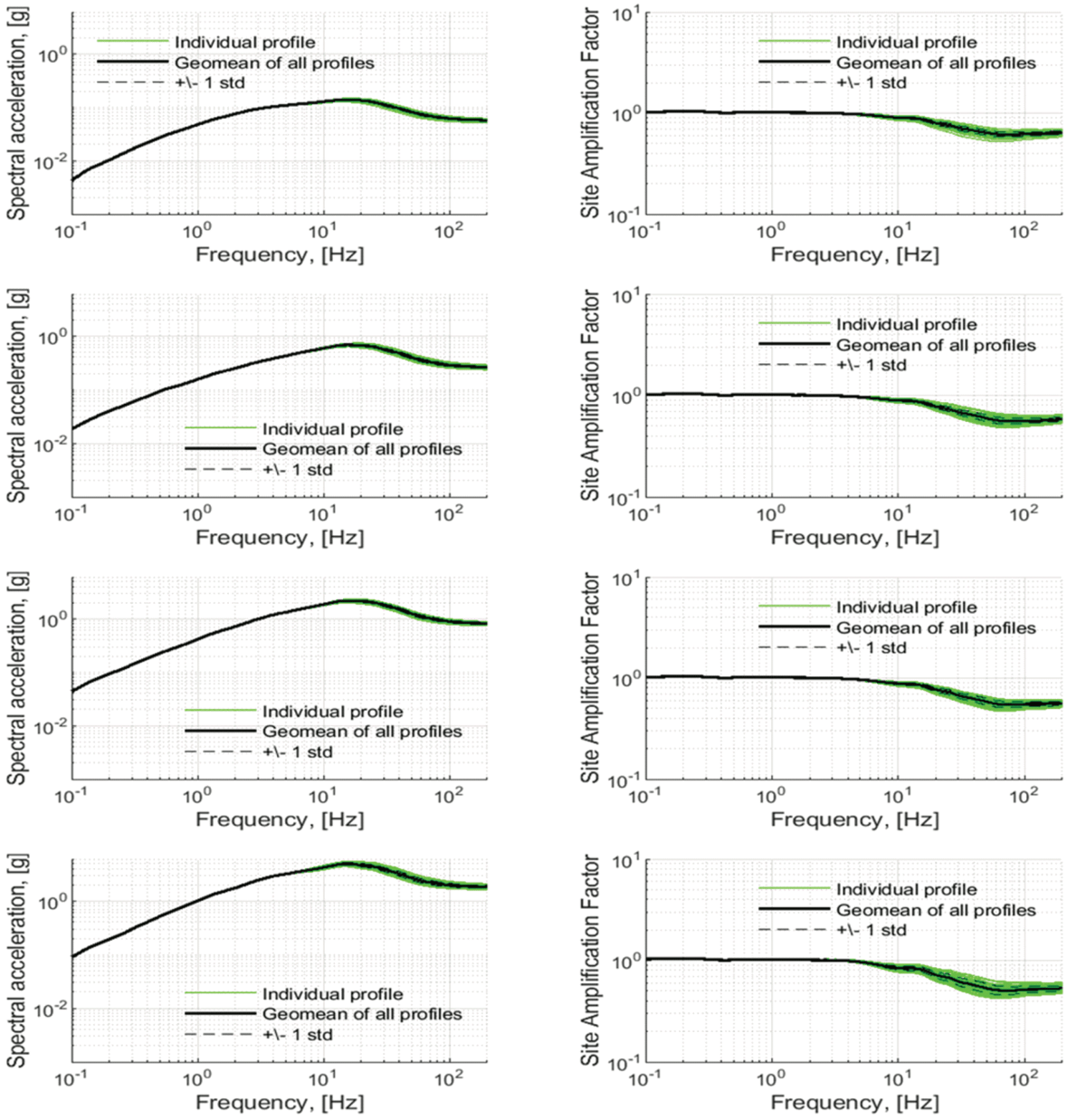


Figure 2.5.2-69R Amplified Response Spectra (Left) and Site Amplification Factors (Right) for the Upper-Range Profile and Material Model 1 at AFEs of 1E-3 (First Row), 1E-4 (Second Row), 1E-5 (Third Row), and 1E-6 (Fourth Row) for FIRS

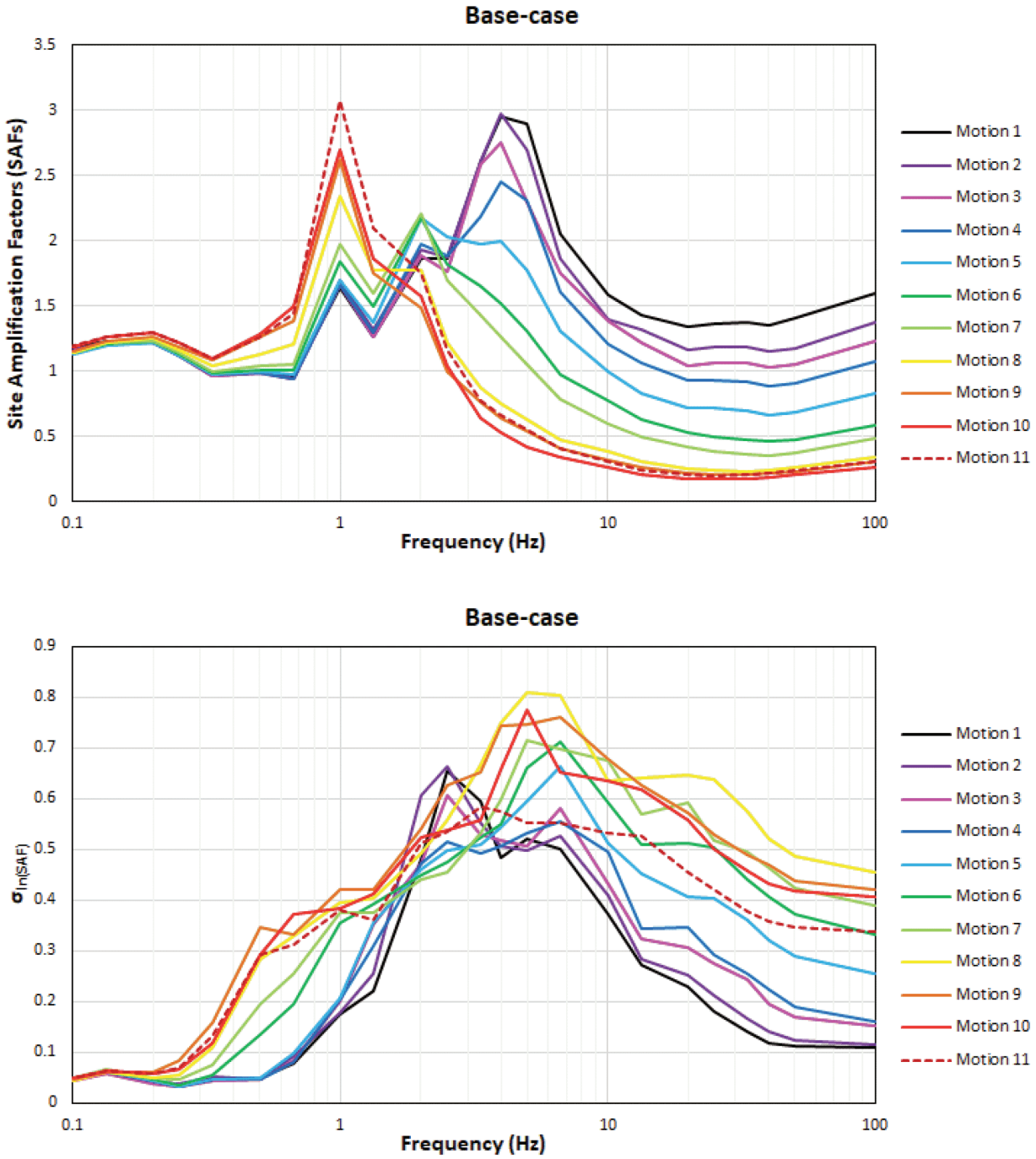


Figure 2.5.2-70R Median and Standard Deviation Site Amplification Factors for Base-Case Profile Using All 11 Input Control Motions for PBSRS

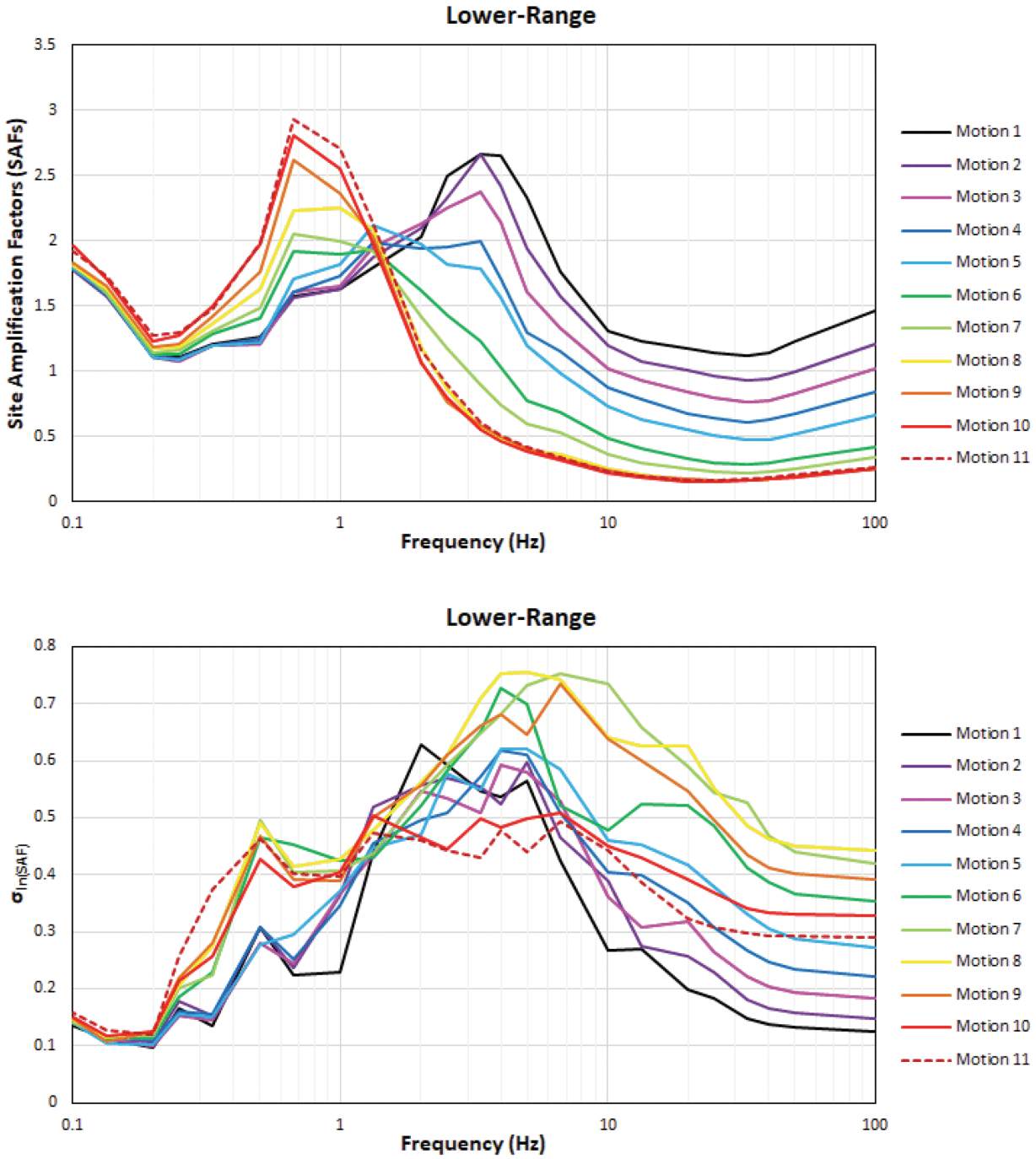


Figure 2.5.2-71R Median and Standard Deviation Site Amplification Factors for Lower-Range Profile Using All 11 Input Control Motions for PBSRS

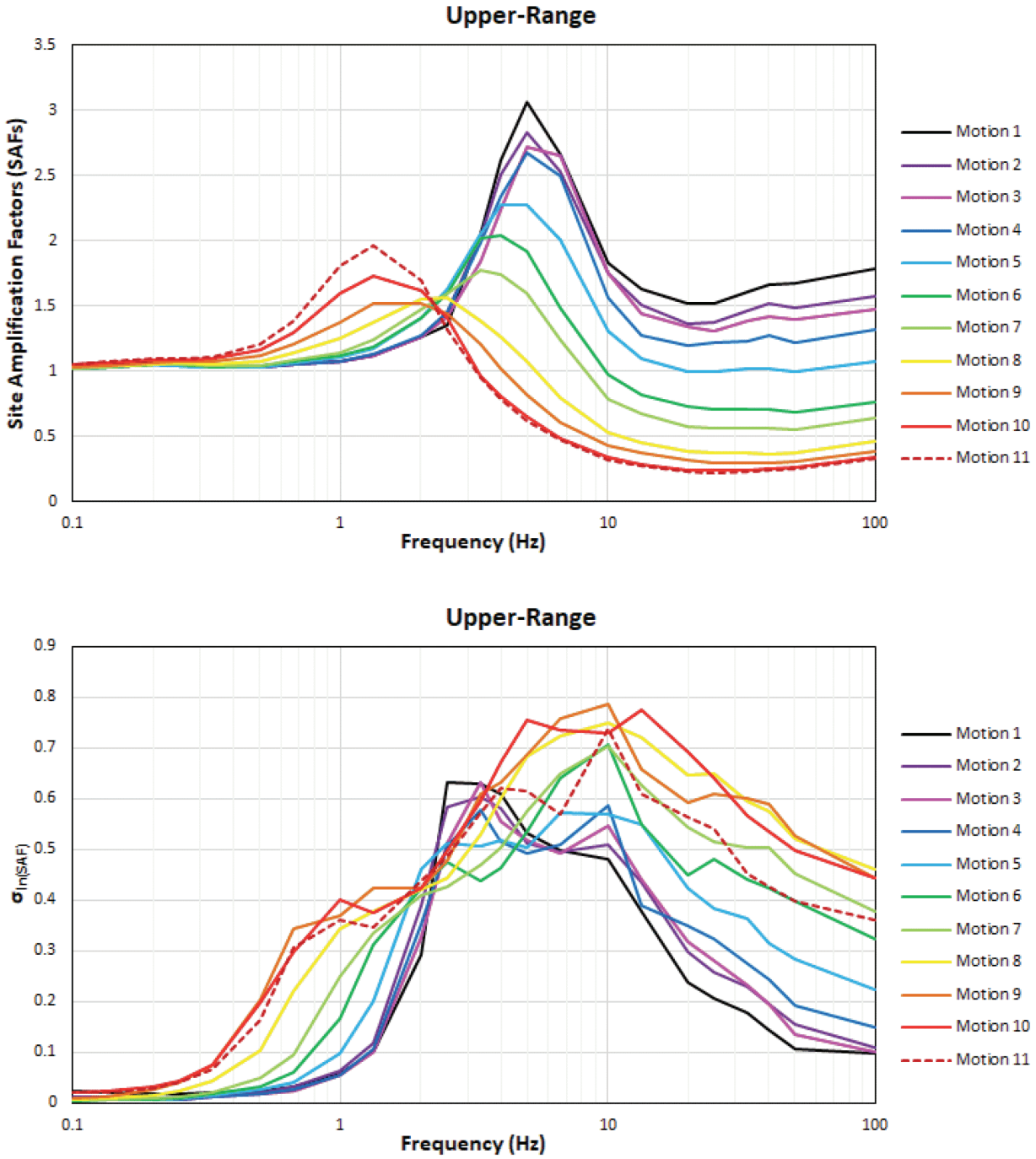


Figure 2.5.2-72R Median and Standard Deviation Site Amplification Factors for Upper-Range Profile Using All 11 Input Control Motions for PBSRS

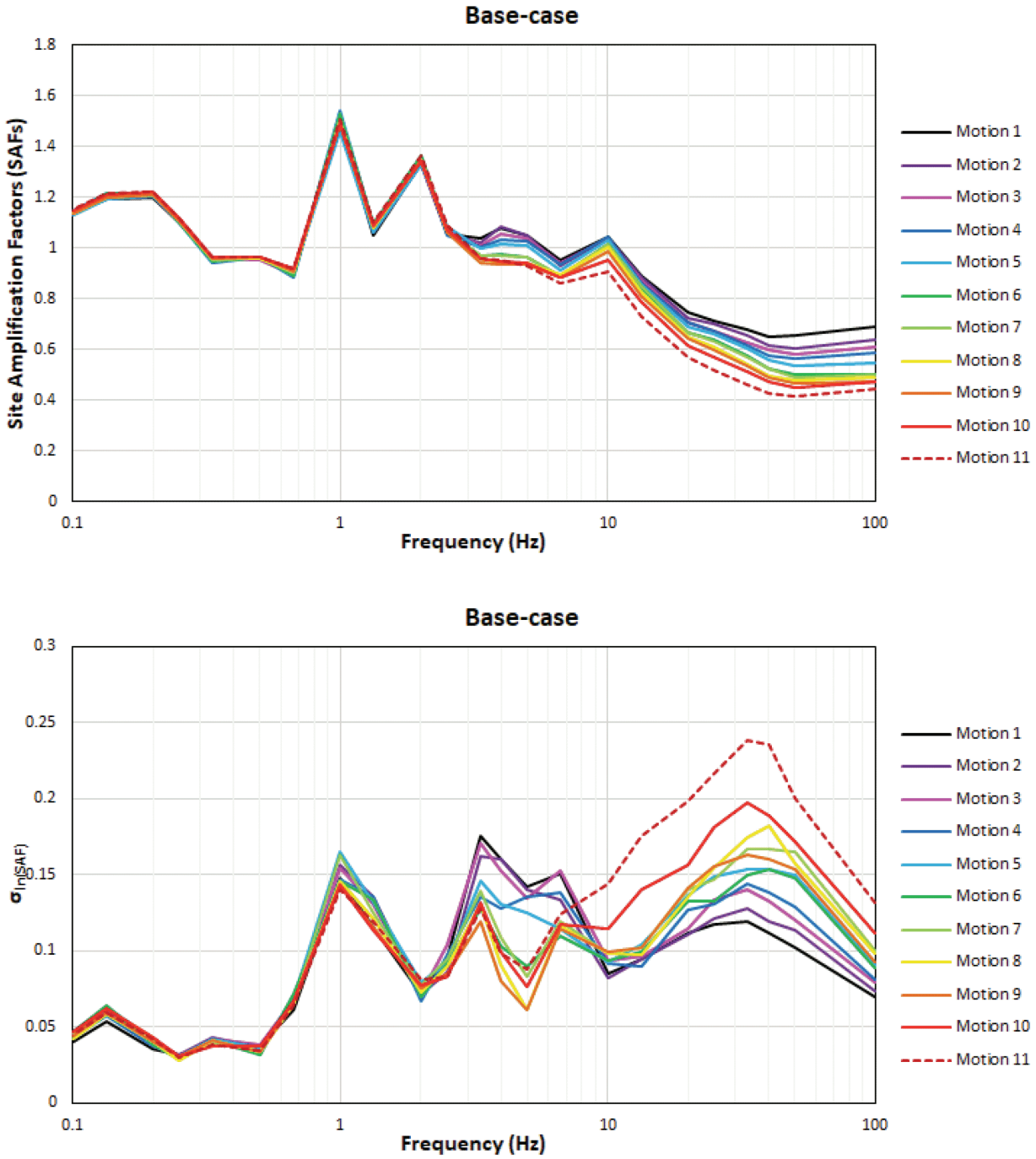


Figure 2.5.2-73R Median and Standard Deviation Site Amplification Factors for Base-Case Profile Using All 11 Input Control Motions for PBIRS

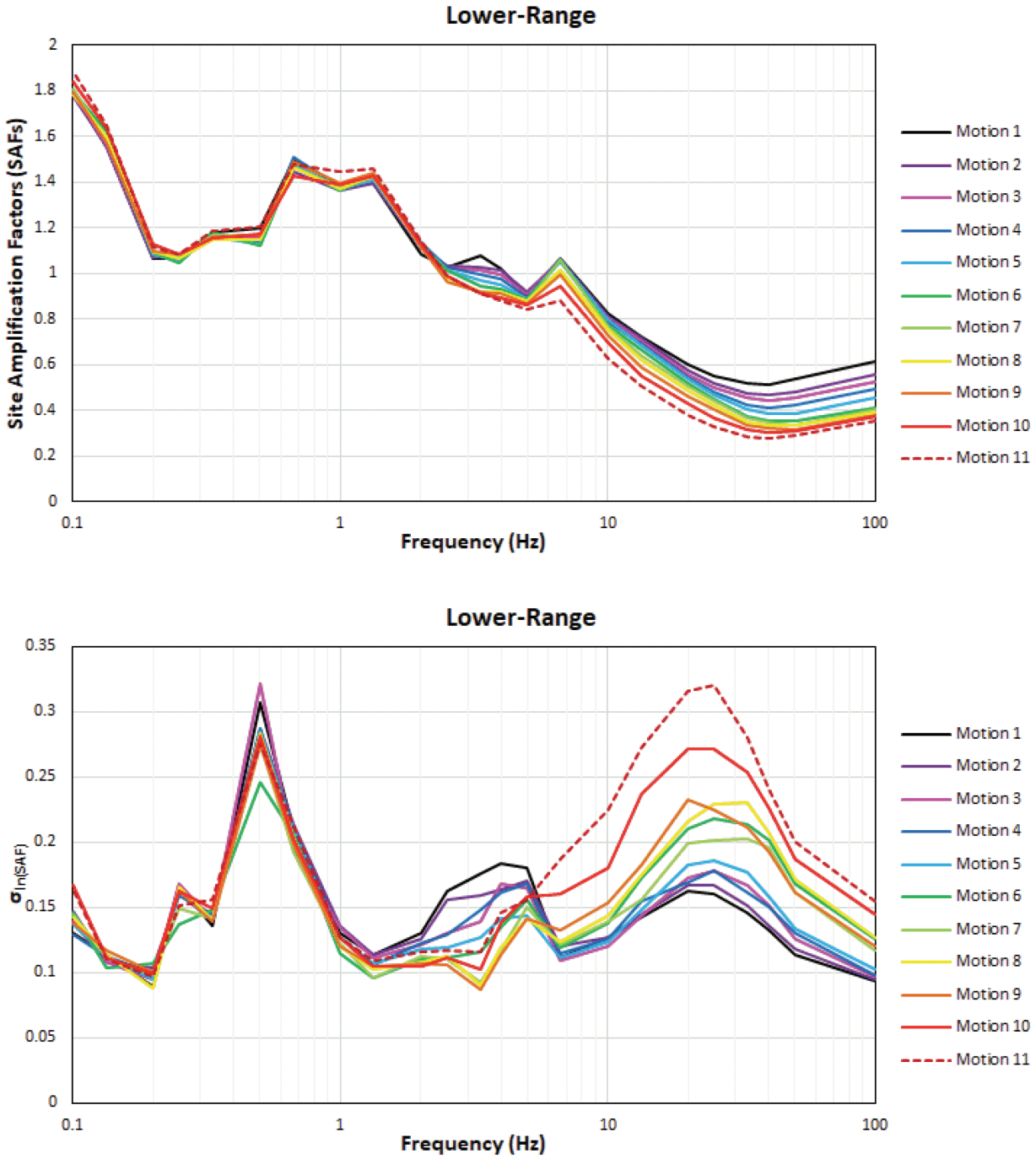


Figure 2.5.2-74R Median and Standard Deviation Site Amplification Factors for Lower-Range Profile Using All 11 Input Control Motions for PBIRS

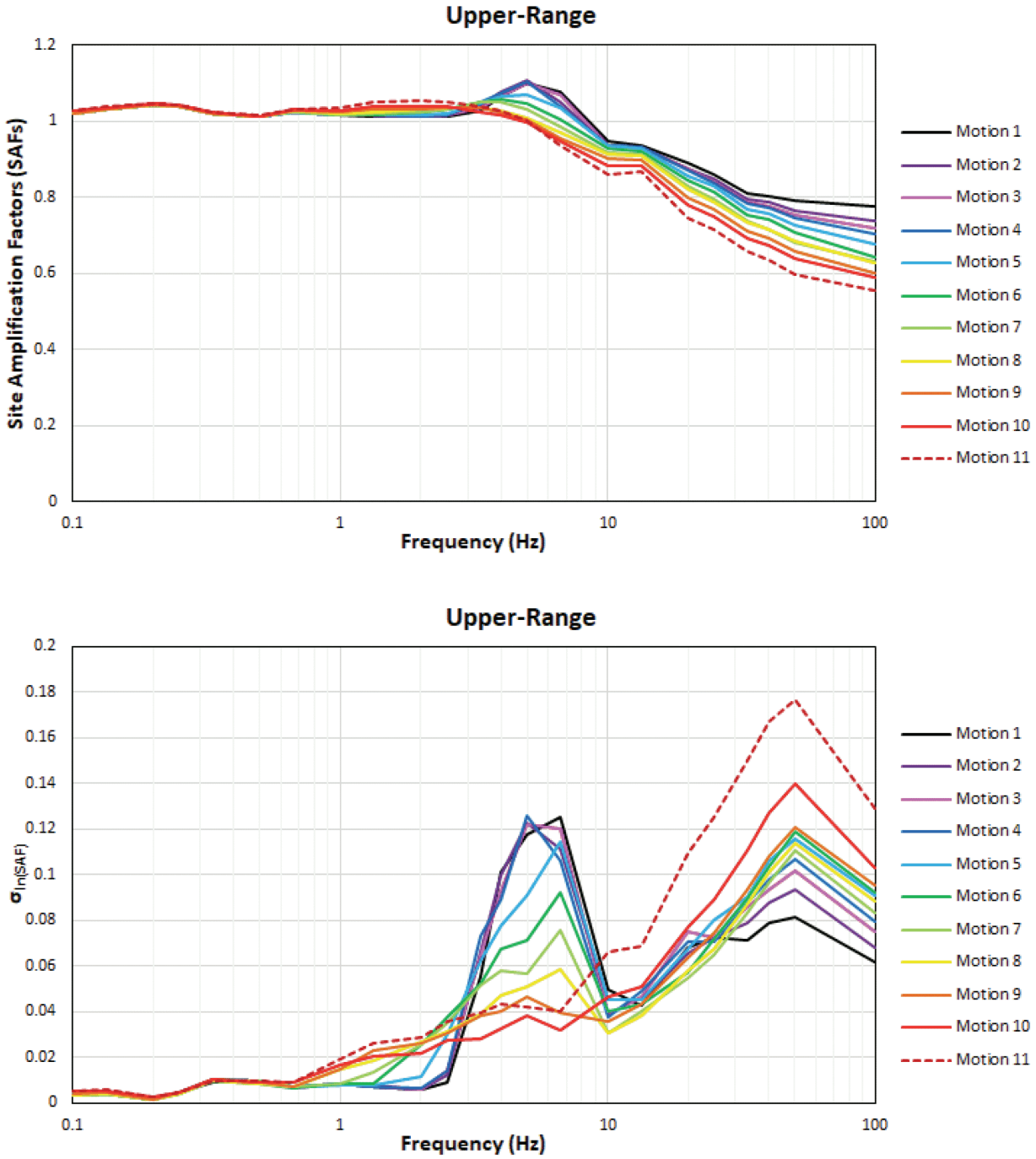


Figure 2.5.2-75R Median and Standard Deviation Site Amplification Factors for Upper-Range Profile Using All 11 Input Control Motions for PBIRS

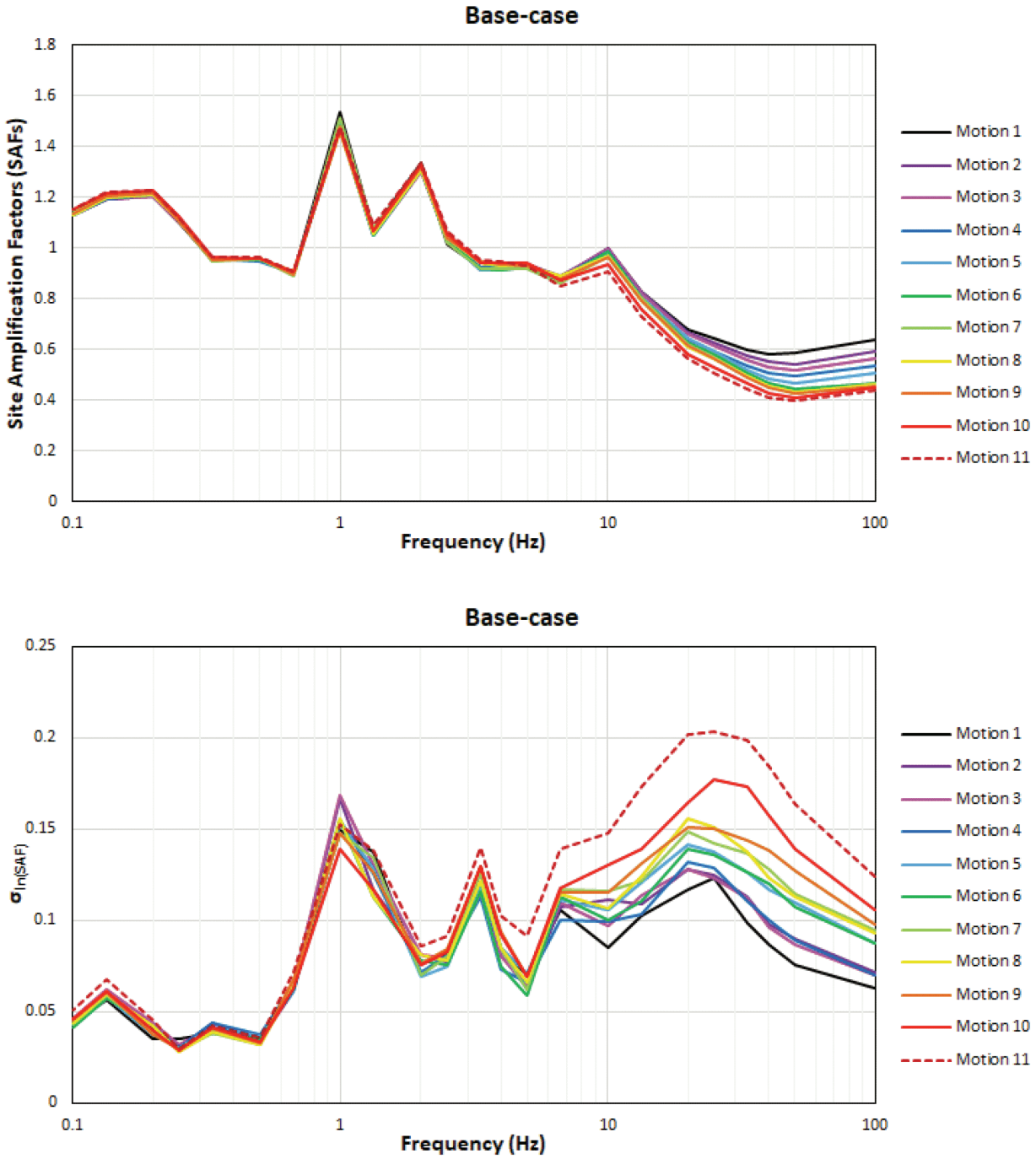


Figure 2.5.2-76R Median and Standard Deviation Site Amplification Factors for Base-Case Profile Using All 11 Input Control Motions for FIRS

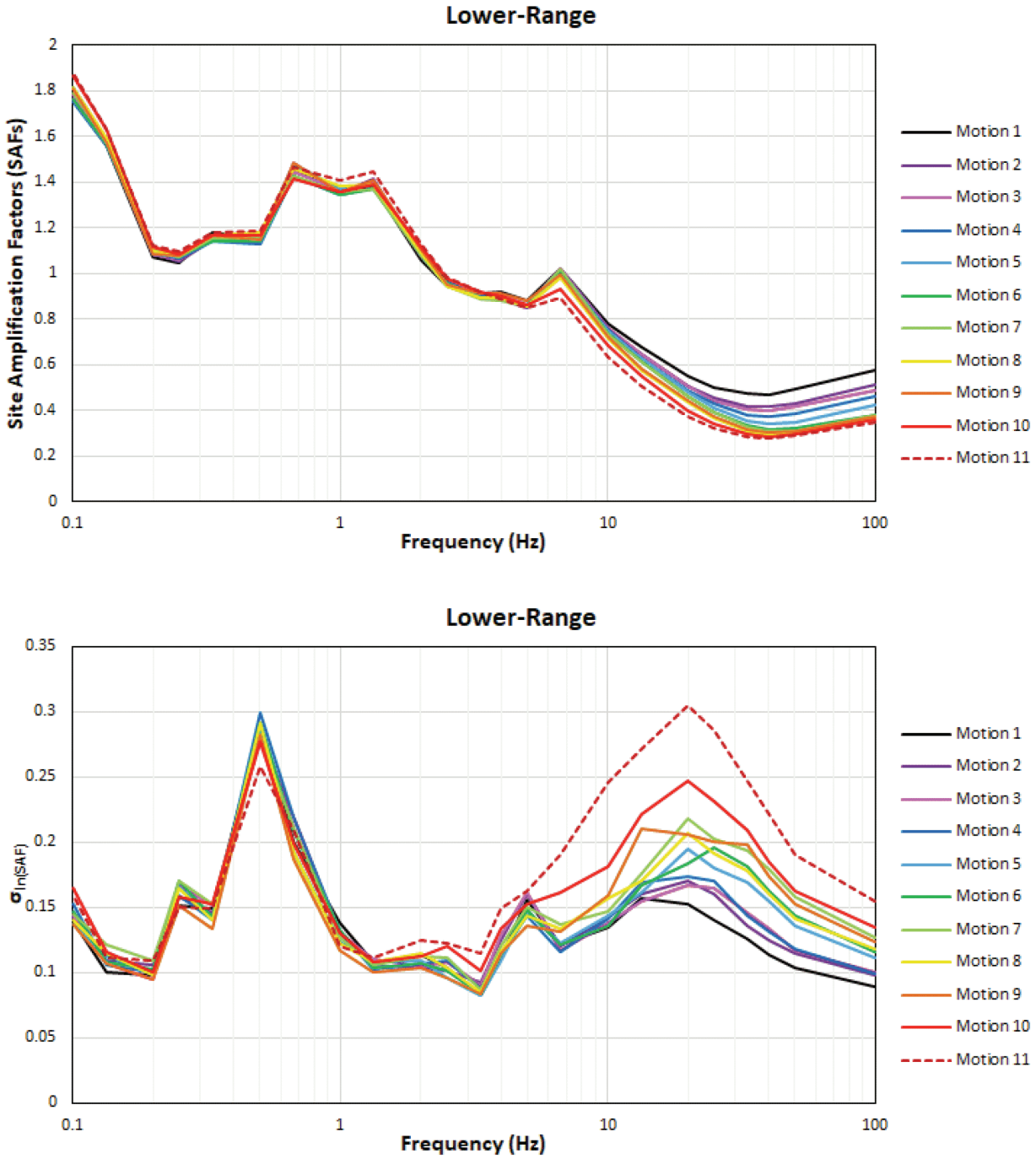


Figure 2.5.2-77R Median and Standard Deviation Site Amplification Factors for Lower-Range Profile Using All 11 Input Control Motions for FIRS

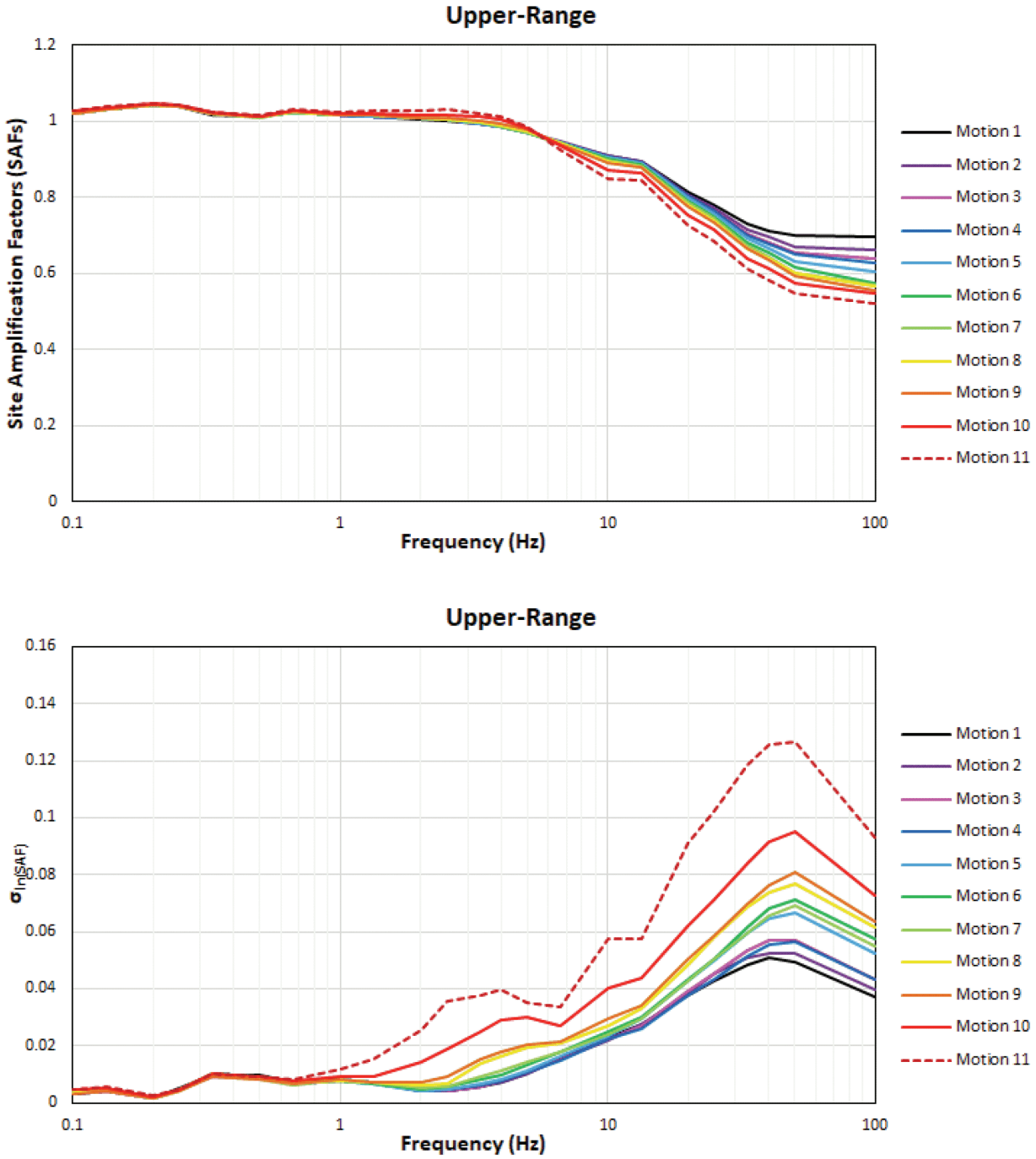


Figure 2.5.2-78R Median and Standard Deviation Site Amplification Factors for Upper-Range Profile Using All 11 Input Control Motions for FIRS

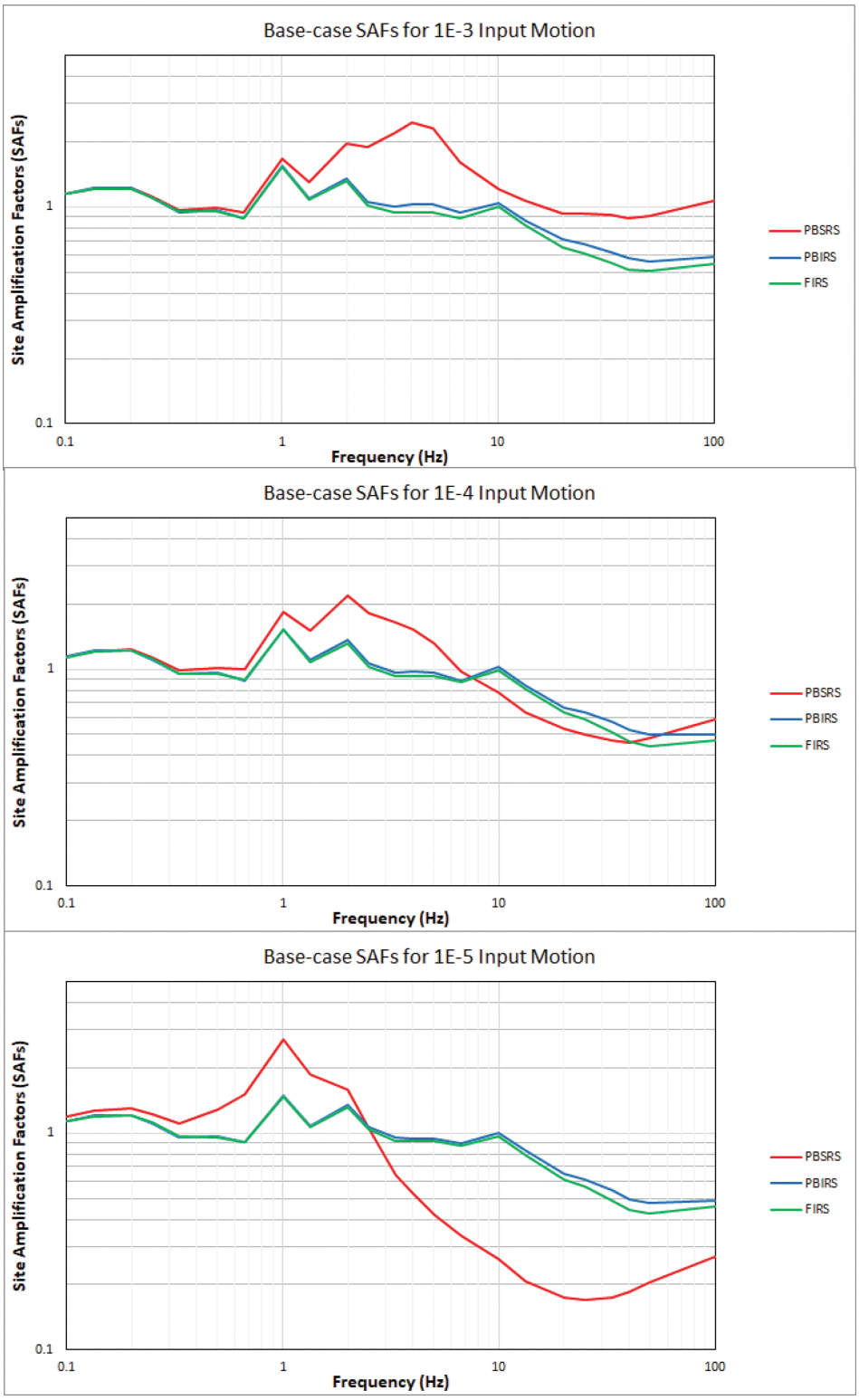


Figure 2.5.2-79R Median Site Amplification Factors for Base-Case Profile Using 1E-3, 1E-4, and 1E-5 Input Control Motions for PBSRS, PBIRS, and FIRS

Renumber Figures 2.5.2-80 through 2.5.2-110 to Figures 2.5.2-105 through 2.5.2-135.

Add new Figures 2.5.2-80R through 2.5.2-104R:

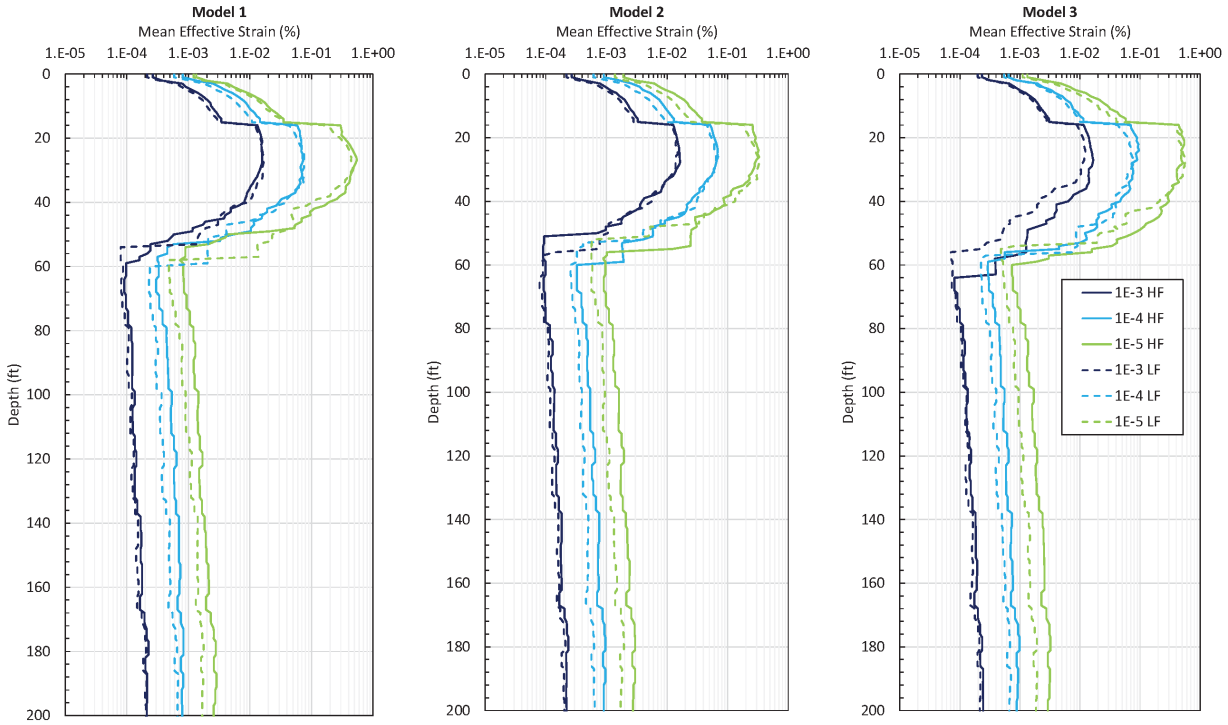


Figure 2.5.2-80R Mean Effective Strain for the Top 200 Feet of Best-Case Profile for 1E-3, 1E-4, and 1E-5 Input Motions, Material Models 1, 2 and 3 (From the PBSRS Analysis)

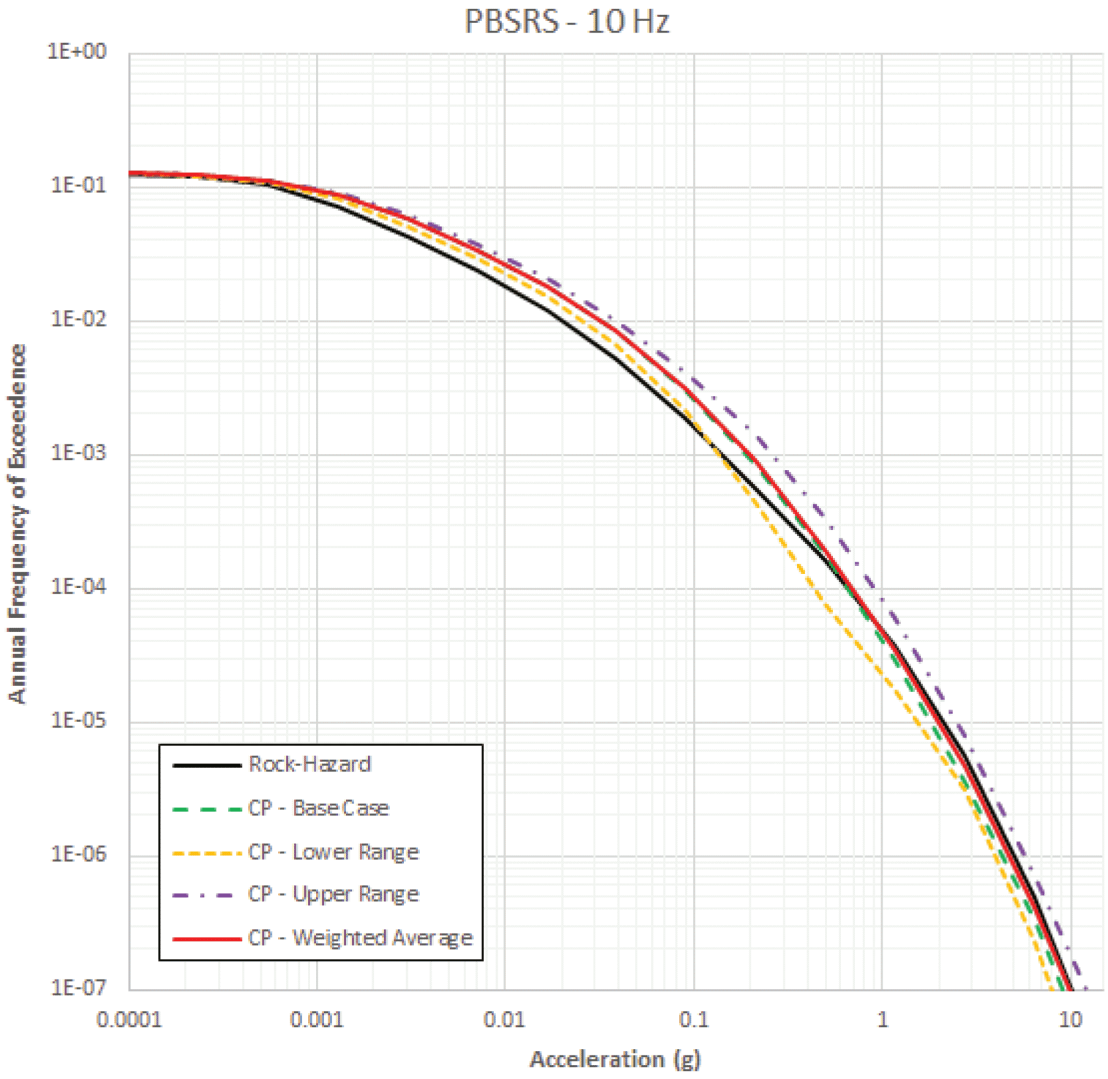


Figure 2.5.2-81R Rock and Control Point (CP) Hazard Curves at 10 Hz SA for PBSRS

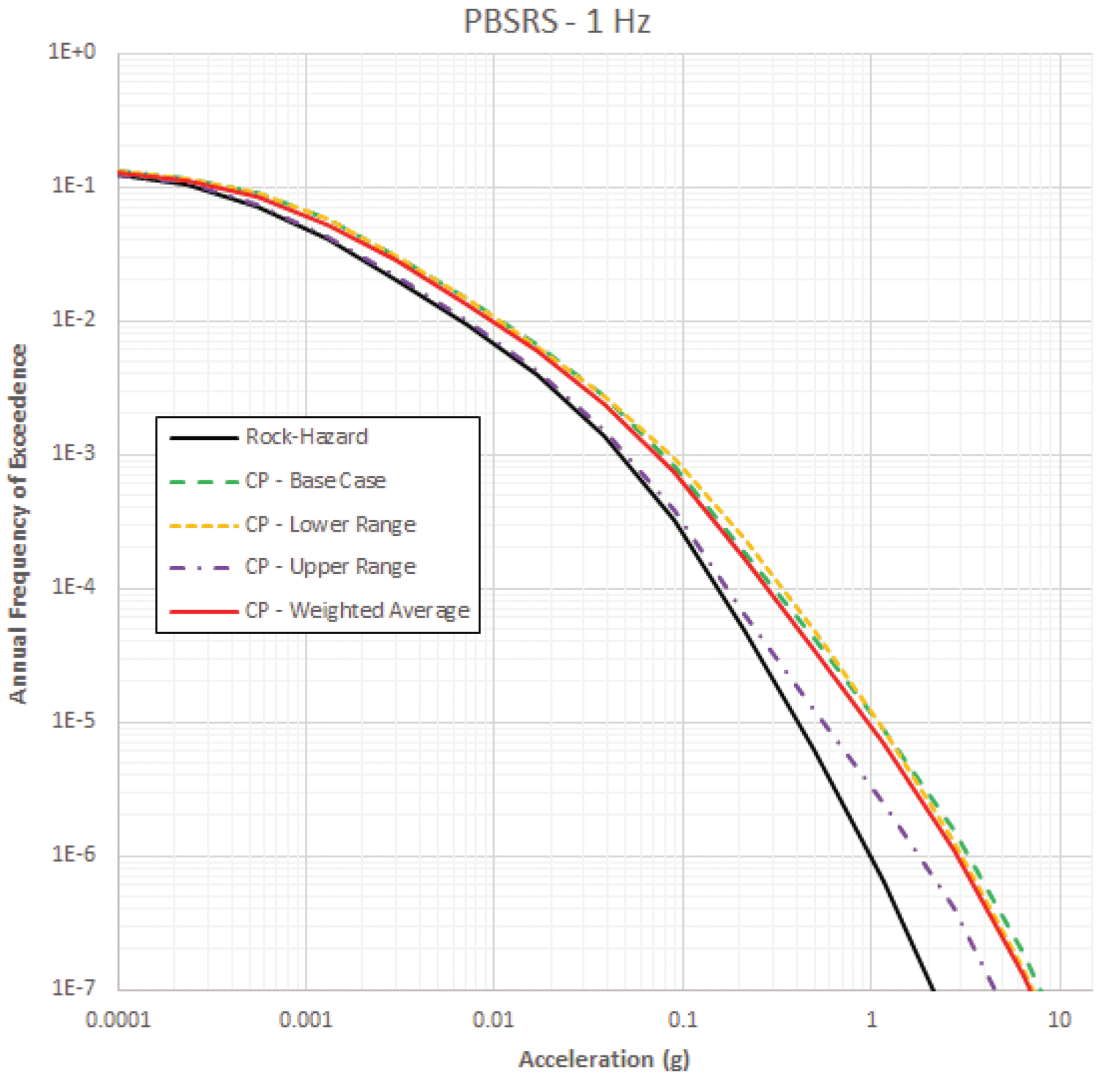


Figure 2.5.2-82R Rock and Control Point (CP) Hazard Curves at 1 Hz SA for PBSRS

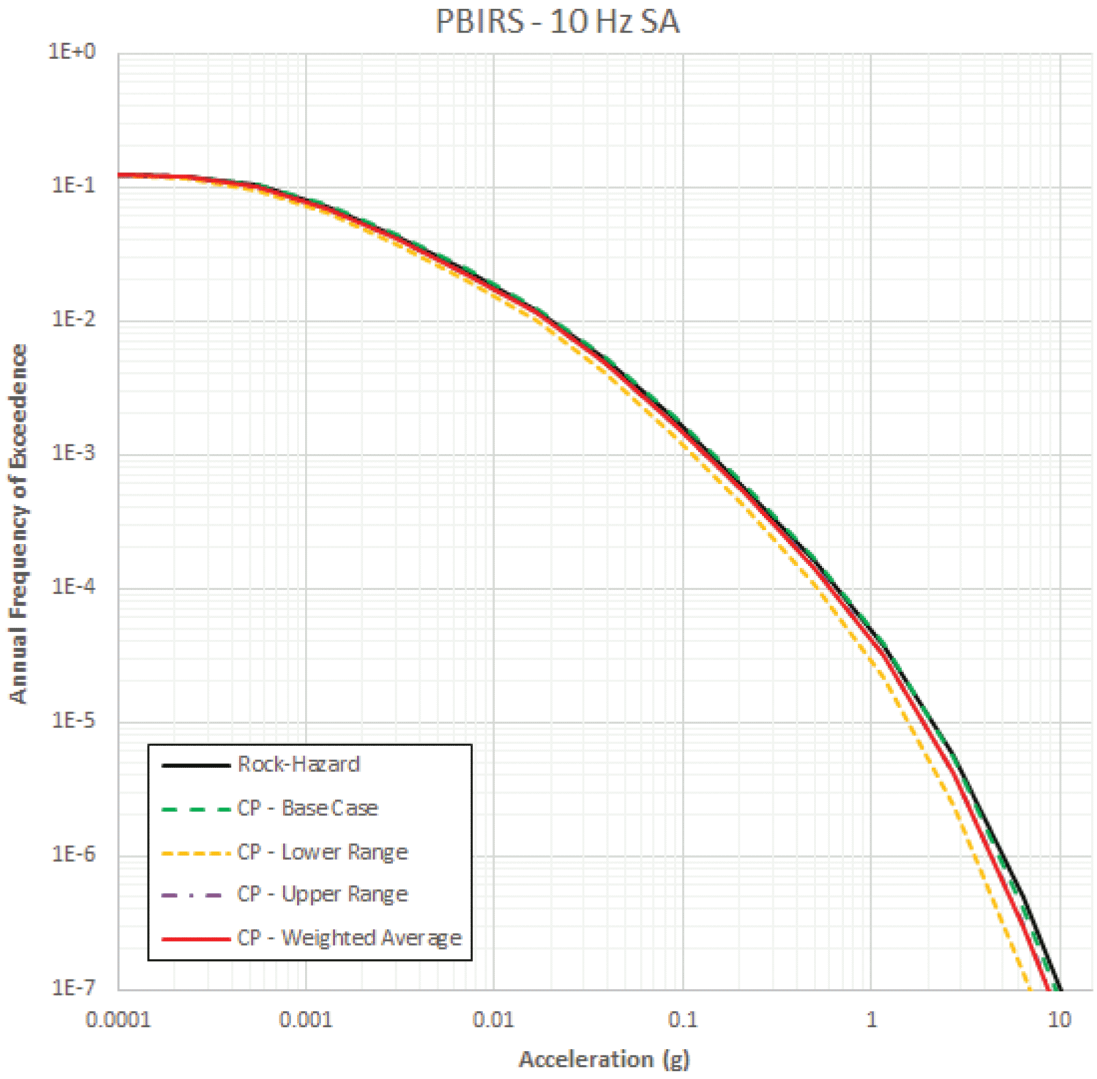


Figure 2.5.2-83R Rock and Control Point (CP) Hazard Curves at 10 Hz SA for PBIRS

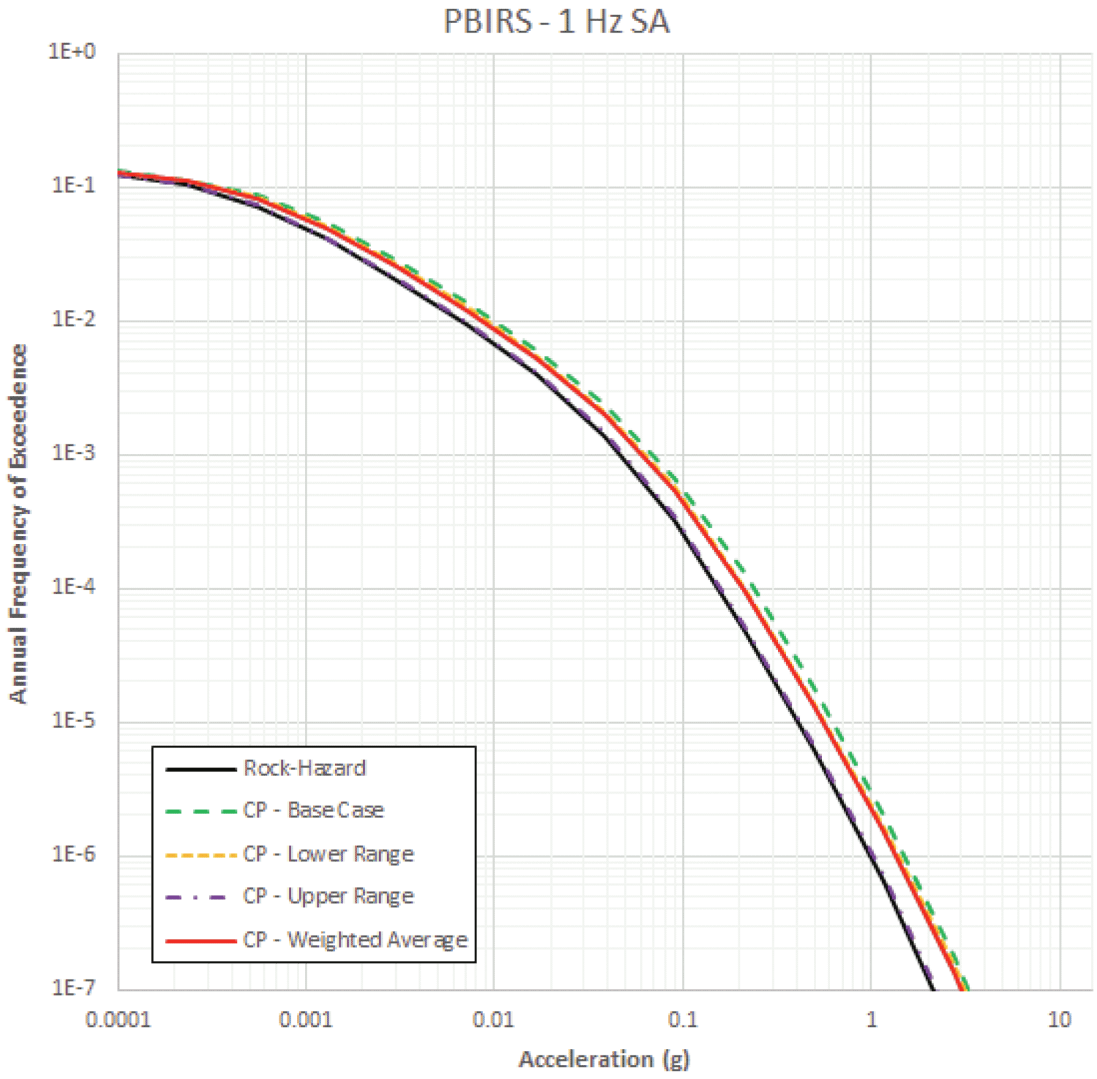


Figure 2.5.2-84R Rock and Control Point (CP) Hazard Curves at 1 Hz SA for PBIRS

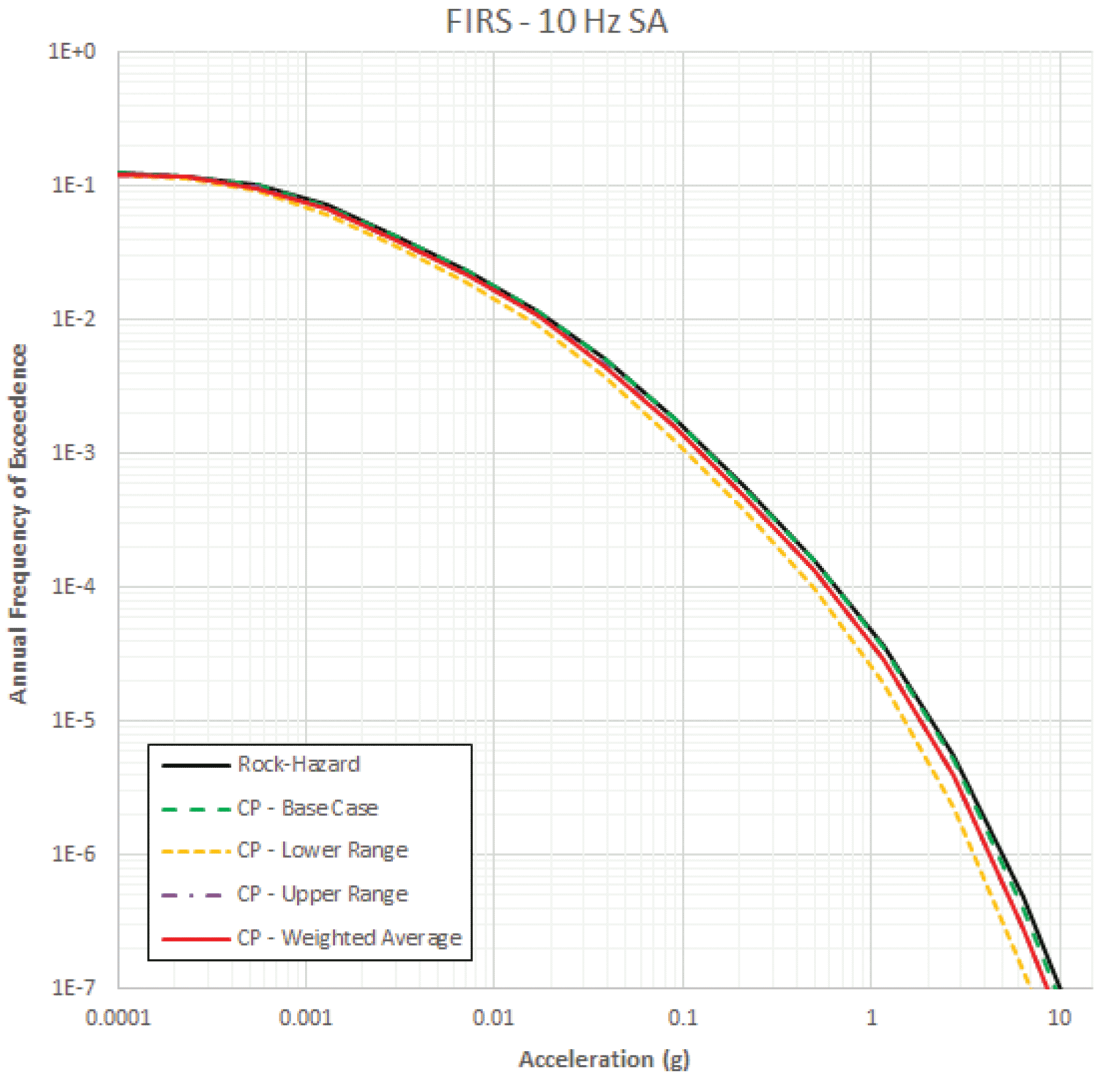


Figure 2.5.2-85R Rock and Control Point (CP) Hazard Curves at 10 Hz SA for FIRS

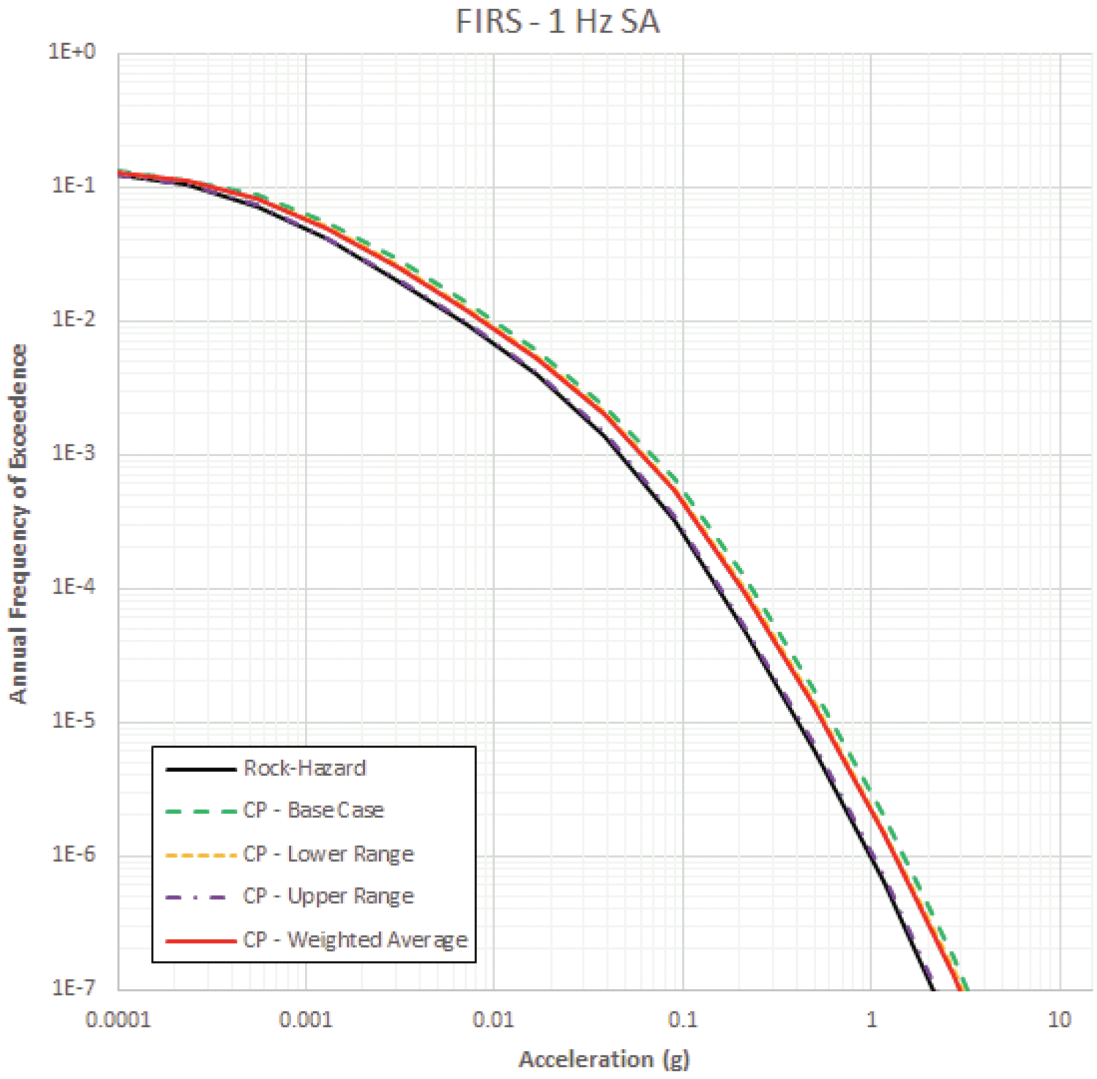


Figure 2.5.2-86R Rock and Control Point (CP) Hazard Curves at 1 Hz SA for FIRS

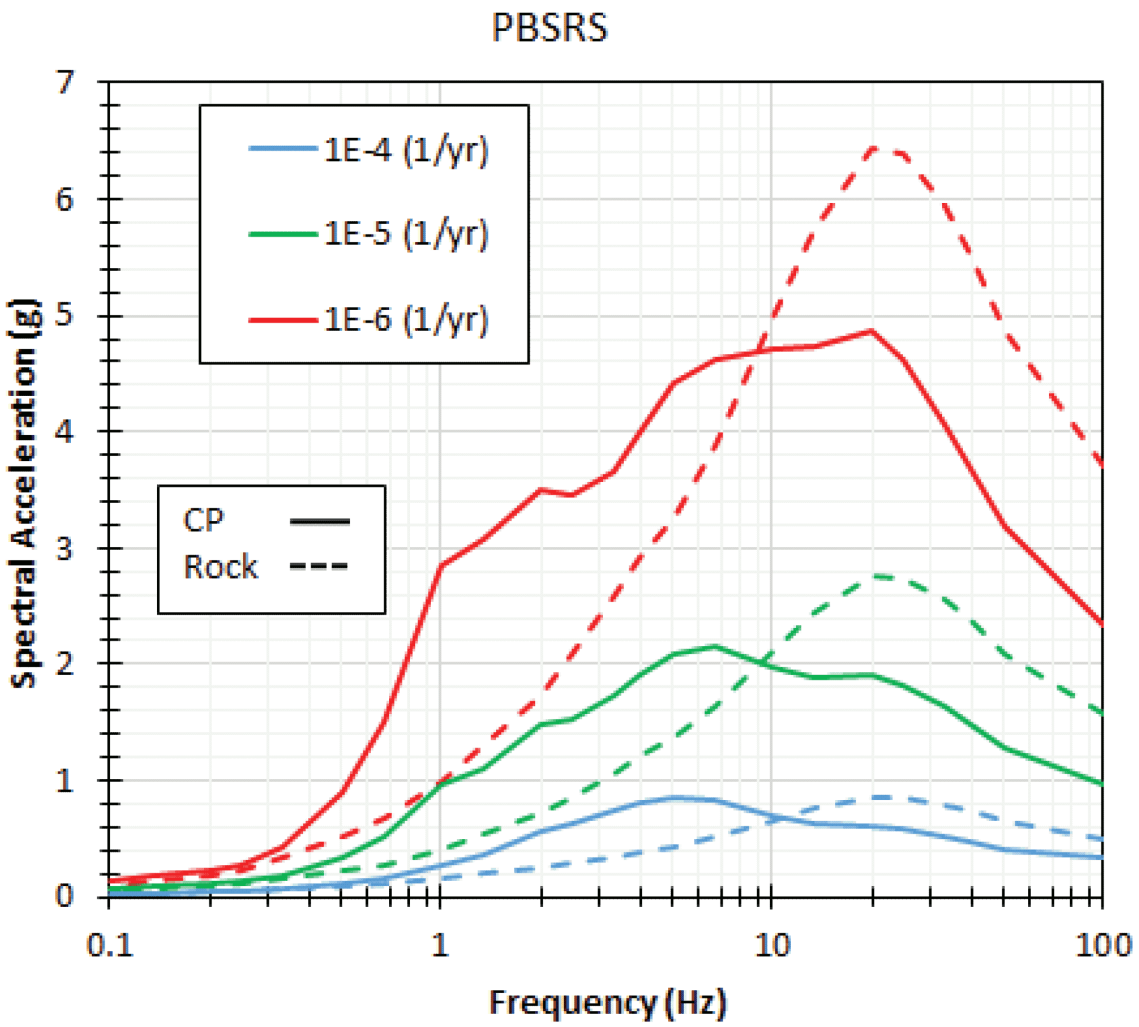


Figure 2.5.2-87R Rock and Control Point (CP) Horizontal Response Spectra at AFEs of 1E-4, 1E-5, and 1E-6 for PBSRS

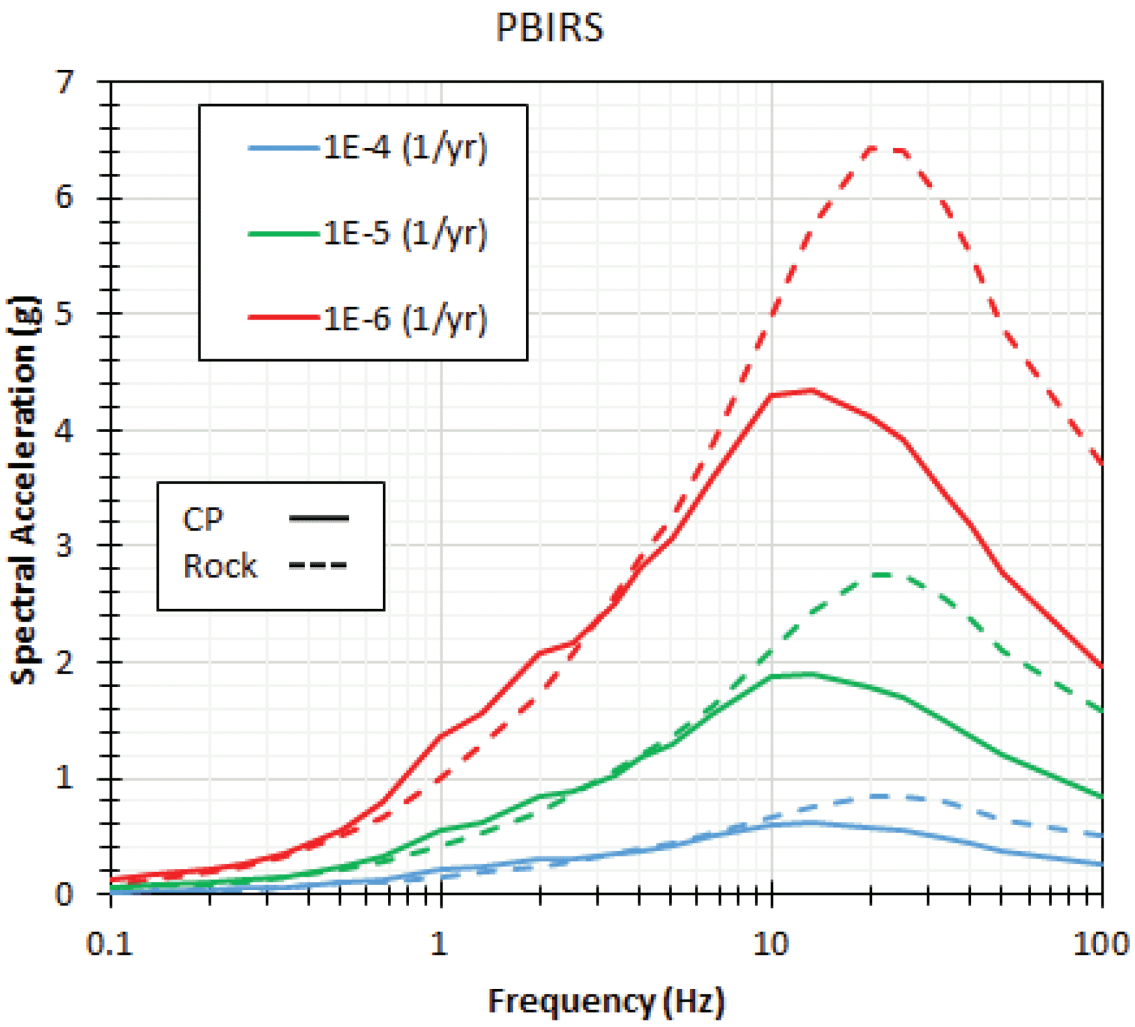


Figure 2.5.2-88R Rock and Control Point (CP) Horizontal Response Spectra at AFEs of 1E-4, 1E-5, and 1E-6 for PBIRS

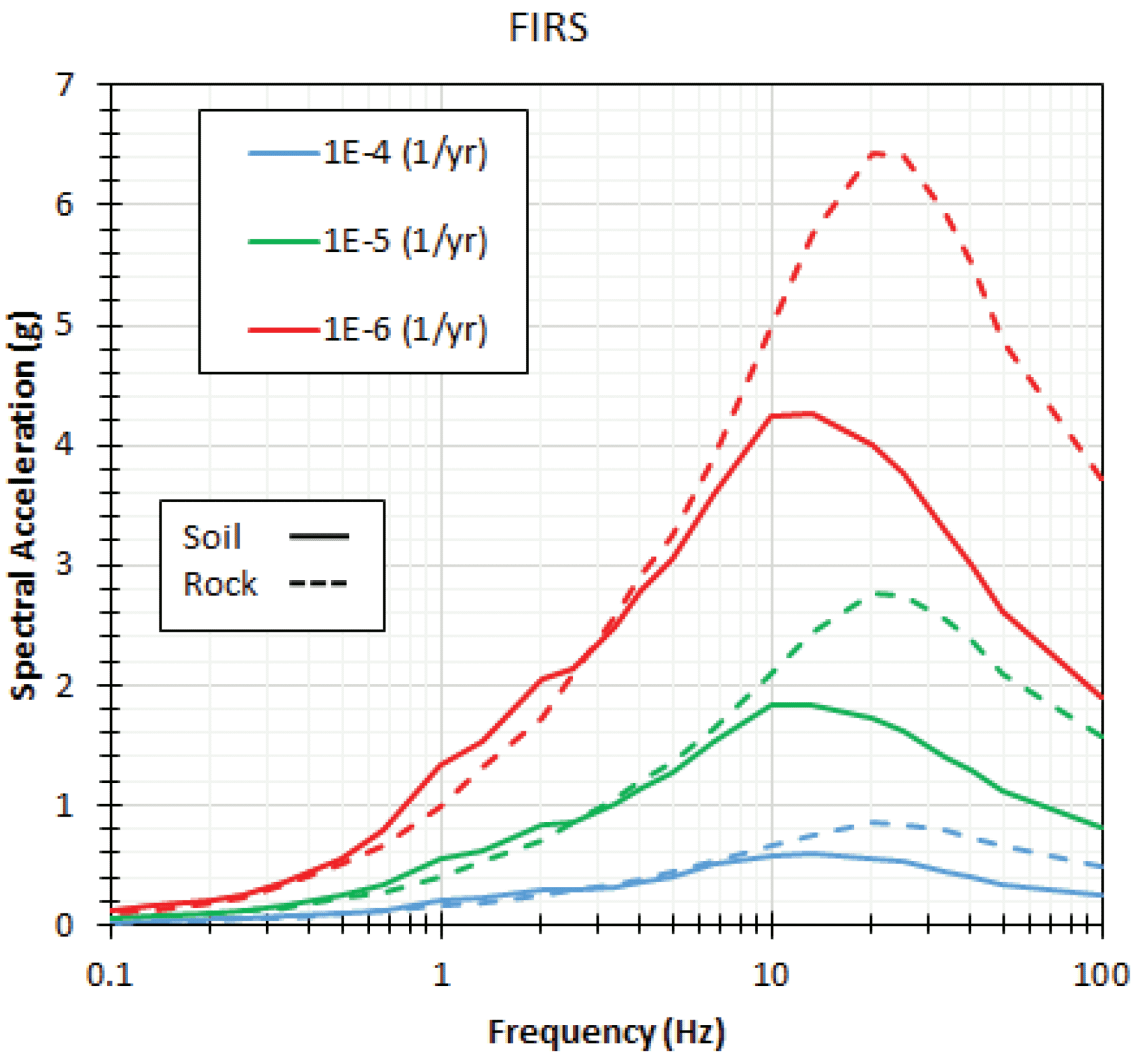


Figure 2.5.2-89R Rock and Control Point (CP) Horizontal Response Spectra at AFEs of 1E-4, 1E-5, and 1E-6 for FIRS

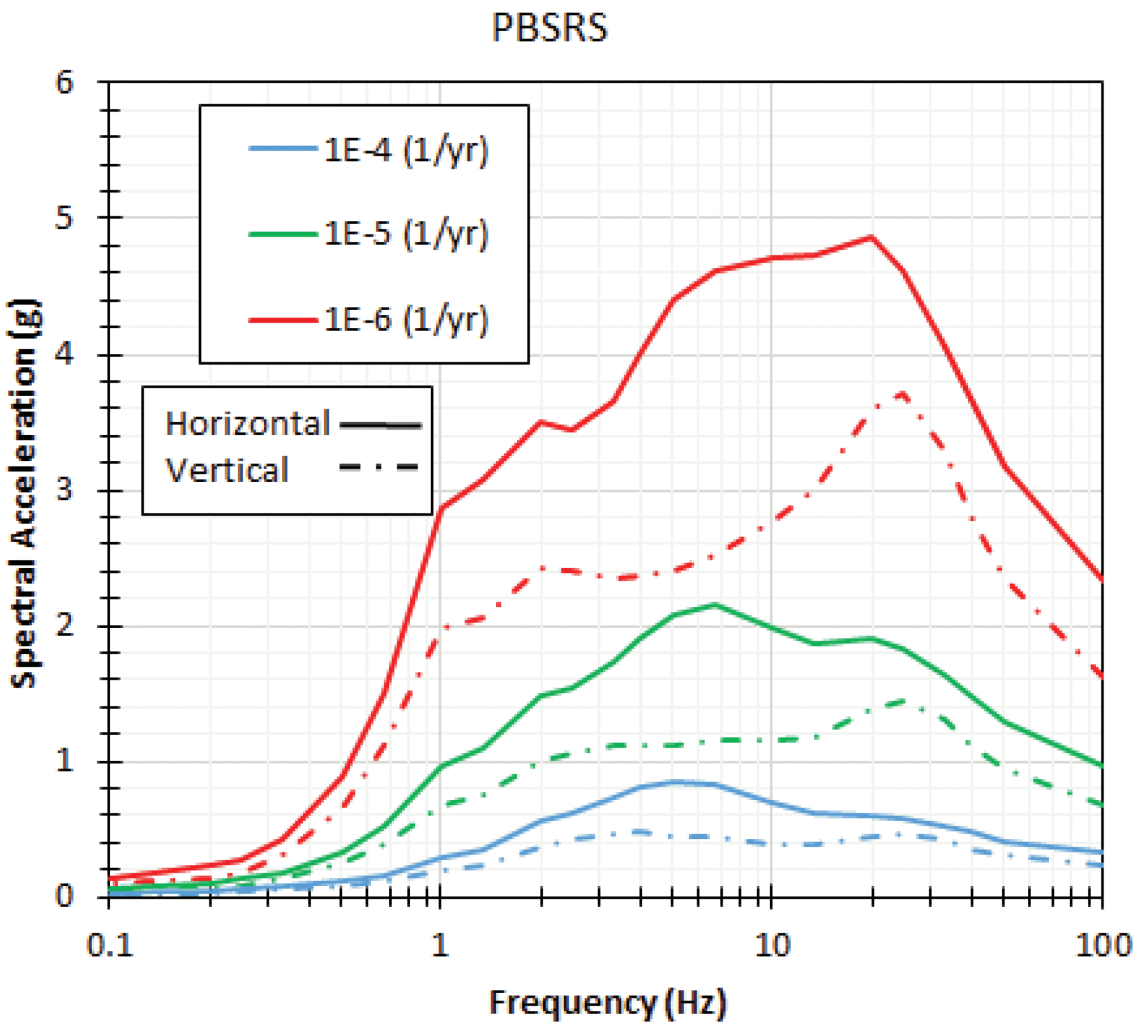


Figure 2.5.2-90R Control Point (CP) Horizontal and Vertical Response Spectra at AFEs of 1E-4, 1E-5, and 1E-6 for PBSRS

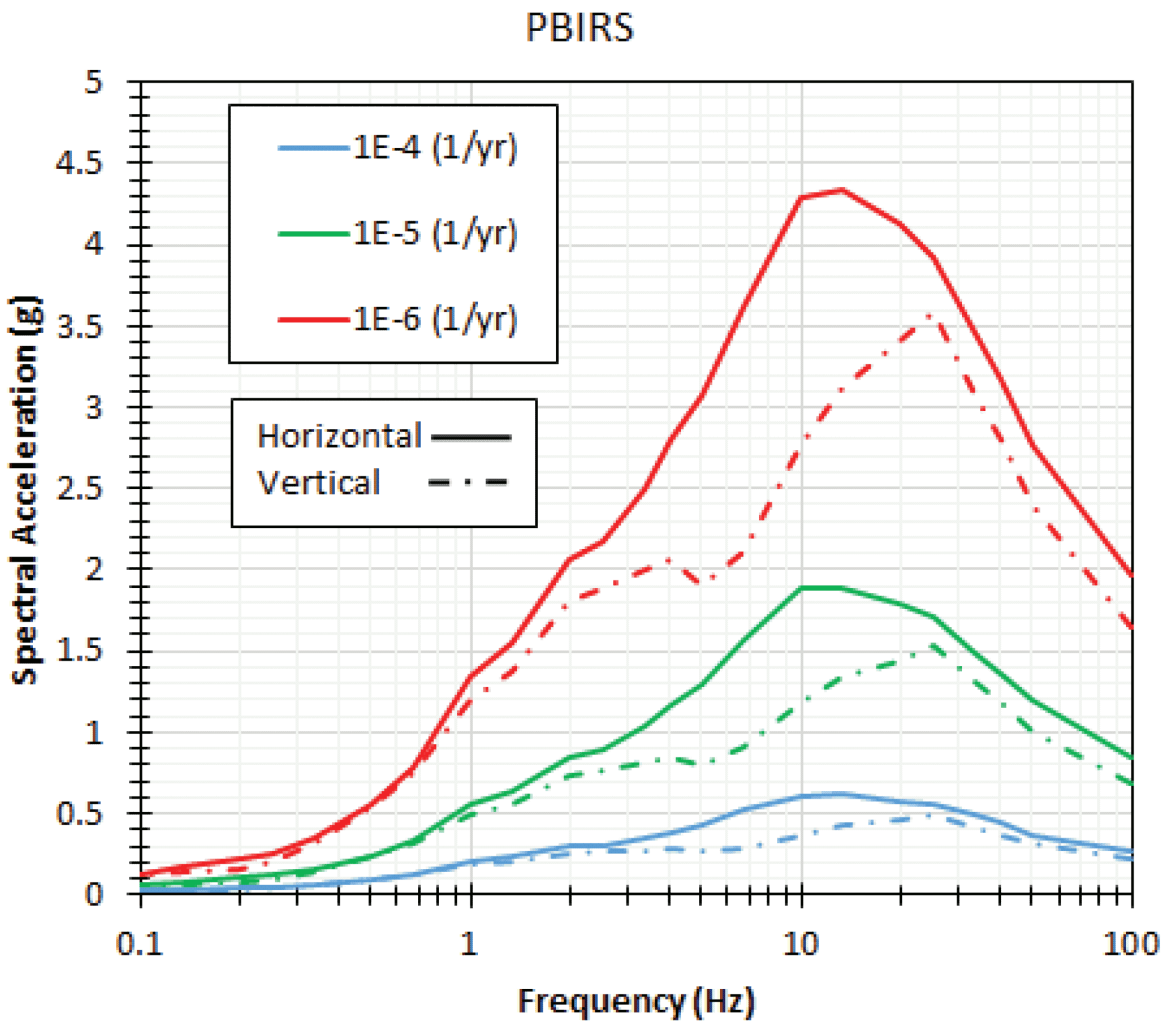


Figure 2.5.2-91R Control Point (CP) Horizontal and Vertical Response Spectra at AFEs of 1E-4, 1E-5, and 1E-6 for PBIRS

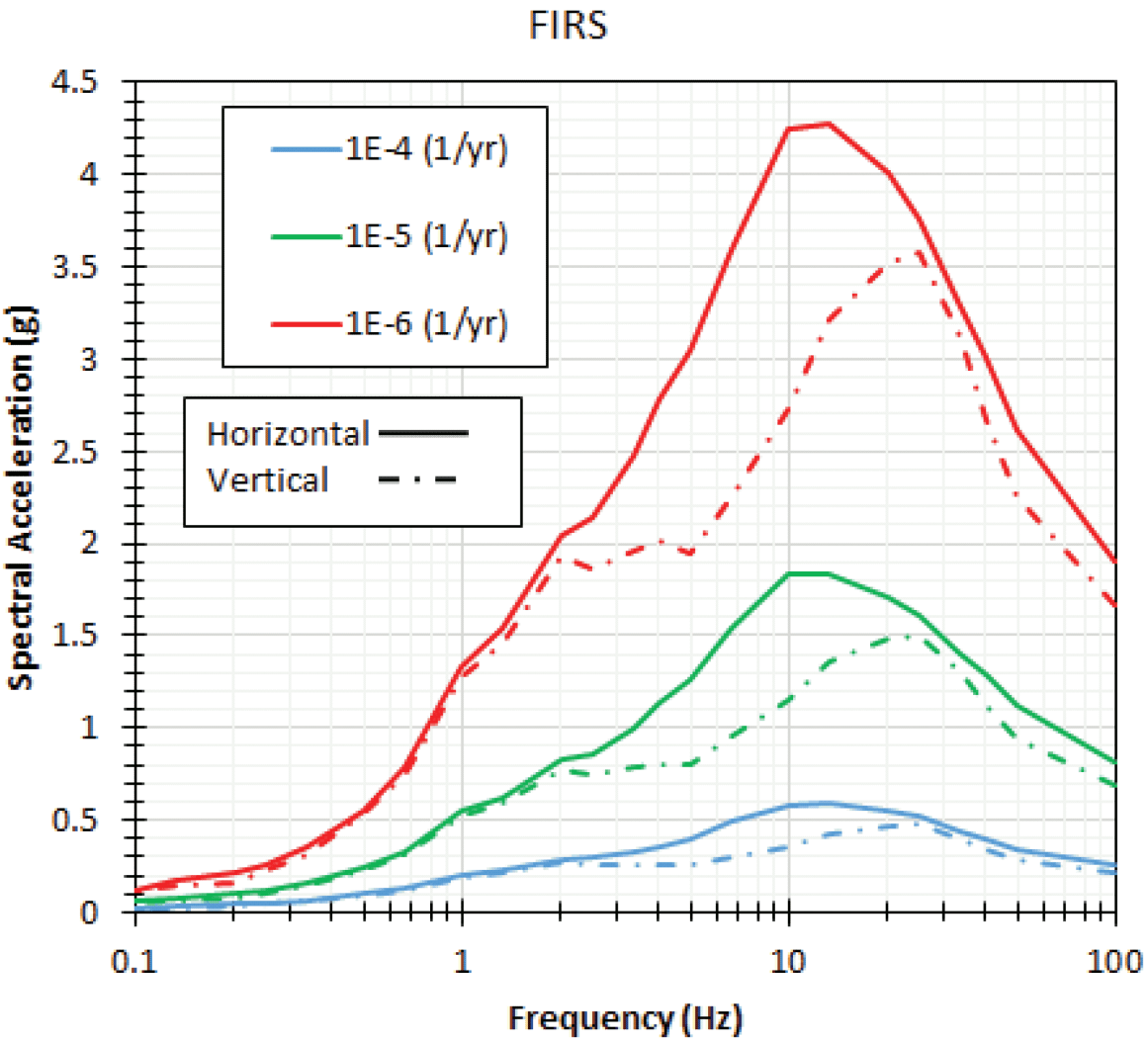


Figure 2.5.2-92R Control Point (CP) Horizontal and Vertical Response Spectra at AFEs of 1E-4, 1E-5, and 1E-6 for FIRS

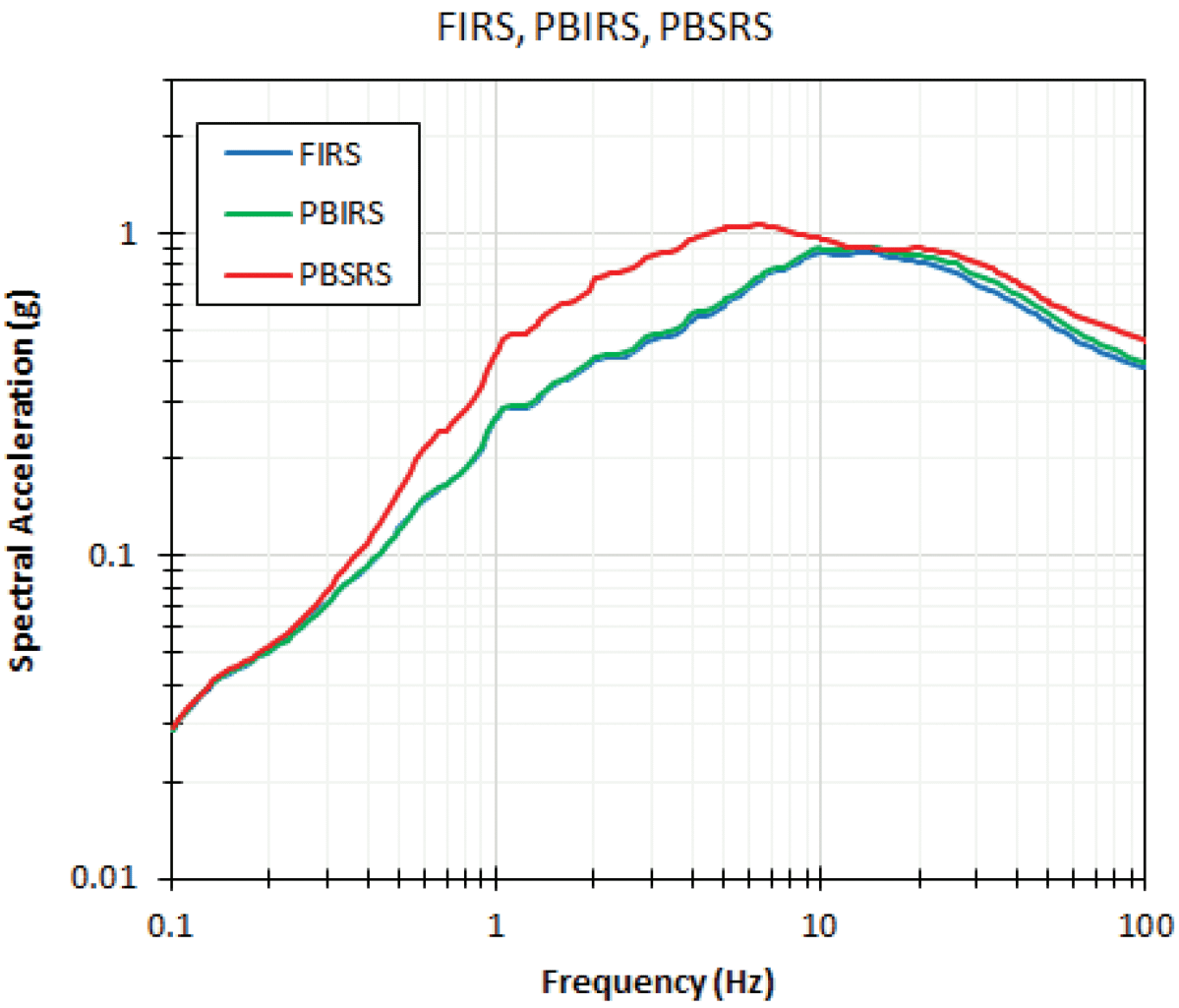


Figure 2.5.2-93R Horizontal PBSRS, PBIRS, and FIRS for 5% Damping at CRN Site

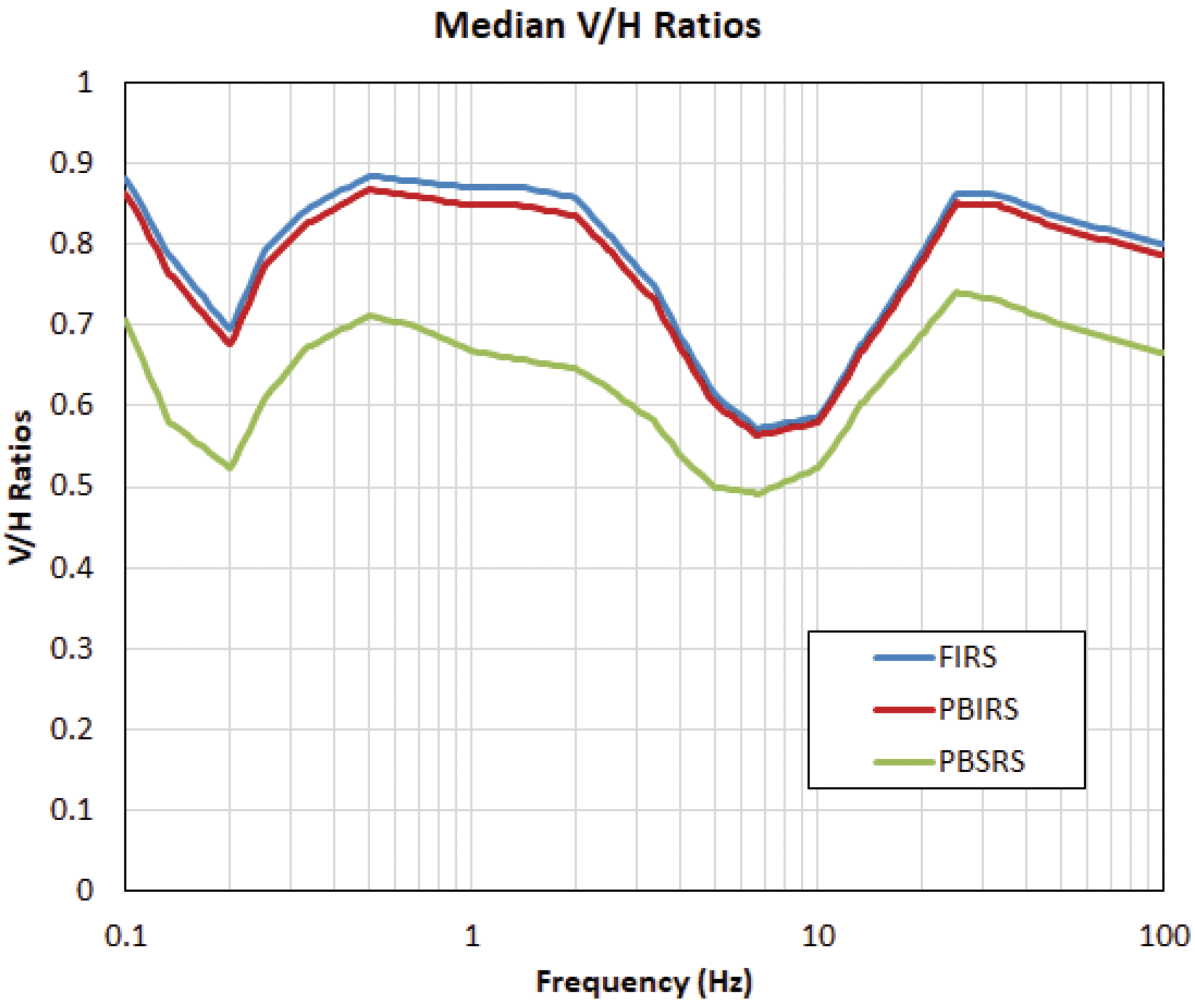


Figure 2.5.2-94R Median V/H ratios for FIRS, PBIRS, and PBSRS

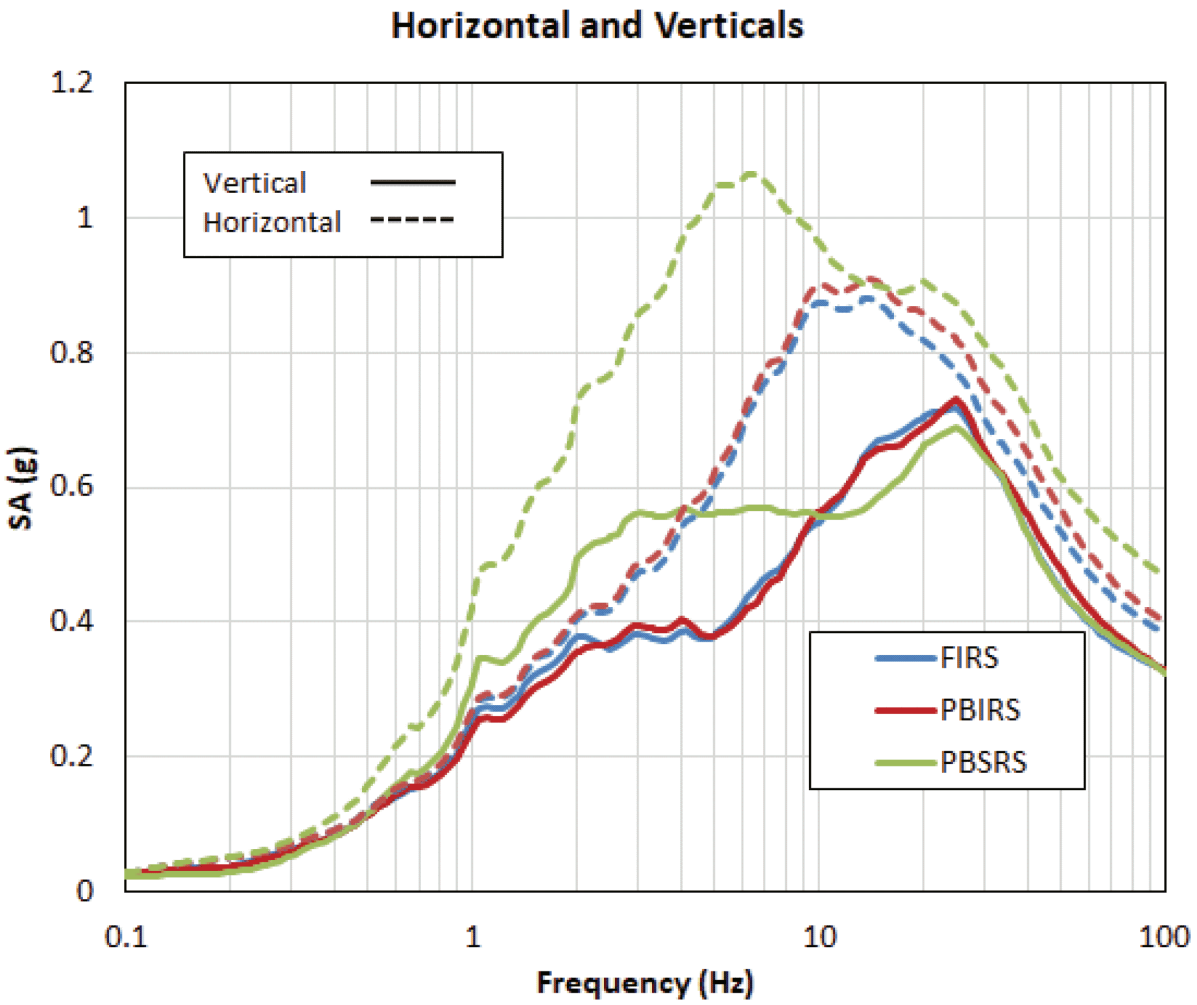


Figure 2.5.2-95R Horizontal and Vertical Spectra for FIRS, PBIRS, and PBSRS

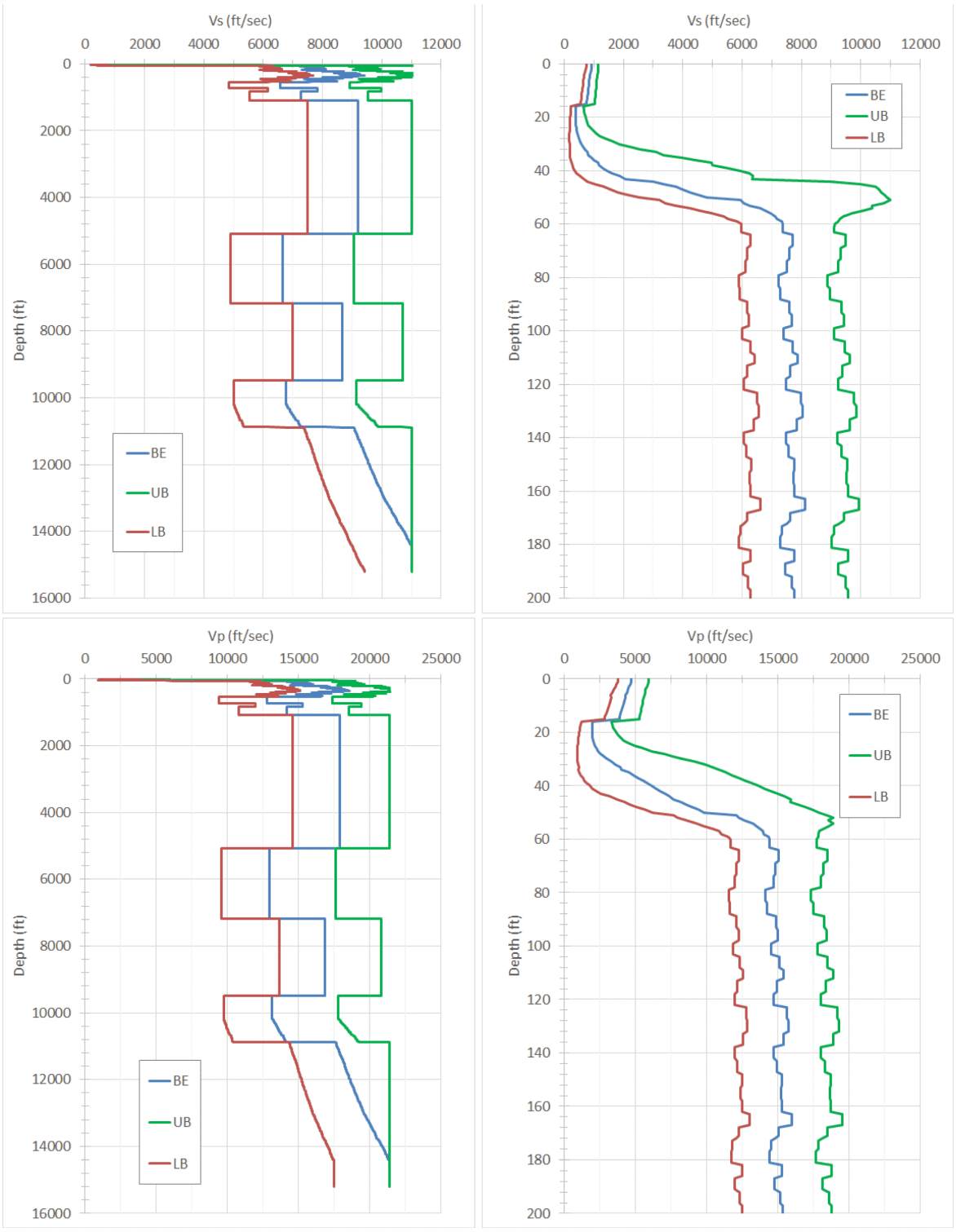


Figure 2.5.2-96R Hazard Consistent Strain Compatible Properties V_S and V_P Corresponding to the PBSRS at CRN Site. BE, LB, and UB Profiles are Included. Left: V_S and V_P Profiles Down to the Hardrock Basement. Right: V_S and V_P Profiles in the Top 200 ft

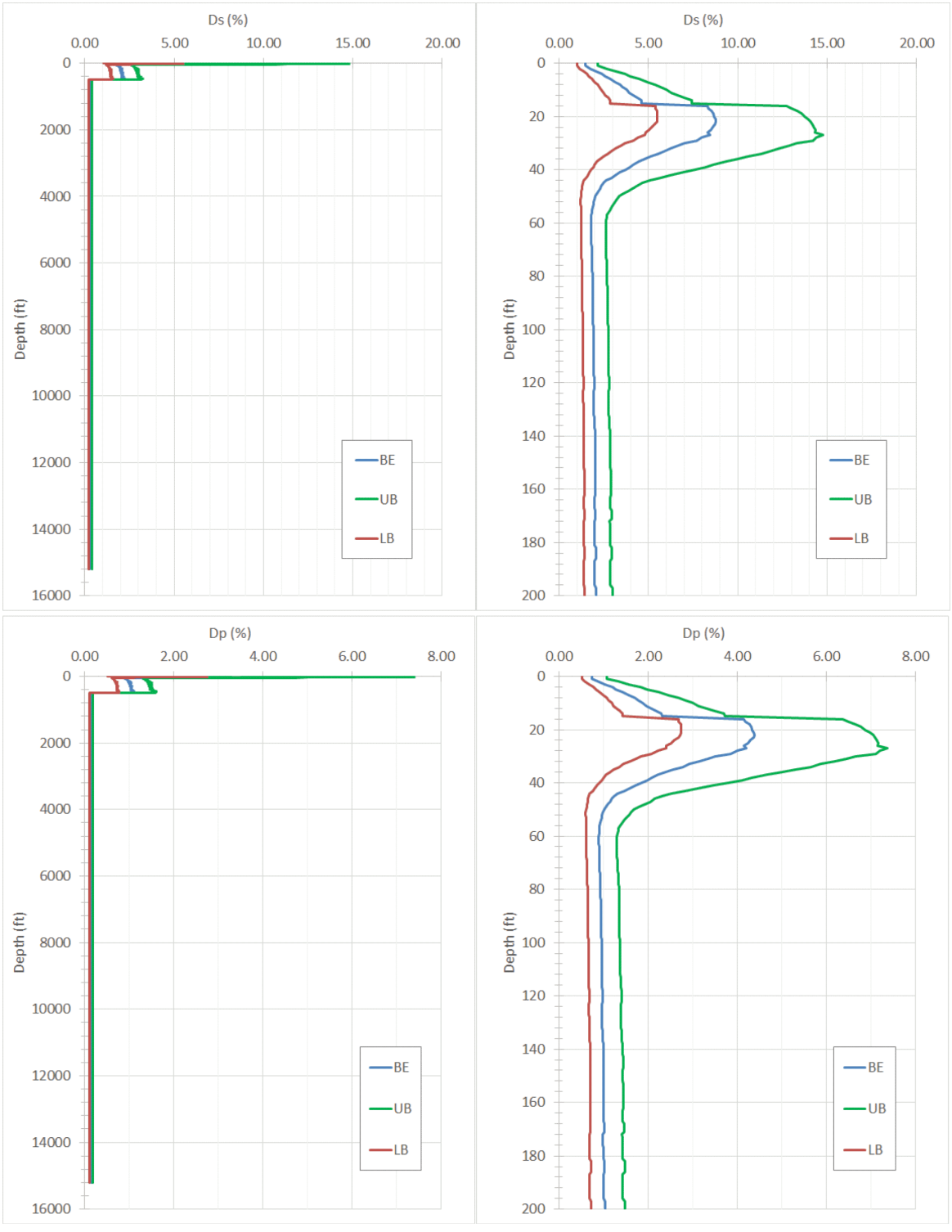


Figure 2.5.2-97R Hazard Consistent Strain Compatible Properties V_S and V_P Corresponding to the FIRS at CRN Site. BE, LB, and UB Profiles are Included. Left: V_S and V_P Profiles Down to the Hardrock Basement. Right: V_S and V_P Profiles in the Top 200 ft

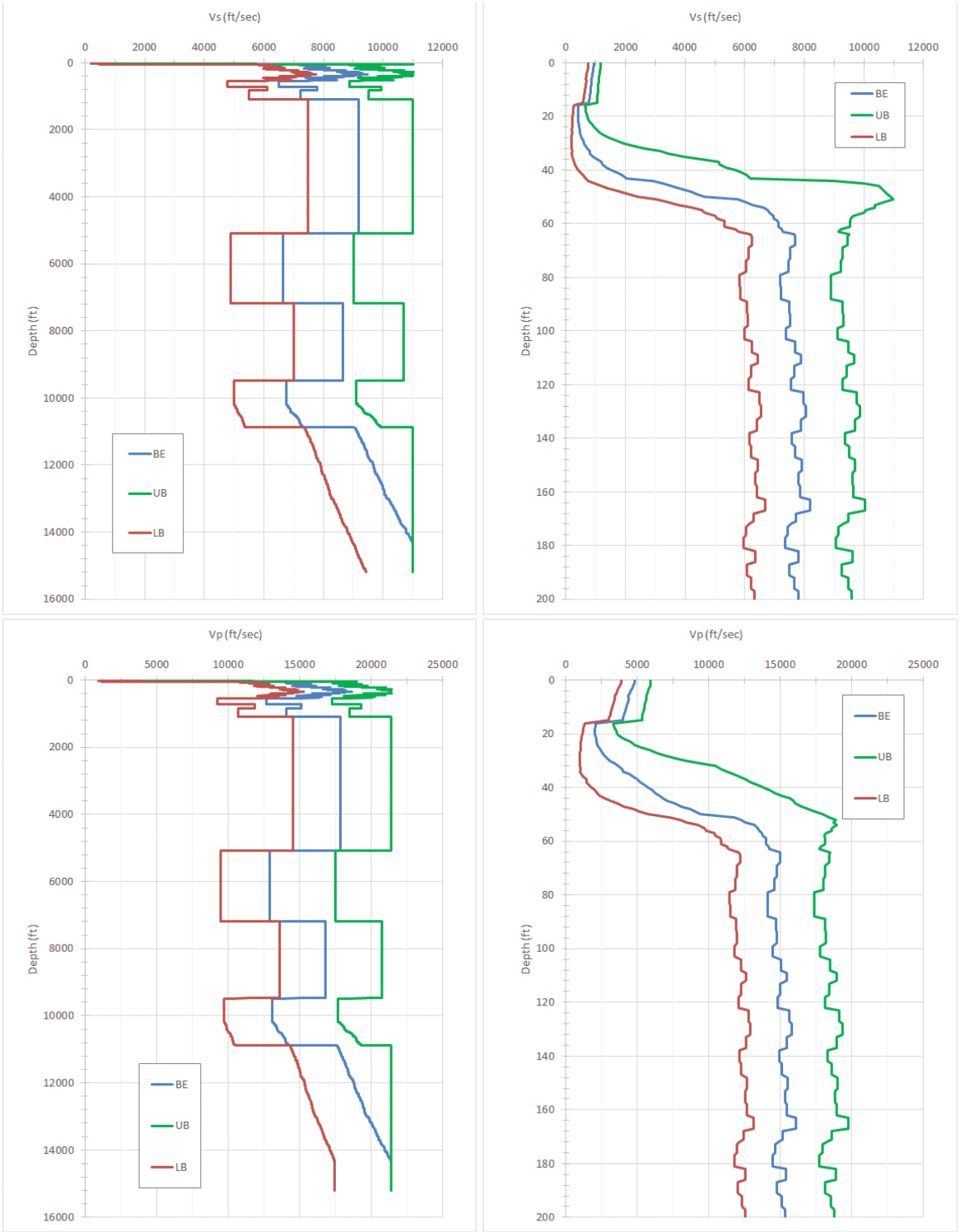


Figure 2.5.2-98R Hazard Consistent Strain Compatible Properties V_S and V_P Corresponding to the FIRS at CRN Site. BE, LB, and UB Profiles are Included. Left: V_S and V_P Profiles Down to the Hardrock Basement. Right: V_S and V_P Profiles in the Top 200 ft

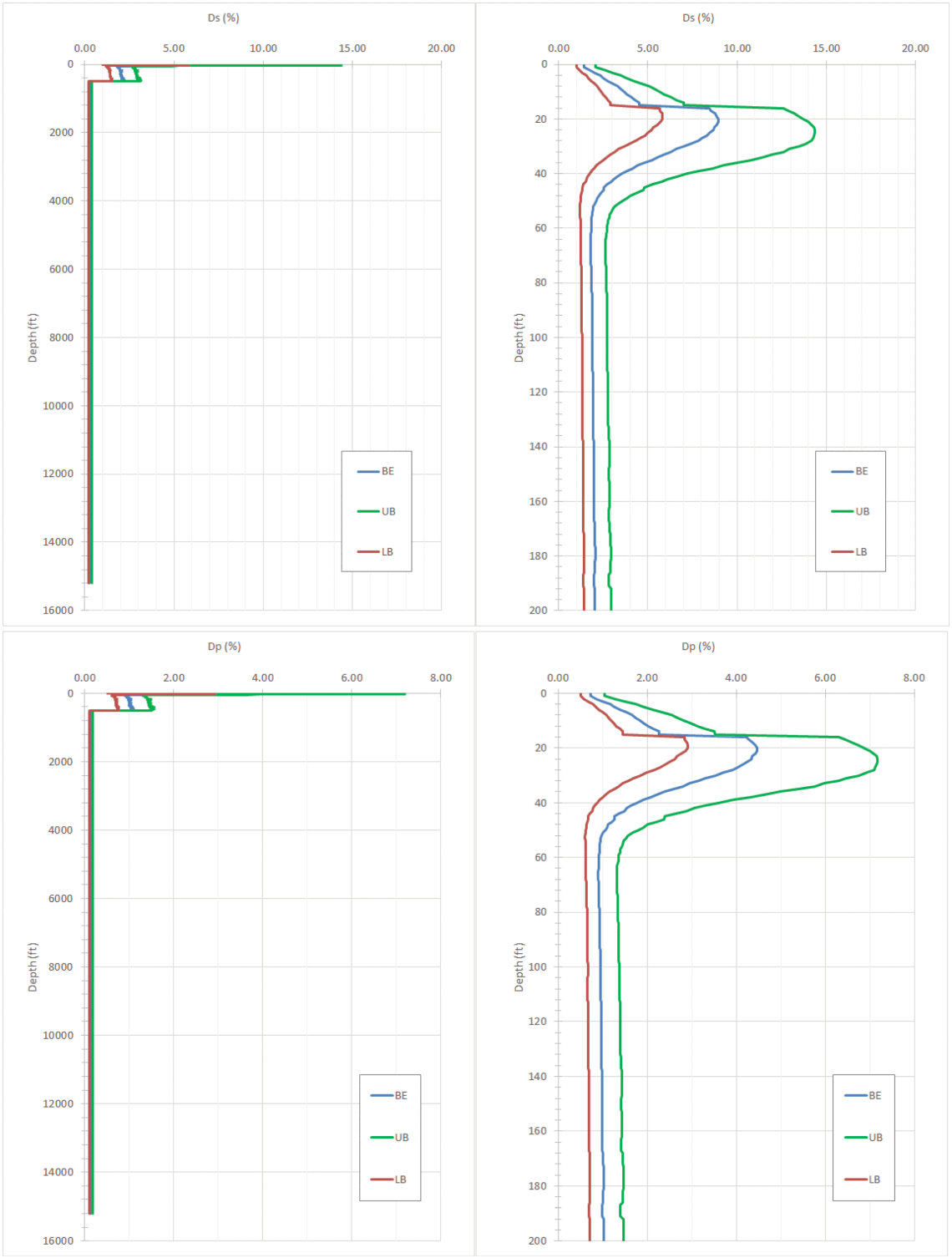


Figure 2.5.2-99R Hazard Consistent Strain Compatible Properties DS and DP Corresponding to the FIRS at CRN Site. BE, LB, and UB Profiles are Included. Left: DS and DP Profiles Down to the Hardrock Basement. Right: DS and DP Profiles in the Top 200 ft

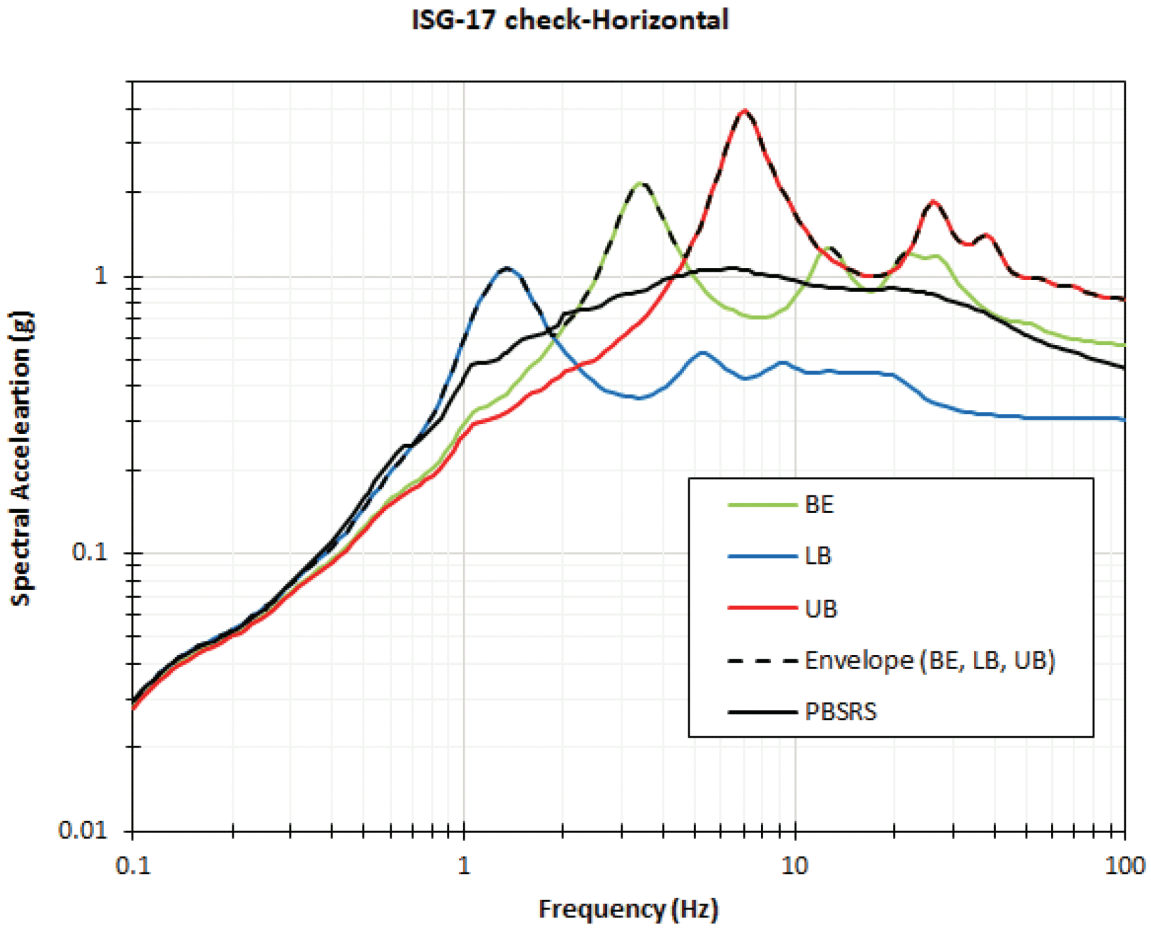


Figure 2.5.2-100R The Horizontal SRS for the BE, LB, and UB Profiles Along with Their Envelope and PBSRS Using the Original FIRS

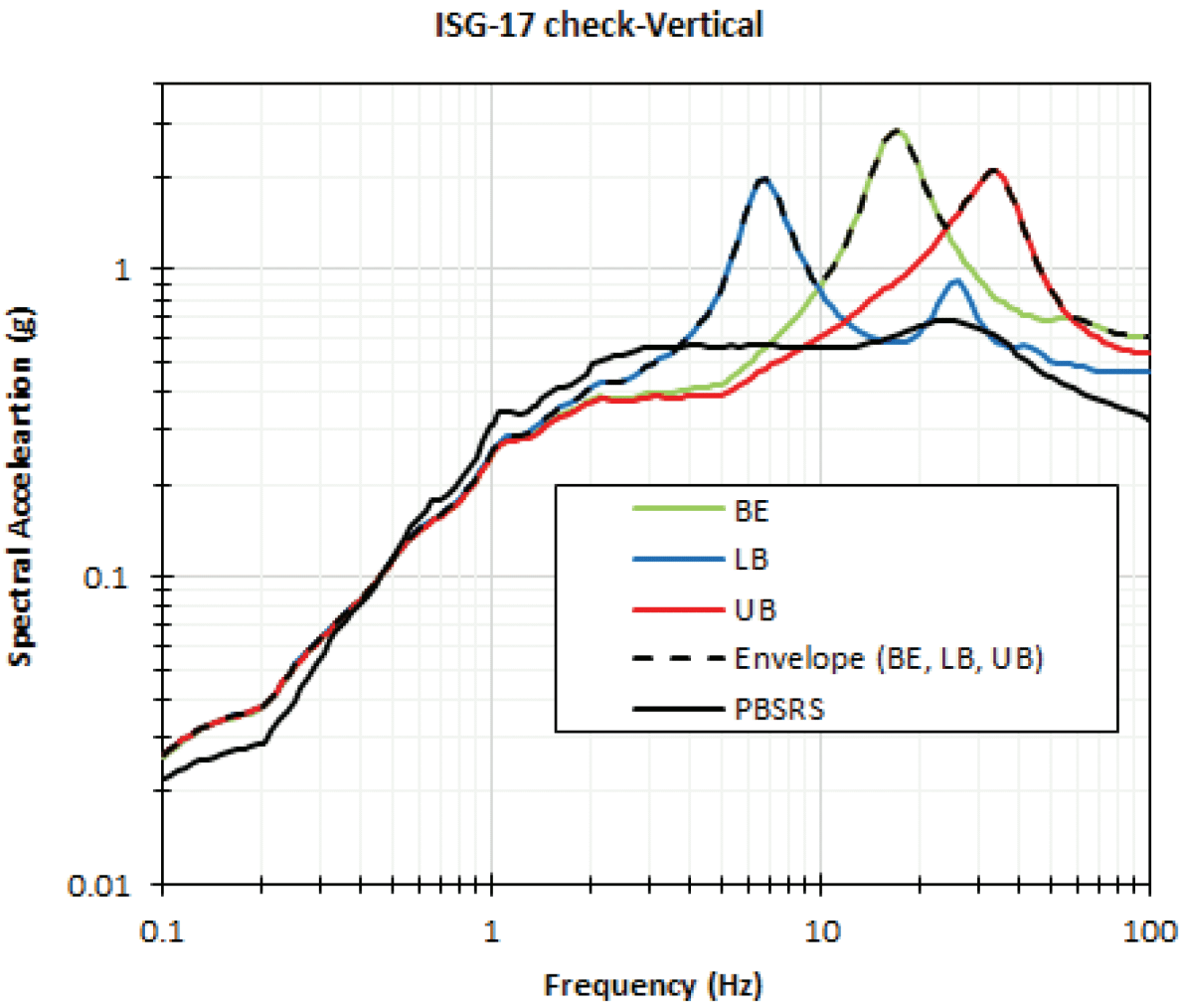


Figure 2.5.2-101R The Vertical SRS for the BE, LB, and UB Profile Along with Their Envelope and PBSRS Using the Original FIRS

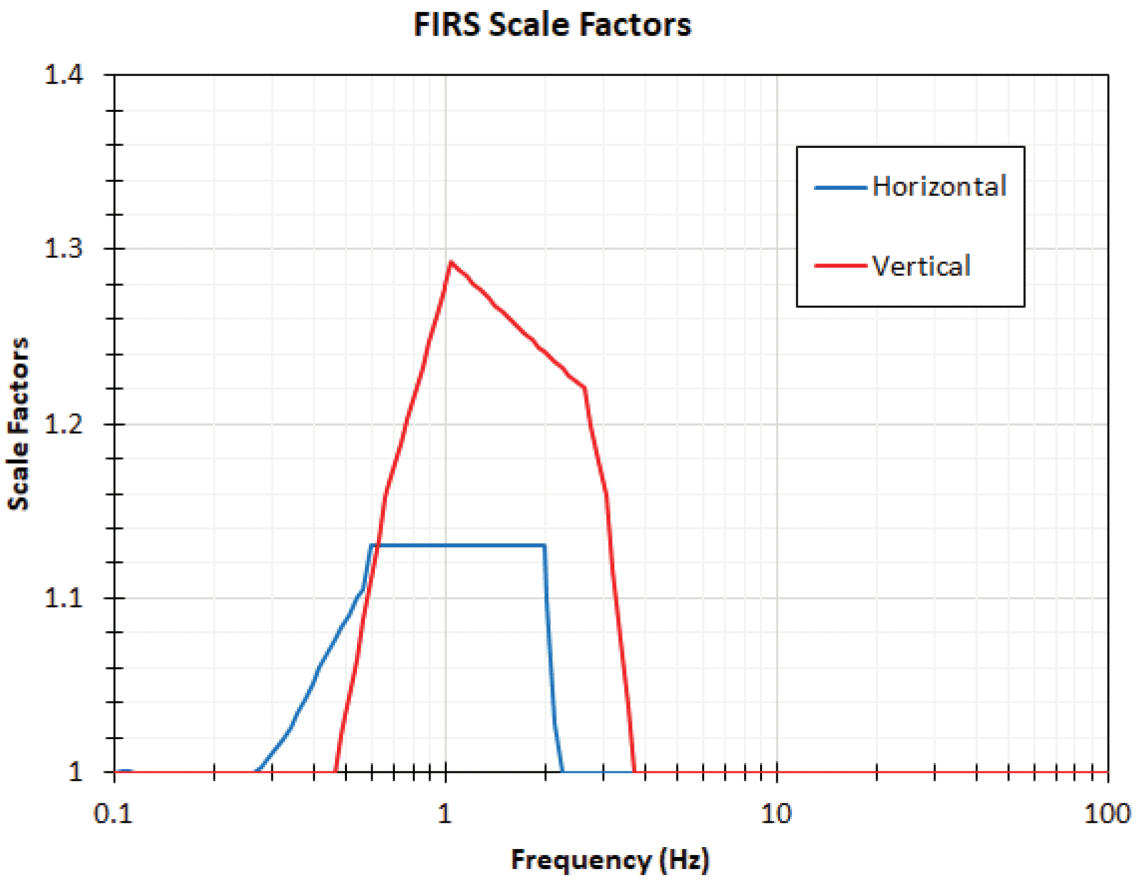


Figure 2.5.2-102R Scale Factors for Adjusting the Horizontal and Vertical FIRS

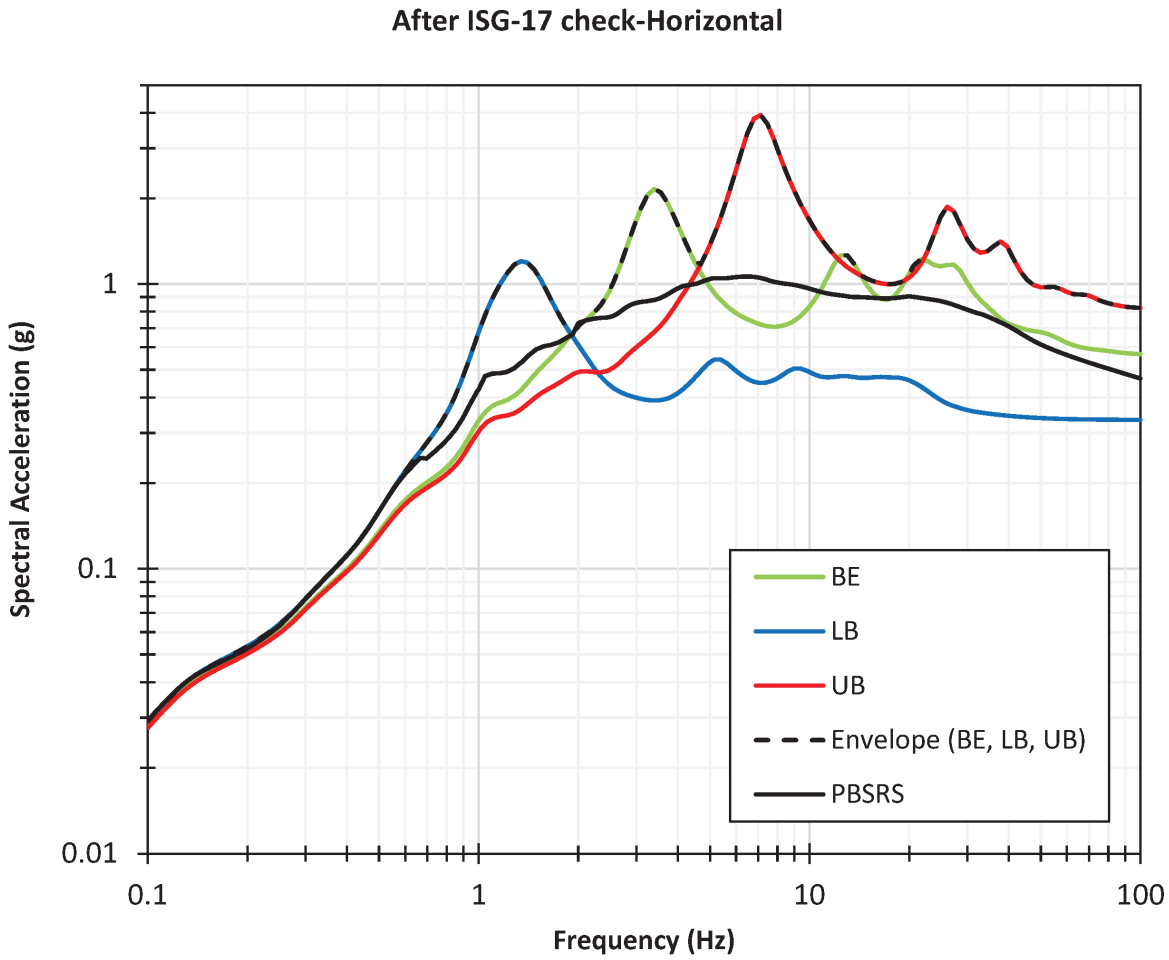


Figure 2.5.2-103R The Horizontal SRS for the BE, LB, and UB Profile Along with Their Envelope and PBSRS Using the Modified FIRS

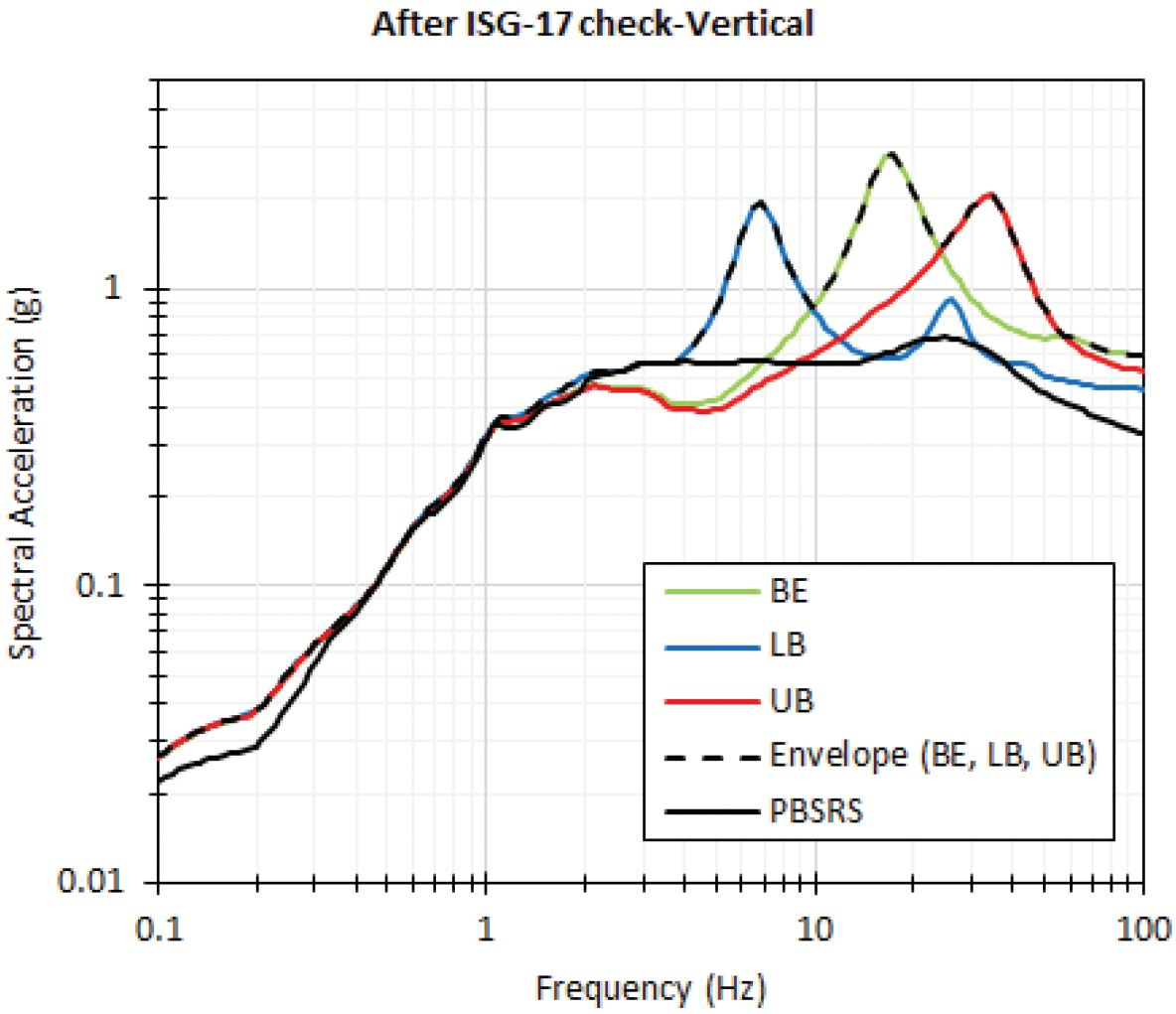


Figure 2.5.2-104R The Vertical SRS for the BE, LB, and UB Profile Along with Their Envelope and PBSRS Using the Modified FIRS

Replace the second Note of ESPA SSAR Figure 2.5.2-130 with the following:

Dashed line corrected for amplification (see Figure 2.5.2-127) and attenuation (Q(f) = (630 f0.5, see Table 2.5.4-30R)

Replace the Note of ESPA SSAR Figure 2.5.2-131 with the following:

Note: Corrected for amplification (see Figure 2.5.2-127) and attenuation (Q(f) = 630 f0.5, see Table 2.5.4-30R), along with kappa fits over the bandwidths considered reliable

2.5.3 Surface Deformation

Replace the second paragraph of ESPA SSAR Section 2.5.3 with the following:

RG 1.208 states, “Comprehensive geological, seismological, geophysical, and geotechnical engineering investigations should be performed.” It also states that these investigations should be performed at four levels with the degree of detail based on distance from the site.

Replace the sixth paragraph of ESPA SSAR Section 2.5.3 with the following:

This section summarizes the evaluation of the site to the above criteria and will establish that there are no significant neotectonic features within the 200-mi CRN Site region that have a potential to impact the site. Several features within the CRN Site region have been proposed to exhibit evidence for Quaternary deformation and are discussed in more detail in Subsection 2.5.3.7.

2.5.3.1 Geological, Seismological, and Geophysical Investigations

Insert the following after the last bullet of ESPA SSAR Section 2.5.3.1:

- Information on the August 9th, 2020, Sparta earthquake including the Department of Energy Y-12 Probabilistic Seismic Hazard Analysis (PSHA) from 2021 (Reference 2.5.3-19).

2.5.3.1.2 Regional and Local Geologic Studies

Replace the last sentence of ESPA SSAR Section 2.5.3.1.2 with the following:

More recent research has been published describing possible paleoseismic evidence for large magnitude ETSZ paleoearthquakes near Douglas Reservoir, approximately 50 mi (80 km) east of the CRN Site (References 2.5.3-12, 2.5.3-13, and 2.5.3-14). Hatcher et al. (Reference 2.5.3-12) describe possible evidence of multiple paleoearthquakes from paleoseismic trenches and geologic mapping (faults, fluidization structures, bleached or clay-filled fractures, and sand dikes), suggesting the most-recent event may have occurred sometime after approximately 73 ka. Warrell et al. (Reference 2.5.3-13) provides additional information on possible paleoseismic features at Douglas Reservoir. Description of these features is presented in Subsection 2.5.3.2.6.

Cox et al. (Reference 2.5.3-20 and Reference 2.5.3-92) describe evidence for surface faulting with ≥6 ft (2 m) of throw exposed in an alluvial terrace along the Little Tennessee River near Vonore, Tennessee. The age of the terrace is not well constrained, and Cox et al. (Reference 2.5.3-20) do not report evidence for multiple events at this site.

In another abstract, Glasbrenner et al. (Reference 2.5.3-35) describe geologic mapping and paleoseismic trenching along the Tellico Reservoir about 22 mi (34 km) southeast of the CRN Site. Based on limited geologic field observations and an apparent seismicity alignment, Cox

et al. (Reference 2.5.3-72), Warrell et al. (Reference 2.5.3-73), and Cox et al. (Reference 2.5.3-92) propose an active, 70 mi (110-km)-long “Quaternary faulting corridor” (referred to as the Dandridge-Vonore fault zone; DVFZ) in the ETSZ. They suggest that this fault corridor may be responsible for at least some of the paleoearthquakes interpreted from exposures near Dandridge, Alcoa, and Vonore, Tennessee. Additional details regarding the proposed DVFZ are provided in a recent field trip guidebook (Reference 2.5.3-74); and were further published in Cox et al. (Reference 2.5.3-92).

Other studies of the ETSZ that post-date development of the CEUS-SSC model have examined the deep crustal structure (Reference 2.5.3-75; Reference 2.5.3-76; Reference 2.5.3-77) and the shallow crustal structure (Reference 2.5.3-78; Reference 2.5.3-79) of the ETSZ. Carpenter et al. (Reference 2.5.3-80) suggests that the **M** 4.2 Perry County, Kentucky, earthquake is related to the ETSZ, thus extending the zone farther north than previously recognized. Yang and Salem (Reference 2.5.3-81) used Permanent Scatterers Synthetic Aperture Radar Interferometry (PSInSAR) data to suggest that land surface deformation may be spatially associated with areas of higher seismicity rate in eastern Tennessee, but caution that many other processes besides earthquake activity can contribute to surface deformation.

A PSHA performed for the nearby Department of Energy Y-12 National Security Complex in Oak Ridge in 2021 (Reference 2.5.3-19) evaluated this new information on the ETSZ to see if it warranted changes to the CEUS-SSC model. This study developed an additional hazard sensitivity analysis to evaluate the postulated Quaternary faulting corridor associated with the Dandridge-Vonore fault zone. The results of DVFZ hazards sensitivity analysis indicate that the DVFZ, as characterized, does not significantly increase hazard at the Y-12 site (Reference 2.5.3-19). The studies further concluded that results from the ongoing paleoseismic studies in the ETSZ are preliminary and do not provide adequate evidence for defining the ages and repeat times of earthquakes and therefore do not provide adequate evidence for defining an RLME seismic source for the ETSZ, and thus no revisions to the CEUS-SSC model are required to account for more recent data from the ETSZ (Reference 2.5.3-19). See Subsection 2.5.2.2.6.1.3 for a discussion of ETSZ Maximum Magnitude (Mmax) sensitivity studies.

2.5.3.2.5.2 Clinch River Terraces

Replace the second sentence of the fourth paragraph of ESPA SSAR Section 2.5.3.2.5.2 with the following:

In order to better constrain the relative ages of terraces along the Clinch River, morphological correlation and longitudinal profiling of terrace elevations was completed along the Clinch River in the CRN Site area, downstream of the Melton Hill Dam (Figure 2.5.3-4).

2.5.3.3 Correlation of Earthquakes with Capable Tectonic Sources

Replace the fourth sentence of the first paragraph of ESPA SSAR Section 2.5.3.2.5.2 with the following:

Twenty-eight earthquakes between **M**2.9 and 4.0 have been recorded within the CRN Site vicinity; of these 28, four have occurred within the site area (Figure 2.5.3-5).

2.5.3.4 Ages of Most Recent Deformation

2.5.3.4.1 Bedrock Faults

Replace the third sentence of the first paragraph of ESPA SSAR Section 2.5.3.4.1 with the following:

In the CRN Site vicinity, there is no evidence for later reactivation of these structures.

Insert the following after the last paragraph of ESPA SSAR Section 2.5.3.4.1:

Within the CRN Site Region, however, there is a recently identified fault related surface rupture attributed to the Mw 5.1 August 9, 2020, Sparta, North Carolina earthquake about 188 mi (302 km) ENE of the CRN Site. This is considered the first documented seismic surface rupture in the eastern United States, leading to the identification of the Little River Fault (Reference 2.5.3-82).

Although the Little River Fault produced strong ground shaking and surface rupture, the geomorphology and focal mechanism analysis of fault seismicity is suggestive of an underlying structural control (Reference 2.5.3-84; Reference 2.5.3-94) and evidence for repeated fault movements within the Quaternary have been documented (Reference 2.5.3-93; Reference 2.5.3-94), it is not considered a hazard to the Site. This is primarily because the Little River Fault only has ~2.5 mi (4 km) of documented surface rupture, which is less than the 4.3 mi (7 km) length criteria needed to be considered a seismic hazard by the U.S. Geological Survey National Seismic Hazard Model (NSHM) for the conterminous United States. For this reason, the Little River Fault is not included as a fault section in the updated 2023 National Seismic Hazard Model database (Reference 2.5.3-84).

Furthermore, the Sparta earthquake is not within the ETSZ (Figure 2.5.2-24R) and is not considered related to the same tectonics as the ETSZ, as it was a shallow event (~0.6 mi (1 km) depth) and all events in the ETSZ have deep sources (4.3-15.5 mi (7-25 km) depth) (Reference 2.5.3-95). Additionally, recent analysis of stress fields in the CEUS using geomechanically constrained stress inversions of earthquake focal mechanisms indicate that the stresses that were responsible for the Sparta earthquake are not compatible with the ETSZ stress field (Reference 2.5.3-96). The location of the Sparta earthquake therefore falls within the Paleozoic Extended Crust seismotectonic zone, in which the CRN Site also resides. The seismic event accordingly does not add any additional hazard for the Site beyond what is already accounted for in the CRN Site PSHA (Subsection 2.5.2). It is speculated that any large event along the Little River Fault near Sparta, North Carolina would most likely be attenuated prior to

reaching the CRN Site, but it is unclear to what degree (Reference 2.5.3-95). The Sparta earthquake and Little River Fault are discussed in more detail in Subsection 2.5.3.7.1.

2.5.3.6 Characterization of Capable Tectonic Sources

Replace the first sentence of the scond paragraph of ESPA SSAR Section 2.5.3.6 with the following:

Seismicity in the CRN Site region is elevated near the CRN Site and is associated with the ETSZ.

Insert the following after the last paragraph of ESPA SSAR Section 2.5.3.6:

The recent 2020 Sparta, North Carolina earthquake occurred within the CRN Site Region and caused surface rupture along the Little River Fault (Reference 2.5.3-82), indicating that it is a capable tectonic source (Reference 2.5.3-95). This Little River Fault is however located outside of the ETSZ, and the Sparta earthquake was determined to have a shallow hypocentral depth (<3 mi (5 km)) (Reference 2.5.3-87) and be a low-angle, NE-vergent reverse fault, indicating that it occurred in different stress field (Reference 2.5.3-96) and therefore does not fit into the characteristics of the ETSZ. The Little River Fault is therefore not considered to pose a hazard to the CRN Site like the ETSZ does (Reference 2.5.3-95). The Sparta earthquake and Little River Fault are discussed in more detail in Subsection 2.5.3.7.1.

2.5.3.7 Designation of Zones of Quaternary Deformation in the Site Region

Insert the following after the first paragraph of ESPA SSAR Section 2.5.3.7:

As of 2020, there is a newly discovered Quaternary fault in northwestern North Carolina named the Little River Fault. This Fault experienced movement, causing surface rupture during the Mw 5.1 August 9, 2020, earthquake near Sparta, North Carolina. The Sparta earthquake produced the first documented seismic surface rupture in the eastern United States allowing identification of the Little River Fault (Reference 2.5.3-82). The epicenter is located approximately 188 mi (302 km) ENE of the CRN Site, placing it within the CRN Site region.

Add new Section 2.5.3.7.1:

2.5.3.7.1 2020 Sparta Earthquake and the Little River Fault

The Little River Fault is of definite Quaternary activity having just been discovered due to its movement causing surface rupture during the August 9, 2020, Mw 5.1 Sparta earthquake, with its epicenter located ~2 mi (~3 km) south of Sparta, North Carolina (36.488° N, 81.106° W) in northwest North Carolina (Reference 2.5.3-82), approximately 188 mi (302 km) ENE of the CRN Site. This earthquake occurred in the allochthonous Neoproterozoic-Ordovician Ashe and Alligator Back metamorphic suites in the Gossan Lead thrust sheet in the eastern Blue Ridge (Reference 2.5.3-85 and Reference 2.5.3-93) and shook much of the eastern United States producing the first documented surface rupture due to faulting east of the New Madrid seismic

zone (Reference 2.5.3-82). The co-seismic surface rupture was initially traceable along a 1.2 mi (2 km) long zone, with offsets indicating predominantly reverse displacement with additional deformation observed south of the main surface rupture as well. High-resolution LiDAR acquired a few months after the earthquake allowed identification of approximately 2.5 mi (4 km) of surface rupture (Reference 2.5.3-93 and Reference 2.5.3-94). Although there were no casualties or significant damage to major infrastructures, damage was recorded to roads, utility lines, and masonry structures, including the collapse of walls and chimneys and the cracking and shifting of foundations (Reference 2.5.3-82).

The Sparta earthquake was the largest in North Carolina in nearly 100 years and the strongest in the eastern United States since the 2011 Mw 5.8 Mineral, Virginia, earthquake. The earthquake sequence included eight foreshocks (Mw 1.8-2.6) during the previous 24 hours before the mainshock and at least 300 aftershocks over the next six months (up to Mw 2.9) (Reference 2.5.3-82). The main shock of the seismic event had a maximum intensity of VI-VII in Sparta and generated peak ground accelerations of at least 0.2 g (Reference 2.5.3-82). According to the USGS, (Reference 2.5.3-86) an intensity of II was recorded at the CRN Site area. This deformation is unique as surface ruptures by faulting are rarely reported for earthquakes with magnitude less than **M** 6.0, and it occurred in a very low strain rate intraplate setting, where earthquake surface deformation is rare (Reference 2.5.3-82).

The mainshock hypocenter and focal mechanism have been estimated independently by several groups (Figure 2.5.3-23), namely the Center for Earthquake Research and Information at the University of Memphis (CERI) (Reference 2.5.3-87), the Saint Louis University Earthquake Center (Reference 2.5.3-88) and the United States Geological Survey (Reference 2.5.3-86). While the first two entities calculated the depth of the hypocenter at 0.6 mi (1 km) or less, the USGS calculated their preferred hypocenter location to be 2.5 mi (4.1 km) (Reference 2.5.3-86 and Reference 2.5.3-82). Although all three calculations resulted in a NW-SE to WNW-ESE striking fault plane with dips between approximately 50-60° SW for the focal mechanism, the preferred USGS (Reference 2.5.3-86) orientation is 176, 48° W for the nodal plane (Reference 2.5.3-82). Figueiredo (Reference 2.5.3-94) discounts the USGS determination of hypocenter depth and nodal plane because their closest stations were approximately 30 mi (50 km) away. These orientations are all oblique to the regional NE-SW structural trend.

A local seismic monitoring network was deployed within 48 hours of the initial earthquake by CERI, and seismic monitoring of continued minor aftershocks (**M** 1-2), which were still taking place years after the mainshock, indicate that while some events occurred at approximately 3.7 mi (6 km) depth, the majority are in the 0.6-2.5 mi (1-4 km) range (Reference 2.5.3-94). Analysis of these aftershocks' seismicity has aided in refining the mainshock hypocenter depth to ~0.8 mi (1.3 km) (Reference 2.5.3-94), considerably shallower than the USGS preferred depth. A depth of around 0.6 mi (1 km) or shallower for the mainshock is however the currently accepted hypocentral depth range (Reference 2.5.3-93 and Reference 2.5.3-94).

This faulting occurred on the previously unidentified and newly named Little River Fault due to its proximity to the Little River which runs through Sparta (Figure 2.5.3-24). The fault is determined to be a west-northwest striking (~110°), south-southwest dipping (45-60°), left lateral oblique reverse fault (Reference 2.5.3-89). The sense of motion is supported by the recorded

displacement of geodetic monuments on the hanging wall and footwall. The regional maximum horizontal stress field (SHmax) is oriented NE-SW to ENE-WSW (Reference 2.5.3-90), consistent with an oblique-reverse focal mechanism and overall reverse fault characteristics on a WNW-ESE striking fault plane.

The majority of the surface rupture was initially identified to the south and southeast of Sparta, stretching for ~1.2 mi (2 km) and across a generally < 80 ft (25-m)-wide zone, but there was no major surface deformation identified near the epicenter (Reference 2.5.3-82). Evidence incudes straight, tens to hundreds of meters long, 110-120°-trending, occasionally en échelon, ground ruptures with an adjacent scarp formed by reverse motion folding and flexure of the topography, with some associated fractures/splays that vary more NW in orientation (Reference 2.5.3-93). The scarp has an average height of 8-12 in (20-30 cm) and a maximum of ~20 in (50 cm) in one anomalous location (Figure 2.5.3-25) (Reference 2.5.3-82 and Reference 2.5.3-93). Along the rupture, which has four different sections (Figure 2.5.3-24), the southern block is consistently uplifted indicating complex oblique reverse motion.

Away from the main fault trace, other co-seismic surface deformation resulting from ground shaking or translational sliding was recognized such as ground fissures in less consolidated material in road cuts and riverbanks southwest of Sparta (Reference 2.5.3-82). Features include the opening of fissures that are centimeters to decimeters in length and millimeters to centimeters wide in the Little River riverbanks and fissures extending for several meters that exhibit shortening and extensional displacement in an industrial parking lot believed to be related to deformation along several preexisting pavement discontinuities (Reference 2.5.3-82).

Subsequent LiDAR analysis of data acquired after the mainshock, and compared to 2016 LiDAR data, revealed that the surface rupture is actually approximately 2.5 mi (4 km) long (Reference 2.5.3-93 and Reference 2.5.3-94). En echelon, subparallel lineaments observed in the LiDAR extending to the northwest (believed to be unruptured in 2020 event) are suspected older segments of the Little River Fault (Figure 2.5.3-24). This data indicates that the Little River Fault length could extend to 3.7 mi (6 km) (Reference 2.5.3-93) or potentially to 7.5 mi (12 km) if all lineaments are actual expressions of the Little River Fault (Reference 2.5.3-94). Comparison of the post-rupture LiDAR data with older LiDAR data (2016) indicates left lateral motion detected up to 1.2 mi (2 km) from rupture and vertical deformation up to 330 ft (100 m) away. However, 2016 data could have other factors influencing the overall differences (e.g. a location of surface lowering is related to erosion) (Reference 2.5.3-94).

Detailed geologic mapping reveals individual fault strands occur in an en-echelon pattern within a 39 ft- (12 m-) wide zone and has identified several strands of brittle faults oriented similar to the co-seismic rupture and fault that are characterized by deformation zones 0.4-10 in (1 to 25 cm) wide, often with several brittle faults localized within several meters in outcrops (Reference 2.5.3-93). The Paleozoic foliation in the bedrock can be seen fractured and rotated into the fault, with microbreccia and manganese-coated slickenside clay seams present along some fault fractures with variable orientations in saprolitic bedrock exposed adjacent to the Little River Fault (Figure 2.5.3-25). Soft-sediment deformation in alluvium has also been documented at multiple locations along river and stream banks along sections of other NW-trending lineaments (Reference 2.5.3-93). Cosmogenic radionuclide analysis of gravel in colluvium

overlying one of the fault strands in the saprolite has given depositional ages of 500 ka, indicating that faulting occurred along the fault zone during the Pleistocene (Reference 2.5.3-93). These observations highly suggest paleoseismicity and reactivation of the Little River Fault through at least Quaternary time.

Four trenches (two initial small excavations and two subsequent traditional paleoseismic trenches) excavated across the surface rupture reveal that this earthquake propagated to the surface along a preexisting structure in the shallow bedrock, which had not been previously identified as an active fault. Additionally, ground-penetration radar and electrical tomography resistivity profiles were acquired normal to the rupture trace, imaging a m-wide south-dipping fault zone (Reference 2.5.3-97). Features in the initial small excavations include a low-angle thrust fault in one trench with ~4 in (10 cm) of fault slip in surficial fill deposits causing 1.5 in (4 cm) of vertical displacement which roots into a steeper preexisting plane (N 115°,S 45°) in underlying weathered bedrock, interpreted to be associated with the Paleozoic fabric (Reference 2.5.3-82). Both dip-slip and oblique slickenlines were identified in the low-angle and steeper fault surfaces (Reference 2.5.3-89). Older fault gouge with manganese-coated surfaces and breccia were observed in the weathered bedrock indicating brittle deformation attributed to previous movement (of unknow age, but probably Cenozoic) along the Little River Fault (Reference 2.5.3-82).

The later traditional paleoseismic trenches revealed a complex fault zone ~50 ft (15 m) wide, with multiple WNW-ESE to NW-SE strands where a low grade metamorphosed greywacke is thrusting amphibolite (Reference 2.5.3-97). They also show that the movement during 2020 Sparta earthquake was accommodated by at least two separate fault splays striking ~110° and dipping 45-60° SSW that offset both bedrock and overlying middle to late Pleistocene soil and colluvium (Reference 2.5.3-94). While one of the fault splays was responsible for the surface rupture, the other splay fault did not rupture the surface but did have movement during the 2020 event (Reference 2.5.3-94).

Based on observations in the trenches as well as electrical resistivity tomography data, the deformation zone along the fault ranges between 33-66 ft (10-20 m) and overall vertical displacement is a combined 20 in (50 cm) including surface and subsurface (Reference 2.5.3-94). At least three separate seismic events on the two fault splays in the two traditional paleoseismic trenches (about 0.6 mi (1 km) apart) were observed, indicating recurrent activity on the Little River Fault (Reference 2.5.3-94). Preliminary optically stimulated luminance (OSL) data on faulted sediments indicates the three separate events took place in the last 50 k.y. equating to about 12 in (30 cm) per event during the Holocene (Reference 2.5.3-94).

While the derived focal mechanisms, Interferometric Synthetic Aperture Radar (InSAR) data and geodetic monuments indicate oblique-reverse motion, the geologic field evidence suggests predominantly reverse faulting and associated folding along a low-angle (~20°) fault plane that roots in a steeper preexistent fabric (~50°) (Reference 2.5.3-82). It is speculated that this discrepancy could indicate complex strain partitioning or a change of slip during the earthquake, with the initial, deep rupture having been predominantly strike-slip along a steeper structure with up-dip failures occurring on shallower preexisting planar discontinuities in the bedrock accommodating most of the reverse component (Reference 2.5.3-82).

Based on continued focal mechanism analysis of seismicity of aftershocks, it was subsequently suggested that the fault that caused surface rupture was a shallow reverse splay fault that roots into a deeper main, blind, subvertical, left lateral, strike-slip fault (Reference 2.5.3-94). The surface rupture fault, as well as the other subparallel reverse fault strands are splays off of this main fault which is oriented ~150 degrees and is believed to reach within 0.6 mi (1 km) of the surface (Reference 2.5.3-94). Total rupture length is estimated to be 2.5 mi (4 km) on reverse fault splays and 2 mi (3 km) on blind subvertical strike slip main fault for a cumulative rupture length of 4.3 mi (7 km) between the main and splay faults (Reference 2.5.3-94).

Modeling of InSAR data deformation field for the Sparta earthquake recorded by the Japan Aerospace Exploration Agency's Advanced Land Observing Satellite #2 is suggested to indicate Little River Fault surface rupture is associated rather with a very shallow normal faulting event (Reference 2.5.3-91). This is in opposition to the seismic movements indicated by Figueiredo et al. (Reference 2.5.3-82) for the Sparta earthquake surface rupture. However, Wick and Chiu (Reference 2.5.3-91) do not contest that the overall seismic motion was not reverse in movement, rather they suggest that the normal fault could be an induced secondary effect of near-surface shear failure on a pre-existing structure, like a fracture or foliation plane. They suggest that the Little River Fault rupture is an observed interaction between the seismically recorded reverse fault and the geodetically recorded normal fault. The normal fault likely progressed aseismically, perhaps induced by the seismic reverse fault with unmapped small-scale tectonic features aiding in localizing the two slip events (Reference 2.5.3-91). Other than the InSAR based argument put forth by Wicks and Chiu (Reference 2.5.3-91), there is no evidence for normal movement, and it is assumed to be inaccurate analysis (Reference 2.5.3-94).

Although the Little River Fault produced strong ground shaking and documented surface rupture, it is not included as a Quaternary fault section in the 2023 National Seismic Hazard Model database because the ~2.5 mi (4 km) long surface rupture is less than the 7 km long criteria to be included in the database. At the time of publishment of the database, however, there was not yet published evidence for recurrent Quaternary deformation along the Fault (Reference 2.5.3-84). Though not yet unequivocally documented (Reference 2.5.3-83), subsequent published abstracts (Reference 2.5.3-97) and unpublished data as discussed here (Reference 2.5.3-93 and Reference 2.5.3-94) describes evidence for repeated fault movements throughout the Quaternary, which now supports classification of the Little River Fault as a Quaternary fault section.

As discussed here, the surface rupture is located along a subtle at least 6 mi (10-km) long WNW-ESE-trending topographic lineament visible in digital elevation models and centered on the fault zone (Reference 2.5.3-82) (Figure 2.5.3-24). The Little River Fault is possibly part of a WNW-ESE set of regional lineaments and faults inferred to have Cenozoic activity that cross the dominant NE-SW Appalachian structural grain, such as the Boone and Mills Gap Faults, located 30 mi (50 km) and 93 mi (150 km) to the southwest, respectively, to which it is subparallel (Reference 2.5.3-83). Other E-W trending lineaments in North Carolina such as the Mesozoic Swannanoa and Laurel Creek lineaments exist that have no known modern seismicity, but there has been no analysis of paleoseismicity along those features that has been found in the literature (Reference 2.5.3-95).

The Sparta earthquake, along with the 2011 Mw 5.8 Mineral, Virginia earthquake, both occurred along structures not included in any database as potentially active, suggesting that the regional seismic hazard may be underestimated in the Eastern United States (Reference 2.5.3-82). However, since the Sparta earthquake is not within the ETSZ (Figure 2.5.2-24R) and therefore falls within the Paleozoic Extended Crust seismotectonic zone, in which the CRN Site also resides, the seismic event does not add any additional hazard for the Site beyond what is already accounted for in the CRN Site PSHA (Subsection 2.5.2).

2.5.3.9 References

Replace the following References in ESPA SSAR Section 2.5.3.9:

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2.5.3-6 Project Management Corporation, Clinch River Breeder Reactor Project - Preliminary Safety Analysis Report, Vol. 3, p. 1982.

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2.5.3-20 Cox, R.T., E. Gamble, R.D. Hatcher Jr., J.C. Glasbrenner, and R. Counts, “Surface faulting and possible kinematics of the East Tennessee Seismic Zone,” Poster and abstract from the 2014 Geological Society of America Meeting in Vancouver, British Columbia, Canada, paper no. 319–10, 2014.

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Add the following References in ESPA SSAR Section 2.5.3.9:

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Add new Figures 2.5.3-23 through 2.5.3-25:

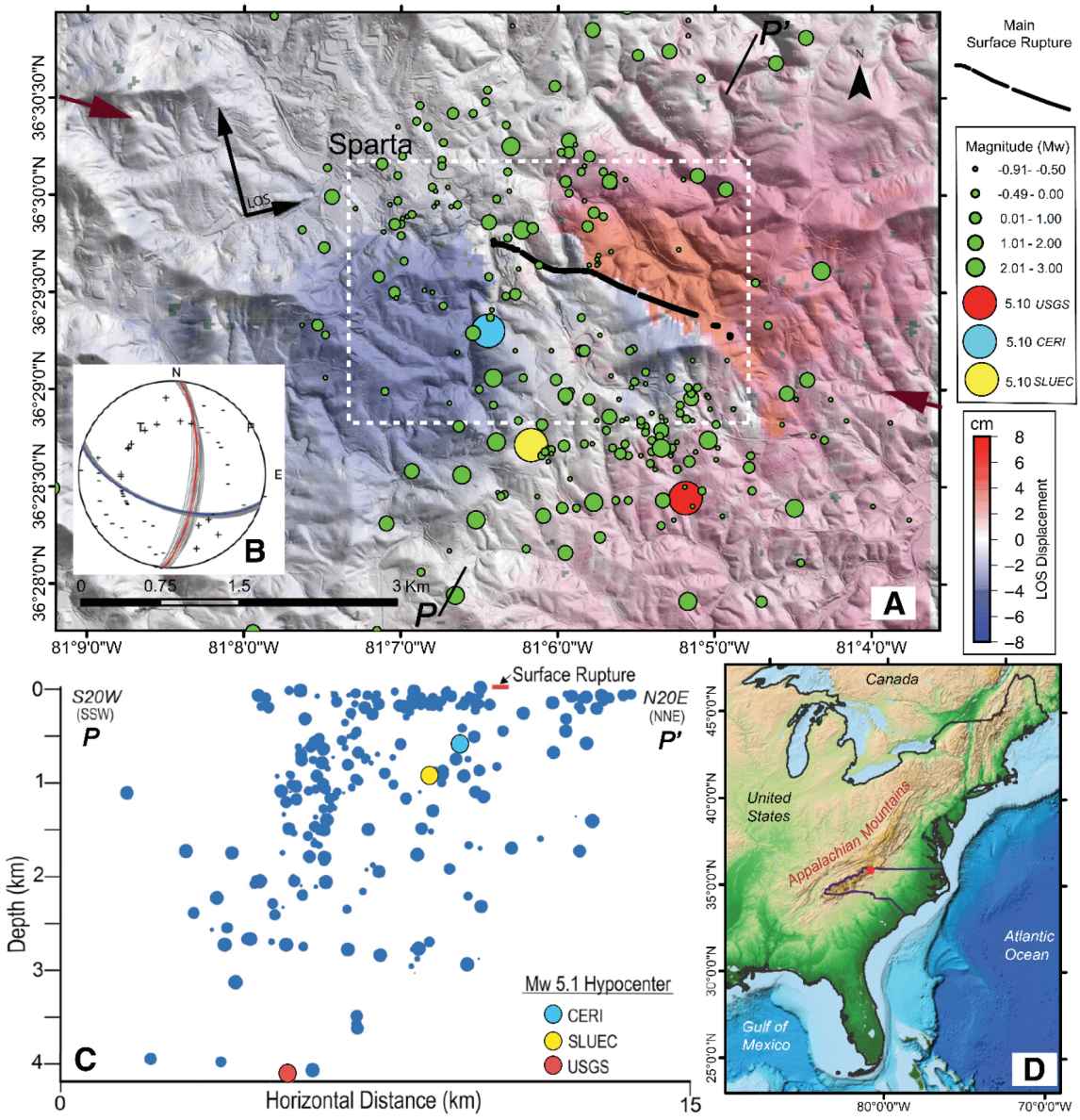


Figure 2.5.3-23 Location, Earthquake Sequence, and Interferometric Synthetic Aperture Radar Interferogram for the Sparta Earthquake

(A) Unwrapped phase interferogram overlaying a lidar-derived hillshade model with the main surface rupture (black line) and August 2020–February 2021 instrumental seismicity (circles; USGS catalog). Topographic lineament marked by brown arrows. Figure 2 location denoted by the dashed white rectangle. Line P-P' indicates the projection plane for seismicity. (B) Focal mechanism solution (Horton et al., 2021). (C) Projection of seismic sequence (USGS catalog) into a plane with azimuth N20°. (D) Location of the earthquake (red) in eastern North America, with North Carolina outlined.

LOS–line of sight; CERI–Center for Earthquake Research and Information, University of Memphis; SLEUC–Saint Louis University Earthquake Center. (Reference 2.5.3-82)

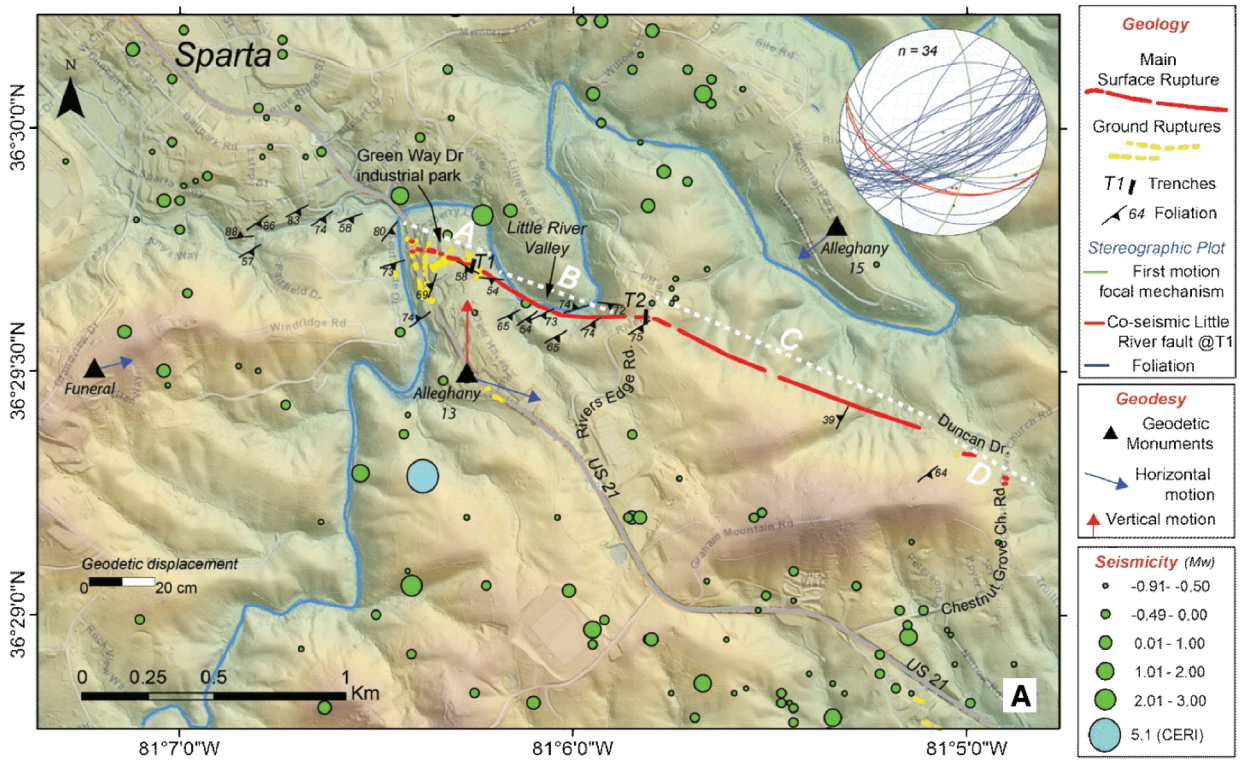


Figure 2.5.3-24 Co-seismic Main Surface Rupture Along the Little River Fault

Main surface and ground ruptures, locations of displaced geodetic monuments, and earthquake sequence. White dashed lines indicate the surface rupture sections (A–D). The stereogram displays the focal mechanism (green), measured fault (red), and foliation (blue) orientations. (Reference 2.5.3-82)

Figure 2.5.3-25 Examples of Deformation Associated with the Little River Fault

A) Location of surface rupture along Little River Fault. Person standing on upthrown side of fault; red line along top of the scarp; arrows point to scarp. B) Location of surface rupture along Little River Fault. persons legs straddling scarp (left foot on upthrown side); red line along top of the scarp; arrows point to scarp. C) Location of surface rupture along Little River Fault; red line along top of the scarp; arrows point to scarp; note foot in upper right for scale. D) Fault surface in saprolite indicating older faulting. Note black manganese mineralization along fault trend.

2.5.4 Stability of Subsurface Materials and Foundations

Replace the first sentence of the first paragraph of ESPA SSAR Section 2.5.3 with the following:

This subsection presents information on the stability of subsurface materials and foundations at the Clinch River Nuclear (CRN) Site for the early site permit application (ESPA).

Replace the first sentence of the second paragraph of ESPA SSAR Section 2.5.3 with the following:

Information presented in this subsection has been developed from a 2013 CRN SMR subsurface investigation (Reference 2.5.4-1) performed between June 2013 and March 2014 (see Subsection 2.5.4.3.2).

Replace the third paragraph of ESPA SSAR Section 2.5.4 with the following:

Supplemental site-specific field exploration (2023 BWRX-300 SMR investigation) and testing were planned and conducted to supplement the existing site characterization data for the selected GVH BWRX-300 reactor technology. The purpose of the additional testing and exploration program is to ensure that the site-specific safety requirements were met, and potential hazards are identified and addressed before construction begins. The raw data from the 2023 BWRX-300 SMR investigation program are presented in Reference 2.5.4-66 and Reference 2.5.4-67. These results provide additional data to supplement the existing datasets (from the ESPA) to inform any necessary modifications to the design (if necessary) at the later stages of the project.

2.5.4.1 Geologic Features

2.5.4.1.1 Stratigraphy

Replace Figure 2.5.4-2 with Figure 2.5.4-2R in ESPA SSAR Section 2.5.4.1.1.

2.5.4.1.2 Previous Loading History

Replace the first two sentences in the second paragraph ESPA SSAR Section 2.5.4.1.2 with the following:

Table 2.5.4-2 shows the approximate differences between the historic ground surface elevations prior to site development for the CRBRP and the 2013 CRN SMR subsurface investigation. The locations of these borings are shown on Figure 2.5.4-1R.

Replace ESPA SSAR Section 2.5.4.1 with the following:

The following discussions focus on geologic features at the CRN Site. More detailed descriptions of geologic features are presented in Subsection 2.5.1. The information is taken from the 2013 CRN SMR investigation program (outlined in Subsection 2.5.4.3) and the 2023 BWRX-300 SMR supplementary investigation. For reference, the existing site elevations range from approximately 855 (MP-406) to 780 feet (ft) (MP-207, MP-211) with an average elevation of about 810 ft for the

CRN Site, and around 800 ft for the Unit 1 location. A finished plant grade elevation of 814.5 ft North American Vertical Datum of 1988 (NAVD88) is expected as the final plant grade elevation. Foundation embedment for the vertical shaft of the Reactor Building (RB) is at a depth of 115 ft (699.5 ft) below finished grade.

All references to elevation given in the following subsections are to NAVD88, with the exception of elevations pertaining to the CRBRP which are with respect to the National Geodetic Vertical Datum of 1929 (NGVD29). In the project area, there is generally a difference of less than 0.5 ft between NAVD88 and NGVD29, with NGVD29 being greater. Likewise, all exploratory point locations are provided relative to Tennessee State Plane Coordinates in North American Datum of 1983 (NAD83) and all references to depth are below the existing ground surface unless stated otherwise.

2.5.4.1.3 Discontinuities, Shear-Fracture Zones and Weathered/Fracture Zones *(Addresses ESP COL Action Item 2.5-1)*

2.5.4.1.3.1 Discontinuities

Replace the first sentence of the second paragraph of ESPA SSAR Section 2.5.4.1.3.1 with the following:

ATV data collected during the 2013 CRN SMR subsurface investigation (Appendix C, Reference 2.5.4-1) indicates that the highest frequency of joints occurs in the upper 100 ft of bedrock.

Insert the following after the third paragraph of ESPA SSAR Section 2.5.4.1.3.1

The supplemental site-specific field exploration (2023 BWRX-300 SMR investigation) and testing have been performed with considerations on the selected BWRX-300 SMR reactor technology. One of the objectives of the additional testing and exploration program was to inform the description of the discontinuities surrounding the vertical shaft of the Reactor Building (RB).

The 2023 BWRX-300 SMR investigation ATV data are broadly consistent with that of the 2013 data, though with a much smaller dataset due to the lesser number of ATV data collected. The strike of the 2023 fractures ranges predominantly between 220- and 250-degrees azimuth with a mean of 234.4 degrees, while the dips predominantly range between 310- and 340-degrees azimuth with a mean of 324.4 degrees. The two dominant sets in the 2023 fracture data are consistent with the dominant sets in the 2013 dataset in that both are oriented parallel to strike. While the most dominant set is most consistent (mean orientation of 234/56), the subordinate set dips SE, close to bedding orientation, but has overall a steeper average dip and slightly rotated strike (mean orientation of 075/54) than the secondary set in the 2013 data. Greater details of the 2023 BWRX-300 SMR subsurface investigation fracture sets and comparison to early investigation data are discussed in Reference 2.5.4-66.

2.5.4.1.3.2 Shear-Fracture Zones

Replace the first sentence of the first paragraph of ESPA SSAR Section 2.5.4.1.3.2 with the following:

A detailed discussion on the shear-fracture zones encountered in the 100- and 200-series borings (from the 2013 investigation at Locations A and B, respectively) at the CRN Site is provided in Subsections 2.5.1.2.4 and 2.5.1.2.6.4.

Replace the second paragraph of ESPA SSAR Section 2.5.4.1.3.2 with the following:

According to the proposed layout of Unit 1, the vertical shaft of the RB will be constructed within the Benbolt Formation. The shear-fracture zones are incorporated in the average Geological Strength Index (GSI) rating for each stratigraphic unit for rock mass characterization (Subsection 2.5.1.2.6) in order to characterize the subsurface condition at the RB location.

Characterization of the shear-fracture zones are based on the 100-, 200-, and 400-series borings drilled at the CRN Site during the 2013 CRN SMR investigation. Similar healed shear-fracture zones were identified during the 2023 field investigation for the selected BWRX-300 reactor technology. They were identified and characterized in only 8 of the 16 rock core boreholes drilled. These healed shear-fractures were identified in depth intervals corresponding to the Rockdell Formation (25 intervals out of 28 described intervals), the Benbolt Formation (2 intervals) and one occurrence in the recently first-time drilled Witten Formation. The presence of these healed shear-fractures is not observed to be laterally continuous zones through multiple borings with shear-fracture zone features in the Benbolt only occurring in the upper strata of the formation, while those identified in the Rockdell Formation appear to comprise different localized zones all throughout the formation thickness. The only occurrence in the Witten Formation was described in the bottom third of the formation. These features as identified in the 2023 subsurface investigation are discussed in more detail in Reference 2.5.1-66.

Based on the determined age of the shear-fracture zone features during Alleghenian shortening, they are not considered an active hazard to the CRN Site. It is furthermore suggested that the previously established term “shear-fracture” for these features is a misnomer within the site, as they are not traditional tectonic shear zones. Regardless, they are factors in the foundation rock mass assessment for bearing capacity and further characterization of such features, if identified, shall be performed via geologic mapping during excavation for the deeply embedded RB shaft.

Delete the third paragraph of ESPA SSAR Section 2.5.4.1.3.2.

2.5.4.1.3.3 Weathered and Fracture Zones

Delete the third paragraph of ESPA SSAR Section 2.5.4.1.3.3.

2.5.4.1.4 Karst Features

Replace the second and third paragraph of ESPA SSAR Section 2.5.4.1.4 with the following:

Contrary to previous drilling investigations at the CRN Site in which many dissolution features larger than 0.1 ft were described, only two (2) of such cavity size were described in cored rock samples recovered and described in the 2023 BWRX-300 SMR investigation. One cavity was described in the rocks of the Witten Formation, while the other cavity was interpreted as a 0.9 ft maximum size based on core loss within rocks of the Rockdell Formation (Reference 2.5.4-66).

A discussion on the cavities encountered in the borings drilled as part of the 2013 and 2023 investigations is provided in Subsection 2.5.1.2.6. A number of cavities encountered in the Rockdell Formation (MP-424) are below the deepest foundation embedment depth (El. 699.5 ft). These (five) cavities were encountered at average cavity elevations between approximately 660 to 680 ft, and they ranged in height from 0.7 to 4.3 ft (Table 2.5.1-11 (Sheets 1-6 of 7) and Table 2.5.1-11R (Sheet 7 of 7)).

Additionally, a three dimensional FE model was developed to evaluate the impact of potentially un-detected karstic cavities within the Benbolt formation, where BWRX-300 Unit 1 RB will be located. The selection of the size and location of the karst cavities and disturbed rock mass zone was in part informed by the ESPA SSAR Subsection 2.5.4 (Reference 2.5.4-64). It is noted that in order to simulate the actual condition at the CRN Site, the size of the cavities in those models were compatible with the height of the cavities detected and observed in the 2013 CRN SMR investigation program. Consistent with ESPA cavity evaluation, two spherical karst cavities were considered to represent the worst-case scenario for the stability of the BWRX-300 RB deeply embedded vertical shaft and surrounding structures: a 15-ft (diameter) cavity located 5 ft beneath the centerline of the RB mat foundation; and a 15-ft (diameter) cavity located 5 ft adjacent to the RB vertical shaft and 2-ft below the top of the unweathered rock below the Radwaste Building (heaviest surface structure). The details are provided in Subsection 2.5.4.13.

A mitigation plan to address possible cavities encountered at and below the foundation levels of SC1 RB during excavation is discussed in Subsection 2.5.1.2.6.10.

2.5.4.2 Properties of Subsurface Materials

2.5.4.2.1 Description of Subsurface Materials

Replace ESPA SSAR Section 2.5.4.2.1 with the following:

The following is a brief description of the subsurface materials and their range in thickness encountered at the site. The information is taken from the borings made for the 2013 CRN SMR subsurface investigation (outlined in Subsection 2.5.4.3.2) contained in Appendix B of Reference 2.5.4-1. Descriptions of the subsurface materials encountered in the 2023 BWRX-300 SMR subsurface investigation are discussed in Reference 2.5.4-66. The two investigations reveal similar lithologic descriptions for the subsurface materials at the CRN Site.

Table 2.5.1-2 (Sheets 1-3 of 4) and Table 2.5.1-2R (Sheet 4 of 4) summarizes the borings drilled and the subsurface material encountered for the 2013 CRN SMR subsurface investigation. For reference, the existing site elevations in the areas explored range from about 762 (MP-407) to 855 ft (MP-406) (both of these borings are located outside of the power block) with an average elevation of approximately 806 ft. Finished grade elevation at the Unit 1 location will be 814.5 ft.

2.5.4.2.1.1 Existing Fill and Residual Soils

Replace Figure 2.5.4-3 with Figure 2.5.4-3R in ESPA SSAR Section 2.5.4.2.1.1.

2.5.4.2.1.2 New Backfill

Replace ESPA SSAR Section 2.5.4.2.1.2 with the following:

New granular backfill will be used to raise grade to the final plant grade elevation (EL. 814.5 ft). This elevation increase near the vertical excavation of the RB is planned to be achieved with the conceptual design of compression rings. The details and assumptions on these elements (at the conceptual design stage) are provided in Subsection 2.5.4.13.

A description of the proposed granular backfill material is contained in Subsection 2.5.4.5.

2.5.4.2.1.3 Weathered Rock

Replace ESPA SSAR Section 2.5.4.2.1.3 with the following:

The depths and thicknesses of weathered rock encountered in the borings are summarized in Table 2.5.4-3R. The 2013 CRN SMR subsurface investigation shows that weathered rock is encountered in most of the borings drilled at the site (Appendix B.1, Reference 2.5.4-1). During the 2013 drilling operations, weathered rock was defined as a material having a SPT of 50 blow counts resulting in less than 6 inches of penetration (Reference 2.5.4-1). However, in the 2023 BWRX-300 SMR investigation, a more objective view was taken as to weathered rock classification from the boring logs, in which a portion of the top of cored rock could be classified in the weathered rock layer thickness, if it was described as highly weathered, exhibited high fracture density and had very poor to poor Rock Quality Designation (RQD).

Subsequent to the 2013 CRN SMR Site investigation, RQD values, ASTM D6032 (Reference 2.5.4-10), shear wave velocity (V_S) values, drill rates and rock core photographs of the 2013 investigation were reviewed to further define the thickness of weathered rock and the corresponding depth to the top of sound rock (hereinafter referred to as rock or bedrock). RQD values equal to or less than 25 percent, representative of very poor quality rock (Reference 2.5.4-10) along with significantly lower-than-average V_S values were also used to define the thickness of weathered rock. Large variations in the thickness of the weathered rock layer at the CRN Site have been observed; in fact, the thickness of the weathered rock from the MP-100 (from the 2013 investigation) set of borings varies from approximately no weathered rock to 19 ft. The maximum thickness of weathered rock at the CRN Site for Unit 1 location is approximately 19.2 ft (MP-106).

Rename and Replace ESPA SSAR Sections 2.5.4.2.1.4 through 2.5.4.2.1.12 with the following:

2.5.4.2.1.4 Witten Formation

The Witten Formation, which was not encountered during the 2013 CRN SMR subsurface investigation drilling program, was encountered in the 2023 investigation, but outside of the power block, along with the Bowen Formation. More details are provided in Reference 2.5.4-66.

2.5.4.2.1.5 Bowen Formation

The Bowen Formation is a maroon calcareous siltstone. Because of its limited thickness and its distinctive color, Hatcher et al. (Reference 2.5.4-6) describe it as a reliable marker for field and subsurface correlations. The 2013 CRN SMR subsurface investigation generally describes the bedrock as a reddish brown to olive brown, laminated to very thinly bedded calcareous siltstone. A detailed description of the bedrock is contained in Subsection 2.5.1.2.3.

The Bowen Formation was encountered in two borings in the 2013 CRN SMR investigation (MP-415 and MP-428) for a total linear footage of 41 ft (Table 2.5.4-1). The top of the formation was encountered in the two borings at elevations of 773 and 765 ft (Table 2.5.1-2 (Sheets 1-3 of 4) and Table 2.5.1-2R (Sheet 4 of 4)). The thickness of the Bowen Formation is illustrated on the subsurface profile on Figure 2.5.4-2R. The average apparent (vertical) thickness of the formation based on these two borings is estimated at 30 ft (Table 2.5.4-1). The Bowen Formation is outside of the footprint of the power block area (Locations A and B as sited in the CRN ESPA).

Two borings from the 2023 BWRX-300 SMR site investigation encountered the Bowen Formation, with one boring sampling its entire thickness (Reference 2.5.4-67). The true thickness of the Benbolt Formation in U1-B15 is approximately 33 ft (Table 2.5.1-3R). Descriptions of the Bowen Formation from the 2023 subsurface investigation are presented in Reference 2.5.4-66.

RQD values for the Bowen Formation range from 14 to 53 percent with average and median values of 26 and 20 percent, respectively, indicative of poor to very poor quality rock.

2.5.4.2.1.6 Benbolt Formation

The Benbolt Formation comprises interbedded fossiliferous nodular limestone, unfossiliferous amorphous micrite within a dark gray siltstone matric, and unfossiliferous calcarenite (Reference 2.5.4-6). The 2013 CRN SMR subsurface investigation describes the bedrock as a gray limestone (micrite/wackestone), strong, very thinly to thinly bedded, locally moderately bedded, and nodular limestone interbedded with little to some laminated to thinly bedded calcareous siltstone. A detailed description of the bedrock is contained in Subsection 2.5.1.2.3.

The Benbolt Formation is encountered in the southeastern portion of the power block area (Locations A and B), predominantly in the 100-series borings but also in a few of the 400-series borings. Twenty-five borings penetrated into the Benbolt Formation in the 2013 site investigation for a total linear footage of 3254 ft (Table 2.5.4-1). The top of the formation encountered in the borings ranges in elevation from about 794 to 739 ft (determined from Table 2.5.1-2

(Sheets 1-3 of 4) and Table 2.5.1-2R (Sheet 4 of 4)). The thickness of the Benbolt Formation is illustrated on the subsurface profile on Figure 2.5.4-2R. The average apparent (vertical) thickness of the formation based on the borings is estimated at 330 ft (Table 2.5.4-1).

RQD values for the Benbolt Formation range from 0 to 100 percent with average and median values of 88 and 98 percent, respectively, indicative of good to excellent quality rock. Rock core recovery and RQD values obtained from the borings are summarized in Table 2.5.4-4 and shown on Figure 2.5.4-3.

Descriptions of the Benbolt Formation from the 2023 BWRX-300 SMR subsurface investigation are presented in Reference 2.5.4-66. The Benbolt Formation is the dominant formation in the Unit 1 footprint, having been encountered in 15 of the 16 borings.

2.5.4.2.1.7 Rockdell Formation

The Rockdell Formation is a thick limestone with bedded and nodular chert that transitions with depth to a calcareous siltstone, a fossiliferous nodular limestone and a micritic limestone (Reference 2.5.4-6). The 2013 CRN SMR investigation generally describes the bedrock as two units, Unit C and Unit D, based on an 11 to 32 ft thick calcareous siltstone layer that forms the contact between the two units (Reference 2.5.4-1). Units C and D are generally described as gray and brownish-gray, strong, laminated to moderately bedded limestone (micrite/wackestone/grainstone), interbedded with few to little, laminated to very thinly bedded calcareous siltstone. A detailed description of the bedrock is contained in Subsection 2.5.1.2.3. With regard to assigning engineering properties, the Rockdell Formation is considered as a single unit.

The Rockdell Formation is encountered in the central and southeastern portions of the power block area (Locations A and B), predominantly in the 100-series borings but also in some of the 400-series borings. Sixteen borings penetrated into the Rockdell Formation for a total linear footage of 1794 ft (Table 2.5.4-1). The top of the formation encountered in the borings ranges in elevation from about 801 to 577 ft (determined from Table 2.5.1-2 (Sheets 1-3 of 4) and Table 2.5.1-2R (Sheet 4 of 4)). The thickness of the Rockdell Formation is illustrated on the subsurface profile on Figure 2.5.4-2R. The average apparent (vertical) thickness of the formation based on the borings is estimated at 287 ft (Table 2.5.4-1).

RQD values range from 0 to 100 percent with average and median values of 88 and 96 percent, respectively, indicative of good to excellent quality rock. Rock core recovery and RQD values obtained from the borings are summarized in Table 2.5.4-4 and shown on Figure 2.5.4-3.

Descriptions of the Rockdell Formation, (Unit C and D members), which underlines the Benbolt Formation within the Unit 1 footprint, are presented from the 2023 BWRX-300 SMR subsurface investigation in Reference 2.5.4-66.

2.5.4.2.1.8 Fleanor Member (Lincolnshire Formation)

The upper member of the Lincolnshire Formation, the Fleanor Member, comprises maroon, calcareous and shaly siltstone with numerous light gray limestone beds (Reference 2.5.4-6). The 2013 CRN SMR investigation generally describes the bedrock as a red, medium strong, laminated to medium bedded, calcareous siltstone with few to little gray micritic limestone layers. A detailed description of the bedrock is contained in Subsection 2.5.1.2.3.

The Fleanor Member is encountered in the central portion of the power block area (Locations A and B), predominantly in the 200-series borings but also in some of the 400-series borings. Twenty-nine borings penetrated into the Fleanor Member for a total linear footage of 2953 ft (Table 2.5.4-1). The top of the Fleanor Member encountered in the borings ranges in elevation from 809 to 381 ft (determined from Table 2.5.1-2 (Sheets 1-3 of 4) and Table 2.5.1-2R (Sheet 4 of 4)). The thickness of the Fleanor Member at the CRN Site is illustrated on the subsurface profile on Figure 2.5.4-2R. The average apparent (vertical) thickness of the member based on the 2013 CRN SMR investigation is estimated at 257 ft (Table 2.5.4-1).

RQD values range from 0 to 100 percent with average and median values of 89 and 98 percent, respectively, indicative of good to excellent quality rock. Rock core recovery and RQD values obtained from the borings are summarized in Table 2.5.4-4 and shown on Figure 2.5.4-3.

Descriptions of the Fleanor Member which underlies the Rockdell Formation in the Unit 1 footprint, are presented from the 2023 BWRX-300 SMR subsurface investigation in Reference 2.5.4-66.

2.5.4.2.1.9 Eidson Member (Lincolnshire Formation)

The lower member of the Lincolnshire Formation, the Eidson Member, comprises massive to nodular limestone with bedded and nodular chert (Reference 2.5.4-6). The 2013 CRN SMR investigation generally describes the bedrock as a gray, medium strong and strong, laminated to thinly bedded, fresh, argillaceous micritic limestone.

The Eidson Member is encountered in the central and northwestern portions of the power block area (Locations A and B), predominantly in the 200-series borings but also in some of the 400-series borings. Seventeen borings penetrated into the Eidson Member for a total linear footage of 1027 ft (Table 2.5.4-1). The top of the Eidson Member encountered in the borings ranges in elevation from about 809 to 604 ft (determined from Table 2.5.1-2 (Sheets 1-3 of 4) and Table 2.5.1-2R (Sheet 4 of 4)). The thickness of the Eidson Member at the CRN Site is illustrated on the subsurface profile on Figure 2.5.4-2R. The average apparent (vertical) thickness of the member based on the borings is estimated at 102 ft (Table 2.5.4-1).

RQD values range from 0 to 100 percent with average and median values of 79 and 88 percent, respectively, indicative of good quality rock. Rock core recovery and RQD values obtained from the borings are summarized in Table 2.5.4-4 and shown on Figure 2.5.4-3.

The Eidson Member was not encountered during the 2023 BWRX-300 SMR drilling investigation.

2.5.4.2.1.10 Blackford Formation

The Blackford Formation comprises purplish to dark maroon calcareous siltstone interbedded with light gray calcarenite (Reference 2.5.4-6). The 2013 CRN SMR investigation describes the bedrock as two units, the Lower and Upper Blackford. The contact between the Lower Blackford and the underlying Newala Formation is a disconformity. The Lower Blackford is generally described as a gray, locally mottled, strong, laminated to thickly bedded, micritic limestone. The Upper Blackford is generally described as a gray, calcareous siltstone, laminated to moderately bedded, interbedded with little to some limestone with few to little chert beds, lenses and nodules. A detailed description of the bedrock is contained in Subsection 2.5.1.2.3. With regard to assigning engineering properties, the Blackford Formation is considered as a single unit.

The Blackford Formation Member is encountered in the central and northwestern portions of the power block area (Locations A and B), in the 200-series and 400-series borings. Fifteen borings penetrated into the Blackford Formation for a total linear footage of 1554 ft (Table 2.5.4-1). The top of the Blackford Formation encountered in the borings ranges in elevation from 840 to 507 ft (determined from Table 2.5.1-2 (Sheets 1-3 of 4) and Table 2.5.1-2R (Sheet 4 of 4)). The thickness of the formation at the CRN Site is illustrated on the subsurface profile on Figure 2.5.4-2R. The average apparent (vertical) thickness of the formation based on the borings is estimated at 254 ft (Table 2.5.4-1).

RQD values range from 0 to 100 percent with average and median values of 81 and 92 percent, respectively, indicative of good to excellent quality rock. Rock core recovery and RQD values obtained from the borings are summarized in Table 2.5.4-4 and shown on Figure 2.5.4-3.

The Blackford Formation was not encountered during the 2023 BWRX-300 SMR drilling investigation.

2.5.4.2.1.11 Newala Formation

The Newala Formation is the uppermost unit of the Knox Group and comprises predominantly fine to medium grained dolomite with some chert nodules (Reference 2.5.4-6). Hatcher et al. (Reference 2.5.4-6) report that the thickness of the Newala Formation is variable due to the erosional unconformity (disconformity) and indicates the relief on this surface is a minimum of about 230 ft. The 2013 CRN SMR investigation generally describes the formation as a fresh, fine to medium grained, gray, locally mottled red, strong to very strong, moderately to thickly bedded crystalline dolomite, with few irregular chert nodules and chert beds. A detailed description of the bedrock is contained in Subsection 2.5.1.2.3.

The Newala Formation is encountered in the northwestern portion of the power block area (Locations A and B), predominantly in the 400-series borings and one 200-series boring (MP-201). Sixteen borings penetrated into the Newala Formation for a total linear footage of 2501 ft (Table 2.5.4-1). The top of the formation encountered in the borings ranges in elevation from approximately 815 to 481 ft (determined from Table 2.5.1-2 (Sheets 1-3 of 4) and Table 2.5.1-2R (Sheet 4 of 4)). Regionally, the thickness of the formation is estimated to range

between 900 and 1200 ft (Reference 2.5.4-11). The thickness of the formation at the CRN Site is unknown as none of the borings drilled at the site penetrated the full thickness of the unit (Figure 2.5.4-2R).

RQD values range from 0 to 100 percent with average and median values of 93 and 98 percent, respectively, indicative of excellent quality rock. Rock core recovery and RQD values obtained from the borings are summarized in Table 2.5.4-4 and shown on Figure 2.5.4-3.

The Newala Formation was not encountered during the 2023 BWRX-300 SMR drilling investigation.

2.5.4.2.1.12 Subsurface Profiles

Figure 2.5.4-2R illustrates a typical subsurface profile through the footprint of the proposed Unit 1 at the CRN Site. The profile is constructed perpendicular to the strike of the bedding planes (parallel to the dip direction). The final plant grade elevation of Unit 1 is expected at EL 814.5 ft and the shaft foundation embedment depth is approximately 115 ft below the plant grade elevation, with the planned excavation depth approximately 129.5 ft below grade elevation.

2.5.4.2.2 Field Investigations

Replace ESPA SSAR Section 2.5.4.2.2 with the following:

The field investigation programs performed previously for the CRBRP and the 2013 CRN SMR field investigation performed at the CRN Site are described in Subsection 2.5.4.3. A summary of the investigations performed for the CRBRP is in Table 2.5.4-5. A summary of the field work performed for the 2013 CRN SMR subsurface investigation is in Table 2.5.4-6. A summary of the soil and rock strata encountered in the borings is in Table 2.5.1-2 (Sheets 1-3 of 4) and Table 2.5.1-2R (Sheet 4 of 4). Geophysical surveys performed for the CRBRP and at the CRN Site are described in Subsection 2.5.4.4. The field investigation at the CRN Site was performed in accordance with guidance presented in RG 1.132, *Site Investigations for Foundations of Nuclear Power Plants*.

2.5.4.2.3 Laboratory Testing *(Addresses ESP COL Action Item 2.5-5)*

Replace the first paragraph of ESPA SSAR Section 2.5.4.2.3 with the following:

As with the field exploration, laboratory tests of soil and rock samples were performed previously for the CRBRP and performed as part of the 2013 CRN SMR subsurface investigation. Previous test results are summarized in the CRBRP PSAR (Reference 2.5.4-2). The types and numbers of tests completed during the 2013 CRN SMR subsurface investigation are shown in Table 2.5.4-7 and the test results are contained in Reference 2.5.4-1.

Insert the following after the fourth paragraph of ESPA SSAR Section 2.5.4.2.3:

The 2023 BWRX-300 SMR investigation was conducted that addresses L&C 8.1 of NEDO-33914-A (Reference 2.5.4-57), and geotechnical laboratory testing was performed at various laboratory locations. Laboratory testing was performed in accordance with ASTM standards or other standards, where applicable and as specified by RIZZO. Laboratory testing results are discussed in further detail in Reference 2.5.4-66. Key parameters, including boring logs, lab testing, soil properties, rock properties, downhole geophysical logging, and spectral analysis of surface waves, exhibited consistent results, confirming the reliability and validity of the data set presented in ESPA.

2.5.4.2.4 Engineering Properties

Replace the first paragraph of ESPA SSAR Section 2.5.4.2.4 with the following:

The engineering properties for the existing fill/residual soil, granular backfill, weathered rock and the bedrock within the footprint of the 2013 power block area (Locations A and B in the CRN ESPA) are derived from the 2013 CRN SMR subsurface investigation and laboratory testing programs and are provided in Table 2.5.4-21.

2.5.4.2.4.1 Soil Properties

Replace the second sentence of the first paragraph of ESPA SSAR Section 2.5.4.2.4.1 with the following:

The properties are developed based on the results of the laboratory testing program from the 2013 CRN SMR investigation program, unless stated otherwise.

Replace the first sentence of the seventh paragraph of ESPA SSAR Section 2.5.4.2.4.1 with the following:

Shear (V_s) and compression (V_p) wave velocity measurements were taken in three borings in the existing fill/residual soil as part of the 2013 CRN SMR field testing program.

Replace the first sentence of the last paragraph of ESPA SSAR Section 2.5.4.2.4.1 with the following:

Maximum dry density and optimum moisture content were determined as part of the 2013 CRN SMR laboratory testing program.

2.5.4.2.4.2 Weathered Rock Properties

Replace the first paragraph of ESPA SSAR Section 2.5.4.2.4.2 with the following:

Weathered rock is described in Subsection 2.5.4.2.1.3. The ESPA was prepared assuming the weathered rock would be excavated during construction and selected engineering properties of the weathered rock were developed from in situ testing (2013 CRN SMR investigation) and material correlations.

2.5.4.2.4.3 Intact Rock Properties

Replace the last sentence of the first paragraph of ESPA SSAR Section 2.5.4.2.4.3 with the following:

The properties are developed based on the results of the laboratory and field testing programs performed for the CRN Site in 2013.

Replace the third sentence of the eighth paragraph of ESPA SSAR Section 2.5.4.2.4.3 with the following:

Pressuremeter testing performed as part of the 2013 CRN SMR field investigation program provides shear moduli at various loadings and thus various levels of strain.

2.5.4.2.4.4 Rock Mass Properties *(Addresses ESP COL Action Item 2.5-5)*

Replace the second paragraph of ESPA SSAR Section 2.5.4.2.4.4 with the following:

The properties are developed using the GSI. The application of the GSI classification and Hoek-Brown failure criterion assumes that the rock mass contains several sets of discontinuities that are closely spaced relative to the proposed structure, such that it behaves as a homogeneous and isotropic mass and that a predetermined failure plane does not exist. In other words, while the behavior of the rock mass is controlled by the movement and rotation of the rock blocks separated by intersecting discontinuities, there are no preferred failure directions (Reference 2.5.4-18).

Replace the sixth paragraph of ESPA SSAR Section 2.5.4.2.4.4 with the following:

The rock mass strength and deformation properties are developed using the GSI classifications of the stratigraphic units that are based on a select number of 100-, 200- and 400-series borings drilled at the 2013 CRN SMR investigation.

The supplemental site-specific field exploration (2023 BWRX-300 SMR investigation) and testing have been conducted with considerations on the selected GVH BWRX-300 reactor technology. The data from this additional testing and exploration program allows for the further assessment of the rock mass properties and characterization of the subsurface conditions under the layout of Unit 1 at the CRN Site in more details. Parallel GSI classifications based on the fracture data of

the 2023 boreholes show that the GSI results for the Rockdell and Benbolt Formations are consistent with GSI classification performed with the 2013 data for the same two formations which addresses Limitation and Condition (L&C) 8.3 of NEDO-33914-A (Reference 2.5.4-57). The data from the 2023 BWRX-300 SMR investigation program, whose results supplement the existing information and datasets, are discussed and compared with the 2013 results in Reference 2.5.4-66.

Renumber the Rock Mass Strength subsection of ESPA SSAR Section 2.5.4.2.4.4 to Section 2.5.4.2.4.5 Rock Mass Strength.

Insert the following after the fifth paragraph of ESPA SSAR Section 2.5.4.2.4.5:

Generalized Hoek-Brown failure criterion, rock mass strength calculations were also performed with the 2023 site investigation data. The results of the rock mass strength calculations with the 2023 data are presented in Reference 2.5.4-66. The results indicate that the data are primarily consistent with the 2013 calculation results.

Renumber the Rock Mass Deformation Modulus subsection of ESPA SSAR Section 2.5.4.2.4.4 to Section 2.5.4.2.4.6 Rock Mass Deformation Modulus.

Insert the following after the first paragraph of ESPA SSAR Section 2.5.4.2.4.6:

The summary of strength properties is presented in Table 2.5.4-24.

Insert the following after the third paragraph of ESPA SSAR Section 2.5.4.2.4.6:

The deformation modulus of the rock mass calculations were also performed with the 2023 site investigation data using the same empirical equations as the 2013 data. The results of the deformation modulus calculations with the 2023 data are presented in Reference 2.5.4-66. The results indicate that the data are primarily consistent with the 2013 calculation results, with only minor deviations that can be explained by the depth of the log data and samples used for analysis.

2.5.4.3 Foundation Interfaces

Replace ESPA SSAR Section 2.5.4.3 with the following:

Subsection 2.5.4.3.1 summarizes subsurface investigations performed for the CRBRP, Subsection 2.5.4.3.2 summarizes the 2013 CRN SMR subsurface investigation program implemented for the CRN Site, while Subsection 2.5.4.3.3 summarizes the 2023 supplemental subsurface investigation program for the BWRX-300 SMR.

2.5.4.3.1 Subsurface Investigation Program for the CRBRP

Replace the second paragraph of ESPA SSAR Section 2.5.4.3.1 with the following:

Over 100 soil and rock core borings were completed for the CRBRP for siting the SC1 RB. A selection of these boring locations is shown on Figure 2.5.4-1R. These existing borings are designated as the B-series borings (Reference 2.5.4-3). The soil borings were advanced using rotary wash drilling techniques and SPTs were performed at 5-ft intervals in accordance with ASTM D1586 (Reference 2.5.4-3). The rock core borings were advanced in accordance with ASTM D2113-70 using NQ-, HQ-, and NX-size core barrels. Rock core recovery and RQD values were recorded.

Rename ESPA SSAR Section 2.5.4.3.2.

2.5.4.3.2 2013 CRN SMR Subsurface Investigation Program

Replace the first paragraph and the first sentence of the second paragraph of ESPA SSAR Section 2.5.4.3.2 with the following:

The 2013 CRN SMR subsurface investigation was performed between June 2013 and March 2014. The investigation was performed over a substantial portion of the CRN Site but predominantly within the footprint of the power block area (Locations A and B in the CRN ESPA). The investigation consisted of a substantial number of exploration points and was originally designed to support the CPA and the locations of the units being sited at that time (Locations A and B). These exploration point locations are identified as the MP-series and are shown on Figure 2.5.4-1R and Figure 2.5.4-2R. The layout for the selected GVH BWRX-300 unit (i.e., Unit 1) has been sited in the vicinity of Location A of power block from the ESPA, allowing to incorporate the existing subsurface investigation data within the footprint of the new site layout.

The scope of the work performed in the 2013 CRN SMR investigation program is listed below:

2.5.4.3.2.1 Borings and Soil Samples/Rock Cores

Replace the first and second paragraphs of ESPA SSAR Section 2.5.4.3.2.1 with the following:

The 82 borings drilled at the CRN Site ranged in depth from about 20 to 540 ft. The deep borings, drilled to depths greater than or equal to about 350 ft, are at least 200 ft deeper than the deepest foundation embedment depth targeted in the CRN ESPA, which was not expected to exceed 138 ft (El. 683 ft). These borings sample even a deeper profile under the anticipated elevation of the BWRX-300 RB foundation for the CRN Site, where the deepest foundation embedment depth is not expected to exceed 115 ft (i.e., approximately El. 699 ft) from the finished site grade El 814.5 ft. Borings include the MP-series borings (100-, 200-, and 400-series) and the CC-B series borings. With the exception of six of the borings (MP-111 PS; MP-111 UD; MP-122 PS-A; MP-122 PS-B; MP-122 UD-A, and MP-122 UD-B) that were drilled in soil only, all of the borings were drilled in soil and extended into the bedrock.

The 100- and 200-series borings drilled at the site are located within the footprint of the CRN Site power block area in Locations A and B, respectively. It is noted that Locations A and B in the ESPA were sited as a general placement without consideration for a specific reactor technology. A number of the 400-series borings are also within the footprint of the 2013 power block area (Locations A and B), while the majority is in the surrounding areas. Seven of the borings (MP-112, -113, -212, -213, -424, -425, and -425) were drilled at inclinations of between 25 and 29 degrees from the vertical. (Figure 2.5.4-1R).

Replace the second sentence of the last paragraph of ESPA SSAR Section 2.5.4.3.2.1 with the following:

Boring locations and depths etc. are summarized in Table 2.5.1-2 (Sheets 1-3 of 4) and Table 2.5.1-2R (Sheet 4 of 4)

2.5.4.3.2.2 Observation Wells/Well Tests

Replace the first sentence of the first paragraph of ESPA SSAR Section 2.5.4.3.2.2 with the following:

Forty-four groundwater wells were installed at the CRN Site in the 2013 investigation.

2.5.4.3.2.3 Groundwater Levels/Sampling

Delete "(Section 3.5 of Reference 2.5.4-1)" in the first paragraph of ESPA SSAR Section 2.5.4.3.2.3.

2.5.4.3.2.4 Test Pits

Replace Figure 2.5.4-1 with Figure 2.5.4-1R in ESPA SSAR Section 2.5.4.3.2.4.

Delete "(Section 2.9 of Reference 2.5.4-1)" in the first paragraph of ESPA SSAR Section 2.5.4.3.2.4.

2.5.4.3.2.6 Direct Shear Tests

Insert the following after the last sentence of ESPA SSAR Section 2.5.4.3.2.6:

Table 2.5.4-7 shows the number of samples and rock formations selected for direct shear tests.

Add new Section 2.5.4.3.3:

2.5.4.3.3 2023 Supplemental Field Investigation Program

The 2023 supplementary investigation program was performed during the period of June 2023 to September 2023 to acquire additional technology specific geotechnical information for a GVH SMR at the CRN Site. The 2023 CRN Site investigation was developed to supplement the existing geotechnical database for the BWRX-300 SMR from the two prior site investigations (References 2.5.4-1, 2.5.4-2 and 2.5.4-3) and to gain technology specific geotechnical information at specific locations where it was not already available. The borings are identified as the U1-B series and are shown on Figure 2.5.4-1R and Figure 2.5.4-2R.

The principal components and sequence of work related to the completion of the supplemental field investigation activities at the CRN Site were as follows:

- Continuous rock and sediment sampling in 16 geotechnical boreholes
- Undisturbed soil sampling in 11 boreholes
- Borehole geophysical surveys in 8 boreholes
- Borehole closure
- Surface geophysical surveys (Spectral Analysis of Surface Waves)
- Site Survey (initial and as-built)
- Laboratory testing of testing of boring soil and rock core samples.

The results of 2023 BWRX-300 SMR investigation were used to develop engineering properties for subsurface layers based on field observations and laboratory testing results. Analysis of the results of the 2023 site investigation are presented in Reference 2.5.4-66.

2.5.4.4 Geophysical Surveys

Replace ESPA SSAR Section 2.5.4.4 with the following:

Subsection 2.5.4.4.2 summarizes geophysical investigations performed for the CRBRP, Subsection 2.5.4.4.3 summarizes the 2013 CRN SMR geophysical program implemented at the CRN Site, while Subsection 2.5.4.4.3 summarizes the 2023 BWRX-300 SMR geophysical program implemented at the CRN Site.

2.5.4.4.1 Previous Geophysical Surveys for CRBRP

Replace the third sentence of ESPA SSAR Section 2.5.4.4.1 with the following:

The downhole V_s and V_p measurements from the CRBRP subsurface investigation program are compared to the 2013 CRN SMR measurements obtained for the CRN Site in Table 2.5.4-16 (Reference 2.5.4-3).

Rename and Replace ESPA SSAR Section 2.5.4.4.2 with the following:

2.5.4.4.2 Geophysical Surveys for the 2013 CRN SMR Investigation

Subsection 2.5.4.4.3.1 summarizes the methods used and results of surface geophysical testing that were performed in the 2013 CRN SMR investigation program. Subsection 2.5.4.4.3.2 summarizes the methods used and results of the downhole geophysical testing that were performed at the CRN Site in 2013.

2.5.4.4.2.1 Surface Geophysical Testing

Replace Figure 2.5.4-1 with Figure 2.5.4-1R in ESPA SSAR Section 2.5.4.4.2.1.

Replace the first sentence of the ESPA SSAR Section 2.5.4.4.2.1 with the following:

Surface geophysical testing at the CRN Site in 2013 consisted of performing a seismic refraction and reflection survey.

2.5.4.4.2.2 Downhole Geophysical Testing

Delete "(Appendix C of Reference 2.5.4-1)" in the Acoustic Televiewer (ATV) Logging subsection of ESPA SSAR Section 2.5.4.4.2.2.

Add new Section 2.5.4.4.3:

2.5.4.4.3 Geophysical Surveys for the 2023 Supplemental Site Investigation

Subsection 2.5.4.4.3.1 summarizes the methods used and results of surface geophysical testing that were performed in the 2023 Supplemental CRN Site investigation program. Subsection 2.5.4.4.3.2 summarizes the methods of the downhole geophysical testing that were performed at the CRN Site in 2023.

2.5.4.4.3.1 Surface Geophysical Testing

The 2023 surface geophysical program consisted of active-source spectral analysis of surface waves (SASW) testing performed at three (3) array locations on the site. The active and passive measurements were combined to develop shear wave velocity profiles. The surface geophysical program test results are provided in Reference 2.5.4-67. Further discussion and analysis of these tests are discussed in Reference 2.5.4-66.

2.5.4.4.3.2 Downhole Geophysical Testing

Downhole geophysical testing was performed during the 2023 Supplemental Site Investigation in 8 geotechnical boreholes involving the following testing:

- V_P and V_S (Suspension P-S velocity logging)
- Downhole Seismic Velocity Testing

- Acoustic televiewer (ATV) and deviation data
- Conductivity and natural-gamma data
- Caliper and natural-gamma data
- Fluid temperature, fluid conductivity and natural-gamma data

Descriptions of the testing methods and subcontractor reports are provided in Reference 2.5.4-67. Further discussion and analysis of these tests are discussed in Reference 2.5.4-66.

2.5.4.5 Excavation and Backfill *(Addresses ESP Permit Condition #4)*

Replace ESPA SSAR Section 2.5.4.5 with the following:

This section discusses the excavation and backfill for the SC1 RB at the CRN Site and includes the following topics:

- The extent (horizontally and vertically) of anticipated excavations, fills and slopes
- Excavation methods and stability
- Backfill sources
- Quality Control
- Construction dewatering impacts
- Retaining walls

2.5.4.5.1 Extent of Excavations, Fill and Slopes *(Addresses ESP COL Action Item 2.5-6)*

Replace ESPA SSAR Section 2.5.4.5.1 with the following:

Existing elevations within the investigated 2013 CRN SMR power block areas shown in Figure 2.5.4-1R, range from 855 (MP-406) to 780 ft (MP-207, MP-211). These relatively large variations in the existing surface elevations occur over a vast area at the CRN Site. Topography generally slopes gently downward from northwest to southeast as shown on Figure 2.5.4-1R. The approximate ground surface and site stratigraphy are shown on Figure 2.5.4-2R, a cross-section through the power block area orientated approximately perpendicular to strike. The proposed Unit 1 at the CRN Site is located in close proximity of Location A (as cited in the CRN ESPA); therefore, the data from the MP-100 group of borings provide a more representative value for range of existing surface elevations at the proposed Unit 1 location. Based on data listed in Table 2.5.4-3R, the existing surface elevation measured at the MP-100 boring series varies between approximately 797 to 802 ft, with an average of around 800 ft (computed average elevation of 799.2 ft). Finished plant grade elevation is expected at elevation 814.5 ft. The bottom of the basemat of the deeply embedded seismic category I structure, i.e., RB, is expected not to exceed a depth of 115 ft below finished grade, i.e., elevation 699.5 ft.

Existing fill/residual soil and weathered rock vary in thickness within the power block area and are generally thickest in the central and southern portions of the site as shown on Figure 2.5.4-2R. Bedrock is shallowest at the northern portion of the Unit 1 area. Table 2.5.4-26R

highlights the variability in the subsurface stratigraphy by summarizing the strata thickness and variability at Locations A and B. It is noted that the proposed location for Unit 1 of the selected technology has been sited in the vicinity of Location A. The construction of the vertical shaft for the RB requires a deep excavation in both soil and rock; however, the horizontal (lateral) extent of this excavation is limited as imposed by the diameter of the vertical shaft and in part to the method of construction (i.e., providing adequate space for the construction). Excavation sidewalls are expected to be vertical or near-vertical due in part to the depth of excavation, requiring the use of surface mounted cranes. The floor of the excavation at the RB location is expected to be irregular since hard unweathered rock will be encountered, requiring the use of dental concrete to establish a level grade.

The final plant grade elevation for the proposed Unit 1 is expected to be at elevation 814.5 ft, which is approximately 14.5 ft higher than the average existing site grade in the vicinity of Unit 1 location at CRN Site. Engineered fill will be used to raise the existing grade to the final plant grade elevation. Additionally, the lateral extent of the backfill will extend to accommodate the construction of the multiple surface (near-surface) built structures around the RB shaft, including the Radwaste, Control, Service, and Turbine Buildings.

The construction of the RB requires preparing a vertical deep shaft by excavating predominantly in an unweathered rock formation (i.e., Benbolt). However, in order to reach the unweathered rock, shallow soil and weathered rock layers must be excavated. As the excavation for the RB proceeds, in order to provide a stable excavation condition in the existing fill and residual soil, as well as retain the engineered fill in the vicinity of the RB shaft, an excavation and construction support system is required. At this preliminary stage, compression rings are considered in a conceptual layout as soil retaining systems. Concrete backfill may be used for dental concrete as well as backfilling around the RB structure.

Excavation of soil and rock at the CRN Site is discussed in the following paragraphs. Design of the excavation and backfill, and subsequently the appropriate construction method, is done during the detailed design stage of the project.

2.5.4.5.2 Excavation Methods and Stability *(Addresses ESP COL Action Item 2.5-7)*

2.5.4.5.2.1 Excavation in Soil

Replace ESPA SSAR Section 2.5.4.5.2.1 with the following:

Excavation in existing fill/residual soil can be made with conventional earthmoving equipment. Excavation must adhere to regulations from Occupational Safety and Health Administration (OSHA), 29 CFR 1926, Safety and Health Regulations for Construction. The excavations in soil for the deep vertical shaft of the RB may require sloped layback approach or vertical cuts supported with temporary systems (such as soldier piles with lagging) and subsequently retained with permanent solutions (such as compression rings). The final selection and design of the appropriate support system is done at the detailed design stage of the project. The side slopes of the ramp for construction access made in soil are checked against slope stability and recommended to be excavated at slope angles of 3 (horizontal) to 1 (vertical).

2.5.4.5.2.2 Excavation in Weathered Rock

Replace ESPA SSAR Section 2.5.4.5.2.2 with the following:

Excavation in the weathered rock can be achieved using conventional excavating equipment. Excavation support methods similar to those used for soil can be used for weathered rock. As described in Subsection 2.4.12, groundwater is generally encountered above the top of bedrock, within the weathered rock. Groundwater control as described in Subsection 2.5.4.6.2 is required during excavation and for excavation support.

2.5.4.5.2.3 Excavation in Rock

Replace ESPA SSAR Section 2.5.4.5.2.3 with the following:

Excavation in rock is likely to be made by using controlled blasting techniques. As noted by Reference 2.5.4-8 the majority of rock excavation (1.1 million cubic yards of primarily laminated siltstone and limestone) for the CRBRP was made by blasting. To minimize rock excavation and provide crane access to the bottom of the excavation, 75-ft high near-vertical rock slopes in the north, south and east portions of the excavation were required. Methods included production and perimeter blasting. Production blasting, to remove the bulk of the excavation, was conducted in 13-ft lifts with burden distances of 5 to 7 ft and spacing between holes of 6 to 8 ft. Perimeter blasting was conducted to form the finished excavation face and foundation grade. Perimeter blast holes were spaced at 2-ft intervals.

The near-vertical rock slope was stabilized with rock bolts. Reference 2.5.4-8 reports that the original rock bolt design for the foundation and excavation support system was based on subsurface data. As the excavation proceeded, geologic mapping of exposed rock surfaces enabled a redesign of the rock bolting program.

Slope movement and foundation performance was monitored with an extensive instrumentation program during and after the excavation. Instrumentation included 20 inclinometers to monitor slope movement around the perimeter of the excavation, 34 horizontal extensometers to monitor deep seated wedge movement of the rock slopes, two vertical extensometers to measure heave and settlement, and 25 piezometers to measure hydrostatic levels behind the excavation slopes and at foundation grade (Reference 2.5.4-8).

The foundation of the RB for the proposed Unit 1 is expected to be resting on rock in Benbolt formation. The construction of the vertical RB shaft will require deep excavations into this rock formation. Therefore, a blasting program specific to this rock type and its rock mass characteristics will be designed to facilitate the excavation similar to CRBRP. As the excavation in rock continues, the exposed faces will be mapped to determine the potential need for additional support and stabilization systems in rock.

The excavation in rock is considered as disturbed zone in the analysis. It is anticipated that the mat foundation overlays a zone of disturbed material with D=0.7, to account for disturbed rock mass from blasting and stress relief. D value mentioned above is a factor which depends on

severity of disturbance caused by blasting and stress relaxation during excavation. The value ranges between 0 and 1, where 0 is used for undisturbed in situ rock masses and 1 is used for very disturbed rock masses. The extent of the disturbed zone is unknown and dependent upon the blasting technique used. However, it is unlikely to extend to depths beyond 10 to 15 ft below the foundation level. Conservatively, this zone was assumed to be 25 ft thick below the foundation, and the D value used in the analysis is conservatively chosen to be 0.7.

It is noted that the supplemental site-specific field exploration (2023 BWRX-300 SMR investigation) and testing was planned and performed with considerations on the selected reactor technology. The objective of the additional data was to supplement the existing information and datasets in characterizing the subsurface conditions. Data from the 2013 CRN SMR investigation program are presented in Reference 2.5.4-66. Design of the excavation sequences and methods, excavation support (such as rock bolting, if necessary), and instrumentation and monitoring system will be developed during the detailed design stage. This design accounts for the specific rock units encountered in the excavation, including the dip angle of these units.

2.5.4.5.3 Backfill Sources *(Addresses ESP COL Action Item 2.5-8)*

Replace ESPA SSAR Section 2.5.4.5.3 with the following:

Granular backfill consists of a processed graded aggregate meeting the gradation requirements of Type A aggregate of the Tennessee Department of Transportation (TDOT) Standard Specifications for Road and Bridge Construction (Reference 2.5.4-37) Section 303 and Section 903. Given the large amount of rock that needs to be excavated across the site, it may be advantageous to set up a crushing and blending plant onsite to produce the crushed aggregate to the required gradation specification. Otherwise, the graded aggregate is imported from nearby quarries.

A detailed field and laboratory test program shall be conducted during the design stage to evaluate backfill sources and their engineering properties. This program will evaluate the use of onsite excavated rock as well as imported backfill. In fact, soundness and abrasion tests along with laboratory tests on characterization of some of the chemical properties of samples from the Benbolt formation have been performed as part of the 2023 BWRX-300 SMR investigation program. These data, presented in Reference 2.5.4-67, will assist in deciding on the suitability of the excavated rock as a source for backfill. The backfill evaluation program will consider gradation (grain size distribution), density, soundness, durability, strength, and the dynamic properties of the backfill. A test pad will be constructed to establish placement and compaction methods. It is expected that the granular backfill is compacted to at least 95 percent of the maximum dry density as determined by the modified Proctor test (ASTM D1557, Reference 2.5.4-15) and that the moisture content of the compacted fill is within 3 percent of its optimum moisture content.

Rename and Replace ESPA SSAR Section 2.5.4.5.4 with the following:

2.5.4.5.4 Quality Control *(Addresses ESP COL Action Item 2.5-8)*

Details, including identification of quality requirements and industry standards, regarding SC1 backfill material and placement specifications will be developed during the detailed design stage. Quality control measures related to backfill will also be developed during the detailed design stage. General requirements for backfill and subgrade quality control are provided in the following paragraphs.

2.5.4.5.4.2 Concrete Backfill

Replace the second sentence of the first paragraph of ESPA SSAR Section 2.5.4.5.4.2 with the following:

The concrete backfill is also used for side fill surrounding the SC1 RB from the basemat elevation to the top of rock.

2.5.4.5.4.3 Foundation Bedrock *(Addresses ESP COL Action Item 2.5-3)*

Add the following to end of the second paragraph of ESPA SSAR Section 2.5.4.5.4.3:

The grouting program is discussed in Section 2.5.1.2.6.10.

2.5.4.6 Groundwater Conditions

2.5.4.6.1 Groundwater Measurements and Elevations

Replace the first sentence of the second paragraph of ESPA SSAR Section 2.5.4.6.1 with the following:

Groundwater observation wells were installed at the CRN Site as part of the 2013 CRN SMR subsurface investigation program.

2.5.4.6.2 Construction Dewatering *(Addresses ESP COL Action Item 2.5-9)*

Add the following to end of the second paragraph of ESPA SSAR Section 2.5.4.6.2:

An dewatering and groundwater control plan, including a monitoring plan, and provide an evaluation of the impact of dewatering on the stability of foundations, will be developed during excavation and construction.

2.5.4.6.3 Groundwater Chemical Properties *(Addresses ESP COL Action Item 2.5-10)*

Insert the following after the last paragraph of ESPA SSAR Section 2.5.4.6.3:

The detailed design for foundation protection based on the chemical characteristics of the groundwater and foundation and fills materials at the site consistent with the applicable industrial standards will be described in the FSAR.

2.5.4.7 Response of Soil and Rock to Dynamic Loading *(Addresses ESP COL Action Item 2.5-11)*

2.5.4.7.2 Velocity Profiles

Replace ESPA SSAR Section 2.5.4.7.2 with the following:

Various geophysical surveys were conducted in 2013 at the CRN Site to characterize in situ dynamic properties of the soil and rock, including seismic refraction and reflection, and P-S Suspension logging as described in Subsection 2.5.4.4.3. The P-S Suspension logging method was used to collect V_S and V_P measurements in the soil and rock. Figures 2.5.4-5 and 2.5.4-6 present V_S and V_P profiles for each stratigraphic unit. These V_S and V_P profiles in unweathered rock have been developed and subsequently used to form the velocity profiles at Locations A and B of the 2013 layout, as sited in the CRN Site ESPA. The new proposed layout of Unit 1 with the selected technology at the CRN Site is located in the vicinity of Location A. In fact, the proposed location of the RB vertical shaft is surrounded by borings MP-110, MP-112, and MP-113 which lie within the extent of Location A. The velocity profiles for Location B are also presented as they provide insight into the subsurface conditions.

Supplemental site-specific field exploration (2023 BWRX-300 SMR investigation) and testing have been conducted to supplement the existing data set. This investigation has been designed and implemented based on the proposed Unit 1 layout with the GVH BWRX-300 reactor technology. The additional testing and exploration program included additional downhole suspension P-S velocity logging and conventional downhole surveys performed in eight borings. This supplemental geophysical investigation provided additional data that confirms the previously acquired information and serves to verify the wave velocity profiles and other dynamic properties of the Benbolt and Rockdell formations, at the CRN Site.

Downhole seismic testing was performed in six borings during the 2023 BWRX-300 SMR investigation to determine P-wave and S-wave velocity profiles. Seismic waves generated on the surface were monitored during downhole testing at depth intervals of 10 ft using a single borehole geophone lowered and raised in the borehole.

Discussion of these geophysical investigations and the associated velocity profiles can be found in Reference 2.5.4-66 and Reference 2.5.4-67.

2.5.4.7.2.1 Shear Wave Velocity Profiles

Replace ESPA SSAR Section 2.5.4.7.2.1 with the following:

The V_S profiles used to develop the performance-based ground motions for the BWRX-300 SMR site are described in Subsection 2.5.2.5.1. Note they differ in the shallow section from the V_S profile developed for Location A.

Details on developing the stratigraphy and V_S profiles for Locations A and B of the 2013 CRN SMR layout within the rock units are provided in this section for context and because the deeper portion of the V_S profile of Location A was adopted for the base-case V_S profile for the BWRX-300 site (Subsection 2.5.2.5.1). These profiles were developed by considering the top of the profile (depth = 0) at the top of unweathered rock at each location and using site-measured V_S data, V_S data measured in similar nearby geologic units, and estimated V_S values from literature from the top of unweathered rock to Precambrian basement rock.

The geologic cross-section illustrating the shallow portion of these profiles is shown on Figure 2.5.4-12 with the full depth shown on Figure 2.5.4-13. The best estimate (mean) base-case V_S profile for the shallow geologic units (rock) was developed for each area by computing the lognormal mean profile for the measured V_S data from the boreholes taken in the area, or just outside the area (within 100 ft), as shown in Figure 2.5.4-11 where the V_S data for each borehole was normalized to the top of unweathered rock, zero depth. The associated borehole data and lognormal average V_S profile for Locations A and B (of the 2013 CRN SMR layout) are shown on Figures 2.5.4-14 and 2.5.4-15, respectively with the top of the profile (depth = 0) at the top of unweathered rock.

Boreholes whose data were used in developing the V_S profile in Figure 2.5.4-14 (Location A) fall within the footprint of the layout of Unit 1 or in its very close proximity and approximately envelop the RB. In fact, boring MP-101 borders the Control buildings, MP-102 is to the west of the RB and Turbine building, MP-111 is located immediately to the North of the Control building. Borehole MP-120 is slightly to the East (considering the plant North) of the Radwaste building. Below the measured data and extending into the Newala, average V_S values provided in Table 2.5.4-21 were assigned to the underlying geologic units. Unless supported by measured data, the Newala Formation and the remainder of the Knox Group were assigned a V_S of 11,000 fps. These profiles are shown on Figures 2.5.4-16 and 2.5.4-17 for Locations A and B, respectively.

Interpretation of the 2023 downhole velocity data reveals that in general, the V_S values range from 1,125 to 9,800 ft/sec, with an average velocity of 5578 ft/sec. No significant velocity reversals with depth were observed. A table of the identified depth intervals (layers), the subsequent seismic velocities over the defined layers, the calculated Poisson's ratio as well as a plot of the V_S profiles at all seven boreholes is presented in Reference 2.5.4-66. The raw data is presented in Reference 2.5.4-67.

Prior to the 2023 BWRX-300 SMR site investigation, limited V_S data was available for the deep geologic units as site specific SASW surveys were not completed during the 2013 CRN SMR investigation. Consequently, measured data from spectral analysis of surface waves (SASW) surveys in the Conasauga shale, Pumpkin Valley shale, and Rome Formation at the nearby Watts Bar facility were used. These data were taken from depths of 500 and 1500 ft and adjusted to the CRN Site using generic V_S profiles for central and eastern U.S. hard rock (Reference 2.5.4-58) and applied to the deeper geologic units of the profiles. These profiles are shown on Figures 2.5.4-18 and 2.5.4-19 for Locations A and B, as sited the ESPA, respectively.

Epistemic uncertainty in the mean base-case (best estimate) profiles is accounted for with upper-and lower-range base-case profiles as described in Subsection 2.5.2.5.1.1.

SASW testing was performed along three (3) array locations on the site during the 2023 Site investigation using a Common-Middle-Receiver Geometry where the second receiver in the array is fixed while the location of the other receivers and source are varied to yield different spacing configurations. Forward and reverse source shots were used for several of the spacing configurations to minimize internal phase shifts between receivers. A detailed description of the surveys is presented in Reference 2.5.4-67.

Interpretation of the SASW velocity data reveal that the three locations can generally be described as a three-layer system with a 25-39 ft thick low velocity layer where V_S is less than 1300 ft/sec, overlying a 67-99 ft thick layer with V_S ranging from 2,230 to 2,840 ft/sec, then V_S increases to between 7,675-8,060 ft/sec down to a modeled depth of 960 ft. The V_S data and related profiles from the SASW survey, along with greater discussion is presented in Reference 2.5.4-66.

Engineered fill will be used to raise the current site grade elevation to the final site grade. The mechanical properties of the backfill will be determined at the detailed stage of the design. As described in Subsection 2.5.2.5.1, at this stage the V_S model for the backfill has been estimated empirically from literature or other site response analysis studies.

2.5.4.7.2.2 Compression Wave Velocity Profiles

Replace ESPA SSAR Section 2.5.4.7.2.2 with the following:

The V_P profiles in rock at the CRN Site are developed in a similar manner to the V_S profiles. The unit V_P profiles as presented on Figure 2.5.4-6 are developed based on the site stratigraphy for the corresponding stratigraphic units as characterized in the 2013 CRN SMR investigation program. Table 2.5.4-11 also presents the P-wave velocity of the weathered rock and existing fill/residual soil at the CRN Site.

Interpretation of the downhole velocity data reveal that in general, the V_P values from 2,040 to 21,270 ft/sec with an average of 11,597 ft/sec. No significant velocity reversals with depth were observed. Discussion of the V_P results as well as a plot of the V_S profiles at all seven boreholes is presented in Reference 2.5.4-66 while the raw data is presented in Reference 2.5.4-67.

V_P results were not directly measured with the SASW method, so the V_P values in the table were computed using an assumed Poisson's ratio for each layer. The V_P data from the SASW survey, along with greater discussion is presented in Reference 2.5.4-66.

2.5.4.7.3 Dynamic Laboratory Tests

Insert the following after the first paragraph of ESPA SSAR Section 2.5.4.7.3:

Resonant column torsional shear (RCTS) tests conducted as part of the 2023 BWRX-300 site investigation have been completed by the University of Texas at Austin (Reference 2.5.2-254 and Reference 2.5.2-255). Ten rock samples collected were tested. The samples were dynamically tested to evaluate the linear and nonlinear dynamic properties in shear. The effects of 1) isotropic confining pressure, 2) time of confinement, and 3) shearing strain amplitude on the shear modulus (G) and material damping ratio (D) were evaluated. The variations of shear-wave velocity, shear modulus (Gmax) and material damping ratio at small strains were analyzed for all ten samples. MRD curves were estimated based on the RCTS results.

2.5.4.7.4.1 Material Damping and Shear Modulus

Replace ESPA SSAR Section 2.5.4.7.4.1 with the following:

A detailed description of the shear modulus reduction and damping (MRD) curves at the CRN Site for the engineered fill, existing fill and residual soil, weathered and unweathered rock is provided in Subsection 2.5.2.5.3.

2.5.4.7.4.2 Kappa

Replace the second through fifth paragraphs of ESPA SSAR Section 2.5.4.7.4.2 with the following:

Subsection 2.5.2.5.2 describes the estimation of kappa.

Subsection 2.5.2.5.3.1 provides a discussion of how kappa is assigned to each shear wave velocity profile.

2.5.4.7.5 Rock Column Amplification/Attenuation Analysis

Replace ESPA SSAR Section 2.5.4.7.5 with the following:

The soil/rock column amplification/attenuation analysis is described in Subsection 2.5.2.5. The generalized soil/rock column for the site response analysis at the CRN Site consists of engineered fill, existing fill/residual soil, in-situ weathered rock, and in-situ unweather rock. This analysis considers a deep rock profile, from the top of unweathered rock to the Precambrian basement rock, with the stratigraphy shown on Figures 2.5.4-12 and 2.5.4-13. Moreover, the amplification factors at the RB foundation level, top of the unweathered rock level, and the final plant grade level (Elevation 814.5 ft) are computed. The V_S profiles, material damping, shear modulus, and kappa values used in this analysis are described in Subsections 2.5.4.7.4 or 2.5.2.5.1.

Unit weights are described in Subsection 2.5.2.5.1.2.

2.5.4.8 Liquefaction Potential

Replace the first sentence of the first paragraph of ESPA SSAR Section 2.5.4.8 with the following:

This section presents the evaluation of the liquefaction potential of the materials adjacent to and under the SC1 RB at the CRN Site.

Replace the fourth paragraph of ESPA SSAR Section 2.5.4.8 with the following:

The SC1 structure (RB) at the CRN Site is embedded at a depth expected not to exceed 115 ft below final grade (El. 699.5 ft). Sound rock (bedrock/rock below the weathered rock) is located at an average elevation of 20 to 30 ft below the existing ground surface around the RB location, more than 70 ft above the foundation level. Furthermore, if any dental work is required to prepare the foundation surface, dental concrete is used. Liquefaction cannot occur in sound rock or concrete; therefore, there is no potential for liquefaction in the foundation materials in unweathered rock.

Replace the fifth paragraph of ESPA SSAR Section 2.5.4.8 with the following:

Subsection 2.5.4.5 describes the plan for backfilling at the CRN Site. Concrete, if used for backfilling, does not pose a potential liquefaction issue. Granular backfill is expected to be used to raise the existing ground surface to the final plant grade elevation. The granular backfill shall be compacted to at least 95 percent of the modified Proctor value. Based on the gradation and degree of compaction, this backfill is not considered to be susceptible to liquefaction.

Replace the seventh paragraph of ESPA SSAR Section 2.5.4.8 with the following:

Test results from Atterberg limits conducted on representative soils from the 2013 CRN SMR investigation are plotted on Figure 2.5.4-10 along with the Polito criteria for liquefiable soils (Reference 2.5.4-41), whereby plasticity is the key factor. If this requirement, whether measured in terms of liquid limit (LL) or plastic limit (PL), is met, the soil does not appear to be susceptible to flow liquefaction. A PL of 10 and a LL of 30 are used as threshold values. All of the CRN Site soils from the 2013 subsurface investigation plotted on Figure 2.5.4-10 fall outside the proposed zone of liquefiable soils and therefore are not susceptible to liquefaction by the Polito criteria.

Replace the eighth paragraph of ESPA SSAR Section 2.5.4.8 with the following:

In summary, rock, backfilled with concrete and compacted (engineering) granular backfill are not susceptible to liquefaction. The CRN Site soils, which include the existing fill and residual soil that consist predominately of cohesive, fine-grained material, are not considered susceptible to liquefaction.

2.5.4.9 Earthquake Design Basis

Replace ESPA SSAR Section 2.5.4.9 with the following:

The development of the site-specific response spectra for Unit 1 at the CRN Site is described in Subsection 2.5.2.5.8. The associated vertical spectra are computed with the V/H ratios as described in Subsection 2.5.2.5.6.

2.5.4.10 Static and Dynamic Stability

Replace the first sentence of the first paragraph of ESPA SSAR Section 2.5.4.10 with the following:

This section presents the evaluation of static and dynamic stability of the SC1 RB including bearing capacity, heave, settlement, and lateral earth pressures in the power block area at the CRN Site.

Replace the second paragraph of ESPA SSAR Section 2.5.4.10 with the following:

Subsection 2.5.4.1.1 describes the stratigraphy at the site. The site is underlain with a succession of stratigraphic units that generally strike N63°E with a dip angle of 33 degrees. Rocks belonging to the Knox Group outcrop to the northwest and progressively younger rocks belonging to the Chickamauga Group outcrop to the southeast. The stratigraphic units at the CRN Site, from northwest to southeast, include the Newala Formation, Blackford Formation, Eidson and Fleanor Members, Rockdell Formation, Benbolt Formation, Bowen Formation and Witten Formation. Benbolt and Rockdell formations are shallowest rock layers under the footprint of the proposed Unit 1 layout. The average thickness of these units, based on the 2013 CRN SMR investigation data, is presented in Table 2.5.4-1.

Replace the fourth and fifth paragraphs of ESPA SSAR Section 2.5.4.10 with the following:

Overlying the dipping bedrock units, a layer of weathered rock and existing fill/residual soil is encountered at the CRN Site. These materials are described in Subsection 2.5.4.2.4. Existing site grades vary within and around the footprint area of Unit 1 with an average elevation of approximately 800 ft. Compacted granular fill is used to establish the expected finished grade and raise the existing surface elevation to the target elevation of 814.5 ft.

Due to the dipping strata at the CRN Site, the stratigraphic units underlying the Unit 1 footprint vary depending on location. For this reason, the location of Unit 1 in the power block area, in the vicinity of Location A of ESPA, is evaluated for static and dynamic stability of the SC1 RB with a foundation embedment of 115 ft below finished grade (El. 699.5 ft) and excavation depth is estimated to be at El. 685 ft.

Replace the seventh paragraph of ESPA SSAR Section 2.5.4.10 with the following:

A Three-dimensional (3D) Finite Element (FE) Foundation Interface Analysis (FIA) was developed to simulate the actual subsurface conditions and construction sequences to provide insights into different aspects including the settlement and heave, bearing capacity, lateral earth pressure, and hydro-static pressure under the footprint of Unit 1 at the CRN Site in Subsection 2.5.4.12. The FIA is used for the evaluation of settlement and heave in Subsection 2.5.4.10.2 and to check the response to amplified service loads in Subsection 2.5.4.10.1.1 and Subsection 2.5.4.10.1.2. The foundation response to the incremental loads provides valuable insights into the high capability of rock in resisting loads. The layout of the selected technology for Unit 1 at the CRN Site (i.e., vertical deep shaft surrounded by surface-founded structures), and site stratigraphy (combination of inclined layering of rock with surficial soil) might render some of the conventional methods for estimating bearing capacity and settlement inapplicable. The FE model attempts to simulate the actual CRN Site subsurface geologic conditions with inclined rock units beneath foundations, impact of excavation on rock disturbance, including a weakened interface between rock units. Rock mass properties based on GSI are considered in evaluating bearing capacity and settlement and heave.

2.5.4.10.1 Bearing Capacity *(Addresses ESP COL Action Item 2.5-12)*

Replace ESPA SSAR Section 2.5.4.10.1 with the following:

The evaluation of the bearing capacity at the CRN Site was presented in Subsection 2.5.4.10.1 of the CRN Site ESPA. In fact, bearing capacity at Locations A and B were computed by considering a plant parameter envelope (PPE). With the BWRX-300 reactor technology, the proposed layout of Unit 1 at the CRN Site consists of a deeply embedded RB vertical shaft with a foundation constructed in the Benbolt formation and several surface-founded structures surrounding the shaft.

The evaluation of bearing capacity of rock is well documented in the literature (References 2.5.4-43, 2.5.4-44, 2.5.4-45, 2.5.4-47, 2.5.4-48, and 2.5.4-49). While authors present different methods, each method generally considers intact rock properties and rock mass

properties. As mentioned in the previous section, the proposed layout for Unit 1 at the CRN Site (i.e., vertical deep shaft surrounded by surface-founded structures) as well as the site stratigraphy (combination of inclined layering of rock with surficial backfill), create a condition that deviates from the assumptions in developing the empirical methods for estimating bearing capacity. Therefore, three-dimensional finite element (FE) modeling is used to complement results from traditional approaches in an attempt to capture conditions such as inclined rock units beneath foundations, impact of excavation on rock disturbance, and effects of interfaces at the CRN Site.

The intact rock properties, obtained from laboratory testing, are described in Subsection 2.5.4.2.4.3 and include unit weight, specific gravity, Poisson's ratio, unconfined compressive strength, elastic modulus, and shear modulus. The rock mass properties which account for the in-situ condition of the rock mass are described in Subsection 2.5.4.2.4.4. These properties include rock mass strength and rock mass deformation modulus. The rock mass strength is developed using the Hoek-Brown failure criteria (Reference 2.5.4-19) and the GSI classification system. The rock mass deformation modulus is developed using empirical equations and a combination of the intact elastic modulus, GSI and a disturbance factor (D). Both of these approaches are described in detail in Subsection 2.5.1.2.6.2.

2.5.4.10.1.1 Ultimate Bearing Capacity

Replace ESPA SSAR Section 2.5.4.10.1.1 with the following:

Ultimate bearing capacity (qu) of the RB foundation is estimated using the bearing capacity equations presented by the U.S. Army Corps of Engineers (USACE), Reference 2.5.4-43. Equation 2.5.4-6 is applicable to a foundation bearing on a moderately dipping (20 to 70 degrees from foundation plane) jointed rock mass with general shear failure along joints or a highly jointed rock mass with general shear failure on irregular failure surface through the rock mass.

$$q_u = 0.5\,\gamma\,\beta\,N_\gamma F_{\gamma d} + \gamma\,D\,N_q F_{qd} \qquad \text{Equation 2.5.4-6}$$

where:

N_γ and N_q are rock capacity factors as function of the friction angle ϕ,

$F_{\gamma d}$ and F_{qd} are the embedment factors,

B and *D* denote the depth of foundation below ground level and foundation width, and

γ represents the total unit weight.

With the dipping bedrock and identifiable joint sets, Equation 2.5.4-6 is adequately suited to calculate bearing capacity. It is noted that considering the depth of two times the width of the RB foundation (2B) as the estimate of the influence depth, the influence depth will be sampling both

Benbolt and Rockdell rock formations below the RB foundation. The general non-uniformity of the rock formations (with the presence of the bedding plane) below the foundation level is recognized as a limitation of this approach.

To mitigate the limitations of the conventional methods, a three-dimensional finite element model was generated to investigate the bearing capacity of the RB foundation. The finite element model developed for Unit 1 incorporates all the surface-founded structures, the RB, and the geometry of the most significant geological and geotechnical units anticipated within the Site. Figure 2.5.4-29R presents the Finite Element 3D model mesh for Unit 1. The model comprises several construction stages that together track the changes in the effective stress, total strain, and displacement fields generated by the excavation of the vertical RB shaft, lowering the groundwater level, material disturbances in the vicinity of the RB's shaft due to blasting, construction of the RB walls and backfilling, restoring of the groundwater table, and anticipated building loads.

The highly sophisticated three-dimensional model demands significant computational resources and substantial computational times. As a result, its usage is mainly reserved to examine the response of the RB and the bearing capacity modeling of the surface founded structures were performed using separate 2D simulations.

When evaluating the bearing capacity of the RB foundation in the 3D FE model, incremental loads up to 11 times the expected service load were applied to the foundation (the service load is estimated at 8.4 ksf). Figure 2.5.4-41 and Figure 2.5.4-42 illustrate the FE mesh and load conditions of the RB mat at the last stage of the simulation. The results were further examined to evaluate the development of plastic zones below the RB's foundation. It was observed that the distribution of the plastic points and the effective plastic strain field within the rock units did not indicate a failure status; therefore, the augmented load is still below the limit capacity.

Two-dimensional (2D) numerical models were developed to evaluate the bearing capacity of surface-founded structures surrounding the RB. These 2D models adequately capture the evolution of the failure surfaces within the uppermost shallow geotechnical layer and the kinematic constraint imposed by the rigid base, which would be otherwise inaccurate using traditional bearing capacity equations. For these structures, the bearing capacity calculation from 2D models follows a load control approach similar to the one presented for the RB. However, since these models are inherently less computationally demanding, pushing the foundations to reach or to get close to the limit state is possible (i.e., numerical instability matches material collapse). A second alternative incorporates the Strength Reduction Method (SRM) and the Stress Analysis Method (SAM) techniques at each load increment. The 2D and 3D numerical analyses were conducted using Midas GTS-NX. Additionally, in some cases for the evaluation of the static bearing capacity of the surface founded structures, PLAXIS 2D was also used.

2.5.4.10.1.2 Allowable Bearing Capacity

Replace ESPA SSAR Section 2.5.4.10.1.2 with the following:

Allowable bearing capacity (q_a) is defined as the ultimate bearing capacity divided by a factor of safety (FS). A FS of 3 is applied to each of the methods described above.

A general bearing failure of the RB foundation is ruled-out due to a net decrease in the bearing pressure at the foundation level. The amount of unloading by removing rock and soil to construct the RB foundation (foundation approximately 115 ft below final plant grade) is comparable to the pressure from the RB structure (a value of 8.4 ksf is anticipated as the load from the structure). Thus, the net change in pressure at the foundation level is expected to be smaller than the anticipated applied pressure. Therefore, general shear failure, including sliding along a predetermined failure plane, such as a bedding plane, is not likely to occur.

The results of the bearing capacity analyses are summarized in Table 2.5.4-27R. Note that lower-bound material properties were used in Equation 2.5.4-6 and the numerical models explained above. Additionally, the friction angle and unit weight in Equation 2.5.4-6 are the weight-averaged values within the depth of influence below the RB mat. It is anticipated that the mat foundation overlays a zone of disturbed material with D=0.7, to account for disturbed rock mass from blasting and stress relief. The extent of this disturbed zone is unknown and dependent upon the blasting technique used. However, it is unlikely to extend to depths beyond 10 to 15 ft below the foundation level. Conservatively, this zone was assumed to be 25 ft thick below the foundation. At the center of the foundation the remaining layers within the depth of influence consists of an 89 ft layer of undisturbed Benbolt that overlays 124 ft of Rockdell formation.

The minimum allowable vertical bearing capacity predicted by the U.S. Army Corps of Engineers (USACE), Reference 2.5.4-43, reaches q_a = 76 ksf, which provides a margin of at least nine times the anticipated service load of the reactor building.

Results from the 3D FEM model align with the positive bearing capacity margins computed from Equation 2.5.4-6. In fact, the 3D findings show that by loading the RB mat with three times its anticipated vertical load (FS=3), the materials below the RB's foundation are absent of any form of nonlinear behavior and consequently remain elastic. Figure 2.5.4-43 (left) illustrates the nonlinear state below the RB mat at three times its vertical load. The 3D model reported failure to reach a converged state at six times the anticipated RB's vertical load. As Figure 2.5.4-44 shows, the lack of plastic points below the RB foundation proves that the material remains within the elastic regime; and therefore, significantly high allowable bearing capacity is expected.

For the RB, the results of the stress-strain analyses from the 3D FE model confirmed that the rock under the RB is capable of withstanding significantly high loads and provided reasonable confidence in the values computed from the conventional methods. Therefore, the allowable bearing capacities estimated using Reference 2.5.4-43 (i.e., 76 ksf) is the recommended q_a at this stage (Table 2.5.4-27R).

A parametric approach was used in evaluating the bearing capacity of the structures surrounding the RB, where the bearing capacity for various foundation widths are evaluated. It is important to note that the existing fill and residual soil layer lies between the shallow engineered backfill and rock in the subsurface profile under these surface-founded structures. A shallow foundation system is also considered for these structures (without deep foundations such drilled shafts). The analysis focused solely on the inherent bearing capacity of the shallow foundations resting on a subsurface profile comprised of backfill, existing fill and residual soil, and rock with lower range estimates of mechanical properties. The results show that the bearing capacity increases with the foundation size and slowly reaches a plateau at larger foundation widths (>60 feet). The average allowable net capacity (by considering an FS = 3) obtained assuming lower bound material properties is approximately 2.0 ksf. Maximum and minimum allowable capacities reached 2.3 ksf and 1.6 ksf, respectively. These correspond to foundations of 100 and 10 ft wide.

SAM and SRM results show similar trends to those of the load control counterpart, although with slightly higher values. Under SAM and SRM methodologies, the average allowable capacity reaches 2.4 ksf. Figure 2.5.4-45 illustrates the results from both methodologies.

Figure 2.5.4-46 consolidates the results for the structures. Since these findings correspond to results under plane strain conditions, these need to be affected by the shape factor to incorporate dimensional effects. After this modification, the admissible capacity for the Power Block structures is $q_a[ksf] = 1.7 \times \left(1 - 0.4\frac{B}{L}\right)$ for B < 40 ft, and $q_a[ksf] = 2.5 \times \left(1 - 0.4\frac{B}{L}\right)$ for B > 60 ft. Capacities vary linearly for foundation widths between 40 and 60 ft.

A summary of q_a values computed for the various structure foundations at Unit 1 location is provided in Table 2.5.4-27R.

2.5.4.10.1.3 Allowable Bearing Capacity of Concrete

Renumber Equation 2.5.4-9 to Equation 2.5.4-7 in ESPA SSAR Section 2.5.4.10.1.3.

Rename and Replace ESPA SSAR Section 2.5.4.10.1.4 with the following:

2.5.4.10.1.4 Ultimate and Allowable Pseudo Static Bearing Capacity

A pseudo-static approach is followed to evaluate the ultimate seismic capacity q_{uE} at the CRN Site. For the reactor building, q_{uE} is obtained using Soubra's method (Reference 2.5.4-62). The values of pseudo-static bearing capacity factors, $N_{c,E}$,$N_{q,E}$, $N_{\gamma,E}$ are obtained as functions of the horizontal seismic coefficient (K_h) and friction angle. Pseudo-static ultimate capacity is obtained replacing the pseudo-static bearing capacity factors in the bearing capacity equation (i.e., $q_{uE} = 0.5\ \gamma \beta\ N_{\gamma.E} F_{yd} + \gamma D\ N_{q.E} F_{qd} + c\ N_{c.E} F_{cd}$).

Notice that pseudo-static capacity takes credit from cohesion contribution; however, as a conservative measure, a 1/3 reduction factor is applied to the cohesion term in the pseudo-static bearing capacity equation.

For surface-founded structures around the RB, pseudo-static capacities are computed following a load control approach similar to the one explained in Subsection 2.5.4.10.1.1. In the pseudo-static method, the soil mass within the failure surface also experiences a lateral body force. This body force, i.e., gravity force, emulates inertia effects generated by horizontal accelerations within the soil.

The applied lateral forces depend on the magnitude of the horizontal seismic coefficient (K_h). For the RB, K_h and the Peak Ground Acceleration ratio (PGA/g) are assumed equal. The PGA for rock foundation is obtained by averaging the median and the bounding spectral acceleration at 100 Hz from the 5 percent damped horizontal site-specific Foundation Input Response Spectra (FIRS). This acceleration value is found to be 0.55g, which translates into a seismic coefficient of K_h=0.55. At this stage of the analysis, for structures surrounding the RB on the other hand, K_h is assumed to be half the seismic coefficient of the RB. Rounding up, the seismic coefficient for the facilities surrounding the RB becomes K_h = 0.3.

Allowable bearing pseudo-static capacity (q_{aE}) is defined as the ultimate pseudo-static bearing capacity divided by the dynamic factor of safety (FS_E). A safety factor no lower than two is typically required for foundation bearing capacity involving seismic input (Reference 2.5.4-46). The safety factor reduction is consequence of the transient nature of the dynamic load. A FS_E of 2 is applied to each of the methods described above.

Soubra's methodology presents an allowable vertical pseudo-static capacity of 168 ksf for the RB. This value corresponds to a horizontal seismic coefficient of K_h = 0.55. The larger capacity compared to the static counterpart comes from differences between the safety factors and contributions from cohesion. Thus, q_{aE} = 168 ksf is the recommended allowable pseudo-static capacity for the RB foundation.

For the mat foundations of other structures within the footprint of Unit 1, Figure 2.5.4-47 illustrates the distribution of effective plastic strain and the percentage of applied load at the last converged state. In the simulations a series of parametric studies were performed in which for a wide range of foundation widths, all foundations were subjected to a target vertical load of 10 ksf and K_h=0.3. Despite differences in the spatial distribution of the effective plastic bands, foundation widths consistently anticipate maximum load ratios varying between 26 and 34 percent of the prescribed load, returning an average ultimate pseudo-static capacity of 3 ksf. Figure 2.5.4-48 summarizes the vertical load ratios at failure. By applying FS_E=2.0, the average allowable capacity reaches 1.5 ksf. This average capacity needs to be further modified by the corresponding shape factor to consider the impact of the foundation's geometry. Thus, q_{aE} =1.5(1-0.4B/L) ksf is the recommended allowable pseudo-static capacity for the mat foundations of the Power Block structures. A summary of q_{aE} values computed for the various structure foundations at Unit 1 location is provided in Table 2.5.4-27R.

The results of the bearing capacity analyses show that the rock layers below the SC1 structure, i.e., RB foundation, show significantly high static and dynamic bearing capacity. Other structures within the footprint of Unit 1 layout that are constructed near the final plant grade did not show high bearing capacity due to the presence of the existing fill and residual soil layer in the subsurface profile. It is also noted that the lower range of mechanical properties of existing fill and residual soil were used in the analyses to provide a more conservative evaluation. In order to provide sufficient bearing capacity for these structures, additional measures will be taken to improve the shallow surface soil conditions at the site. Given the relatively shallow depth to the competent bedrock, drilled shafts socketed in unweathered rock (Benbolt) are anticipated to improve the bearing capacity of the surface founded structures. Since the design of these structures is still in progress, the bearing capacity analyses will be re-evaluated and revised at the final design stage.

2.5.4.10.2 Settlement and Heave Analysis *(Addresses ESP COL Action Item 2.5-13)*

Replace ESPA SSAR Section 2.5.4.10.2 with the following:

Given the deep excavation required for the construction of the RB, an unloading (heave) of the underlying bedrock during construction is expected. Settlement and heave analyses using rock mass properties are evaluated for the RB foundation. Moreover, in addition to the RB, there are several surface-founded structures around the RB shaft for which the settlement is evaluated. Three-dimensional (3D) FE modeling is used by incorporating the subsurface stratigraphy and rock mass properties (as discussed in Subsection 2.5.4.2.4.4) to simulate the subsurface condition and estimate settlement and heave under the structures of Unit 1 at the CRN Site.

2.5.4.10.2.1 Settlement Analysis

Replace ESPA SSAR Section 2.5.4.10.2.1 with the following:

The SC1 structure (RB) at the CRN Site has an embedment depth that is expected not to exceed 115 ft below finished grade; thus, this structure is founded directly on bedrock and settlement, if any, is expected to be small. A foundation contact pressure of 8.4 ksf from the RB is used for the analysis. A 3D finite element model has been developed to evaluate the foundation settlement. The details and results of this analysis are presented in Subsection 2.5.4.13. The 3D model is attempting to simulate the actual field conditions at and around the RB by incorporating the initial state of the subsurface stratigraphy, rock mass properties and rock bedding planes, various interface conditions, potential impact of blasting during excavation, changes due to construction sequences, and applying the structure service loads. The results confirmed that the settlement of the RB foundation remains below 0.05 in. (Table 2.5.4-33R).

The CRN Site ESPA conducted settlement and heave analyses using rock mass properties. Rockmass deformation modulus, E_{rm}, values for combinations of D and GSI (E_{rm} values using the Reference 2.5.4-22 and Reference 2.5.4-23 methods as described in Subsection 2.5.4.2.4.4) were used in the analyses. The Hoek and Diederichs method (Reference 2.5.4-22) incorporates both GSI and D while the Gokceoglu (Reference 2.5.4-23) method incorporates only GSI. This

results in three E_{rm} values for each of the upper-bound and lower-bound GSI with D = 0 and D = 0.7 (two are Hoek and Diederichs, Reference 2.5.4-22, and one is Gokceoglu (Reference 2.5.4-23). According to the CRN Site ESPA findings, using these six E_{rm} values leads to six estimations of settlement and heave for the foundation and stratigraphic unit, representing the three E_{rm} calculation methods and the conditions for an undisturbed and a disturbed rock mass. For settlement and heave analyses, the 2013 CRN SMR analysis considered each stratigraphic unit within the depth of influence of a respective foundation separately, this is, as a single infinite rock layer below the foundation. Consequently, settlements associated with layers of low stiffness exhibited the larger settlement. Per Table 2.5.4-28R, the estimated total settlements range between 0.01 to 0.28 in.

In the case of the surface founded structures, the settlement analysis in the 3D FE model started with a base model, in which the shallow foundation of the structures rest on a layer of engineered backfill that is underlain by the existing fill and residual soil. The results of the settlement analyses from the base model for the surface founded structures showed that the considered total and differential settlement thresholds were exceeded. It is noted that the lower range of mechanical properties of soil and rock were used in the analyses to provide a more conservative evaluation.

In order to overcome the settlement issue for these surface-founded structures, additional measures will be taken to improve the shallow surface soil conditions at the site. Given the relatively shallow depth to the competent bedrock, drilled shafts socketed in unweathered rock (Benbolt) are anticipated. In the preliminary design of the drilled shaft, the designer considered the consolidating layer (existing fill and residual soil) into account in the down-drag calculations to support the design. Therefore, the need for consolidation analysis is eliminated. The impact of drilled shafts is evaluated using equivalent uniform properties as discussed in Subsection 2.5.4.13.

2.5.4.10.2.2 Heave Analysis

Renumber Equation 2.5.4-11 to Equation 2.5.4-8 in ESPA SSAR Section 2.5.4.10.2.2.

Replace the second paragraph of ESPA SSAR Section 2.5.4.10.2.2 with the following:

Table 2.5.4-32R shows that total heave ranges from 0.01 in. to 0.36 in. and the largest estimated total heave is less than 0.5 in. The measured heave is likely closer to the heave value given for the larger GSI since the lower heave value represents a disturbed zone (D = 0.7) for the full depth of influence beneath the foundation; in practice, the disturbed zone due to excavation is considerably less than this.

A more detailed computation of the anticipated heave is obtained by means of the 3D FE model mentioned before. Subsection 2.5.4.13 presents and discusses the findings from numerical simulations.

2.5.4.10.3 Lateral Earth Pressure *(Addresses ESP COL Action Item 2.5-14)*

Renumber Equation 2.5.4-12 to Equation 2.5.4-9 in ESPA SSAR Section 2.5.4.10.3.

Renumber Equation 2.5.4-13 to Equation 2.5.4-10 in ESPA SSAR Section 2.5.4.10.3.

Replace the second paragraph of ESPA SSAR Section 2.5.4.10.3 with the following:

With an embedment depth expected not to exceed 115 ft below final plant grade and assuming the materials behind the walls are in an at-rest condition, the at-rest earth pressure coefficient is used to determine lateral pressure. Reference 2.5.4-14 provides the at-rest lateral earth pressure coefficient:

Replace the last paragraph of ESPA SSAR Section 2.5.4.10.3 with the following:

Evaluation of these components and a full assessment of lateral earth pressure is conducted during the OLA. This evaluation includes finite element modeling using site-specific foundation input response spectra to determine the seismic component of lateral earth pressure.

2.5.4.11 Design Criteria *(Addresses ESP COL Action Item 2.5-15)*

Replace ESPA SSAR Section 2.5.4.11 with the following:

The criteria summarized below are considered geotechnical criteria. These criteria have been initially addressed in the 2013 CRN SMR analysis, they have been re-evaluated and confirmed upon the completion of processing and evaluating the data from the 2023 supplemental site investigation program at the CRN Site and will be evaluated again at the final design stage.

- The liquefaction potential of cohesive residual soils at the CRN Site is evaluated in Subsection 2.5.4.8. RG 1.198 provides that cohesive soils with fines content greater than 30 percent that are either classified as clays or have a PI greater than 30 percent should generally not be considered susceptible to liquefaction. The residual soils are characterized with a fines content of 80 percent and a PI of 40 percent and are thus not considered susceptible to liquefaction. Methods to evaluate the liquefaction potential of fine grained soils (References 2.5.4-41 and 2.5.4-42) confirm this conclusion. The proposed granular backfill, based on the gradation and level of compaction, was also not considered susceptible to liquefaction.
- Bearing capacity and settlement criteria for the BWRX-300 reactor technology and associated structures have not been finalized and will be addressed at the final design stage.
- Subsection 2.5.5 discusses the stability of slopes at the site; with the preliminary site grading information established, the presence of any SC1 slopes is not required for the BWRX-300 reactor technology. The PMF analysis assumes a minimum slope of 3:1.

Since the design of the BWRX-300 reactor technology is still in progress, other geotechnical-related criteria that pertain to structural design (such as wall rotation, sliding, and overturning) are addressed at the final design stage.

2.5.4.12 Techniques to Improve Subsurface Conditions *(Addresses ESP COL Action Items 2.5-7 and 2.5-9)*

Replace the first paragraph of ESPA SSAR Section 2.5.4.12 with the following:

The presence of karst at the CRN Site is described in Subsection 2.5.1.2.5. Given the RB will be constructed inside a deep vertical shaft, the use of geophysical methods from the surface to evaluate the presence of karst below the elevation of the RB foundation is difficult. Instead, a subsurface investigation using other geophysical methods to evaluate the presence of karst is recommended. A combined survey using ground penetrating radar (GPR) and a multi-frequency electromagnetic (EM) survey will be implemented in the base of the excavation as it progresses. The goal of this investigation is to detect any potential voids below the foundation level within a certain zone of influence (void zone of influence). The void zone is developed considering the minimum size and location (both area and depth) of a potential void beneath the foundation that does not detrimentally impact the foundation. The density of GPR/EM lines and the number of surveys (every lift, every other lift, etc.) implemented would depend on the depth of the zone of influence and the resolution required to detect potential voids. If anomalies are identified during the geophysical investigation, core drilling will be conducted to validate the anomalies and identify remedial needs. Remediation may include grouting.

Replace the third, fourth and fifth paragraph of ESPA SSAR Section 2.5.4.12 with the following:

Excavation mapping will be conducted during the entire excavation as described in Reference 2.5.4-38. An instrumentation plan will be developed to monitor lateral and vertical displacement during excavation and construction. Slope inclinometers and horizontal extensometers will be installed to monitor slope movement. Extensometers will be installed to monitor heave in subsurface materials due to the excavation associated relaxation of the rock mass. Settlement monitors will be installed to monitor the vertical movement during and after construction. Piezometers will be installed to monitor changes in pore pressures during excavation and dewatering, and settlement due to construction of the structures. The RB area is underlain by hard rock, where heave and settlement due to construction activities are expected to be minimal and to take place simultaneously with construction. Any monitoring will be developed and implemented prior to the start of construction.

2.5.4.13 Foundation Assessment Model *(Addresses ESP COL Action Items 2.5-2 and 2.5-16)*

Replace ESPA SSAR Section 2.5.4.13 with the following:

A 3D finite element model was developed to simulate the actual subsurface conditions and construction sequences to the extent that is reasonably practical. This numerical model provides insights into different aspects including the settlement and heave, bearing capacity, lateral earth pressure, and hydro-static pressure under the footprint of Unit 1 at the CRN Site. Performing this additional foundation assessment is consistent with the requirements specified in Section 4 of

NEDO-33914-A (Reference 2.5.4-57) for performing foundation interface analyses (FIA). All analyses were performed using Three-Dimensional (3-D) Finite Element (FE) modeling in Midas-GTS-NX program.

The subsurface stratigraphy in the 3D model was developed based on the 2013 CRN SMR borings information and general structural geology of the site at Unit 1 location. The service load from the Reactor Building and all surrounding surface-founded structures (e.g. Control, Radwaste, Turbine, and Service Buildings) are modeled in the 3D FE model. The foundation of the Reactor Building is assumed to be 4.0-ft thick mat foundation, the outside diameter of the RB shaft is 36.4 m (119.4 ft), and the vertical shaft the RB is made of 2.0-ft thick composite steel brick. The top of the foundation of the surface founded structures is assumed at the final plant grade, with mat thickness that varies between 1.22 m (4.0 ft) to 2.54 m (8 ft 4 in.), depending on the structure. Figure 2.5.4-27R presents Midas-GTS-NX model layout for Unit 1.

Two cases are evaluated with the 3D model: a base model, and an alternative model (or bounding model). The base model includes a relatively shallow soil profile from the final plant grade to top of rock, which consists of engineered backfill underlain by the existing fill and residual soil layer without any other foundation improvement. After running the simulation of this base model, it was observed that the total and differential settlement of the surface founded structures exceeded the considered thresholds. To limit the settlement, an alternative foundation system with improved soil condition was developed. The improvement is intended to simulate the equivalent stiffness of 36-in-diameter drilled shafts (caissons or drilled piers) that transfer the load from surface-founded structures to unweathered rock. The equivalent stiffness was estimated by performing a series of parametric studies. Therefore, the drilled shafts were not modeled explicitly in the current 3D model. It is noted that because of the relatively large number of drilled shafts, simulating them would significantly increase the computational demands of the 3D model and since their design at this stage is considered preliminary, the computation of such model would not be flexible to any potential design changes of the drilled shaft before the design is finalized.

The Support of Excavation (SOE) system is assumed to maintain a stable excavation and limit the lateral displacement of the surrounding shallow soil and weathered rock masses during construction of the deeply embedded vertical shaft at CRN Site. This system includes temporary excavation support (anchored soldier pile wall with lagging) and compression rings. Figure 2.5.4-28R shows a conceptual schematic of the RB vertical shaft, shoring system, and caissons in a cross-section view. Figure 2.5.4-29R presents the Finite Element 3D model mesh for Unit 1 (note that the drilled shafts were modeled using equivalent stiffness properties).

To account for the disturbance in rock caused by blasting during the rock mass excavation, a 25-ft thick zone of disturbed rock mass was assumed around the RB shaft perimeter and below the bottom of excavation. The disturbed rock mass properties for the Benbolt formation were calculated using the lower bound GSI equivalent properties and a Disturbance Factor (D) of 0.7.

An initial groundwater level at El. 800 ft was assumed coinciding with the existing ground surface. A long-term design groundwater level at El. 814.5 ft coinciding with the final plant grade elevation was assumed as a final condition after all surface building construction. Temporary dewatering is required to maintain a dry excavation; therefore, it was assumed dewatering progresses with

excavation and the impact of changes in the hydrostatic pressure during and after construction is evaluated. The RB shaft excavation will be performed in lifts. Prior to excavation of each lift, the water level inside the excavation will be lowered to the base of each lift, while maintaining the water elevation outside the excavation at the global initial groundwater level of El. 800 ft. A seepage analysis was then performed to estimate the pore pressures which are used as an input for the following stress-deformation excavation construction stage.

Figure 2.5.4-30R presents the excavation stages assumed for the construction of the RB vertical shaft. Figure 2.5.4-31 presents the compression rings (conceptual layout) modeled for the construction of the RB vertical shaft. Figure 2.5.4-32 presents the disturbed rock mass zone surrounding the Reactor Building vertical shaft.

The interaction between the diaphragm wall and surrounding soil/rock was modeled via interface elements in Midas-GTS-NX. Regarding the material properties of the interface, the strength reduction factor of the interface (R_{inter} = 0.7) was used to relate the interface stiffness and strength to the adjacent soil/rock stiffness and strength properties. The rock mass was modeled using an elasto-plastic Mohr-Coulomb model, in which the discontinuities within each rock mass are accounted for through the GSI values specified for each geologic unit (and subsequently the equivalent Mohr-Coulomb failure parameters φ' and c' are used). The lower bound estimates of rock mass properties were used in this model. Moreover, discontinuities along stratigraphic boundaries (such as the contact between the Benbolt and Rockdell formations) were simulated using interface elements (R_{inter} = 0.7) with reduced stiffness compared to those of the adjacent rock formations to represent potential discontinuities and fracture zones between the geologic units.

2.5.4.13.1 Foundation Assessment Results

The 3D FE model provided a large amount of data on the variation of stresses and deformations in soil and rock throughout the several construction stages. Some of the post-processed outputs from the 3D model are presented in this subsection. Figure 2.5.4-27R shows the locations and cross sections where the results are extracted.

Figure 2.5.4-33 presents the lateral pressures acting on the vertical walls in the RB shaft after loading Reactor Building and surrounding surface-founded structures to the design loads. The lateral pressures were extracted from the normal stresses acting on the interface elements from the alternative 3D model that uses the equivalent properties for soil and weathered rock (i.e., with the impact of drilled shafts). In addition, the RB vertical shaft experiences lateral hydrostatic pressure over its full depth. Total and effective lateral stresses acting on the RB vertical shaft walls in the Benbolt formation are very similar from all directions. However, in the subsurface soils, the lateral pressures from the Turbine and Radwaste buildings sides are higher compared to the Control and Service buildings' sides due to the higher building loads of the Turbine and Radwaste buildings and the bigger footprint of the Turbine building.

Figure 2.5.4-34 presents the lateral deformations corresponding to the loading condition in Figure 2.5.4-33. Lateral deformations of the vertical shaft walls in the subsurface soils and Benbolt formation are very similar from all directions. The lateral deformations from the Turbine and Radwaste buildings sides are higher compared to the Control and Service buildings sides

due to the higher building loads of the Turbine and Radwaste buildings and the bigger footprint of the Turbine building. These results confirm that stable conditions in rock during the excavations and construction sequences (as they were modeled in this 3D FE model) can be accomplished.

Bearing pressure under the foundations, in particular under the RB foundation, at the end of final construction stage is another important output from the 3D simulations. Figure 2.5.4-35 presents the distribution of total bearing pressure beneath all foundations by post processing the outputs of the 3D model that uses equivalent properties for soil and weathered rock (i.e., with the impact of drilled shafts). The values of the pressures at the locations of cross sections A-A', B-B', and C-C', were extracted and plotted in Figure 2.5.4-36 and Figure 2.5.4-37.

The other important output of the 3D FE model is estimates of total and differential settlement for the foundations of the RB and surface founded structures under the service loads. As anticipated, the stiff unweathered rock under the RB foundation did not deform noticeably under the applied service load. The estimated settlement was less than 0.05 inches for the RB foundation in both the base case (without the impact of drilled shafts) and alternative model (bounding model with the impact of drilled shafts) models. On the other hand, the differential settlements computed in the base case model for the surface founded structures exceeded the considered threshold of 0.5 inches in 50 ft.

The amount of unloading from excavation versus the amount of reloading from the new structures results in a net decrease in bearing pressure. The calculated settlement and heave values presented in Table 2.5.4-33R are small compared to reported values in ESPA (Table 2.5.4-31R and Table 2.5.4-32R) and differences in magnitude are due to the different analytical approaches available to calculate settlement and heave. The settlement is largely attributed to recompression of the material as opposed to new elastic or consolidation settlement. The estimated heave and settlement are expected to be instantaneous, occurring during and shortly after construction. No long-term settlement is expected after construction.

Table 2.5.4-33R shows the estimated settlement in the Benbolt formation at depth 115 ft below finished grade (at base of Reactor Building foundation mat) after loading the Reactor Building with the design loads. Additionally, Table 2.5.4-33R shows the estimated heave in the Benbolt formation at depth 119.5 ft below finished grade (at base of Reactor Building foundation mud-mat) from the excavation sequence. Figure 2.5.4-38 through Figure 2.5.4-40 show plot of the estimated Benbolt settlement distribution beneath the Reactor Building foundation mat. Figure 2.5.4-49 shows plot of the estimated Benbolt heave distribution beneath the Reactor Building foundation mud-mat.

The Foundation Interface Analysis is performed for Unit 1 at Location A only. It considers lower-bound GSI values and a D value of 0.7 for the disturbed zone due to excavation activities. The total estimated settlement in the Benbolt formation at depth 115 ft below finished grade (at base of Reactor Building foundation mat) after loading the Reactor Building with the design loads is less than 0.05 in. The total estimated heave in the Benbolt formation at depth 119.5 ft below finished grade (at base of Reactor Building foundation mud-mat) from the excavation sequence is less than 0.022 in. The estimated settlement and heave values from the Foundation Interface

Analysis are smaller and comparable to the values obtained from the analytical methods presented in Subsection 2.5.4.1.1.1 and Subsection 2.5.4.1.1.2 for Location A and lower-bound GSI values, respectively.

Figure 2.5.4-38 presents the settlement distribution computed for all foundations within the footprint of Unit 1 by post processing the outputs of the alternative 3D model that uses equivalent properties for soil and weathered rock (i.e., with the impact of drilled shafts). The settlement values at the locations of cross sections A-A', B-B' and C-C' (illustrated on Figure 2.5.4-38) were extracted and plotted in Figure 2.5.4-39 and Figure 2.5.4-40. The settlement values computed in the alternative model are smaller than the corresponding values from the base case model. Additionally, the differential settlements resulted from the alternative 3D model (with the equivalent properties for soil and weathered rock, i.e., with the impact of drilled shafts) were all below the considered threshold of 0.5 inches in 50 ft. It is anticipated that stiff drilled shaft elements will most likely provide smaller settlement values; therefore, the improvement introduced by the drilled shaft will provide stable conditions.

The impact of potentially undetected karst features under or around the RB foundation on the stability of the SC1 RB was evaluated in detail by considering the BWRX-300 technology initial general layout (Reference 2.5.4-65) at CRN Site. According to the subsurface data from the 2013 and 2023 investigations, approximately 99 percent of the cavities observed in the borings are significantly smaller than 11 ft and the inferred height and the maximum observed cavity height did not exceed 17 ft. Moreover, cavity development in the CRN Site areas is generally limited to the most markedly weathered zone immediately below ground surface, to depths less than 100 ft; in fact, about 75 percent of the reported cavities in the borings from the 2013 and 2023 investigations had been observed at depths less than 55 ft.

The selection of the size and location of the karst cavities and disturbed rock mass zone was in part informed by the 2013 CRN SMR investigation data (Reference 2.5.4-1). It is noted that in order to simulate the actual condition at the CRN Site, additional 3D FE models were developed to investigate the impact of undetected cavities on the stability of the new proposed system. The size of the cavities in the FE models are compatible with the height of the cavities detected and observed in the 2013 CRN SMR investigation program. Consistent with the previous cavity evaluation, two spherical karst features (cavities) were considered to examine the stability of the BWRX-300 RB deeply embedded vertical shaft and surrounding structures. One of the karst features, modeled as a 15-ft (diameter) cavity, is located 5 feet beneath the centerline of the RB mat foundation and the second one, with similar dimensions, is located 5 feet adjacent to the RB vertical shaft and 2 feet below the top of the unweathered rock below the Radwaste Building (heaviest surface structure). To account for the rock disturbance and stress release from blasting activities during excavation, a 25 feet thick zone of disturbed rock mass (D=0.7) is assumed around the shaft perimeter and 25 feet below the RB mat foundation base. The two cavities are embedded in these weakened zones. Additionally, the lower range mechanical properties have been assigned to the soil and rock mass in these simulations.

The results of the 3D FE models were evaluated in the context of the impact of the potential cavities on the deformations and stress distribution in the domain at different stages of construction and loading. Similar to the previous results, these simulations showed that the

postulated cavities did not have any meaningful impact on the stability of the foundation. This study was based on the first generation layout (Reference 2.5.4-65); however, at the final design stage, the 3D finite element model will be re-evaluated and updated based on any changes on structures in terms of location, dimension, elevation of foundations, and subgrade material properties.

2.5.4.14 References

Replace the following References in ESPA SSAR Section 2.5.4.14:

2.5.4-46 IAEA Safety Standards, “Geotechnical Aspects of Site Evaluation and Foundations for Nuclear Power Plants, Safety Guide No NS-G-3.6,” International Atomic Energy Agency, 2004.

2.5.4-52 Not Used.

2.5.4-57 NEDO-33914-A, “BWRX-300 Advanced Civil Construction and Design Approach,” GE Hitachi Nuclear Energy, Revision 2, June 2022.

2.5.4-59 Not Used.

2.5.4-60 Not Used.

Add the following References in ESPA SSAR Section 2.5.4.14:

2.5.4-62 Soubra, A.H., “Upper-Bound Solutions for Bearing Capacity of Foundations,” Journal of Geotechnical and Geoenvironmental Engineering, 125:59–68, 1999.

2.5.4-63 Not Used.

2.5.4-64 CRN ESPA, “Clinch River Nuclear Site Early Site Permit Application, Part 2, Site Safety Analysis Report,” Tennessee Valley Authority (TVA), Revision 2, January 18, 2019.

2.5.4-65 Black and Veatch, “Site Single Unit Plot Plan, Drawing ID: 410214-1GAU-G1001, Rev. 0.”

2.5.4-66 RIZZO, “Supplemental Geotechnical Field Investigation, Data Analysis Report, Clinch River Nuclear Site, RIZZO International Inc., Pittsburgh, Pennsylvania, Revision 0,” December 20, 2024.

2.5.4-67 S&ME, “Data Report, CRN Supplemental Geotechnical Investigation, Oak Ridge, Tennessee, Document Control No. CRN-23-0140-A, Revision C,” March 25, 2024.

Rename Table 2.5.4-3 (Sheets 1-3 of 3) to ESPA SSAR Table 2.5.4-3R (Sheets 1-3 of 3), Summary of Thicknesses of the Existing Fill/Residual Soil and Weathered Rock and Depth to the Top of Bedrock from the 2013 CRN SMR Investigation

Renumber Table 2.5.1-1 as Table 2.5.1-1R in ESPA Table 2.5.4-18.

Rename and Replace Table 2.5.4-27 with the following:

Table 2.5.4-27R Summary of Allowable Bearing Capacity Values at Unit 1 Location

Structure	Static Bearing Capacity (ksf)		Dynamic Bearing Capacity (ksf)	
	Calculation Method	**Allowable Capacity (ksf)**	**Calculation Method**	**Allowable Capacity (ksf)**
Reactor Building	U.S. Army Corps	76.0	Soubra (K_h = 0.55)	168
Power block Structures	2D FEM Models	1.7*(1-0.4*B/L) for B<40 ft 2.5*(1-0.4*B/L) for B>60 ft	2D FEM Models (K_h = 0.3)	1.5*(1-0.4B/L)

Renumber Table 2.5.4-28 to Table 2.5.4-31R.

Renumber Table 2.5.4-29 to Table 2.5.4-32R.

Renumber Table 2.5.4-30 to Table 2.5.4-28R and Replace with the following:

Table 2.5.4-28R Smoothed Base-case Shear Wave Velocity Profiles and Densities of Unweathered Rock for Location A
(Sheet 1 of 2)

Thickness (m)	Depth (m)	BE V_S (m/s)	LR V_S (m/s)	UR V_S (m/s)	Density (g/cm^3)
3.00	3.00	2321.50	1857.20	2901.88	2.69
3.00	6.00	2321.50	1857.20	2901.88	2.69
2.25	8.25	2196.20	1756.96	2745.25	2.69
2.26	10.52	2196.20	1756.96	2745.25	2.69
2.50	13.01	2398.10	1918.48	2997.63	2.69
2.50	15.51	2398.10	1918.48	2997.63	2.69
2.99	18.50	2398.10	1918.48	2997.63	2.69
3.00	21.51	2398.10	1918.48	2997.63	2.69
3.00	24.51	2398.10	1918.48	2997.63	2.69
3.00	27.51	2398.10	1918.48	2997.63	2.69
3.00	30.51	2398.10	1918.48	2997.63	2.69
3.00	33.51	2398.10	1918.48	2997.63	2.69
3.00	36.51	2398.10	1918.48	2997.63	2.69
2.75	39.26	2398.10	1918.48	2997.63	2.69
2.75	42.02	2398.10	1918.48	2997.63	2.69
2.75	44.77	2398.10	1918.48	2997.63	2.69
2.75	47.52	2398.10	1918.48	2997.63	2.69
3.10	50.61	2615.30	2092.24	3269.13	2.69
3.10	53.71	2615.30	2092.24	3269.13	2.69
3.10	56.81	2615.30	2092.24	3269.13	2.69
3.10	59.90	2615.30	2092.24	3269.13	2.69
3.10	63.00	2615.30	2092.24	3269.13	2.69
3.25	66.25	2894.30	2315.44	3520.00	2.69
3.25	69.51	2894.30	2315.44	3520.00	2.69
3.25	72.76	2894.30	2315.44	3520.00	2.69
3.25	76.01	2894.30	2315.44	3520.00	2.69
3.10	79.11	2894.30	2315.44	3520.00	2.69
3.10	82.21	2894.30	2315.44	3520.00	2.69
3.10	85.31	2894.30	2315.44	3520.00	2.69
3.10	88.40	2894.30	2315.44	3520.00	2.69
3.10	91.50	2894.30	2315.44	3520.00	2.69
3.47	94.97	2743.07	2194.45	3428.83	2.69
3.47	98.45	2743.07	2194.45	3428.83	2.69

Table 2.5.4-28R Smoothed Base-case Shear Wave Velocity Profiles and Densities of Unweathered Rock for Location A
(Sheet 2 of 2)

Thickness (m)	Depth (m)	BE V_S (m/s)	LR V_S (m/s)	UR V_S (m/s)	Density (g/cm^3)
3.47	101.92	2743.07	2194.45	3428.83	2.69
3.47	105.40	2743.07	2194.45	3428.83	2.69
3.47	108.87	2743.07	2194.45	3428.83	2.69
3.47	112.35	2743.07	2194.45	3428.83	2.69
3.47	115.82	2743.07	2194.45	3428.83	2.69
3.47	119.30	2743.07	2194.45	3428.83	2.69
3.47	122.77	2743.07	2194.45	3428.83	2.69
3.47	126.25	2743.07	2194.45	3428.83	2.69
3.47	129.72	2743.07	2194.45	3428.83	2.69
3.47	133.19	2743.07	2194.45	3428.83	2.69
3.13	136.33	2194.45	1755.56	2743.07	2.69
3.13	139.46	2194.45	1755.56	2743.07	2.69
3.13	142.60	2194.45	1755.56	2743.07	2.69
3.13	145.73	2194.45	1755.56	2743.07	2.69
3.13	148.87	2194.45	1755.56	2743.07	2.69
3.13	152.00	2194.45	1755.56	2743.07	2.69
59.53	211.53	2194.45	1755.56	2743.07	2.69
31.09	242.62	2743.07	2194.45	3428.83	2.69
77.42	320.04	2499.24	1999.39	3124.05	2.69
1218.23	1538.26	3352.64	2682.11	3520.00	2.69
640.05	2178.31	2194.45	1755.56	2743.07	2.72
701.01	2879.32	3093.87	2475.10	3520.00	2.80
213.35	3092.67	2234.99	1787.99	2793.74	2.72
213.35	3306.02	2234.99	1787.99	2793.74	2.72
548.13	3854.15	3117.34	2493.87	3520.00	2.80
-	3854.15	3520.00	3520.00	3520.00	2.71

Notes:

Dash (–) is representative of the half-space with an infinite thickness and no damping.
V_S= shear wave velocity

BE = Best Estimate, LR = Lower Range, UR = Upper Range

Renumber Table 2.5.4-31 to Table 2.5.4-29R and Replace with the following:

Table 2.5.4-29R Smoothed Base-case Shear Wave Velocity Profiles and Densities of Unweathered Rock For Location B
(Sheet 1 of 2)

Thickness (m)	Depth (m)	BE V_S (m/s)	LR V_S (m/s)	UR V_S (m/s)	Density (g/cm^3)
2.01	2.01	1917.70	1534.16	2397.13	2.69
2.00	4.01	2237.30	1789.84	2796.63	2.69
2.00	6.00	2237.30	1789.84	2796.63	2.69
3.51	9.51	2237.30	1789.84	2796.63	2.69
3.51	13.01	2237.30	1789.84	2796.63	2.69
2.50	15.51	2465.80	1972.64	3082.25	2.69
3.50	19.01	2465.80	1972.64	3082.25	2.69
3.50	22.51	2465.80	1972.64	3082.25	2.69
3.50	26.01	2465.80	1972.64	3082.25	2.69
3.50	29.51	2465.80	1972.64	3082.25	2.69
3.50	33.01	2465.80	1972.64	3082.25	2.69
3.50	36.51	2465.80	1972.64	3082.25	2.69
3.50	40.02	2465.80	1972.64	3082.25	2.69
3.50	43.52	2465.80	1972.64	3082.25	2.69
3.50	47.03	2465.80	1972.64	3082.25	2.69
3.50	50.53	2465.80	1972.64	3082.25	2.69
2.82	53.35	2477.00	1981.60	3096.25	2.69
2.82	56.18	2477.00	1981.60	3096.25	2.69
2.82	59.00	2477.00	1981.60	3096.25	2.69
3.30	62.31	2838.20	2270.56	3520.00	2.69
3.30	65.61	2838.20	2270.56	3520.00	2.69
3.30	68.91	2838.20	2270.56	3520.00	2.69
3.30	72.22	2838.20	2270.56	3520.00	2.69
3.30	75.52	2838.20	2270.56	3520.00	2.69
3.16	78.68	2838.20	2270.56	3520.00	2.69
3.16	81.84	2838.20	2270.56	3520.00	2.69
3.16	85.00	2838.20	2270.56	3520.00	2.69
3.16	88.15	2838.20	2270.56	3520.00	2.69
3.16	91.31	2838.20	2270.56	3520.00	2.69
3.26	94.57	2838.20	2270.56	3520.00	2.69
3.26	97.83	2838.20	2270.56	3520.00	2.69
3.26	101.08	2838.20	2270.56	3520.00	2.69
3.26	104.34	2838.20	2270.56	3520.00	2.69

Table 2.5.4-29R Smoothed Base-case Shear Wave Velocity Profiles and Densities of Unweathered Rock For Location B
(Sheet 2 of 2)

Thickness (m)	Depth (m)	BE V_S (m/s)	LR V_S (m/s)	UR V_S (m/s)	Density (g/cm^3)
3.66	108.00	2838.20	2270.56	3520.00	2.69
3.70	111.70	3352.64	2682.11	3520.00	2.80
3.70	115.40	3352.64	2682.11	3520.00	2.80
3.70	119.10	3352.64	2682.11	3520.00	2.80
3.70	122.80	3352.64	2682.11	3520.00	2.80
3.70	126.50	3352.64	2682.11	3520.00	2.80
3.70	130.20	3352.64	2682.11	3520.00	2.80
3.70	133.90	3352.64	2682.11	3520.00	2.80
3.70	137.59	3352.64	2682.11	3520.00	2.80
3.70	141.29	3352.64	2682.11	3520.00	2.80
3.70	144.99	3352.64	2682.11	3520.00	2.80
3.70	148.69	3352.64	2682.11	3520.00	2.80
3.70	152.39	3352.64	2682.11	3520.00	2.80
1175.25	1327.64	3352.64	2682.11	3520.00	2.80
640.05	1967.69	2173.12	1738.49	2716.40	2.72
426.70	2394.39	3080.77	2464.61	3520.00	2.80
76.20	2470.59	2224.32	1779.46	2780.40	2.72
365.74	2836.33	3097.84	2478.27	3520.00	2.80
228.59	3064.92	2234.99	1787.99	2793.74	2.72
228.59	3293.51	2234.99	1787.99	2793.74	2.72
548.13	3841.64	3116.43	2493.14	3520.00	2.80
-	3841.64	3520.00	3520.00	3520.00	2.71

Notes:

Dash (–) is representative of the half-space with an infinite thickness and no damping.
V_S = shear wave velocity

BE = Best Estimate, LR = Lower Range, UR = Upper Range

Renumber Table 2.5.4-32 to Table 2.5.4-30R.

Replace Table 2.5.4-33 with the following:

Table 2.5.4-33R Total Estimated Settlement and Heave Beneath the Reactor Building Foundation from the Foundation Interface Analysis

Location A, Lower-bound GSI	
Settlement (in.)	**Heave (in.)**
0.05	0.022

Delete Table 2.5.4-34.

Replace Figures 2.5.4-1 and 2.5.4-2 with the following:

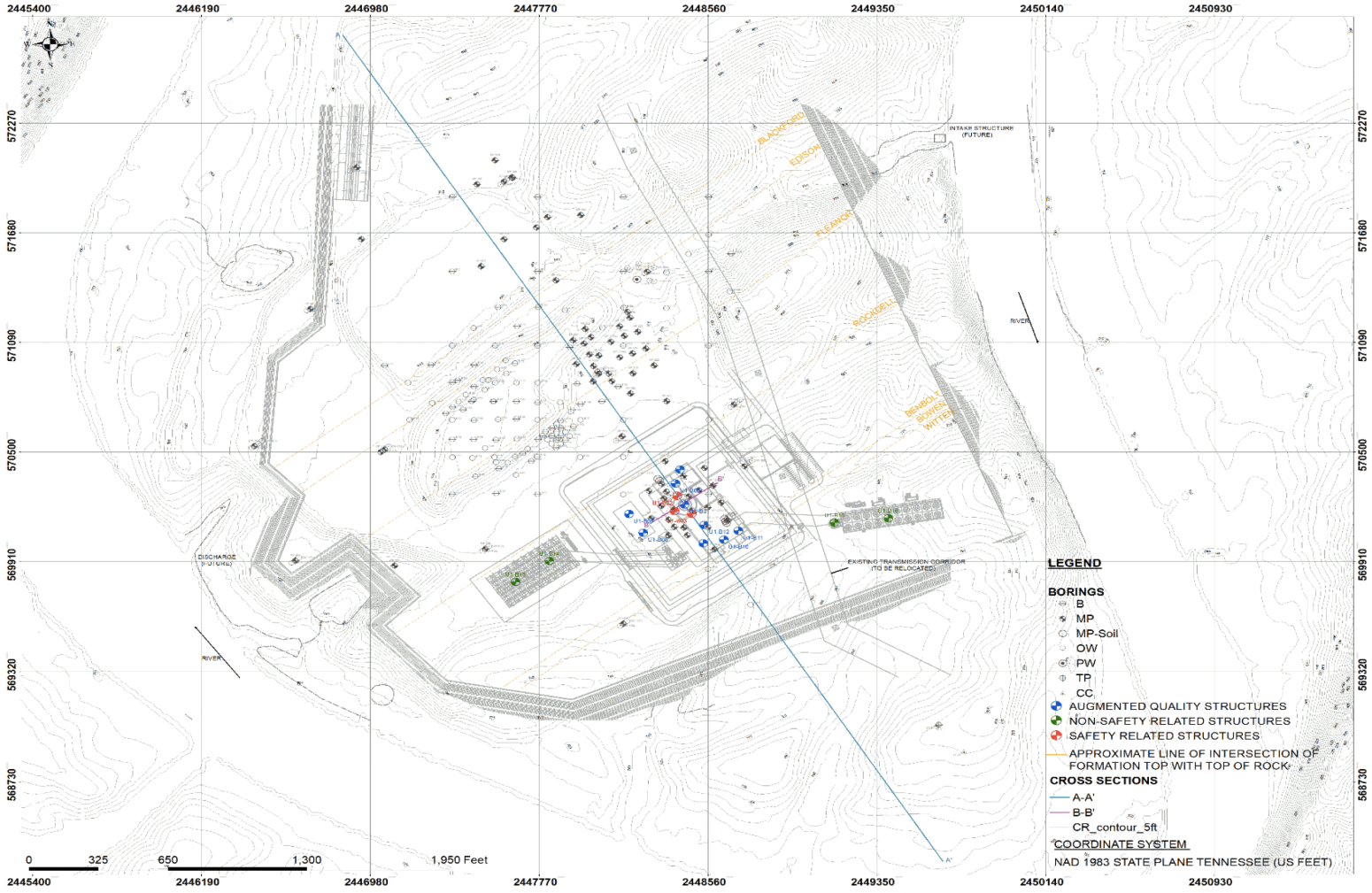


Figure 2.5.4-1R Site Layout and Boring Location Plan
(Power Block Layout and Boring Location from the 2013 and 2023 Investigations)

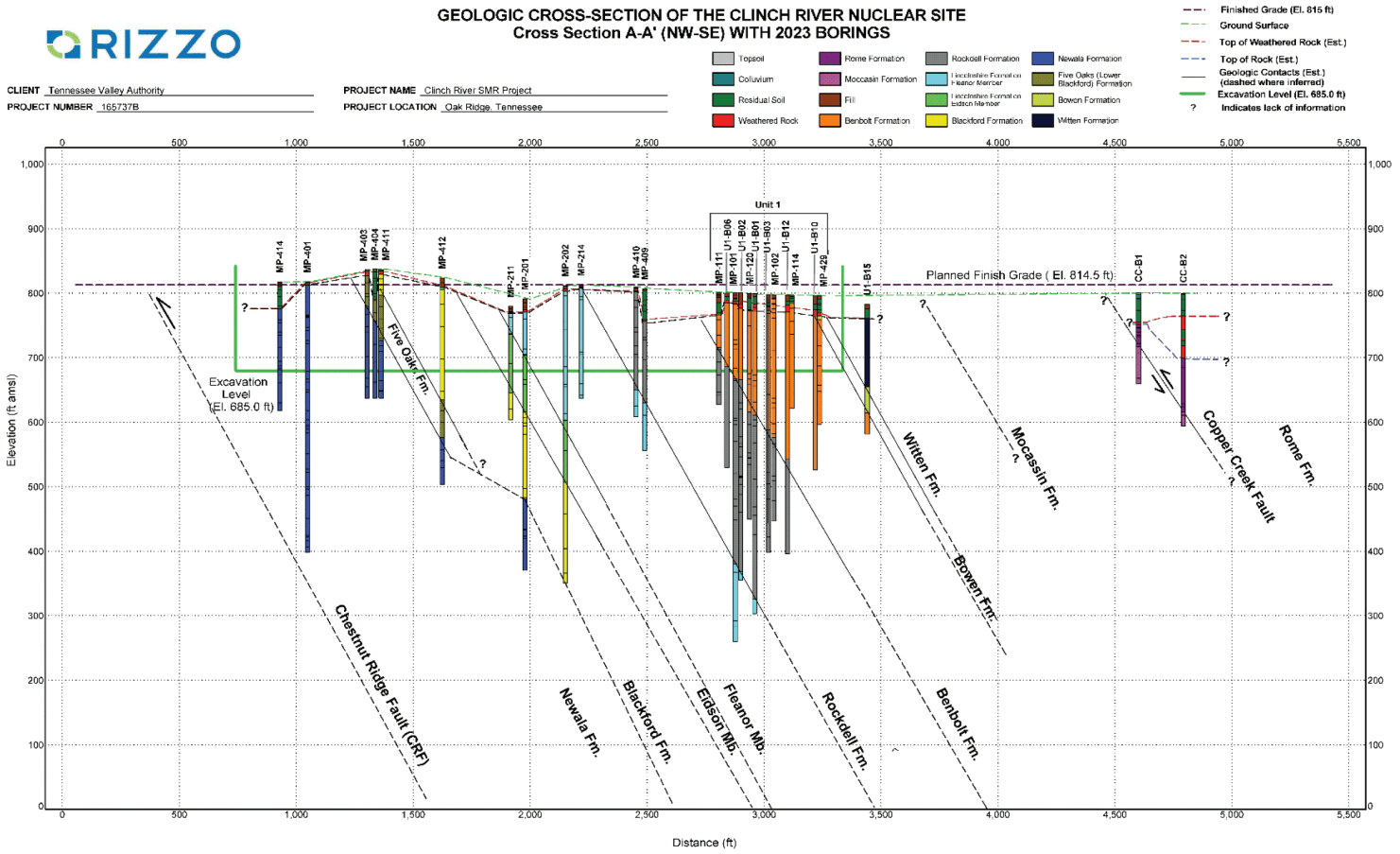


Figure 2.5.4-2R Geotechnical Cross-Section Through the Proposed Unit 1 Area

(Boreholes from the 2013 and 2023 Investigations)

Rename Figure 2.5.4-11, Boreholes from the 2013 CRN SMR Investigation at Locations A and B (as Sited in the CRN ESPA)

Rename Figure 2.5.4-12, Cross-Section Showing Borehole Locations and Location of Profiles (Locations A and B as Sited in the CRN ESPA and Marked up in Figure 2.5.4-11)

Rename Figure 2.5.4-13, Geologic Cross-Section and Location of Profiles (Locations A and B as Sited in the CRN ESPA and Marked up in Figure 2.5.4-11)

Rename Figure 2.5.4-14, Shear Wave Velocity Profiles for Location A (Data from 2013 CRN SMR Investigation, Adopted for Unit 1)

Rename Figure 2.5.4-15, Shear Wave Velocity Profiles for Location B (Data from 2013 CRN SMR Investigation)

Rename Figure 2.5.4-16, Shallow Shear Wave Velocity Profile for Location A (Adopted for Unit 1)

Rename Figure 2.5.4-18, Geologic and Shear Wave Velocity Profile for Location A (Adopted for Unit 1)

Rename Figure 2.5.4-20, Base-case Shear Wave Velocity Profiles for Location A (Adopted for Unit 1)

Replace Figures 2.5.4-27 through 2.5.4-30 with the following:

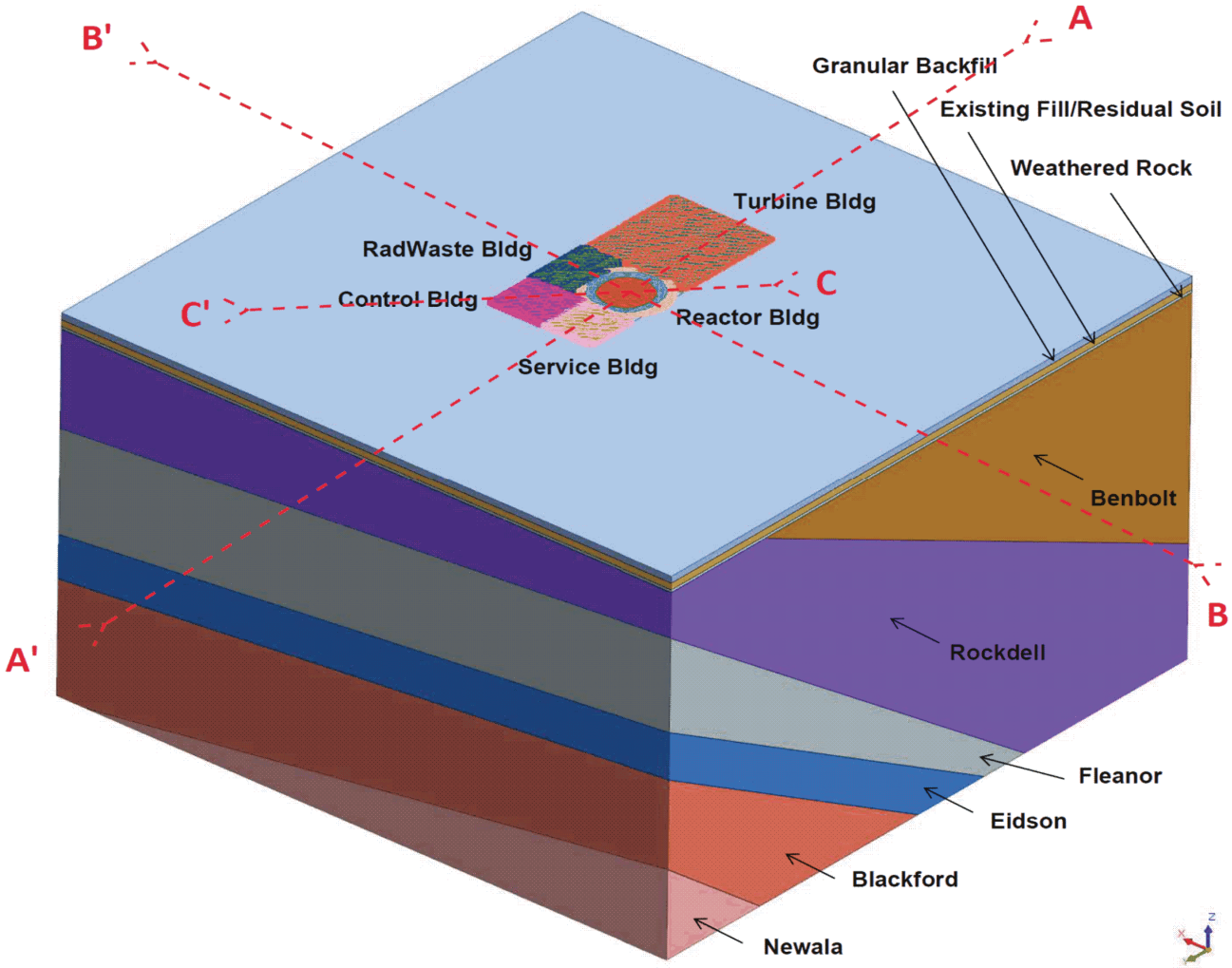


Figure 2.5.4-27R Finite Element 3D Model Layout of Unit 1 and Cross Sections Analyzed

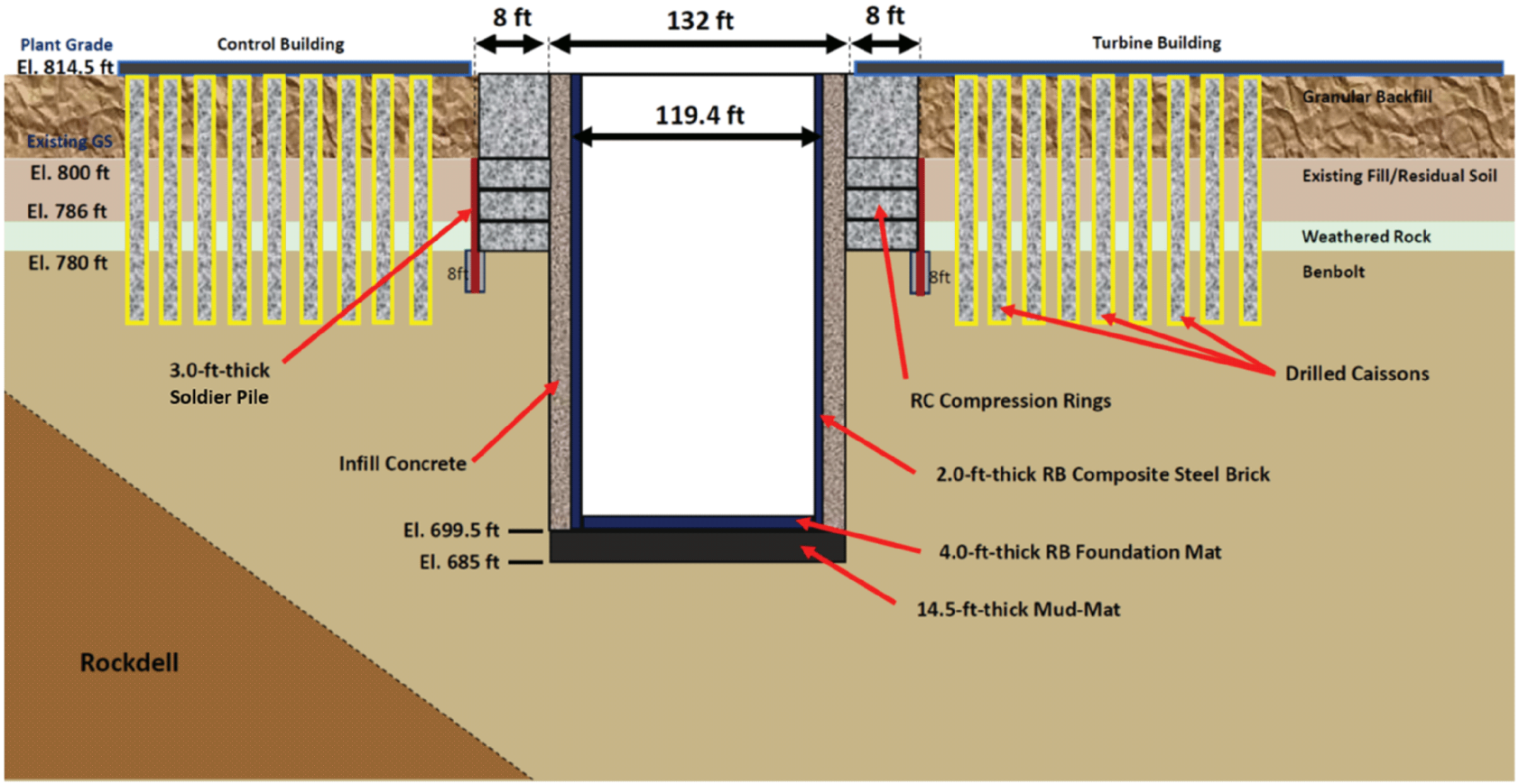


Figure 2.5.4-28R Reactor Building Vertical Shaft Conceptual Layout

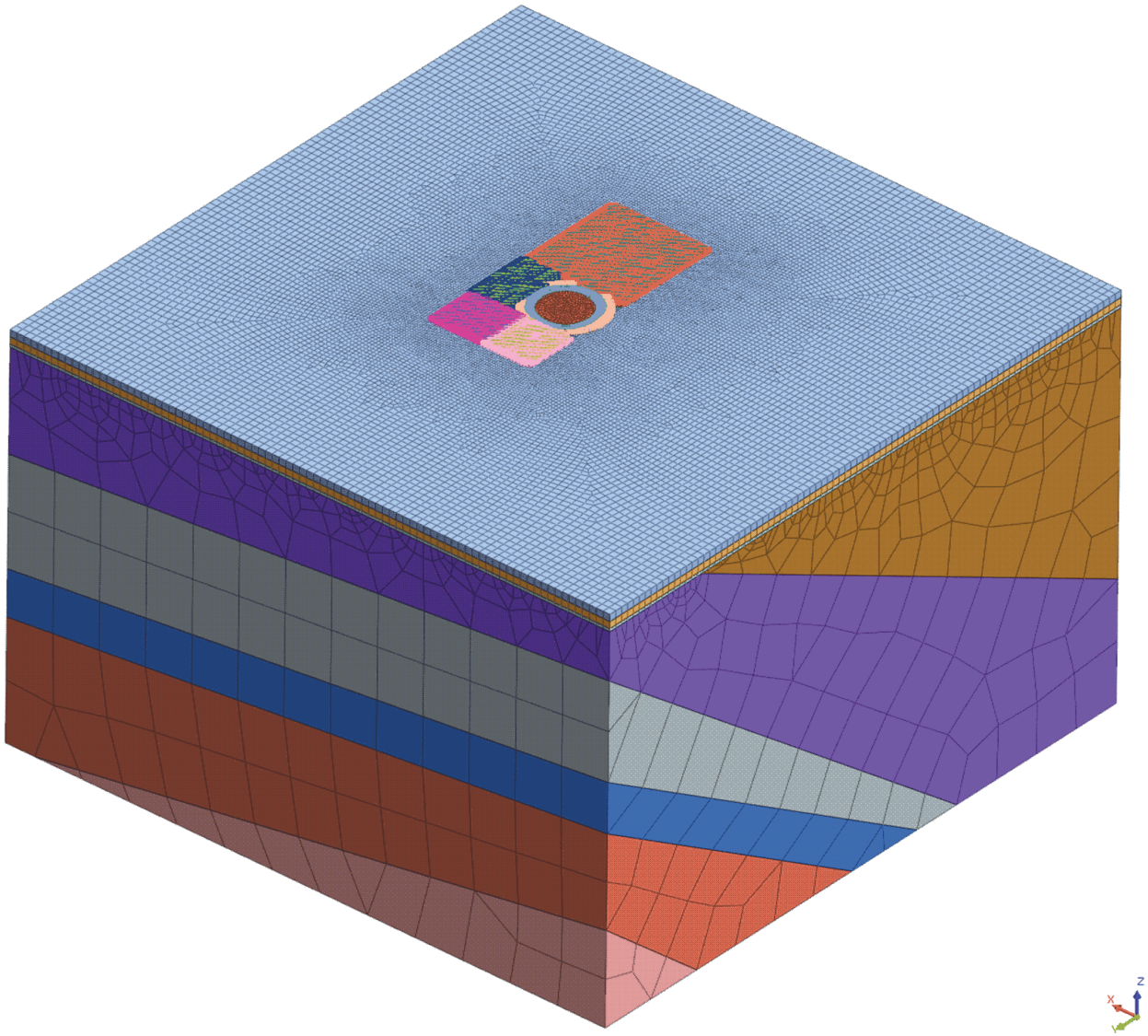


Figure 2.5.4-29R Finite Element 3D Model Mesh of Unit 1

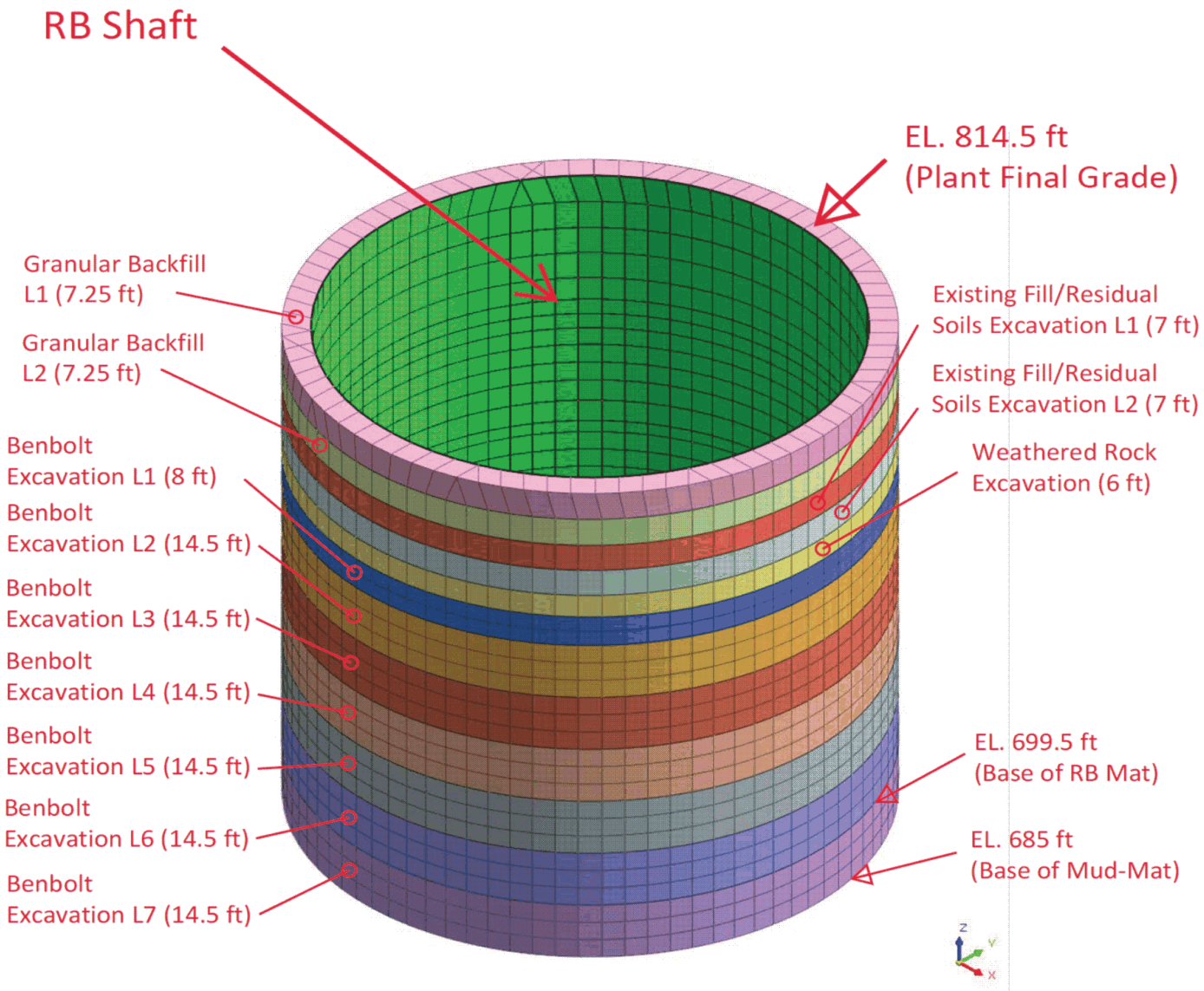


Figure 2.5.4-30R Excavation Stages for Construction of Reactor Building Vertical Shaft

Add new Figures 2.5.4-31 through 2.5.4-49:

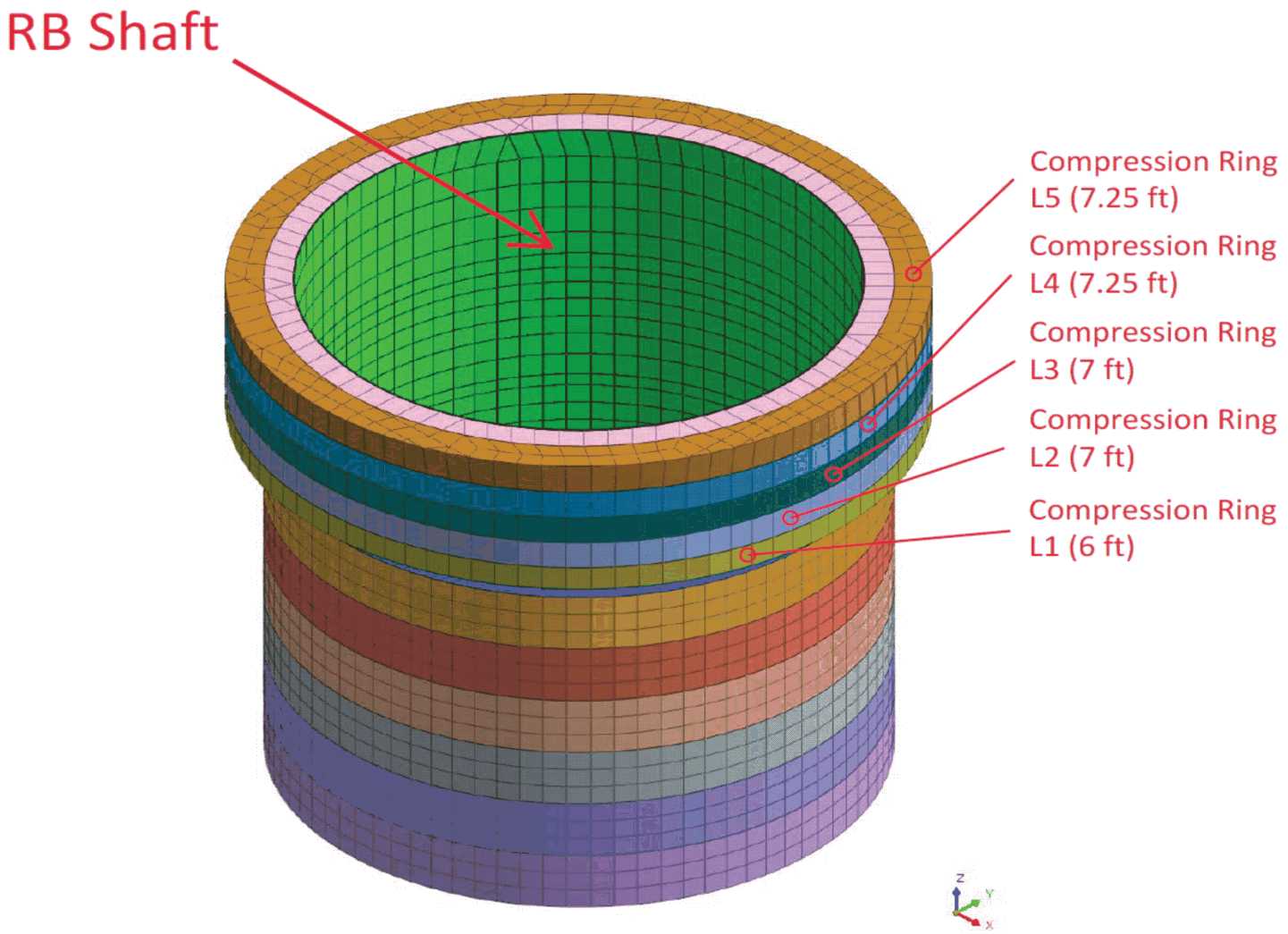


Figure 2.5.4-31 Compression Rings Layout During Construction of Reactor Building Vertical Shaft

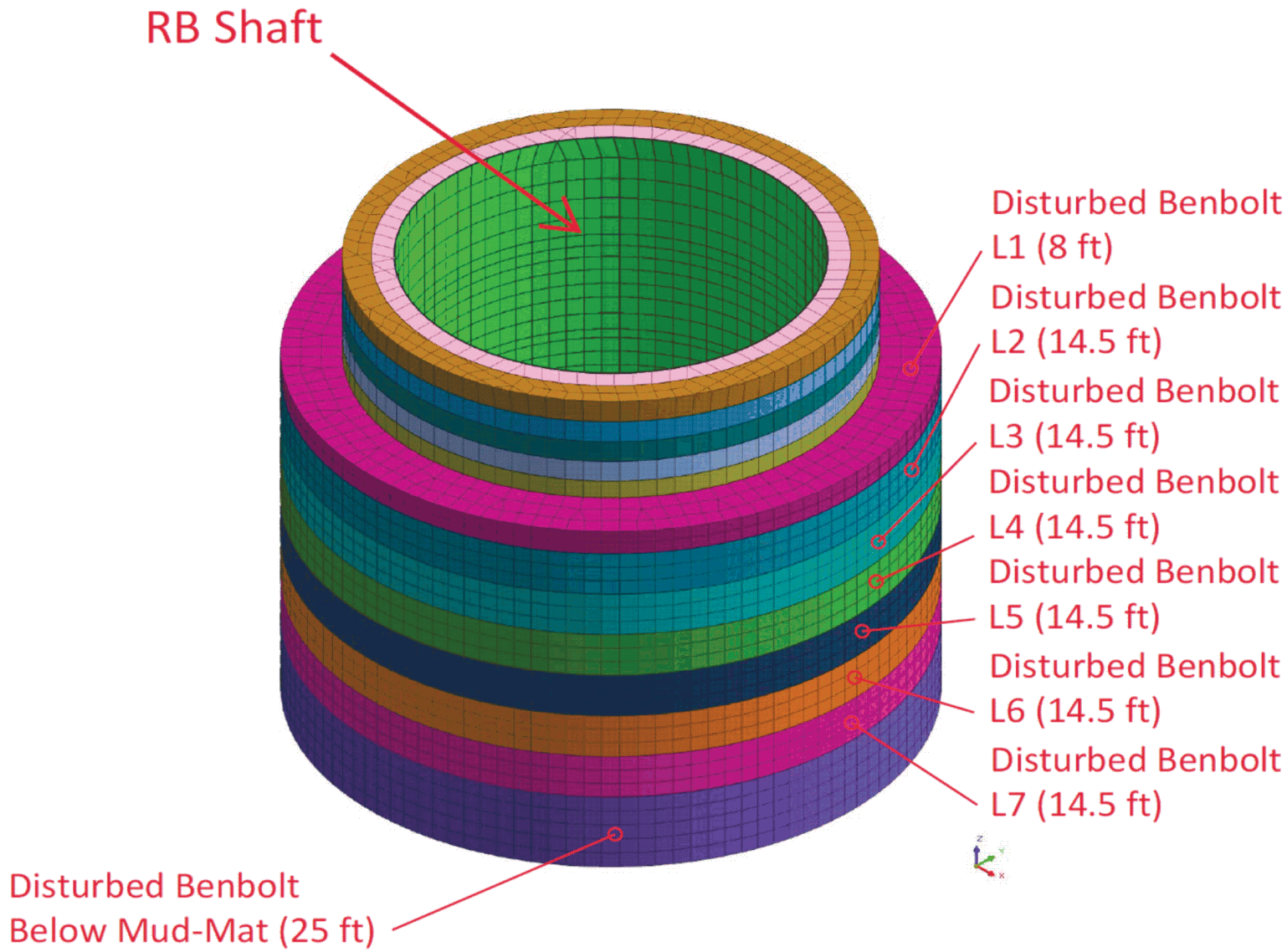


Figure 2.5.4-32 Disturbed Rock Mass Zone Surrounding Reactor Building Vertical Shaft

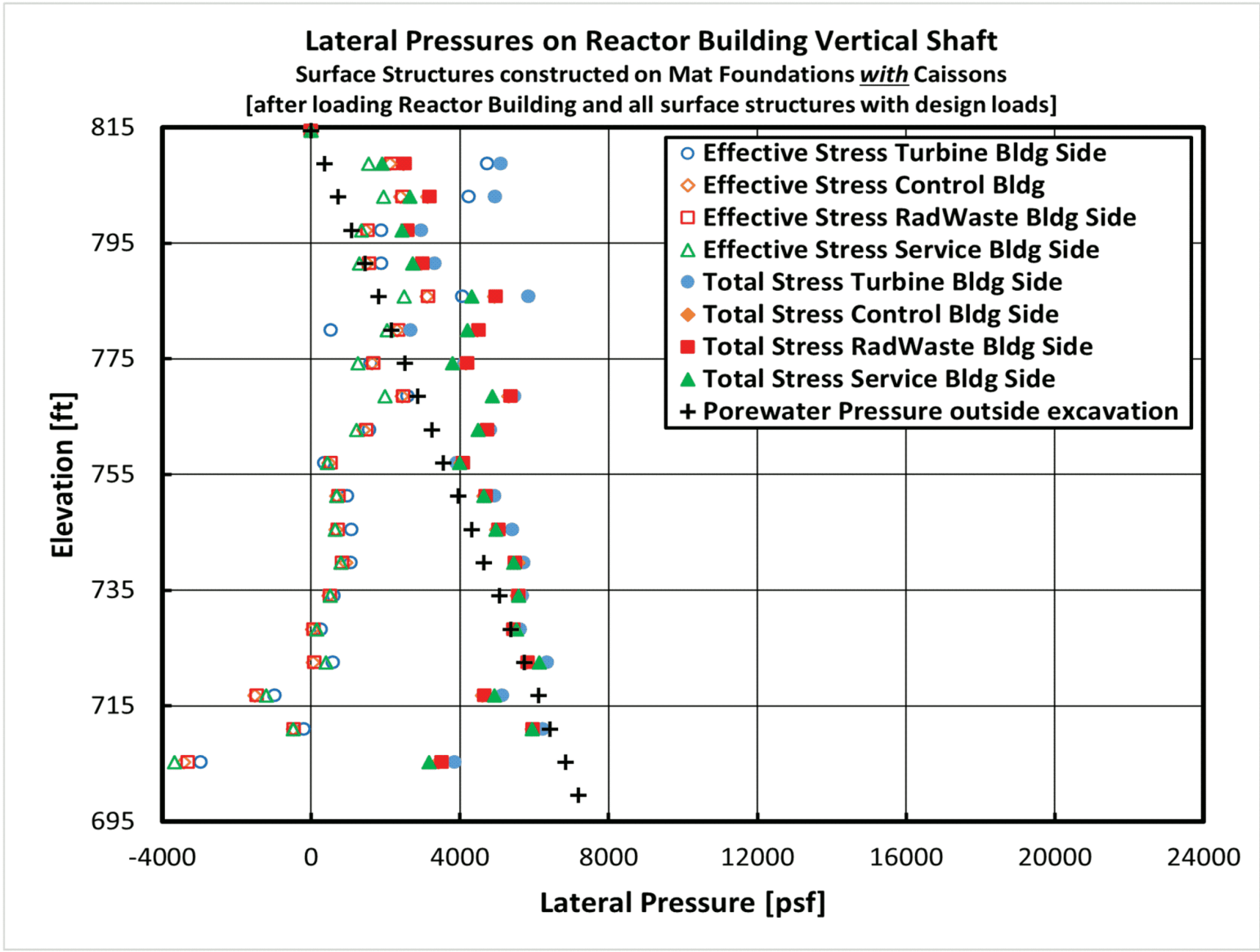


Figure 2.5.4-33 Lateral Pressures on Reactor Building Vertical Shaft after Loading all Structures (Mats With Caissons)

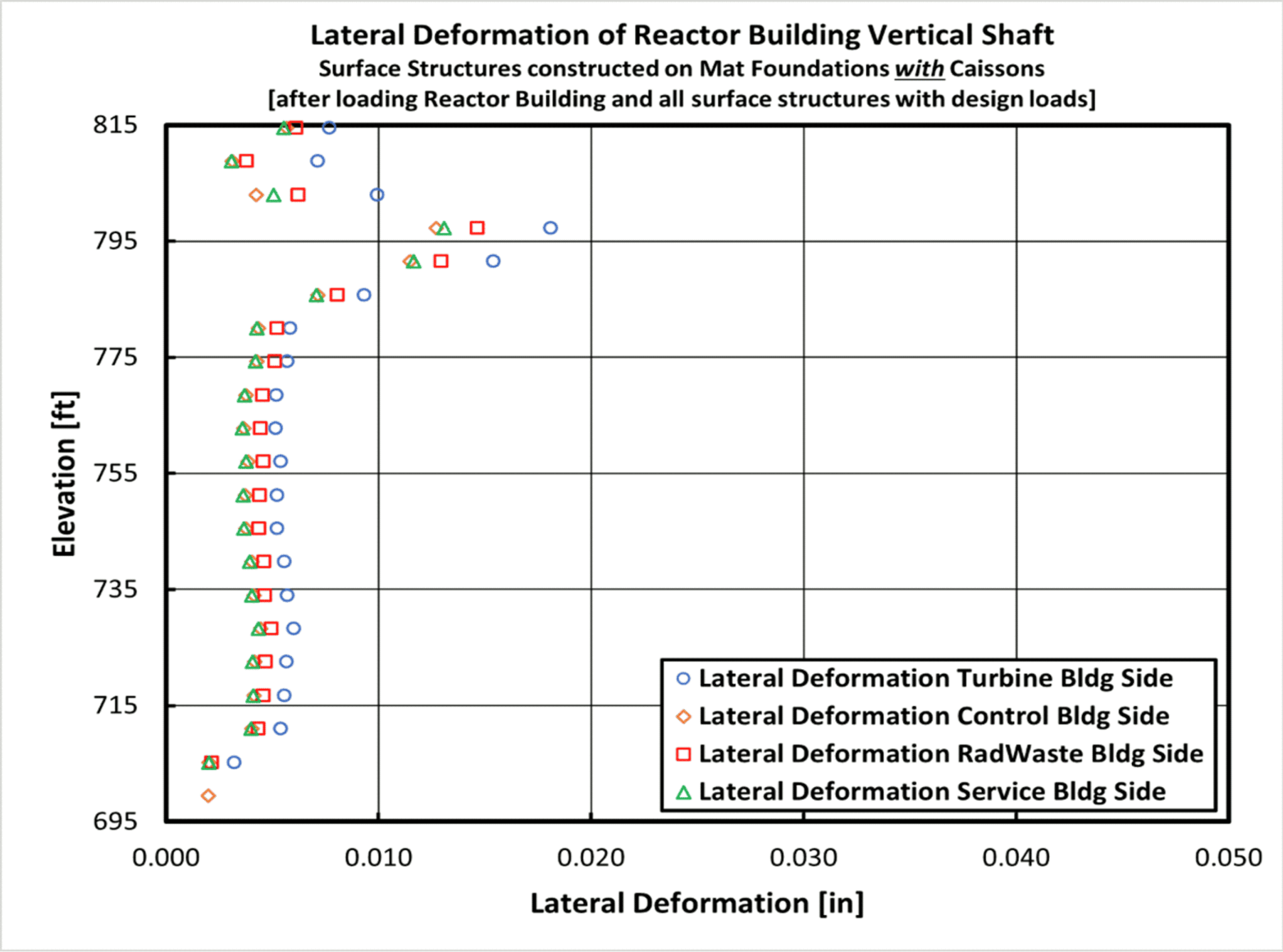


Figure 2.5.4-34 Lateral Deformation of Reactor Building Vertical Shaft after Loading all Structures (Mats With Caissons)

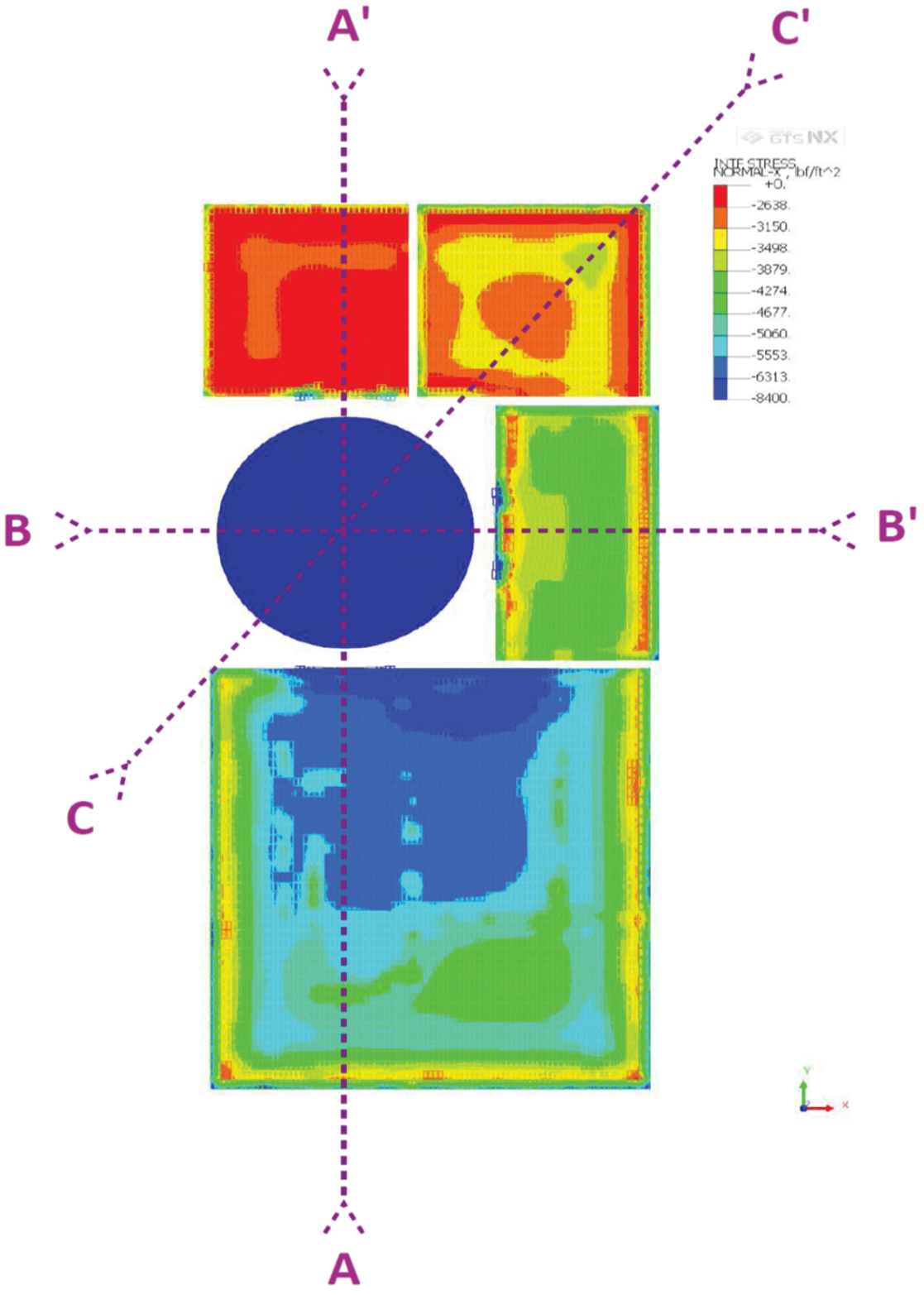


Figure 2.5.4-35 Total Bearing Pressure Distribution Beneath all Buildings Mat Foundations (Mats With Caissons)

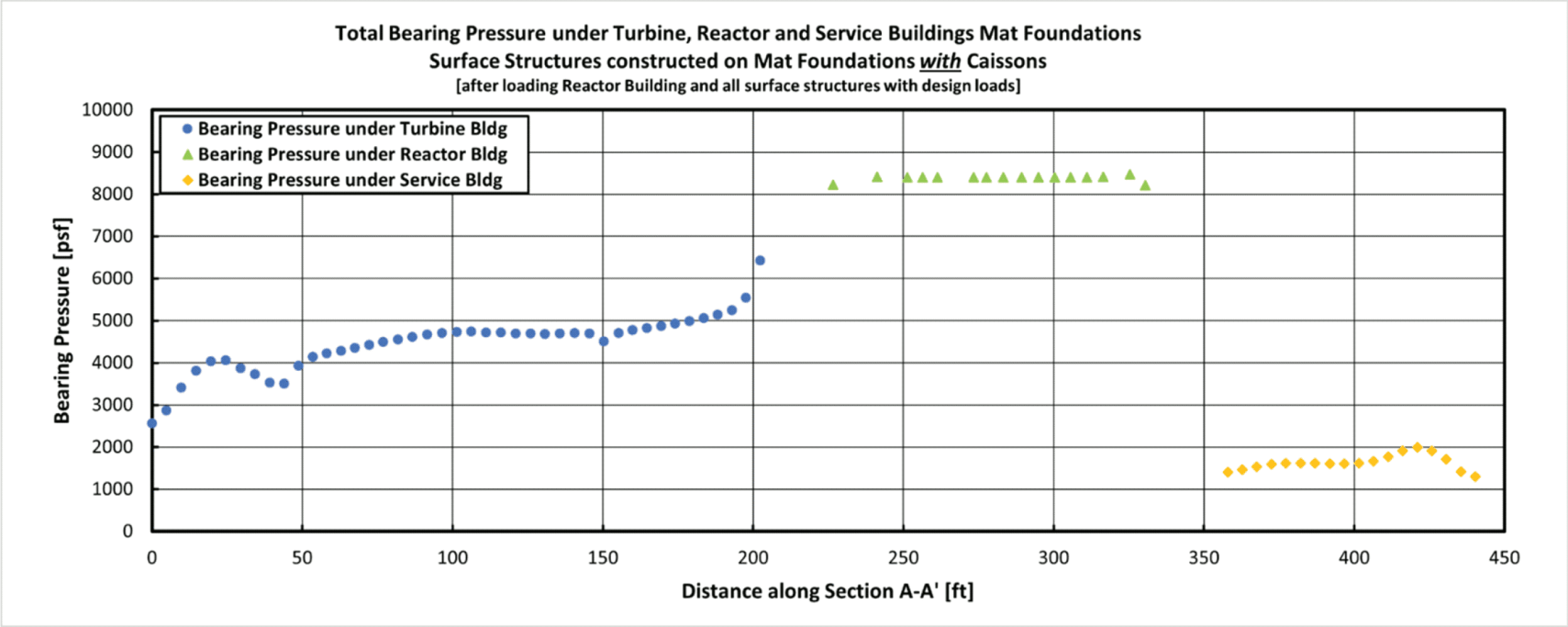


Figure 2.5.4-36 Total Bearing Pressure Distribution along Cross Section A-A' (Mats With Caissons)

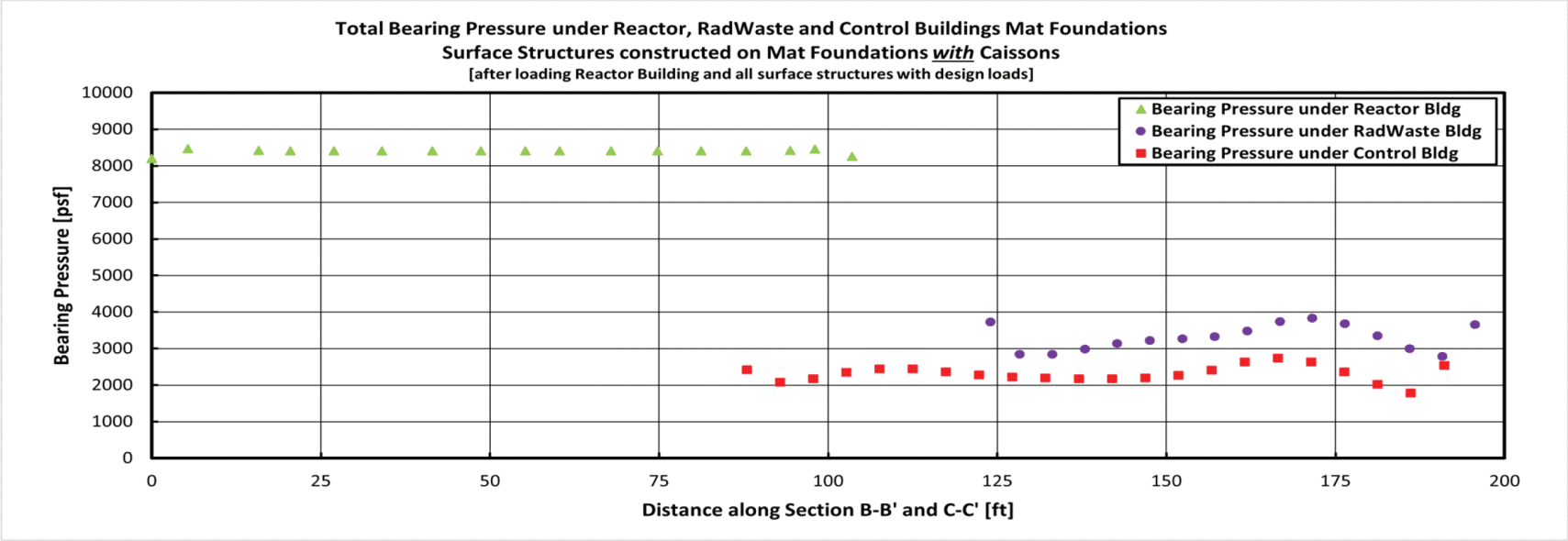


Figure 2.5.4-37 Total Bearing Pressure Distribution along Cross Sections B-B' and C-C' (Mats With Caissons)

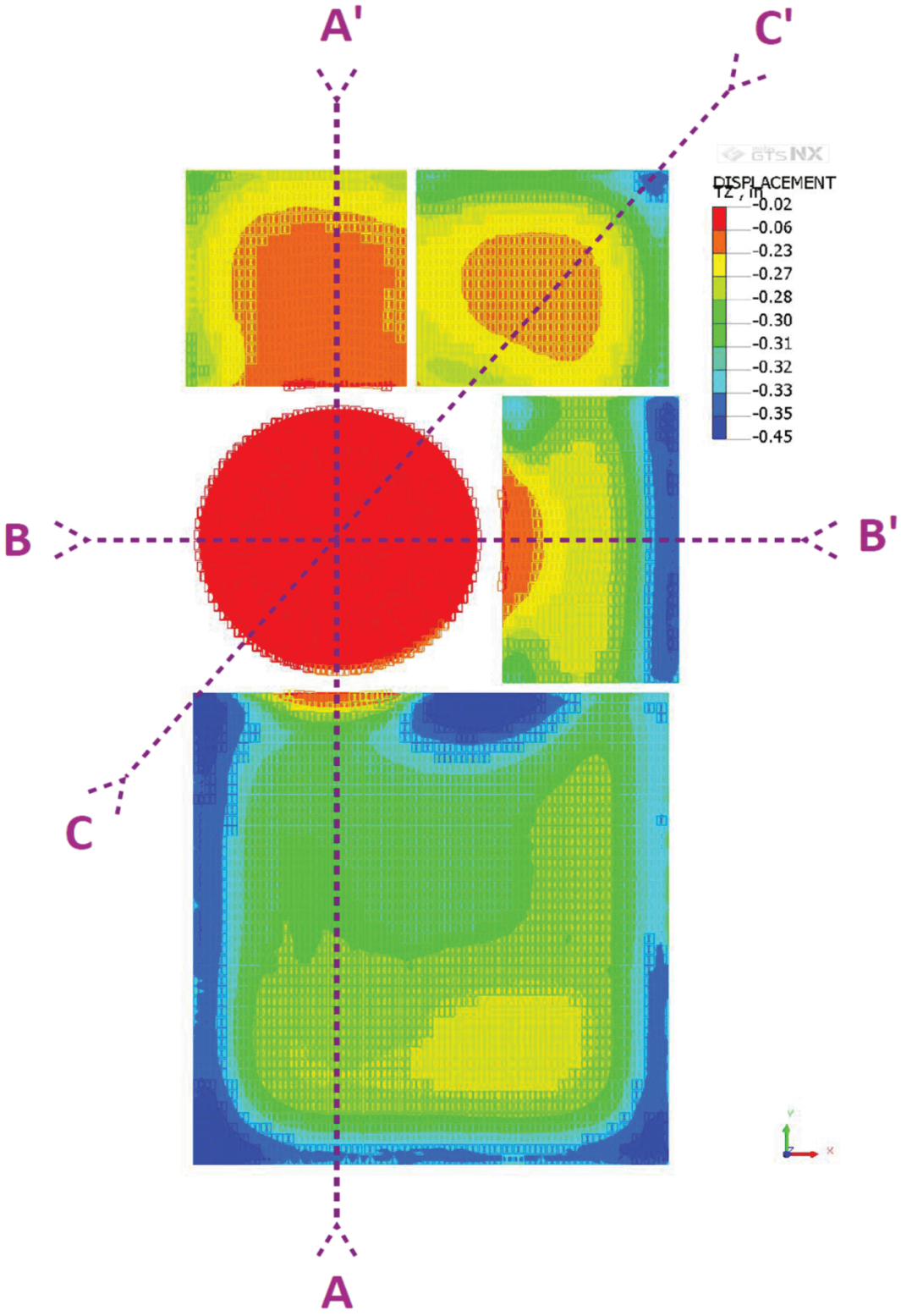


Figure 2.5.4-38 Expected Settlement Distribution Beneath all Buildings Mat Foundations (Mats With Caissons)

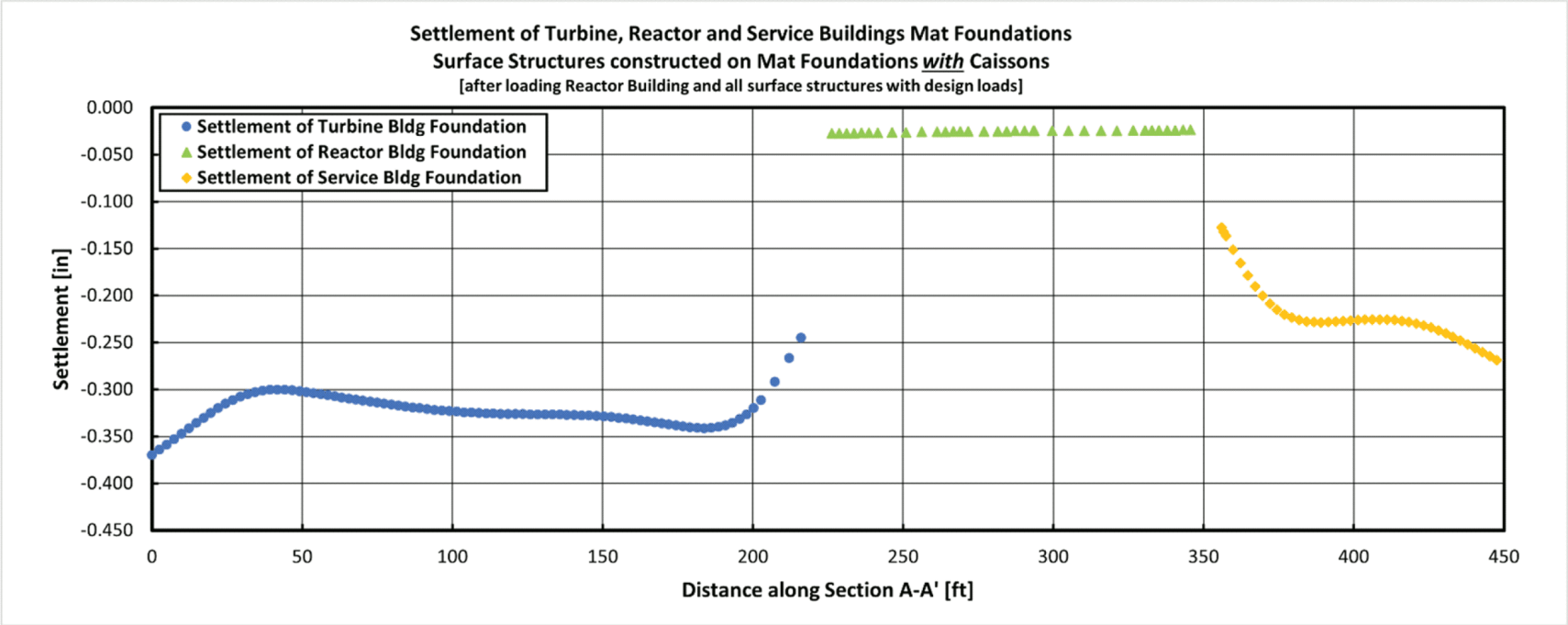


Figure 2.5.4-39 Expected Settlement Distribution along Cross Section A-A' (Mats With Caissons)

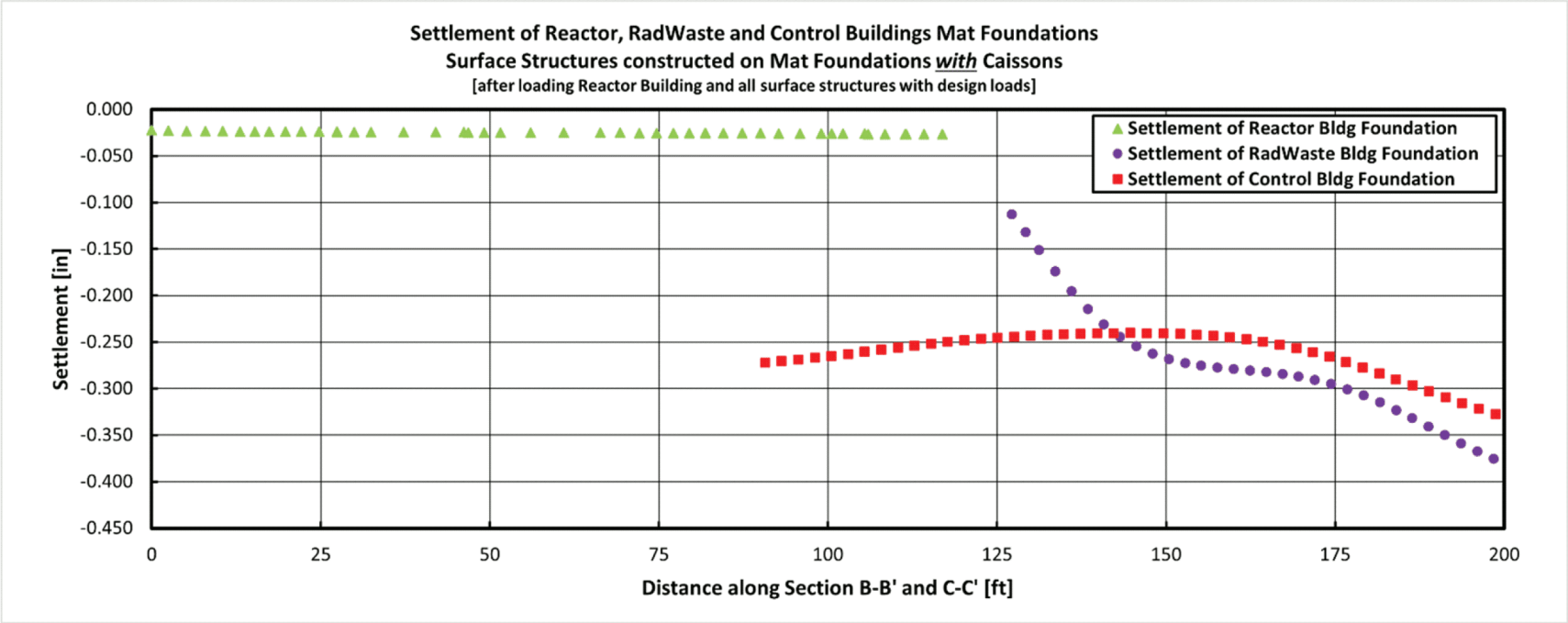


Figure 2.5.4-40 Expected Settlement Distribution along Cross Sections B-B’ and C-C’ (Mats With Caissons)

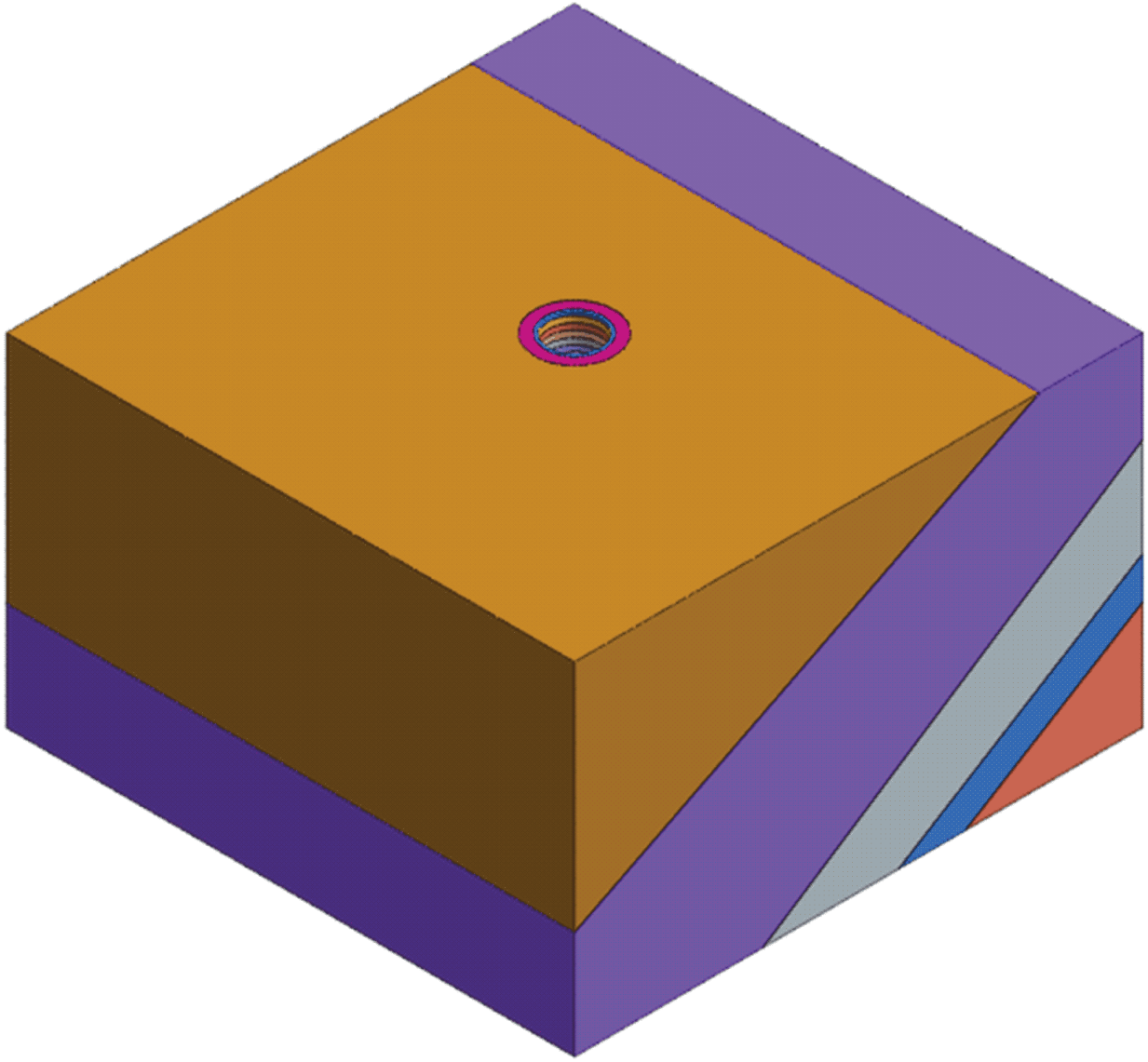


Figure 2.5.4-41 3D Finite Element Model up to El. 780 ft (Subsurface Soils and Weathered Rock hidden)

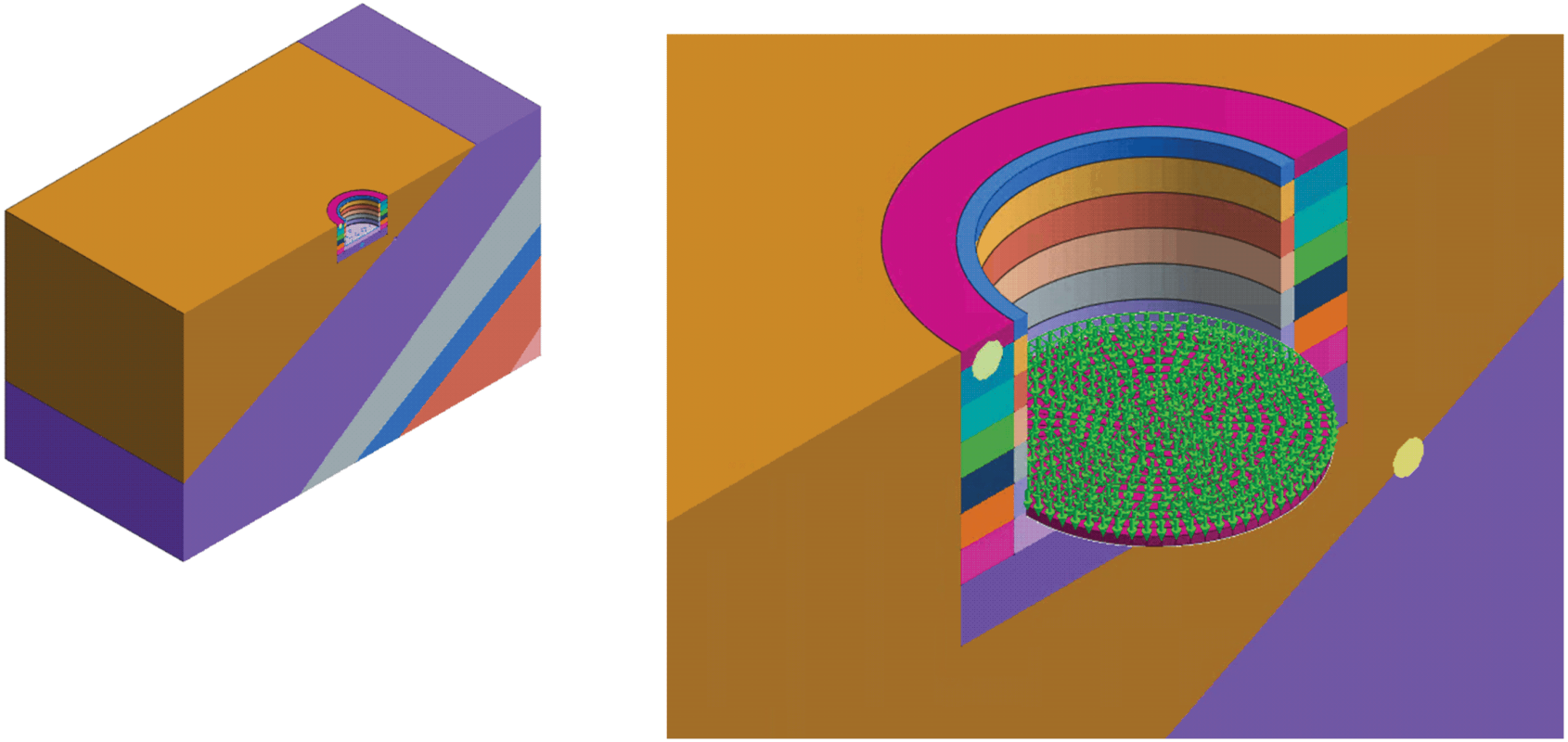


Figure 2.5.4-42 Section Cut and Magnified view through the Reactor Building up to El. 780 ft (Subsurface Soils and Weathered Rock hidden)

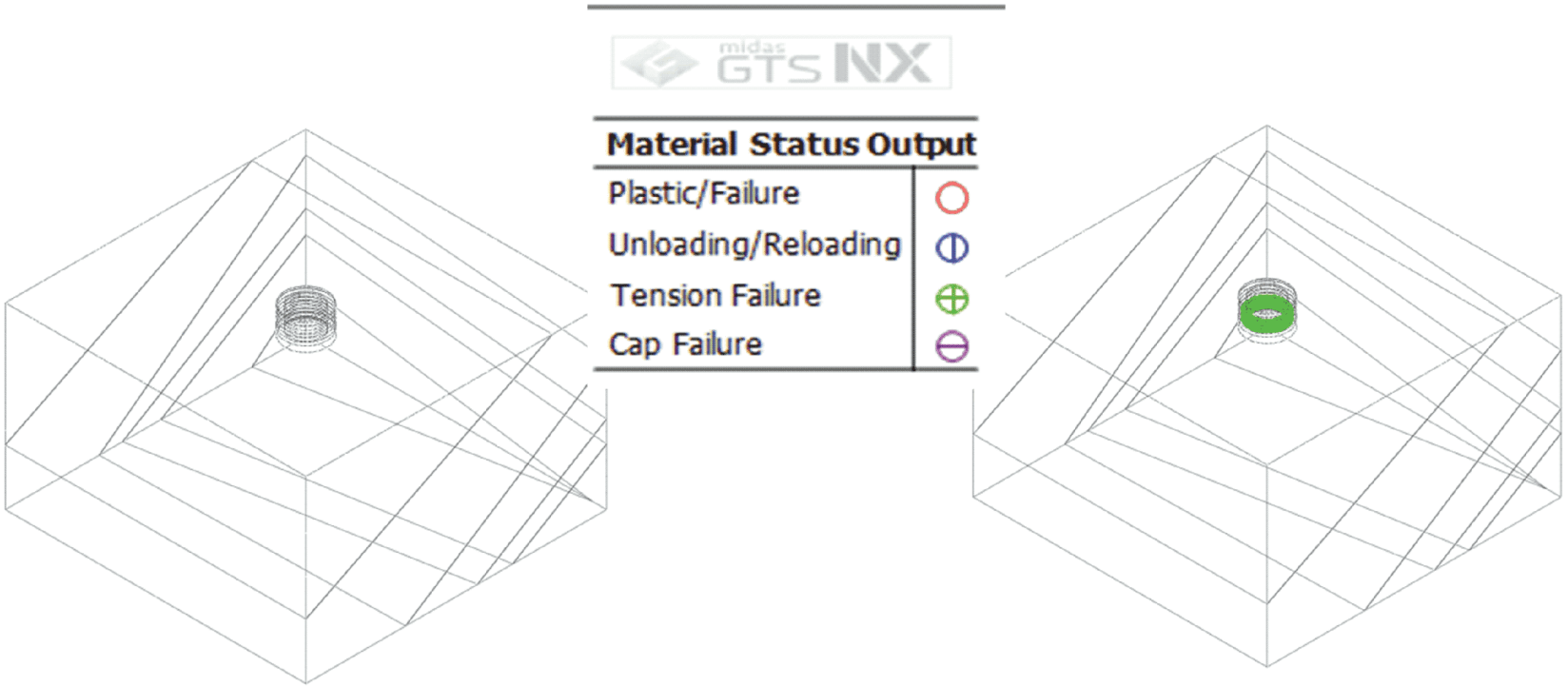


Figure 2.5.4-43 Plastic Point Zones due to Amplified RB's Service Load

Left: Service Load 25.2 ksf (x3).
Right: Service Load 50.4 ksf (x6).

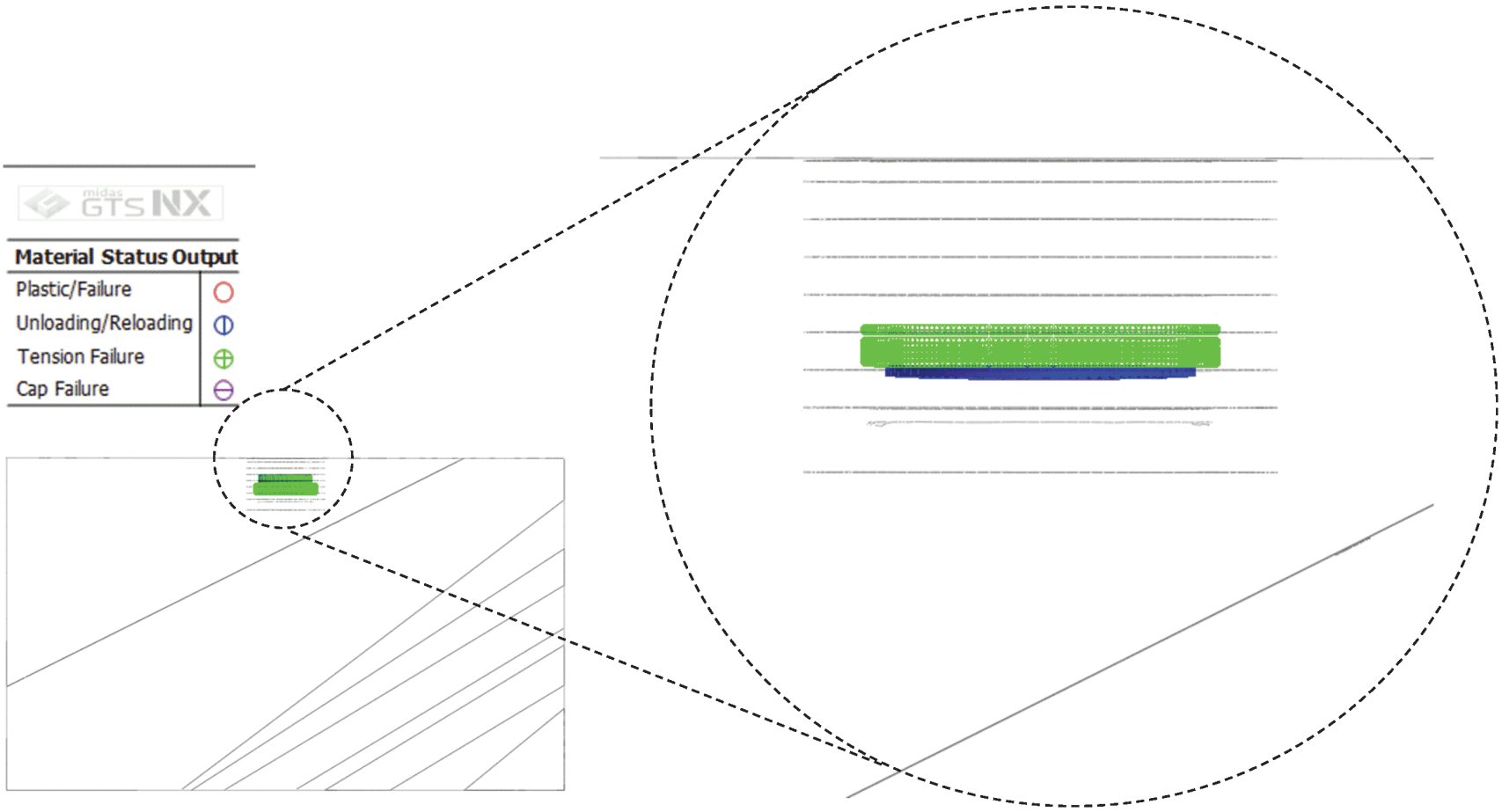
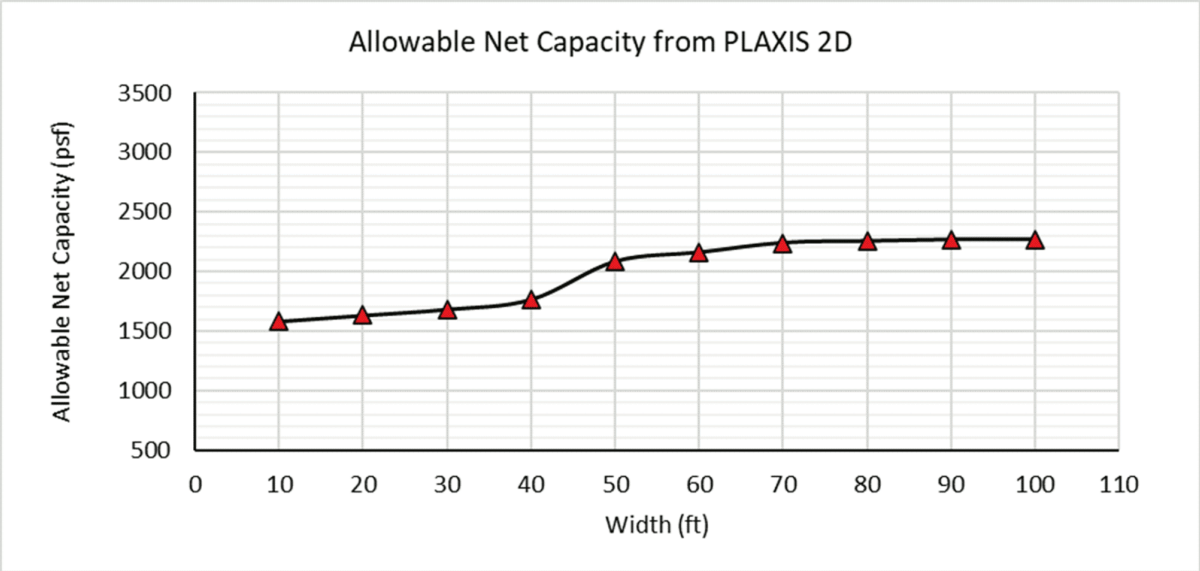


Figure 2.5.4-44 Plastic Point Zones Generated by a Service Load at the RB Foundation of 50.4 ksf. Loads of the Surface Founded Structures at Service Values. The Blue Region on the Right Depicts the RB's Mat Foundation.



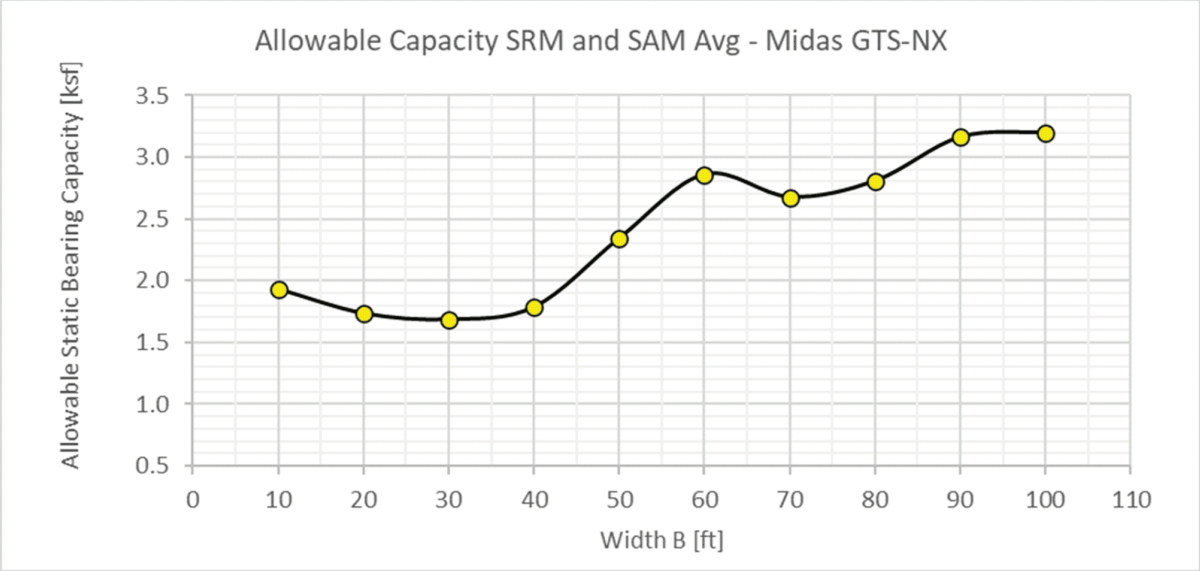


Figure 2.5.4-45 Allowable Bearing Capacity Of Surface Founded Structures. Top: Load Control Results From PLAXIS 2D. Bottom: SAM And SRM Results From Midas GTSNX.

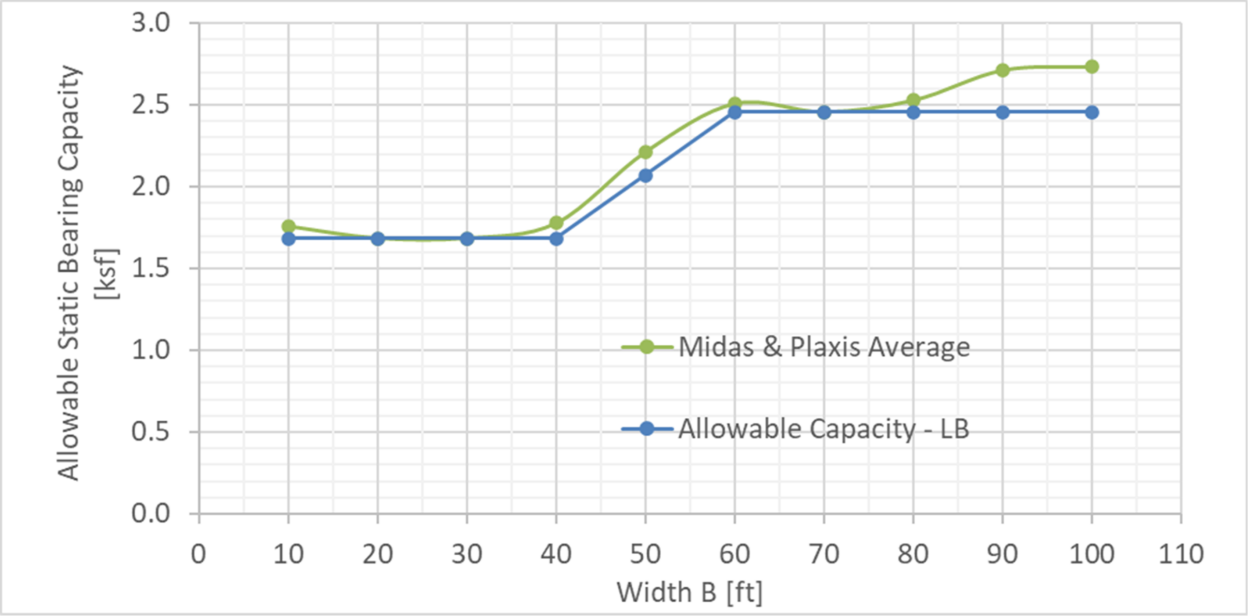
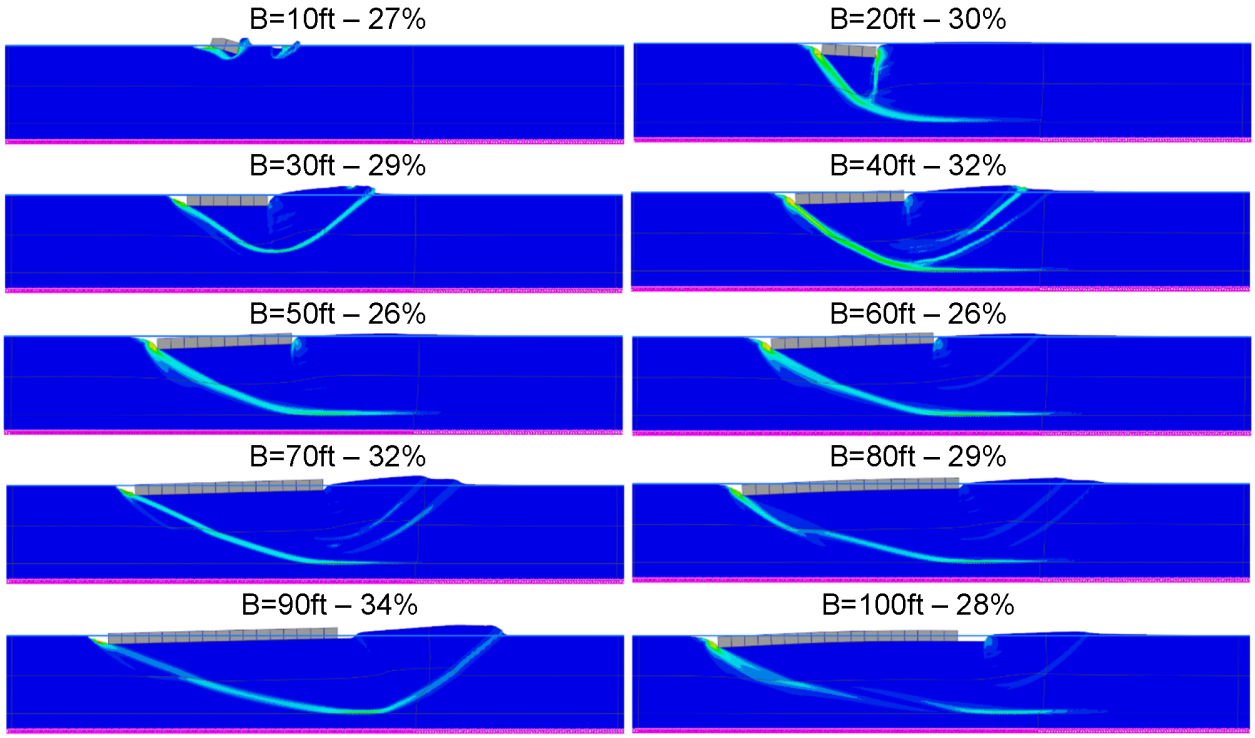


Figure 2.5.4-46 Foundations of Surface Founded Structures Allowable Bearing Capacity.



All Foundations Subjected To A Target Vertical Load Of 10 ksf And A Horizontal Seismic Coefficient Of Kh=0.3. Percentages On The Top Represent The Vertical Load Ratio At Failure.

Figure 2.5.4-47 Foundations Of Surface Founded Structures. Plastic Strain Distribution At Maximum Pseudo-Static Capacity

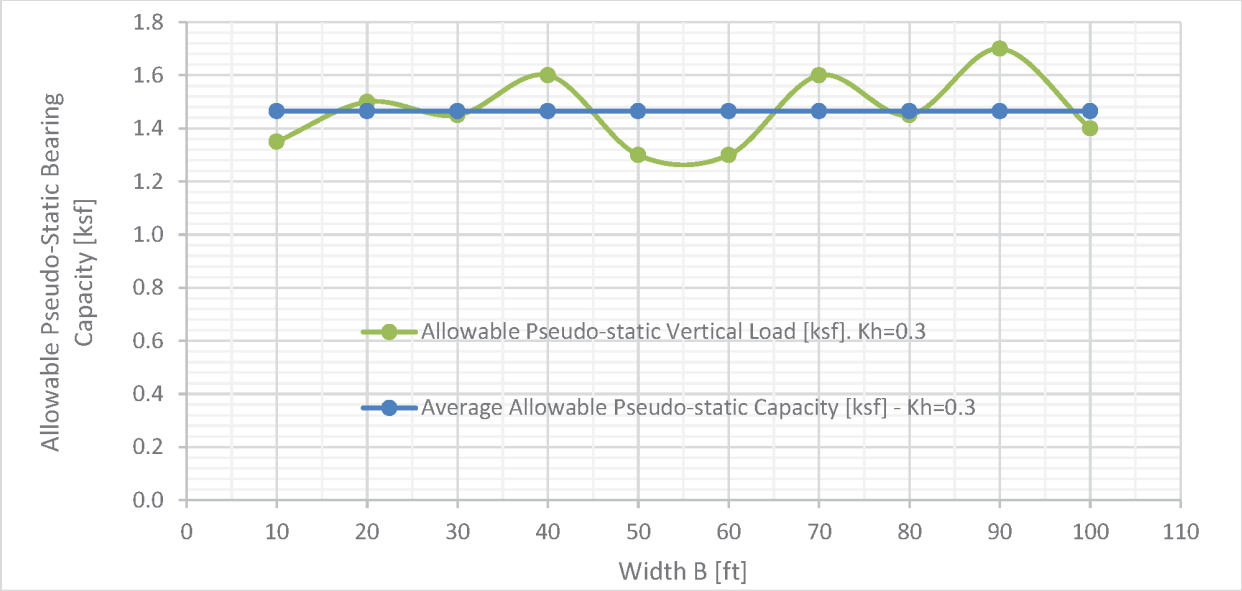


Figure 2.5.4-48 Surface Founded Structures Allowable Pseudo-Static Bearing Capacity

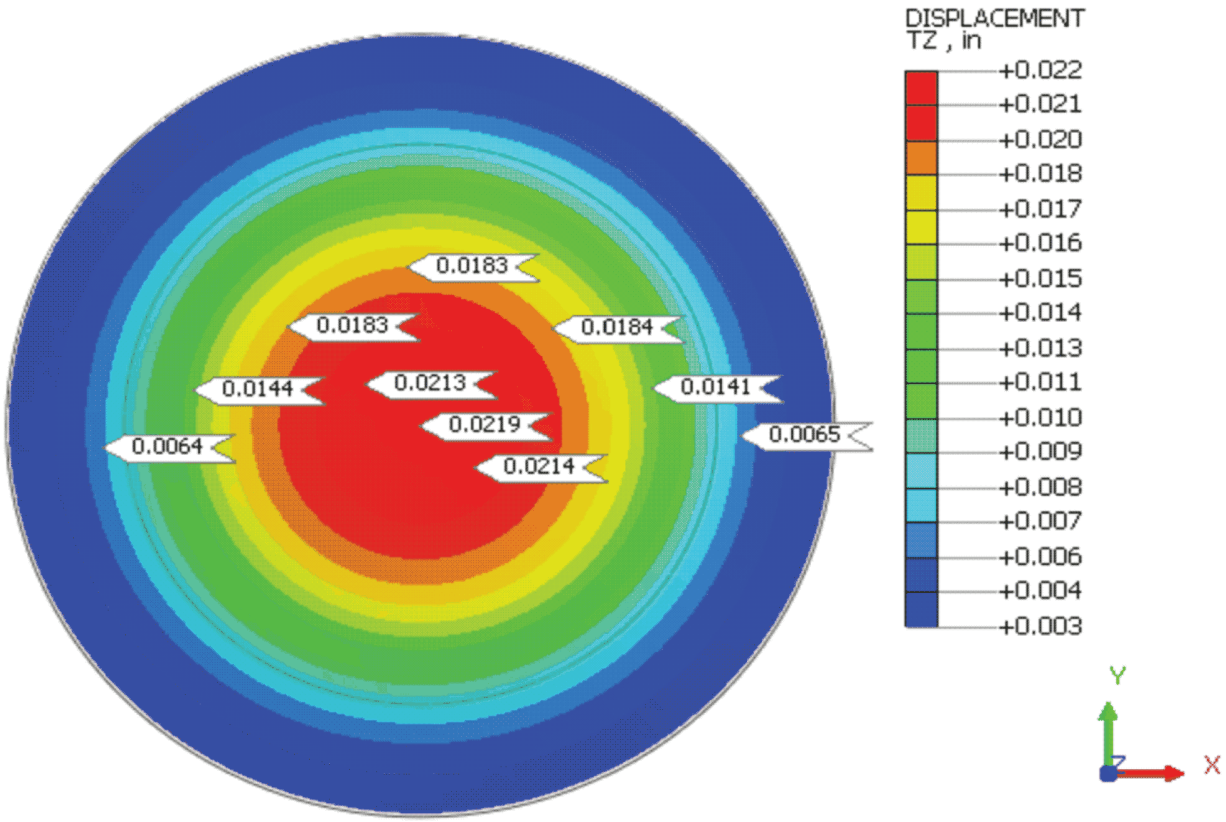


Figure 2.5.4-49 Total Estimated Heave Distribution Beneath the Reactor Building Foundation Mud-Mat from the Foundation Interface Analysis

2.5.5 Stability of Slopes *(Addresses ESP COL Action Item 2.5-17)*

Replace ESPA SSAR Section 2.5.5 with the following:

The information presented in this subsection is based on the results of the site-specific subsurface investigation performed at the Clinch River Nuclear Site, which is described in Subsection 2.5.4. Note that the detailed data collected from this 2013 and the 2023 investigations are contained in Reference 2.5.5-1, Reference 2.5.5-4, and Reference 2.5.5-5, respectively.

The Clinch River Nuclear Site is located along the Clinch River arm of the Watts Bar Reservoir within the City of Oak Ridge, Tennessee. The topography in this area consists of a series of northeast to southwest trending alternating ridges and valleys. During previous site grading at the Clinch River Property, as described in Subsection 2.5.4.1, portions of the ridges were excavated and some of the valleys were filled resulting in a gently sloping topography within the power block area. To the west of the power block area, an excavation approximately 35 ft deep remains from the previous site grading. Several ridges are located east of the power block area.

As described in Subsection 2.5.4.1, the existing ground surface at and around the power block area at the time of the 2013 CRN Site investigation program is generally gently sloping downward from northwest to southeast, ranging from approximately Elevation 855 to 780 ft North American Vertical Datum of 1988 (NAVD88) (Reference 2.5.5-1). Note that all references to elevations given in this subsection are to NAVD88. The power block finish grade elevation is at 814.5 ft.

Details of the proposed site grading are developed for the BWRX-300 technology. The planned finished grade elevation is at elevation of 814.5 ft, and a flat table-top site with no SC1 slopes is proposed (Reference 2.5.5-2).

2.5.5.1 Slope Characteristics

The planned finished grade elevation of the power block area is set at 814.5 ft. The power block configuration and site grading do not include any permanent slopes.

During construction, a temporary excavation cut slope will be made in the power block area to construct foundations. This excavation will be made in soil, weathered rock, and hard rock. The excavation will include vertical faces for the Reactor Building shaft. The excavation will be backfilled, and no permanent slopes will remain within close proximity (the shortest distance between permanent slope and SC1 structure is greater than 900 ft, Reference 2.5.5-3) to the SC1 structure after construction in the power block area.

Groundwater conditions in the power block area, including seepage and high and low groundwater levels, are discussed in Subsection 2.4.12.

2.5.5.2 Design Criteria and Analyses

No permanent SC1 slopes are identified. Construction excavation cut slopes will be required in the power block area for construction of the foundations. Soil and rock will be removed and replaced with compacted granular backfill and lean concrete, respectively. The final design of the excavation, including excavation support and slope stabilization details, is developed in the OLA. The construction excavation cut slopes will be temporary since they are for the construction period only. The excavation will be backfilled and none of these slopes will remain after construction in the power block area (Reference 2.5.5-3).

2.5.5.3 Results of the Investigation

As described in Subsection 2.5.4, a detailed subsurface investigation encompassing the power block area was conducted and included material characterization, foundation analysis, and rock mass characterization. These data will be used for design of the foundation excavation cut slopes.

2.5.5.4 Properties of Borrow Material

Information related to backfill and borrow material for SC1 backfill is provided in Subsection 2.5.4.5.

2.5.5.5 References

2.5.5-1 AMEC (AMEC Environmental and Infrastructure, Inc.), Geotechnical Exploration and Testing, Clinch River SMR Project, Oak Ridge, Tennessee, Data Report, Rev. 4, October 2014.

2.5.5-2 Black and Veatch, “Drawing Title: Grading & Drainage – Site Plan – Overall Unit 1, ID: 410214-DS-3001A, Rev. 0,” March 24, 2023.

2.5.5-3 Black and Veatch, “Drawing Title: Grading & Drainage – Site Grading Profiles, ID: 410214-DS-3950, Rev. 0,” March 24, 2023.

2.5.5-4 RIZZO, “Supplemental Geotechnical Field Investigation, Data Analysis Report, Clinch River Nuclear Site,” RIZZO International Inc., Pittsburgh, Pennsylvania, Revision 0, December 20, 2024.

2.5.5-5 S&ME, “Data Report, CRN Supplemental Geotechnical Investigation, Oak Ridge, Tennessee Document Control No. CRN-23-0140-A, Revision C,” March 25, 2024.

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LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
1D	One-Dimensional
2D	Two-Dimensional
3D	Three-Dimensional
AC	Alternating Current
ACI	American Concrete Institute
AISC	American Institute of Steel Construction
ALWR	Advanced Light Water Reactor
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
ARI	Alternate Rod Insertion
ARS	Acceleration Response Spectra
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASTM	ASTM International
ATF	Acceleration Transfer Function
ATH	Acceleration Time History
ATWS	Anticipated Transient Without Scram
BE	Best Estimate
BIS	Boron Injection System
BL-DSA	Baseline Deterministic Safety Analysis
BPVC	Boiler and Pressure Vessel Code
BTP	Branch Technical Position
BWR	Boiling Water Reactor
CB	Control Building
CCDP	Conditional Core Damage Probability
CCF	Common Cause Failure
CDF	Core Damage Frequency
CEPSS	Containment Equipment and Piping Support Structure
CFR	Code of Federal Regulations
CFS	Condensate and Feedwater Heating System
CIS	Containment Inerting System
CMon	Containment Monitoring Subsystem
CIV	Containment Isolation Valve
CMTR	Certified Material Test Report
CN-DSA	Conservative Deterministic Safety Analysis
CRD	Control Rod Drive

LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
CRDA	Control Rod Drop Accident
CRDH	Control Rod Drive Housing
CRN	Clinch River Nuclear
CUF	Cumulative Usage Factor
CUW	Reactor Water Cleanup
CWE	Chilled Water Equipment
CWS	Circulating Water System
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
DL3	Defense Line 3
DL4a	Defense Line 4a
DLF	Dynamic Load Factor
DP-SC	Diaphragm Plate Steel-Plate Composite
DSA	Deterministic Safety Analysis
ECCS	Emergency Core Cooling System
EFS	Equipment and Floor Drain System
EMC	Electromagnetic Compatibility
EMI/RFI	Electromagnetic Interference/Radio Frequency Interference
EPRI	Electric Power Research Institute
EQ	Environmental Qualification
FCU	Fan Coil Unit
FE	Finite Element
FIA	Foundation Interface Analysis
FIRS	Foundation Input Response Spectra
FIV	Flow Induced Vibration
FMCRD	Fine Motion Control Rod Drive
FPC	Fuel Pool Cooling and Cleanup System
FPS	Fire Protection System
FSAR	Final Safety Analysis Report
FSF	Fundamental Safety Function
FW	Feedwater
GDC	General Design Criteria

LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
GNF	Global Nuclear Fuels
GVH	GE Vernova Hitachi Nuclear Energy
HCU	Hydraulic Control Unit
HELB	High Energy Line Break
HRA	Human Reliability Analysis
HVS	Heating, Ventilation, and Cooling System
HWC	Hydrogen Water Chemistry
HX	Heat Exchanger
I&C	Instrumentation and Control
IBC	International Building Code
IC	Isolation Condenser
ICC	Isolation Condenser Pools Cooling and Cleanup System
ICS	Isolation Condenser System
IE	Initiating Event
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronics Engineers
ILRT	Integrated Leak Rate Test
ISI	Inservice Inspection
ISRS	In-Structure Response Spectra
IST	Inservice Testing
L&C	Limitation and Condition
LB	Lower Bound
LMS	Lumped Mass Stick
LMSM	Lumped Mass Stick Model
LOCA	Loss-of-Coolant Accident
LV	Low Voltage
LWM	Liquid Waste Management System
MAAP	Modular Accident Analysis Program
MCA	Main Condenser and Auxiliaries
MCC	Motor Control Center
MCR	Main Control Room
MS	Main Steam
MSCIV	Main Steam Containment Isolation Valve
MSL	Main Steam Line
MSRIV	Main Steam Reactor Isolation Valve
MTE	Main Turbine Equipment
NBS	Nuclear Boiler System

LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
NDE	Non-Destructive Examination
NDTT	Nil-Ductility Transition Temperature
NEI	Nuclear Energy Institute
NHS	Normal Heat Sink
NPS	Nominal Pipe Size
NRIC	National Reactor Innovation Center
NUREG	Nuclear Regulatory Report
OBE	Operating Basis Earthquake
OLNC	On-Line NobleChem™
OM	Operations & Maintenance
OPEX	Operating Experience
PAM	Post-Accident Monitoring
PBIRS	Performance Based Intermediate Response Spectra
PBSRS	Performance Based Surface Response Spectra
PCCS	Passive Containment Cooling System
PCS	Primary Containment System
PCW	Plant Cooling Water System
PDC	Principal Design Criteria
PIE	Postulated Initiating Event
PPS	Plant Pneumatics System
PRA	Probabilistic Risk Assessment
PREMS	Process Radiation and Environmental Monitoring System
PSA	Probabilistic Safety Assessment
PSAR	Preliminary Safety Analysis Report
PSHA	Probabilistic Seismic Hazard Assessment
QA	Quality Assurance
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
RRS	Required Response Spectra
RWB	Radwaste Building
RWST	Refueling Water Storage Tank
SAFDL	Specified Acceptable Fuel Design Limits

LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
SBLOCA	Small Break Loss-of-Coolant Accident
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
SEI	Structural Engineering Institute
SIR	Seismic Interface Restraint
SIT	Structural Integrity Test
SRA	Site Response Analysis
SRP	Standard Review Plan
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
SW	Service Water
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
TSV	Turbine Stop Valve
UB	Upper Bound
UHS	Ultimate Heat Sink
UPS	Uninterruptible Power Supply
USNRC	U.S. Nuclear Regulatory Commission
V&V	Verification and Validation
V/H	Vertical to Horizontal
WRNM	Wide Range Neutron Monitor
ZPA	Zero Period Acceleration

LIST OF EFFECTIVE SECTIONS

CHAPTER 3 DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

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CHAPTER 3 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 (AOO) and Design Basis Accident (DBA) Postulated Initiating Events (PIE). 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) Structures, Systems, and Components (SSCs), and their Safety Category 1 functions, are subject to the GVH quality assurance 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 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.

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 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 or assure adequate protection in the event of a fire:

- 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 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 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

CRN-1 is a single unit and does not share SSCs with other nuclear power units.

Accordingly, the requirements of 10 CFR 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 (OPEX), 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. 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, OPEX, 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 BPVC requirements 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 to 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 modelled in evaluation of SAFDLs.

Accordingly, the requirements of 10 CFR 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 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 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 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 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 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 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, OPEX, and qualification test results.

Accordingly, the requirements of 10 CFR 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 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 Section 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 Distributed Control and Information System (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 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 U.S. 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 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. Design features, such as self-testing routines in digital equipment, that permit inservice testing 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 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 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 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 in-core 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 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 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 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, and
- Vulnerabilities to single equipment failures which could cause inadvertent movement of a single control rod are minimized.

Accordingly, the requirements of 10 CFR 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 System Redundancy and 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 Control Rod Drive (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 Fine Motion Control Rod Drives (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 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. The BIS is a complementary design feature used to mitigate the

consequences of Design Extension Conditions (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 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 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.

Pursuant to 10 CFR 50.12 an Exemption to 10 CFR 50, Appendix A, GDC 28 is requested from the GDC 28 requirement to postulate a CRDA and control rod ejection as a Design Basis Accident on the basis that multiple positive means exist to prevent the prerequisite conditions. The exemption request is provided in Enclosure 4.

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 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 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 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 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 inservice inspection. 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 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 Technical Specifications (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 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 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 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 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 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 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 per ASME BPVC, Section XI, Subsection IWC, as applicable.

Accordingly, the requirements of 10 CFR 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 Inservice Inspection (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 inservice inspection 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 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 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 inservice inspection during all normal plant states to assure integrity and functionality of components.

Accordingly, the requirements of 10 CFR 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 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 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 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 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 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 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 ANSI/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 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 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 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 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 inservice inspection, 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 inservice testing 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 Nuclear Boiler System (NBS), Reactor Water Cleanup System (CUW), and Condensate and Feedwater Heating System (CFS) have inboard integral RIVs and outboard CIVs. The supply lines for Shutdown Cooling System (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 Boron Injection System (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 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 Containment Monitoring Subsystem (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 50, Appendix A, GDC 56 are met.

Additional information discussing the compliance approach to 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 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 and the ventilation systems. Noble gases and radioiodine activity that enters the Offgas System are held-up by activated carbon charcoal adsorber beds, allowing for the decay of these gases. Ventilation releases are through the Plant Vent Stack located in the Turbine Building. 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 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.

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

Accordingly, the requirements of 10 CFR 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 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 System, 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 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 Containment Monitoring Subsystem, 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 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
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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
II	13	Instrumentation and Control	15.2.4.3	Reactivity and Power Distribution Anomalies Bounding Event Selection	FSAR

Table 3.1-1 Compliance with General Design Criteria/Principal Design Criteria
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GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
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 Code 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 Inservice Testing 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

Table 3.1-1 Compliance with General Design Criteria/Principal Design Criteria
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GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
II	14	Reactor Coolant Pressure Boundary	5.2.3	Reactor Coolant Pressure Boundary Materials	Complies
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 Demineralizer 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

Table 3.1-1 Compliance with General Design Criteria/Principal Design Criteria
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GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
II	15	Reactor Coolant System Design	3.9.3	ASME BPVC Class 1, 2, and 3 Components, Component Supports, and Core Support Structures	Complies
II	15	Reactor Coolant System Design	3.9.6	Functional Design, Qualification, and Inservice Testing 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

Table 3.1-1 Compliance with General Design Criteria/Principal Design Criteria
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GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
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
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

Table 3.1-1 Compliance with General Design Criteria/Principal Design Criteria
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GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
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
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

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GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
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
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

Table 3.1-1 Compliance with General Design Criteria/Principal Design Criteria
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GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
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
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

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GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
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
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 Systems	Complies
III	PDC 28	Reactivity Limits	7.1	I&C Systems	Complies

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GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
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 - CRDA is excluded by Exemption Request
III	PDC 29	Protection Against Anticipated Operational Occurrences	3.9.4	Control Rod Drive 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 - CRN-1 complies 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 - CRN-1 complies with PDC 29
IV	30	Quality of Reactor Coolant Pressure Boundary	3.10	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	Complies
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

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GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
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 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 Reactor Coolant Pressure Boundary	LTR 4.1.10(2)	10 CFR 50 Appendix A, GDC 31	Complies
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

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GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
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 50 Appendix A, GDC 33	Complies
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

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GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
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
IV	36	Inspection 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	3.9.6	Functional Design, Qualification, and Inservice Testing 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

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GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
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
IV	39	Inspection of Containment Heat Removal System	6.6	Preservice and Inservice Inspection of Class 2 and 3 Components	Complies

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GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
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

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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	Inspection of Containment Atmosphere Cleanup Systems	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 Containment atmosphere cleanup is not an ESF function
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

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GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
IV	44	Cooling Water	6.2.2	Passive Containment Cooling System	Complies
IV	44	Cooling Water	6.3	Emergency Core Cooling System (Isolation Condenser System)	Complies
IV	44	Cooling Water	7.1	I&C Systems	N/A I&C systems not required 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 Subsystem	N/A No cooling functions provided by Subsystem

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GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
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
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

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GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
IV	45	Inspection of Cooling Water System	9.2.6	Condensate Storage and Transfer Subsystem	N/A No cooling functions provided by Subsystem
IV	45	Inspection 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	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 function not relied upon in safety analysis
IV	46	Testing of Cooling Water System	3.9.6	Functional Design, Qualification, and Inservice Testing 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	Preservice and Inservice Inspection of Class 2 and 3 Components	Complies

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GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
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 Subsystem	N/A No cooling functions provided by Subsystem
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 Vessel	Complies
V	50	Containment Design Basis	3.8.2	Class MC Steel Components of the BWRX-300 Containment	Complies

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GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
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
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	Fracture Prevention of Containment Pressure Boundary	Complies
V	51	Fracture Prevention of Containment Pressure Boundary	LTR 2.1.1.11(5)	10 CFR 50 Appendix A, GDC 51	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 51	Complies
V	53	Provisions for Containment Testing and Inspection	3.8.1	Steel-Plate Composite Containment Vessel	Complies

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GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
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 Inservice Testing 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
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

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GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
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
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

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GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
VI	60	Control of Releases of Radioactive Materials to the Environment	10.4.3	Turbine Gland Sealing System	Complies
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

Table 3.1-1 Compliance with General Design Criteria/Principal Design Criteria
(Sheet 25 of 26)

GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
VI	62	Prevention of Criticality in Fuel Storage and Handling	9.1.4	Light Load Handling System	Complies
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	Primary Containment System	Complies
VI	64	Monitoring Radioactivity Releases	6.8	Containment Monitoring Subsystem	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
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

Table 3.1-1 Compliance with General Design Criteria/Principal Design Criteria
(Sheet 26 of 26)

GDC Group	GDC #	Title	PSAR or LTR Section	Topic	Comments
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) Containment Vessel (SCCV) and Reactor Building 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 deterministic safety analysis (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 probabilistic safety analysis (PSA).

The BWRX-300 Safety Strategy defines the fundamental safety functions (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 anticipated operational occurrence (AOO) PIEs.
- DL3 functions—Detect and mitigate design basis accident (DBA) PIEs and event sequences comprising AOO PIEs and failure of DL2 functions.
- DL4a functions—Detect and mitigate design extension conditions (DEC), including event sequences associated with some DBA PIEs and failure of DL3 functions.
- 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, reactor coolant pressure boundary (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, and summarized using nomenclature selected for the CRN site. 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.

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 10 Code of Federal Regulations (CFR) 50 Appendix A, General Design Criteria (GDC) 2, following the NUREG-0800 Standard Review Plan (SRP) 3.3.1 and 3.3.2. Overall compliance to GDC 2, for wind and tornado is provided in Subsection 3.1.1. The BWRX-300 design meets the GDC 2 requirements by following the applicable guidance provided in 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. Safety Class 1 (SC1) Structures, Systems, and Components (SSCs) are listed in Table 3A-1.

3.3.1 Wind Loadings

As discussed in NUREG-0800, 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 used 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 of 71.5 m/s (160 mph) 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 listed in Table 2.0-1R is from the Early Site Permit Application that was calculated using ASCE/SEI 7-05 and has a recurrence interval of 100 years and remains the design-basis wind speed for the Clinch River site. The basic wind speed used in the design of Seismic Category I structures, (i.e., 71.5 m/s (160 mph)) 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, NUREG-0800, SRP 3.3.1 refers to the ASCE/SEI 7-05 version. The BWRX-300 design uses the ASCE/SEI 7-16 version. A comparison performed between these two ASCE/SEI versions concludes that the ASCE/SEI 7-16 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 are listed in Table 2.0-1R. The tornado wind design parameters bound the hurricane wind design parameters and 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 are designed for the full pressure drop as provided in Table 2.0-1R.

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 NEDO-33914-A, “BWRX-300 Advanced Civil Construction and Design Approach,” Section 6.0 (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, 2017.

3.3-2 ANSI/AISC N690-18, “Specification for Safety-Related Steel Structures for Nuclear Facilities,” American National Standards Institute/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,” GE-Hitachi Nuclear Energy Americas, LLC, Revision 2, June 2022.

3.4 Water Level (Flood) Design

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3.4.1 Internal Flood Protection for On-Site Equipment Failures

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3.4.1.1 Internal Flooding Evaluation Methodology

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3.4.1.2 Internal Flooding Analysis

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

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3.4.3 References

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

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 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 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 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. 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 SRP 3.5.1.1, “Internally Generated Missiles (Outside Containment),” 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 SC1 equipment, components, or structures, whose failure could result in a missile, do not cause failure of SC1 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 with 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, or
3. The plant's capability to prevent accidents which could result in potential offsite exposures comparable to those referred to in 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 1E-07 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 thru 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 is provided in Table 2.0-1R as 130 mph. However, the design hurricane wind speed used for determination of hurricane missile velocities for the BWRX-300 is 200 mph, which bounds the site-specific value.

3.5.1.5 Site Proximity Missiles (Except Aircraft)

As discussed in Section 2.2.3.2, the evaluations performed on potential accidents that could generate missiles (i.e., explosions) do not pose a threat to the CRN Site. Site proximity missiles are therefore dismissed from further consideration.

3.5.1.6 Aircraft Hazards

In accordance with the acceptance criteria in 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 1E-07 per year, a detailed review of aircraft hazards is to be performed. This review determined that the total aircraft impact frequency is 5.13E-05 per year, which is greater than 1E-07 per year.

Therefore, an evaluation against a second criterion (Core Damage Frequency (CDF) less than 1E-08 per year) is performed for large aircraft and small aircraft impact. The initiating event frequency for large aircraft impact is 4.92E-07 per year. Risk CDF due to aircraft accident hazards is estimated using the following equation:

CDF = IE frequency * CCDP

Where:

IE frequency is the Initiating Event frequency,

CCDP is the Conditional Core Damage Probability

The CDF for the large aircraft impact is 6.4E-11 per year which is below 1.0E-08 per year.

For the small aircraft impact evaluation, individual CCDPs are developed for the Turbine Building (TB), Control Building (CB), RWB, Service Building, and RB. For all buildings except the RB, the CCDP is calculated assuming all SSCs associated with that building are assumed to fail. The RB is capable of withstanding a large aircraft crash and its associated engine without perforation. Therefore, Safety Class 1 (SC1) components inside the RB would remain intact and maintain their intended functions following the impact of a large commercial aircraft, which is bounding for the small aircraft impact scenario. The highest CDF for the small aircraft impact is for the CB with a CDF of 1.27E-09 per year, which is below the screening criterion of 1.0E-08 per year. Therefore, no further analysis for small aircraft impact is required for the BWRX-300 design.

A summary description of the large aircraft impact assessment required per 10 CFR 50.150 is provided in Appendix 3N.

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 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 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 349-13, "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 349-13 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 3.5-9) 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 3.5-7): An acceptable procedure for such an analysis, where the impact is assumed to be plastic, is presented in Reference 3.5-7. In this reference, the deformability of the missile is not considered.

    b. BC-TOP-9A: 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, Clause F.2.1 of ACI 349 (Reference 3.5-3) and Clause Appendix N9.1.6a of American National Standards Institute/American Institute of Steel Construction (ANSI/AISC) N690-18, "Specification for Safety-Related Steel Structures for Nuclear Facilities," (Reference 3.5-8).

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 3.5-7) and BC-TOP-9A, Section 3.5, "Structural Response by Energy Balance Method." 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, including the modifications required by Limitation and Condition 8.7.

3.5.4 References

3.5-1 ANSI/ANS-2.3, "Estimating Tornado, Hurricane, and Extreme Straight Line Wind Characteristics at Nuclear Facility Sites," American National Standards Institute/ American Nuclear Society, 2011 (R2016).

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, “Design of Structures for Missile Impact,” Bechtel Power Corporation, Revision 2, September 1974.

3.5-7 Williamson, R.A. and Alvy, R.R, “Impact Effect of Fragments Striking Structural Elements,” Holmes and Narver, 1973.

3.5-8 ANSI/AISC N690-18, “Specification for Safety-Related Steel Structures for Nuclear Facilities,” American National Standards Institute/American Institute of Steel Construction, 2018.

3.5-9 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.

3.5-10 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
(Sheet 1 of 2)

Building	Reactor Building, Main Control Room, and Main Control Room to Secondary Control Room Egress[3][4]		Radwaste Building[4]
	Tornado Missile Spectrum [1]	Hurricane Missile Spectrum [1]	Tornado Missile Spectrum
Missile Velocity:			
Horizontal Missile Velocity, V_h	41 m/s (135 ft/s) for Automobile and Pipe, 8 m/s (26 ft/s) for Solid Sphere	V_h per Table 2 of RG 1.221 by linear interpolation for Design Hurricane Wind speed 89.4 m/s (200 mph) [5] 56 m/s (183 ft/s) for Automobile 43.5 m/s (142 ft/s) for Pipe 38 m/s (125 ft/s) for Sphere	V_h per Table 2 of RG 1.143 for Design Basis Tornado Wind speed 76 m/s (170 mph) [6] 30 m/s (100 ft/s) for Automobile and Pipe
Vertical Missile Velocity, V_y	27.5 m/s (90.5 ft/s) for Automobile and Pipe, 5.4 m/s (17.6 ft/s) for Solid Sphere	26 m/s (85.3 ft/s) for all missiles (Per note Table 2 of RG 1.221)	V_y per Table 2 of RG 1.143 for Design Basis Tornado Wind speed 76 m/s (170 mph) [6] 20 m/s (67 ft/s) for Automobile & Pipe
Missile Characteristics:			
High Kinetic Energy Missile	Automobile 1810 kg (4000 lb) 5m x 2m x 1.3m (16.4 ft x 6.6 ft x 4.3 ft) C_DA/m [7] = 0.0070 m^2/kg (0.0343 ft^2/lb)	Automobile 1810 kg (4000 lb) 5m x 2m x 1.3m (16.4 ft x 6.6 ft x 4.3 ft)	Automobile 1810 kg (4000 lb) Frontal Area = 1.86 m^2 (20 ft^2)
Rigid Missile to Test Penetration Resistance	Schedule 40 Pipe 130 kg (287 lb) 0.168 m diameter x 4.58 m long (6.625 in diameter x 15 ft long), C_DA/m [7] = 0.0043 m^2/kg (0.0212 ft^2/lb)	Schedule 40 Pipe 130 kg (287 lb) 0.168 m diameter x 4.58 m long (6.625 in diameter x 15 ft long)	Schedule 40 Pipe [8] 130 kg (287 lb) 0.168 m diameter x 4.58m long (6.625 in diameter x 15 ft long)

Table 3.5-1 Design Extreme Wind-Generated Missile Spectrum
(Sheet 2 of 2)

Building	Reactor Building, Main Control Room, and Main Control Room to Secondary Control Room Egress[3][4]		Radwaste Building[4]
	Tornado Missile Spectrum [1]	Hurricane Missile Spectrum [1]	Tornado Missile Spectrum
Small Rigid Missile to Test Opening Size	Solid Steel Sphere 0.0669 kg (0.147 lb) 2.54 cm (1 in) diameter C_DA/m [7] = 0.0034 m^2/kg (0.0166 ft^2/lb)	Solid Steel Sphere 0.0669 kg (0.147 lb) 2.54 cm (1 in) diameter	Not Applicable [9]
Applicable Height of Missile	Automobile: Up to 9.14 m (30 ft) above grade within 0.5 mile (0.8 kilometer) of the plant structures. Pipe and Sphere: Full height of building	Automobile: Up to 9.14 m (30 ft) above grade within 0.5 mile (0.8 kilometer) of the plant structures. Pipe and Sphere: Full height of building	Automobile: Up to 9.14 m (30 ft) above grade. Pipe: Full height of building
Applicable RG	RG 1.76[2]	RG 1.221	RG 1.143 and ANSI-2.3 [6]

Notes:

(1) Design for missile impact uses the bounding missile speeds (tornado or hurricane).

(2) Values are for Region I from RG 1.76.

(3) Only for portions requiring storm hardening (at a minimum the MCR and MCR to SCR egress route)

(4) Design basis wind speeds are nominal 3-second gust in (m/s) mph at 33 ft (10m) above ground over open terrain.

(5) The design hurricane wind speed used for determination of hurricane missile velocities for the BWRX-300 is 89.4 m/s (200 mph), which bounds the site-specific value of 58.12 m/s (130 mph).

(6) In accordance with Table 2 of RG 1.143, Design Basis tornado wind speed is based on Table 3 of ANSI/ANS-2.3 at a probability of 10^{-5}/yr consistent with RG 1.76. Horizontal and vertical velocity coefficients are based on Table 4 of ANSI/ANS-2.3. In accordance with Table 3 of ANSI/ANS-2.3 hurricanes do not occur at a probability of $1e^{-5}$/yr or less for all regions.

(7) C_DA/m missile flight parameter including drag coefficient (C_D) x projected frontal area of missile (A)/mass of missile (m).

(8) Schedule 40 pipe missile per Table 4 of ANSI/ANS-2.3 and consistent with Table 2 of RG 1.76.

(9) In accordance with Table 2 of RG 1.143, the design-basis missiles for the Radwaste building do not include the small rigid missile.

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
(Sheet 1 of 3)

Barrier Material	Limit State	Permissible Ductility Ratio (μ_p)	Reference
Steel Structures Barriers	Tension(1)	$\leq \min(0.25\varepsilon_u/\varepsilon_y, 0.1/\varepsilon_y)$	ANSI/AISC N690 Table NB3.1 (Reference 3.5-8)
	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 $F_e \geq 4.5\ F_y$	$\leq \min\left(\frac{0.225}{F_y/F_e}, \frac{\varepsilon_{st}}{\varepsilon_y}, 10\right)$	
Reinforced Concrete Structures	Flexure(3)-Steel beams, walls, and slabs	$\min\left(\frac{0.05}{\rho - \rho'}, 10\right)$ 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	

Table 3.5-4 Permissible Ductility Ratios-Steel and Reinforced Concrete
(Sheet 2 of 3)

Barrier Material	Limit State	Permissible Ductility Ratio (μ_p)	Reference
Reinforced Concrete Structures (continued)	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)
	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	$\min\left(\frac{0.05}{\rho-\rho'}\ 10\right)$ 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)

Table 3.5-4 Permissible Ductility Ratios-Steel and Reinforced Concrete
(Sheet 3 of 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] (Sheet 1 of 2)

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]
Steel-Plate Composite 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
Steel-Plate Composite 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

Table 3.5-5a Structural Acceptance Criteria for Flexure in Terms of Support Rotations and for Shear in Terms of Ductility[10][11] (Sheet 2 of 2)

Notes:

(1) Steel-Plate Composite 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 and ductility, 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) One (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 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 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:

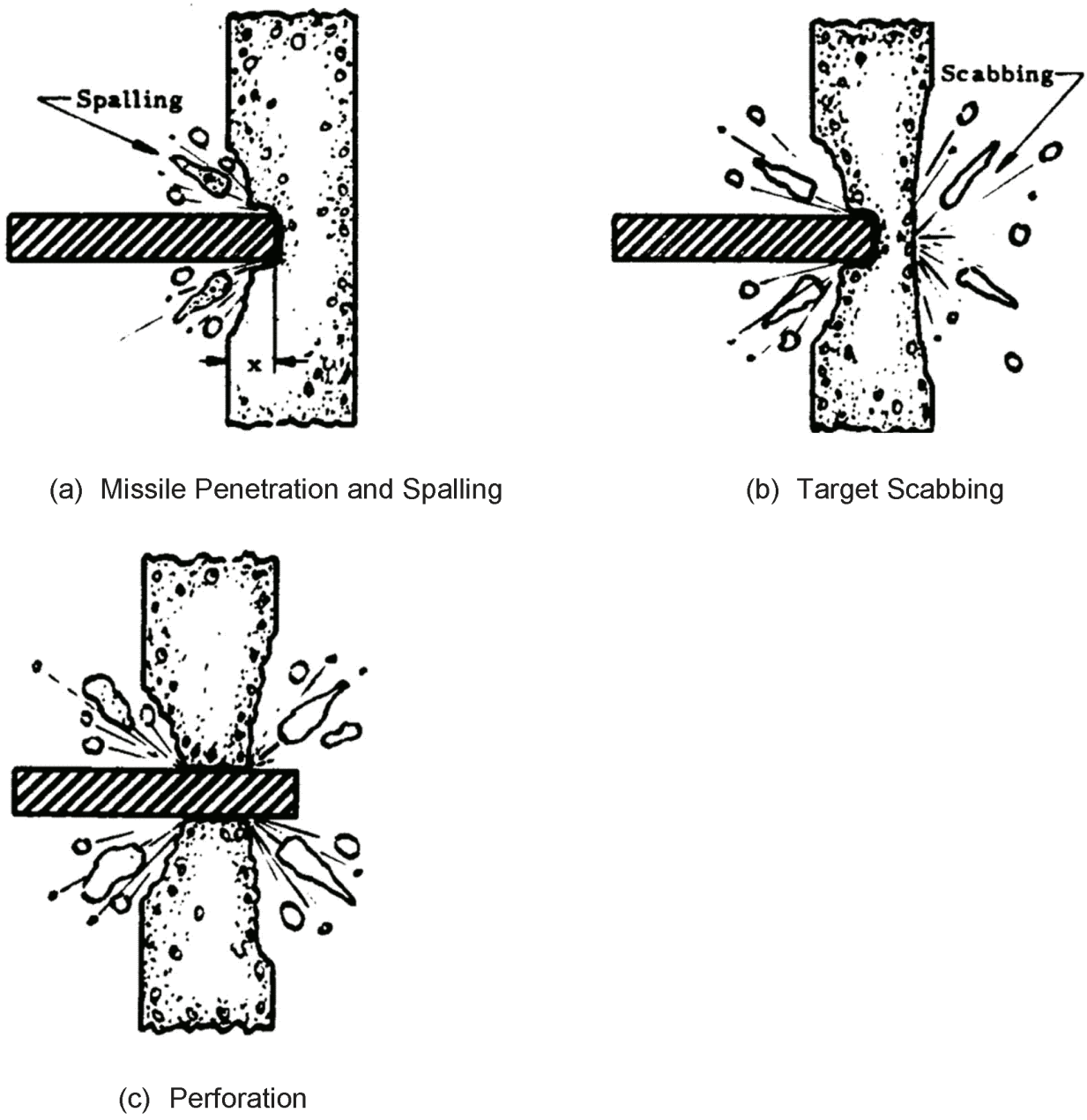
(1) The tri-axiality factor, TF, is defined as

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

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

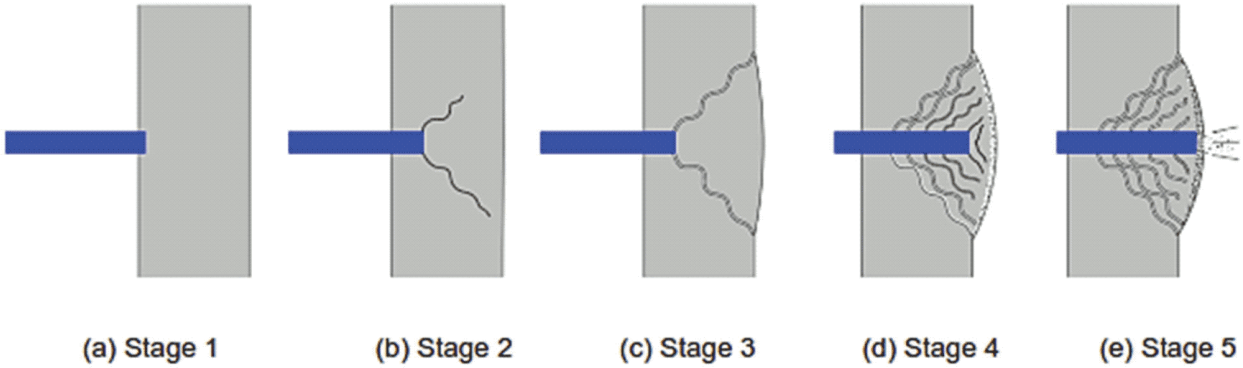


Figure 3.5-2 BWRX-300 Missile Impact Local Damage Modes for the Steel-Plate Composite Barrier

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

In accordance with NUREG-0800 SRP 3.6.1 and 3.6.2, the plant is designed for protection against piping failures inside and outside containment so 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 Subsection 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, 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 BWRX-300 SSCs, their location, seismic category, and Quality Group is provided in Table 3A-1. Overall compliance for BWRX-300 design to GDC 4 is provided in Subsection 3.1.1.4. An analysis of pipe break events is 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 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 piping, 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 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 3.6-2) discusses that each RIV 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 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, B.1(iii)(2), and identifying leakage cracks as specified in BTP 3-4, B.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).

However, 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 seismic restraint 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 USNRC Safety Evaluation Report for NEDC-33911P-A (Reference 3.6-1) has a limitation and condition specified in Section 6 which states “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 3.6-2 and further outlined in Subsection 5.4.3), GDC 4 dynamic effects for the area between the piping weld at the flanged connection to the RIVs and the RPV are eliminated by design provisions and inservice inspections 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 B.1(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 inservice examination of all pipe welds should be conducted during each inspection interval as defined in American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, IWA-2400.” The welds and base metal in the break exclusion zone are pre-examined and subject to Inservice Inspection (ISI) in accordance with ASME Boiler and Pressure Vessel Code (BPVC), Section XI, Division 1, “Rules for Inspection and Testing of Components of Light-Water-Cooled Plants,” (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 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 inservice 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 Components,” 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 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 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 inservice 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 subcompartment 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 License 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 NUREG 0800, 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 subsection 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

Because 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 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, Division 1, “Rules for Construction of Nuclear Facility Components,” 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 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, Division 1, “Rules for Construction of Nuclear Facility Components,” 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 Piping

Breaks in seismically analyzed high energy non-ASME BPVC Section III piping 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 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 containment penetrations consider analytical concepts to eliminate the need to consider postulated cracks as described in BTP 3-4.

ASME BPVC Section III Class 1, 2, 3 and non-ASME BPVC Section III 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 BPVC Section III 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 Section III Class 1, 2, 3 and non-ASME BPVC Section III 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 BPVC Section III 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 under loading).

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, which are 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 nonconservatisms 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% 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 nonconservatisms 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 designed to ASME BPVC, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” Subsection NF, “Supports.”
- Pipe whip restraints that also act as restraints for other loads (e.g., deadweight, seismic) are designed to ASME BPVC, Section III, Subsection NF, (Reference 3.6-9). 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 inservice examinations of pressure-retaining components as required by the requirements 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 thermo-hydraulic 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 NUREG 0800, 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 “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.

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

3.6-3 ASME BPVC, Section XI, Division 1, “Rules for Inspection and Testing of Components of Light-Water-Cooled Plants,” 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, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” Subsection NB, “Class 1 Components,” American Society of Mechanical Engineers, 2021.

3.6-6 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.

3.6-7 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.

3.7 Seismic Design

The seismic design of the BWRX-300 Structures, Systems, and Components (SSCs) complies with 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 their 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 guidance of 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 1/2 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 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) 43-19, “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 SRP 3.7.1, “Seismic Design Parameters.”

Subsections 3.7.1.1.1 and 3.7.1.1.2 present the SSE design response spectra and time histories used for the design of the BWRX-300 Seismic Category I SSCs at the Clinch River Nuclear (CRN) Site.

The OBE at the CRN Site is set as 1/3 of the SSE. In accordance with 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 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 subsection define the amplitude and the frequency content of the design ground motion used for the seismic design of the BWRX-300 at the CRN Site. 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 consisting of engineered backfill on top of the existing fill, in-situ residual soil, weathered rock and base rock. 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 43-19 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 CRN 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, at the top of the rock elevation having a significant contrast between rock and overlaying soil shear wave velocities.

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 as discussed in Subsection 2.5.2. Figure 2.5.2-93R presents the horizontal FIRS, PBSRS and PBIRS developed for the BWRX-300 at the CRN Site.

The vertical FIRS, PBSRS and PBIRS for the BWRX-300 at the CRN Site are developed by applying Vertical to Horizontal (V/H) ratios calculated using Pezeshk et al. V/H model (Reference 3.7-4) to the horizontal spectra. Table 2.5.2-40 presents the rock and soil V/H ratios used for calculation of the vertical ground motion design spectra.

The Nuclear Energy Institute (NEI) checks 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 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.

To ensure the propagated SSI foundation input motions envelop the horizontal and vertical PBSRS, the frequency dependent factors shown in Figure 3.7-1 are used to augment the FIRS. As shown in Figure 3.7-2, which presents the NEI check of the augmented horizontal and vertical FIRS, the envelope of the 5% damped spectra for the responses at the surface of the Lower Bound (LB), Best Estimate (BE) and Upper Bound (UB) profiles do not fall below the PBSRS. This confirms that the DC/COL-ISG-017 requirement is met.

Figure 3.7-3 presents the augmented ground motion spectra defining the amplitude and frequency content of the horizontal and vertical SSI input control motion applied to the SSI model at the RB foundation bottom. Figure 3.7-4 and Figure 3.7-5 present the PBIRS at the top of the un-weathered rock layer and the PBSRS at the finished plant grade elevation, respectively.

The derived horizontal and vertical FIRS envelop the spectra from RG 1.60, “Design Response Spectra for Seismic Design of Nuclear Power Plants,” anchored at 0.1 g, as required by Appendix S to 10 CFR Part 50 and demonstrated in Figure 3.7-3.

3.7.1.1.2 Design Time Histories

Design ground motion Acceleration Time Histories (ATH) used as input to the seismic SSI analyses of the integrated RB at the CRN Site are developed by spectral matching seed ground motion records to the ground motion design response spectra presented in Figure 3.7-3. In accordance with Section 5.2.3 of NEDO-33914-A, five sets of three design motion ATHs, in the two horizontal and in the vertical directions, are developed for the design to mitigate uncertainties due to the phasing of the time history frequency components. This number of design ground motion ATHs exceeds the minimum of four allowed in NUREG/CR-5347, “Recommendations for Resolution of Public Comments, Seismic Design Criteria.”

Design ATHs are developed by fitting recorded seed time histories to the 5% damped target design spectra. Developed ATHs meet the acceptance criteria for Option 2 in Section II.1.B of SRP 3.7.1, Appendix F of RG 1.208, and the additional guidance of Section 5.2.3 of NEDO-33914-A.

Table 3.7-2 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 the CRN Site. These are selected from the NGA-West2 ground motion database maintained by the Pacific Earthquake Engineering Research Center (Reference 3.7-5). In accordance with NEDO-33914-A guidance, the selected seed time histories include records with different magnitude and distance bins that have spectral shapes reasonably consistent with the spectral shape of the design target spectrum over the frequency range of interest and characteristics that reasonably represent the earthquake motions expected at the CRN Site.

Using the seed time histories in Table 3.7-2, time histories compatible to the design ground motion spectra in Figure 3.7-3 are generated using the time domain spectral matching method, “An Improved Method for Nonstationary Spectral Matching,” (Reference 3.7-6), where adjustment of initial time histories (seed motions) is made by adding wavelet functions to the initial acceleration time history in time domain. This adjustment is repeated until its spectrum becomes compatible with the target spectrum over the desired frequency range. In accordance with Section 5.2.3 of NEDO-33914-A, the time step of the modified time histories is refined to 0.0025 seconds for the purposes of accurately calculating high frequency in-structural responses.

Response spectra of the generated ATHs are computed at a minimum of 100 points per frequency decade, uniformly spaced over the log frequency scale, and compared to the appropriate target response spectra. Cross-correlation coefficients, peak values, arias intensity, and power spectrum density functions are then computed for the spectrally matched time histories to ensure the modified ATHs meet the following SRP 3.7.1 and RG 1.208 criteria:

- The time step for the modified time histories are less than 0.005 seconds.
- The average of the scaled response spectra of the five sets of modified ATHs do not fall more than 10% below the target spectrum at any one frequency,
- The modified ATH response spectra do not exceed the target response spectrum at any frequency by more than 30% in the frequency range of interest,
- Power spectrum density functions of the modified ATHs do not have significant gaps in energy at any frequency over the frequency range of interest,
- Modified time histories have typical strong motion durations (defined by 5-75% arias intensity), peak ground acceleration, velocity and displacement, and ratios of peak ground acceleration, velocities and displacements that are generally consistent with characteristics for the magnitude-distance pairs for controlling events considered by the PSHA, and
- The modified ATHs are statistically independent with the directional correlation coefficients between pairs of time histories not exceeding a value of 0.16.

Figure 3.7-6, Figure 3.7-7, and Figure 3.7-8 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 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. The plot demonstrates that there are no gaps in the power spectrum density for any of the time histories.

Table 3.7-3 presents the ground motion parameters of the matched time histories for FIRS. These parameters are consistent with the typical values for magnitude and distance range for the controlling earthquake as listed in Table 3-5 of NUREG/CR-6728, “Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Hazard- and Risk-Consistent Ground Motion Spectra Guidelines,” U.S. Nuclear Regulatory Commission. 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.

Figure 3.7-10 through Figure 3.7-24 show the matched time history for the selected records listed in Table 3.7-2 matched to the horizontal and vertical FIRS. These plots demonstrate that the selected acceleration, velocity, and displacements are compatible and do not result in displacement baseline drift in conformance with the guidance of Section 3.7.1, Section II.1.B.

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. To account for the primary non-linearity of the subgrade materials, stiffness and damping properties of subgrade materials compatible to the free-field strains generated by design level earthquake event are developed based on results of Approach 3 SRA discussed in Subsection 2.5.2. The strain-compatible damping of the subgrade materials is limited to 15% in accordance with the regulatory guidance of SRP 3.7.1, RG 1.208, Appendix E and the recommendations of ASCE/SEI 4-16, “Seismic Design of Safety-Related Nuclear Structures,” (Reference 3.7-7), Section C5.2.

Following the 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 may be used for development of seismic demands for structural design per ASCE/SEI 43-19, 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 and the rock below. For a description of the supporting media at the CRN Site, 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, presented 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 the CRN Site. The LB and UB shear wave velocities reflect a

minimum coefficient of variation of each layer properties of 0.5. 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. The groundwater level considered for the seismic SSI analysis at the CRN Site is Elevation 814.5 ft per Table 2.0-1R. Figure 3.7-25 and Figure 3.7-26 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 4-16, is implemented for the seismic design of the integrated RB structures. Seismic structural stress demands are obtained directly from the results of SSI analyses performed on models that include Three-Dimensional (3D) Finite Element (FE) representations of the integrated RB and the surrounding subgrade. 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 4-16, 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. Material damping is introduced using a complex stiffness matrix with complex elastic moduli. For each element, the real part of the elastic modulus represents stiffness, and the imaginary part represents damping, which leads to effective damping ratios that are frequency independent.

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 presented in Subsection 2.5.2. The profiles of strain-compatible subgrade properties are presented Subsection 3.7.1.3. The uncertainties related to variation of soil and rock properties are addressed as described in Subsection 2.5.2.5.1.

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 discussed in Subsection 2.5.2. The requirement of 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 the CRN Site 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 the CRN Site.

The seismic responses of the RB at the CRN Site 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 SRP 3.7.1, SRP 3.7.2 and RG 1.61. The model also meets the structural modeling requirements of Section 3 of ASCE/SEI 4-16, and the one-step modeling requirements and guidelines of Section 5.1.1 of NEDO-33914-A.

Appendix 3B presents the details of the 3D FE model used in the seismic SSI analysis of the integrated RB. This model consists of the integrated RB structure, the surrounding subgrade, the excavated volume of the subgrade materials replaced by the embedded portion of the RB structure and the near field backfill materials. Certain construction elements of the CRN excavation plan are also included in the integrated RB 3D FE model. These consist of a concrete mud mat under the integrated RB common mat foundation, an annulus of concrete fill adjacent to the RB wall extending the full RB embedment height and a concrete shelf increasing the thickness of concrete fill around the annulus from top of rock to surface. Solid elements are used to model the concrete fill material. The inclusion of the concrete fill deviates from the seismic analysis modeling approach described in Section 5.3.8 of NEDO-33914-A. However, the presence of concrete fill per the CRN excavation plan is expected to affect the seismic response of the RB, especially at elevations above the top of un-weathered rock. Modeling these features results in more realistic seismic response consistent with the guidance of SRP 3.7.2.

The CRN excavation plan considers the use of temporary excavation support for the soil around the RB excavation above top of rock. To assess the effects of the excavation support on the RB seismic response (see Subsection 3.7.2.9.2), additional solid elements are also included in the RB SSI model to model the temporary excavation support. These solid elements are assigned soil material properties for the base case seismic analysis and concrete material properties for the sensitivity analysis discussed in Subsection 3.7.2.9.2.

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 4-16, 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 (sloshing) components are considered in accordance with Section II.1.A of SRP 3.7.2 and following the provisions of Section 3.6.3 of ASCE/SEI 4-16, and Chapter 5 of American Concrete Institute (ACI) 350.3, "Code Requirements for Seismic Analysis and Design of Liquid-Containing Concrete Structures," (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 4-16.

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 4-16.

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 modeling 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 4-16.

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 4-16, 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 4-16. 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 consisting of engineered backfill placed on top of the existing fill, in-situ residual soil, weathered rock, and in-situ base rock as stated in Subsection 3.7.1.1.1. The SSI analysis model described in Subsection 3.7.2.3 does not consider the SSSI effects of the surrounding Power Block structures resting on foundations supported by drilled shafts that transfer the vertical loads from these buildings to the bedrock.

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 where the in-situ soil and weathered rock are excavated down to competent rock and replaced with engineered fill up to plant grade Elevation 814.5 ft. Properties assigned to the engineered fill are competent for transferring the vertical loads from the Power Block near surface foundations to the bedrock. The consideration of two different free-field site property conditions for the SSI analyses is an alternative approach adopted to add margin in RB seismic demands to account for the absence of drilled shafts from the analyzed models due to their low design maturity. 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 which is considered to be the more representative of the expected actual plant conditions.

The following are key requirements and approaches considered in the seismic SSI analyses to ensure the structural integrity and stability of the deeply embedded 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 DC/COL-ISG-017 guidance and addressing current limitations in 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 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 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 4-16. 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 4-16, 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-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 4-16, 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 4-16, 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 4-16, 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 4-16, and following the guidance of Section 5.3 of NEDO-33914-A.

The NEDO-33914-A specified sensitivity analysis cases are modified to adequately address the SSI parameter uncertainties related to the specific layout of the CRN seismic SSI analysis model described in Subsection 3.7.2.3. The following subsections present the modified approach for evaluating the effects of parameter variations on the integrated RB responses at the CRN Site.

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 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 the CRN Site, 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.

If the estimated stresses in the steel-plate composite structural members in the design basis BE case meet the applicable stress level criteria for Response Level 2 per ASCE/SEI 43-19, a sensitivity SSI analysis is performed to address the effects of structural stiffness variations on models where the steel-plate composite structural members are assigned LB stiffness and higher SSE damping properties. In accordance with Section 5.3.5 of NEDO-33914-A, the sensitivity analysis is performed for BE subgrade profile.

The effects of structural stiffness variations are assessed by comparing the 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 with a few modifications given that the concrete fill around the RB is included in the RB SSI model as discussed in Subsection 3.7.2.3. Refer to Subsection 3.7.2.9.7 for a discussion on how uncertainties related to the concrete fill are addressed.

As stated in Subsection 3.7.2.3, the CRN excavation plan considers the use of temporary excavation support for the soil around the RB excavation above top of rock. The uncertainty related to the excavation support may be investigated by performing a sensitivity SSI analysis using the RB model with adjacent Power Block structures supplemented by the explicit modeling of the excavation support and BE properties of surrounding subgrade materials. Shell and beam elements or solid elements may be used to represent the BE dynamic properties of the excavation support.

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 the 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 the CRN site. 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 4-16 and Section 5.3.9 of NEDO-33914-A.

A sensitivity SSI analysis is performed on a model where portions of the concrete fill that may experience separation from the surrounding subgrade soil are assumed to remain unbonded for the total duration of the earthquake. 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. Alternatively, the separation may be conservatively considered for the full depth of soil, down to the top of un-weathered rock.

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 the CRN Site is assessed following the guidelines in Section 5.3.3 of NEDO-33914-A, based on the geological and seismological conditions of the site.

As discussed in Subsection 2.5.2.6, a Two-Dimensional (2D) sensitivity analysis is performed to investigate the effect of the approximately 33-degree dip of rock layers on site response results. The response spectra results from the 2D analysis indicate that the response spectra from the 1D analysis presented in Subsection 3.7.1 adequately envelop the effects of non-vertically propagating seismic waves on the input ground motion.

3.7.2.9.6 Non-Linear Sensitivity Evaluation

Results of the foundation interface analysis presented in Subsection 2.5.4.13, indicate that the RB at the CRN Site is mainly embedded in rock and exhibits nearly linear elastic behavior during SSE level event. The ground motion at the CRN site is characterized by high energy content in high frequencies resulting in subgrade strains with a low amplitude. Therefore, the effects of the non-linearity of subgrade materials on the seismic response and design of RB are insignificant. A nonlinear sensitivity SSI analysis described in NEDO-33914-A (Reference 3.7-1) is not required and NEDO-33914-A Limitation and Condition (L&C) 8.5 does not apply.

3.7.2.9.7 Concrete Fill Strength

The inclusion of the concrete fill in the RB 3D FE model introduces the additional uncertainty over the strength of the concrete fill which is not designed to maintain its structural integrity throughout the operational life of the plant. To address this uncertainty, an additional sensitivity evaluation is performed using a reduced value of the design concrete fill strength. Responses calculated from this sensitivity analysis are compared to results obtained from the SSI analysis of the RB model with adjacent Power Block structures with design concrete fill strength. If the comparisons show significant exceedances (> 10%) in the RB seismic response due to the change in concrete fill strength, the results of this sensitivity analysis are included in the RB seismic design basis.

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 4-16, 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:

$$[M] \{\ddot{u}(t)\} + [C] \{\acute{u}(t)\} + [K] \{u(t)\} = \{P(t)\}$$

where:

$\{u(t)\}$ = time dependent displacement of nonsupport points relative to the supports.

$\{\acute{u}(t)\}$ = 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 generalize 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 4-16.

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 Sections 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 4-16, 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 plate 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 regularity guidance of Section 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 43-19 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 Subsection 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 4-16. 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 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,” GE-Hitachi Nuclear Energy Americas, LLC, Revision 2, 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 Pezeshk, S., Farhadi, A., and Haji-Soltani, A., “A new model for vertical-to-horizontal response spectral ratios for central and eastern North America,” Bulletin of the Seismological Society of America, v. 112, p. 2018-2030, 2022.

3.7-5 PEER, The Pacific Earthquake Engineering Research Center NGA-West2 ground motion database, http://ngawest2.berkeley.edu/, 2022.

3.7-6 Al Atik, L., and Abrahamson, N., “An Improved Method for Nonstationary Spectral Matching,” Earthquake Spectra, Vol. 26, No. 3, pp 601-617, 2010.

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 Nuclear 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, Division 1, “Rules for Construction of Nuclear Facility Components,” American Society of Mechanical Engineers, 2021.

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
(Sheet 1 of 2)

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	1/2 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
TB Structure	Seismic Category II	IBC		
	To meet II/I Seismic Interaction requirements (4)	ASCE/SEI 43 and ASCE/SEI 4	SSE	LS-C

Table 3.7-1 Seismic Categories and Design Basis of BWRX-300 Structures
(Sheet 2 of 2)

Structure	Seismic Category/Evaluation	Design/Evaluation Basis	Design Basis Earthquake (1)	Limit State (2)
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 43-19:
   - 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 Seismic 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

Record Sequence Number	Year	Earthquake Name	Station Name	Mag	Rrup (km)	V_S30 (m/sec)	Comp	PGV/PGA (cm/sec/g)	PGA*PGD/ PGV^2	5-75% Dur(sec)	Scale Factor
2113	2002	Denali, Alaska	TAPS Pump Station #09	7.9	54.8	383	13	205.60	3.84	30.33	4.05
							103	162.61	5.49	29.98	3.61
							UP	162.60	3.14	34.49	3.86
28	1966	Parkfield	Cholame - Shandon Array #12	6.2	17.6	409	50	111.56	5.46	14.33	5.62
							320	115.23	5.32	14.72	5.15
							DWN	85.49	4.85	13.54	5.26
302	1980	Irpinia_ Italy-02	Rionero In Vulture	6.2	22.7	575	000	151.01	1.75	11.11	2.86
							270	66.49	3.68	9.37	3.70
							UP	98.67	2.03	10.82	4.49
522	1986	N. Palm Springs	Indio	6.1	35.6	308	225	93.68	3.68	9.66	5.23
							315	100.05	2.50	7.45	2.87
							UP	37.30	10.49	7.77	4.94
3513	1999	Chi-Chi_ Taiwan-06	TCU145	6.3	59.7	240	N	67.94	7.24	12.45	6.10
							W	110.90	7.61	16.68	5.51
							V	128.09	5.76	14.84	7.42

Table 3.7-3 Ground Motion Parameters of the Matched Time Histories

Record Sequence Number	Year	Earthquake Name	Station Name	Mag	Rrup (km)	V_S30 (m/sec)	Comp	PGV/PGA (cm/sec/g)	PGA*PGD/PGV^2	5-75% Dur(sec)
2113	2002	Denali, Alaska	TAPS Pump Station #09	7.9	54.8	383	13	98.50	7.28	29.68
							103	102.82	8.84	29.10
							UP	96.82	3.36	33.72
28	1966	Parkfield	Cholame - Shandon Array #12	6.2	17.6	409	50	76.13	9.15	12.15
							320	65.95	10.36	12.64
							DWN	68.92	4.47	12.05
302	1980	Irpinia_ Italy-02	Rionero In Vulture	6.2	22.7	575	000	64.18	6.86	13.91
							270	56.62	5.64	12.32
							UP	69.19	2.74	13.89
522	1986	N. Palm Springs	Indio	6.1	35.6	308	225	57.07	7.17	10.59
							315	57.67	5.52	6.16
							UP	69.40	5.12	11.64
3513	1999	Chi-Chi_ Taiwan-06	TCU145	6.3	59.7	240	N	48.75	12.96	12.37
							W	74.37	9.89	13.34
							V	105.85	6.69	15.69

Table 3.7-4 Correlation Coefficients Between Pairs of Modified Time Histories

Set Number	Record Sequence Number (RSN)	Earthquake Name	Correlation Coefficient between V & H1	Correlation Coefficient between V & H2	Correlation Coefficient between H1 & H2
1	2123	DENALI	-0.085	0.069	0.013
2	28	PARKF	0.096	0.037	0.050
3	302	ITALY	0.145	0.027	0.012
4	522	PALMSPR	0.072	0.073	0.026
5	3513	CHICHI	0.093	0.104	0.080

Table 3.7-5 Seismic Damping Values for BWRX-300 Structures

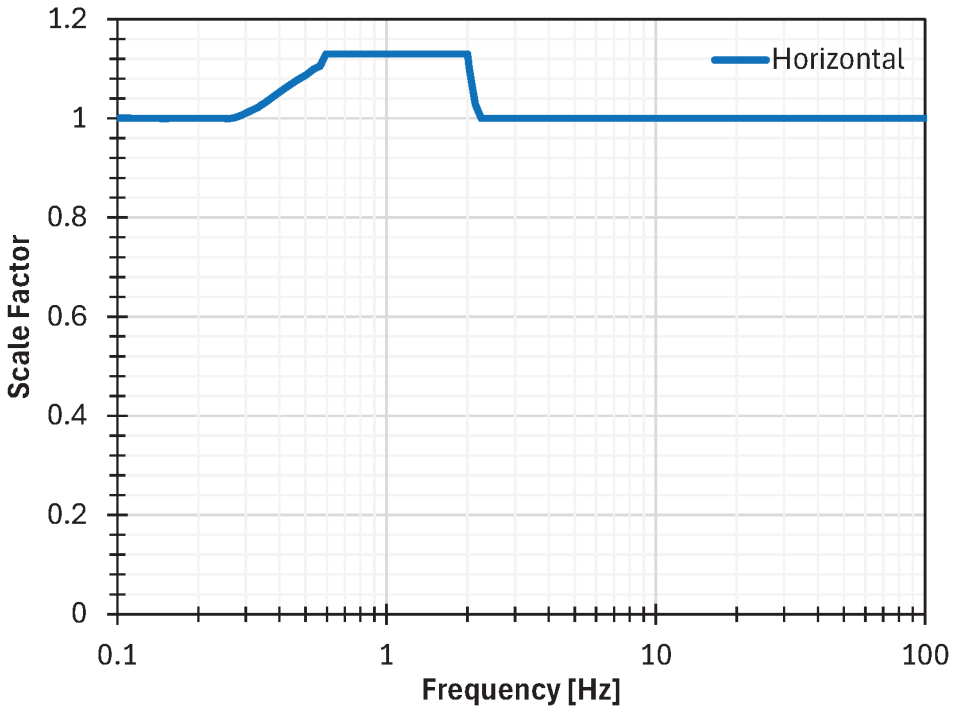
Material	Critical Damping	
	OBE (%)	SSE (%)
Steel-plate composite structures	3	5
Welded and Friction-bolted steel structures	3	4
Bearing-bolted steel structures	5	7
Reinforced concrete structures	4	7

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



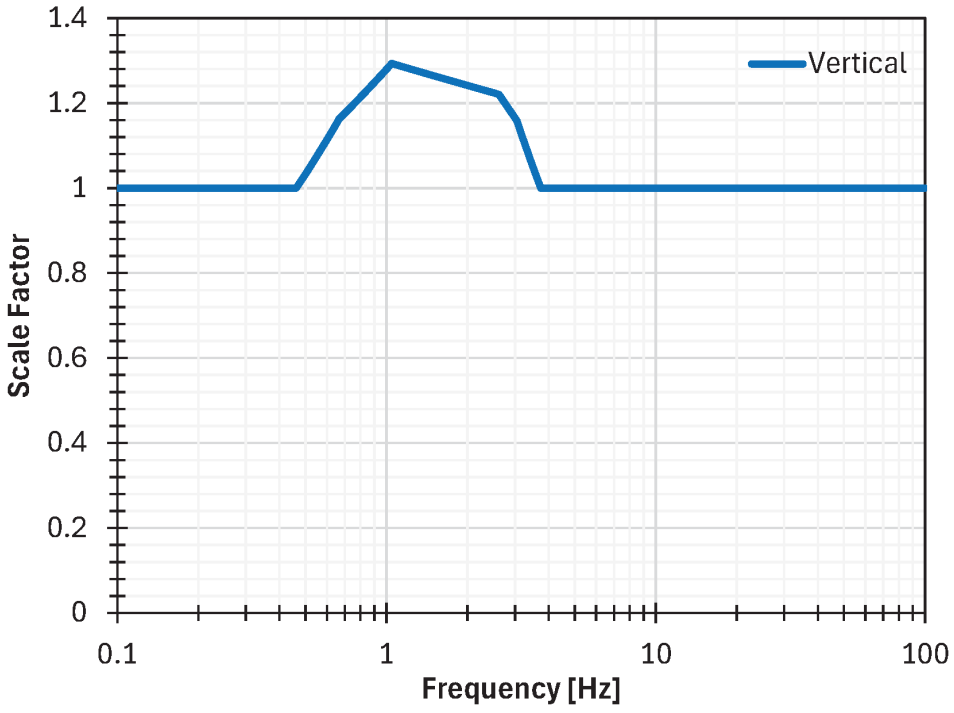
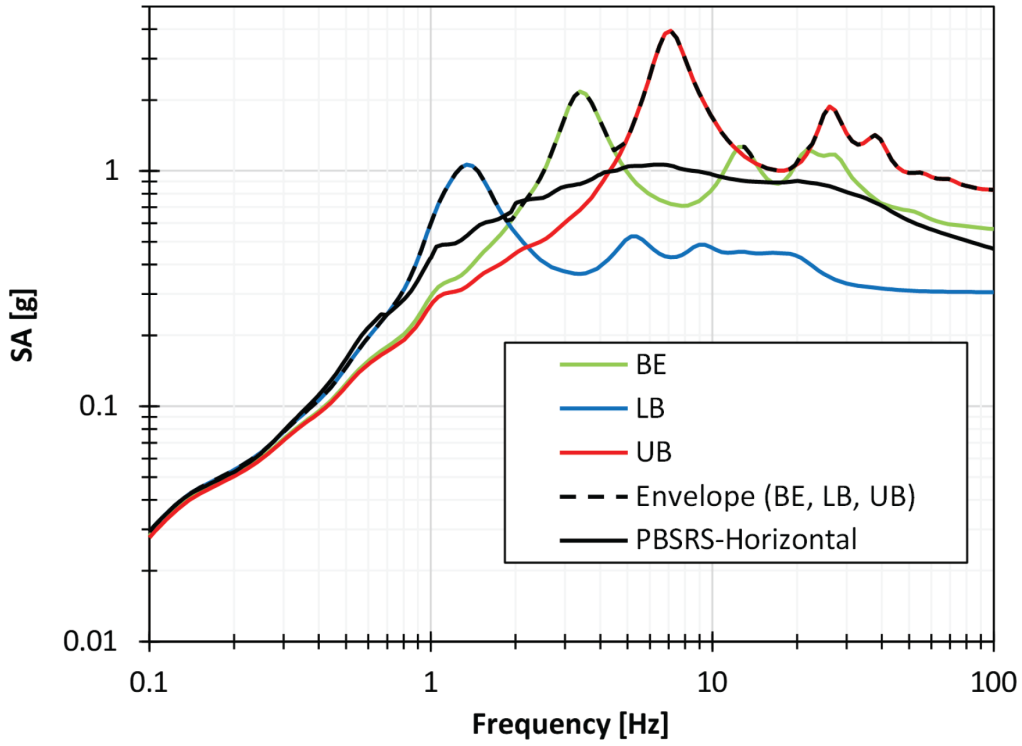


Figure 3.7-1 Scale Factors for Adjusting the Foundation Input Response Spectra (FIRS)



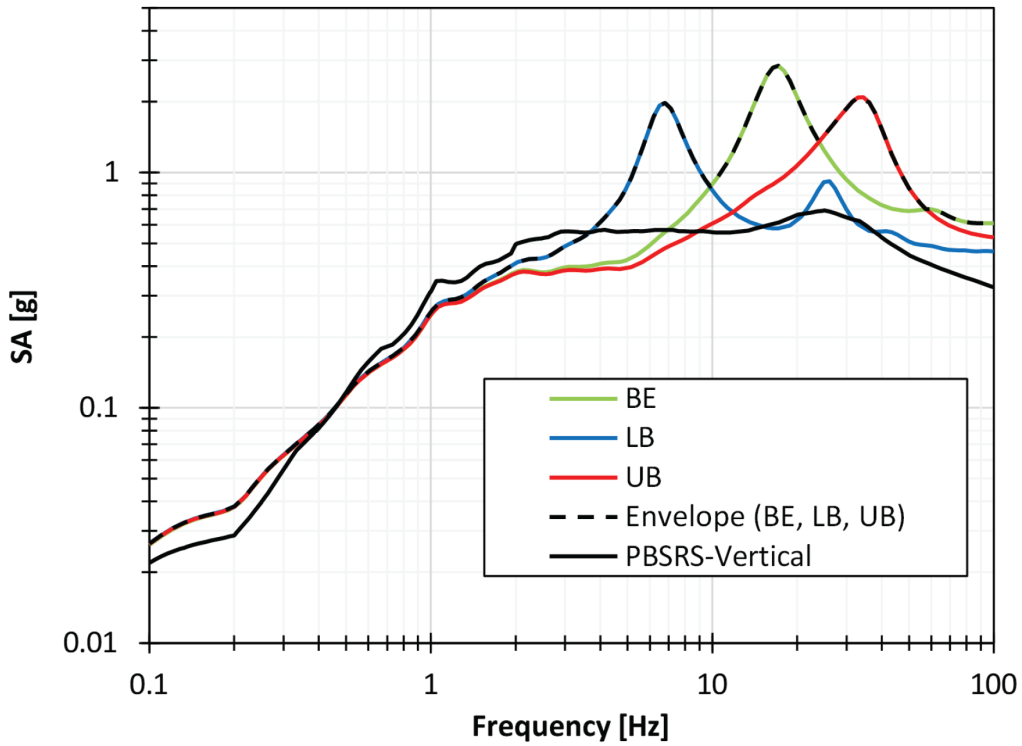
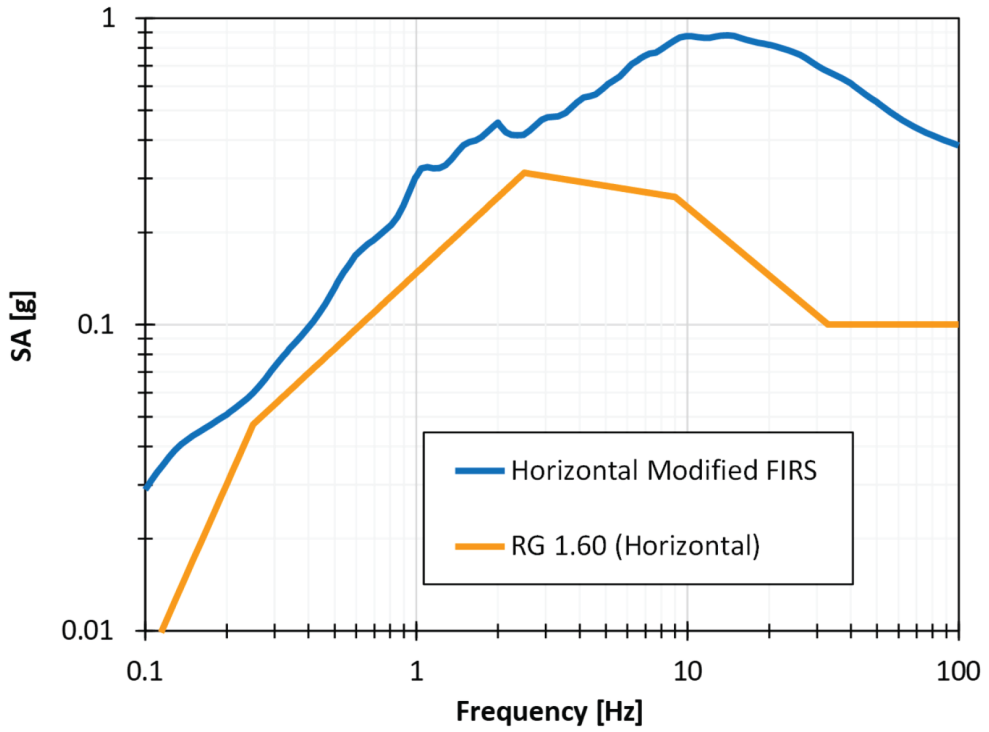


Figure 3.7-2 Nuclear Energy Institute (NEI) Check Results for the Ground Motion Response Spectra



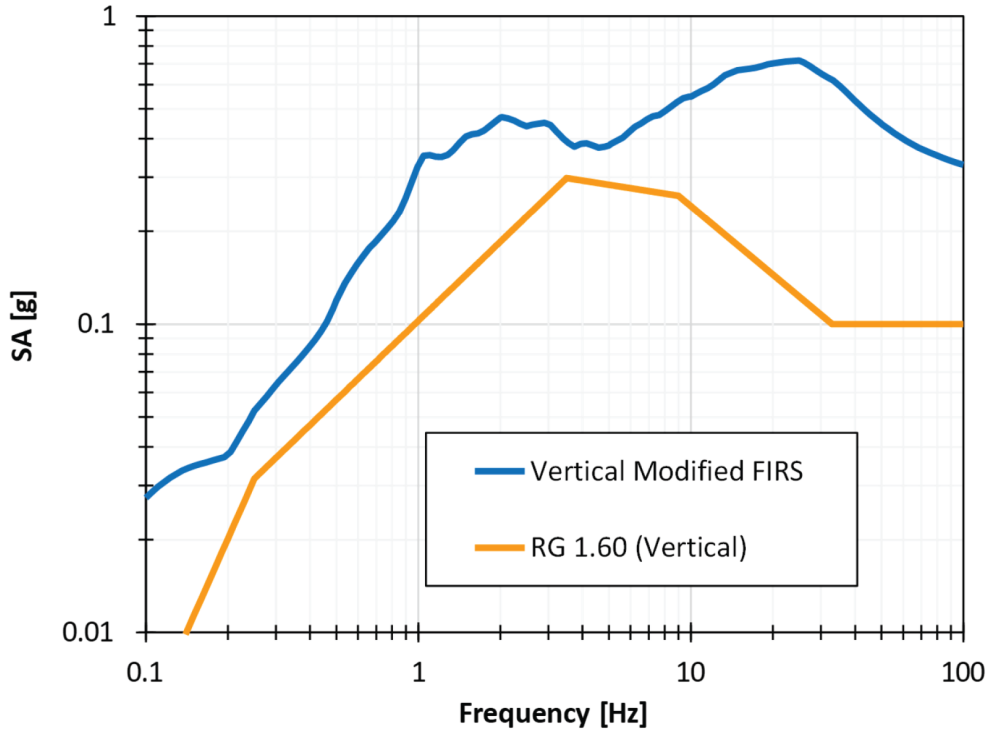


Figure 3.7-3 Augmented Safe Shutdown Earthquake Ground Motion Input Response Spectra at Bottom of Reactor Building Foundation

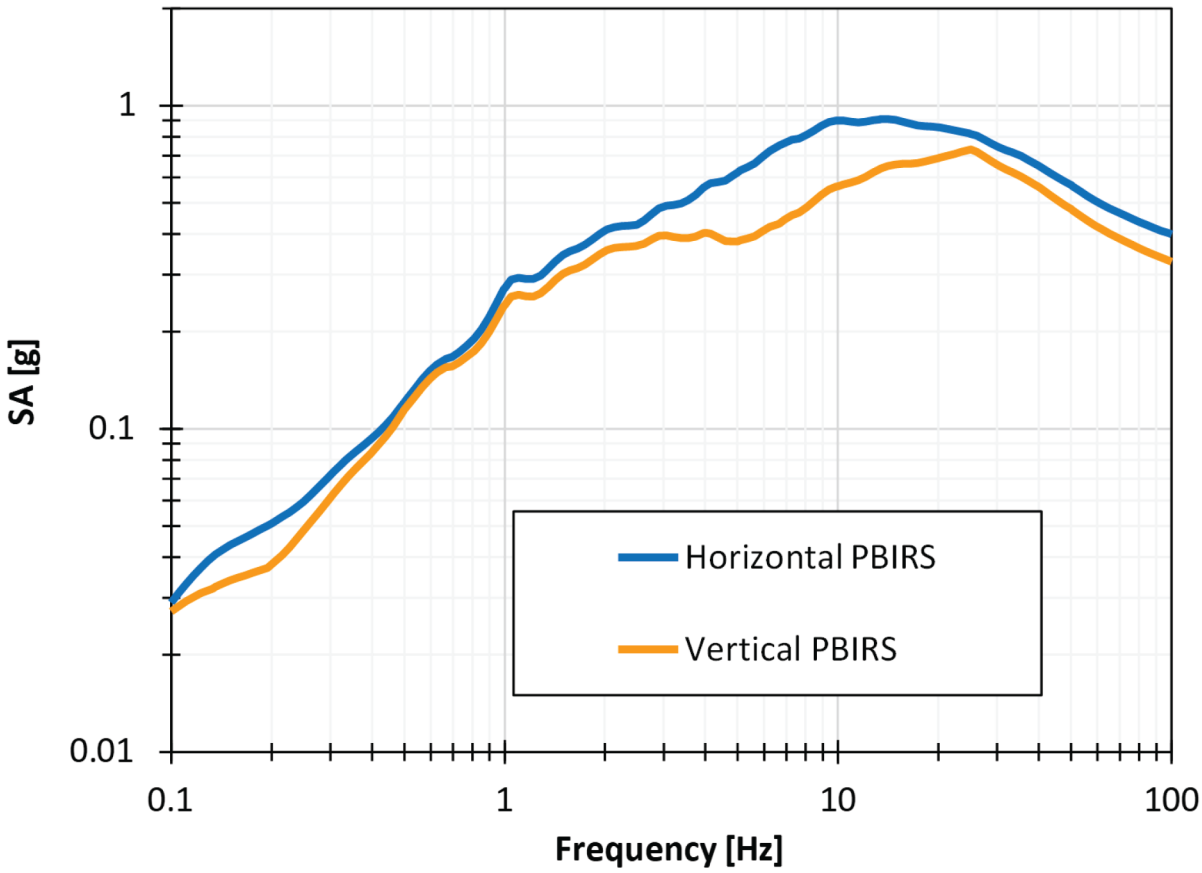


Figure 3.7-4 Safe Shutdown Earthquake Ground Motion Input Response Spectra at Top of Un-Weathered Rock

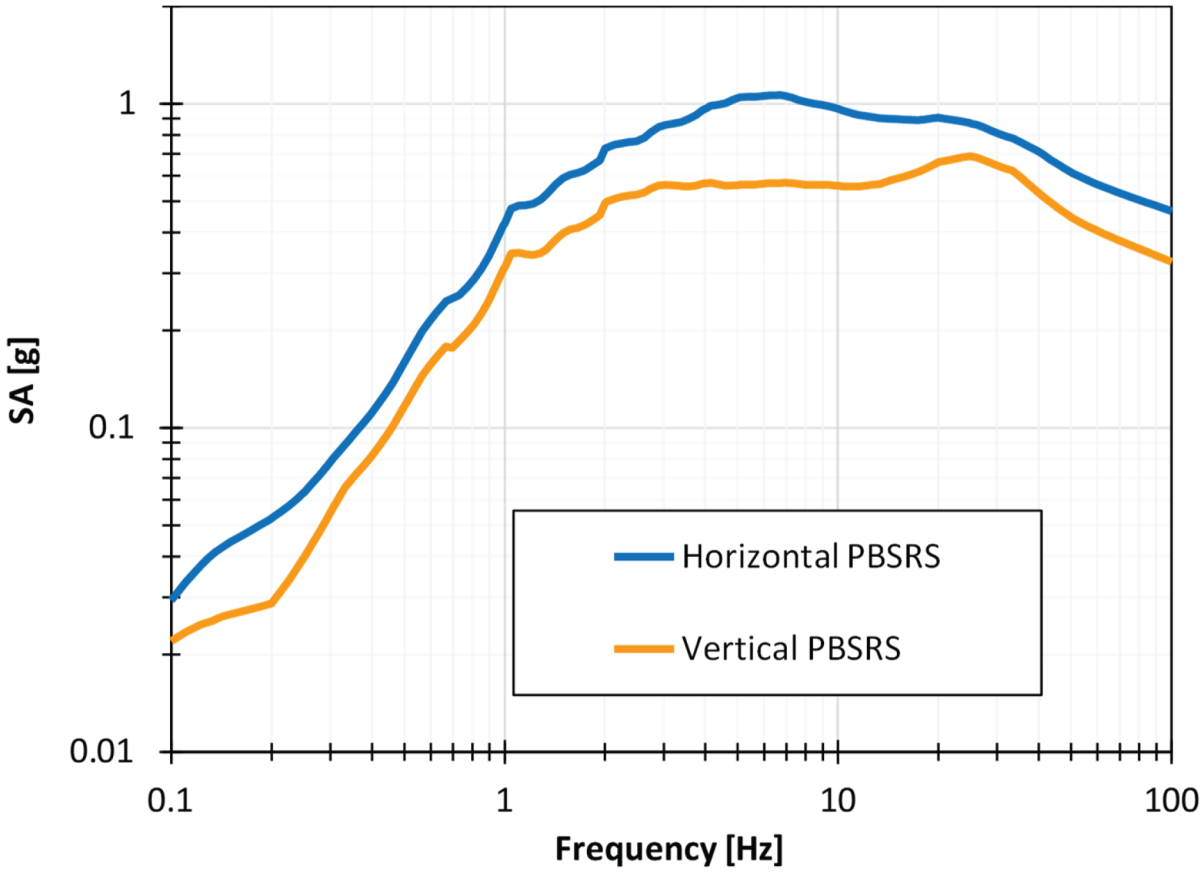


Figure 3.7-5 Safe Shutdown Earthquake Ground Motion Input Response Spectra at Plant Finished Grade Elevation

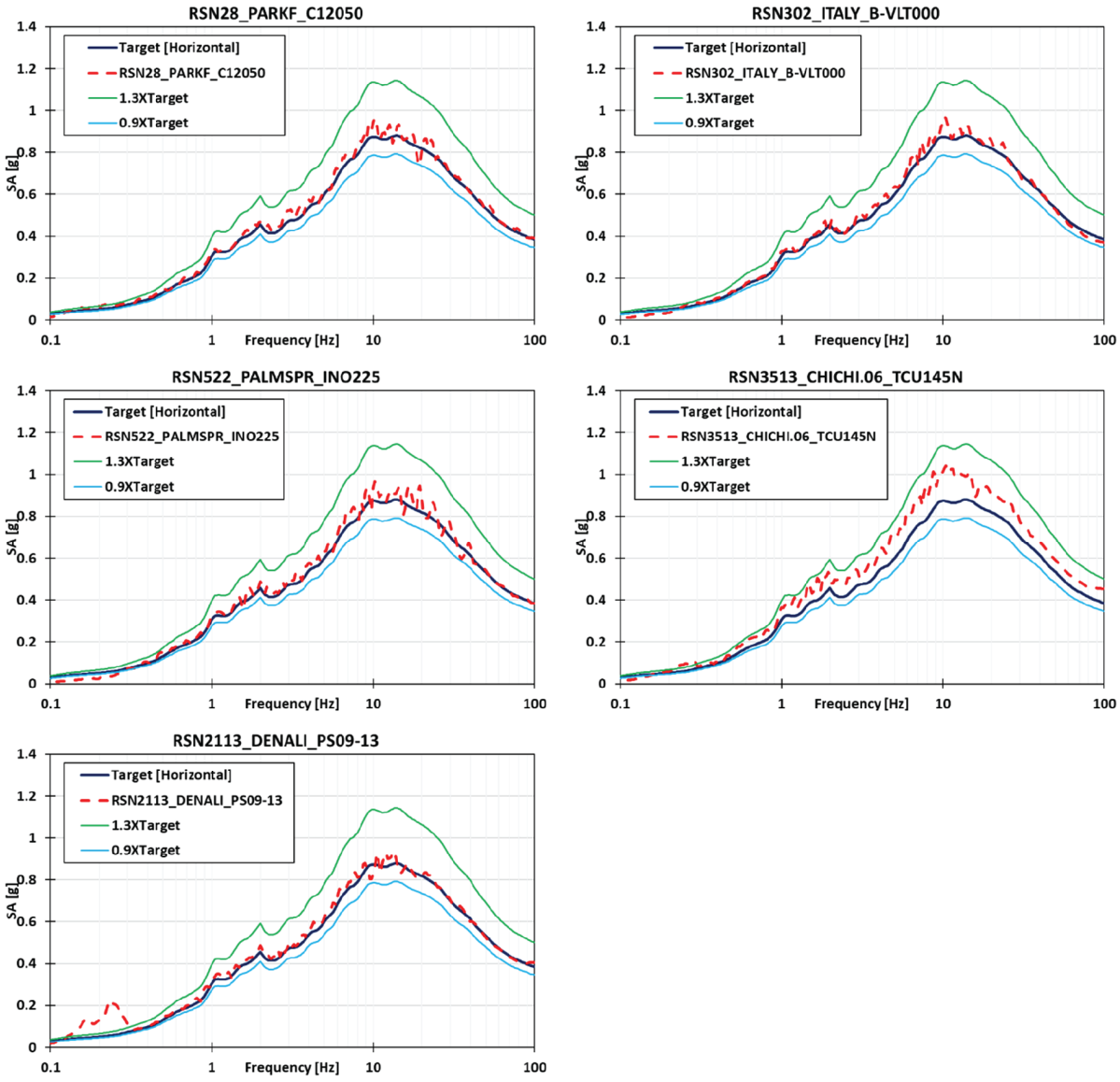


Figure 3.7-6 Comparison of Scaled Time Histories with Target Spectra for First Horizontal Component

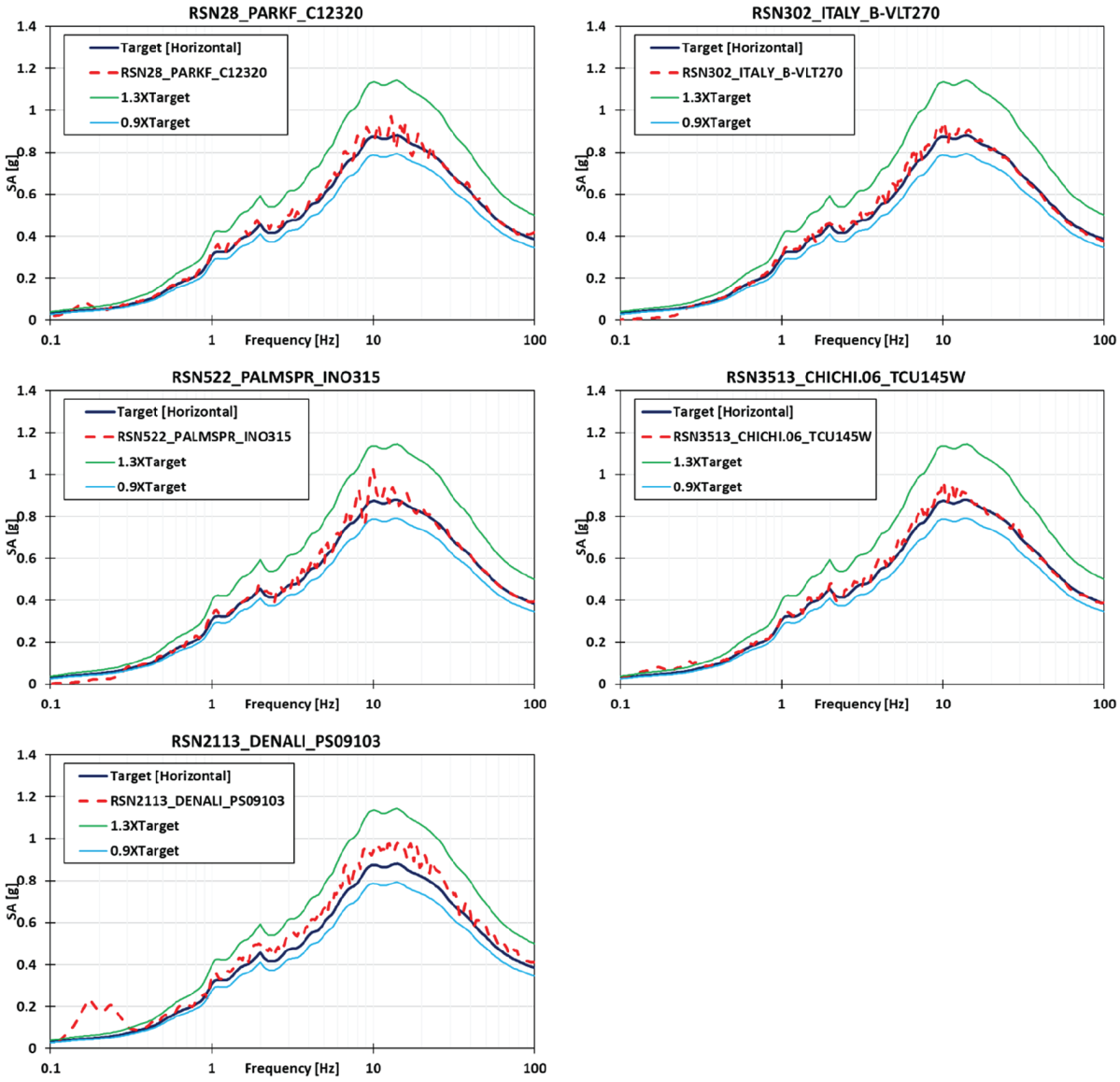


Figure 3.7-7 Comparison of Scaled Time Histories with Target Spectra for Second Horizontal Component

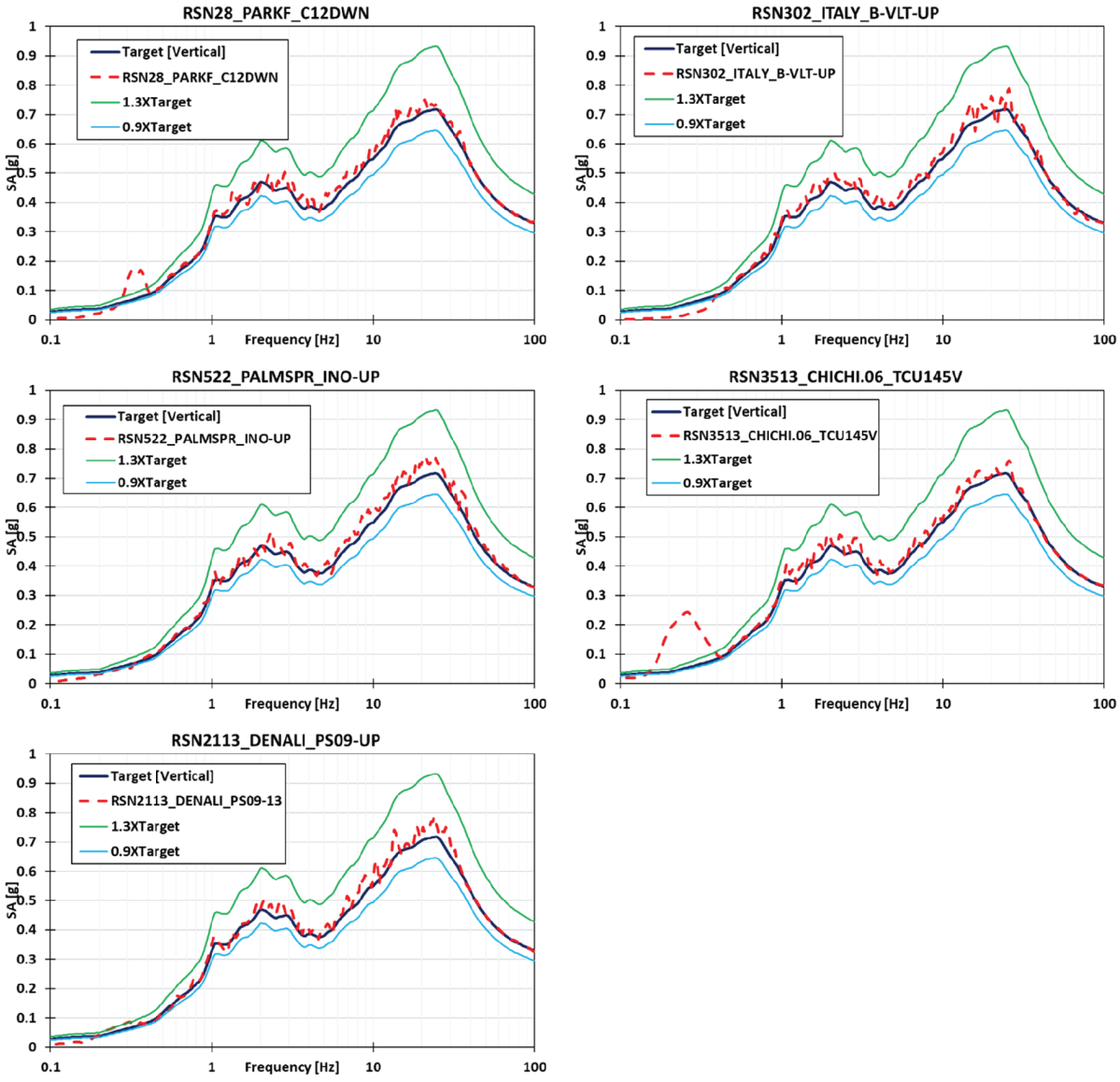


Figure 3.7-8 Comparison of Scaled Time Histories with Target Spectra for the Vertical Component

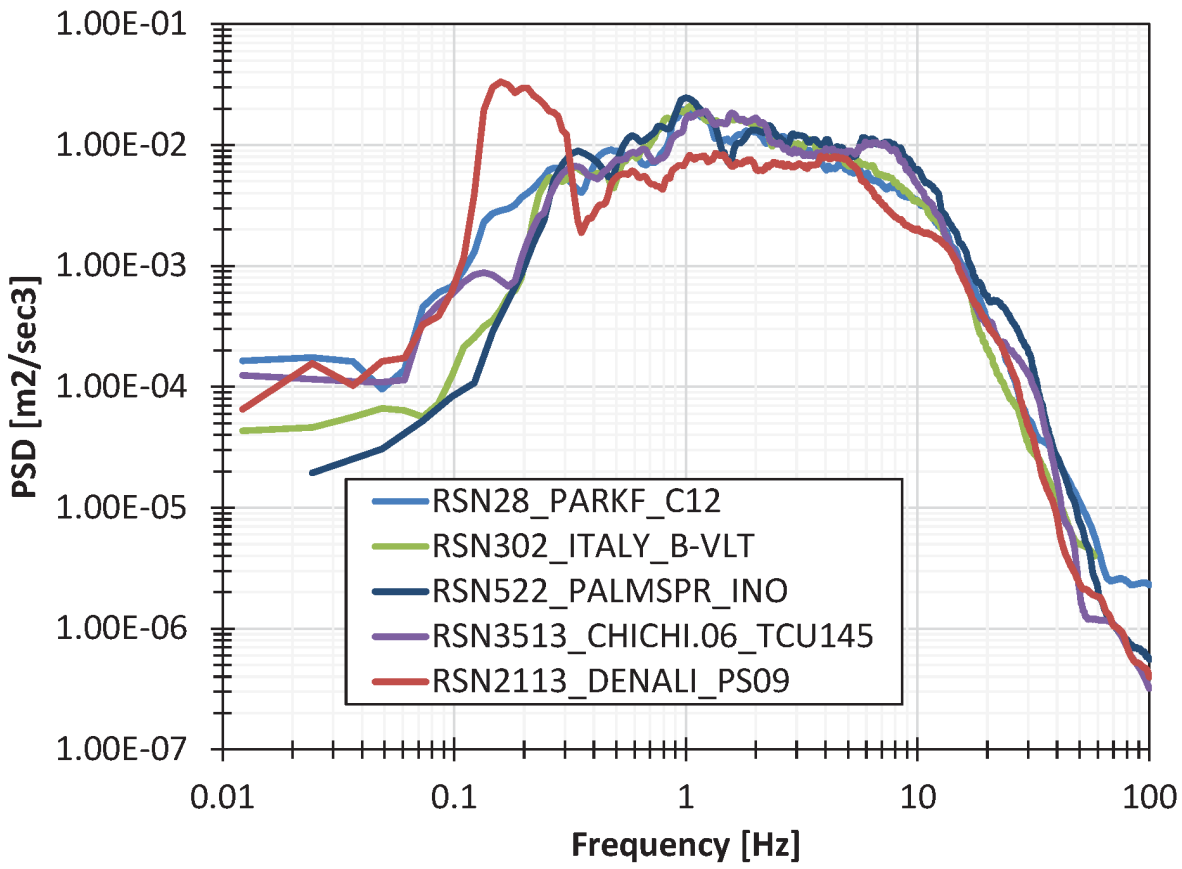


Figure 3.7-9 Smoothed Power Spectrum Densities of the Geometric Mean of Modified Time Histories

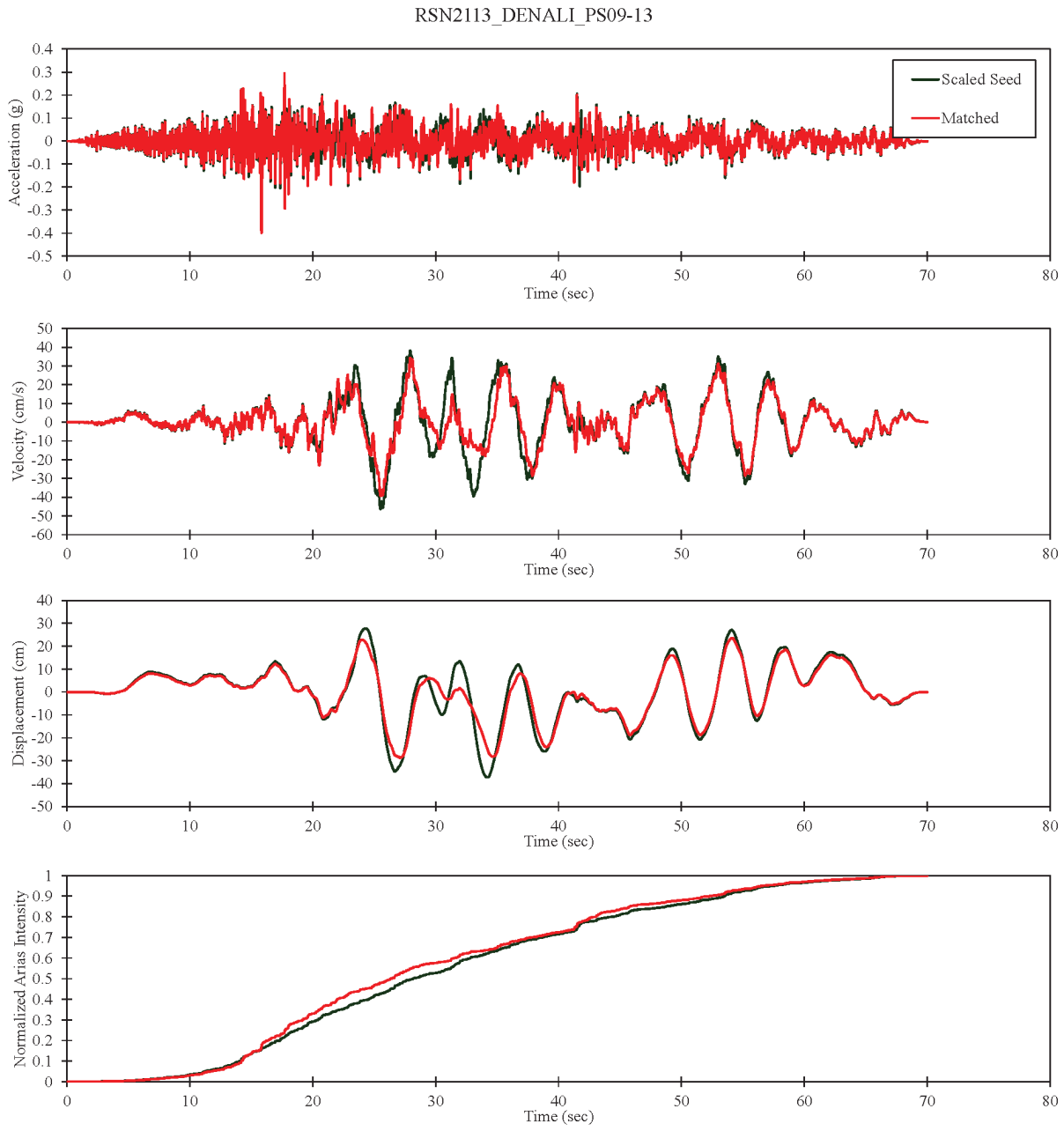


Figure 3.7-10 Acceleration, Velocity, Displacement and Normalized Arias Intensity of the Matched First Horizontal Time History for the Denali, Alaska Earthquake

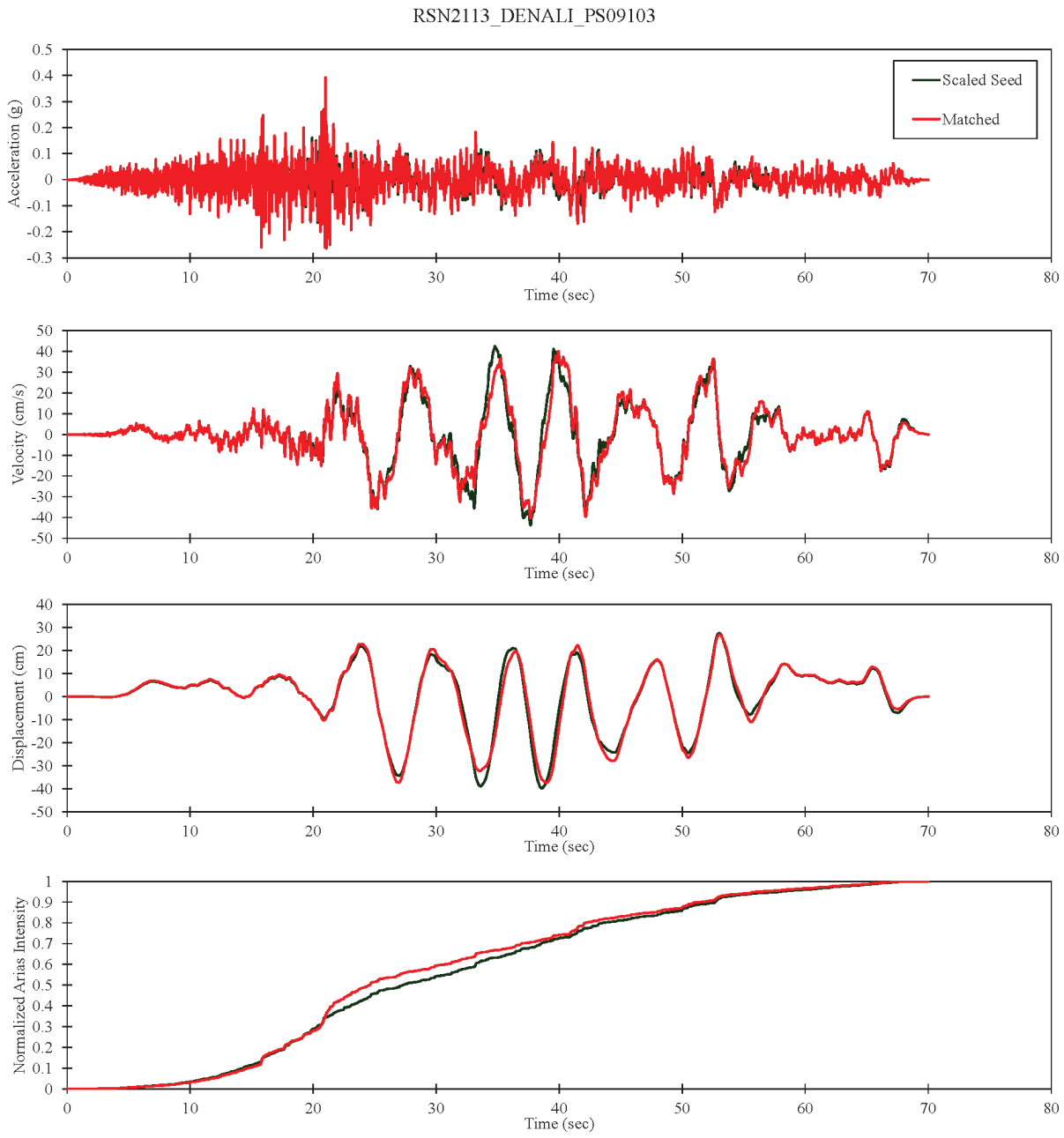


Figure 3.7-11 Acceleration, Velocity, Displacement and Normalized Arias Intensity of the Matched Second Horizontal Time History for the Denali, Alaska Earthquake

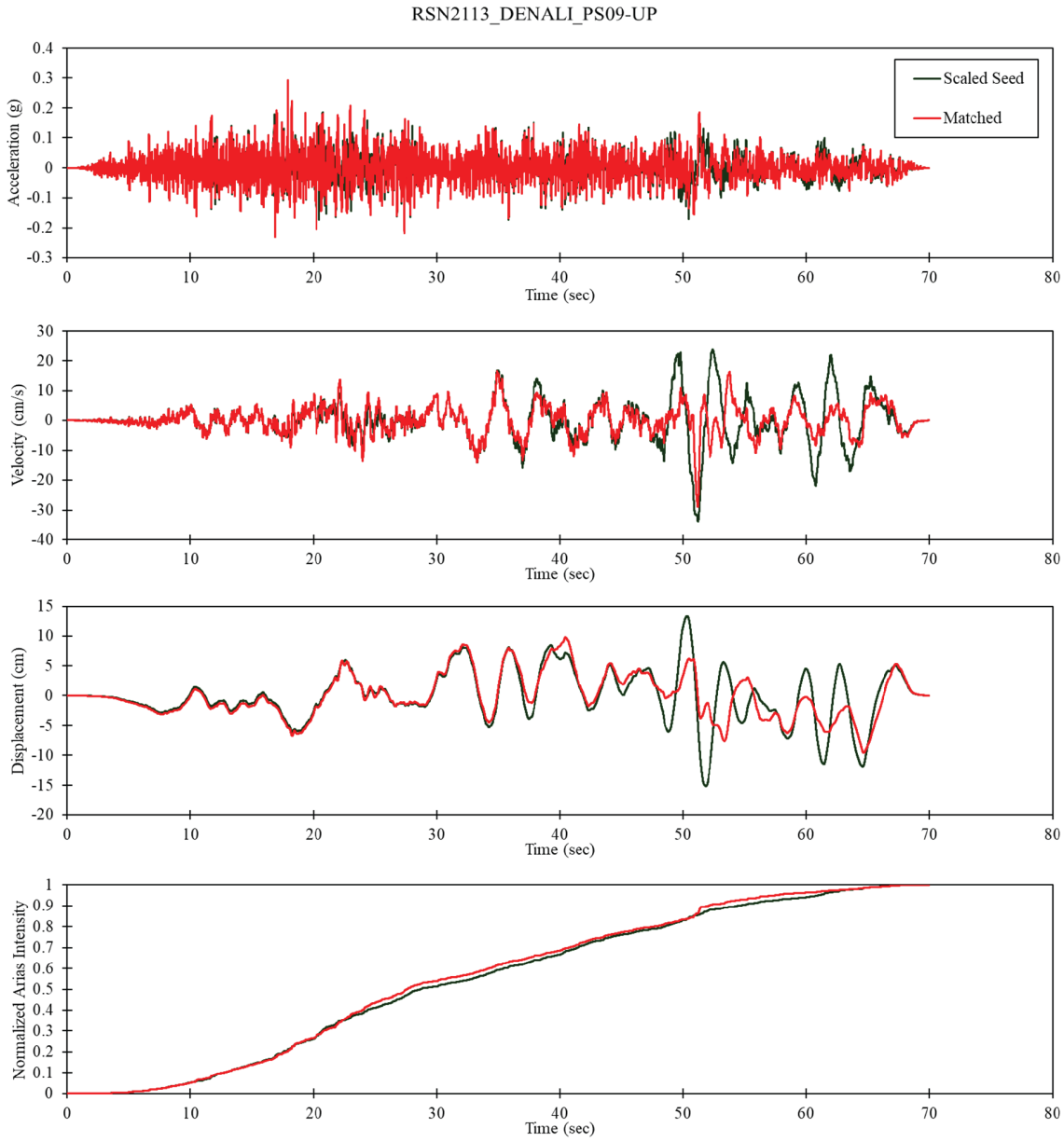


Figure 3.7-12 Acceleration, Velocity, Displacement and Normalized Arias Intensity of the Matched Vertical Time History for the Denali, Alaska Earthquake

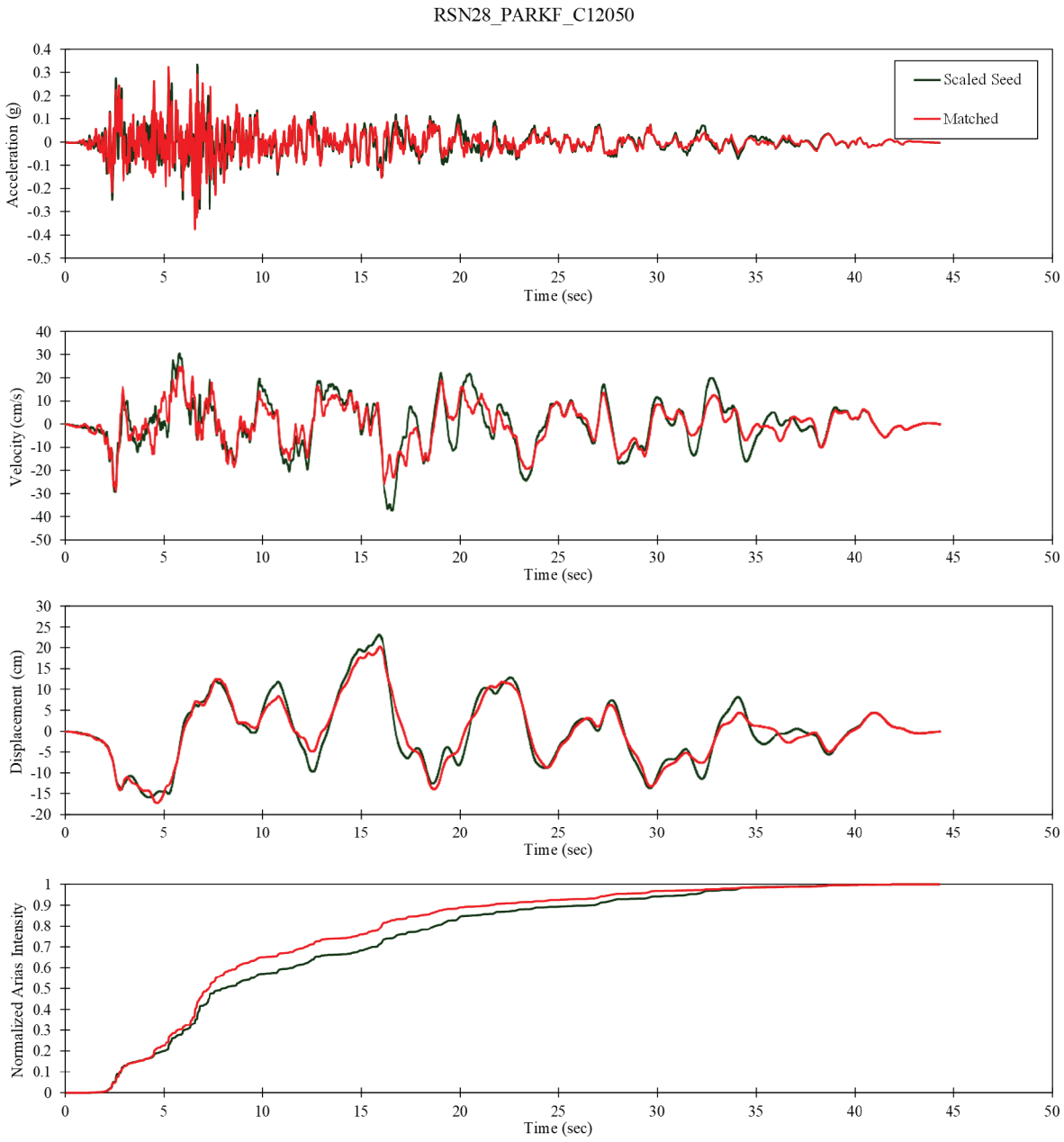


Figure 3.7-13 Acceleration, Velocity, Displacement and Normalized Arias Intensity of the Matched First Horizontal Time History for the Parkfield Earthquake, Cholame

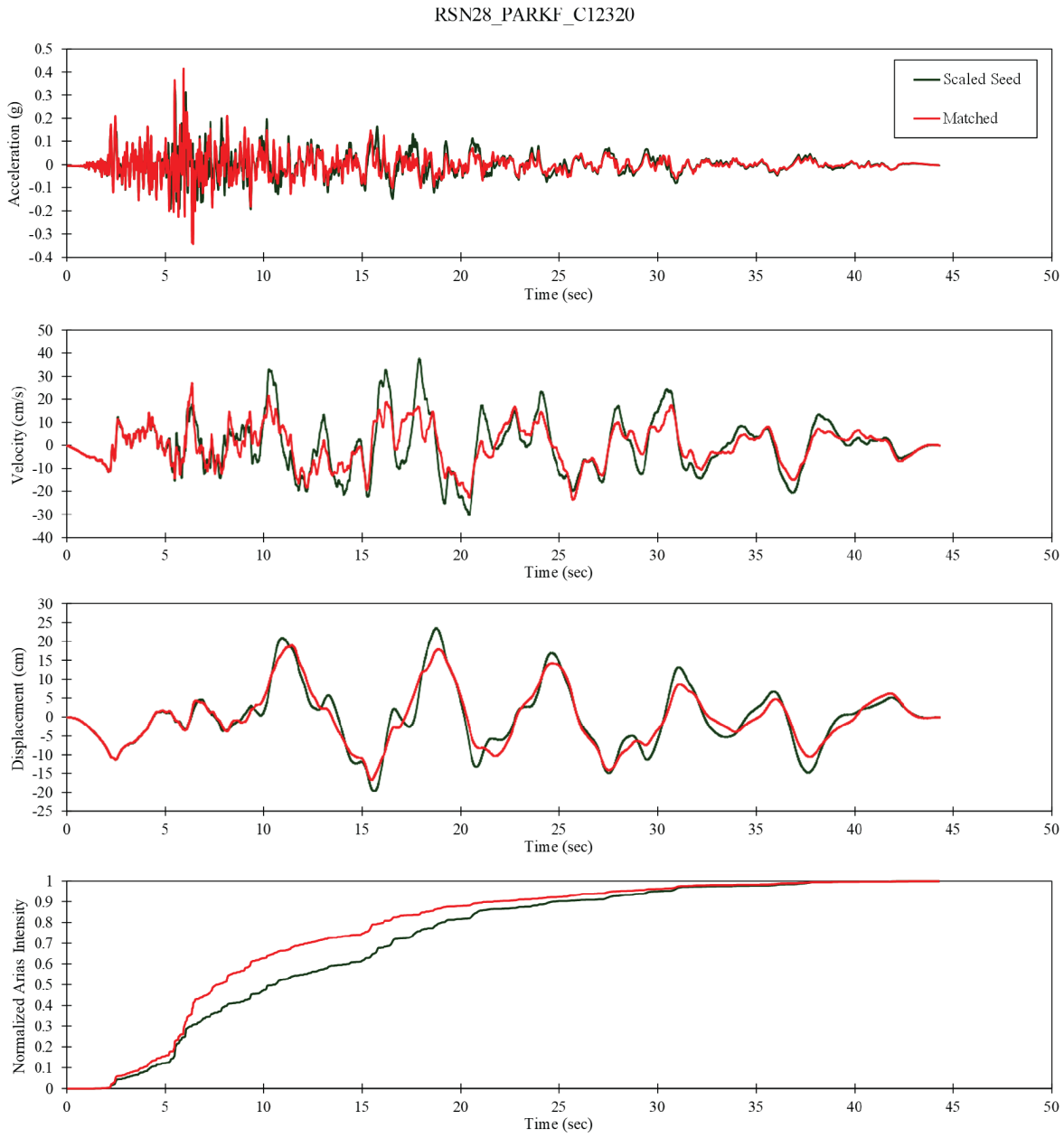


Figure 3.7-14 Acceleration, Velocity, Displacement and Normalized Arias Intensity of the Matched Second Horizontal Time History for the Parkfield Earthquake, Cholame

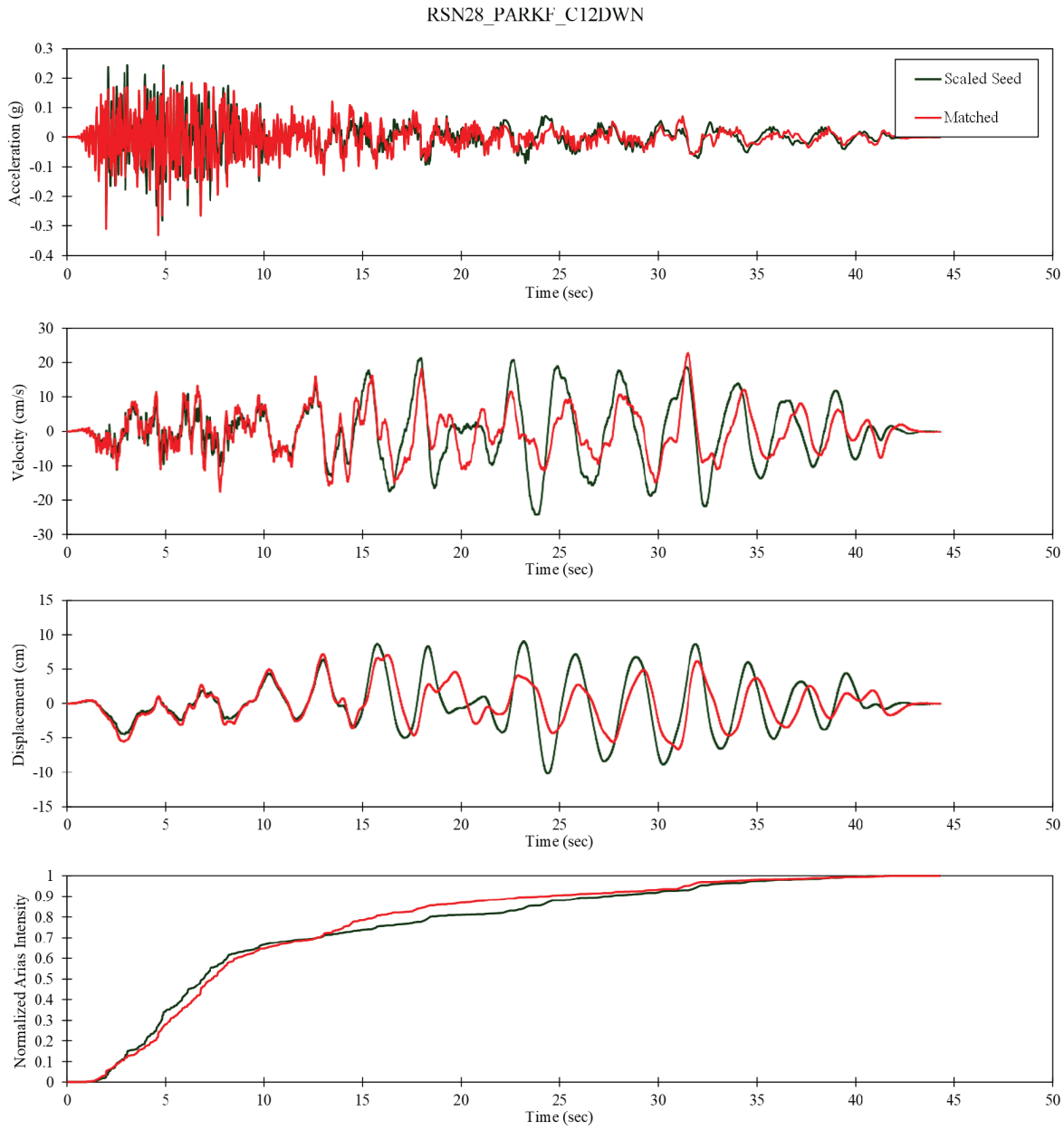


Figure 3.7-15 Acceleration, Velocity, Displacement and Normalized Arias Intensity of the Matched Vertical Time History for the Parkfield Earthquake, Cholame

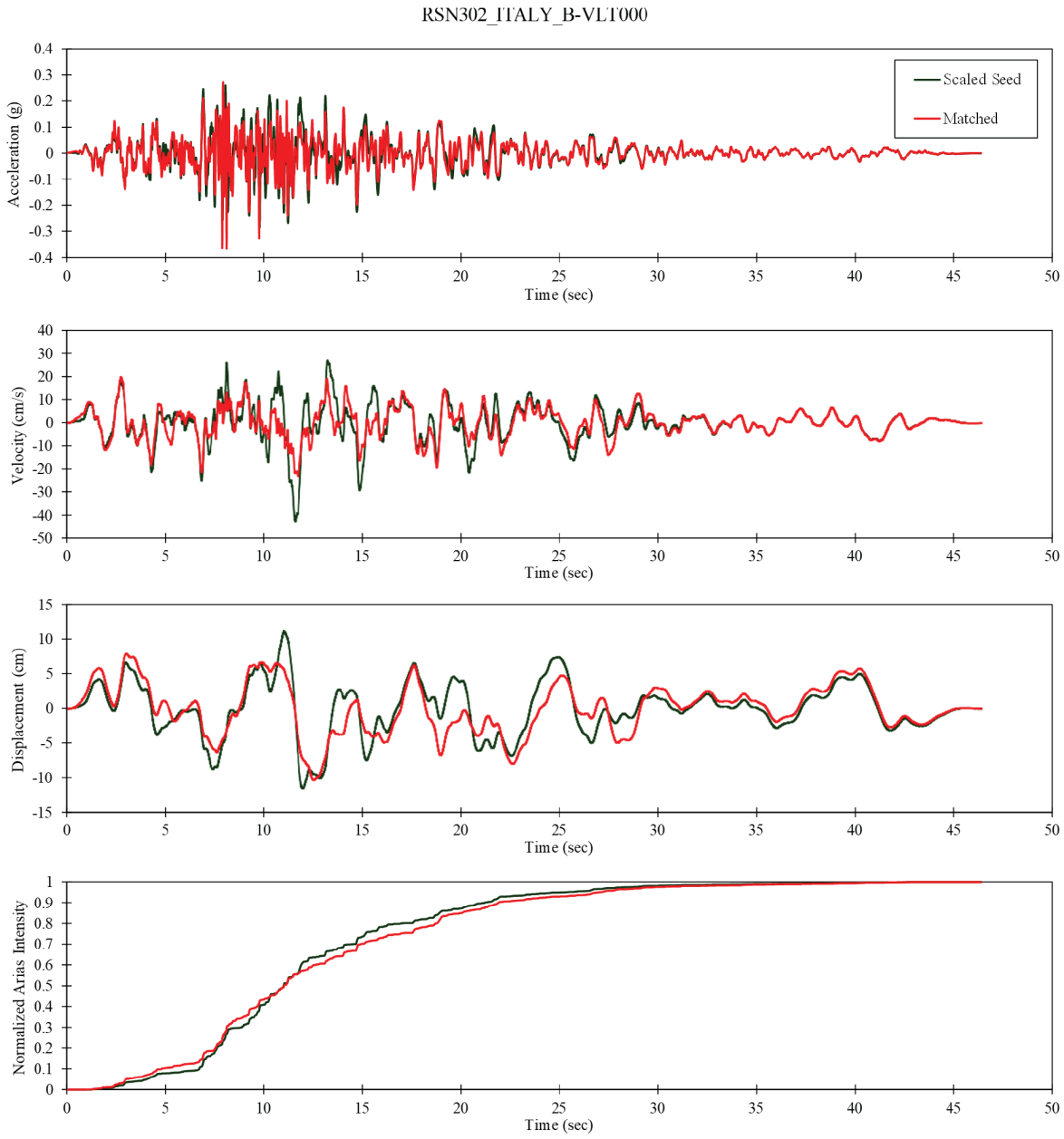


Figure 3.7-16 Acceleration, Velocity, Displacement and Normalized Arias Intensity of the Matched First Horizontal Time History for the Irpinia Earthquake

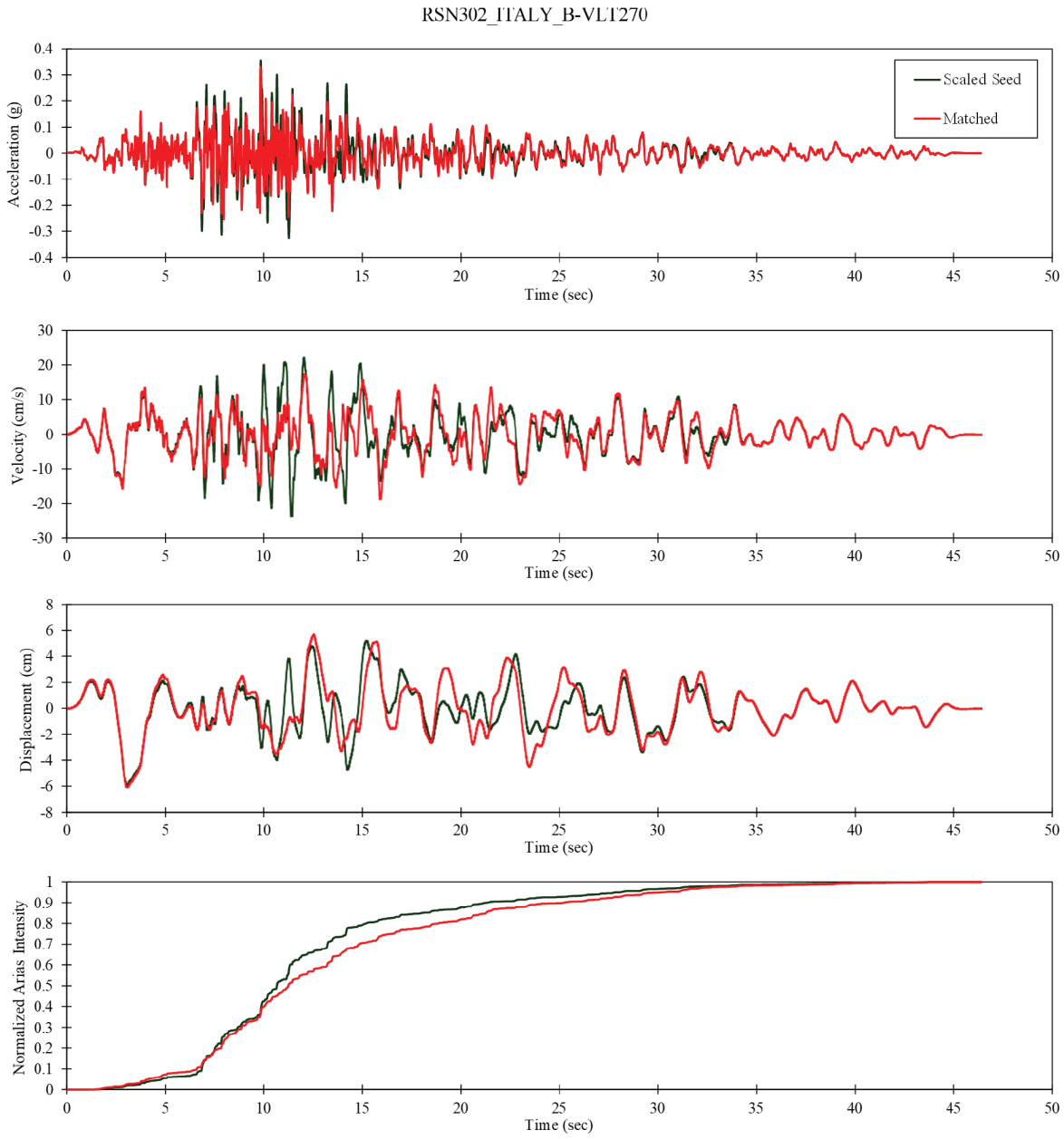


Figure 3.7-17 Acceleration, Velocity, Displacement and Normalized Arias Intensity of the Matched Second Horizontal Time History for the Irpinia Earthquake

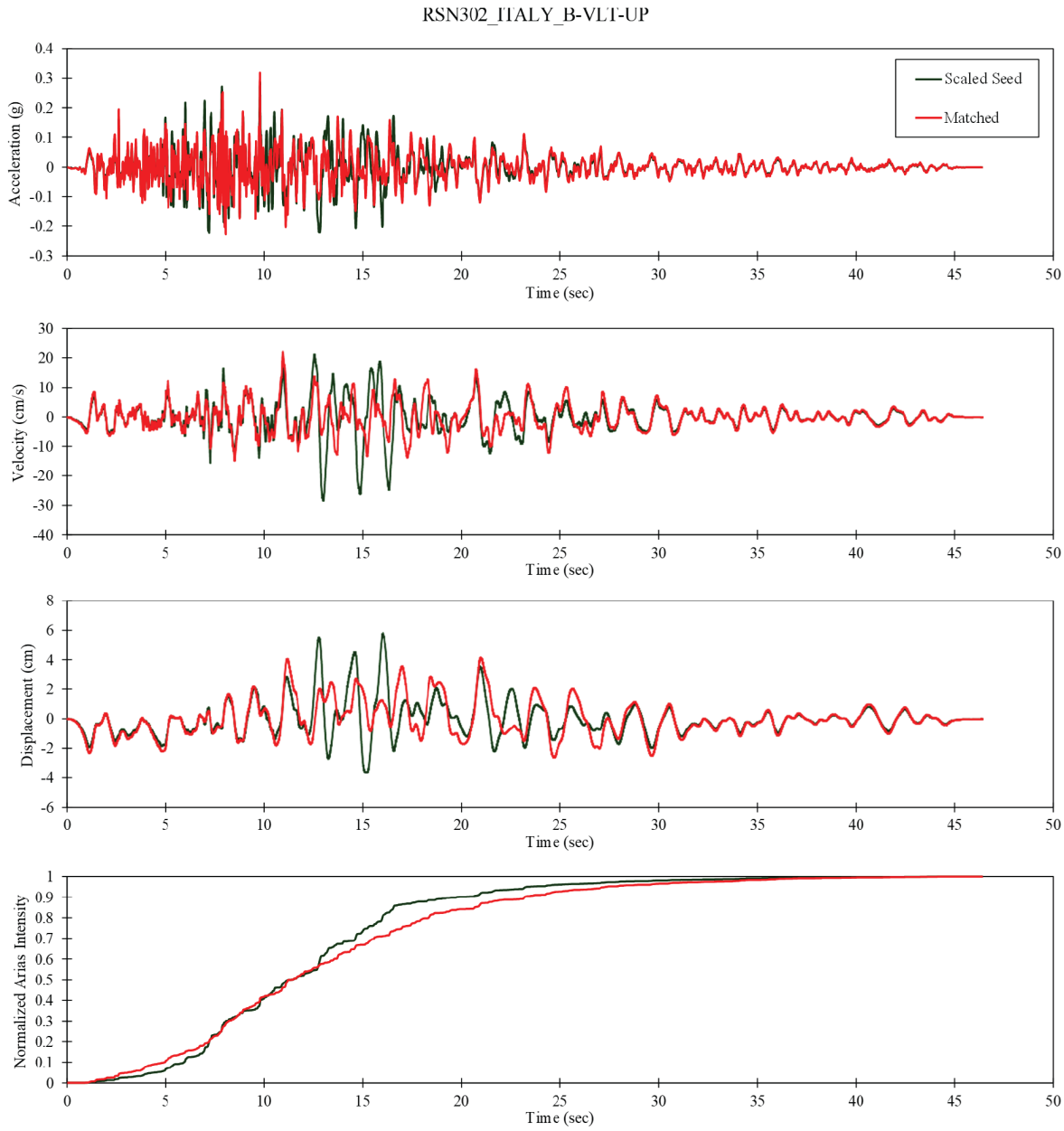


Figure 3.7-18 Acceleration, Velocity, Displacement and Normalized Arias Intensity of the Matched Vertical Time History for the Irpinia Earthquake

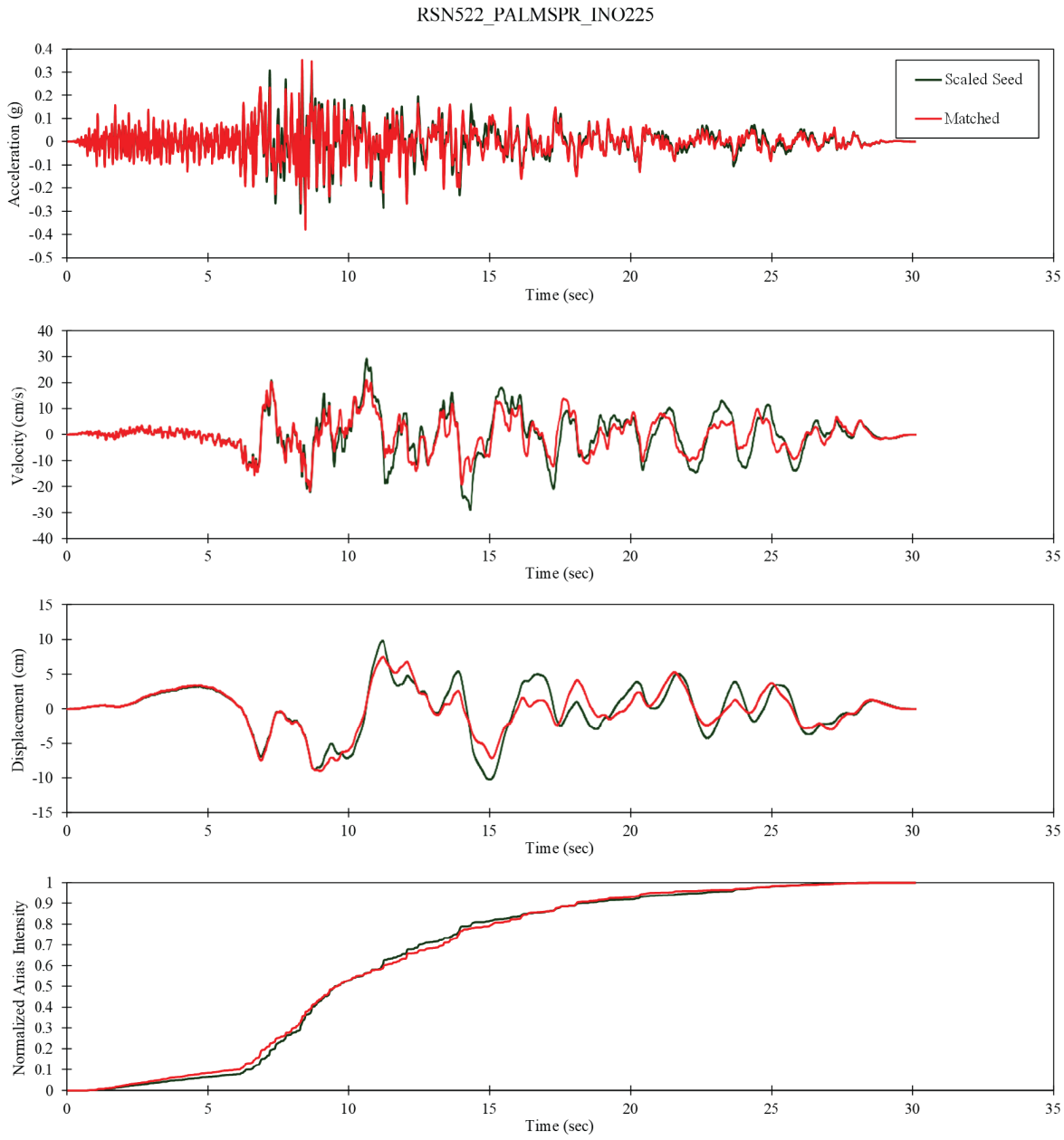


Figure 3.7-19 Acceleration, Velocity, Displacement and Normalized Arias Intensity of the Matched First Horizontal Time History for the Palm Springs Earthquake

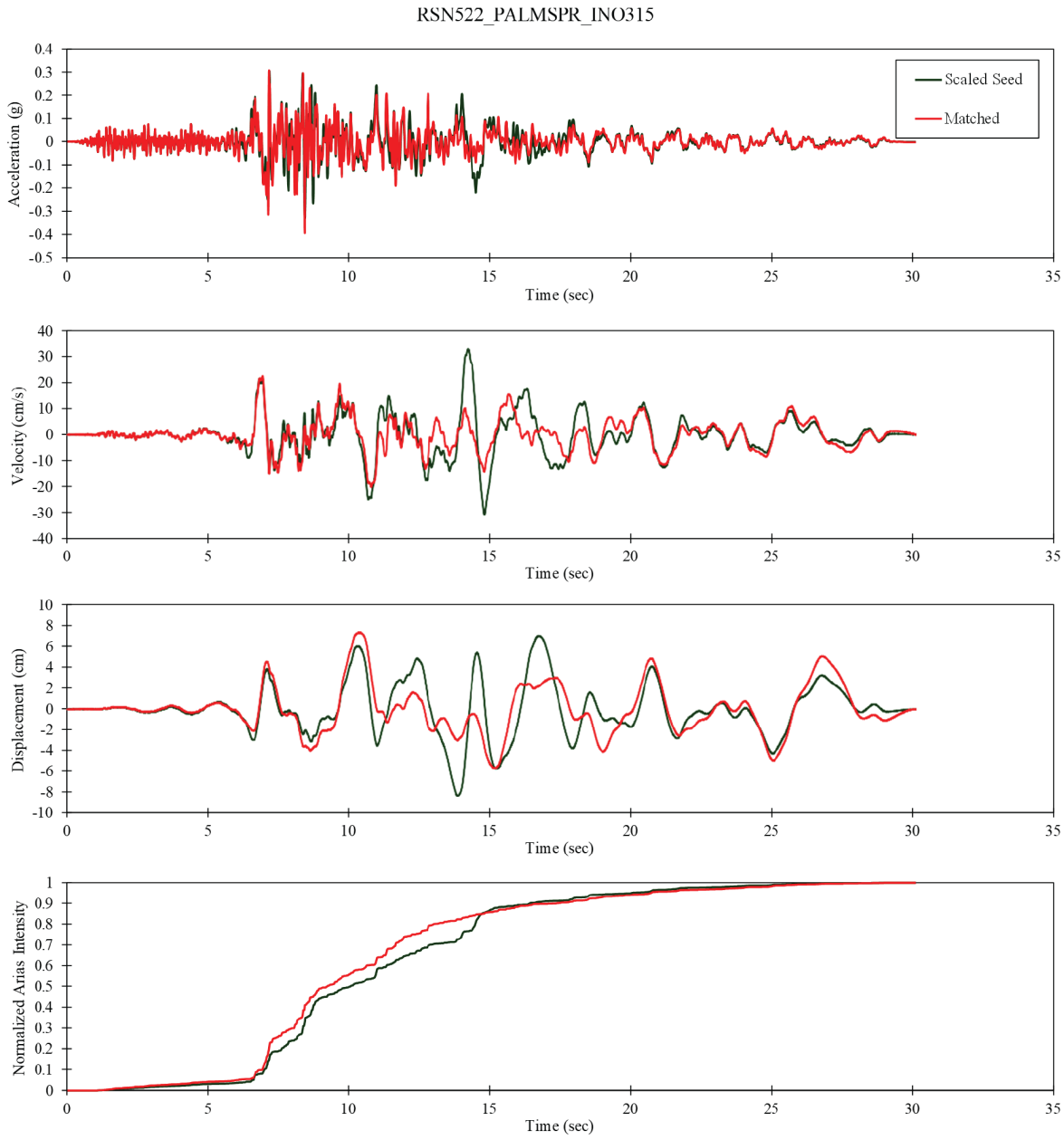


Figure 3.7-20 Acceleration, Velocity, Displacement and Normalized Arias Intensity of the Matched Second Horizontal Time History for the Palm Springs Earthquake

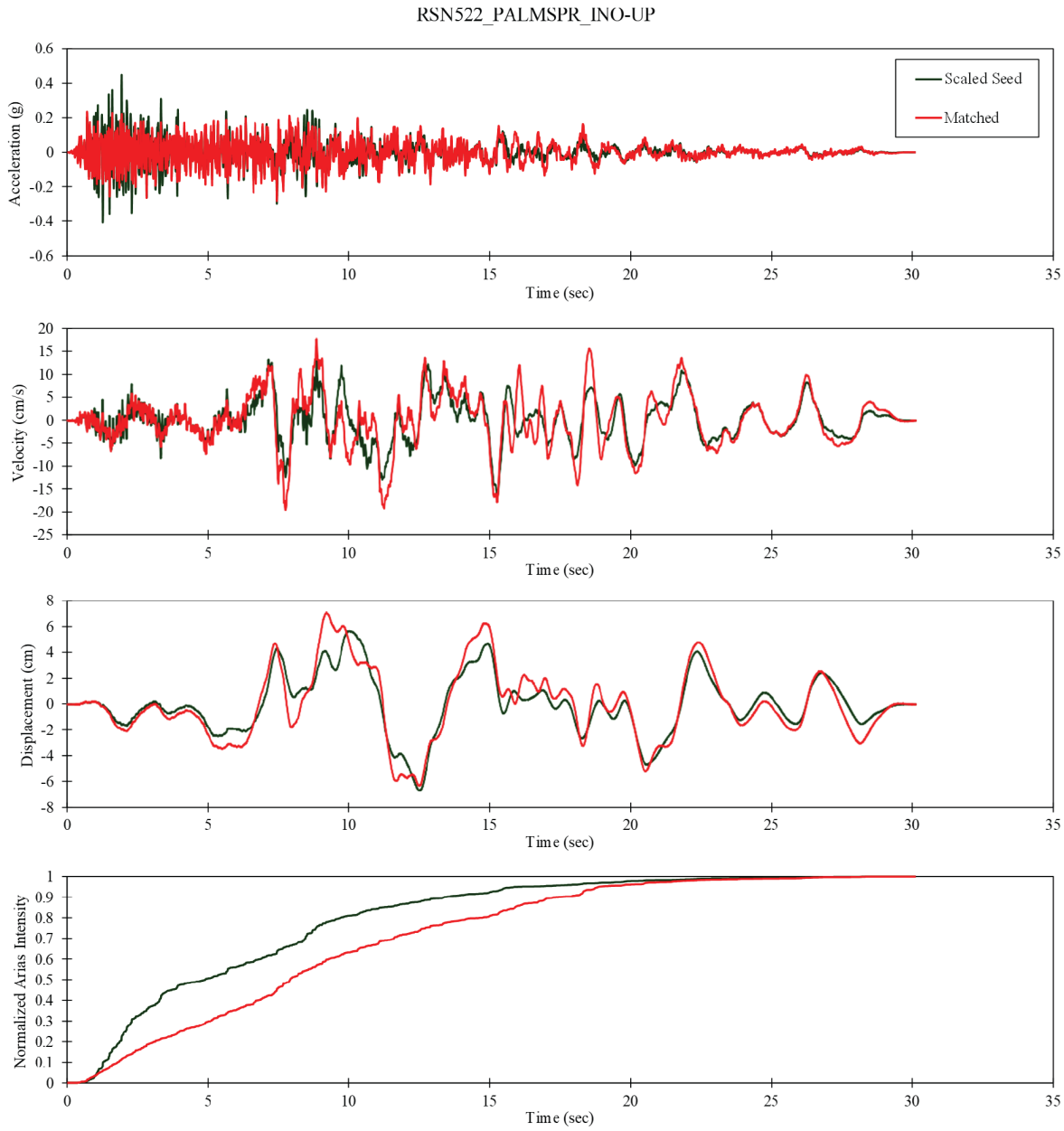


Figure 3.7-21 Acceleration, Velocity, Displacement and Normalized Arias Intensity of the Matched Vertical Time History for the Palm Springs Earthquake

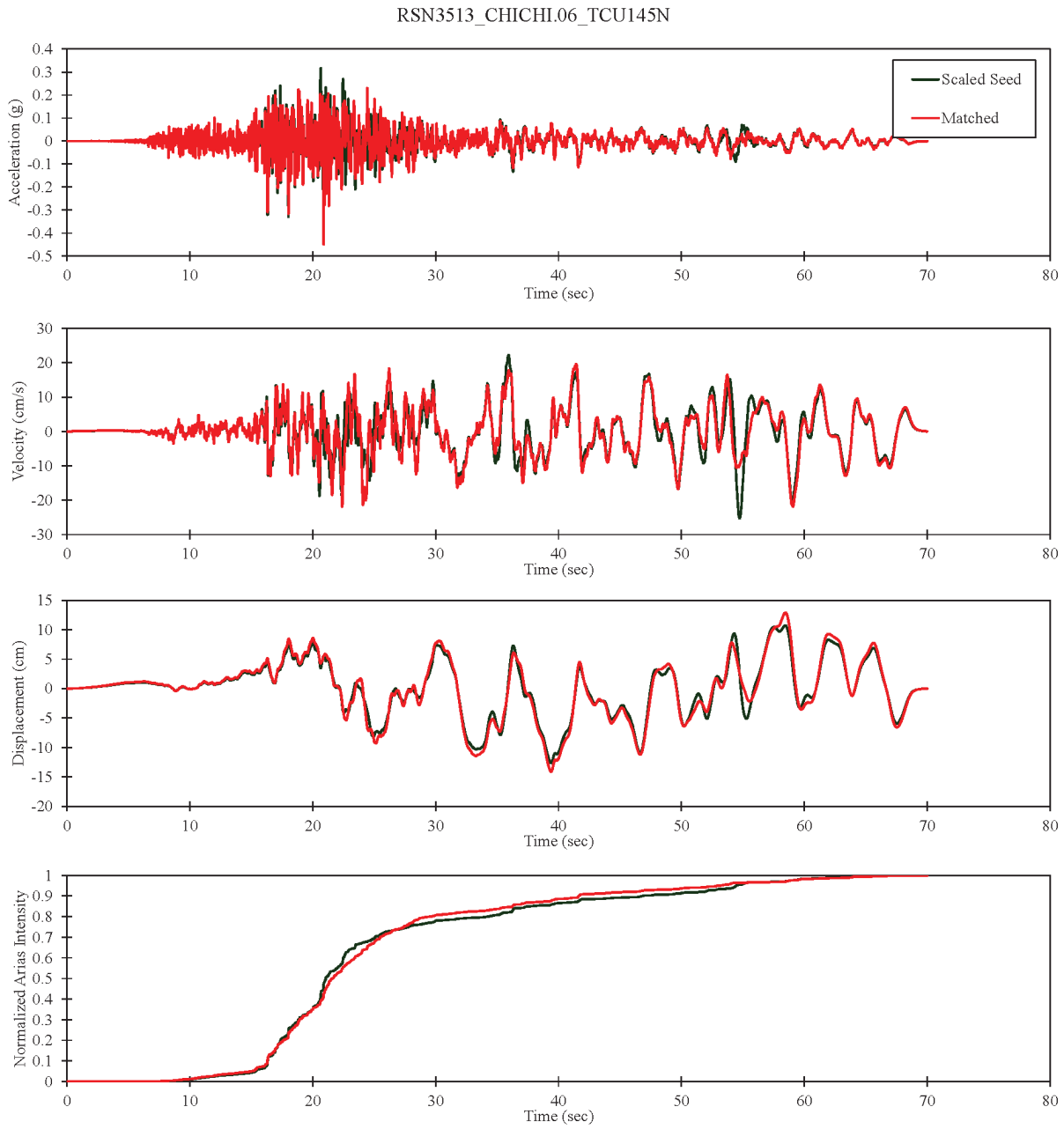


Figure 3.7-22 Acceleration, Velocity, Displacement and Normalized Arias Intensity of the Matched First Horizontal Time History for the Chi-Chi Earthquake

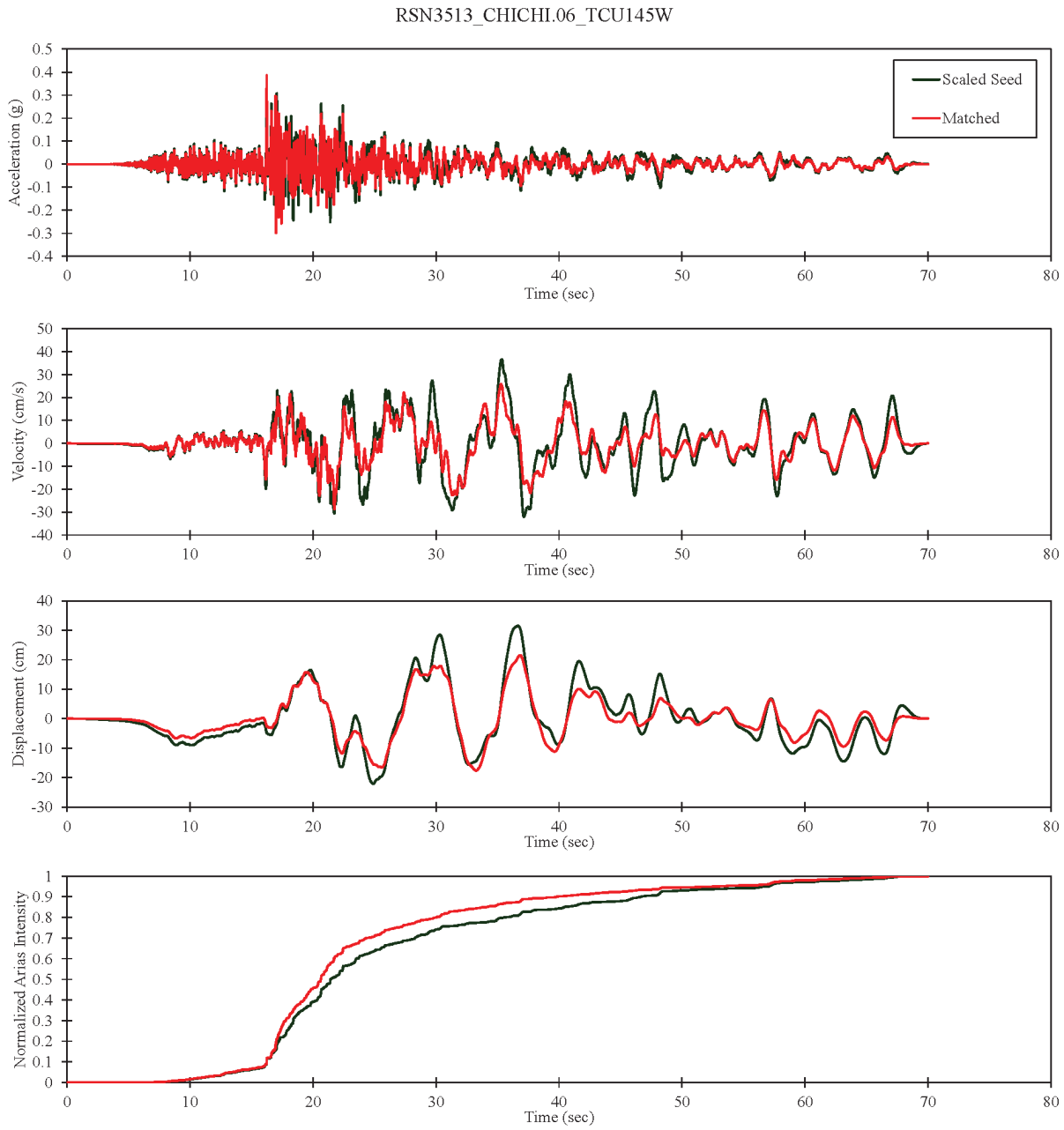


Figure 3.7-23 Acceleration, Velocity, Displacement and Normalized Arias Intensity of the Matched Second Horizontal Time History for the Chi-Chi Earthquake

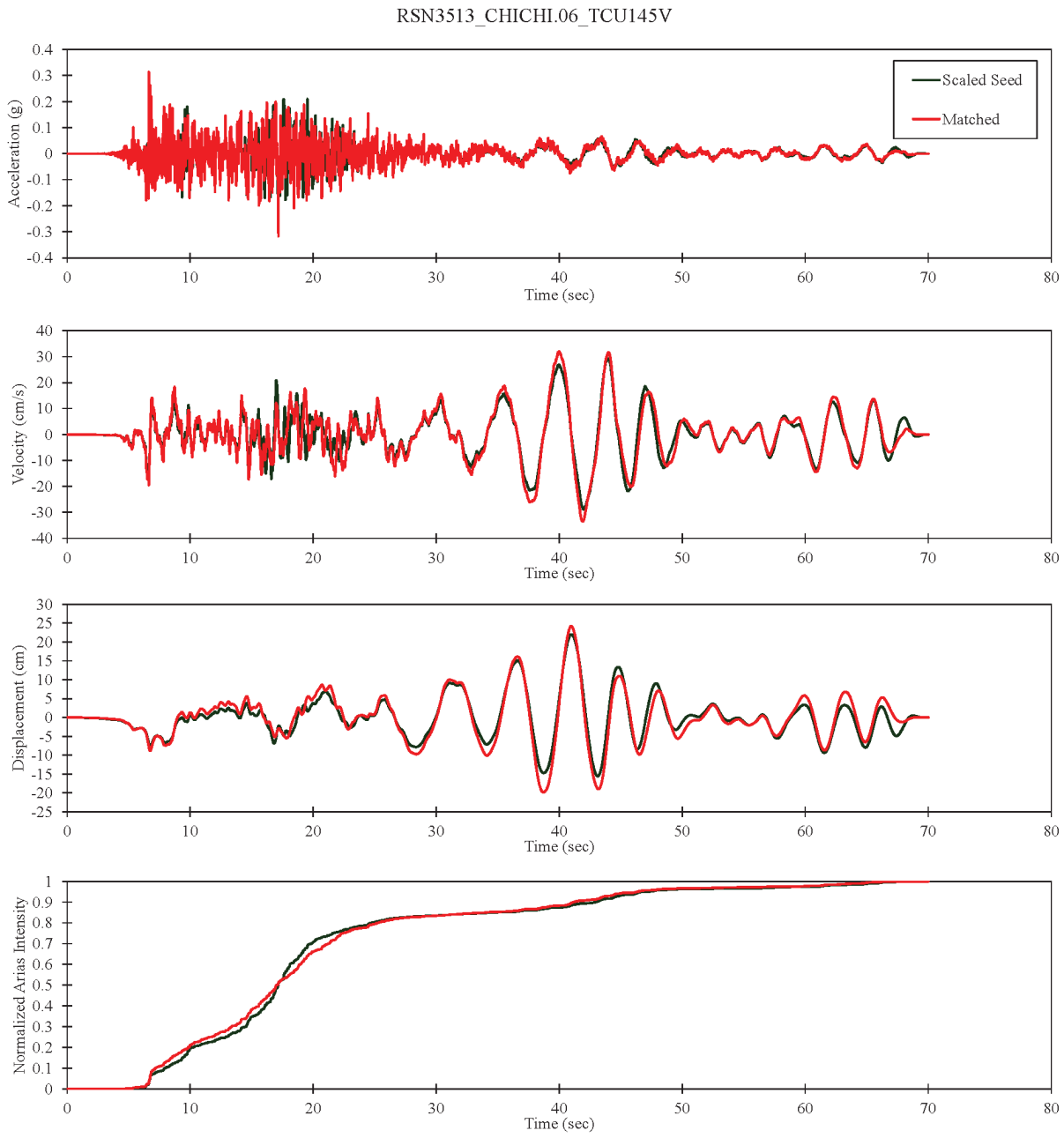


Figure 3.7-24 Acceleration, Velocity, Displacement and Normalized Arias Intensity of the Matched Vertical Time History for the Chi-Chi Earthquake

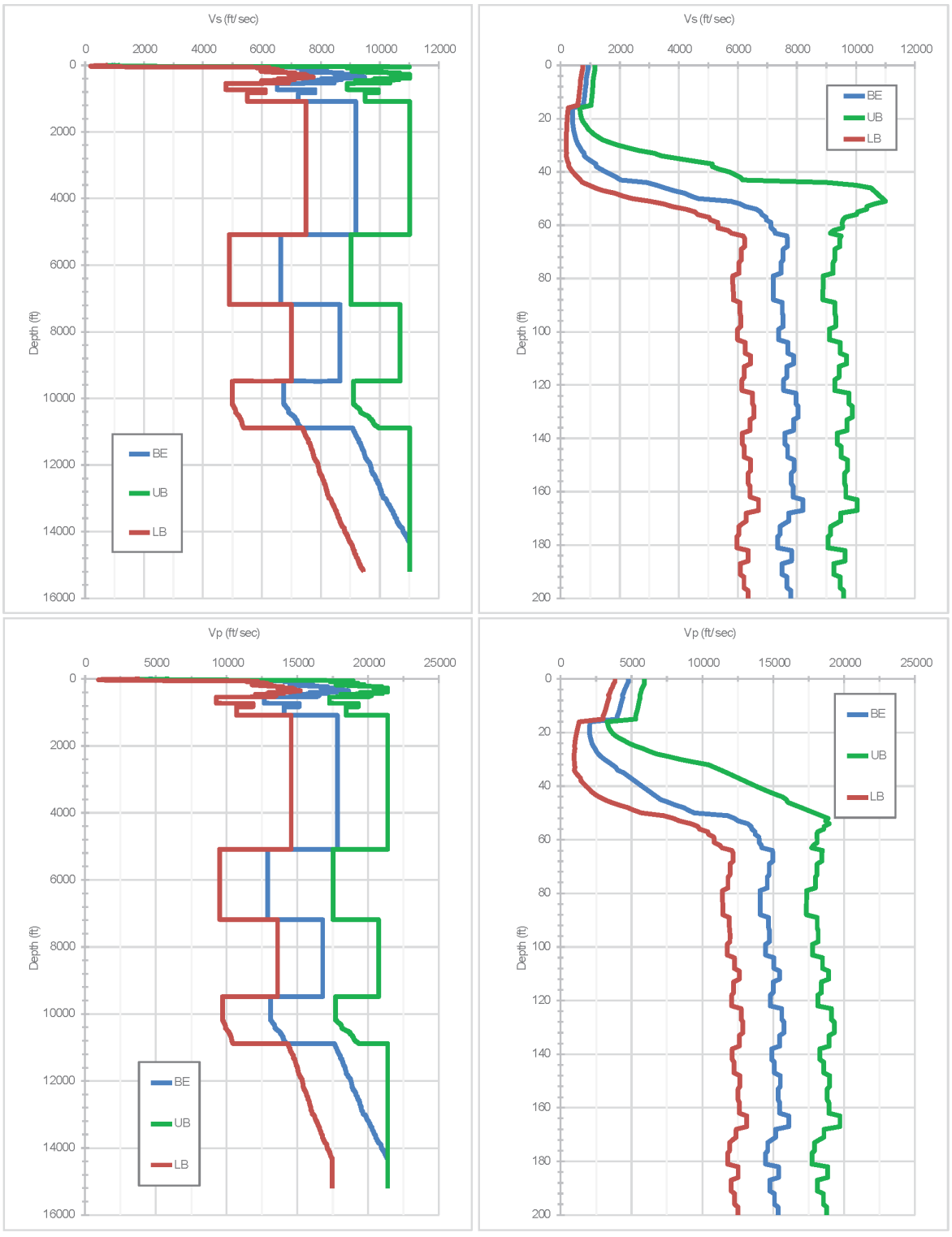


Figure 3.7-25 Hazard Consistent Strain-Compatible Shear Wave Velocity (V_s) and Compression (P)-Wave Velocity (V_p) Profiles

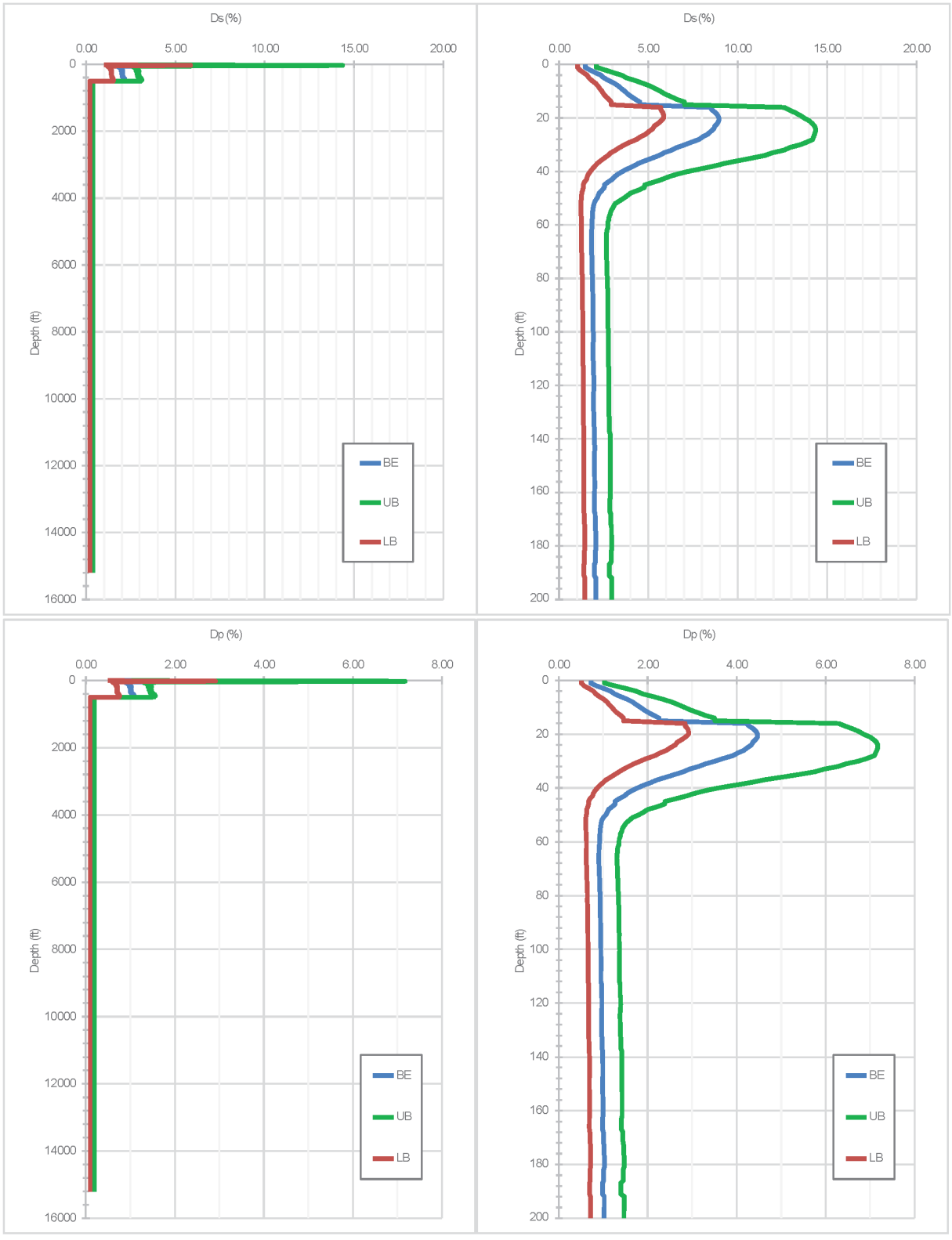


Figure 3.7-26 Hazard Consistent Strain-Compatible Damping Ratios

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 internal structures 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 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 plates 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 USNRC regulatory requirements and guidance is 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:

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, 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 LOCAs as discussed in Subsection 3.8.1.3.

GDC 16 - The SCCV, including the containment closure head, access openings 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 Subsection 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) 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 Subsection 3.8.1.4 and Subsection 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 Subsection 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 are 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

SRP 3.8.1, “Concrete Containment,” and 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, access openings 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 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

1. 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 Clinch River Nuclear (CRN) 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 RG 1.136 and are consistent with the guidance of 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, access openings and penetrations, is determined to ensure its structural integrity and leak tightness under DEC internal pressure loads following the guidance of SRP 3.8.1 and 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 guidance of Regulatory Position 1 of 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 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 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.

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 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) 43-19, “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 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 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 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-21, “Standard Specification for Structural Steel for Bridges,” Grade HPS 70W (Reference 3.8-13), per ASME BPVC, Code Case N-763, “ASTM A709-06, 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 Type B , “Standard Specification for Steel Bar, Carbon and Alloy, Cold-Finished,” (Reference 3.8-15), 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 Grade 50, “Standard Specification for High-Strength Low-Alloy Columbium-Vanadium Structural Steel,” (Reference 3.8-16), per ASME BPVC, Code Case N-632, “Use of ASTM A572 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 face plates 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, RG 1.28, “Quality Assurance Program Criteria (Design and Construction)” and 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 RG 1.136 and 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, access openings 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, access openings 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 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, access openings 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 SRP 3.8.2 and 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 Section 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 SRP 3.8.2. OBE loads are also considered for cyclic loading considerations following the guidance of 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, access openings 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 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 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 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 penetration 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 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 SRP 3.8.2, 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 face plates 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 by 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 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 containment internal structural steel is in accordance with ANSI/AISC N690 consistent with 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 Subsection II.3 of 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 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 Subsection II.4 of 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 SRP 3.8.3 and 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 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 A500, “Standard Specification for Cold-Formed Welded and Seamless Carbon Steel Structural Tubing in Rounds and Shapes,” (Reference 3.8-20) or ASTM 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 SRP 3.8.3, 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 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. DP-SC 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 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 RG 1.243 and SRP 3.8.4, “Other Seismic Category I Structures.”

The design of the RB DP-SC walls, slabs and mat foundation 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 Subsection II.2 of SRP 3.8.4.

In addition to meeting the requirements of ANSI/AISC N690 and guidance of RG 1.243, the RB polar crane rail clips, runway girders, and support corbels are also designed to meet the requirements of ASCE/SEI 7-16, “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 7-16.

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 Subsection II.3 of 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 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 4-16, “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 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

Analysis Method

The RB, containment and the containment internal structures are analyzed as one integrated structure, using ANSYS and ACS SASSI computer programs, to determine structural design demands resulting from various design loads and design load combinations. Evidence of qualification of these computer programs, including a description of the programs and extent of use, is presented in Appendix 3I.

Four types of analyses are performed to obtain design demands for the integrated RB structures:

- 1-g static SSI analyses
- Static and quasi-static analyses
- Thermal stress analyses
- Seismic SSI analyses

These analyses are performed on integrated RB structural models that have the same node and element type configurations. The use of linear elastic models with identical FE configuration enables the demands obtained from different analysis cases to be combined on an element-by-element basis for the applicable design load combinations per governing design codes. For more information on the analysis models, analysis cases and model requirements, including boundary conditions considered, refer to Appendices 3B, 3C and 3E.

Integrated RB Finite Element Model

The integrated RB detailed structural model used in the various structural analyses is presented in Section 3B.1. This model is developed following the general FE modeling guidelines discussed in Subsection 3.7.2.3 and Section 5.1.1 of NEDO-33914-A. The model adequately represents the RB structural configuration for all main structural members, including their foundations, penetrations, and openings. The model meets the mesh refinement and quality attributes required for calculation of structural stress demands.

Materials properties assigned to the model depend on the analyzed loads and resulting stress responses. Unit weight properties are assigned to the models used for the 1-g static SSI analyses to adequately simulate gravity and earth pressure loads. Seismic mass inertia properties assigned to the dynamic model used for the seismic SSI analyses are discussed in Subsection 3.7.2.3.

Stiffness properties are assigned to the SCCV and RB to reflect effective stiffness for load combinations without accidental thermal load. For load combinations with accidental thermal load, reduced stiffness is considered to account for the cracking effects on the redistribution of forces and moments. The effective stiffnesses of DP-SC and conventional steel-plate composite structural elements and their connections are computed as discussed in Sections 5.5 and 5.6 of NEDC-33926P-A.

The effect of interaction with the surrounding subgrade is incorporated in the analyses of the deeply embedded integrated RB by considering the surrounding soil and rock as a layered half-space continuum. Elastic spring elements are used to model the interfaces of the RB and surrounding subgrade. These spring elements provide results that are used to calculate static and dynamic earth pressures. The geotechnical design parameters used as input for the static and thermal analyses are discussed in Subsection 3.8.5.4.1. Dynamic subgrade properties developed for the seismic SSI analyses of the integrated RB are discussed in Subsection 3.7.1.3 and Appendix 3C. Excavation support and backfill effects on the analysis results are evaluated as discussed in Subsection 3.7.2.9.

Integrated Reactor Building Load Path

The load path to foundation for vertical loads within the integrated RB and SCCV structures is as follows:

- Vertical loads from the RB roof slab are transferred to the roof steel beams that are supported by the RB exterior wall
- The refueling floor vertical loads are transferred to the RB exterior wall and pool walls
- Vertical loads on the pools are transferred through the pool slab to the SCCV wall, wing walls, and RB exterior wall
- Vertical loads on the RB floors are primarily transferred to SCCV wall, wing walls, and the RB exterior wall, with some of the vertical loads from the slabs transferred to the elevator and staircase shaft wall
- The crane vertical load is transferred to the crane corbels integrated into the RB exterior wall
- The RPV vertical loads and vertical loads coming from the containment equipment and piping supported by the CEPSS framing are transferred to the RPV pedestal
- Vertical loads from the support platforms inside containment are transferred to the SCCV wall and RPV pedestal
- All vertical loads on the RB exterior wall, wing walls, SCCV wall, and the RPV pedestal are transferred to the common mat foundation

The load path to foundation for lateral loads is as follows:

- The RB roof lateral loads are transferred through the roof diaphragm to the RB exterior wall
- The refueling floor lateral loads are transferred to the RB exterior wall and pool walls
- The hydrostatic and hydrodynamic fluid loads in pools are transferred to the pool walls which then transfer the loads to the slabs above or below
- Lateral loads on the RB floors are transferred to the SCCV wall, wing walls, and the RB exterior wall which, in turn transfer the loads to the common mat foundation
- The crane lateral load transverse to the crane girder is transferred to the crane corbels at both ends. The crane lateral load parallel to the crane girder is transferred to one side of the crane corbel.

- The RPV lateral load is transferred tangentially only through the top stabilizers at the top to the CEPSS, the bottom stabilizers at mid-height of the RPV pedestal, and through the RPV skirt to the top of the pedestal. The transfer of only tangential shear at the stabilizers is to allow RPV vertical and radial thermal expansion relative to the skirt while still providing lateral and rotational rigidity and load transfer.
- Lateral loads from the containment equipment and piping are transferred through the CEPSS framing to the RPV pedestal and SCCV wall. The CEPSS framing transfers lateral loads to the SCCV through tangential only supports to allow the vertical and radial relative expansion between the CEPSS and SCCV.
- Lateral loads from the RPV pedestal are primarily transferred to the SCCV (inner) mat foundation, with some of the loads being transferred through the CEPSS
- The SCCV wall and RPV pedestal transfer lateral load from the support platforms inside containment to the inner mat foundation
- The static and dynamic lateral earth pressures are mainly resisted by the below grade portion of the RB cylindrical exterior wall. Due to the Integrated RB design, some portions of these pressures are transferred indirectly to the SCCV through the wing walls and floors.

1-g Static SSI Analyses

The 1-g static SSI analyses provide demands from the Earth gravity (1-g) loads due to:

- The dead load (D) from the weight of the RB structure and equipment
- The vertical fluid loads (F) from the hydrostatic pressure on the pool slabs
- The static lateral earth pressure load (H) on the below grade RB exterior wall including surcharge lateral pressures loads applied from the surrounding Power Block structures and foundations, excluding groundwater loads

The 1-g static SSI analyses utilize a model including all Power Block structures with the subgrade. Lower Bound (LB) equivalent linear stiffness properties and Upper Bound (UB) unit weight properties assigned to the subgrade model used in the 1-g static SSI analyses are discussed in Subsection 3.8.5.4.1.

Stress demands obtained from the one-step 1-g static SSI analyses include the effects of static earth pressures simulated by the interaction of the integrated RB structural model with the subgrade FE model. The applicable overburden inertia loads from the surrounding Power Block foundations and other surcharge loads are included in the 1-g analysis since models of the Power Block structures and foundations are explicitly included in the 1-g static SSI model.

Following the guidance of Section 5.1.2 of NEDO-33914-A, the following stiffness properties are assigned to the contact springs in the models used for the 1-g static SSI analyses to provide UB lateral soil pressures on the RB below grade exterior wall:

- The contact springs in the direction normal to the RB exterior walls are assigned properties representing UB stiffness conditions at the SSI interfaces.

- The friction at the RB exterior walls is not considered by assigning very low stiffness properties to the contact springs in vertical and tangential direction.

Subgrade Modeling Assumptions for Deeply embedded RB

In accordance with Section 5.1.2 of NEDO-33914-A, the following assumptions related to the modeling of the subgrade are introduced in the 1-g static SSI analyses. These assumptions enable an efficient calculation of stress demands on the RB structure due to pressure loads from soil and rock surrounding and supporting the RB shaft:

1. The properties of the subgrade materials are represented by linear elastic constitutive models.

2. The non-linearities at soil-structure interfaces are not considered.

3. The rock mass is assumed continuous and the presence of cavities, fracture zones, joints, bedding planes, discontinuities and other weak zones is not considered.

The soil and rock strata in the 1-g static SSI models are modeled based on the principles of continuum mechanics using isotropic linear elastic properties. Possible fracture zones, joints, bedding planes, discontinuities and cavities in the rock are not explicitly included in the design SSI analyses models. Static properties assigned to the soil and rock materials are discussed in Subsection 3.8.5.4.1. Stiffness properties assigned to contact springs at the interface of the RB and the surrounding subgrade follow the guidelines of Section 5.1.2 of NEDO-33914-A.

Rock with disadvantageous fracture zones, joints, bedding planes and discontinuities is reinforced to create a more self-supporting rock mass. If needed, rock reinforcements are provided as initial ground support. The rock reinforcements and other supports provided during the excavation and construction may degrade and are inaccessible after construction. Therefore, the design addresses the rock loads remaining after the initial ground support degrades by including the potential weight of the rock in the static 1-g SSI analysis or by applying additional pressures on the RB exterior wall.

RB Design Earth Pressure Load Validation

Validations of the earth pressure loads are to be performed following the guidelines in Section 5.1.3 of NEDO-33914-A to ensure the 1-g SSI static analysis provides conservative earth pressure design demands on the deeply embedded RB structure.

Following the guidance of Section 4 of NEDO-33914-A, Foundation Interface Analysis (FIA) is performed as described in Subsection 2.5.4.13, on models capturing the non-linear constitutive behavior of soil and rock materials surrounding the RB shaft. These models employ non-linear interface modeling features capable of capturing the effects of non-linearities at the subgrade structure contact surfaces. The results of the FIA are to be used for validation of the design earth pressures following the guidance of Section 5.1.3 of NEDO-33914-A to address Limitation and Condition (L&C) 8.2 of NEDO-33914-A.

Static and Quasi-Static Analyses

The static and quasi-static analyses provide stress demands due to:

- Live loads
- Crane loads
- SIT and accident condition containment internal pressure load including differential containment and RB sub-compartment loads
- Horizontal hydrostatic pressure loads on pool walls
- Groundwater pressure loads on the integrated RB mat foundation and below-ground exterior wall
- Extreme wind loads on RB roof and exterior wall
- Rain and snow loads
- Seismic water sloshing and breathing mode quasi-static pressure loads on pool walls
- Quasi-static pressure loads due to high-energy line break (jet impingement, blast loads)
- Equipment and pipe reaction loads including RPV reaction loads
- Post-accident internal flooding loads

The analyses of global static and quasi-static loads that can affect the global response of the integrated RB consider the effect of subgrade stiffness. Following the sub-structuring methodology, design demands from these loads are obtained from subgrade stiffness impedance analyses performed on models consisting of two parts:

- Super-element representing LB stiffness of the subgrade surrounding the RB
- Integrated FE model of the RB, containment, and containment internal structures

The super-elements define the stiffness of the subgrade at the nodes of the RB interfaces with the surrounding soil. The stiffness properties of the super-elements are developed using a layered 3D solid FE model. Subgrade stiffness properties assigned to the super-elements are described in Subsection 3.8.5.4.1. To adequately simulate half-space boundary conditions, the depth of these models is deeper than three times the largest foundation dimension, or two times the depth of the RB embedment. The horizontal extent of these models is more than three times the RB shaft diameter.

The nodes of the super-element are coincident with the nodes of the integrated RB FE structural model. The coincident super-element and structural model nodes are connected by contact spring elements. LB stiffness properties are assigned to these contact spring elements to yield larger structural deformations and conservative design stress demands.

Fixed bases analyses are performed for the local loads with smaller magnitudes that do not affect the Integrated RB mat foundation or global response.

Demands due to hydrostatic lateral pressure loads are obtained from static analyses of the integrated RB model with vertical supports applied to all mat foundation nodes. Demands from the upward buoyant pressures on the mat foundation are obtained from a static analysis of the

integrated RB structural model with vertical supports at the nodes connecting the RB exterior wall with the mat foundation and horizontal supports established at the central node of the mat. The results from the two groundwater load analyses are combined with the results of the 1-g SSI analysis cases to obtain earth pressure and groundwater load demands for the design of integrated RB structure.

Thermal Stress Analyses

The structural stress demands caused by component temperatures resulting from normal operating conditions and DBA events are calculated from analyses of the integrated RB FE structural model including the subgrade super-element.

Equivalent stiffness properties are assigned to the steel-plate composite shell elements to account for the stiffness reduction effects under normal operating and DBA temperatures.

Static subgrade properties representing both UB and LB stiffness are considered for the super-element modeling the subgrade. The variation of subgrade modeling representing UB and LB properties is considered and enveloped in order to develop conservative thermal internal force demands for the design of the RB and containment structures. Equivalent linear subgrade stiffness properties assigned for the thermal stress analyses are discussed in Subsection 3.8.5.4.1.

Seismic SSI Analyses

Design internal force effects caused by SSE seismic demands and dynamic subgrade pressures are obtained directly from the results of the SSI seismic analyses performed using the one-step approach discussed in Subsection 3.7.2.1. Dynamic subgrade properties used as input in the seismic SSI analysis are discussed in Subsection 3.7.1.3 and Appendix 3C.

Design Method

The design of the RB DP-SC structures conforms to the requirements of Section 5.0 of NEDC-33926P-A and meets the acceptance criteria discussed in Subsection 3.8.4.1.5. Conventional steel-plate composite structures are designed per the requirements of ANSI/AISC N690 as supplemented by RG 1.243. Steel-plate composite connections in the RB are designed per the requirements of Section 5.11 of NEDC-33926P-A and are further discussed in Appendix 3B and Appendix 3H. The design of the RB steel structures is per ANSI/AISC N690 and meets the acceptance criteria discussed in Subsection 3.8.4.1.5.

Membrane forces, shear forces and bending moments used in the design of the RB structure are obtained from the linear elastic SSI analyses for the integrated RB FE model for the loads and load combinations defined in Subsection 3.8.4.1.3.

Results from the FE analyses are evaluated to identify critical cross-sections where maximum structural demands occur for different controlling loads and load combinations. Key responses reviewed include:

- Vertical bending moment demands on the RB outer shaft and SCCV wall, at base of walls and at cross sections near the plant grade elevation
- Out-of-plane bending demands for major floor slabs and the RB foundation mat at mid-span and near the slab supports
- In-plane shear demands on major interior walls
- Bending moment demands on roof girders

Appendix 3H presents design demands for critical RB sections identified per the criteria above. The structural demands at the critical locations are used to evaluate the design of the critical cross-sections and connections using the applicable codes of record.

The integrated RB curved walls are designed and detailed to have a radius-to-wall panel thickness greater than 2.0 which minimizes the effects of curvature on the section behavior, as discussed in Section 5.12, and in compliance with L&C 8.11, of NEDC-33926P-A. The steel faceplates of DP-SC modules are designed to be non-slender as referenced in Section 5.3 of NEDC-33926P-A. As a result, the faceplates can accommodate the residual stress distribution due to rolling of materials during fabrication of curved sections without compromising their membrane stress capacity in both tensions and compression. Any residual stresses and strains resulting from rolling of the curved plates are evaluated and incorporated into detailed design.

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

In accordance with Table 3.7-1, the seismic design of the RB structure considers LS-D response in accordance with ASCE/SEI 43-19 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 43-19.

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 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 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 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 349-13, “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 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 349-13.

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 Grade A325 Type 1, “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).

Bolts of the following grades may be used otherwise, when deemed necessary:

- ASTM F3125 Grade A490 Type 1
- ASTM A354 Grade BD, “Standard Specification for Quenched and Tempered Alloy-Steel Bolts, Studs, and Other Externally Threaded Fasteners,” (Reference 3.8-29)
- ASTM A449 Type 1, “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)
- 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 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)
  - ASTM A516 Grade 70, “Standard Specification for Pressure Vessel Plates, Carbon Steel, for Moderate- and Lower-Temperature Service,” (Reference 3.8-33)
  - ASTM A240 Type 304L, “Standard Specification for Chromium and Chromium-Nickel Stainless-Steel Plate, Sheet, and Strip for Pressure Vessels and for General Application,” (Reference 3.8-34)
- 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 exterior wall and common mat foundation to protect them 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, RG 1.28 as required by 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 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 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 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 resting on a shallow reinforced concrete mat foundation supported by drilled shafts. 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.

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 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 RG 1.143, which establishes the appropriate 10 CFR 50 requirements, following the guidance of SRP 3.8.4. Analysis procedures for the RWB are in accordance with ASCE/SEI 4-16 and ASCE/SEI 43-19. 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 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 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 4-16 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 SRP 3.8.4 and RG 1.143. Design and detailing of the RWB structural steel is in accordance with ANSI/AISC N690 as supplemented by RG 1.243. Design and detailing of the RWB structural concrete is in accordance with ACI 349-13, as supplemented by 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 RG 1.243, and ACI 349-13 as endorsed and modified by RG 1.142, following the guidance of 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 349-13 and ANSI/AISC N690, following the guidance of SRP 3.8.4 and RG 1.143.

The RWB quality control procedures meet the guidance of RG 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” 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 CB is founded on a near surface reinforced concrete mat foundation supported by drilled shafts.

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 Subsection II.2 of SRP 3.7.2 and 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 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

This section presents general design rules for the DP-SC mat foundation supporting the integrated RB. Adjacent Power Block structures’ foundations are also discussed in this section due to their importance to the stability and structural integrity of the integrated RB.

3.8.5.1 Description of the Foundations

3.8.5.1.1 RB and Containment Common Mat Foundation

The integrated RB structure is founded on a circular mat foundation that supports the containment, containment internal structures and RB. The mat foundation is made of DP-SC modules and is deeply embedded in the rock. The mat foundation is a Seismic Category I structure, same as the superstructures it supports. Figure 3.8-8 provides a plan view of the mat foundation, while Figure 3.8-1 provides an elevation sectional view of the foundation. Foundation dimensions and embedment depth are provided in Figure 3.8-8. As stated in Subsection 3.8.1.2.1 and shown in Figure 3.8-2 and Figure 3.8-8, the containment boundary for the mat foundation extends to the outer perimeter of the SCCV referred to as the inner mat foundation. The mat foundation portion outside of the containment boundary is referred to as the outer mat foundation.

3.8.5.1.2 Other Power Block Foundations

The RWB, CB, Turbine Building (TB), Service Building and Reactor Auxiliary Structures are supported near the ground surface by drilled shaft deep foundations. 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 349-13. Structural design and analysis of CB, TB, Service Building and Reactor Auxiliary Structures foundations are in accordance with the IBC and ACI 318-22, "Building Code Requirements for Structural Concrete and Commentary," (Reference 3.8-37).

Subsection 2.5.4.10.1 provides the bearing capacity of the in-situ soil materials supporting the Power Block foundations. 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 is assessed in Subsection 2.5.4.8

Groundwater effects on the Power Block structures' foundations are discussed in Subsections 2.5.4.10 and 2.5.4.13.

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 SRP 3.8.5, “Foundations.”

3.8.5.4 Design and Analysis Procedures

The design and analysis procedures for the DP-SC common mat foundation conform to the guidance in Subsection II.4 of SRP 3.8.5.

As outlined in Subsection 3.8.4.1.4, the common mat foundation supports the integrated RB structures and provides the mechanism for transferring all applicable loads from the integrated RB to the surrounding subgrade.

The mat foundation is analyzed using the methods where the transfer of loads from the foundation mat to the supporting foundation media is determined by elastic methods. Demands for the design of the mat foundation are obtained from the structural analyses described in Subsection 3.8.4.1.4. These analyses are performed on the integrated RB structural model that include the effects of interaction of the structure with the surrounding subgrade and consider the surcharge loads from the foundations of the surrounding Power Block structures as discussed in Subsection 3.8.4.1.4. Subgrade modeling assumptions used in the analyses are also discussed in Subsection 3.8.4.1.4.

The DP-SC mat foundation is represented by thick shell elements in the integrated RB FE model. Properties assigned to the foundation shell elements in the dynamic FE model used for the seismic SSI analyses are discussed in Subsection 3.7.2.3.1. Properties assigned to the foundation shell elements in the integrated RB FE models used for the static and thermal stress analyses are discussed in Subsection 3.8.4.1.4. Subgrade properties used for the seismic SSI analysis of the mat foundation are discussed in Subsection 3.7.1.3 and Appendix 3C. Subgrade properties used for the static and thermal stress analyses of the mat foundation are discussed in Subsection 3.8.5.4.1.

As stated in Subsection 3.8.4.1.4, elastic springs are used to represent the stiffness properties of the foundation-subgrade interface. Vertical spring force results obtained from these spring elements serve for calculations of foundation bearing stresses. Methodology used to validate the subgrade pressure loads on the common mat foundation is described in Section 5.1.3 of NEDO-33914-A.

The containment (inner) mat foundation is designed in accordance with Section 6.0 of NEDC-33926P-A, consistent with RG 1.136. The non-pressure retaining portion of the common mat foundation (outer mat foundation) is designed to ANSI/AISC N690, supplemented by RG 1.243 and Section 5.0 of NEDC-33926P-A.

Allowable bearing capacities for the common mat foundation are addressed in Subsection 3.8.5.4.2. Foundation stability evaluations are performed using the criteria in Subsection 3.8.5.4.3 and results from the FIA discussed in Subsection 2.5.4.13.

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 existing CRN 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 are representative of at-rest lateral pressure conditions

Two stratigraphy profiles are developed to bound the geotechnical conditions across the CRN site:

- Profile A uses the lower-range estimate of the one-dimensional stratigraphy for the CRN site. This profile considers a thicker layer of existing fill and residual soil and no weathered rock above the un-weathered rock. Profile A represents a bounding lower subgrade stiffness condition where the thickness of the new fill, and existing fill/residual soils, and the depth to the un-weathered rock are maximized. This subgrade stratigraphy is considered for the analysis load cases with LB subgrades as described in Subsection 3.8.4.1.4.

- Profile B uses the upper-range estimate of the one-dimensional stratigraphy for the CRN site. This profile considers a thinner layer of weathered rock above the un-weathered rock. Profile B represents a bounding upper subgrade stiffness condition where the existing fill/residual soils are not present, weathered rock is present directly beneath the new fill, and the depth to the un-weathered rock is minimized. This subgrade stratigraphy is considered for the analysis load cases with UB subgrades as described in Subsection 3.8.4.1.4.

The equivalent linear subgrade static profiles reflect anticipated as-built conditions at the site after construction of the BWRX-300 small modular reactor and include an appropriate level of conservatism to ensure the design bounds possible changes in site parameters.

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 ratios (υ_{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:

$$\upsilon_{st} = \frac{K_0}{1 + K_0}$$

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 is evaluated in Subsection 2.5.4.10 per the guidance of SRP 2.5.4, “Stability of Subsurface Materials and Foundations.”

The bearing capacity of the rock supporting the integrated RB mat foundation at the CRN site is provided in Subsection 2.5.4.10.1.

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 Subsection II.4 of 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 43-19, 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 Subsection II.5 of 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

Materials used for the construction of the containment and RB common mat foundation are the same as those of the superstructures described in Subsections 3.8.1.6.1 and 3.8.4.1.6.

The common mat foundation DP-SC modules are designed with a sacrificial thickness to account for corrosion over the RB operating life as discussed in Subsection 3.8.4.1.6.

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 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 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,” GE Vernova Hitachi Nuclear Energy Americas, LLC, Revision 4, December 2025.

3.8-2 ASME BPVC, Section III, Division 2, “Code for Concrete Containments,” American Society of Mechanical Engineers, 2021.

3.8-3 ASME BPVC, Section II, “Materials,” American Society of Mechanical Engineers, 2021.

3.8-4 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.

3.8-5 ASME BPVC, Section V, “Non-Destructive Examination,” American Society of Mechanical Engineers, 2021.

3.8-6 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.

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.

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3.8-10 NEDO-33914-A, “BWRX-300 Advanced Civil Construction and Design Approach,” GE-Hitachi Nuclear Energy Americas, LLC, Revision 2, 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.

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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, 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,” Subsection NF, “Supports,” American Society of Mechanical Engineers, 2021.

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-13, “Standard Specification for Cold-Formed Welded Carbon Steel Hollow Structural Sections,” ASTM International, 2015.

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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.

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3.8-36 IBC, “International Building Code,” International Code Council, 2021.

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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	

(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 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.

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)

(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-3421.1of 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)

(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

(1) Based on SRP 3.8.2 and 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) As per 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 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

(1) Based on SRP 3.8.2 and RG 1.57

(2) OBE consideration only for post-flooding conditions and cyclic loading considerations, in accordance with 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 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 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]

*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, Table 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

*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 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, Table 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
(Sheet 1 of 2)

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									
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

Table 3.8-7 Load Combinations, Load Factors and Acceptance Criteria for Reactor Building Steel-Plate Composite Modules and Steel Structures
(Sheet 2 of 2)

(1) Load combinations are in accordance with ANSI/AISC N690 and 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 of ANSI/AISC N690.

Table 3.8-8 As-Built Subgrade Properties for Static 1-g Soil-Structure Interaction and Impedance Analyses

Layer	Layer Thickness	Parameters for Static 1-g SSI Analysis (LB E_{st} & UB ν_{st})			Parameters for Static Impedance Analysis (LB E_{st} & LB ν_{st})		
		Total Unit Weight	Equivalent shear wave velocity	Equivalent P- wave velocity	Total Unit Weight	LB Static Poisson Ratio	LB Elastic Modulus
		$W_{S,Total}$	V_S	V_P	$W_{S,Total}$	$\nu_{ST,LB}$	E_{st-LB}
	m	kN/m^3	m/s	m/s	kN/m^3	-	MPa
New Engineered Backfill	4.4	23.1	132	269	23.1	0.24	111
Existing Fill/Residual Soil	8.5	20.6	62	178	20.6	0.37	23
Benbolt Formation (Upper)	23.5	26.6	1,141	2,065	26.6	0.18	9,036
Benbolt Formation (Lower)	24.9	26.6	1,437	2,599	26.6	0.18	14,318
Rockdell Formation	87.5	26.6	1,064	2,161	26.6	0.12	8,230
Fleanor Member	78.3	26.7	797	1,399	26.7	0.10	4,348
Eidson Member	31.1	26.5	909	1,596	26.5	0.16	5,623
Blackford Formation	77.4	26.4	1,160	2,010	26.4	0.13	9,067
Nawala Formation	Half-space	28.0	1,917	4,358	28.0	0.16	28,946

Table 3.8-9 As-Built Static Subgrade Properties for Thermal Analysis

Layer	Layer Thickness	Total Unit Weight	UB Static Poisson Ratio	UB Elastic Modulus
		$W_{S,Total}$	$\nu_{ST,UB}$	$E_{ST,UB}$
	m	kN/m^3	-	MPa
New Engineered Backfill	4.4	23.1	0.34	325
Weathered Rock	3.7	24.0	0.34	803
Benbolt Formation (Upper)	28.3	26.6	0.28	40,181
Benbolt Formation (Lower)	24.9	26.6	0.28	40,181
Rockdell Formation	87.5	26.6	0.34	50,991
Fleanor Member	78.3	26.7	0.26	19,227
Eidson Member	31.1	26.5	0.26	39,806
Blackford Formation	77.4	26.4	0.25	28,745
Nawala Formation	Half-space	28.0	0.38	63,893

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

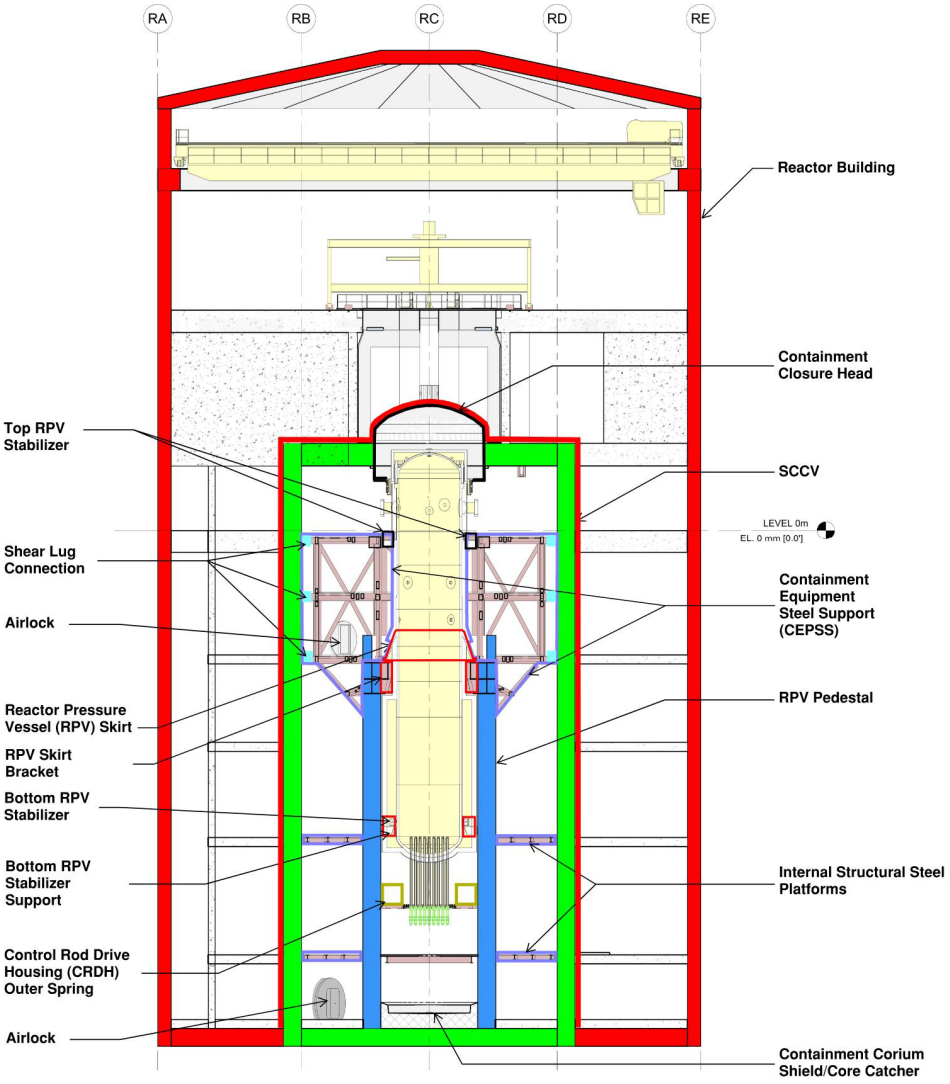
where

D = Dead Load, W = Wind

H = Lateral soil pressure, E’ = Safe Shutdown Earthquake

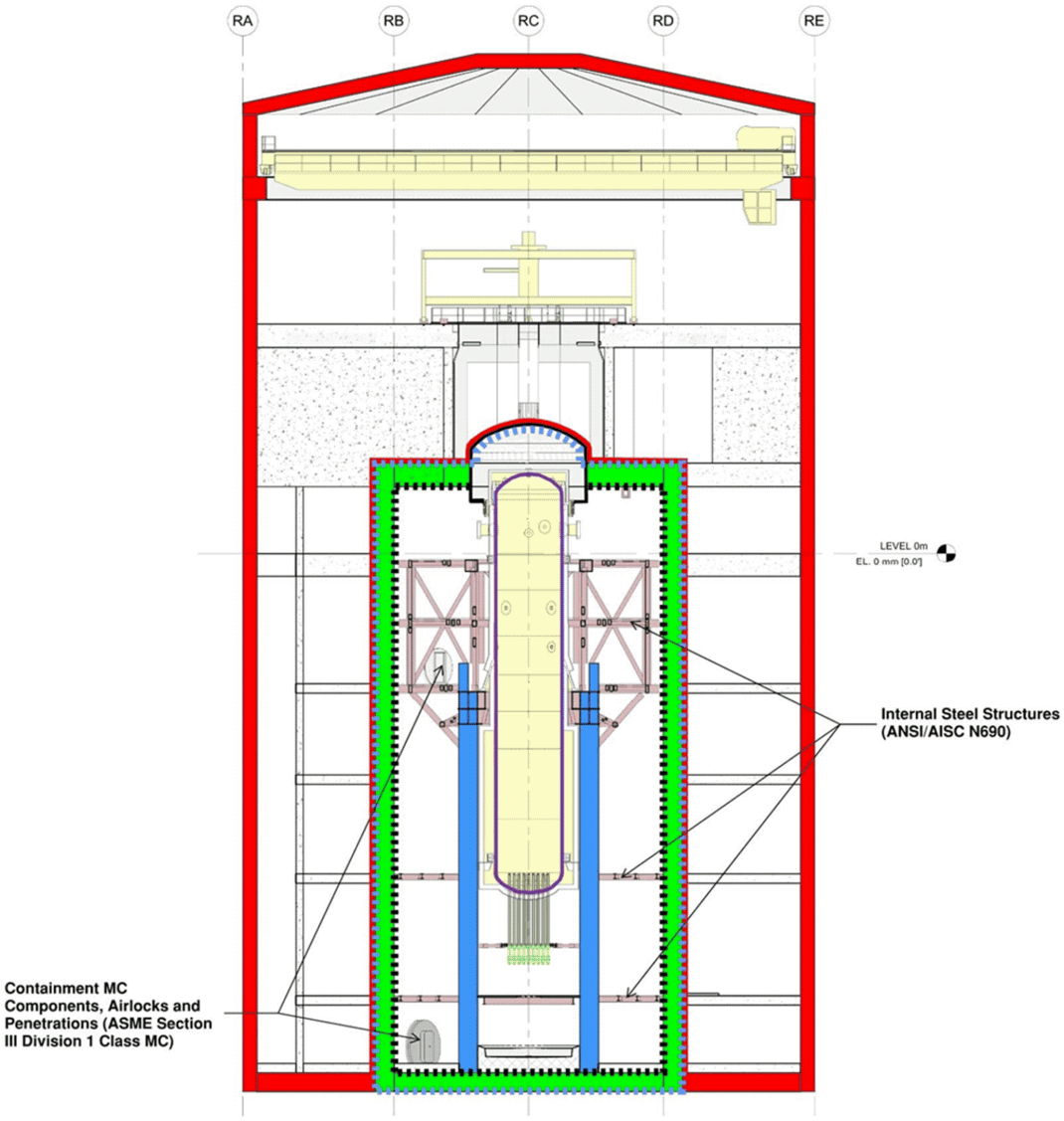
W_t = Tornado or Hurricane Wind

F’ = Buoyant forces of design basis flood



*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.

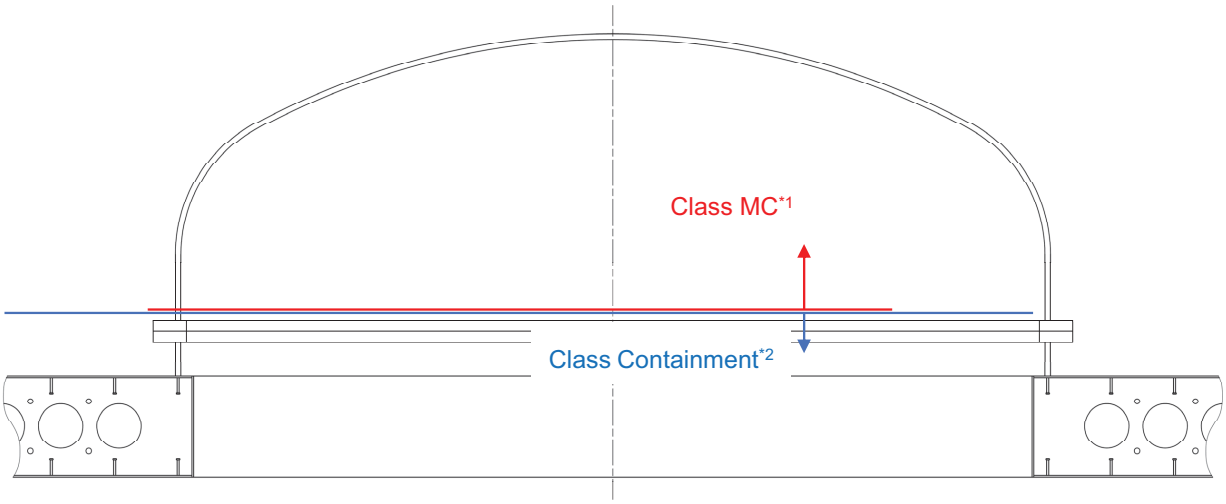
Figure 3.8-1 BWRX-300 Integrated Reactor Building Structures Including Mat Foundation



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	

*For illustration purposes only. Integrated RB layout is based on current design and may change as the design progresses

Figure 3.8-2 Structural Boundary of the BWRX-300 Containment, Containment Internal Structures and Reactor Building

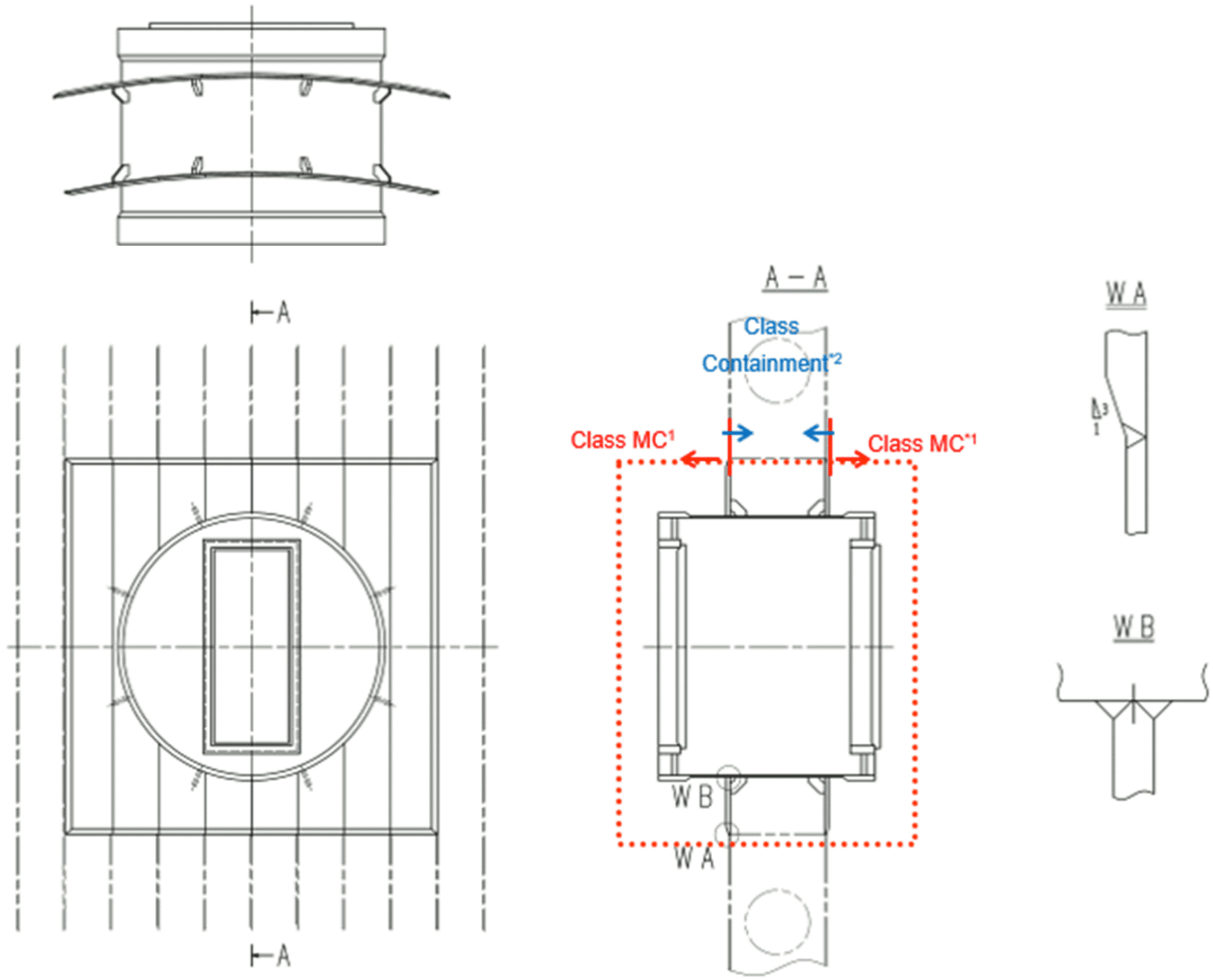


*1: Designed in accordance with ASME BPVC, Section III, Subsection NE (for Class MC)

*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 3.8-3 Containment Closure Head Structure Boundary

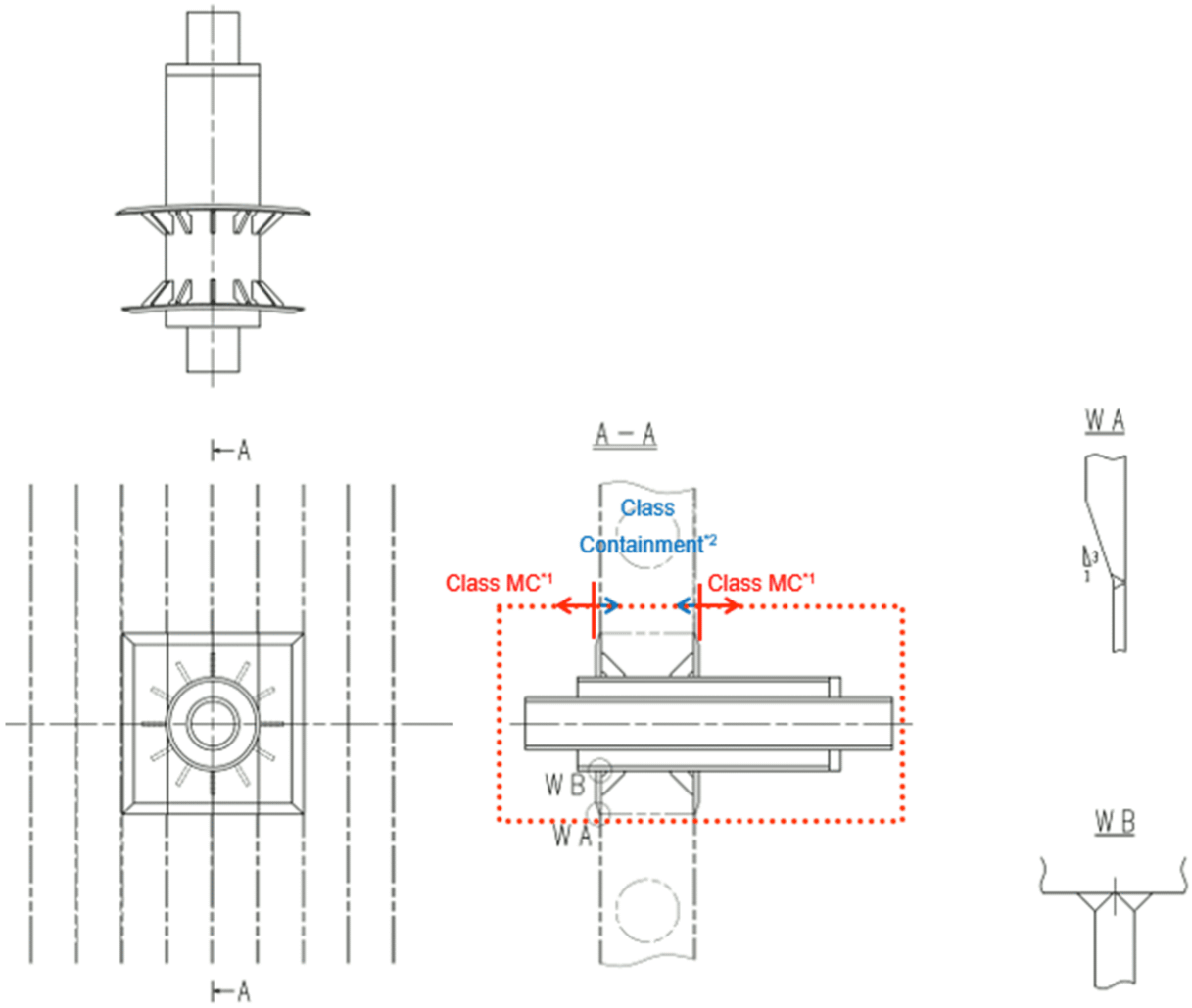


*1: Designed in accordance with ASME BPVC, Section III, Subsection NE (for Class MC)

*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

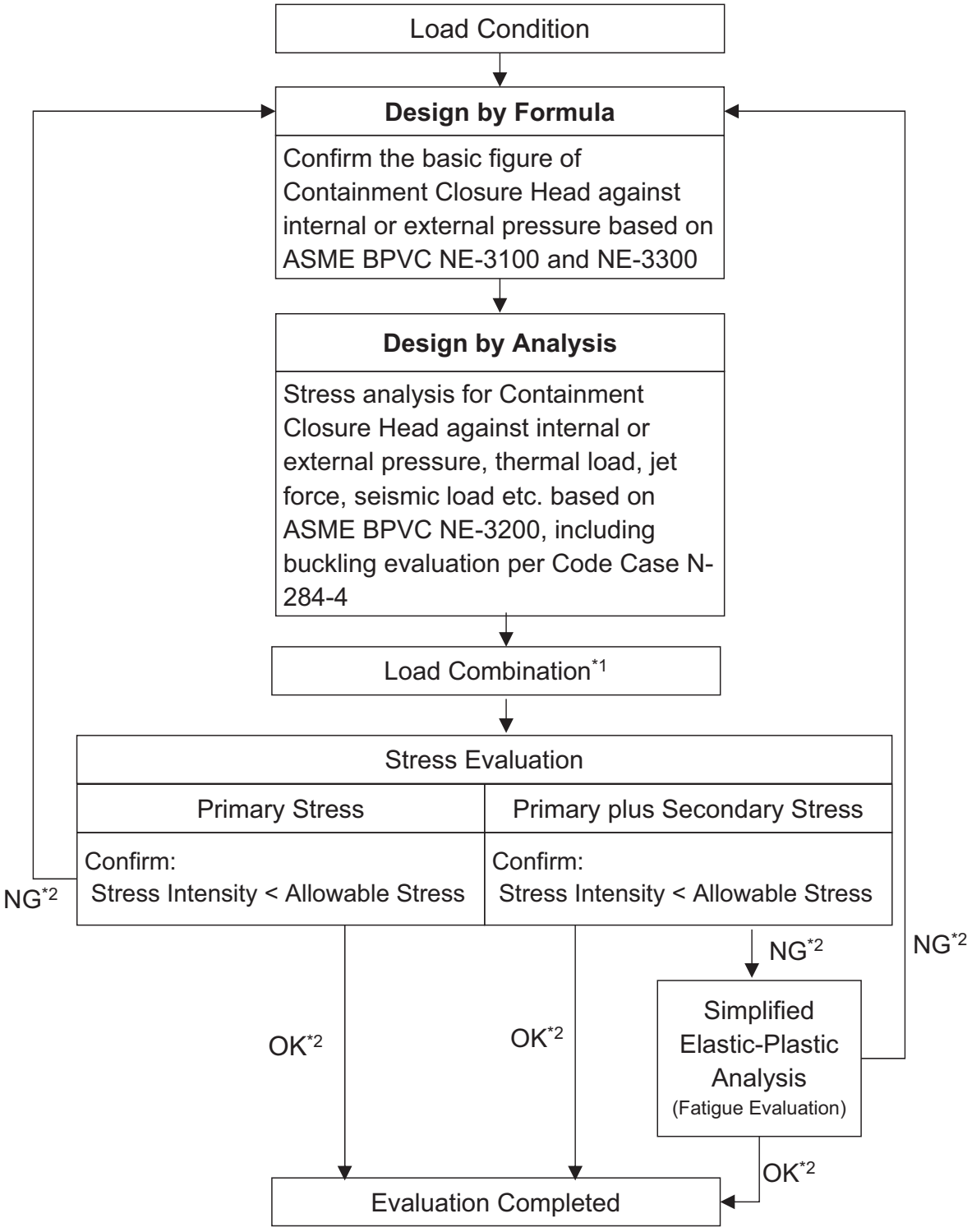


*1: Designed in accordance with ASME BPVC, Section III, Subsection NE (for Class MC)

*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

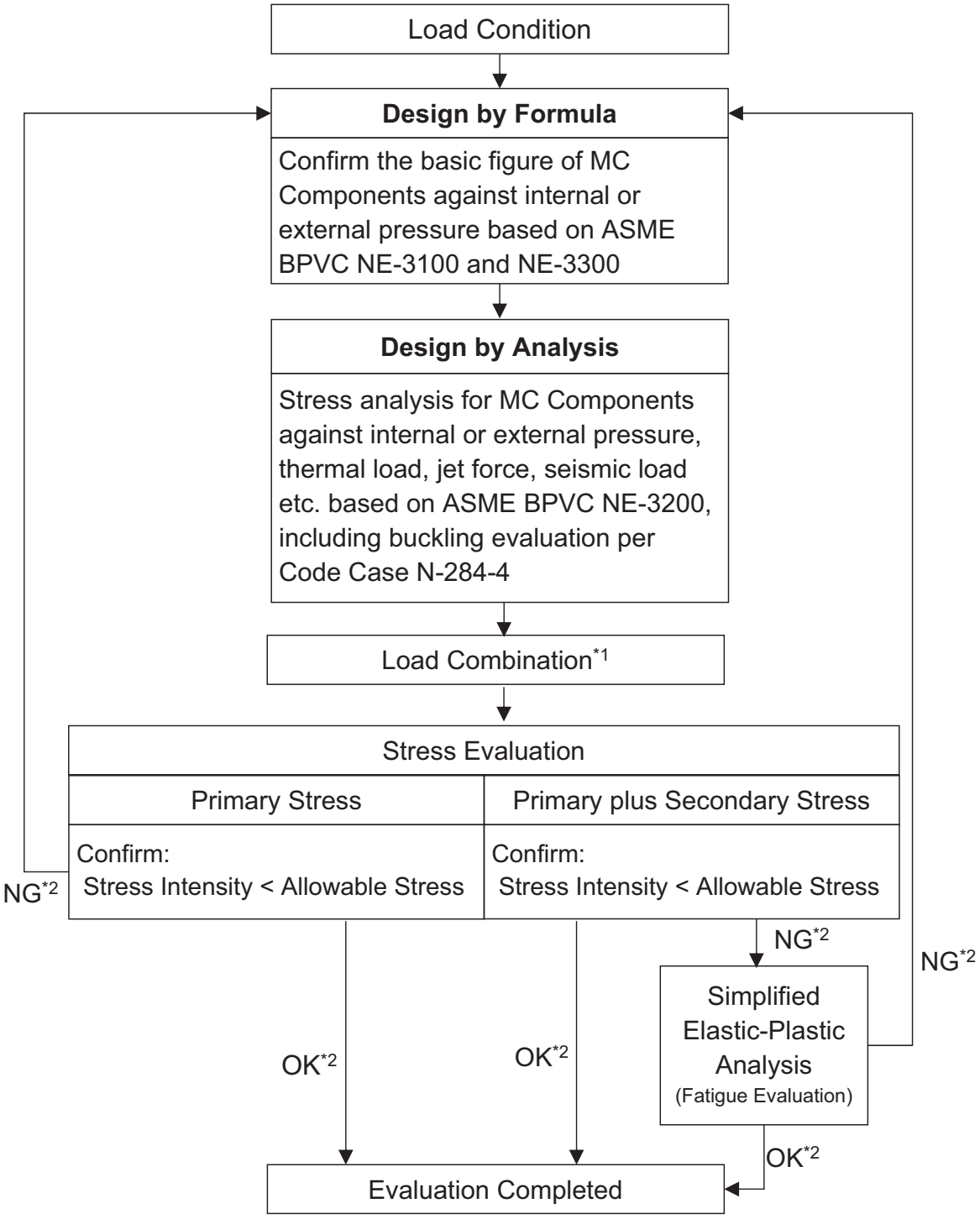
Figure 3.8-5 Penetrations Jurisdictional Boundary



*1: SRP 3.8.2 and RG 1.57

*2: Ok refers to “Okay” and NG refers to “No Good” in the flowchart

Figure 3.8-6 Design Procedures for the Containment Closure Head

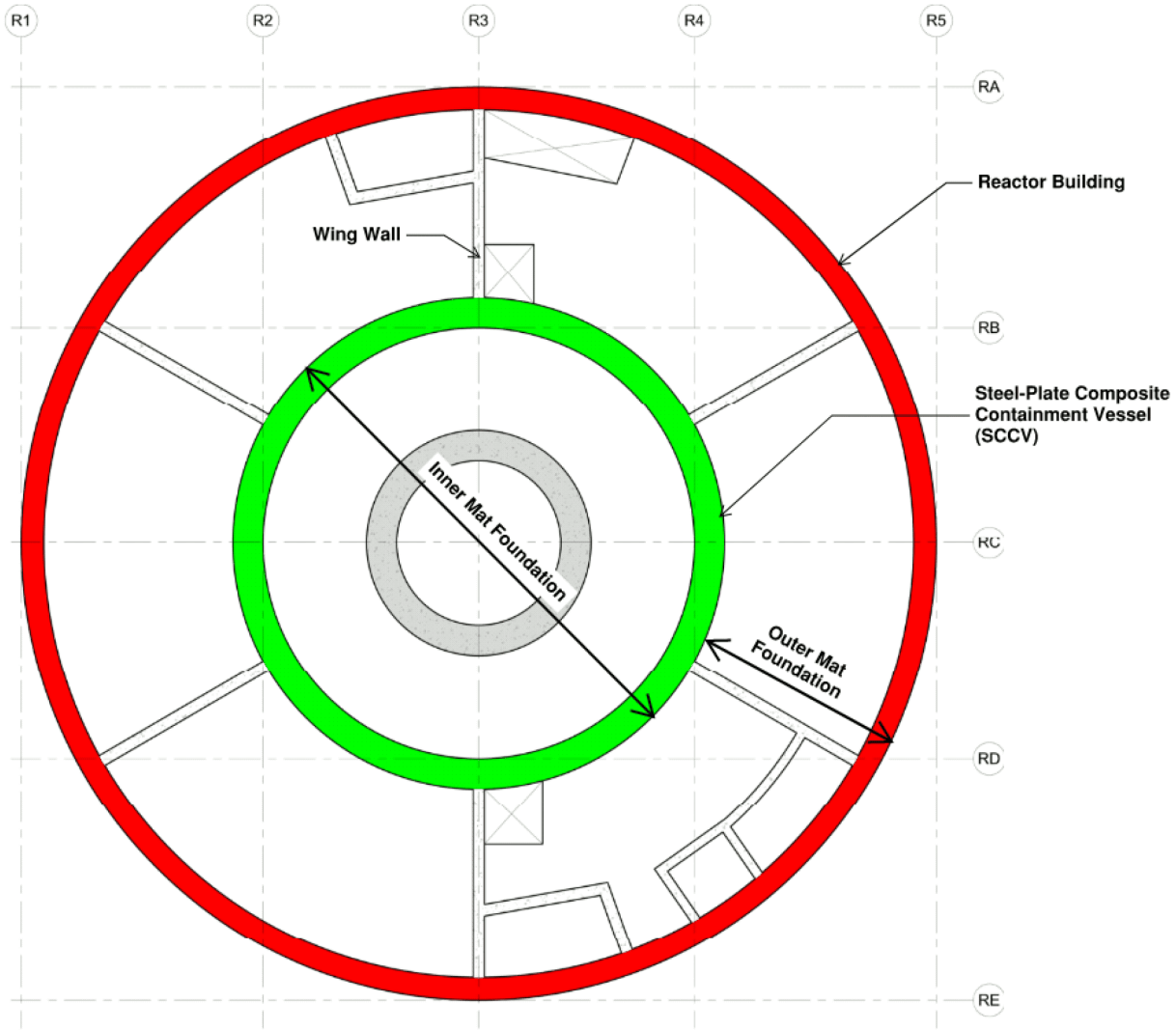


*1: SRP 3.8.2 and RG 1.57

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 3.8-7 Design Procedures for Other Class MC Steel Components



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

Figure 3.8-8 Plan View of BWRX-300 Mat Foundation

3.9 Mechanical Systems and Components

Section 3.9 provides the general design aspects used for Safety Class 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 SC2 and SC3 equipment follows ASME B31.1, “Power Piping,” (Reference 3.9-8) and ASME B31.3, “Process Piping,” (Reference 3.9-9).

Section 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 Safety Class 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 Safety Class 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 Safety Class 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 Safety Class and SCN SSCs. Valves include the valve body assembly, actuators, appurtenances, and all non-electrical connections.

Rotating Equipment

Rotating equipment includes Safety Class 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 Safety Class and SCN heat exchangers and tanks, including their supports, filter assemblies, and nozzles.

Ductwork Systems

Safety Class 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

Safety Class 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: Anticipated Operational Occurrence (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 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 Safety Class 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 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 Section 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 instrument, 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 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 Section 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 Section 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 Section 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 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 60-year life period 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 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 Safety Class 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 Analysis 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 Section 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 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 NRC 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 - 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 NRC 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 Section 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 Section 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 Section 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. This section will be further developed in the FSAR stage.

3.9.3 ASME 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 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 have an extremely low probability of abnormal leakage.

GDC 15 – Reactor coolant system design. The mechanical components of the Reactor Coolant System (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 Safety Class 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 Reactor Building Vibration (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 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 for RPV 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 Safety Class, 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, Steam Tunnel and Turbine Building 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 Subsection 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 Subsection 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 Main Steam (MS) piping trains extending from the outboard Main Steam Reactor Isolation Valve (MSRIV) up to the Seismic Interface Restraints (SIR) that are outboard of the Main Steam Containment Isolation Valves (MSCIV) 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.

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 Piping Supports

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
15	Hot Shutdown	200	Normal
17	Cold Shutdown	200	Normal
18	Vessel Flooding	72	Normal
19	Unbolt	72	Normal
20	Refuel	72	Normal
21	Design Basis Accident Pressurization With Feedwater	2	Emergency
22	Design Basis Accident Pressurization Without Feedwater	1	Emergency
23	Design Basis Accident Without Pressurization	3	Emergency
24	Inadvertent Isolation Condenser Initiation – All Trains	5	Emergency
25	Small Pipe Rupture — Loss-of-Coolant Accident	1	Emergency
26	Bounding Transient Without Scram	N/A	N/A
27	Large Pipe Rupture – Loss-of-Coolant Accident	1	Faulted
28	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
(Sheet 1 of 2)

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
29	SCRAM Solenoid	2
57	FMCRD Ball Check Valve	1
4	Isolation Condenser (IC) Inner Pool A Check	2

Table 3.9-4 Preliminary Active Valve List
(Sheet 2 of 2)

Quantity	Description	ASME Code Class
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 are identified from the perspective of their normal position and safety position.

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 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. Section 3.9.2 addresses dynamic testing of components of the Reactor Coolant Pressure Boundary (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 International Electrotechnical Commission (IEC)/Institute of Electrical and Electronics Engineers (IEEE) 60980-344 “Nuclear Facilities – Equipment Important to Safety – Seismic Qualification,” (Reference 3.10-1). 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” (Reference 3.10-12) 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 3.10-3). Prior to issuance of the FSAR, the differences resulting from the revised reference documents will be reconciled along with the NRC 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 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 3.10-2) 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 3.10-3) 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 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, “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 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, etc.) 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 inservice 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 420-2023, “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 383-2023, “IEEE Standard for Qualifying Electric Cables and Splices for Nuclear Facilities,” (Reference 3.10-6) and IEEE 384-2018, “IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits,” (Reference 3.10-7), and test requirements in IEEE 1202-2006, “IEEE Standard for Flame-Propagation Testing of Wire and Cable,” (Reference 3.10-8).

Supports provided by the equipment supplier will be qualified in accordance with this section by the equipment supplier.

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 628-2020, “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 572-2019, “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 that 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 382-2019, “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, “Nuclear Facilities – Equipment Important to Safety - Seismic Qualification,” International Electrotechnical Commission/Institute of Electrical and Electronics Engineers, 2020.

3.10-2 ASME BPVC, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” Subsections NB and NCD, American Society of Mechanical Engineers, 2021.

3.10-3 ASME QME-1, “Qualification of Active Mechanical Equipment Used in Nuclear Facilities,” American Society of Mechanical Engineers, 2023.

3.10-4 IEC/IEEE 60780-323, “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 Power Generating Stations and Other Nuclear Facilities,” Institute of Electrical and Electronics Engineers, 2019.

3.10-12 IEEE 344-2013, “IEEE Standard for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating,” Institute of Electrical and Electronics Engineers, 2013.

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 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 BWRX-300 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 IEC 63147:2017/IEEE Std 497-2016, "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 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 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 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, Division 1, “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 535-2022, “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 334-2006, “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 317-2013, “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 382-2019, “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 that includes IEEE 638-2013 “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 649-2006, “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 650-2006, “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 and 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 420-2023, “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 7-4.3.2-2016, “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.

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 383-2023, “IEEE Standard for Qualifying Electric Cables and Splices for Nuclear Facilities,” (Reference 3.11-17) and IEEE 384-2018, “IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits,” (Reference 3.11-18) are considered. The test requirement guidance provided in IEEE 1202-2023 “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 628-2020 “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, shall be analyzed to withstand the effects of an SSE.

SC1 connection assemblies consider the qualification guidance provided in IEEE 572-2019, “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, “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 382-2019. Line mounted equipment also includes active mechanical equipment subjected to ASME QME-1 including Non-mandatory Appendix QR-A.

3.11.3 References

3.11-1 IEC/IEEE 60780-323, “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,” International Electrotechnical Commission/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, Section III, Division 1, “Rules for Construction of Nuclear Facility 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 Electronics 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 Electronics 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 Electronics 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 Electronics 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 Electronics 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 Electronics 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 Electronics 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 Electronics 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 Electronics Engineers, 2016.

3.11-17 IEEE 383-2023, “IEEE Standard for Qualifying Electric Cables and Splices for Nuclear Facilities,” Institute of Electrical and Electronics Engineers, 2023.

3.11-18 IEEE 384-2018, “IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits,” Institute of Electrical and Electronics Engineers, 2018.

3.11-19 IEEE 1202-2023, “IEEE Standard for Flame-Propagation Testing of Wire and Cable,” Institute of Electrical and Electronics 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 Electronics 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 Electronics Engineers, 2019.

3.11-22 IEC/IEEE 60980-344, “Nuclear Facilities Equipment Important to Safety-Seismic Qualification,” International Electrotechnical Comission/Institute of Electrical and Electronics Engineers, 2020.

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 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 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 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 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 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 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 III.

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 Clamp-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 accident.

(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-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 Break 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 (Sheet 1 of 2)

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)

Table 3.12-3 Load Combinations and Acceptance Criteria for ASME B31.1 Piping and Components (Sheet 2 of 2)

Description	Load Combination	Acceptance Criteria	ASME Code Reference
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 Equation 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 Equation 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 $S_A = f (1.25 S_C + 1.25 S_h - S_L)$.

(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 (Sheet 1 of 2)

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)

Table 3.12-4 Load Combinations and Acceptance Criteria for ASME B31.3 Piping and Components (Sheet 2 of 2)

Description	Load Combination	Acceptance Criteria	ASME Code Reference
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.

(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 Equation 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	NB-2322.1	NCD-2322.1
	Orientation of Impact Test Specimens	NB-2322.2(a)(3)	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.

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).

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

Table 3A-1 Preliminary BWRX-300 Component Classification List
(Sheet 1 of 23)

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	

Table 3A-1 Preliminary BWRX-300 Component Classification List
(Sheet 2 of 23)

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	

Table 3A-1 Preliminary BWRX-300 Component Classification List
(Sheet 3 of 23)

Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes
FMCRD Separation Switches	SC3	PCS	N/A	Non-Seismic	
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	
REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS					
Nuclear Boiler System					
Reactor Pressure Vessel (RPV)	SC1	PCS	A	I	

Table 3A-1 Preliminary BWRX-300 Component Classification List
(Sheet 4 of 23)

Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes
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 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	
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	

Table 3A-1 Preliminary BWRX-300 Component Classification List
(Sheet 5 of 23)

Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes
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 Safety 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, TB	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	
Quench Tank Isolation Valves	SC1	PCS	B	I	
Quench Tank	SC3	PCS	D	II	
Head Vent Quench Tank Vacuum Breaker	SC3	PCS	D	II	

Table 3A-1 Preliminary BWRX-300 Component Classification List
(Sheet 6 of 23)

Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes
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	
Steam Supply and Condensate Return Valves	SC1	PCS	A	I	
Standby Gas Purge Valves	SC1	PCS	A	I	

Table 3A-1 Preliminary BWRX-300 Component Classification List
(Sheet 7 of 23)

Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes
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, SC3 and SCN Instrumentation and Control Systems					
Equipment that supports Defense Line 2 (DL2) functions	SC3	RB, CB, TB	N/A	Note 11	
Equipment that supports Defense Line 4a (DL4a) functions	SC2	RB, CB, TB	N/A	Note 11	
Equipment that supports DL4b functions	SC3	FSAR	N/A	Non-Seismic	
Non-Safety I&C System	SCN	RB, CB, TB	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	
Dry Type Transformers	SC3	CB, TB	N/A	Non-Seismic	

Table 3A-1 Preliminary BWRX-300 Component Classification List
(Sheet 8 of 23)

Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes
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	

Table 3A-1 Preliminary BWRX-300 Component Classification List
(Sheet 9 of 23)

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 and Lightning Protection System					
All components	SCN	ALL	N/A	Non-Seismic	

Table 3A-1 Preliminary BWRX-300 Component Classification List
(Sheet 10 of 23)

Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes
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	
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	

Table 3A-1 Preliminary BWRX-300 Component Classification List
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Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes
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	
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 asociated 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	

Table 3A-1 Preliminary BWRX-300 Component Classification List
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Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes
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	
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 Suction Surge Tank and Return Guard Pipe	SC1	RB	B	I	
Unidirectional Pool Makeup Conduits	SC1	RB	B	I	

Table 3A-1 Preliminary BWRX-300 Component Classification List
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Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes
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	
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

Table 3A-1 Preliminary BWRX-300 Component Classification List
(Sheet 14 of 23)

Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes
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	
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	

Table 3A-1 Preliminary BWRX-300 Component Classification List
(Sheet 15 of 23)

Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes
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	
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 Tank 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, Valves, and Other Pressure Retaining Components with no Safety Function	SCN	RB	D	Non-Seismic	
Instrumentation with no Safety Class Function	SCN	RB	N/A	Non-Seismic	

Table 3A-1 Preliminary BWRX-300 Component Classification List
(Sheet 16 of 23)

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	
DL 4a 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	

Table 3A-1 Preliminary BWRX-300 Component Classification List
(Sheet 17 of 23)

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, and Sprinklers	SCN	RB, TB, CB, RWB, OL	Note 10	Non-Seismic	
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	

Table 3A-1 Preliminary BWRX-300 Component Classification List
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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		

Table 3A-1 Preliminary BWRX-300 Component Classification List
(Sheet 19 of 23)

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	

Table 3A-1 Preliminary BWRX-300 Component Classification List
(Sheet 20 of 23)

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, with 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	

Table 3A-1 Preliminary BWRX-300 Component Classification List
(Sheet 21 of 23)

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 SYSTEMS					
Process Radiation Monitoring Subsystem					
Radiation Monitoring Equipment (non- pressure boundary)	SC3	RB, CB,TB, RWB	N/A	Non-Seismic	
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	

Table 3A-1 Preliminary BWRX-300 Component Classification List
(Sheet 22 of 23)

Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes

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 System 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.

Table 3A-1 Preliminary BWRX-300 Component Classification List
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Principal Component	Safety Class (Note 1)	Location (Note 2)	Quality Group (Notes 3,5)	Seismic Category (Notes 4,5,7,9,12)	Notes

9. Components classified as SC1 may be assigned to a Seismic Category lower than Category 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	Seismic Category	Seismic Demand	Interaction Requirement
RB	I	SSE	Not Applicable
SCCV			
Containment Internal Structures			
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 I SSCs during a Safe Shutdown Earthquake (SSE) or extreme wind event.

APPENDIX 3B INTEGRATED REACTOR BUILDING FINITE ELEMENT MODEL

This appendix presents the Three-Dimensional (3D) Finite Element (FE) models developed for the analysis and design of the integrated Reactor Building (RB) structures which consist of the RB structure outside containment, containment, containment internal structures and their common mat foundation.

Section 3B.1 presents the integrated RB 3D FE model used to perform the various structural analyses discussed in Subsection 3.8.4.1.4. This model is developed using appropriate element types to represent the mass, stiffness, damping characteristics of the integrated RB structures for the analyzed loads being evaluated and resulting stress responses. The model has a sufficiently refined mesh to accurately capture the dynamic response of the structures over the frequency range of interest and to enable accurate calculation of structural stress demands. Mesh refinements in areas of geometric structural discontinuities are used, as applicable.

Table 3B-1 summarizes the integrated RB modeling requirements for the various analysis cases performed. Section 3B.2 presents the 3D FE model used for the seismic Soil-Structure Interaction (SSI) analysis of the integrated RB. Section 3B.3 presents the 3D FE models used for the 1-g static SSI analyses. Section 3B.4 presents the 3D FE models used for static and quasi-static analyses, while Section 3B.5 presents the 3D FE models used for thermal stress analysis. These models are developed per the methodology presented in Subsection 3.7.2.3 and Subsection 3.8.4.1.4.

3B.1 Integrated Reactor Building Finite Element Model

3B.1.1 Model Overview

The integrated RB 3D FE model is developed using ANSYS (See Appendix 3I for description) and represents the major structural elements of the integrated RB structures as described in Subsection 3.7.2. The origin of the model is defined at the center of the integrated RB structures with the Z-origin defined at grade (Elevation 0.0 m (0.0 ft)) and the global X- and Y-axes pointed in the North-South, and East-West directions, respectively.

The integrated RB FE model is developed using wall centerlines and approximate centerlines of slabs. The elevation of slabs is adjusted so that the centerline of the slab at grade is at Elevation 0.0 m (0.0 ft). The centerline elevations for the roof slab approximate the weighted average of the slab thickness.

Figure 3B-1 is a 2/3 slice of the integrated RB FE model showing the major structural components of the model. Containment mechanical and electrical piping penetrations are incorporated in the integrated RB FE model as shown in Figure 3B-2.

The RPV and internals, including fuel, valve mass and supports, are integrated into the integrated RB FE model using a Lumped Mass Stick (LMS) model developed as described in Subsection 3.7.3.3.3 and shown in Figure 3B-3. Subsection 3B.1.4 discusses the connections used to integrate the RPV LMS and the integrated RB models.

3B.1.2 Reactor Building Steel-Plate Composite Structures

The integrated RB structures are primarily constructed using Diaphragm Plate Steel-Plate Composite (DP-SC) modules as described in Section 3.8. Geometric details of a typical DP-SC module are illustrated in Figure 3B-4. A DP-SC module system consists of multiple steel components arranged and welded together to form a module. The DP-SC steel faceplates serve as the main reinforcement and permanent formwork for the DP-SC concrete infill. Diaphragm plates with holes allow the flow of concrete inside the module. The diaphragm plates and headed steel studs provide the composite action between the DP-SC steel faceplates and concrete infill, with the diaphragm plates also serving as out-of-plane shear reinforcement for the composite section. The DP-SC modules are spliced together using welded connections to form structural walls, floors, or mat foundation sections as shown in Figure 3B-4. The RB exterior wall and floors, pool walls (excluding fuel pool walls), 36 in thick steam tunnel wing walls, the Steel-Plate Composite Containment Vessel (SCCV) wall and top slab, the RPV pedestal and the common mat foundation are constructed using DP-SC modules.

Conventional steel-plate composite modules, where the two faceplates are connected using discrete tie bars instead of diaphragm plates as is the case for DP-SC modules, are used in the construction of the RB remaining wing walls, partition walls, elevator walls, stair walls and fuel pool walls.

3B.1.3 Model Properties

Material properties used in the construction of the integrated RB FE model are described in Section 3.8 and presented in Table 3B-2 and Table 3B-3. Properties assigned to the integrated RB FE model reflect these properties and are computed in accordance with the guidelines of Section 5.0 of NEDC-33926P-A, “BWRX-300 Steel-Plate Composite (SC) Containment Vessel (SCCV) and Reactor Building Structural Design,” (Reference 3B-1).

Thermal conductivity is determined with ACI/TMS 122R-14, “Guide to Thermal Properties of Concrete and Masonry Systems,” (Reference 3B-2) and following the guidance of Section 5.5 of NEDC-33926P-A.

OBE and SSE damping values applicable to the integrated RB structural model are listed in Table 3.7-5. However, more conservative OBE damping values of 2% and 4% are used for welded/friction-bolted steel structures and bearing-bolted steel structures, respectively.

3B.1.3.1 Steel-Plate Composite Equivalent Properties

The steel-plate composite elements of the integrated RB structures are modeled using elastic shell elements with an equivalent thickness, elastic modulus, and material density calibrated to match the steel-plate composite effective stiffness and mass properties developed using the procedures described in Section 5.5 of NEDC-33926P-A. The effect of stainless-steel liners used for corrosion or shielding protection is not considered in the calculation of the FE equivalent section properties.

Effective material properties used for the operating conditions represent the Best Estimate (BE) structure stiffness and combine both the steel and concrete stiffnesses as discussed in Section 5.5 of NEDC-33926P-A. Effective material properties for the fully cracked condition are referred to as the Lower Bound (LB) structure stiffness. Most analyses are performed using the operating/effective (BE) section properties. Thermal accident conditions utilize partially cracked models with section properties that use the operating/effective properties over most of the RB but fully cracked properties over the components subjected to accident conditions. The fully cracked condition is simulated by not considering the contribution of the concrete infill for out-of-plane flexural stiffness and a LB shear stiffness as discussed in Section 5.5 of NEDC-33926P-A.

Wall curvatures are not considered in the mathematical formulations of steel-plate composite properties but are included in the 3D FE model as discussed in Section 5.12 of NEDC-33926P-A.

3B.1.4 Steel-Plate Composite Connections

The RB FE model utilizes centerline shell element models of steel-plate composite components where adjacent component elements share nodes at their intersecting lines to simulate the steel-plate composite welded connections. This modeling approach effectively suggests rigid connection behavior, with no relative deformations between components.

The below grade slabs and wing walls are connected to the SCCV using six independent springs at each intersecting node to represent the stiffness in each of the deformation degree of freedom. These springs are assigned a large stiffness value to simulate rigid connection behavior.

The centerlines of shell elements representing slabs and walls are connected in the RB model. The RB model includes an enhanced feature within the below grade slabs that incorporates rigid shell elements at the slabs' inner and outer radial edges to account for the rigidity that exists within the slab-to-wall intersection. The slab mesh radial dimension is set equal to half the wall thickness and the elements assigned an elastic modulus equal to 20 times the slab modulus to provide the rigid offset.

3B.1.5 Reactor Pressure Vessel and Containment Equipment and Piping Support Structure Connections

The BWRX-300 Reactor Pressure Vessel (RPV) is supported vertically and laterally by a steel skirt anchored using anchor bolts to structural steel that lines the top of the pedestal as seen in Figure 3.8-1. Top and bottom stabilizers (shown in Figure 3.8-1) provide additional lateral support to the RPV. The Control Rod Drive Housing (CRDH) located below the RPV (see Figure 3.8-1) is also supported by lateral supports. The RPV is free for radial and vertical movement at the stabilizers and CRDH supports. The skirt, bottom stabilizers, and CRDH lateral support transfer lateral load directly to the pedestal. The upper stabilizers transfer lateral load to the Containment Equipment and Piping Support Structure (CEPSS).

The RPV LMS model shown in Figure 3B-1 and Figure 3B-3 is developed to represent the dynamic mechanical behavior of the BWRX-300 RPV, reactor internals, fuel, valve mass, and its supports stiffness. During its incorporation into the integrated RB FE model, the LMS model was adapted to ensure a proper transfer of RPV loads to the pedestal and CEPSS framing shown in Figure 3B-1 and Figure 3B-3.

The CEPSS consists of a 3D space frame that supports three platform levels. Each platform includes radial and circumferential beams along with infill beams that will support the platform grating and large equipment and piping supports. The CEPSS framing includes vertical bracing members around the CEPSS outer perimeter (near SCCV wall) and inner perimeter (above the pedestal). Horizontal bracing members are also included integral to the platform framing which are critical for lateral force transfer to the SCCV wall. The CEPSS framing is vertically and laterally supported by the pedestal and tangentially restrained by the SCCV wall through shear lugs that allow the vertical and radial relative expansion between the CEPSS and SCCV.

3B.1.6 Modeling of Openings and Penetrations

The model mesh size meets the ANSI/AISC N690-18, “Specification for Safety-Related Steel Structures for Nuclear Facilities,” (Reference 3B-3) modeling requirements for the purposes of element strength evaluation including modeling in the connection regions and around penetrations (mesh size < t_{sc}, where t_{sc} is the depth of the steel-plate composite module as indicated in Figure 3B-4).

Where known large penetrations (> $t_{sc}/2$) exist, the opening is included in the model using the actual or approximate opening size. Where known small penetrations (< $t_{sc}/2$) exist, the opening is not explicitly modeled and evaluation around the small openings depends on whether the penetration is detailed as “free edge” or “fully developed.”

3B.1.7 Static and Dynamic Mass Properties

The integrated RB model is analyzed for the design loads described in Section 3.8. These loads are assigned to the integrated RB model as static and dynamic masses.

3B.1.7.1 Equipment and Non-Structural Mass

The dead inertial mass assigned to the integrated RB model includes the self-weight and supplemental weight added to represent equipment and non-structural dead loads not represented in the model. Static dead load analysis is performed by applying a 1-g acceleration to the SSI model including dead load masses. The dead load inertial masses also contribute to the seismic inertial loads and remain in the integrated RB seismic SSI analyses.

The non-structural dead load mass assigned to the integrated RB model includes the mass of staircases, partition walls, the basemat 2 ft thick normal weight concrete topping slab, the pedestal refractory material (weighing approximately 820 psf), and the containment and CEPSS platforms. The assigned mass of the two platforms inside containment is equivalent to 65 psf.

The CEPSS top and middle platforms are assigned 75 psf for equipment dead load and 16 psf for light weight steel grating. The CEPSS lower platform is assigned 75 psf for equipment dead load and 32 psf for heavy steel grating.

In areas where the concentrated equipment loads are larger than 100 psf, a uniform mass is added to the shell elements by increasing the material density. Equipment masses are accounted for by increasing floor shell element density. Concentrated mechanical and electrical equipment load mass is assigned in two ways in the model, either by increasing the density of the floor shell elements, or by assigning the mass to a set of nodes in the model.

Outside of the heavy equipment areas, a uniformly distributed load of 100 psf is applied to the floors of the Integrated RB to simulate the additional static mass from mechanical, electrical, and miscellaneous equipment supported from the floors.

The polar crane composed is of an overhead bridge of two deep girders supporting a trolley, with the polar crane's parked position aligned with the Y axis of the RB. The seismic analysis model includes vertical and horizontal mass at each girder end representing the crane girders. The trolley weight is also included as vertical and horizontal masses with the trolley parked at the end of the crane in the +Y direction. Vertical mass representing the full 125-ton payload is also included for the seismic model and positioned at the end of the crane in the +Y direction. Due to the pendulum effect of the suspended payload, the horizontal inertia mass of the payload is not included in the seismic model. The 1-g analysis model only includes the vertical mass of the crane girders and trolley (parked at the end of the crane in the +Y direction) as this analysis model is intended to include the structural dead loads. As indicated in Table 3B-1, the crane payloads and operational loads are considered in a separate load case defined in Subsection 3E.1.8.

3B.1.7.2 Equivalent Live and Snow Loads

A 100 psf live load is considered for all floors except for the refueling floor where the live load is 200 psf. The seismic model includes fifty percent of the live load on floor areas without heavy equipment and twenty-five percent of snow loads are modeled.

No equivalent live load mass is considered on the CEPSS for the seismic model as it is within the containment and no live load is present during normal operating conditions.

3B.1.7.3 Hydrostatic and Hydrodynamic Mass

The hydrostatic and hydrodynamic loads applied on the integrated RB 3D FE model are computed using best estimate water levels in pools under normal operating conditions. Methodology used to account for hydrodynamic effects on the integrated RB pools is discussed in Subsection 3.7.2.3.1.

Hydrostatic loads on pool floors are captured by modeling the fluid volume in each of the pools as a vertical only mass during 1-g static SSI analyses. Hydrostatic loads on pool walls are analyzed as a pressure load during the static structural analysis as demonstrated in Subsection 3E.3.2.

The vertical fluid volume mass on pool floors also acts as the seismic inertial mass in the vertical direction during seismic analysis. The hydrodynamic impulsive loads on pool walls are captured during seismic response analysis by assigning masses to the pool walls consistent with the impulsive pressure distribution. Hydrodynamic convective (sloshing) loads and the “breathing” mode lateral pressures on pool walls are analyzed as a pressure load during the static structural analysis. The “breathing” mode lateral pressures are the result of vertical seismic excitation on the pools causing an increased vertical pressure and lateral pressures with distributions matching the hydrostatic pressures but with magnitudes based on the seismic response at the pool levels.

3B.2 Seismic Structure-Soil-Structure Interaction Analysis Models

As the integrated RB is deeply embedded, the interaction of the structure with the surrounding subgrade is important for its structural integrity and its response under static and dynamic loads. In accordance with the guidance of Section 5.1 of NEDO-33914-A, “BWRX-300 Advanced Civil Construction and Design Approach,” (Reference 3B-4), the one-step approach, as defined in Section 3.1.2 of ASCE/SEI 4-16, “Seismic Design of Safety-Related Nuclear Structures,” (Reference 3B-5), is implemented for the design of the BWRX-300 integrated RB structures. To adequately account for the SSI effects, a 3D FE model that includes FE representations of the integrated RB (see Section 3B.1 and Figure 3B-1) and the surrounding subgrade is developed as described in Subsection 3.7.2. The design basis analysis case model for the RB seismic SSI analysis, shown in Figure 3B-5, includes the explicit representation of certain construction elements in the Clinch River Nuclear (CRN) excavation plan which are illustrated in Figure 3B-6. These include the mud mat under the RB mat foundation shown in red in Figure 3B-5 and Figure 3B-6, and the concrete fill in the annulus and shelf shown in purple and red in the same figures. The model also includes solid elements, shown in blue in Figure 3B-5 and Figure 3B-6, that may be assigned material properties corresponding to soil or simulating the temporary excavation support to assess their effects on the RB seismic response as discussed in Subsection 3.7.2.3 and Subsection 3.7.2.9.2. This model is paired with the range of in-situ as-built subgrade conditions discussed in Subsection 3.7.1.3 to perform the Case 1 seismic SSI analysis documented in Appendix 3C.

Coarse surface mounted FE models representing the global dynamic properties and weight of the surrounding Power Block structures and foundations are coupled in ANSYS with the Integrated RB and subgrade model shown in Figure 3B-5 to capture the Structure-Soil-Structure Interaction (SSSI) effects as discussed in Subsection 3.7.2.4. The coarse Power Block models are inserted into the integrated RB model with the centerline of their mat foundations set a grade elevation (Elevation 0.0 m (0.0 ft)). This combined SSI model, shown in Figure 3B-7, is paired with a second case of as-built subgrade conditions where the in-situ soil and weathered rock are excavated down to competent rock and replaced with engineered fill up to plant grade Elevation 814.5 ft to provide adequate support for supporting the adjacent Power Block structures. The results of this Case 2 seismic SSI analysis are also summarized in Appendix 3C.

The Turbine Building (TB), one of the Power Block structures included in the combined SSI model, is divided into three separate structural systems consisting of the TB shell structure, radiological Shield Wall Area, and the turbine generator pedestal. The turbine generator pedestal

including foundation is structurally isolated from both the TB shell and the shielding walls for vibration control. Figure 3B-8 shows the LMS model used to represent the turbine generator pedestal structure, and the isolated FE model used to model its isolated foundation in the combined seismic SSI analysis model.

The Case 1 and Case 2 seismic SSI models, also referred to as the dynamic SSI models, are assigned BE structure stiffnesses with seismic and dead load inertial masses, and lower OBE-level damping properties. The models are developed in ANSYS and then translated into ACS SASSI (see Appendix 3I) format.

3B.2.1 Modeling of Soil-Structure Interfaces

The soil and soil-structure interfaces of the dynamic SSI analysis models in SASSI consist of the following:

- Far-field subgrade model representing the properties of the subgrade materials that are approximated as continuum infinite horizontal layers resting on surface of uniform half-space.
- Near-field subgrade model representing the properties of subgrade materials supporting the adjacent Power Block foundations and around the RB below grade exterior wall.
- Excavated soil volume representing the properties of the volume of far-field subgrade replaced by the RB and adjacent near-field volume elements.
- Linear spring elements to connect the near-field subgrade elements to structural elements, which are assigned stiffness properties that can be adjusted to simulate different interface conditions. The excavated volume and the near-field volumes are also connected with linear spring elements at the near-field (backfill) perimeter.

The soil elements used in dynamic and subgrade impedance calculations use a consistent layer thickness governed by the dynamic response which is sufficiently refined to transmit the entire frequency range of interest. The minimum layer thickness is approximately double the shell element mesh size.

3B.2.1.1 Far-Field Subgrade

The far-field subgrade is represented in ACS SASSI by half-space continuum of horizontally infinite layers with uniform linear elastic properties resting on uniform elastic half-space. A viscous lower boundary of the subgrade model is established at a depth of 482 ft meeting the regulatory guidance of Subsection II.4 of SRP 3.7.2 as discussed in Subsection 3.7.2.3.1.

3B.2.1.2 Near-Field Subgrade Volume

Figure 3B-9 presents the single layer of near-field solid elements placed below the BWRX-300 Power Block foundations, which are assigned subgrade material properties, and the solid FEs around the below grade portion of the RB exterior wall that represent the properties of the concrete backfill in the two seismic SSI analysis cases.

At the excavated volume perimeter, the FE of the excavated soil volume (see Subsection 3B.2.1.3) and near-field volume share the same nodes. The mesh of these two subgrade volumes aligns with every other element of the RB exterior wall and mat foundation mesh. Having the FE mesh of the near-field volume and excavated volume being twice the size of the refined one-step approach RB structural model helps reduce the ACS SASSI dynamic soil impedance calculations to reasonable durations. The mesh of the near-field volume also aligns with the mesh of the surface mounted Power Block foundations.

3B.2.1.3 Far-Field Subgrade Excavated Volume

The excavated volume model (see Figure 3B-10) consists of 3D solid FEs representing the part of the horizontally layered subgrade displaced by the near-field volume (see Subsection 3B.2.1.2) and the embedded portion of the RB structure. The mesh of the excavated subgrade volume aligns with the layering of the far-field subgrade profiles and the near-field subgrade volume shown in Figure 3B-9. The meshes of the excavated and near-field volumes are aligned at the perimeter, where the excavated volume and near-field volume solid elements share the same nodes.

3B.2.1.4 Near-Field - Structure Interfaces

Three linear spring elements are assigned to nodes between the near-field volume and the FE models of the RB and adjacent Power Block structures as well at the nodes of the RB exterior wall to represent the interaction mechanisms at the structure and soil interfaces. These spring elements are assigned to approximately every other node on the RB exterior wall and mat foundation mesh.

For seismic analysis, interface springs in the dynamic SSI models are assigned high stiffnesses to simulate bonded conditions between the near-field volume solid elements and the structural shell elements and the near-field elements and the excavated volume.

3B.3 1-g Static Soil-Structure Interaction Analysis Model

The 1-g static SSI analysis provides design demands for gravity inertia loads and static earth pressure loads including surcharge loads from the foundations of adjacent Power Block structures.

Two sets of 1-g static SSI analysis are performed to evaluate the unfactored and factored gravity demands on the integrated RB (see Table 3B-1) requiring the development of two 1-g static SSI analysis models. Factored gravity demands are obtained from load combinations with different load factors assigned to dead and soil pressure loads (1.2D+1.6H or D+1.3H).

The two 1-g static SSI models have the same configuration and FE attributes as the combined dynamic model developed to capture the SSSI effects described in Section 3B.2. The structural members of the 1-g static SSI models are assigned the same BE effective stiffness properties as

the dynamic models. Differences between the static and dynamic models consist of the following adjustments made to the unit weight properties of shell and lumped mass elements of the 1-g SSI models:

- The removal of the live, roof snow and crane mass from the 1-g static SSI Models. The fluid horizontal dynamic masses are not removed from the models as these are not activated in 1-g gravity analysis.
- The total mass of adjacent Power Block structures is increased by 10% in the unfactored and factored 1-g SSI analysis models for added safety margin. The factored 1-g SSI analysis also uses unit weight properties of the subgrade and surrounding structures increased by an additional 33%.

Per recommendations of Section 5.1.2 of NEDO-33914-A, stiffness properties assigned to the spring elements in the 1-g static SSI models provide SSI interface conditions that emphasize the lateral earth pressure loads on the RB exterior wall below grade. An Upper Bound (UB) stiffness is assigned to the spring elements in the direction normal to the RB exterior wall and mat foundation to simulate bonded conditions. The friction at the RB exterior wall and mat foundation is not considered by assigning very low stiffness properties to the contact springs in the vertical and tangential direction.

3B.4 Static and Quasi-Static Analysis Models

The static and quasi-static analyses provide design demands for static normal operating and accident conditions loads, excluding thermal loads, not captured in the 1-g static SSI analysis. For more information on load cases considered in the analyses, refer to Subsection 3.8.4.1.4 and Table 3B-1.

As indicated in Table 3B-1, static and quasi-static analyses are performed on standalone models of the integrated RB with either BE structure stiffness properties for normal operating and environmental loads or partially cracked conditions consisting of effective stiffness properties for all components except those subjected to temperature demands that create fully cracked concrete conditions.

Static and Quasi-static analyses are performed using the subgrade stiffness sub-structuring method as discussed in Subsection 3.8.4.1.4 in which the subgrade represented by solid FEs (see Figure 3B-11) is condensed into a single matrix element, called a super-element. This super-element represents the SSI stiffness at nodes coincident with every other node of the mat foundation and every node at the outside of the near-field volume. For static and quasi-static analyses, the super-element is assigned LB subgrade stiffness properties to emphasize the deformations at the subgrade-structure interfaces. This results in conservative estimate of the member force and moment demands on the RB integrated structure.

3B.5 Thermal Stress Analysis Model

Thermal analyses provide stress demands due to normal operating and accident load conditions as indicated in Table 3B-1.

Thermal stress analyses are performed on standalone models of the integrated RB that have the same configuration and FE attributes as the models developed for static and quasi-static analyses discussed in Section 3B.4. The standalone integrated RB model used for normal operating thermal analysis is developed with BE structure stiffness properties. The standalone integrated RB model used for the accident thermal analysis is developed with lower structure stiffness properties (fully cracked or lower bound section properties) assigned to walls and slabs within DBA regions affected by DBA thermal loads (e.g., the exterior pool walls, SCCV wall, RPV pedestal, inner mat foundation and SCCV top slab), and BE structure stiffness properties elsewhere.

In addition to the modeling requirements of Table 3B-1, the following modifications are also included in the FE models used for thermal stress analyses:

- The SCCV and RPV pedestal rigid beams used to support the dynamic analysis are removed or modified to have a very low elastic modulus and thermal coefficient to match the steel-plate composite they are embedded into.
- The RPV LMS model developed for the dynamic analysis is also removed from the models to obtain accurate thermal results. The removal of the RPV LMS model is acceptable as the RPV is designed to allow nearly unrestrained thermal growth through the stabilizers and RPV skirt.
- Crane rigid support offsets are removed to eliminate artificially high thermal stresses.

Various conditions for normal operating and DBA thermal stress analyses are considered and use super-elements assigned LB and UB subgrade stiffness properties to obtain bounding internal forces for the design of the integrated RB structures.

3B.6 Validation of Finite Element Models

This section documents the verification of the integrated RB ANSYS seismic analysis models and their translation from ANSYS to SASSI.

The verification process includes detailed review of the assigned finite elements to create the models' structural components, structural connectivity, and the non-structural masses along with static, modal, and harmonic analyses to confirm model static and dynamic behavior. The analysis models include the masses, structural stiffness, and damping appropriate for seismic analyses. The analysis models are translated to SASSI for use in the SSI analyses documented in Appendix 3C. These models are additionally used as a basis for static and thermal structural analysis in ANSYS with some adaptations as described in Section 3B.3 through Section 3B.5.

In addition to a detailed review of the RB FE model, the model validation includes the following steps:

- Modal analysis of the RB fixed base model
- 1-g analysis of the RB fixed base model in X-, Y-, and Z-direction
- Modal analysis of the isolated RPV LMS model for verification of adaptation into the integrated RB model

- Modal analysis of the Power Block structures for verification of adaptation into the integrated RB model
- Acceleration Transfer Function (ATF) comparison between ANSYS and SASSI RB fixed base models for verification of translation of the integrated RB model to SASSI

The verification steps discussed in Subsection 3B.6.1 through Subsection 3B.6.5 demonstrate that the integrated RB ANSYS structural models adequately represent the RB static and dynamic structural behavior and confirm the accurate translation of the model into SASSI.

3B.6.1 Fixed Base Modal Analysis of Reactor Building

The objective of the modal analysis is to understand the key dynamic properties of the model. The Integrated RB significant mode frequencies, and mass participation ratios are presented in Table 3B-4. These modal frequencies agree reasonably well with simplified hand calculations which provides confidence in the model.

The integrated RB refined mesh ensures an accurate representation of the dynamic response and accurate computations of structural demands as discussed in the introductory paragraph of this Appendix. In addition, modeling equipment and non-structural masses as a combination of distributed and nodal mass as discussed in Subsection 3B.1.7.1 captures the participation of all mass and ensures an adequate representation of the dynamic modes for the structures.

3B.6.2 Fixed Base 1-g Analysis of Reactor Building

A 1-g acceleration analysis is performed for the fixed base RB model in each of the three orthogonal directions to assess the total seismic mass and to examine the deformed shape and ensure component connectivity in all three directions.

The total seismic mass/weight for the fixed base RB model is summarized in Table 3B-5. The deformed shapes of the RB for the 1-g X, 1-g Y and 1-g Z analyses are shown in Figure 3B-12 through Figure 3B-14. The deformed shapes shown in Figure 3B-12 through Figure 3B-14 confirm that element connectivity is as expected between the various structural components, including the steel-plate composite components, the RPV LMS model and the CEPSS.

3B.6.3 Validation of Incorporation of Reactor Pressure Vessel Lumped Mass Stick Model

The incorporation of the RPV LMS model into the integrated RB model is validated by comparing the dynamic characteristics of the RPV LMS model within the integrated RB model with benchmark results of the eigenvalue analysis of the standalone RPV LMS model. To perform this comparison, the RPV LMS model is “isolated” in the global RB FE model by applying fixed boundary conditions to all nodes up to LMS model connection points (i.e., skirt to the top of pedestal, top and bottom stabilizers).

Figure 3B-15 compares the modal properties (frequencies and cumulative masses) between the RPV LMS model before and after integration into the integrated RB model. As demonstrated by Figure 3B-15, the dynamic coupled response between the integrated RPV LMS model and the RB is adequately captured.

3B.6.4 Validation of Dynamic Models of Adjacent Power Block Structures

The coarse FE models of the adjacent Power Block structures that are coupled with the integrated RB are required to represent their global dynamic properties for use in seismic SSI analyses.

To validate the translation of each Power Block structure into the combined seismic SSI analysis model, a comparison of the dynamic frequencies of the primary modes of vibration in each direction between the provided coarse mesh models and each Power Block structure “isolated” in the global structural model is performed. The isolation of the global structural models is achieved by fixing the degrees of freedom at each building foundation and the degrees of freedom associated with the structures.

Validation results demonstrate that the dynamic characteristics of the coarse dynamic models match reasonably well in total mass and dominant dynamic response with those from the refined models with the largest frequency difference being 1.7% for the Turbine Building Y-direction response. Therefore, the coarse FE models of the adjacent Power Block structures provide an adequate representation of the global dynamic properties for the SSI analyses of the integrated RB.

3B.6.5 SASSI Model Translation Verification

To validate the translation of the integrated RB model from the ANSYS to the SASSI, the total ATFs obtained from the ANSYS fixed based RB and the SASSI fixed base model are compared.

The ATFs are defined as the ratio of the acceleration at a particular point in the structure to the amplitude of the input acceleration at the base. Harmonic analysis is performed in ANSYS to determine the ATF while SASSI utilizes ATFs as a key calculated quantity in its response calculation approach.

Figure 3B-16 and Figure 3B-17 compare the two pairs of ATFs for the center of the RB roof and middle of RPV fuel. As shown in Figure 3B-16 and Figure 3B-17, the ATF peaks occur at frequencies consistent with the primary mode frequencies obtained from modal analysis, thus confirming the translation of each model from ANSYS to SASSI.

3B.6.5.1 ANSYS Fixed Base Harmonic Analysis

The RB model used in the ANSYS fixed base harmonic analysis is the model shown in Figure 3B-5 excluding the Power Block backfill layer. The harmonic analysis model intentionally keeps the RB backfill, shelf, and additional solid elements along with the inter-connecting spring elements to ensure proper translation of the finite elements up to the connection with the soil layering in SASSI.

ATFs are obtained at key locations, such as the pedestal, RPV, CEPSS, and fuel, to ensure the translation can capture the global structural behavior of the integrated RB structures. A full integration harmonic analysis is then performed, where a harmonic 1-g acceleration input is used

to excite the model with frequencies varying from 0.1 to 20 Hz for horizontal input and 0.1 to 40 Hz for vertical input, at increments of 0.1 Hz. The results of harmonic analysis include the amplitude of the steady-state relative displacement and response phase-angle.

3B.6.5.2 SASSI Fixed Base Frequency Domain Analysis

SASSI analyses are performed on FE models of the integrated RB mounted on a series of rigid soil layers. To simulate fixed base conditions, the following properties are assigned to the soil layers beneath the integrated RB common mat foundation:

- Shear wave velocity (V_S) of 32,808 ft/s
- Compression Wave Velocity (V_P) 65,617 ft/s
- Damping ratio of 1%

SASSI analyses of the “fixed base” models are performed for analysis frequencies ranging from 0.1 to 70 Hz to calculate ATF amplitudes for the responses in the three orthogonal directions at the same key nodal locations.

3B.7 References

3B-1 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.

3B-2 ACI/TMS 122R-14, “Guide to Thermal Properties of Concrete and Masonry Systems,” American Concrete Institute, 2014.

3B-3 ANSI/AISC N690-18, “Specification for Safety-Related Steel Structures for Nuclear Facilities,” American National Standards Institute/American Institute of Steel Construction, 2018.

3B-4 NEDO-33914-A, “BWRX-300 Advanced Civil Construction and Design Approach,” GE-Hitachi Nuclear Energy Americas, LLC, Revision 2, June 2022.

3B-5 ASCE/SEI 4-16, “Seismic Design of Safety-Related Nuclear Structures,” American Society of Civil Engineers, 2017.

Table 3B-1 Integrated RB Analyses Cases and Model Requirements
(Sheet 1 of 6)

Loading Type	Applicable Design Load(s)	Analysis Method (Computer Program)	Structural Stiffness	Subgrade Properties	Embedded RB Exterior Wall Contact Springs	Mat Foundation Contact Springs	Structural Model
Seismic Inertial, earth pressure, hydrodynamic: impulsive pressure on pool walls and slabs	E	Seismic SSI (ACS SASSI)	Effective BE	BE strain compatible	Rigid in all three directions	Rigid in all three directions	Standalone (for SSI) and Coupled dynamic models of RB and adjacent Power Block structures (for SSSI effects)
				LB strain compatible(1)			
				UB strain compatible(1)			
Unfactored dead load, vertical hydrostatic load, and lateral static earth pressure including adjacent building surcharge	D, F_v, H	1-g static SSI (ANSYS)	Effective BE	UB unit weight	Rigid in radial (bearing) directions. Soft in vertical/tangential (frictional) directions	Rigid in vertical direction only	Coupled static models of the RB and adjacent Power Block structures
				UB Poisson's ratio			
Factored dead load, vertical hydrostatic load and lateral static earth pressure including adjacent building surcharge				LB Young's modulus			

Table 3B-1 Integrated RB Analyses Cases and Model Requirements
(Sheet 2 of 6)

Loading Type	Applicable Design Load(s)	Analysis Method (Computer Program)	Structural Stiffness	Subgrade Properties	Embedded RB Exterior Wall Contact Springs	Mat Foundation Contact Springs	Structural Model
Live load on floors	L	Static subgrade stiffness sub-structuring analysis (ANSYS)	Effective BE	LB static massless LB Poisson's ratio LB Young's modulus	Rigid in radial (bearing) directions. Soft in vertical/tangential (frictional) directions	Rigid in vertical direction only	Standalone RB model
Crane	C						
Hydrostatic and hydrodynamic breathing mode pressures on pool walls	F_h or Eb						
Wind load on above grade RB exterior wall and roof	W						
Tornado wind load above grade RB exterior wall and roof	W_t						
Containment accident flooding	H_a						
Subgrade slab flooding	H_g						
Normal operating and Structural Integrity Test uniform containment pressure	P_0, P_{op}, P_t						
Accident uniform containment pressure	P_a		LB for walls/ slabs within DBA region affected by DBA thermal loads and effective BE elsewhere				

Table 3B-1 Integrated RB Analyses Cases and Model Requirements
(Sheet 3 of 6)

Loading Type	Applicable Design Load(s)	Analysis Method (Computer Program)	Structural Stiffness	Subgrade Properties	Embedded RB Exterior Wall Contact Springs	Mat Foundation Contact Springs	Structural Model
Accident seismic torsion	E_t	Static subgrade stiffness sub-structuring analysis (ANSYS)	Effective BE	LB static massless LB Poisson's ratio LB Young's modulus	Soft in all three directions	Rigid in all three directions	Standalone RB model
Seismic hydrodynamic sloshing load on pool walls	E_s						
Normal snow load on roof	S						
Extreme precipitation (snow) pressure on roof	S_x						
Groundwater hydrostatic pressure on mat foundation (H_b) and RB exterior wall (H_g) below groundwater level	H_g, H_b					Rigid in vertical direction only except at mat foundation center rigid in all three directions	
Normal operating and DBA nozzle, equipment and piping reaction loads	R_o, R_a						
Local loads on containment during DBA	R_r, R_b						
Additional lateral rock pressure on RB exterior wall	H_r						

Table 3B-1 Integrated RB Analyses Cases and Model Requirements
(Sheet 4 of 6)

Loading Type	Applicable Design Load(s)	Analysis Method (Computer Program)	Structural Stiffness	Subgrade Properties	Embedded RB Exterior Wall Contact Springs	Mat Foundation Contact Springs	Structural Model
Normal operating thermal loads	TC1 - Summer	Thermal stress subgrade stiffness sub-structuring analysis (ANSYS)	Effective BE	UB static massless UB Poisson's ratio UB Young's modulus	Rigid in all three directions	Rigid in all three directions	Standalone RB model
	TC2 - Summer			LB static massless LB Poisson's ratio LB Young's modulus	Rigid in normal directions. Soft in the other two directions		
	TC3 - Winter			UB static massless UB Poisson's ratio UB Young's modulus	Rigid in all three directions		
	TC4 - Winter			LB static massless LB Poisson's ratio LB Young's modulus	Rigid in normal directions. Soft in the other two directions		

Table 3B-1 Integrated RB Analyses Cases and Model Requirements
(Sheet 5 of 6)

Loading Type	Applicable Design Load(s)	Analysis Method (Computer Program)	Structural Stiffness	Subgrade Properties	Embedded RB Exterior Wall Contact Springs	Mat Foundation Contact Springs	Structural Model
Design basis accident inside containment - DBA-1	TC5 - Summer	Thermal stress subgrade stiffness sub-structuring analysis (ANSYS)	Effective except SCCV and RPV pedestal use LB	UB static massless UB Poisson's ratio UB Young's modulus	Rigid in all three directions	Rigid in all three directions	Standalone RB model
	TC6 - Summer			LB static massless LB Poisson's ratio LB Young's modulus	Rigid in normal directions. Soft in the other two directions		
	TC7 (10 hours duration) and TC9 (72 hours duration) - Winter			UB static massless UB Poisson's ratio UB Young's modulus	Rigid in all three directions		
	TC8 (10 hours) and TC10 (72 hours) - Winter			LB static massless LB Poisson's ratio LB Young's modulus	Rigid in normal directions. Soft in the other two directions		
Design basis accident inside steam tunnel - DBA-2	TC11 (10 hours) and TC13 (72 hours) - Winter	Thermal stress subgrade stiffness sub-structuring analysis (ANSYS)	Effective except Steam Tunnel uses LB	UB static massless UB Poisson's ratio UB Young's modulus	Rigid in all three directions	Rigid in all three directions	Standalone RB model
	TC12 (10 hours) and TC14 (72 hours) - Winter			LB static massless LB Poisson's ratio LB Young's modulus	Rigid in normal directions. Soft in the other two directions		

Table 3B-1 Integrated RB Analyses Cases and Model Requirements
(Sheet 6 of 6)

Loading Type	Applicable Design Load(s)	Analysis Method (Computer Program)	Structural Stiffness	Subgrade Properties	Embedded RB Exterior Wall Contact Springs	Mat Foundation Contact Springs	Structural Model
Design basis accident in isolation condenser pools and/or reactor cavity and equipment pools - DBA-3	TC15 (10 hours) and TC17 (72 hours) - Winter	Thermal stress subgrade stiffness sub-structuring analysis (ANSYS)	Effective except fuel pool walls/ slabs use LB	UB static massless UB Poisson's ratio UB Young's modulus	Rigid in all three directions	Rigid in all three directions	Standalone RB model
	TC16 (10 hours) and TC18 (72 hours) - Winter			LB static massless LB Poisson's ratio LB Young's modulus	Rigid in normal directions. Soft in the other two directions		
Design basis accident in fuel pool - DBA-4	TC19 (10 hours) and TC21 (72 hours) - Winter	Thermal stress subgrade stiffness sub-structuring analysis (ANSYS	Effective except fuel pool walls/ slabs use LB	UB static massless UB Poisson's ratio UB Young's modulus	Rigid in all three directions	Rigid in all three directions	Standalone RB model
	TC20 (10 hours) and TC22 (72 hours) - Winter			LB static massless LB Poisson's ratio LB Young's modulus	Rigid in normal directions. Soft in the other two directions		

Notes:

(1) 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.

Table 3B-2 Steel-Plate Composite Material Properties

Material Properties		Value in Imperial Units	
Steel Unit weight		490	lb/ft^3
Concrete unit weight	Normal Density Concrete	145	lb/ft^3
	High Density Concrete	219	lb/ft^3
Compressive strength of concrete 8 ksi		8	ksi
Steel Yield Stress grade 50		50	ksi
Steel Yield Stress grade 60		60	ksi
Steel Yield Stress grade 65		65	ksi
Concrete Elastic modulus	Normal Density Concrete	$E_c = \gamma_c^{1.5}\sqrt{f_c'}$ where f_c' is in ksi and γ_c in lb/ft^3	
	High Density Concrete	$E_c = 234\sqrt[3]{f_c'}\left(\frac{\gamma_c}{144}\right)^{1.17}$ where, E_c is in ksi, f_c' is in psi and γ_c in lb/ft^3	
Steel Elastic modulus		29,000	ksi
Concrete Poisson's ratio		0.17	unitless
Steel Poisson's ratio		0.3	unitless
Thermal expansion coefficient		5.5×10^{-6}	/°F
Thermal conductivity coefficient		$k_c = 0.5e^{0.02\gamma_c}$ γ_c in lb/ft^3	Btu·in./(h·ft^2·°F)

Table 3B-3 Structural Steel Properties

Engineering Properties	Value
Unit weight	490 lbf/ft^3
Young's modulus	29,000 ksi
Poisson's ratio	0.3
Thermal expansion coefficient	7.8 x 10^{-6}/ °F

Table 3B-4 Significant Modes of the Fixed Base Integrated Reactor Building Model

Mode	Frequency (Hz)	X	Y	Z	ROTX	ROTY	ROTZ	Mode Shape Description
		Effective Mass/Total Mass (%)						
1	2.42	40	23	0	4	8	0	Primary X mode, RB, SCCV, CEPSS
2	2.47	23	41	0	7	4	0	Primary Y mode, RB, SCCV, CEPSS
5	5.23	0	0	0	0	0	78	Torsion Mode
6	6.83	0	0	19	0	0	0	RB Roof vertical mode
7	7.15	5	8	0	17	10	0	Y second mode, and Pedestal Y
8	7.19	9	5	0	10	17	0	X second mode, and Pedestal X
13	9.03	0	0	50	0	0	0	Primary Z mode, RB, and SCCV
23	13.01	1	3	0	6	3	0	Y third mode, and Pedestal Y
24	13.15	2	1	0	4	5	0	X third mode, and Pedestal X

Table 3B-5 Static Analysis of Fixed Base Integrated Reactor Building Model

Direction	Seismic Weight (kip)	Max Displacement at Roof (in)
X-direction	1.71E5	2.56
Y-direction	1.73E5	2.49
Z-direction	1.72E5	0.40

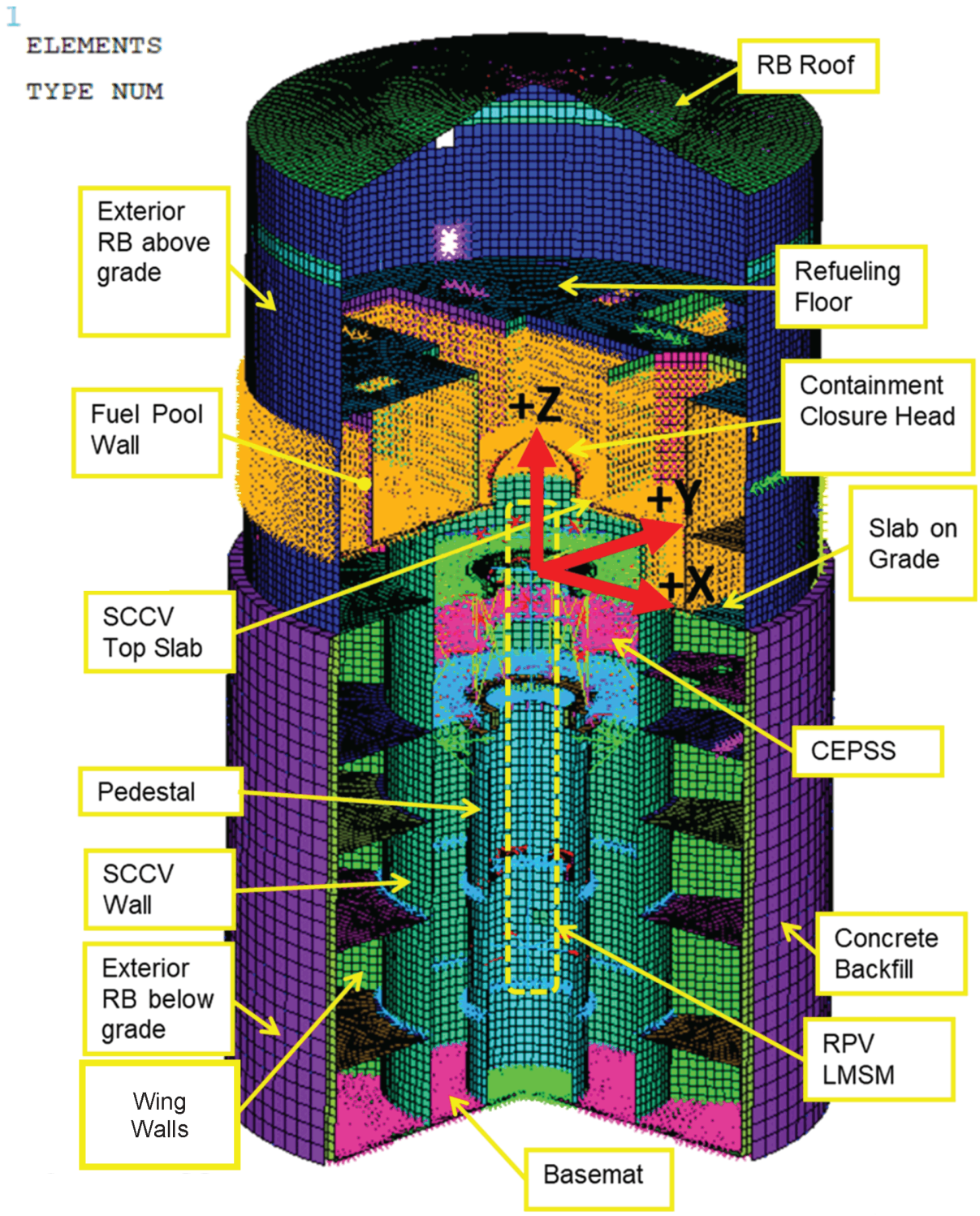


Figure 3B-1 Overview of Integrated RB Finite Element Model

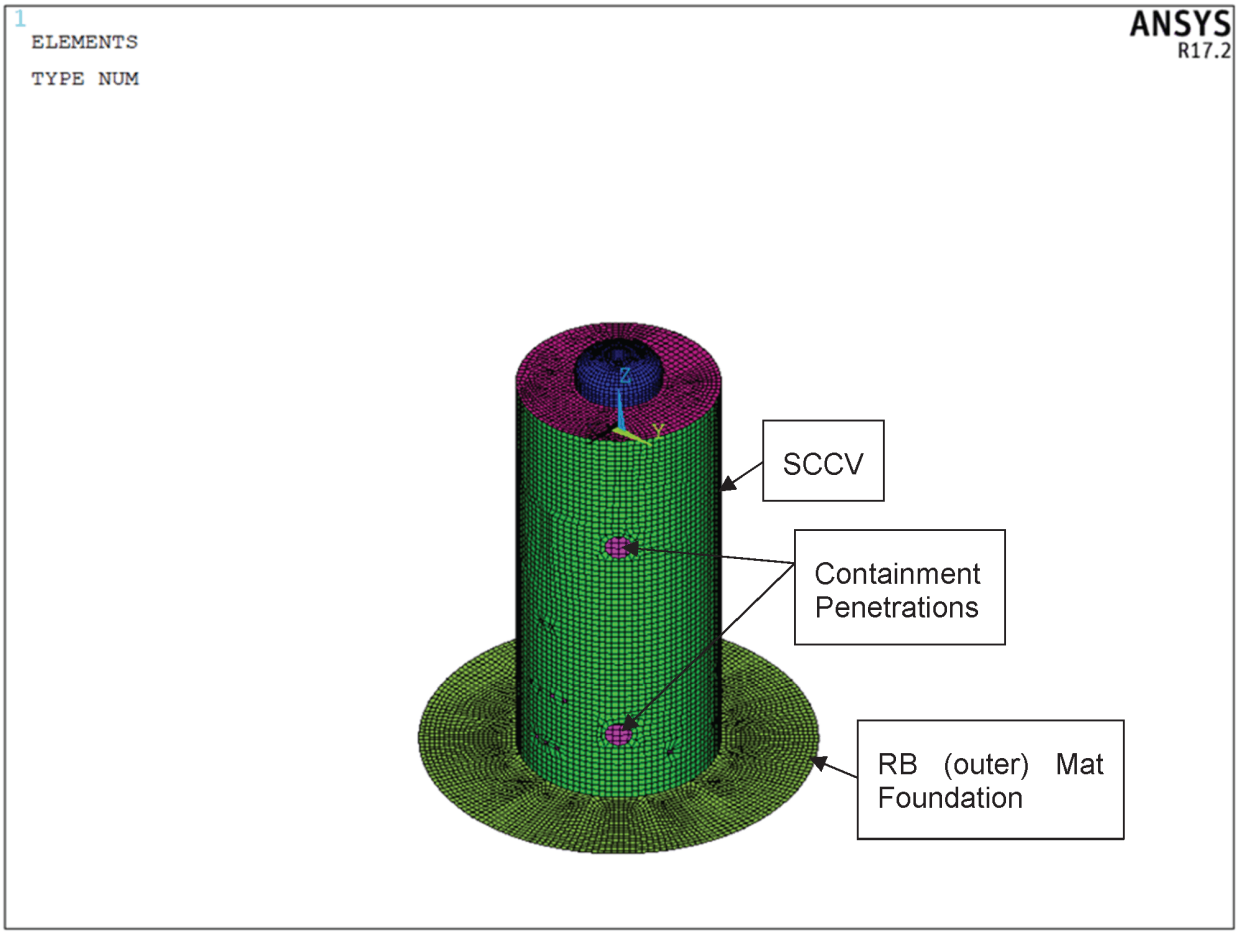


Figure 3B-2 Steel-Plate Composite Containment Vessel Structural Model with Penetrations

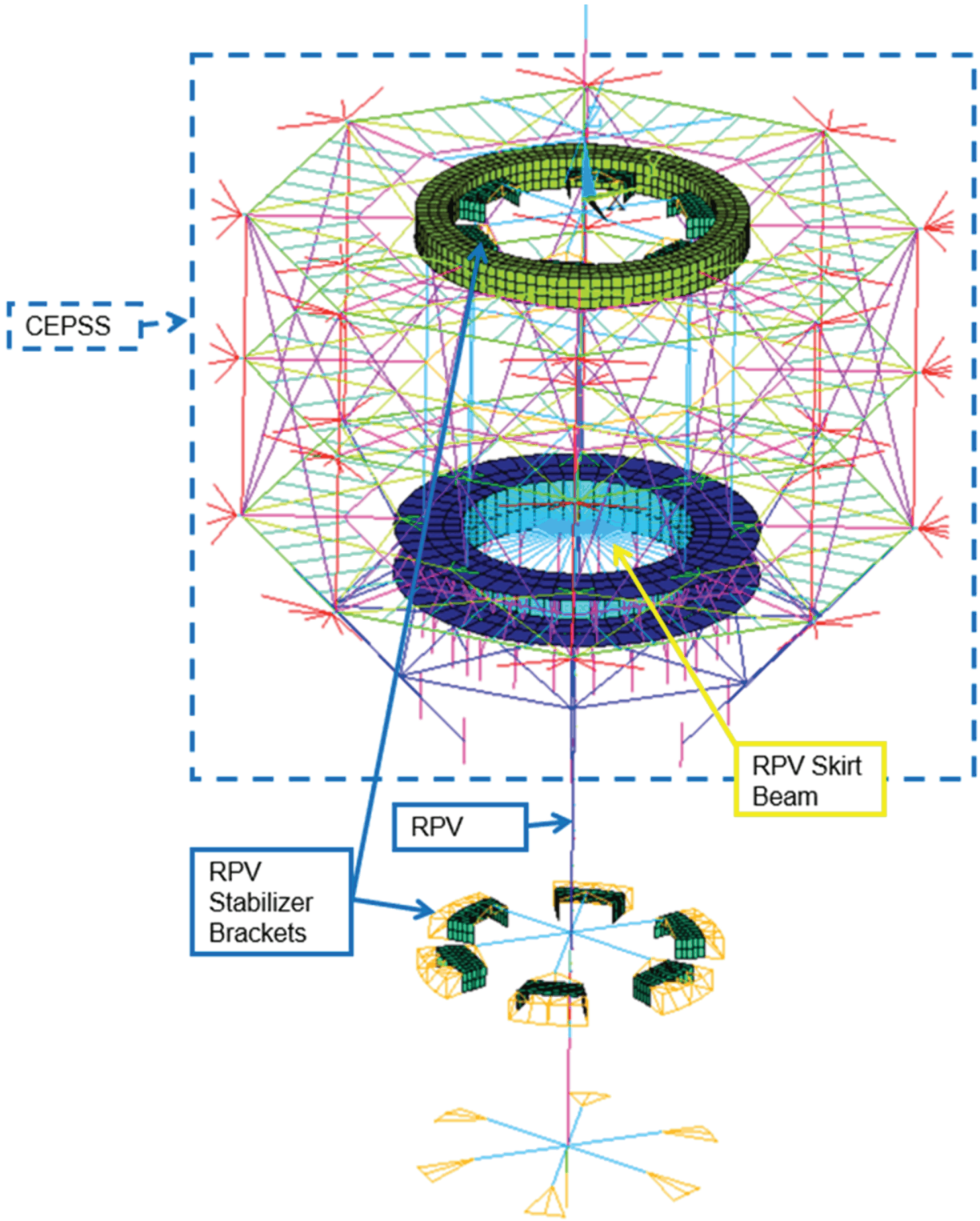
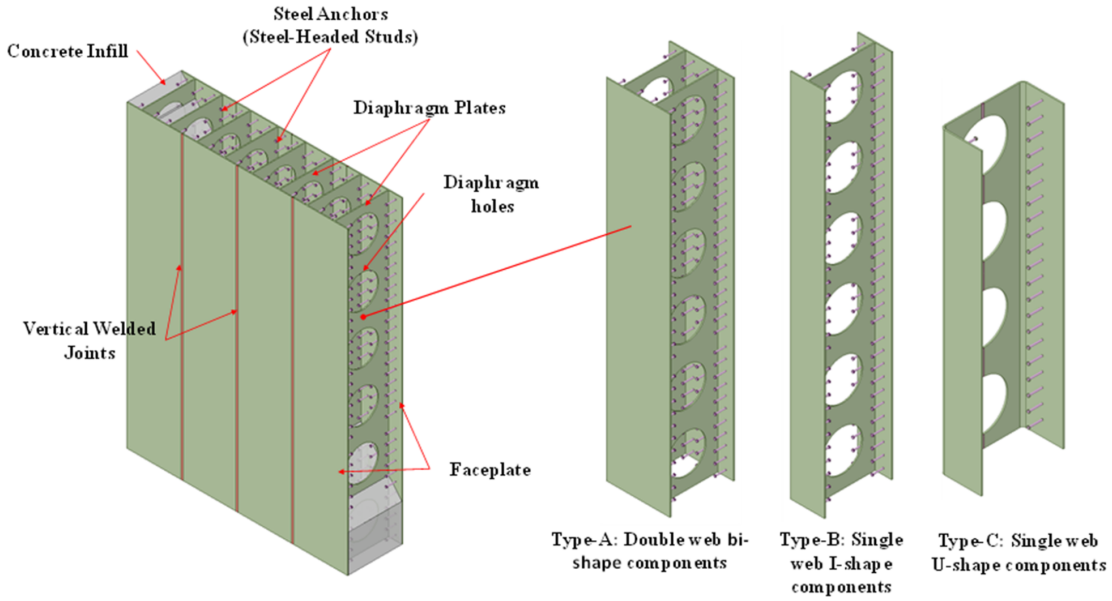
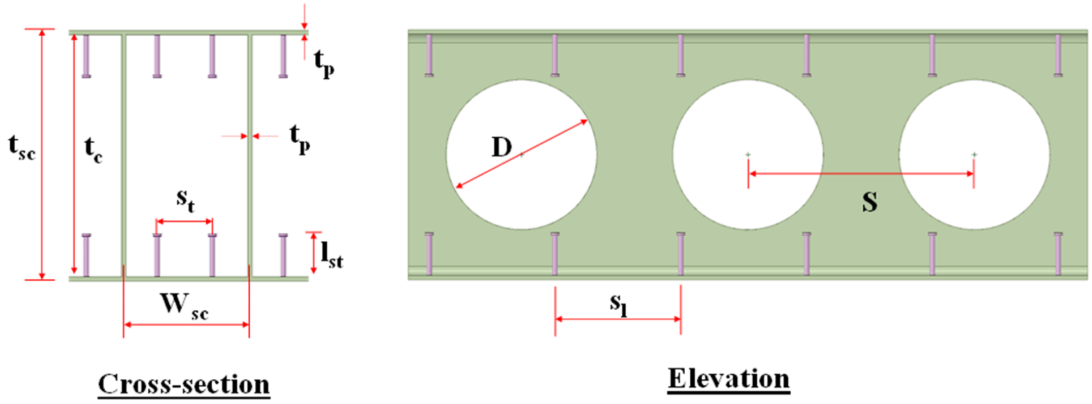


Figure 3B-3 Lumped Mass Stick Model of Reactor Pressure Vessel and Containment Equipment and Piping Support Structure



(a) Diaphragm Plate Steel-Plate Composite Module System



(b) Geometrical Details

Figure 3B-4 Typical Diaphragm Plate Steel-Plate Composite Section and Module System

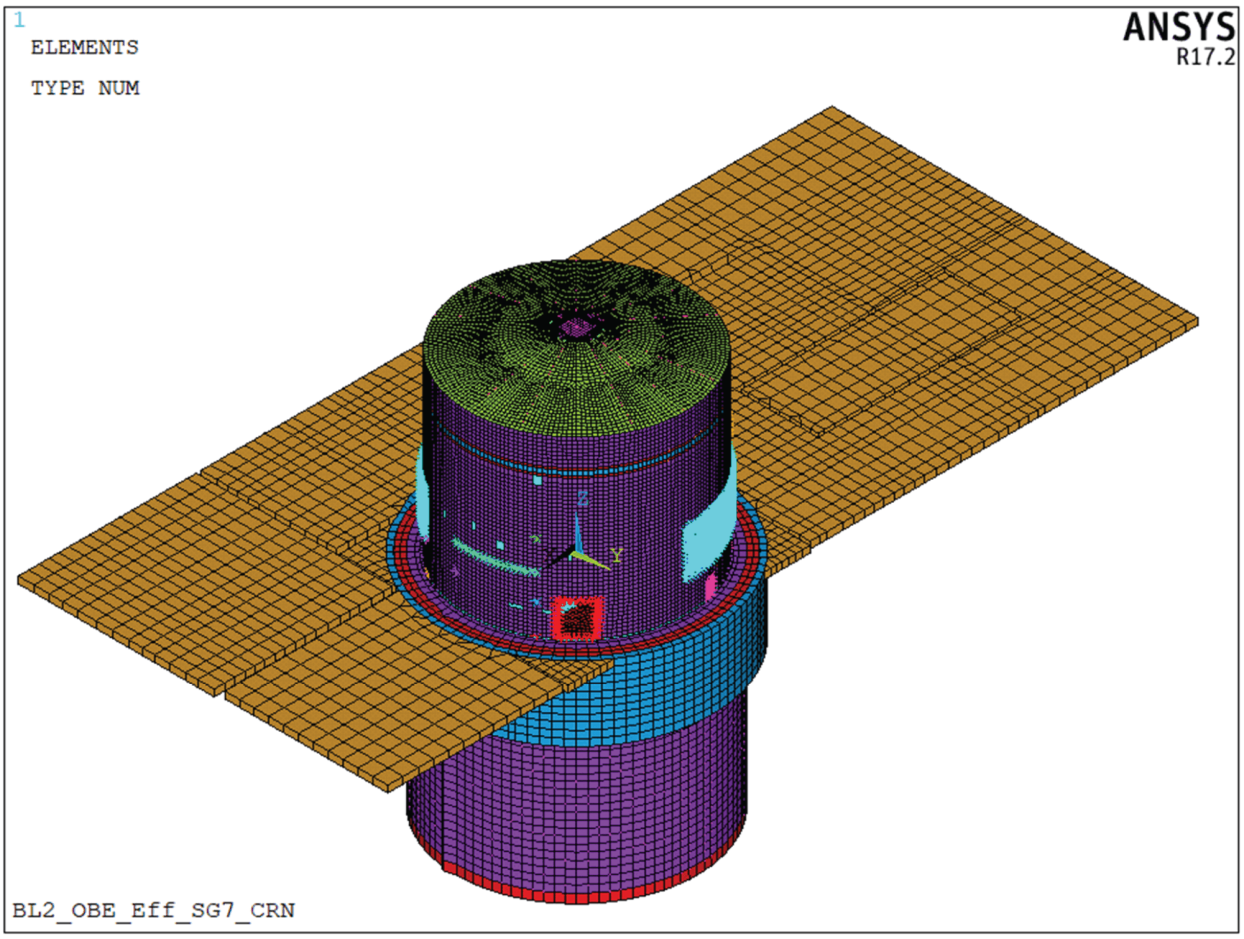
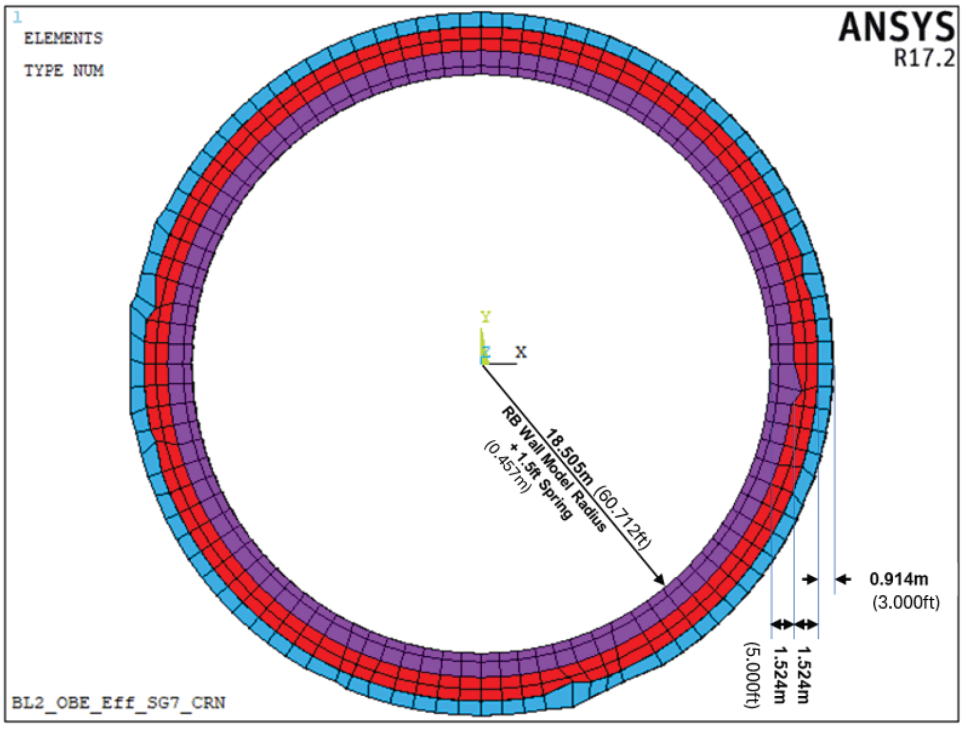
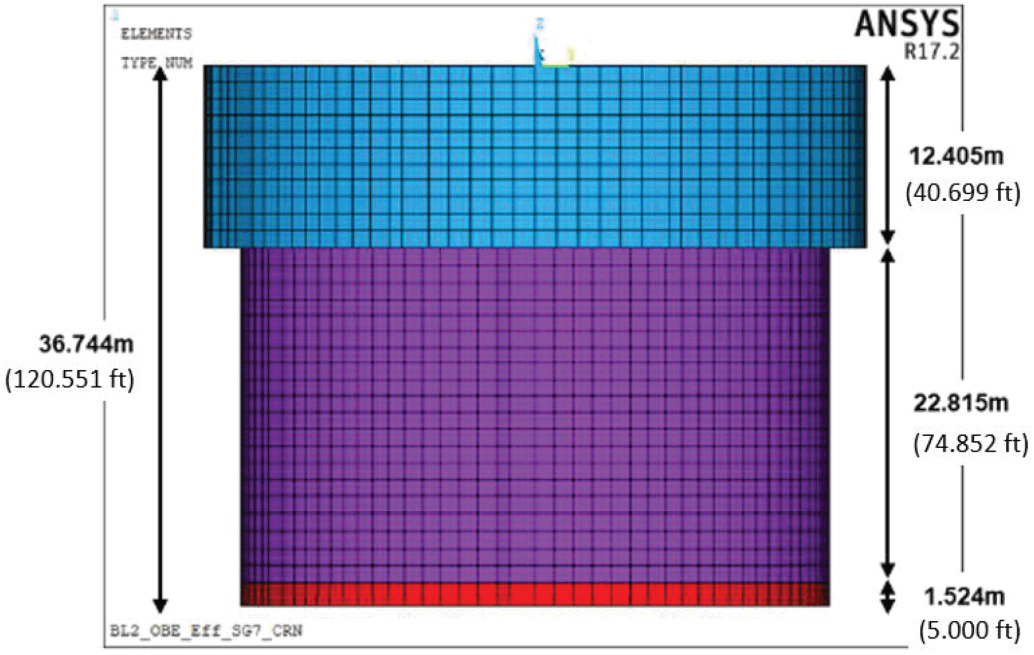


Figure 3B-5 Reactor Building Standalone Soil-Structure Interaction Model (for Case 1)



Plan View



(a) Side View

Figure 3B-6 Backfill, Shelf and Additional Solid Elements Added to Reactor Building Design Basis Seismic Model

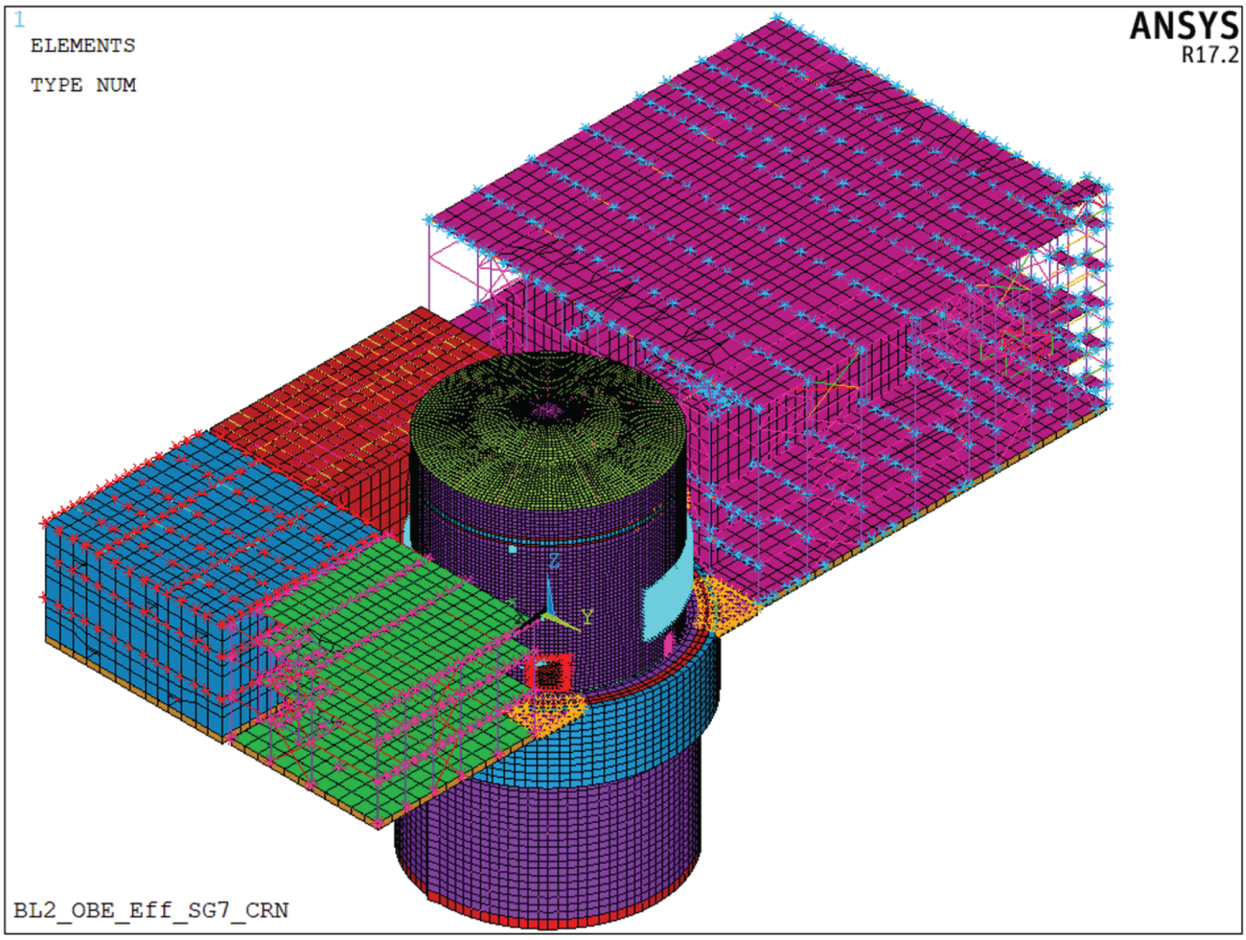


Figure 3B-7 Combined Seismic Soil-Structure Interaction Analysis Model with Clinch River Nuclear Excavation Layout (For Case 2)

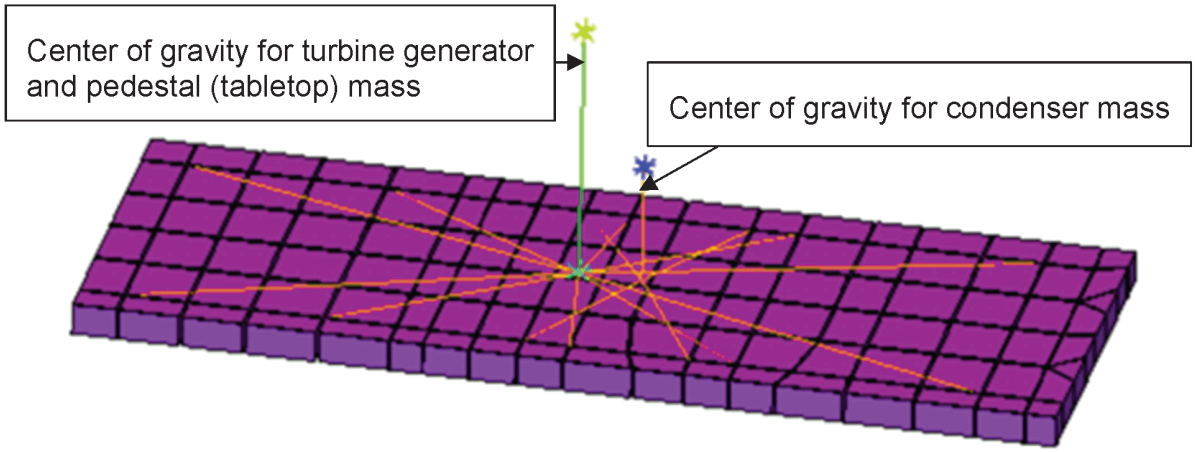


Figure 3B-8 Turbine Generator Pedestal and Isolated Foundation at Grade

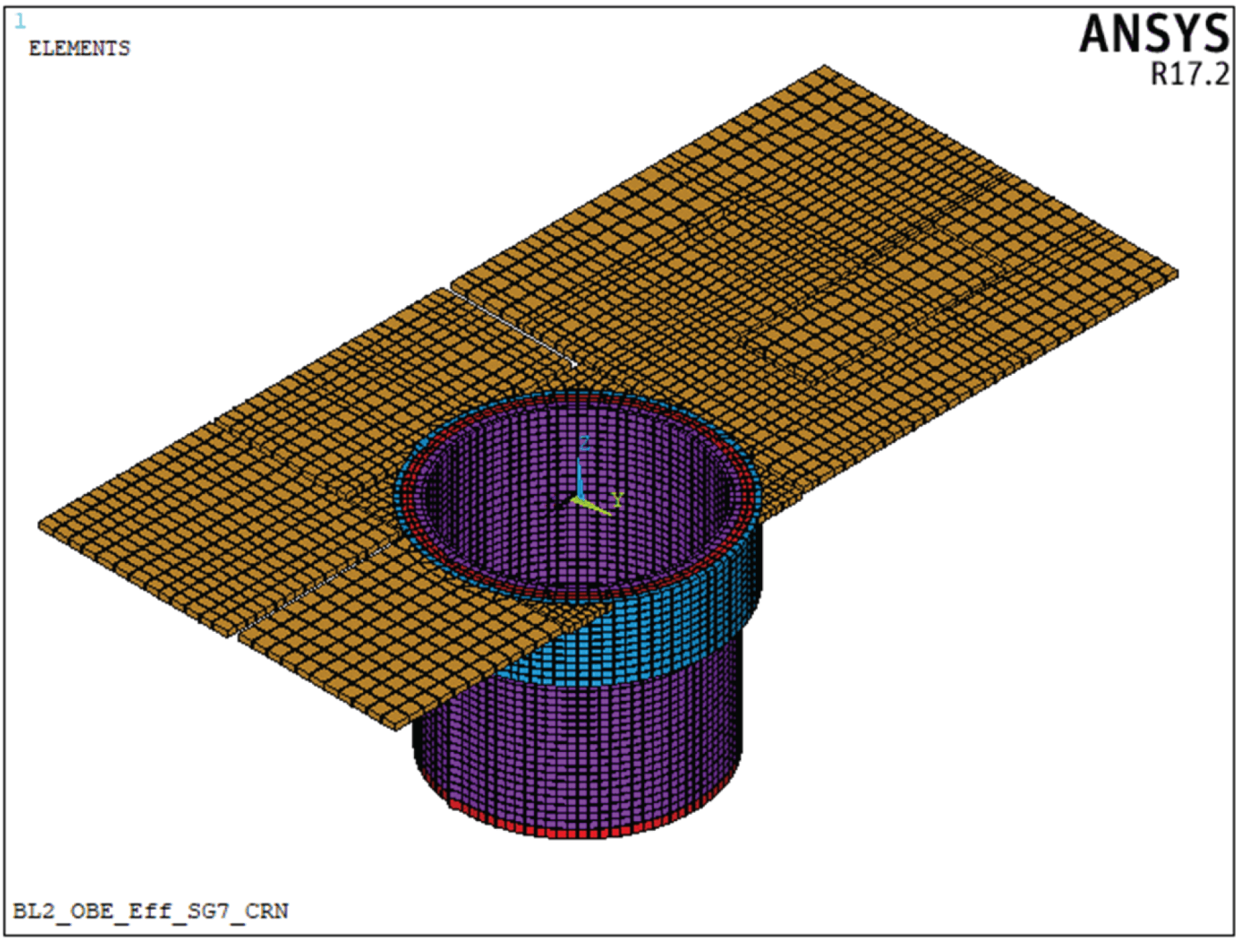


Figure 3B-9 Reactor Building and Power Block Near-Field Subgrade Volume Elements

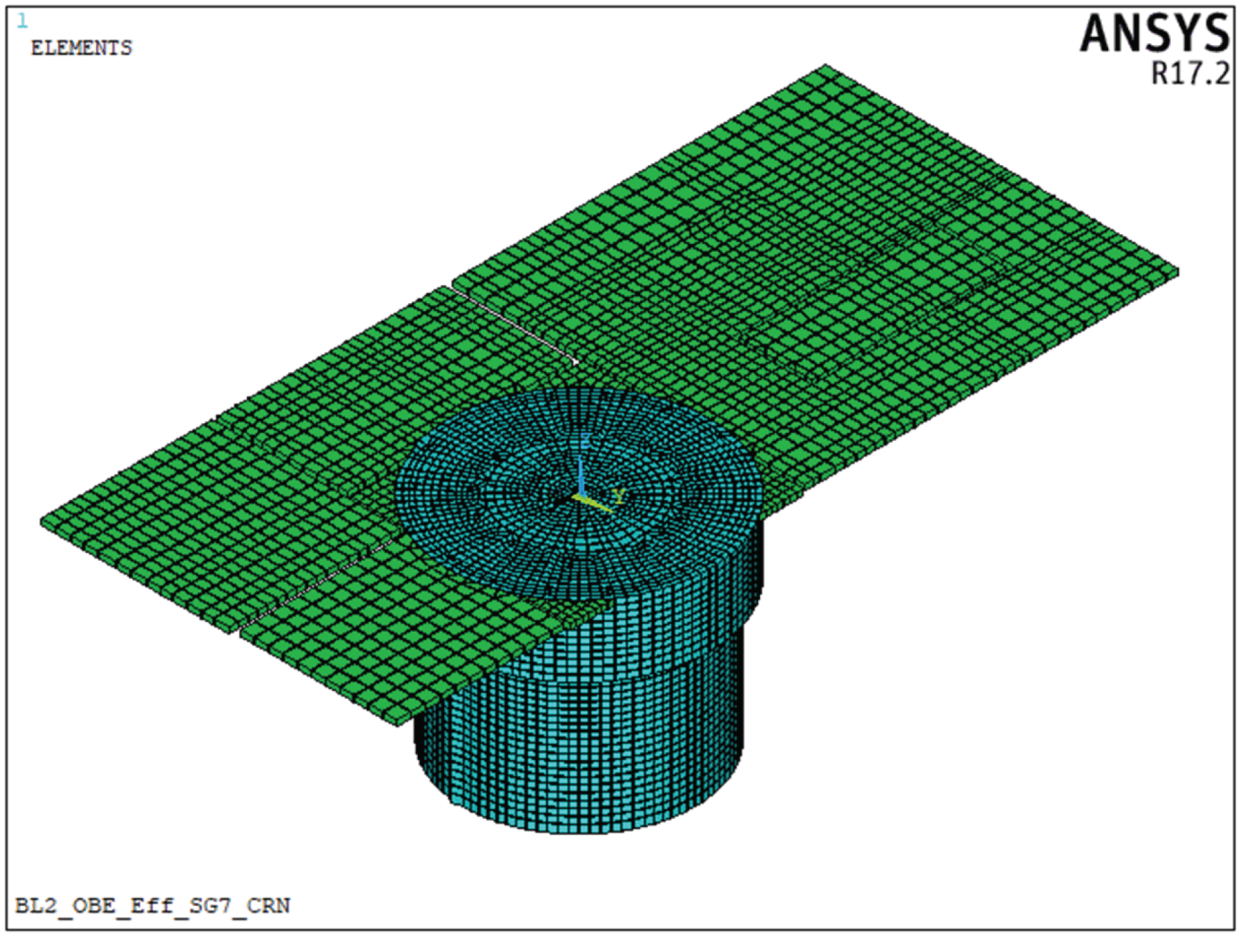


Figure 3B-10 Reactor Building and Power Block Excavated Subgrade Volume Element Model

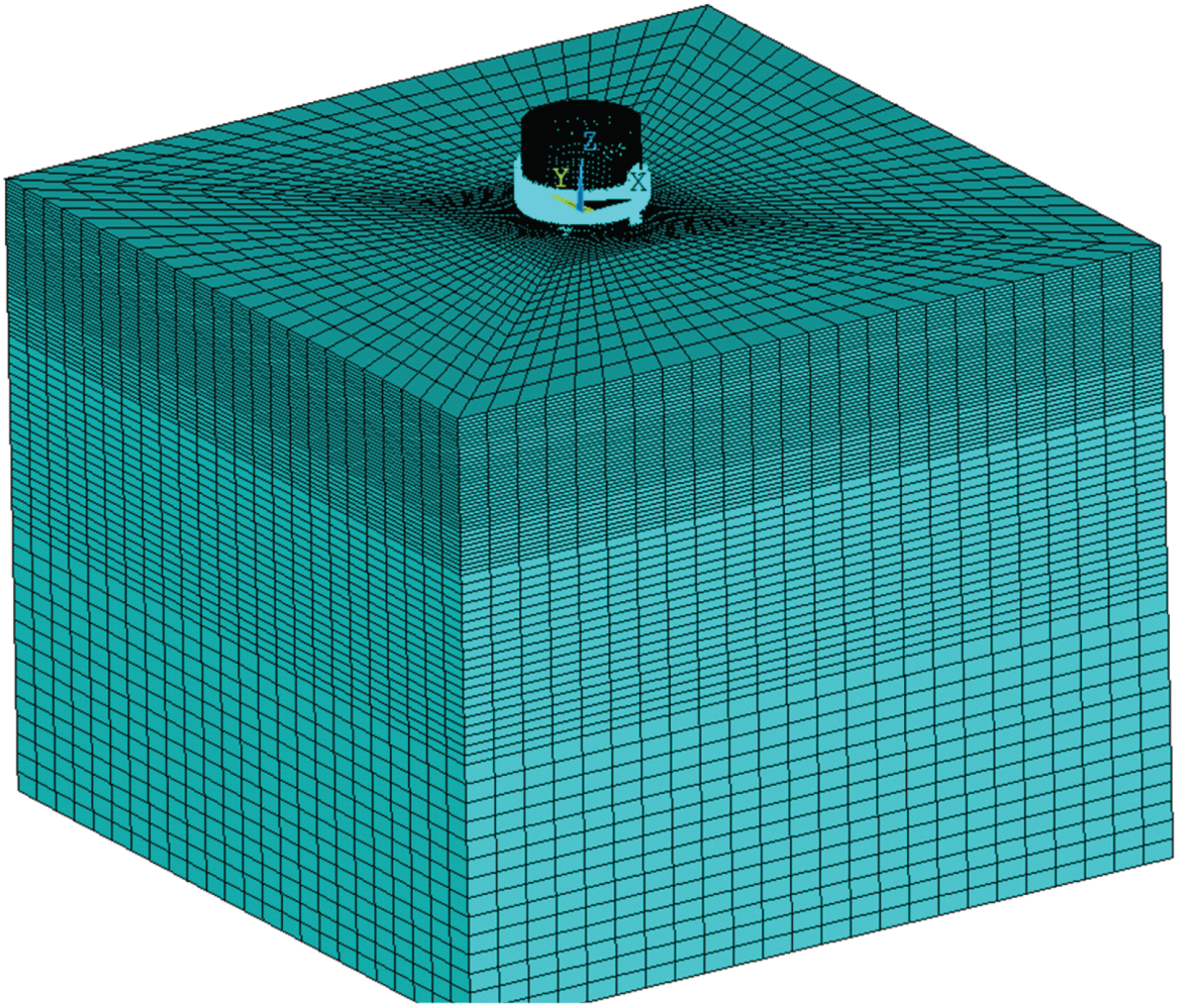


Figure 3B-11 Finite Element Model for Calculation of Subgrade Impedance

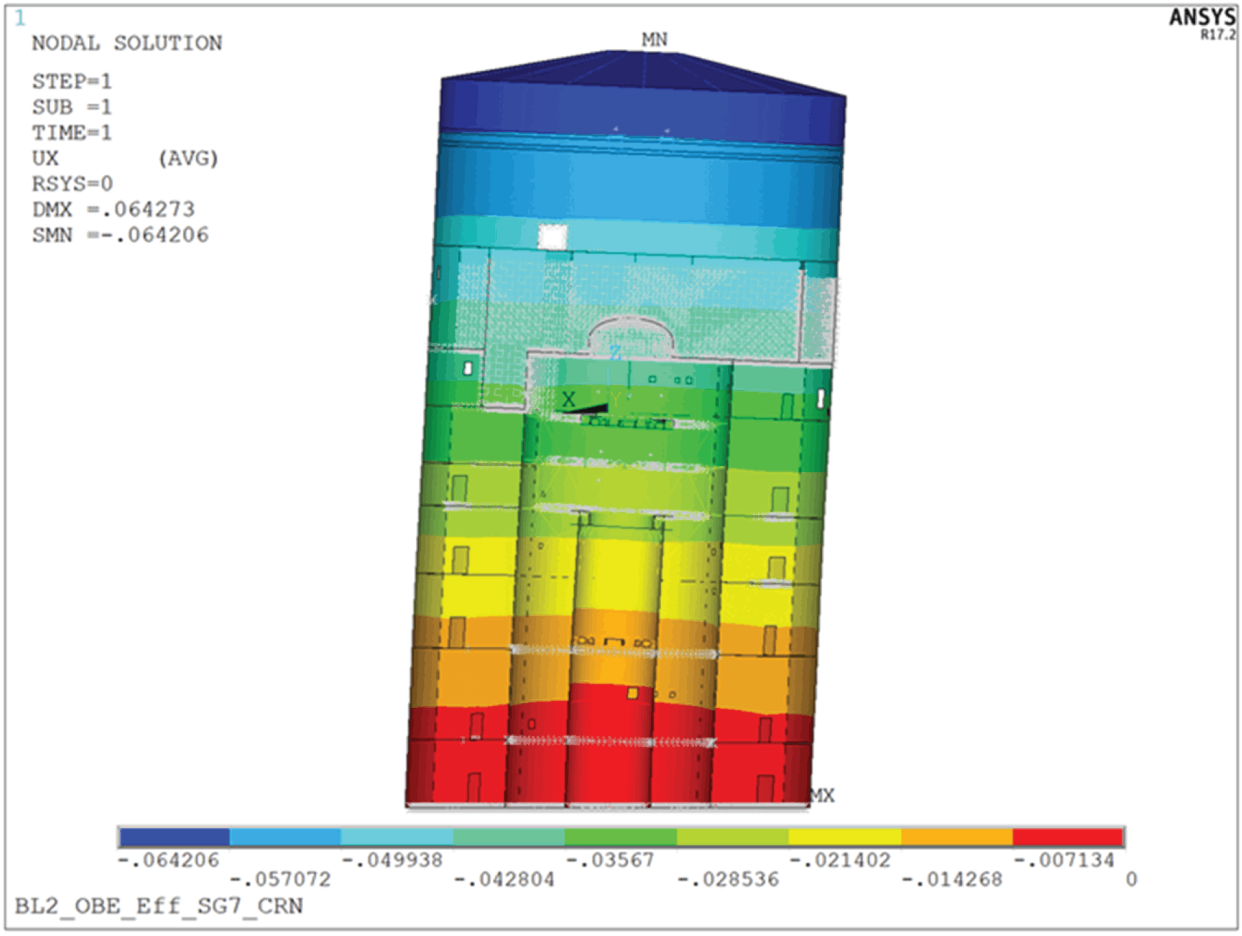


Figure 3B-12 1-g X Primary Deformed Shape (m)

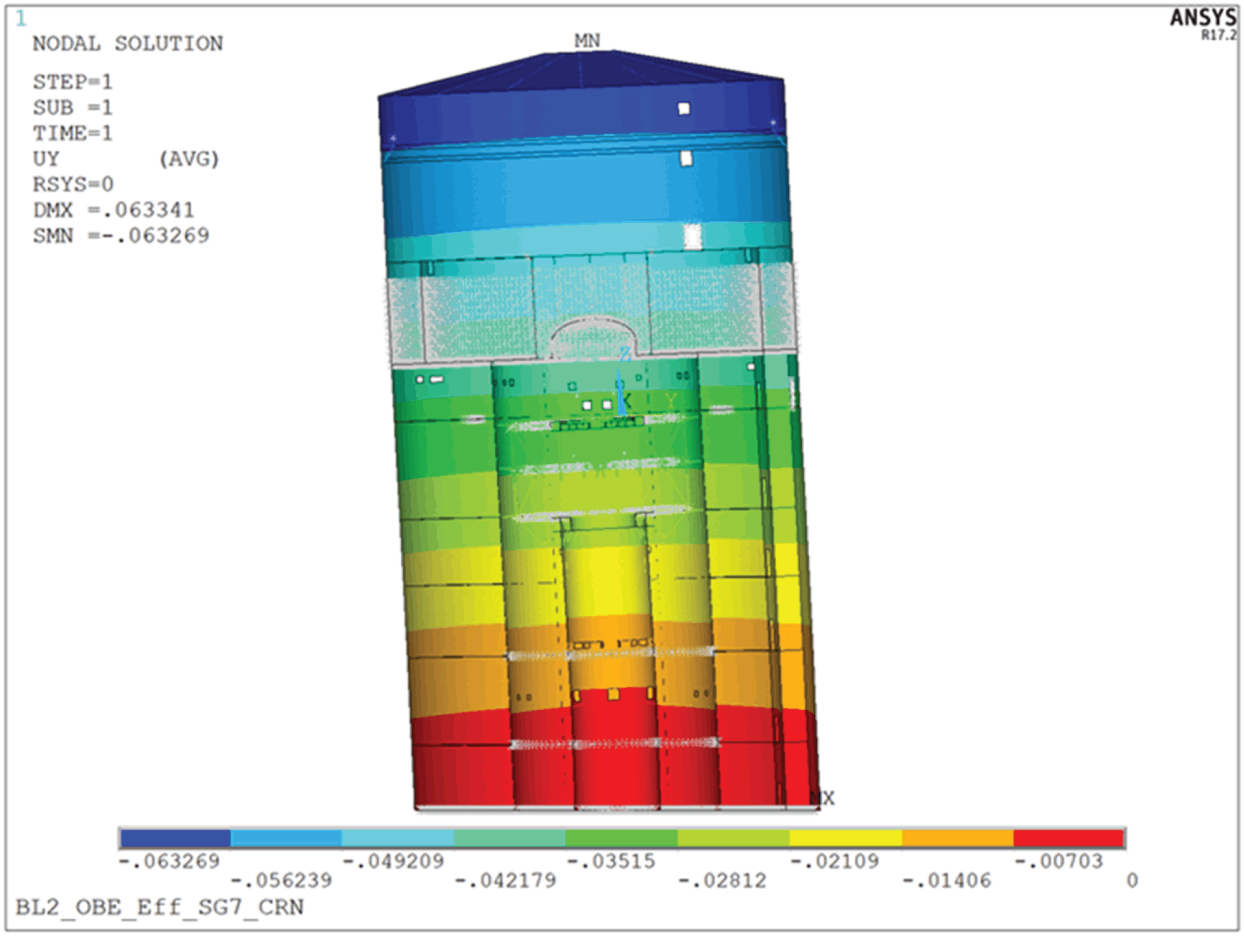


Figure 3B-13 1-g Y Primary Deformed Shape (m)

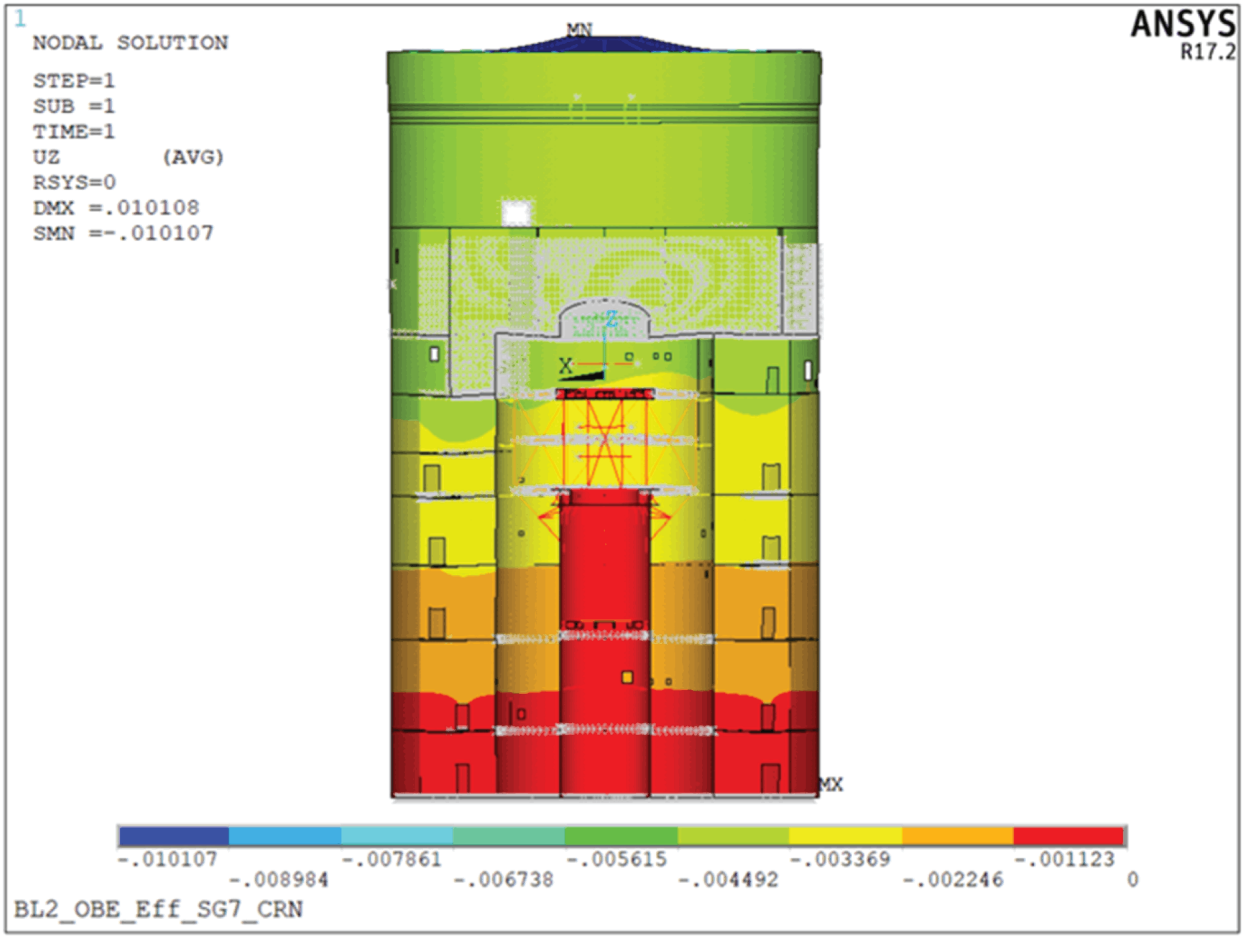


Figure 3B-14 1-g Z Primary Deformed Shape (m)

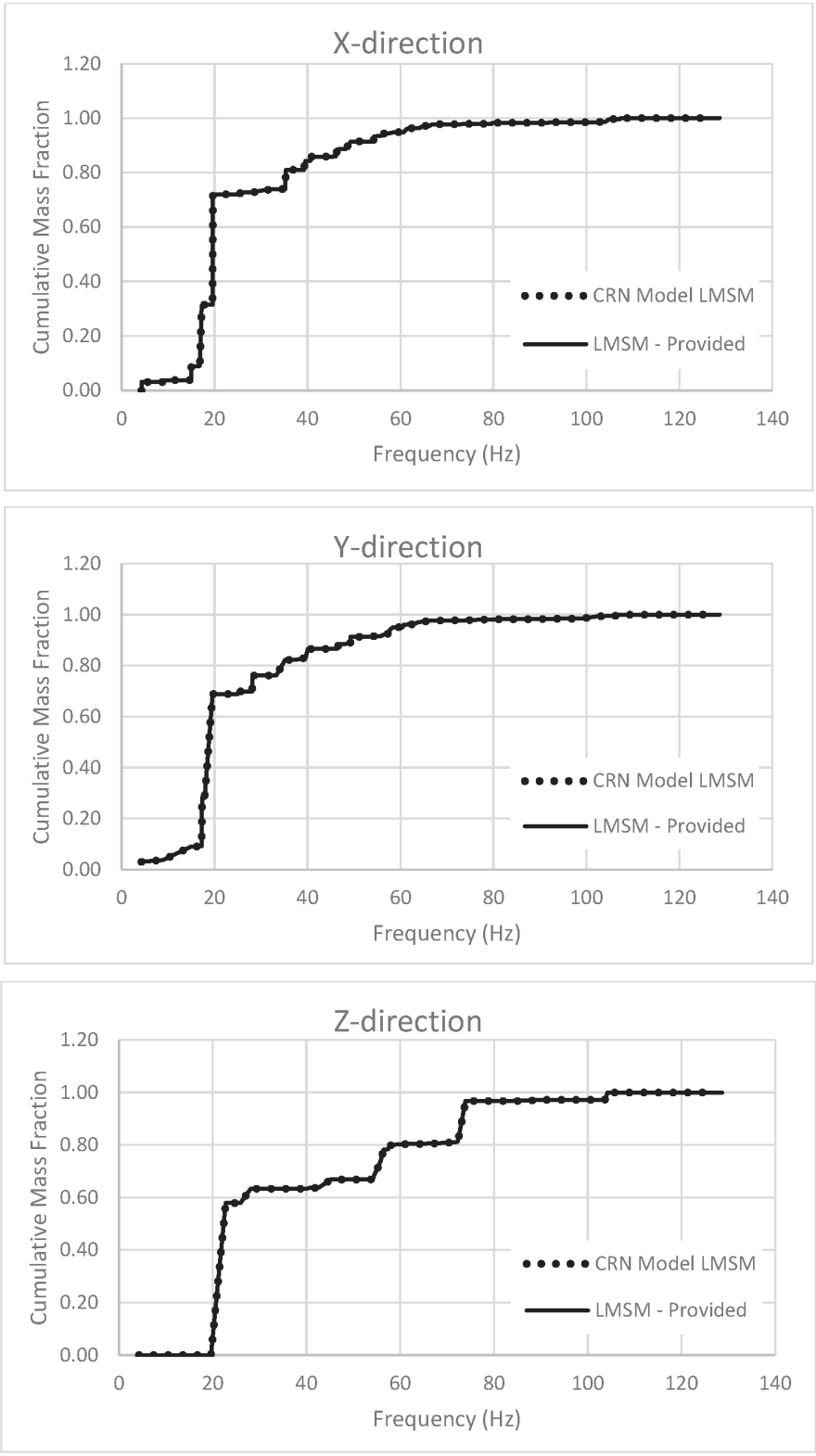


Figure 3B-15 Cumulative Mass Participation Ratio for the Lumped Mass Stick Model of the Reactor Pressure Vessel

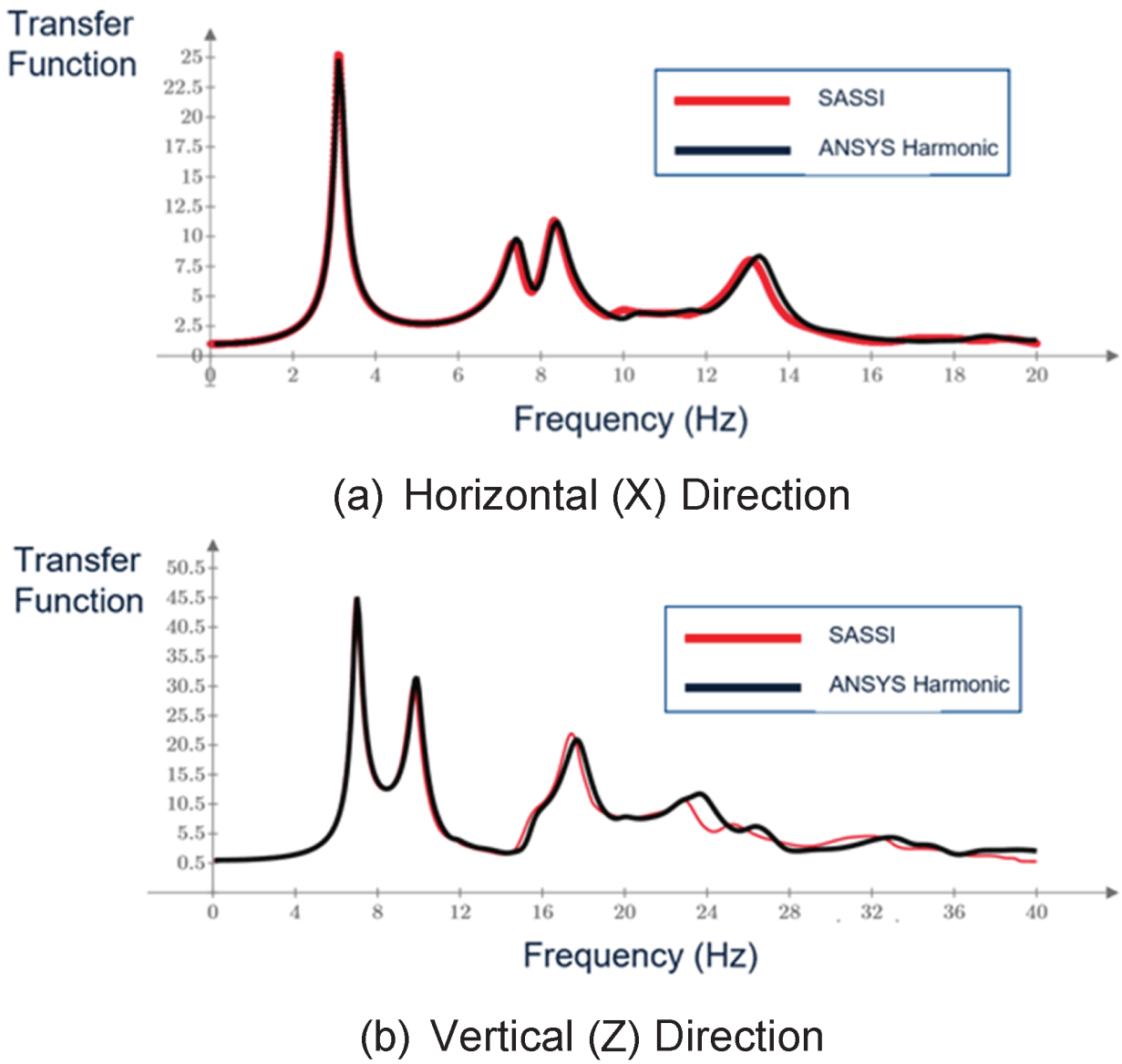


Figure 3B-16 Total Acceleration Transfer Functions at Center of Reactor Building Roof

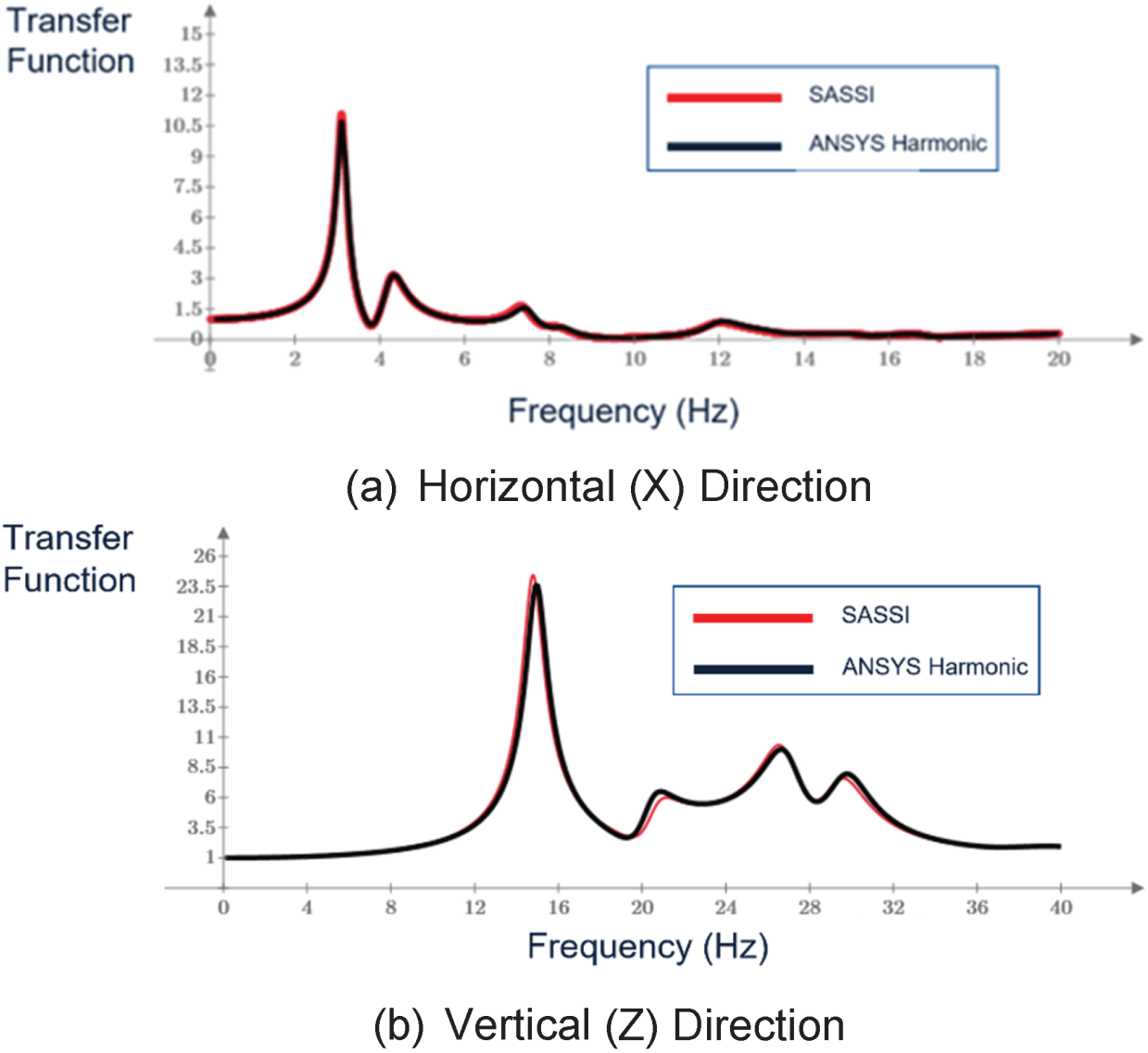


Figure 3B-17 Total Acceleration Transfer Functions at Middle of Fuel

APPENDIX 3C SEISMIC SOIL STRUCTURE INTERACTION ANALYSIS RESULTS

This appendix documents the results of the seismic Soil-Structure Interaction (SSI) analysis of the integrated Reactor Building (RB).

3C.1 Seismic SSI Analysis Methodology

Seismic demands for the design of the Seismic Category I SSCs at the Clinch River Nuclear (CRN) Site are obtained from the SSI analysis of the Three-Dimensional (3D) Finite Element (FE) models discussed in Section 3B.2 evaluated for two free-field site property conditions. Case 1 considers the in-situ soils to be left intact and engineered backfill is added to raise the plant finished grade elevation to 814.5 ft, while Case 2 considers the condition where the residual soil and weathered rock are removed down to competent rock and replaced with engineered backfill up to the plant finished grade. As stated in Section 3B.2, the Case 1 profile is paired with the standalone RB analysis model (excluding other Power Block structures), while the Case 2 profile is paired with the full FE model combining the RB with the adjacent Power Block structures. The subgrade parameters are developed to conservatively cover the variabilities and address uncertainties in the geotechnical properties of the subsurface materials and seismicity of the CRN Site. The consideration of Cases 1 and 2 provides additional margin in the estimated seismic demands for the design of the Seismic Category I structures in the absence of caissons in the analyzed models.

The Case 1 and Case 2 seismic SSI analyses are performed using the one-step analysis approach discussed in Subsection 3.7.2.1 which allows seismic stress demands to be obtained directly from the results of the SSI analyses. In both cases, the control motion is applied to the model at the bottom of the RB with the vertical propagation of horizontal seismic motions through the subgrade considered to be controlled by the shear-wave velocities of the soil/rock layers, and the propagation of vertical motions considered to be controlled by compression-wave velocities.

As stated in Section 3B.2, the dynamic SSI FE models are assigned structural material damping ratios corresponding to Response Level 1 per Table 3-1 of ASCE/SEI 43-19, “Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities,” (Reference 3C-1) for steel-plate composite elements (3%) and metal structures (4%) in lieu of the larger damping ratios corresponding to Response Level 2 as permissible per Regulatory Guide (RG) 1.61, “Damping Values for Seismic Design of Nuclear Power Plants.”

Analysis frequencies selected for each of the subgrade profile cases in the seismic SSI analyses are provided in Section 3C.5. Methodology used to calculate the Case 1 and Case 2 Seismic responses is summarized in Subsection 3C.1.1 through Subsection 3C.1.3.

3C.1.1 Stress Calculation

Shell element forces and moments obtained from the Case 1 and Case 2 seismic SSI analyses are provided at the center of the element with respect to the element local coordinate system. To calculate the co-directional stress demands, the time histories of responses due to the three earthquake components are combined algebraically for each element using the

time-step-by-time-step approach as discussed in Subsection 3.7.2.6. The combined stress demands are then transformed from the local coordinate system into a global cartesian or cylindrical coordinate system for use as input for the design of the structural members.

Average stress results from the corresponding five sets of Acceleration Time Histories (ATHs) represent the response of the structure for each of the SSI analysis cases considered. Stress results obtained from the analyses of the three subgrade profiles for Case 1 and Case 2, and from the sensitivity analysis discussed in Subsection 3C.4.3, are enveloped to provide enveloping CRN site-specific seismic stress demands that account for the uncertainties and variations of the subgrade properties and the seismological conditions at the site.

3C.1.2 In-Structure Response Spectra Calculation

Per Subsection 3.7.2.5.1, the In-Structure Response Spectra (ISRS) are calculated at 301 frequency points equally distributed on the logarithmic scale at the frequency range from 0.1 Hz to 100 Hz to ensure the ISRS are sufficiently close to the peak response frequencies of the supporting structure. The ISRS are calculated at equipment locations for multiple damping ratios spanning the range corresponding to that of equipment under consideration.

Each SSI analysis case is performed separately for each one of the three directional components of input ground motion using five sets of ATHs. For each set, the ISRS obtained from the analysis of each of the three ground motion components are combined to get the total co-directional response using the Square-Root-of-the-Sum-of-Squares method. For each subgrade profile analysis case, average spectra from the corresponding sets of ATHs are computed representing the amplitude and frequency content of the in-structure motion at a particular location as illustrated in Figure 3C-1. The ISRS are calculated as the envelope of the results obtained from the Case 1 and Case 2 SSI analysis cases. To address uncertainties related to the modeling of natural frequencies of the supporting structure and the SSI analysis methodology, peaks of the enveloping ISRS are broadened by the 15% requirement specified in Subsection 3.7.2.5.1 as illustrated in Figure 3C-2. Following the guidelines of ASCE/SEI 4-16, “Seismic Design of Safety-Related Nuclear Structures,” (Reference 3C-2), sharp valleys between peaks may also be filled for use in the design and qualification of SSCs.

If the comparisons of the sensitivity analysis results discussed in Subsection 3C.4.3 and Section 3C.7 indicate that the seismic in-structure responses exceed the design basis seismic responses obtained from the Case 1 and Case 2 seismic SSI analyses by more than 10%, the results of the sensitivity analyses are also included in the design basis ISRS.

3C.1.3 Relative Displacement Calculation

Relative displacements are calculated for key nodal locations throughout the RB relative to the free field at mat foundation elevation. Relative displacements are calculated using the same methodology used for ISRS calculation described in Subsection 3C.1.2, excluding broadening which is not applicable.

3C.2 Key Nodal Locations

As stated in Subsections 3.7.2.2 and 3.7.2.9, responses at key nodal locations are used to demonstrate the accuracy of the seismic SSI analysis results, to determine analysis cutoff frequencies, and to evaluate the effects of different site conditions on the RB SSI response based on comparisons of results from the different seismic SSI analysis cases.

Key nodal locations selected per the criteria provided in Section 5.3.1 of NEDO-33914-A, "BWRX-300 Advanced Civil Construction and Design Approach," (Reference 3C-3) for the verification and validation of the integrated RB seismic SSI analyses and results are provided in Table 3C-1.

3C.3 Key Structural Members

Seismic stress demands are calculated for key structural members and compared between the different SSI analysis cases to understand the effects of site conditions on the seismic response of the soil-structure system.

Table 3C-2 presents the key structural members selected based on their critical support function of Safety Class 1 equipment, critical importance within the lateral force resisting system, or critical location along the seismic load path. These key structural members are also illustrated in Figure 3C-3.

3C.4 Seismic Analysis Cases

The seismic analysis cases performed for the integrated RB are presented in Table 3C-3 and discussed in Subsection 3C.4.1 through Subsection 3C.4.3. Note that the Upper Bound (UB) SSI analysis profiles are used in conjunction with the Lower Bound (LB) damping values, while the LB SSI analysis profiles are used in conjunction with the UB damping values.

The envelope of the responses obtained from the six primary SSI analysis cases presented in Table 3C-3 forms the design basis for the BWRX-300 integrated RB at the CRN site. The other six analysis cases presented in Table 3C-3 are the sensitivity analysis cases performed to evaluate the effects of uncertainties and different site conditions discussed in Subsection 3.7.2.9.

3C.4.1 Case 1

The Case 1 seismic SSI analyses are performed using the standalone integrated RB FE model presented in Figure 3B-5 evaluated using the input response spectra presented in Figure 3.7-3 through Figure 3.7-5 and the three sets of subgrade profiles presented in Figure 3.7-25 and Figure 3.7-26.

The concrete backfill and mud mat dynamic properties assigned to the SSI FE model used in the Best Estimate (BE) Case 1, LB Case 1 and UB Case 1 analyses are presented in Table 3C-4.

3C.4.2 Case 2

The Case 2 seismic SSI analyses are performed using the combined SSI FE model presented in Figure 3B-7 evaluated using the Foundation Input Response Spectra (FIRS) at the bottom of the integrated RB foundation, the Performance Based Surface Response Spectra (PBSRS) at the plant finished grade elevation and the Performance Based Intermediate Response Spectra (PBIRS) at top of the un-weathered rock presented in Figure 3C-4.

Figure 3C-5 and Figure 3C-6 present the strain-compatible shear wave velocity profiles, compression wave velocity and damping ratio profiles used for the Case 2 seismic analyses of the integrated RB.

The concrete backfill and mud mat dynamic properties assigned to the SSI FE model used in the BE Case 2, LB Case 2 and UB Case 2 analyses are the same as those for Case 1 presented in Table 3C-4.

3C.4.3 Sensitivity Cases

The six sensitivity analyses listed in Table 3C-3 and discussed in Subsection 3C.4.3.1 through Subsection 3C.4.3.6 are performed using the combined SSI FE model presented in Figure 3B-7 and the BE Case 2 subgrade profile.

3C.4.3.1 Cracked Sensitivity Analysis

In this analysis, the BE Case 2 model is modified by considering that all concrete made members are fully cracked. Higher SSE-level damping values are assigned to the structural members reflecting higher energy dissipation associated with cracking of concrete. BE soil and rock properties and BE properties of the concrete fill are used. This case is performed to evaluate the variability in structural stiffness on the RB seismic response by considering LB structural stiffness for the RB.

3C.4.3.2 Excavation Support Sensitivity Analysis

Given that the concrete fill around the RB is included in the BE Case 2 model, the uncertainty related to the excavation support provided by the temporary excavation support is investigated by evaluating the base case model supplemented by the explicit modeling of the temporary excavation support. Solid elements are used to represent the BE dynamic properties of the excavation support corresponding to a concrete compressive strength of 3,500 psi, which is the same as that of the BE concrete backfill properties.

3C.4.3.3 No-Friction Sensitivity Analysis

The technique used for the RB shaft construction and corrosion allowance can affect the friction at the interfaces between the RB exterior wall and the surrounding concrete fill material. This case is performed to investigate the potential effects on the RB seismic response when, for short instances of time, parts of the below-grade concrete backfill may slip in tangential or vertical

directions with respect to the RB exterior wall. The analysis is performed by modifying the BE Case 2 model such that it simulates a complete lack of traction between the concrete backfill and the exterior wall of the RB from ground surface to the bottom of the foundation.

3C.4.3.4 Dry Sensitivity Analysis

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 considered to be located at finished grade as discussed in Subsections 3.7.1.3 and 3.7.2.9.3. The potential effects of groundwater level variability on the seismic design are addressed by comparing the seismic responses obtained from the SSI analysis of the dry soil profile. The dry soil profile uses BE soil dynamic properties representative of the conditions of the LB groundwater level at the site. In this analysis, the groundwater is conservatively considered to be located below the RB foundation bottom elevation, thus unsaturated soil Poisson's ratios are used.

3C.4.3.5 Separation Sensitivity Analysis

The separation sensitivity analysis is performed to investigate the effects on the RB seismic response when, for short instances of time, parts of the below-grade concrete backfill may separate from the surrounding subgrade soil. This separation would occur on one side of the backfill and along a portion of the embedment depth. The analysis is performed by modifying the BE Case 2 model such that it simulates a complete separation between the concrete backfill and the surrounding top layers of subgrade soil. The separation is conservatively considered for the full depth of soil, down to the top of un-weathered rock, and for the entire potential separation surface around the backfill.

3C.4.3.6 Lean Concrete Sensitivity Analysis

The use of concrete fill as the design basis analysis case introduces the additional uncertainty over the strength of the concrete fill which is not designed to maintain its structural integrity throughout the entire operational life of the plant. To address this uncertainty, a sensitivity evaluation is performed using a reduced value of the design concrete fill strength corresponding to a concrete compressive strength of 1,500 psi.

3C.5 Design Basis Seismic Analysis Cases and Selected Frequencies of Analysis

As stated in Section 3C.4, the six primary SSI analysis cases presented in Table 3C-3 form the design basis for the integrated RB at the CRN Site.

The six primary seismic SSI analysis cases are performed for frequencies starting at 0.012 Hz to the respective cutoff frequency for each of the subgrade profiles. Following the selection criteria in Section 5.3.2 of NEDO-33914-A and as indicated in Table 3C-5, the following cutoff frequencies are adopted for the analysis:

- 35 Hz for the LB Cases 1 and 2
- 51 Hz for the BE Cases 1 and 2
- 70 Hz for the UB Cases 1 and 2

For the UB Cases 1 and 2, the 70 Hz cutoff frequency allows for the adequate calculation of ISRS up to a minimum of 50 Hz.

The SASSI frequency domain analysis calculates the response of the system at selected "Structure" frequencies defined as the number of frequencies necessary to adequately characterize the response of the full SSI system. The results for the response at frequencies between the selected frequencies are interpolated.

The calculated frequencies are narrowly spaced with a total of 197 frequencies for the UB case, which ensures that Acceleration Transfer Function (ATF) peaks and troughs are adequately captured and minimizes ATF interpolation concerns.

3C.5.1 Subgrade Properties Variation Effects

Figure 3C-7 compares the ATFs representing the co-directional responses at the RB exterior wall at grade in the three orthogonal directions obtained from the six primary SSI analysis cases.

The plots in Figure 3C-7 show distinct dominant frequencies for the SSI system depending on the subgrade profile analysis. Following the expected trend, the ATF peak frequencies obtained from the LB profile are associated with the lowest frequency, followed by higher ATF peak frequencies for the BE profile and with the highest ATF peak frequency associated with the UB profile as shown in Figure 3C-7. The dots in the ATF plots in Figure 3C-7 represent the calculated "Structure" frequencies, and the smooth curves in between show the interpolated ATFs.

Figure 3C-8 compares the ATF results obtained from the analysis of the six primary and six sensitivity cases listed in Table 3C-3. The peak at the smallest frequency of around 3.5 Hz is associated with the "No-Friction" sensitivity case. Figure 3C-9 presents the ATF normalized by the ATF of the free-field outcrop motion corresponding to the RB exterior wall at grade. These ATFs are normalized with respect to the free-field response at surface therefore removing the large amplitude ATF peaks at the embedment soil column frequencies to better depict the peak responses of the RB structure. The amplitudes of the normalized ATFs fall well below unity at frequencies that are smaller than their respective cut-off frequencies thus providing confirmation of the adequacy of the selected cut-off frequencies of analysis. The exception in the LB Case 1 normalized ATF is due to the amplitude of the free-field ATF (used in the denominator to calculate the normalized ATF) approaching zero at frequencies larger than 20 Hz, which results in the artificially high amplitude of the normalized ATF.

3C.6 Seismic Response Characteristics of the Integrated Reactor Building at the Clinch River Nuclear Site

3C.6.1 Seismic Structural Stress Responses for Cases 1 and 2

This section presents plots of main stress components in key structural members to depict the characteristics of how the integrated RB structures resist the CRN site-specific seismic loads. Member force and moment results obtained from the analysis of the primary SSI cases are compared to evaluate the effect of geotechnical and seismological site conditions on the seismic stress responses of the integrated RB structures.

Figure 3C-10 through Figure 3C-21 compare the in-plane membrane forces and out-of-plane bending moments on the below grade portions of the RB exterior wall obtained from the LB, BE and UB Cases 1 and 2. Due to the cylindrical shape of the wall, the out-of-plane dynamic pressure forces applied on the below grade portion of the RB exterior wall are resisted by the tangential (hoop) membrane forces as shown in Figure 3C-10 through Figure 3C-12 and in-plane shear forces as shown in Figure 3C-13 through Figure 3C-15. Comparing the hoop forces due to the LB, BE, and UB cases in Figure 3C-10 through Figure 3C-12, the LB case is found to provide the largest seismic demands followed by the BE and UB cases. This can be attributed to the larger dynamic earth pressures applied on the deeply embedded structure supported by the softer soil layers. A similar pattern is observed for the in-plane shear forces shown in Figure 3C-13 through Figure 3C-15, and vertical forces shown in Figure 3C-16 through Figure 3C-18. The plots in Figure 3C-19 through Figure 3C-21 show that the out-of-plane horizontal moments are of a small magnitude with peak amplitudes at the connections of the RB exterior wall with the interior floors and wing walls.

The RB cylindrical shape helps reduce the demands on the RB wing walls that run in the radial direction connecting the RB exterior wall to the SCCV wall. This is demonstrated by the in-plane shear forces in wing walls illustrated in Figure 3C-22 through Figure 3C-24.

3C.6.2 Seismic In-Structure Responses for Cases 1 and 2

Four percent (4%) damped ISRS for responses at key nodal locations obtained from the analysis of the six primary SSI cases and the six sensitivity cases (see Section 3C.7) are compared as shown in Figure 3C-25 for the integrated RB common mat foundation to evaluate the effect of subgrade variations on the RB in-structure response.

At the mat foundation, the response is dependent on the subgrade profile, as can be seen by the increasing peak frequencies with increasing profile stiffness in Figure 3C-25. The BE Case 2 and the six sensitivity cases which share the same subgrade properties have similar in-structure responses at the center of the common mat foundation, with the exception of the vertical ISRS for the “Dry” case. This exception is due to the differences in the compression wave velocities of the top 37.4 ft of soil layers.

3C.6.3 Ground Contact Ratio under RB Foundation

In accordance with SRP 3.7.2.II.4, the ground contact ratio under the RB foundation is calculated to confirm the linear SSI analysis methodology is acceptable for the BWRX-300 design. The minimum ground contact ratio is calculated for each of the six primary SSI cases and for each of the 5 time-histories associated with each case by comparing the seismic vertical forces in the springs below the RB foundation with those obtained from static analysis under operating condition gravity loads. Net tension load in a spring indicates that the seismic component exceeds the static component, i.e., uplift. Conversely, net compression in a spring indicates that there is contact with the ground. Because the springs are approximately uniformly distributed under the RB foundation, the ratio of number of springs in compression to the total is considered a reasonable proxy for the ground contact ratio.

A review of the ground contact ratios calculated for all cases confirms the minimum ground contact ratio is 86.6%, which is greater than the minimum acceptance criterion of 80% per SRP 3.7.2.II.4. Thus, the linear SSI analysis methodology is acceptable for the BWRX-300 design.

3C.7 Sensitivity Analysis Cases

As stated in Subsection 3.7.2.9, sensitivity evaluations are performed to investigate the effects of structural stiffness variation, soil-structure interface conditions, soil separation, groundwater variations, and excavation and backfill properties on the seismic response and design of the CRN RB SSCs.

The sensitivity evaluations are based on results of sensitivity SSI analyses performed using the BE subgrade profile properties and models of the combined SSI model shown in Figure 3B-7 representing conditions that bound the variation of the evaluated SSI parameter. The sensitivity evaluations are performed by comparing in-structure responses at key locations and stress demands for key structural members within the RB. Subsection 3C.7.1 through Subsection 3C.7.4 summarize the findings of these sensitivity evaluations.

Per guidance of Section 5.3 of NEDO-33914-A, the results of the sensitivity analysis are included in the RB seismic design basis when comparisons show significant exceedances, greater than 10%, in the RB seismic response due to any of the examined effects. To do so, results of the sensitivity cases are enveloped with those of the Case 2 design basis analyses of LB, BE, and UB subgrade profiles. The enveloped results are then used to develop seismic member force and moment demands, and design ISRS, following the methodology in Subsection 3C.1.1 and Subsection 3C.1.2 for the Seismic Category I SSCs at the CRN Site.

In general, the comparisons show that seismic forces and moments do not differ significantly from the base case BE Case 2 with some exceptions, notably for the No-Friction and Cracked sensitivity cases as demonstrated in Subsection 3C.7.1 through Subsection 3C.7.5.

3C.7.1 Structural Stiffness and Damping Variation Effects on Responses

Figure 3C-26 through Figure 3C-30 compare stress responses obtained from the cracked sensitivity analysis discussed in Subsection 3C.4.3.1 and the BE Case 2 analysis discussed in Subsection 3C.4.2 at the RB exterior wall, SCCV top slab and RPV pedestal.

The comparisons of stress responses obtained from the models with BE (Effective) and LB (Reduced) stiffness properties at the RB exterior wall, the RPV pedestal and at the SCCV top slab show a noticeable decrease in stress demands due to the reduction in stiffness. In the case of the RB exterior wall, the decrease in stress demands is around 35% for vertical forces (Figure 3C-26) and 25% for in-plane shear forces (Figure 3C-27). The out-of-plane moment decrease in the SCCV top slab ranges from 10% to 15% (Figure 3C-28), while the RPV pedestal demand decrease is approximately 20% for vertical (Figure 3C-29) and 15% for in-plane shear forces (Figure 3C-30).

3C.7.2 Excavation Support, Backfill and Friction Effects on Responses

Figure 3C-31 through Figure 3C-35 compare stress responses obtained from the excavation support sensitivity analysis discussed in Subsection 3C.4.3.2 and the BE Case 2 analysis discussed in Subsection 3C.4.2 at the RB exterior wall, SCCV top slab and RPV pedestal. The comparisons show that the overall effects of excavation support on these structures are small.

Figure 3C-36 through Figure 3C-40 compare stress responses obtained from the no-friction sensitivity analysis discussed in Subsection 3C.4.3.3 and the BE Case 2 analysis discussed in Subsection 3C.4.2 at the RB exterior wall, SCCV top slab and RPV pedestal. The comparisons show a large increase in vertical forces in the below-grade segment of the RB exterior wall (about 4 times that of BE Case 2, see Figure 3C-36) due to the absence of frictional bond between the wall and the surrounding concrete backfill. An increase of about 30% (Figure 3C-38) is also observed in the SCCV top slab out-of-plane moments. Similarly, for the RPV pedestal, Figure 3C-39 and Figure 3C-40 indicate an increase of about 25% for vertical and 15% for in-plane forces.

3C.7.3 Groundwater Variation Effects on Responses

Figure 3C-41 through Figure 3C-45 compare stress responses obtained from the “Saturated” and “dry” subgrade profiles (see Subsection 3C.4.3.4) at the RB exterior wall, SCCV top slab and RPV pedestal.

The comparisons show that the overall effect of groundwater variation on the structures is small, with the largest effect being an increase in the SCCV top slab out-of-plane moments of about 15% (Figure 3C-43).

3C.7.4 Soil Separation Effects on Responses

Figure 3C-46 through Figure 3C-50 compare stress responses obtained from the separation sensitivity analysis discussed in Subsection 3C.4.3.5 and the BE Case 2 analysis discussed in Subsection 3C.4.2 at the RB exterior wall, SCCV top slab and RPV pedestal.

The comparisons show that the overall effect of soil separation on the structures is small with the largest increase being in the order of 10% in vertical forces in the below-grade segment of the RB exterior wall (Figure 3C-46).

3C.7.5 Concrete Fill Strength Effects on Responses

Figure 3C-51 through Figure 3C-55 compare stress responses obtained from the lean concrete sensitivity analysis discussed in Subsection 3C.4.3.6 and the BE Case 2 analysis discussed in Subsection 3C.4.2 at the RB exterior wall, SCCV top slab and RPV pedestal.

The comparisons show a 10% to 15% increase in the RB wall vertical and in-plane shear forces due to the reduced strength of the concrete backfill (Figure 3C-51 and Figure 3C-52) but no increase on stress responses for the SCCV top slab or RPV pedestal (Figure 3C-53 through Figure 3C-55).

3C.8 SSE Structural Stress Demands

To illustrate the seismic demands on the Seismic Category I integrated RB structures, critical seismic member force and moment demands are presented in this section for key structural members.

3C.8.1 Containment Stress Demands

Figure 3C-56 presents the seismic moment demands for the SCCV top slab, while Figure 3C-57 and Figure 3C-58 present the bounding seismic membrane forces and moment demands for the SCCV wall.

3C.8.2 Containment Internal Structures Stress Demands

Figure 3C-59 presents the seismic membrane forces for the RPV pedestal.

3C.8.3 Reactor Building Stress Demands

Figure 3C-60 and Figure 3C-61 present the bounding seismic membrane forces and moment demands for the RB exterior wall.

3C.9 Maximum SSE Displacements

Table 3C-6 presents the amplitudes of seismic displacements at key locations within the integrated RB structures. These displacements are computed relative to the free-field at the bottom of the RB foundation and are an envelope of results obtained from the six primary SSI analysis cases and six sensitivity cases described in Section 3C.4.

3C.10 References

3C-1 ASCE/SEI 43-19, “Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities,” American Society of Civil Engineers, 2019.

3C-2 ASCE/SEI 4-16, “Seismic Design of Safety-Related Nuclear Structures,” American Society of Civil Engineers, 2017.

3C-3 NEDO-33914-A, “BWRX-300 Advanced Civil Construction and Design Approach,” GE-Hitachi Nuclear Energy Americas, LLC, Revision 2, June 2022.

Table 3C-1 Key Nodal Locations

Node Number	Location Description	Elevation [m (ft)]
54526	Center of RB mat foundation	-35.22 (-115.6)
57403	Edge of RB mat foundation	-35.22 (-115.6)
97673	RPV support at top of pedestal	-8.95 (-29.4)
88881	RB exterior wall at El. -15m (-49 ft) floor	-14.805 (-48.6)
122772	RB exterior wall at grade	0.00 (0.00)
141410	RB exterior wall at operating floor	14.238 (46.7)
146384	Edge of RB roof	29.115 (95.5)
150753	Center of RB roof	32.258 (105.8)

Table 3C-2 Key Structural Members

Structural Member	Description	Critical Stress Components
Mat Foundation Slab	Circular slab	Out-of-plane moment
Steel-Plate Composite Containment Vessel (SCCV) Top slab	Circular slab	Out-of-plane moment
SCCV wall	Cylindrical wall	Membrane forces, in-plane shear, out-of-plane moment
RPV pedestal	Cylindrical wall	Membrane forces, in-plane shear
RB exterior wall	Cylindrical wall	Membrane forces, in-plane shear, out-of-plane moment
Wing walls	Rectangular walls	Membrane forces, in-plane shear

Table 3C-3 Seismic Analysis Cases

Analysis Case	Soil Profile	Category
BE Case 1	BE Case 1	Primary
LB Case 1	LB Case 1	Primary
UB Case 1	UB Case 1	Primary
BE Case 2	BE Case 2	Primary
LB Case 2	LB Case 2	Primary
UB Case 2	UB Case 2	Primary
Separation	BE Case 2	Sensitivity
No-Friction	BE Case 2	Sensitivity
Excavation Support	BE Case 2	Sensitivity
Lean Concrete	BE Case 2	Sensitivity
Dry	BE Case 2-Dry	Sensitivity
Cracked	BE Case 2	Sensitivity

Table 3C-4 Concrete Fill Dynamic Properties

Case	Shear-wave Velocity [ft/s]	Compression-wave Velocity [ft/s]	Damping Ratio [%]
BE	6673	10580	1.0
LB	6135	9731	1.8
UB	7106	11270	0.5

Table 3C-5 Dominant Frequencies for Soil-Structure Interaction System Response

Soil Case	V_s at FIRS [ft/sec]		Cut-off Frequency [Hz]
	Case 1	Case 2	
LB Cases 1 and 2	6,122	6,194	35.16
BE Cases 1 and 2	7,549	7,621	51.27
UB Cases 1 and 2	9,311	9,380	70.19
BE Case 2-Dry	N/A	7,621	51.27

Table 3C-6 Enveloped Seismic Relative Displacements at Key Locations

Node Number	Location Description	Seismic Relative Displacement (in)		
		X	Y	Z
54526	Center of RB mat foundation	0.01	0.01	0.01
57403	Edge of RB mat foundation	0.01	0.01	0.01
97673	RPV support at top of pedestal	0.22	0.23	0.04
88881	RB exterior wall at El. -15m (-49ft) floor	0.19	0.17	0.13
122772	RB exterior wall at grade	0.20	0.27	0.22
141410	RB exterior wall at operating floor	0.46	0.55	0.27
146384	Edge of RB roof	0.72	0.86	0.28
150753	Center of RB roof	0.76	0.95	0.55

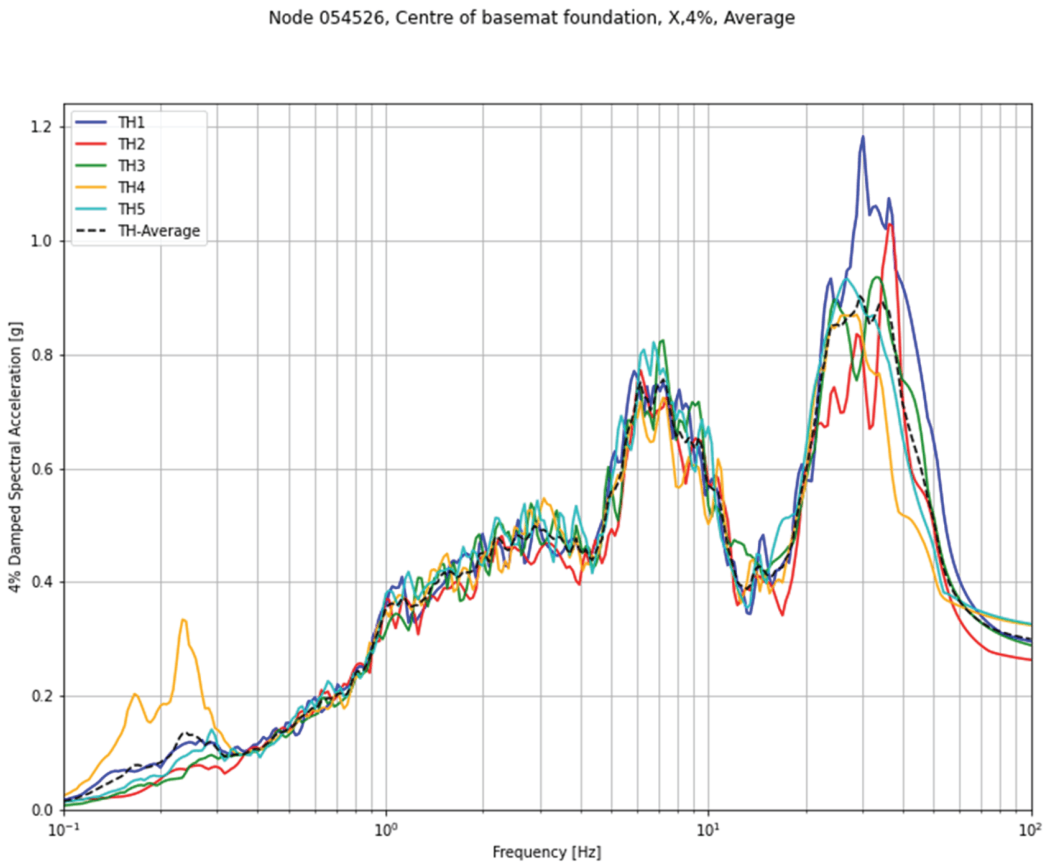


Figure 3C-1 Averaging of In-Structure Response Spectra from Five Sets of Acceleration Time History Results

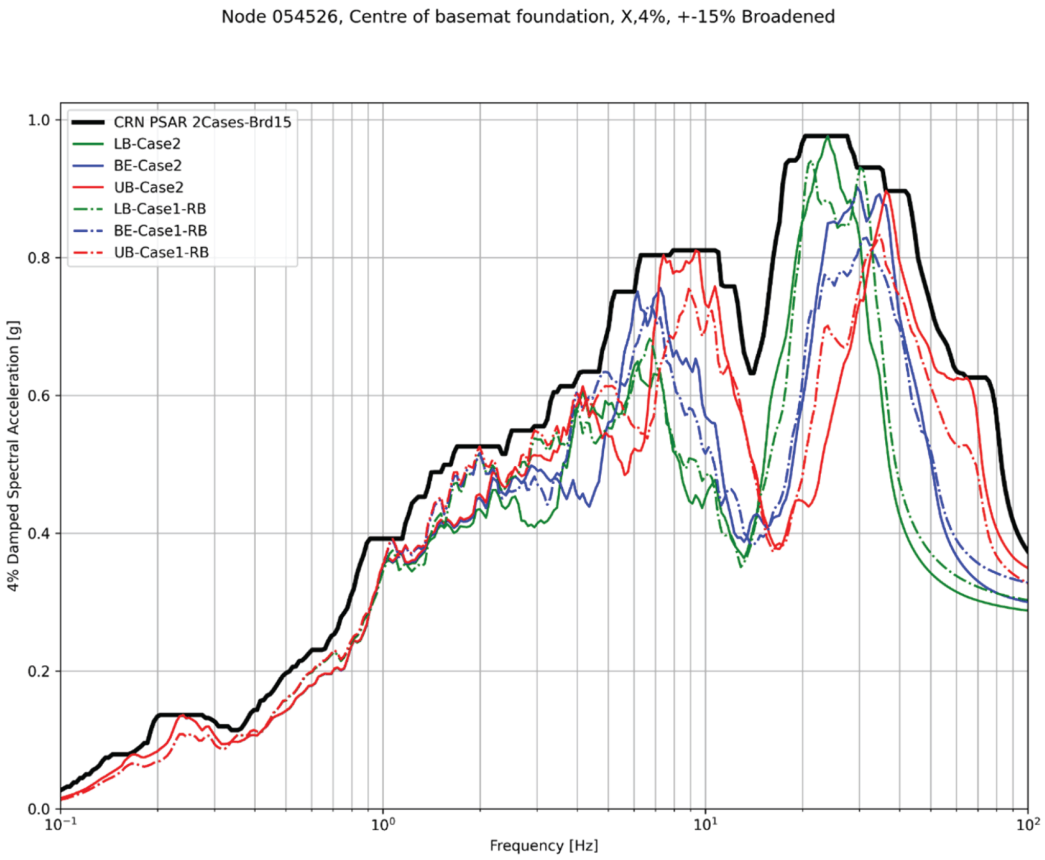


Figure 3C-2 Enveloping and Broadening of In-Structure Response Spectra

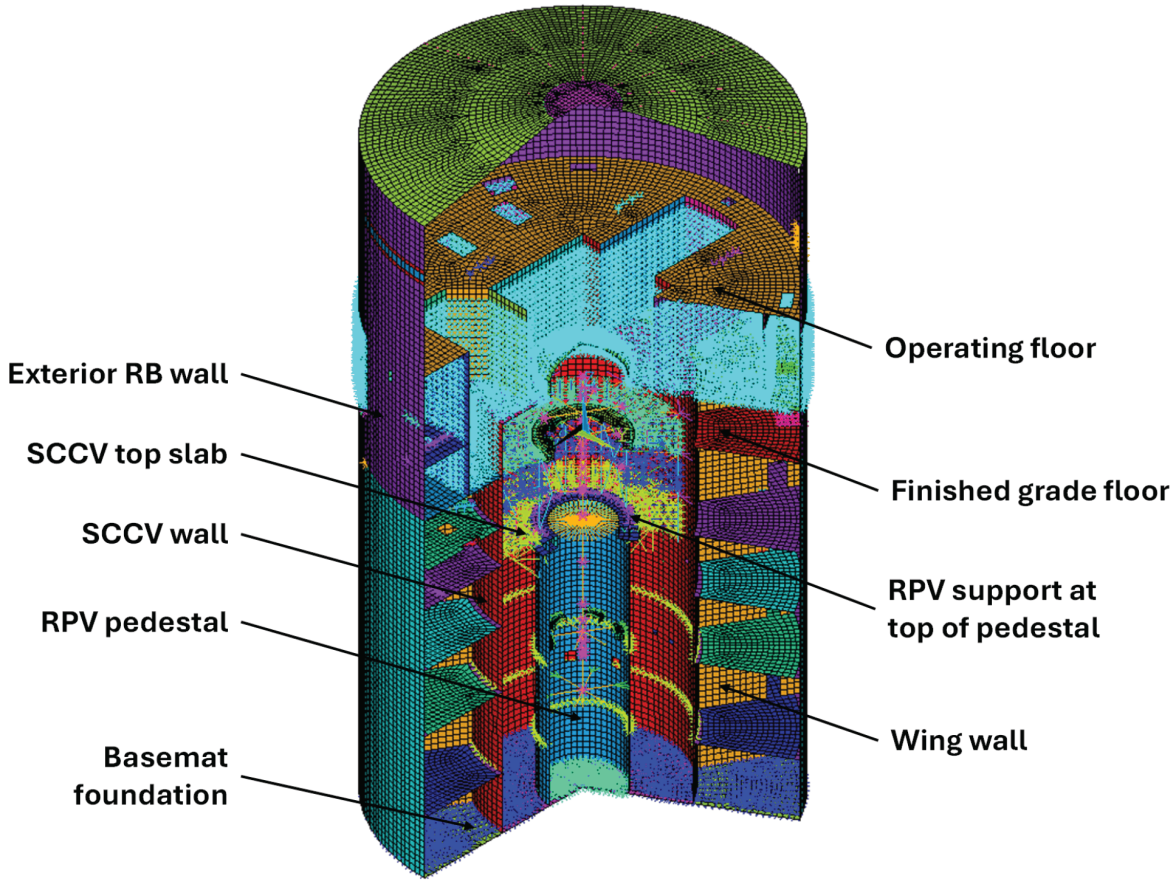


Figure 3C-3 Reactor Building Finite Element Model

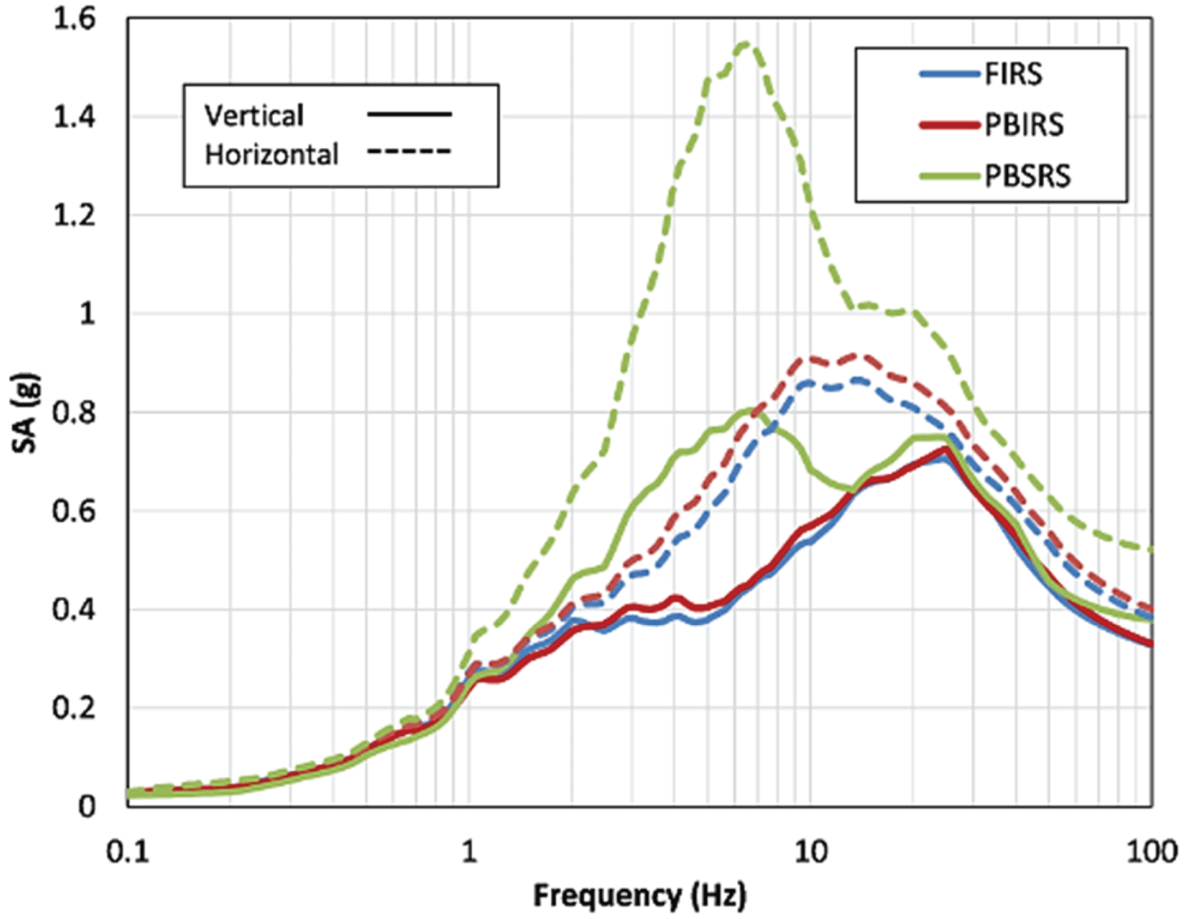


Figure 3C-4 Case 2 Horizontal and Vertical Input Response Spectra at Bottom of Reactor Building Foundation, at Top of Un-Weathered Rock and at Plant Finished Grade Elevation

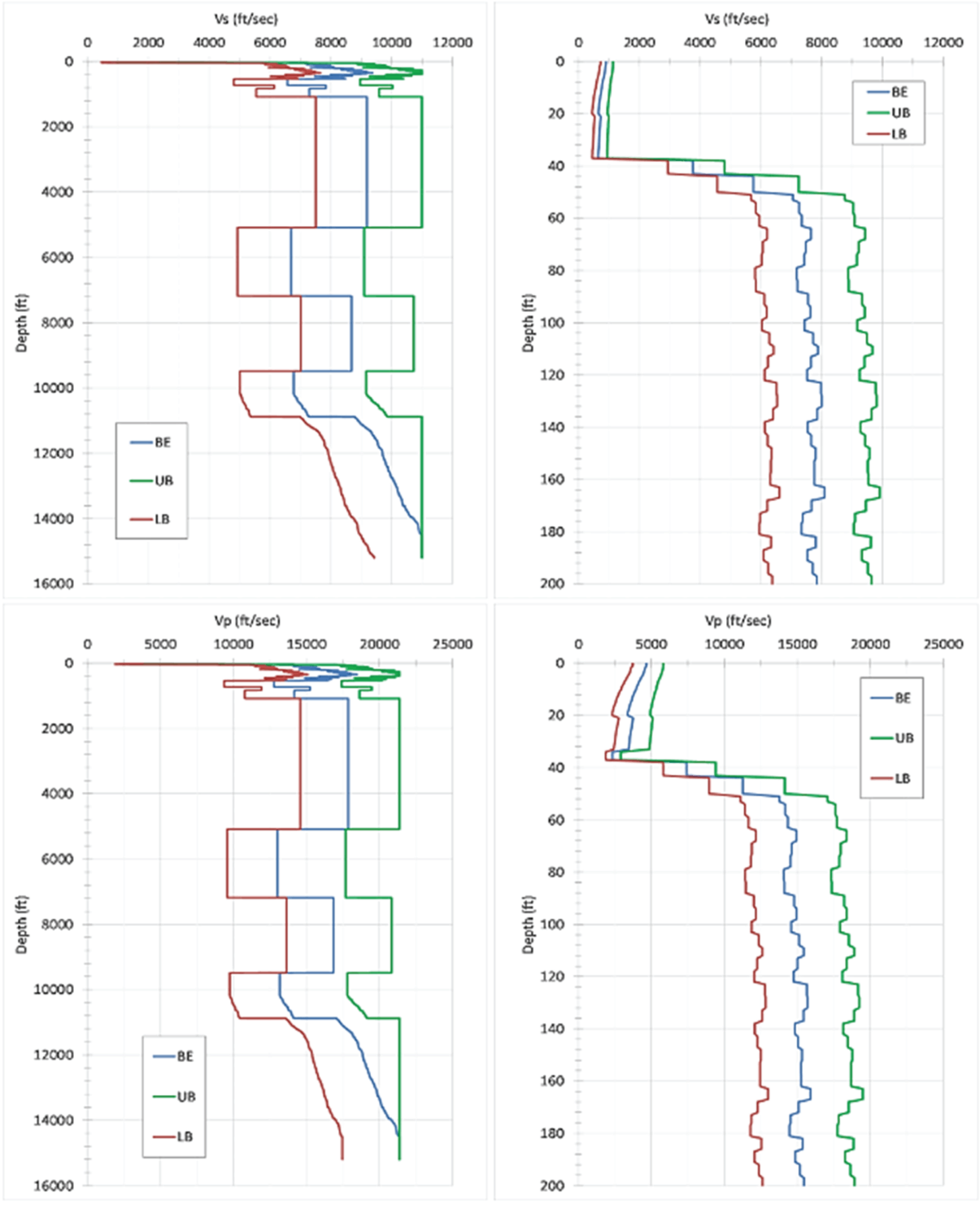


Figure 3C-5 Case 2 Hazard Consistent Strain-Compatible Shear Wave Velocity (V_s) and Compression (P)-Wave Velocity (V_p) Profiles

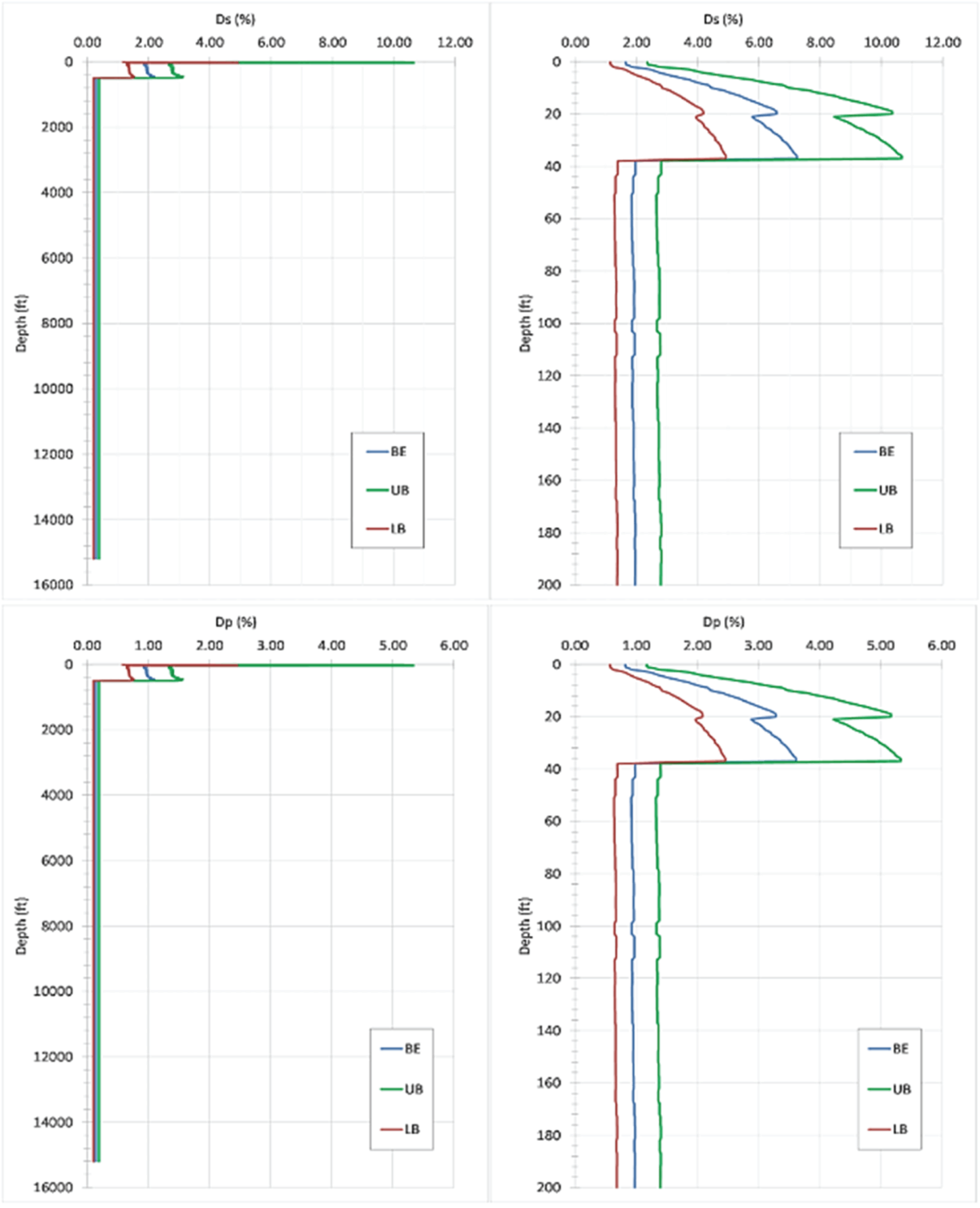


Figure 3C-6 Case 2 Hazard Consistent Strain-Compatible Damping Ratios

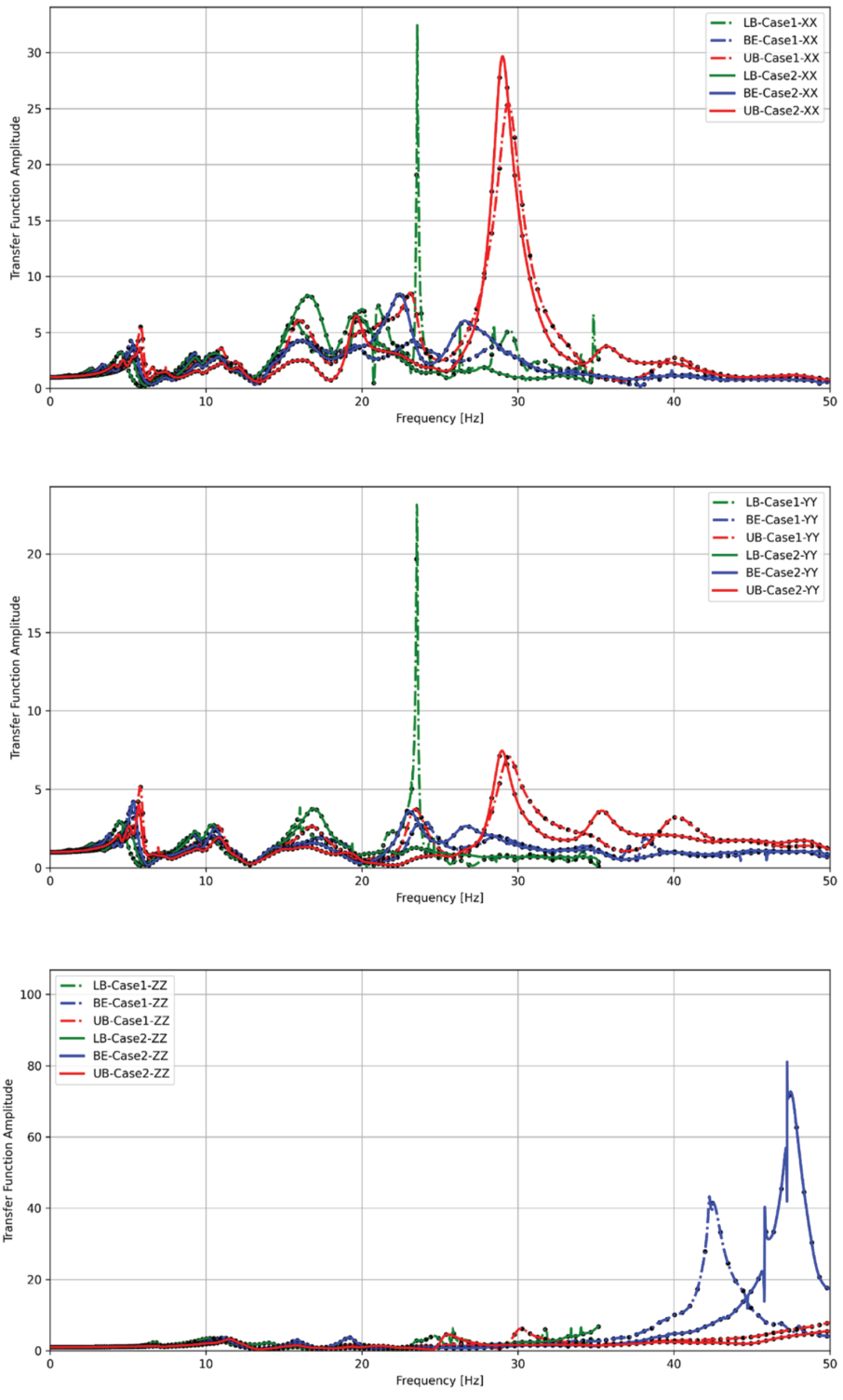


Figure 3C-7 Effects of Subgrade Variation on the Acceleration Time Histories - Reactor Building Exterior Wall at Grade

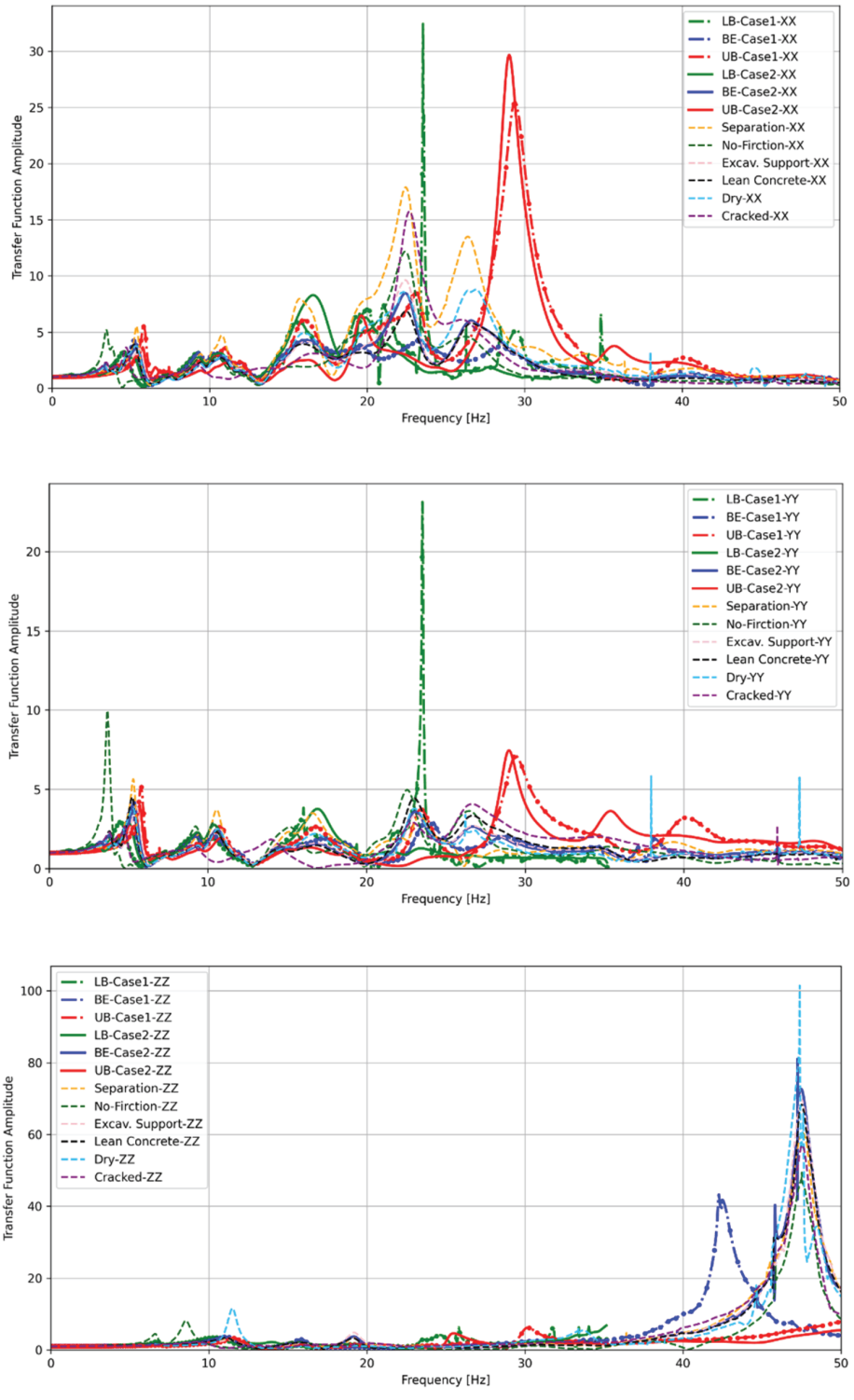


Figure 3C-8 Comparison of Acceleration Transfer Function for Primary and Sensitivity Cases - Reactor Building Exterior Wall at Grade

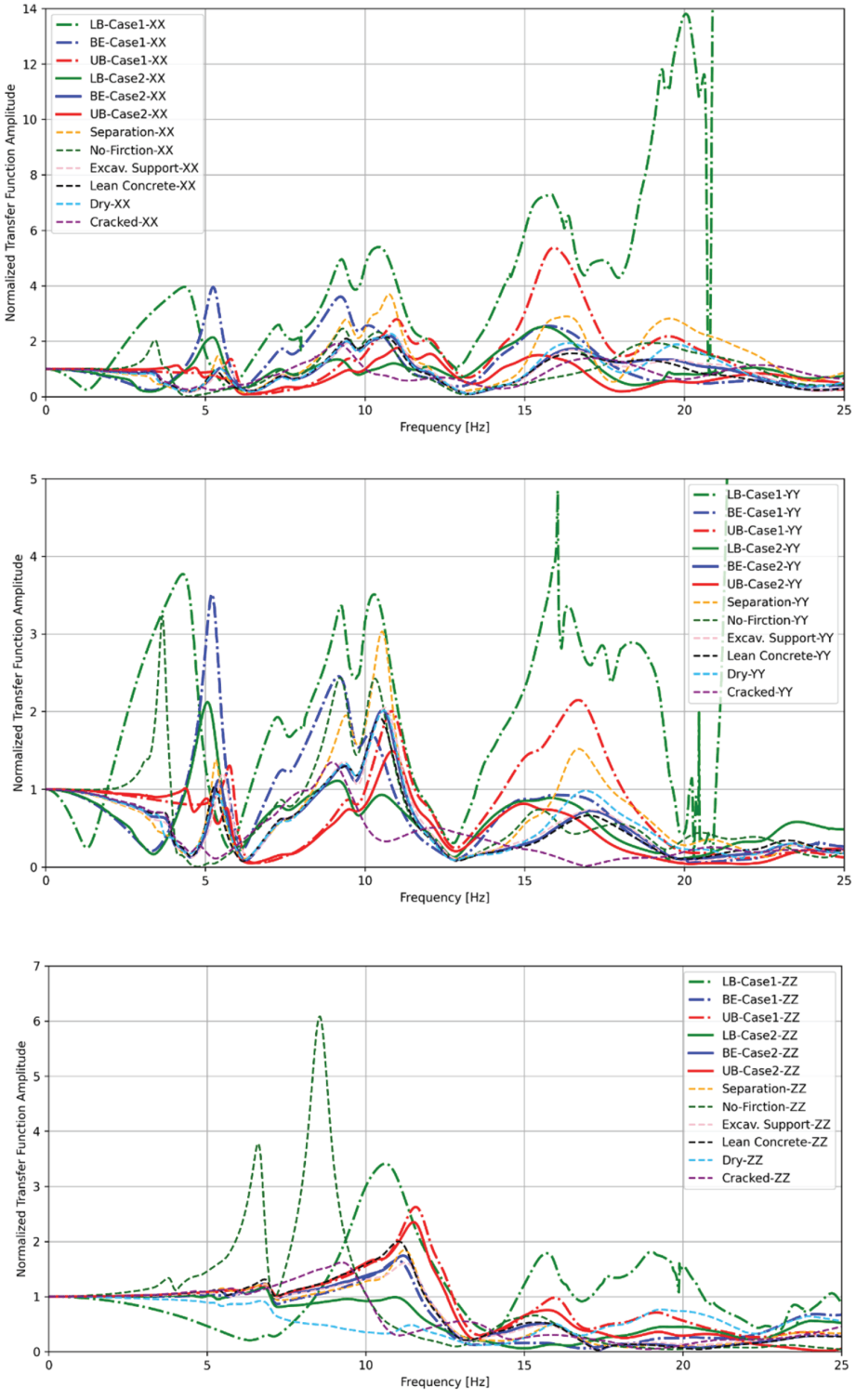


Figure 3C-9 Effects of Subgrade Variation on Normalized Acceleration Transfer Function - Reactor Building Exterior Wall at Grade

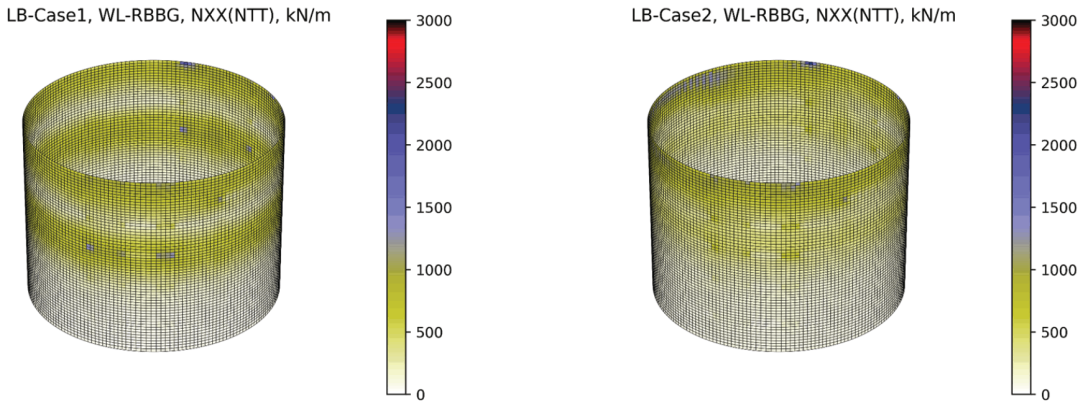


Figure 3C-10 Membrane Tangential (Hoop) Forces - Exterior Wall Below Grade - Lower Bound Cases 1 and 2

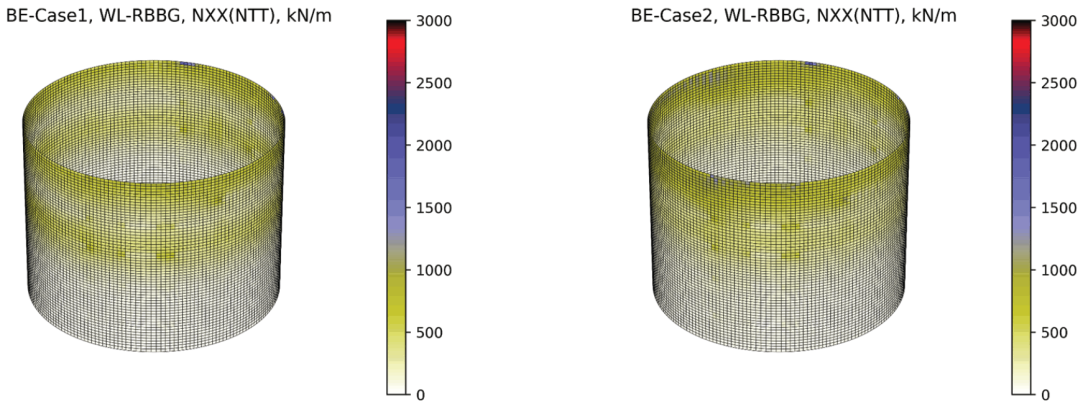


Figure 3C-11 Membrane Tangential (Hoop) Forces - Exterior Wall Below Grade - Best Estimate Cases 1 and 2

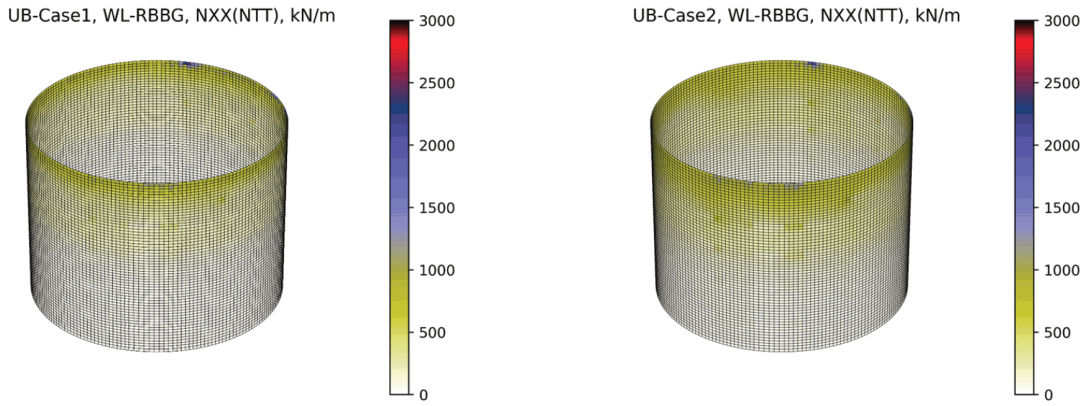


Figure 3C-12 Membrane Tangential (Hoop) Forces - Exterior Wall Below Grade - Upper Bound Case 1 and 2

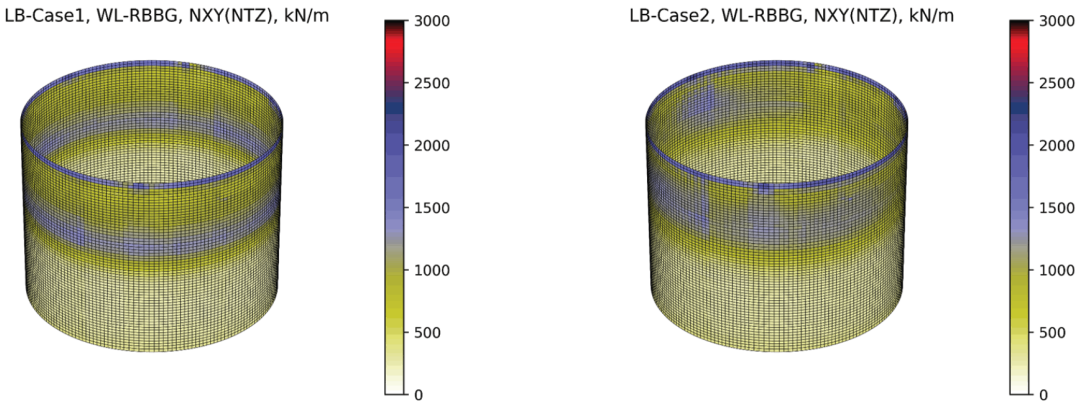


Figure 3C-13 In-plane Shear Forces - Exterior Wall Below Grade - Lower Bound Cases 1 and 2

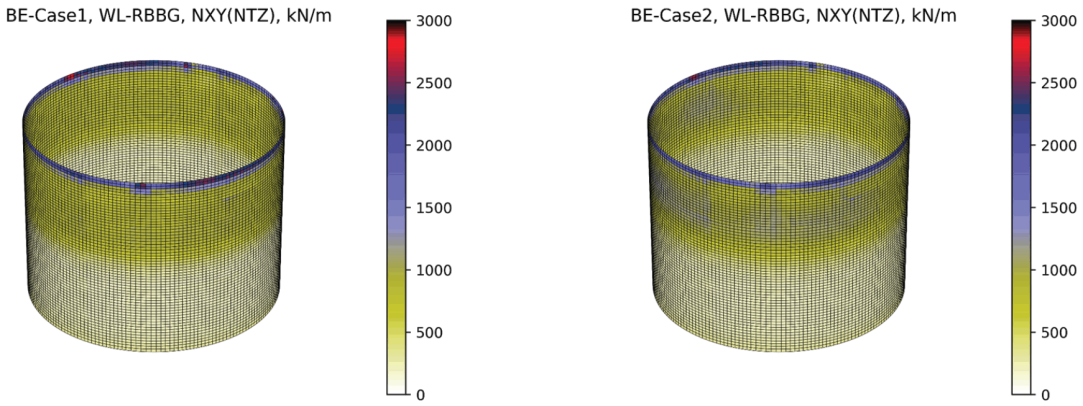


Figure 3C-14 In-plane Shear Forces - Exterior Wall Below Grade - Best Estimate Cases 1 and 2

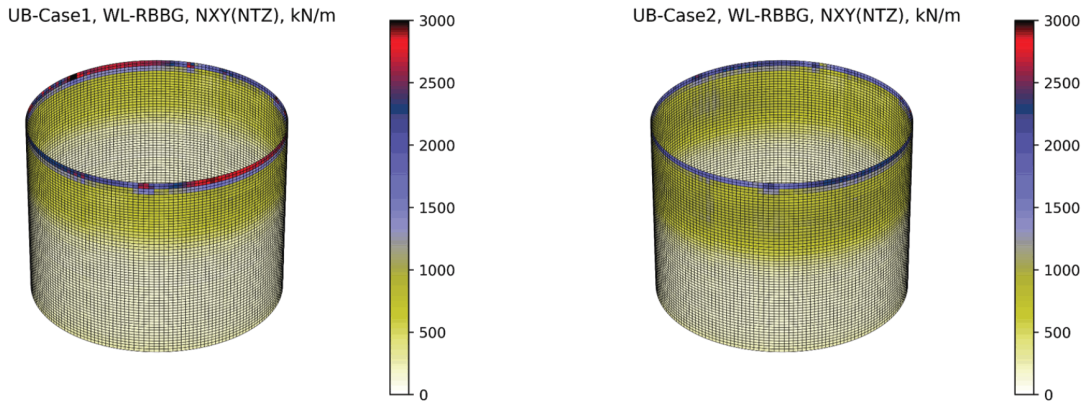


Figure 3C-15 In-plane Shear Forces - Exterior Wall Below Grade - Upper Bound Case 1 and 2

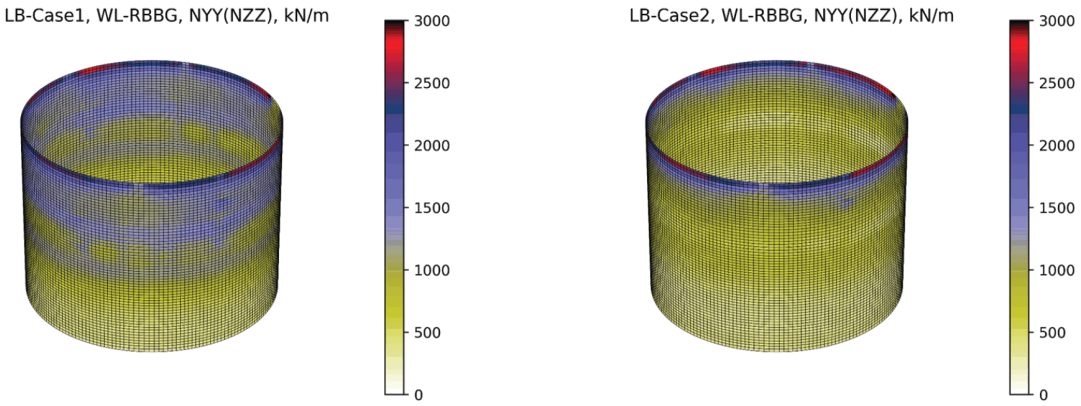


Figure 3C-16 Membrane Vertical Forces - Exterior Wall Below Grade - Lower Bound Cases 1 and 2

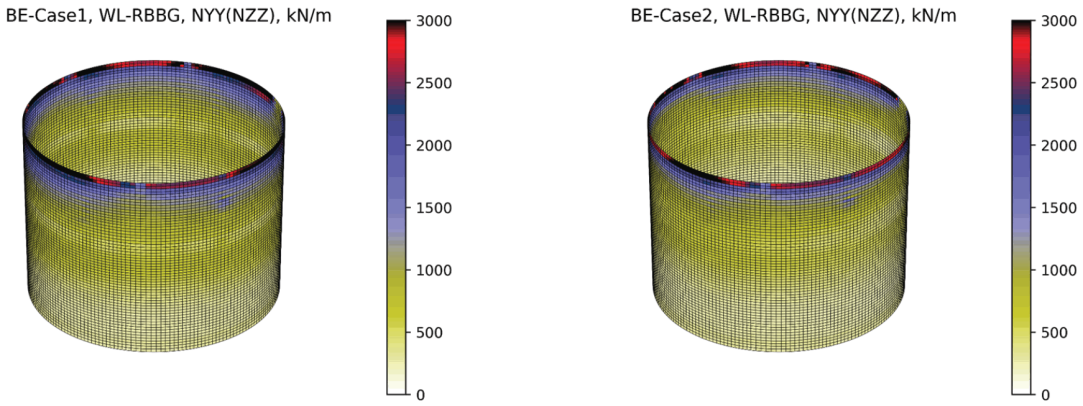


Figure 3C-17 Membrane Vertical Forces - Exterior Wall Below Grade - Best Estimate Cases 1 and 2

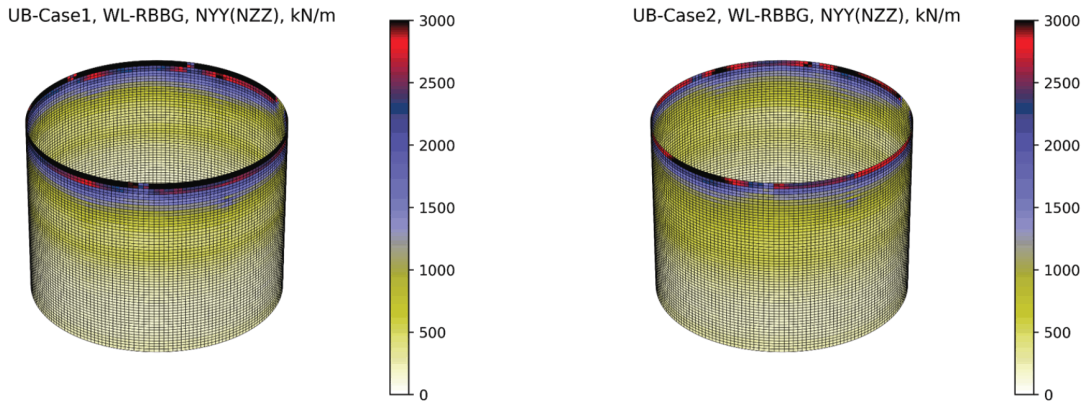


Figure 3C-18 Membrane Vertical Forces - Exterior Wall Below Grade - Upper Bound Cases 1 and 2

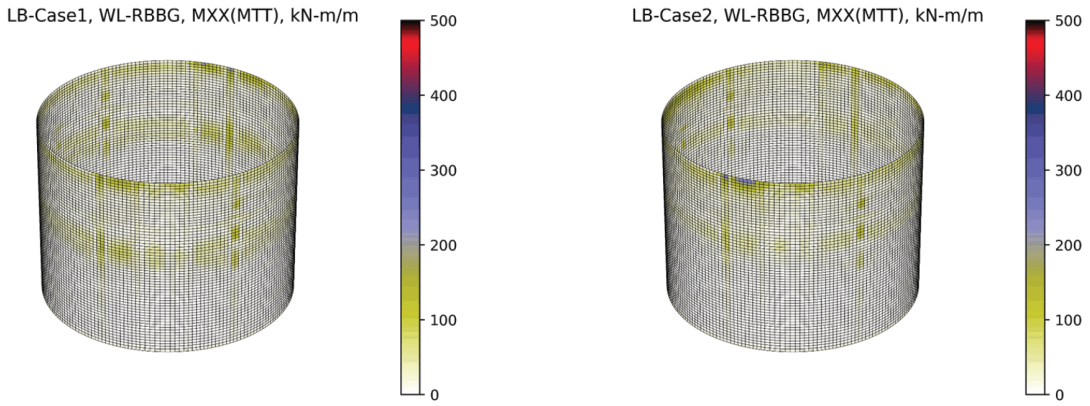


Figure 3C-19 Out-of-Plane Horizontal Moments - Exterior Wall Below Grade - Lower Bound Cases 1 and 2

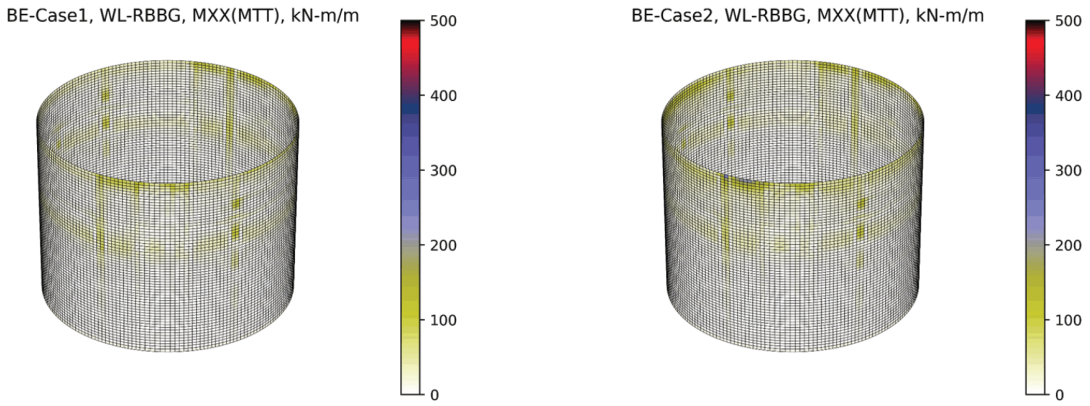


Figure 3C-20 Out-of-Plane Horizontal Moments - Exterior Wall Below Grade - Best Estimate Cases 1 and 2

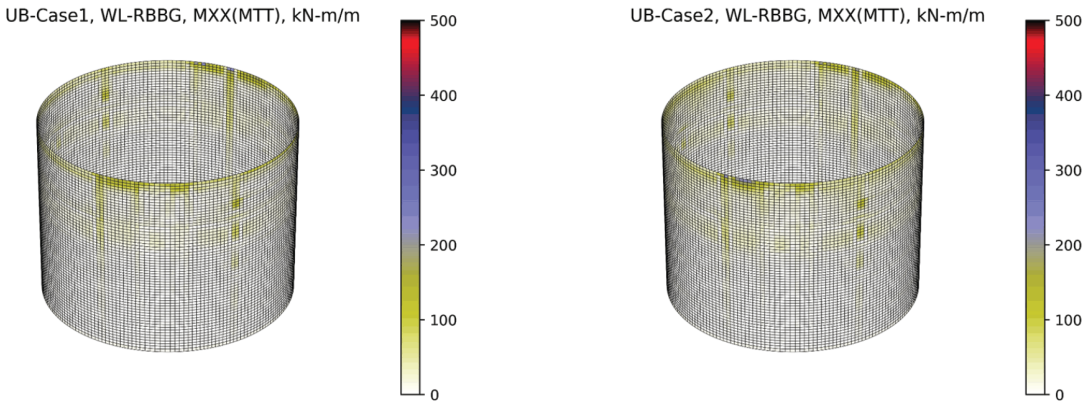


Figure 3C-21 Out-of-Plane Horizontal Moments - Exterior Wall Below Grade - Upper Bound Cases 1 and 2

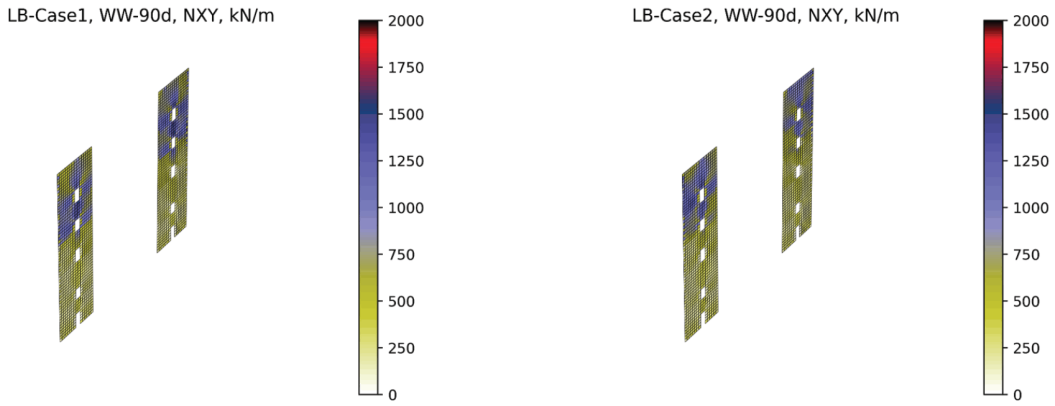


Figure 3C-22 In-Plane Shear Forces - Wing Walls - Lower Bound Cases 1 and 2

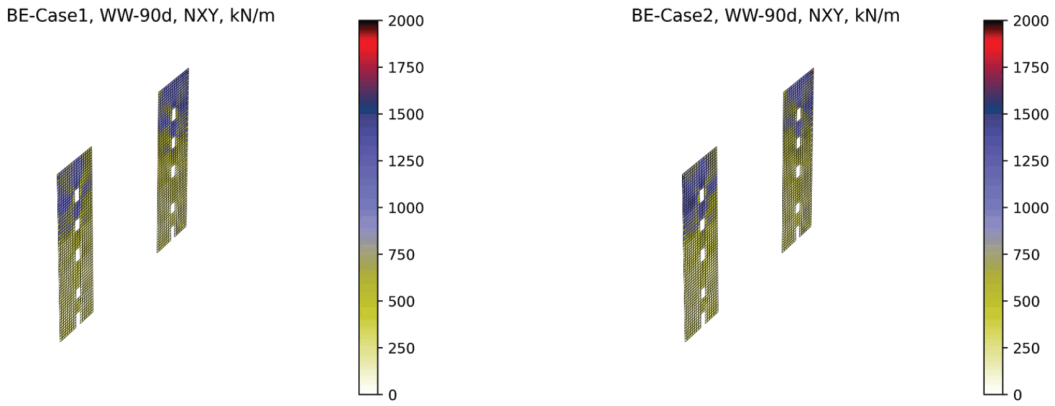


Figure 3C-23 In-Plane Shear Forces - Wing Walls - Best Estimates Cases 1 and 2

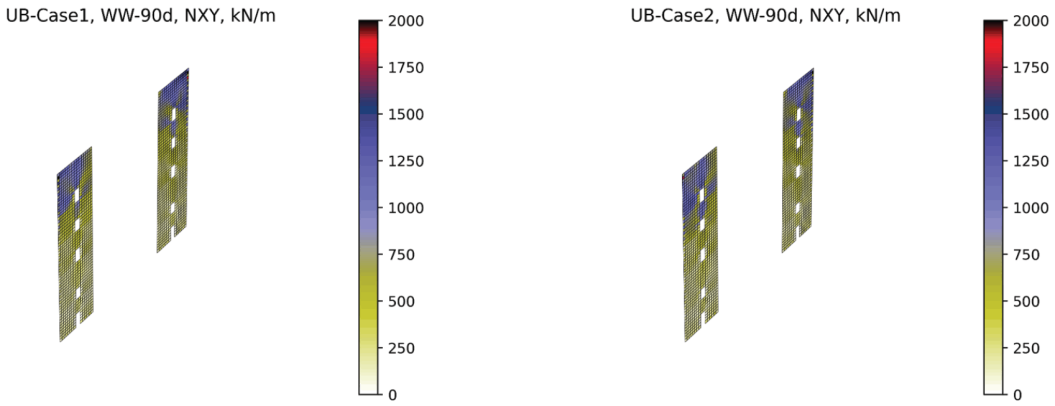


Figure 3C-24 In-Plane Shear Forces - Wing Walls - Upper Bound Cases 1 and 2

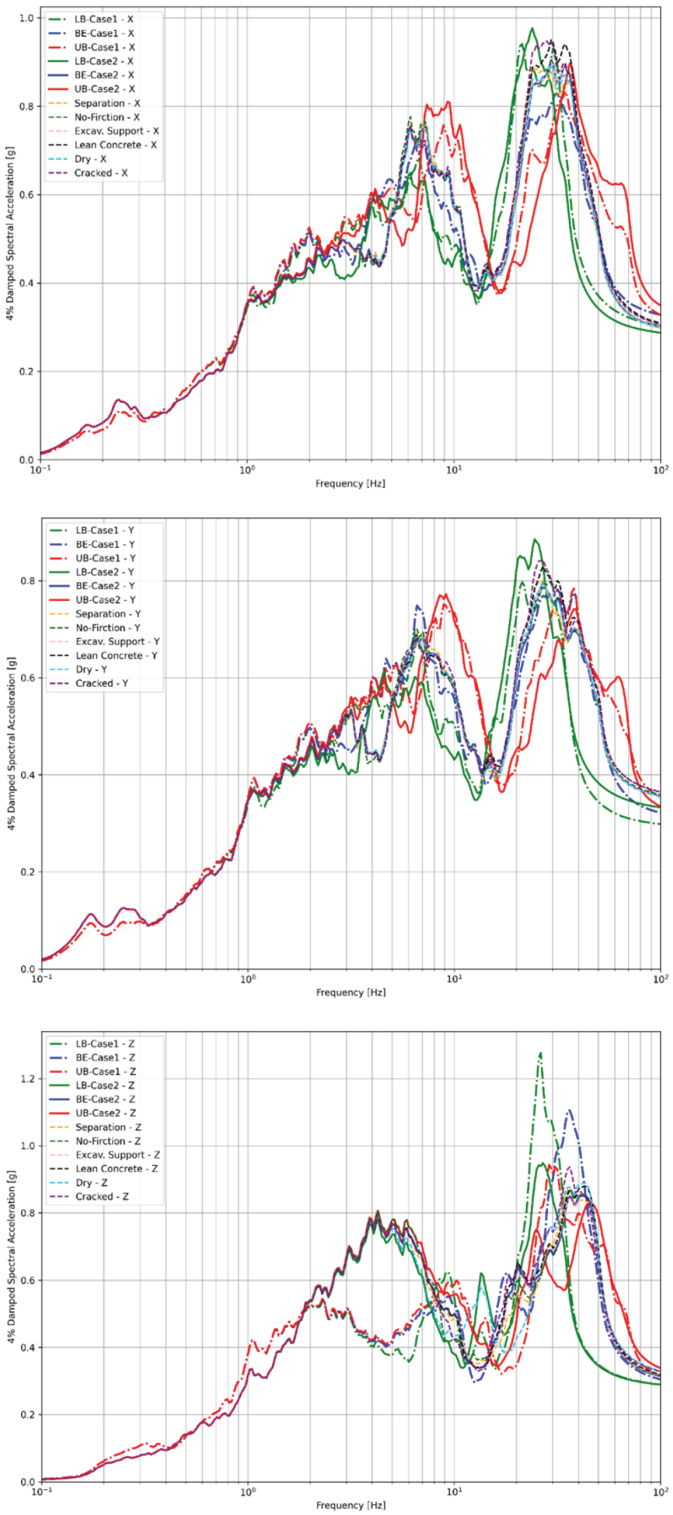


Figure 3C-25 In-Structure Response at Center of Common Mat Foundation

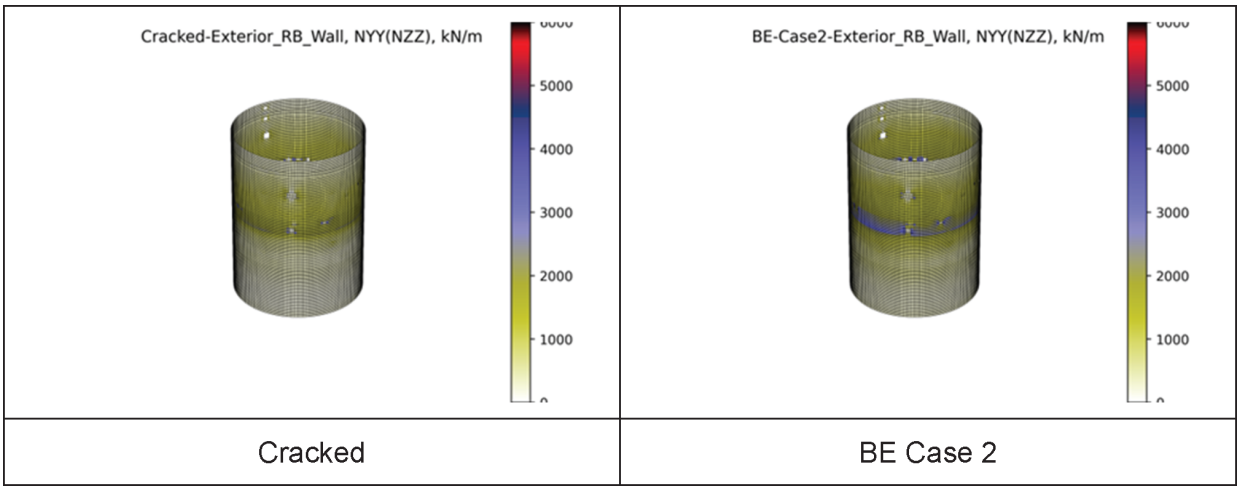


Figure 3C-26 Structural Stiffness Variation Effects on Vertical Forces in Reactor Building Exterior Wall

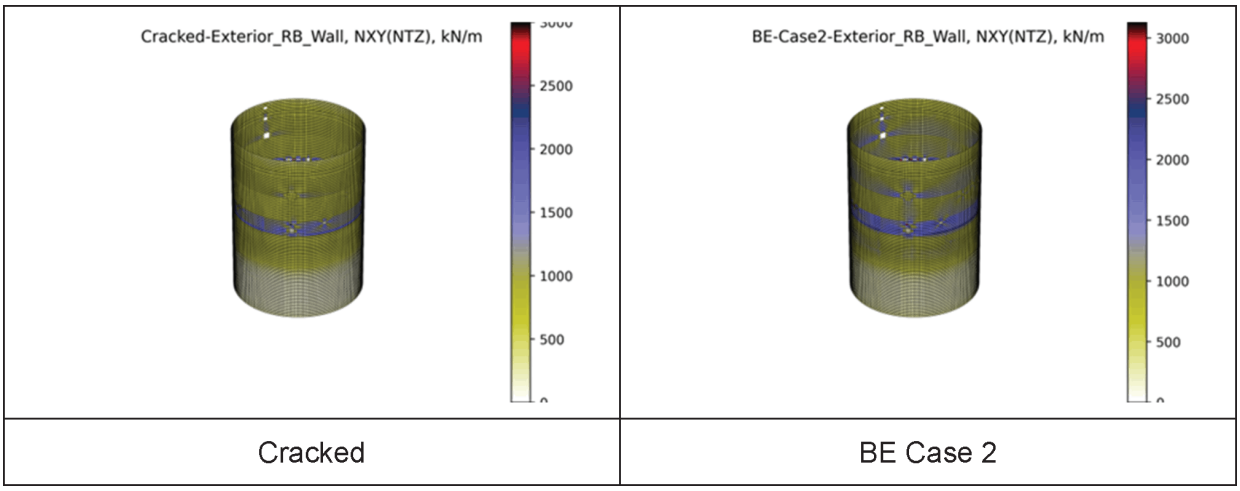


Figure 3C-27 Structural Stiffness Variation Effects on In-Plane Shear Forces in the Reactor Building Exterior Wall

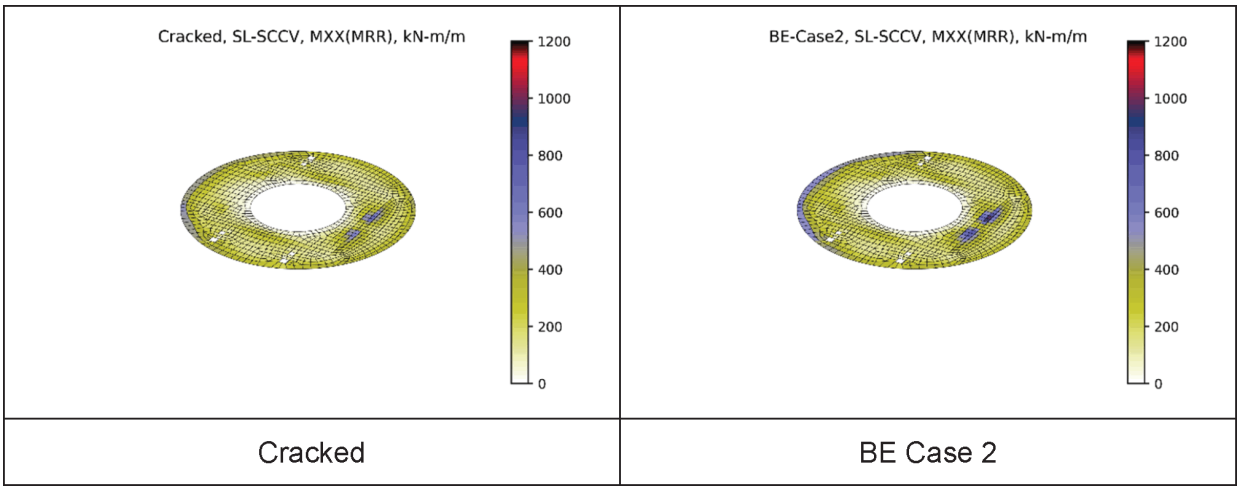


Figure 3C-28 Structural Stiffness Variation Effects on Containment Top Slab Out-of-Plane Moment

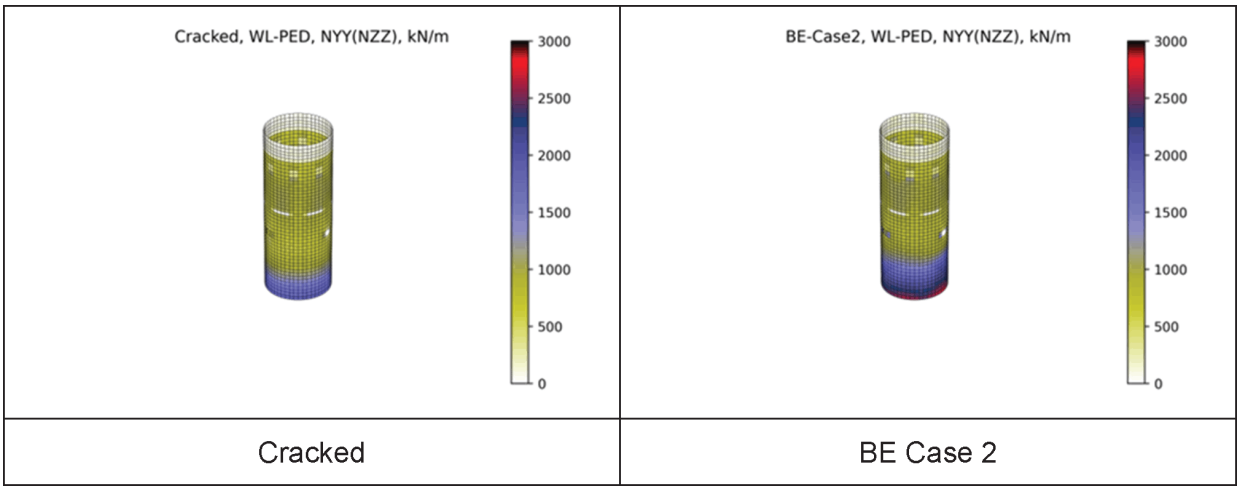


Figure 3C-29 Structural Stiffness Variation Effects on Vertical Forces in the Reactor Pressure Vessel Pedestal

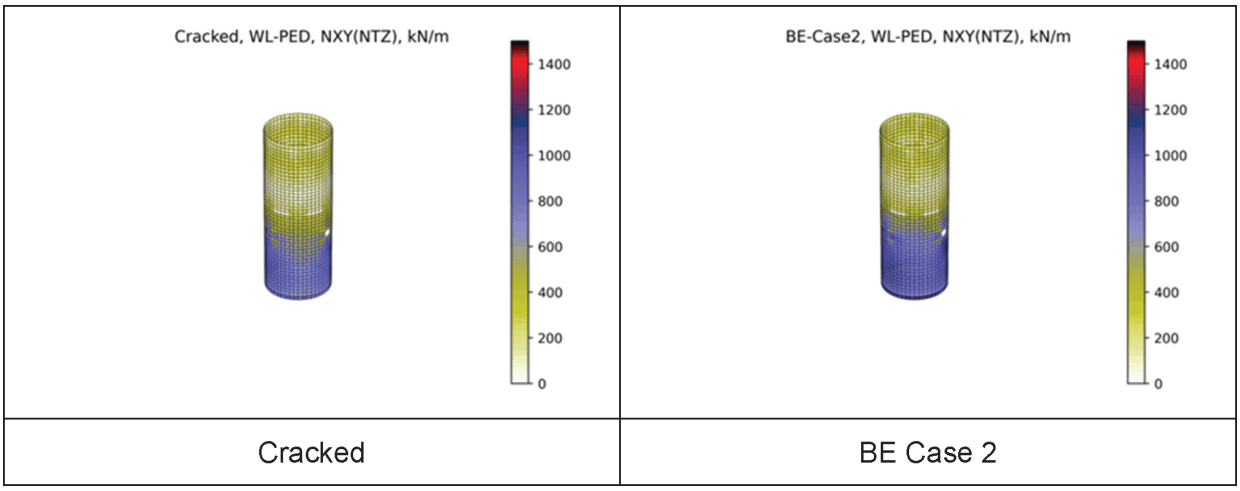


Figure 3C-30 Structural Stiffness Variation Effects on In-Plane Shear Forces in the Reactor Pressure Vessel Pedestal

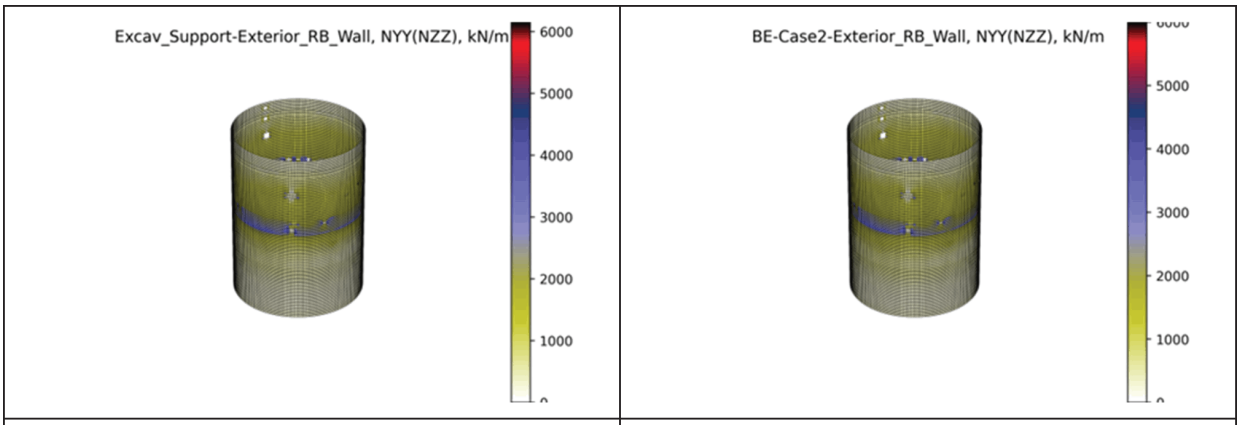


Figure 3C-31 Excavation Support Effects on Vertical Forces in the Reactor Building Exterior Wall

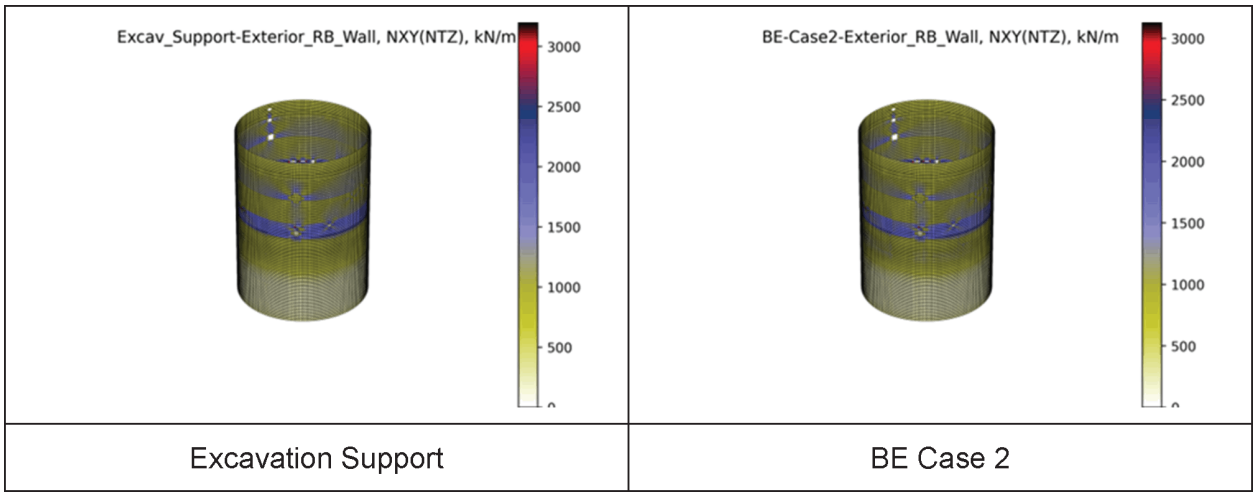
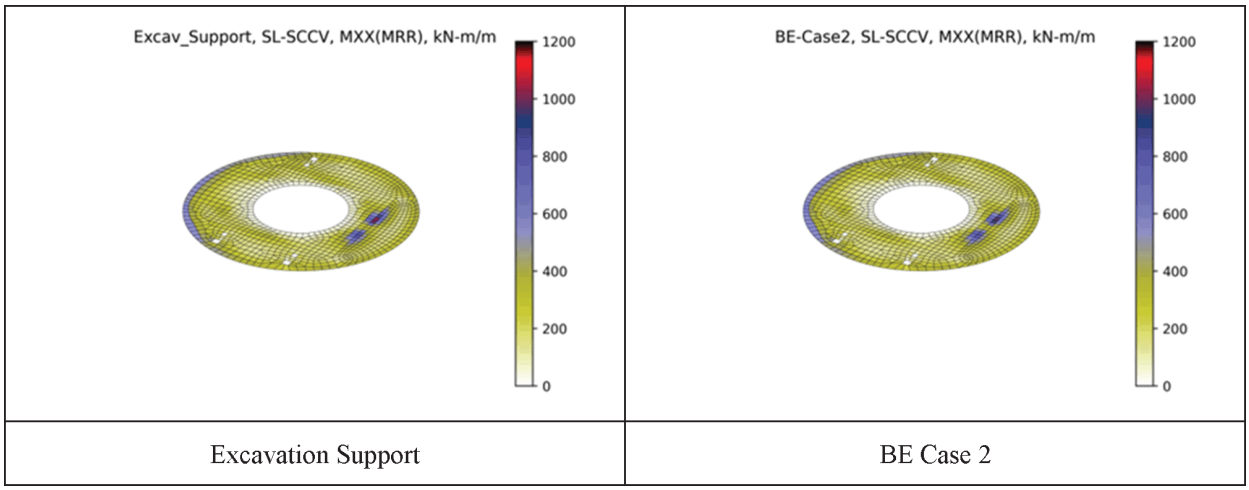
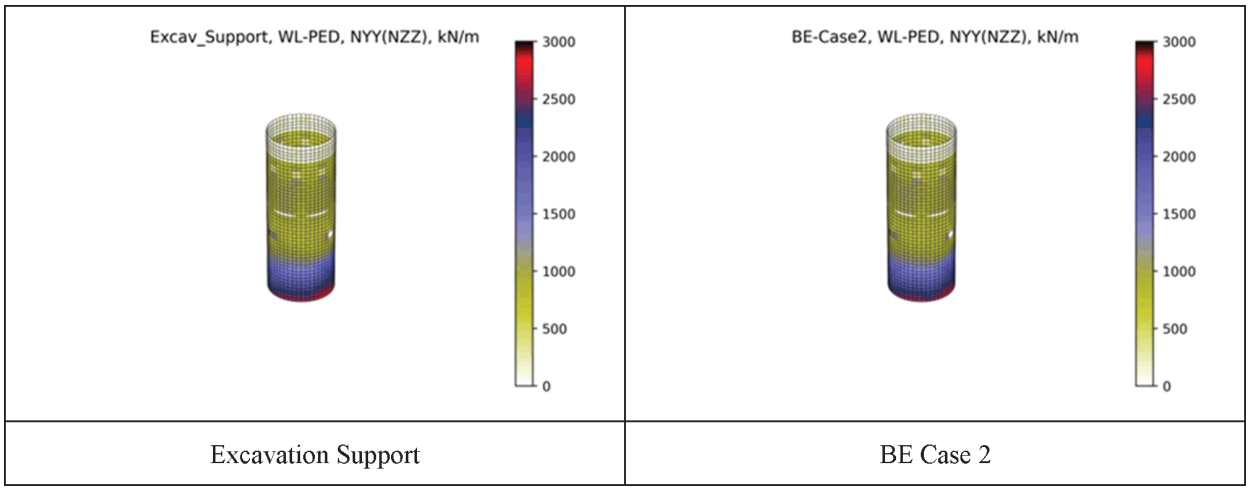


Figure 3C-32 Excavation Support Effects on In-Plane Shear Forces in the Reactor Building Exterior Wall



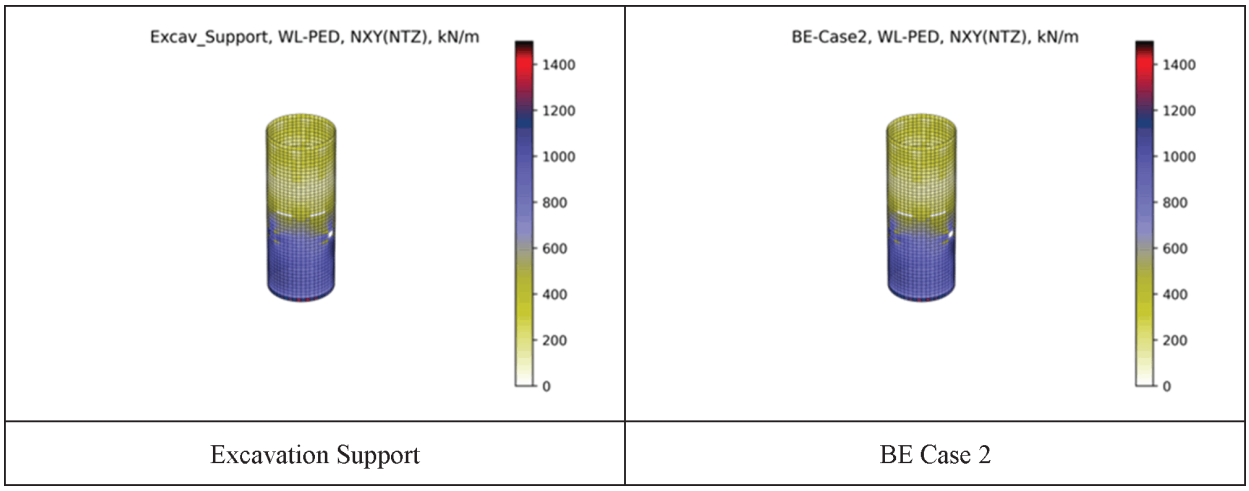
Excavation Support	BE Case 2

Figure 3C-33 Excavation Support Effects on Containment Top Slab Out-of-Plane Moment



Excavation Support	BE Case 2

Figure 3C-34 Excavation Support Effects on Vertical Forces in the Reactor Pressure Vessel Pedestal



Excavation Support	BE Case 2

Figure 3C-35 Excavation Support Effects on In-Plane Shear Forces in the Reactor Pressure Vessel Pedestal

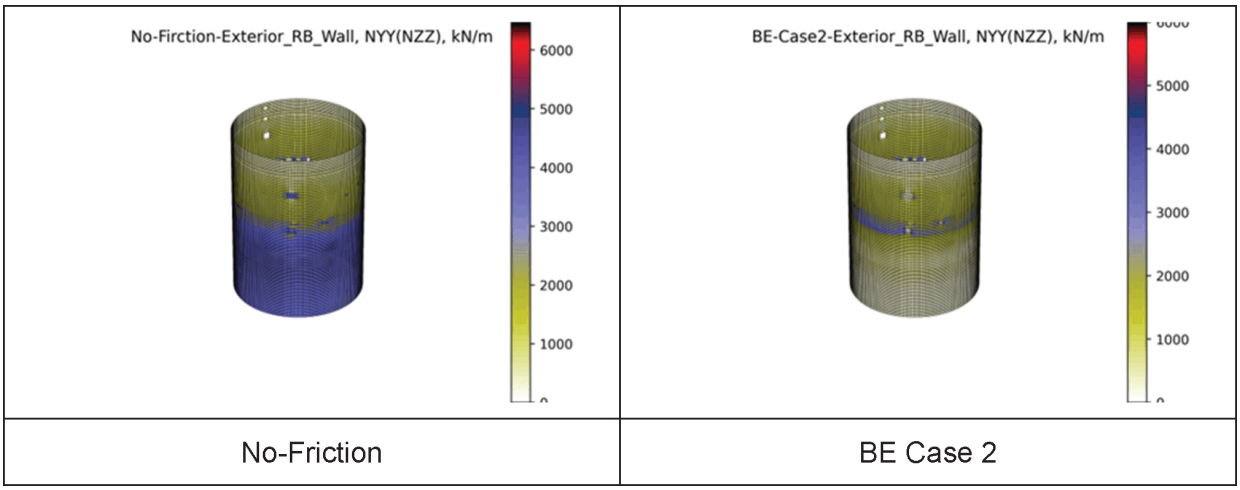
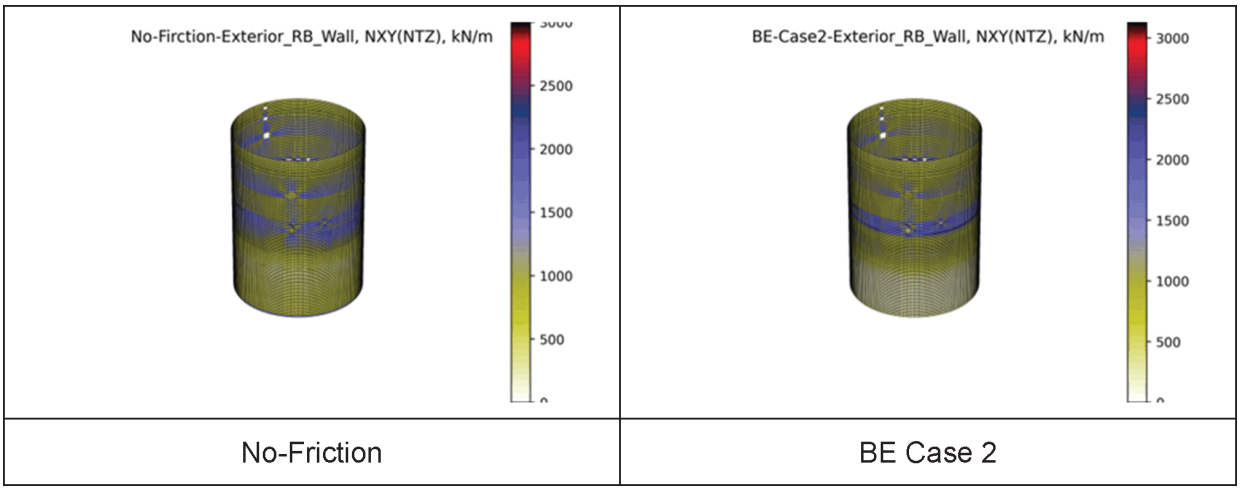


Figure 3C-36 Friction Effects on Vertical Forces in the Reactor Building Exterior Wall



No-Friction	BE Case 2

Figure 3C-37 Friction Effects on In-Plane Shear Forces in the Reactor Building Exterior Wall

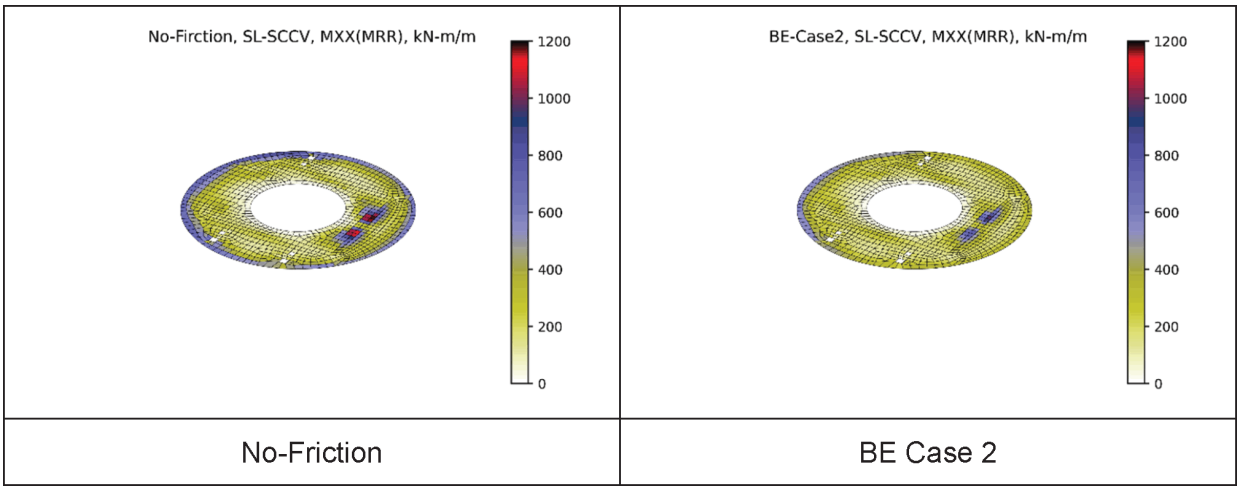


Figure 3C-38 Friction Effects on the Containment Top Slab Out-of-Plane Moment

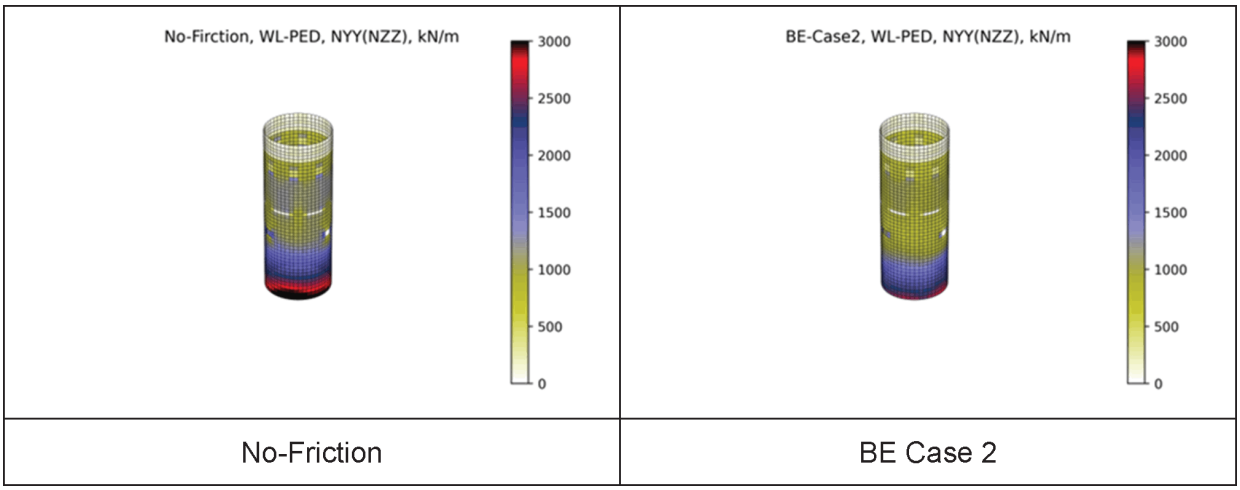
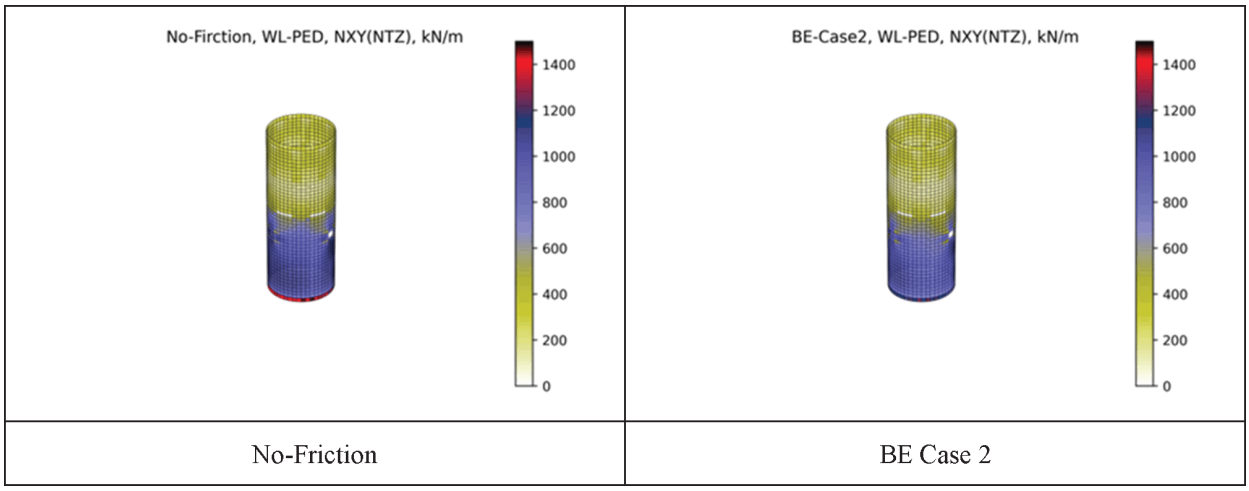


Figure 3C-39 Friction Effects on Vertical Forces in the Reactor Pressure Vessel Pedestal



No-Friction	BE Case 2

Figure 3C-40 Friction Effects on In-Plane Shear Forces in the Reactor Pressure Vessel Pedestal

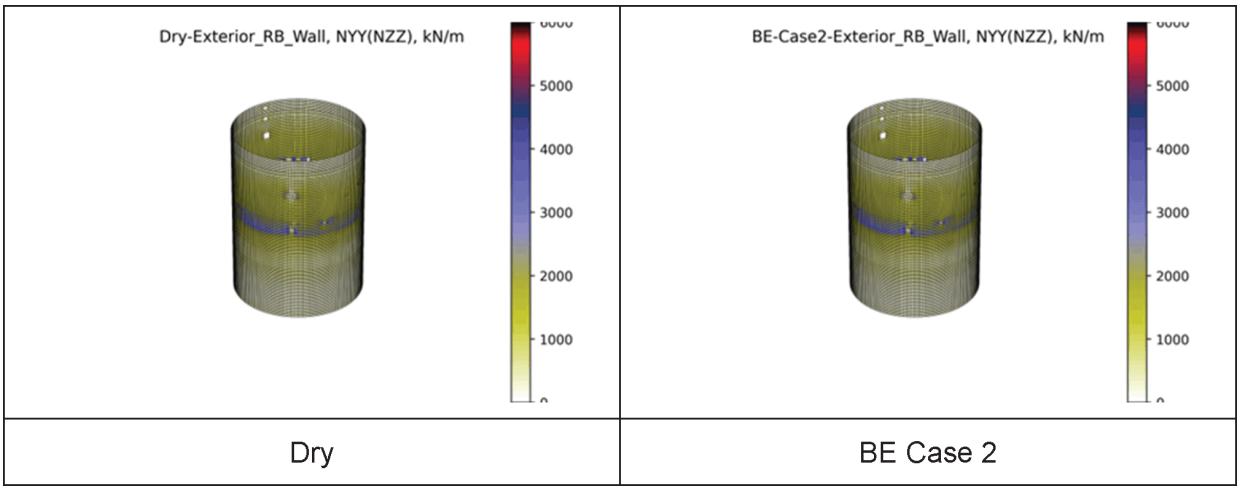


Figure 3C-41 Groundwater Variation Effects on Vertical Forces in the Reactor Building Exterior Wall

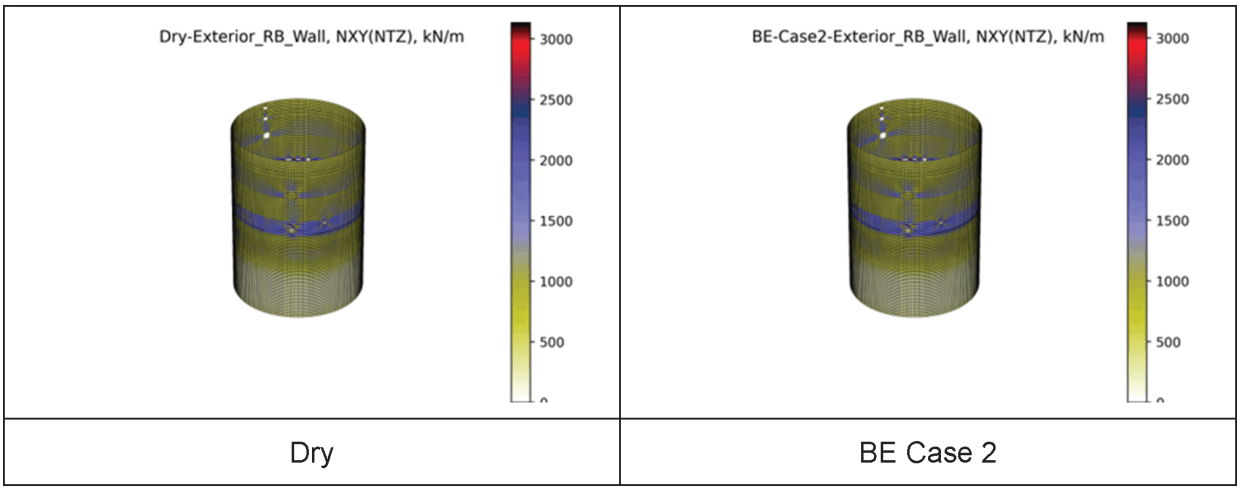


Figure 3C-42 Groundwater Variation Effects on In-Plane Shear Forces in the Reactor Building Exterior Wall

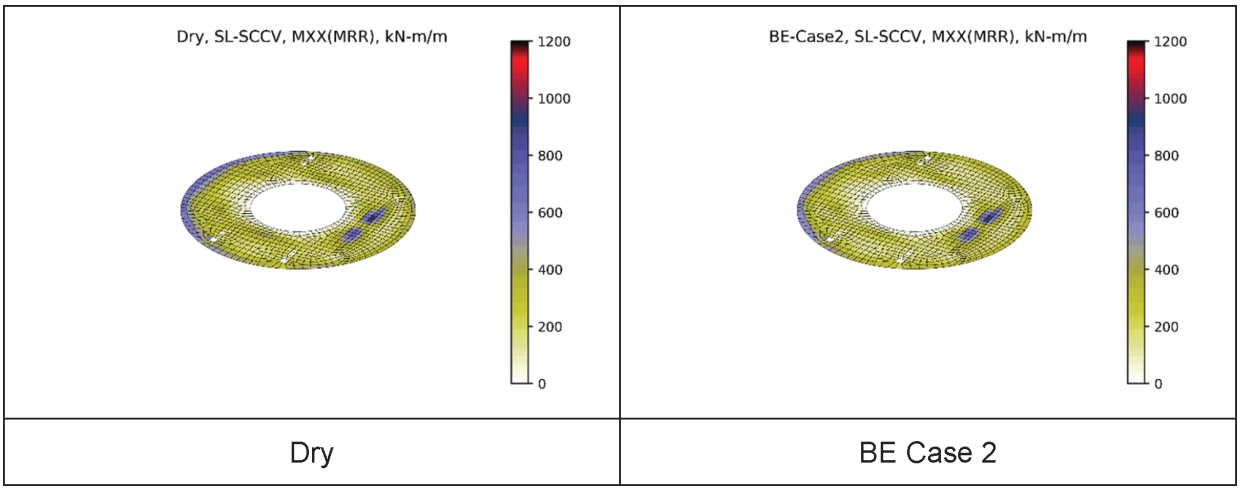


Figure 3C-43 Groundwater Variation Effects on Containment Top Slab Out-of-Plane Moment

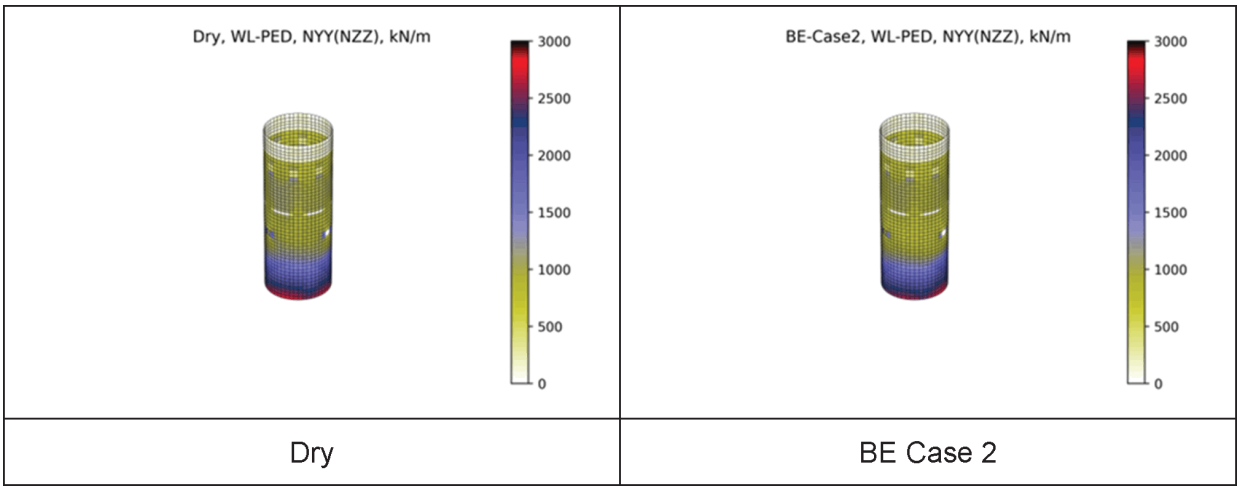


Figure 3C-44 Groundwater Variation Effects on Vertical Forces in the Reactor Pressure Vessel Pedestal

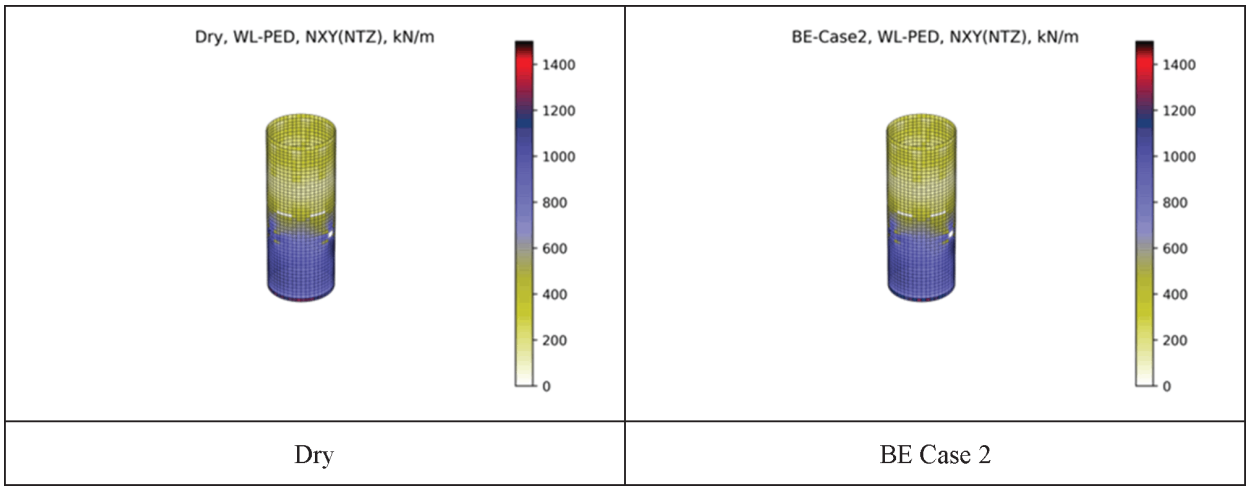


Figure 3C-45 Groundwater Variation Effects on In-Plane Shear Forces in the Reactor Pressure Vessel Pedestal

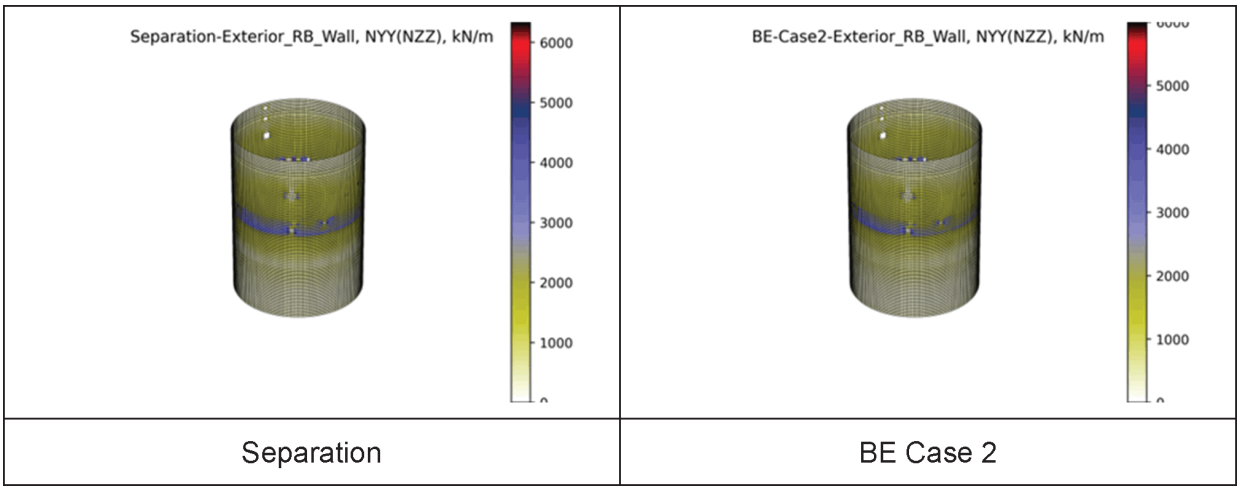


Figure 3C-46 Separation Effects on Vertical Forces in the Reactor Building External Wall

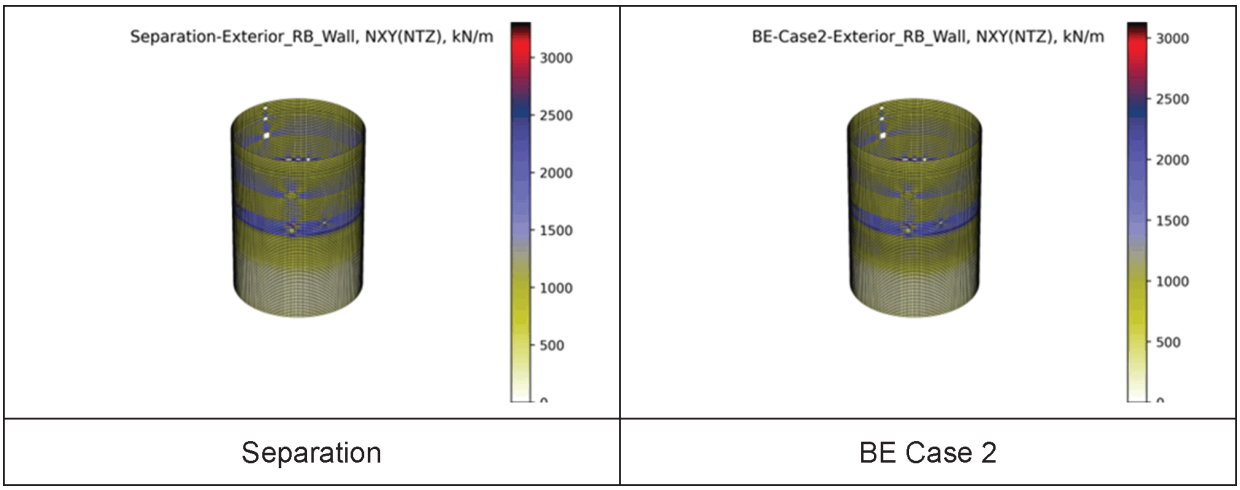


Figure 3C-47 Separation Effects on In-Plane Shear Forces in the Reactor Building Exterior Wall

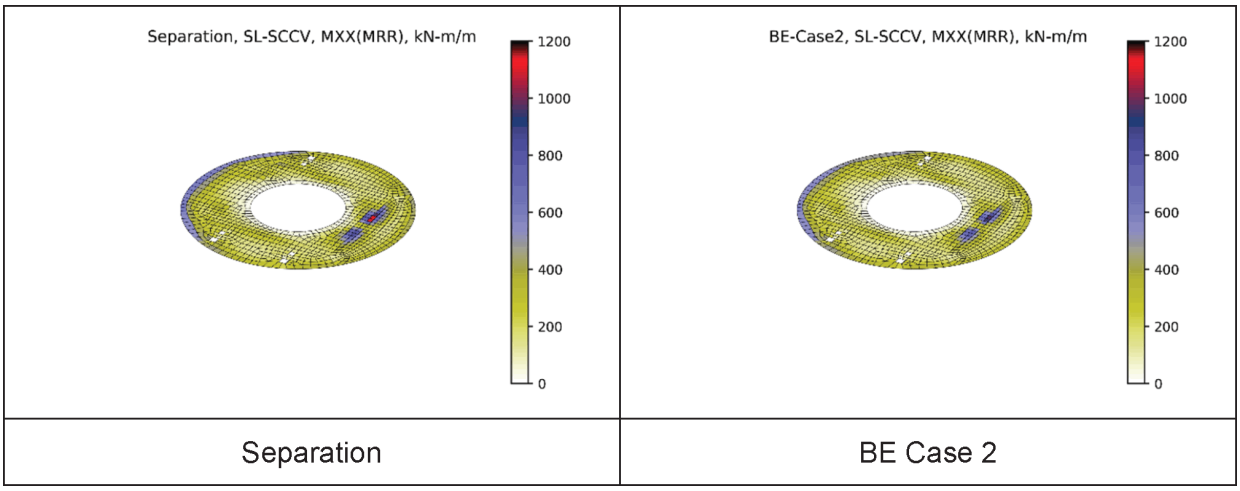


Figure 3C-48 Separation Effects on Containment Top Slab Out-of-Plane Moment

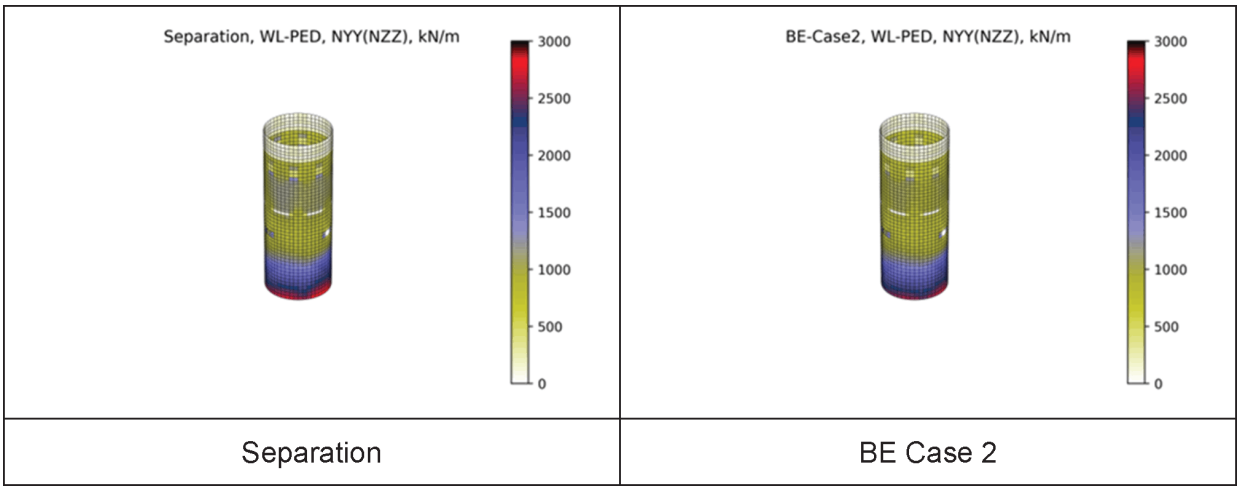
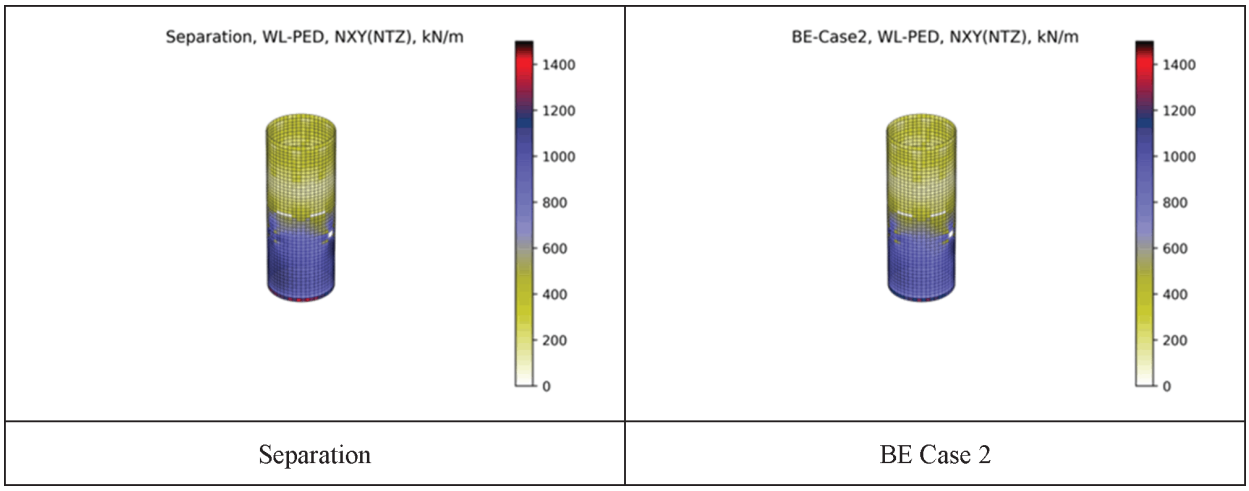
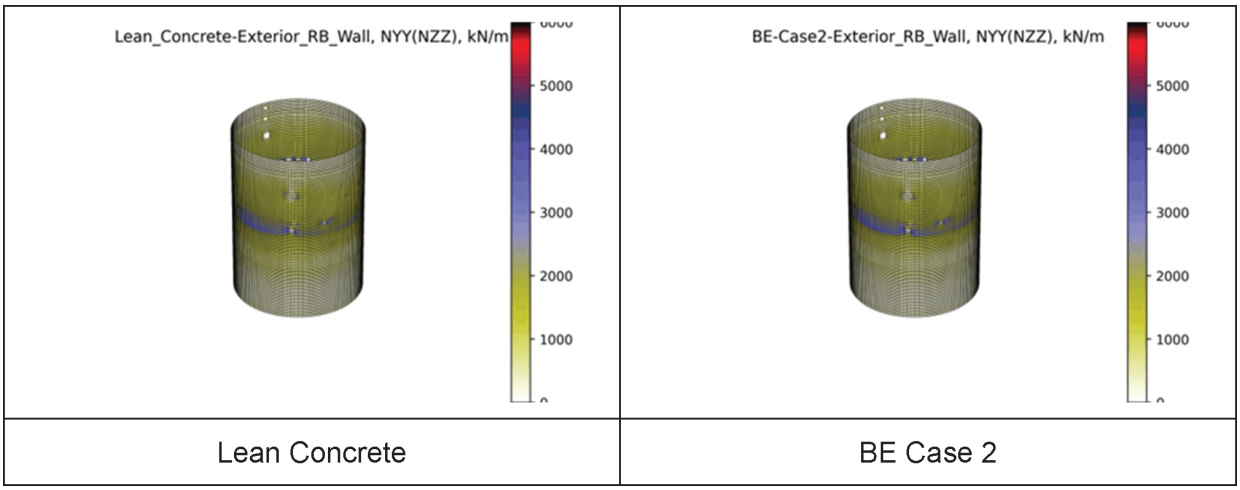


Figure 3C-49 Separation Effects on Vertical Forces in the Reactor Pressure Vessel Pedestal



Separation	BE Case 2

Figure 3C-50 Separation Effects on In-Plane Shear Forces in the Reactor Pressure Vessel Pedestal



Lean Concrete	BE Case 2

Figure 3C-51 Concrete Fill Strength Effects on Vertical Forces in the Reactor Building Exterior Wall

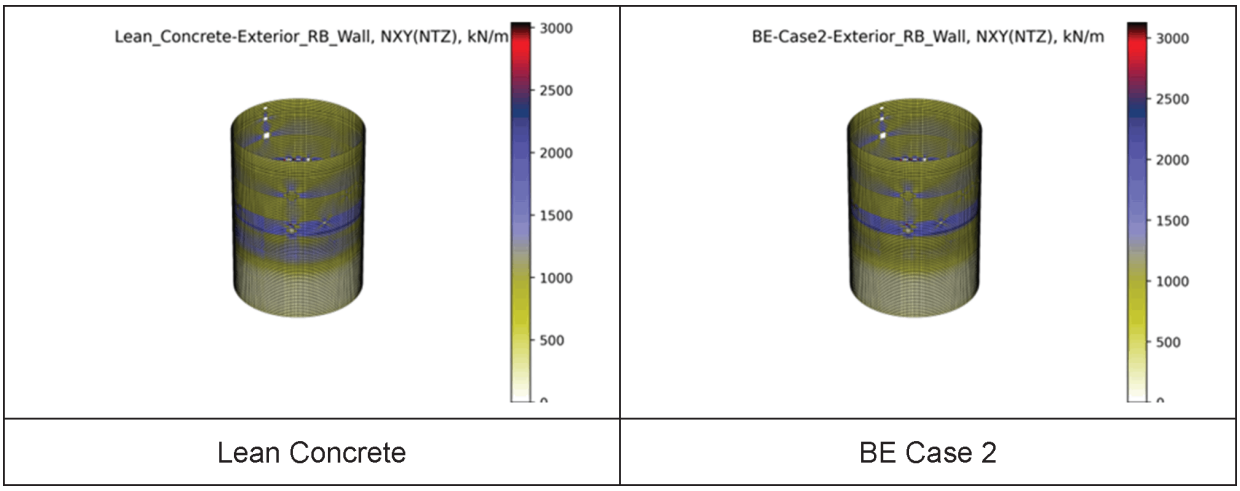


Figure 3C-52 Concrete Fill Strength Effects on In-Plane Shear Forces in the Reactor Building Exterior Wall

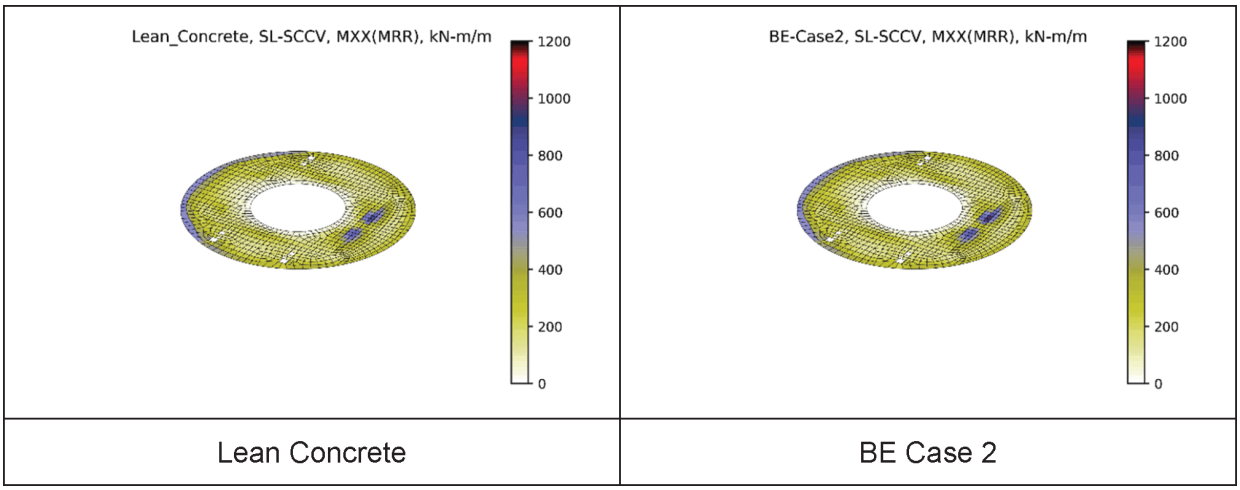
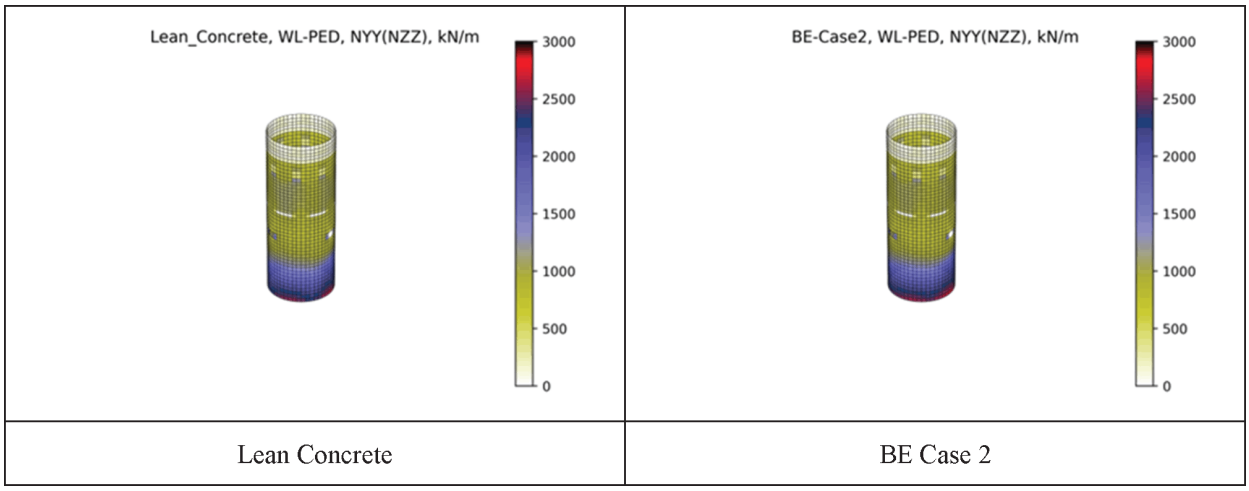


Figure 3C-53 Concrete Fill Strength Effects on Containment Top Slab Out-of-Plane Moment



Lean Concrete	BE Case 2

Figure 3C-54 Concrete Fill Strength Effects on Vertical Forces in the Reactor Pressure Vessel Pedestal

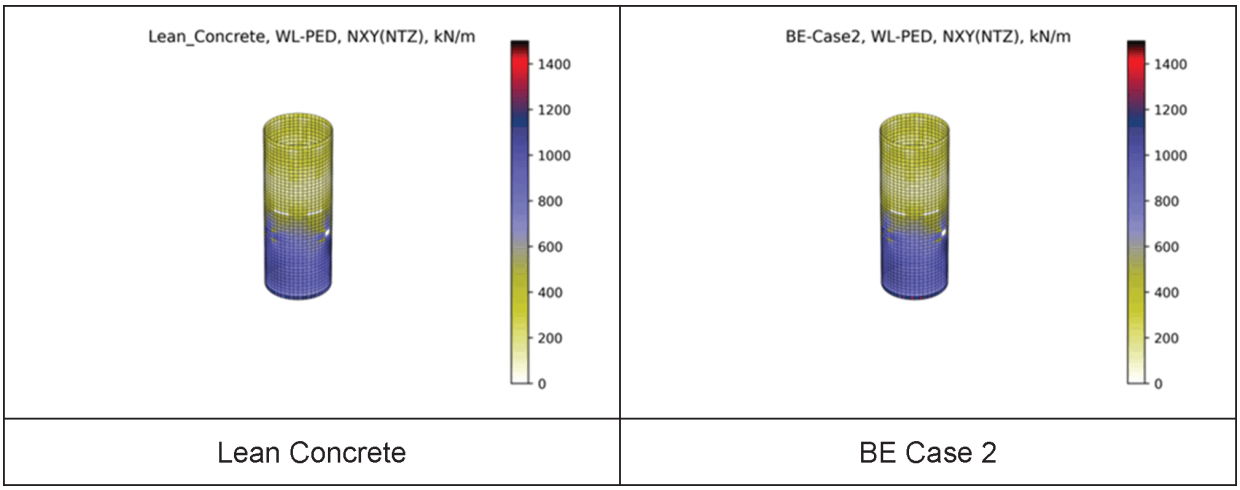
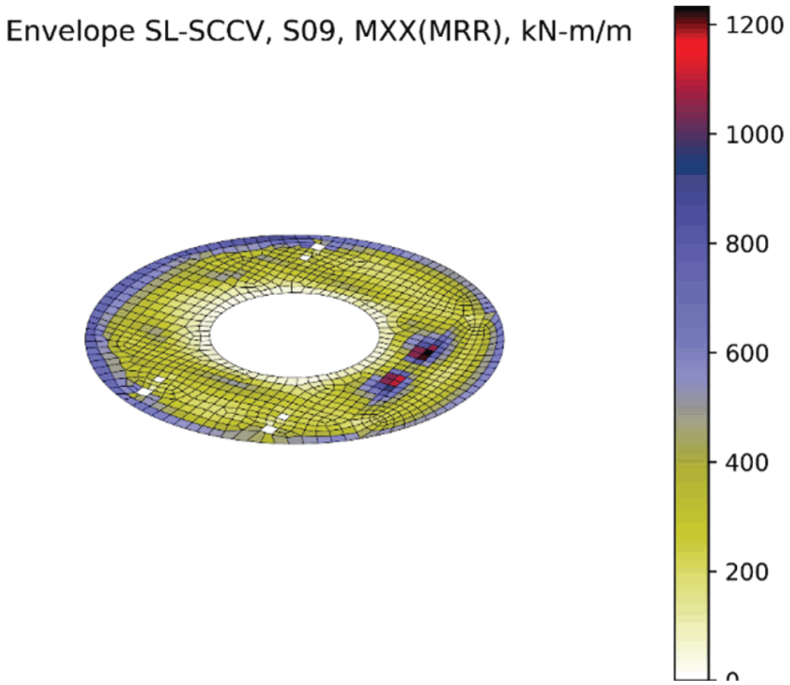
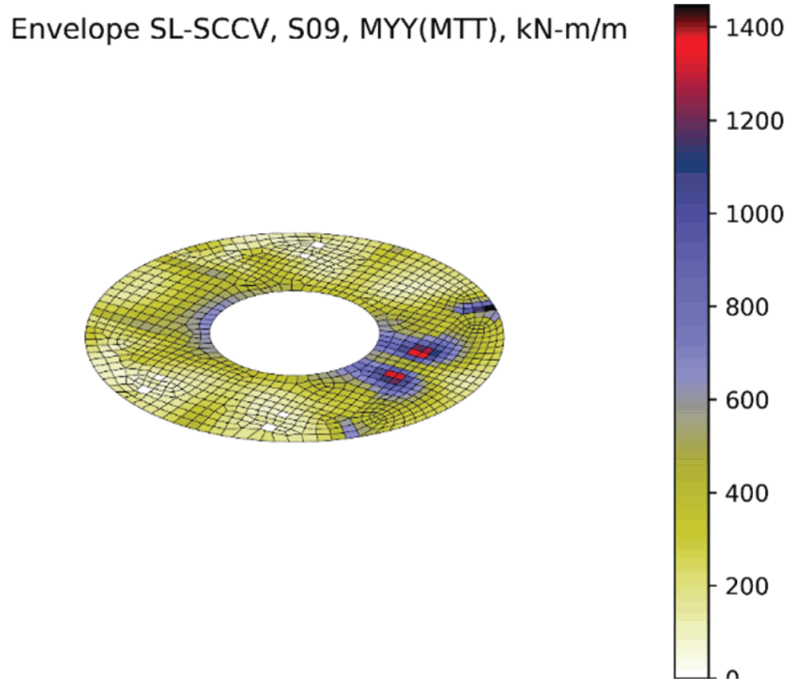


Figure 3C-55 Concrete Fill Strength Effects on In-Plane Shear Forces in the Reactor Pressure Vessel Pedestal

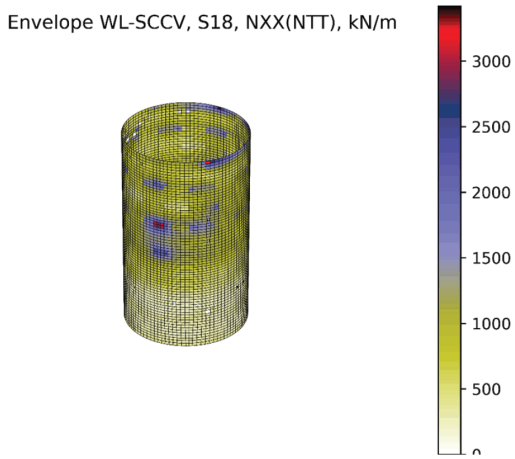


(a) Radial Moment

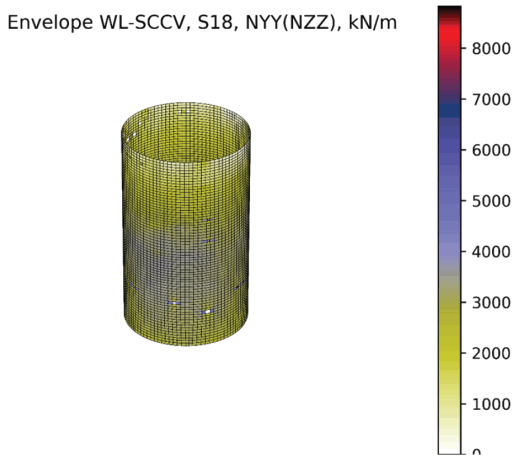


(b) Tangential Moment

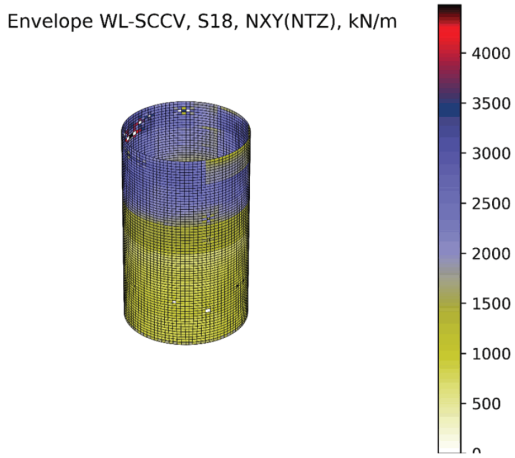
Figure 3C-56 Seismic Design Basis Moment Demands for the Containment Top Slab



(a) Tangential (Hoop) Forces

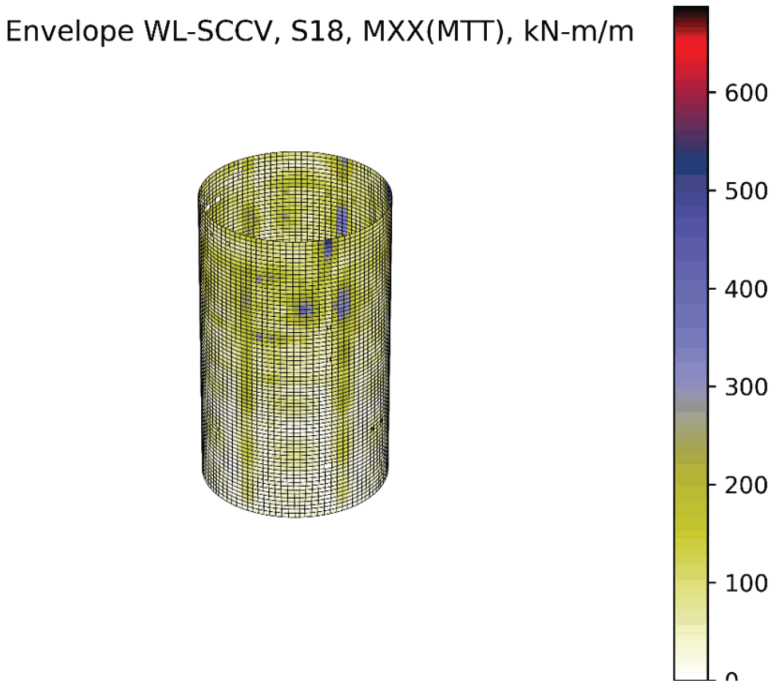


(b) Vertical Forces

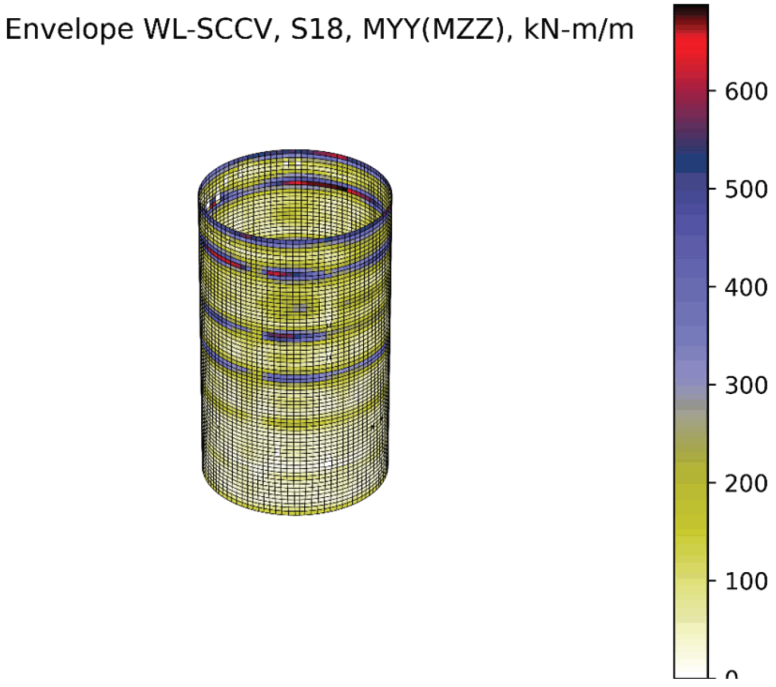


(c) In-Plane Shear Forces

Figure 3C-57 Seismic Design Basis Membrane Forces for the Containment Wall

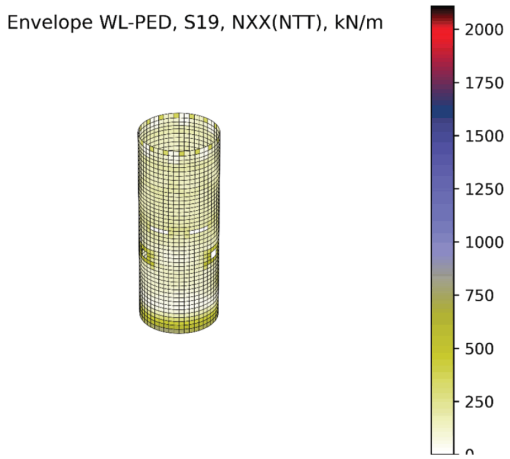


(a) Tangential Moments

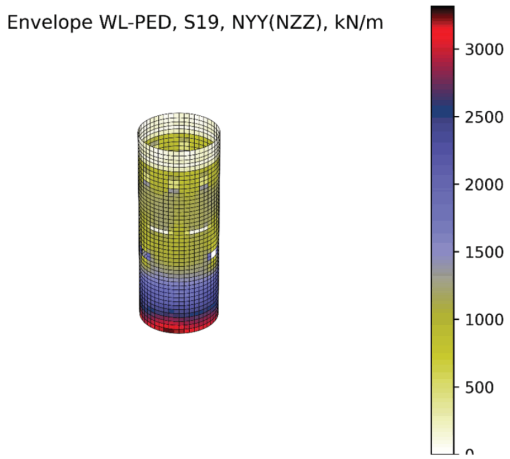


(b) Vertical Moments

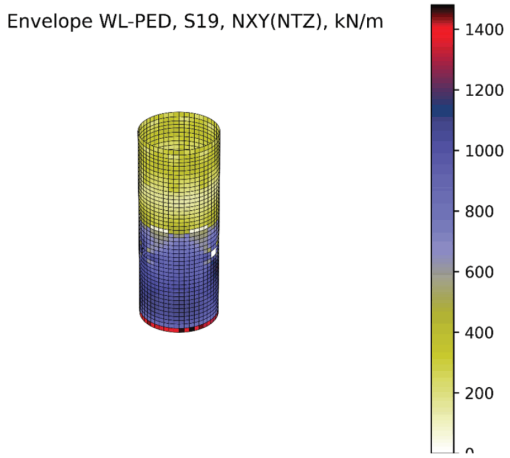
Figure 3C-58 Seismic Design Basis Moments for the Containment Wall



(a) Tangential (Hoop) Forces

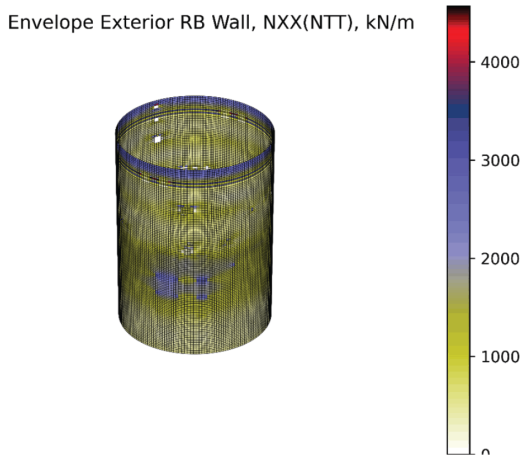


(b) Vertical Forces

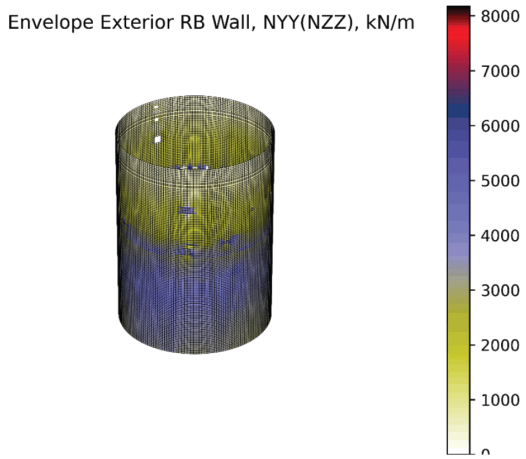


(c) In-Plane Shear Forces

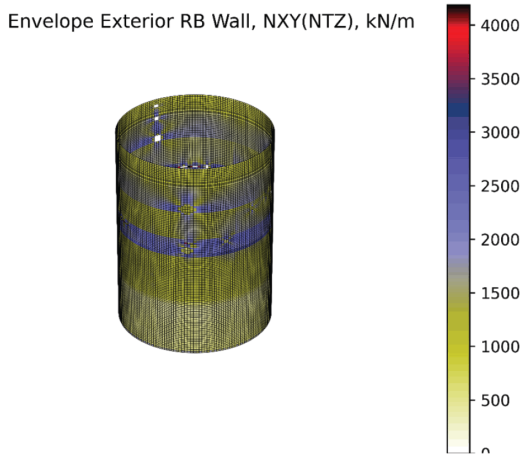
Figure 3C-59 Seismic Design Basis Membrane Forces for the Reactor Pressure Vessel Pedestal



(a) Tangential (Hoop) Forces

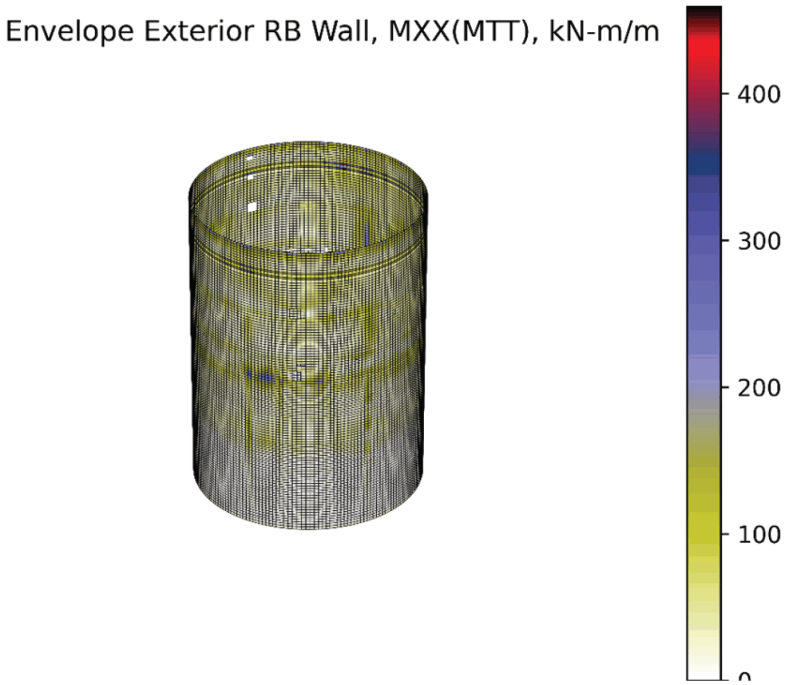


(b) Vertical Forces

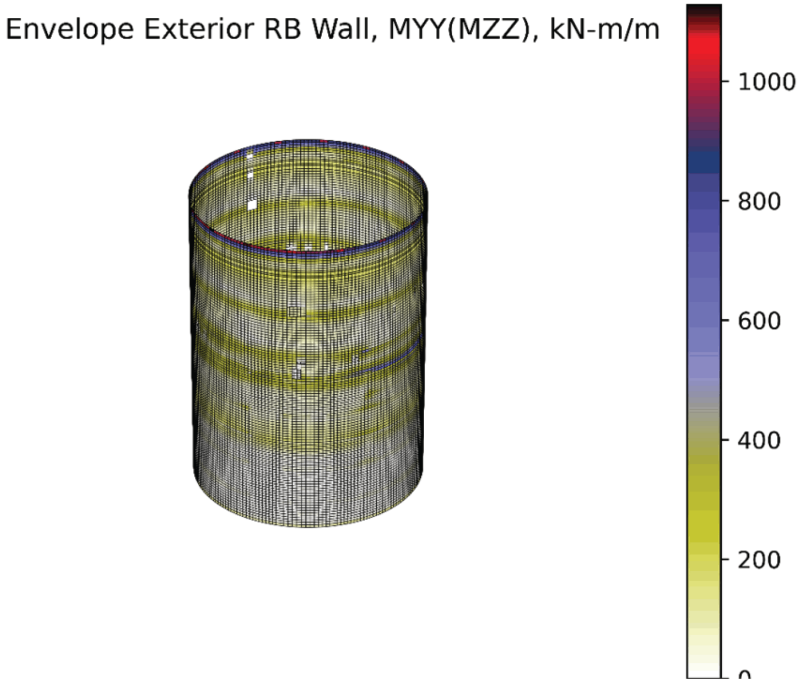


(c) In-Plane Shear Forces

Figure 3C-60 Seismic Design Basis Membrane Forces for the Reactor Building Exterior Wall



(a) Tangential Moments



(b) Vertical Moments

Figure 3C-61 Seismic Design Basis Moments for the Reactor Building Exterior Wall

APPENDIX 3D INTERACTION EVALUATIONS RESULTS

3D.1 Introduction

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

This appendix documents the 1-g static SSI, static and quasi-static and thermal analyses performed for the design of the integrated Reactor Building (RB) structures. The analyses are performed per the methodology presented in Subsection 3.8.4.1.4, using the Finite Element (FE) models described Section 3B.3 through Section 3B.5, and design parameters discussed in Section 3E.1.

The 1-g static SSI, static and quasi-static and thermal analyses are performed for the applicable loads listed in Section 3.8, excluding construction, refueling and piping reaction loading. These loads are judged to be non-governing for the design of the integrated RB structures and will be considered in the final design of the structures. Tornado missiles loadings are also not considered in the analyses as the thickness of the RB exterior wall above grade and the roof thickness is 2-3 times larger than the minimum acceptable barrier thickness requirements listed in Table 3.5-2. Tornado missile loading effects are also not expected to govern the design of the integrated RB and this assumption will be validated in the final design of structures.

Beyond design basis events are also not included in the load cases summarized in this appendix.

3E.1 Design Parameters Considered in the Structural Analyses

The following are the design parameters considered in the 1-g static SSI, static and quasi-static and thermal analyses. For live loads considered in the analyses, refer to Subsection 3B.1.7.2.

3E.1.1 Equivalent Linear Static Subgrade Properties

The site-specific geotechnical parameters presented in Table 3.8-8 and Table 3.8-9 are used to develop the input for the 1-g static SSI and subgrade impedance analyses used to evaluate the structural response to significant static and thermal design loads.

Table 3E-1 identifies the equivalent linear static subgrade profiles used for each of the structural analyses. The Lower Bound (LB) and Upper Bound (UB) elastic moduli (E_{st}) and Poisson's ratios (ν_{st}) in Table 3E-1 are the LB and UB as-built static subgrade properties provided in Table 3.8-8 and Table 3.8-9. The effective soil unit weight (γ) is calculated as the total soil/rock unit weights provided in Table 3.8-8 and Table 3.8-9 minus the unit weight of water.

LB subgrade properties are used for the static and quasi-static analyses to emphasize the deformations at the subgrade-structure interfaces resulting in UB estimates of member forces and moment demands. Both LB and UB subgrade stiffness properties E_{st} and ν_{st} are considered for the thermal stress analyses to bound the thermal stress demands on the RB integrated structures.

To evaluate the RB exterior wall, the 1-g SSI analyses use UB soil unit weight, LB E_{st} and UB ν_{st} to emphasize the lateral earth pressures applied from the subgrade on the RB exterior wall.

3E.1.2 Nominal Groundwater Table Elevation

The groundwater at the Clinch River Nuclear (CRN) site is located at plant finished grade (Elevation 814.5 ft) as per Table 2.0-1R.

3E.1.3 Additional Horizontal Rock Pressure

Residual horizontal stresses that exceed the magnitude of the vertical stresses may be present in the rock masses at the CRN site due to past seismic activities. The results of past studies indicate these stresses are anisotropic, i.e., the magnitude of the pressures in the two orthogonal horizontal directions is different.

Rock excavation can relief the residual horizontal stresses due to deformation of the rock mass. However, the magnitude of the rock mass displacements and resulting relief of residual rock stresses depends on the duration of the excavation and rock reinforcement that may be used to secure the excavation. The durability of the temporary rock reinforcement elements during the life of the plant is not guaranteed. If the rock reinforcement decays with time, the in-situ residual stress not already relieved during excavation can transfer from the reinforcement to the RB exterior wall. As a result, residual rock pressures, ranging circumferentially between 43.5 psi and 14.5 psi, are applied on the portion of the RB embedded in rock to account for the additional rock pressure from unstable rock. These pressures are applied uniformly along the depth of the embedded RB exterior wall starting at depth of 12.9 m (42.3 ft) below plant grade down to the bottom of the mat foundation but non-symmetrically around the RB circumference within the rock layers.

3E.1.4 Ambient Temperature

The stress-free temperature for the design of the RB and Steel-Plate Composite Containment Vessel (SCCV) at the CRN site is 60°F.

The design temperatures representing the extreme external atmospheric conditions considered in the thermal analyses are -40°F in winter and 104°F in summer. Although the governing summer ambient temperature at the CRN Site is 107°F per Table 2.0-1R, the 3°F difference is not significant as thermal stress demands from the summer condition do not govern the thermal stress demands from winter conditions which create the most severe thermal gradients on steel-plate composite components.

3E.1.5 Precipitation Design Parameters

The CRN RB design considers a normal and extreme ground snow loading of 52.2 psf and 104 psf, respectively. These loads bound the CRN design-basis ground snow loads of 21.93 psf.

3E.1.6 Wind and Extreme Wind Parameters

As stated in Subsection 3.3.1.1, the integrated RB is designed for a wind load of 160 mph based on the ASCE/SEI 7-16, “Minimum Design Loads and Associated Criteria for Buildings and Other Structures,” (Reference 3E-1) Risk Category IV, basic wind speed maps (3-second gust). This wind speed bounds the design basis wind speeds including hurricane for the CRN site (130 mph) listed in Table 2.0-1R.

The design basis tornado parameters are those listed in Table 2.0-1R. Extreme wind-generated missiles considered in the design of the integrated RB are those listed in Table 3.5-1.

Tornado wind load calculations were performed using both the BC-TOP-3-A, “Tornado and Extreme Wind Design Criteria for Nuclear Power Plants,” (Reference 3E-2) approach as discussed in Section 3.3.2.2 and the ASCE/SEI 7-22, “Minimum Design Loads and Associated Criteria for Buildings and Other Structures,” (Reference 3E-3) approach, with the conclusion that the latter provides bounding pressure parameters. The bounding parameters were applied to the analysis model.

3E.1.7 Pool Water Levels

Pool water levels used for the hydrostatic load case are based on normal operating levels and are as follows:

- Isolation condenser pool water depth is 23.6 ft
- Equipment and reactor cavity pools water depth is 25.6 ft
- Fuel pool water depth is 44.9 ft

3E.1.8 Crane Loading

Table 3E-2 summarizes the crane loads applied on the crane supporting structures as nodal forces in the FE model.

3E.1.9 Containment Internal Pressure Loading

The containment pressure loadings considered in the analyses are the normal operating pressure loads P_o, the design basis accident pressure P_{a1} generated by a Loss-of-Coolant Accident (LOCA) inside containment, and the test pressure load (P_t) presented in Table 3E-3.

Conservatively, the negative containment pressure (P_{o-}) which does not occur under normal operation but can occur during Anticipated Operational Occurrences (AOOs) is considered for the normal operating load case as indicated in Table 3E-3.

Test pressure load (P_t) applied to the SCCV is defined as 1.15 times the containment accident pressure (P_{a1}).

3E.1.10 Internal Pressure Loading Outside Containment

An AOO exists in which the isolation condenser pools will develop internal pressure as the pool water boils and the enclosed pools develop internal pressure. In the absence of more information about the event precipitating the AOO, an internal pressure in the isolation condenser pools ($P_{op,}$ see Table 3E-3) of 4.5 psi is also considered in the analyses.

An accident inside the steam tunnel causing internal pressure P_{a2} is also considered in the analyses as indicated in Table 3E-3.

3E.1.11 Internal Flooding

The RB subgrade floors are not expected to flood simultaneously due to a water line rupture. Based on preliminary design evaluations, flooding of the floor slab above the common mat foundation is observed to be the most critical. As a result, a lone flood depth of 4.9 ft is considered to act on the floor slab above the common mat foundation.

Figure 3E-1 presents the hydrostatic loads considered in the design of the integrated RB structures as a result of a containment flooding generated by LOCA.

3E.1.12 Thermal Conditions

Thermal stress analyses are performed by applying temperature loads based on heat transfer analysis.

The thermal stress analyses consider two cases, winter and summer and four accident conditions, with two accident durations each, as indicated in Table 3B-1.

Table 3E-4 present the summer and winter thermal conditions inside and outside the integrated RB during normal operations.

Summer/winter thermal conditions inside containment during normal operations are taken as 149°F. For accident conditions, the maximum design internal temperature considered is 330°F, which is in compliance with the Limitation and Condition (L&C) 8.6 of NEDC-33926P-A, “BWRX-300 Steel-Plate Composite (SC) Containment Vessel (SCCV) and Reactor Building Structural Design” (Reference 3E-5).

3E.2 1-g Static Soil-Structure Interaction Analysis

The 1-g static SSI analysis is performed in ANSYS and considers the interaction of the deeply embedded RB with the surrounding Power Block structures and subgrade under 1-g loads (See Appendix 3I).

The model used for the 1-g static SSI analyses is described Section 3B.3. The soil and rock layers of the model are assigned the equivalent linear elastic subgrade properties summarized in Table 3E-1. Structural elements of the model are assigned BE stiffness properties as discussed Section 3B.3. Submerged unit weight properties are assigned to the subgrade materials located

below the bounding nominal groundwater level specified at plant finished grade as noted in Subsection 3E.1.2. Boundary conditions considered for the 1-g static SSI analysis are described in Table 3B-1.

3E.2.1 Analysis Cases

Two sets of vertical 1-g static SSI analyses are performed for the unfactored and factored gravity inertia load conditions as discussed in Section 3B.3 and shown in Table 3B-1. Demands obtained from the 1-g static SSI analysis are from:

- The dead load from the self-weight and weight of permanent equipment and components.
- The vertical fluid load on pool slabs from the weight of the water in the pools.
- The static earth pressure load due to submerged weight of the soil and rock.
- The surcharge lateral pressure load from the surrounding Power Block surface mounted foundations conservatively captured by assigning higher mass density properties as discussed Section 3B.3.
- Crane girders and trolley weights applied as vertical forces with the trolley parked at the end of the crane.

In the model used to analyze the factored load combination, the submerged unit weight properties of the subgrade and surrounding structures are increased by 33% to account for the higher factored lateral earth pressure loads also as described in Section 3B.3.

3E.2.2 Response and Demands from Dead Load and Earth Pressure Loads

Figure 3E-2 and Figure 3E-3 present the magnitude of internal forces obtained from the 1-g SSI analysis in the RB exterior wall and SCCV wall, respectively.

3E.3 Static and Quasi-Static Analysis

The analyses of static and quasi-static loads are performed on the FE models described Section 3B.4. As stated in Subsection 3.8.4.1.4, the analysis of the static and quasi-static loads that have global effect on the response of the integrated RB structural model are performed using the subgrade impedance sub-structuring method that accounts for the subgrade stiffness at the interfaces with the RB structure. Linear elastic contact springs connect the integrated RB structural model and subgrade FE models developed using the subgrade stiffness impedance methodology. Stiffness properties are assigned to the contact springs to adequately represent the interaction mechanism between the RB structure, backfill, and subgrade.

Analyses performed to determine force/moment demands due to load cases for which the interaction with the surrounding soil is not important are performed on models with prescribed boundary conditions.

Table 3B-1 summarizes the modeling requirements for the various static and quasi-static analysis cases required for the design of the integrated RB structures.

3E.3.1 Analysis Cases

The static and quasi-static analysis cases performed are those described in Table 3B-1, excluding analysis cases associated with normal operating and Design Basis Accident (DBA) nozzle, equipment and piping reaction loads and local loads on containment during DBA for which information is not available.

The following subsections present responses and demands obtained from load cases governing the design of the integrated RB structures.

3E.3.2 Response and Demands from Horizontal Hydrostatic and Hydrodynamic Loads

Horizontal pool water loads applied on the integrated RB model consist of the horizontal hydrostatic loads (F_h) and sloshing and breathing mode hydrodynamic pressures due to seismic (Safe Shutdown Earthquake (SSE)) excitation.

Hydrostatic pressures based on the pool water levels listed in Subsection 3E.1.7 are applied as an outward normal pressure with a gradient equal to the density of water at approximately 60°F (62.4 lb/ft^3) multiplied by the distance from top of water. No pressure is applied on the intermediate pool walls that restrain fluid on both sides as the net force on the intermediate wall is negligible.

The deformed shape response of the pool walls due to hydrostatic loads is shown in Figure 3E-4(a). The outward expansion of the pool walls shown in Figure 3E-4(a) is consistent with the loading.

Hydrodynamic pressures on pool walls due to vertical earthquake excitation, referred to as breathing water pressure loads, are also applied as outward normal pressures on pool walls using the same approach as for hydrostatic pressures. The magnitude of the fluid pressure is scaled based on the vertical Zero Period Acceleration (ZPA) the pools experience presented in Table 3E-5. Breathing water pressures are applied on intermediate pool walls that restrain fluid from both sides as these walls may experience a net pressure depending on the differential ZPA in various regions.

As illustrated in Figure 3E-4(b), the outward expansion of the pool walls under breathing water pressures is also consistent with the loading.

Sloshing pressure is obtained based on the 0.5% damped spectral acceleration corresponding to the sloshing mode frequency at the pool elevation. Sloshing pressure is based on a bounding spectral acceleration of 0.5 g. Sloshing pressure loads are applied to the ANSYS model as nodal forces in the corresponding sloshing directions for X (North-South) and Y (East-West) loading directions. The sloshing pressure force magnitudes are dependent on the pool plan geometry, water depth, and seismic input. For the calculation of the sloshing pressures, the RB pools are divided in multiple regions of equivalent rectangular dimensions. For each pool region, the general sloshing parameters are calculated, then the pressure for each wall or floor shell is calculated by considering the relative water depth and location. Corner forces are calculated for each shell using the projected shell area in the sloshing direction.

3E.3.3 Responses and Demands from Containment Internal Pressure Load Cases

The containment pressure loadings listed in Table 3E-3 are applied normal to the SCCV wall, SCCV top slab, inner mat foundation, and containment closure head.

Figure 3E-5 presents the accident pressure P_{a1}, selected as representative of other pressure loads applied on the containment, and the resulting deformed shape of the integrated RB due to the accident pressure.

3E.3.4 Responses and Demands from Groundwater and Rock Pressure Loads

The lateral groundwater pressure is applied as an inward normal pressure with a gradient equal to the density of water at approximately 60°F (62.4 lb/ft^3) multiplied by the distance starting from grade down to the centerline of the common mat foundation.

The 114.8 ft of groundwater causing buoyancy uplift pressure on the mat foundation is applied as an upward normal load on the mat foundation model.

Rock pressures described in Subsection 3E.1.3 are applied as an inward normal pressure on the embedded RB with a gradient applied circumferentially between 43.5 psi and 14.5 psi with linear interpolation in between as illustrated in Figure 3E-6. The rock pressures are applied as nodal forces on the RB exterior wall starting at a depth of 12.9 m (42.3 ft) below the grade centerline.

Responses and demands due to groundwater and rock pressure loads are consistent with the loading as demonstrated by Figure 3E-7.

3E.4 Thermal Stress Analysis

Thermal stress analyses are performed by applying body loads to the shell and beam elements of the models described in Section 3B.5 to induce thermal strains. Shell elements are assigned temperatures to each face to induce a linear thermal gradient through the shell thickness while beam elements are assigned a uniform temperature throughout their length and cross-section. Modeling requirements for thermal load cases considered for the design of the integrated RB structures are presented in Table 3B-1.

3E.4.1 Analysis Cases

Table 3B-1 presents the thermal stress load cases considered for the design of the BWRX-300 integrated RB structures.

As indicated in Table 3B-1, four design basis accident conditions are considered in the heat transfer analyses in addition to the normal operating thermal conditions, with DBA-1 being a design basis accident inside containment. The heat transfer analysis only considers the extreme winter conditions with an outdoor temperature of -40°F which create the most severe thermal gradients on steel-plate composite components due to the larger differences between the outside temperature and the temperatures in the BWRX-300 containment and RB. Summer conditions are only considered for the DBA-1 analysis cases.

The FE models analyzed for normal operating thermal conditions are assigned effective structural stiffnesses, while partially cracked models are used in DBA analyses. The four DBAs result in temperatures on the walls and slabs within the DBA region which exceed the threshold temperature defined in ANSI/AISC N690-18, “Specification for Safety-Related Steel Structures for Nuclear Facilities,” (Reference 3E-4) and require fully cracked equivalent section properties to be assigned to the models. Under normal operating temperature load conditions, the steel-plate composite elements are not further cracked beyond the effective structural stiffness because gradient loads are small and develop over significant time.

For both normal operating and DBA thermal load cases, the RB model is considered with LB and UB subgrade stiffness to determine the bounding conditions for determining internal design forces.

3E.4.2 Response and Demands from Thermal Loads

The deformed shapes for normal operating and DBA winter thermal conditions are shown in Figure 3E-8.

The expansion and contraction of the RB exterior wall is consistent with the effect of the differential temperatures exterior and interior to the RB. The RPV pedestal experiences the highest expansion deformations because of the higher temperatures in the SCCV and the essentially unrestrained thermal expansion of the cantilevered component.

Internal forces from the DBA-1 thermal analysis with summer exterior temperatures and an UB subgrade condition are presented in Figure 3E-9.

3E.5 References

3E-1 ASCE/SEI 7-16, “Minimum Design Loads and Associated Criteria for Buildings and Other Structures,” American Society of Civil Engineers, 2016.

3E-2 BC-TOP-3-A, “Tornado and Extreme Wind Design Criteria for Nuclear Power Plants,” Bechtel Power Corporation, Revision 3, August 1974.

3E-3 ASCE/SEI 7-22, “Minimum Design Loads and Associated Criteria for Buildings and Other Structures,” American Society of Civil Engineers, 2022.

3E-4 ANSI/AISC N690-18, “Specification for Safety-Related Steel Structures for Nuclear Facilities,” American National Standards Institute/American Institute of Steel Construction, 2018.

3E-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.

Table 3E-1 Equivalent Linear Static Subgrade Profiles

Analysis Case	Parameters Used
1-g SSI analysis	UB γ, LB E_{st} and UB ν_{st}
Static and quasi-static analyses	LB E_{st} and LB ν_{st}
Thermal analysis (Operational and Accident)	UB E_{st} and UB ν_{st} LB E_{st} and LB ν_{st}

Notes:

(1) γ = Effective unit weight for subgrade materials below groundwater table

(2) E_{st} = Young Modulus representing linearized stiffness properties of the soil and rock for long-term static loading conditions

(3) ν_{st} = Poisson's ratio representative of at-rest lateral earth pressure conditions

Table 3E-2 Crane Loading

Crane load case	Crane payload	Load application direction	Note
Crane vertical payload	251.3 kip	Downwards (-Z)	Crane payload capacity
Crane transverse load	62.5 kip	+Y	25% of vertical payload taken as Crane transverse load
Crane Hoisting impact load	41.2 kip	Downwards (-Z)	Considering lifted load, load block, and attachment
Crane horizontal load	62.5 kip	+X	25% of vertical payload taken as Crane horizontal load

Table 3E-3 Containment Pressure Loading

Pressure Load		Pressure value	
		kPa(g)	Psi(g)
Normal positive pressure	P_{o+}	18.7	2.7
Normal negative pressure	P_{o-}	-14	-2
Design Basis Accident 1 Peak Pressure	P_{a1}	413.7	60
Design Basis Accident 2 Peak Pressure	P_{a2}	839	122
Test Peak Pressure	P_t	475.73	69
Isolation Condenser Pool Pressurization	P_{op}	31.0	4.5

Table 3E-4 Summer/Winter Thermal Conditions Inside and Outside Integrated Reactor Building During Normal Operations

REGION	SUMMER (or Maximum)	WINTER (or Minimum)
Main Steam/Feedwater Penetration Room	104°F	50°F
Interior	104°F	50°F
RB Outdoor Temperature	104°F	-40°F
Isolation Condenser Pools	110°F	110°F
Ground	60°F	60°F

Note:

Although the governing summer ambient temperature at the CRN Site is 107°F per Table 2.0-1R, the 3ºF difference is not significant as thermal stress demands from the summer condition do not govern the thermal stress demands from winter conditions which create the most severe thermal gradients on steel-plate composite components.

Table 3E-5 Breathing Mode Accelerations

Floor Elevation [m(ft)]	Breathing Mode Vertical ZPA [g]
0 (0)	0.69
5.138 (16.857)	0.69 (1)

Note:

(1) For area outside the connected reactor cavity pool and equipment pool. For the reactor cavity pool and equipment pool regions, 1.76 g is used.

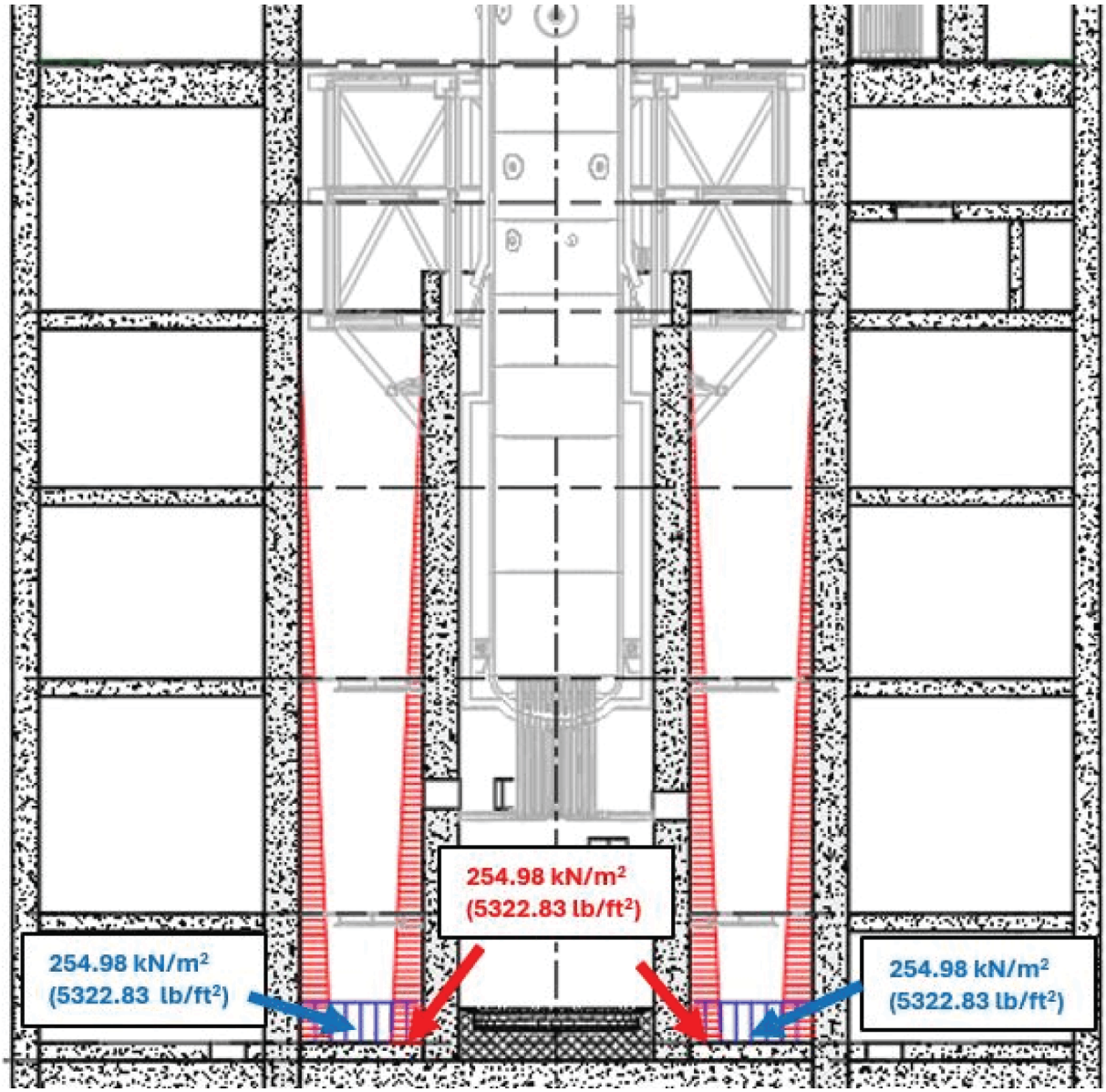


Figure 3E-1 Loss-of-Coolant Accident Containment Flooding

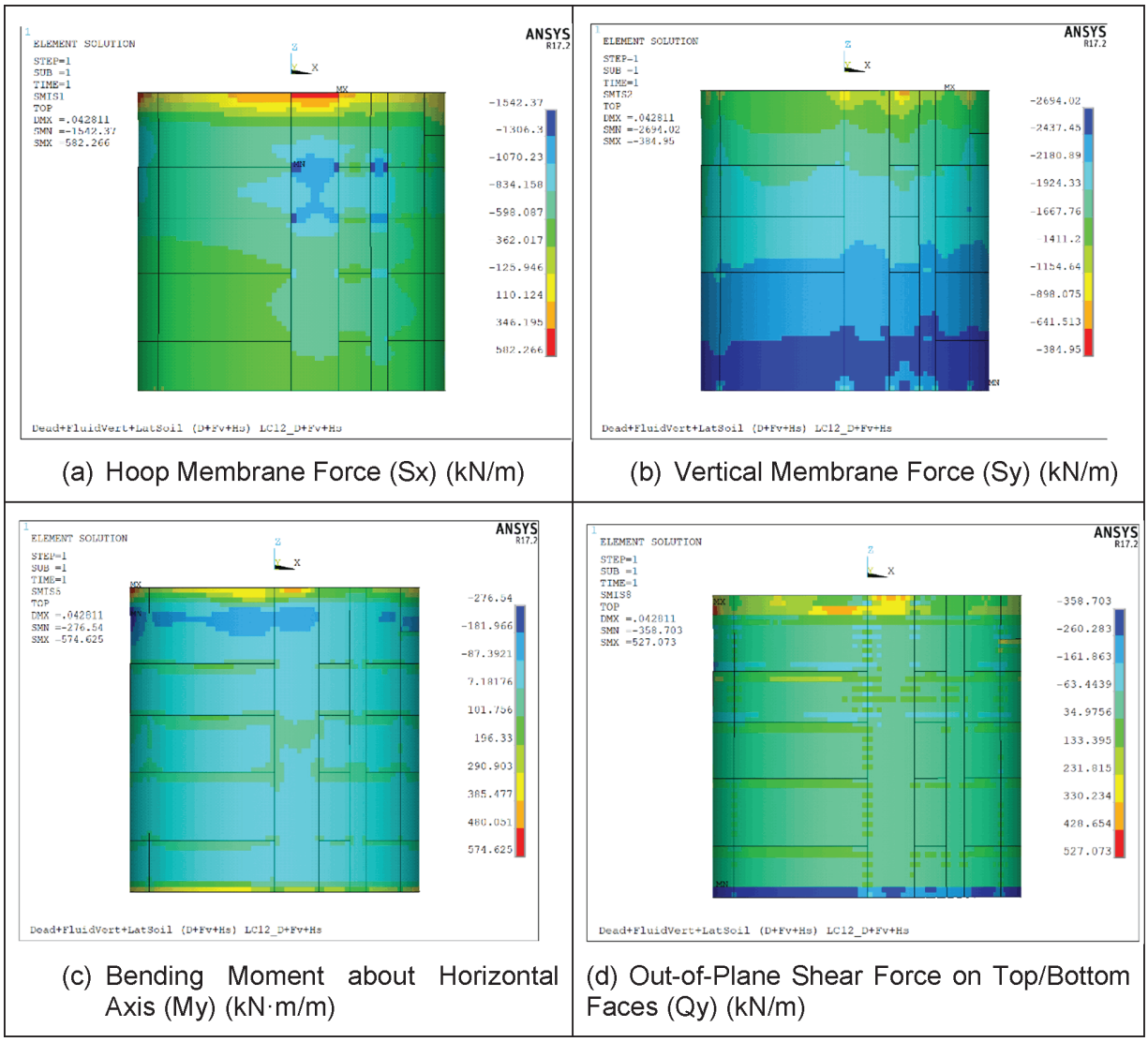


Figure 3E-2 Internal Forces from 1-g Soil-Structure Interaction Analysis in Reactor Building Exterior Wall

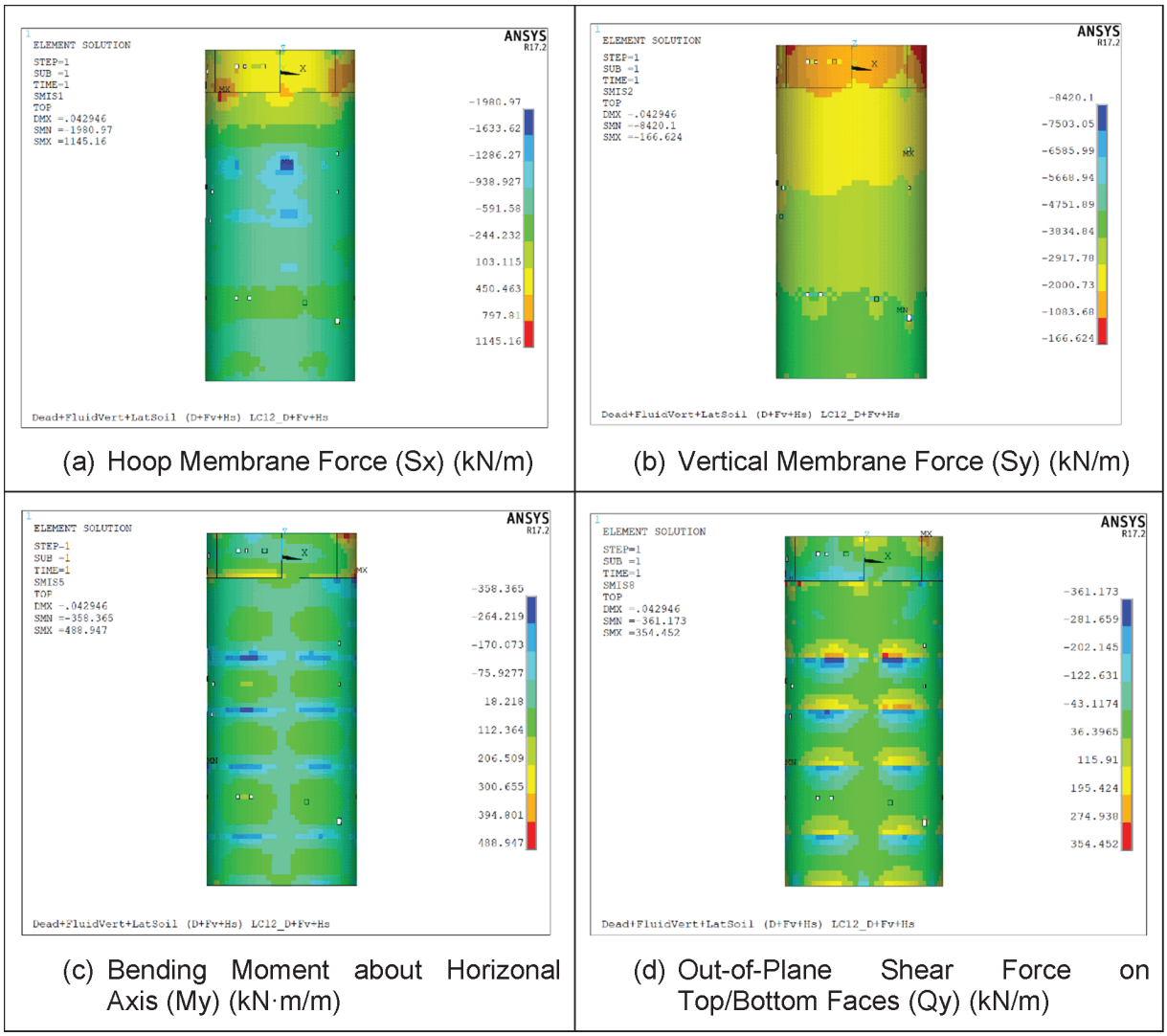
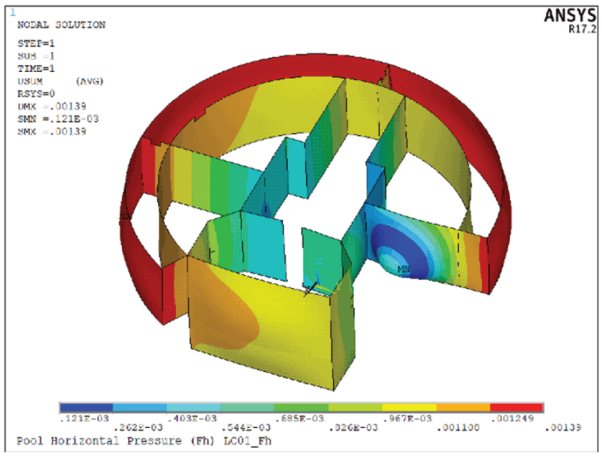
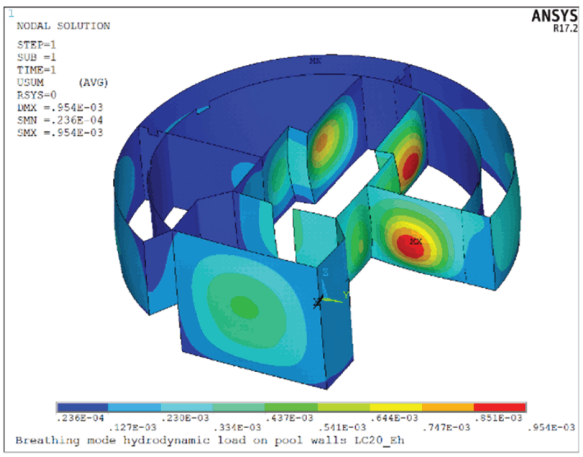


Figure 3E-3 Internal Forces from 1-g Soil-Structure Interaction Analysis in Containment Wall

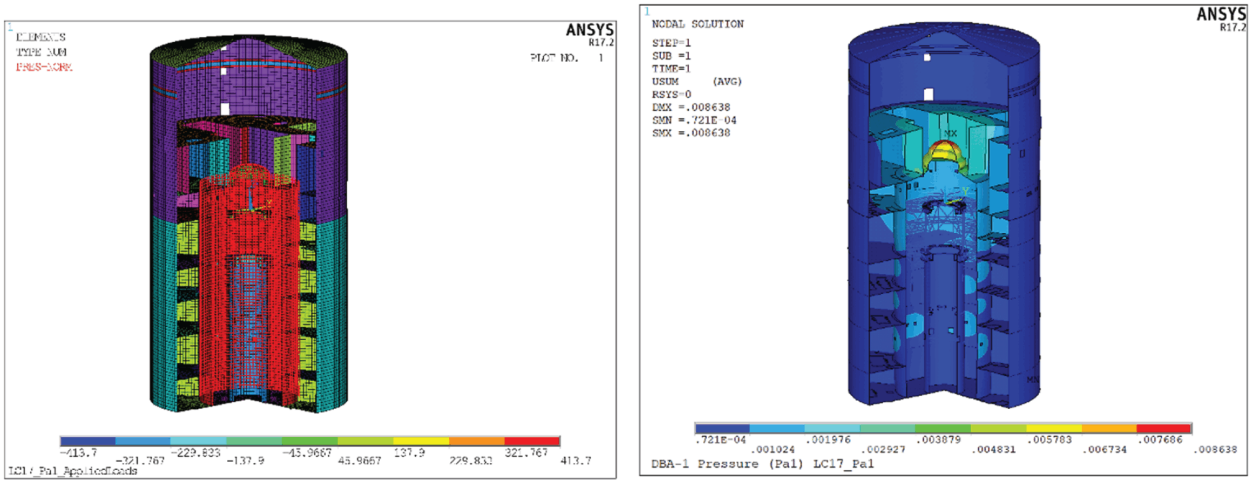


(a)Hydrostatic



(b) Hydrodynamic

Figure 3E-4 Deformed Shape for Hydrostatic and Hydrodynamic Loads (Units: m)



(a) Accident Pressure Loads (kPa) (b) Deformed Shape (m)

Figure 3E-5 Accident Pressure Loading Inside Containment and Corresponding Deformed Shape of the Integrated Reactor Building

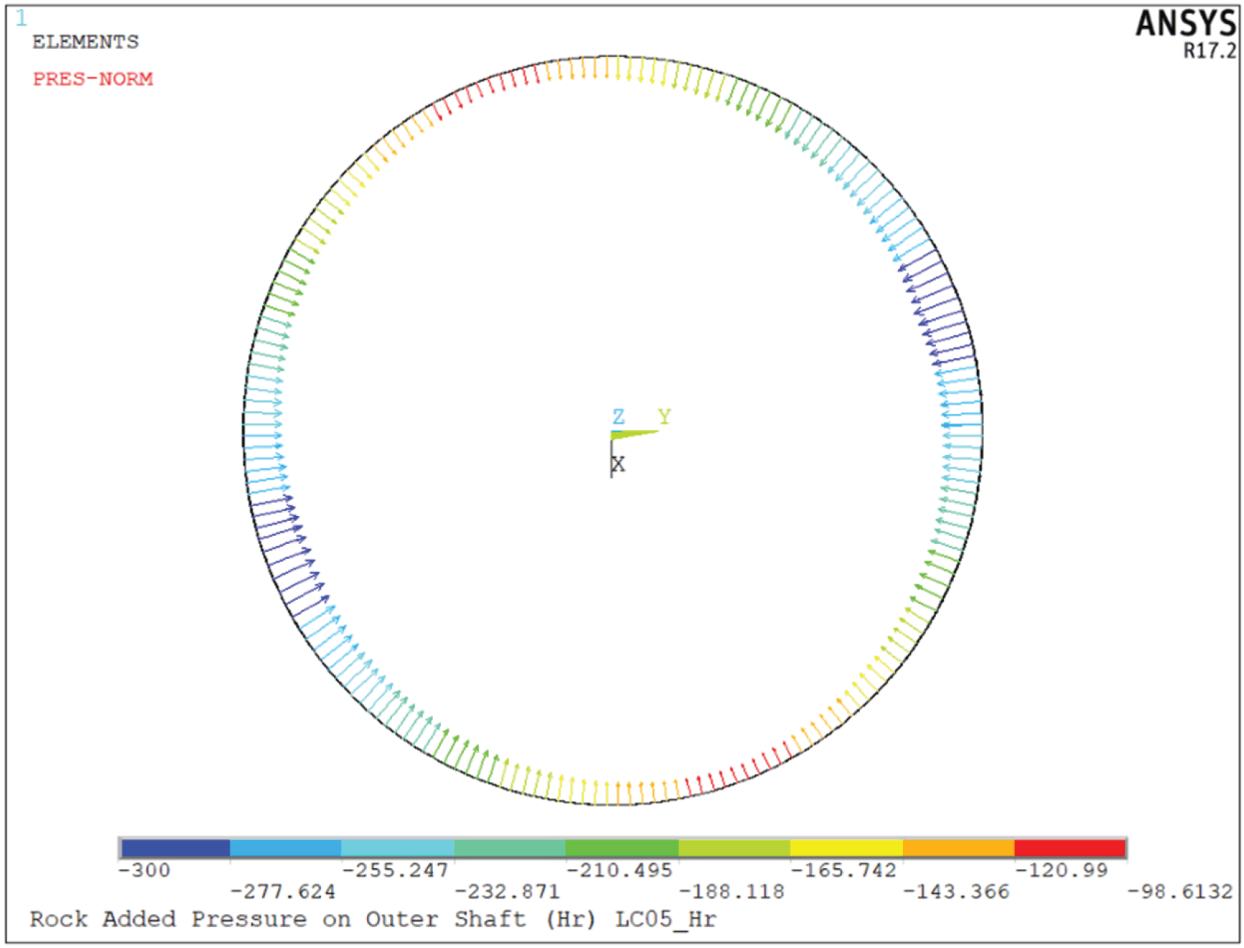
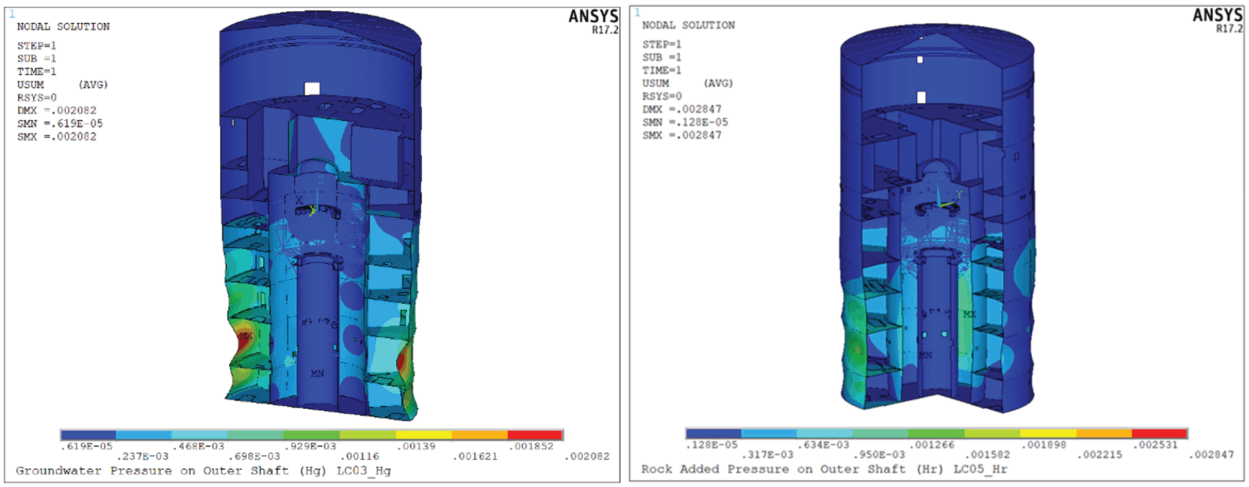
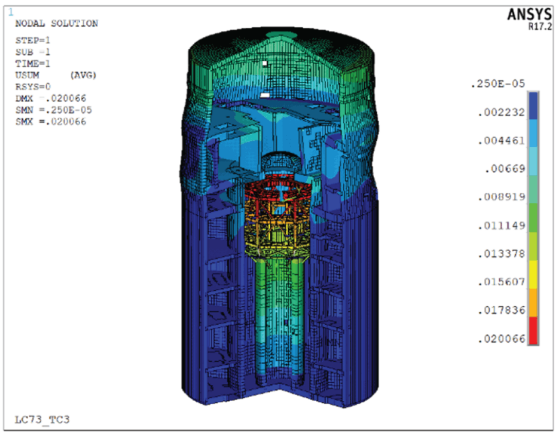


Figure 3E-6 Rock Pressure on Integrated Reactor Building Exterior Wall (Units: kPa)

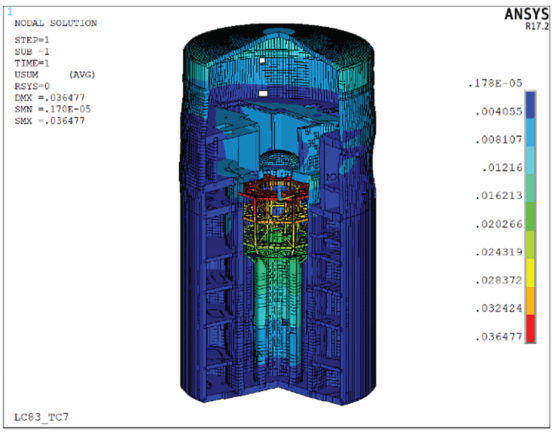


(a) Due to Groundwater (b) Due to Rock Pressure

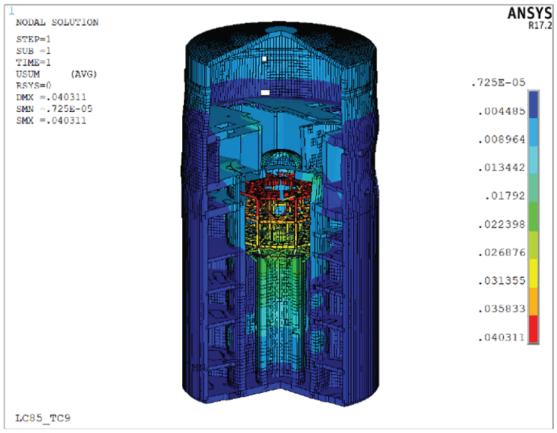
Figure 3E-7 Deformed Shape of Integrated Reactor Building Due to Groundwater and Rock Pressures (Units: m)



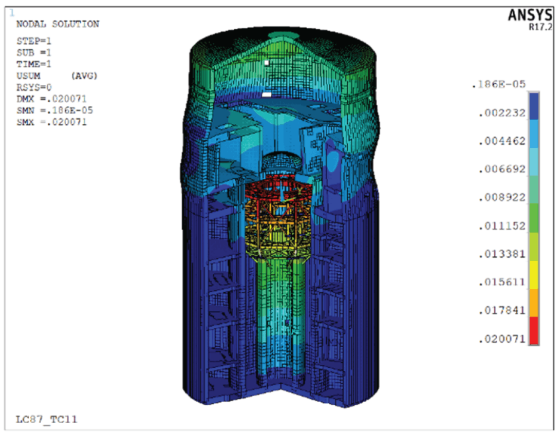
(a) Normal Operating



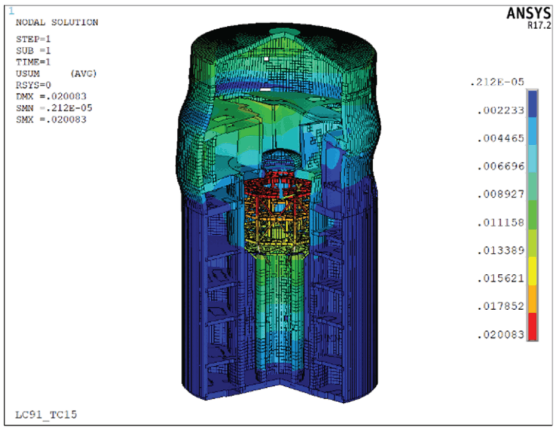
(b) DBA-1 10 hours



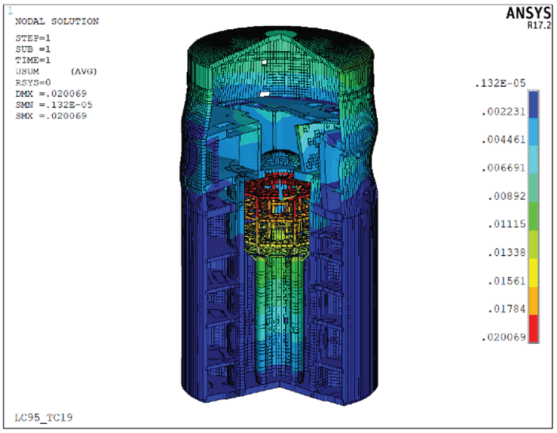
(c) DBA-1, 72 hours



(d) DBA-2, 10 hours

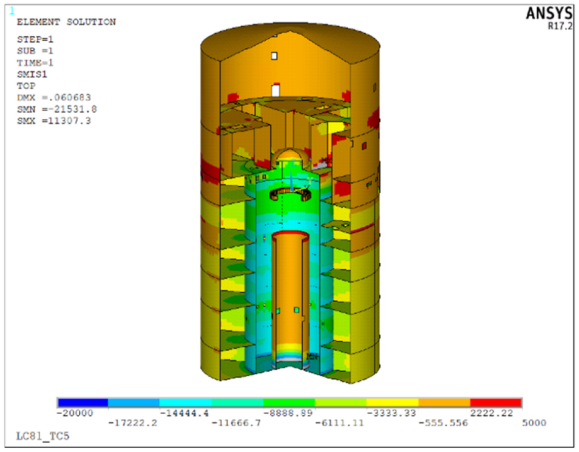


(e) DBA-3, 10 hours

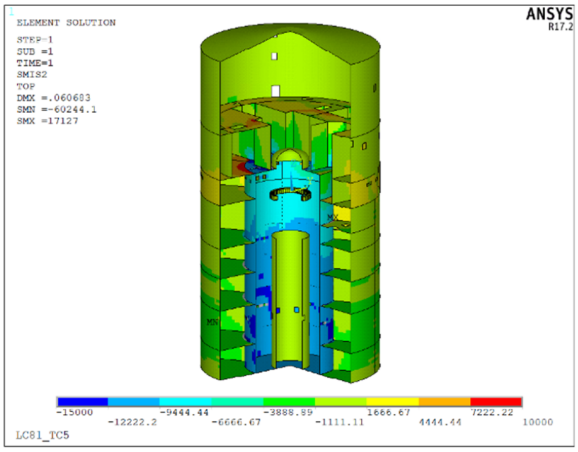


(f) DBA-4, 10 hours

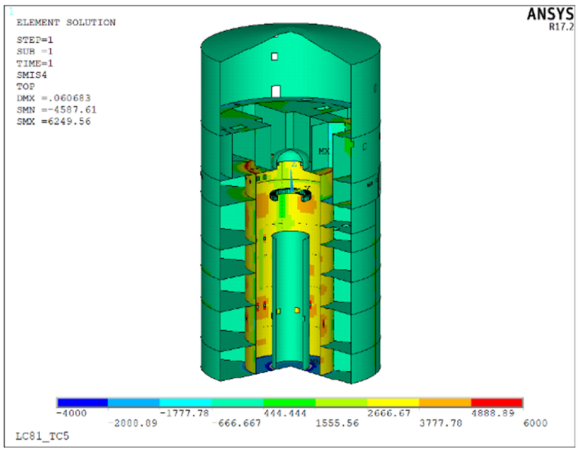
Figure 3E-8 Deformed Shape of Integrated Reactor Building Due to Normal Operating Thermal Loads During Winter with Upper Bound Subgrade (Units: m)



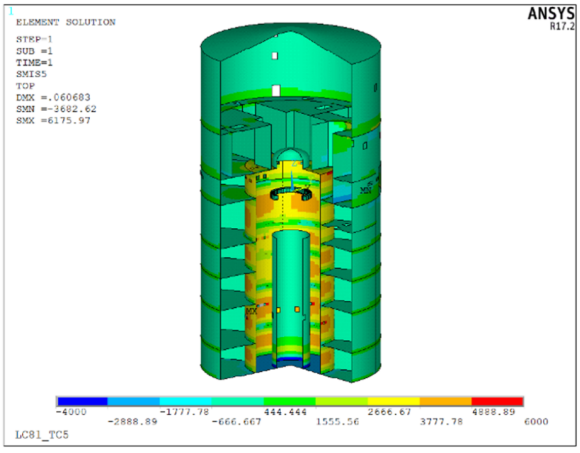
Hoop Membrane Force (Sx) (kN/m)



Vertical Membrane Force (Sy) (kN/m)



Bending Moment about Vertical Axis (Mx) (kN·m/m)



Bending Moment about Horizontal Axis (My) (kN m/m)

Figure 3E-9 Reactor Building Internal Forces Due to a Design Basis Accident Inside Containment during Summer

APPENDIX 3F DESIGN DETAILS AND EVALUATION RESULTS FOR THE CONTAINMENT

This appendix focuses on the design evaluation of the Steel-Plate Composite Containment Vessel (SCCV) portion of the containment and provides estimates of the available margins for the structure.

Design demands used in the evaluations are obtained from the structural analyses discussed in Appendix 3C and Appendix 3E performed using the one-step approach and the Finite Element (FE) models discussed in Appendix 3B. The design demands are obtained from design load combinations that assess the operational and accident structural demands on the structure.

As discussed in Section 3.8.2, the Class MC components of the BWRX-300 containment consist of the containment closure head, the two containment airlocks and the containment penetrations. The major Class MC components are represented in the integrated Reactor Building (RB) finite element model described in Appendix 3B to ensure accurate loading values can be derived from the structural analyses summarized in the Chapter 3 appendices. The preliminary loads on MC components have been reviewed and determined reasonable to continue with the current design. The design and analysis of Class MC components will be performed per the methodology described in Section 3.8.2, using the demands from the finite element models, by the fabricator during the procurement process. A summary of the results and margins for those components will be provided during future licensing activities.

3F.1 Design Methodology

The design of the SCCV structure is in accordance with Section 6.0 of NEDC-33926P-A, “BWRX-300 Steel-Plate Composite (SC) Containment Vessel and Reactor Building Structural Design,” (Reference 3F-1).

Table 3F-1 presents the design parameters used in the DP-SC design evaluation, which are in compliance with Limitation and Condition (L&C) 8.3 and L&C 8.4 of NEDC-33926P-A (Reference 3F-1).

3F.2 Load Combinations and Design Demands

The evaluation of the SCCV is based on analysis results using load combinations as presented in Table 3.8-1.

Additional detail of the individual load demands considered is described in Subsection 3.8.1.3. Discussion on analytical methods is captured in Subsection 3.8.1.4.

For evaluation purposes, design demands are conservatively presented on element-by-element basis without averaging of the results across adjacent elements as allowed by the NEDC-33926P-A.

3F.3 Structural Capacity of SCCV Diaphragm Plate Steel-Plate Composite Sections

Design capacities for SCCV DP-SC components are determined using the material properties and geometric configuration of each section and are computed using the design rules for BWRX-300 containment DP-SC structures described in Section 6.0 of NEDC-33926P-A.

These results provide capacities for out-of-plane shear strength and the combined capacities considering out-of-plane shear forces interaction, as well as the stress capacities for steel plates and concrete infill.

The SCCV final panel section design will address how each required strength or demands, namely, axial tension, axial compression, out-of-plane flexure, and in-plane (tangential) shear compares to the corresponding allowable section strength (capacity) consistent with the acceptance criteria for all force action types specified in Table 6-1(a) and Table 6-1(b) of NEDC-33926P-A for factored and service load combinations to comply with L&C 8.17 of NEDC-33926P-A.

As stated in Subsection 3.8.4.1.4, the integrated RB DP-SC faceplates are designed to accommodate the residual stress distribution due to rolling of materials during fabrication of curved sections without compromising their membrane stress capacity in both tensions and compression. An evaluation was performed in compliance with L&C 8.11 of NEDC-33926P-A to quantify the effects of residual stresses due to rolling on the curved plates within the integrated RB structures. The evaluation determined that residual stresses and strains resulting from rolling of the curved plates are minimal and do not need to be accounted for further in the design of the BWRX-300 DP-SC curved walls.

These capacities are used to calculate the demand-to-capacity ratios described in Section 3F.4.

3F.4 Structural Design Evaluation

The evaluation of the SCCV structure is performed by calculating and plotting demand-to-capacity ratios for the individual loads and interaction checks discussed in Section 3F.3 for the SCCV wall, top slab, and inner mat foundation. The demand-to-capacity ratios identify critical locations and vulnerabilities for specific limit states.

Demand-to-capacity ratios are calculated for each load combination, and envelope ratios are calculated as the maximum element demand-to-capacity ratio over all the load combinations. The enveloping ratios are used to produce an enveloped contour plot for each design criteria.

Figure 3F-1 through Figure 3F-12 present the demand-to-capacity ratios for the SCCV wall, top slab, and inner mat foundation for the selected load combinations.

These figures show the side of the SCCV wall where the main steam line penetrations are located near the top of the wall. Each set of plots show the demand-to-capacity ratios for the critical criteria for the SCCV including the principal compressive stress, faceplate Von Mises yield stress, out-of-plane shear, and out-of-plane shear interaction on the diaphragm plates.

The SCCV principal compressive stress demand-to-capacity ratios are generally less than 0.5 and the faceplate stresses are above 0.75 near the top of the SCCV with some localized exceedances around the main steam line penetrations. The out-of-plane shear plots show locally high demand-to-capacity ratios where the RB slabs and wing walls intersect the SCCV wall as is expected. Design exceedances exist in these areas, however, some of the elements with a design exceedance are within the thickness of the intersecting component or are outside the critical shear section. Furthermore, the finite element mesh size is approximately half the SCCV wall thickness which would allow element averaging across four elements for future evaluation.

Similar observations can be made about the SCCV top slab and inner mat foundation which shows shear-critical behavior. As noted for the SCCV wall, many of these high shear regions in the slabs lie within the intersecting walls and outside the critical shear regions.

A summary of additional evaluations performed, or design updates developed to address these exceedances will be included in the FSAR.

3F.5 Critical SCCV Connections Design

The integrated RB DP-SC modules splices and floor-to-wall, wall-to-wall and wall-to-mat foundation connections are designed per Sections 5.11 and 6.14 NEDC-33926P-A and meet the requirements of Section N9.4 of ANSI/AISC N690-18, “Specification for Safety-Related Steel Structures for Nuclear Facilities,” (Reference 3F-2), ANSI/AISC 360-16, “Specification for Structural Steel Buildings” (Reference 3F-3) and AISC Design Guide 32, “Design of Modular Steel-Plate Composite Walls for Safety Related Nuclear Facilities,” (Reference 3F-4).

Two types of connections are used to connect the various integrated RB steel-plate composite components as allowed by Appendix N9.4.2 of ANSI/AISC N690-18:

1. Full-strength connections, where the required strength for the connections is 125% of the smaller of the corresponding nominal strengths of the connected parts computed per Section 5.7 of NEDC-33926P-A, and

2. Overstrength connections, where the connection required strength is 200% of the required strength due to seismic loads plus 100% of the required strength due to non-seismic loads (including thermal loads).

Full-strength connections ensure a ductile behavior with yielding and inelasticity occurring away from the connection in one of the connected members. Overstrength connections are used where it is not feasible to provide a full-strength connection, such as when the sizing of the structural members is governed by non-structural requirements (e.g., shielding requirements) and the members are oversized. Demands used to design the overstrength connections are extracted from the FE model at the connection location for the applicable load combinations considered in the design and by doubling the seismic demands as required by Section N9.4.2 of ANSI/AISC N690-18.

In accordance with Section N9.4.3 of ANSI/AISC N690-18, the connections strength is calculated using the applicable force transfer mechanism and the available strength of the connectors contributing to the force transfer.

For conventional steel-plate composite module-to-module joint connections, joint shear strength is conservatively calculated per Section 21.7.4.1(c) of American Concrete Institute (ACI) 349-13 "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary" (Reference 3F-5). For DP-SC module-to-module connections where the faceplates are continuous, or where continuity is provided through stiffener plates, such that the confinement action of the section, surrounded by the faceplates, represents a closed-filled-tube at intersections, the available joint shear capacity at intersection is calculated using Equation 5-49 of NEDC-33926P-A, Section 5.11, without increasing the contribution from the concrete.

The SCCV wall is connected to the inner and outer mat foundations using a rigid full-strength connection, in which the SCCV wall is the continuous member, and the inner and outer mat foundation modules are connected to the SCCV wall. This connection, referred to as a full-strength rigid T-connection, has been tested under cyclic loading with ambient and elevated temperature conditions mimicking the DBA thermal load and was found to exceed the strength and ductility acceptance criteria. See Figure 3F-13 for an illustration of these connections.

The wing walls connection to SCCV wall and floor connections to SCCV wall are identical to those connecting the wing walls and floors to the RB exterior wall discussed in Section 3H.5.

3F.6 References

3F-1 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.

3F-2 ANSI/AISC N690-18, "Specification for Safety-Related Steel Structures for Nuclear Facilities," American National Standards Institute/American Institute of Steel Construction, 2018.

3F-3 ANSI/AISC 360-16, "Specification for Structural Steel Buildings," American National Standards Institute/American Institute of Steel Construction, 2016.

3F-4 AISC Steel Design Guide 32, "Design of Modular Steel-Plate Composite Walls for Safety Related Nuclear Facilities," American Institute of Steel Construction, 2017.

3F-5 ACI 349-13, "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary," American Concrete Institute, 2014.

Table 3F-1 Design Parameters for the SCCV Diaphragm Plate Steel-Plate Composite Modules

Structural Component	Member Thickness (in)	Faceplate/ Diaphragm Plate Thickness (in)	Yield Strength (ksi)	Concrete Compressive Strength (ksi)	Diaphragm Plate Spacing (in)
SCCV wall	48	0.75	60(1)	8	24
Inner mat foundation	48	0.5		8	24
SCCV top slab	60	0.75		8	30

Note:

(1) For load combinations with thermal loading, temperature-dependent material properties are used for containment.

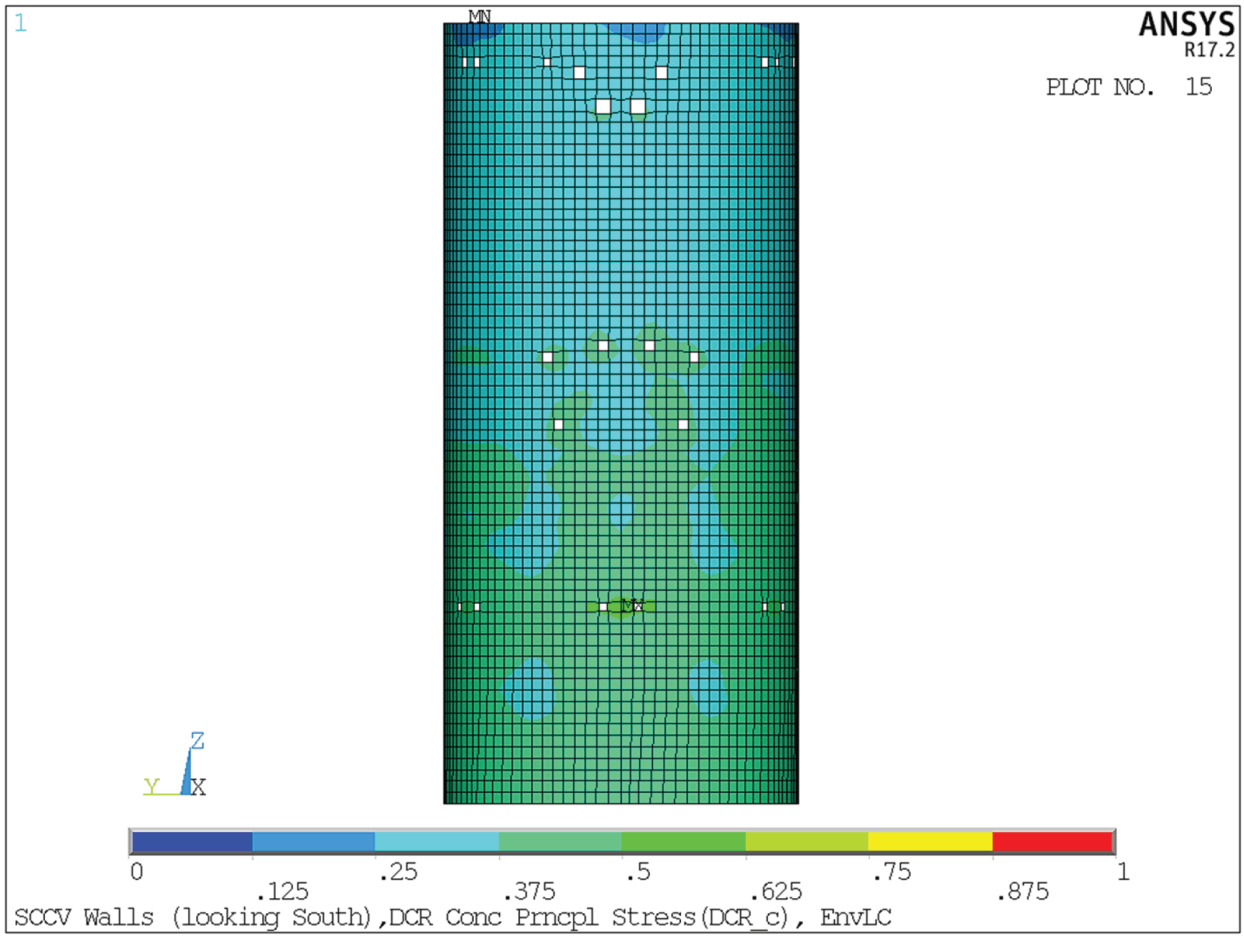


Figure 3F-1 Containment Wall Envelope Demand-to-Capacity Ratios - Principal Compression

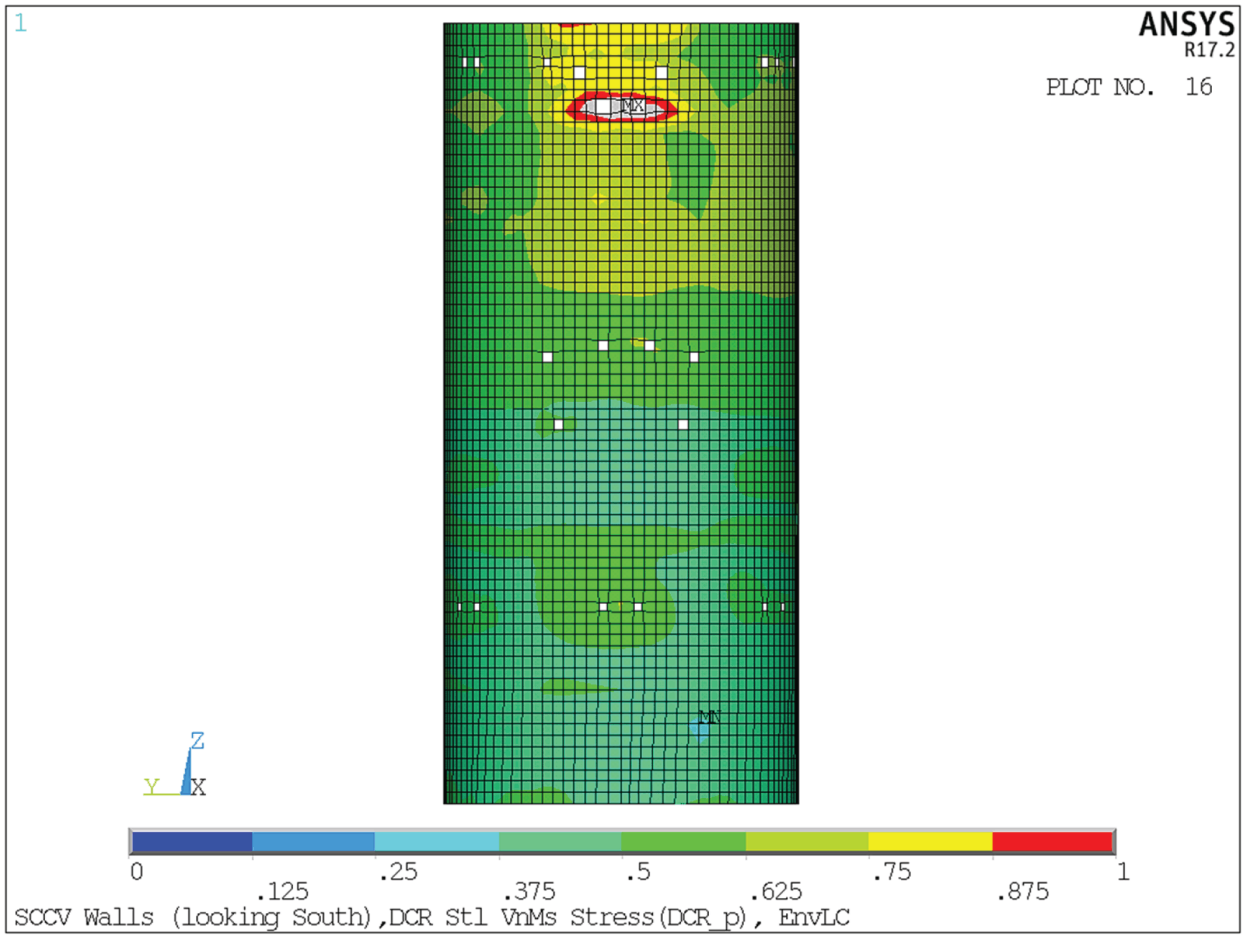


Figure 3F-2 Containment Wall Envelope Demand-to-Capacity Ratios - Von Mises Stress

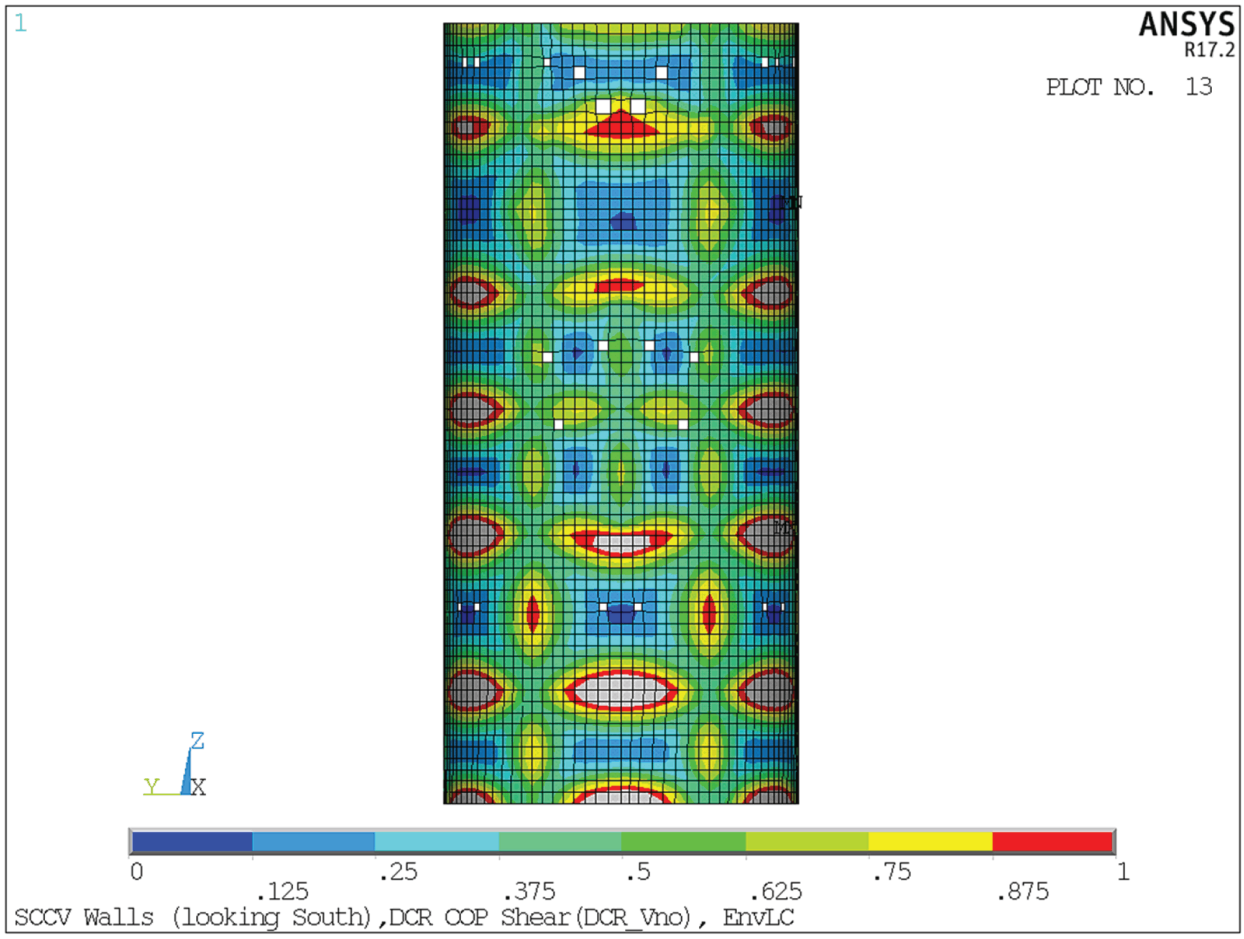


Figure 3F-3 Containment Wall Envelope Demand-to-Capacity Ratios - Out-of-Plane Shear

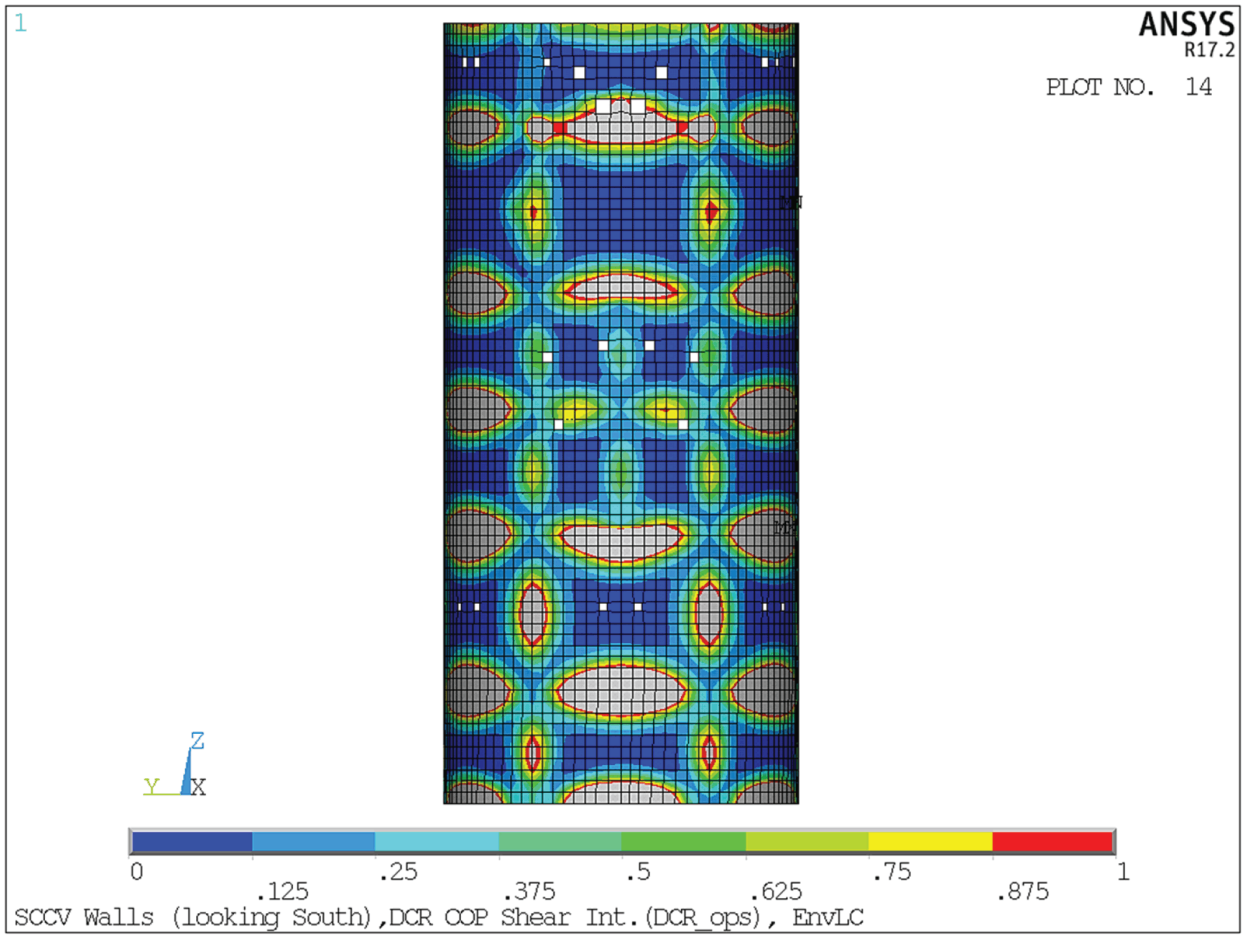


Figure 3F-4 Containment Wall Envelope Demand-to-Capacity Ratios - Out-of-Plane Shear Interaction

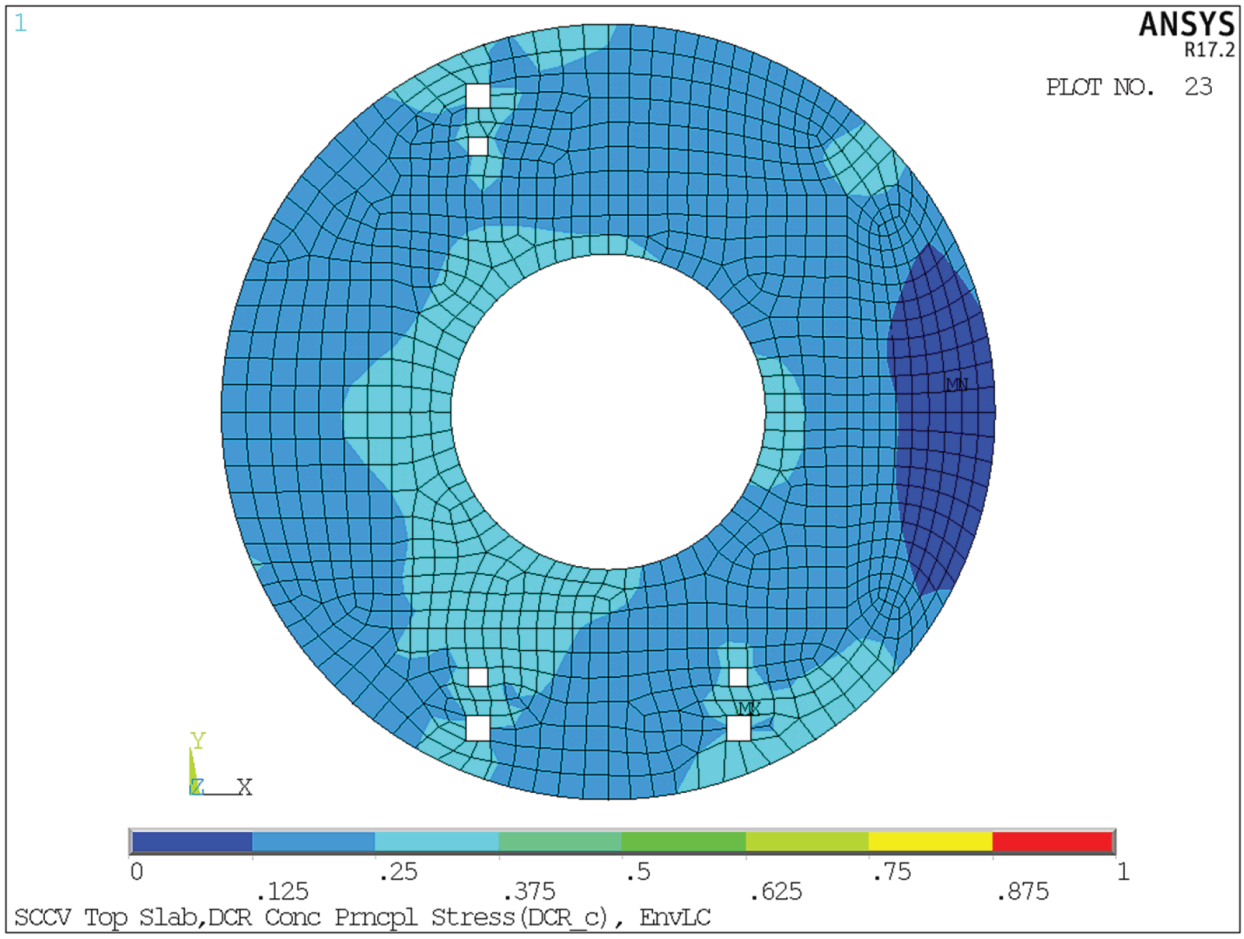


Figure 3F-5 Containment Top Slab Envelope Demand-to-Capacity Ratios - Principal Compression

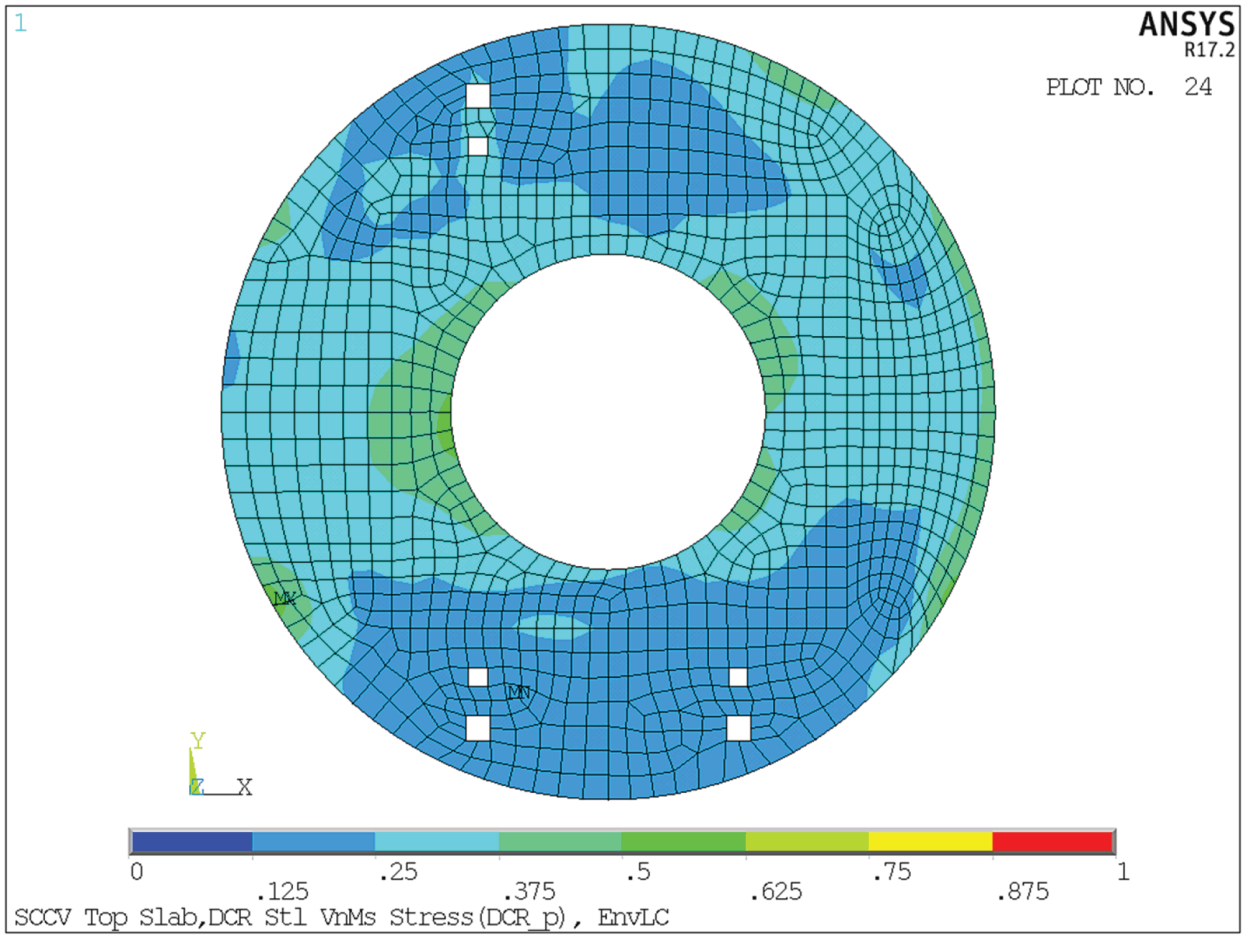


Figure 3F-6 Containment Top Slab Envelope Demand-to-Capacity Ratios - Von Mises Stress

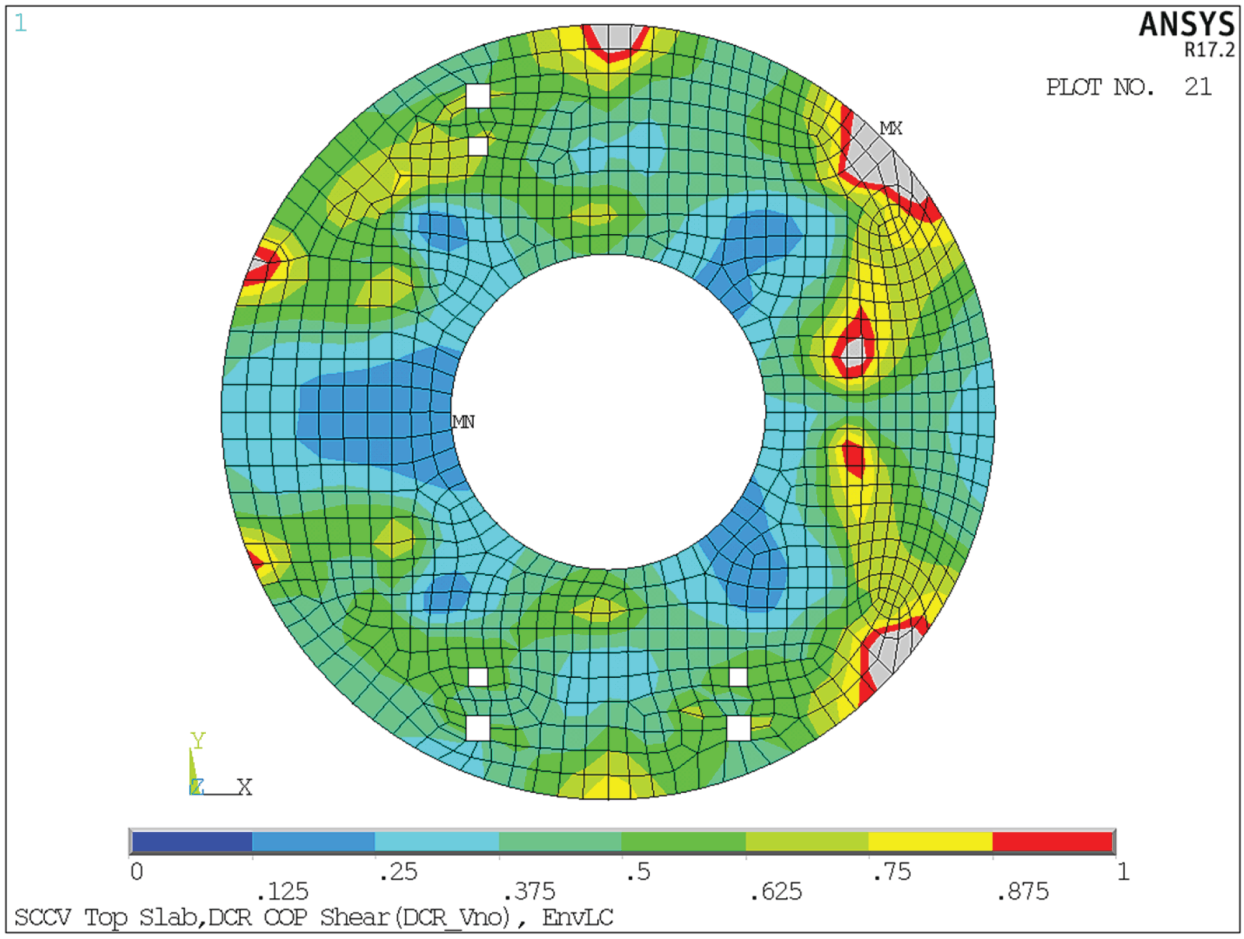


Figure 3F-7 Containment Top Slab Envelope Demand-to-Capacity Ratios - Out-of-Plane Shear

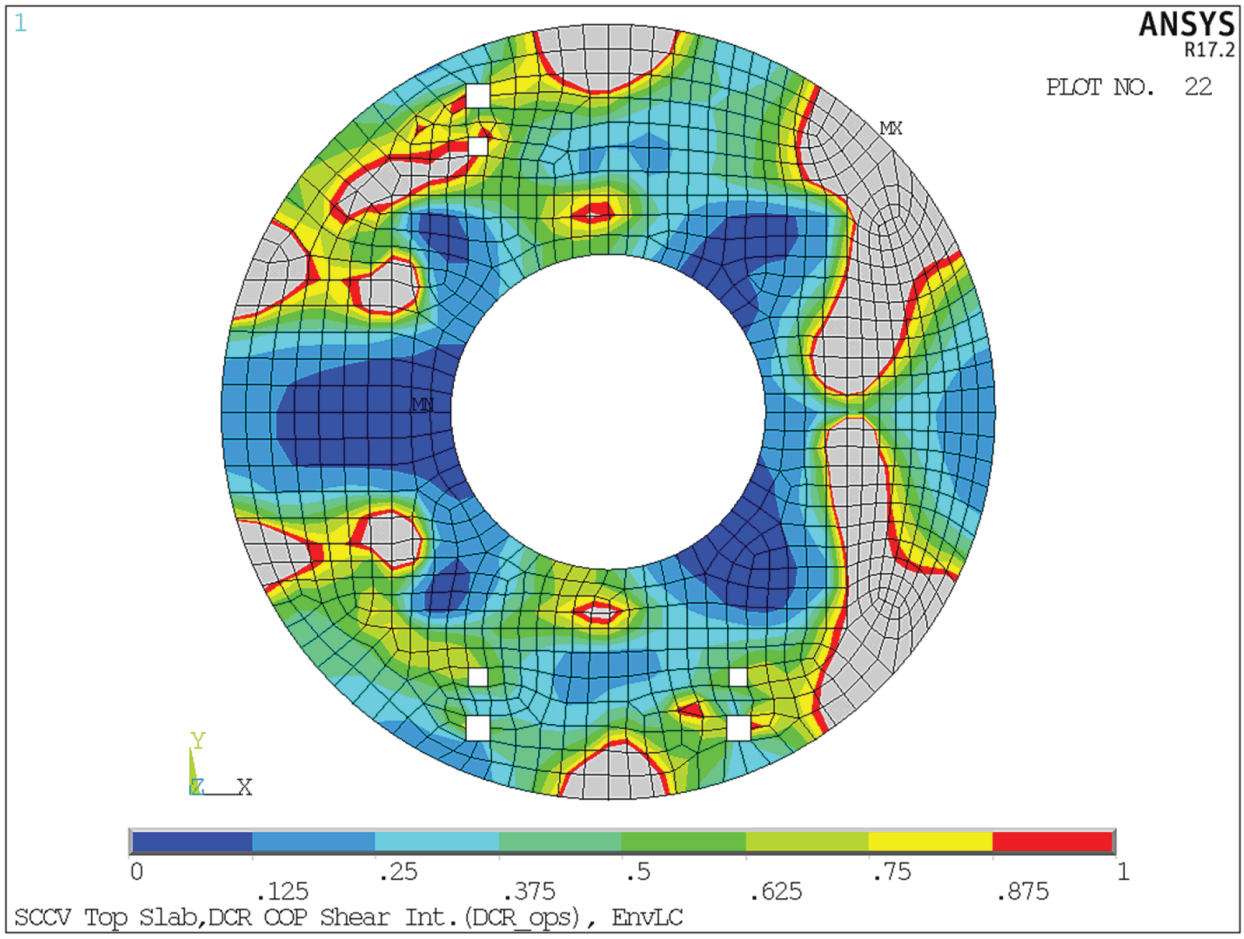


Figure 3F-8 Containment Top Slab Envelope Demand-to-Capacity Ratios - Out-of-Plane Shear Interaction

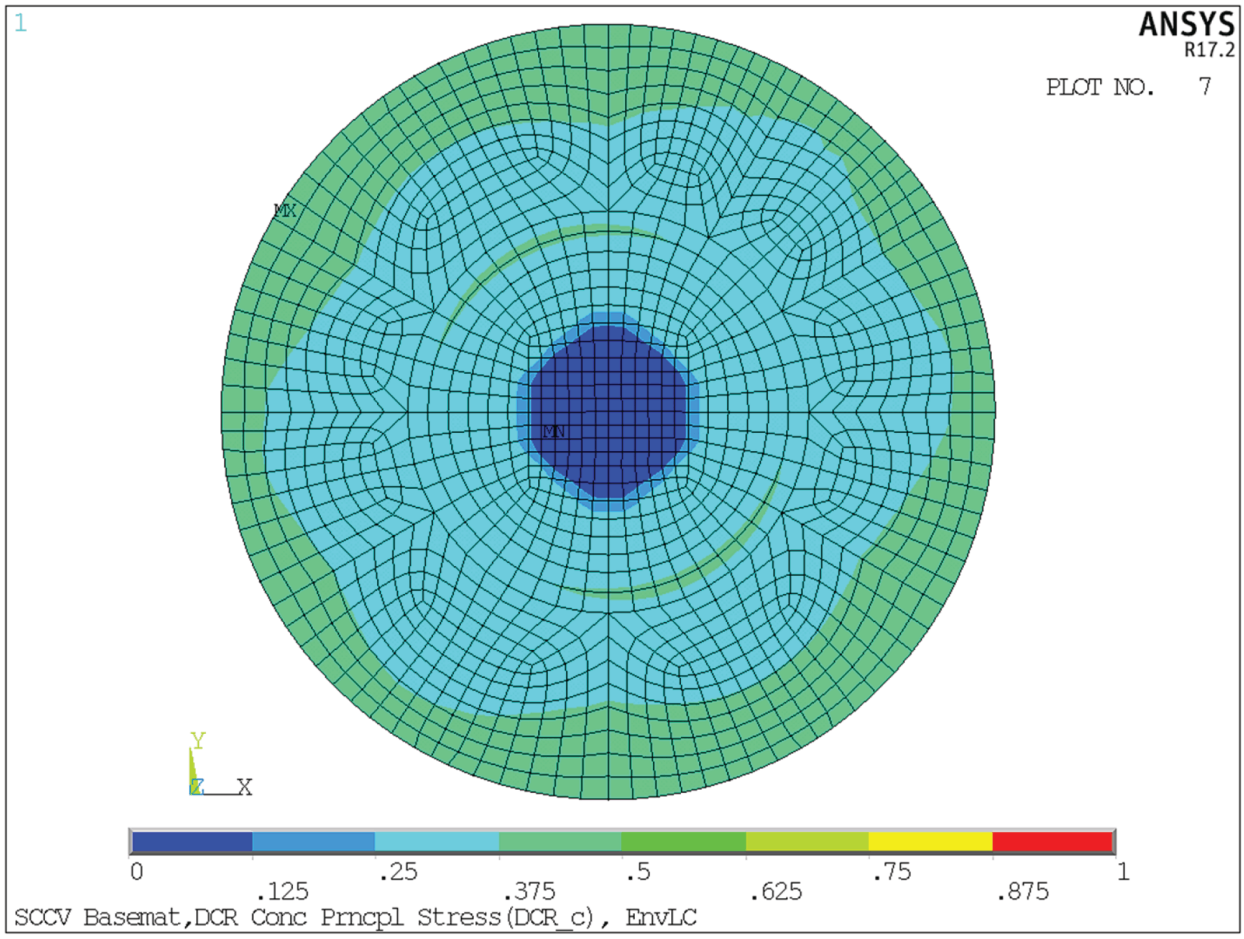


Figure 3F-9 Containment Basemat (Inner Mat Foundation) Envelope Demand-to-Capacity Ratios - Principal Compression

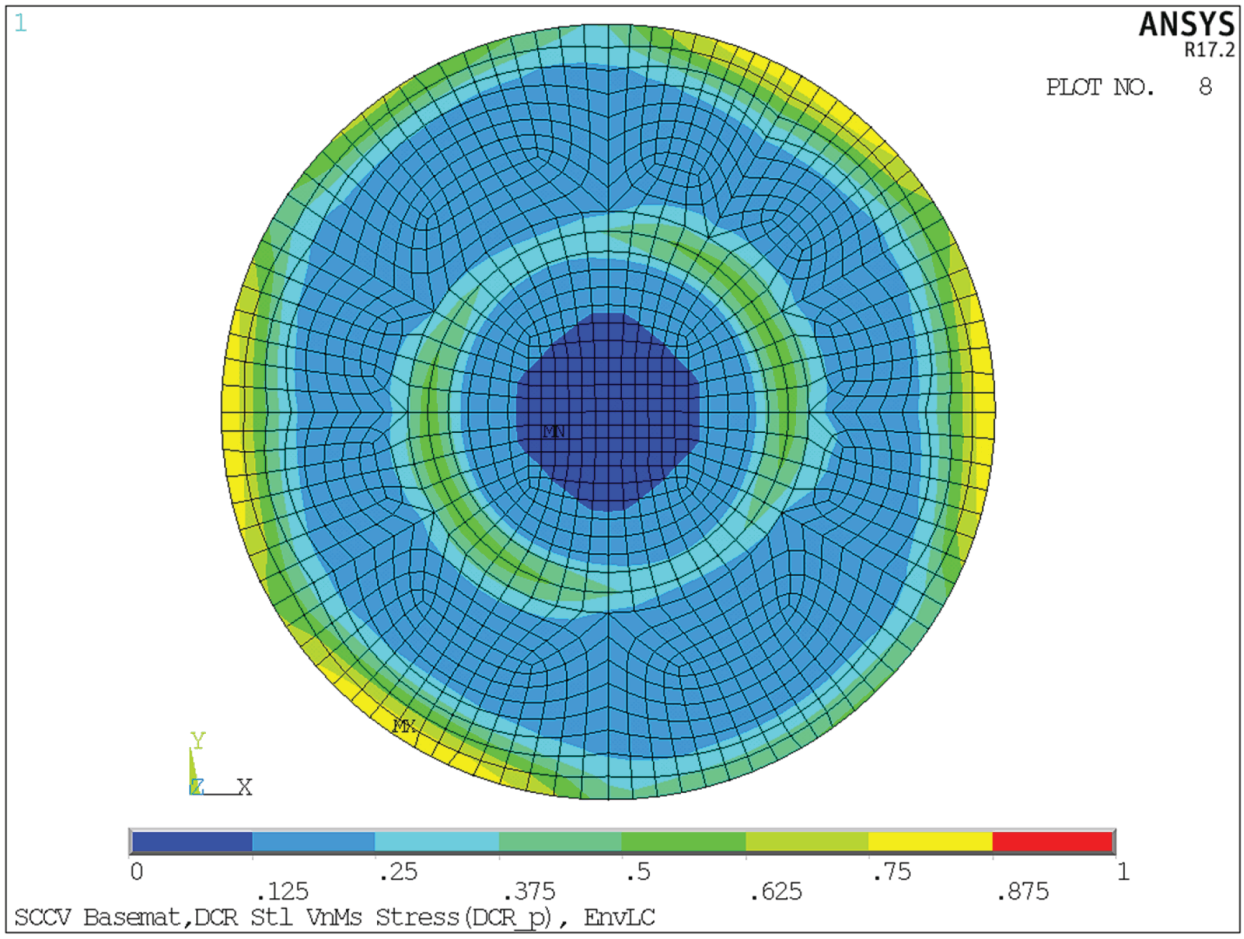


Figure 3F-10 Containment Basemat (Inner Mat Foundation) Envelope Demand-to-Capacity Ratios - Von Mises Stress

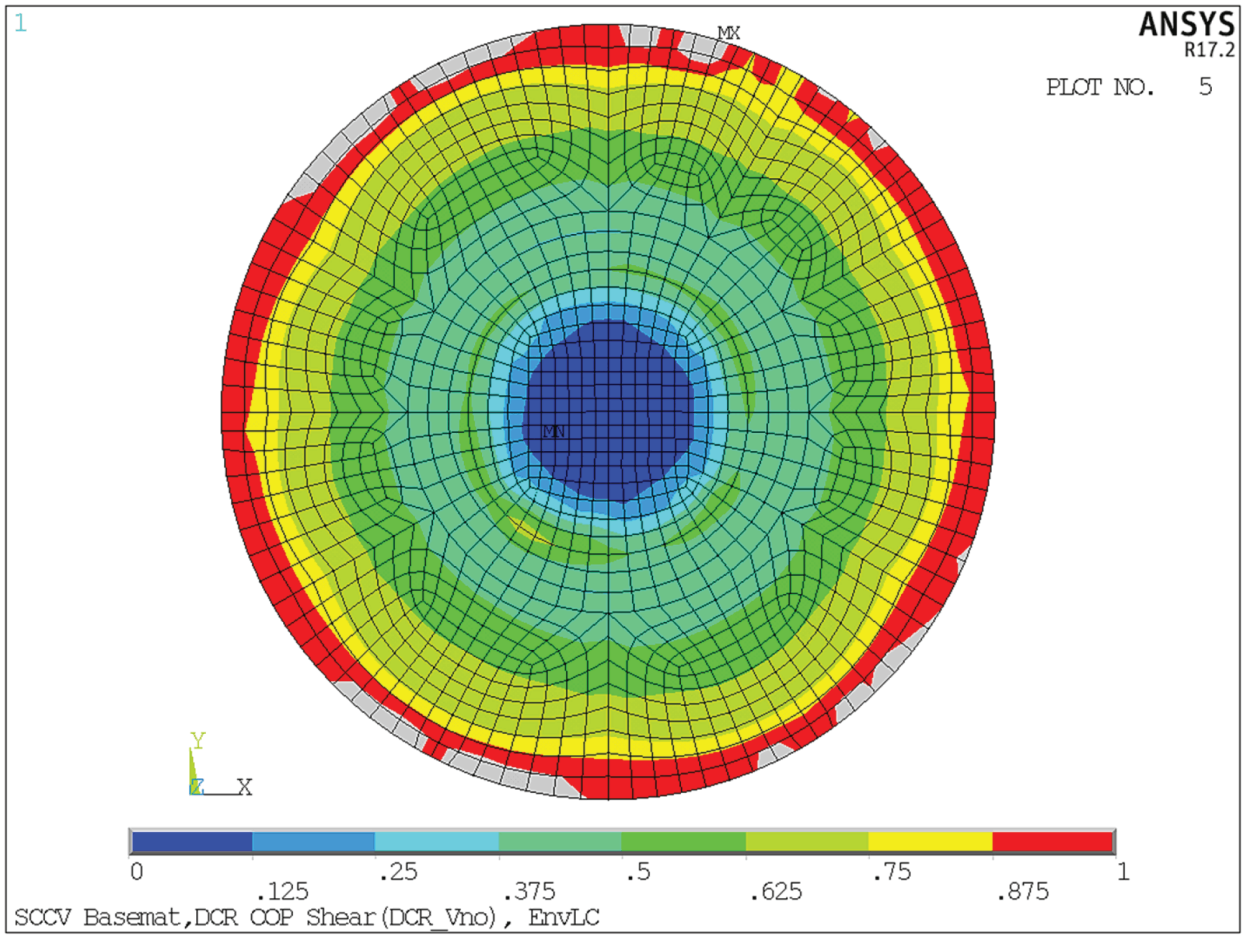


Figure 3F-11 Containment Basemat (Inner Mat Foundation) Envelope Demand-to-Capacity Ratios - Out of Plane Shear

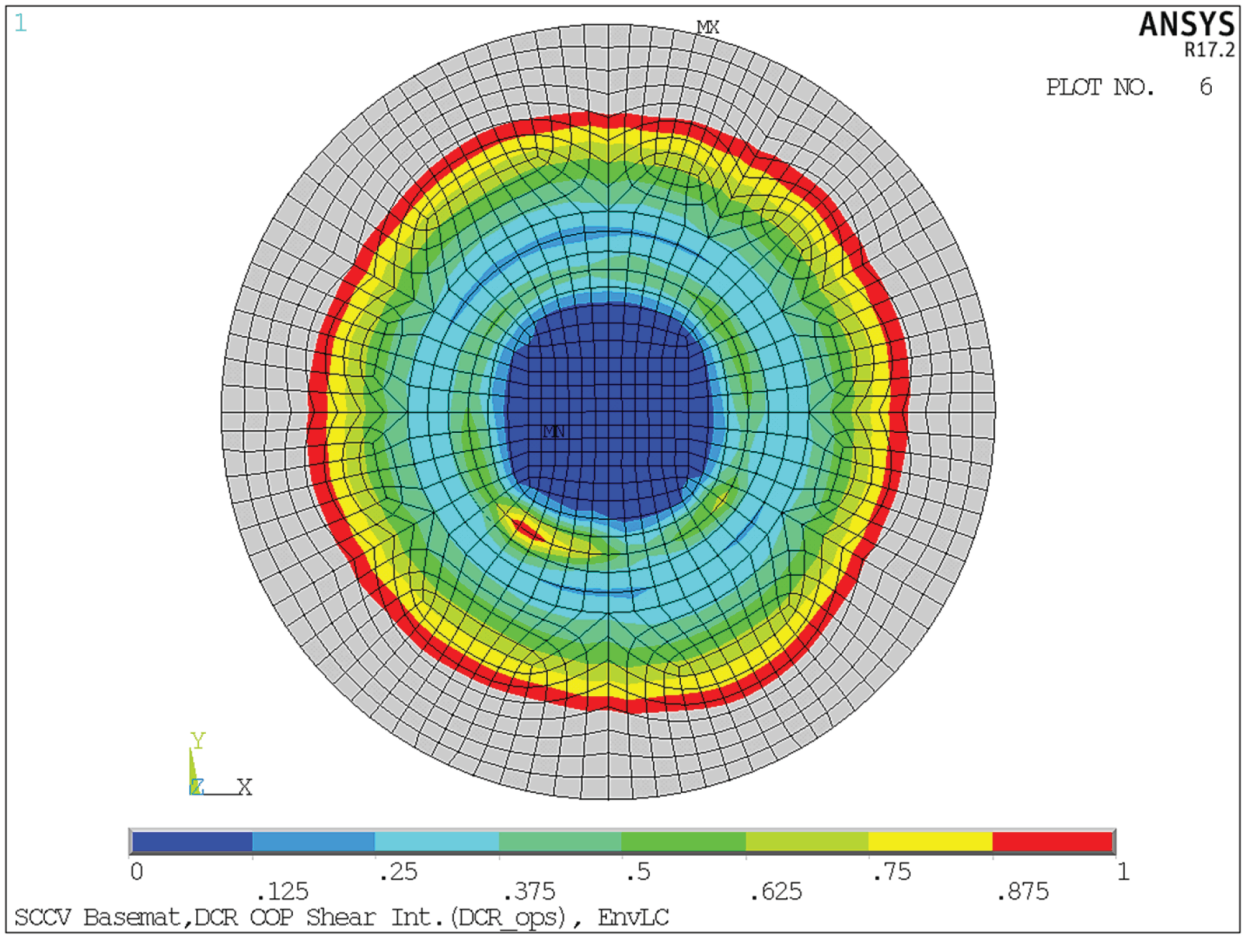


Figure 3F-12 Containment Basemat (Inner Mat Foundation) Envelope Demand-to-Capacity Ratios - Out of Plane Shear Interaction

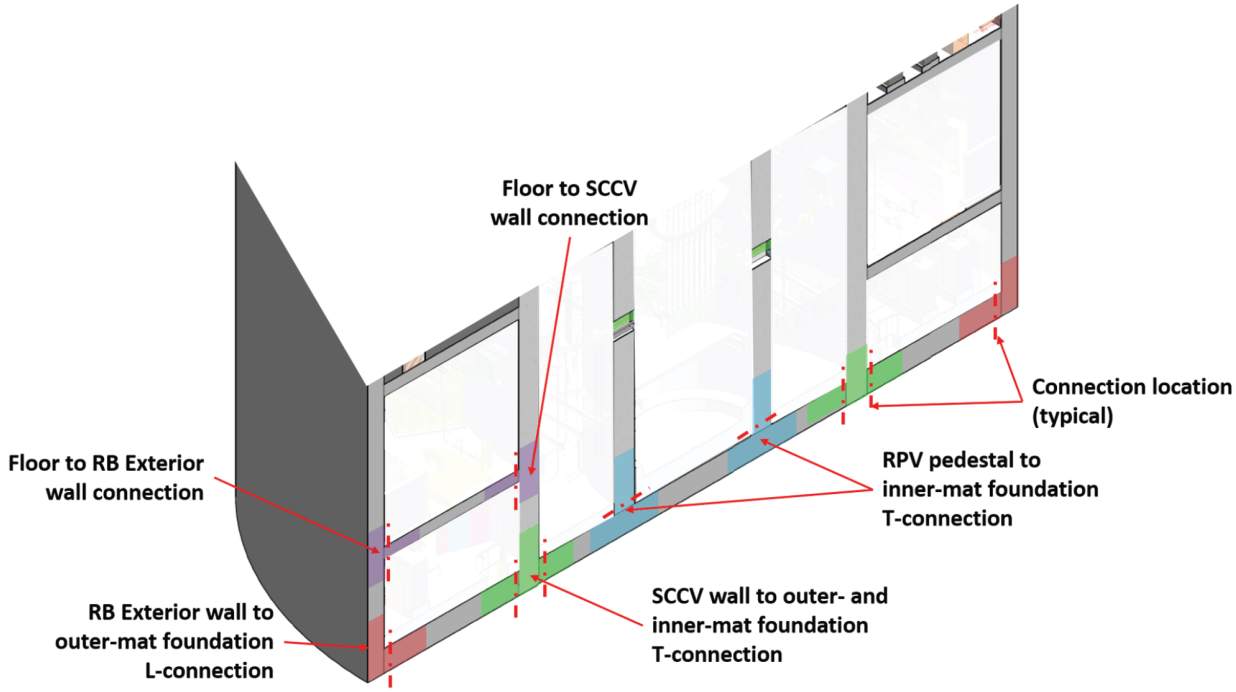


Figure 3F-13 Floor to Wall Connection Location Examples

APPENDIX 3G DESIGN DETAILS AND EVALUATION RESULTS FOR THE CONTAINMENT INTERNAL STRUCTURES

This appendix focuses on the design evaluation of the Diaphragm Plate Steel-Plate Composite (DP-SC) Reactor Pressure Vessel (RPV) pedestal and provides estimates of the available margins for the structure. Design evaluations for other containment internal structures are not provided for preliminary evaluation as these structures do not contribute to the global response of the integrated Reactor Building (RB) and their design is still progressing.

Design demands used in the evaluations are obtained from the structural analyses discussed in Appendix 3C and Appendix 3E, performed using the one-step approach and the Finite Element (FE) models discussed in Appendix 3B. The design demands are obtained from design load combinations that assess the operational and accident structural demands on the structure.

3G.1 Design Methodology

The design of the RPV pedestal is in accordance with Section 5.0 of NEDC-33926P-A, “BWRX-300 Steel-Plate Composite (SC) Containment Vessel and Reactor Building Structural Design,” (Reference 3G-1), similar to the Reactor RB structure.

Table 3G-1 presents the design parameters used in the DP-SC design evaluation, which are in compliance with Limitation and Condition (L&C) 8.3 and L&C 8.4 of NEDC-33926P-A (Reference 3G-1).

3G.2 Load Combinations and Design Demands

The evaluation of the RPV pedestal is based on design demands obtained from the load combinations in Table 3.8-7.

Additional detail of the individual load demands considered is described in Subsection 3.8.3.3. Discussion on analytical methods is discussed in Subsection 3.8.3.4.

3G.3 Structural Capacity of Diaphragm Plate Steel-Plate Composite Sections

Design capacities for the RPV pedestal DP-SC components are determined using the material properties and geometric configuration of each section and the design rules for BWRX-300 non-containment DP-SC structures described in Section 5.0 of NEDC-33926P-A.

These results provide capacities for uniaxial tensile strength, compressive strength, out-of-plane flexural strength, in-plane shear strength, and out-of-plane shear strength, as well as the combined capacities considering out-of-plane shear forces interaction, and in-plane membrane forces and out-of-plane moments interaction.

These capacities are used to calculate the demand-to-capacity ratios described in Section 3G.4.

3G.4 Structural Design Evaluation

Demand-to-capacity ratios are calculated for each load combination, and envelope ratios are calculated as the maximum element demand-to-capacity ratio over all considered load combinations. The enveloping ratios are used to produce an enveloped contour plot for each design criterion.

Figure 3G-1 through Figure 3G-4 present the demand-to-capacity ratios for the design of the RPV pedestal for the load combinations in Table 3.8-7.

The demand-to-capacity ratios for the ANSI/AISC N690-18 (Reference 3G-2) criteria shown in Figure 3G-1 through Figure 3G-4 include the two notional half checks, out-of-plane shear, and out-of-plane shear interaction on the diaphragm plates.

Demand-to-capacity ratios for the notional half checks are approximately 0.75 near the base of the RPV pedestal. These checks include the in-plane membrane, in-plane shear, and out-of-plane bending effects. The out-of-plane shear demand-to-capacity ratios are observed to be high only at the top and bottom of the pedestal. The higher ratios at the top are due to the local interaction of the pedestal with the CEPSS structure while those at the bottom are due to the interaction with the Steel-Plate Composite Containment Vessel (SCCV) inner mat foundation. The exceedances observed at the inner mat foundation all lie within the thickness of the SCCV mat foundation.

A summary of additional evaluations performed, or design updates developed to address these exceedances will be included in the FSAR.

3G.5 Pedestal Connection Design

A full-strength rigid T-connection (see Section 3F.5 for design methodology and Figure 3F-13) is used to connect the RPV pedestal to the common mat foundation. In this connection, the inner mat foundation is the continuous member, with the RPV pedestal bearing on the mat foundation.

3G.6 References

3G-1 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.

3G-2 ANSI/AISC N690-18, “Specification for Safety-Related Steel Structures for Nuclear Facilities,” American National Standards Institute/American Institute of Steel Construction, 2018.

Table 3G-1 Design Parameters for the RPV Pedestal Diaphragm Plate Steel-Plate Composite Modules

Structural Component	Member Thickness (in)	Faceplate/ Diaphragm Plate Thickness (in)	Yield Strength (ksi)	Concrete Compressive Strength (ksi)	Diaphragm Plate Spacing (in)
RPV Pedestal	48	0.75	50(1)	8	24

Note:

(1) For load combinations with thermal loading, temperature-dependent material properties are used.

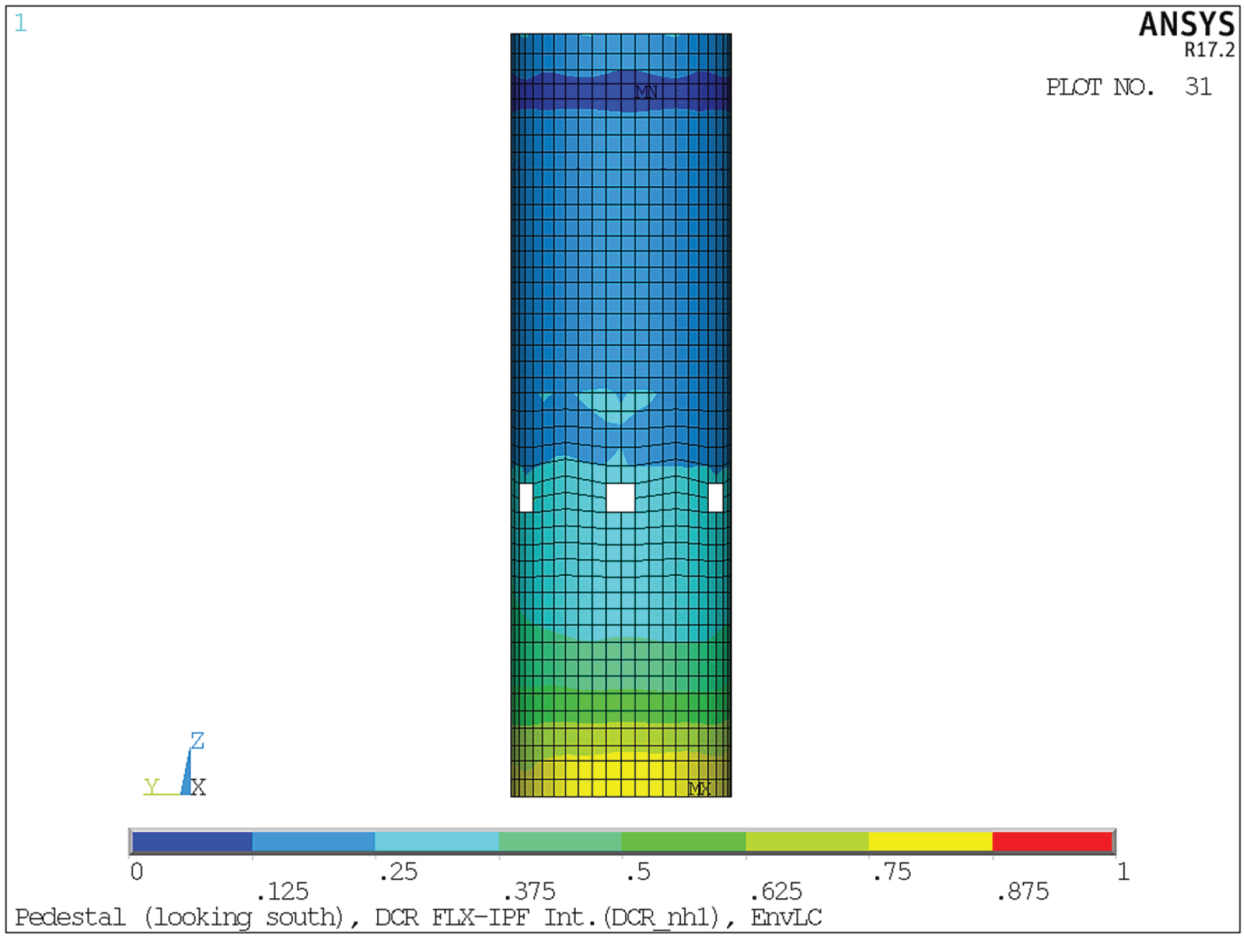


Figure 3G-1 Envelope Demand-to-Capacity Ratios for Reactor Pressure Vessel Pedestal - Notional Half 1

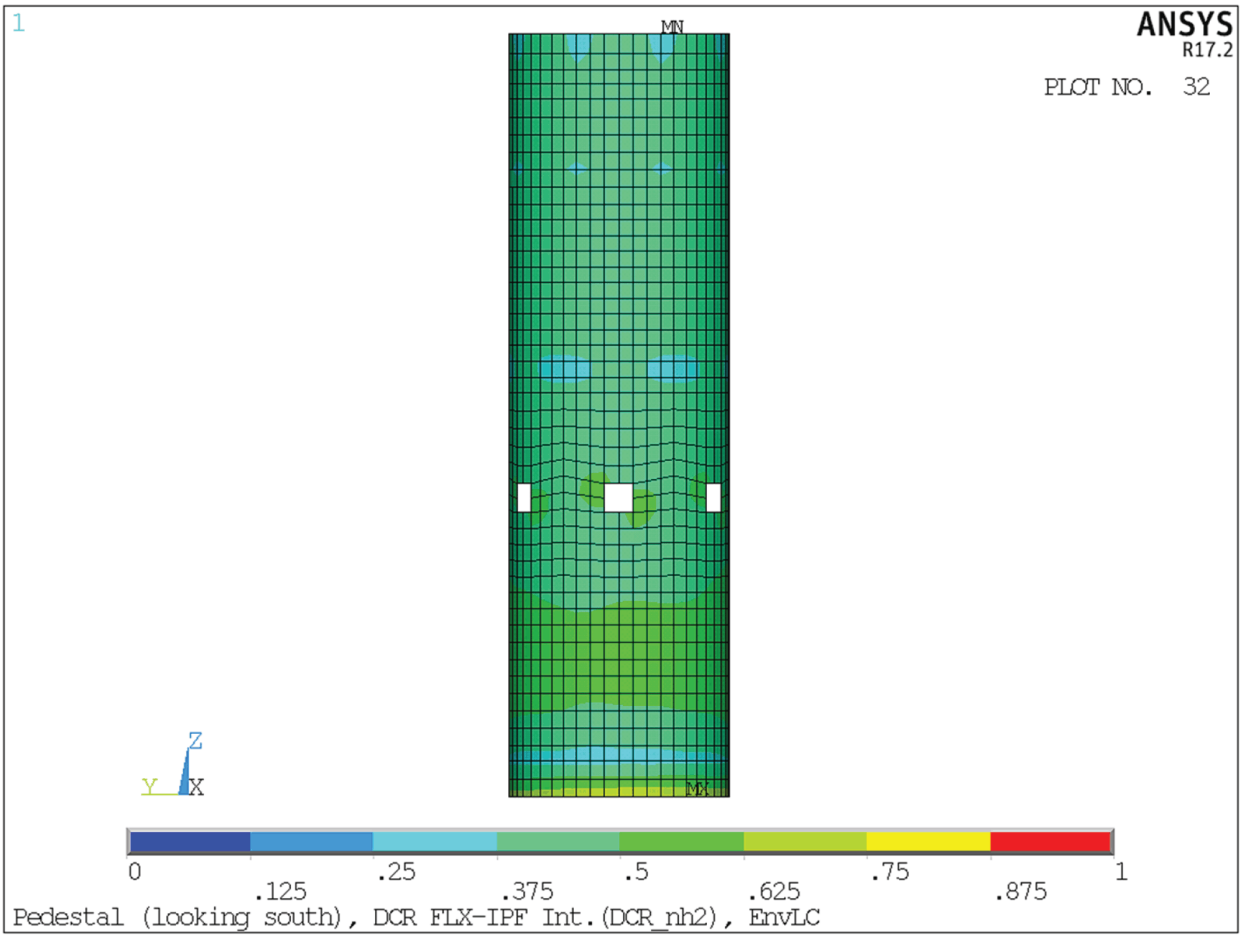


Figure 3G-2 Envelope Demand-to-Capacity Ratios for Reactor Pressure Vessel Pedestal - Notional Half 2

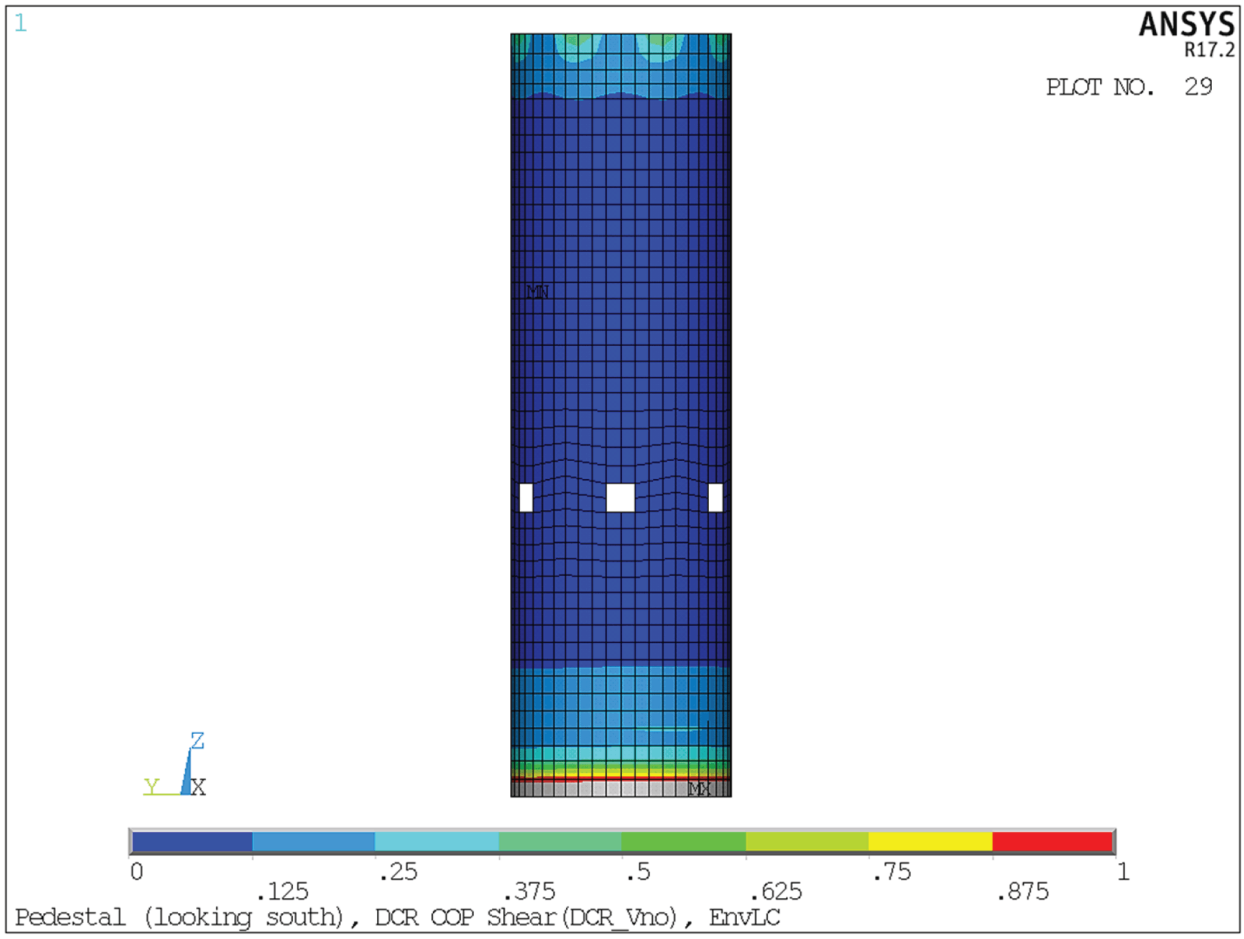


Figure 3G-3 Envelope Demand-to-Capacity Ratios for Reactor Pressure Vessel Pedestal - Out-of-Plane Shear

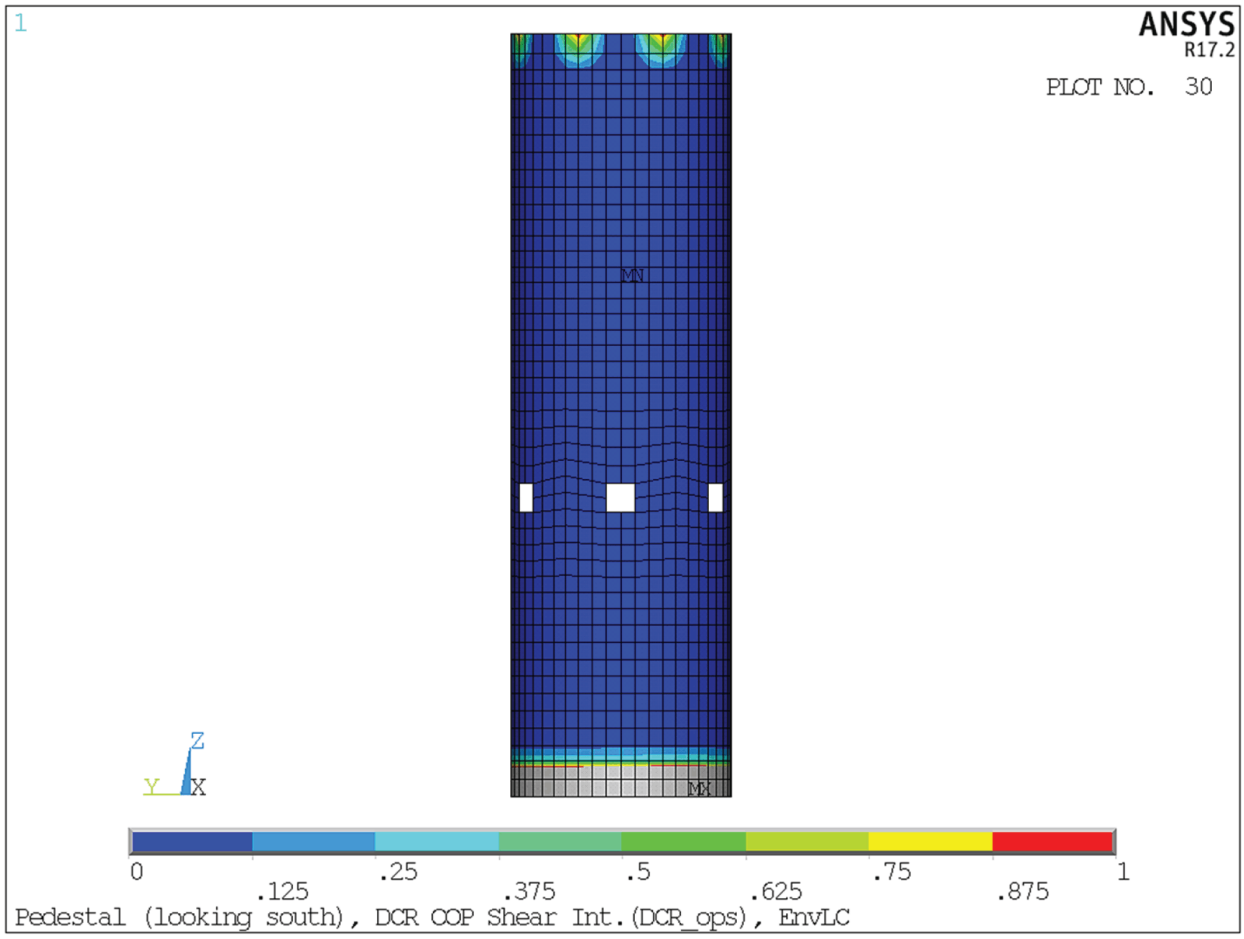


Figure 3G-4 Envelope Demand-to-Capacity Ratios for Reactor Pressure Vessel Pedestal - Out-of-Plane Shear Interaction

APPENDIX 3H DESIGN DETAILS AND EVALUATION RESULTS FOR THE REACTOR BUILDING STRUCTURE

This appendix evaluates the design and provides estimates of the available margins for key structural members of the Reactor Building (RB).

Design demands used in the evaluations are obtained from the structural analyses discussed in Appendix 3C and Appendix 3E, performed using the one-step approach and the Finite Element (FE) models discussed in Appendix 3B. The design demands are obtained from design load combinations which assess the operational and accident structural demands on the structure.

3H.1 Design Methodology

The design of the RB structure using Diaphragm Plate Steel-Plate Composite (DP-SC) is in accordance with Section 5.0 of NEDC-33926P-A, “BWRX-300 Steel-Plate Composite (SC) Containment Vessel and Reactor Building Structural Design,” (Reference 3H-1) developed to address the particularities of DP-SC construction.

The design for those portions of the RB structure using conventional steel-plate composite components is in accordance with ANSI/AISC N690-18 (Reference 3H-2), Appendix N9 as supplemented by RG 1.243, “Safety-Related Steel Structures and Steel-Plate Composite Walls for other than Reactor Vessels and Containments.”

Table 3H-1 presents the design parameters used in RB steel-plate composite design evaluations, which are in compliance with the Limitation and Condition (L&C) 8.3 and L&C 8.4 of NEDC-33926P-A (Reference 3H-1).

3H.2 Load Combinations and Design Demands

The evaluation of the RB structure is based on results from the load combinations presented in Table 3.8-7.

Additional detail of the individual load demands considered is described in Subsection 3.8.4.1.3. Discussion on analytical methods is captured in Subsection 3.8.4.1.4.

Similar to the Steel-Plate Composite Containment Vessel (SCCV), design demands are conservatively presented on element-by-element basis without averaging of the results across adjacent elements as allowed by the ANSI/AISC N690-18.

3H.3 Structural Capacity of RB Diaphragm Plate Steel-Plate Composite Sections

Design capacities for RB steel-plate composite components are determined using the material properties and geometric configuration of each section, and are computed as follows:

- DP-SC sections are evaluated using the design rules for BWRX-300 non-containment DP-SC structures described in Section 5.0 of NEDC-33926P-A.

- Conventional steel-plate composite sections are evaluated based on ANSI/AISC N690-18 Appendix N9 as supplemented with additional guidance from AISC Steel Design Guide 32, "Design of Modular Steel-Plate Composite Walls for Safety Related Nuclear Facilities," (Reference 3H-3) and RG 1.243.

These results provide capacities for uniaxial tensile strength, compressive strength, out-of-plane flexural strength, in-plane shear strength, and out-of-plane shear strength, as well as the combined capacities considering out-of-plane shear forces interaction, and in-plane membrane forces and out-of-plane moments interaction.

These capacities are used to calculate the demand-to-capacity ratios described in Section 3H.4.

3H.4 Structural Design Evaluation

Demand-to-capacity ratios are calculated for each load combination, and envelope ratios are calculated as the maximum element demand-to-capacity ratio over all the load combinations. The enveloping ratios are used to produce an enveloped contour plot for each design criteria. Key RB components considered in the evaluation include the RB exterior wall above and below ground, the outer mat foundation, and wing walls.

Figure 3H-1 through Figure 3H-6 present the demand-to-capacity ratios for key RB components for the load combinations in Table 3.8-7.

The demand-to-capacity ratios for the RB exterior wall below grade and above grade portions are provided in Figure 3H-1 and Figure 3H-2. The side of the RB exterior wall looking from the -X to +X direction is shown in the figures.

The RB exterior wall shows acceptable demand-to-capacity ratios for the below grade portions but locally high notional half demand-to-capacity ratios where the subgrade slabs intersect the wall. Some notional half exceedances are observed near grade where the thick grade slab intersects the RB exterior wall around the steam tunnel. The accident in the steam tunnel and pools produces high thermal and pressure demands in this area.

The demand-to-capacity ratios for the RB outer mat foundation is provided in Figure 3H-3. Similar to the inner mat foundation, the notional half demand-to-capacity ratios of the outer mat foundation are quite low, with high out-of-plane shear demand-to-capacity ratios near the intersections with the walls above.

The demand-to-capacity ratios for a set of the RB wing walls are provided in Figure 3H-4. These represent the wing walls that lie within the second quadrant (30 degree clockwise from -X to +Y) and the fourth quadrant (30 degree clockwise from +X to -Y). The notional half demand-to-capacity ratios show values generally between 0.5 and 0.75. Some locally high out-of-plane shear is observed near the top where the wing walls are attached to the thick slabs near grade.

The RB subgrade floor at EL-14.8m (-48.6 ft) and the refueling floor are provided in Figure 3H-5 and Figure 3H-6. The demand-to-capacity ratios for the subgrade floor are all quite low as these floors have been thickened to increase their vertical stiffness and natural frequency for vibratory purposes. The refueling floor has generally acceptable demand-to-capacity ratios with some exceedances around the south side where the thick fuel pool walls below intersect.

A summary of additional evaluations performed. or design updates developed to address these exceedances will be included in the FSAR.

3H.5 Critical Reactor Building Connections Design

The RB exterior wall-to-outer mat foundation connection is a full-strength rigid connection, in which the RB wall is the continuous member, with the mat foundation connected to the wall. This connection is referred as a full-strength rigid L-connection.

Wing walls, and slab at grade-to-RB and SCCV walls, are also connected using full-strength rigid T-connections, while below-grade slabs are connected to the RB and SCCV walls using overstrength rigid T-connections for out-of-plane moment transfer.

The RB roof-to-RB exterior wall is a full-strength L-connection. In this connection, the RB reinforced concrete roof is the continuous member and bearing on the RB exterior wall. This connection is classified as rigid connection for out-of-plane moment transfer due to the continuity of the concrete and headed dowels used to transfer the out-of-plane moment.

For the design methodology of these connections, refer to Section 3F.5. See Figure 3F-13 for an example of these connections.

3H.6 References

3H-1 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.

3H-2 ANSI/AISC N690-18, “Specification for Safety-Related Steel Structures for Nuclear Facilities,” American National Standards Institute/American Institute of Steel Construction, 2018.

3H-3 AISC Steel Design Guide 32, “Design of Modular Steel-Plate Composite Walls for Safety Related Nuclear Facilities,” American Institute of Steel Construction, 2017.

Table 3H-1 Design Parameters for the Reactor Building Steel-Plate Composite Components

Structural Component	Component Type	Member Thickness (in)	Faceplate Thickness (in)	Yield Strength (ksi)	Concrete Compressive Strength (ksi)	Diaphragm Plate Spacing (in)
RB exterior wall (elevation -5 m (-16.4 ft) and lower)	DP-SC	36	0.75	65	8	18
RB exterior wall (elevation -5 m (-16.4 ft) to 0.0 m (0.0 ft))	DP-SC	36	1.25			18
RB exterior wall (elevation 0.0 m (0.0 ft) to 18.3 m (60.0 ft))	DP-SC	36	1.25			18
RB exterior wall (elevation 18.3 m (60.0 ft) and above)	DP-SC	36	0.75			18
Outer mat foundation	DP-SC	48	0.5			24
Wing Walls	Conventional	18	0.5	50		9(1)
Steam Tunnel (Thick Wing) Walls	DP-SC	36	0.875	50		18

Note:

(1) Tie bar spacing provided vs the diaphragm spacing provided for DP-SC components.

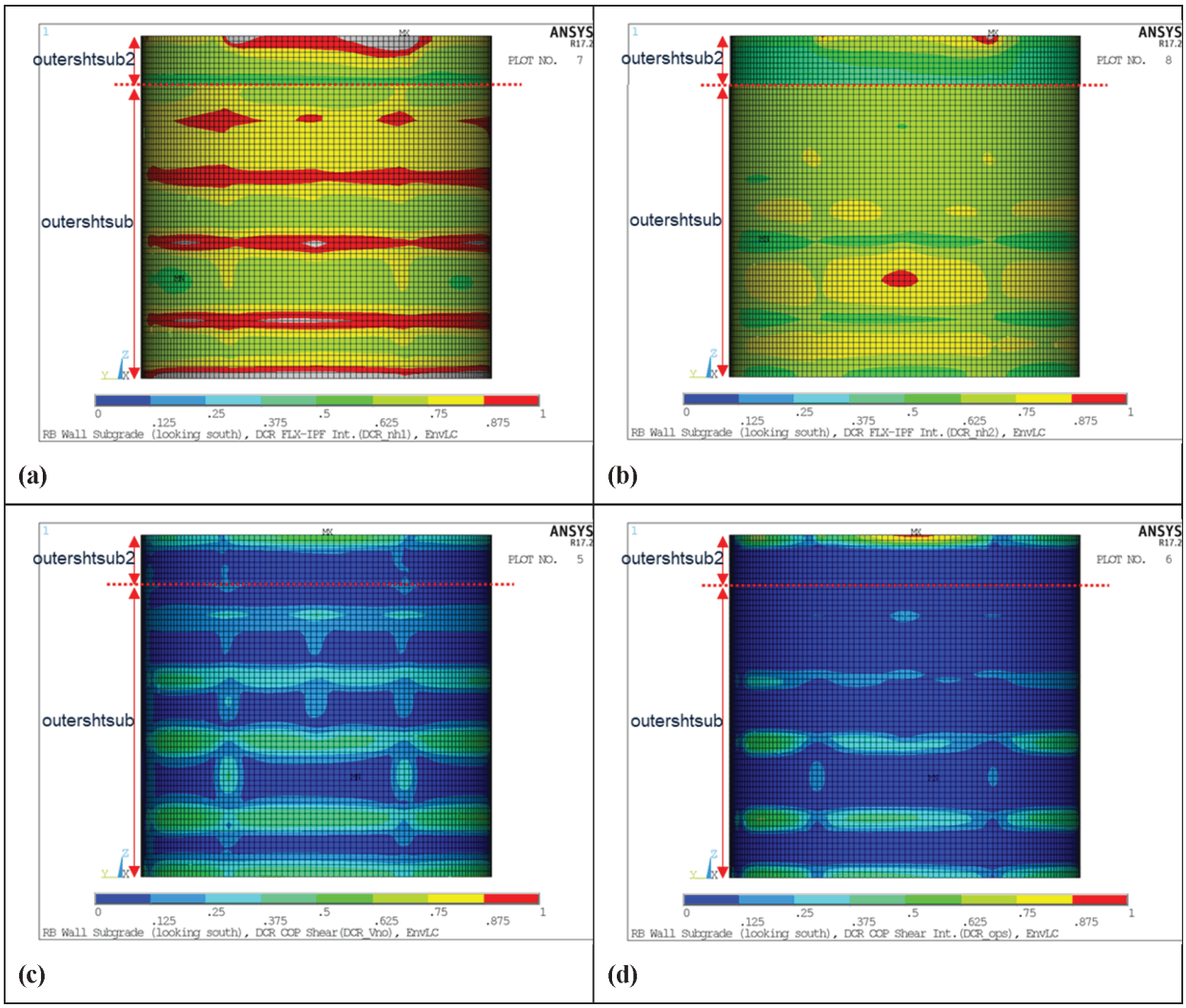


Figure 3H-1 Envelope Demand-to-Capacity Ratios for Reactor Building Exterior Wall Below Grade (a) Notional Half 1, (b) Notional Half 2, (c) Out-of-Plane Shear, and (d) Out-of-Plane Shear Interaction

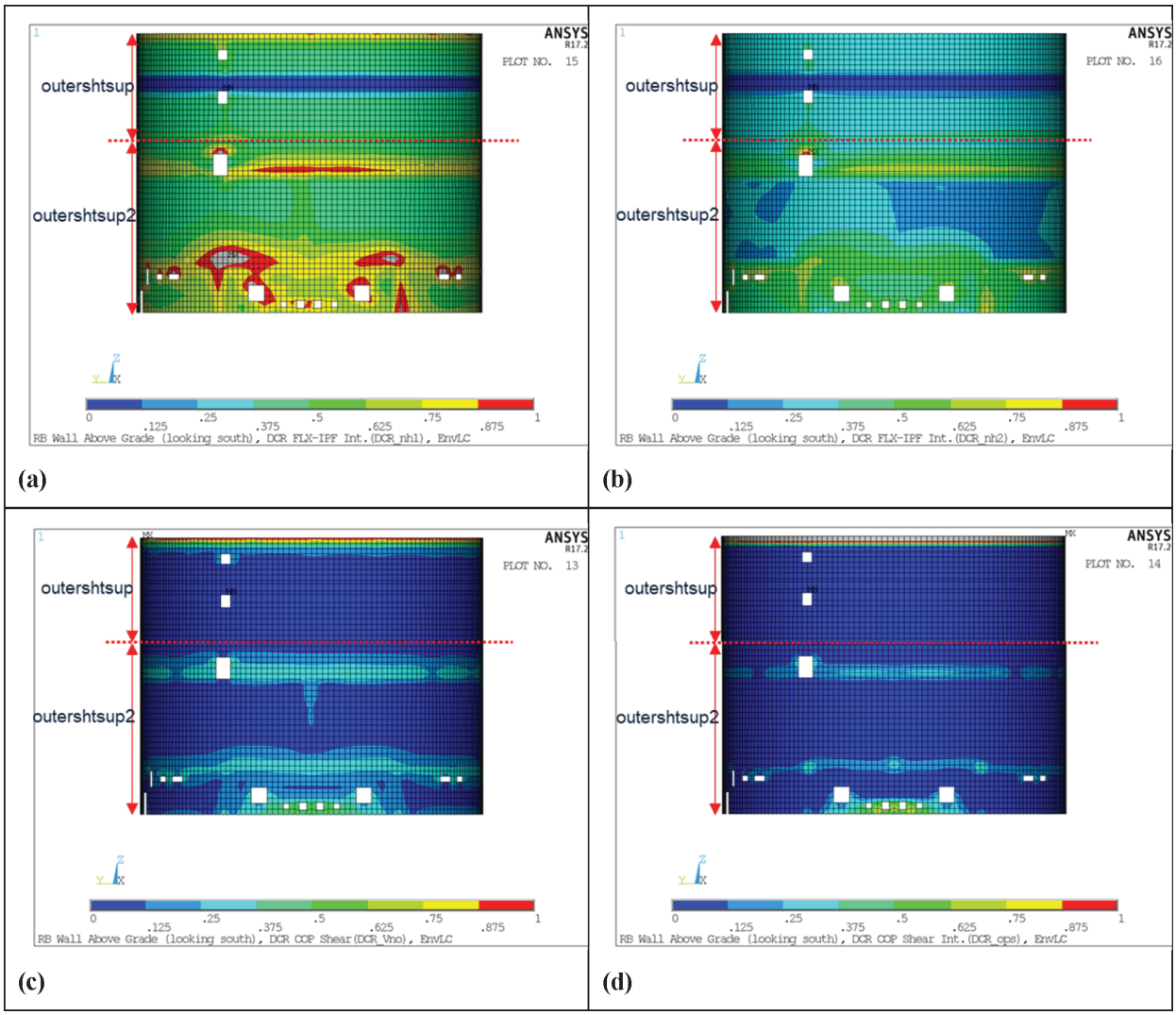


Figure 3H-2 Envelope Demand-to-Capacity Ratios for the Reactor Building Exterior Wall Above Grade (a) Notional Half 1, (b) Notional Half 2, (c) Out-of-Plane Shear, and (d) Out-of-Plane Shear Interaction

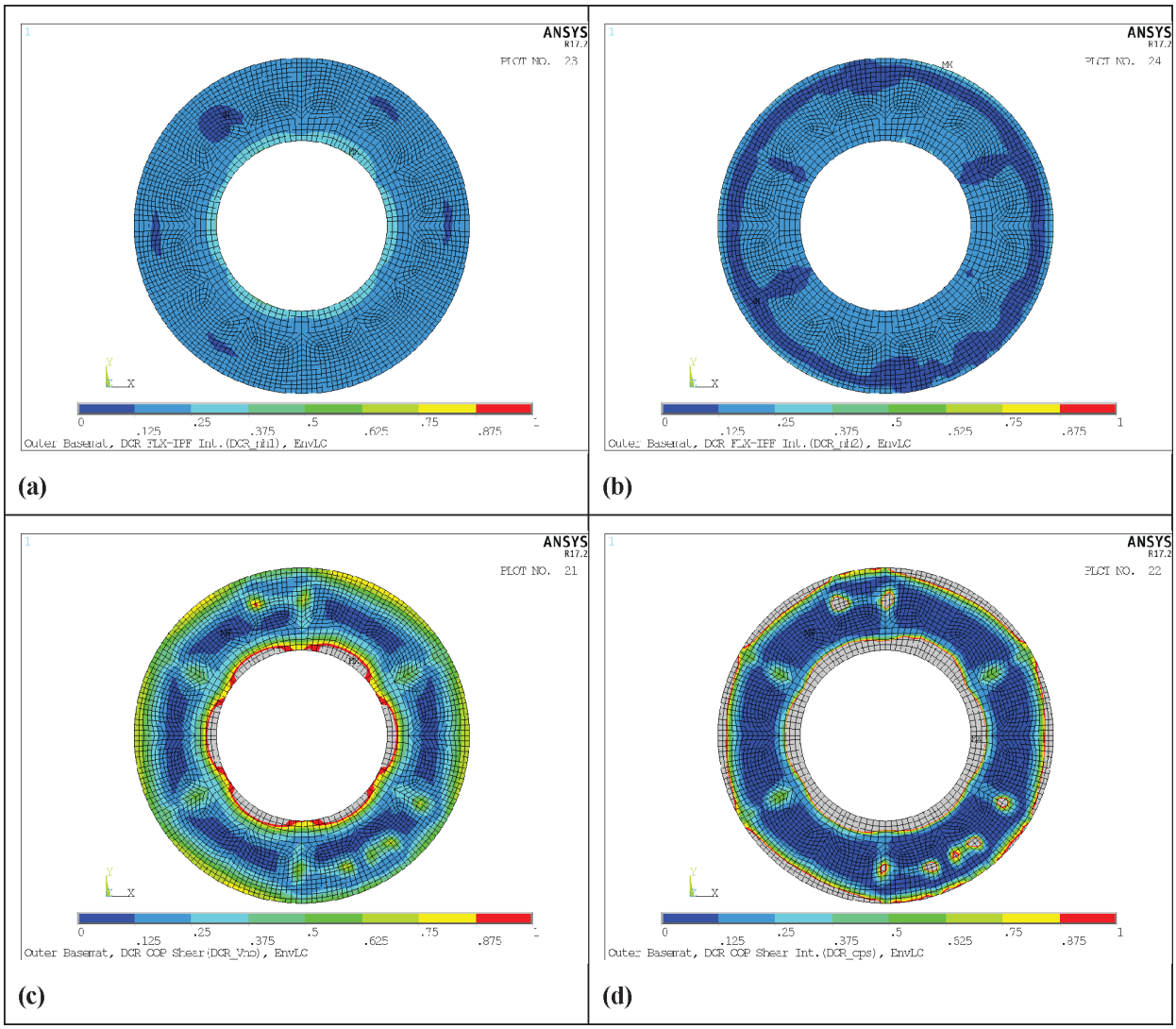


Figure 3H-3 Outer Mat Foundation Envelope Demand-to-Capacity Ratios (a) Notional Half 1, (b) Notional Half 2, (c) Out-of-Plane Shear, and (d) Out-of-Plane Shear Interaction

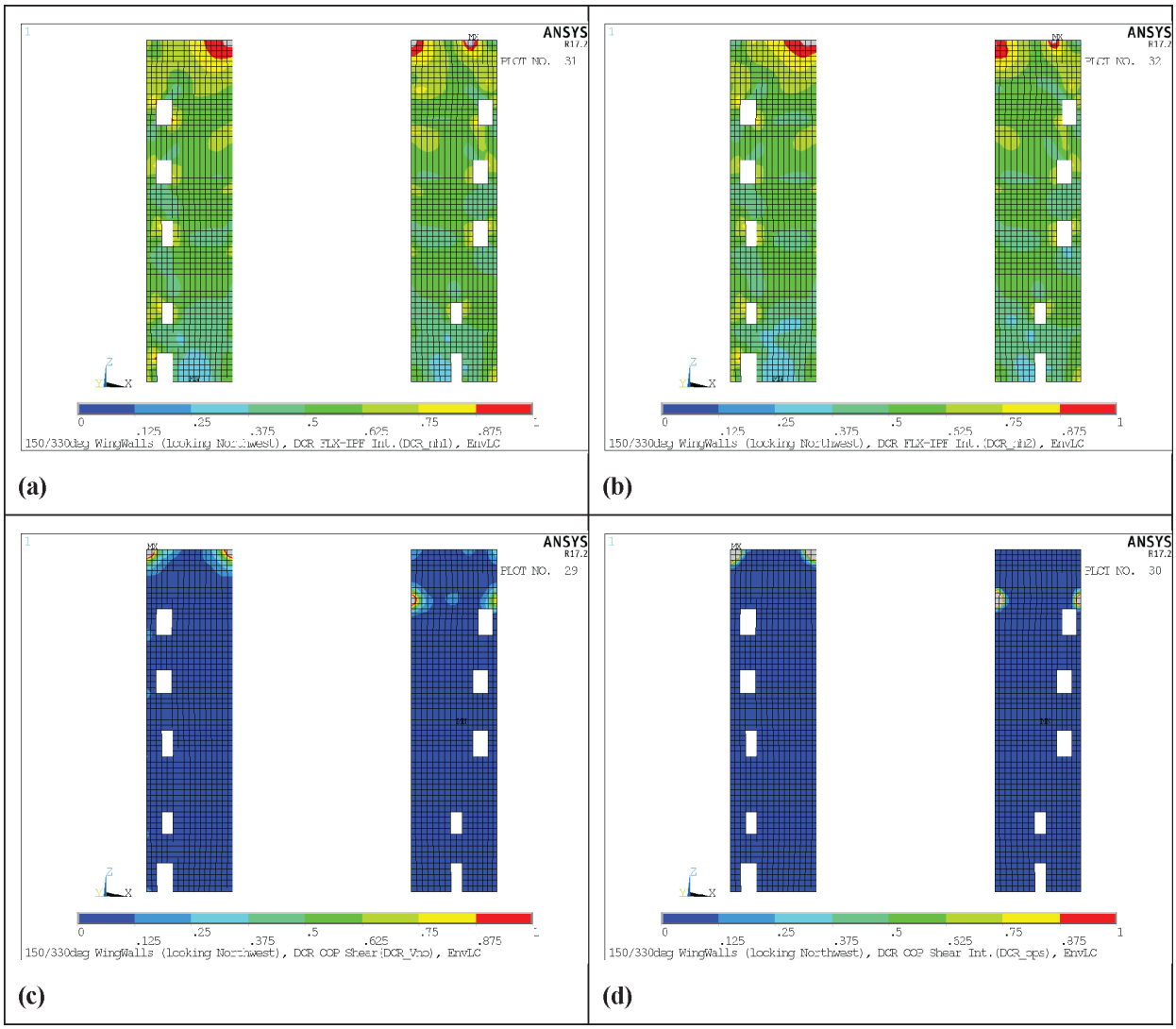


Figure 3H-4 Envelope Demand-to-Capacity Ratios for Reactor Building Wing Walls (a) Notional Half 1, (b) Notional Half 2, (c) Out-of-Plane Shear, and (d) Out-of-Plane Shear Interaction

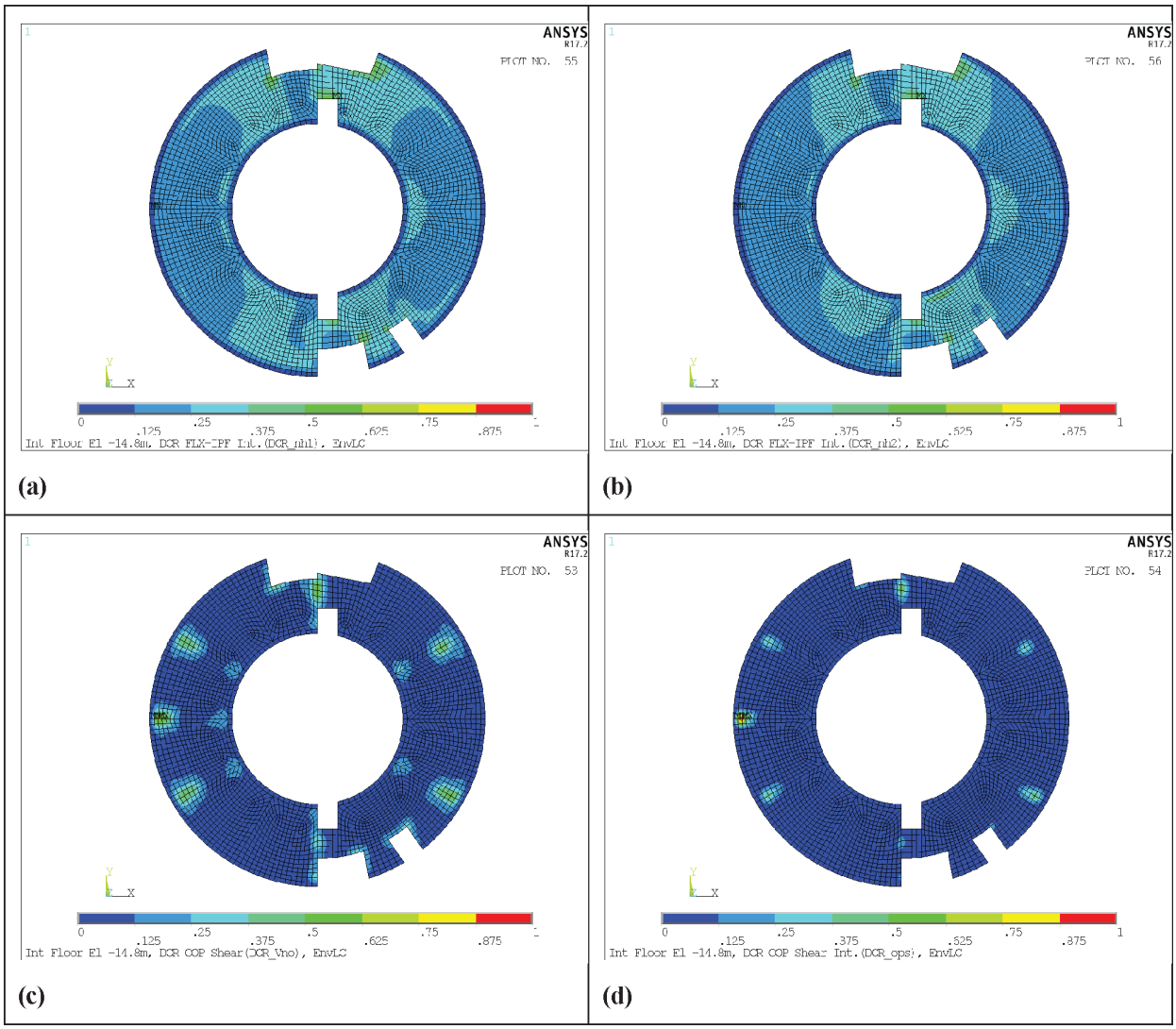


Figure 3H-5 Envelope Demand-to-Capacity Ratios for Reactor Building Subgrade Floor EL-14.8m (a) Notional Half 1, (b) Notional Half 2, (c) Out-of-Plane Shear, and (d) Out-of-Plane Shear Interaction

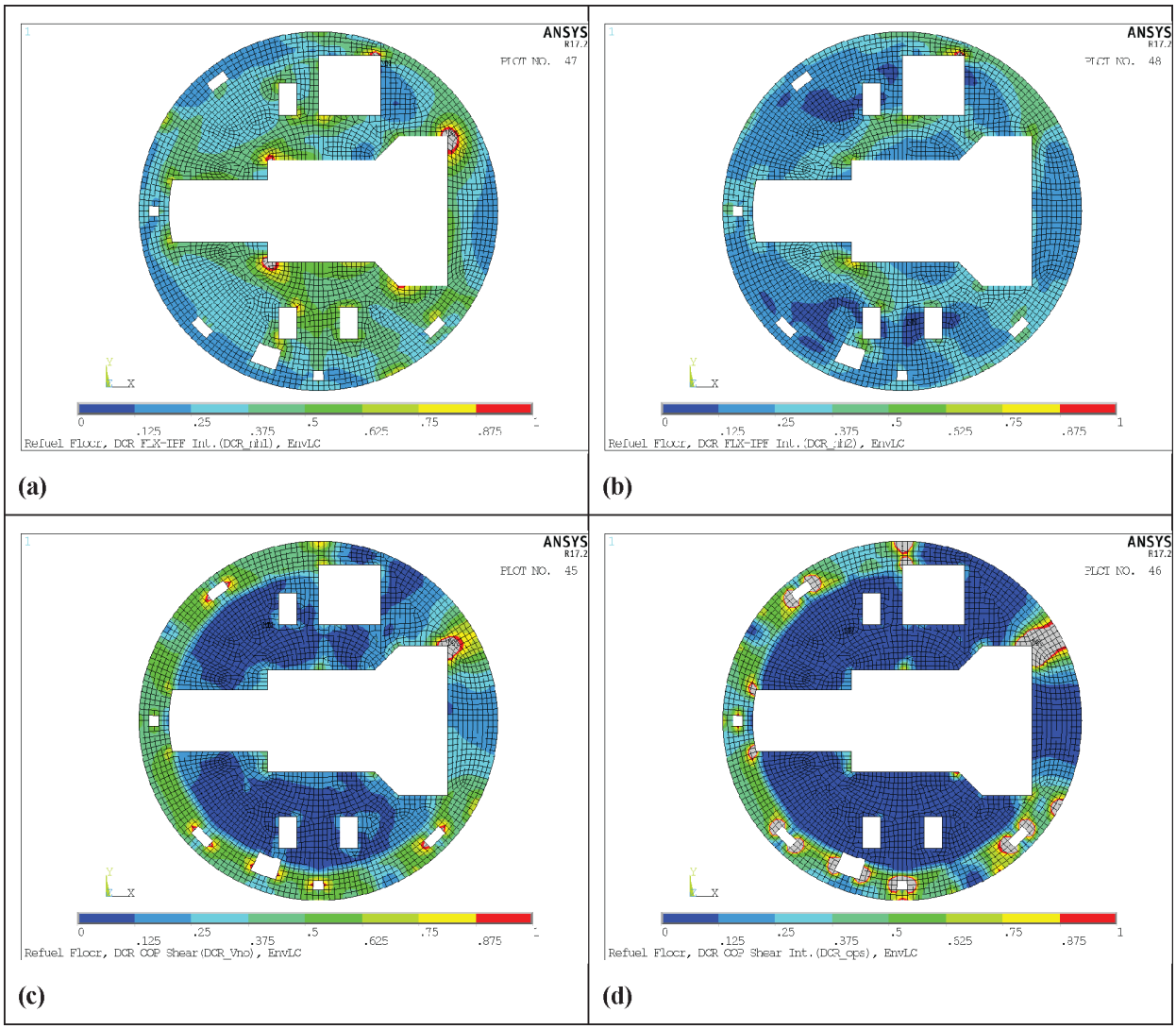


Figure 3H-6 Envelope Demand-to-Capacity Ratios for Reactor Building Refuel Floor (a) Notional Half 1, (b) Notional Half 2, (c) Out-of-Plane Shear, and (d) Out-of-Plane Shear Interaction

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 STRUCTURES, SYSTEMS AND COMPONENTS – MECHANICAL DESIGN

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 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 STRUCTURES, SYSTEMS AND COMPONENTS – 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% or 30% 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 STRUCTURES, SYSTEMS AND COMPONENTS – 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 STRUCTURES, SYSTEMS AND COMPONENTS - 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 GVH quality assurance program.

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 Principal Design Criteria 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.

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, and
- 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 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 SSCs required to support core cooling and fuel pool cooling based on the methodology in Section 3 of NEI 07-13, and
- 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 Section 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:

- Reactor Building, 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, and
- 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), and
  - 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, and
- 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, and
- 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).

3N.7 References

3N-1 NEI 07-13, "Methodology for Performing Aircraft Impact Assessment for New Plant Designs," Nuclear Energy Institute, Revision 8, April 2011.

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LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
1D	One-Dimensional
2D	Two-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
BOC	Beginning of Cycle
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
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
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
GNF	Global Nuclear Fuel
GT	Gamma Thermometer
GVH	GE Vernova Hitachi Nuclear Energy
HCU	Hydraulic Control Unit
I&C	Instrumentation and Control
IASCC	Irradiation Assisted Stress Corrosion Cracking

LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
ICS	Isolation Condenser System
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
MCPR	Minimum Critical Power Ratio
MCR	Main Control Room
MFLPD	Maximum Fraction of Limiting Power Density
NBS	Nuclear Boiler System
OLMCPR	Operating Limit Minimum Critical Power Ratio
PCV	Pressure Control Valve
PLR	Partial Length Fuel Rod
PDC	Principal Design Criteria
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
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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CHAPTER 4 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 Section 4.6), 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 Subsection 4.3. The thermal-hydraulic design bases requirements and important methodologies are described in Subsection 4.4. Reactor internals materials and the Control Rod Drive (CRD) system structural materials are discussed in Subsection 4.5. The functional design of the CRD system is discussed in Subsection 4.6. The thermal-hydraulic stability is discussed in Appendix 4A.

The Safety Classifications, Quality Groups, and Seismic Categories of the reactor components are listed 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 (FMCRD).

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 of 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, “Rules 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 Tubes

The control rod guide tubes fit into core plate holes and rest on their associated CRD housings that are welded to the reactor bottom head. The control rod guide tubes provide lateral support and channel the 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 the 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 support 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 provide 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 (LOCA).

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 steam 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 startup, power operation, and plant shutdown modes of operation. The FW spargers also provide a uniform distribution of FW flow within the downcomer annulus and deliver 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 connections inside the vessel on 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, 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 an 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.

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 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. GTs represents a diverse technology to the neutron detecting Local Power Range Monitors (LPRM). 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,” (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, “Rules for Construction of Nuclear Facility Components,” Subsection NG, “Core Support Structures,” American Society of Mechanical Engineers, 2021.

4.1-2 ASME BPVC, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” Subsection NF, “Supports,” American Society of Mechanical Engineers, 2021.

4.1-3 NEDC-32983P-A, “General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations,” GE Nuclear Energy, Revision 2, January 2006.

Table 4.1-1 Analytical Techniques in Core Design

Analysis	Technique	Computer Code
Fuel rod design Fuel performance characteristics (temperature, internal pressure, and clad stress)	Numerical solutions of 1D steady-state and transient heat transfer and finite element mechanical analysis	PRIME03P
Nuclear design Cross-sections and group constants	Lattice physics	TGBLA06
X-Y and X-Y-Z power distribution, reactivity coefficients Axial power distributions, control rod worth Fuel rod power	Steady-state coupled nuclear thermal hydraulics Quasi 2 group diffusion theory	PANAC11
Thermal-hydraulic design steady state	Multi-dimensional, two-fluid model thermal hydraulics 3D reactor kinetics	TRACG v4

Note: 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

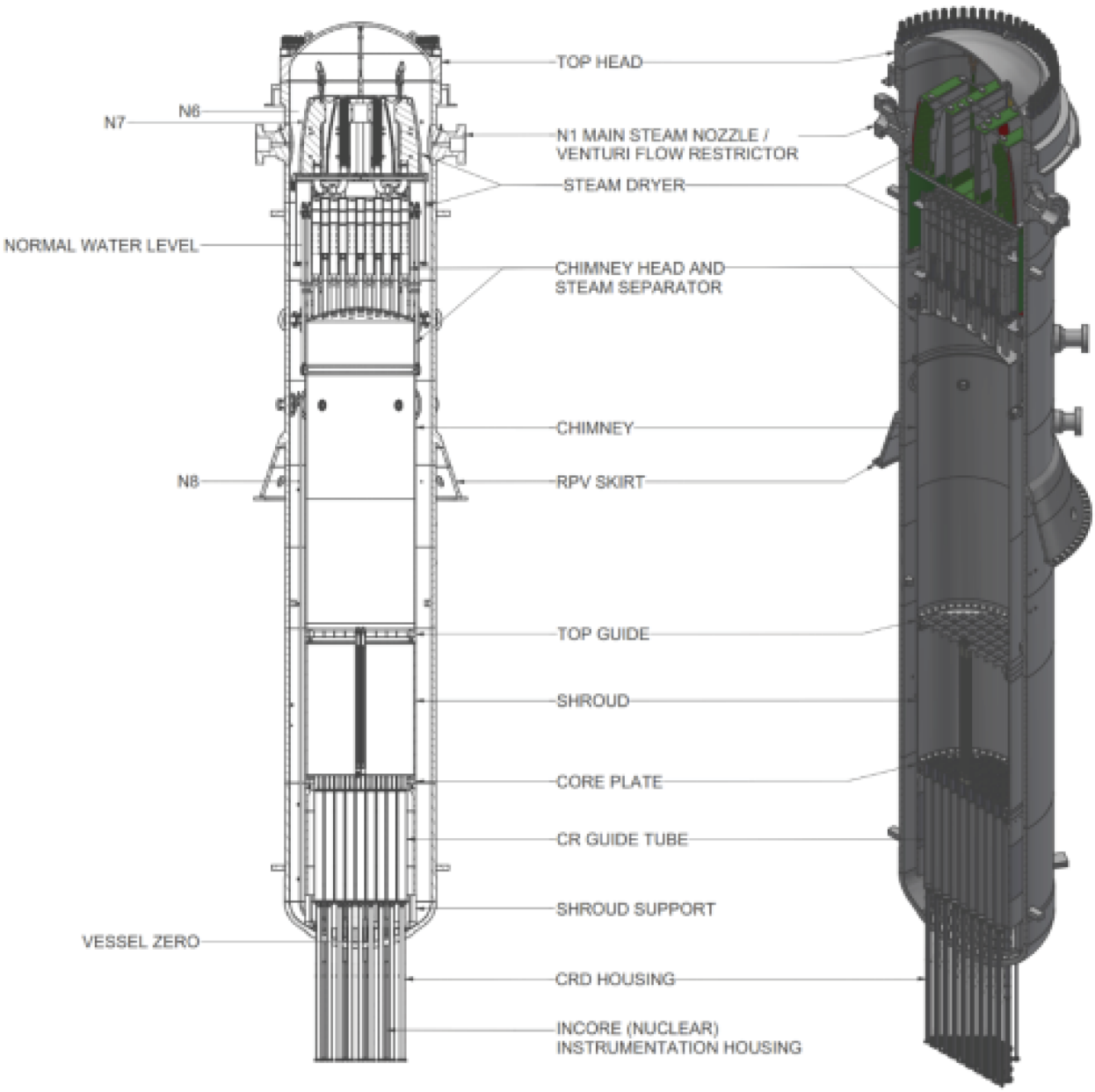


Figure 4.1-1 BWRX-300 Reactor Pressure Vessel and Internals Schematic

Note: Schematic is not to scale. Some features are omitted or relocated for display purposes.

4.2 Fuel System Design

This section discusses 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, "General Electric Standard Application for Reactor Fuel (GESTAR II) (Supplement for United States)," (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 fuel rods, thick corner/thin wall channel, and axial enrichment loading. New or improved features included in GNF2 are:

- Partial length fuel 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 is incorporated in GE6 through GE14 designs is standard in the GNF2 design. The combination and number of partial length fuel rods, as well as the lengths of partial length fuel 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 (70 standard and 8 tie 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 fuel bundle during fuel handling operations when the fuel 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 standard 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 fuel bundle operation cycle.

4.2.2.1.4 Partial Length Fuel Rods

Fourteen partial length fuel 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 (PLR) 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™ 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 CRN 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 (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 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% 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 the thermal-mechanical design criteria specified by NEDE-24011-P-A-31-US (Reference 4.2-1).

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 and compliance is addressed as follows:

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 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 a 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 Isolation Condenser System (ICS) function and is addressed in Section 6.3.

Section 3.1 summarizes compliance with GDC.

4.2.6 Testing, Inspection and Surveillance Plans

The GE Vernova Hitachi Nuclear Energy (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),” Global Nuclear Fuel - Americas, LLC, November 2020.

4.2-2 NEDC-33270P, “GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II),” Global Nuclear Fuel - Americas, LLC, Revision 11, August 2020.

4.2-3 NEDE-33798P, Supplement 1, “NSF Channel Annual Experience Summary Report,” Global Nuclear Fuel - Americas, LLC, Revision 7, January 2024.

4.2-4 NEDC-33256P-A, “The PRIME Model for Analysis of Fuel Rod Thermal - Mechanical Performance Part 1 - Technical Bases,” Global Nuclear Fuel - Americas, LLC, Revision 2, October 2021.

4.2-5 NEDC-33257P-A, “The PRIME Model for Analysis of Fuel Rod Thermal - Mechanical Performance Part 2 - Qualification,” Global Nuclear Fuel - Americas, LLC, Revision 2, October 2021.

4.2-6 NEDC-33258P-A, “The PRIME Model for Analysis of Fuel Rod Thermal - Mechanical Performance Part 3 - Application Methodology,” Global Nuclear Fuel - Americas, LLC, Revision 2, October 2021.

4.2-7 NEDC-33840P-A, “The PRIME Model for Transient Analysis of Fuel Rod Thermal-Mechanical Performance,” Global Nuclear Fuel - Americas, LLC, Revision 1, August 2017.

4.2-8 NEDC-34042P, “BWRX-300 GNF2 Fuel Assembly Thermal-Mechanical Design Report,” GE-Hitachi Nuclear Energy Americas, LLC, Revision 0, 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),” General Electric Company, Revision 0, October 1984.

4.2-10 NEDC-33139P-A, “Cladding Creep Collapse,” Global Nuclear Fuel - Americas, LLC, Revision 0, July 2005.

4.2-11 NEDC-34041P, “BWRX-300 GNF2 Fuel Assembly Mechanical Design Report,” GE-Hitachi Nuclear Energy Americas, LLC, Revision 0, April 2024.

4.2-12 NEDC-33934P, “BWRX-300 Safety Strategy,” GE-Hitachi Vernova Nuclear Energy Americas, LLC, Revision 2, March 2026.

4.2-13 NEDC-33912P-A, “BWRX-300 Reactivity Control,” GE-Hitachi Nuclear Energy Americas, LLC, Revision 2, February 2023.

4.2-14 NEDE-24011-P-A-31, “General Electric Standard Application for Reactor Fuel, (GESTAR II),” Global Nuclear Fuel - Americas, LLC, Revision 31, November 2020.

Table 4.2-1 GNF2 Fuel Assembly Material

Component	Material (Reference 4.2-11)
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

Table 4.2-2 GNF2 Fuel Bundle Key Attributes

Component	GNF2
Fuel Bundle	
Total number of fuel rods	92
Number of full length standard rods	70
Number of full length 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

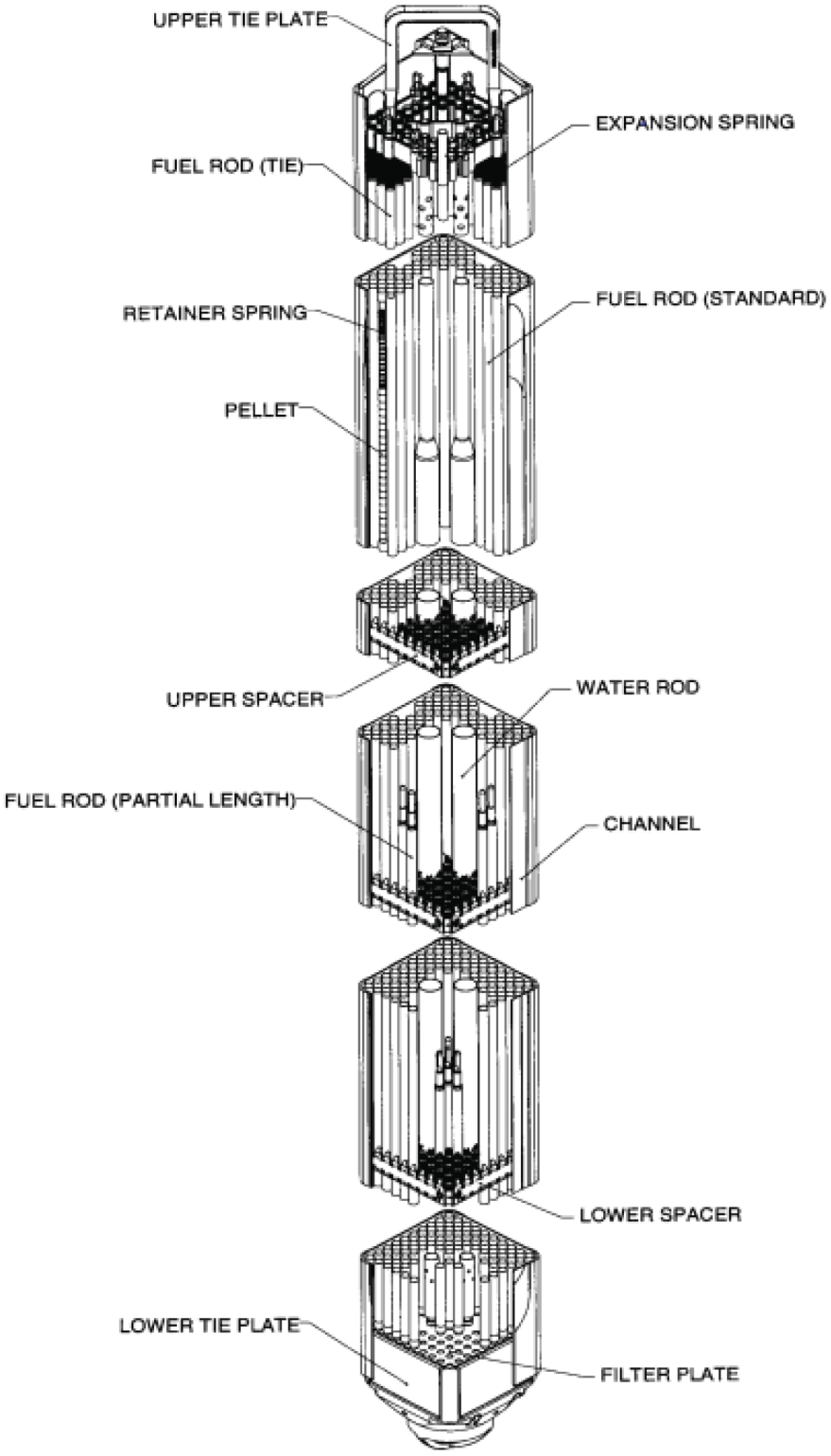


Figure 4.2-1 GNF2 Fuel Assembly

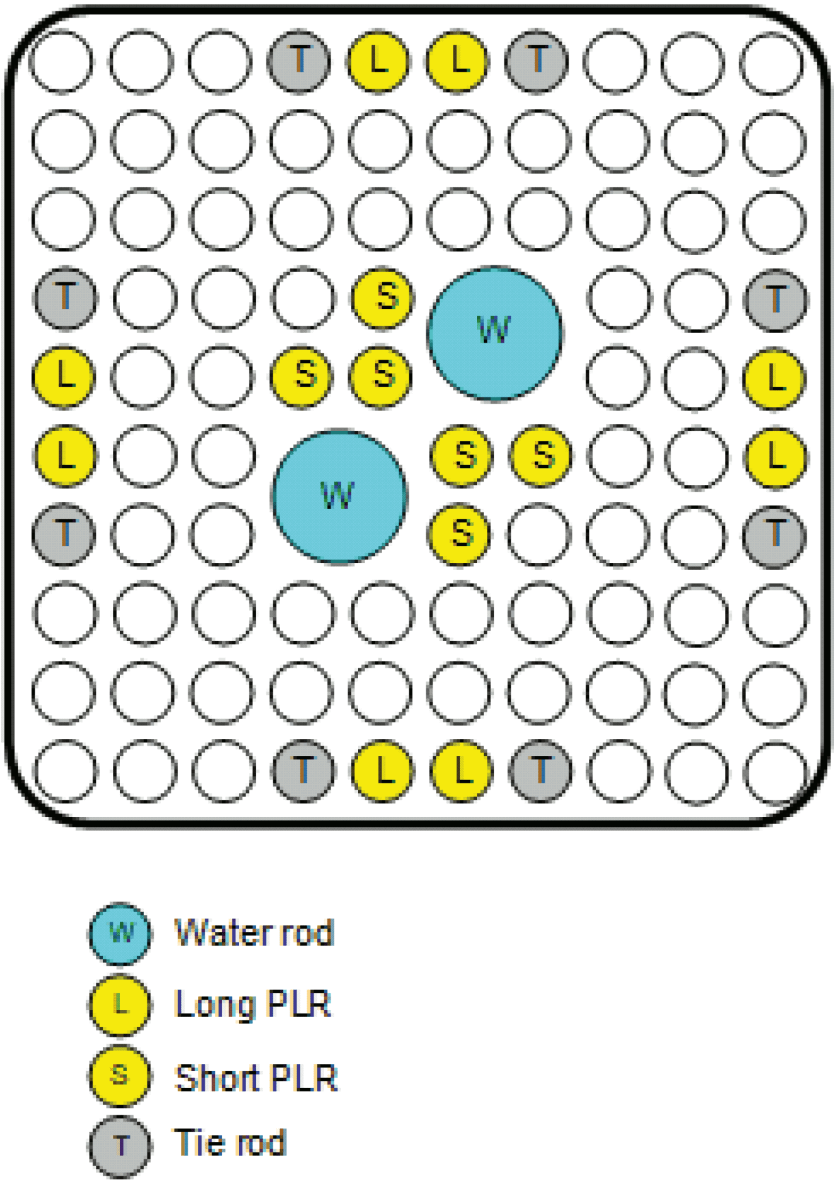


Figure 4.2-2 GNF2 Lattice Arrangement

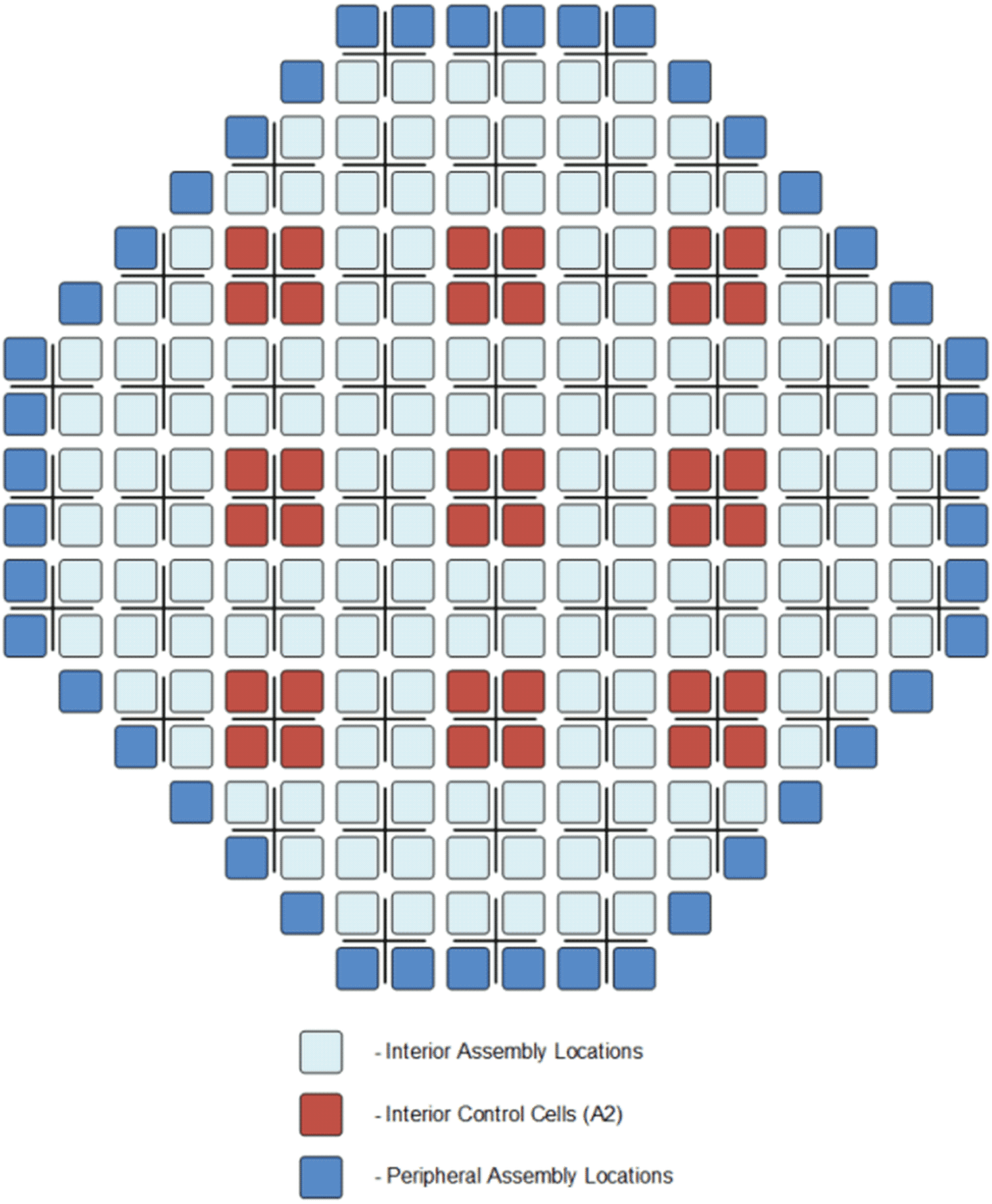


Figure 4.2-3 Control Rods Distributed Throughout Core

Note: All available control cells are shown in red.

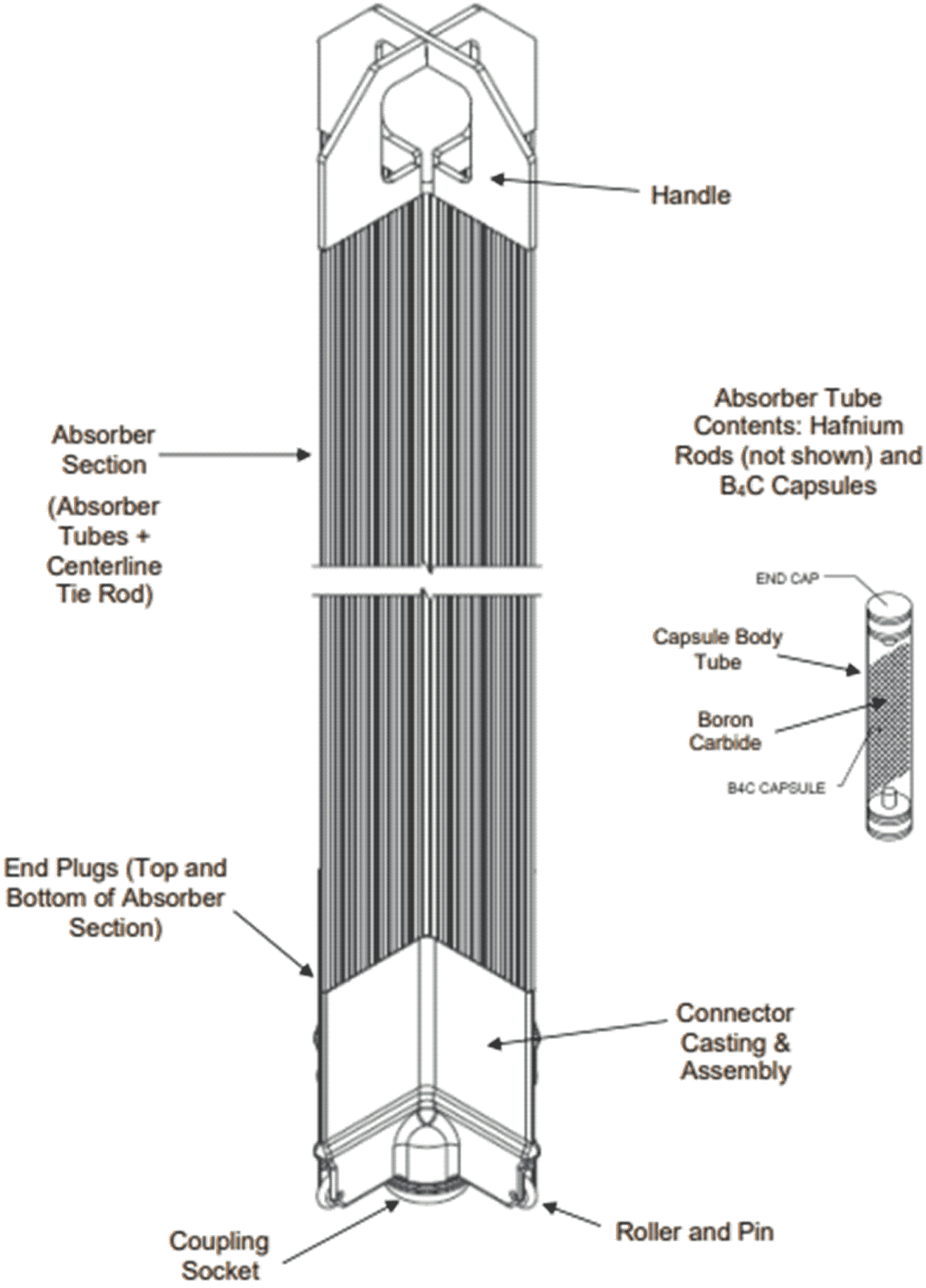


Figure 4.2-4 BWRX-300 Control Rod

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 reactivity control 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 (GESTAR II) (Supplement for United States),” (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 a single 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 below.

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 Subsection 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). A representative nuclear instrumentation design is depicted in Figure 4.3-1. A representative control rod pattern at Beginning of Cycle (BOC) is shown on Figure 4.3-3. 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 (ECP) used in the steady-state nuclear core analysis are the Two-Dimensional (2D) lattice physics code TGBLA06 and the 3D BWR core simulator code 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 PANAC11 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, and
- 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 reactivity control 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 BWR core simulator code 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 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% $\Delta k/k$. 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 Section 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 Mhleberg (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 Principal Design Criteria (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 (Reference 4.3-7) 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 Section 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 PDC 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,” GE-Hitachi Nuclear Energy Americas, LLC, Revision 0, April 2024.

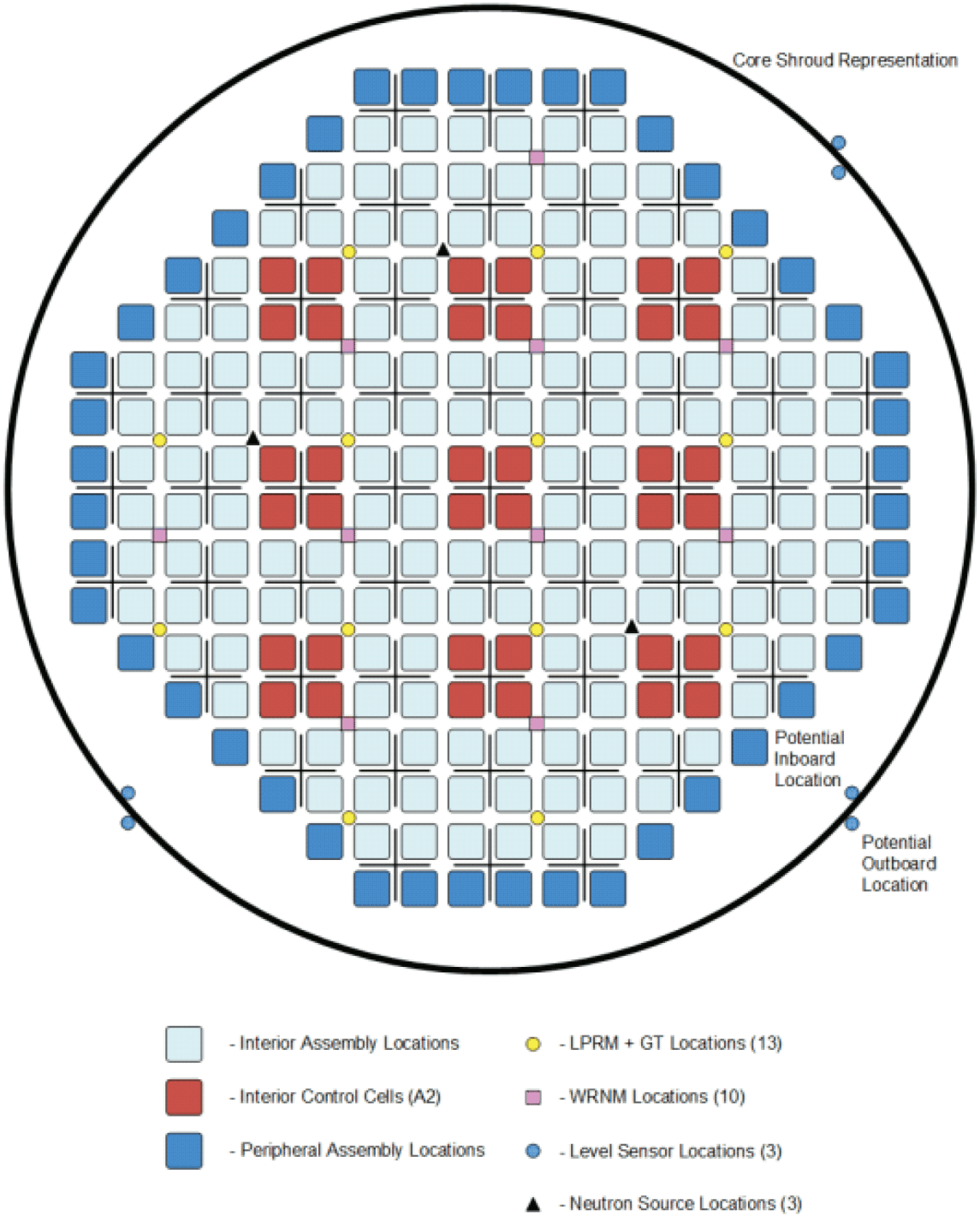
4.3-3 NEDC-34045P, “BWRX-300 GNF2 Fuel Bundle Information Report for Equilibrium 12-Month Cycle,” GE-Hitachi Nuclear Energy Americas, LLC, Revision 0, April 2024.

4.3-4 NEDC-34039P, “BWRX-300 GNF2 Steady State Nuclear Methods: TGBLA06/PANAC11 Application Methodology,” GE-Hitachi Nuclear Energy Americas, LLC, Revision 0, April 2024.

4.3-5 NEDC-33270P, “GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II),” Global Nuclear Fuel - Americas, LLC, Revision 11, August 2020.

4.3-6 NEDC-34042P, “BWRX-300 GNF2 Fuel Assembly Thermal-Mechanical Design Report,” GE-Hitachi Nuclear Energy Americas, LLC, Revision 0, April 2024.

4.3-7 NEDE-24011-P-A-31, “General Electric Standard Application for Reactor Fuel, (GESTAR II),” Global Nuclear Fuel - Americas, LLC, Revision 31, November 2020.



Notes:

1: Penetration locations are approximate.

2: All available control cells are shown in red.

Figure 4.3-1 Core Map Instrumentation Layout

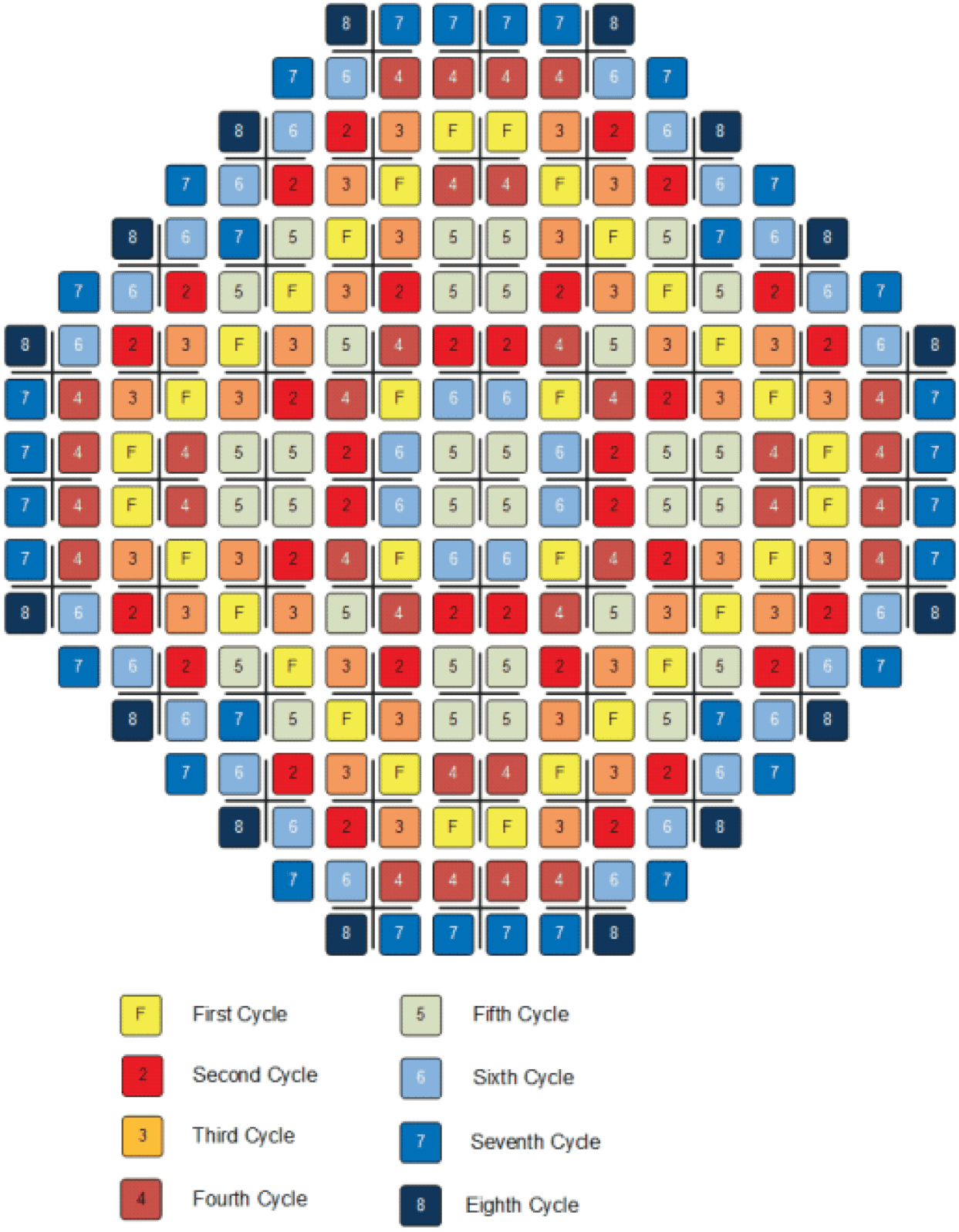


Figure 4.3-2 Core Loading Map – Reference Equilibrium Cycle

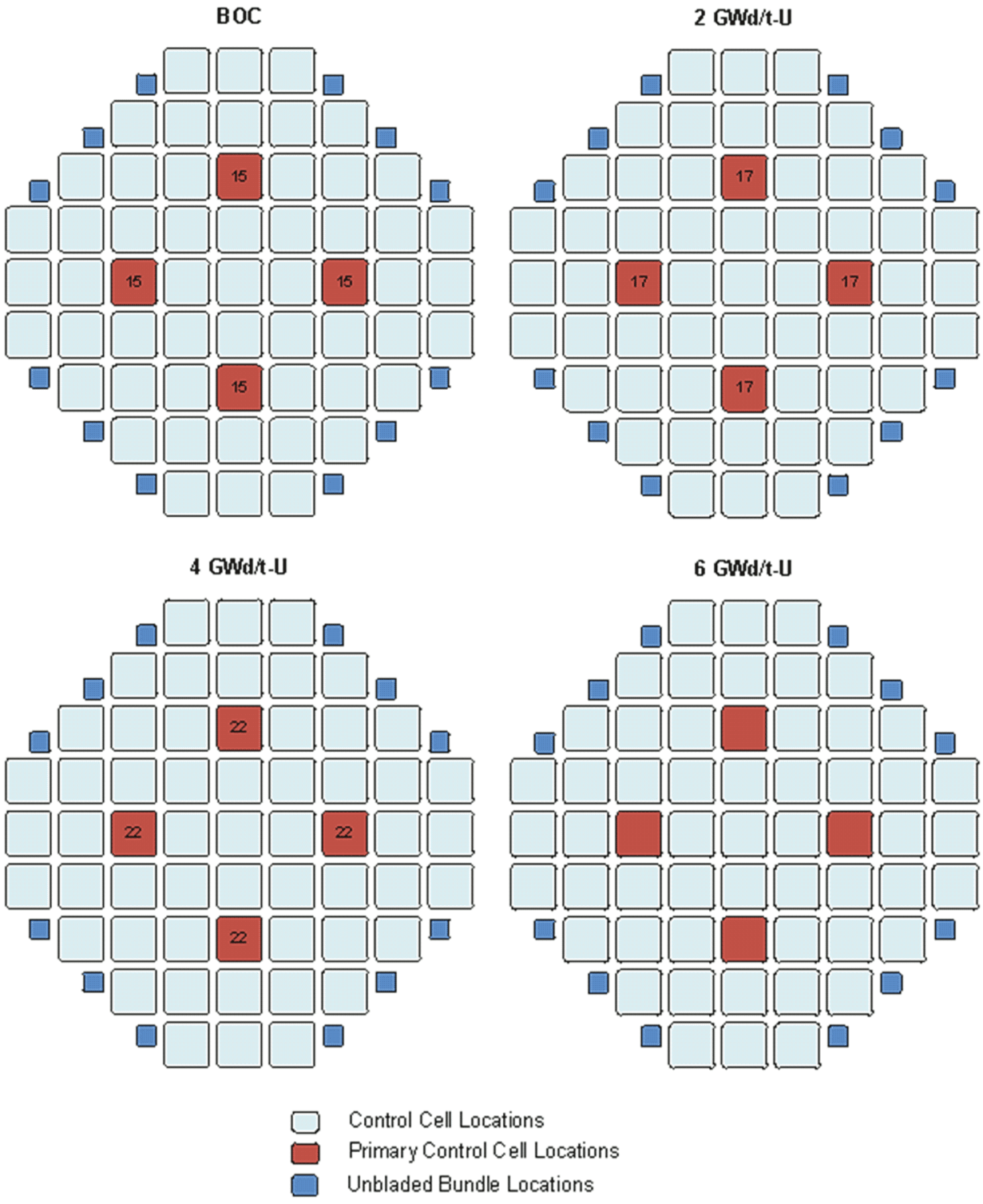


Figure 4.3-3 Representative Control Rod Patterns

Note: Control cells selected for use in the representative core depletion are shown in red.

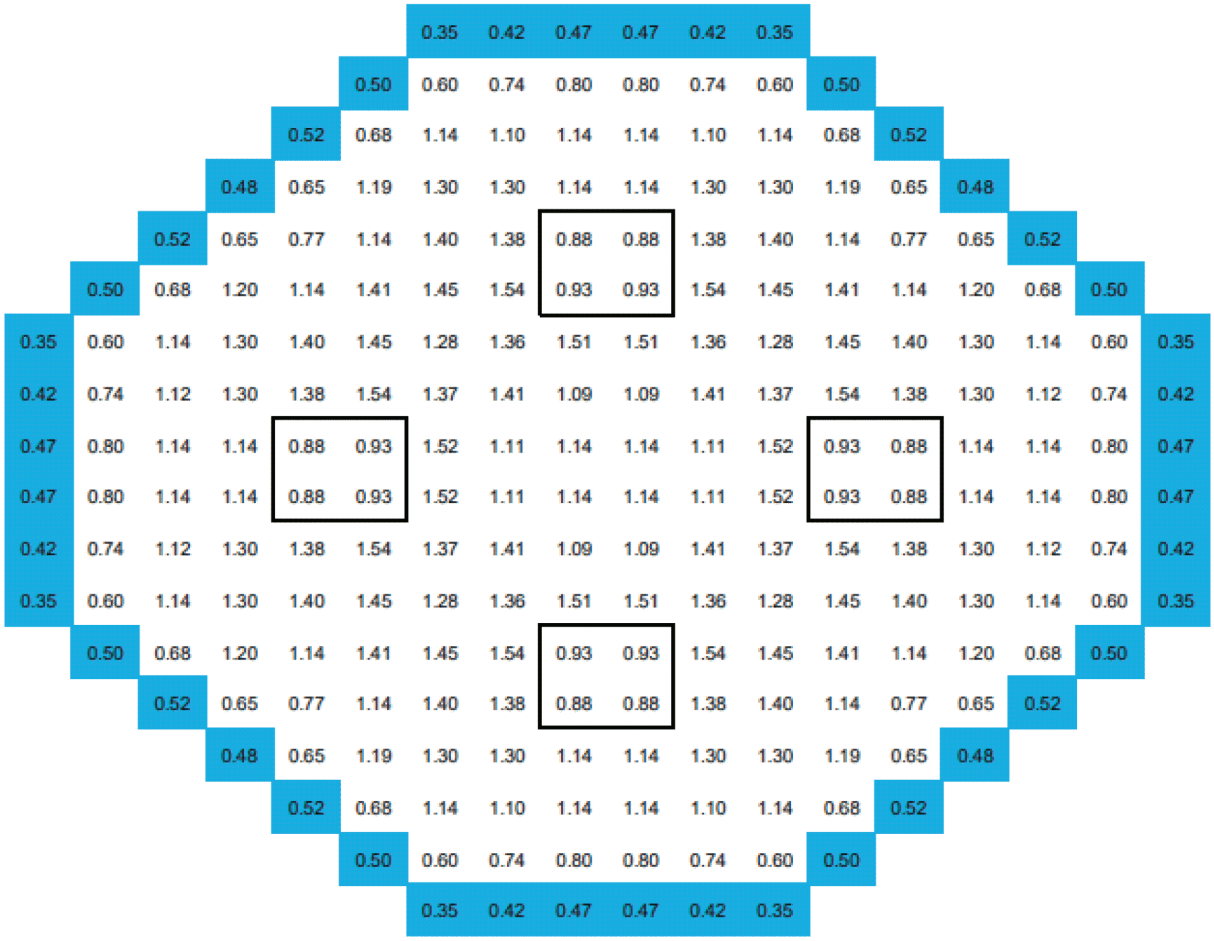


Figure 4.3-4a Peak to Average Bundle Power Ratio at Beginning of Cycle (Average Bundle Power – 3.625 MW_{th})

Note: Control cells selected for use in the representative core depletion are outlined.

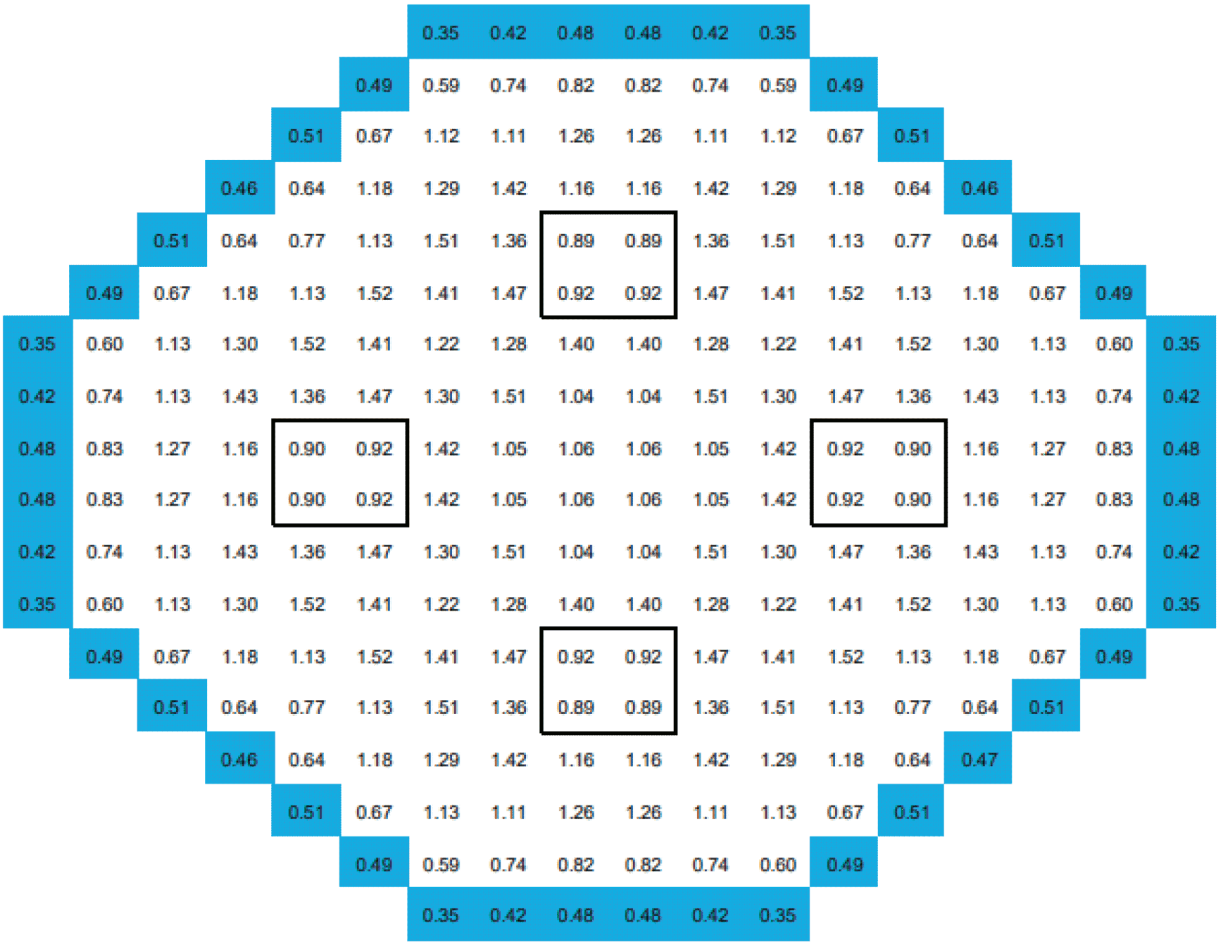


Figure 4.3-4b Peak to Average Bundle Power Ratio at Middle of Cycle (Average Bundle Power – 3.625 MW_{th})

Note: Control cells selected for use in the representative core depletion are outlined.

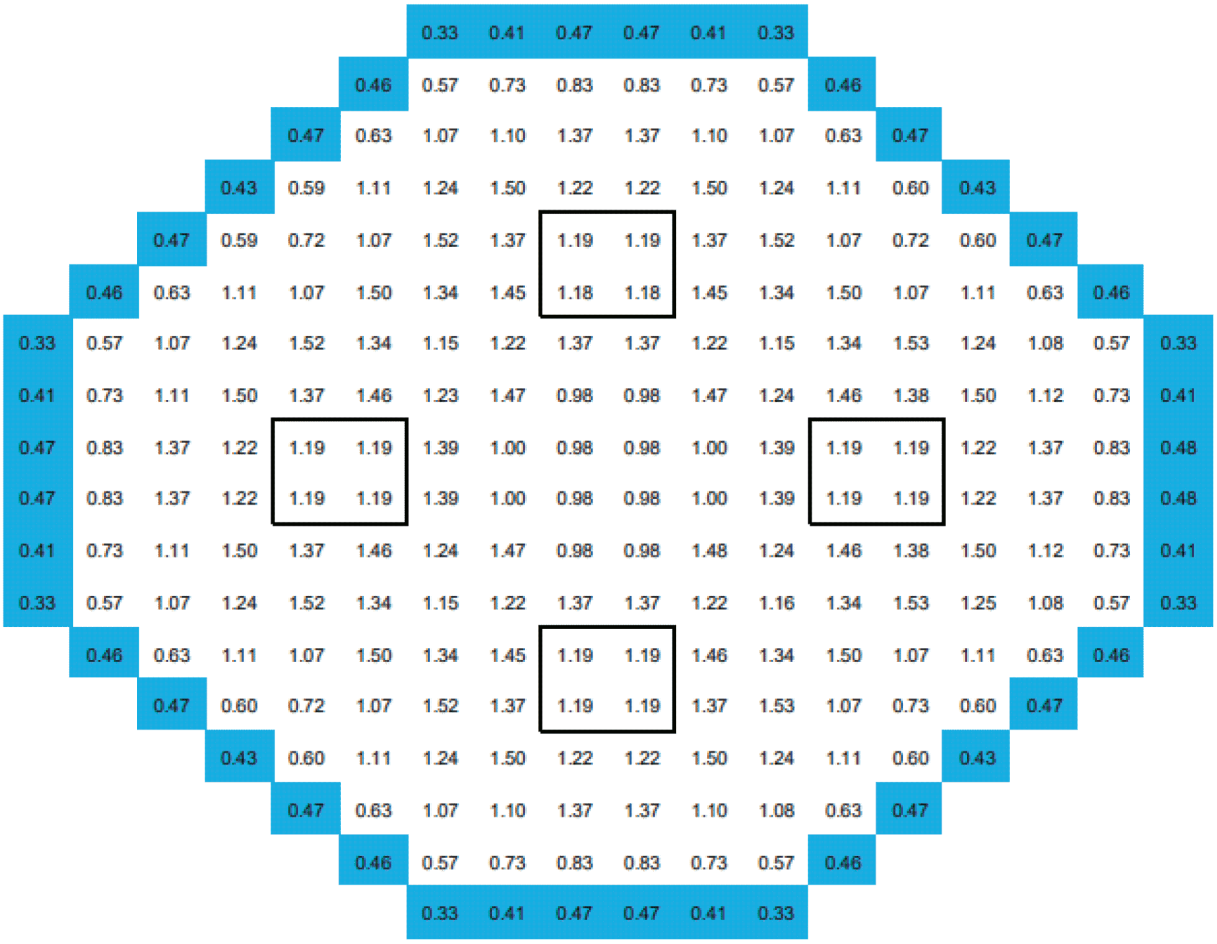


Figure 4.3-4c Peak to Average Bundle Power Ratio at End of Cycle (Average Bundle Power – 3.625 MW_{th})

Note: Control cells selected for use in the representative core depletion are outlined.

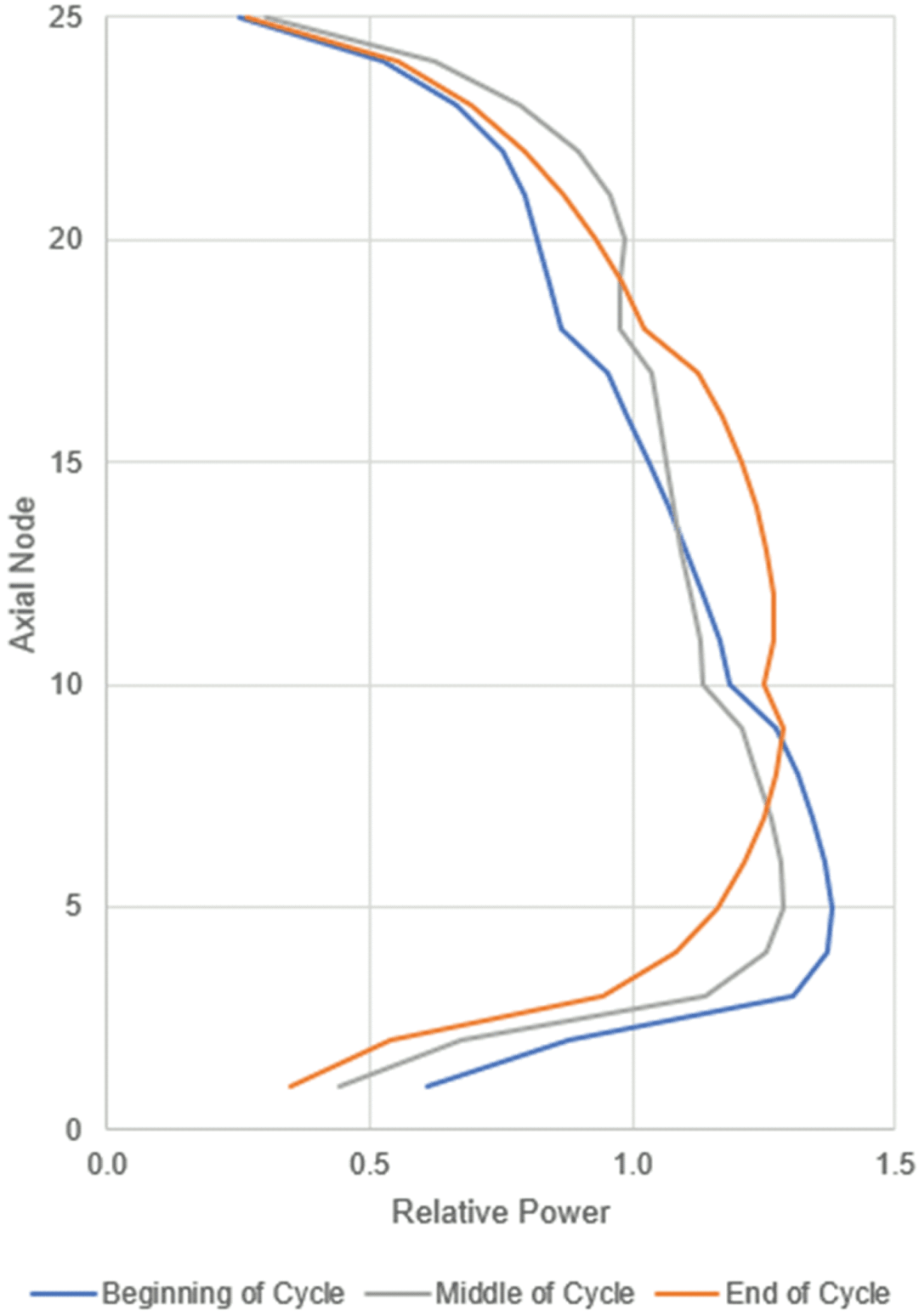


Figure 4.3-5 Axial Power Shapes

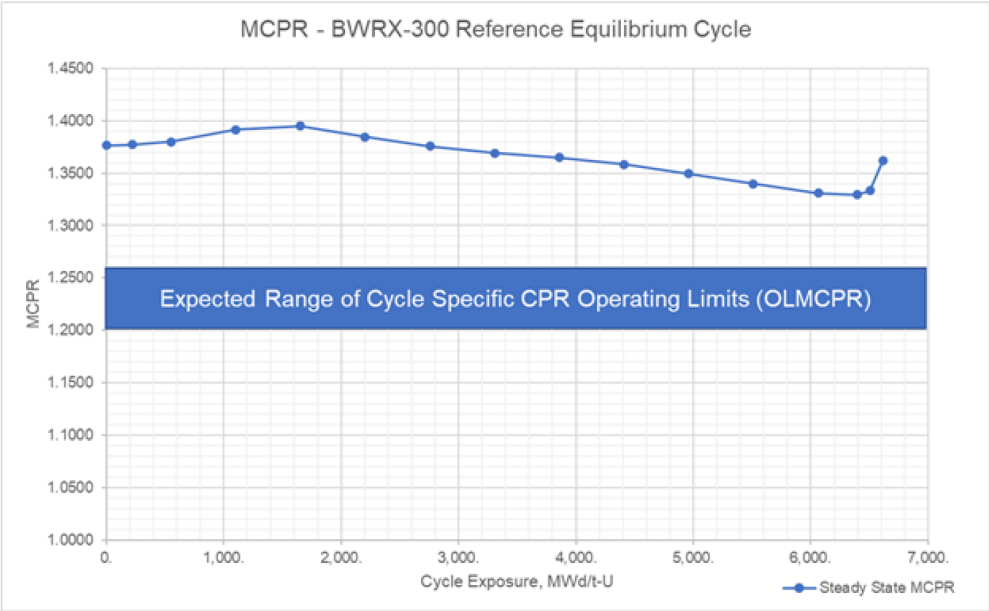


Figure 4.3-6 MCPR vs. Cycle Exposure for the Reference Equilibrium Cycle

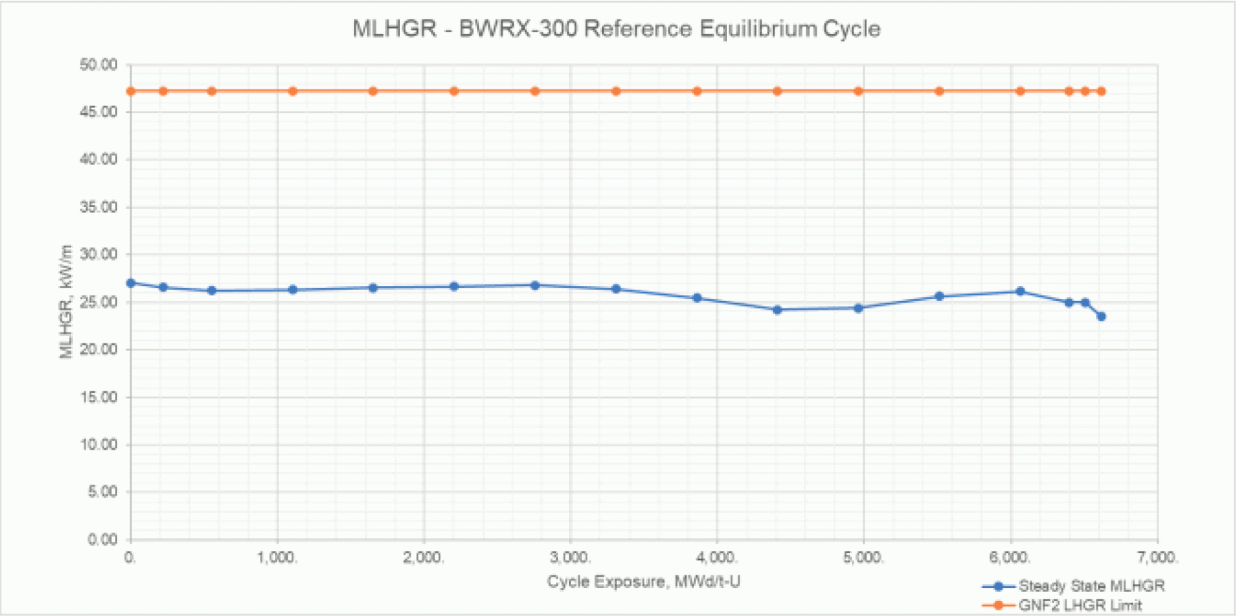


Figure 4.3-7 Reference Equilibrium Cycle Thermal Margin – Maximum Linear Heat Generation Rate (MLHGR)

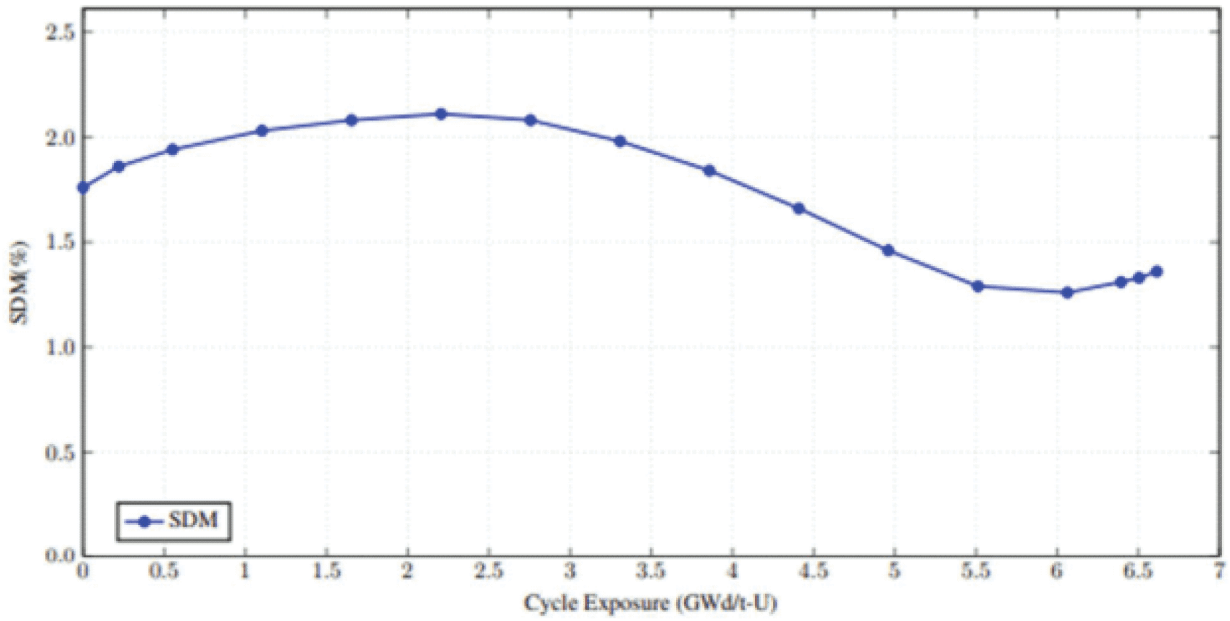


Figure 4.3-8 Shutdown Margin vs Cycle Exposure

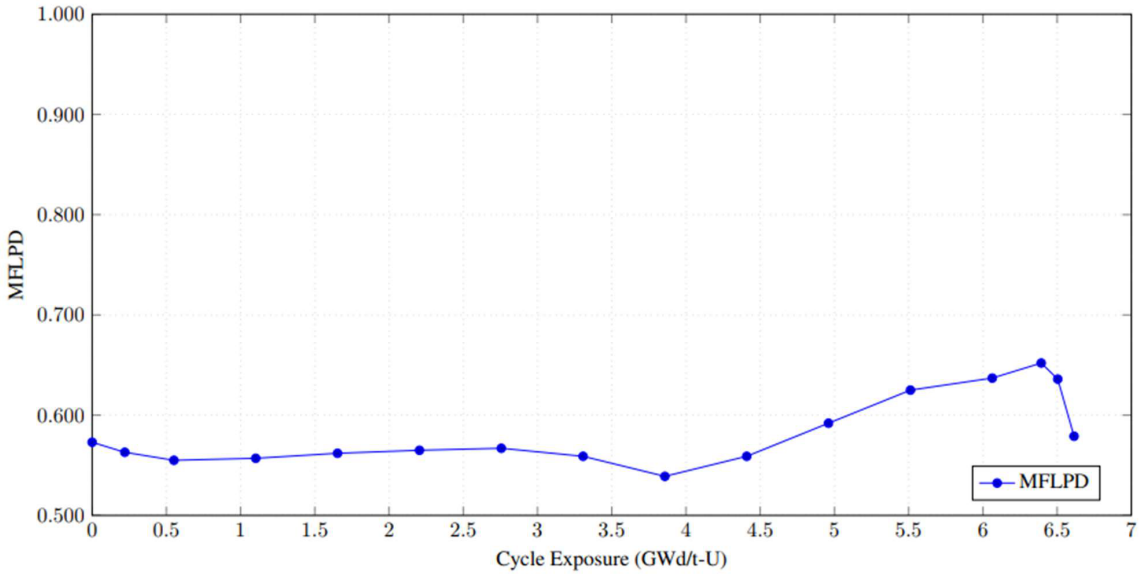


Figure 6: MFLPD vs. Cycle Exposure

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 $MCPR_{99.9\%}$ provides assurance 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 fuel bundle critical power performance methodology was originally described in NEDE-10958-A, “General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application,” (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 (GESTAR II) (Supplement for United States),” (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 $MCPR_{95/95}$, establishes 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 stage 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 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, Nuclear 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 $MCPR_{99.9\%}$

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," GE-Hitachi Nuclear Energy Americas, LLC, Revision 4, January 2008.

4.4-2 NEDC-34039P, "BWRX-300 GNF2 Steady State Nuclear Methods: TGBLA06/PANAC11 Application Methodology," GE-Hitachi Nuclear Energy Americas, LLC, Revision 0, April 2024.

4.4-3 NEDC-34040P, "BWRX-300 GNF2 Fuel Assembly Pressure Drop Characteristics," GE-Hitachi Nuclear Energy Americas, LLC, Revision 0, April 2024.

4.4-4 NEDE-10958-A, "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application," General Electric Company, January 1977.

4.4-5 NEDC-33292P, "GEXL17 Correlation for GNF2 Fuel," Global Nuclear Fuel - Americas, LLC, Revision 3, 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)," Global Nuclear Fuel - Americas, LLC, Revision 31, November 2020.

4.4-8 NEDC-34043P, "BWRX-300 TRACG Application," GE-Hitachi Nuclear Energy Americas, LLC, Revision 1, May 2025.

4.4-9 NEDC-33256P-A, "The PRIME Model for Analysis of Fuel Rod Thermal - Mechanical Performance Part 1 - Technical Bases," Global Nuclear Fuel - Americas, LLC, Revision 2, October 2021.

4.4-10 NEDC-34044P, "BWRX-300 GNF2 Equilibrium 12 Month Cycle Nuclear Design Report," GE-Hitachi Nuclear Energy Americas, LLC, Revision 0, April 2024.

4.4-11 NEDC-33912P-A, "BWRX-300 Reactivity Control," GE-Hitachi Nuclear Energy Americas, LLC, Revision 2, 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 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

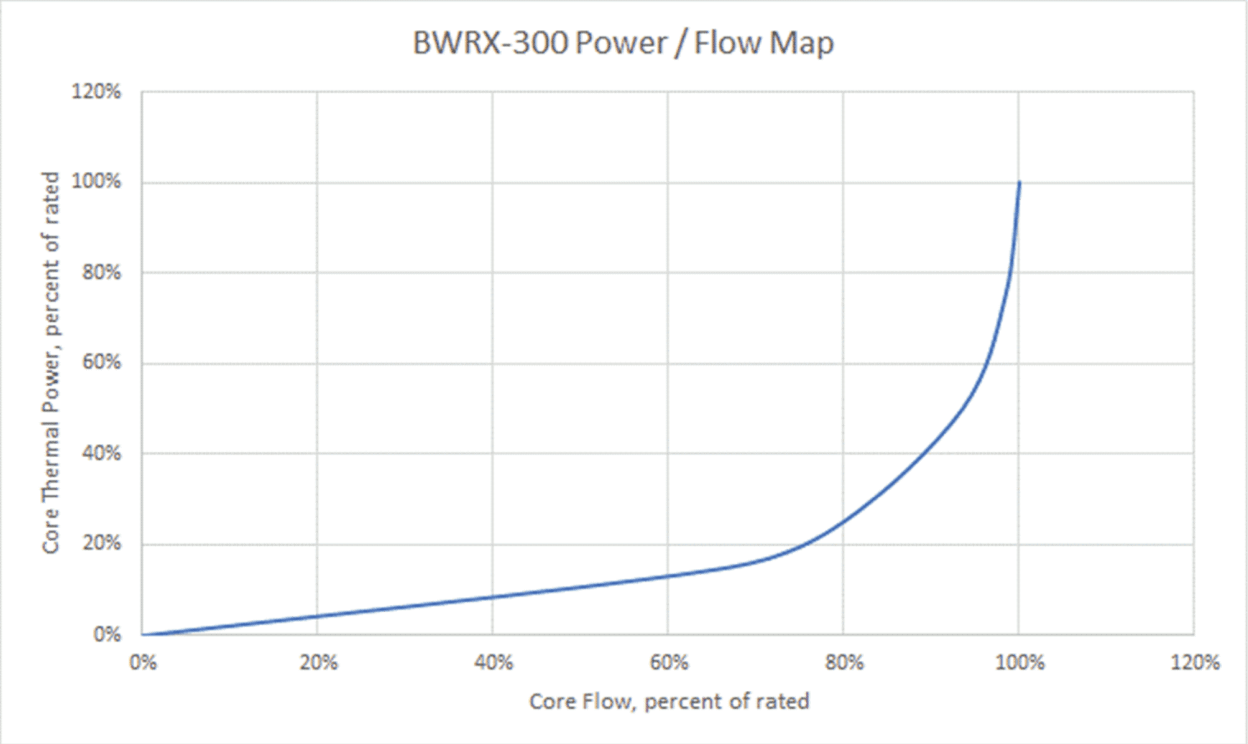


Figure 4.4-1 BWRX-300 Power / Flow Map

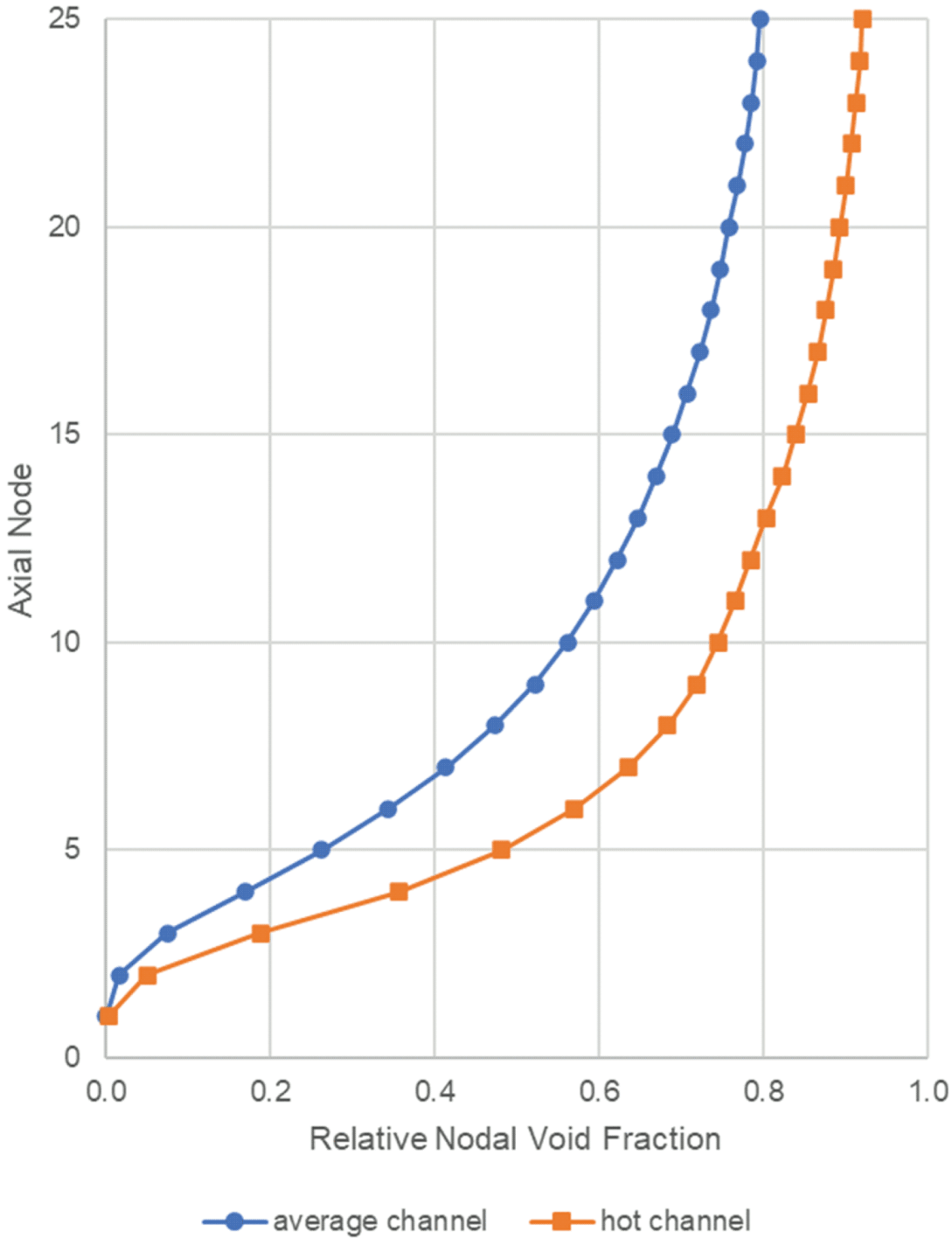


Figure 4.4-2 Relative Void Fraction for Analyzed Node (Hot Channel and Average Channel)

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 Reactor Coolant Pressure Boundary (RCPB) components of the CRD, consist of the FMCRD middle flange including the ball check valve, and the lower component housing, which encloses 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 (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 CRDM 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 CRDM are equivalent to those given in Parts A, B and D 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. Cold work concerns 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 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 FMCRD components and is intended to support design requirements. The FMCRD material specification is used with the design specification required by ASME BPVC, Section III, NCA-2021.

When applicable, such as for ASME Class 1 components, ASME BPVC, Section III, NB-2300 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 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. 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 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 the 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 GVH quality assurance 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 quality assurance 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 materials comply with the following GDC, which has plant-wide applicability, as discussed in Section 3.1:

GDC 1 – Quality standards and records.

The following GDC 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. Contamination protection and cleaning of austenitic stainless steel is described in Subsection 5.2.3. Other reactor materials used 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 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 assembly, top guide, core support structures, CRD housings, and incore 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 ASME BPVC, 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 Procedures; Welders; Brazers; and Welding, Brazing, and Fusing Operators,” (Reference 4.5-4), and when applicable, ASME BPVC, 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 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 BPVC, Section III. Ultrasonic examination shall be conducted in accordance with ASME BPVC, Section V, Article 4 and Article 5 as required and modified by ASME BPVC, Section III.

Liquid Penetrant/Magnetic Particle Examinations

All liquid penetrant and magnetic particle examination shall be conducted in accordance with ASME BPVC, Section V, Article 6 and Article 7, respectively, as required by ASME BPVC, Section III.

Visual Examination

Visual examinations shall be conducted in accordance with the ASME BPVC, 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 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 stainless steel materials, additional service temperature tensile tests are required to verify that these materials still meet the strength requirements at service temperature per ASME BPVC, Section II (Reference 4.5-2), Section III (Reference 4.5-1) and Section XI,

Division 1, “Rules for Inspection and Testing of Components of Light-Water-Cooled Plants,” (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). 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 system and the reactor internal structural materials.

Regulatory Requirements

The reactor internal structural materials are designed and fabricated 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 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 ASME BPVC, Section III (Reference 4.5-1) and Section IX (Reference 4.5-4), 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 the ASME BPVC, Section II (Reference 4.5-2) and Section III (Reference 4.5-1), 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, Division 1, “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 Procedures; Welders; Brazers; and Welding, Brazing, and Fusing Operators,” American Society of Mechanical Engineers, 2021.

4.5-5 ASME BPVC, Section XI, Division 1, “Rules for Inspection and Testing of Components of Light-Water-Cooled Plants,” American Society of Mechanical Engineers.

4.5-6 ASTM A262-15(2021), “Standard Practices for Detecting Susceptibility to Intergranular Attack in Austenitic Stainless Steels,” ASTM International, 2015 (R2021).

Table 4.5-1 Reactor Internal Components and Materials
(Sheet 1 of 2)

Component(s)	Form	Material Type (Note 1)	ASME Specification
Steam Dryer Assembly	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
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

Table 4.5-1 Reactor Internal Components and Materials
(Sheet 2 of 2)

Component(s)	Form	Material Type (Note 1)	ASME Specification
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 system
- 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 CRD 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 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 and System Classification

The Safety Classifications, Quality Groups, and Seismic Categories of the CRD system 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 electro-mechanical 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 diagram of the HCU is provided in Figure 4.6-8. The major components of each HCU are described 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 provide assurance 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 simplified diagram of the CRD hydraulic subsystem 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. 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 operations, the CRD hydraulic subsystem is in normal operation as described below. During power operations, 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 operations, 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 hydraulic subsystem 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 hydraulic subsystem 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 GDC and PDC 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 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 PDC 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.

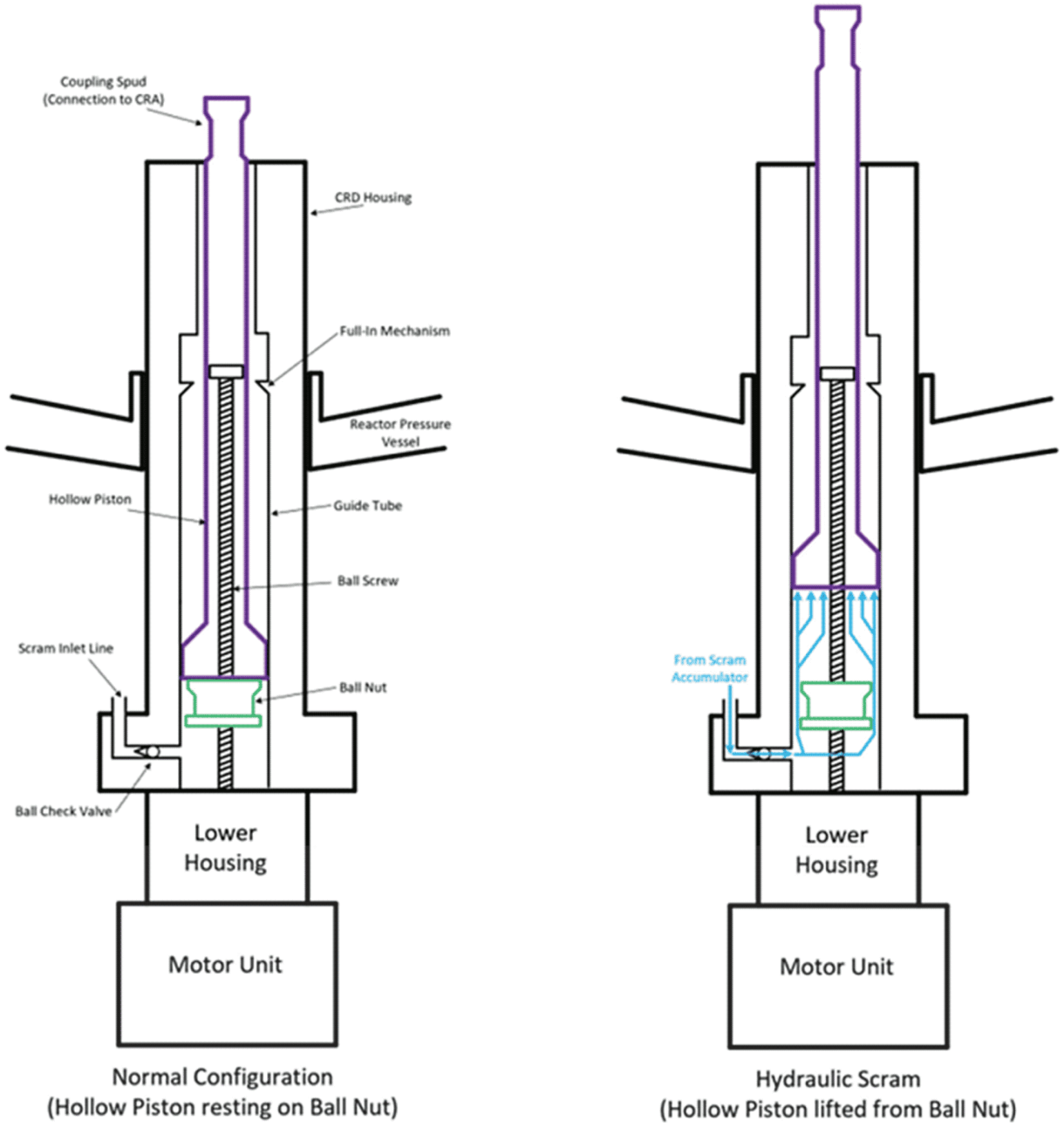


Figure 4.6-1 BWRX-300 Simplified View of FMCRD with Hydraulic Scram

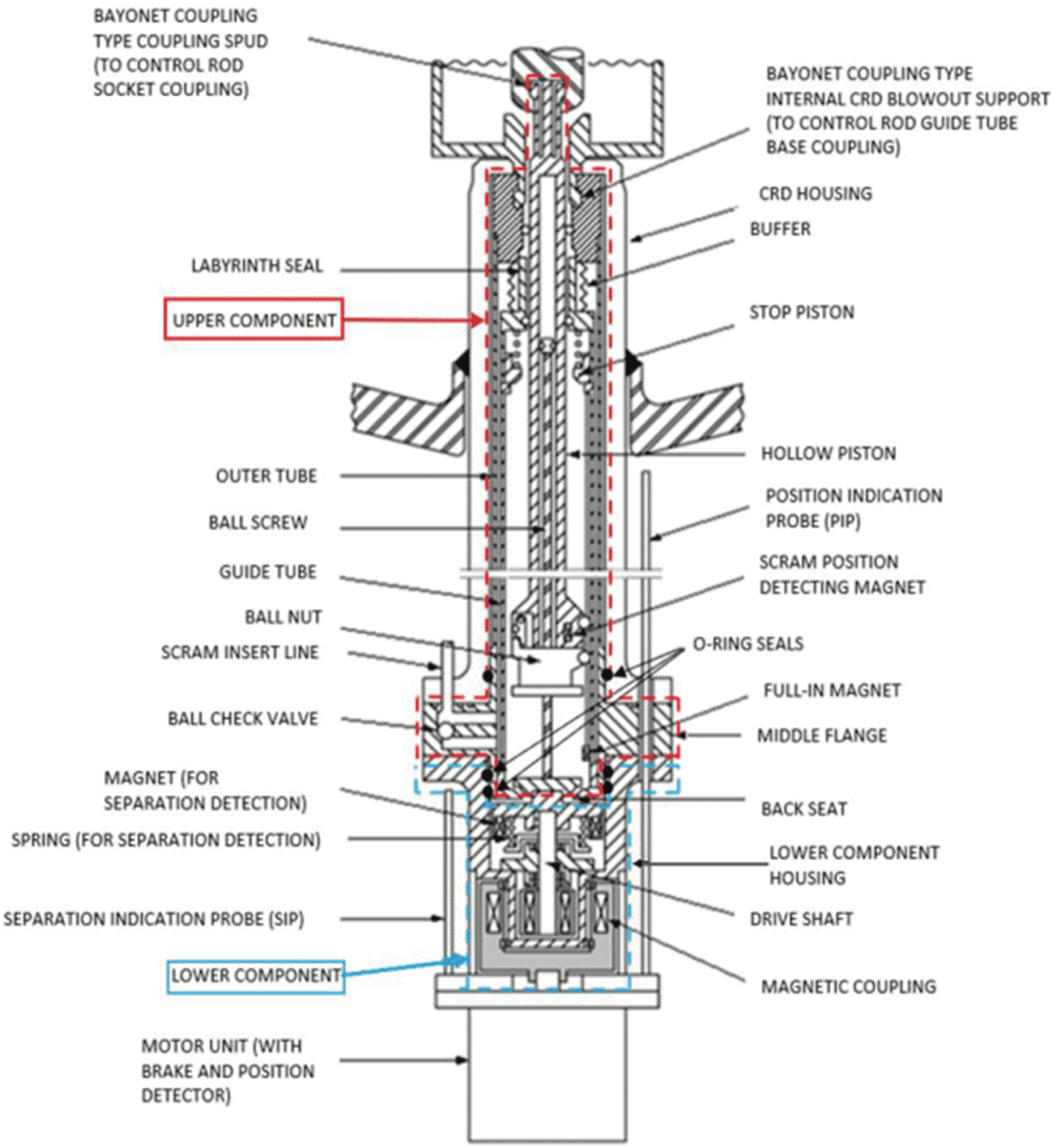


Figure 4.6-2 Schematic Representation of Magnet Coupling Fine Motion Control Rod Drive
Note:

The red and blue dashed lines indicate the upper and lower components that are bolted together at the middle flange.

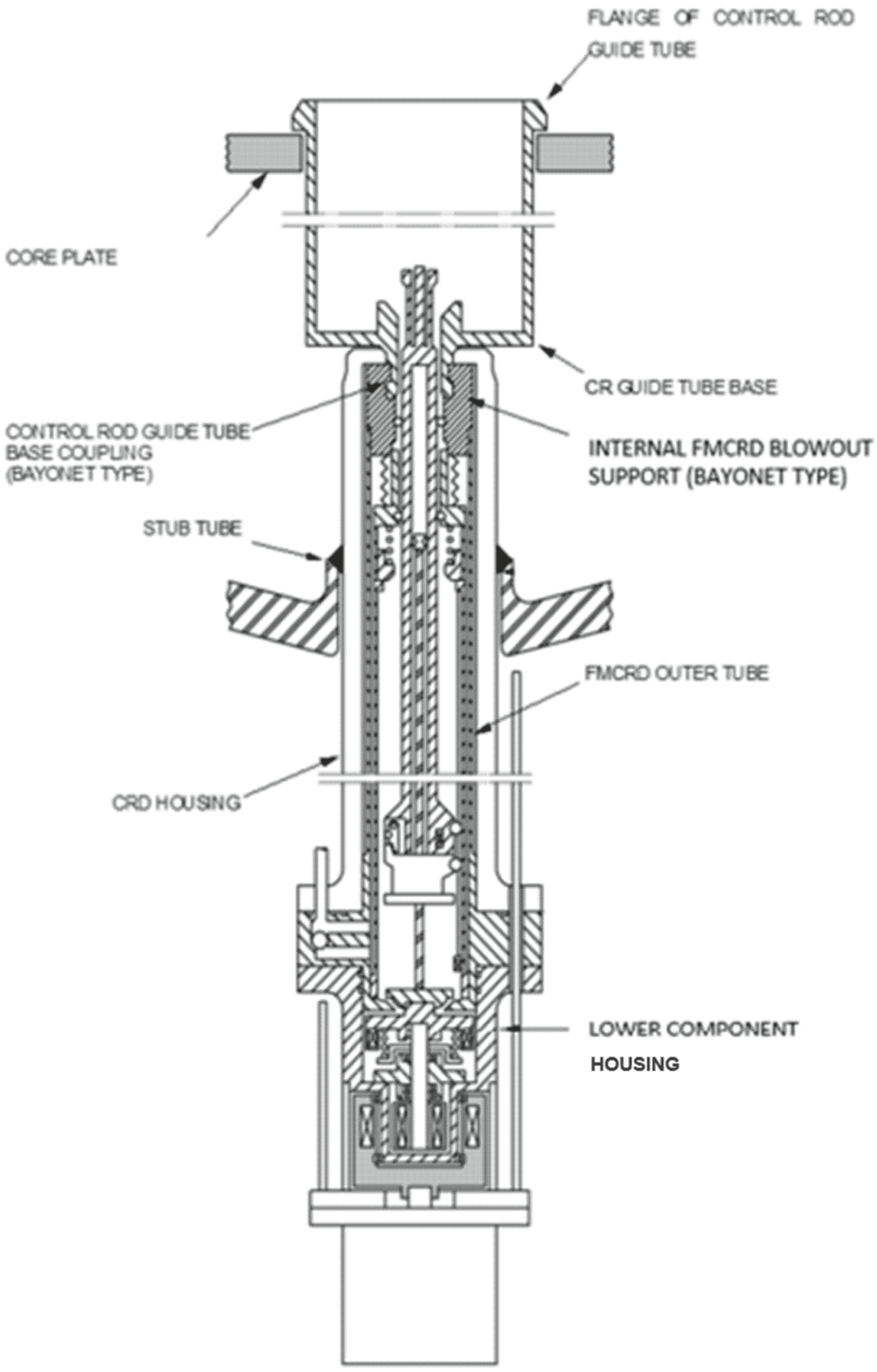


Figure 4.6-3 Representative Internal Control Rod Drive Blowout Support Schematic

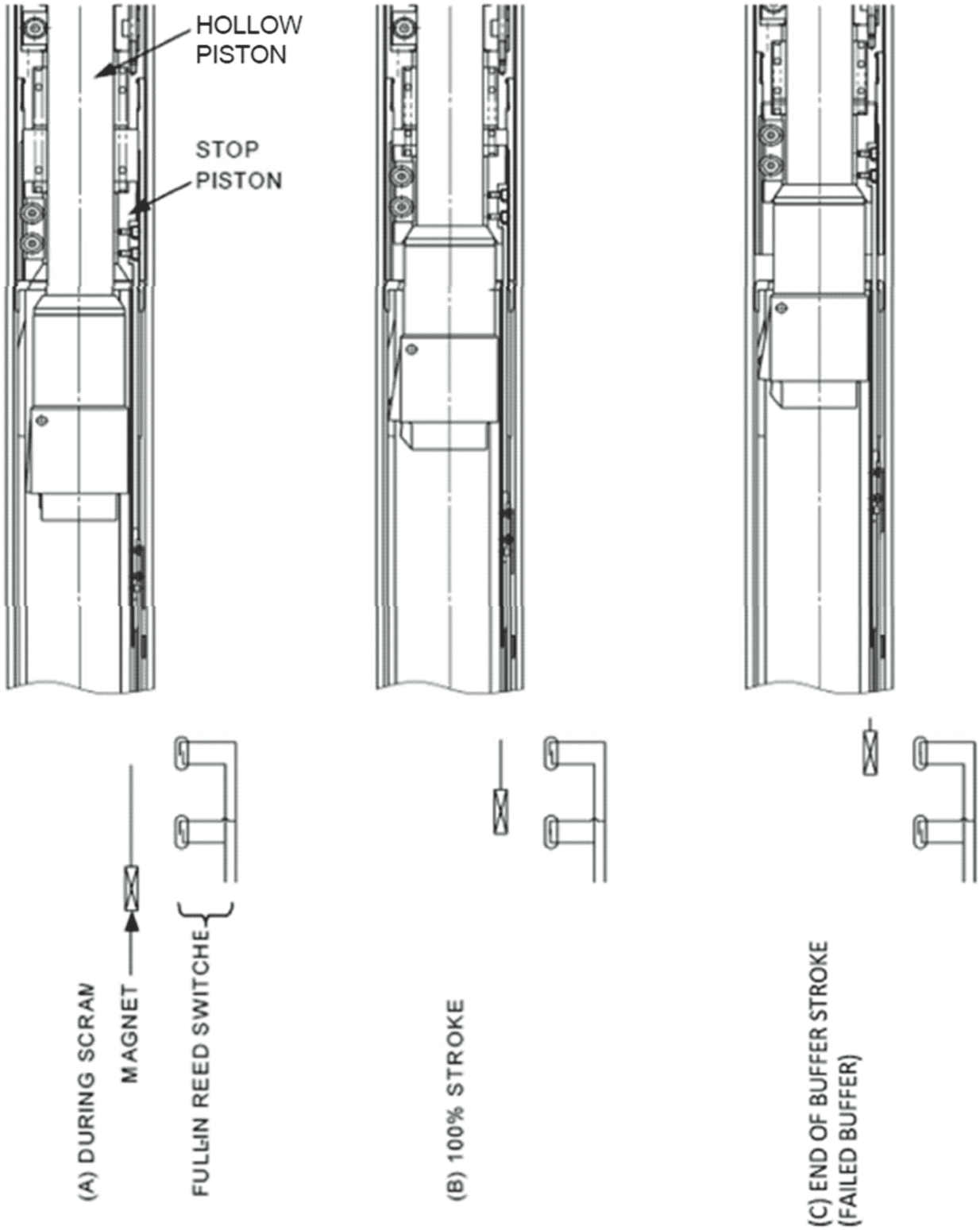


Figure 4.6-4 Representative Continuous Full-in Indicating Device

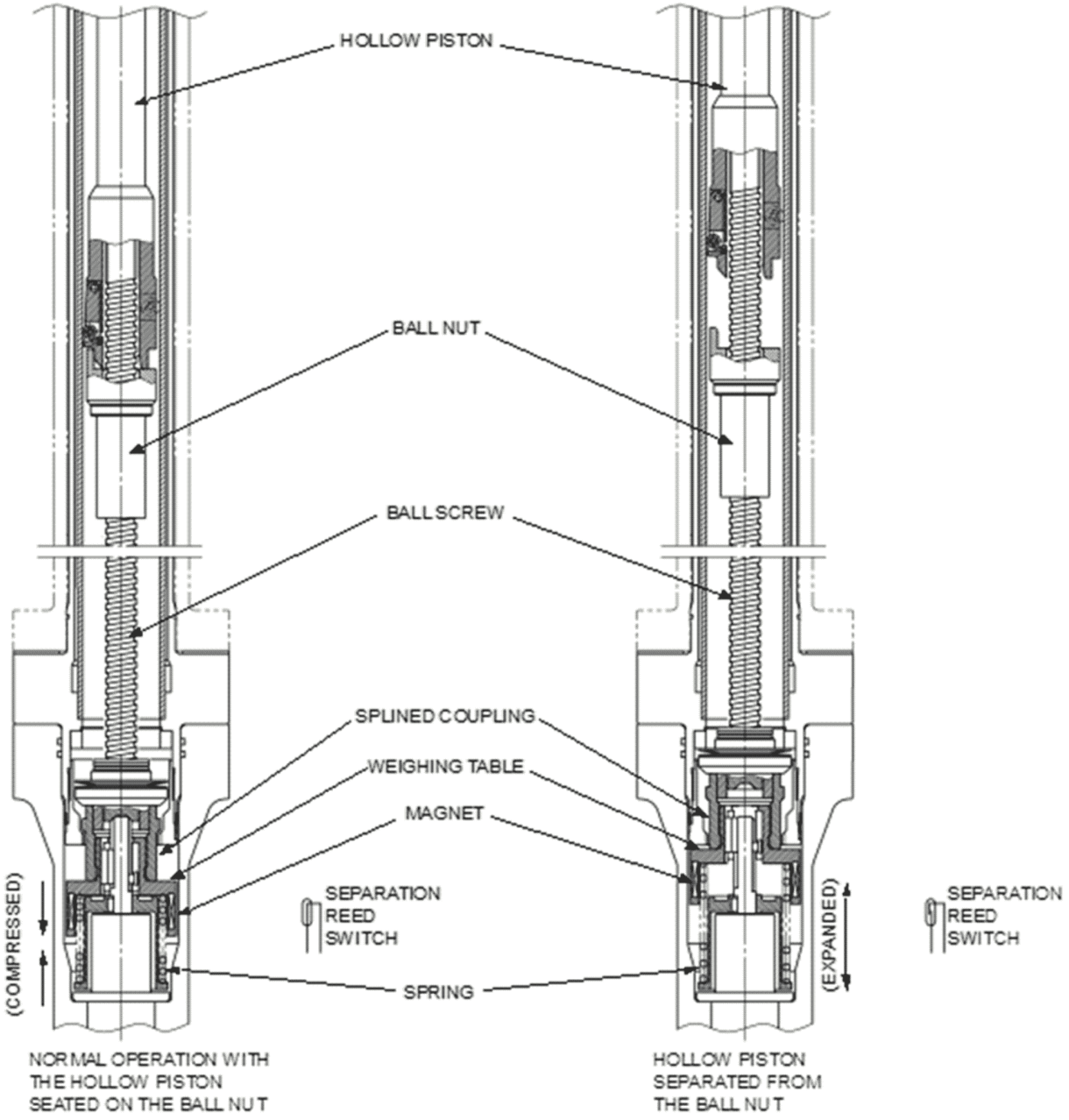


Figure 4.6-5 Representative Control Rod Separation Detection

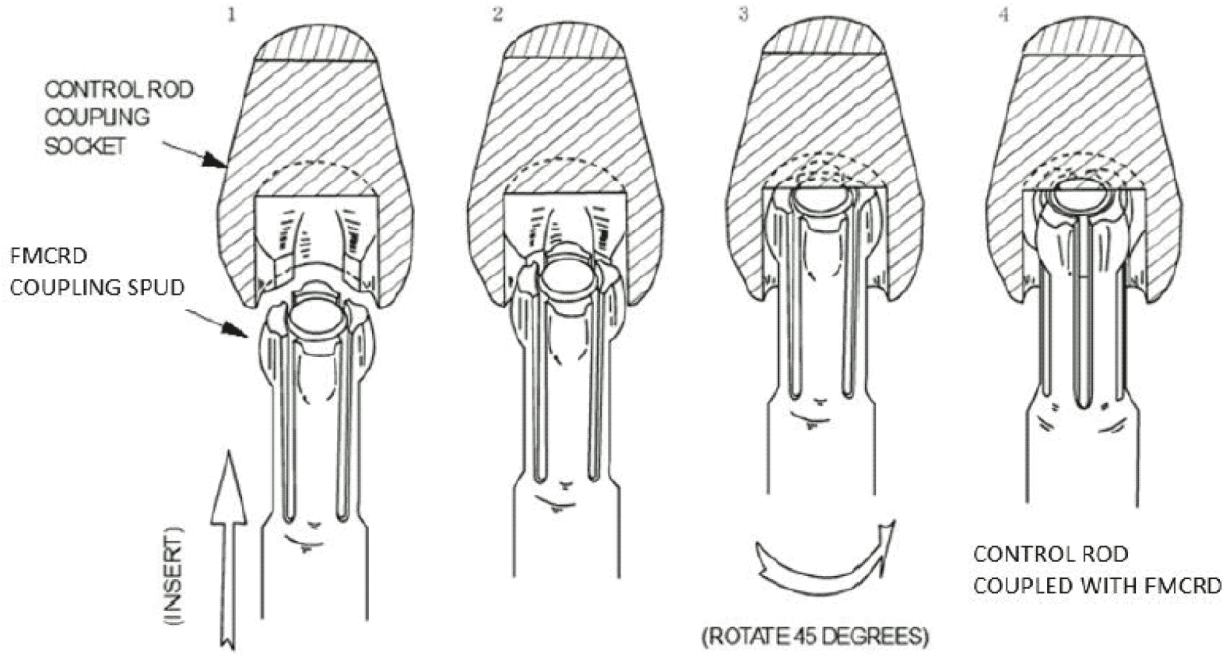


Figure 4.6-6 Representative Control Rod Coupling Socket to FMCRD Coupling Spud

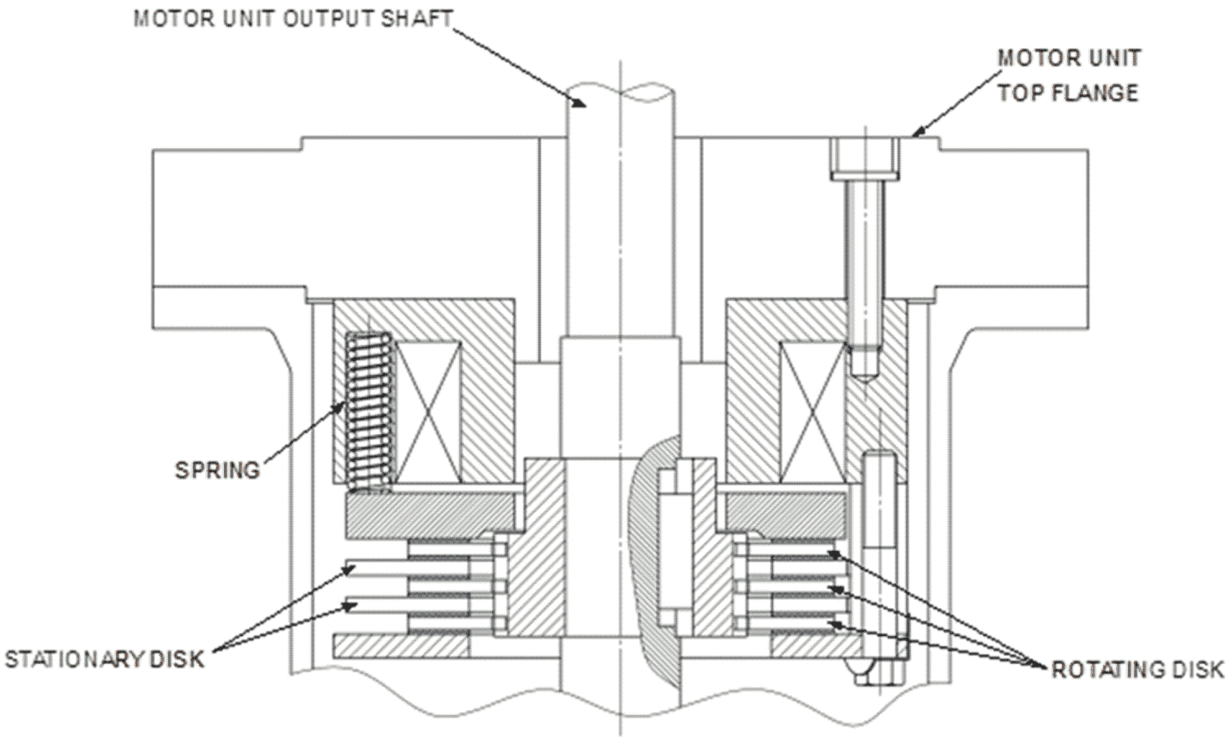


Figure 4.6-7 Representative Fine Motion Control Rod Drive Electro-Mechanical Brake

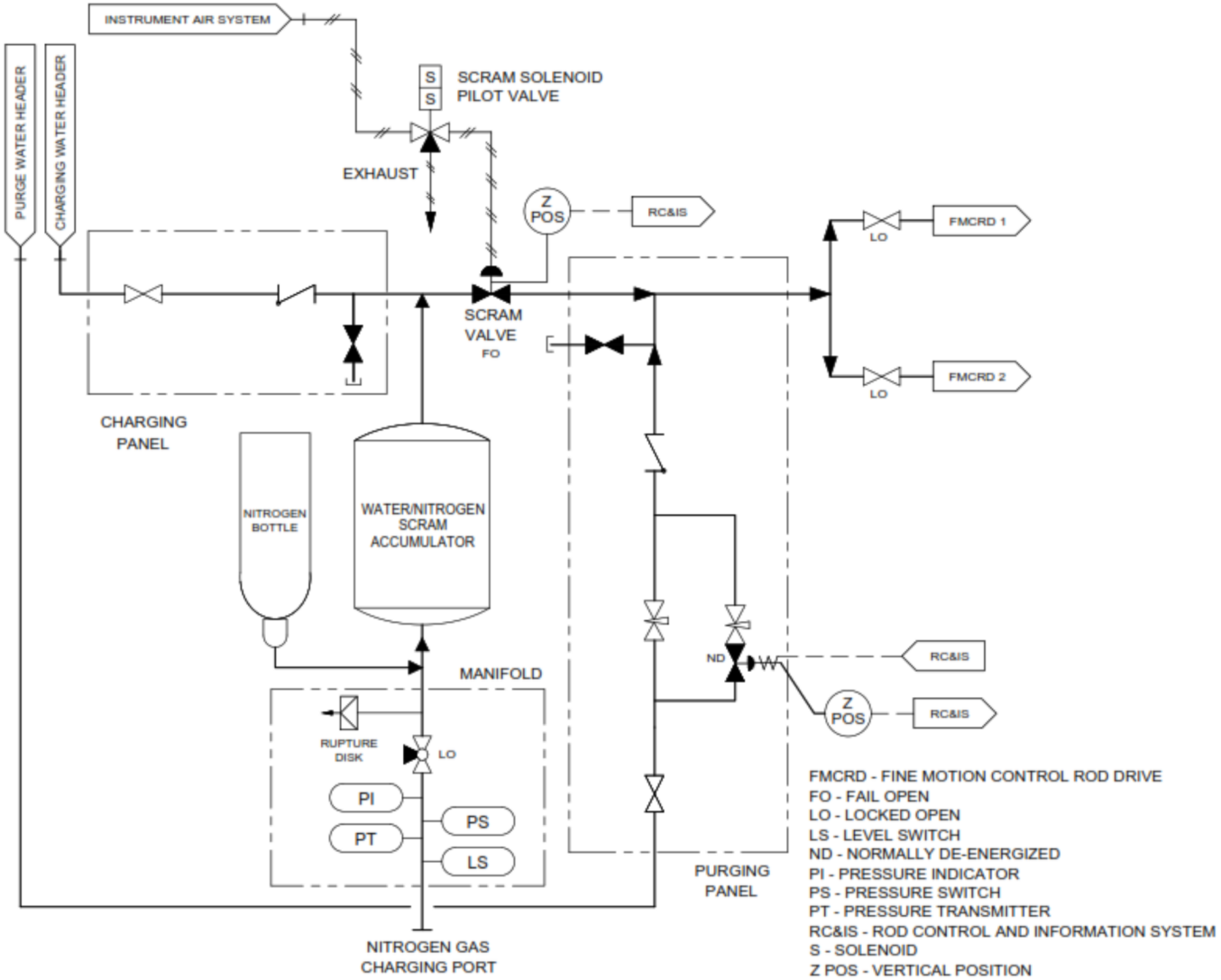


Figure 4.6-8 Hydraulic Control Unit Simplified Diagram

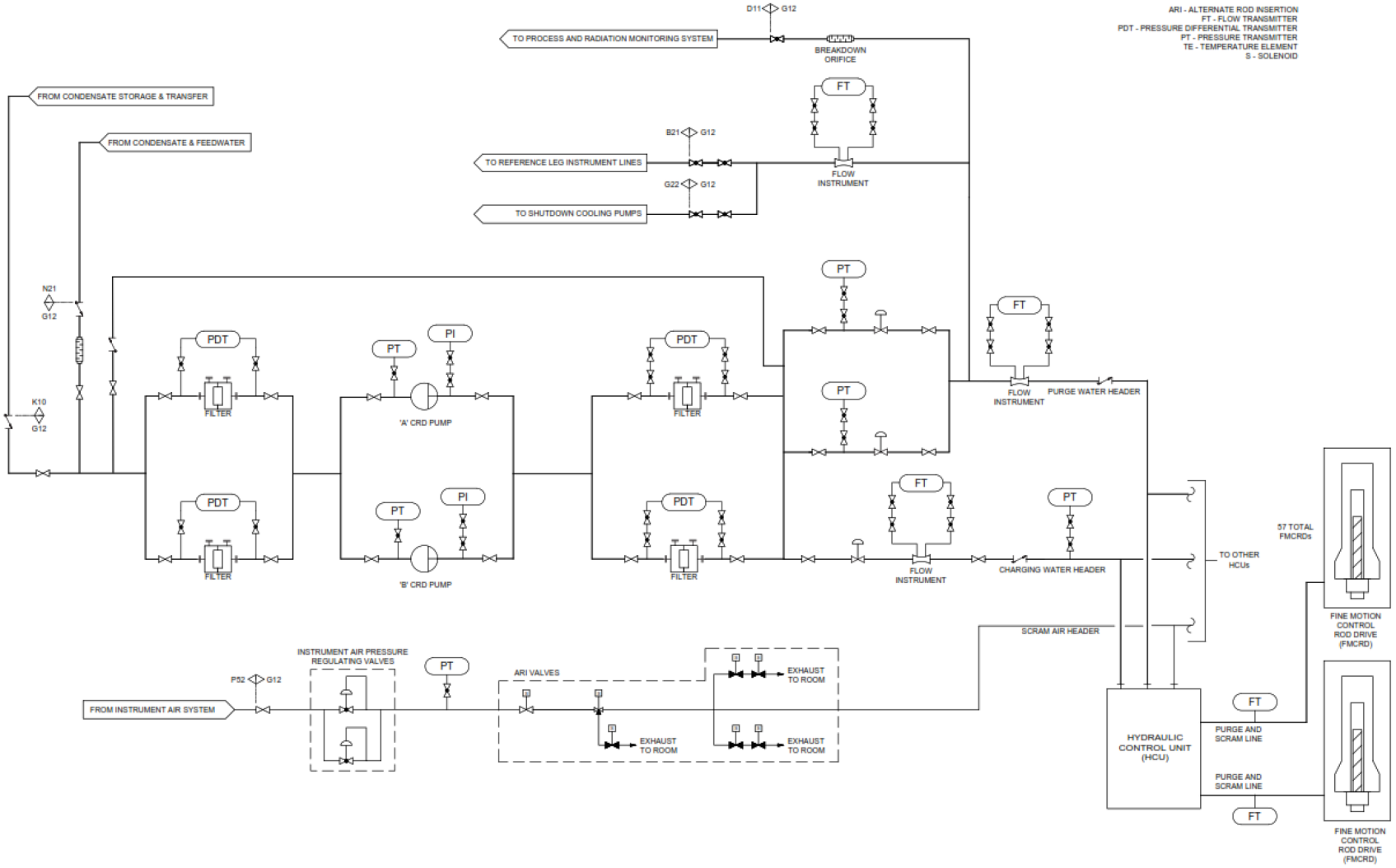


Figure 4.6-9 Control Rod Drive Hydraulic Subsystem Simplified Diagram

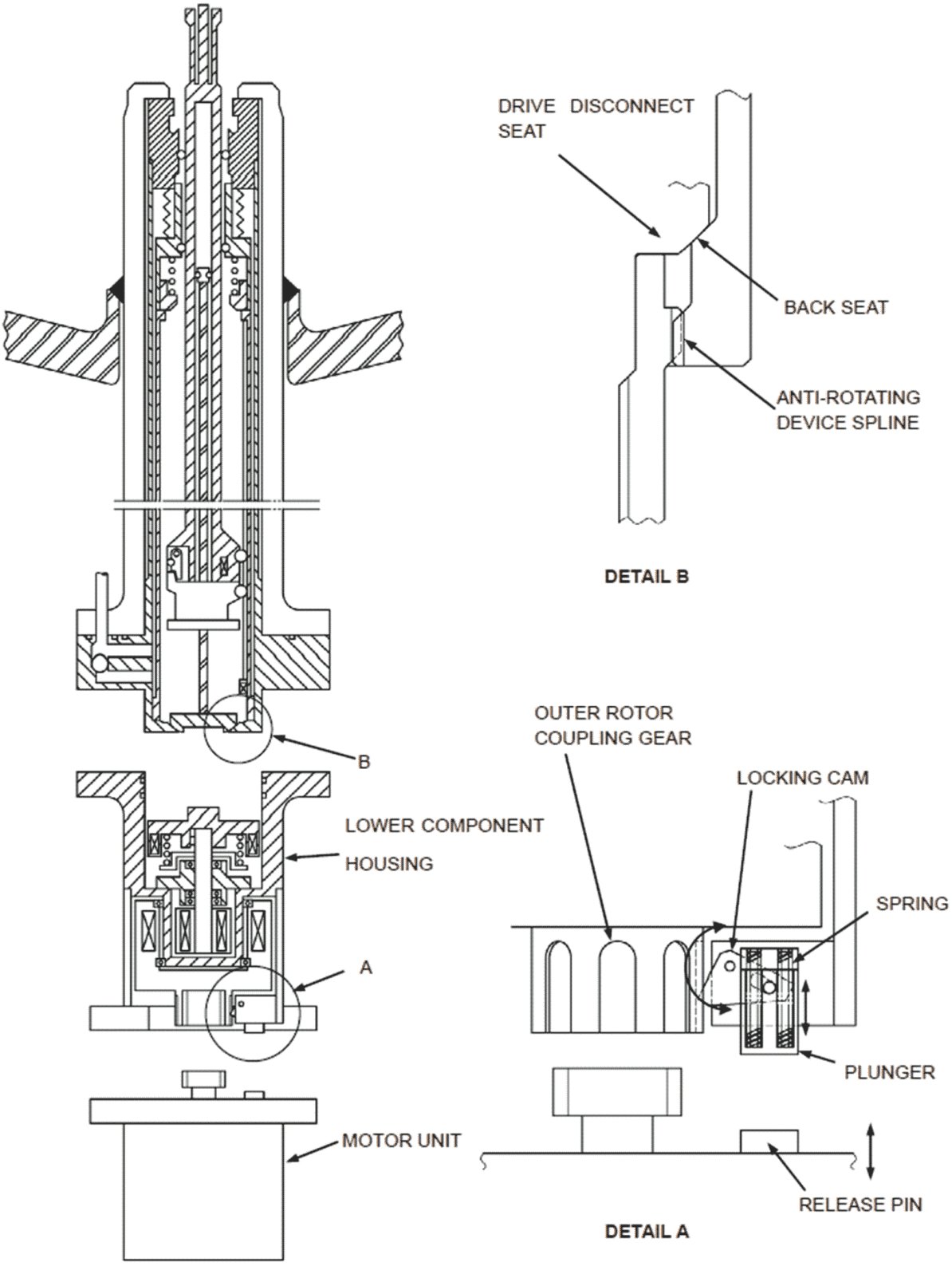


Figure 4.6-10 Representative Fine Motion Control Rod Drive Anti-Rotation Device

APPENDIX 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 GDC 12 was previously addressed in the USNRC-approved 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-7) 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 report. NEDC-34270P 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. A limiting core-wide dominance evaluation 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-7). 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-3), 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-4).

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-5). 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-6).

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,” GE-Hitachi Nuclear Energy Americas, LLC, Revision 2, February 2023.

4A-2 NEDC-33922P-A, “BWRX-300 Containment Evaluation Method,” GE-Hitachi Nuclear Energy Americas, LLC, Revision 3, June 2022.

4A-3 NEDC-34043P, “BWRX-300 TRACG Application,” GE-Hitachi Nuclear Energy Americas, LLC, Revision 1, May 2025.

4A-4 NEDE-32176P, “TRACG Model Description,” GE-Hitachi Nuclear Energy Americas, LLC, Revision 4, January 2008.

4A-5 NEDE-32177P, “TRACG Qualification,” GE-Hitachi Nuclear Energy Americas, LLC, Revision 3, August 2007.

4A-6 NEDC-33083P-A, “TRACG Application for ESBWR,” GE Nuclear Energy, Revision 1, March 2005.

4A-7 NEDC-34270P, “BWRX-300 Stability Analysis,” GE Vernova Hitachi Nuclear Energy Americas, LLC, Revision 1, March 2026.

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LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
ALARA	As Low As Reasonably Achievable
ANSI	American National Standards Institute
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
EPRI	Electric Power Research Institute
FAC	Flow Accelerated Corrosion
FE	Flow Element
FPC	Fuel Pool Cooling and Cleanup System
FSAR	Final Safety Analysis Report
FT	Flow Transmitter
FW	Feedwater
FWCIV	Feedwater Containment Isolation Valve
FWRIV	Feedwater Reactor Isolation Valve
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

LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
ICS	Isolation Condenser System
ICSRIV	Isolation Condenser Supply Reactor Isolation Valve
IGSCC	Intergranular Stress Corrosion Cracking
ISI	Inservice Inspection
IST	Inservice Testing
LOCA	Loss-of-Coolant Accident
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
MSRIV	Main Steam Reactor Isolation Valve
NBS	Nuclear Boiler System
NDE	Nondestructive Examination
NPSH	Net Positive Suction Head
OBE	Operating Basis Earthquake
OLNC	On-Line NobleChem™
OPEX	Operating Experience
PCS	Primary Containment System
PCW	Plant Cooling Water System
PDC	Principal Design Criteria
PRM	Process Radiation Monitoring Subsystem
PS	Process Sampling Subsystem
PSI	Preservice Inspection
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
RWB	Radwaste Building
SC1	Safety Class 1
SCC	Stress Corrosion Cracking
SDC	Shutdown Cooling System

LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
SSCs	Structures, Systems, and Components
SSE	Safe Shutdown Earthquake
TB	Turbine Building
USNRC	U.S. Nuclear Regulatory Commission

LIST OF EFFECTIVE SECTIONS

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CHAPTER 5 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 discussed in Section 4.4.

The Reactor Coolant System (RCS) is comprised of the Safety Class 1 (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 RCS is illustrated on 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 illustrated on 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 to the RPV
- The Boron Injection System (BIS) which interfaces with ICS for a 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), i.e., Containment. The ICS is discussed in Section 6.3. The Shutdown Cooling System (SDC), which interfaces with the ICS and CFS outside the PCS for the flow path from the RPV, is discussed in Subsection 5.4.7. The CUW is discussed in Subsection 5.4.8. The BIS is discussed in Subsection 9.3.10. System interfaces are illustrated on 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 (MSCIV), is illustrated on Figure 5.1-3. The MS subsystem and the CFS piping and components outboard of the CIVs are discussed in 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 illustrated on Figure 5.1-4. A schematic flow diagram of the RCS is illustrated on Figure 5.1-5 and depicts nominal pressures, temperatures, and flow rates for normal steady-state operating conditions at rated power.

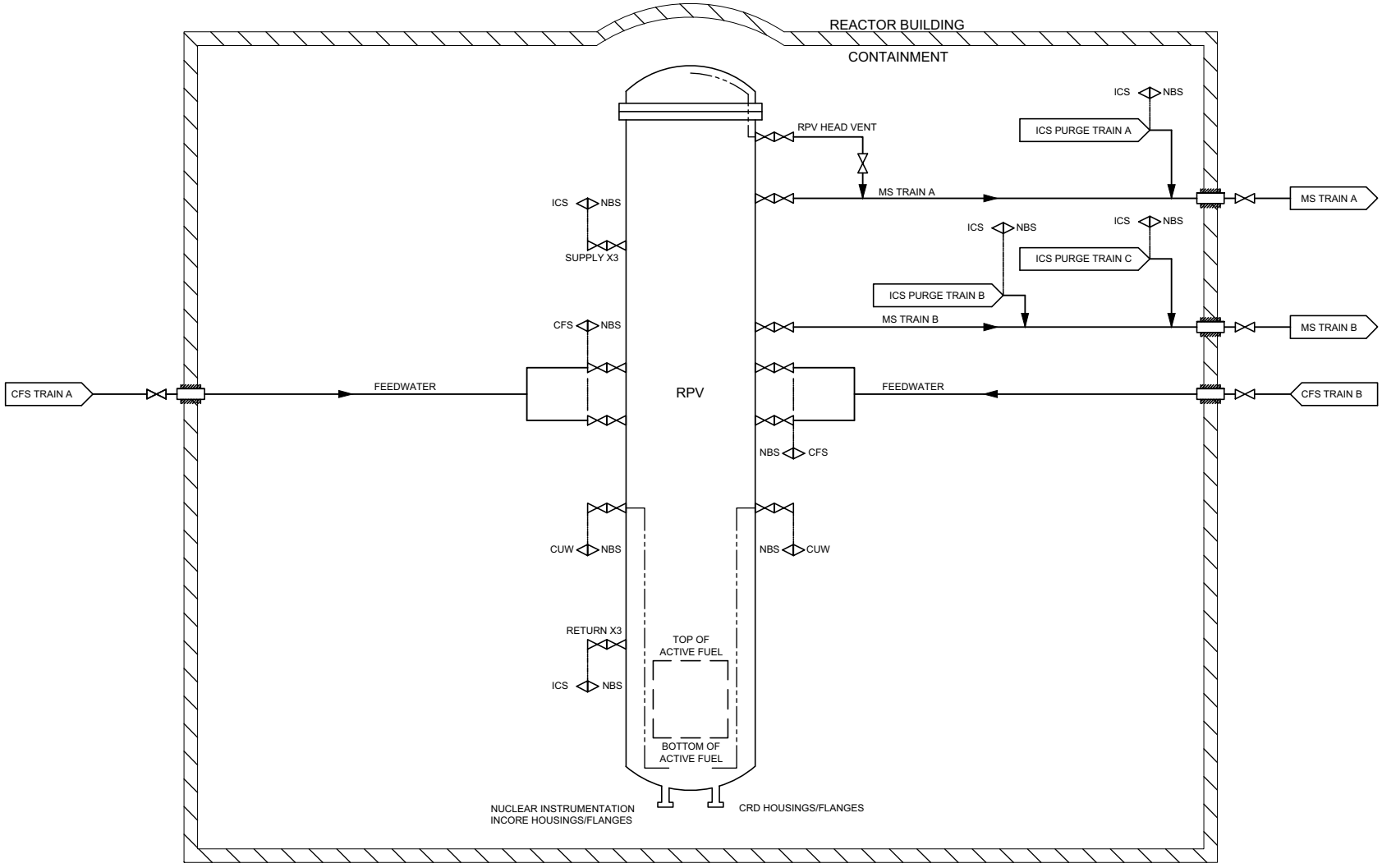
5.1.2 Simplified Diagrams

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.



Notes for this figure are provided on the next page.

Figure 5.1-1 BWRX-300 Reactor Coolant System Simplified Diagram

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 are shown in phantom line.

2. Interfacing system breaks are provided to identify interfaces for the NBS, ICS, CFS, and CUW.

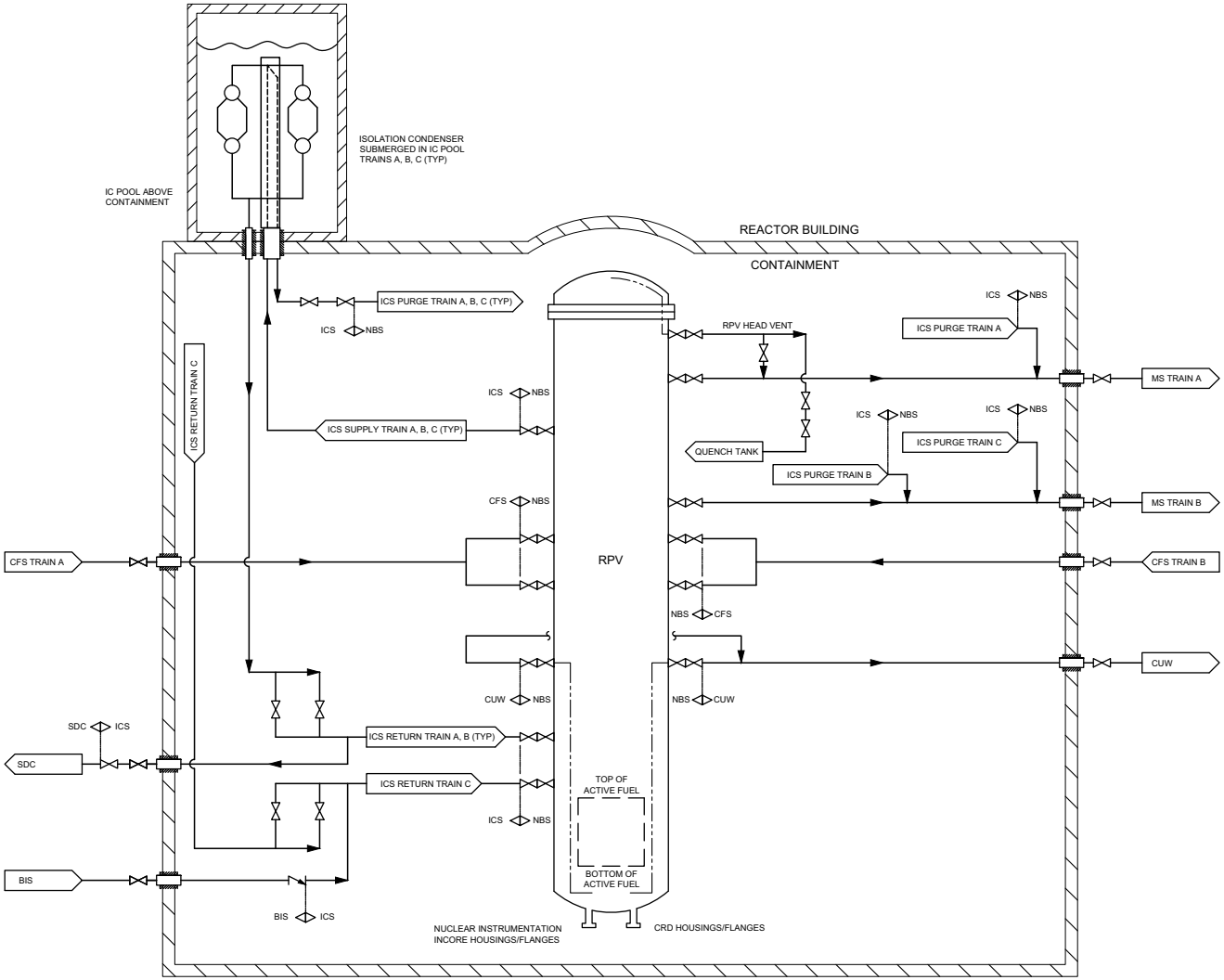
3. Reactor Isolation Valves (RIV) are two valves in a single body and directly bolted to RPV nozzles. RIVs are further discussed in Subsection 5.4.3.

4. All valves shown have power operators.

5. The RCS flow path starts at and includes the Feedwater Containment Isolation Valves (FWCIV), 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.



Notes for this figure are provided on the next page.

Figure 5.1-2 BWRX-300 Reactor Coolant Pressure Boundary Simplified Diagram

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.

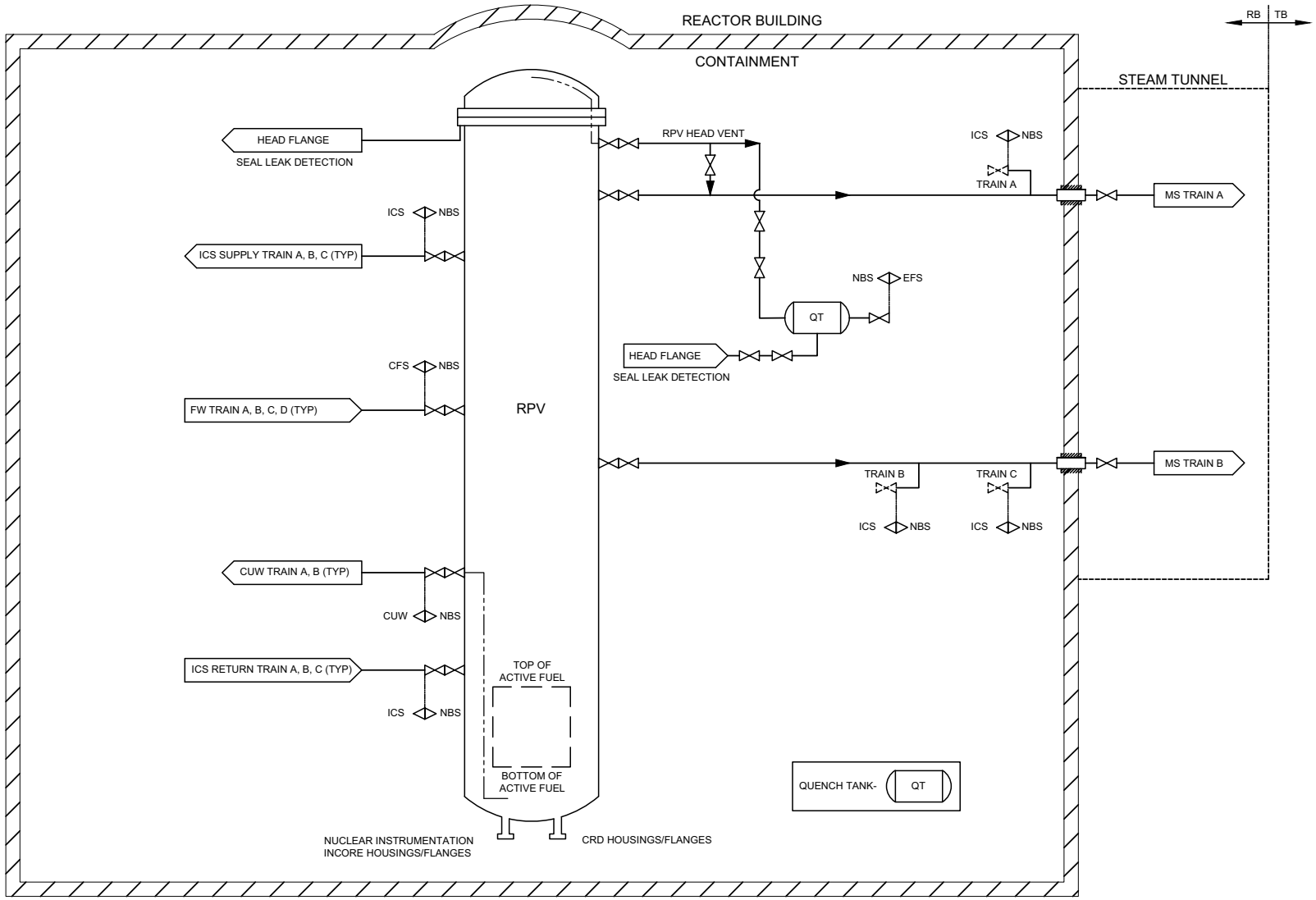
4. ICS consists of three independent trains (identified as Trains A, B, and C), each containing an IC.

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.



Notes for this figure are provided on the next page.

Figure 5.1-3 BWRX-300 Nuclear Boiler System Simplified Diagram

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.

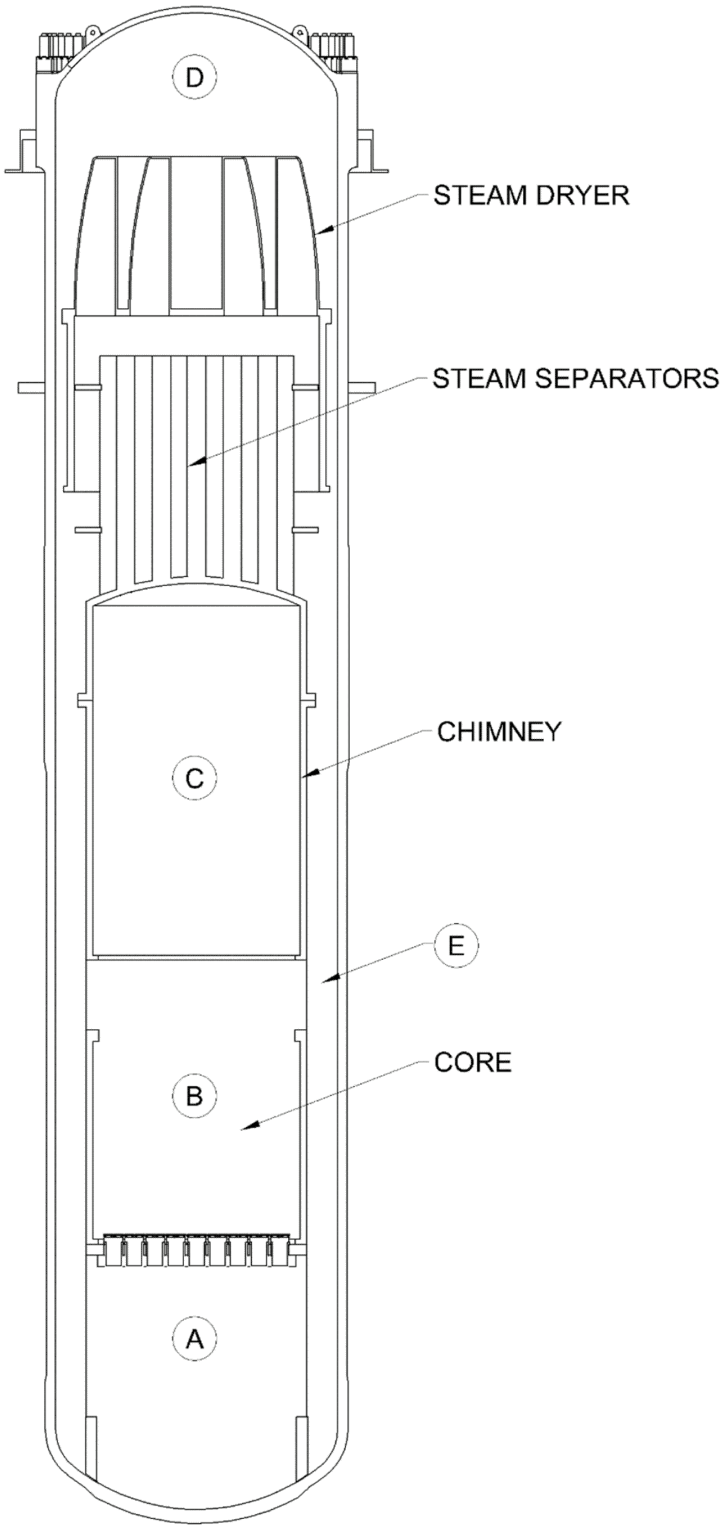
5. MSCIVs are located within the MS Tunnel shown in phantom line.

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.



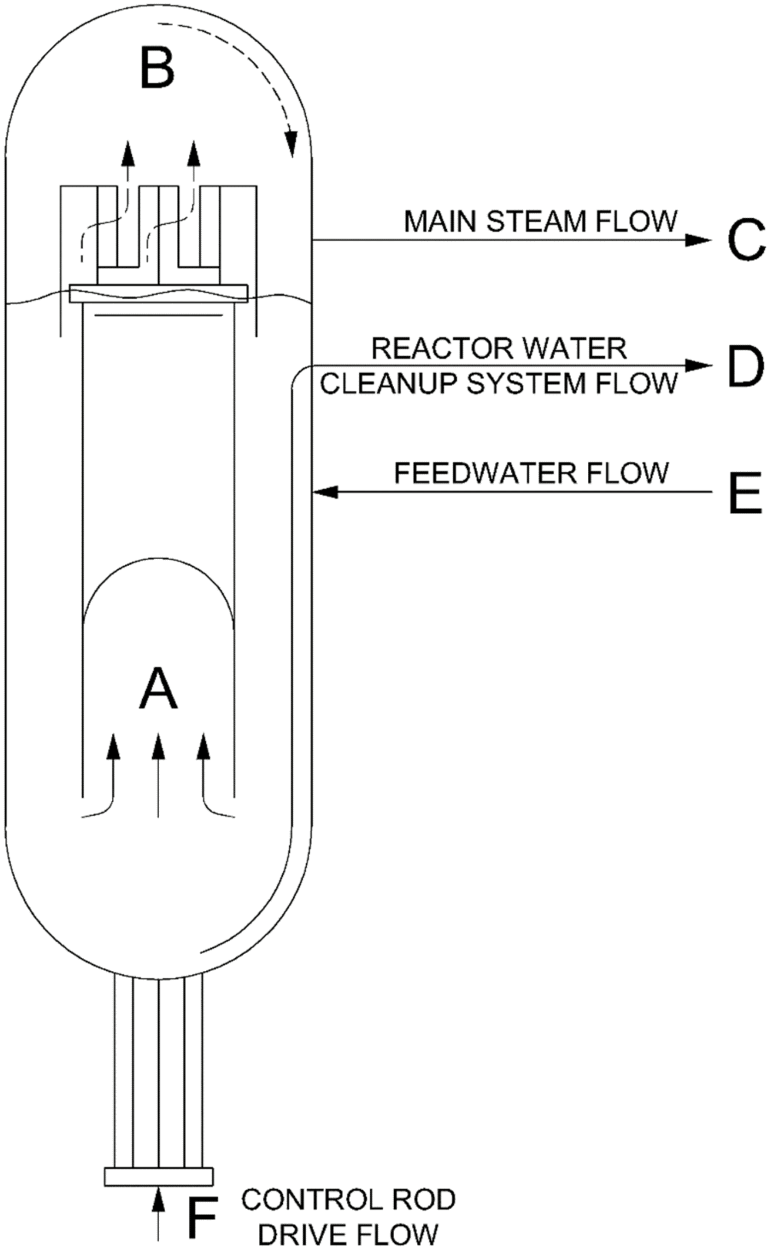
Note: Volume descriptions and their values at locations A, B, C, D, and E are provided on the next page.

Figure 5.1-4 BWRX-300 RPV Volumes at 100% Power

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

Note: Volume locations are identified on the schematic on the previous page.

Figure 5.1-4: BWRX-300 RPV Volumes at 100% Power



Note: Flow parameters and their values at locations A, B, C, D, E, and F are provided on the next page.

Figure 5.1-5 BWRX-300 RCS Schematic Flow Diagram at 100% Power

Location (**Note 1**)	Parameter (**Note 1**)	Nominal Value	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 (544.7) Saturation Temperature	^{o}C (^{o}F)
D (Reactor Water Cleanup System)	CUW Flow	5.1 (0.040)	kg/s (Mlbm/hr)
	CUW Temperature	276.6 (529.9)	^{o}C (^{o}F)
E (Feedwater)	CFS Flow	507.0 (4.029)	kg/s (Mlbm/hr)
	CFS Temperature	241.8 (467.3)	^{o}C (^{o}F)
F (Control Rod Drive (CRD) System)	CRD Flow	1.2 (0.0096)	kg/s (Mlbm/hr)
	CRD Temperature	43.4 (110.0)	^{o}C (^{o}F)

Notes:

1. Parameters are grouped by location and locations are identified on the schematic on the previous page.
2. MS Flow and MS Pressure are measured upstream of the Turbine Stop Valve and MS Temperature is measured at the turbine inlet.

Figure 5.1-5 BWRX-300 RCS Schematic Flow Diagram at 100% Power

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 illustrated on Figure 5.1-1.

The RCPB, as illustrated on 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 Criteria (GDC) 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 listed in 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 listed 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 ASME BPVC, Section XI, Division 1, “Rules for Inspection and Testing of Components of Light-Water-Cooled Plants,” (Reference 5.2-6), subject to the conditions identified and incorporated by reference in 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, will be addressed in the Final Safety Analysis Report (FSAR).

The RCPB is comprised of ASME Class 1 and ASME Class 2 components as discussed below.

The following RCPB components are Quality Group A, ASME Class 1 components designed to meet the requirements of ASME BPVC, Section III, 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 downstream of the CIV), and ICS to BIS interface piping, including CIVs
- 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, Subsection NCD:

- The piping from the RIVs up to and including each respective outboard CIV except for ICS compontents (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 to and from the ICs, as allowed by 10 CFR 50.55a(c)(2)(ii). The reactor water level instrumentation lines are classified as ASME Class 2 per 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 system, Quality Group B components are ASME Class 2 components and Quality Group C components are ASME Class 3 components.

Subsection 5.2.4 discusses compliance with the ISI and IST requirements of ASME BPVC, Section XI for ASME Class 1 components. Section 6.6 discusses 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 discussed 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 applicable 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 applicable revision of this RG will be addressed in the FSAR as discussed below.

ASME Code Cases provide alternatives to ASME BPVC provisions and are developed and approved by ASME. The RGs pursuant to 10 CFR 50.55a(a)(3) identify ASME 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, will 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, on the date no more than 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 will 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 (AOO) and Design Basis Accident event sequences where the RCPB 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 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, 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 discussed 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 discussed 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 discussed in Subsection 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 RCPB component, e.g., vessels, piping, and valves. The adequacy and suitability of the ferritic materials, austenitic 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, 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 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 considers 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-based alloy material selection. Further discussion of the materials associated with the RPV pressure-retaining components 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 lists 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 part 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 assembly, chimney, and core support structures, are listed 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 was 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 was considered. Similarly, consideration was 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 subsection 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 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 a 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 austenitic 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 minimizes the rate of progression. To aid in maintaining water quality, the BWRX-300 is designed to cascade feedwater heater 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 with upstream filters has a high efficiency for the removal of particulate and ionic impurities.

Materials of construction and their compatibility with reactor coolant is further discussed 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 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

Injection of noble metals was developed by GE Vernova Hitachi Nuclear Energy (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 BWRX-300 water chemistry program is 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 will 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 water chemistry under normal operation. The BWRX-300 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.

The most direct approach to ensure 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 2¼ Cr-1 Mo, can be used for piping that continuously meet the five 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 to preclude any risk of FAC. Although

not a part of the RCPB, stainless steel is also 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 100ºC to 150ºC.

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 materials 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 generally 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). 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 locating 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 will 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, e.g., 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 or Ni-Cr-Fe alloy 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 primarily used to minimize levels of radiation from corrosion products generated in the core. Where higher strength requirements than those for austenitic stainless steels are imposed, nickel-based alloys, nitrogen strengthened alloys, or precipitation hardening stainless steels are used.

The stainless steel structural materials used for 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 alloys (including niobium-modified Alloy 600M with filler metal Alloy 82, Alloy 625 with filler metal ERNICMO-3, Alloy 718, and Alloy X-750)
- Precipitation hardening stainless steels such as Type 630 and Type XM-13
- Carbon steels and low-alloy steels

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 RCPB such as SCC, general corrosion, and FAC. The specific categories of materials that are used are wrought austenitic stainless steels, cast austenitic stainless steels, martensitic stainless steels, precipitation hardening stainless steels, and nickel-based 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-based 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

The following RG has specific applicability to the design of the RCPB:

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 Nondestructive Examination (NDE) of ferritic wrought seamless tubular products used for ASME Class 1 and ASME Class 2 components as specified in the ASME BPVC.

5.2.3.3.1 Fracture Toughness

ASME Class 1 and ASME 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:

1. 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-2331 and NB-2332 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.

2. 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.

3. The RPV 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 ASME 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 is 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.

4. Calibration of instruments and equipment meets the requirements of ASME BPVC, Section III, NB/NCD-2360.

Fracture toughness tests will be performed for the RCPB, and the test results are 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 requirements of ASME BPVC, Section III and Section IX, “Qualification Standard for Welding, Brazing, and Fusing Procedures; Welders; Brazers; and Welding, Brazing, and Fusing Operators,” (Reference 5.2-5). 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 meet ASME BPVC, Section III (Subsection NB for Class 1 components and Subsection NCD for Class 2 components and Nonmandatory Appendix D, D-1000); and ASME BPVC, Section IX, 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 RCPB components 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 ensuring 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 RCPB components. 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 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, penetrants) are completely removed and cleanliness is ensured. 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 controlled during fabrication as well as operation.

Miscellaneous non-metallic materials, used for the fabrication and assembly of RCPB components, are also controlled to minimize contamination with species known to have detrimental effects on alloys such as stainless steels and nickel-based alloys.

Contaminant materials, which come in contact with stainless steel or nickel-based 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 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 SCC. 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 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 SCC. 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 requirements of ASME BPVC, Sections III and IX. 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 SCC.

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 discusses an acceptable method of implementing requirements with regard to the control of welding when fabricating and joining austenitic stainless steel components.

Written welding procedures are required for RCPB welds performed for material fabrication and plant construction. These procedures comply with the requirements of ASME BPVC, Sections III and IX 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 ASME BPVC, Sections III and IX. 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 ASME BPVC, Sections III and IX. The BWRX-300 meets the intent of RG 1.71 directly or by an alternate approach 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:

1. The welder performance qualification test assembly shall be welded under simulated access conditions.

2. 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.

3. 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 50.55a, as it relates to the requirements for inspecting and testing ASME Class 1 and ASME Class 2 components of the RCPB as specified in ASME BPVC, Section XI. The BWRX-300 ISI program complies with 10 CFR 50.55a as indicated in Subsection 5.2.1.1 and Subsection 5.2.1.2.

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 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 discussed in Subsection 5.2.1.1.

The RCPB consists of ASME Class 1 and ASME Class 2 components as discussed in Subsection 5.2.1.1 and illustrated on Figure 5.1-2. 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, Subsection IWB, and Subsection IWC. Examination results are evaluated and accepted in accordance with Subsection IWA, Subsection IWB, and Subsection IWC.

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 Regulatory Guide 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 Regulatory Guide 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 NEDC-33910P-A, "BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection," (Reference 5.2-2) discusses conformance to RG 1.45, "Guidance on Monitoring and Responding to Reactor Coolant System Leakage," 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 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 was 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 leakage 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 Specifications 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 GDC 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 discussed separately in the following sections.

5.2.5.1.1 Detection of Leakage Within Containment

The primary methods used for monitoring 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 leakage 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. Leakage 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 discussed 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 for breaks downstream of the MS Flow Elements (FE).

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 RPV 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 discussed in Subsection 5.4.7 and Subsection 5.4.8, respectively.

The ambient temperatures in the MS Tunnel area, SDC area, CUW area, and FW heater area are monitored for indications of steam leakage. The MS 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 leakage 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 leakage 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 leakage detection capability with the ability to trend the leakage rate downstream of quench tank isolation valves. This line is normally isolated during normal operations and, therefore, is 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 Containment Monitoring Subsystem (CMon) monitors containment pressure and provides signals 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 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 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 leakage rates inside containment during normal operations. The CCS is discussed 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 each ICS steam supply line to each IC and in each ICS condensate return line to the RPV. ICS flow detection is train specific.

Containment Sump Level

The Equipment and Floor Drain System (EFS) monitors containment sump level to measure and trend containment leakage. Excessive containment leakage is designed to be detected by the containment sump level changes, and frequency and duration of the pump runs. The EFS 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 MS Tunnel and TB steam area.

Process Radiation Level

The Process Radiation Monitoring Subsystem (PRM) includes leakage detection sensors for radiation levels in the following:

- Refuel Floor Area
- RB Heating, Ventilation, and Cooling System (HVS) Exhaust

- Refueling Floor HVS Exhaust
- IC Pool Area

The PRM provides radiation monitoring of the lower RB HVS exhaust.

The PRM provides radiation monitoring of the Refueling Floor HVS exhaust.

The PRM monitors the IC 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 train isolation.

Area temperature sensors are provided for the SDC rooms containing the pumps, heat exchangers, and piping. The system continuously monitors for leakage utilizing area temperature sensors. 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 discussed 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, associated capabilities will be addressed in the FSAR.

5.2.5.4 Limits for Reactor Coolant Leakage Rates Within the Containment

Limits for reactor coolant leakage rates will be addressed in the FSAR.

5.2.5.5 Criteria to Evaluate the Adequacy and Margin of Leakage Detection System

Leakage 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. 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 leakage detection systems will be addressed 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 leakage 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.

Leakage detection provided for SDC and CUW is discussed in Subsection 5.2.5.2.2.

Leakage past closed RIVs 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 monitoring systems. Although monitored leakage parameters are included in Table 5.2-6, associated capabilities will be addressed in the FSAR.

5.2.5.8.2 Leakage-Monitoring-Related Positions

Position 2.1

Information included in plant procedures will be addressed 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 2.2 is satisfied.

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 2.3 is satisfied.

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 leakage 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 addressed in the FSAR.

5.2.5.8.3 Operations-Related Positions

Position 3.1

Operating plant activities will be addressed in the FSAR.

Position 3.2

Information included in plant procedures will be addressed in the FSAR.

Position 3.3

Leakage monitoring system alarms, testing and calibration, and operator interface information will be addressed in the FSAR.

Position 3.4

Operating plant activities will be addressed in the FSAR.

5.2.5.8.4 Technical Specification Position

Position 4.1

Technical Specifications will be addressed 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 (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 A572 Grades 50 and 65 for Structural Attachments to Class CC Containment Liners, Section III, Division 2
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 NRC for incorporation into RG 1.84.

Table 5.2-2a Reactor Coolant Pressure Boundary Materials
(Sheet 1 of 8)

Component	Form	Material (Note 1)	ASME Specification
Reactor Pressure Vessel			
RPV Shells and Heads	Forging Plate	Low-Alloy Steel	SA-508/SA-508M Gr 3 Class 1, SA-533/SA-533M Type B 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 (MSRIV)			
MSRIV Body	Forging	Low-Alloy Steel	SA-350/SA-350M Gr LF2 Class 1
MSRIV Cover	Forging	Low-Alloy Steel	SA-350/SA-350M Gr LF2 Class 1
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)			
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
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 Bolting	Bolting	Alloy steel	SA-193/SA-193M Gr B7
MSCIV Hex Nuts	Bolting Nuts	Alloy steel	SA-194/SA-194M Gr 7

Table 5.2-2a Reactor Coolant Pressure Boundary Materials
(Sheet 2 of 8)

Component	Form	Material (Note 1)	ASME Specification
Main Steam Line (RCPB Portion)			
MS Piping	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 Fittings	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)
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 Housings	Forging	Stainless Steel	SA-965/SA-965M Gr F304
Middle Flange	Forging	Stainless Steel	SA-965/SA-965M Gr F304
Upper Flange	Forging	Stainless Steel	SA-965/SA-965M Gr F316
Ball Check Valve Retainer	Bar and Shapes	Stainless Steel	SA-479/SA-479M Type 316
CRD Mounting Bolts	Bolting	Alloy steel	SA-193/SA-193M Gr B7

Table 5.2-2a Reactor Coolant Pressure Boundary Materials
(Sheet 3 of 8)

Component	Form	Material (Note 1)	ASME Specification
CRD Penetration Stub Tubes	Bar, Seamless Pipe, Forging	Ni-Cr-Fe	SB-166, SB-167, SB-564 Nickel Alloy 600M (Modified per ASME Code Case N-580-2)
RPV Appurtenances Incore Instrumentation and CRD Penetration 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)
Incore Housings	Seamless Pipe, Forging	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 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 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 Type 630

Table 5.2-2a Reactor Coolant Pressure Boundary Materials
(Sheet 4 of 8)

Component	Form	Material (Note 1)	ASME Specification
IC Drum and Drum Covers	Forging	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	Forging	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
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)

Table 5.2-2a Reactor Coolant Pressure Boundary Materials
(Sheet 5 of 8)

Component	Form	Material (Note 1)	ASME Specification
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 (CIV)			
ICS CIV	Forging	Stainless Steel	SA-182/SA-182M Gr F304/F304L Gr F316/F316L (Dual Grade)
SDC Interface Redundant Isolation Valve (ICS Outboard CIV)	Forging	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 Pipe	Stainless Steel	SA-312/SA-312M Gr TP316/ TP316L (Dual Grade)
ICS Purge Fittings	Forging or Fittings	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 (FWRIV)			
FWRIV Body	Forging	Low-Alloy Steel	SA-350/SA-350M Gr LF2
FWRIV Cover	Forging	Low-Alloy Steel	SA-350/SA-350M Gr LF2

Table 5.2-2a Reactor Coolant Pressure Boundary Materials
(Sheet 6 of 8)

Component	Form	Material (Note 1)	ASME Specification
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
Condensate and Feedwater Heating System Feedwater Containment Isolation Valves (FWCIV)			
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

Table 5.2-2a Reactor Coolant Pressure Boundary Materials
(Sheet 7 of 8)

Component	Form	Material (Note 1)	ASME Specification
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
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 Bolting	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 Type 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.

Table 5.2-2a Reactor Coolant Pressure Boundary Materials
(Sheet 8 of 8)

6. Not Used.
7. Not Used.
8. Code Case N-943 for use of Alloy 625 (UNS N06625) 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 (Sheet 1 of 2)

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, E316L-XX
	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

**Table 5.2-2b Reactor Coolant Pressure Boundary Welding Filler Metals
(Sheet 2 of 2)**

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-based alloy and carbon/low-alloy steel or stainless steel are performed using Nickel Alloy 82. Welds between carbon steel 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 Forging	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	Forging, Plate	Low-Alloy Steel	SA-508/SA-508M Gr 3 Class 1, SA-533/SA-533M Type B Class 1
Vessel Stabilizer Brackets	Forging, Plate	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. (Note 1)	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:

1. 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 (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
(Sheet 1 of 2)

System	Instrument Type	Leak Size	Process Measurement	Location (Note 1)
Nuclear Boiler	Differential pressure transmitter	Break	MSL 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 and Environmental Monitoring	Pressure transmitter	Leak/Break	Containment pressure	I
	Radiation monitor	Leak	Refuel floor area radiation	O
	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	IC 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
Containment Cooling	Differential pressure transmitter	Leak/Break	Containment cooler drain flow	I

Table 5.2-6 Monitored Leakage Parameters
(Sheet 2 of 2)

System	Instrument Type	Leak Size	Process Measurement	Location (Note 1)
Reactor Water Cleanup	Temperature element	Break	Mass flow balance	O
	Differential pressure transmitter	Break		O
	Temperature element	Leak/Break	Ambient temperature	O
Condensate and Feedwater Heating	Differential pressure transmitter	Break	Reverse pressure from RPV to FW	O
	Temperature element	Leak/Break	Ambient temperature	O
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 = Inside 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 5.3-2).

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 50.55a as it relates to quality standards for design and determination and monitoring of fracture toughness, by compliance with Appendix G to 10 CFR 50, ASME BPVC, Section III, Section IX, and Section XI, 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 A, GDC 1 and GDC 30, as they relate to quality standards for design, fabrication, erection, and testing of SSCs, by compliance with ASME BPVC, Section III, Section IX, "Qualification Standard for Welding, Brazing, and Fusing Procedures; Welders; Brazers; and Welding, Brazing, and Fusing Operators," (Reference 5.3-3), and Section XI, "Rules for Inspection and Testing of Components of Light-Water-Cooled Planted," (Reference 5.3-4) 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 reactor vessel material surveillance program, by compliance with 10 CFR 50, Appendix H and ASTM International (ASTM) E185, "Standard Practice for Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels," (Reference 5.3-5). 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 listed in Table 5.2-2a and Table 5.2-3, with the applicable specifications.

The RPV materials comply with ASME BPVC, Section III; ASME BPVC, Section II, "Materials," (Reference 5.3-1); 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 examined, and surface examined by magnetic particle methods or liquid penetrant methods as required by ASME BPVC, Section III, Subsection NB.

Fracture toughness properties of materials are also measured and controlled in accordance with ASME BPVC, Section III, Division 1.

Fabrication of the RPV is performed in accordance with the GVH quality assurance program. 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, Sections III and IX. 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, Nonmandatory Appendix D. PWHT of low-alloy steel 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.

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 to 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 discussed in Subsection 5.2.3.4.2 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 RPV 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 5.3-6) and control of stresses. These controls conform to the guidance of Generic Letter 88-01 and NUREG-0313, Revision 2 as discussed 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 Subsection 5.3.1.6 for additional information.

5.3.1.5 Fracture Toughness

Appendix G to 10 CFR 50 is applied to ASME 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 based on 10 CFR 50, Appendix G, Section IV.A.2. The FSAR will address bounding pressure-temperature limit curves either as part of the Technical Specifications 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 with 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 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 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 E185-21 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 ASTM E185-21. 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 E185-21, Paragraph 5.4.1, are used to determine the reference temperature curve of the core beltline base material and weld metal.

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 normal plant operation including AOOs.

Bolting Material

All bolting material exceeding 25 mm (1 in.) diameter has special material requirements as presented in Table 5.3-1.

5.3.1.6 Material Surveillance

The reactor vessel material 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 (1) monitor irradiation embrittlement at the end of the period of extended operation and (2) 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 material surveillance 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 installed in removable specimen capsules at the inside vessel wall opposite the active core. The remaining unirradiated baseline specimens are used to establish material reference data.

Material surveillance program data is used as discussed 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 reactor vessel material 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 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 contains 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 also includes 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 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 was 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 material 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 is 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 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 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 addressed 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 surveillance specimen capsules. 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 reactor vessel flange. After installation of the reactor vessel closure head, a nut and washer are installed on the upper end of each stud. The proper amount of preload is applied to the studs by sequential tensioning.

Hardness tests are performed on RPV main closure bolting to demonstrate that heat treatment was properly performed.

The BWRX-300 design conforms with RG 1.65, “Materials and Inspections for Reactor Vessel Closure Studs.” 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 complies with ASME BPVC, Section III, Subsection NB requirements. The RPV main closure bolting materials are specified in Table 5.2-2a. The bolting undergoes NDE per ASME BPVC, Section III, NB-2580 before being placed into service. The NDE includes visual examination, magnetic particle or liquid penetrant examination, and ultrasonic 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 applications. Lubricants shall 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 reactor vessel shell flange are 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.

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 ensure 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 material 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 regulations:

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 met with acceptable upper-shelf energy values throughout the life of the vessel.

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 ensuring 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 ensuring 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 material surveillance program, by compliance with 10 CFR 50, Appendix H and ASTM E185 as discussed in Subsection 5.3.1.6.

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 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 addressing bounding pressure-temperature limit curves as discussed in Subsection 5.3.1.5, the FSAR will address the methodology used to determine these limits.

The ferritic materials in the RCPB, including pressure-temperature limits, 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 heads and flange areas, FW nozzles, 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, Nonmandatory 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, Nonmandatory 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 addressed in the FSAR.

Temperature Limits for Inservice Inspection Hydrostatic and Leak Pressure Tests

Bounding pressure-temperature limit curves using plant-specific data, such as materials, fluences and stresses, will be developed and documented as appropriate in Technical Specification requirements.

Operating Limits During Heatup, Cooldown, and Core Operation

Bounding pressure-temperature limit curves using plant-specific data such as materials, fluences and stresses will be developed and documented as appropriate in Technical Specification 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 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 reactor vessel material surveillance program will be conducted in accordance with the requirements of ASTM E185. The material surveillance program includes specimens of base metal and weld metal of the beltline forging. Subsection 5.3.1.6 provides additional detail on the material surveillance program.

5.3.3 Reactor Vessel Integrity

This subsection provides a 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 discussed in Subsection 5.3.1. The basic design of the reactor vessel establishes compatibility with required inspections as discussed 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 Subsection 5.3.1.5 and Subsection 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 discussed in Subsection 5.2.3 and Subsection 5.3.1. Conformance with RG 1.99 is discussed in Subsection 5.3.1 and Subsection 5.3.2.

Fabrication Methods

The processes used to fabricate the reactor vessel, including forming, welding, cladding, and machining, are discussed in Subsection 5.3.1.

Inspection Requirements

The fabrication inspection and test methods and requirements are discussed 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 material surveillance of the reactor vessel are discussed 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. The acceptance criteria in these subsections discuss methods that meet the requirements of the following regulations:

- 10 CFR 50.55a
- 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

The GVH quality assurance program 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 complies with the requirements of ASME BPVC, Section II. The shells, RPV heads, 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 alloy overlay as discussed in Subsection 5.2.3.2.2. The core beltline limiting forging is made of SA-508/SA-508M Grade 3 Class 1 material. The RPV head fasteners are discussed in Subsection 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, Subsection NB requirements. The shells, RPV heads, 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, Sections III and IX. Weld test samples support each procedure for full penetration welds. Fabrication of the RPV is performed in accordance with the GVH quality assurance program.

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 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. PWHT of low-alloy steel welds is performed in accordance with ASME BPVC, Section III as specified in Table 5.3-1.

Other fabrication processes such as cutting, bending, and forming, are performed in accordance with the reactor vessel design documents.

5.3.3.3 Inspection Requirements

Plates, forgings, and bolting are ultrasonically examined, 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 will 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 ensures 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 II, “Materials,” American Society of Mechanical Engineers, 2021.

5.3-2 ASME BPVC, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” American Society of Mechanical Engineers, 2021.

5.3-3 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.

5.3-4 ASME BPVC, Section XI, Division 1, “Rules for Inspection and Testing of Components of Light-Water-Cooled Plants,” American Society of Mechanical Engineers.

5.3-5 ASTM E185-21, “Standard Practice for Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels,” ASTM International, 2021.

5.3-6 ASTM A262-15(2021), “Standard Practices for Detecting Susceptibility to Intergranular Attack in Austenitic Stainless Steels,” ASTM International, 2015 (R2021).

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 Subsection 5.3.1.2 and Subsection 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 Subsection 5.2.3.3.2, Subsection 5.3.1.2, and Subsection 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 Subsection 5.2.3.3.1, Subsection 5.3.1.5, and Subsection 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 welds 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 (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 an illustration of the RPV and reactor internals.
2. Elevation zero is located at the inside of the RPV bottom head. Height dimensions provided in this table are relative to elevation zero.

Table 5.3-4 Reactor Vessel Estimated Fluence Values

Parameter (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)	$4.63x10^{18}$
Estimated peak neutron fluence at inside surface of limiting RPV beltline shell weld (Note 1)	$1.33x10^{18}$
Estimated peak neutron fluence at RPV beltline shell 1/4 wall thickness (Note 2)	$3.43x10^{18}$
Estimated peak neutron fluence at limiting RPV beltline shell weld 1/4 wall thickness (Note 2)	$9.85x10^{17}$

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. Material 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 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 there are 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 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 illustrated on 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 criteria for the RPV isolation functions.

The RIVs allow Loss-of-Coolant Accidents (LOCA) 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 Regulatory Guide 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 Regulatory Guide 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.

The 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 3.11 provides information on environmental qualification of mechanical and electrical equipment. Section 3.12 provides information on the design and analysis, and testing, of RCS piping, and associated piping supports.

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 discusses 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 function of the ICS. Section 6.6 discusses preservice inspection and ISI of ASME Class 2 and ASME 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 MS 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 subsystem consists of two MSLs. Figure 5.1-2 and Figure 5.4-5 provide illustrations of the MS subsystem.

An MSL flow restrictor is installed in each of the two RPV MS nozzles. The flow restrictor limits steam flow in the event of an break in the MS subsystem. The flow restrictor, together with the fast closing MSRIVs (discussed 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 MSL flow restrictor is designed in accordance with “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 MSL flow restrictors meet material and construction standards as discussed in Subsection 5.2.3.

5.4.4.4 Testing and Inspection Requirements

Because the MSL flow restrictor forms a permanent part of the RPV main steam outlet nozzle, and has no moving components, no testing program beyond RPV ISI is planned.

5.4.4.5 Instrumentation Requirements

The MSL flow instruments are differential pressure Flow Transmitters (FT) used to determine MS flow for the purpose of MS isolation and Reactor 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, a 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

- Safety Category N functions for MS and FW are discussed in Section 10.3 and Subsection 10.4.7, respectively.

The Safety Classifications, Quality Groups, and Seismic Categories of MS and FW Isolation SSCs are listed in Table 3A-1.

5.4.5.2 System Description

Reactor Isolation Valves

As discussed 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 containment isolation valves are designed to ASME BPVC, Section III, Subsection NCD, and Quality Group B requirements.

The containment isolation function is 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. Figure 5.1-2 and Figure 5.4-5 provide illustrations of MS subsystem. Only the portion of the MSL from the MSRIV up to and including the MSCIV is addressed in this chapter, 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.

As discussed in Subsection 5.4.4, an 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 MS subsystem. 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 illustrates a simplified diagram of the FW Subsystem, and system interfaces located in the RB. Only the portion of the CFS and the FW subsystem from the FWRIV up to and including the FWCIV is addressed in this chapter, while the remainder of the CFS and the FW subsystem 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 subsystem isolation valves for loss of FW flow and leakage 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 leakage detection, isolation, and containment capabilities. MS and FW isolation subsystems are provided with Safety Category 1 leakage 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 regulatory guidance has aspects specific to MS and FW isolation subsystems:

Branch Technical Position (BTP) 3-4, “Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment.” BTP 3-4 conformance is discussed 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 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 discusses the RCPB overpressure protection and ICS emergency core cooling and decay heat removal functions.

GDC 33 compliance related to reactor coolant makeup for small breaks is 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 (discussed in Section 6.3) in conjunction with SDC.

ICS provides the 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. The SDC provides the Safety Category 3 residual and decay heat removal function.

5.4.7.1 Design Bases

Safety Category 1 Functions

- Leakage 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 ICS containment isolation valves and an FE located outside the PCS, each SDC suction line is routed to an SDC pump in a room outside the PCS. Train A and B pumps 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 to the respective SDC Train A or B heat exchanger. The heat exchangers are located in adjoining rooms above the SDC pumps. A heat exchanger bypass line is installed to increase operational flexibility.

After exiting the SDC heat exchanger or heat exchanger bypass line, flow is either directed back to the CFS Train A or B injection line upstream of the CFS CIVs 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 Liquid Waste Management System (LWM) through the CUW.

SDC Decay Heat Removal Subsystem

The SDC decay heat removal 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 returned to the RPV through the CFS as illustrated on 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 Leakage 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 was sized to ensure it can withstand the maximum pressure and temperature combination, and the diameter was 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 leakage 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 for decay heat removal. The pump is installed below the ICS condensate RPV return 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 during startup operations. 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 leakage detection and isolation function is Safety Category 1. The ICS performs the Safety Category 1 heat removal function. The ICS is discussed in Section 6.3.

Regulatory Requirements

10 CFR 50, Appendix A

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 and Principal Design Criteria (PDC) have aspects specific to the SDC 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. SDC leakage 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. PDC 19 does not apply for SDC normal instrumentation.

GDC 34 – Residual heat removal, as it relates to provision of a residual heat removal system. SDC does not perform a Safety Category 1 heat removal function. The ICS (discussed in Section 6.3) performs the Safety Category 1 heat removal function.

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 discussed in Section 6.3.

Regulatory Guidance

The following regulatory guidance have 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 (discussed in Section 6.3). Additional information on conformance with SECY-94-084 is provided in NEDC-33934P, “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, leakage 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 discussed 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 reactor vessel and heat exchanger cooling water limitations, as well as allowing the operator to input a temperature rate setpoint that allows automatic reactor vessel 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 power operations. CUW does not clean or filter reactor water. Reactor water cleanup, filtration, and ion removal is performed by the CFD as discussed in Subsection 10.4.6. CUW provides an overboarding flow path to the MCA hotwell or LWM directly from the RPV lower region.

5.4.8.1 Design Bases

Safety Category 1 Functions

- Leakage 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 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 illustrates a simplified diagram of the CUW and interfacing systems.

Reactor water cleanup, filtration, and ion removal are performed by CFD as discussed 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 but relies on the SDC for motive pressure to generate flow, or sufficient reactor pressure is 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's inlet is independently connected to RPV penetrations located at about the mid-vessel height and takes suction from piping located near the RPV bottom head. The suction piping connects to the two CUW RIVs and then combines inside the PCS to form one suction 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 leakage detection and isolation subsystem to close upon a detected leak.

Downstream of the CUW containment isolation valve, the flow continues through a Regenerative Heat Exchanger (RHX) and flow control valves. The CUW provides process water to the PS for online chemistry monitoring. The RHX and flow control valves are designed to condition the water to acceptable temperatures, flows, and pressures for processing to CFS or overboarding. The RHX recovers heat back from the CFS.

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 MCA 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 discussed in Subsection 10.4.1.

Subsection 6.2.4 discusses the containment isolation function and CIVs.

CUW Leakage Detection and Isolation Subsystem

CUW leakage detection and isolation subsystem performs the Safety Category 1 function to identify leaks in CUW and initiates automatic response actions. The CUW leakage detection and isolation subsystem continuously monitors for leakage utilizing density-compensated differential flow measurements. If a leak is detected, the Safety Category 1 isolation function isolates the system from the RCPB through the closure of the CUW RIVs and CUW CIV.

The leakage 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 leakage 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 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 leakage detection features upstream and downstream locations allows for detection of differential flow between the locations, or detection of a leak 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 for 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 throttling valves. These valves are 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 FW 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 the Reactor Water Cleanup alignment, 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 reactor pressure. If not, SDC can be used to letdown the excess reactor inventory. The primary SDC overboard line interfaces with CUW downstream of the RHX. Overboard flow can be directed to the MCA hotwell or LWM.

Hot Shutdown

During hot shutdown operations, CUW is in operation 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 ensure the completion of fundamental CUW isolation functions. The RIVs and CIV 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, CUW leakage detection and isolation subsystem identifies the leak condition and provides 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. Reactor water cleanup, filtration, and ion removal is performed by the CFD as discussed 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 discussed 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 leakage detection, isolation, and containment capabilities. CUW is provided with leakage 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 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 a reactor water purification function and does not retain radioactive materials generated from reactor water purification. Reactor water cleanup, filtration, and ion removal is performed by the CFD as discussed 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 CFD or overboards excess “swell,” when required, to the MCA hotwell or LWM. 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 CUW 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 and Feedwater Piping

MSL and FW lines from the RIVs to the CIVs, including isolation functions, are discussed in Subsection 5.4.3 and Subsection 5.4.5. MS and FW piping beyond the CIVs is discussed in Section 10.3 and Subsection 10.4.7, respectively.

5.4.10 Pressurizer

The BWRX-300 design does not include a pressurizer.

5.4.11 Pressurizer Relief Discharge System

The 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 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 performs 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 MSL Train A, 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 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 shutdown. 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 and 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 discussed 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 subsystem 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. PDC 19 does not apply for the RPV head vent subsystem normal instrumentation.

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 discussed 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, 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).

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 leakage detection capability.

During reactor shutdown periods, reactor water level can be measured using the RPV head vent connection as the upper tap for the RPV level instrumentation sensors.

5.4.14 Reactor Coolant System Pressure Relief Devices/Reactor Coolant Depressurization Systems

The BWRX-300 design utilizes the ICS 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,” 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,” International Electrotechnical Commission/Institute of Electrical and Electronics Engineers, 2020.

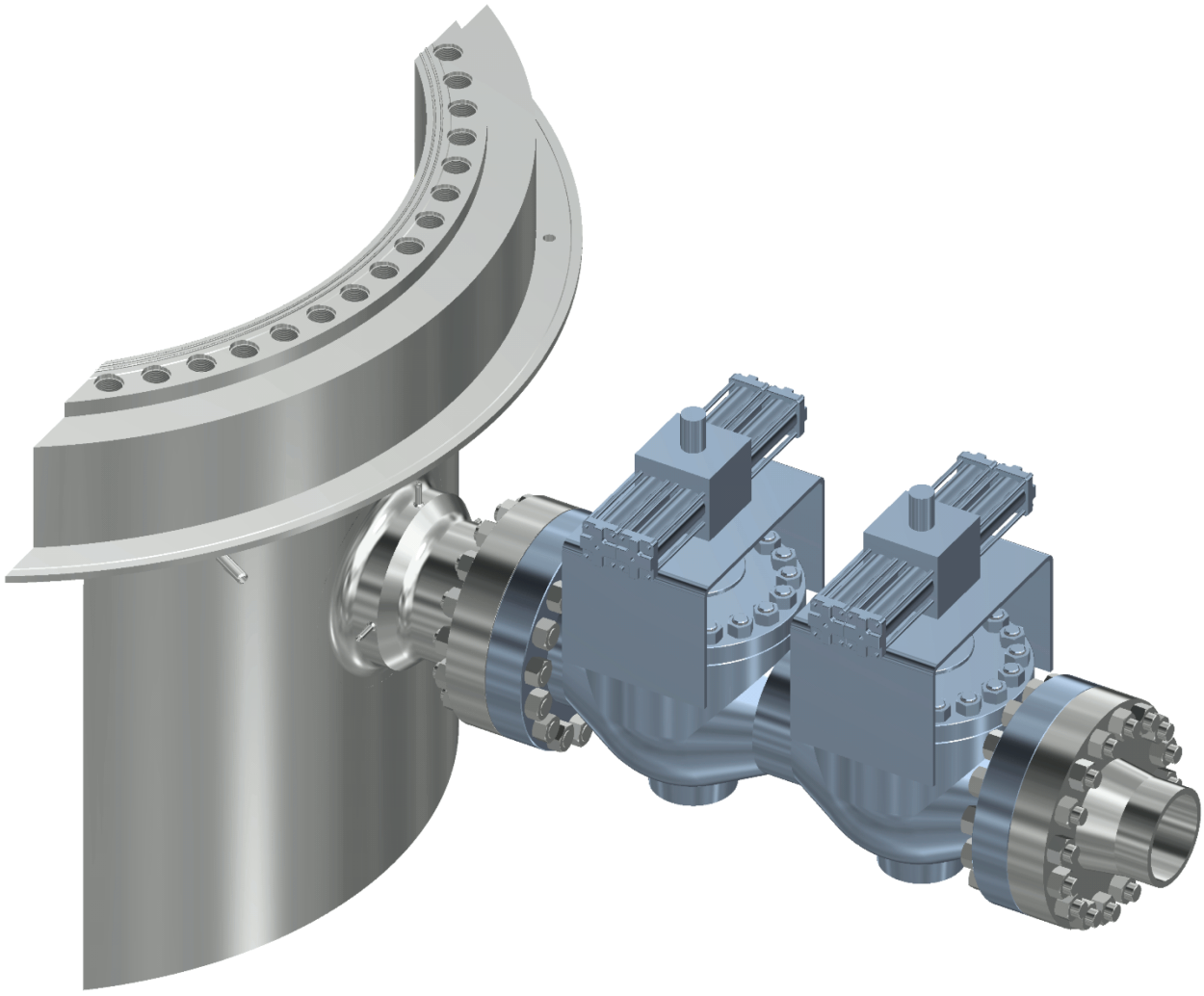
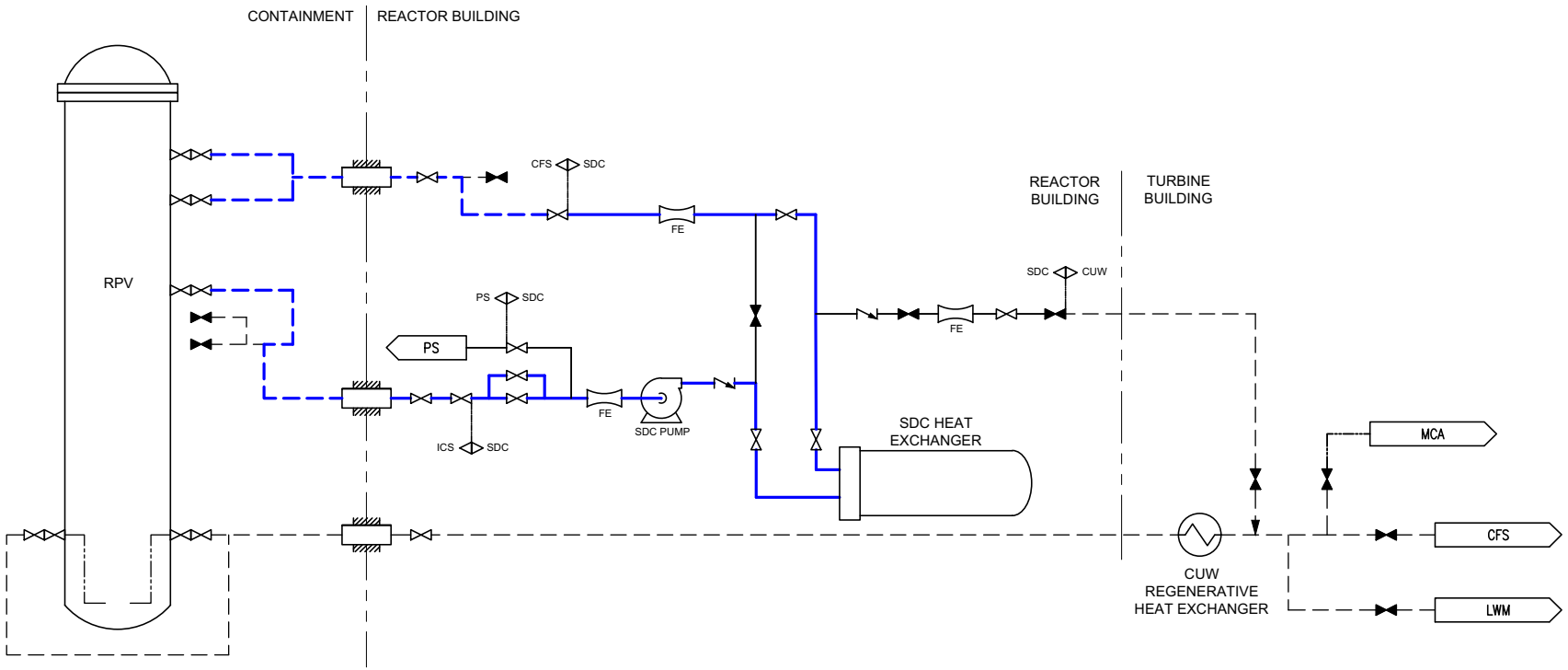


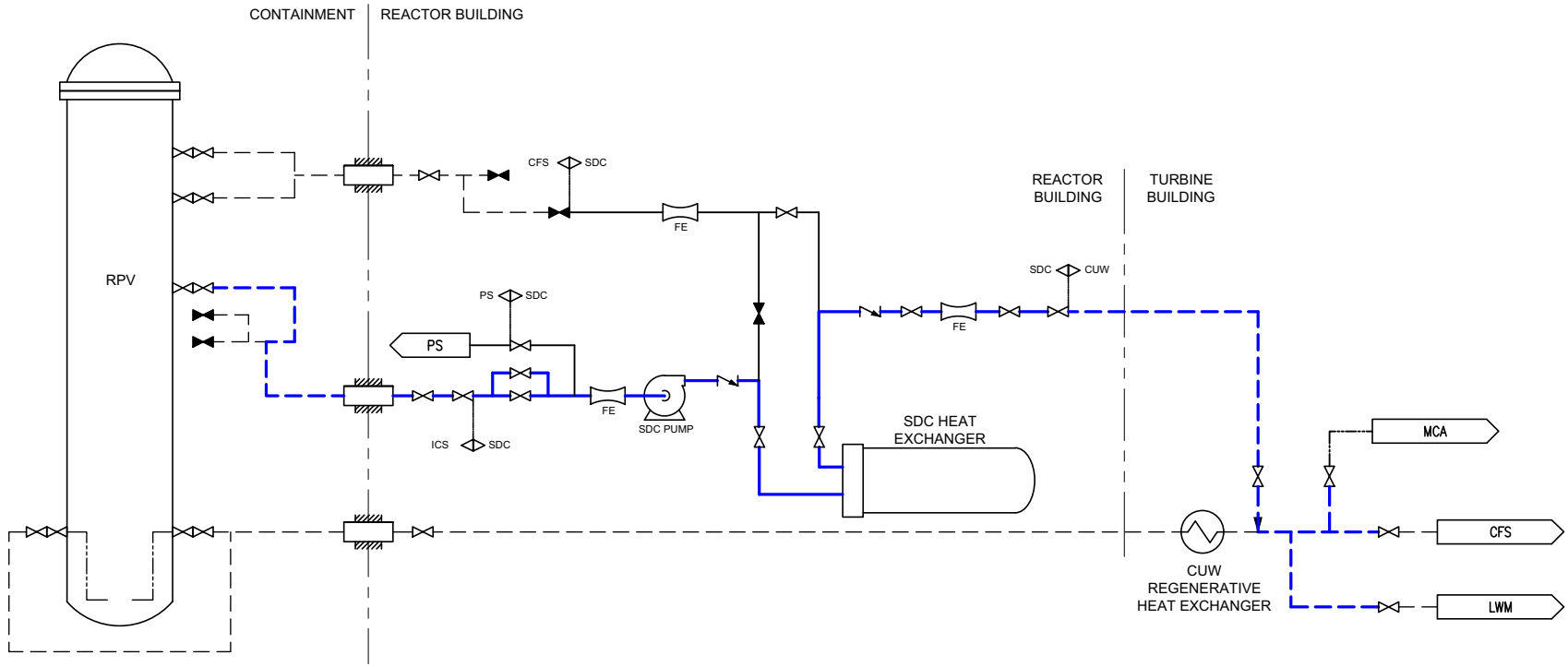
Figure 5.4-1 BWRX-300 Reactor Pressure Vessel Reactor Isolation Valve



Notes:

1. All 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.

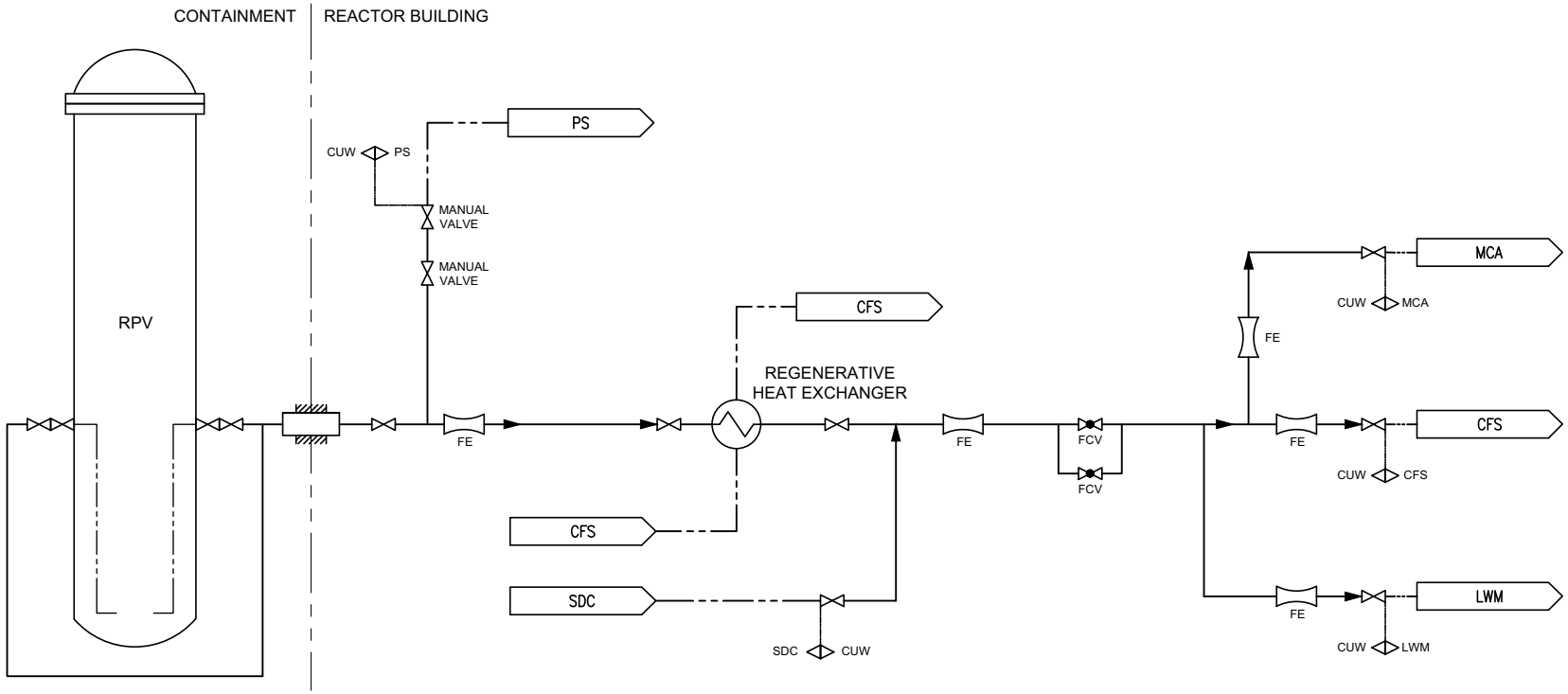
Figure 5.4-2 Shutdown Cooling System Simplified Diagram – Decay Heat Removal



Notes:

1. All valves shown, except check valves, have power operators.
2. Safety Class/Quality Group/Seismic Category breaks are not shown.
3. Blue line represents overboarding flow path. Multiple flowpaths shown.
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. All 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 5.4-4 Reactor Water Cleanup System Simplified Diagram

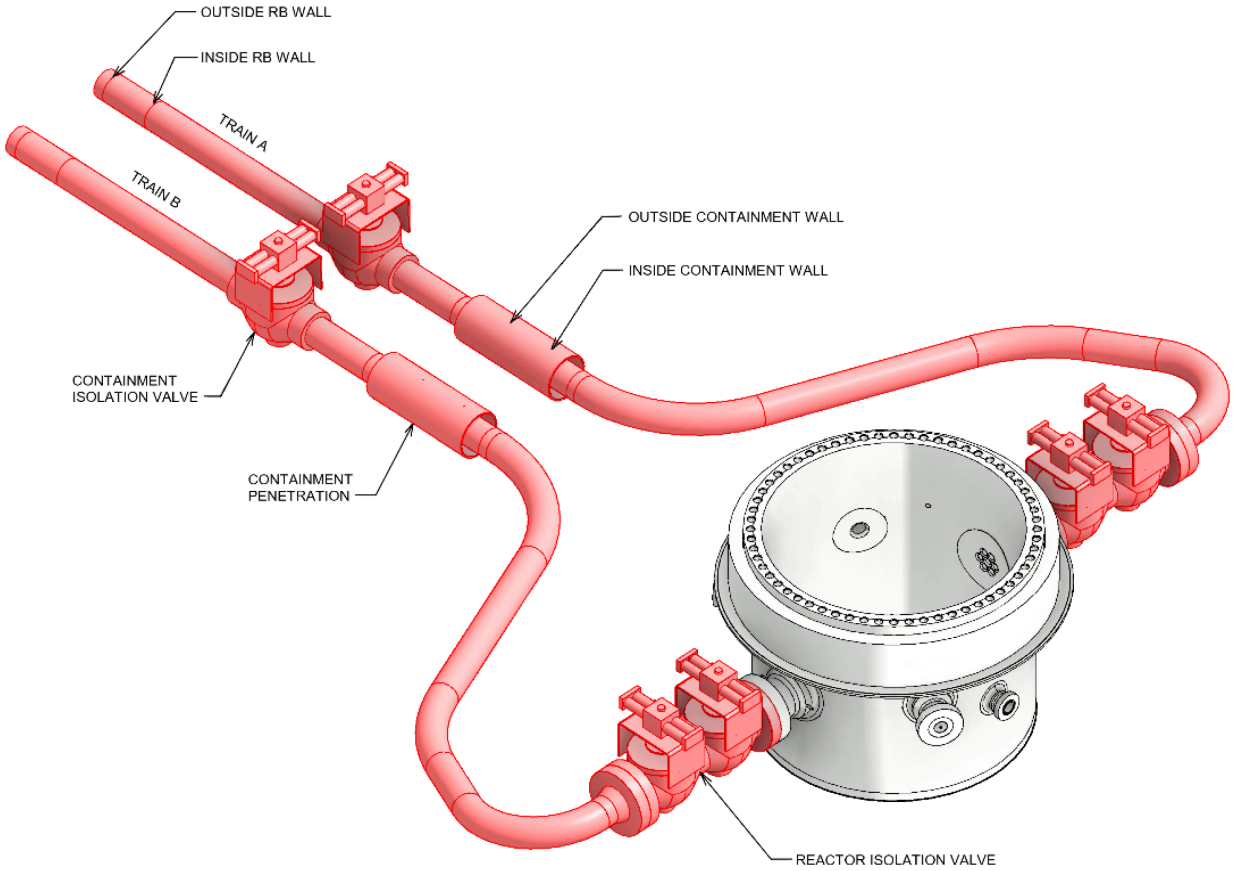


Figure 5.4-5 Main Steam System

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LIST OF ACRONYMS AND ABBREVIATIONS

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
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

LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
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
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

LIST OF EFFECTIVE SECTIONS

CHAPTER 6 ENGINEERED SAFETY FEATURES

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CHAPTER 6 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-Alloy 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 A5.20	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 will be 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 will be 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.
- Thecontainmentisolationfunctionalityisoperational,includingallassociated 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,” GE-Hitachi Nuclear Energy Americas, LLC, Revision 3, June 2022.

6.2-3 NEDC-33911P-A, “BWRX 300 Containment Performance,” GE-Hitachi Nuclear Energy Americas, LLC, Revision 3, January 2022.

6.2-4 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.

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
(Sheet 1 of 3)

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

Table 6.2-2 Process Line Isolation
(Sheet 2 of 3)

Line	Inside Containment CIV (Note 1)	Outside Containment CIV (Note 1)
Containment Inerting System		
Inlet to Containment	N/A	CIV, CIV
Outlet from Containment to Plant Vent Stack	N/A	CIV, CIV
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

Table 6.2-2 Process Line Isolation
(Sheet 3 of 3)

Line	Inside Containment CIV (Note 1)	Outside Containment CIV (Note 1)

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 (Sheet 1 of 3)

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 A, B, and C	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

Table 6.2-4 Process Line Isolation Configuration and GDC Compliance (Sheet 2 of 3)

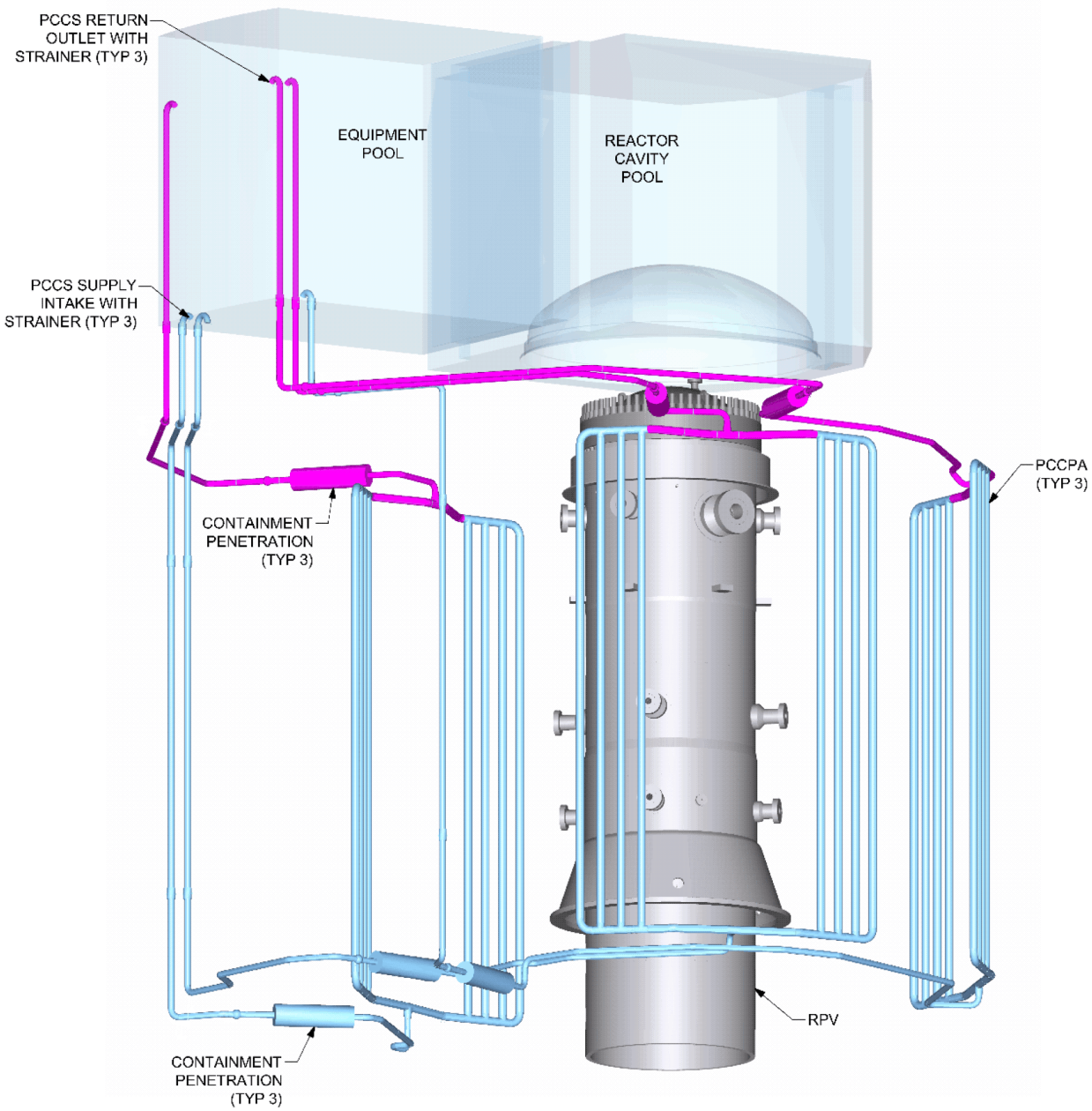
System/GDC Compliance Basis	Description	CIV Position			
		Normal	Hot Shutdown	Post-Accident	Loss of Motive Power
PPS GDC 56	Service Air	Locked Closed	Locked Closed	Locked Closed	N/A
	Instrument Air (includes Nitrogen Supply)	Open	Open	Closed	Closed
CIS GDC 56 (Note 3)	Inlet to Containment	Open/ Closed (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

Table 6.2-4 Process Line Isolation Configuration and GDC Compliance (Sheet 3 of 3)

System/GDC Compliance Basis	Description	CIV Position			
		Normal	Hot Shutdown	Post-Accident	Loss of Motive Power
PCS GDC 56	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.



Note: Valves are not shown.

Figure 6.2-1 Passive Containment Cooling System Configuration

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

In order to demonstrate the functioning of the recombiner, two tests will be conducted for the BWRX-300: an integral test and a component test.

The integral test uses a scaled prototype to simulate the geometry, elevations and materials of the lower and upper drums to demonstrate that the flow through the recombiner is in fact in the direction and rate as predicted by the calculations. Additionally, gasses will be injected into the system and concentrations measured at various locations to confirm that the radiolytic gases will collect in the lower drum.

The recombiner component testing will be performed by the vendor using the actual recombiner vessel and recombiner plates to demonstrate the recombiner effectiveness at the specified pressures and steam flow rates determined by GOTHIC analyses and confirmed by the integral testing.

A full description of the validation testing is described in NEDC-34349P, “BWRX-300 Darlington New Nuclear Project (DNNP) Joint Report on GE Hitachi’s Containment Evaluation Method: Implementation of Licensing Considerations” (Reference 6.3-2).

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 6.3-1) 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.

Subsections 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,” GE-Hitachi Nuclear Energy Americas, LLC, Revision 2, June 2021.NEDC-33922P-A, “BWRX-300 Containment Evaluation Method,” GE-Hitachi Nuclear Energy Americas, LLC, Revision 3, June 2022.

6.3-2 NEDC-34349P, “BWRX-300 Darlington New Nuclear Project (DNNP) Joint Report on GE Hitachi's Containment Evaluation Method: Implementation of Licensing Considerations,” GE-Hitachi Nuclear Energy Americas, LLC, Revision 0, June 2025.

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)

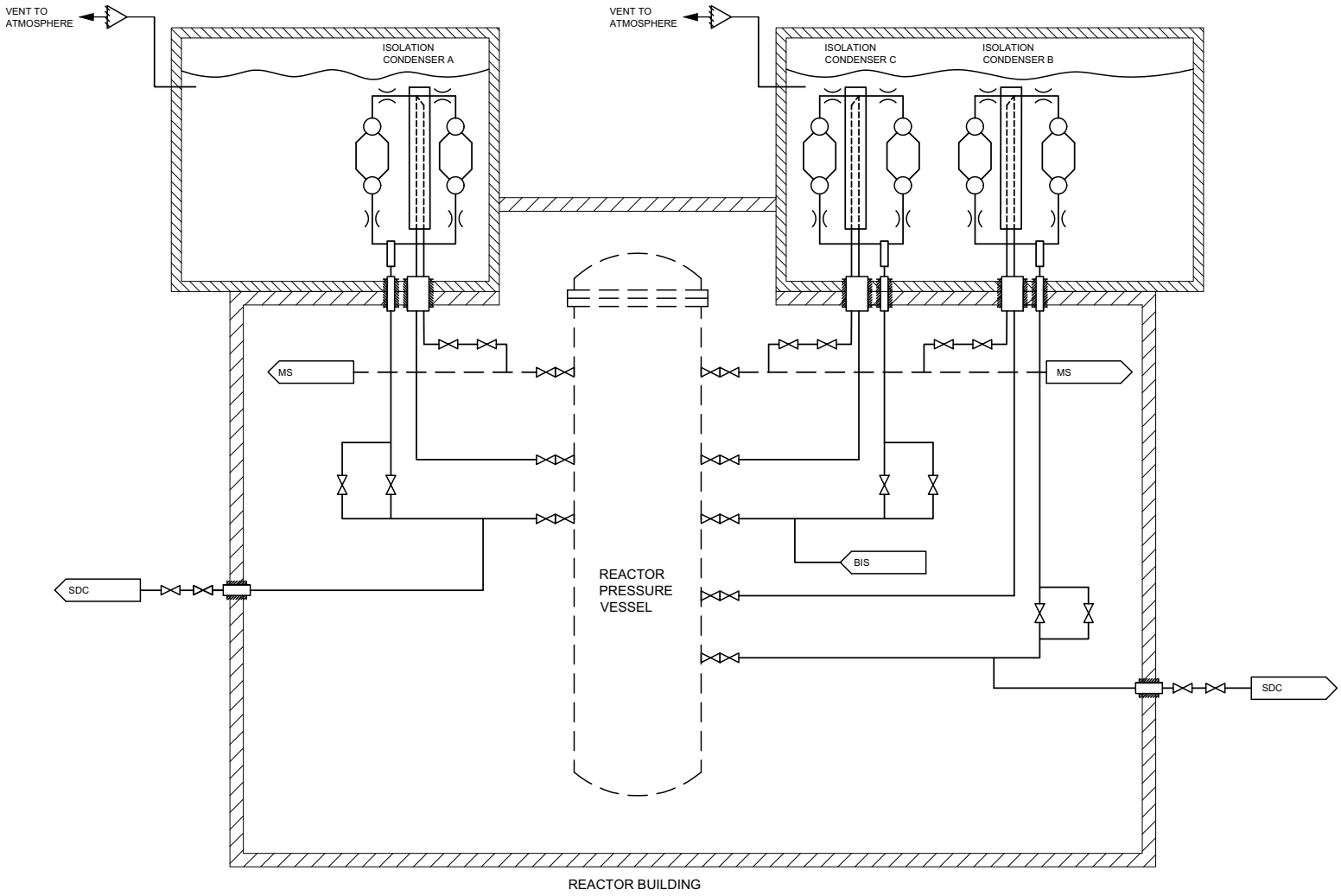


Figure 6.3-1 Isolation Condenser System Simplified Diagram

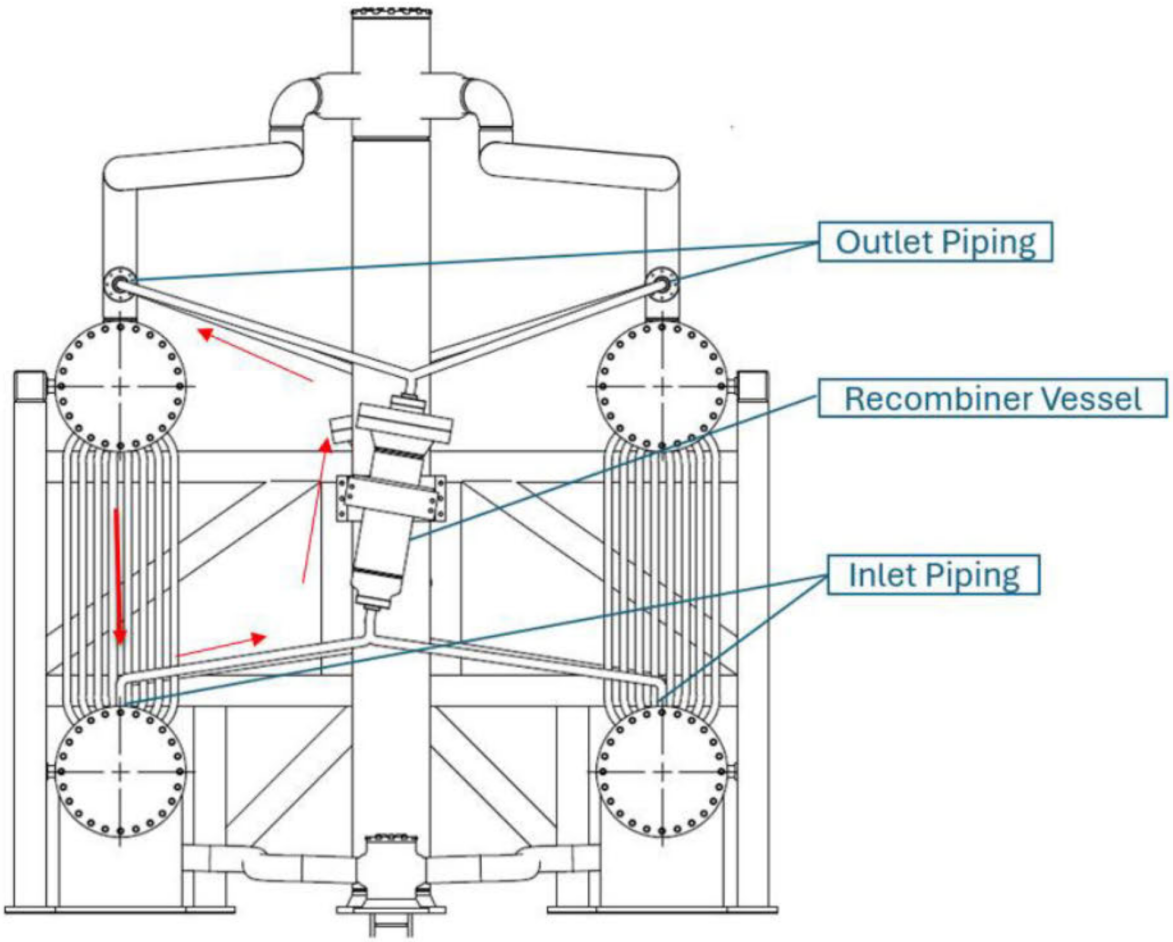


Figure 6.3-2 Preliminary Hydrogen Recombiner Configuration

Note: Red arrows show flow direction.

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, Revision 2, 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.

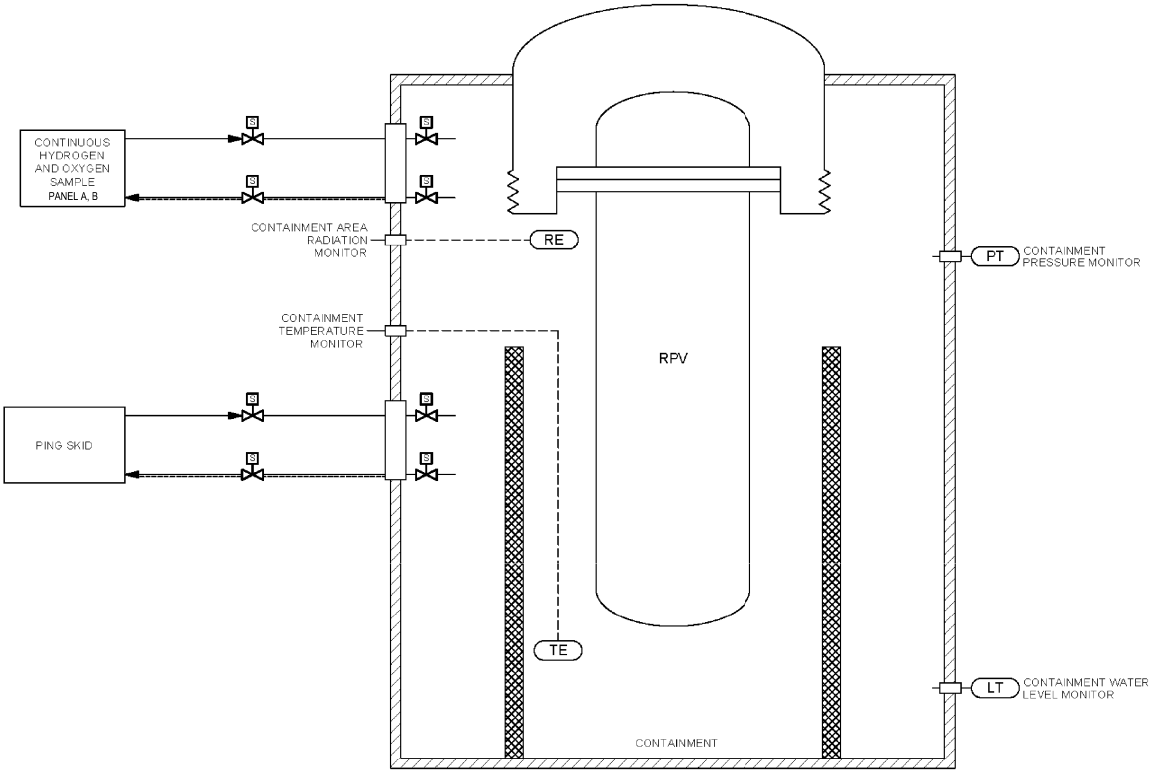
RG 1.21 – Measuring, Evaluating, and Reporting Radioactive Material in Liquid and Gaseous Effluents and Solid Waste: The design conforms to the guidance provided in RG 1.21 with regard to monitoring for potentially hazardous contamination such as radioactivity, noble gases, air particulates, and halogens.

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.



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LIST OF ACRONYMS AND ABBREVIATIONS

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
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
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

LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
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
GT	Gamma Thermometer
GVH	GE Vernova Hitachi Nuclear Energy
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
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

LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
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
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
TVA	Tennessee Valley Authority
UDH	Unit Data Highway
UPS	Uninterruptible Power Supply
USNRC	U.S. Nuclear Regulatory Commission
V&V	Verification and Validation
VDU	Visual Display Unit
WRNM	Wide Range Neutron Monitor

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CHAPTER 7 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 (DL). 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 (DBA) 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 Table 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 (PIE) in the Anticipated Operational Occurrences (AOO) 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 (LPRM) 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 (CCF) 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 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. TVA 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 has been 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) 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 603-2018, “Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” (Reference 7.1-2) as an alternative to the IEEE 603-1991 version to support a 3-division SC1 architecture. Subsection 7.3.1.3.2 describes the approach for addressing IEEE 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 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 safety classes. 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 and IEEE 603-2018. 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 and IEEE 603-2018.

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 (SC1, SC2, and SC3)
- GDC 2 – Design bases for protection against natural phenomena (SC1)
- GDC 4 – Environmental and dynamic effects design bases (SC1, SC2, and SC3)

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 (SAFDL) 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 Section 7.5 and Section 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 Subsection 7.3.3.2 and Subsection 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., LPRM) 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 safety class. 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 Subsection 7.3.1.2, Subsection 7.3.2.2, and Subsection 7.3.3.2, respectively. 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 (HCU) and the FMCRD motors for reactivity control during normal operation, planned power changes, AOOs, and postulated accidents.

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 Subsection 7.3.1.2, Subsection 7.3.2.2, and Subsection 7.3.3.2, respectively.

PDC 29 – Protection against anticipated operational occurrences. For events that result in AOOs, the DL2 functions provide 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 and with a scram follow by the SC2 FMCRD control system. The combination of DL2, independent DL3 functions, and diverse DL4a functions, provide layered protection against AOO PIEs, and results in extremely high functional reliability for performance of the Fundamental Safety Functions (FSF) during and after design basis events, including AOOs. The SC1, SC2, and SC3 system functions are described in Subsection 7.3.1.2, Subsection 7.3.2.2, and Subsection 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 (RIV). 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 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 (ARM) 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 Subsection 7.3.3.4 and Subsection 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 603-2018, "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 63147:2017/IEEE 497-2016, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations," International Electrotechnical Commission/Institute of Electrical and Electronics Engineers, 2017.

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 – Part 4: Testing and Measurement Techniques," International Electrotechnical Commission, 2012.

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, “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, “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,” Consolidated Version, International Electrotechnical Commission, 2016.

7.1-24 NEI 08-09, “Cyber Security Plan for Nuclear Power Reactors,” Nuclear Energy Institute, Revision 6, April 2010.

7.1-25 IEEE 603-1991, “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 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	-	-

Notes:

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 (Sheet 1 of 3)

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 63147:2017/IEEE 497-2016 "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 – Part 4: Testing and Measurement Techniques," (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)

Table 7.1-2 Instrumentation and Control System and Equipment Standards (Sheet 2 of 3)

Safety Class	Description	Systems	Equipment
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 603-2018 (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)
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 61508, "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)

Table 7.1-2 Instrumentation and Control System and Equipment Standards (Sheet 3 of 3)

Safety Class	Description	Systems	Equipment
SC3	I&C equipment supporting Safety Category 3 functions	• IEC 62138 (Reference 7.1-15) (Category C software)	• IEC/IEEE 60980-344 (Reference 7.1-13) (seismic) - Note 1 • IEC 61508, (Reference 7.1-16) (can be considered for use on pre-developed equipment for SC3 systems)

Note:

1. If required for mitigation of seismic-related events.

Table 7.1-3 Alignment of Regulatory Guides to BWRX-300 Alternatives
(Sheet 1 of 3)

Regulatory Guides	BWRX-300 Alternative
RG 1.22, Revision 0, "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, Revision 1, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems"	• IEEE 603-1998 (Reference 7.1-2)
RG 1.53, Revision 2, "Application of the Single-Failure Criterion to Safety Systems"	
RG 1.62, Revision 1, "Manual Initiation of Protective Actions"	
RG 1.75, Revision 3, "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, Revision 2, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants"	• IEC/IEEE 60780-323 (Reference 7.1-12)
RG 1.97, Revision 5, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants"	• IEC 63147:2017/IEEE 497-2016 (Reference 7.1-6)
RG 1.100, Revision 4, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants"	• IEC/IEEE 60980-344 (Reference 7.1-13)
RG 1.105, Revision 4, "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, Revision 3, "Periodic Testing of Electric Power, and Protection Systems"	• IEC 60671 (Reference 7.1-17) • IEC 62385 (Reference 7.1-18)

Table 7.1-3 Alignment of Regulatory Guides to BWRX-300 Alternatives (Sheet 2 of 3)

Regulatory Guides	BWRX-300 Alternative
RG 1.152, Revision 4, "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 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 1.168, Revision 2, "Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants"	
RG 1.169, Revision 1, "Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants"	
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 63147:2017/IEEE 497-2016 (Reference 7.1-6)

Table 7.1-3 Alignment of Regulatory Guides to BWRX-300 Alternatives (Sheet 3 of 3)

Regulatory Guides	BWRX-300 Alternative
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 safety class equipment are not used to perform higher safety category functions in the higher safety class 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.

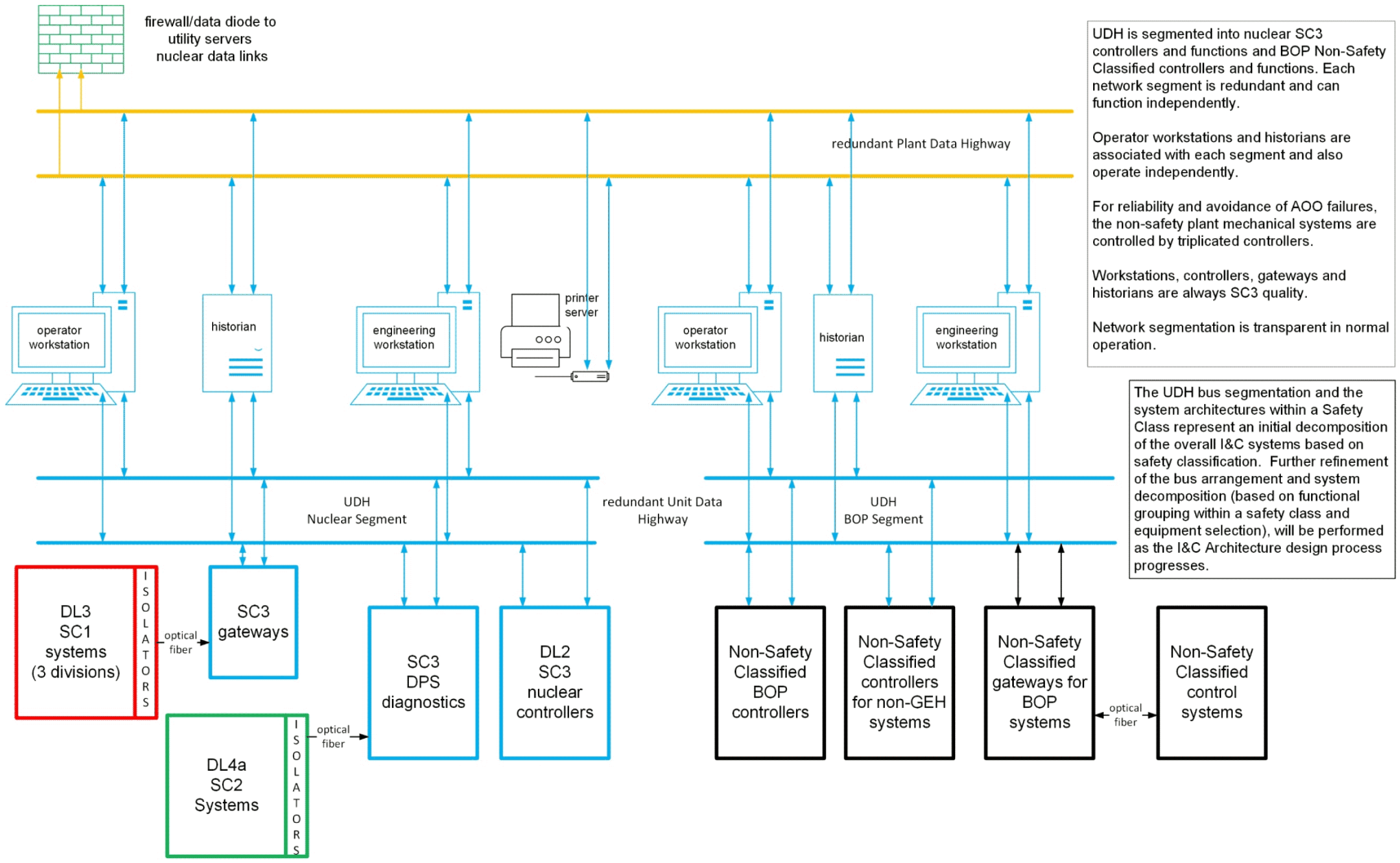

Figure 7.2-1 BWRX-300 Distributed Control and Information System Network Architecture

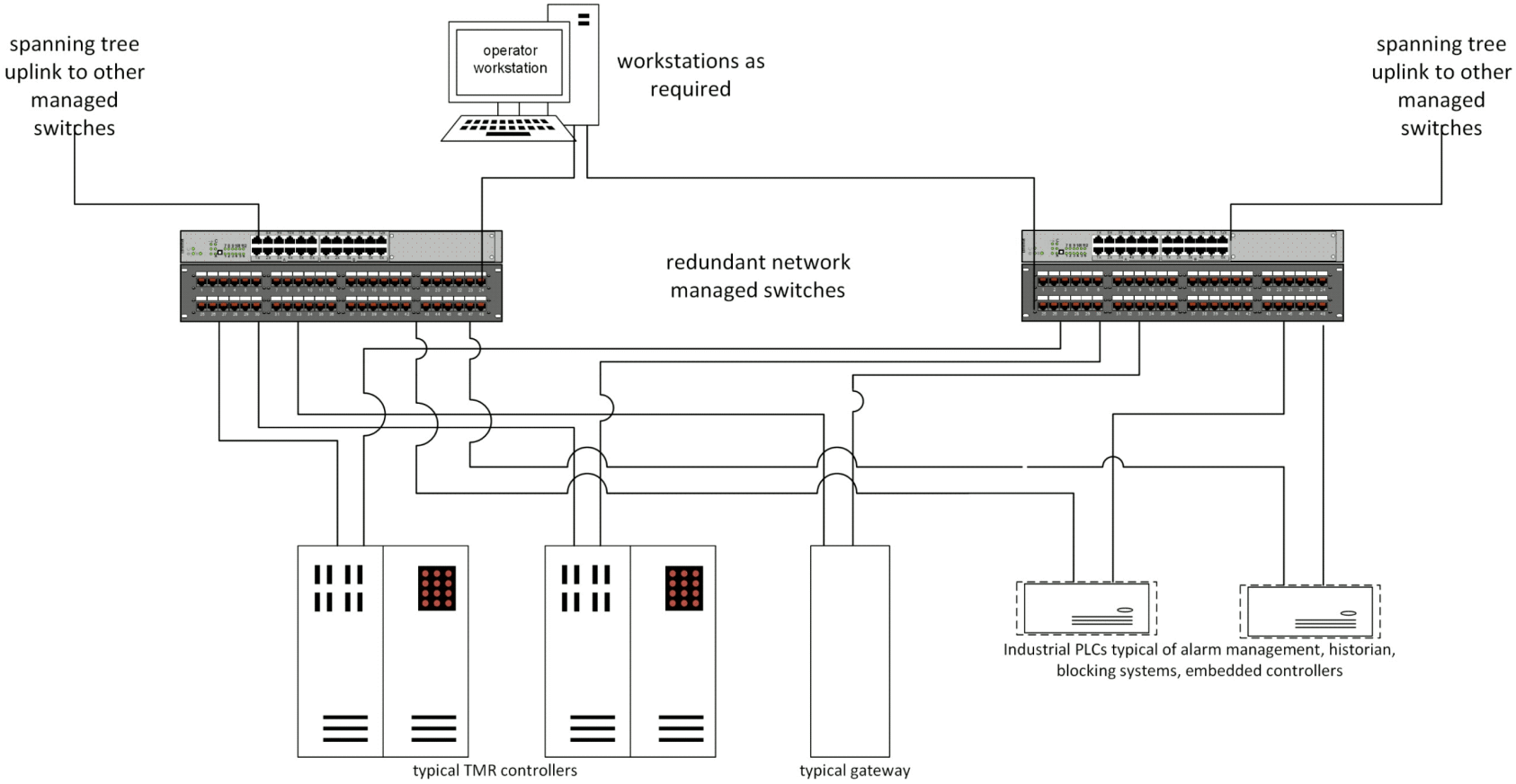

Figure 7.2-2 Typical Instrumentation and Control Controller and Component Network Connections

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.

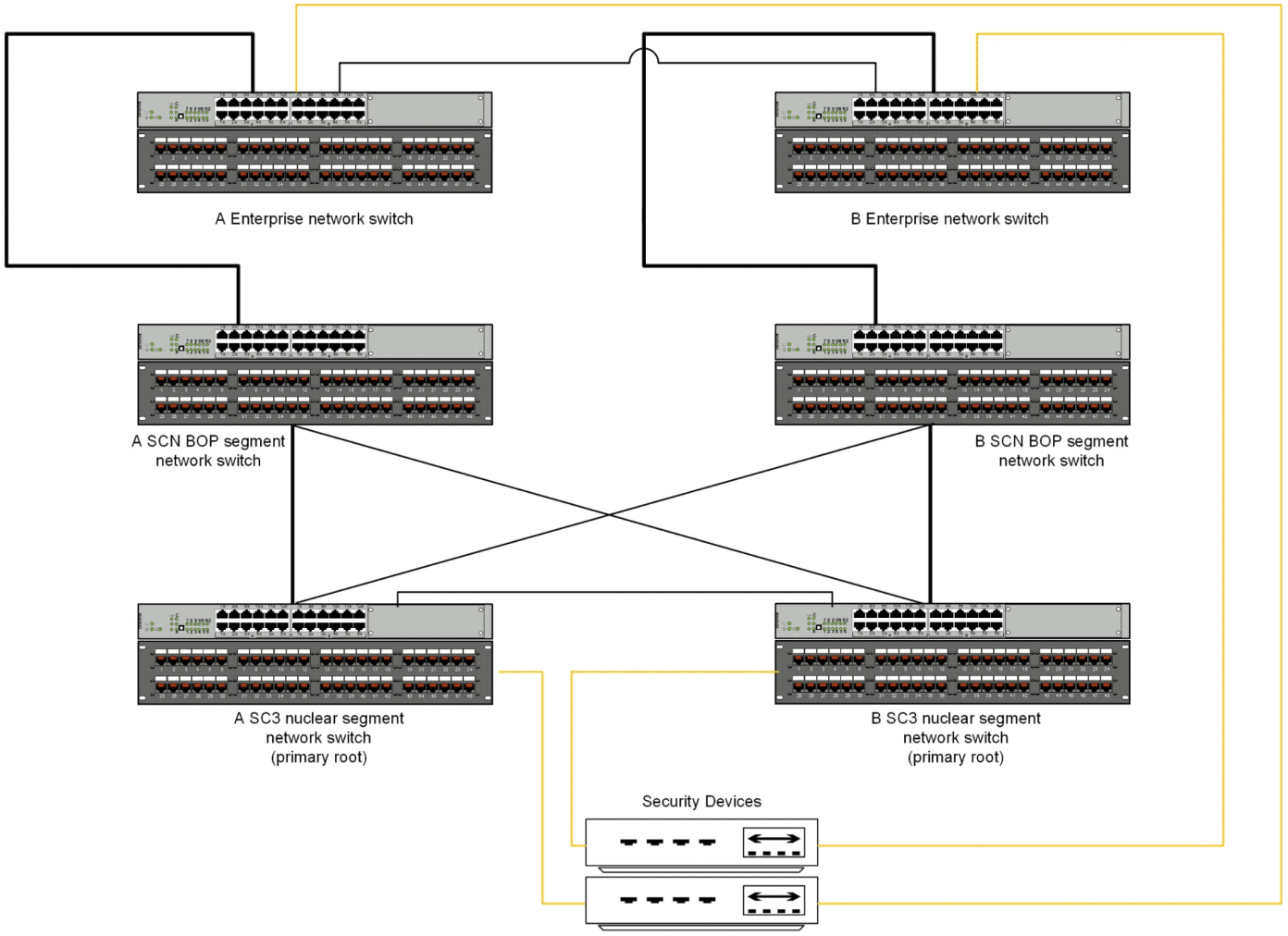

Figure 7.2-3 Rapid Spanning Tree Network Managed Switches

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 (GT) arranged radially throughout the BWRX-300 core. Each string has four LPRM detectors and eight GT 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 7.3.3.2.

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 (MSRIV), Main Steam Containment Isolation Valves (MSCIV), Feedwater Reactor Isolation Valves (FWRIV), Feedwater Containment Isolation Valves (FWCIV), 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 (IC). 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 63147:2017:2017/IEEE 497-2016, “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 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 safety class 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 Subsection 9.4.1 and Subsection 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,” (Reference 7.3-1 through Reference 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 GE Vernova Hitachi Nuclear Energy (GVH) 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 safety class 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 Section 7.5 and Section 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 “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 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 (ERIP), 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 (ERICP) – 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 (Reference 7.3-1 through Reference 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 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). 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 provides 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 "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 603-2018.

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:

   i. Bring reactor to critical

   j. Heatup and pressurization

   k. Increase power to 15%

   l. Roll turbine to rated speed

   m. Synchronize turbine-generator

   n. Increase electrical load to ~ 20%

   o. 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 generators 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 safety class boundaries when the use of data links and networks is not indicated. These nonsoftware 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 (Reference 7.3-1 through Reference 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 63147:2017/IEEE 497-2016. 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 Section 7.5 and Section 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 603-2018.

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 (Reference 7.3-1 through Reference 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 and IEEE 603-2018.

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 NEDC-33911P-A, “BWRX-300 Containment Performance,” GE-Hitachi Nuclear Energy Americas, LLC, 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 63147:2017/IEEE 497-2016, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations,” International Electrotechnical Commission/ Institute of Electrical and Electronics Engineers, 2017.

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 603-2018, “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
(Sheet 1 of 2)

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
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

Table 7.3-1 Safety Category 1 Instrumentation and Control Functions and Initiation Signals
(Sheet 2 of 2)

Initiating Signal	Reactor Mode Switch Position
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
(Sheet 1 of 3)

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

Table 7.3-2 BWRX-300 Candidate Accident Monitoring Variables
(Sheet 2 of 3)

Variable	Type	Selection Reason	Remarks
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) NEDC-33911-A, "BWRX-300 Containment Performance," (Reference 7.3-18)	Provided for severe accident monitoring
Containment Oxygen Concentration	C, F	10 CFR 50.44(c)(4) NEDC-33911-A, (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

Table 7.3-2 BWRX-300 Candidate Accident Monitoring Variables
(Sheet 3 of 3)

Variable	Type	Selection Reason	Remarks
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 FMCRDs 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
IC Pools Water Level	D	BWR legacy knowledge	N/A
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 (Sheet 1 of 2)

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
FMCRD 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
Feedwater and Condensate Pump Trip	
Detection of a FW line break	RUN, STARTUP, or SHUTDOWN
High Flux after Scram	RUN or STARTUP

Table 7.3-3 Safety Category 2 Diverse Protection System Functions and Initiating Signals (Sheet 2 of 2)

Initiating Signal	Reactor Mode Switch Position
Feedwater and SDC Isolation	
Indication of a 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
Reactor Cavity Pool Makeup from IC 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
WRNM Short reactor period (at low power)	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 I&C Compliance Alignment
(Sheet 1 of 2)

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

Table 7.3-5 Safety Class 1 I&C Compliance Alignment
(Sheet 2 of 2)

I&C Topic	IEEE 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 I&C 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 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 I&C 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 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 I&C 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

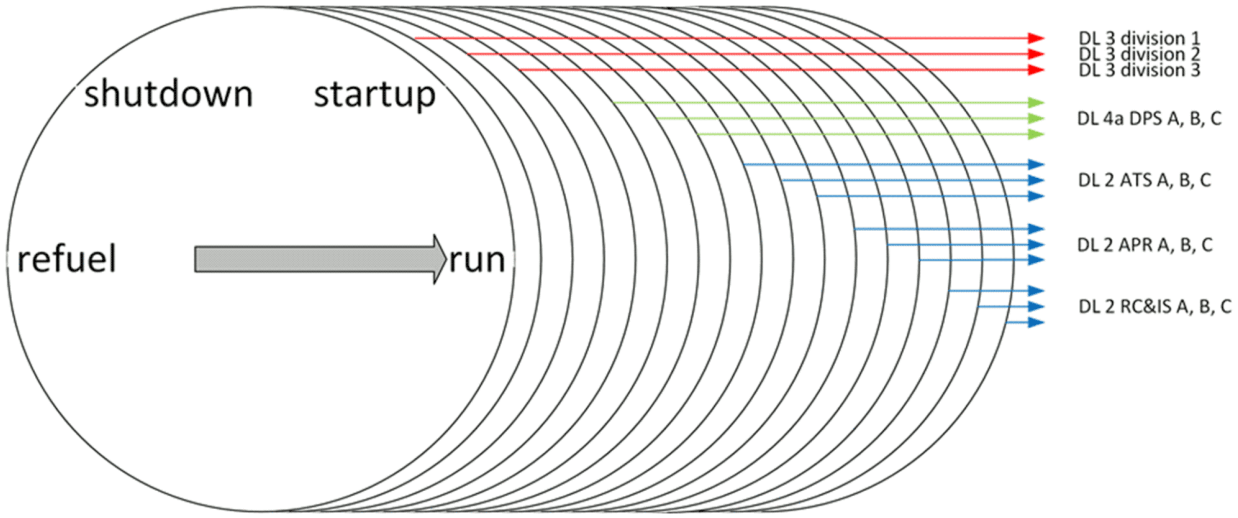

Figure 7.3-1 Reactor Mode Switch

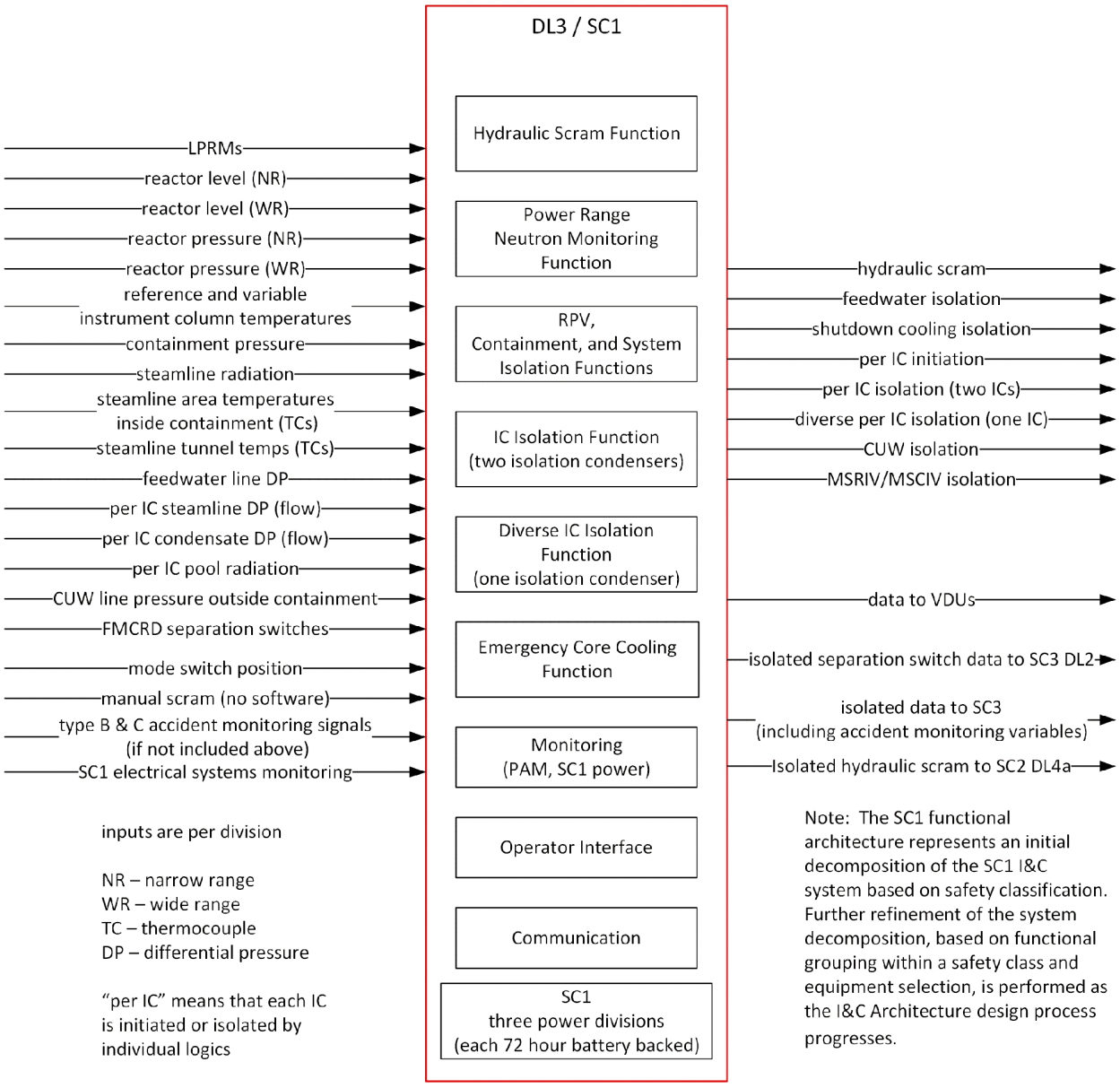

Figure 7.3-2 Safety Class 1 Functional Architecture

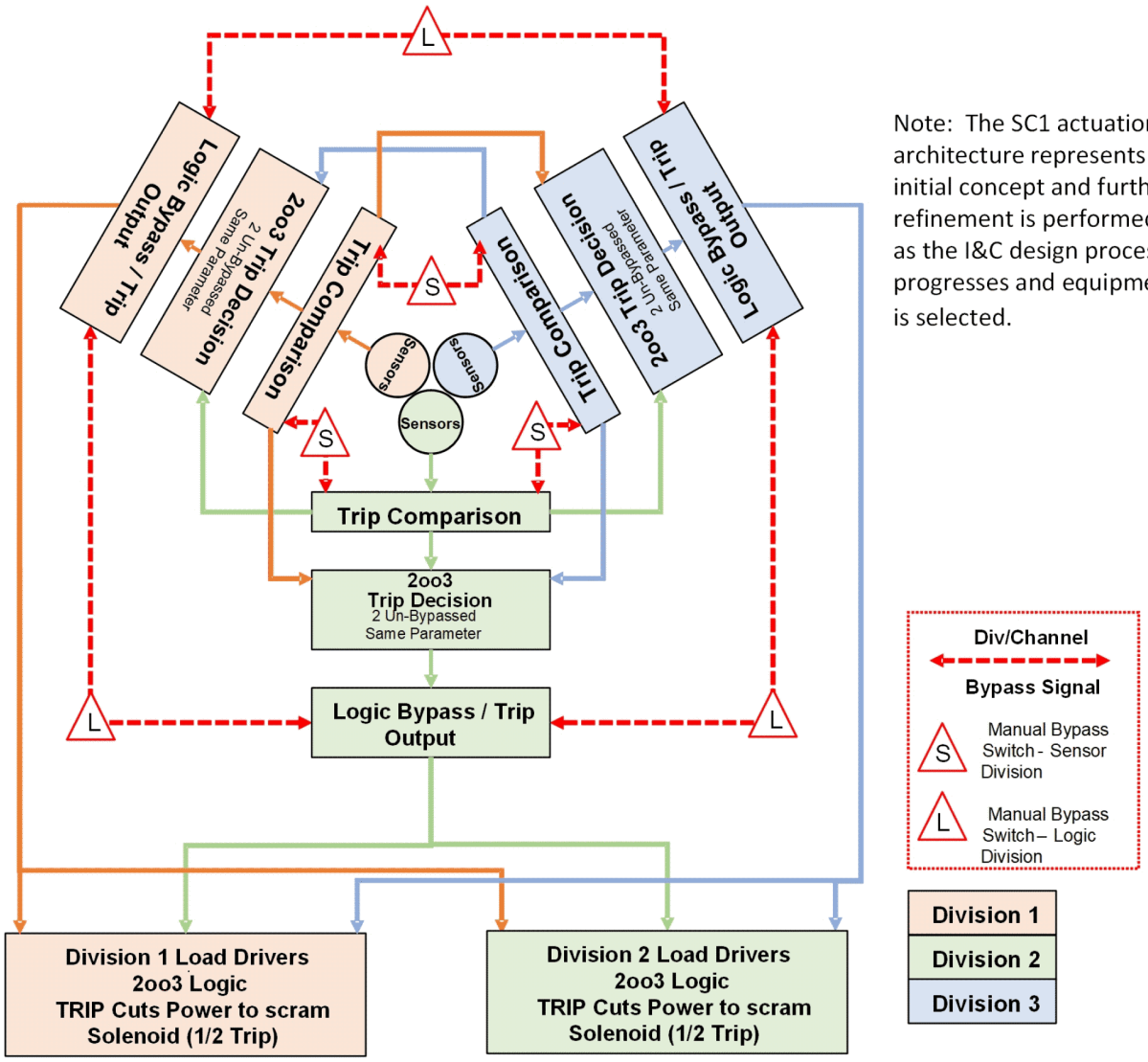

Figure 7.3-3 BWRX-300 Safety Category 1 Actuation Logic

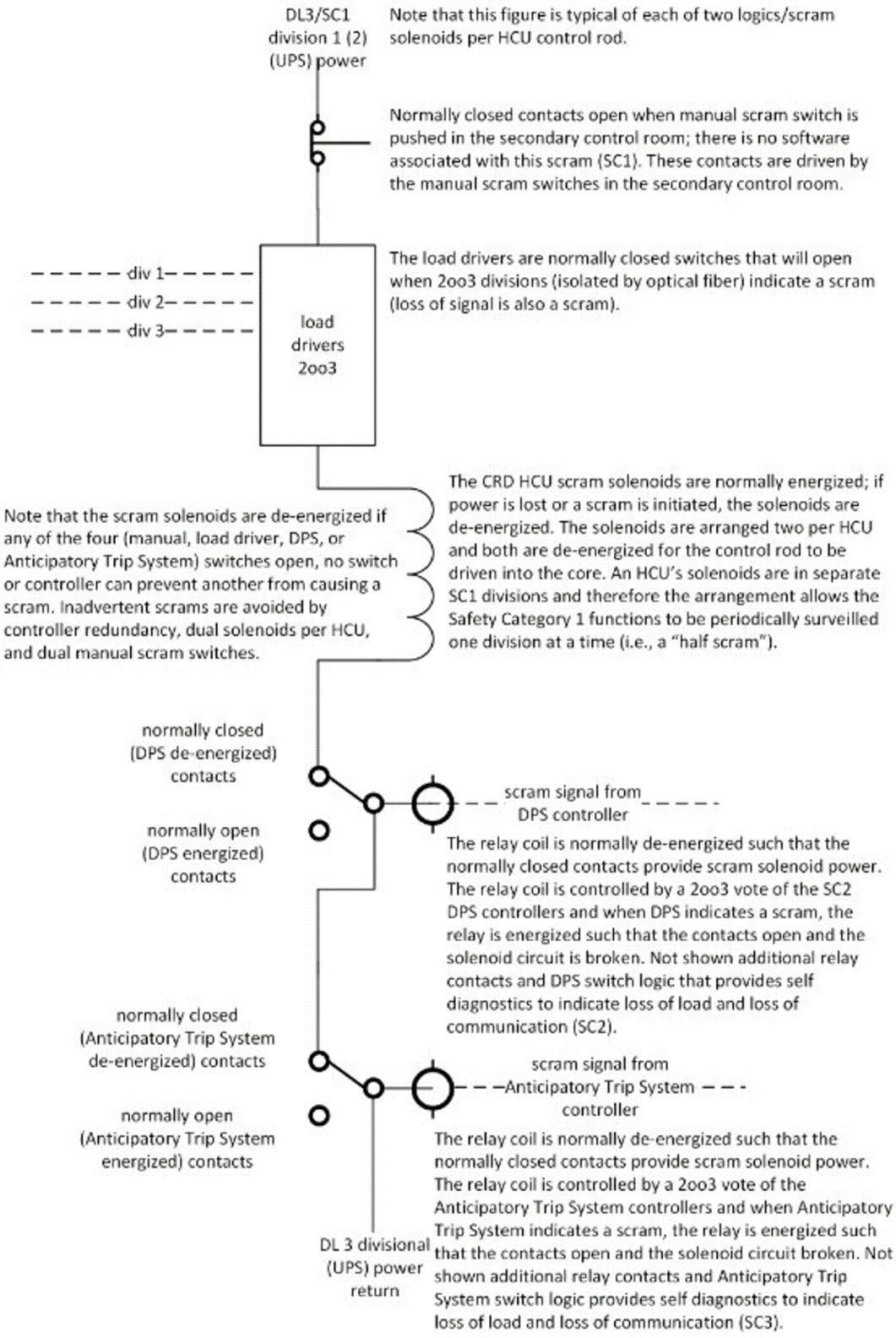

Figure 7.3-4 BWRX-300 Hydraulic Scram Solenoid Control and Power

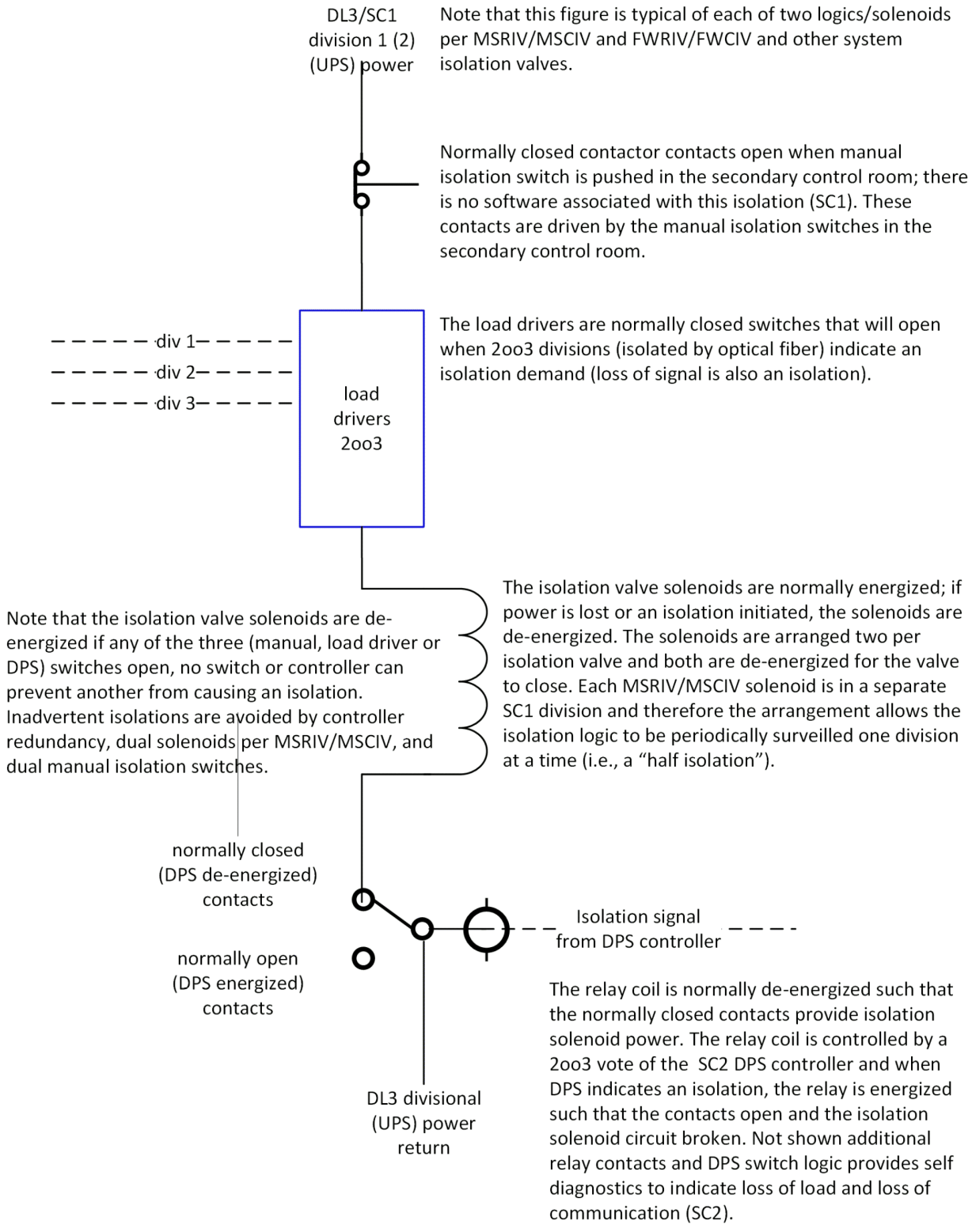

Figure 7.3-5 BWRX-300 Isolation Valve Solenoid Control and Power

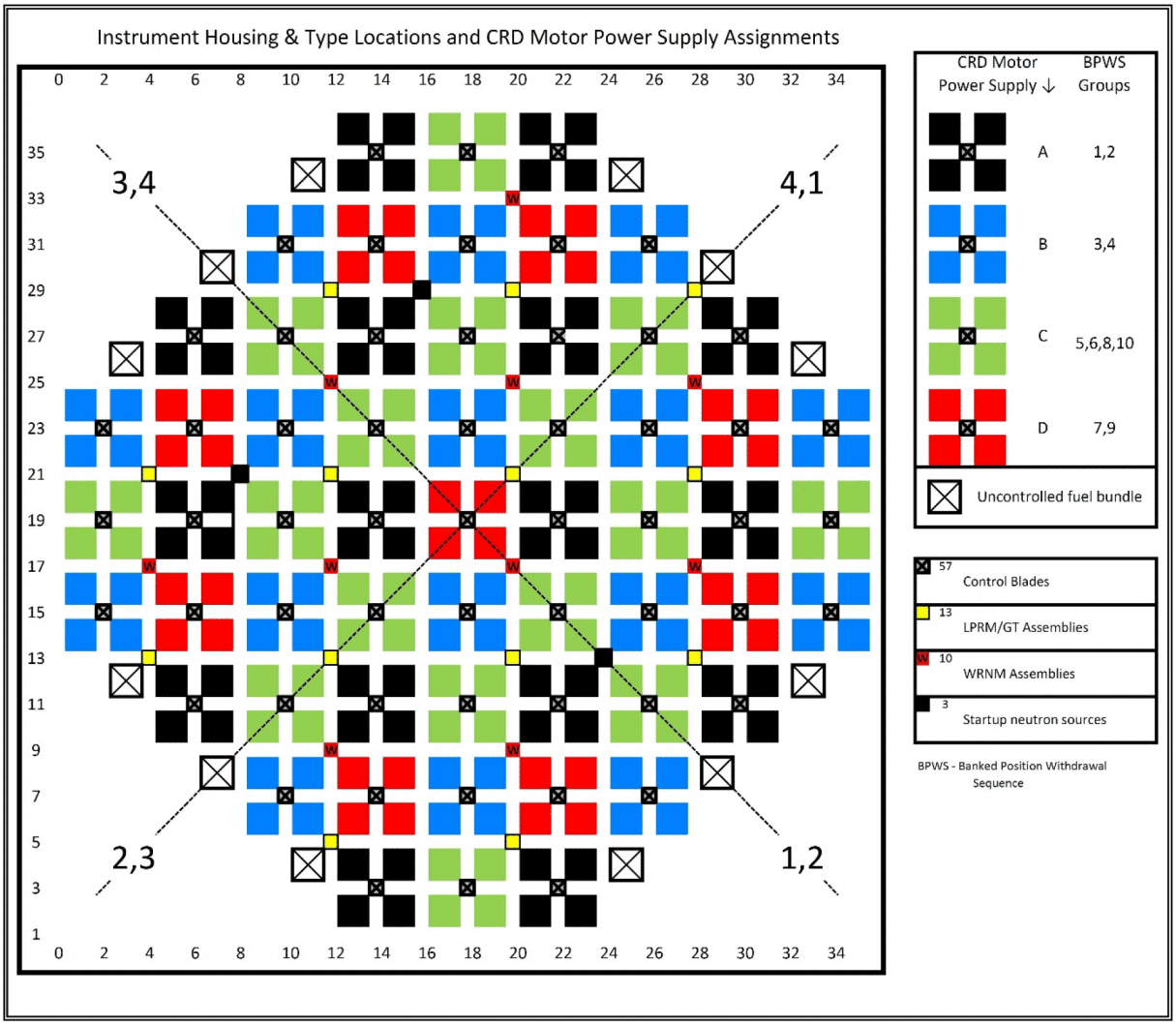

Figure 7.3-6 BWRX-300 Core Map with Monitoring Locations and Fine Motion Control Rod Drive Power Assignments

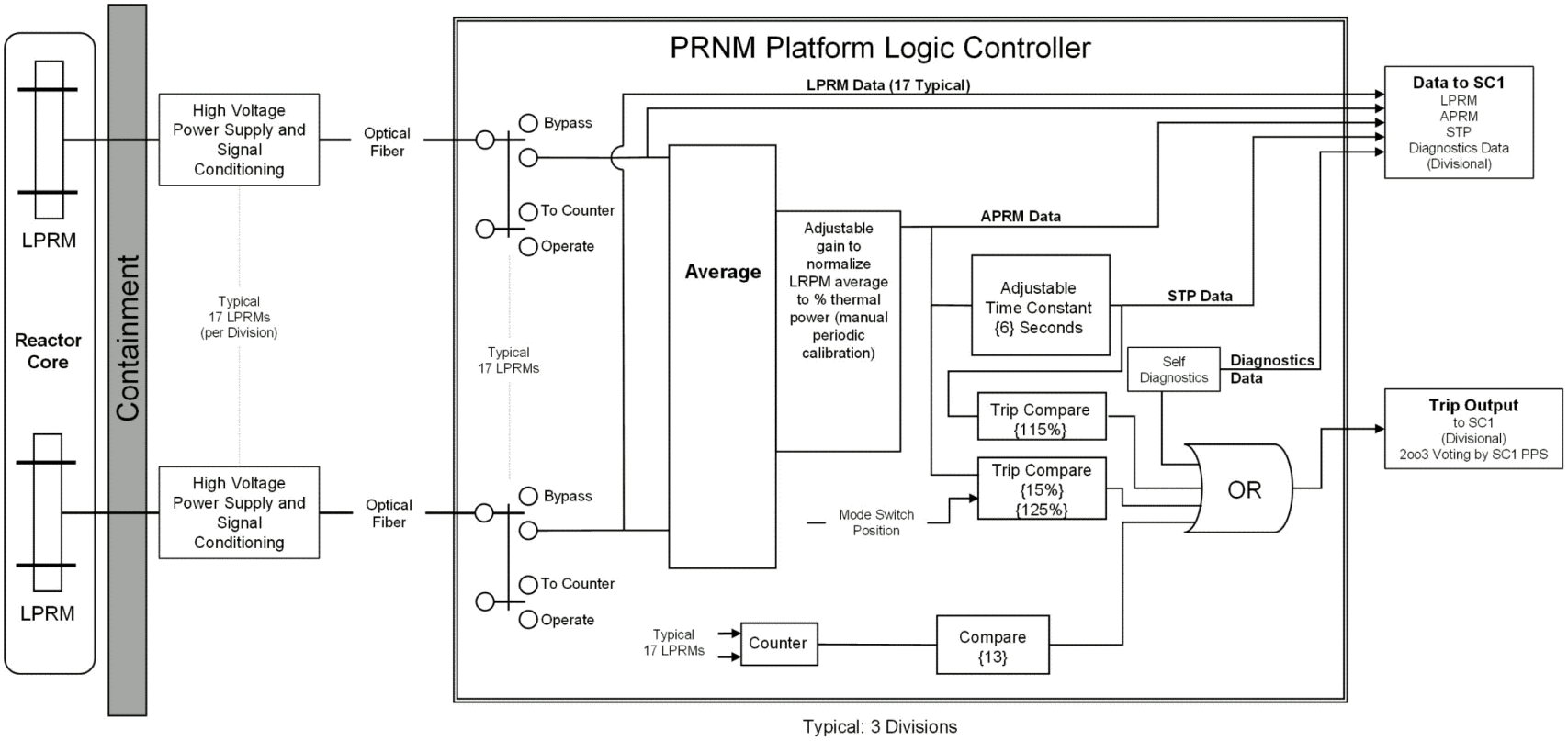

Figure 7.3-7 Representative Safety Class 1 Power Range Neutron Monitoring Block Diagram

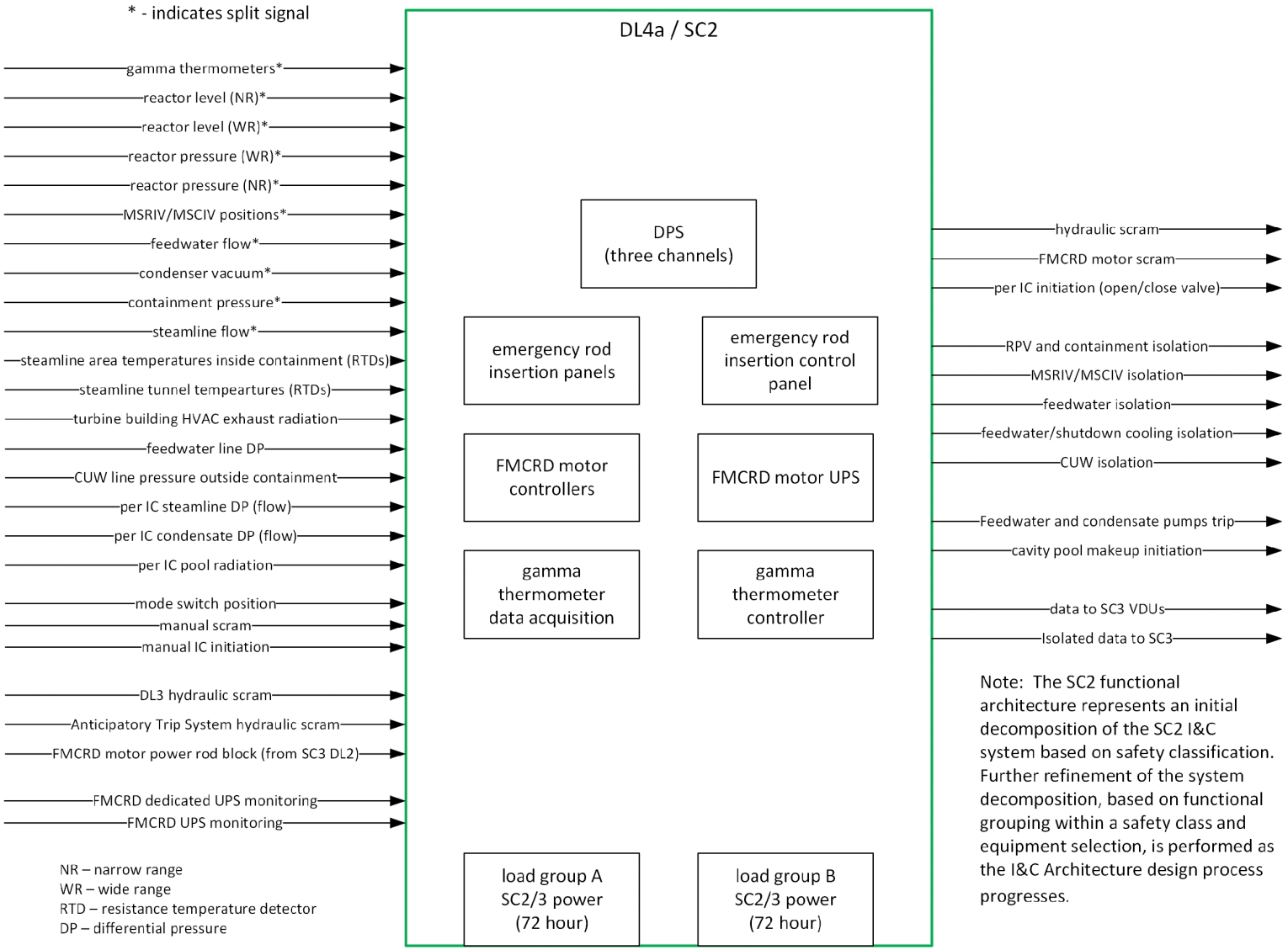

Figure 7.3-8 Safety Class 2 Functional Architecture

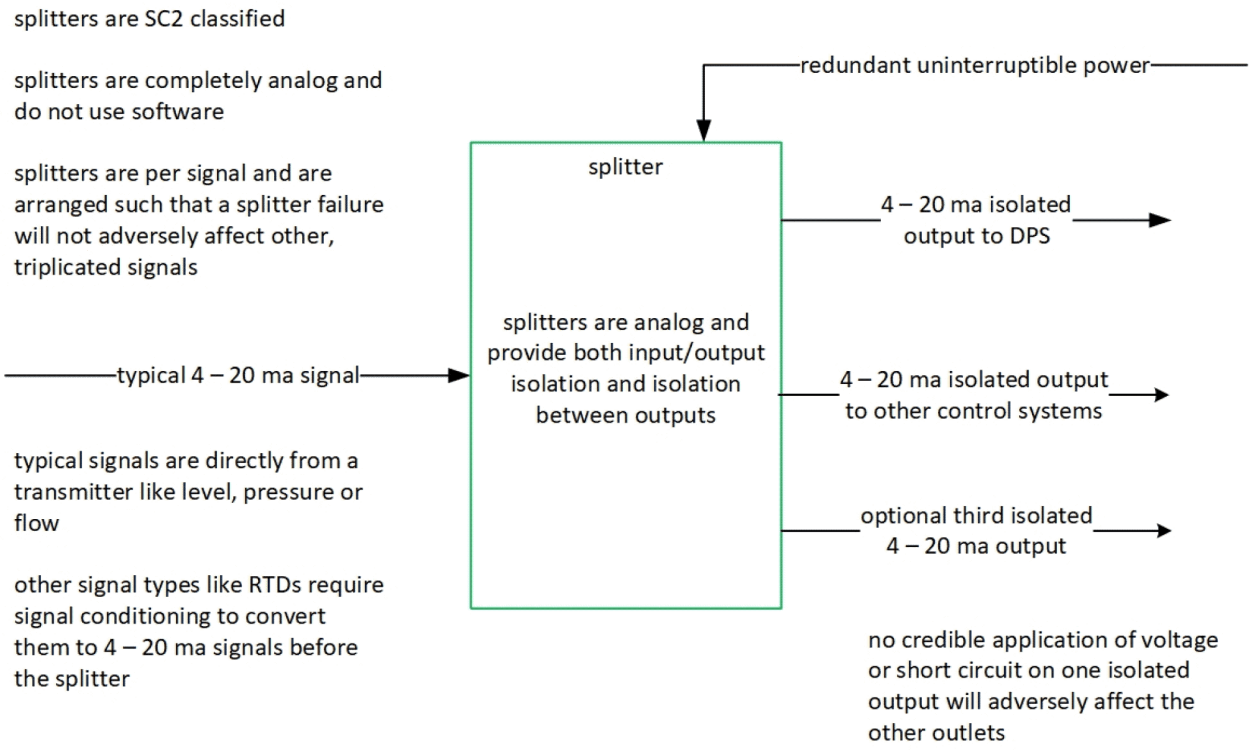

Figure 7.3-9 Safety Class 2 Analog Signal Splitters

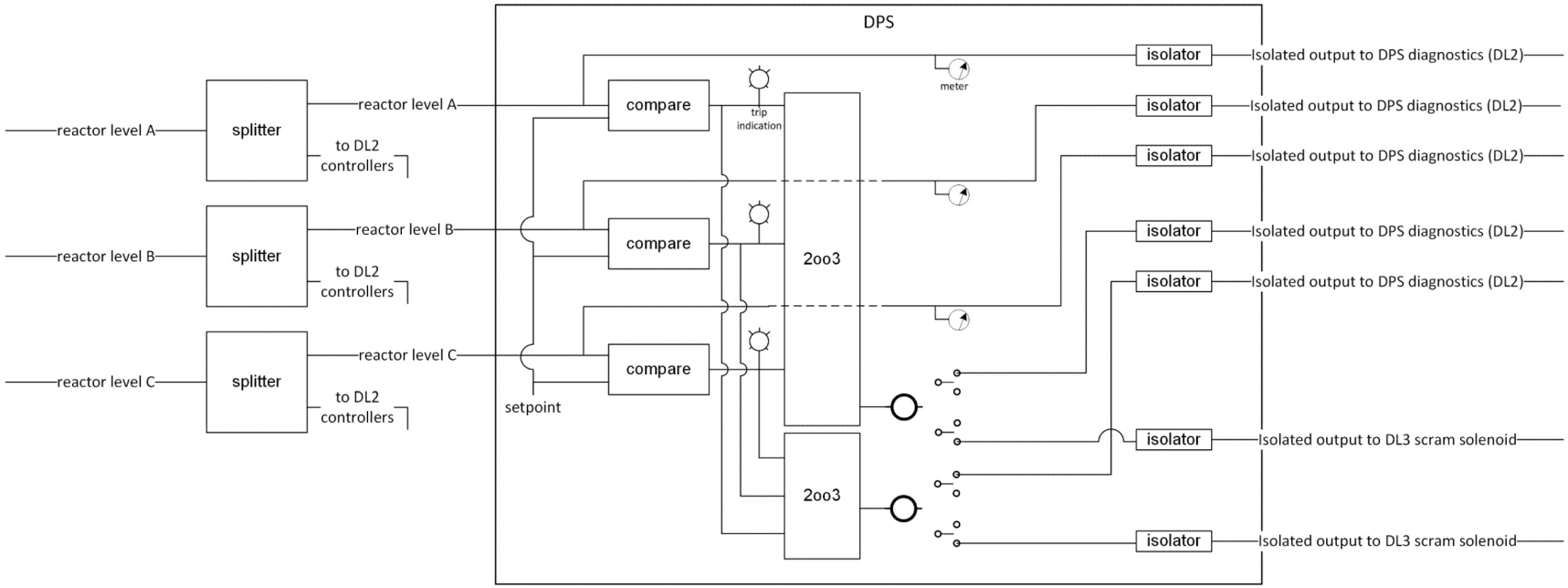

Figure 7.3-10 Representative Diverse Protection System Scram Block Diagram

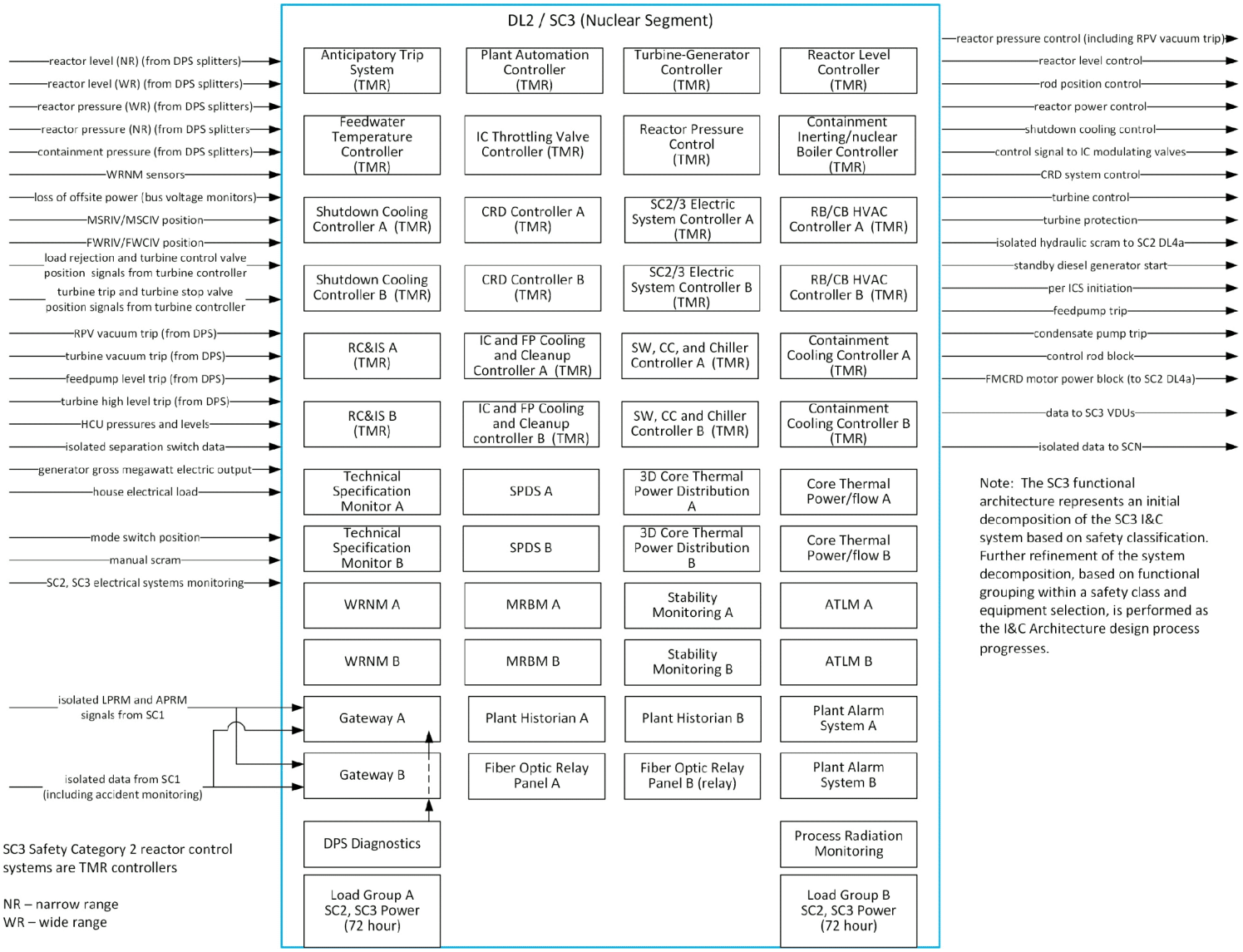

Figure 7.3-11 Safety Class 3 Nuclear Segment Functional Architecture

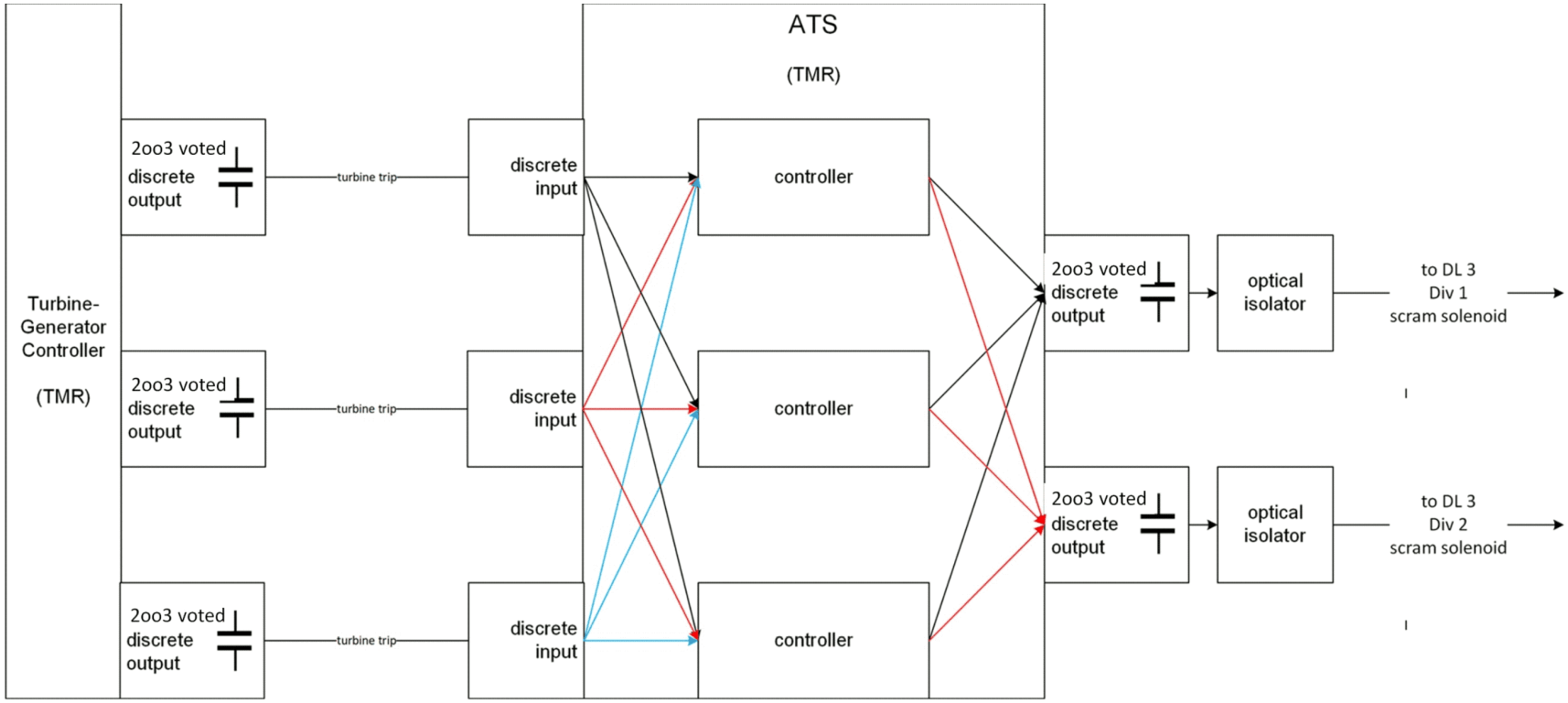

Figure 7.3-12 Representative Anticipatory Trip System Scram Block Diagram

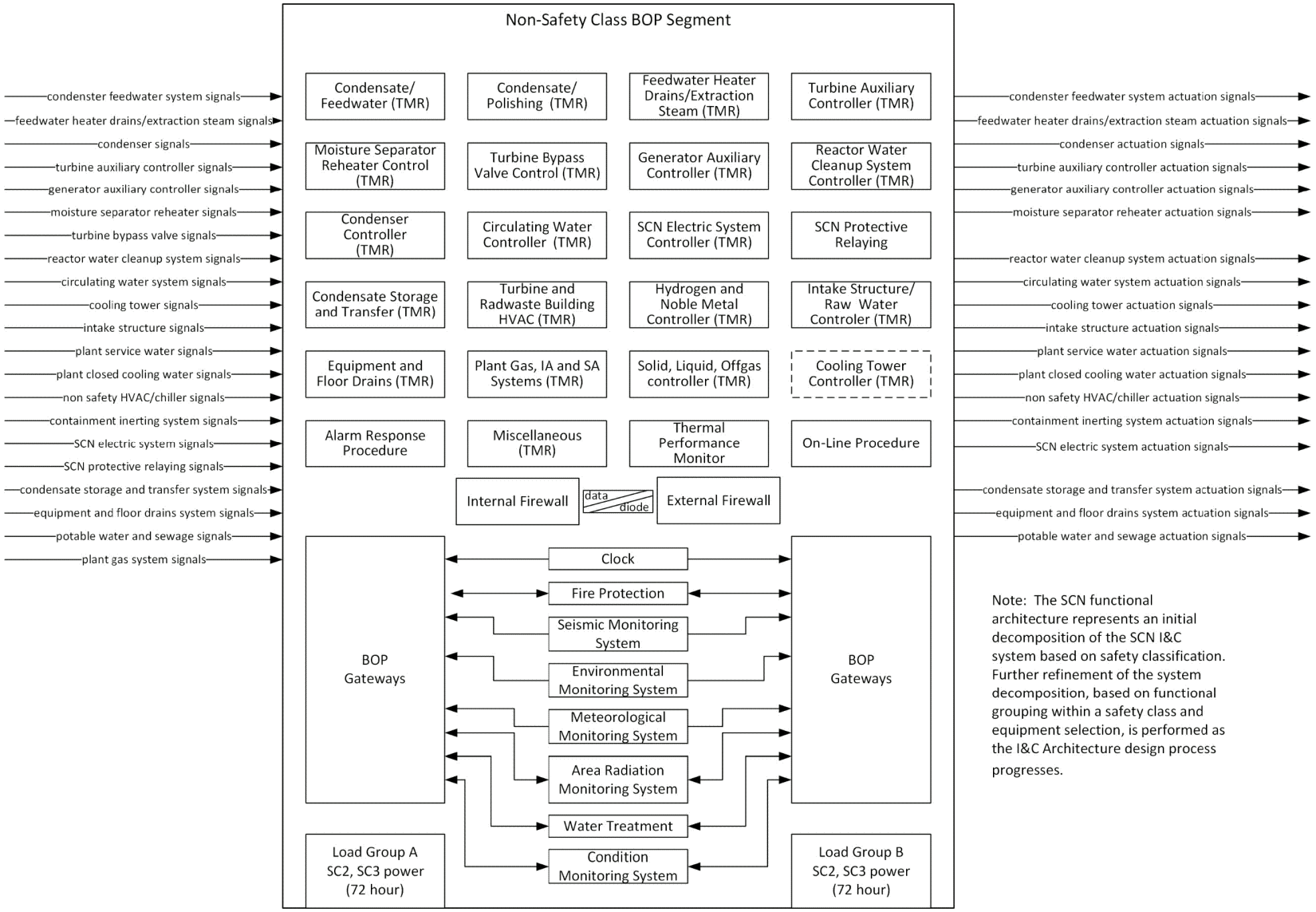

Figure 7.3-13 Non-Safety Class Balance of Plant Segment Functional Architecture

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 7.4-4).

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.” 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 “DRG: Instrumentation and Controls for Non-Light-Water Reactor (Non-LWR) Reviews,” (Reference 7.4-2) and the key industry I&C standards IEC 61513 (Reference 7.4-1) and IEEE 603-2018 (Reference 7.4-3), 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 “Design Review Guide (DRG): Instrumentation and Controls for Non-Light-Water Reactor (Non-LWR) Reviews,” U.S. Nuclear Regulatory Commission, February 2021.

7.4-3 IEEE 603-2018, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,” Institute of Electrical and Electronics Engineers, 2018.

7.4-4 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.

7.4-5 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.

7.4-6 IEEE 7-4.3.2-2003, “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 7.4-4) and IEC 62138 (Reference 7.4-5)
I&C Topic	**IEEE 603-2018 Clause (Reference 7.4-3)**
Qualified I&C	Clauses 5.3 and 5.4 as supplemented by IEEE 7-4.3.2-2003 (Reference 7.4-6)

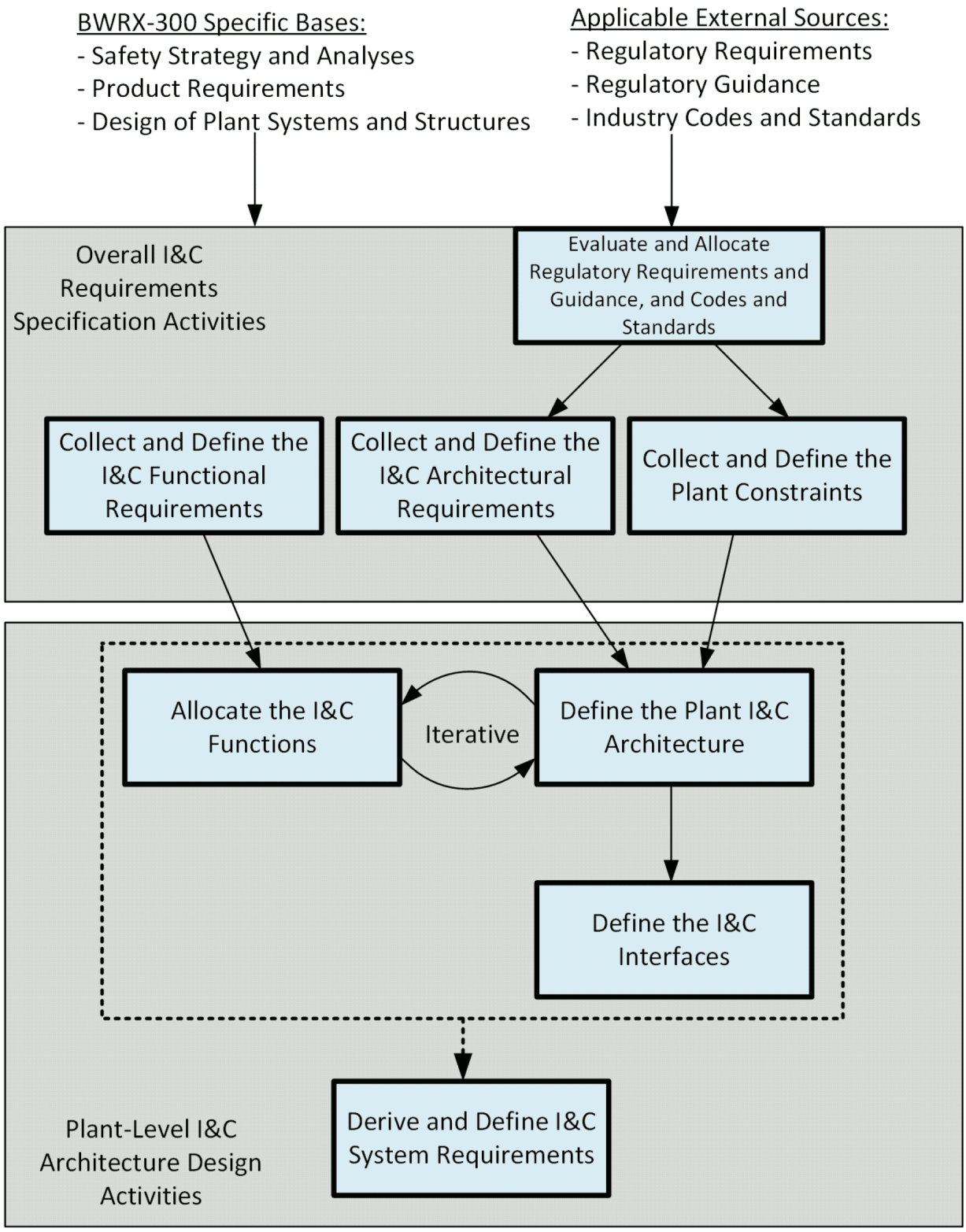

Figure 7.4-1 BWRX-300 Instrumentation and Control Architecture Design Process

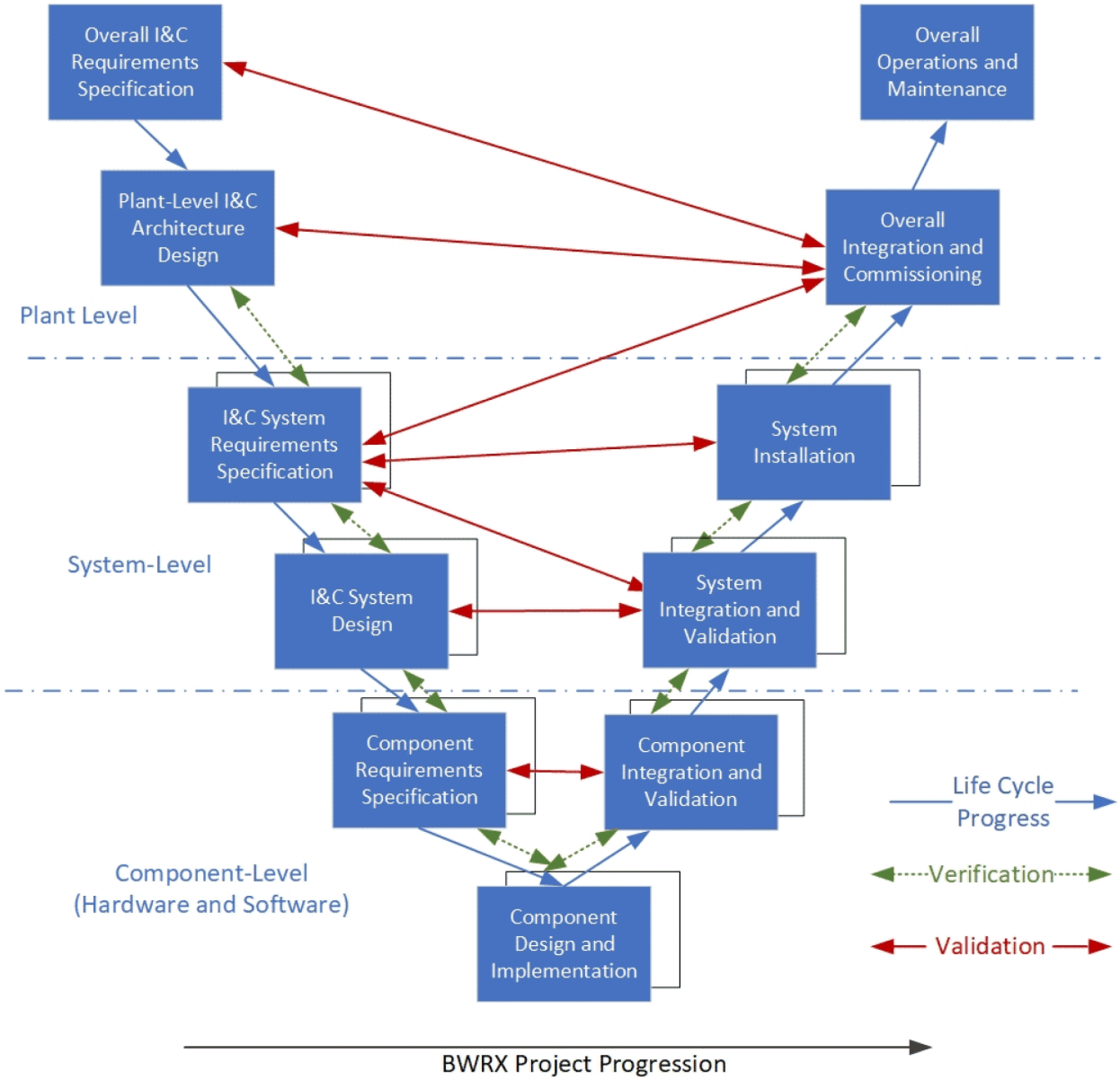

Figure 7.4-2 Overall BWRX-300 Instrumentation and Control System Life Cycle

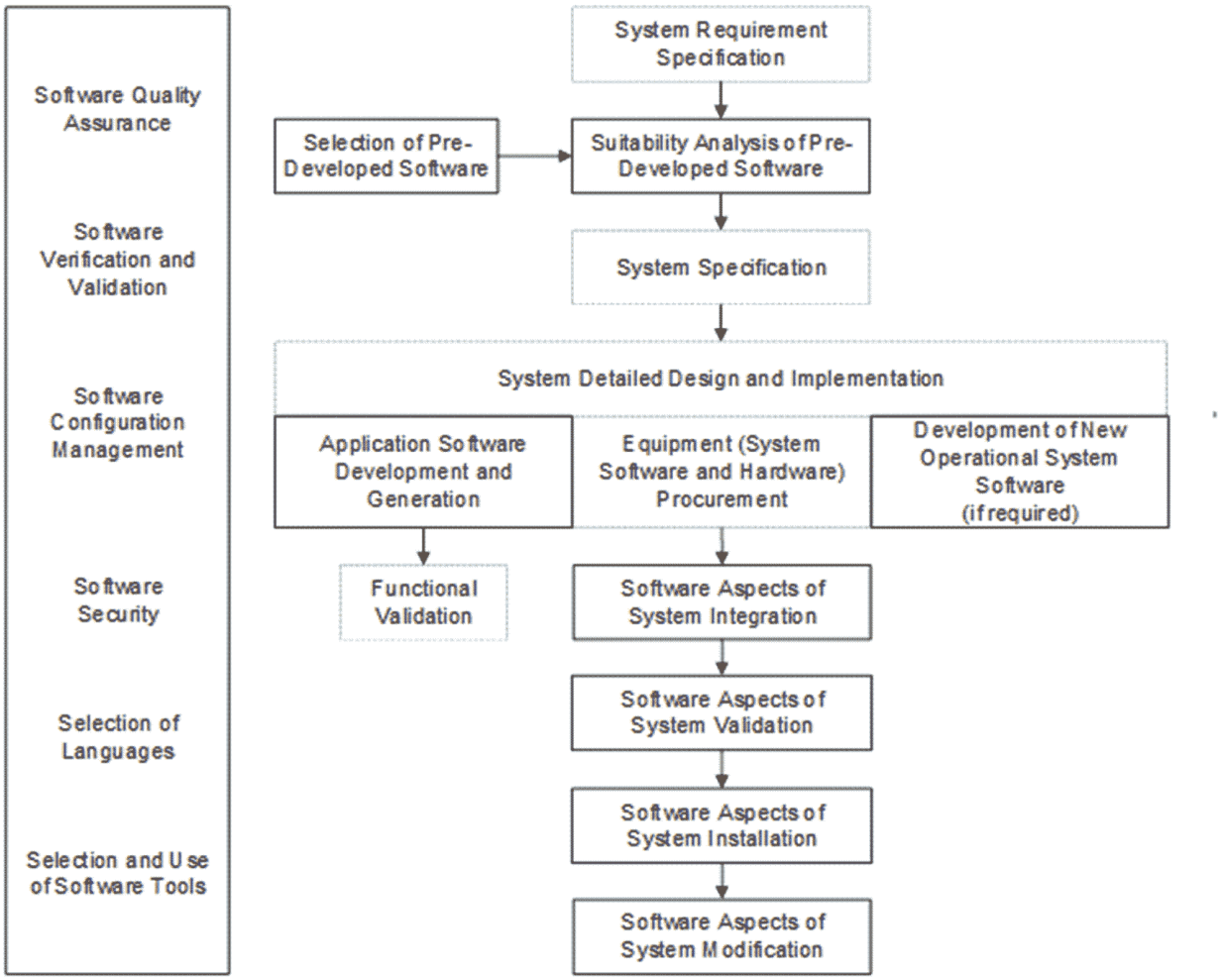

Figure 7.4-3 Software Related Activities in Instrumentation and Control System Life Cycle

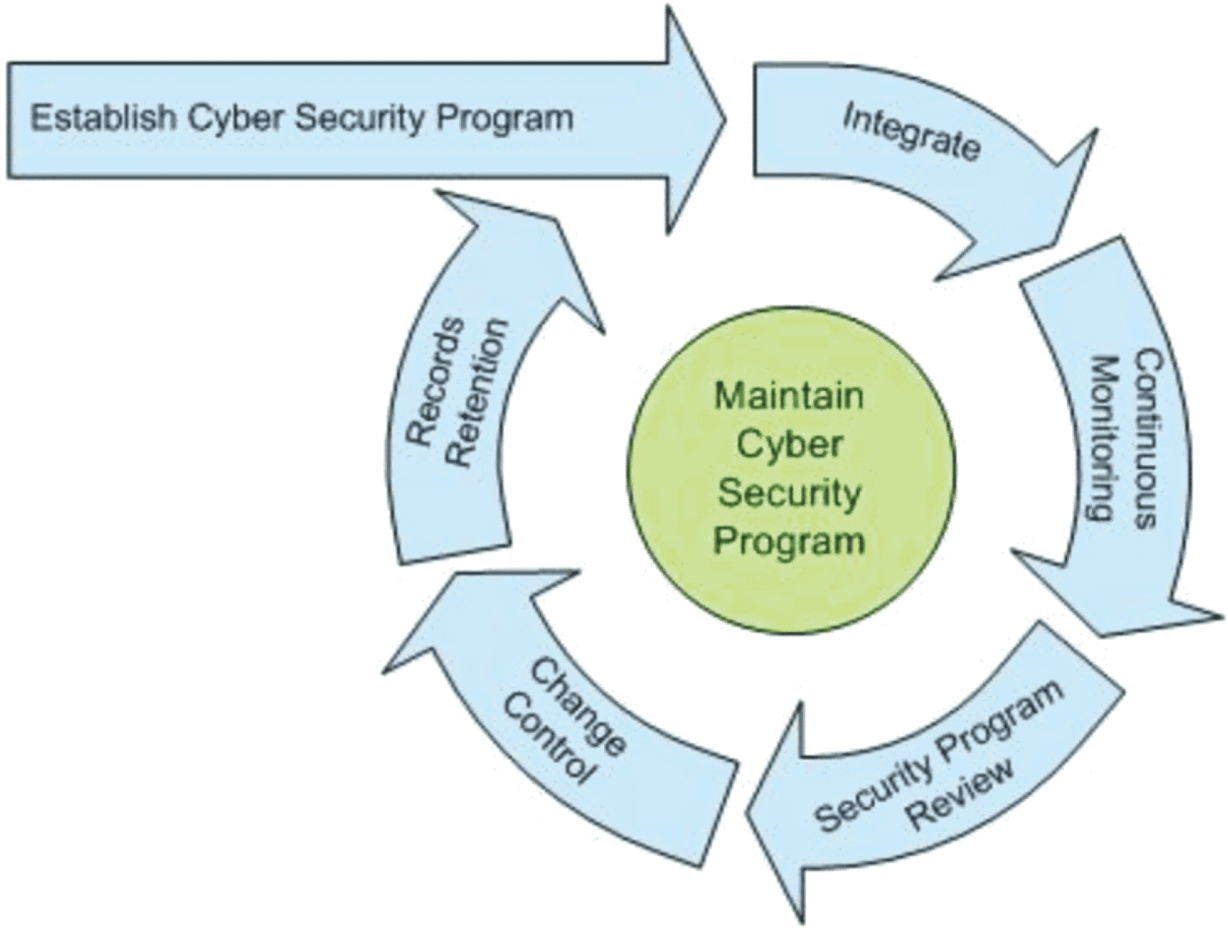

Figure 7.4-4 Cyber Security Life Cycle

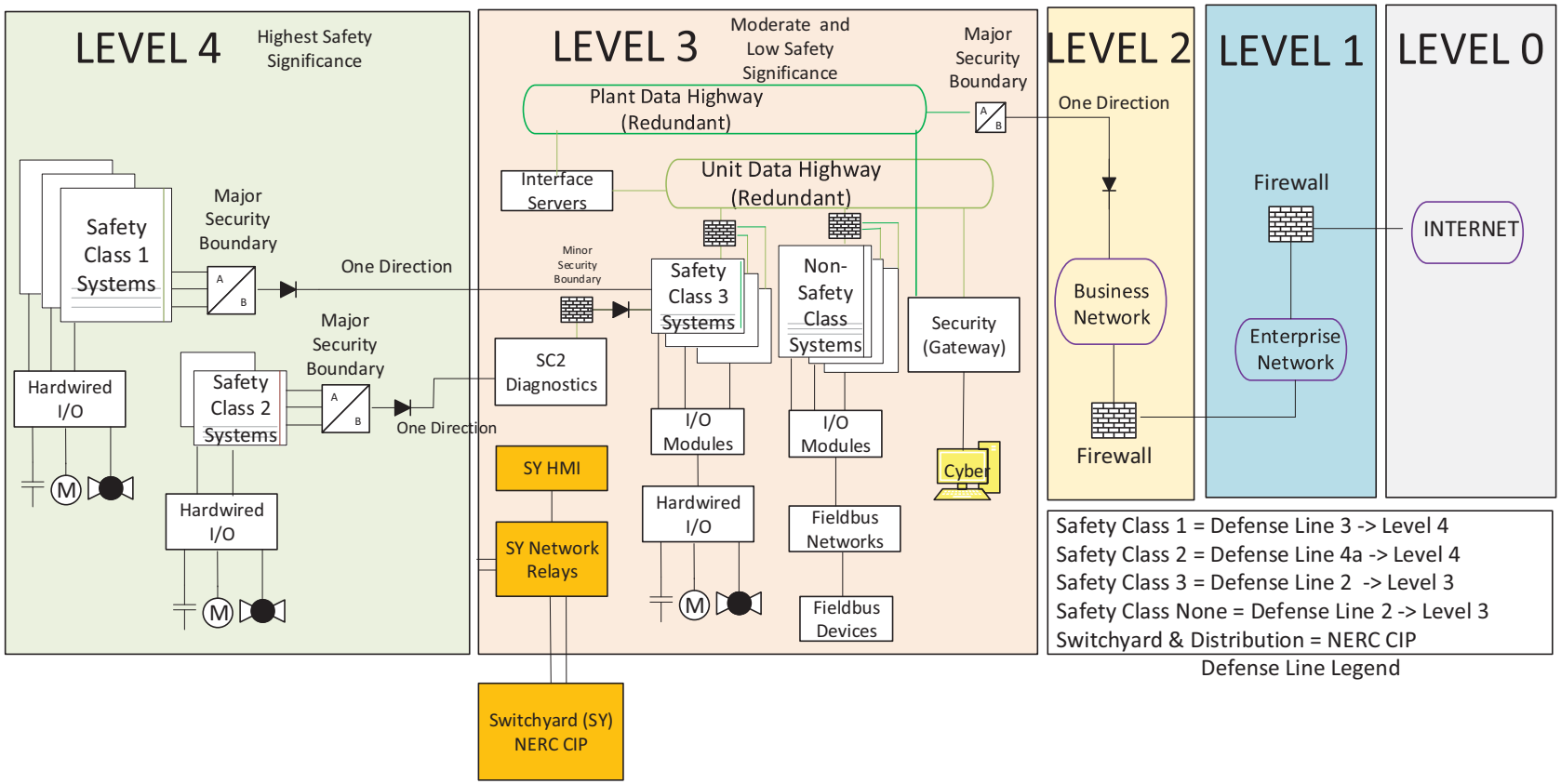

Figure 7.4-5 Cyber Security Architecture Security Boundaries

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.

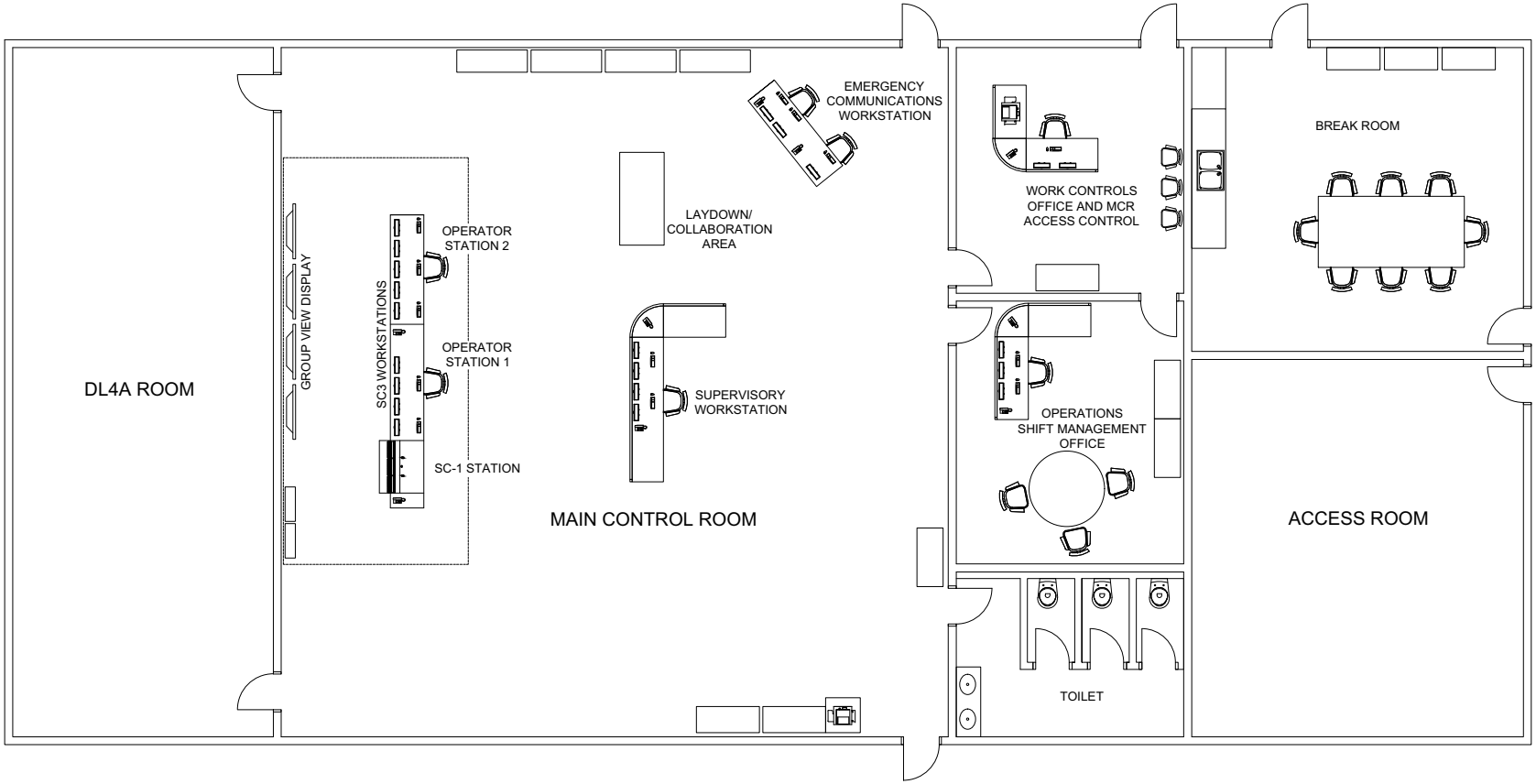

Figure 7.5-1 Main Control Room and Surrounding Areas (Top View)

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 Table 7.3-1, Table 7.3-3, and Table 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.

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 (HA).

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.

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 7B-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 7B-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 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.

Methodology

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.

7B.1 REFERENCES

7B-1 IAEA-TECDOC-668, “The Role of Automation and Humans in Nuclear Power Plants,” International Atomic Energy Agency, 1992.

7B-2 IAEA SSG-51, “Human Factors Engineering in the Design of Nuclear Power Plants,” International Atomic Energy Agency, 2019.

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LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
AC	Alternating Current
AOO	Anticipated Operational Occurrence
BOP	Balance of Plant
BTP	Branch Technical Position
CRD	Control Rod Drive
D-in-D	Defense-in-Depth
DBA	Design Basis Accident
DC	Direct Current
DCIS	Distributed Control and Information 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
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

LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
PDC	Principal Design Criteria
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 (STS)
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
VRLA	Valve-Regulated Lead-Acid

LIST OF EFFECTIVE SECTIONS

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CHAPTER 8 ELECTRIC POWER

This chapter describes the CRN-1 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 and subsystems:

- Generator and Exciter
- Preferred Power System
  - Balance of Plant (BOP) Distribution Subsystem
  - Preferred Power 250 Volts Direct Current (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 Raceway System
- Grounding and Lightning Protection System
- Non-Distributed Control and Information System (Non-DCIS) Communication System
- Freeze and Cathodic Protection System

In addition to describing the electrical systems and associated requirements, this chapter includes 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 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 DBAs.

The Defense-in-Depth (D-in-D) concept is achieved by defined safety classes that are further established in Section 3.2. 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 Safety Category 3 functions that support the control of abnormal operation and the detection of failures 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 the consequence of severe accidents. 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

Grounding and Lightning Protection System

Freeze and Cathodic Protection System

Figure 8.1-1 depicts the main BWRX-300 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 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, onsite 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 Criteria (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 onsite 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 typical BWRX-300 switchyard interface 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 Medium Voltage 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 UAT 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 standby diesel generator 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 an LOPP for loads which have a function required during an 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 GDC 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 SC1 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 standby diesel generator 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 Control Rod Drive (CRD) system pumps can provide reactor coolant makeup 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). 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 standby diesel generator 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 RIV and ICS control power is supplied from the Emergency Power System (SC1 DC Power Subsystem and UPS Subsystem). Neither offsite nor standby diesel generator 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 RIV and ICS control power is supplied from the Emergency Power System (SC1 DC Power Subsystem and UPS Subsystem). Neither offsite nor standby diesel generator 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) Action Item I.D.3, Safety System Status Monitoring: This regulation is met by compliance with RG 1.47 and BTP 8-5, and compliance with IEEE 603-2018.

10 CFR 50.34(f)(2)(xiii), TMI Action Item II.E.3.1, Emergency Power Supply for Pressurizer Heater: This regulation is applicable only to Pressurized Water Reactors (PWR) 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 regulation is applicable only to PWRs and does not apply to the BWRX-300.

10 CFR 50.55a, Codes and Standards: This regulation is 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, standby diesel generator 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

U.S. Nuclear Regulatory Commission (USNRC) endorsed standards are used to show conformance to applicable USNRC Regulatory Guides (RGs). Table 1.9-20 lists conformance to 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, “IEEE Standard Criteria for Safety Systems for Nuclear Power Generation Stations,” 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 standby diesel generator 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 to the guidance 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 are 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 recommendations 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 requirements of IEEE 603-2018. See also conformance to 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 to the 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 standby diesel generator AC power needed to achieve safe shutdown. Therefore, the NUREG is not applicable. However, D-in-D principles such as redundancy and diversity are incorporated in the design and integration of BWRX-300 systems.

BTP 8-1, Requirements on Motor-Operated Valves in the ECCS Accumulator Lines: The BWRX 300 design 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 8.3-33) 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 SC1 equipment 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: The BWRX-300 design does not include SC1 diesel generators. 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 conforms to 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 Vernova Hitachi Nuclear Energy Americas, LLC, Revision 2, March 2026.

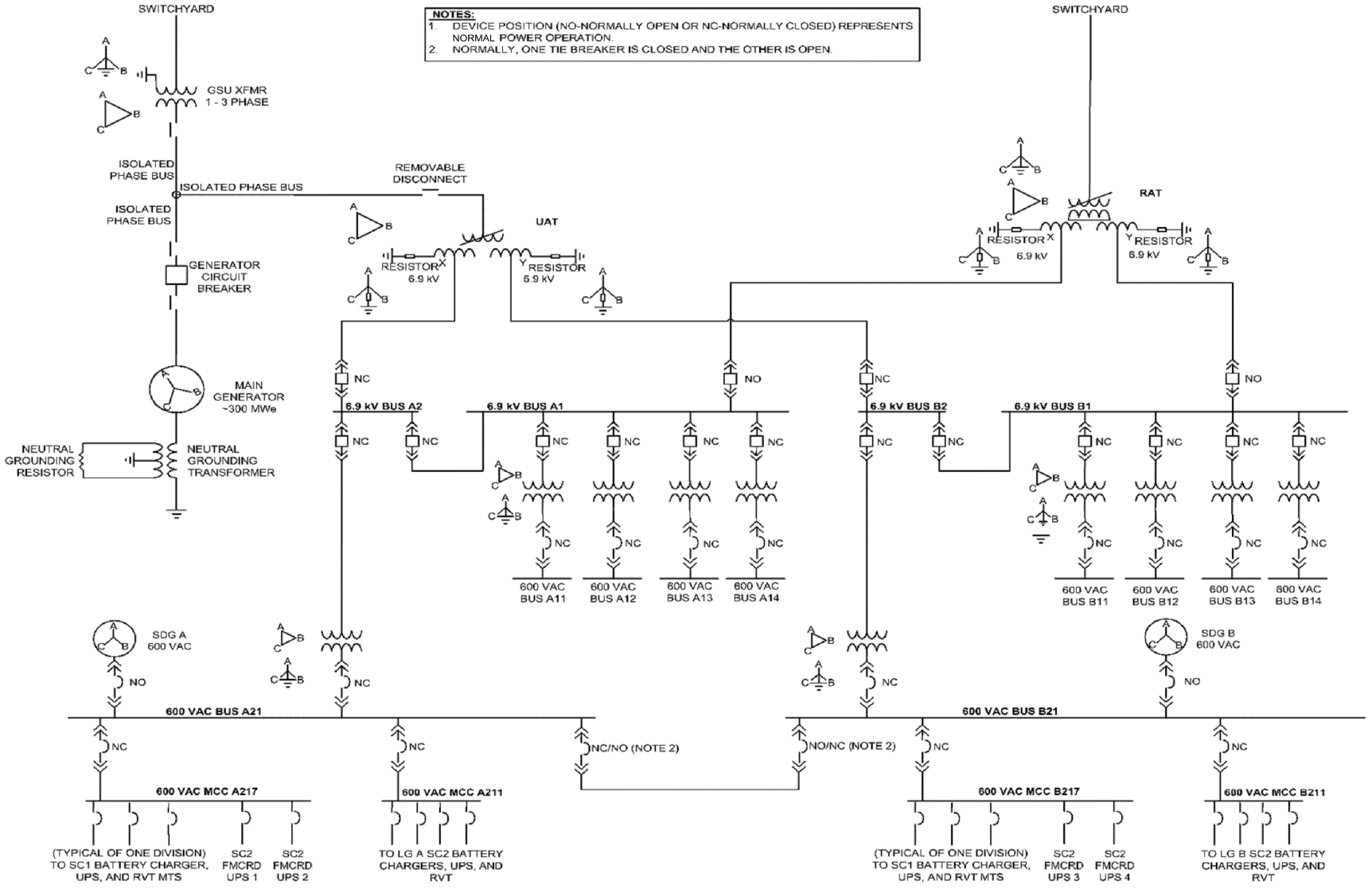

Figure 8.1-1 Main Electrical One-Line

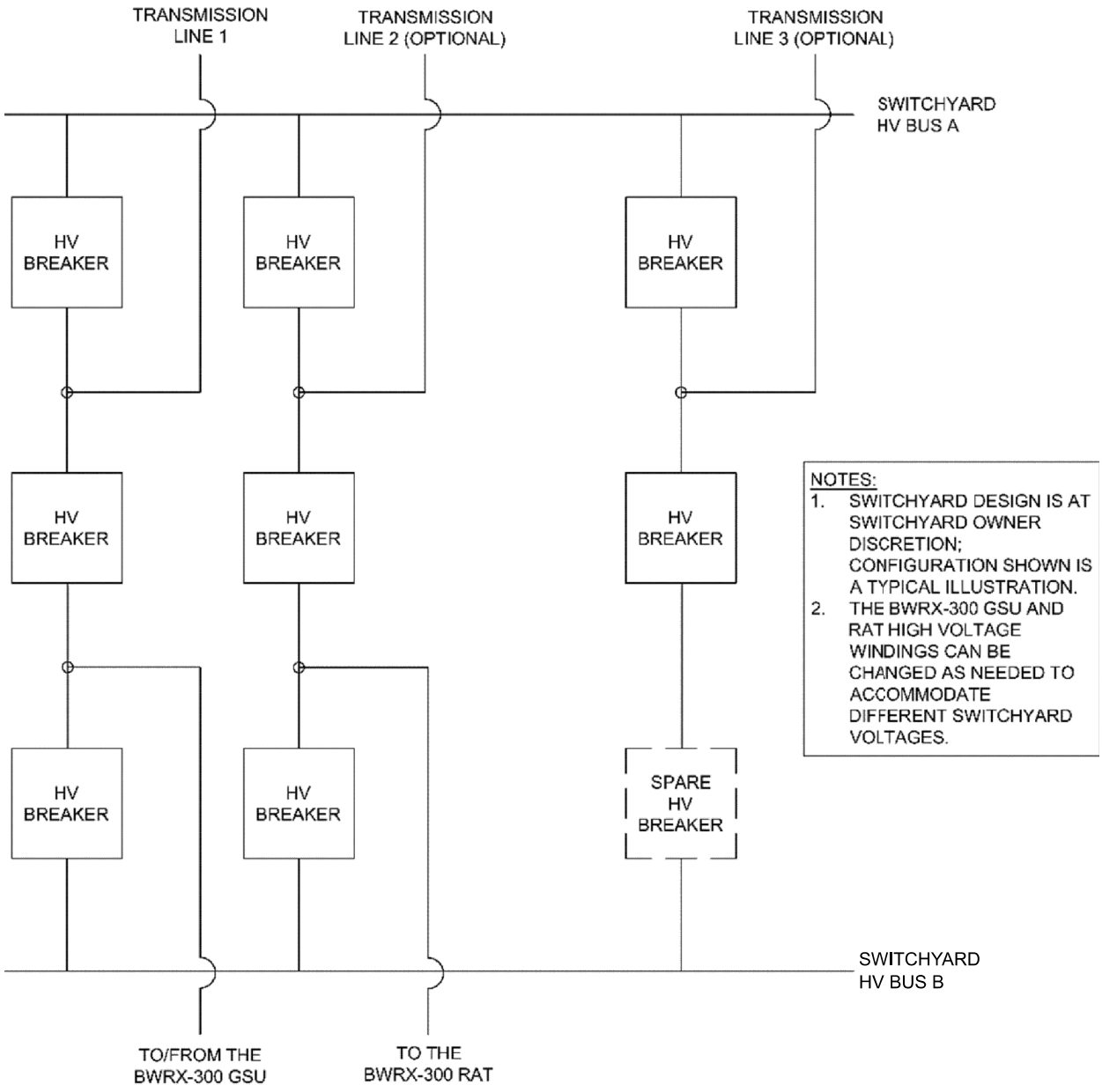

Figure 8.1-2 Typical BWRX-300 Switchyard Interface

8.2 Offsite Power Systems

Power is supplied to the plant from the switchyard connected to the transmission grid offsite power sources. The plant 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 IEC/IEEE 62271-37-013, “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 this Preferred Power System 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. The system is designed and located to minimize the likelihood of simultaneous failure during DBAs and adverse environmental conditions. There are multiple transmission lines that interconnect the switchyard to the grid.

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. Line and bus relaying has sufficient redundancy and diversity to prevent the failure of a breaker to open from causing a loss of the switchyard and total offsite power to the plant. The switchyard breaker control and protective relaying is in a weather-enclosed switchyard house with its own power source for control and protection.

The preliminary switchyard design includes four 161 KV transmission lines which interconnect the switchyard to the grid; Fort Loudoun, Kingston 1, Kingston 2, and Bethel Valley lines. The Fort Loudoun line is on a separate right of way from the other three lines. The Kingston 1, Kingston 2, and Bethel Valley lines share right of way. Kingston 2 and Bethel Valley lines are on the same transmission structures at the Clinch River Site. The Fort Loudoun line enters the yard

from the opposite direction of the other lines and does not cross the other lines to the switchyard. Each of the lines connect to a separate section of the switchyard bus and each line can be isolated from the switchyard individually without removing the other lines from service.

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 AC power system is difficult to detect and has the potential to adversely affect downstream equipment. The BWRX-300 conforms to the guidance provided within BTP 8-9 for new reactors by including open phase detection for both the GSU and RAT. The open phase detection detects an open phase during operational electrical system configurations and plant load conditions. Following detection of an open phase, an alarm is generated in the Main Control Room (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, “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 confirm 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 standby diesel generator AC power 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 IEC/IEEE 62271-37-013, “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, “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 safety 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 onsite AC power system and electrical distribution equipment. Figure 8.1-1 shows the main electrical one line diagram. The onsite AC power 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 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, onsite AC power system, and the offsite power source. In addition, the Preferred Power System includes the entirety of the Balance of Plant (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 onsite AC 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 onsite AC power system includes four 6.9 kV MV buses. The first supply is connected through the UAT to the MV buses. The second supply 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 Preferred Power System 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 250 VDC Subsystem. The primary purpose of this subsystem is to provide backup power in the event of an LOPP for loads which have a function required during an 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 the three SC1 EDS 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 the 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 MG 1-2021, “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 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 qualified as discussed 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 the subsystem prevents or decreases the rate of corrosion of a metal in an electrolyte.

The Freeze and Cathodic Protection System is classified as 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 Grounding Subsystem consists of a plant ground grid and grounding 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 establish equal potential and are 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 Subsystem 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 plant 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 Grounding Subsystem and Lightning Protection Subsystem are designed and installed to the applicable sections of the following guides and standards:

1. IEEE 80-2013, “IEEE Guide for Safety in AC Substation Grounding,” (Reference 8.3-3).
2. IEEE 81-2012, “IEEE Guide for Measuring Earth Resistivity, Ground Impedance, and Earth Surface Potentials of a Grounding System,” (Reference 8.3-4).
3. IEEE 665-1995, “IEEE Guide for Generation Station Grounding,” (Reference 8.3-5).
4. IEEE 1050-2004, “IEEE Guide for Instrumentation and Control Equipment Grounding in Generating Stations,” (Reference 8.3-6).
5. NFPA 780, “Standard for the Installation of Lightning Protections Systems,” (Reference 8.3-7).
6. IEEE C62.23-2017, “IEEE Application Guide for Surge Protection of Electric Generating Plants,” (Reference 8.3-8).

7. IEEE 666-2007, “IEEE Design Guide for Electric Power Service Systems for Generating Stations,” (Reference 8.3-9).

8. IEEE 142-2007, “IEEE 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 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 (Reference 8.3-11). This supports conformance to 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 1202-2023, “IEEE 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 1202-2023.

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 onsite 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 Preferred Power 250 VDC 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 onsite DC power system description for the SC1 DC Power Subsystem and 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 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 provide assurance that 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 the three Emergency Power System divisions. By division, SDG sizing considers one charger operating at rated capacity. Independent of offsite or Standby Power System, 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 the 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, “IEEE Standard for Criteria for the Protection of Class 1E Power Systems and Equipment in the Nuclear Power Generating Stations,” (Reference 8.3-15).

Note that IEEE 741-2022 Annex C and Annex D are not applicable to CRN-1 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 Safety 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 Series, “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, “IEEE 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 (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, “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 verify 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 conformance to RG 1.204.

In accordance with IEEE 1050-2004 (Reference 8.3-6), adequate filtering is provided by the battery charger, UPS, and regulating voltage transformers to prevent adverse effects on other plant equipment through the propagation of Electromagnetic Interference/Radio Frequency Interference (EMI/RFI) of the levels allowed by IEC 61000-3 Series, “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 conformance to RG 1.6.

The Emergency Power System batteries meet the qualification requirements of IEC/IEEE 60780-323, “Nuclear Facilities – Electrical Equipment Important to Safety – Qualification,” (Reference 8.3-26), IEC/IEEE 60980-344, “Nuclear Facilities – Equipment Important to Safety – Seismic Qualification,” (Reference 8.3-27), and IEEE 535-2022, “IEEE 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, “IEEE 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 System 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 UPSs 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 (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 of SC1 SSCs support conformance to 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 onsite 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 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 batteries 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. The 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 onsite DC 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 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 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 electrical systems accommodates online testing and inspection to the extent practical. Condition monitoring equipment is included where appropriate. For the Emergency Power System, the UPS can be manually bypassed for each division, and testing can be done online without losing the divisional output.

Required electrical system testing is governed by the Technical Specifications 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 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 Raceway System also includes the primary containment electrical penetration assemblies.

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 (Reference 8.3-11). This supports conformance to 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 1202-2023 (Reference 8.3-12) and IEEE 383-2023 (Reference 8.3-13). Fiber optic cables within the Power Block are qualified to IEEE 1202-2023 (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 lines and performed in accordance with the requirements of IEEE 384-2018 (Reference 8.3-10) and IEC 60709 (Reference 8.3-11). Separation for SCN components and systems are derived from the application of IEC 60709 (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 Cable and Raceway 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 lines and performed in accordance with the requirements of IEEE 384-2018 (Reference 8.3-10), IEC 60709 (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 anticipated normal and abnormal conditions.

Installation of the cabling and raceways is per approved procedures and testing requirements to provide assurance that 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 conforms to RG 1.118.

Condition monitoring equipment is included where appropriate to support inspection activities and confirm that equipment performance and health is as desired. Required electrical system testing is governed by the 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 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, 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 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 UPSs 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 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 Specifications 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

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8.3-19 IEEE 379-2014, “IEEE Standard for Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems,” Institute of Electrical and Electronics Engineers, 2014.

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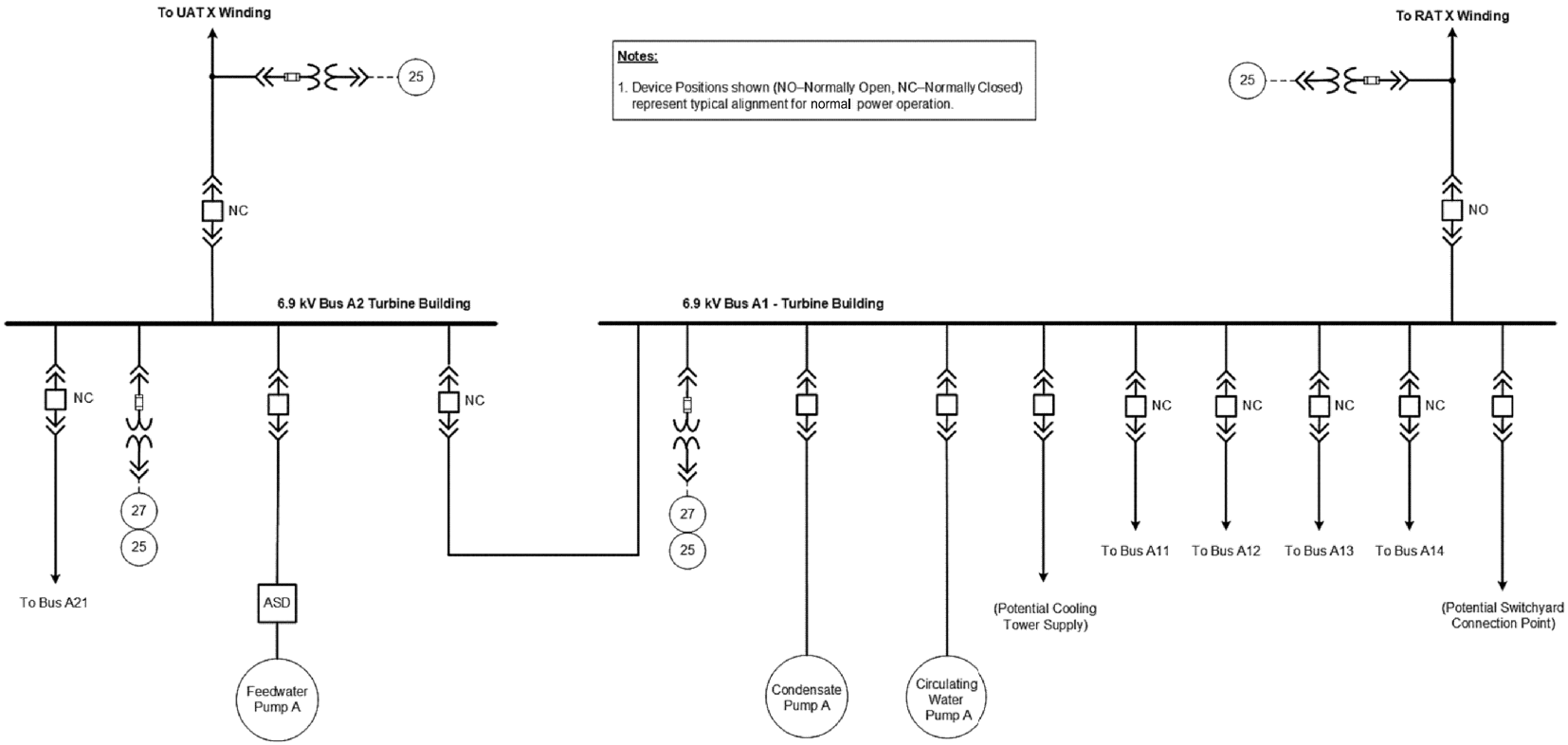

Figure 8.3-1 Preferred Power System Medium Voltage Bus A Train

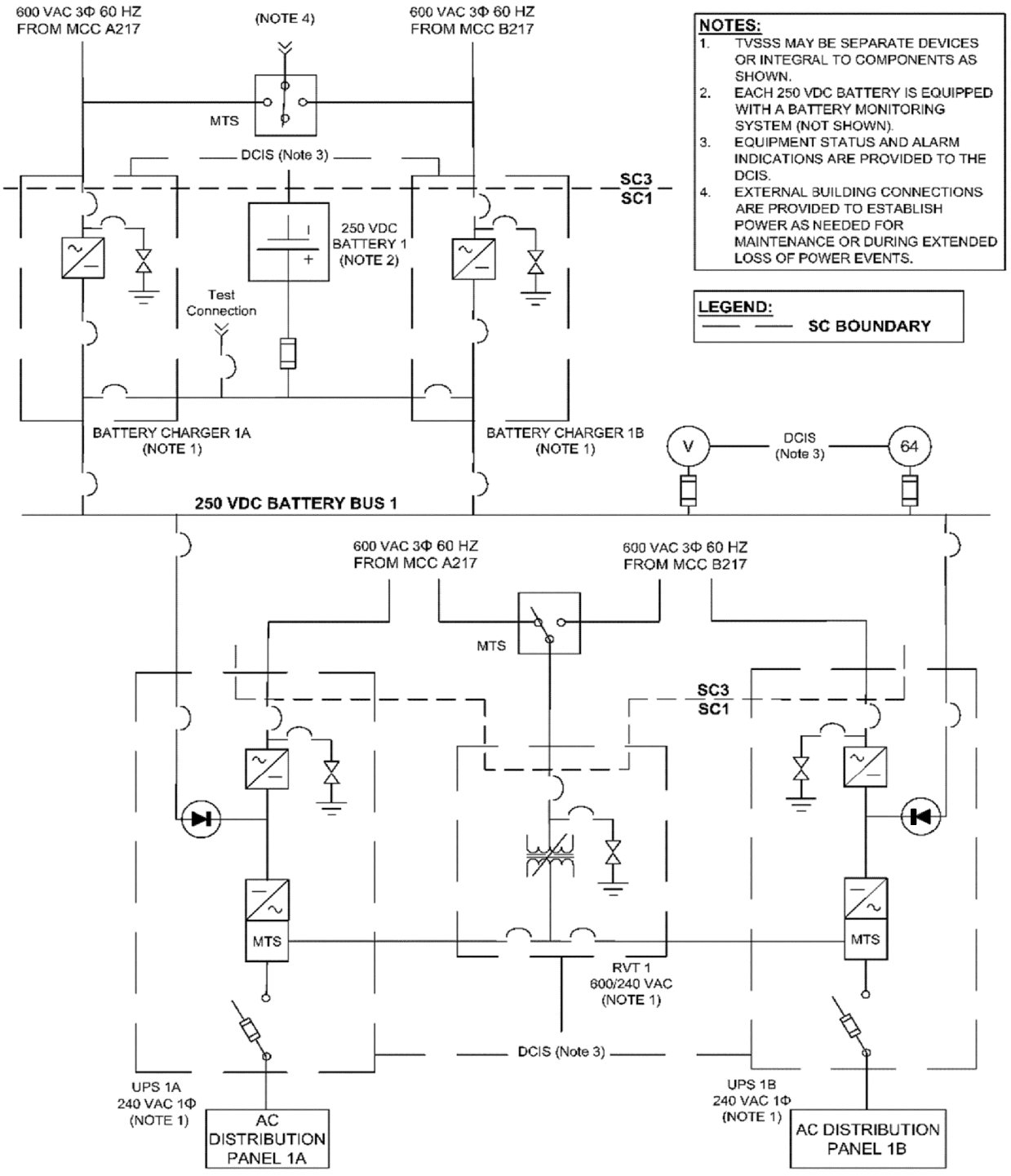

Figure 8.3-2 Emergency Power System

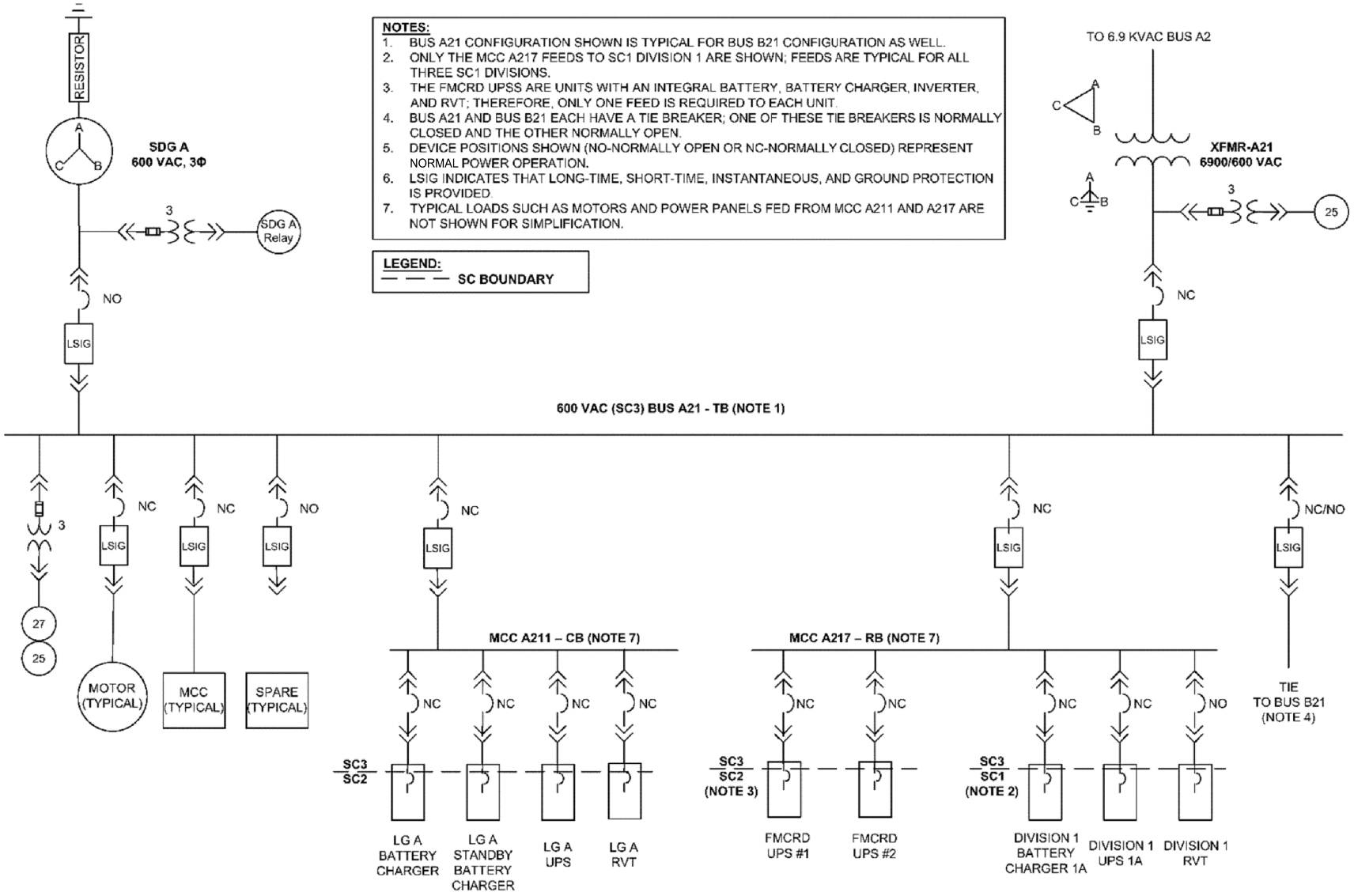

Figure 8.3-3 Standby Power System Low Voltage Distribution Train A

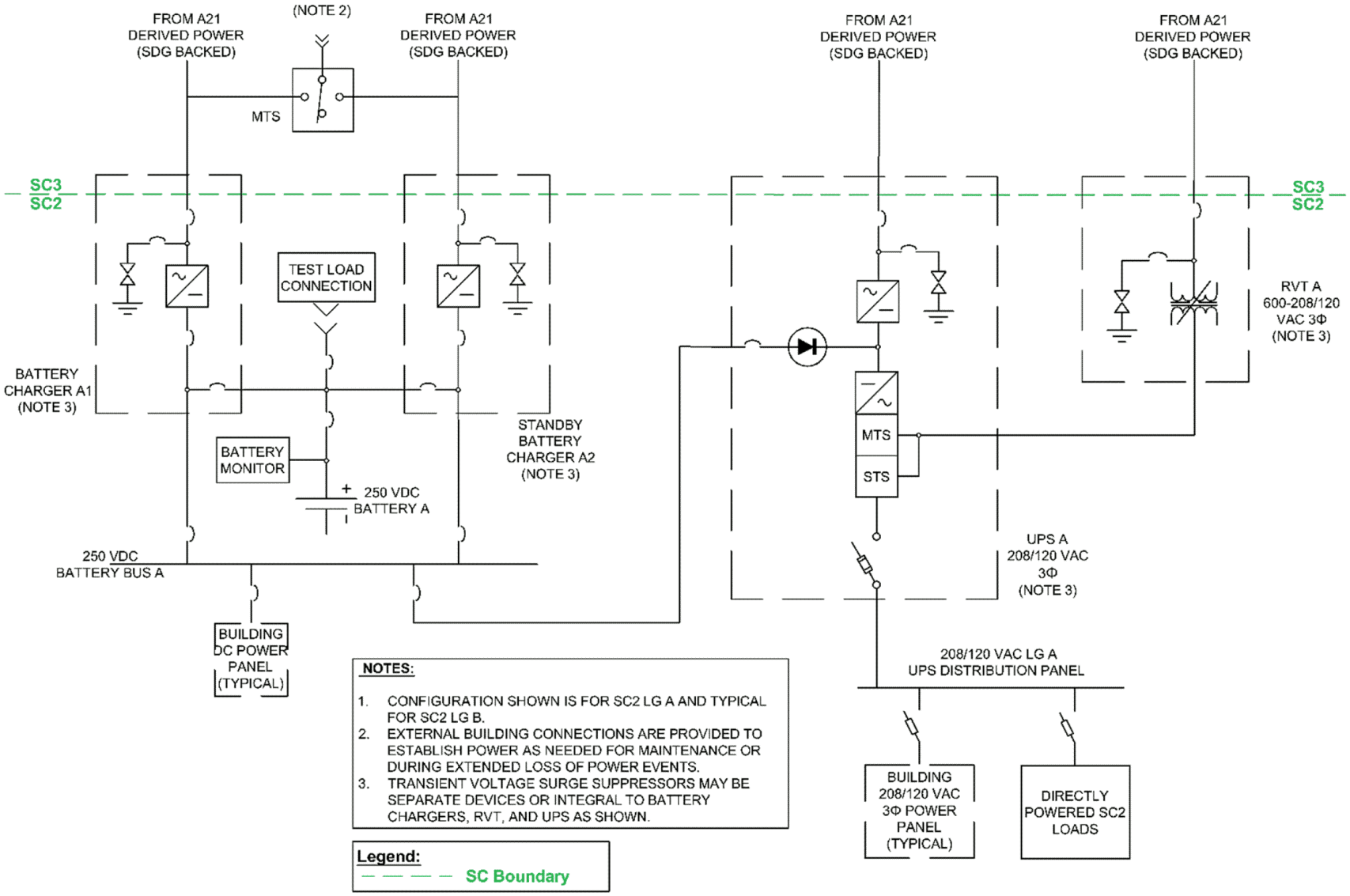

Figure 8.3-4 Standby Power 250 VDC Subsystem

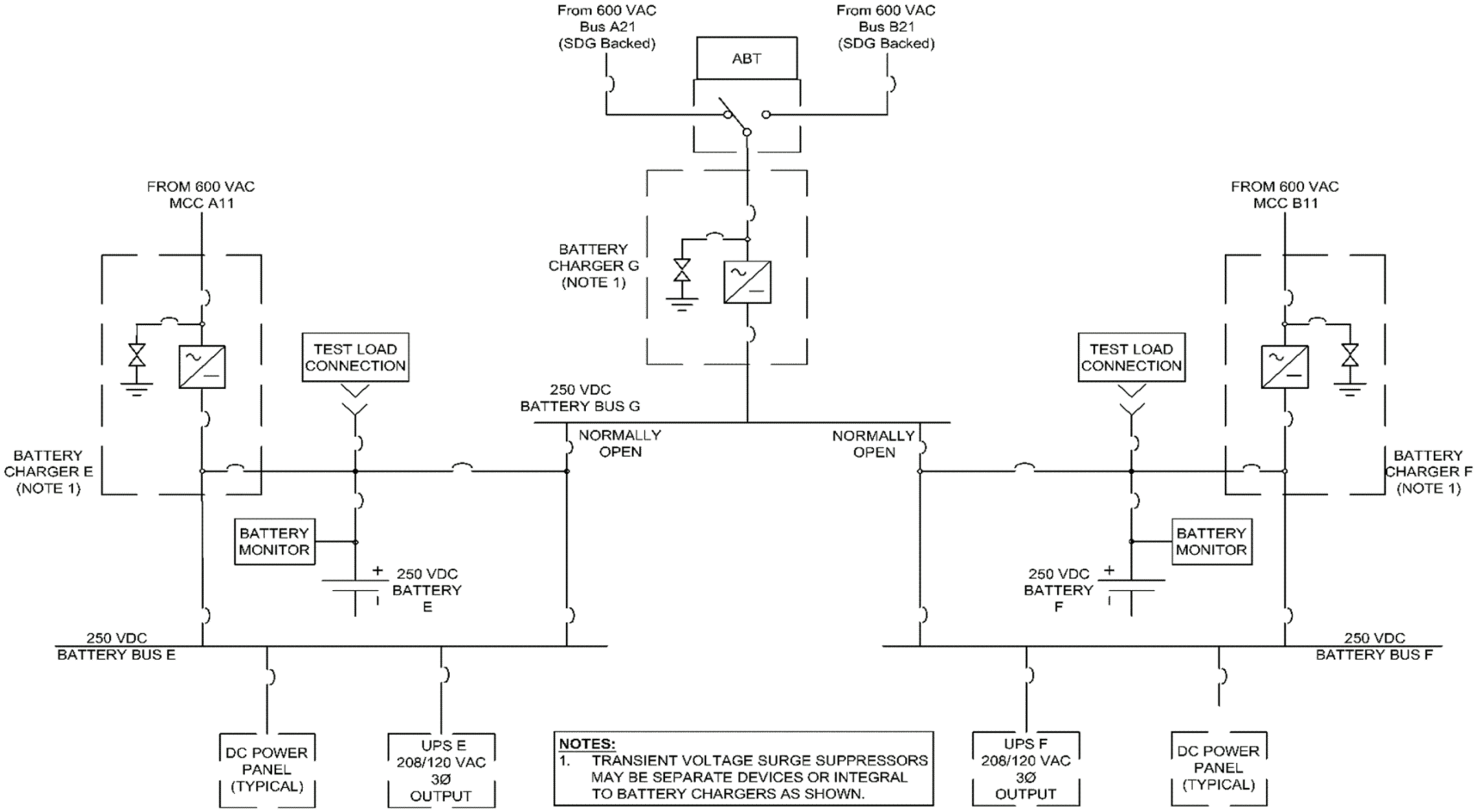

Figure 8.3-5 Preferred Power 250 VDC Subsystem Part One

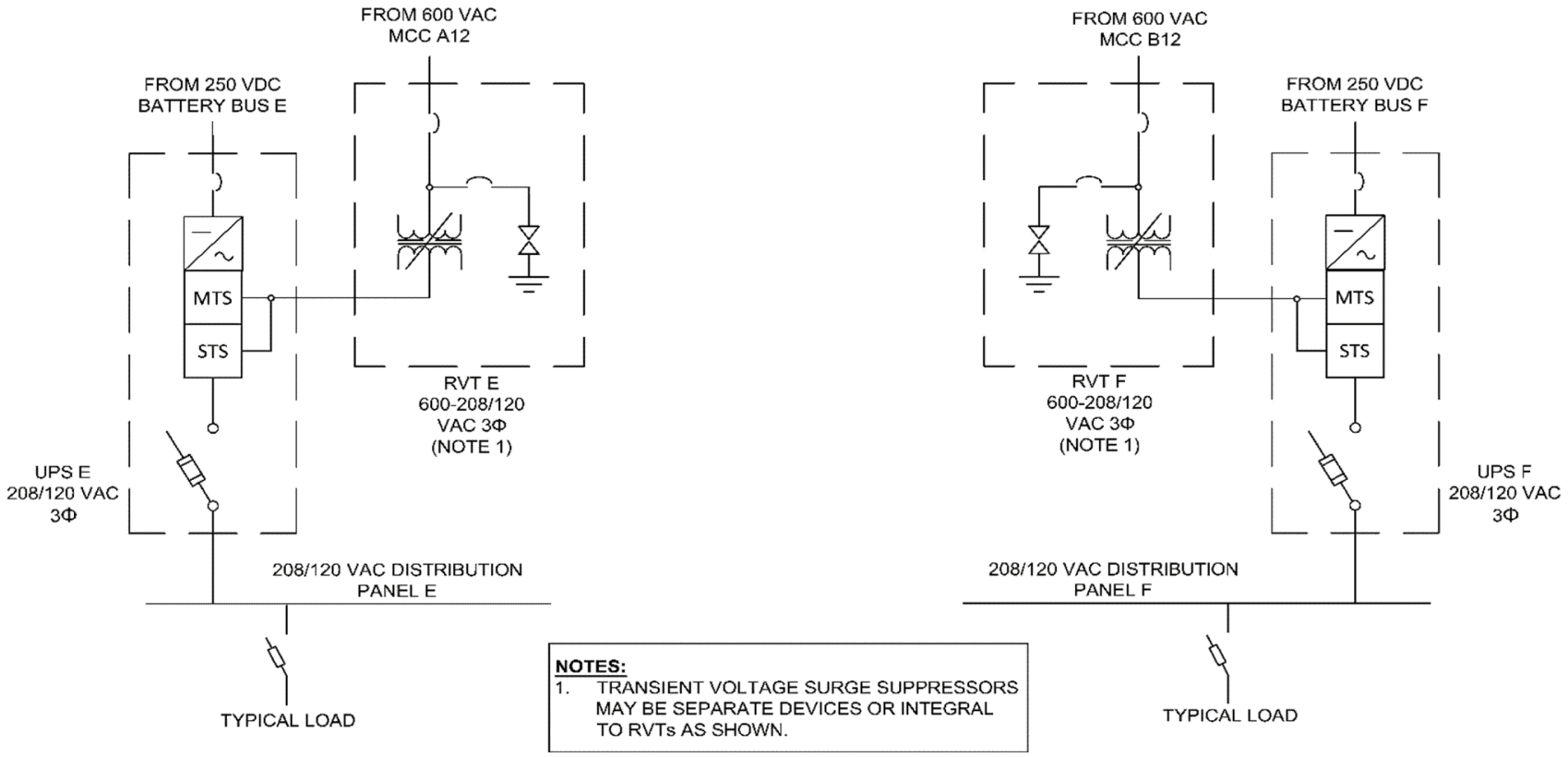

Figure 8.3-6 Preferred Power 250 VDC Subsystem Part Two

8.4 Station Blackout

The BWRX-300 design minimizes the reliance on electrical power to support safety category functions. The passive design of the plant is not dependent upon offsite 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 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 GDC.

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LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
AC	Alternating Current
AHRI	Air-Conditioning, Heating, and Refrigeration Institute
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
API	American Petroleum Institute
ARM	Area Radiation Monitoring Subsystem
ASD	Adjustable Speed Drive
ASHRAE	American Society of Heating, Refrigerating, and Air-Conditioning Engineers
ASME	American Society of Mechanical Engineers
AWWA	American Water Works Association
BIS	Boron Injection System
BPVC	Boiler and Pressure Vessel Code
BWR	Boiling Water Reactor
CB	Control Building
CCS	Containment Cooling System
CFD	Condensate Filters and Demineralizers System
CFR	Code of Federal Regulations
CFS	Condensate and Feedwater Heating System
CHE	Cranes, Hoists, and Elevators
CIS	Containment Inerting System
CIV	Containment Isolation Valve
CRD	Control Rod Drive
CRN-1	Clinch River Nuclear Unit 1
CST	Condensate Storage Tank
CUW	Reactor Water Cleanup System
CWE	Chilled Water Equipment
CWS	Circulating Water System
DBA	Design Basis Accident
DCIS	Distributed Control and Information System
D-in-D	Defense-in-Depth
EFS	Equipment and Floor Drain System
EFU	Emergency Filter Unit

LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
EME	Emergency Mitigating Equipment
EPRI	Electric Power Research Institute
EQ	Environmental Qualification
FCU	Fan Coil Unit
FPC	Fuel Pool Cooling and Cleanup System
FPP	Fire Protection Program
FPS	Fire Protection System
FSAR	Final Safety Analysis Report
FSSA	Fire Safe Shutdown Analysis
GDC	General Design Criteria
GNF	Global Nuclear Fuel
HCW	High Conductivity Waste
HVS	Heating, Ventilation, and Cooling System
HWC	Hydrogen Water Chemistry
I&C	Instrumentation and Control
IA	Instrument Air
IC	Isolation Condenser
ICC	Isolation Condenser Pools Cooling and Cleanup System
ICS	Isolation Condenser System
IGSCC	Intergranular Stress Corrosion Cracking
ISA	International Society of Automation
ISI	Inservice Inspection
IST	Inservice Testing
ITP	Initial Test Program
LOCA	Loss-of-Coolant Accident
LOPP	Loss-of-Preferred Power
LWM	Liquid Waste Management System
MCR	Main Control Room
MCRE	Main Control Room Envelope
MOX	Mixed Uranium-Plutonium Oxide
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NHS	Normal Heat Sink
NS	Non-Seismic
NUREG	Nuclear Regulatory Report
OGS	Offgas System
OHLHS	Overhead Heavy Load Handling System

LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
OLNC	On-Line NobleChem™
PCCS	Passive Containment Cooling System
PCW	Plant Cooling Water System
PDC	Principal Design Criterion
PPS	Plant Pneumatics System
PREMS	Process Radiation and Environmental Monitoring System
PRM	Process Radiation Monitoring Subsystem
PS	Process Sampling Subsystem
PSA	Probabilistic Safety Assessment
PVS	Plant Vent Stack
RB	Reactor Building
RCPB	Reactor Coolant Pressure Boundary
RG	Regulatory Guide
RPV	Reactor Pressure Vessel
RWB	Radwaste Building
RWST	Refueling Water Storage Tank
SA	Service Air
SBO	Station Blackout
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
SRP	Standard Review Plan
SSCs	Structures, Systems, and Components
SSE	Safe Shutdown Earthquake
SW	Service Water
SWM	Solid Waste Management System
TB	Turbine Building
TGFU	Toxic Gas Filtration Unit
UHS	Ultimate Heat Sink
USNRC	U.S. Nuclear Regulatory Commission
WGC	Water, Gas, and Chemical Pads

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CHAPTER 9 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 fuel storage racks located in the fuel pool. The fuel 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 discussed 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 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 Classifications, 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 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 reactor cavity, holds the steam dryer assembly and steam separator assembly 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, the equipment required to be stored in the pool including various refueling outage equipment, and lay-down areas. 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 the effective multiplication factor (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 fuel 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 fuel pool 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 storage 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 cavity pool 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 for cooling. The fuel storage 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 a fuel bail extended above the rack. The fuel storage 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 assembly 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 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, Division 1, “Rules for Construction of Nuclear Facility Components,” (Reference 9.1-1), Subsection NF, “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 spent fuel.

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 racks. A program for monitoring the effectiveness of the neutron poison present in the neutron absorbing panels is to be maintained.

The fuel storage racks 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 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 storage racks is permanently marked with identification traceable to the material certifications. The rack design ensures the fuel assembly bails 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 storage racks, and compliance is addressed as follows:

GDC 61 – Fuel storage and handling and radioactivity control, as it relates to the design for fuel storage and radioactivity control

GDC 63 – Monitoring fuel and waste storage, 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, fuel pool liner and fuel storage racks are designed to comply with the following regulations:

10 CFR 20.1101(b) as to radiation doses kept as low as reasonably achievable.

10 CFR 50.68 as to design to preclude criticality accidents.

The design of the fuel pool liner and fuel storage 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 fuel pool 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 fuel pool 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 fuel 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 racks. The fuel storage racks are top loading, allowing for identification of individual fuel assemblies from the poolside or refueling 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” (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 are maintained at least 23 feet above the top of the active fuel in the fuel 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. 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 storage 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 that 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. In this configuration, the design conforms to position C.6 of RG 1.13. 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 to the surge tank 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 pool, equipment pool, and fuel pool as it overflows weirs located near the top of the reactor cavity and 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 fuel cask evolutions.

During plant outages, there are alternate configurations of the fuel pool. The adjacent reactor cavity pool may be drained with the fuel pool gate installed to allow access to the containment head and RPV head 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 will be evaluated in the Final Safety Analysis Report (FSAR).

Due to the fuel storage racks being 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 5.6.4 of ANSI/ANS-57.2, 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 Monitoring) 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, 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
- Maintaining 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
- Maintaining the water level of the fuel pool for shielding and cooling during accident conditions

Safety Category N Functions

- Maintaining the water quality of the fuel pool, reactor cavity pool, and equipment pool through filtration and demineralization
- Restoring 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. Normal heat load operation 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 normal heat load operation, 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 fuel cask equipment.

Mode B – High Heat Load

High heat load operation 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 normal heat load operation, coolant inventory can be added or overboarded 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. 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 breaches 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 fuel 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 particulate 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 and high heat load modes. However, the demineralizer can be bypassed if the FPC temperature exceeds the operational limits of the demineralizer.

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:

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 Refueling Platform area, a radiation monitor alarm initiates closure of the RB isolation dampers and secures the RB upper-level supply Air Handling Units (AHU).

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 is provided to alarm both locally and in a continuously manned location if the water level in the fuel 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 pool, 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 fuel 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, reactor cavity pool, and fuel pool.

The Safety Classifications, Quality Groups, and Seismic Categories of the equipment associated with the Light Load Handling System are listed 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 spent 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 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 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 assembly 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 Figure 9.1-5 through Figure 9.1-9. These illustrated activities include:

- Containment Closure Head Removal
- Reactor Pressure Vessel Head Removal
- Steam Dryer Assembly Removal
- Steam Separator Assembly Removal
- Fuel Handling

Overhead heavy load lifts of the containment closure head, RPV head, steam dryer assembly, and steam separator assembly 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 steam 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. Fuel is moved 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 fuel 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 fuel grapple that extends from normal up to its full extension to reach the top guide (Subsection 4.1.2.2). The fuel pool water level requirements and component clearance over the travel path establishes the shielded water depth for spent fuel transfers.

Fuel storage racks are provided in the fuel pool for the temporary and long-term storage of new and spent fuel 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 fuel storage 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 reactor core to fuel pool, to adequately shield the upper elevations of the 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 prep machines adjacent to the fuel pool. The boom supports the channel handling tool and load balancer over the fuel prep 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 spans the length of the bridge, where refueling, servicing, and inspection personnel perform their work tasks. 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 hoist, 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 fuel grapple are suspended from a trolley system and are used to lift and orient fuel assemblies for placement in the core or fuel 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 assembly 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 lifted fuel assembly during transit; and (3) prevent lifting of fuel assemblies without fuel grapple 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, cranes and hoists provide the means to lift and lower equipment and materials, including heavy loads (Subsection 9.1.5), and move them horizontally along safe load paths. Administrative controls (e.g., lift procedures) are used to determine the load paths and equipment used for anticipated 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 spent fuel assembly 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.

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 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 fuel grapple are 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, such that the fuel assembly 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 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 and compliance with GDC 1, 2, 4 and 5.

Lifting and handling operations using overhead handling systems are performed consistent with ASME NML-1, “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 NML-1 use a risk-informed approach by 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 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 NML-1 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 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 NOG-1, “Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder),” (Reference 9.1-6) as endorsed by RG 1.244, and lifting devices, which may include special lifting devices designed to ASME 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 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 spent fuel in the reactor vessel or fuel pool, and safe shutdown equipment.

A heavy load handling program is developed consistent with ASME NML-1. 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 NOG-1 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. A RB circular rail supports the two girders allowing the polar 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 the polar crane consist of the components associated with reactor vessel refueling, and spent fuel cask 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 polar crane has access to a truck bay for fuel transfer to spent fuel 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 OHLHS 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, or otherwise complies with the requirements of 10 CFR Part 50 Appendix B. As a Type I crane, the RB polar crane complies with 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 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, Division 1, “Rules for Construction of Nuclear Facility Components,” Subsection NF, “Supports,” 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 Americas, LLC, 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, 1992 (R2024).

9.1-5 ASME NML-1, “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, “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, “Design of Below-The-Hook Lifting Devices,” American Society of Mechanical Engineers, 2017.

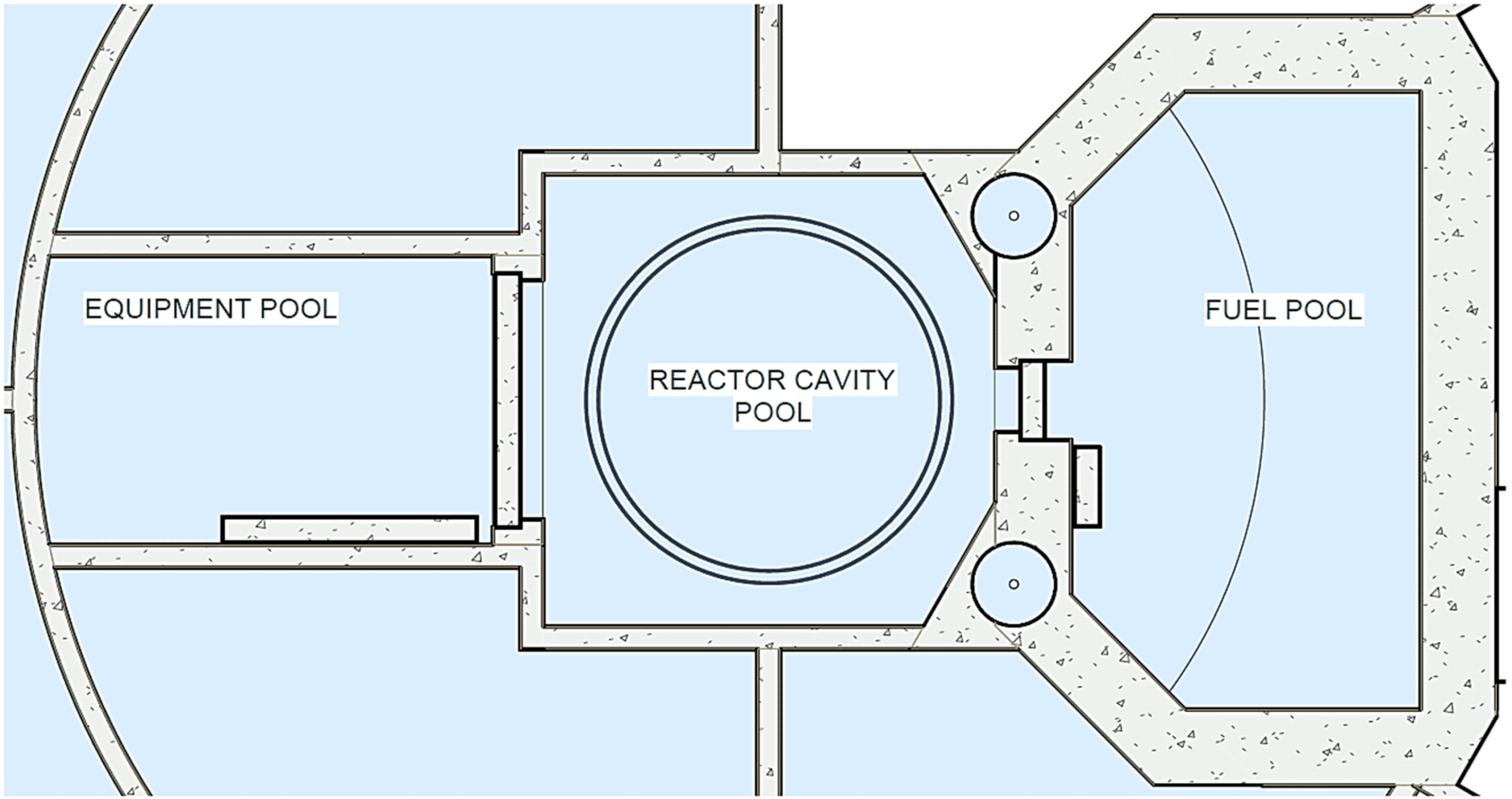

Figure 9.1-1 Arrangement of Fuel, Reactor Cavity and Equipment Pools

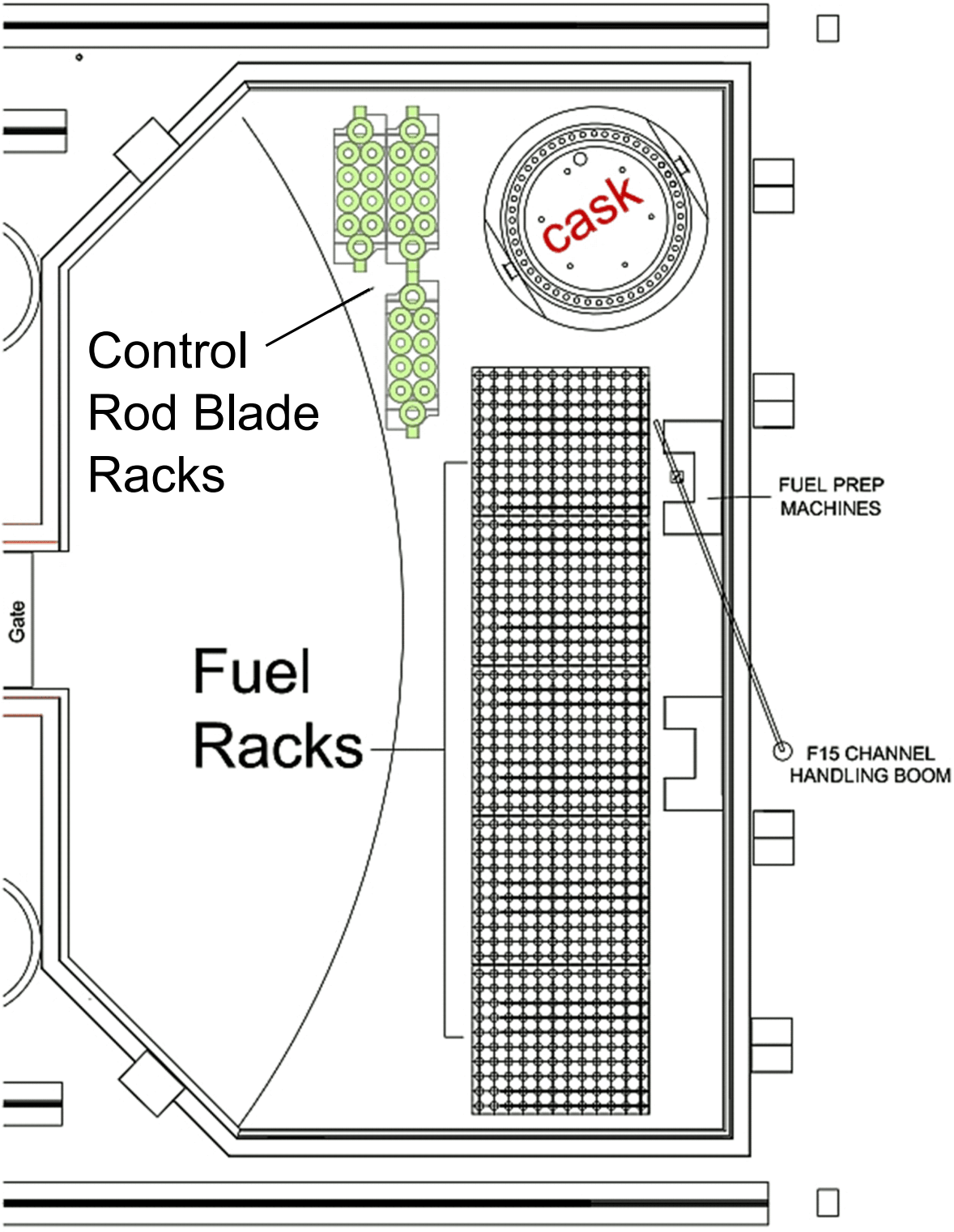

Figure 9.1-2 Fuel Pool Arrangement

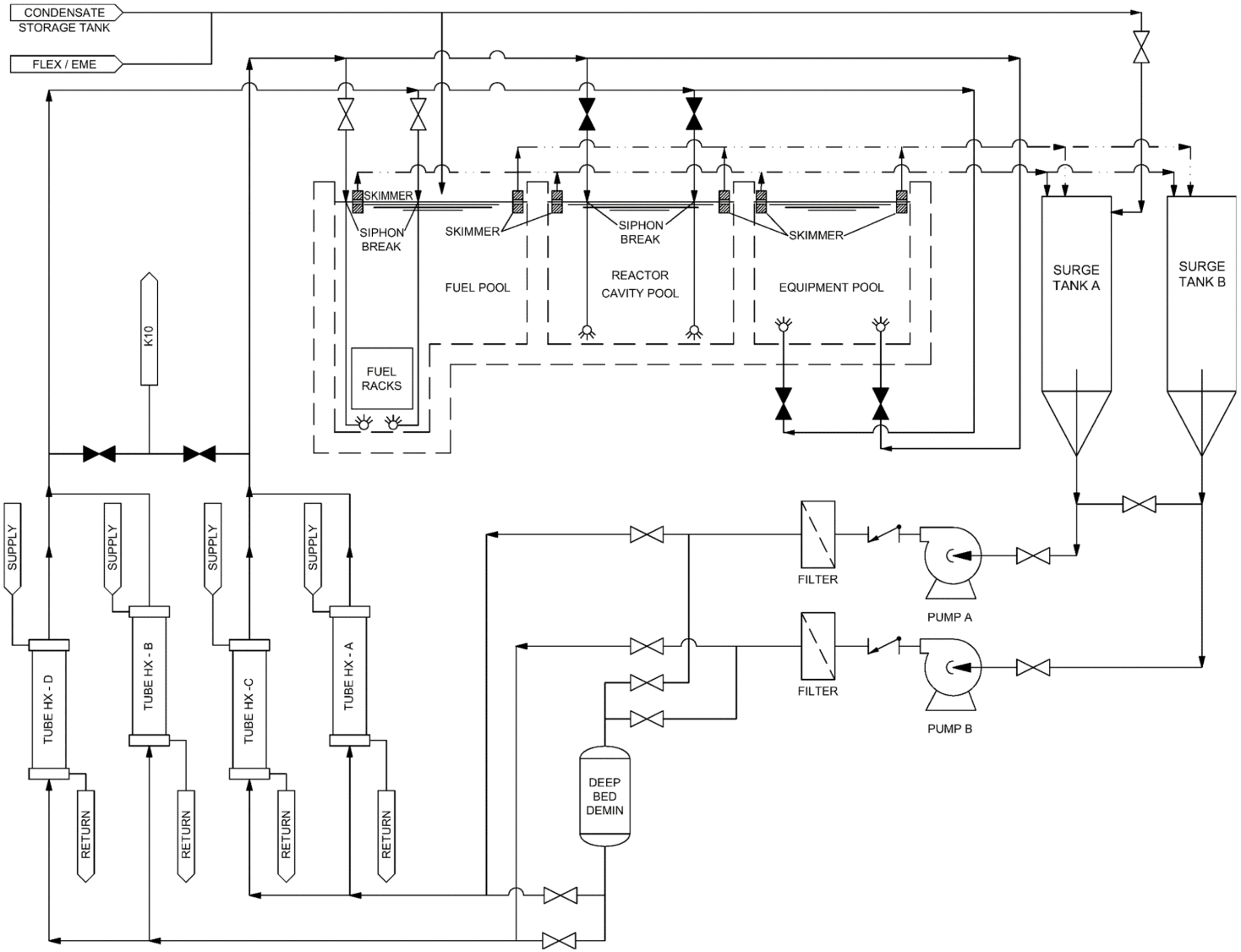

Figure 9.1-3 Fuel Pool Cooling System Simplified Diagram

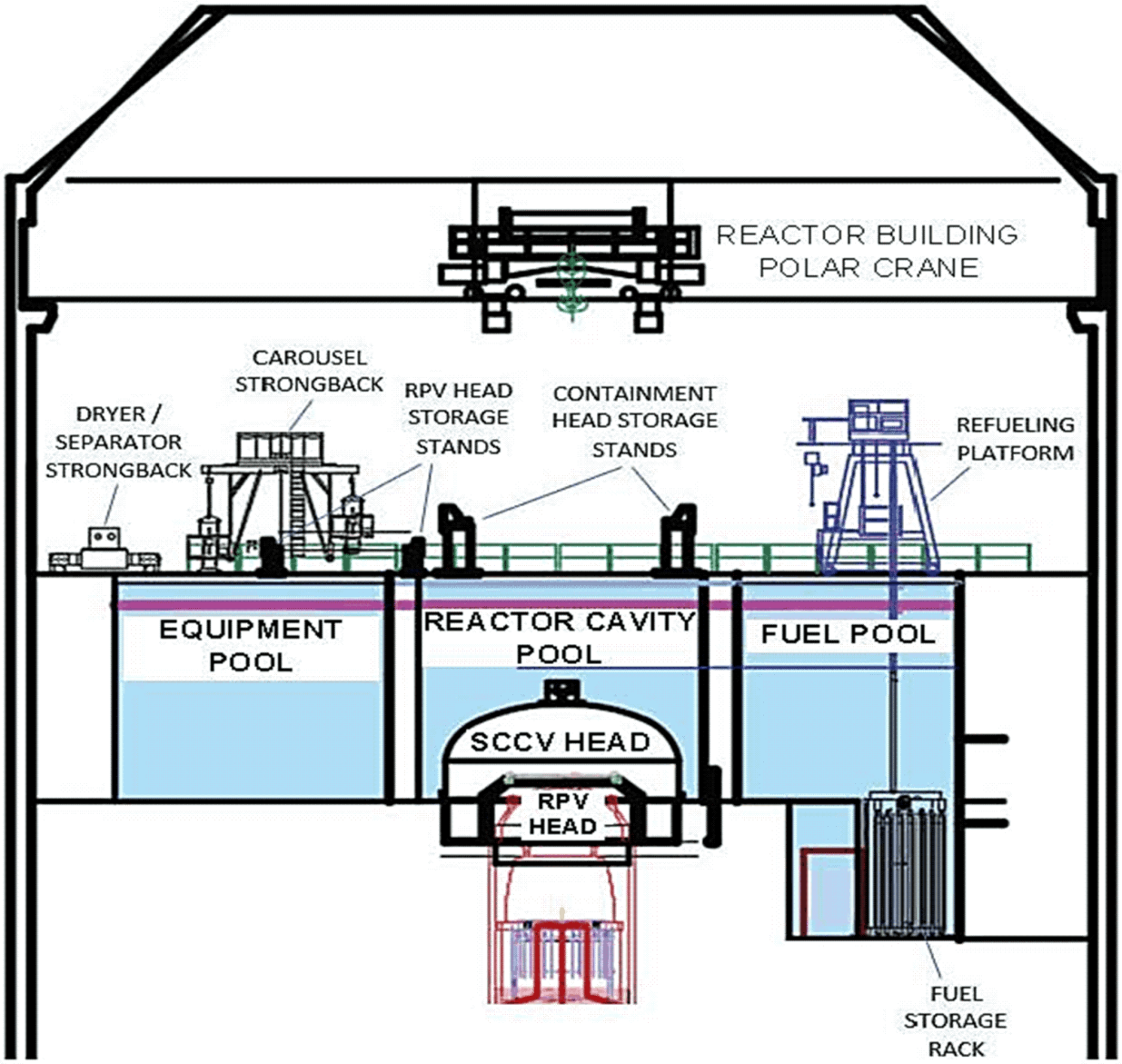

Figure 9.1-4 Location of Equipment on Refuel Floor

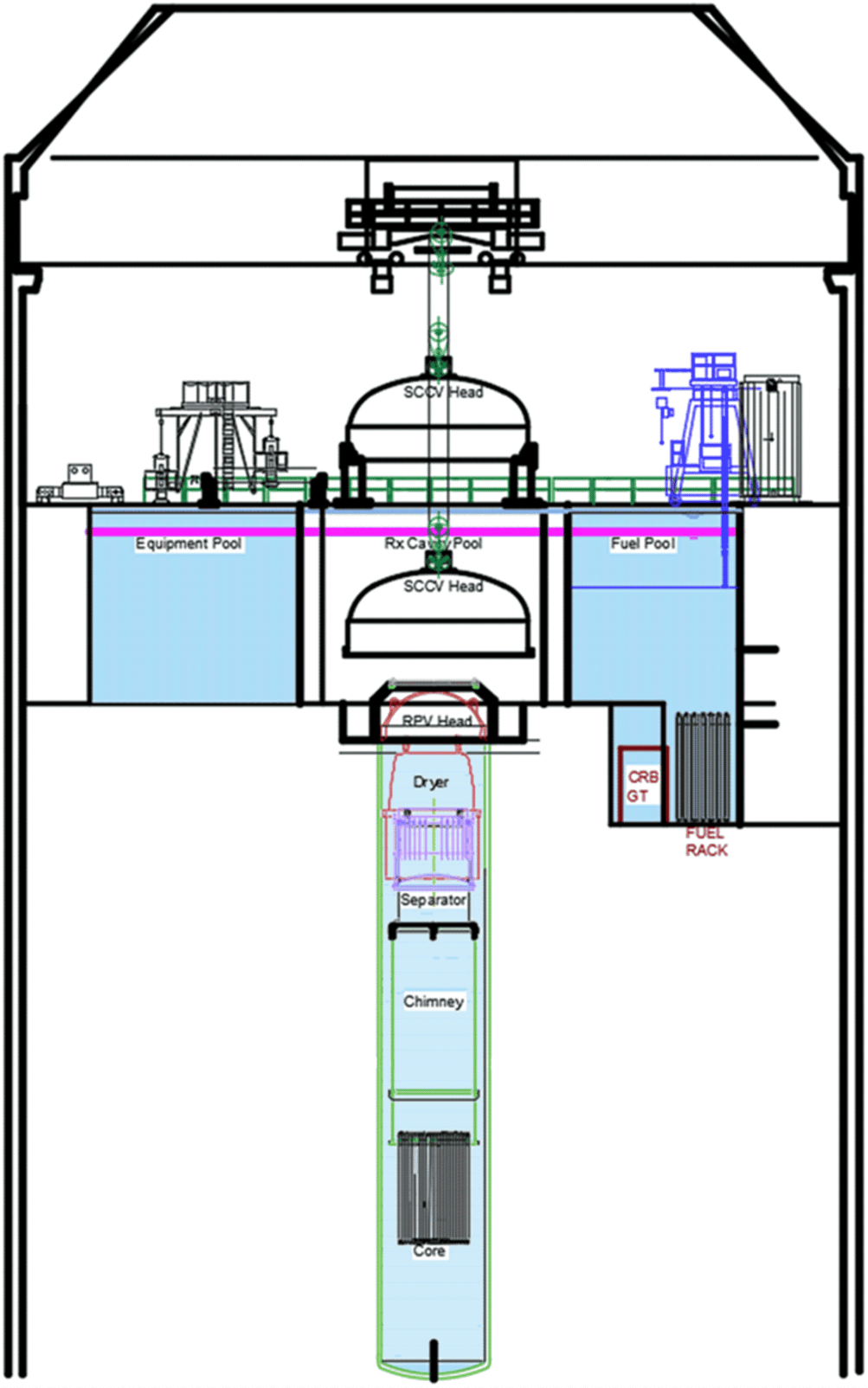

Figure 9.1-5 Containment Closure Head Removal

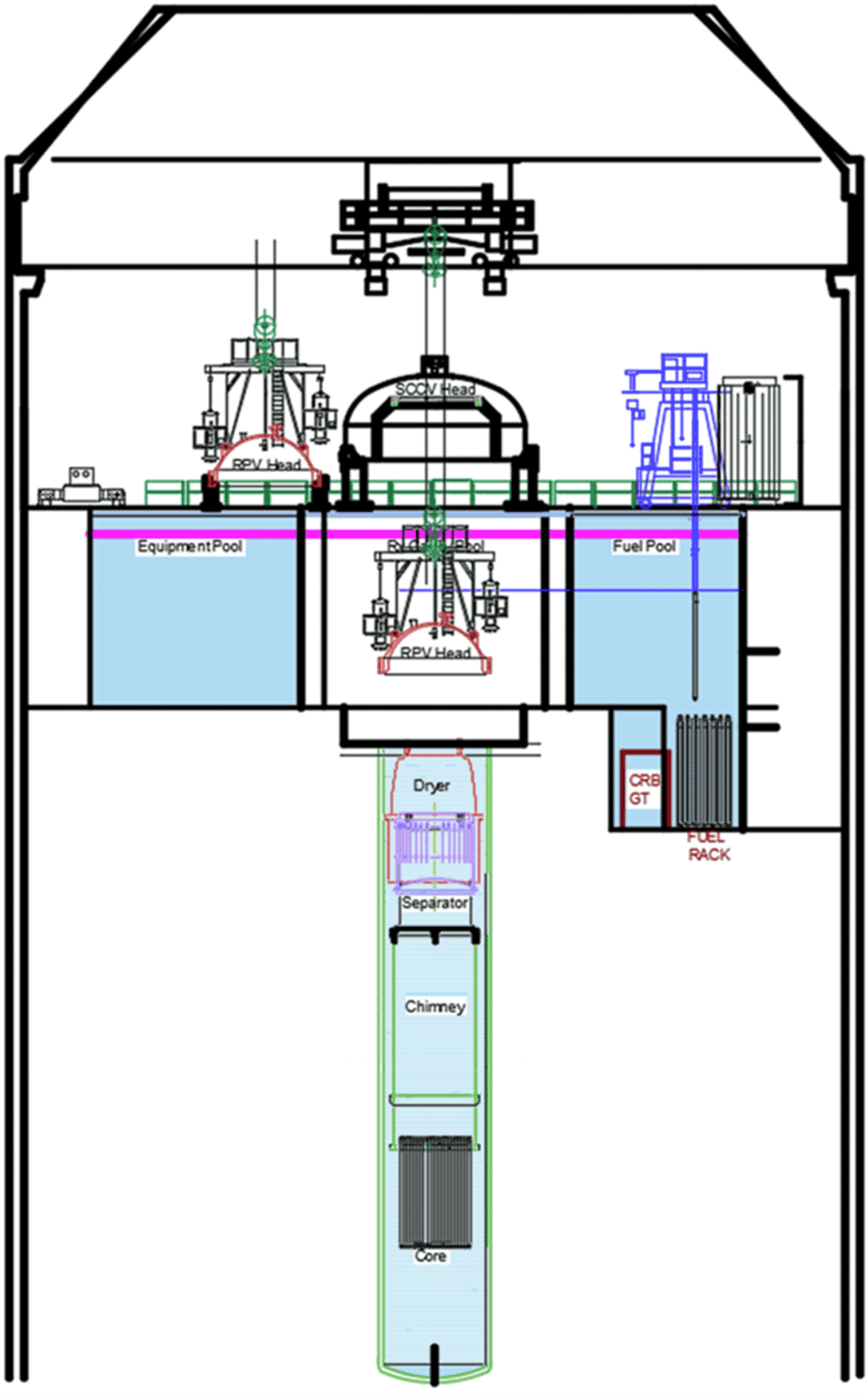

Figure 9.1-6 Reactor Pressure Vessel Head Removal

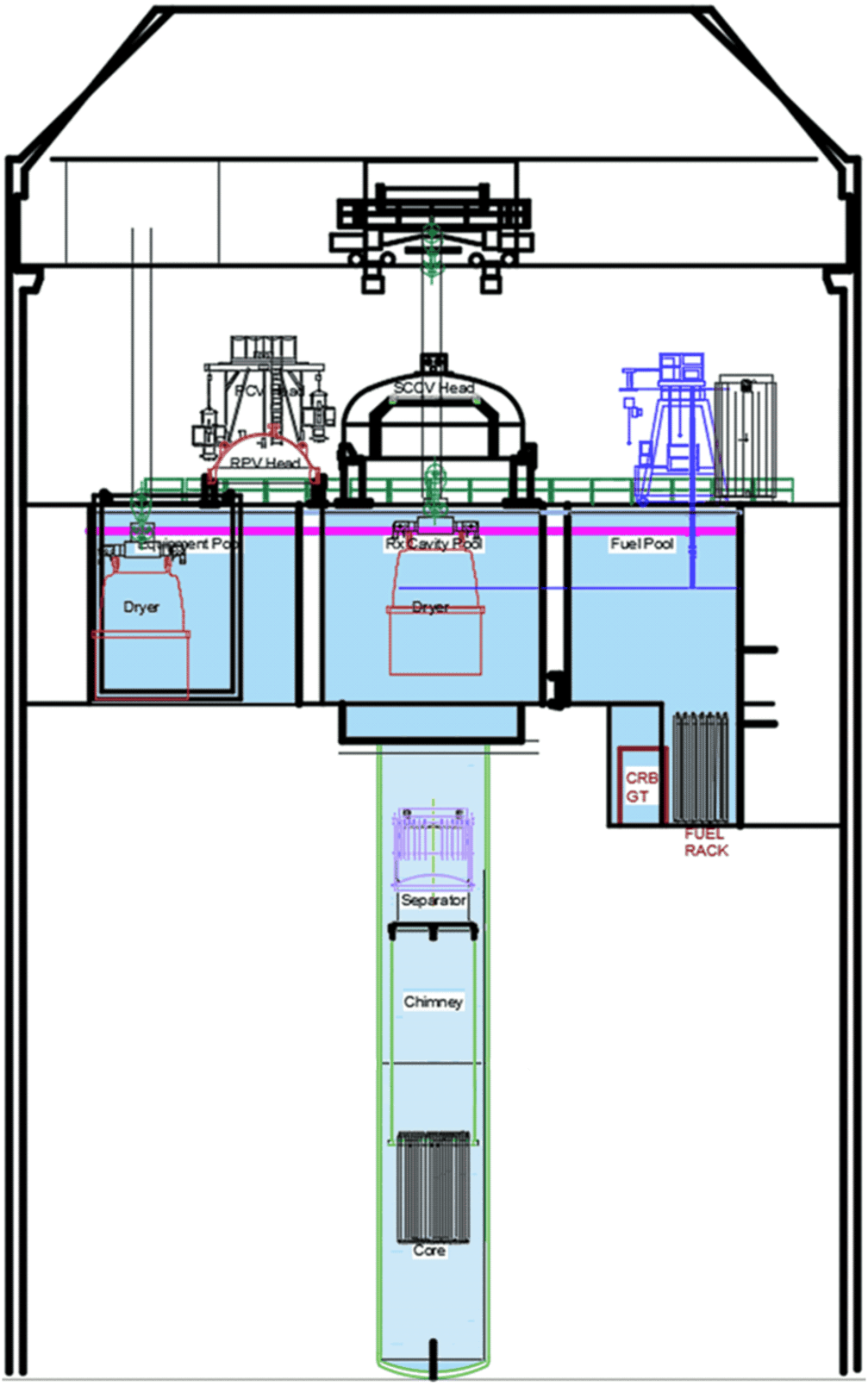

Figure 9.1-7 Steam Dryer Assembly Removal

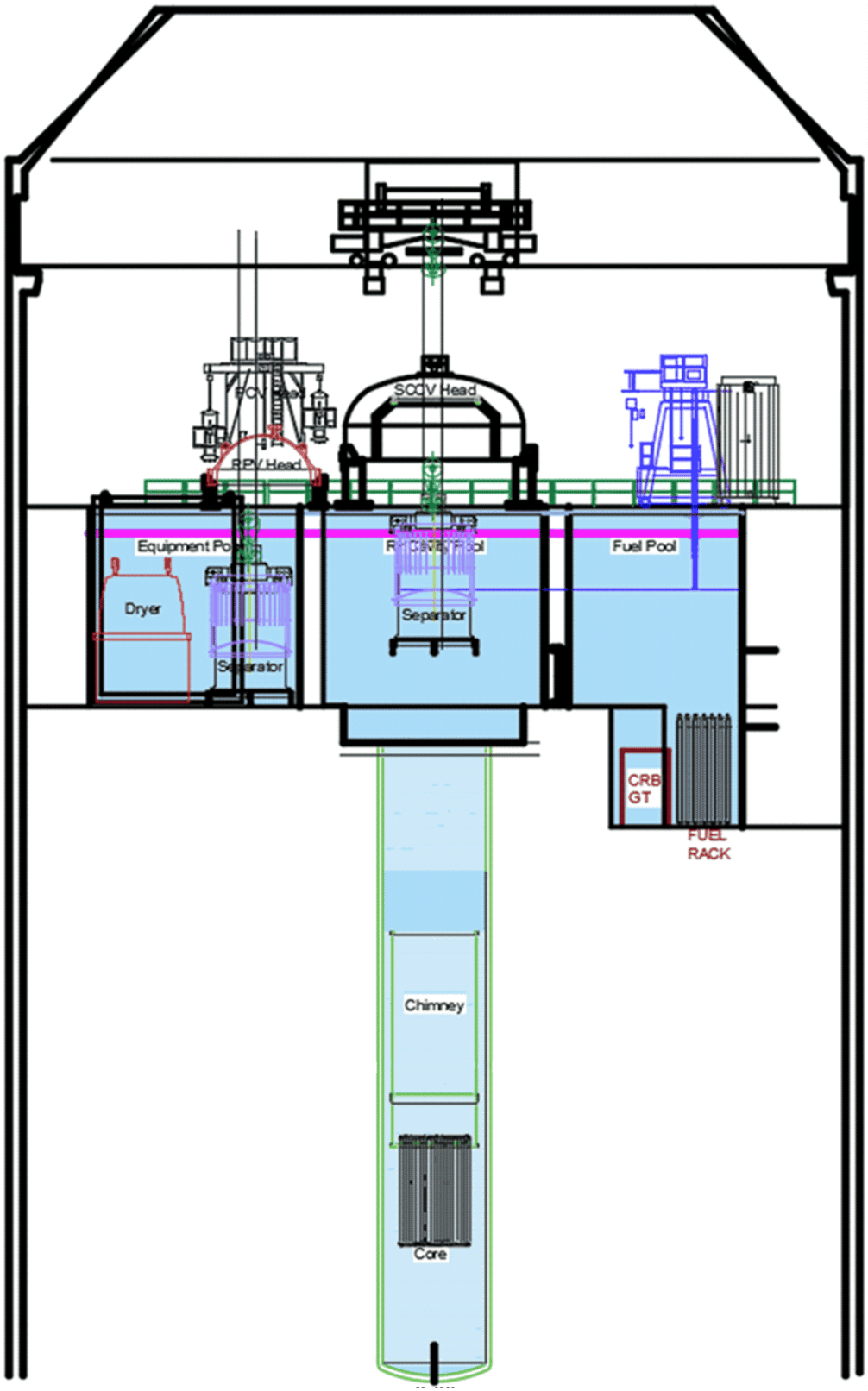

Figure 9.1-8 Chimney Head and Steam Separator Assembly Removal

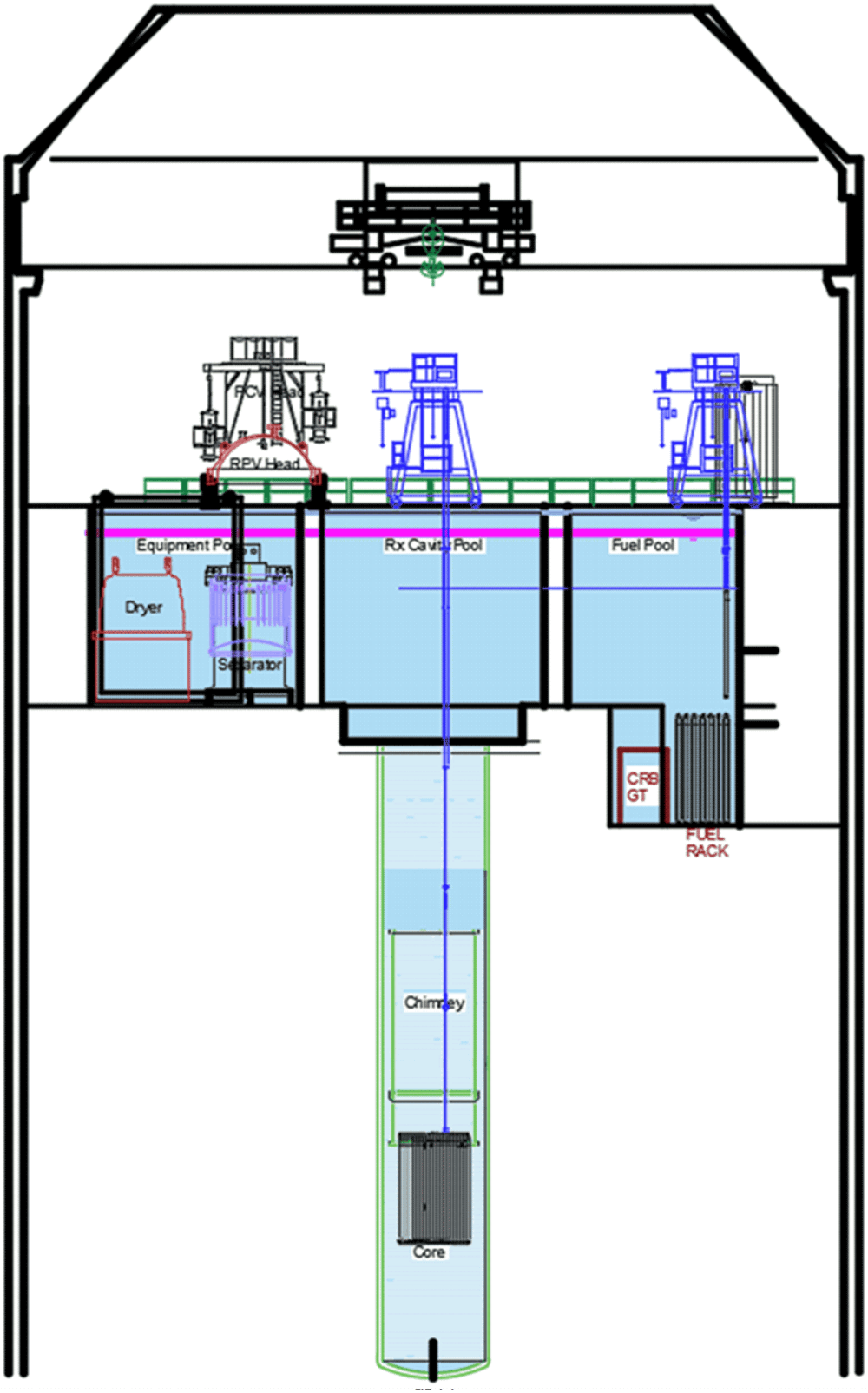

Figure 9.1-9 Fuel Handling

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, “Design Requirements for Light Water Reactor Fuel Handling Systems,” (Reference 9.1.A-1)
- ANSI/ANS-57.2, “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 9.1.2.3 describes the protection of the fuel storage racks from natural phenomena. Non-seismic SSCs in the vicinity of the 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 and the fuel pool. 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 light load handling equipment, and Subsection 9.1.5 describes heavy load handling equipment.

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 and ANSI/ANS-57.2, 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 to handle fuel assemblies 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 procedures.

The fuel storage racks comply with the applicable guidance, including scope of the nuclear criticality safety analysis of ANSI/ANS-57.2. 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 storage rack vendor, the fuel storage rack design is evaluated to verify that the parameters used remain bounding. Relevant information includes materials, dimensional data, spacing between fuel storage racks, material/dimensional details of structures near the racks, which may provide neutron reflection (such as pool walls and floors). The fuel storage 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 storage racks are analyzed using the Monte Carlo neutron transport program (MCNP) and ENDF/B-VII.0 nuclear cross-section library. The methodology used in this analysis is the peak standard cold core geometry incore eigenvalue ($k\infty$) criterion. MCNP 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 storage 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 MCNP.

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 fuel 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 National Standards Institute/American Nuclear Society, 1992 (R2024).

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, Revision 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.

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

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.
- Following a Loss-of-Preferred Power (LOPP), once power is restored from the Standby Diesel Generators (SDG), the SW Subsystem provides a cooling water source and heat rejection functions to support PCW rejecting heat to the NHS System.
- The NHS System has provisions to tie-in with the SW Subsystem supply and return piping to provide once-through cooling flow to a PCW heat exchanger when the NHS System cooling tower is not available.

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 Classifications, Quality Groups, and Seismic Categories of the SW Subsystem SSCs are listed 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 Subsystem 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 Subsystem pump suction lines. During normal operation, only one PCW heat exchanger is in service, and the corresponding SW Subsystem pump is used to provide cooling water flow. During a plant shutdown, both PCW heat exchangers and SW Subsystem pumps are typically in service to facilitate a plant shutdown. In the event of an LOPP, the SW Subsystem pumps can be powered by the SDGs to provide continuous cooling

support to the PCW. The SW Subsystem pumps provide sufficient cooling water to meet the performance requirements of the PCW heat exchangers.

Two individual SW Subsystem pump discharge lines each contain pressure instrumentation, air release valves, and an isolation valve. Each line supplies service water independently to its related PCW heat exchanger. SW Subsystem piping is designed to ASME B31.1, “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 blowdown piping 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 discussed 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 Subsystem pumps and associated valves. Radiation monitors on SW Subsystem piping downstream of each PCW heat exchanger allow for the detection of high radiation levels.

SW Subsystem pumps and remote valves are operated and monitored from the Main Control Room (MCR). If a SW Subsystem 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

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, Quality Groups, and Seismic Categories of the PCW SSCs are listed 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 PCW 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 an 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

- FPC Heat Exchangers
- 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 Subsystem that interfaces with PCW through the PCW heat exchangers. The PCW is backed up by SDGs to support the system during an 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.

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 discussed 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 functions 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.

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 other GDCs. Additional information on GDC compliance is discussed 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 Subsystem provides potable water for areas of the facility outside of radiologically controlled areas. The Sanitary Sewer Handling Subsystem collects sanitary water from outside radiologically controlled areas and transfers it offsite.

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

- Provides potable water for various SCN systems including the Fire Protection System (FPS) storage tanks, Power Block areas outside of radiologically controlled areas, and other users outside of the Power Block
- Collects sanitary water from areas outside the radiologically controlled areas in the Power Block

The Safety Classifications, Quality Groups, and Seismic Categories of the Potable Water Distribution and Sanitary Sewer Handling Subsystems SSCs are listed in Table 3A-1.

9.2.4.2 System Description

The Potable Water Distribution Subsystem provides potable water to the facility.

The Potable Water Distribution Subsystem consists of supply line piping, valves, and instrumentation. The potable water supply line from the municipality is sized for 400 percent capacity. Heat tracing is provided for outside above ground piping and instrumentation. The subsystem is supplied potable water from the municipal potable water line and provides potable water to the following areas:

- FPS Fire Water Storage Tanks
- Power Block areas outside of radiologically controlled areas
- Other users outside of the Power Block (e.g., Administration Building)

Potable water is supplied to the restroom facility within the MCR envelope.

The Sanitary Sewer Handling Subsystem consists of piping and valves to collect sanitary water from the facility washrooms and break areas and transfers it off site to the tie-point with the municipality.

9.2.4.3 Safety Evaluation

These subsystems serve no Safety Category functions (classified as SCN) and are not credited for mitigation of a DBA and have no safe shutdown functions.

The Potable Water Distribution and Sanitary Sewer Handling Subsystems 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 GDC has aspects specific to the Potable Water Distribution and Sanitary Sewer Handling Subsystems and compliance is addressed as follows:

GDC 2 – Design Bases for Protection Against Natural Phenomena. Meeting GDC 2 ensures that flooding due to failures of non-seismic piping does not affect the ability of the plant to shut down safely and remain in safe shutdown condition. Conformance regarding the Potable Water Distribution and Sanitary Sewer Handling Subsystems piping inside the MCR envelope will be confirmed in the FSAR during the internal flooding analysis discussed in Section 3.4.1.

GDC 60 – Control of releases of radioactive materials to the environment. The Potable Water Distribution and Sanitary Sewer Handling Subsystems have no interconnection with radioactive water systems. This separation precludes the possibility of inadvertent contamination and meets the requirements of GDC 60.

Additional information on GDC compliance is discussed in Section 3.1.

9.2.4.4 Inspection and Testing

Inspection and testing are performed in accordance with applicable codes and industry standards.

9.2.4.5 Instrumentation and Controls

The Potable Water Distribution Subsystem provides monitoring and control for the capability to ensure that there is an adequate source of potable water available 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 the 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 (LOCA) 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 System 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 vented to atmosphere. The vent can be isolated in response to a high radiation alarm, or due to an 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

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 discussed 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.

Safety Category N 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 (Subsection 10.4.1)
- Alternate supply to the Control Rod Drive (CRD) pumps (Section 4.6)
- Makeup supply and backwash/flushing water for the FPC (Subsection 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) (Subsection 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 discussed 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) AHUs (Subsection 9.4.7).
- During an 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 Classifications, Seismic Categories and Quality Groups of the CWE SSCs are listed in Table 3A-1.

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 (FCU) 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 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 exchangers 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 discussed 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 pool makeup 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 discussed in Section 3.1.

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 Suction 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 System

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).
- Following an LOPP event, the NHS System provides a cooling tower with two cooling cells that can be powered from the SDGs.
- The NHS System provides once-through cooling water flow to a PCW heat exchanger during off-normal operations when the cooling tower is unavailable, utilizing makeup water and CW Supply piping to discharge to the holding pond.

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 and the SW Subsystem. The NHS System includes the cooling tower, intake structure, makeup water pumps, holding pond (also known as cooling tower blowdown pond), diffuser, and other associated equipment. The NHS System supports heat rejection for the CW Supply and SW Subsystem during normal operation and shutdown. The NHS System also provides the water source for the Demineralized Water Storage and Distribution Subsystem. A simplified diagram of the NHS System is provided in Figure 9.2-7.

The NHS System consists of four makeup water pumps that provide makeup water to the cooling tower basin. The flow rate to the basin is regulated with a level control valve, with the excess flow going to the holding pond or the weir box. The makeup water is pumped through filters as

needed. Backwash lines are provided for filters. The makeup water lines also have a connection for emergency filling of the fire water storage tanks. A discharge line is provided for the demineralized water trailer to the holding pond.

Cooling tower basin blowdown water is discharged to the holding pond via the CW Supply pumps or SW Subsystem pumps. The NHS System design supports heat removal at full power operation and normal decay heat removal during plant shutdown. The NHS System includes mechanical draft cooling towers that remove heat from the CW Supply and SW Subsystem and then returns the water to the cooling tower basin. Two of the cooling tower cells can be powered from the SDGs.

During shutdown, a once-through cooling configuration can be selected to cool the PCW heat exchangers. A connection with appropriate valving to the CW Supply is provided to receive water from the makeup water pumps. This water cools the PCW heat exchangers and returns the water to the NHS System via a connection on the cooling tower blowdown line. This water then flows into the blowdown holding pond to be discharged into the Clinch River. The system is designed to operate during normal modes of plant operation. NHS System equipment is designed for the plant normal operating environmental conditions specified for its location.

9.2.9.3 Safety Evaluation

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 System. No SC1 SSCs require cooling by this system to mitigate DBAs. The design and layout of the NHS System 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 System, or cooled by the system, is required to mitigate DBAs and, therefore, the NHS System 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 discussed in Section 3.1.

9.2.9.4 Testing and Inspection Requirements

Equipment and components of the NHS System are designed to facilitate inspection and maintenance during plant operation. Instrumentation is provided to support monitoring thermal performance of the cooling tower and also to support performance testing.

9.2.9.5 Instrumentation Requirements

The following defines the basic I&Cs required for system operation and control:

- Level is monitored in the NHS System intake structure, cooling tower basin and blowdown holding pond.
- Local temperature transmitters monitor the water temperature in the intake structure, cooling tower basin and discharge.
- Pressure transmitters measure makeup water pump discharge pressure.
- Pressure differential is monitored across filters and traveling screens.
- Flow transmitter measures flow in the blowdown line and diffuser line.

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. The RWST function 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 discussed 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, “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,” American Petroleum Institute, Thirteenth Edition, 2021.

9.2-6 ASME B31.5, “Refrigeration Piping and Heat Transfer Components,” American Society of Mechanical Engineers, 2019.

9.2-7 ANSI/AHRI Standard 550/590 (I-P), “Performance Rating of Water-chilling and Heat Pump Water-heating Packages Using the Vapor Compression Cycle,” American National Standards Institute/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 1, “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.

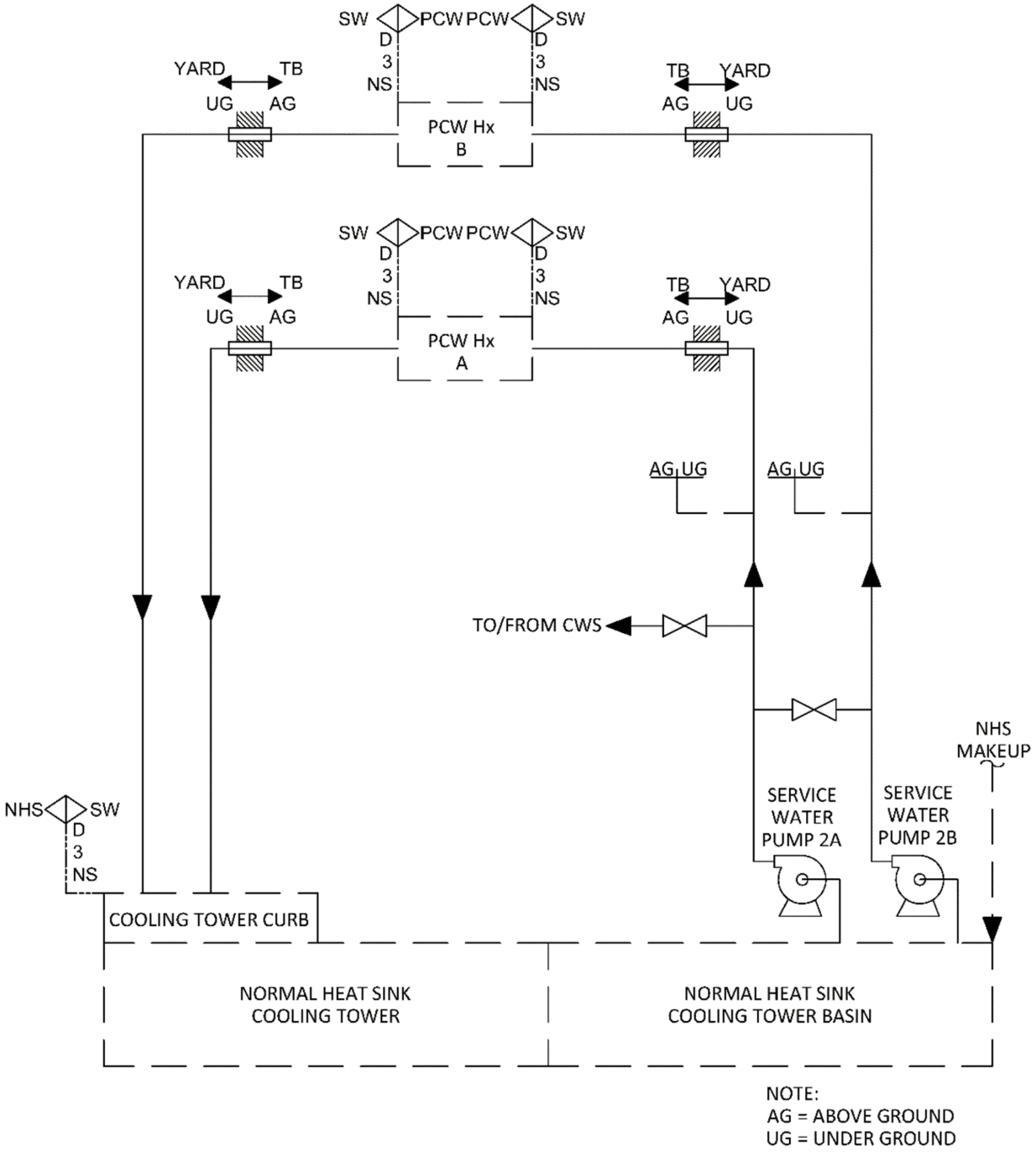

Figure 9.2-1 Service Water Subsystem Simplified Diagram

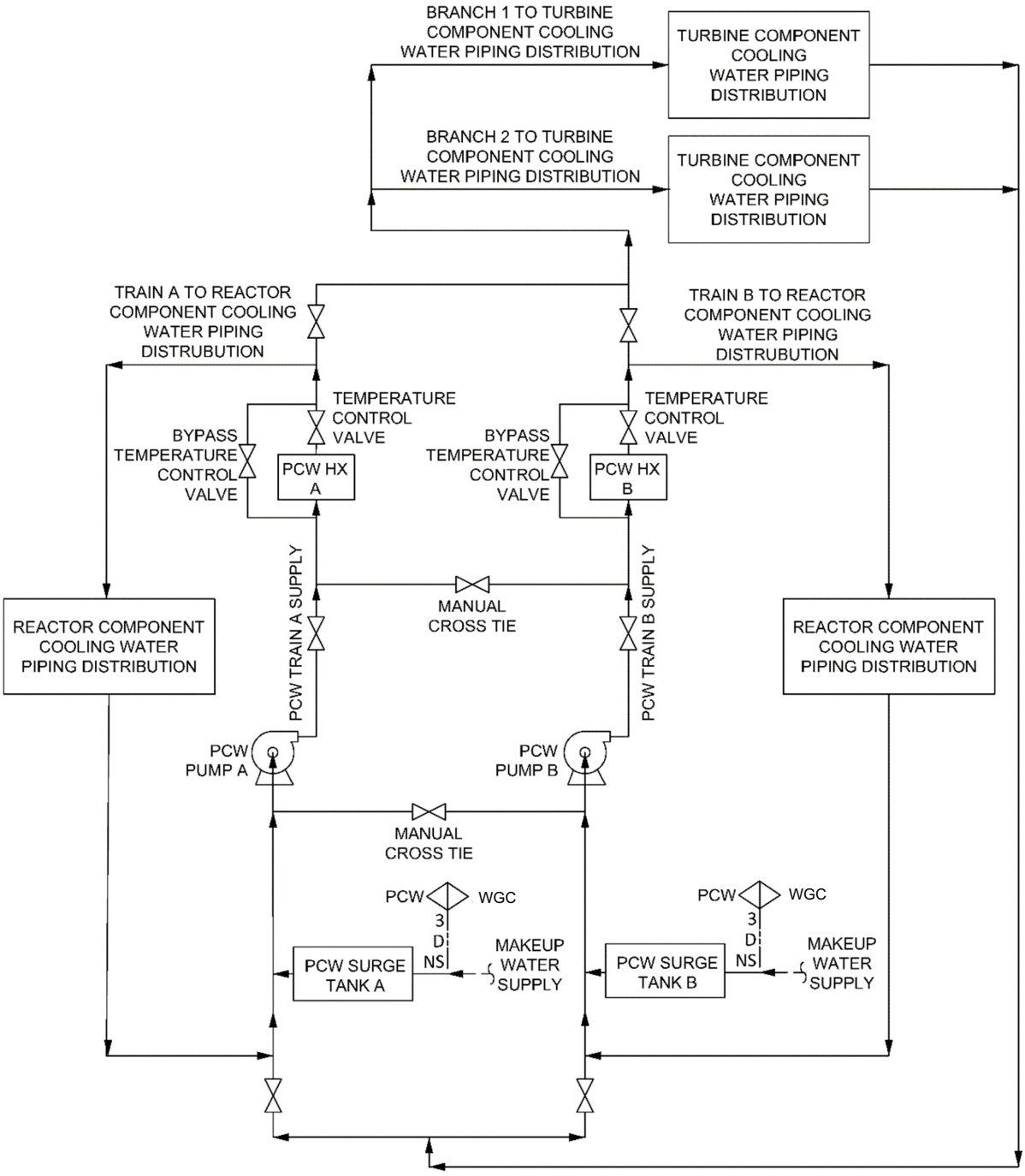

Figure 9.2-2 Plant Cooling Water System Simplified Diagram

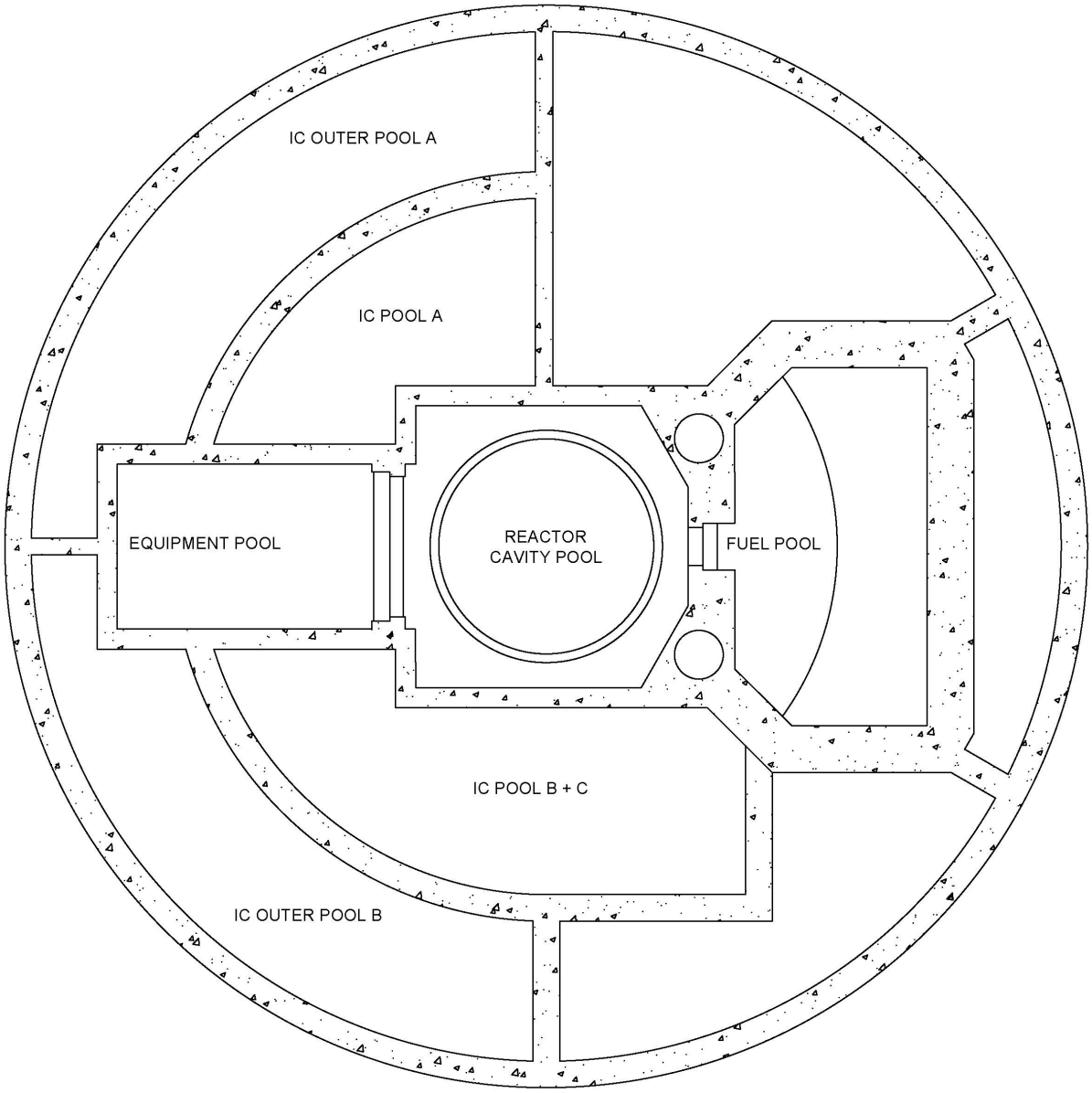

Figure 9.2-3 Ultimate Heat Sink Pools Simplified Diagram

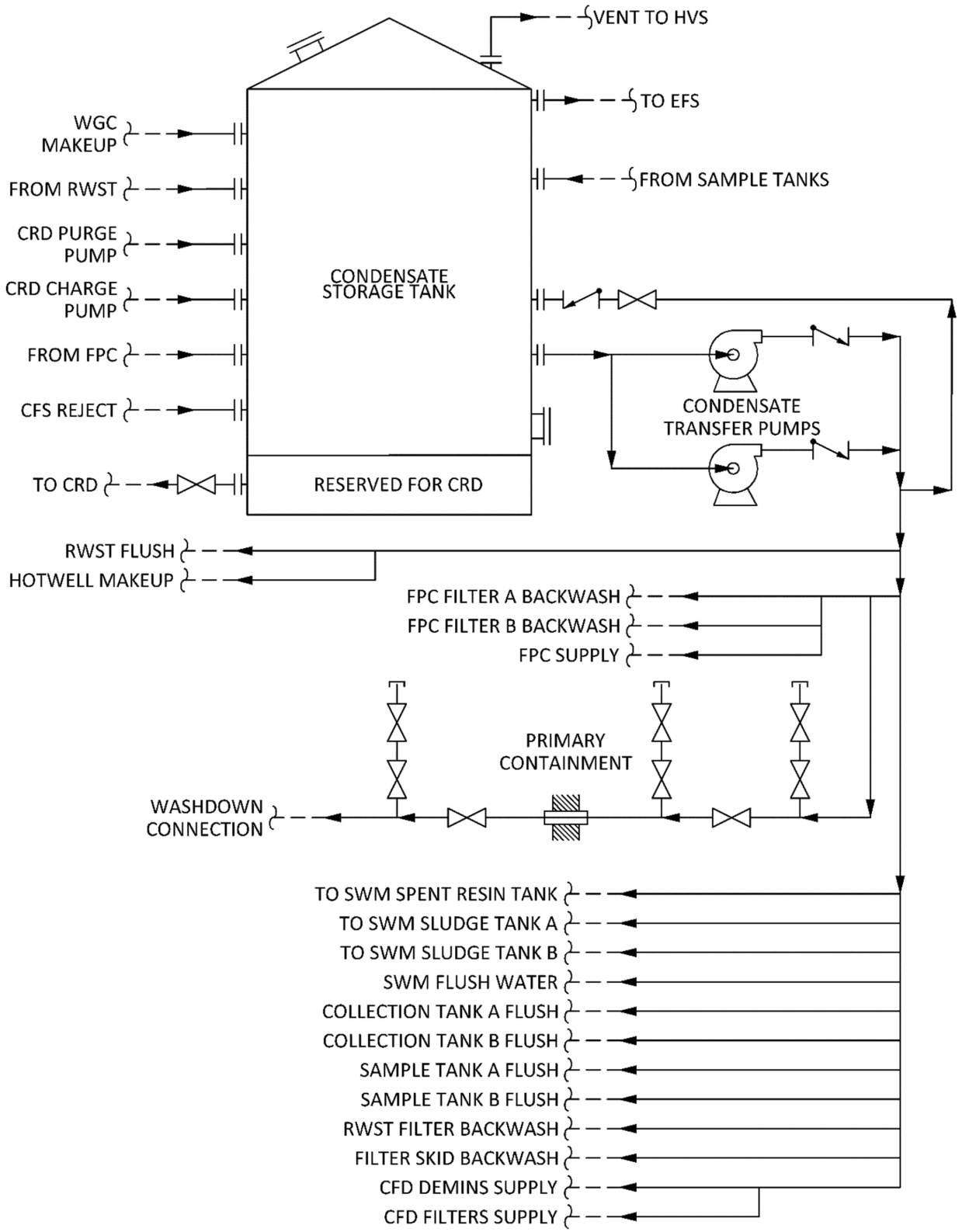

Figure 9.2-4 Condensate Storage and Transfer Subsystem Simplified Diagram

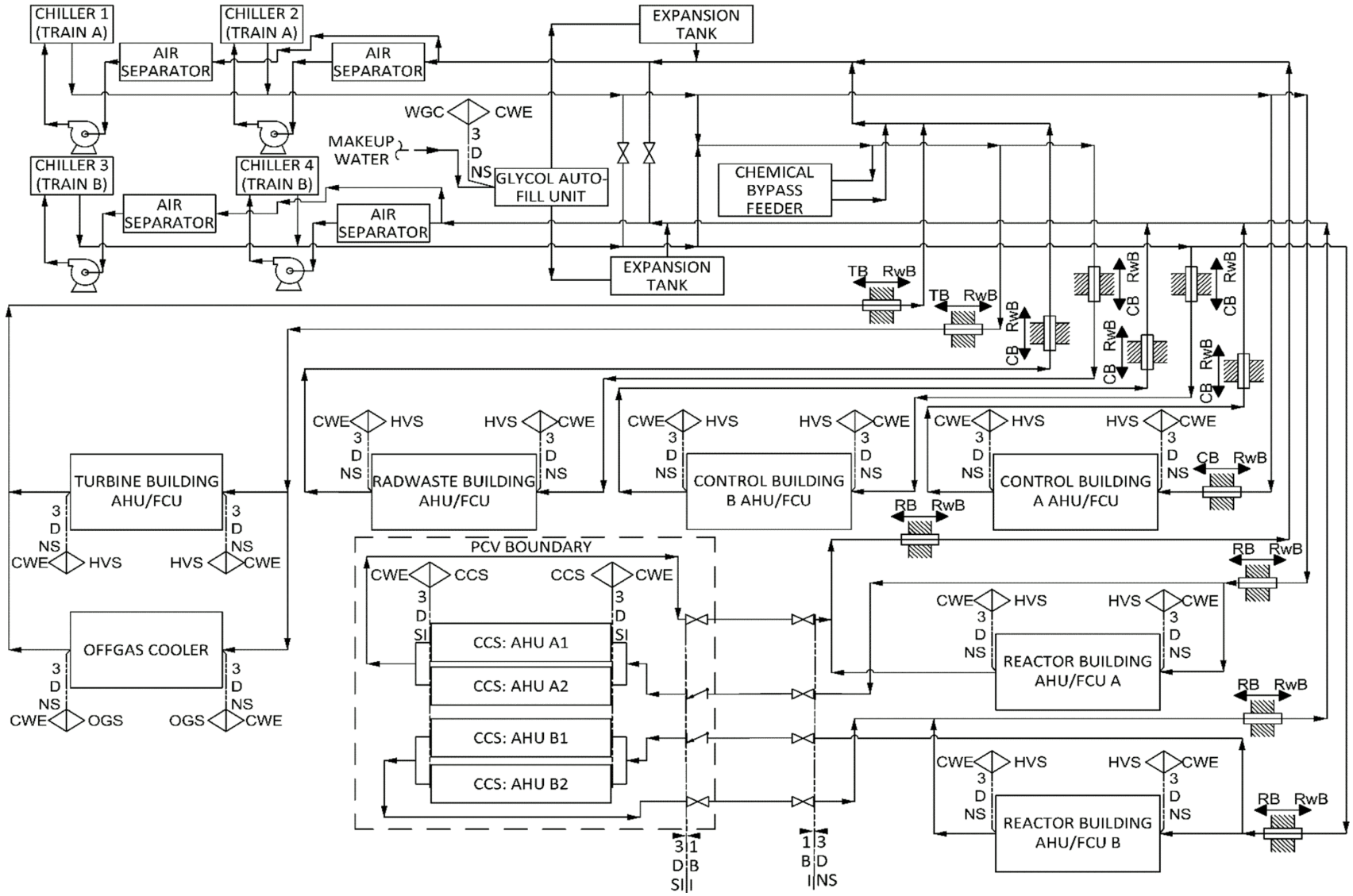

Figure 9.2-5 Chilled Water Equipment Simplified Diagram

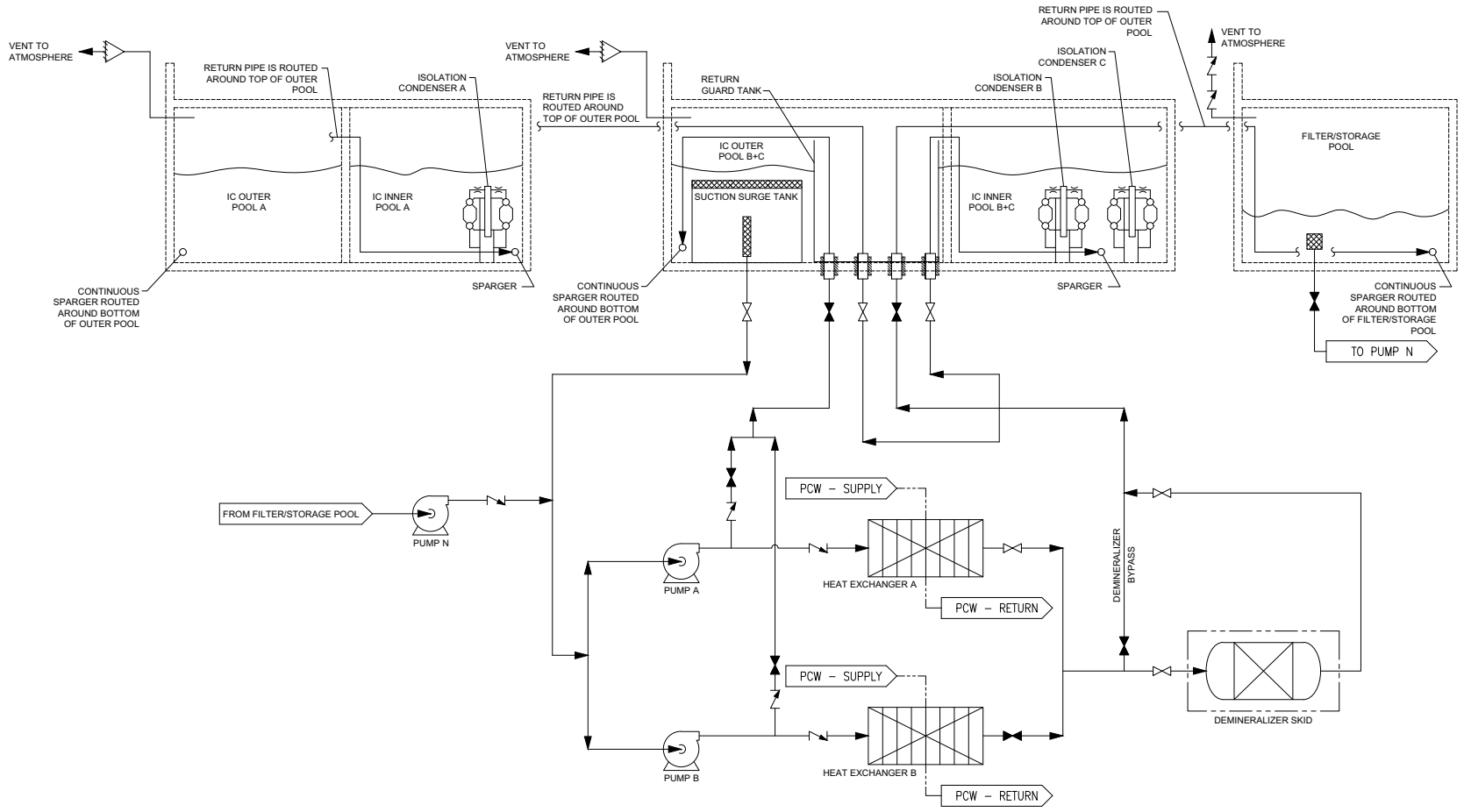

Figure 9.2-6 Isolation Condenser Pools Cooling and Cleanup System Simplified Diagram

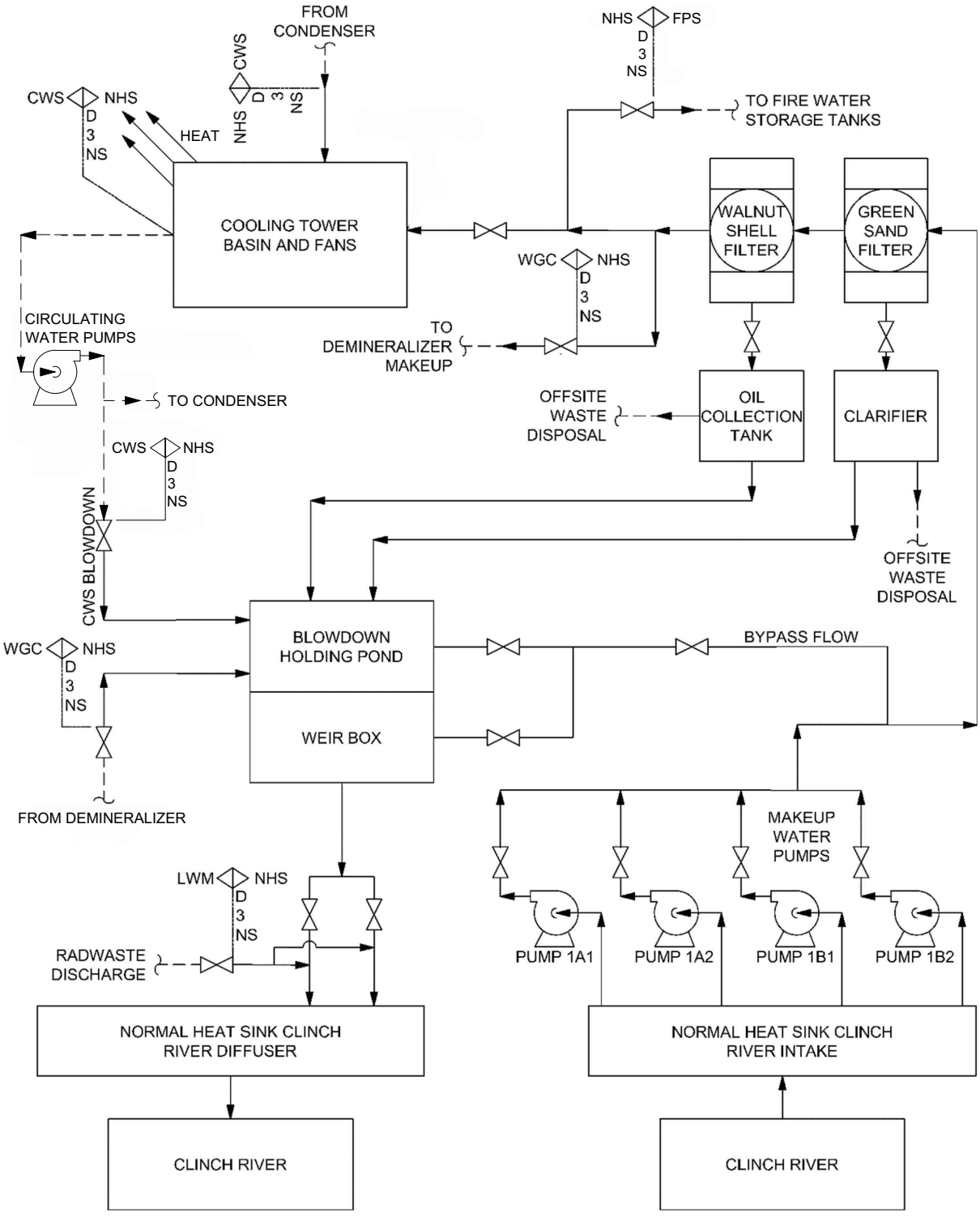

Figure 9.2-7 Normal Heat Sink System Simplified Diagram

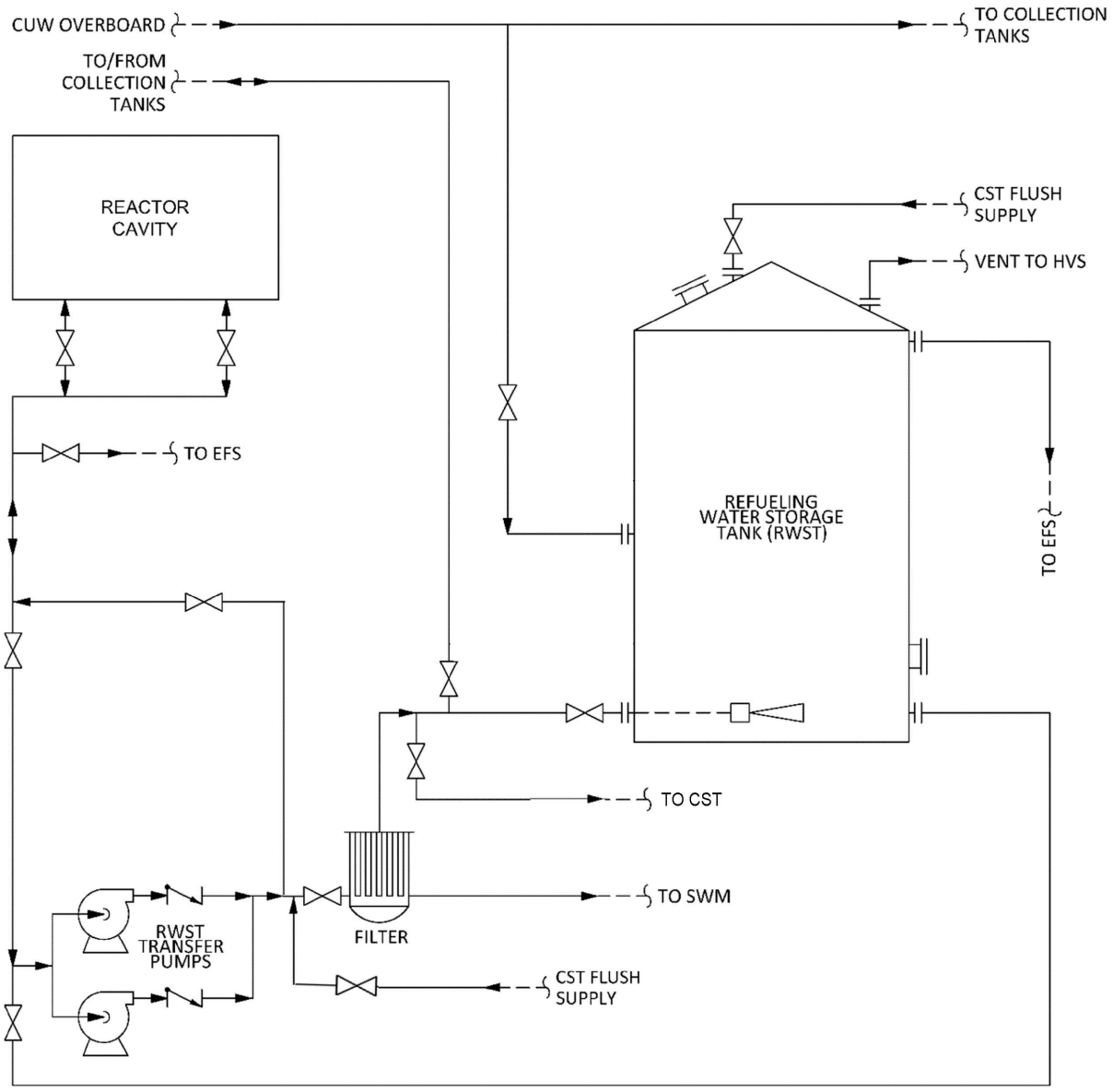

Figure 9.2-8 Refueling Water Storage and Cleanup Subsystem Simplified Diagram

9.3 Process Auxiliaries

9.3.1 Plant Pneumatics System

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 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 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 on 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 Safety Class 3 (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 9.3-7).

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 9.3-5), the U.S. 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 9.3-6) 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, defense-in-depth 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 Design Extension Conditions (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 Defense-in-Depth 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, February 2023.

9.3-5 “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.

9.3-6 EPRI NP-5283-SR-A, “Guidelines for Permanent BWR Hydrogen Water Chemistry Installations,” Electric Power Research Institute.

9.3-7 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.

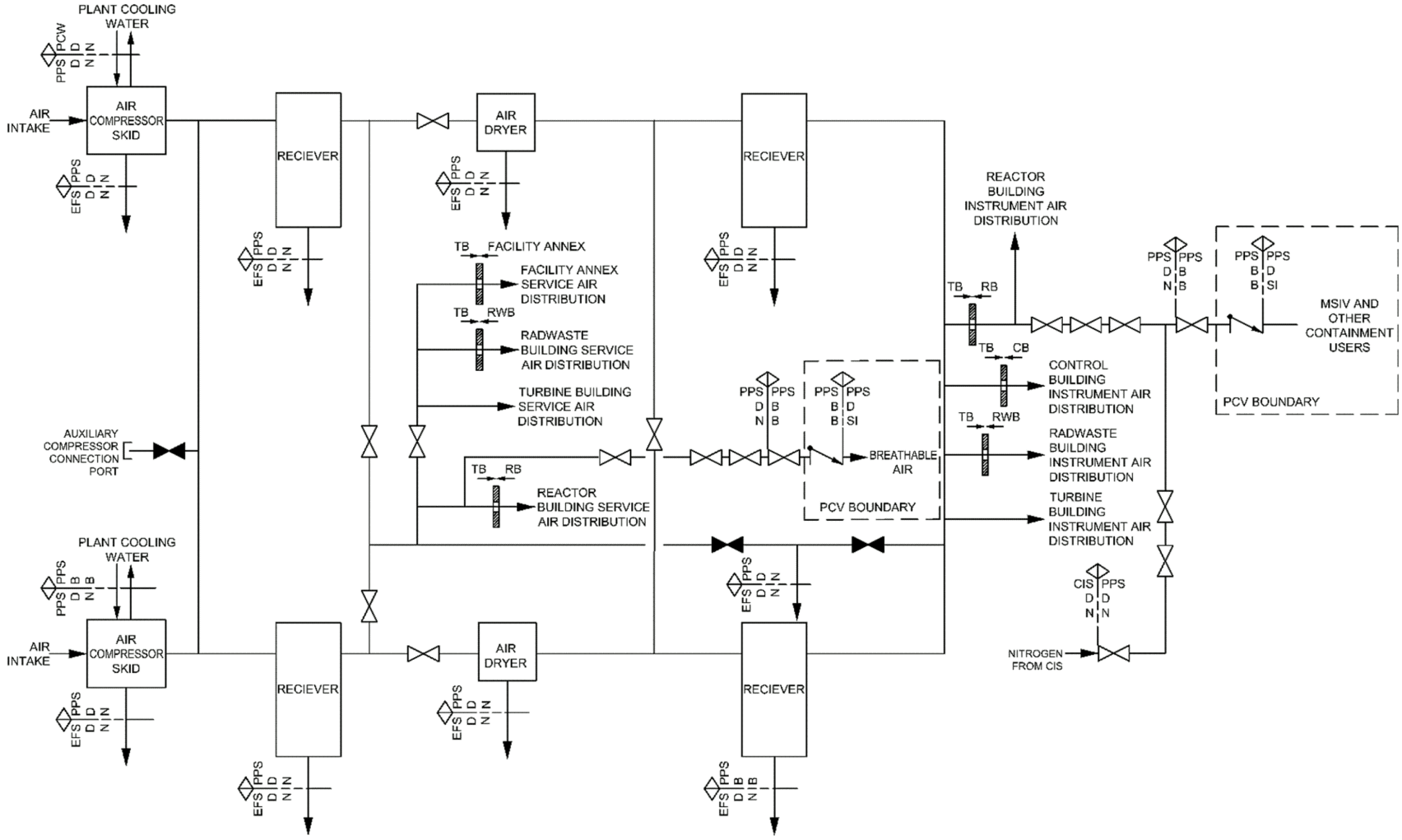

Figure 9.3-1 Plant Pneumatics System Simplified Diagram

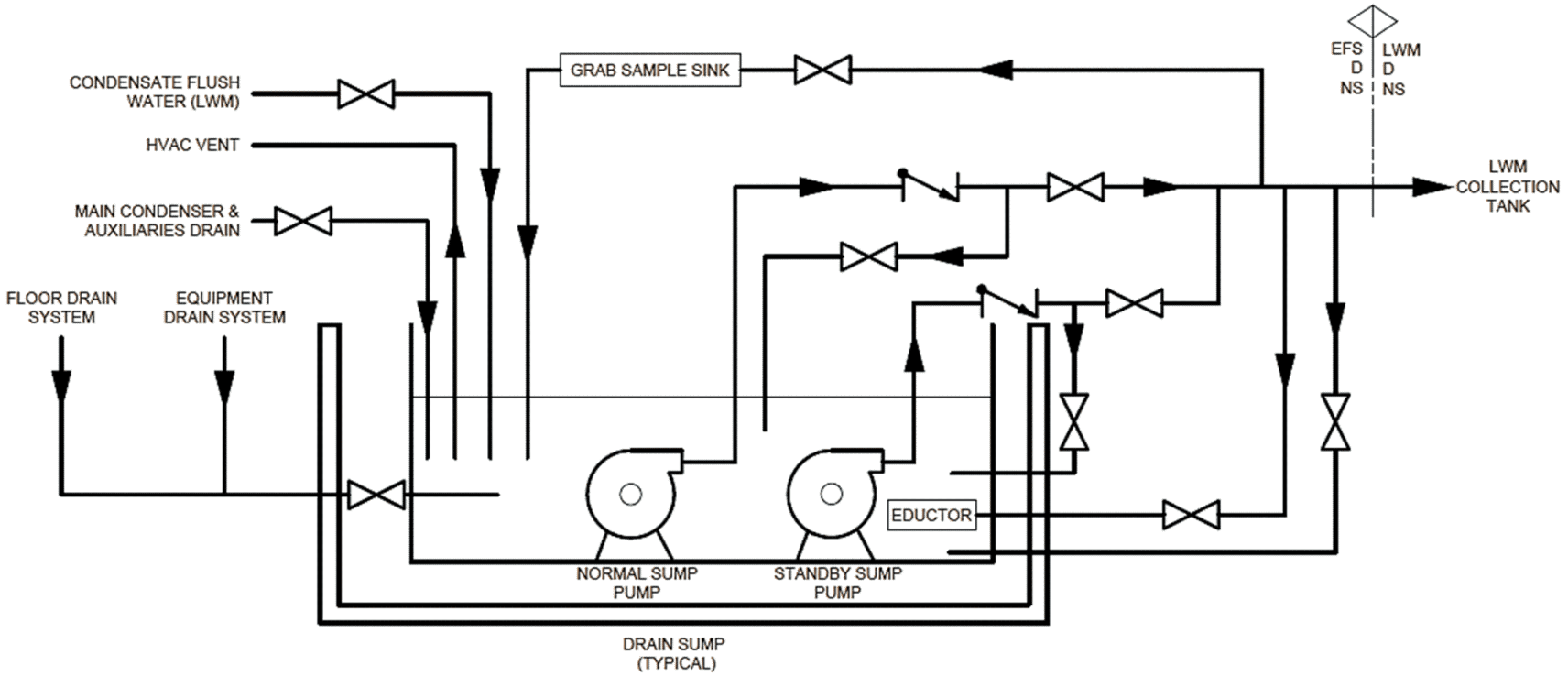

Figure 9.3-2 Normal Sump Simplified Diagram

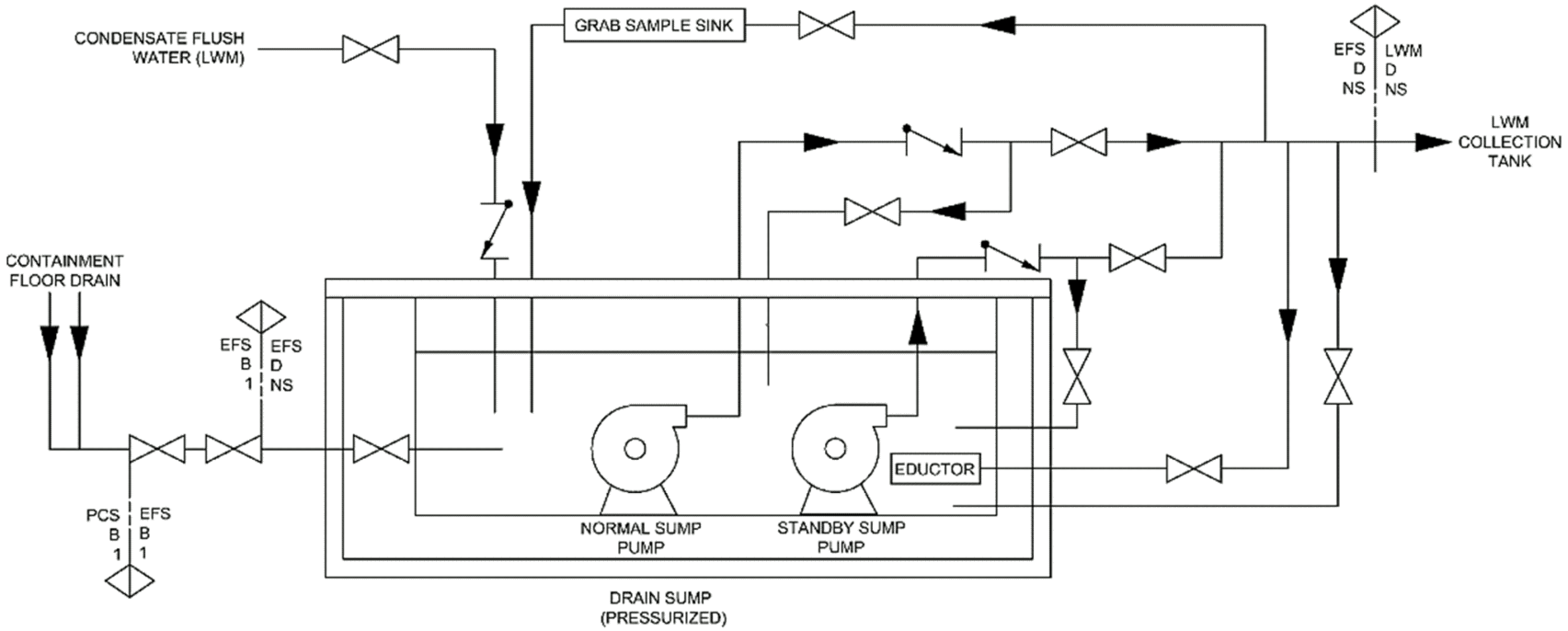

Figure 9.3-3 Pressurized Sump Simplified Diagram

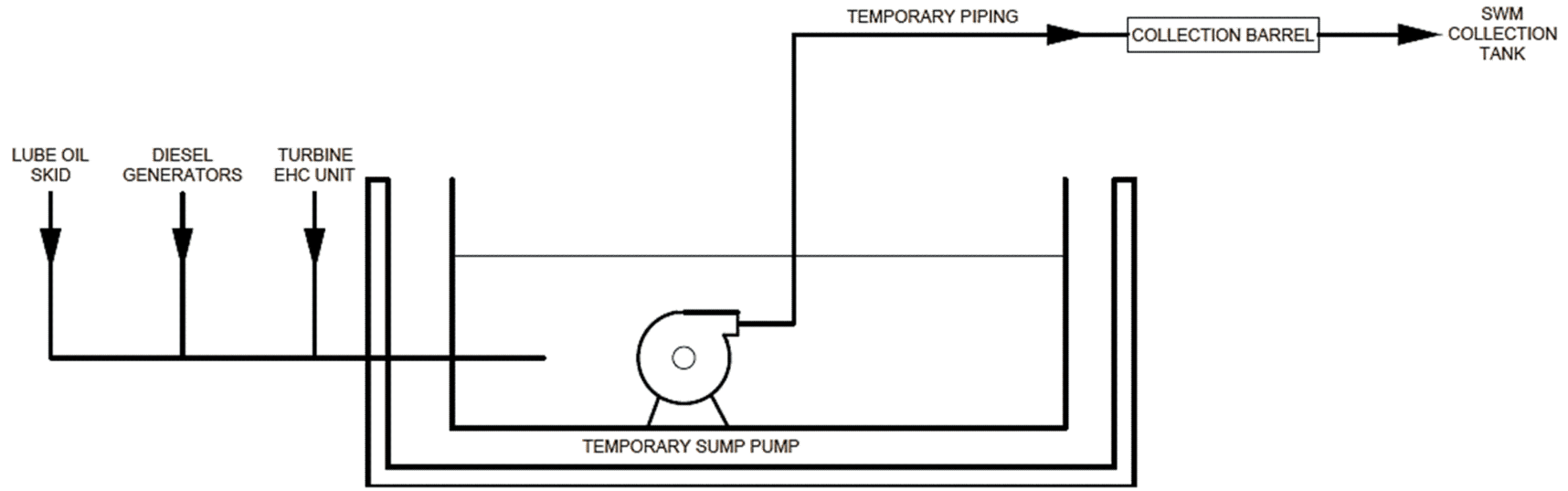

Figure 9.3-4 Oily Waste Sump Simplified Diagram

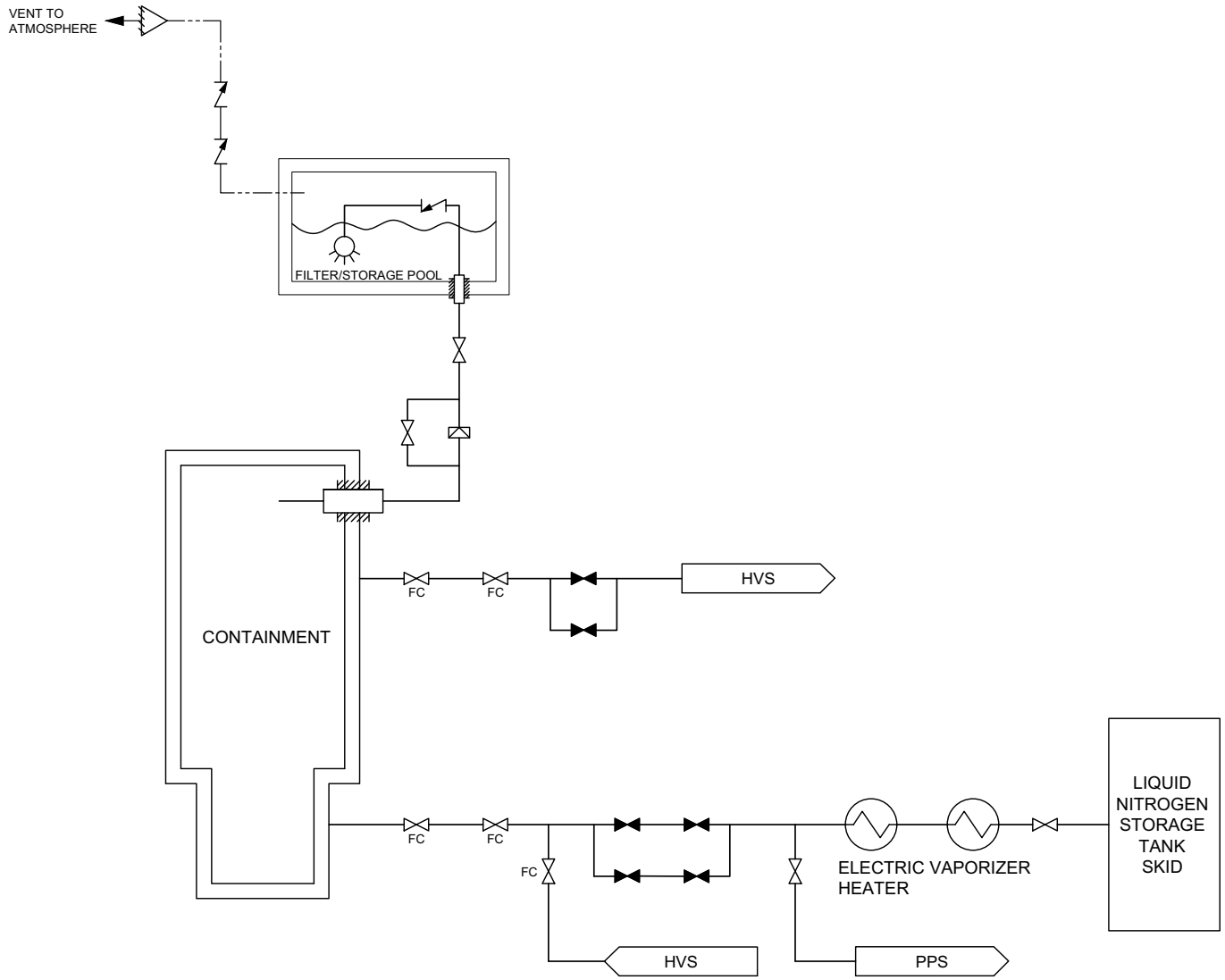

Figure 9.3-5 Containment Inerting System Simplified Diagram

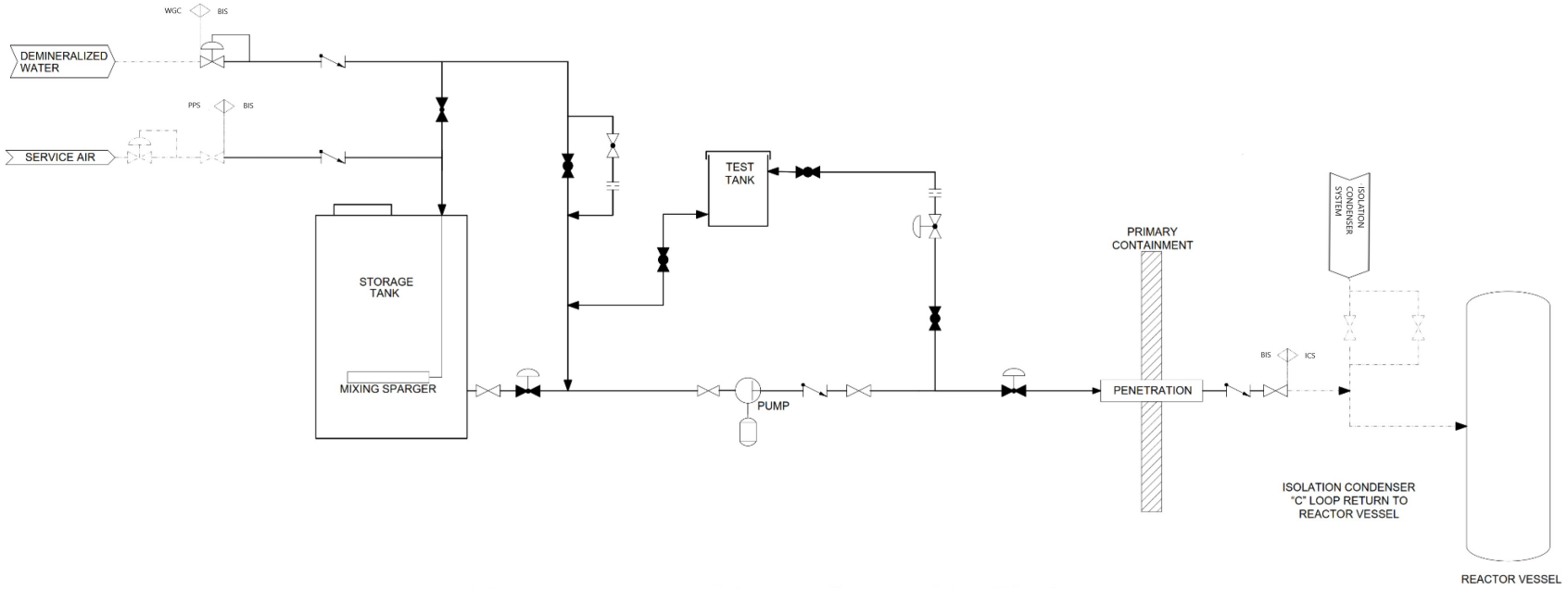

Figure 9.3-6 Boron Injection System Simplified Diagram

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 Classifications, Quality Groups, and Seismic Categories of the CB HVS 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. Per the offsite chemical hazards screening for Clinch River Nuclear Unit 1 (CRN-1), the release of chemicals from nearby storage sites need not be considered a design basis event.

Mobile sources of hazardous chemicals were also considered in an offsite transportation hazards screening for CRN-1. Most hazardous chemicals are screened out with the exceptions being anhydrous ammonia, chlorine, hydrochloric acid, or nitric acid due to the proximity of highway I-40 to the site (within 5 miles). For these postulated offsite chemical releases via a traffic accident, design features of the CB HVS offer protection for operators in the MCR. 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 10 Code of Federal Regulations (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 Classifications, 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 Classifications, 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 Classifications, Quality Groups, and Seismic Categories of the RB HVS SSCs 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 for CRN-1 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 Classifications, 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, “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, “American National Standard for Boiling Water Reactor Containment Ventilation Systems,” American National Standards Institute/American Nuclear Society, 1978 (R1987).

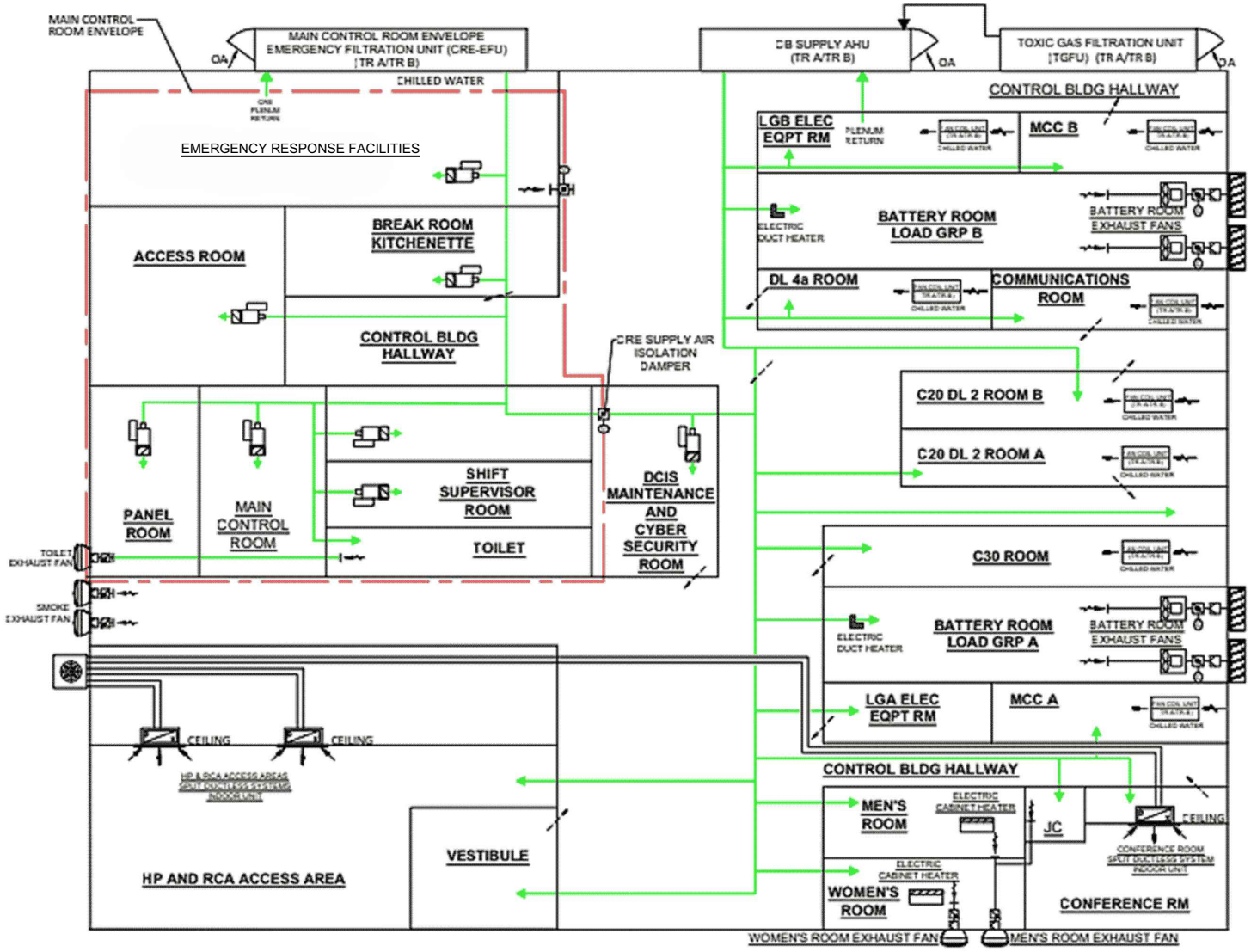

Figure 9.4-1 Control Building Heating, Ventilation, and Cooling Simplified Diagram

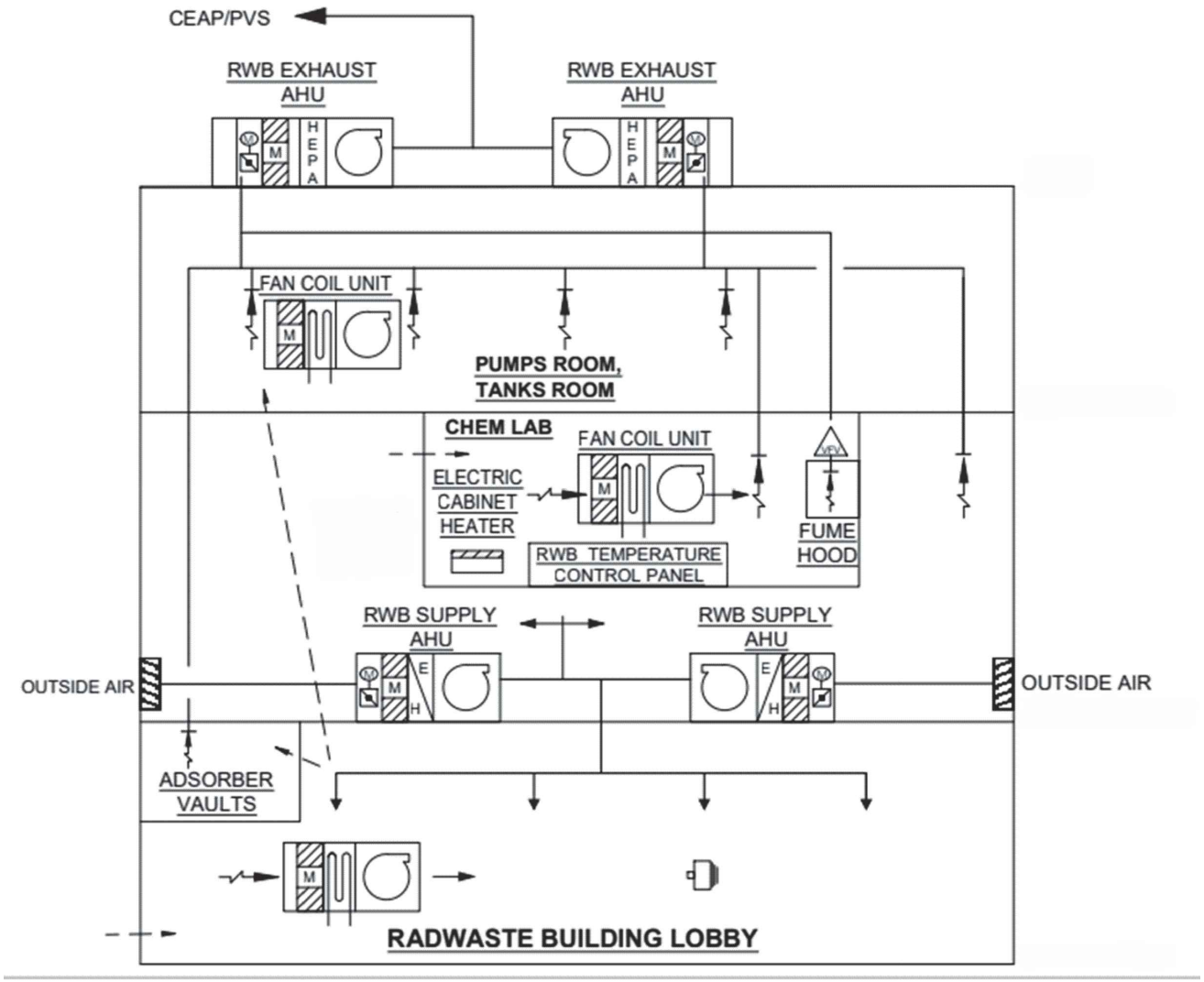

Figure 9.4-2 Radwaste Building Heating, Ventilation, and Cooling Simplified Diagram

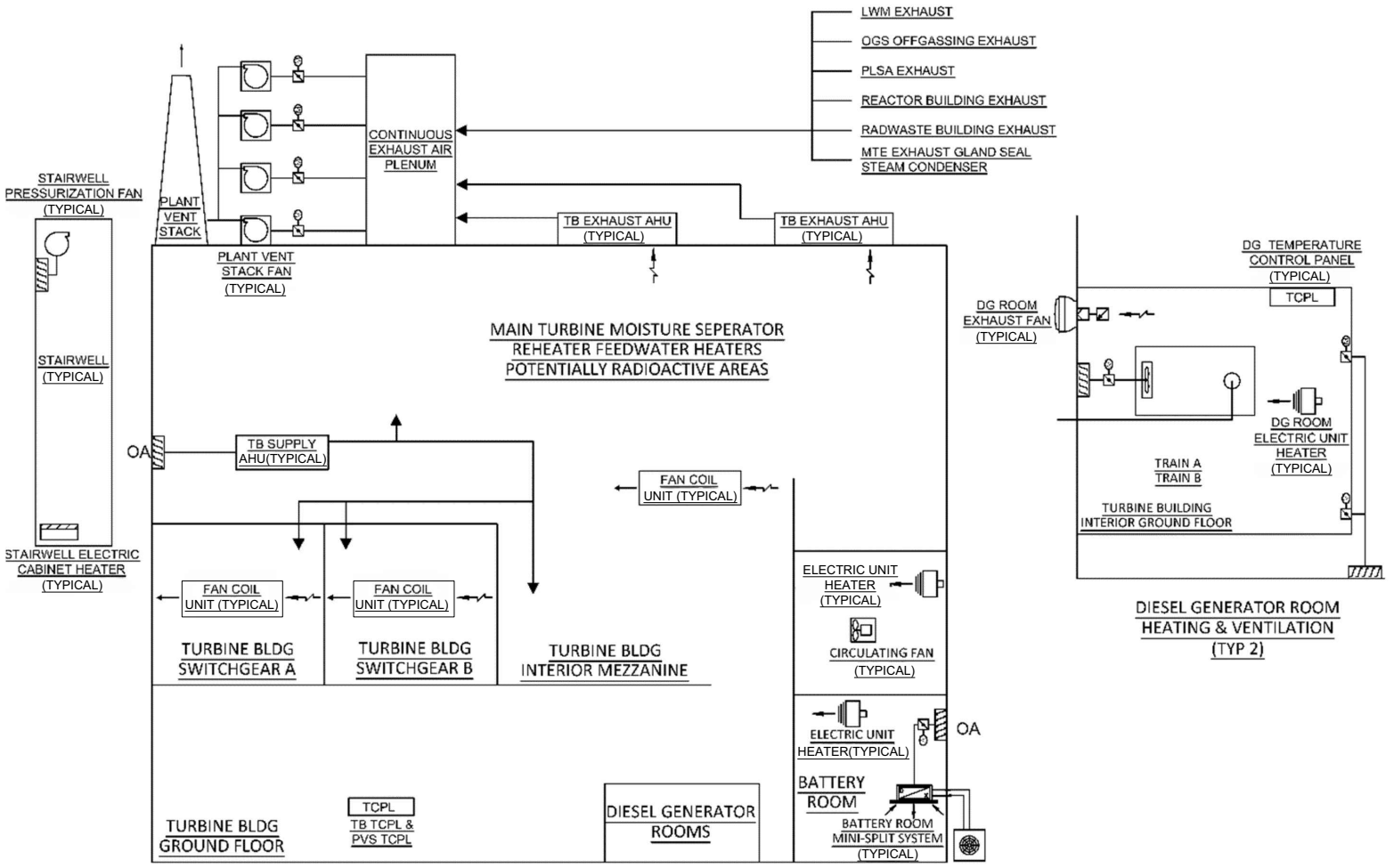

Figure 9.4-3 Turbine Building Heating, Ventilation, and Cooling and Plant Vent Stack Simplified Diagram

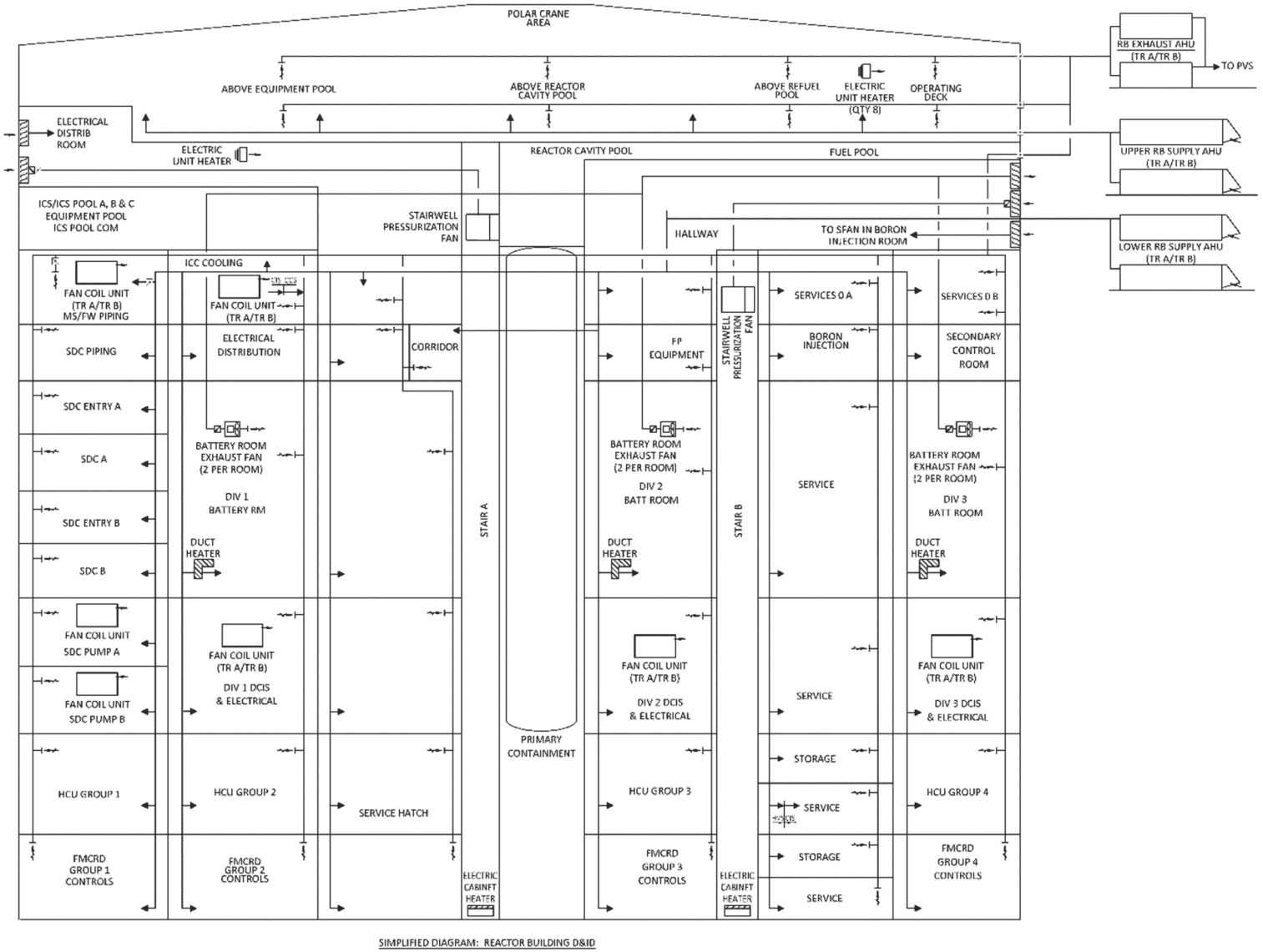

Figure 9.4-4 Reactor Building Heating, Ventilation, and Cooling Simplified Diagram

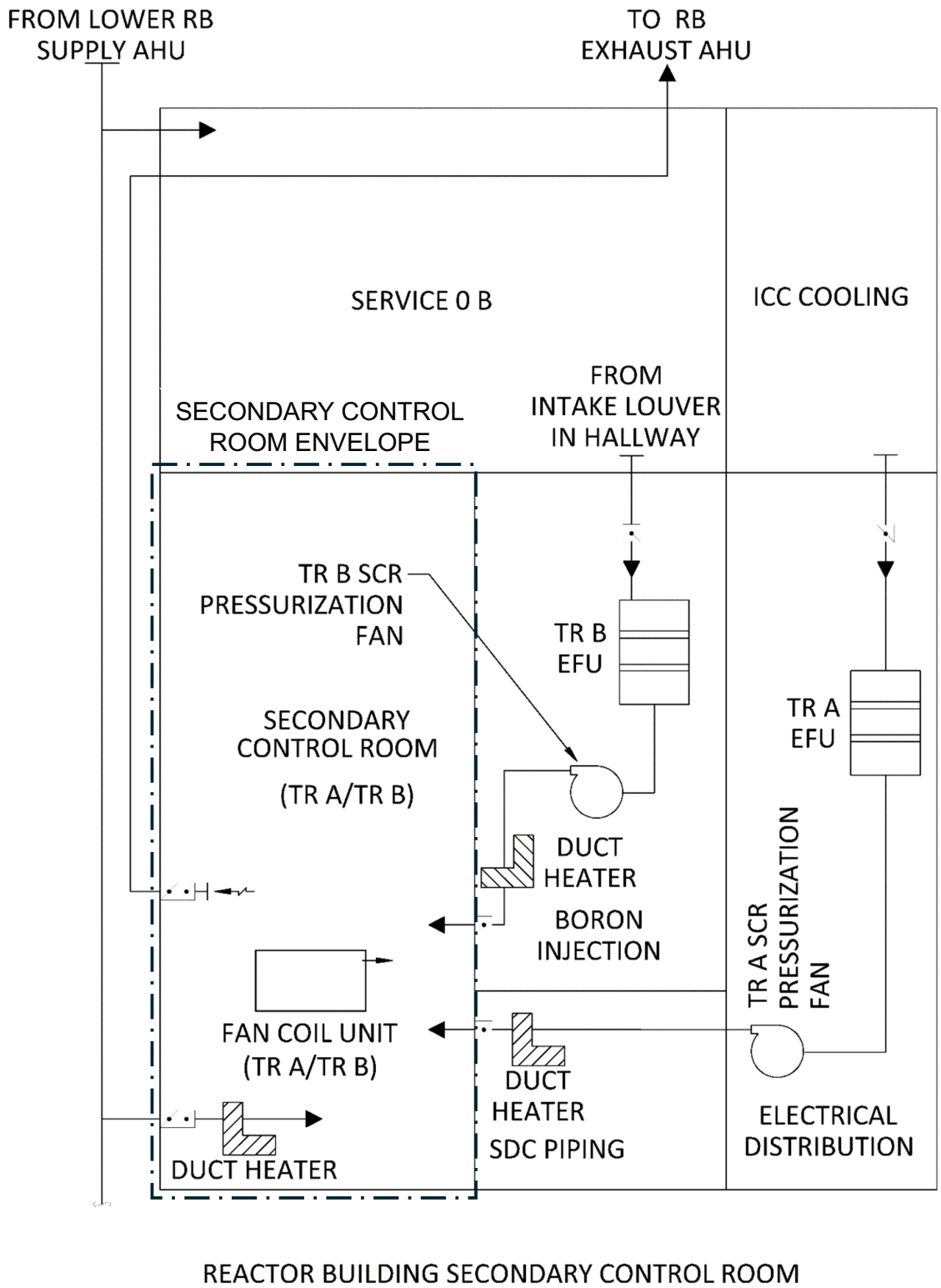

Figure 9.4-5 Secondary Control Room Heating, Ventilation, and Cooling Simplified Diagram

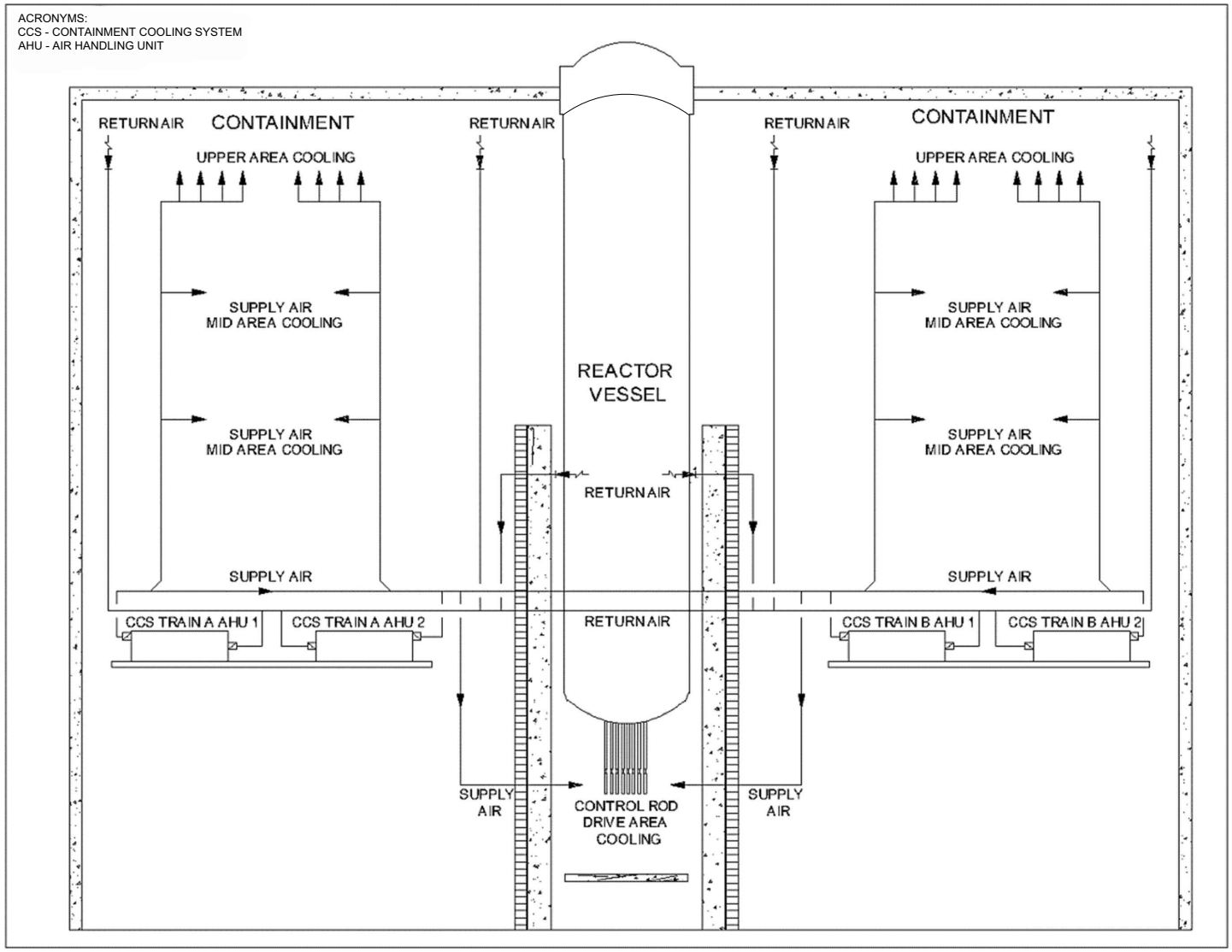

Figure 9.4-6 Containment Cooling System Simplified Diagram

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 and Subsection 9A.3.2 describes the methodology for the Fire Safe Shutdown Analysis (FSSA). The Fire Hazards Analysis and FSSA will be completed for Clinch River Nuclear Unit 1 (CRN-1) prior to Issuance of the 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.

Section 15.6 evaluates fire as a potential internal hazard in the Probabilistic Safety Assessment (PSA).

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 Fire Hazards Analysis
- 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 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 effect.

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 Fire Hazards Analysis 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 (for example, of 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 Fire Hazards Analysis 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 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 complex is separated from the remainder of the building by a fire barrier having a fire rating of three hours.

The Main Control Room (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 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 Fire Hazards Analysis. 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 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 tank 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, Factory Mutual approved for fire main service, 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.

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 Fire Hazards Analysis.

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 Fire Hazards Analysis. 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 Fire Hazards Analysis. 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 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 Safety Class 1 (SC1) SSCs or significant combustible loading according to the Fire Hazards Analysis 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 Fire Hazards Analysis.

Smoke detection in ventilation systems is provided to meet NFPA 90A/92 requirements. 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 with integrated, supervised, one-way communications within the protected area. 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.

Manual fire alarm station (pull box station) locations are determined by the Fire Hazards Analysis.

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. 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. 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 Fire Hazard Analysis.

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 Fire Hazard Analysis.

Smoke removal from areas containing SSCs required for fire safe shutdown is provided by the HVS system operating in smoke removal mode. An HVS mode pressurizes fire rated stairwells as required by the Fire Hazard Analysis.

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 10 Code of Federal Regulations (CFR) 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 effect 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 Fire Hazards Analysis 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 4, May 2021

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 Fire Hazards Analysis.

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, and testing descriptions are provided in the FSAR.

The following NFPA tests, which are required by code, are performed as a minimum part of the inspection and maintenance requirements 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 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 relevant requirements of the following regulations:

The design of the Non-DCIS Communication System complies with 10 CFR Part 50, Appendix E, IV.E(9). 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 withNuclear 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 are 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

- During a LOPP, fuel oil is transferred from the bulk storage tank to the standby diesel generator day tanks.

Safety Category N Functions

- The Fuel Oil Storage and Transfer System provides bulk fuel oil storage and distribution to the day tanks.

The Safety Classifications, Quality Groups, and Seismic Categories of the Fuel Oil Storage and Transfer System 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.

The Diesel Fuel Oil Storage and Transfer System provides storage and transfer of diesel fuel oil to the day tanks of the SDGs. This subsystem is comprised of one (1) Diesel Fuel Oil Storage Tank, two (2) fully redundant Diesel Fuel Oil Transfer Pumps, and associated piping, valves, and instrumentation. The fuel oil transfer pumps are powered from the Standby Power System to ensure diesel availability. To prevent freezing, the Fuel Oil Storage Tank is equipped with an immersion heater and a recirculation line from the transfer pumps. A piping tie-in is provided to the Diesel Fuel Oil Storage Tank for FLEX connections.

The Diesel Fuel Oil Storage Tank is double-walled or provided with a containment berm capable of holding the contents of the tank if a leak develops. The offloading area is bermed to catch the contents of the transport truck if leakage occurs. The bermed area(s) contain(s) a sump pit which can be pumped out following sampling. Provisions are made for sampling of the bermed area(s) for residual oil. The pumps transfer fuel oil underground from the Diesel Fuel Oil Storage Tank to the SDG day tanks located in the diesel generator rooms.

Underground fuel oil lines are designed to allow for leak detection.

The Diesel Fuel Oil Transfer Pumps distribute diesel fuel oil from the Diesel Fuel Oil Storage Tank to the SDG day tanks and provides recirculation as appropriate. The SDG day tanks are double walled to provide containment and minimize the chance of a fuel spill in the SDG rooms.

Figure 9.5-2 provides a simplified diagram of the Diesel Fuel Oil Storage and Transfer System.

The Fuel Oil Storage Tank and Transfer Pumps are located outside of the protected area. This allows fuel deliveries without needing to enter the protected area. Additionally, this increases the distance between a potential external combustible source and the plant.

The Fuel Oil Storage Tank satisfies the requirements of API-650, “Welded Steel Tanks for Oil Storage” (Reference 9.5-2).

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 is to be finalized at 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. Piping and equipment layout is to be finalized at 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.

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 Association, 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 4, May 2021.
16	American Society of Mechanical Engineers, ASME A17.1, “Handbook on Safety Code for Elevators and Escalators,” 2022.

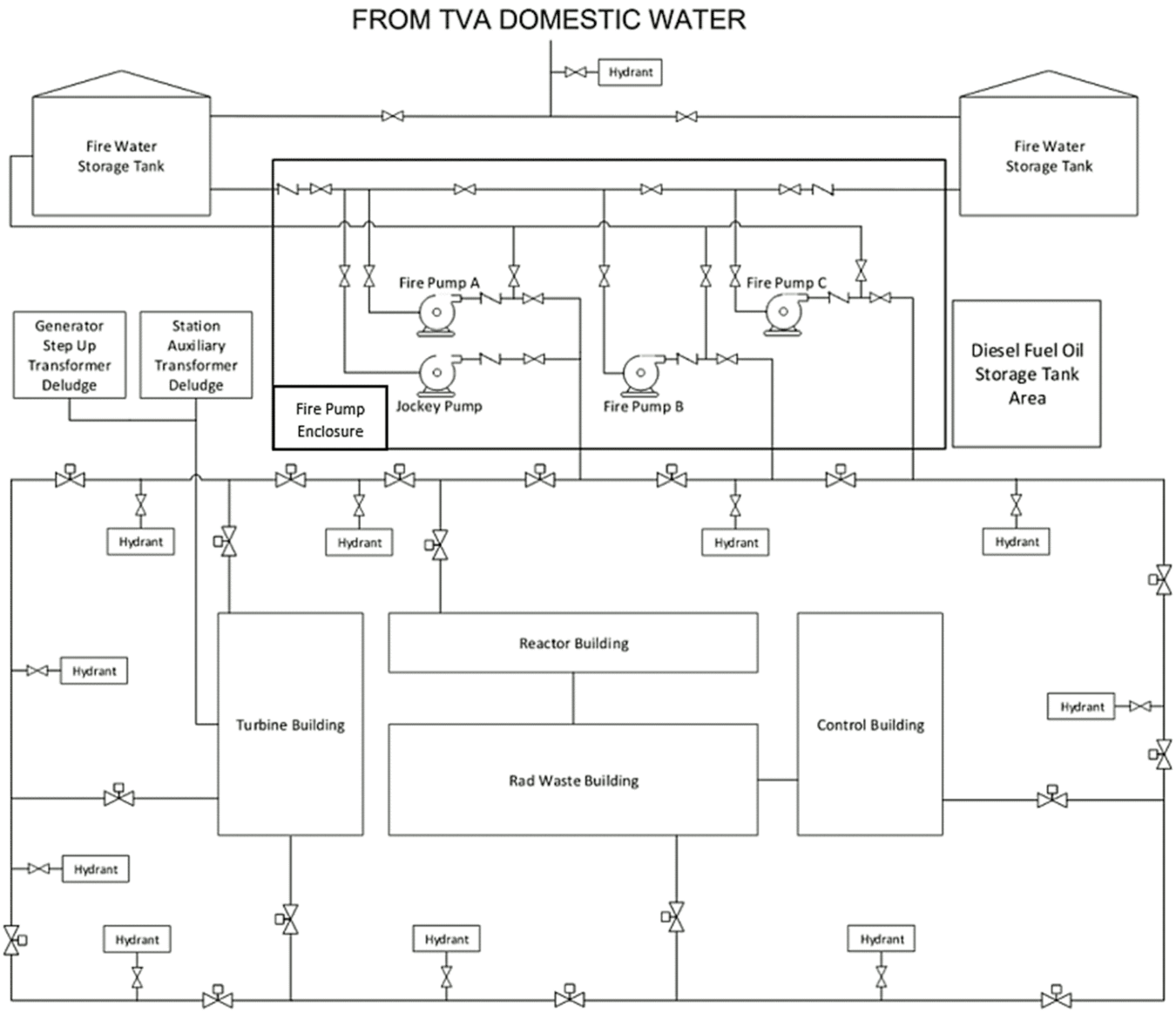

Figure 9.5-1 BWRX-300 Fire Protection System Simplified Diagram

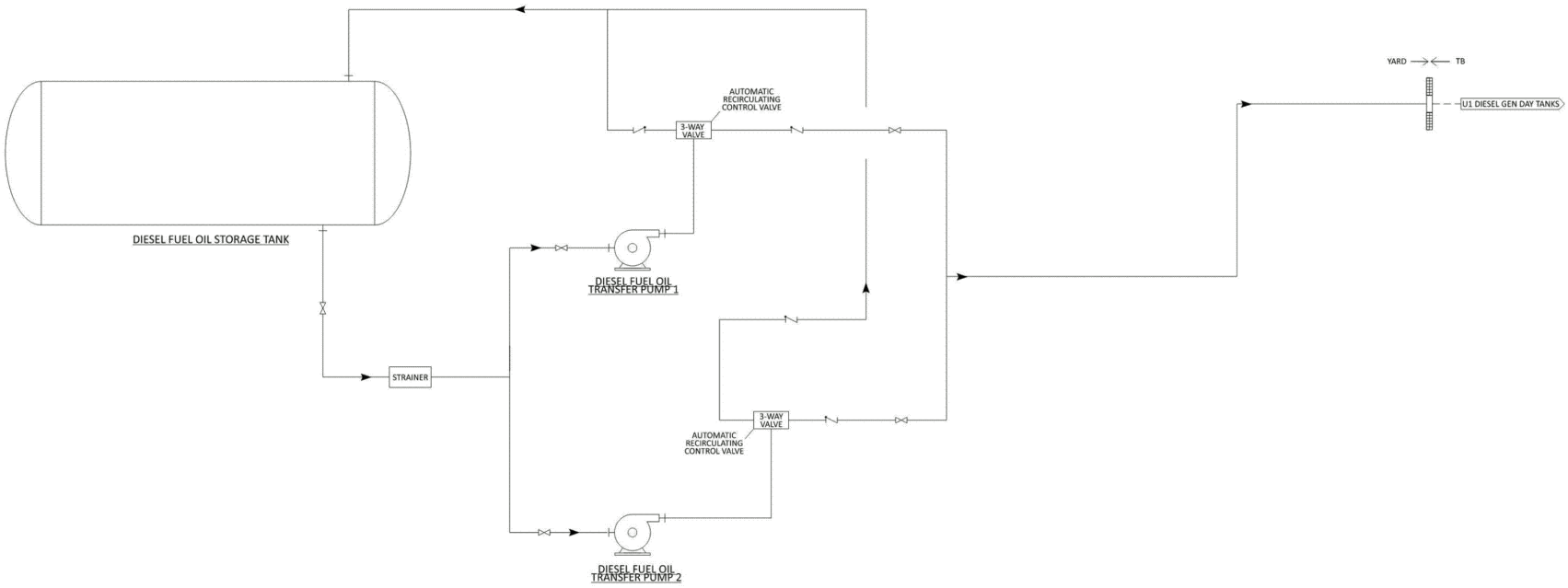

Figure 9.5-2 Diesel Fuel Oil Storage and Transfer System Simplified Diagram

APPENDIX 9A FIRE HAZARDS ANALYSIS

This section describes the content of the Fire Hazard Analysis for the Clinch River Nuclear Unit 1 (CRN-1). The Fire Hazard Analysis 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 Fire Hazard Analysis and confirms suitability. This section does not include all the analyses that are necessary for a complete Fire Hazards Analysis and Fire Safe Shutdown Analysis (FSSA). The completed documents will be issued prior to the issuance of the FSAR.

9A.1 Introduction

Performing a Fire Hazard 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 4, “Fire Protection for Nuclear Power Plants” provides guidance in the preparation of the Fire Hazard Analysis. The Fire Hazard Analysis includes fire protection requirements for the design, construction, commissioning, and operation of nuclear power plants, including 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 Fire Hazard Analysis 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 Fire Hazard Analysis 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 Fire Hazard Analysis is updated as necessary to reflect plant design and operational changes. The Fire Hazard Analysis 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 Fire Hazard Analysis 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 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) and the intake pumping station, are in the scope.

9A.2.2 Codes and Standards

The Codes and Standards utilized in the FPS for the CRN-1 BWRX-300 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 CRN-1 BWRX-300 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 Fire Hazard Analysis 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 Fire Hazard Analysis 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 Fire Hazard Analysis and FSSA is based on the methodologies described below.

9A.3.1 Methodology for Fire Hazard 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 Fire Hazard Analysis.

The content of the Fire Hazard Analysis complies with applicable guidance in RG 1.189, including elements and attributes listed in Regulatory Position 1.2. The Fire Hazard Analysis 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 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:

1. Identify the fire safe shutdown functions.

2. 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.

3. 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.

4. 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 Fire Hazard Analysis document contains the evaluation and analysis data for each fire zone in the CRN-1 BWRX-300 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 Fire Hazard Analysis and FSSA as applicable.

9A.4 Fire Hazard Analysis and Fire Safe Shutdown Analysis Summary

Fire Protection Analyses by Room, Fire Area, or Fire Zone are provided in the Fire Hazard Analysis and FSSA documentation.

It is a general conclusion of the FSSA that the station has the capability to safely shutdown the operating unit in the event of a fire. The resulting analysis documented prior to the FSAR is to 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 Fire Hazards Analysis and FSSA documents.

9A.6 References

9A-1 NFPA 804, “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.

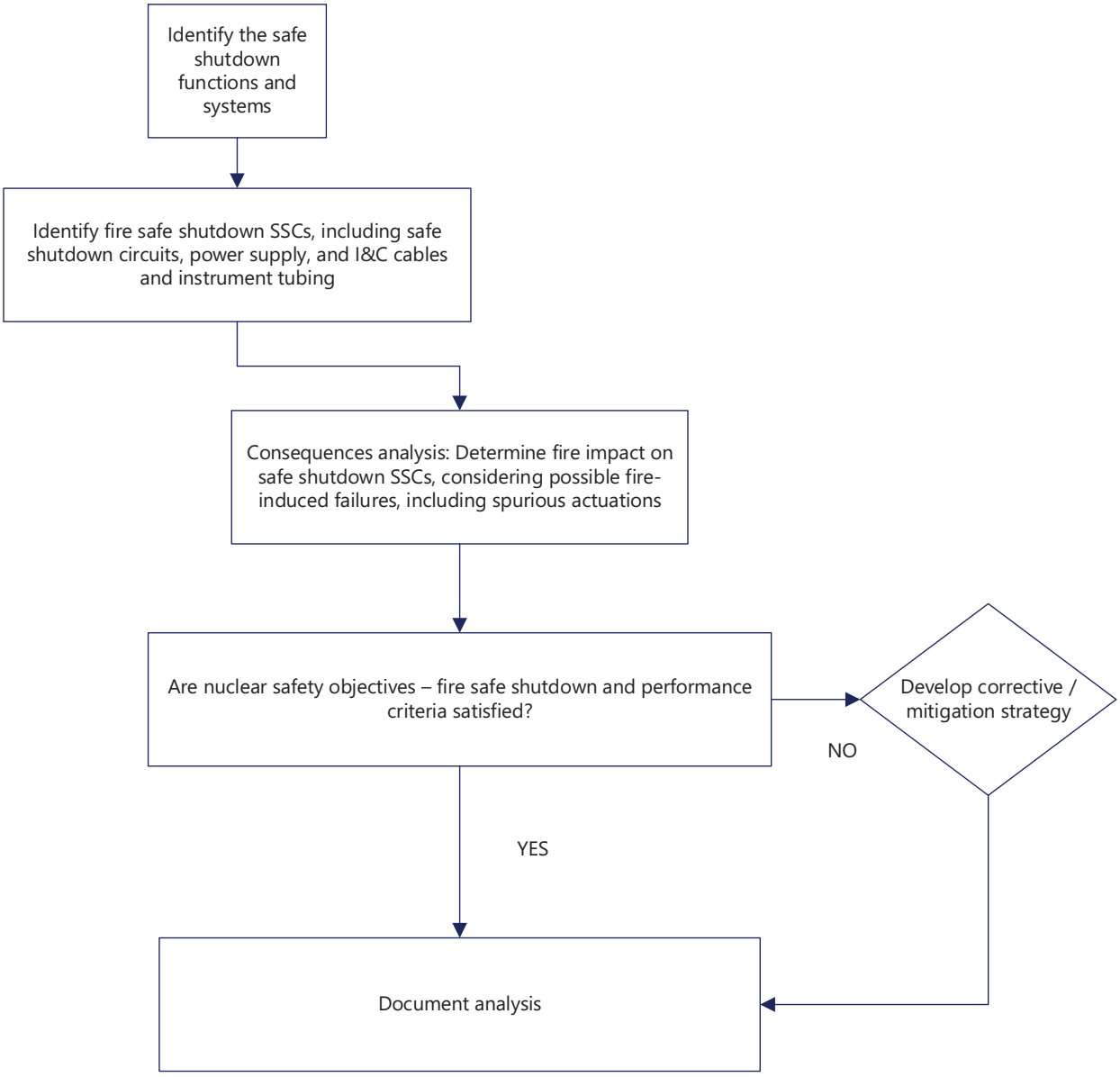

Figure 9A-1 BWRX-300 Fire Safe Shutdown Analysis Methodology

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LIST OF ACRONYMS AND ABBREVIATIONS

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
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

LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
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
NUREG	Nuclear Regulatory Report
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
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

LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
TSV	Turbine Stop Valve
USNRC	U.S. Nuclear Regulatory Commission

LIST OF EFFECTIVE SECTIONS

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CHAPTER 10 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 (MS) subsystem 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 System (CFD), and the Condensate and Feedwater Heating System (CFS).

A simplified flow diagram with representative heat balance of the steam and power conversion system is provided in Figure 10.1-1. A summary of the important design and representative 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) turbine 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.

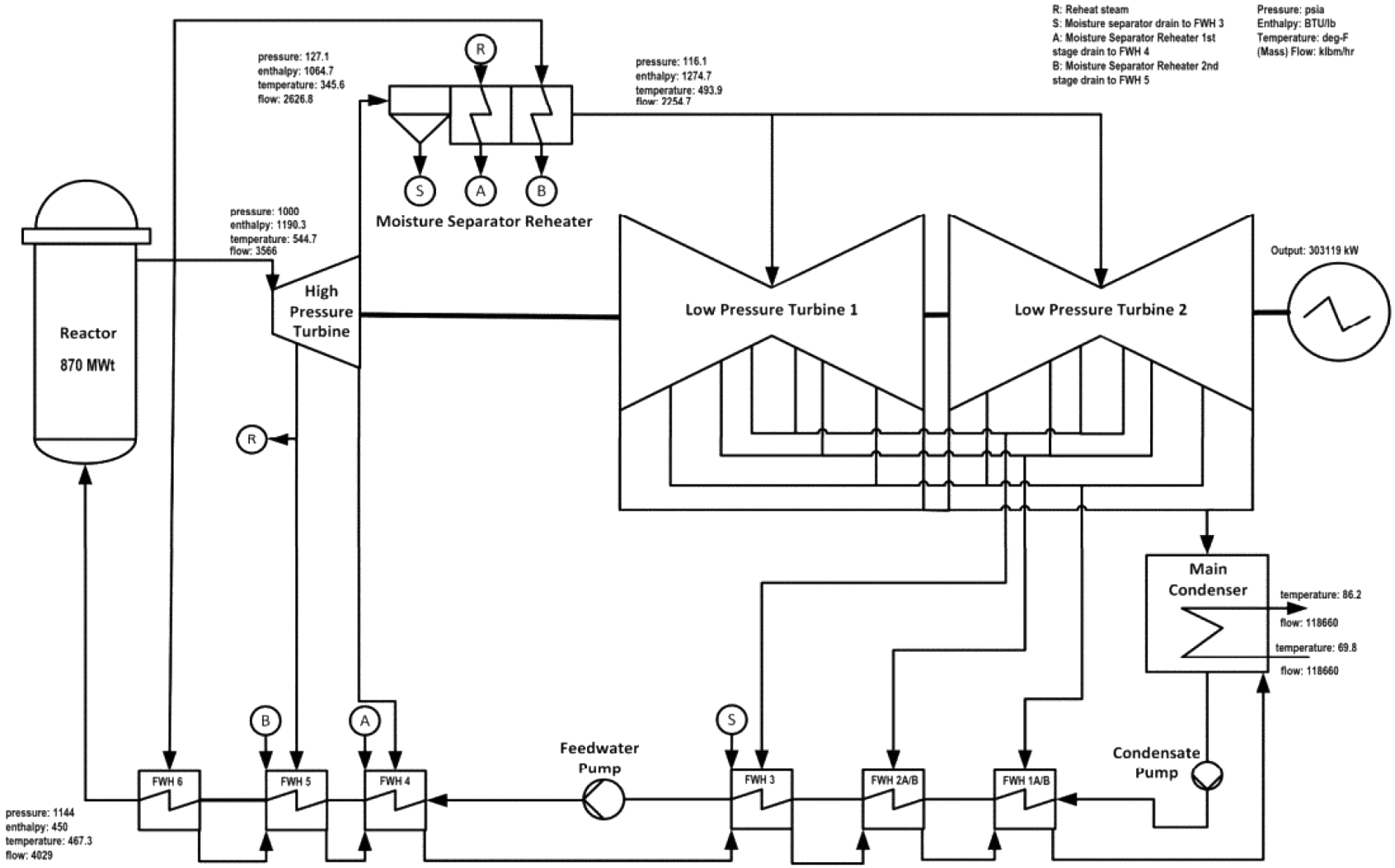
Other features of the steam and power conversion system are presented as follows in Section 10.4:

- 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 – 25% capacity each
Circulating water flow rate, Mlbm/hr	118.7
Circulating water temperature rise, ºF	16.4
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.



Note: Values shown on this diagram 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

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 Classifications, 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 Subsystem
- EHC Subsystem

Figure 10.2-1 is a simplified 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 (FWH)
- 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 Subsystem 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) or Design Basis Accident (DBA). These event classifications correspond to American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” (Reference 10.2-5), Service Levels of Upset and Emergency 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.

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) 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) A370-24a, “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 E208-20, “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-lbf.
- 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 $\sqrt{}$ 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. Position C.5 of RG 1.115 states that the U.S. 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 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 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 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 A370-24a, “Standard Test Methods and Definitions for Mechanical Testing of Steel Products,” ASTM International, 2024.

10.2-2 ASTM E208-20, “Standard Test Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels,” ASTM International, 2020.

10.2-3 EPRI NP-5479, “Application Guide for Check Valves in Nuclear Power Plants,” Electric Power Research Institute.

10.2-4 ASME TDP-2, “Prevention of Water Damage to Steam Turbines Used for Electric Power Generation: Nuclear-Fueled Plants,” American Society of Mechanical Engineers, 2012 (R2017).

10.2-5 ASME BPVC, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” American Society of Mechanical Engineers, 2021.

10.2-6 ASME B31.1, “Power Piping,” American Society of Mechanical Engineers, 2022.

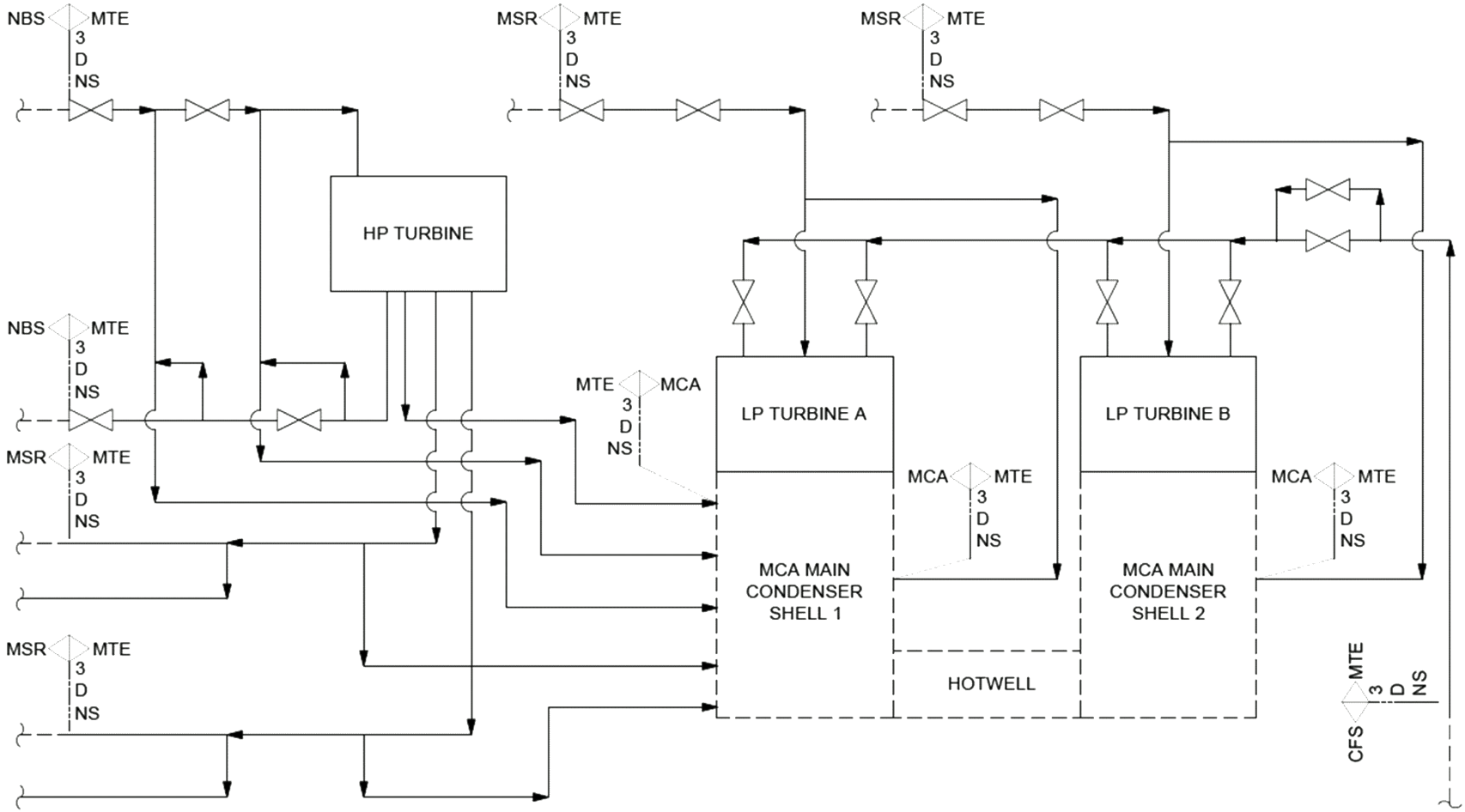


Figure 10.2-1 Main Turbine Equipment Simplified Diagram

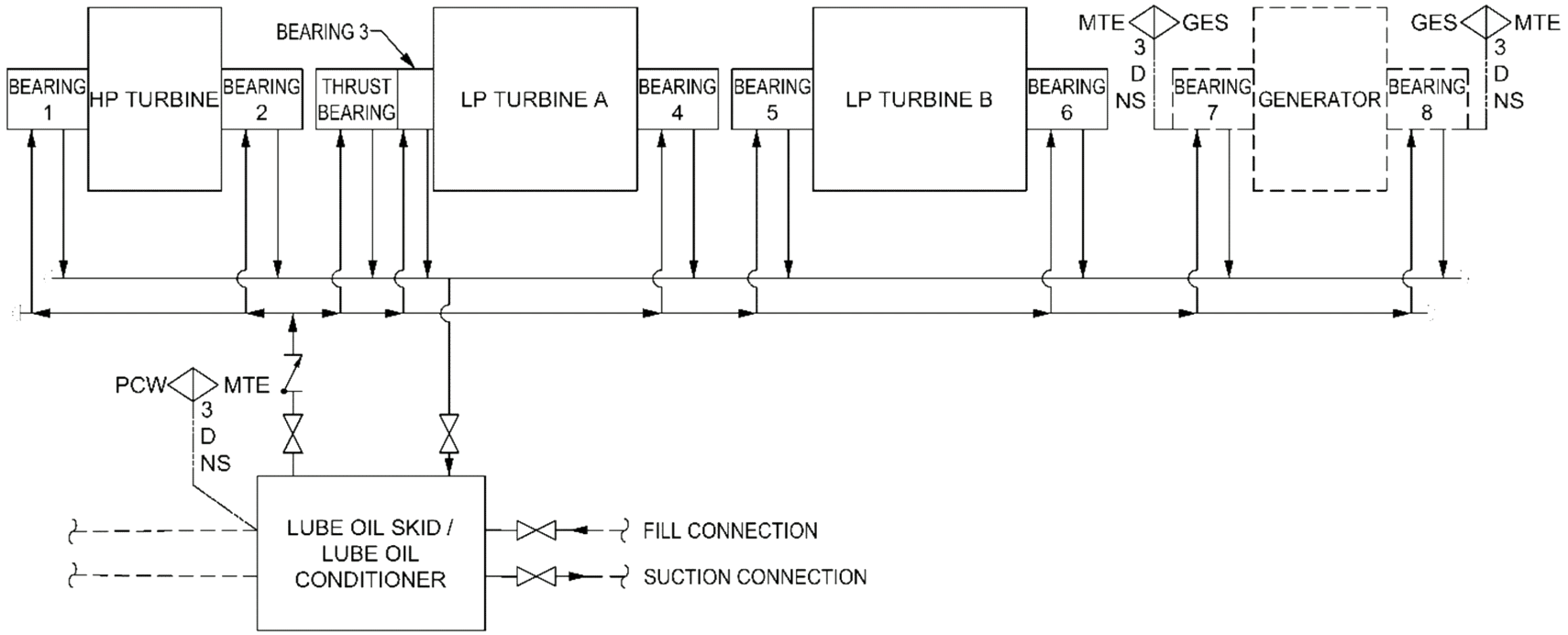


Figure 10.2-2 Turbine Lube Oil System Simplified Diagram

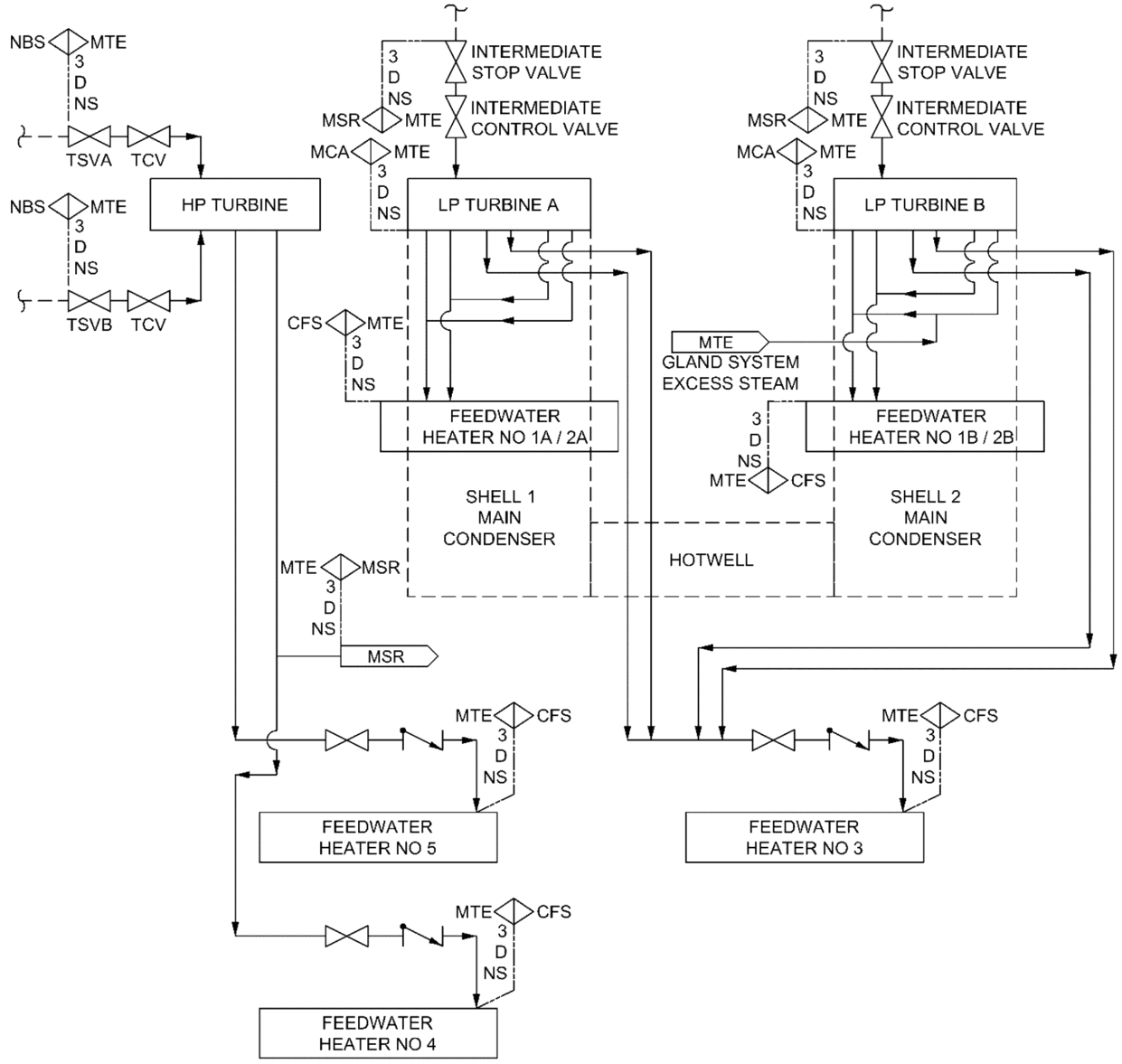


Figure 10.2-3 Extraction Steam Subsystem Simplified Diagram

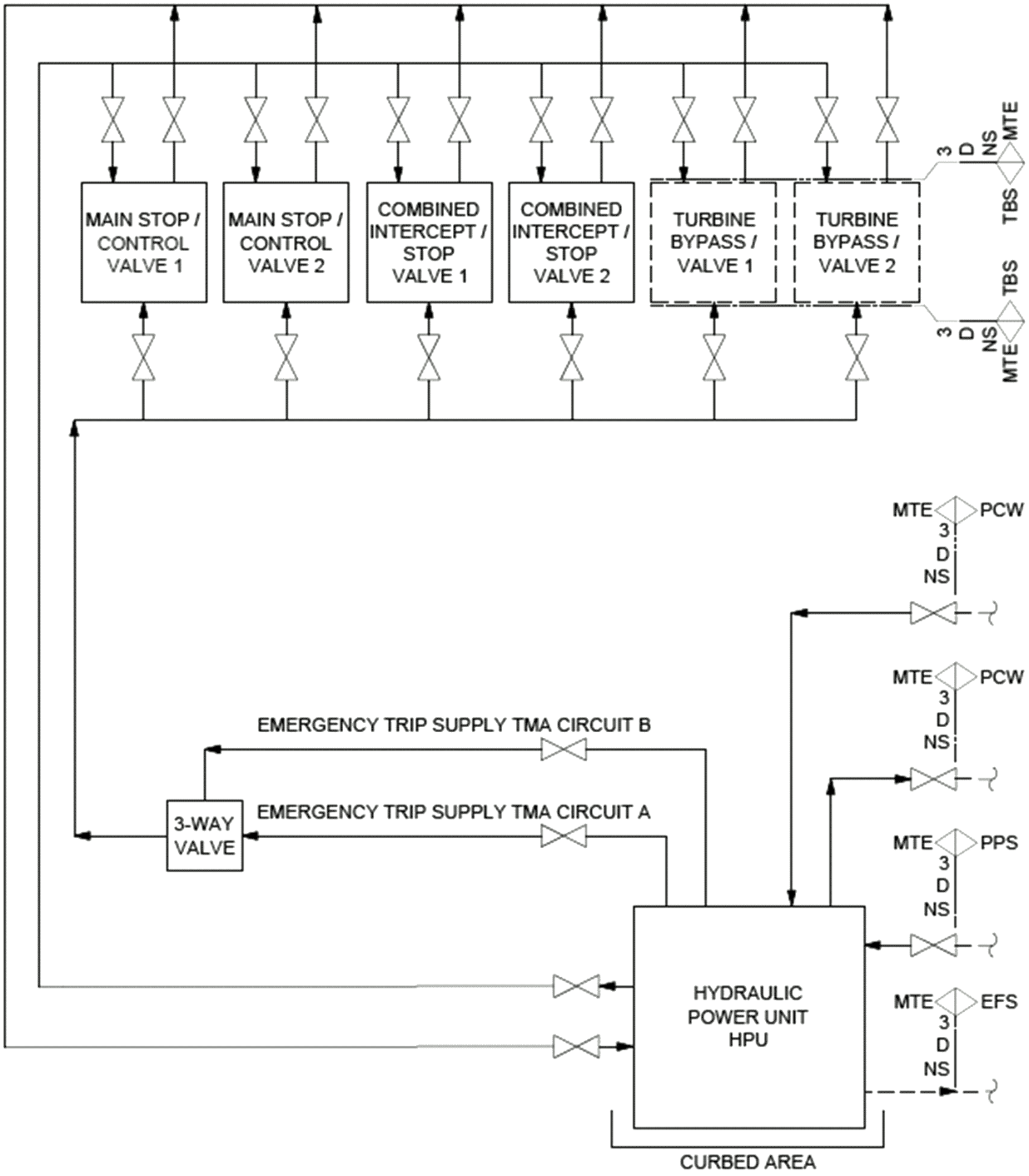


Figure 10.2-4 Electro-Hydraulic Control Subsystem Simplified Diagram

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 Classifications, 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 MSCIVs 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 diagram showing the 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, and the No. 6 FWH. 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 Nuclear Regulatory Report (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,” Subsection NF, “Supports,” (Reference 10.3-2) requirements.

The MS piping in the TB is designed to 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.

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, 2021.

10.3-2 ASME BPVC, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” Subsection NF, “Supports,” American Society of Mechanical Engineers, 2021.

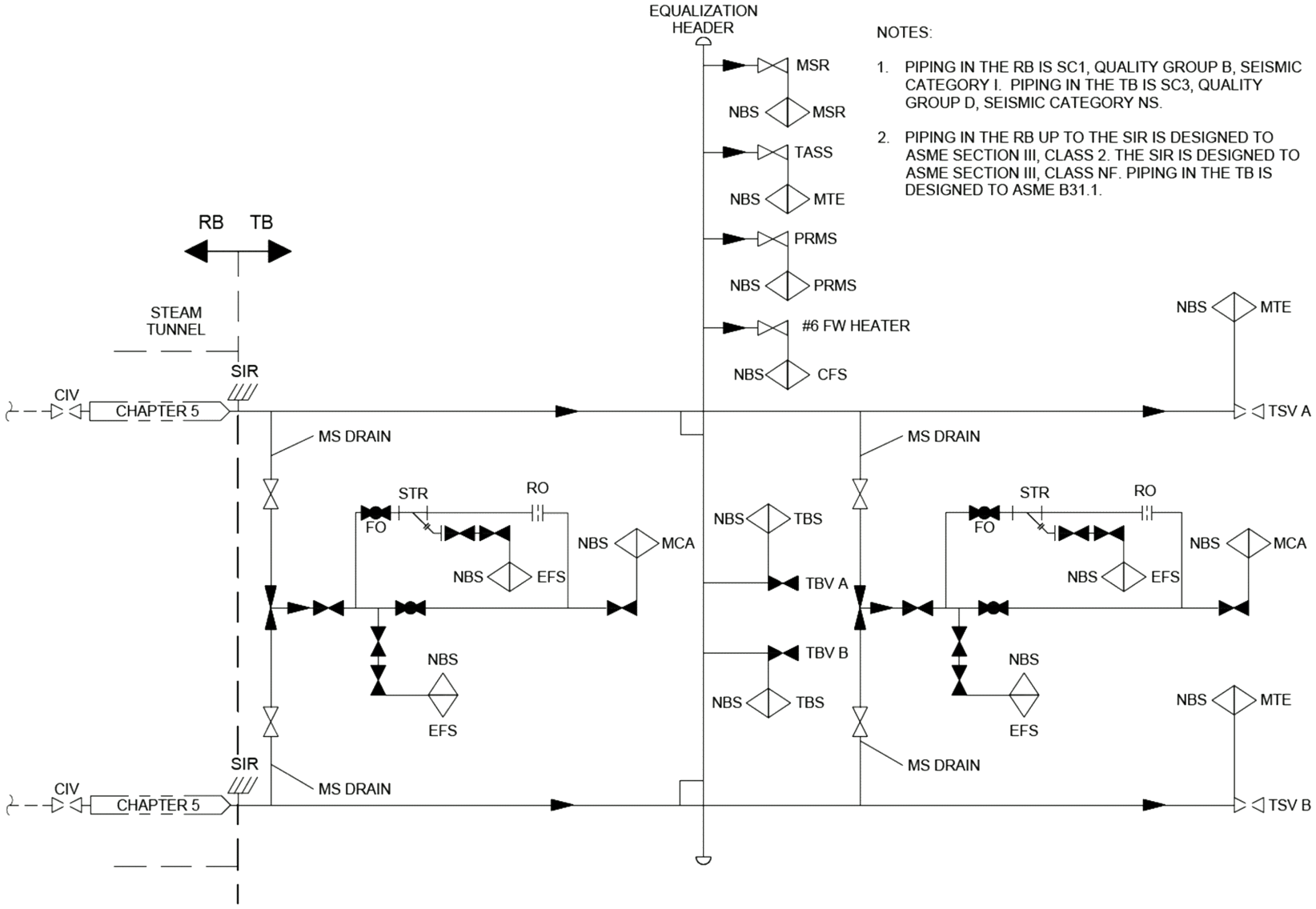


Figure 10.3-1 Main Steam Subsystem Simplified Diagram (RB & TB Portion)

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 Classifications, 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. 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 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 Classifications, 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 diagram of 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 to minimize the potential of water damage to the turbines.

The MSR is protected by a relief valve, per ASME BPVC, Section VIII, UG-150.

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 Classifications, 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 Classifications, 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 sub-systems: 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 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 MCA to the environment through the Normal Heat Sink (NHS) System
- The NHS System has provisions to utilize the CWS piping to provide once-through cooling flow to a Plant Cooling Water System heat exchanger (see Subsection 9.2.9)

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

The CW Supply, shown in Figure 10.4-5, provides cooling water to the main condenser and transfers heat from the condenser to the environment through the NHS System. The NHS System is described in Subsection 9.2.9. The CW Supply uses four 25% pumps to provide cooling water to the MCA. The CW Supply and SW Subsystem pumps are located in individual pump bays in the cooling tower basin of the NHS System.

The CW Supply pumps draw cooling water from the cooling tower basin and discharge through a common underground header. In the TB, the piping is routed above grade to the tube side of the main condenser to provide the cooling water flow for condensing of the main turbine exhaust steam. Cooling water exiting the main condenser is routed to the cooling tower header underground and then aboveground where it interfaces with the NHS System at the top of the cooling tower curb. The NHS System provides the cooling tower risers and spray circuits for this line.

A condenser tube cleaning system is provided to prevent fouling, scaling, and corrosion of the condenser tubes.

The CW Supply includes water box vents to help fill the condenser water boxes during startup and remove accumulated air and other gases from the water boxes during normal operation. The water box drain pump and associated piping provides the means to drain the condenser water boxes during outages or maintenance.

A fill line connecting the SW Subsystem pumps to the CW Supply piping 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 main CW Supply pumps.

Material and equipment selection for the CW Supply is based on the range of environmental conditions that exist at the plant location.

Strainers, filters, and chemical treatment, included within the NHS System and Water, Gas, and Chemical Pads System, are used to mitigate small plant aquatic life and other debris in the CW Supply. Chemicals are also used to reduce biological growth, reduce alkalinity, and inhibit scale formation. The water treatment chemicals added to the CW Supply allow the water to meet or exceed the water quality requirements specified in the BWRX-300 water quality specifications.

Screens are included between the cooling tower basin and the CW Supply and SW Subsystem pumps. The screens are sized to remove debris that could be introduced into the NHS System cooling tower basin. This reduces the potential for damage to the pumps or condenser as well as the load on the SW Subsystem strainers.

The CW Supply continues to provide Safety Category 3 functions during off-normal conditions. It is designed to maintain the reactor operation during a turbine trip up to a 25% load rejection, whereby 25% of the main steam bypasses 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. The CW Supply provides sufficient circulating water to the main condenser to support full power operation throughout the year. The CW Supply, together with the NHS System, provides sufficient circulating water to the main condenser to support rated TG power.

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:

Qualification of pumps 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.

Below ground piping meets applicable American Water Works Association standards based on pipe material and size.

10.4.5.4 Tests and Inspections

Equipment and components of the CW Supply are designed for easy inspection and maintenance during plant operation. Routine testing of the CW Supply is conducted in accordance with normal power plant requirements for demonstrating system and component operability and integrity.

10.4.5.5 Instrumentation and Controls

The CW Supply is operated and monitored from the MCR. Instrumentation for major system parameters provides indication and alarms in the MCR. 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 Classifications, 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 specifications.

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 CFD 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 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 RPV at the nominal conditions derived in the plant heat balance
- Modulate the flow of FW to the RPV 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
- 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 Classifications, 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. Figure 10.4-7 and Figure 10.4-8 show the condensate portion of the CFS and the FW portion of the CFS, respectively. Figure 10.4-9 and Figure 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, and HEI-2622, “Standards for Closed Feedwater Heaters” / HEI 2629, “Standards for Steam Surface Condensers” (Reference 10.4-10 and Reference 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-R4),” (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

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10.4-15 ASME B16.34, “Valves Flanged, Threaded and Welding End,” American Society of Mechanical Engineers, 2020.

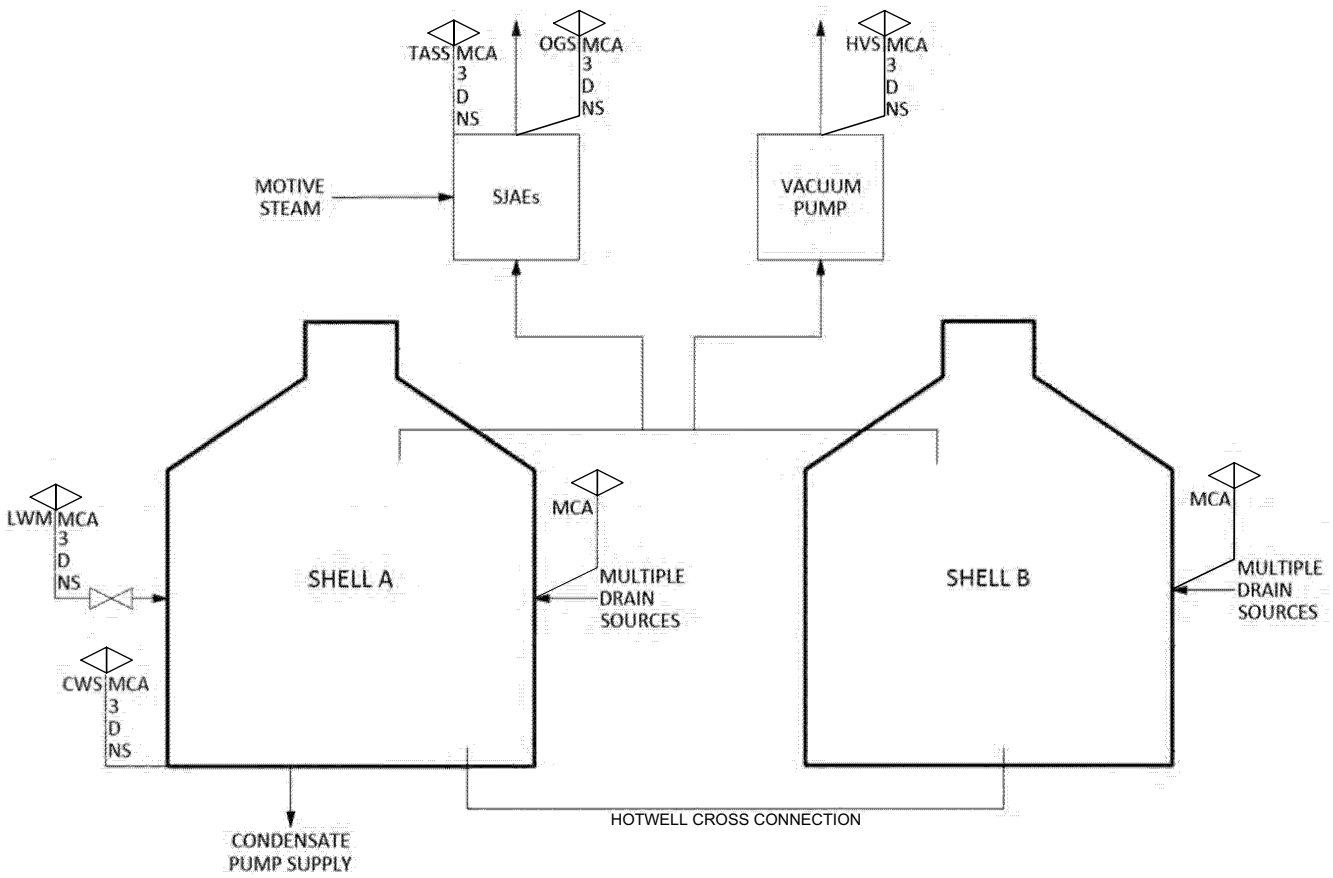


Figure 10.4-1 Main Condenser and Auxiliaries Simplified Diagram

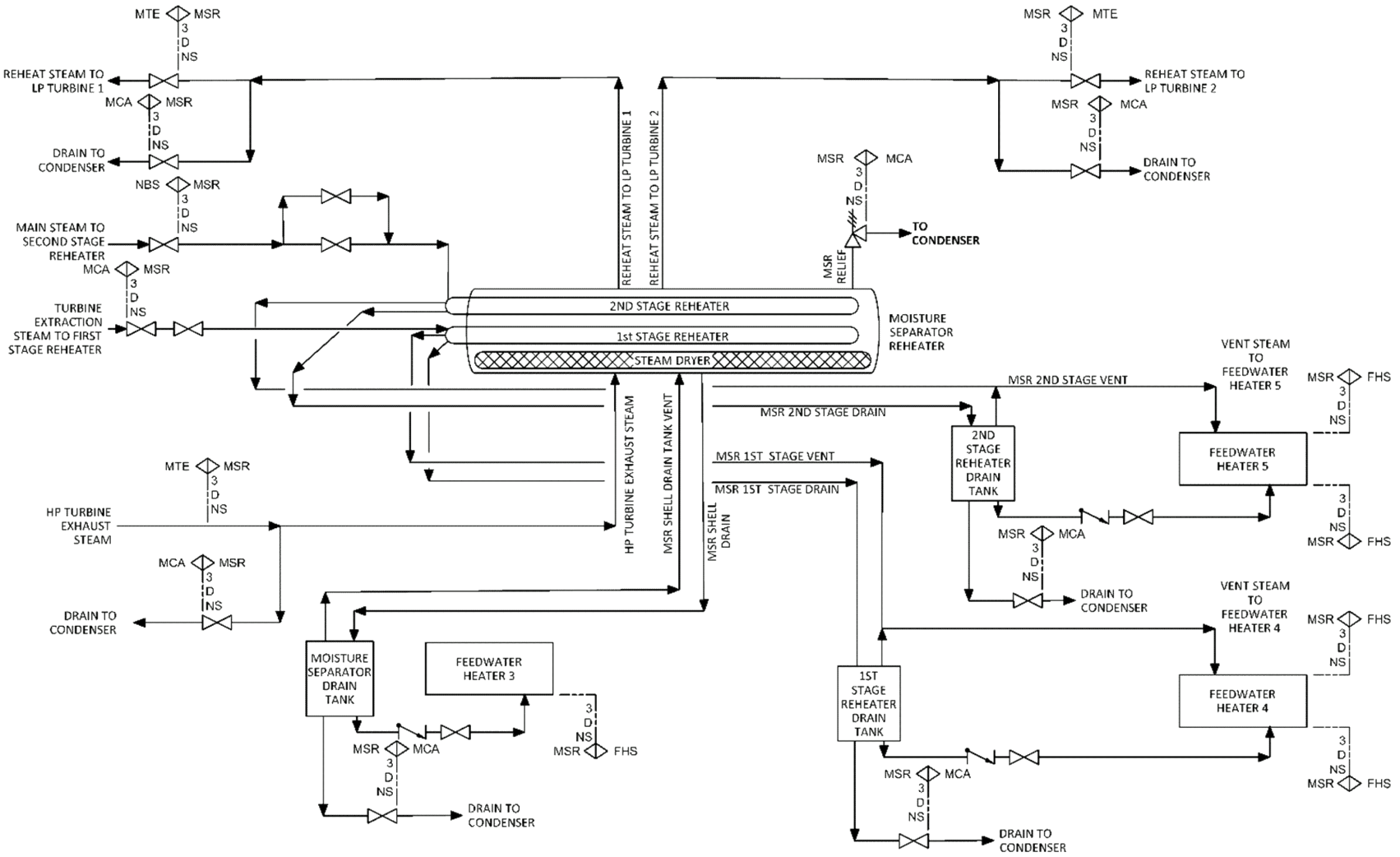


Figure 10.4-2 Moisture Separator Reheater System Simplified Diagram

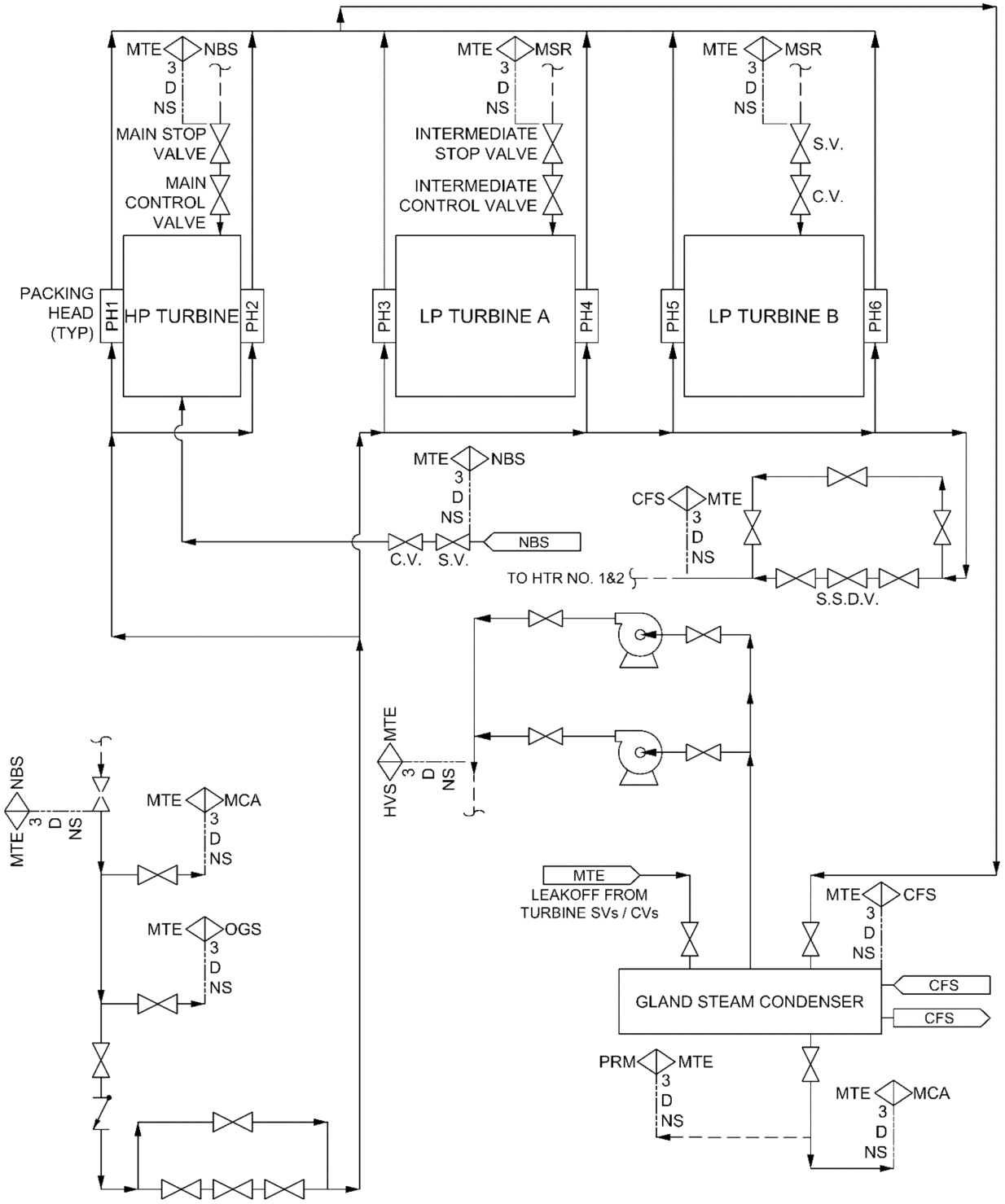


Figure 10.4-3 Turbine Gland Seal System Simplified Diagram

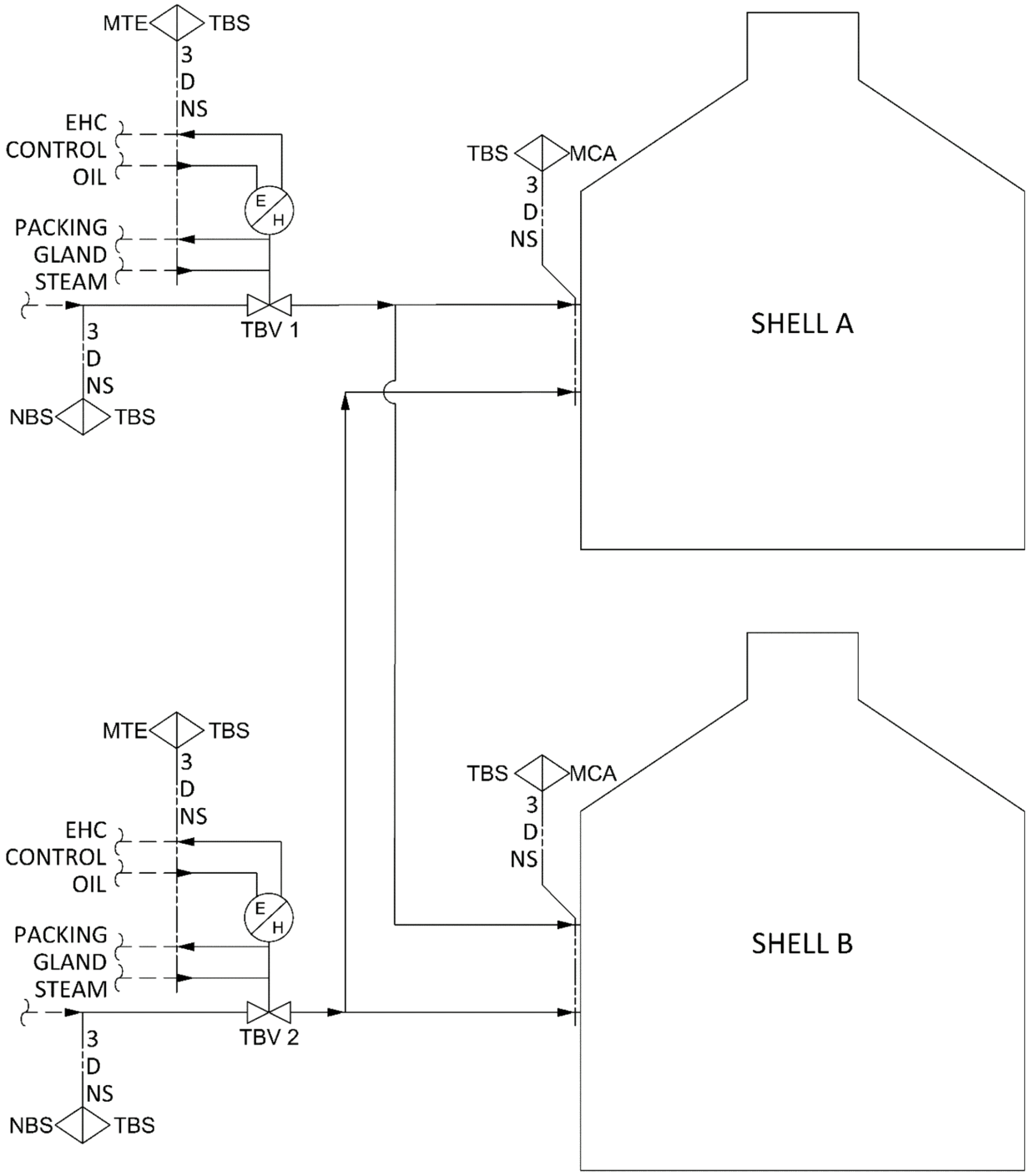


Figure 10.4-4 Turbine Bypass System Simplified Diagram

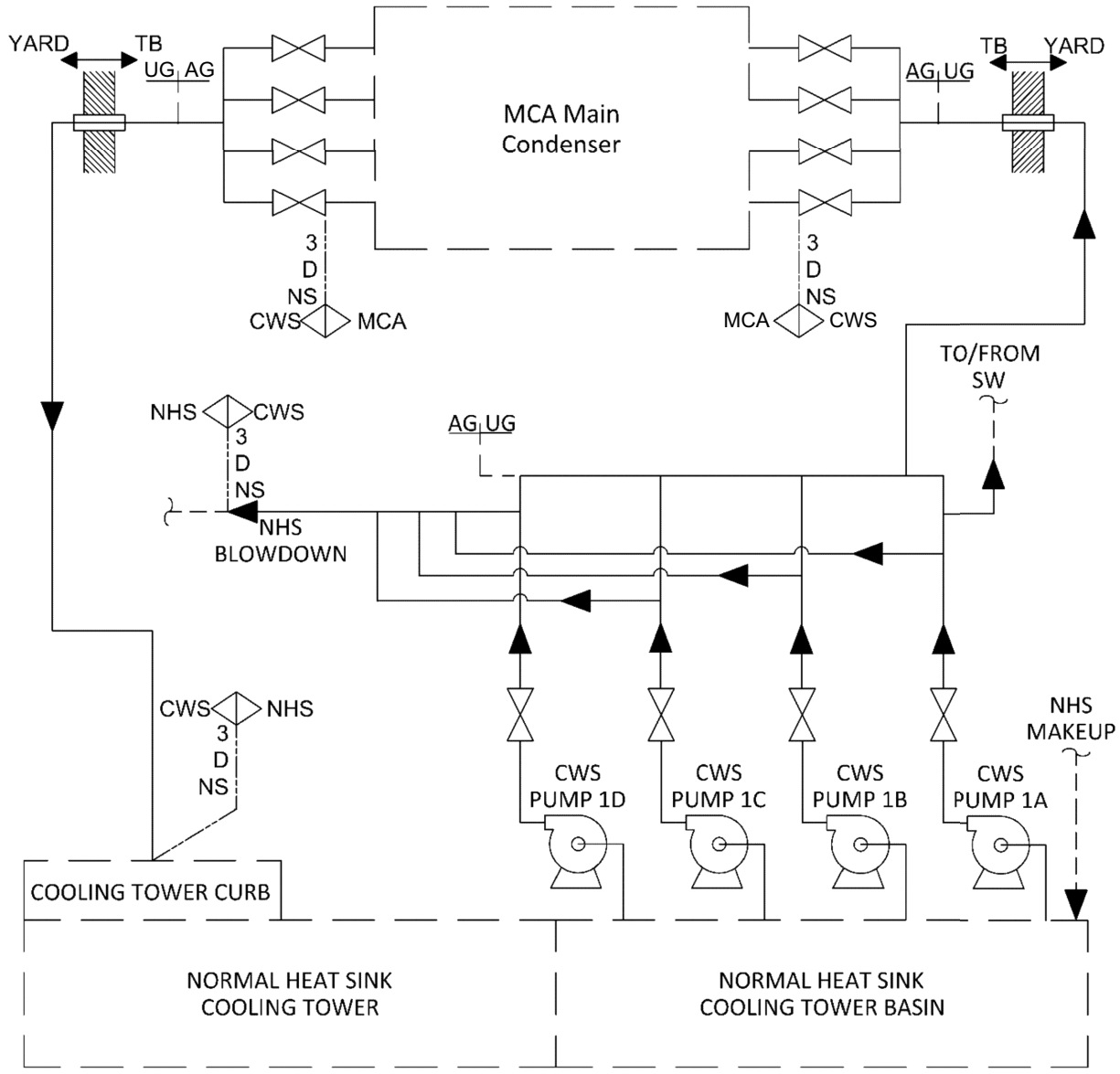
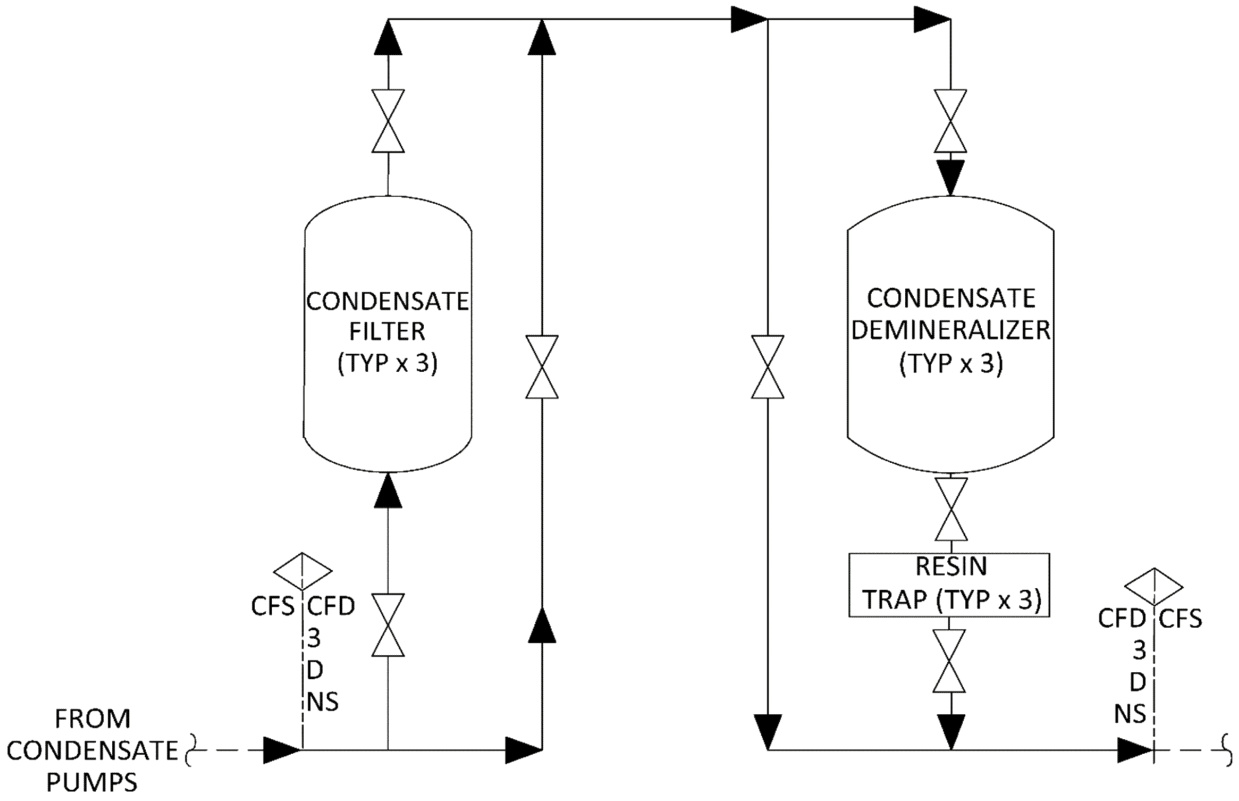


Figure 10.4-5 Circulating Water Supply Simplified Diagram



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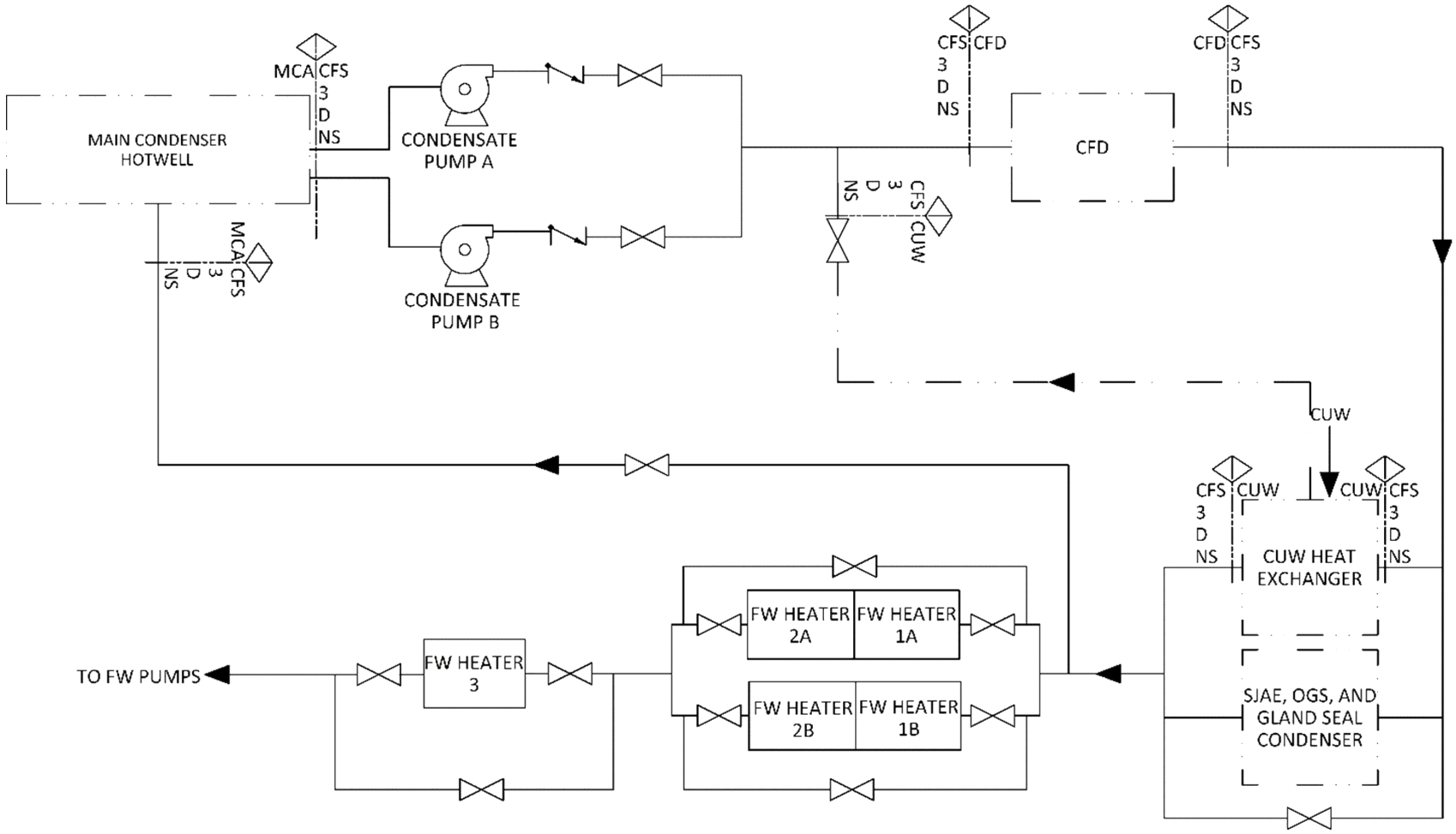
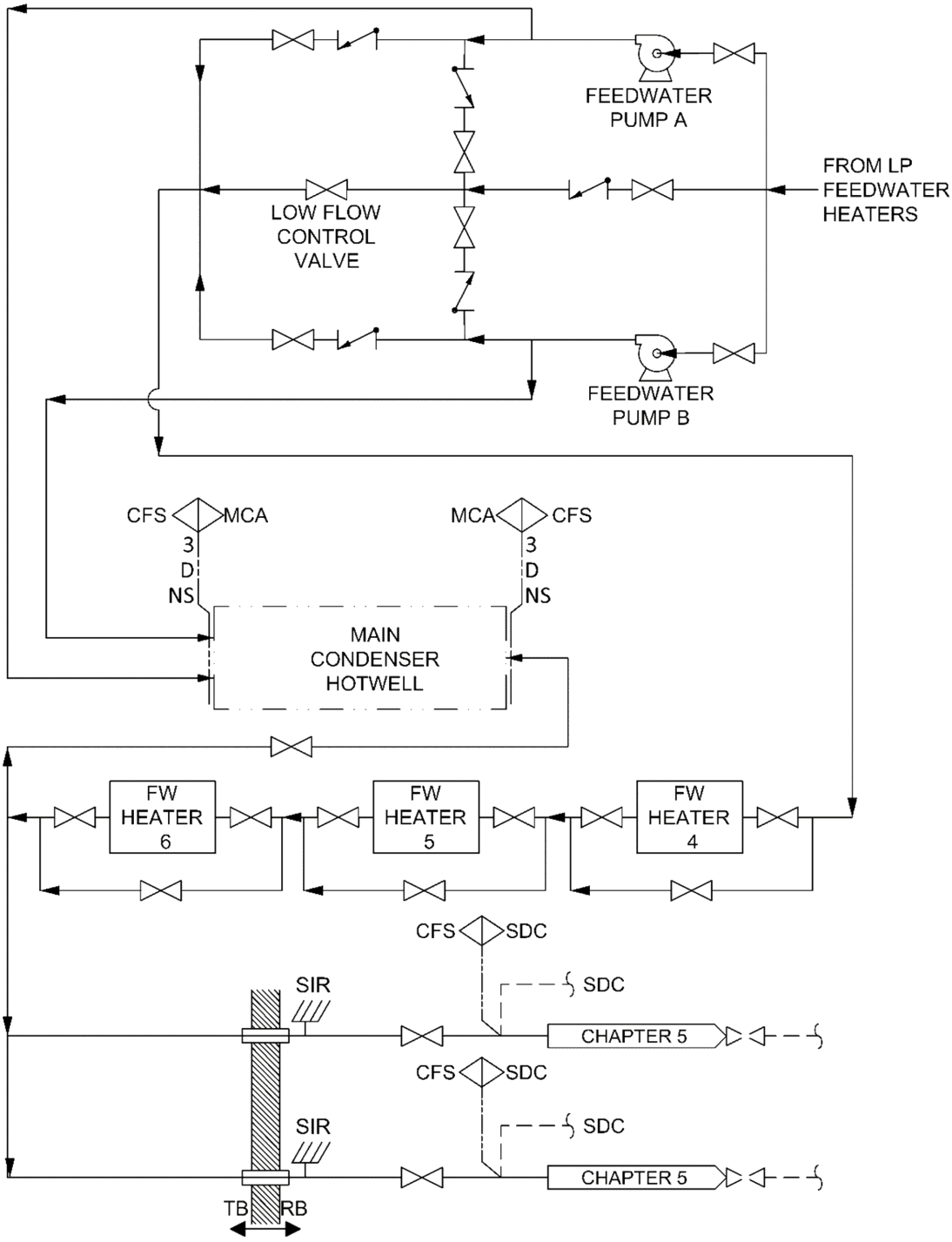


Figure 10.4-7 CFS Condensate Portion Simplified Diagram



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

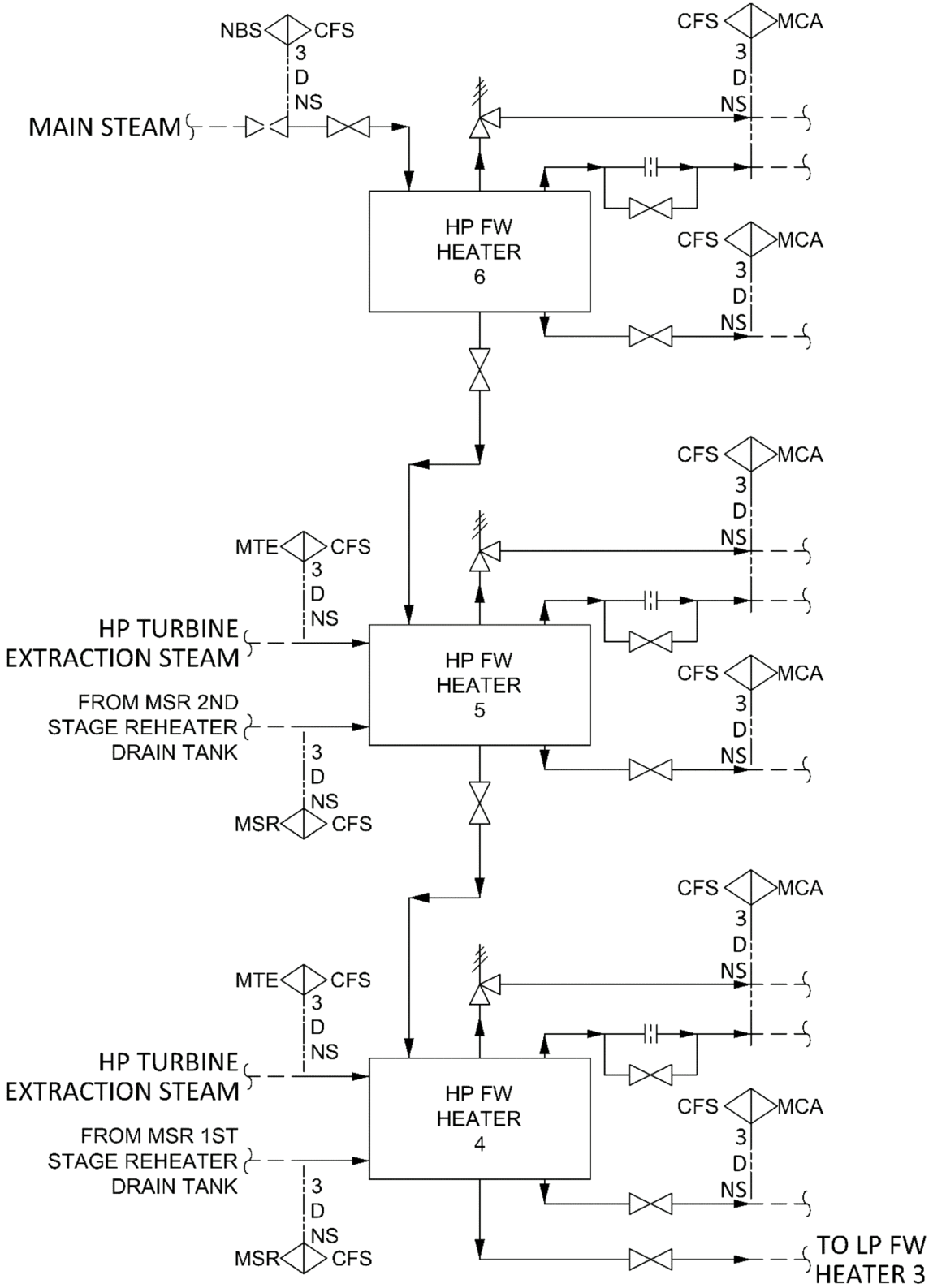
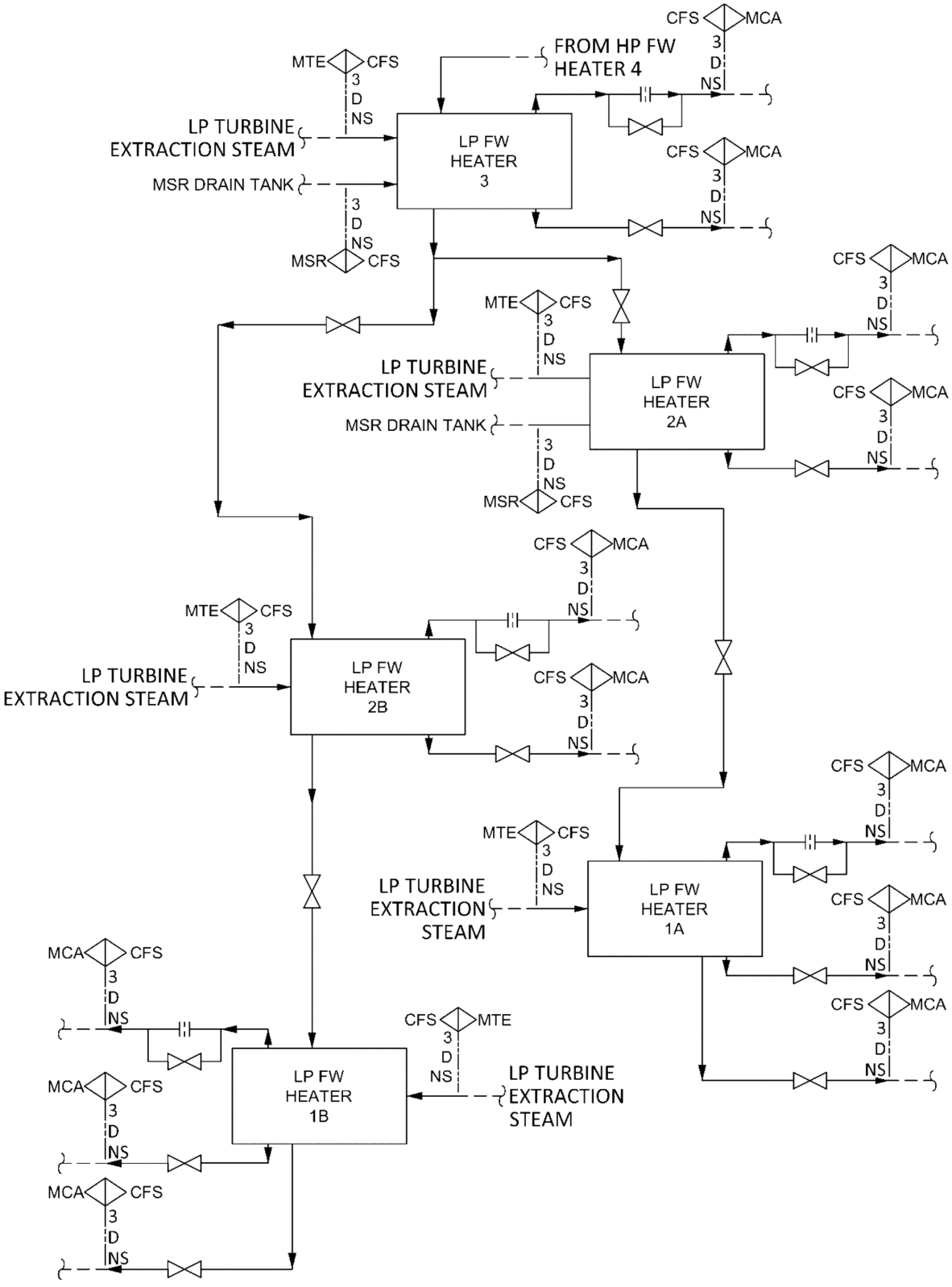


Figure 10.4-9 CFS HP Heaters Vents and Drains Simplified Diagram



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LIST OF ACRONYMS AND ABBREVIATIONS

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
COL	Combined License
CR	Control Room
CRE	Control Room Envelope
CRN	Clinch River Nuclear
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
ESP	Early Site Permit
FPC	Fuel Pool Cooling and Cleanup System
GDC	General Design Criteria
GNF	Global Nuclear Fuel
GSC	Gland Seal Condenser
GVH	GE Vernova Hitachi Nuclear Energy

LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
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
MSL	Main Steam Line
NHS	Normal Heat Sink
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
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
SMR	Small Modular Reactor
SRP	Standard Review Plan
SSAR	Site Safety Analysis Report
SSCs	Structures, Systems, and Components
SWM	Solid Waste Management System
TB	Turbine Building
TGFU	Toxic Gas Filtration Unit
TMR	Triple Modular Redundant
USNRC	U.S. Nuclear Regulatory Commission

LIST OF EFFECTIVE SECTIONS

CHAPTER 11 RADIOACTIVE WASTE MANAGEMENT

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CHAPTER 11 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 source term is a conservative design basis model that utilizes an established GE Vernova Hitachi Nuclear Energy (GVH) fuel clad defect model.

The realistic model, which is based on available measured nuclide concentrations during normal operation at operating Boiling Water Reactors (BWR), is determined from the American National Standards Institute (ANSI)/American Nuclear Society (ANS), ANSI/ANS-18.1, “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; it 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, 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 System (CFD). 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 ANSI/ANS-18.1. 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 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, 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 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 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/ANS-18.1 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/s after a 30-minute decay
- I-131 fuel leak rate of 250 µCi/s

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 by 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 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. 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/s 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 NUREG-0800, Standard Review Plan (SRP), 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/s 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 GVH fuel clad defect model I-131 release rate of 250 µCi/s. 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 BWR 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 11.1-6 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 and the BWRX-300 operating parameters in Table 11.1-4. 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 employs On-line NobleChem™ technology along with the HWC which allows 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 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 vessel 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/s. 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 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:

$$C = \frac{s}{m(\lambda + R)} \qquad \text{Equation 11.1-1}$$

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:

$$Adjustment\ Factor = \frac{P \cdot WP_n \cdot (R_{ni} + \lambda)}{WP \cdot P_n \cdot (R_i + \lambda)} \qquad \text{Equation 11.1-2}$$

where:

R = removal rates of the radionuclide

The subscript “n” refers to the BWR 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 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

WP = weight of water in the reactor vessel

However, the BWRX-300 coolant cleanup system configuration differs from the ANSI/ANS-18.1 BWR Reference Plant model. 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 vessel and sends it to the CFD which purifies the condensate, as required, to maintain 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 CUW 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 maintain airborne radiation exposures to plant personnel and releases to the environment 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 National Standards Institute/American Nuclear Society, 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	**BWR 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/s (lb/hr)	1.60E+01 (1.30E+05)	5.07E+00 (4.02E+04)
Steam Flow Rate, kg/s (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 Operation 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

* 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

* 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

* 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

* 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

* Normal operation concentrations are the same as the design basis concentrations.

11.2 Liquid Waste Management System

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 (CIV), which are not described in this section. The CIVs, their safety functions, and testing are described in Subsection 6.2.4. 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 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 description of this subsystem is provided in Subsection 9.2.10.

11.2.1 Design Bases

The Waste Collection and Filtering Subsystem and the Waste Sampling Subsystem perform the following Safety Category functions:

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 plants 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 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

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. The subsystem is 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 1.143, “Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants.” 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) System 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, “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 will consist of 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 BPVC, 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 LWM 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 System where it mixes with blowdown flow prior to discharge. This interface point between the blowdown and the LWM is upstream of the plant outfall to the Clinch River arm of the Watts Bar Reservoir. The LWM interfaces with other systems including:

- Process Radiation and Environmental Monitoring System (PREMS)
- SDC
- CFS
- 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. Clean, oily, and contaminated wastes are collected separately via the EFS 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, Control releases of radioactive materials to the environment, 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 (AOO), as described in Subsection 11.2.4.

GDC 61, Fuel storage and handling and radioactivity control, 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.

The Clinch River Nuclear (CRN) Site location, characteristics, and site-related design parameters description provided in Section 2.0 provides sufficient detail to support a safety assessment.

Evaluation Results

Section 2.0 includes a Plant Parameter Envelope (PPE) Evaluation that provides criteria for demonstrating compliance with Combined License (COL) Action Item 2.4-4 of the CRN Early Site Permit, ESP-006 (Reference 11.2-3). The PPE Evaluation for the accidental liquid release was reperformed using the bounding source term for the BWRX-300 design presented in Table 2.0-5R. The updated BWRX-300 accidental liquid release evaluation and results are presented in Section 2.4.13.

11.2.4 Radioactive Effluent Releases

Methodology

The analysis is performed with the GALE-BWR methodology (see Appendix 3L) which implements the assumptions outlined in RG 1.112, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Nuclear Power Reactors.” The GALE-BWR methodology is described in 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.

Section 2.0 includes a PPE Evaluation that provides criteria for demonstrating compliance with COL Action Item 11-1 of CRN Early Site Permit, ESP-006 (Reference 11.2-3). A PPE Evaluation for the total annual liquid effluent releases provided in Table 11.2-3 is presented in Table 2.0-6R.

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.

Dose Evaluation Results

Clinch River Nuclear Site Early Site Permit Application Part 2, Site Safety Analysis Report (ESPA SSAR) Subsection 11.2.3.1 describes the exposure pathways by which radiation and radioactive effluents can be transmitted from the proposed nuclear units to individuals living near the site. ESPA SSAR Subsection 11.2.3.2 provides an estimate of the maximum doses to the public and evaluates the effects of these doses by comparing them to applicable regulatory limits.

As shown in ESPA SSAR Table 11.2-4, the projected CRN Site liquid radioactive effluent concentrations in the Clinch River arm of the Watts Bar Reservoir are within the Effluent Concentration Limits of 10 CFR 20, Appendix B, Table 2, Column 2.

The projected annual doses to the Maximally Exposed Individual from liquid radioactive effluents from each reactor and for the site are shown in ESPA SSAR Table 11.2-5 and Table 11.2-6.

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, as shown in ESPA SSAR Table 11.2-7.

The projected total site doses due to liquid and gaseous radioactive effluents are within the environmental standards of 40 CFR 190, as shown in ESPA SSAR Table 11.2-8.

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.

Projected annual collective doses to the population within 50 miles of the CRN Site are shown in ESPA SSAR Table 11.2-9.

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 maintain 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 B31.3, "Process Piping," American Society of Mechanical Engineers, 2022.

11.2-2 ASME B31.1, "Power Piping," American Society of Mechanical Engineers, 2022.

11.2-3 "Clinch River Nuclear Site Early Site Permit, ESP-006," U.S. Nuclear Regulatory Commission, December 2019 (ML19352D868) and Final Safety Evaluation Report, June 2019 (ML19162A157).

Table 11.2-1 Equipment Codes (from RG 1.143, Table 1)

Component	Design and Construction	Materials[1]	Welding	Inspection and Testing
Piping and Valves	ASME B31.3 [2,3]	ASME-Section II[4]	ASME, Section IX	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. RW-IIa and RW-IIb piping systems are to be designed as category "M" systems.
3. 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, Subsection NCA are not required. Therefore, these components are not classified as ASME Code Class 3.
6. Flexible Hoses should only be used in conjunction with Mobile Radwaste Processing Systems.
7. Alternative requirements such as those in ASME BPVC, 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

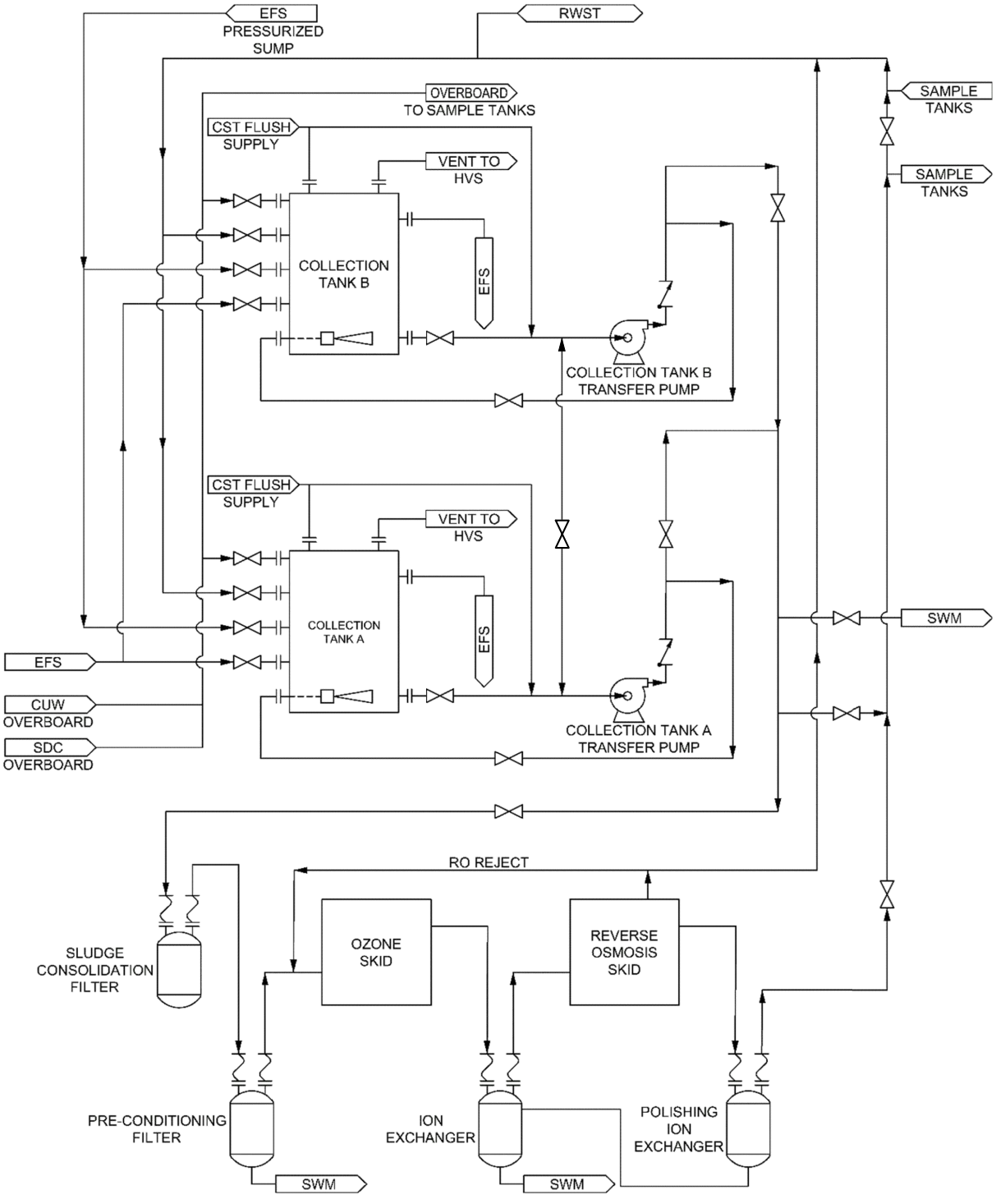


Figure 11.2-1 Waste Collection and Filtering Subsystem Simplified Diagram

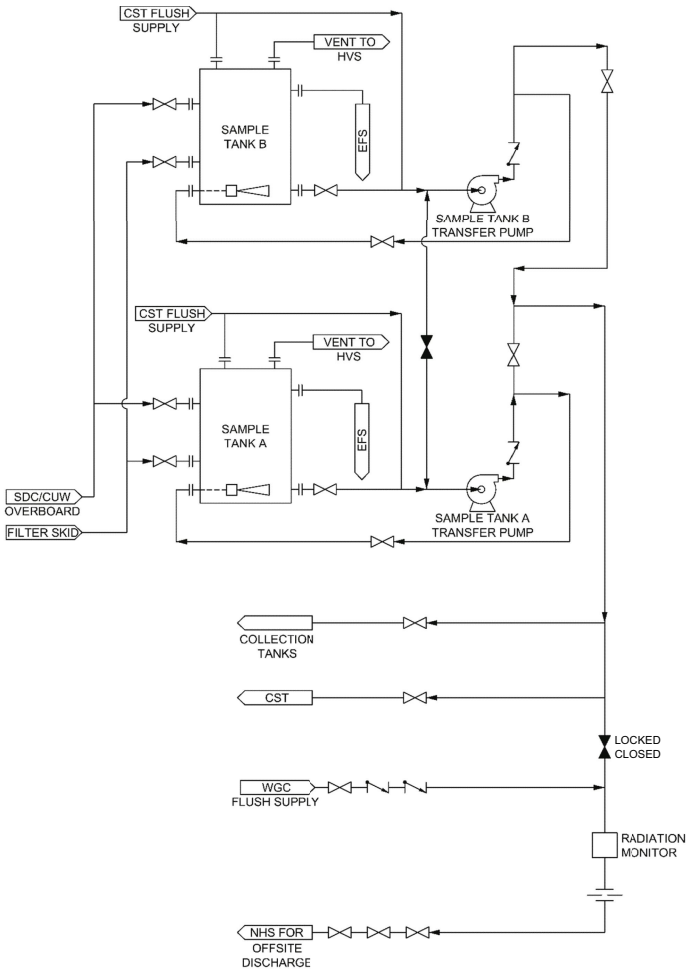


Figure 11.2-2 Waste Sampling Subsystem Simplified Diagram

11.3 Gaseous Waste Management System

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 (SJAE), 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

- 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 (MCA) 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 such 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 MCA at the exit of the SJAEs within the TB. 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 OGS 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 is condensed from a saturated vapor state to a saturated liquid state. The condensed Main Steam 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 provided such that only 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) (Subsection 9.2.7) enters the tube side of the heat exchanger. The chilled water is used to cool the offgas effluent and is controlled by CWE valves. Chilled water passes through the Cooler Condenser and returns to

the CWE. 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 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. 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 (HWC) 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 OGS preparation mode occurs when the plant is in startup operations. 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 includes:

- Catalytic Recombiner section heater: activated to produce a sufficiently high temperature in the Recombiner section prior to startup operations
- Charcoal Adsorber Vault ventilation: activated to achieve desired atmospheric conditions in the Charcoal Adsorber Vault prior to power operations

11.3.2.3.3 System Startup

The OGS startup mode occurs between the time that the SJAEs are placed into service as the reactor power rises (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 this 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 power operations, radiolytic gas production varies linearly with thermal power.

During the beginning of the OGS normal operation mode, 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 and then 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. During normal operation

containment air is circulated 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, potentially radioactive subsystem's exhaust to a common Plant Vent Stack during normal operation. 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

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, Control releases of radioactive materials to the environment, 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, Fuel storage and handling and radioactivity control, as it relates to the ability of the OGS design to provide 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 Subsection 11.3.2 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 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, Sections 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

The radiological consequences are determined for an OGS accident as specified in SRP 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. Even with the extended release period, the dose results are well within the BTP 11-5 acceptance criterion.

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 Main Control Room

(MCR) 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. The dose results are within the limiting 25 mSv (2.5 Rem) whole body dose for an OGS designed to withstand explosions and earthquakes, per BTP 11-5 and RG 1.143.

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. 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 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.

Section 2.0 includes a PPE Evaluation that provides criteria for demonstrating compliance with COL Action Item 11-1 of CRN Early Site Permit, ESP-006 (Reference 11.3-2). A PPE Evaluation for the total airborne effluent releases provided in Table 11.3-5 is presented in Table 2.0-4R.

Dose Evaluation Results

ESPA SSAR Subsection 11.3.3.1 describes the exposure pathways by which radiation and radioactive effluents can be transmitted from the proposed nuclear units to individuals living near the site. ESPA SSAR Subsection 11.3.3.2 provides an estimate of the maximum doses to the public and evaluates the effects of these doses by comparing them to applicable regulatory limits.

As shown in ESPA SSAR Table 11.3-3, projected gaseous radioactive effluent concentrations at the CRN Site Exclusion Area Boundary (EAB) are within the Effluent Concentration Limits of 10 CFR 20, Appendix B, Table 2, Column 1.

The projected annual doses to the Maximally Exposed Individual from gaseous radioactive effluents from each reactor and from the site are shown in ESPA SSAR Tables 11.3-4 and 11.3-5, respectively.

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, as shown in ESPA SSAR Table 11.2-7.

The projected total site doses due to liquid and gaseous radioactive effluents are within the environmental standards of 40 CFR 190, as shown in ESPA SSAR Table 11.2-8.

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.

Projected annual collective doses to the population within 50 miles of the CRN Site are shown in ESPA SSAR Table 11.2-9.

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

OGS instrumentation requirements are described as follows:

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 and HWC. 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 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.

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 Institute/American Nuclear Society, 2017.

11.3-2 “Clinch River Nuclear Site Early Site Permit, ESP-006,” U.S. Nuclear Regulatory Commission, December 2019 (ML19352D868) and Final Safety Evaluation Report, June 2019 (ML19162A157).

Table 11.3-1 Offgas Failure Environmental Releases

Radionuclide	Release Rates		Environmental Releases (2 hours)		Environmental Releases (74 hours)	
	µCi/s	MBq/s	µCi	MBq	µCi	MBq
Kr-83m	5.7E+00	2.1E-01	4.1E+04	1.5E+03	1.5E+06	5.6E+04
Kr-85m	2.4E+01	8.8E-01	1.7E+05	6.3E+03	6.3E+06	2.3E+05
Kr-85	2.0E+02	7.4E+00	1.4E+06	5.3E+04	5.3E+07	2.0E+06
Kr-87	2.5E-02	9.2E-04	1.8E+02	6.6E+00	6.6E+03	2.4E+02
Kr-88	1.2E+01	4.6E-01	9.0E+04	3.3E+03	3.3E+06	1.2E+05
Kr-89	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Xe-131m	6.0E+01	2.2E+00	4.3E+05	1.6E+04	1.6E+07	5.9E+05
Xe-133m	4.9E-01	1.8E-02	3.5E+03	1.3E+02	1.3E+05	4.8E+03
Xe-133	1.6E+02	6.1E+00	1.2E+06	4.4E+04	4.4E+07	1.6E+06
Xe-135m	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Xe-135	8.3E-10	3.1E-11	5.9E-06	2.2E-07	2.2E-04	8.1E-06
Xe-137	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
Xe-138	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00

Table 11.3-2 Offgas System Failure Dose Results

Release Duration	Whole Body Dose	BTP 11-5 Dose Limit
2 hours	2.3E-03 mSv (2.3E-04 rem)	25mSv (2.5 rem)
74 hours	8.6E-02 mSv (8.6E-03 rem)	25mSv (2.5 rem)

Table 11.3-3 Offgas System Failure Accident Parameters

I. Data and Assumptions Used to Estimate Source Terms	
a. Power Level	870 MWt
b. Offgas Release Rate	87,000 µCi/s (3219 MBq/s)
c. Duration of Release	2 hours/74 hours
II. Dispersion and Dose Data	
d. EAB X/Q	4.96E-03 s/m^3
e. Dose Methodology	BTP 11-5
f. Dose Conversion Assumptions	BTP 11-5
g. Activity Releases	Table 11.3-1
h. Dose Results	Table 11.3-2

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
Normal Operation Noble Gas Source Term	Table 11.1-2

Table 11.3-5 BWRX-300 Annual Airborne Releases Including AOOs (Ci/yr)
(Sheet 1 of 3)

Nuclide	Reactor Building	Service Building	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
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

Table 11.3-5 BWRX-300 Annual Airborne Releases Including AOOs (Ci/yr)
(Sheet 2 of 3)

Nuclide	Reactor Building	Service Building	Turbine Building	Radwaste Building	Mechanical Vacuum Pump	Turbine Seal	Offgas System	Containment	Total[1]
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
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

Table 11.3-5 BWRX-300 Annual Airborne Releases Including AOOs (Ci/yr)
(Sheet 3 of 3)

Nuclide	Reactor Building	Service Building	Turbine Building	Radwaste Building	Mechanical Vacuum Pump	Turbine Seal	Offgas System	Containment	Total[1]
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.

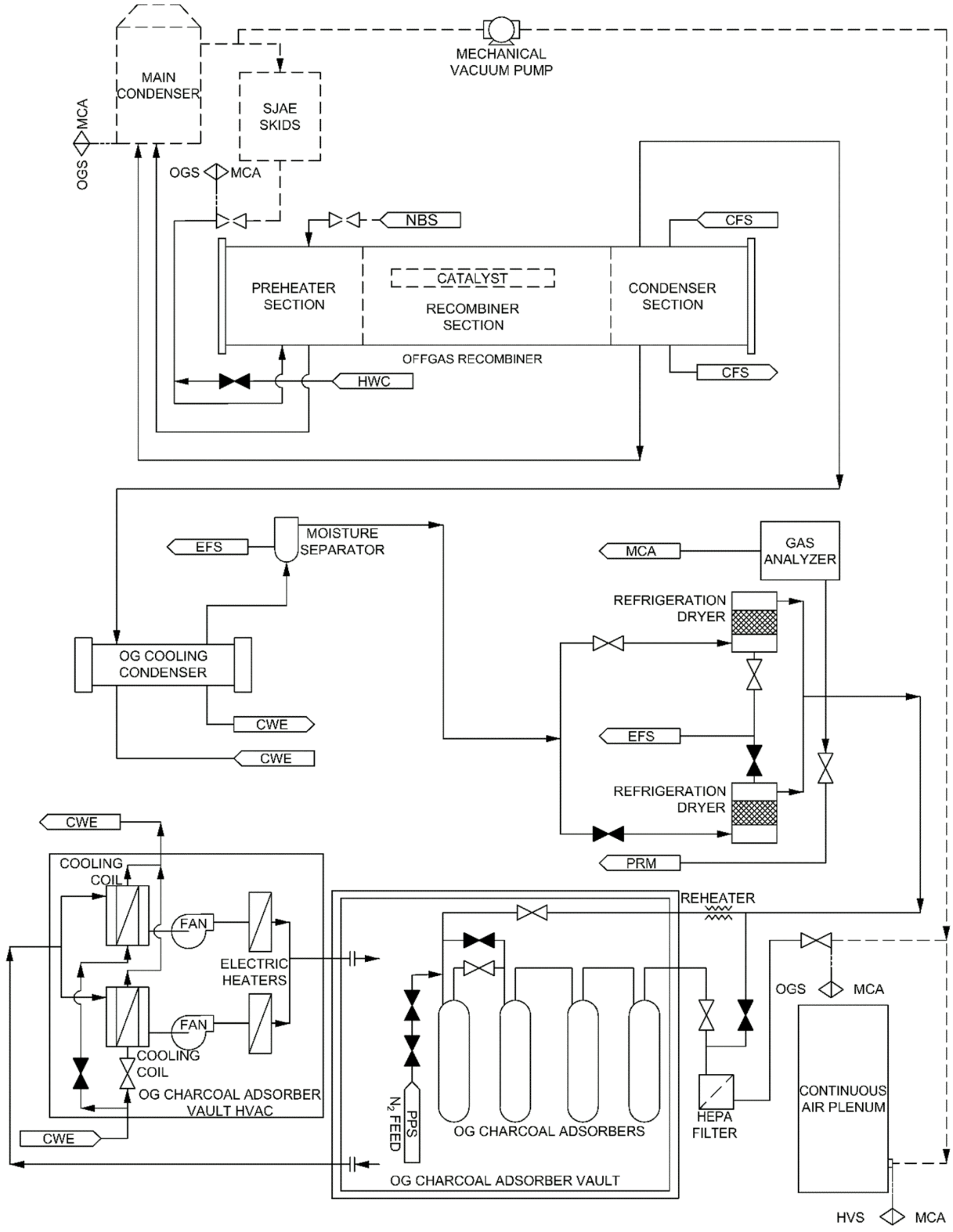


Figure 11.3-1 Offgas System Simplified Diagram

11.4 Solid Waste Management System

The SWM 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. 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

- 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 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 diagram for the SWM. The SWM 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, 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 (HIC) (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. 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. 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 performs 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 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 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. 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 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.

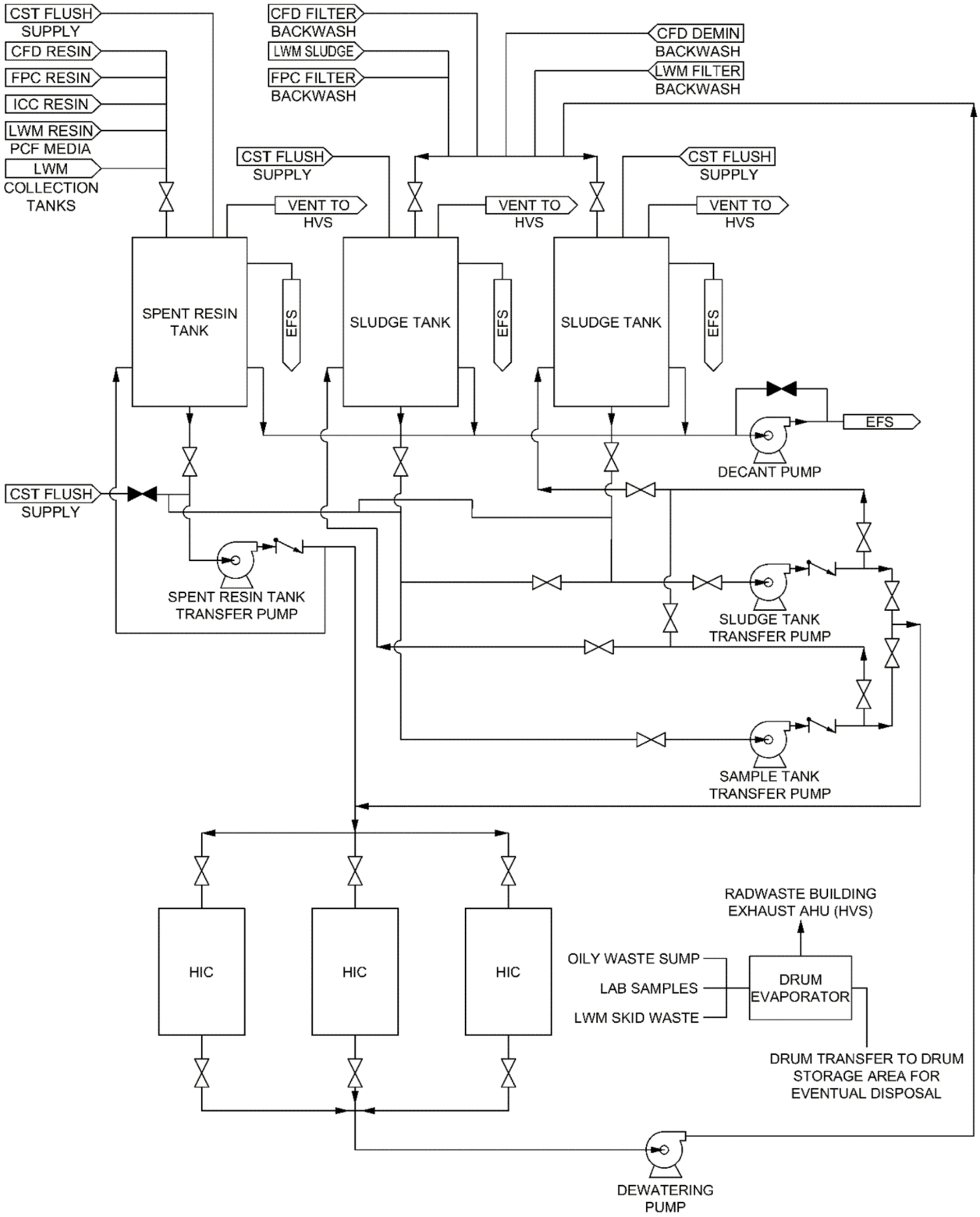


Figure 11.4-1 Solid Waste Management System Simplified Diagram

11.5 Process Radiation Monitoring

The PRM Subsystem is part of the PREMS. In addition to PRM, the PREMS includes the following subsystems:

- Area Radiation Monitoring Subsystem (ARM), Subsection 12.3.4
- Containment Monitoring Subsystem (CMon), Section 6.8
- 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 System is secured if high radiation is detected at the offsite discharge line.
- RB HVS Intake: HVS Secondary Control Room (SCR) Emergency Filter Units (EFU) 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 (AHU).
- 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 or toxic gas is detected at either of the CB supply AHUs.

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 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 plant 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.

The PRM 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 and PDC 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 and PDC 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 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

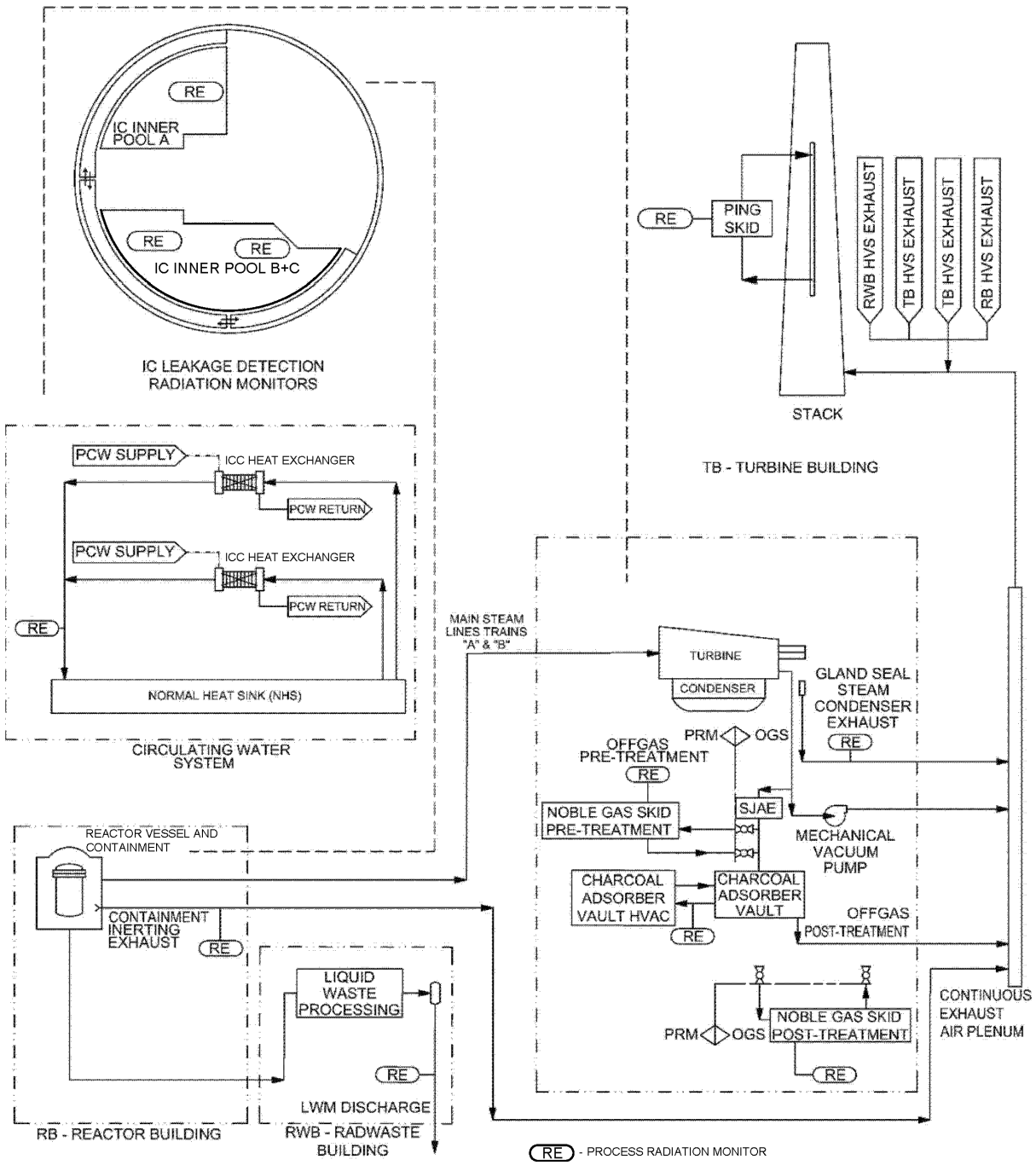
Instrument modules are designed to facilitate calibration checks and troubleshooting. Accessibility for power supply adjustments is provided.

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+C	Area	2	No
IC Pool Atmospheric Vents	At the IC Pool Vents	Offline (PING skid)	3	No
LWM Discharge	Upstream of discharge to NHS System	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



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LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
ALARA	As Low As Reasonably Achievable
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
EFS	Equipment and Floor Drain System
FMCRD	Fine Motion Control Rod Drive
FPC	Fuel Pool Cooling and Cleanup System
GDC	General Design Criteria
GT	Gamma Thermometer
GVH	GE Vernova Hitachi Nuclear Energy
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
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
PREMS	Process Radiation and Environmental Monitoring System
PRM	Process Radiation Monitoring Subsystem
PS	Process Sampling Subsystem

LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
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
USNRC	U.S. Nuclear Regulatory Commission

LIST OF EFFECTIVE SECTIONS

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CHAPTER 12 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 U.S. Nuclear Regulatory Commission (USNRC) 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 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 Analysis 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 (HCU), 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 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 (SJAE) 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 Figure 12.3-11 and Figure 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 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 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 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
(Sheet 1 of 2)

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
Sr-91	5.12E+03
Sr-92	1.10E+04
Y-91	1.10E+02

Table 12.2-2 SDC Heat Exchanger Tube Side Activity
(Sheet 2 of 2)

Isotope	Activity (MBq)
Class 6 (continued)	
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]
(Sheet 1 of 2)

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]
(Sheet 2 of 2)

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]
(Sheet 1 of 2)

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
Y-93	9.90E+00	2.68E-04

Table 12.2-4 FPC Surge Tank Activity Concentrations[1]
(Sheet 2 of 2)

Isotope	Concentration (Bq/g)	Concentration (µCi/g)
Class 6 (continued)		
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]
(Sheet 1 of 2)

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
Y-92	1.08E+07	2.93E+02
Y-93	1.18E+06	3.19E+01

Table 12.2-5 FPC Heat Exchanger Activity[1]
(Sheet 2 of 2)

Isotope	Activity (Bq)	Activity (µCi)
Class 6 (continued)		
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
(Sheet 1 of 2)

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
Y-93	9.90E-06	2.68E-10

Table 12.2-6 Isolation Condenser Pool Activity Concentrations
(Sheet 2 of 2)

Isotope	Concentration (Bq/g)	Concentration (µCi/g)
Class 6 (continued)		
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
(Sheet 1 of 2)

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
Y-92	1.16E+02	3.14E-03
Y-93	1.26E+01	3.42E-04

Table 12.2-7 ICC Heat Exchanger Activity
(Sheet 2 of 2)

Isotope	Activity (Bq)	Activity (µCi)
Class 6 (continued)		
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
(Sheet 1 of 2)

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
Sr-91	3.31E+03
Sr-92	7.10E+03
Y-91	7.10E+01

Table 12.2-8 CUW Regenerative Heat Exchanger Tube Side Activity
(Sheet 2 of 2)

Isotope	Activity (MBq)
Class 6 (continued)	
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
(Sheet 1 of 2)

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
Sr-91	8.84E+02
Sr-92	1.89E+03
Y-91	1.89E+01

Table 12.2-9 CUW Regenerative Heat Exchanger Shell Side Activity
(Sheet 2 of 2)

Isotope	Activity (MBq)
Class 6 (continued)	
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
(Sheet 1 of 2)

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
Co-58	4.29E-01
Co-60	8.45E-01
Ni-63	1.87E-03

Table 12.2-10 Main Condenser Activity
(Sheet 2 of 2)

Isotope	Activity (MBq)
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
(Sheet 1 of 2)

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
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

Table 12.2-11 SJAE Component Activity
(Sheet 2 of 2)

Nuclide	First Stage Ejector (MBq)	Intercondenser (MBq)	Second Stage Ejector (MBq)
Class 6 (continued)			
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
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
(Sheet 1 of 3)

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
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

Table 12.2-12 OGS Normal Operation Equipment Activity
(Sheet 2 of 3)

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)
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
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

Table 12.2-12 OGS Normal Operation Equipment Activity
(Sheet 3 of 3)

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)
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
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
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]
(Sheet 1 of 2)

Nuclide	Activity (MBq/m^3)	Activity ($\mu Ci/m^3$)
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

Table 12.2-13 EFS Containment Sump Activity[1]
(Sheet 2 of 2)

Nuclide	Activity (MBq/m^3)	Activity ($\mu Ci/m^3$)
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]
(Sheet 1 of 2)

Nuclide	Activity (MBq/m^3)	Activity ($\mu Ci/m^3$)
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

Table 12.2-14 EFS Reactor Building Sump Activity[1]
(Sheet 2 of 2)

Nuclide	Activity (MBq/m^3)	Activity ($\mu Ci/m^3$)
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]
(Sheet 1 of 2)

Nuclide	Activity (MBq/m^3)	Activity ($\mu Ci/m^3$)
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

Table 12.2-15 EFS Radwaste Building Sump Activity[1]
(Sheet 2 of 2)

Nuclide	Activity (MBq/m^3)	Activity ($\mu Ci/m^3$)
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]
(Sheet 1 of 2)

Nuclide	Activity (MBq/m^3)	Activity ($\mu Ci/m^3$)
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

Table 12.2-16 EFS Turbine Building Sump Activity[1]
(Sheet 2 of 2)

Nuclide	Activity (MBq/m^3)	Activity ($\mu Ci/m^3$)
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]
(Sheet 1 of 2)

Nuclide	Activity (MBq/m^3)	Activity ($\mu Ci/m^3$)
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

Table 12.2-17 LWM Collection Tank Activity[1]
(Sheet 2 of 2)

Nuclide	Activity (MBq/m^3)	Activity ($\mu Ci/m^3$)
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]
(Sheet 1 of 2)

Nuclide	Activity (MBq/m^3)	Activity ($\mu Ci/m^3$)
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

Table 12.2-18 LWM Sample Tank A Activity[1]
(Sheet 2 of 2)

Nuclide	Activity (MBq/m^3)	Activity ($\mu Ci/m^3$)
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]
(Sheet 1 of 2)

Nuclide	Activity (MBq/m^3)	Activity ($\mu Ci/m^3$)
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

Table 12.2-19 LWM Sample Tank B Activity[1]
(Sheet 2 of 2)

Nuclide	Activity (MBq/m^3)	Activity ($\mu Ci/m^3$)
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]
(Sheet 1 of 2)

Nuclide	Activity (MBq/m^3)	Activity ($\mu Ci/m^3$)
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

Table 12.2-20 LWM Condensate Storage Tank Activity[1]
(Sheet 2 of 2)

Nuclide	Activity (MBq/m^3)	Activity ($\mu Ci/m^3$)
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]
(Sheet 1 of 2)

Nuclide	Activity (MBq/m^3)	Activity ($\mu Ci/m^3$)
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

Table 12.2-21 LWM Refueling Water Storage Tank Activity[1]
(Sheet 2 of 2)

Nuclide	Activity (MBq/m^3)	Activity ($\mu Ci/m^3$)
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
(Sheet 1 of 2)

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
Y-93	7.35E+10

Table 12.2-22 LWM Filter Skid Resin Activity
(Sheet 2 of 2)

Nuclide	Activity (Bq)
Class 6 (continued)	
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]
(Sheet 1 of 2)

Nuclide	Activity (Bq/m^3)	Activity ($\mu Ci/m^3$)
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

Table 12.2-23 SWM Spent Resin Tank Activity[1]
(Sheet 2 of 2)

Nuclide	Activity (Bq/m^3)	Activity ($\mu Ci/m^3$)
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]
(Sheet 1 of 2)

Nuclide	Activity (Bq/m^3)	Activity ($\mu Ci/m^3$)
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

Table 12.2-24 SWM Sludge Storage Tank Activity[1]
(Sheet 2 of 2)

Nuclide	Activity (Bq/m^3)	Activity ($\mu Ci/m^3$)
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
(Sheet 1 of 2)

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
Y-92	7.8E-11	3E-06	2.6E-05

Table 12.2-25 Reactor Building Upper Level Airborne Activity
(Sheet 2 of 2)

Isotope	Airborne Concentration (µCi/mL)	10 CFR 20 DAC Value (µCi/mL)	DAC Fraction
Class 6 (continued)			
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
(Sheet 1 of 2)

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
Y-92	1.2E-11	3E-06	4.1E-06

Table 12.2-26 Reactor Building Lower Levels Airborne Activity
(Sheet 2 of 2)

Isotope	Airborne Concentration (µCi/mL)	10 CFR 20 DAC Value (µCi/mL)	DAC Fraction
Class 6 (continued)			
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
(Sheet 1 of 2)

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
Y-91	6.3E-14	5E-08	1.3E-06
Y-92	1.9E-12	3E-06	6.4E-07

Table 12.2-27 Radwaste Building Airborne Activity
(Sheet 2 of 2)

Isotope	Airborne Concentration (µCi/mL)	10 CFR 20 DAC Value (µCi/mL)	DAC Fraction
Class 6 (continued)			
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
(Sheet 1 of 3)

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
Mn-56	6.8E-12	6E-06	1.1E-06

Table 12.2-28 Turbine Building Airborne Activity
(Sheet 2 of 3)

Isotope	Airborne Concentration (µCi/mL)	10 CFR 20 DAC Value (µCi/mL)	DAC Fraction
Class 6 (continued)			
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

Table 12.2-28 Turbine Building Airborne Activity
(Sheet 3 of 3)

Isotope	Airborne Concentration (µCi/mL)	10 CFR 20 DAC Value (µCi/mL)	DAC Fraction
Class 6 (continued)			
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 the 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 90° elbows is avoided whenever possible. They are replaced by large-radius curved sections at least five times the diameter or two 45° 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 Alloy 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 Figure 12.3-1 through Figure 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

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 objective 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 Figure 12.3-1 through Figure 12.3-19. Plant features are not depicted to scale and areas not shown in the figures are non-radiological control zones.

The dose rate applicable for a particular zone is based on OPEX and represents a 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 1 meter from a radiation source or 1 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 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 of 500 rads or more within a period of one hour at 1 meter from the radiation source or 1 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 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 (HVS) 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 the 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. 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 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 Classifications, 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
- 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 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 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 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 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 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 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.

Figure 12.3-1 Reactor Building Level -34.0 Meters Radiation Zones

Figure 12.3-2 Reactor Building Level -29.0 Meters Radiation Zones

Figure 12.3-3 Reactor Building Level -21.0 Meters Radiation Zones

Figure 12.3-4 Reactor Building Level -14.5 Meters Radiation Zones

Figure 12.3-5 Reactor Building Level -8.5 Meters Radiation Zones

Figure 12.3-6 Reactor Building Level -4.8 Meters Radiation Zones

Figure 12.3-7 Reactor Building Level 0.0 Meters Radiation Zones

Figure 12.3-8 Reactor Building Level 4.9 Meters Radiation Zones

Figure 12.3-9 Reactor Building Level 13.0 Meters Radiation Zones

Figure 12.3-10 Reactor Building Level 22.7 Meters Radiation Zones

Figure 12.3-11 Turbine Building Level 0.0 Meters Radiation Zones

Figure 12.3-12 Turbine Building Level 6.1 Meters Radiation Zones

Figure 12.3-13 Turbine Building Level 13.0 Meters Radiation Zones

Figure 12.3-14 Radwaste Building Level 0.0 Meters Radiation Zones

Figure 12.3-15 Radwaste Building Level 6.1 Meters Radiation Zones

Figure 12.3-16 Radwaste Building Level 13.0 Meters Radiation Zones

Figure 12.3-17 Service Building Level 0.0 Meters Radiation Zones

Figure 12.3-18 Service Building Level 6.1/6.7 Meters Radiation Zones

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., ≥ 0.1 mrem/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 Table 12.4-2 through Table 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 (LPRM) / Gamma Thermometers (GT)
- 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 Storage 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

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 Part 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 1.2-5.

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, 8.4, 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:

1. A documented management commitment to keep exposures ALARA

2. A trained and qualified organization with sufficient authority and well-defined responsibilities

3. 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:

Organization – A radiation protection supervisor and at least one (1) radiation protection technician, each selected, trained, and qualified consistent with the guidance in RG 1.8.

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.

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.

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.

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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LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
ALARA	As Low as Reasonably Achievable
ANS	American Nuclear Society
BDBE	Beyond-Design-Basis Event
BWR	Boiling Water Reactor
BWRX-300	Boiling Water Reactor, 10th Design - 300 MWe
CFR	Code of Federal Regulations
CPA	Construction Permit Application
CRN	Clinch River Nuclear
EOF	Emergency Operations Facility
EPZ	Emergency Planning Zone
ER	Environmental Report
ESP	Early Site Permit
ESPA	Early Site Permit Application
ETE	Evacuation Time Estimate
FEMA	Federal Emergency Management Agency
FFD	Fitness for Duty
FSAR	Final Safety Analysis Report
GM	General Manager
GVH	GE Vernova Hitachi Nuclear Energy
HFE	Human Factors Engineering
INPO	Institute of Nuclear Power Operations
NEI	Nuclear Energy Institute
NNP	New Nuclear Program
USNRC	U.S. Nuclear Regulatory Commission
OLA	Operating License Application
PEP	Plume Exposure Pathway
PM	Plant Manager
PSAR	Preliminary Safety Analysis Report
PSP	Physical Security Plan
QA	Quality Assurance
RAP	Reliability Assurance Program
REAC/TS	Radiation Emergency Assistance Center/Training Site
SSCs	Structures, Systems and Components
SGI	Safeguards Information
SMR	Small Modular Reactor
SSAR	Site Safety Analysis Report
T&Q	Training and Qualification

LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
TSC	Technical Support Center
TVA	Tennessee Valley Authority
USCB	U.S. Census Bureau

LIST OF EFFECTIVE SECTIONS

CHAPTER 13 CONDUCT OF OPERATIONS

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CHAPTER 13 CONDUCT OF OPERATIONS

13.1 Organizational Structure

The Tennessee Valley Authority (TVA) Clinch River Nuclear Unit 1 (CRN-1) organization supporting the CRN-1 construction permit application (CPA) is discussed in Section 17.5. The organizational structure described herein for the design, construction and operations of CRN-1 is based on the GE Vernova Hitachi Nuclear Energy (GVH) BWRX-300 passive plant design. Internal and external functional responsibilities, including oversight during various stages of design, construction and plant operations are performed by onsite resources and offsite resources, such as TVA Nuclear Fleet services and contractors.

13.1.1 Management and Technical Support Organization

TVA has several decades of experience as an electric utility provider. Currently, TVA is the licensee of a fleet of three (3) commercial nuclear power plants. TVA is an agency of the federal government whose major policies, programs, and organization are determined by a Board of Directors structure pursuant to the TVA Governance Restructuring provisions of the Consolidated Appropriations Act of 2005. The Board of Directors members are nominated by the President of the United States and confirmed by the Senate for five-year terms. The Board of Directors selects a Chief Executive Officer who also serves as President to manage TVA's day-to-day business. The Board of Directors shape the long-term business strategies and recommends major program initiatives that guides TVA's day-to-day operations.

TVA has contracted, along with other investors, with GVH to develop a standard design for the BWRX-300 Small Modular Reactor technology. GVH provides engineering and technical services to the fleet of operating Boiling Water Reactors (BWR) 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 also provides services for spent fuel storage. 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 CRN-1. Operating experience from design, construction, and operation of earlier GE BWRs is applied in the design, construction, and operation of the BWRX-300.

The contractor(s) responsible for the construction and additional design engineering for selected site-specific portions of the plant are discussed in Section 1.4. Support contractor(s) are selected based on experience and proven technical capability in nuclear construction projects or projects of similar scope and complexity.

13.1.1.1 Design, Construction and Operating Responsibilities

The President/Chief Executive Officer has overall responsibility for the design, construction, and operation of TVA nuclear plants, including the new nuclear project activities for preparation of the CRN-1 CPA. During the PSAR/Construction stage of the project, the Senior Vice President has the overall responsibility for the CRN-1 CPA, site preparations, design, construction, and testing. The CRN-1 Construction Management organization is shown in Figure 13.1-2. At the appropriate

time after construction, the Chief Nuclear Officer accepts responsibility for the CRN-1 and maintains direct control of nuclear plant operation through the site executive. TVA is responsible for the safe operation of the CRN site. Both the Senior Vice President and the Chief Nuclear Officer will have no ancillary responsibilities that might detract from nuclear safety matters.

At key project milestones (e.g., beginning of construction, fuel load, and commercial operation), senior management ensures there are sufficient qualified personnel available to perform project activities.

Key executive and corporate management positions, functions, and responsibilities, including the construction management organization are addressed in Section 17.5. The Final Safety Analysis Report (FSAR) will identify the TVA corporate officer responsible for nuclear activities that will have no ancillary responsibilities that might detract attention from nuclear safety matters.

Qualifications and training requirements for the design and construction staff will be consistent with Regulatory Guide 1.8, “Qualification and Training of Personnel for Nuclear Power Plants,” for comparable positions.

13.1.1.1.1 Design and Construction Activities

TVA design and construction management teams provide oversight of design and construction activities prior to transitioning from the design/construction phase to the operational phase. The principal site engineering activities during design and construction activities are described below. Engineering activities associated with site characteristics supplement data collected and evaluated to support the Early Site Permit. Section 17.5 describes the responsibilities for transition planning, including responsibilities for the design change processes and the division of responsibilities for design-related activities.

1. Meteorology

Site meteorological data is collected, evaluated, and applied to assess the impact of the station on local meteorological conditions. An onsite meteorological measurements program is also employed to produce data for making atmospheric dispersion estimates for postulated accidents and expected routine airborne releases of effluents.

2. Geology

Site and regional geotechnical characteristics are catalogued and evaluated to determine if geologic conditions could present a challenge to the safety of the plant. Items of interest include geologic structure, seismicity, geological history, and groundwater conditions.

3. Seismology

Seismological data is collected and evaluated to determine if the site location and area surrounding the site is appropriate for the construction and operation of a nuclear power plant. Information regarding tectonics, seismicity, correlation of seismicity with tectonic structure, characterization of seismic sources, and ground motion are assessed to estimate the potential for strong earthquake ground motions or surface deformation at the site.

4. Hydrology

Hydrological conditions at the plant site and the surrounding area are mapped and evaluated. The study includes hydrologic characteristics of streams, lakes, shore regions, the regional and local groundwater environments, and existing or proposed water control structures that could influence flood control and plant safety.

5. Demography

Local and surrounding area population distribution are developed and evaluated. The data is used to determine if requirements are met for establishment of exclusion area, low population zone, and population center distance.

6. Environmental Effects

Monitoring programs are developed to enable the collection of data necessary to determine possible impact on the environment due to construction, startup, and operational activities and to establish a baseline from which to evaluate future environmental monitoring.

7. Design of Plant and Ancillary Systems

Responsibility for design and construction of systems outside the Power Block such as circulating water, service water, switchyard, and fire water supply are delegated to qualified contractors.

8. Review and Approval of Plant Design Features

GVH is the reactor technology vendor and is responsible for design control of the Power Block, including human-factors engineering considerations. Design engineering review and approval is performed in accordance with the GVH approved quality assurance program and procedures. Design issues arising during construction are addressed and implemented with notification and communication of changes to the TVA engineering organization. Design control becomes the responsibility of the TVA engineering organization prior to loading fuel.

9. Review and Approval of Material and Component Specifications

Material and component specifications for structures, systems, and components (SSCs) designed by vendors are reviewed and approved in accordance with an approved vendor QA program or are controlled for review and approval consistent with Section 17.5.

10. Procurement of Materials and Equipment

Procurement of BWRX-300 materials during the construction phase is the responsibility of GVH and the constructor. The process is controlled by the construction QA programs of these organizations. Oversight of the inspection and receipt of materials process is procedurally controlled as described in the construction QA programs.

11. Management and Review of Construction Activities

Overall management and responsibility for construction activities is assigned to the CRN-1 General Manager (GM) Construction/Engineering. The CRN-1 GM Construction/Engineering is accountable to the CRN-1 Site Vice President and directs the organizations responsible for construction oversight. For construction oversight positions and functional area descriptions reporting to the GM Construction/Engineering during the construction phase, see Figure 13.1-2.

Monitoring and reviewing construction activities by CRN-1 personnel are continuous processes. Contractor performance is monitored to provide objective data to CRN-1 management to identify problems early and develop proper corrective actions and solutions. Monitoring of construction activities verifies that contractors comply with contractual obligations for quality, schedule, and cost. Monitoring and review of construction activities is divided functionally across the various disciplines of the construction staff, e.g., electrical, mechanical, instrumentation and control, and tracked based on system and major plant components/areas.

Specific hazards, impacted SSCs, and managerial and administrative controls are reviewed on a recurring basis and, if necessary, controls are revised/developed and implemented and maintained current as work progresses on site. For example, prior to construction activities that involve the use of large construction equipment such as cranes, managerial and administrative controls are in place to prevent adverse impacts on the site or site environs.

12. Document Control

Document control requirements to ensure retention and retrievability of records are in accordance with an approved vendor QA program or consistent with Section 17.5.

13.1.1.1.2 Preoperational Activities

Preoperational activities are performed to transition the CRN-1 from the construction phase to the operational phase and include, but are not limited to preoperational testing, schedule management, test procedure development, and integrated startup testing.

GVH will provide technical assistance for preoperational activities for the BWRX-300 prior to fuel load. After each system is turned over to plant staff, the construction organization relinquishes responsibility for that system. At that time, construction activities will be completed and construction staff available to provide support for preoperational and start-up testing, as necessary. Preoperational activities are performed by the Initial Test Program described in Chapter 14.

13.1.1.1.3 Technical Support for Operations

Site executives are responsible for establishing the organization of managers, functional managers, supervisors, and staff sufficient to perform and support safe plant operations. The

composition, staffing levels, and responsibilities of CRN-1 technical support organizations for the operation of CRN-1 will be described in the FSAR for the following functions:

- Nuclear, mechanical, structural, electrical, thermal-hydraulic, metallurgical, and material, and instrumentation and controls engineering
- Plant chemistry
- Radiation protection
- Fueling and refueling operations support
- Maintenance support
- Operations support
- Quality assurance
- Training
- Safety review
- Fire protection
- Emergency organization
- Outside contractual assistance

13.1.1.1.4 Human Factors Engineering Considerations

Human Factors Engineering (HFE) principles are initially developed by GVH. GVH's HFE organization consists of a core team, with participation from multiple engineering disciplines. The GVH HFE organization's roles and responsibilities align with those in NUREG-0711, and the organization is positioned within the BWRX-300 team such that HFE has the same level of placement and authority as the traditional engineering disciplines. The HFE process, including organizational responsibilities during plant operations will be described in the FSAR.

Section 7.1.3.3 identifies industry standards used to ensure that proven engineering design practices and state-of-the-art methods are used to design the CRN-1 control rooms. In addition, the results from HFE analyses and HFE design requirements are incorporated in the control room design as described in Sections 7.5 and 7.6.

13.1.1.2 Organizational Arrangement

The organizational arrangement for key executive and corporate management positions are described in Section 17.5. The CRN-1 Project Management organization is shown in Figure 13.1-1. The CRN-1 Construction Management organization is shown in Figure 13.1-2.

13.1.1.3 Qualifications

CRN-1 personnel will meet the qualification and training requirements in Regulatory Guide 1.8, "Qualification and Training of Personnel for Nuclear Power Plants," for equivalent positions.

13.1.2 Operating Organization

The overall QA Program requirements for the CRN-1 operation phase is consistent with the guidance in Regulatory Guide 1.33, “Quality Assurance Program Requirements (Operation).” The CRN-1 is a small modular reactor designed for simplicity of operations and relies primarily on passive safety features as part of its safety design approach.

The CRN-1 on-site plant operations’ organizational structure, allocations, responsibilities, and staffing plan will be developed to support detailed design and described in the FSAR. The functions required for plant operations are based on regulatory requirements in 10 CFR Part 50 and 10 CFR Part 55 supplemented by needed functions to operate the plant safely and efficiently consistent with the CRN-1 Construction Permit and Operating License.

The organizational structure and staffing plan will provide assurance that an organization will be established and maintained of adequate size with the competence to protect the health and safety of the public. The functional areas will encompass the regulatory programs at the stages they are required.

Currently, TVA Nuclear Fleet Services provide support for several technical areas to TVA nuclear plants. To the extent practical, staffing plans will consider the use of off-site TVA Nuclear Fleet Services for centralized technical support areas such as procurement, engineering, outage planning, records management, and QA auditors. In addition, staffing plans will consider:

1. Human Factors Engineering
2. Digital Strategy
3. Concept of Operations
4. Maintenance Plan
5. Design Basis Accident Analysis
6. Probabilistic Risk Assessment

Conformance to regulatory guidance is demonstrated and/or exemptions to regulations will be sought to modify or eliminate traditional light water reactor staff positions, qualifications or training requirements that do not align with the BWRX-300 passive plant design. Industry initiatives/guidance and passive plant operating experience will be used to inform the CRN-1 staffing plan, qualifications, and training requirements.

13.1.2.1 Plant Organization

The CRN-1 functional organizational structure shown in Figure 13.1-3 takes into account the passive CRN-1 design. Qualifications and training requirements for the CRN-1 plant staff will be consistent with Regulatory Guide 1.8, “Qualification and Training of Personnel for Nuclear Power Plants,” for comparable positions.

Table 13.1-1 shows the functional positions and qualifications that correlate to comparable positions in Regulatory Guide 1.8. Licensed Senior Operators and Licensed Operators will be trained using the licensed operator training program. Figure 13.1-3 shows the operational phase organizational divisions and the representative functions within each division.

The fire brigade organization, staffing requirements and functional responsibilities are addressed in the Fire Protection Program described in Section 9.5. Radiation protection staffing and functional responsibilities are described in Section 12.5. The emergency preparedness and security staffing and functional responsibilities are described in Appendix 13A and Section 13.6, respectively.

13.1.2.1.1 Plant Manager

The Plant Manager (PM) is responsible for the overall facility operations, activities, and safety of all personnel on site, the protection of the facility, and the safe operation of the facility in a manner that is consistent with the facility license, site policies/procedures and the protection of the health and safety of the public.

13.1.2.1.2 Operations Division

The Operations Division is responsible for management of the planning, directing, and coordinating of the operating activities in accordance with applicable regulations, policies, and procedures. The Operations Division includes shift managers and supervisors, licensed operators, non-licensed operators, and support staff.

13.1.2.1.3 Maintenance Division

The Maintenance Division is responsible for overseeing the planning, directing, and coordinating of routine and corrective maintenance of plant equipment in accordance with applicable regulations, policies, and procedures. The Maintenance Division includes electrical, mechanical, and instrumentation and control supervisors, engineers, craft, and support personnel.

13.1.2.1.4 Technical Division

The Technical Division is responsible for apprising management of the operational and regulatory status and current and projected performance of the site to ensure the plant is operating within the design and licensing basis. The Technical Division includes managers, supervisors, engineering, licensing, fire protection, safety, emergency, and support staff.

13.1.2.1.5 Radiation Protection and Chemistry Division

The Radiation Protection and Chemistry Division is responsible for overseeing the programs and processes designed to protect onsite personnel and the public from radiological hazards and for monitoring and control of the various plant water systems to ensure that the water chemistry remains within acceptable limits. The Radiation Protection and Chemistry Division includes managers, supervisors, specialists, technicians, and support staff that develop, implement, or maintain site programs.

13.1.2.1.6 Generation and Planning Division

The Generation and Planning Division oversees the development and management of the schedules for maintenance work activities and outage activities including the associated programs, processes, and staff used for at power and during outages. The Generation and Planning Division includes managers, supervisors, planners, schedulers, and support staff.

13.1.2.1.7 Site Support Division

The Site Support Division is responsible for directing work activities and ensuring that the work activities are conducted in a timely and efficient manner to provide safe, reliable, and efficient generation to meet system needs for security, performance improvement, training, procurement, document control/records, and business support services. The Site Support Division includes managers, supervisors, instructors, specialists, security officers, and support staff.

13.1.2.2 Plant Personnel Responsibilities and Authorities

Managers and supervisors within the plant operating organization establish goals and expectations for their organization and to reinforce behaviors that promote radiation protection and procedure adherence.

13.1.2.3 Operating Shift Crews

Sufficient staffing is assigned to safely operate and maintain the plant during normal, abnormal, and emergency plant operations. Licensed and senior licensed operators are assigned to the control room continuously using a crew rotation system. Non-licensed staff provide support activities (e.g., administrative, maintenance, radiation protection) to control room staff. Shifts will include a Shift Technical Advisor, who provides advice and counsel to the operations shift and assists in determining causes and mitigating the consequences of plant accidents. A minimum shift crew is established for normal, abnormal, and emergency plant operations. Specific duties, functions, and responsibilities of the operations' staff are described in plant administrative procedures and the Technical Specifications. The minimum shift crew requirements for all plant modes of operation are also identified in the Technical Specifications.

13.1.3 Qualifications of Nuclear Plant Personnel

13.1.3.1 Qualification Requirements

Qualification requirements for functional positions are shown in Table 13.1-1. Specific training and qualification requirements for staff positions will be provided in the Training and Qualification Program.

13.1.3.2 Qualifications of Plant Personnel

Qualification, education, and experience requirements for positions providing corporate technical support are maintained in approved position descriptions. The number of assigned personnel will be contained in approved staffing plans.

An indoctrination and training program for CRN-1 personnel is established to ensure that suitable proficiency is achieved and maintained for safe operations of the plant. Records of personnel training and qualifications are maintained. A description of the training program and the required minimum qualifications for plant staff will be provided in the FSAR, consistent with 10 CFR 50.34(b)(6)(i). The licensed operator training program, including the requalification training program, is addressed in Section 13.2.

Table 13.1-1 Functional Positions and Qualifications
(Sheet 1 of 2)

Nuclear Function	ANS Position	ANS Section
EXECUTIVE MANAGEMENT		
Executive Vice President/Chief Nuclear Officer		N/A
Senior Executive (Sr. Vice President, Vice President)		N/A
PLANT MANAGEMENT		
Plant Manager	Senior Manager	3.1-2014 Section 4.2.1
OPERATIONS DIVISION		
Operations Manager	Senior Manager	3.1-2014 Section 4.2.2
Shift supervisor/manager	Middle Manager	3.1-2014 Section 4.3.7
Shift Technical Advisor	Engineering Personnel	3.1-2014 Section 4.6.2
Non-licensed operator	Operator	3.1-2014 Section 4.5.2
MAINTENANCE DIVISION		
Maintenance Manager	Senior Manager	3.1-2014 Section 4.2.3
Supervisors	First Line Supervisor	3.1-2014 Section 4.4.7
Field Engineers	First Line Supervisor	3.1-2014 Section 4.4.7
Craft Foreman/Leads	First Line Supervisor	3.1-2014 Section 4.4.7
Technician (electrical, mechanical, I&C)	Technician	3.1-2014 Sections 4.5.7 & 4.5.3
RADIATION PROTECTION AND CHEMISTRY DIVISION		
Radiation Protection and Chemistry Manager	Middle Manager	3.1-2014 Sections 4.3.2 & 4.3.3
RP Supervisors	First Line Supervisor	3.1-2014 Section 4.4.6
RP Specialists/Technician	Technician	3.1-2014 Section 4.5.3.2
Chemistry Specialist/Technician	Technician	3.1-2014 Section 4.5.3.1
TECHNICAL DIVISION		
Manager	Senior Manager	3.1-2014 Section 4.2.4
Supervisors	First Line Supervisor	3.1-2014 Section 4.4.8
Discipline Engineering	Engineering	3.1-2014 Section 4.6.1
Safety and Fire Protection Engineer	Engineering	3.1-2014 Section 4.6.1
Performance Monitoring	Engineering	3.1-2014 Section 4.6.1
Emergency Preparedness Supervisor	First Line Supervisor	3.1-2014 Section 4.4.8
Regulatory Compliance/Licensing		NA

Table 13.1-1 Functional Positions and Qualifications
(Sheet 2 of 2)

Nuclear Function	ANS Position	ANS Section
GENERATION AND PLANNING DIVISION		
Manager		NA
Supervisors/Leads	First Line Supervisor	3.1-2014 Section 4.4.8
Schedulers		NA
Planners		NA
Outage support (electrical, mechanical, I&C)	Technician	3.1-2014 Section 4.5.3 & 4.5.7
SITE SUPPORT DIVISION		
Site Support Manager		NA
Training Manager	Senior Manager	3.1-2014 Section 4.2.5
Training Instructors	Instructor	3.1-2014 Section 4.5.4
Performance Improvement Specialist		NA
Procurement Specialist		NA
Security Officers	10 CFR 73.55	NA
Document Control and Records Management		NA
Administrative Support		NA
QUALITY ASSURANCE		
Manager	Senior Manager	3.1-2014 Section 4.2.6
Quality Control	NA	3.1-2014 Section 4.5.5
Quality Assurance	NA	3.1-2014 Section 4.5.6

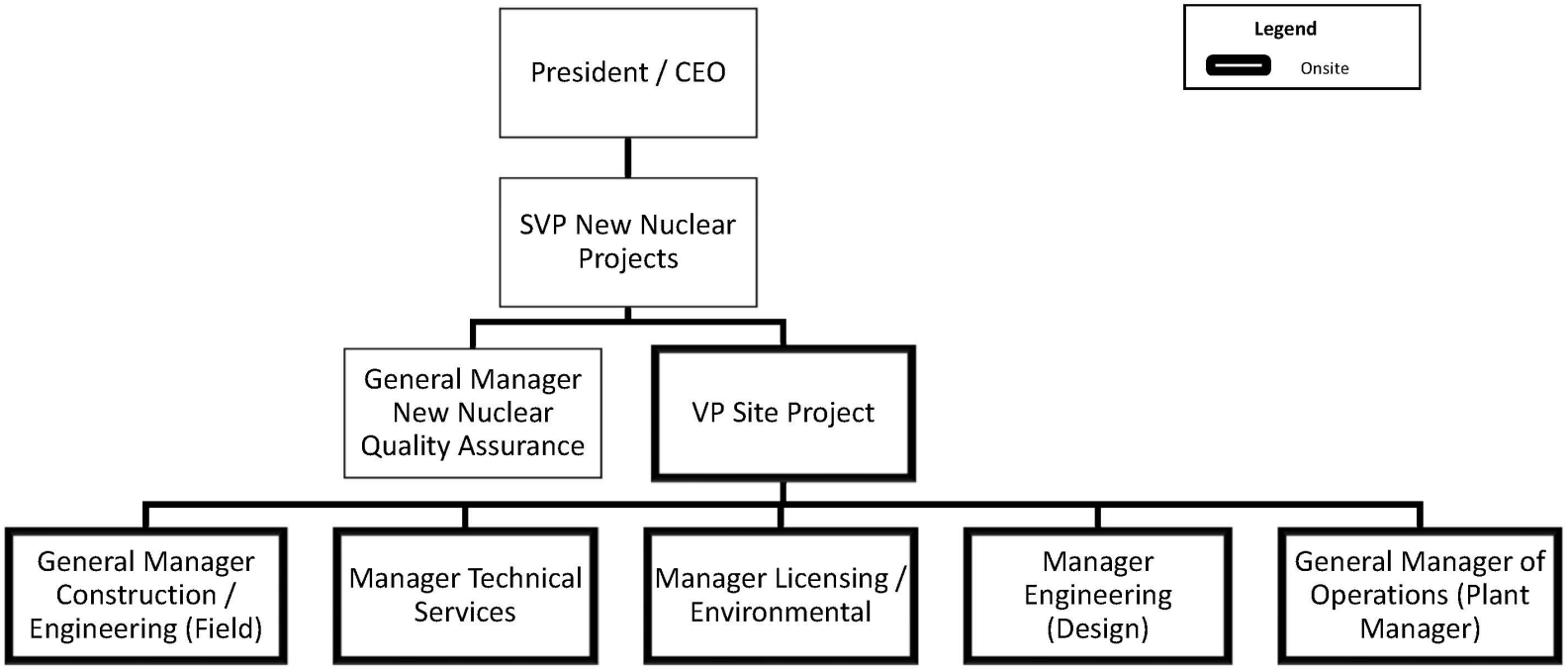

Figure 13.1-1 CRN-1 Project Management Organization

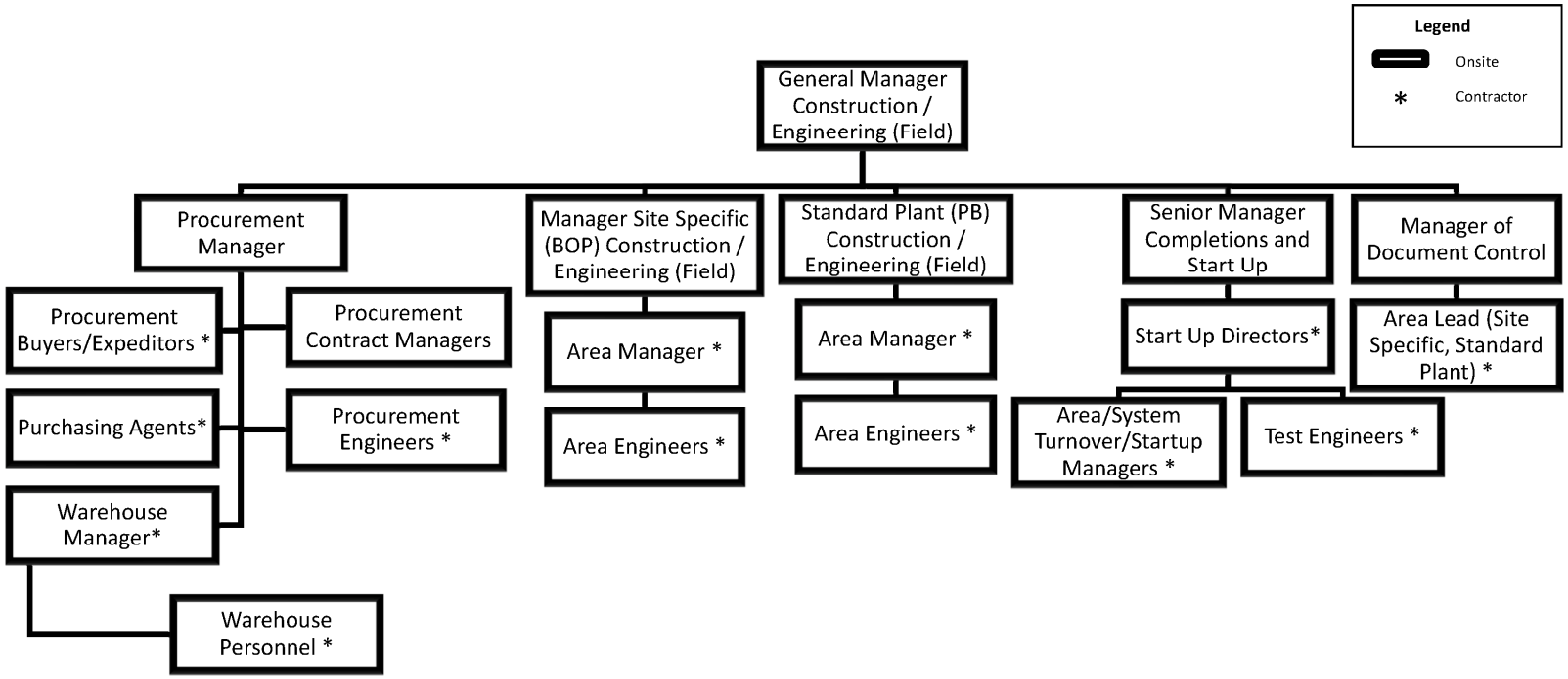

Figure 13.1-2 CRN-1 Construction Management Organization

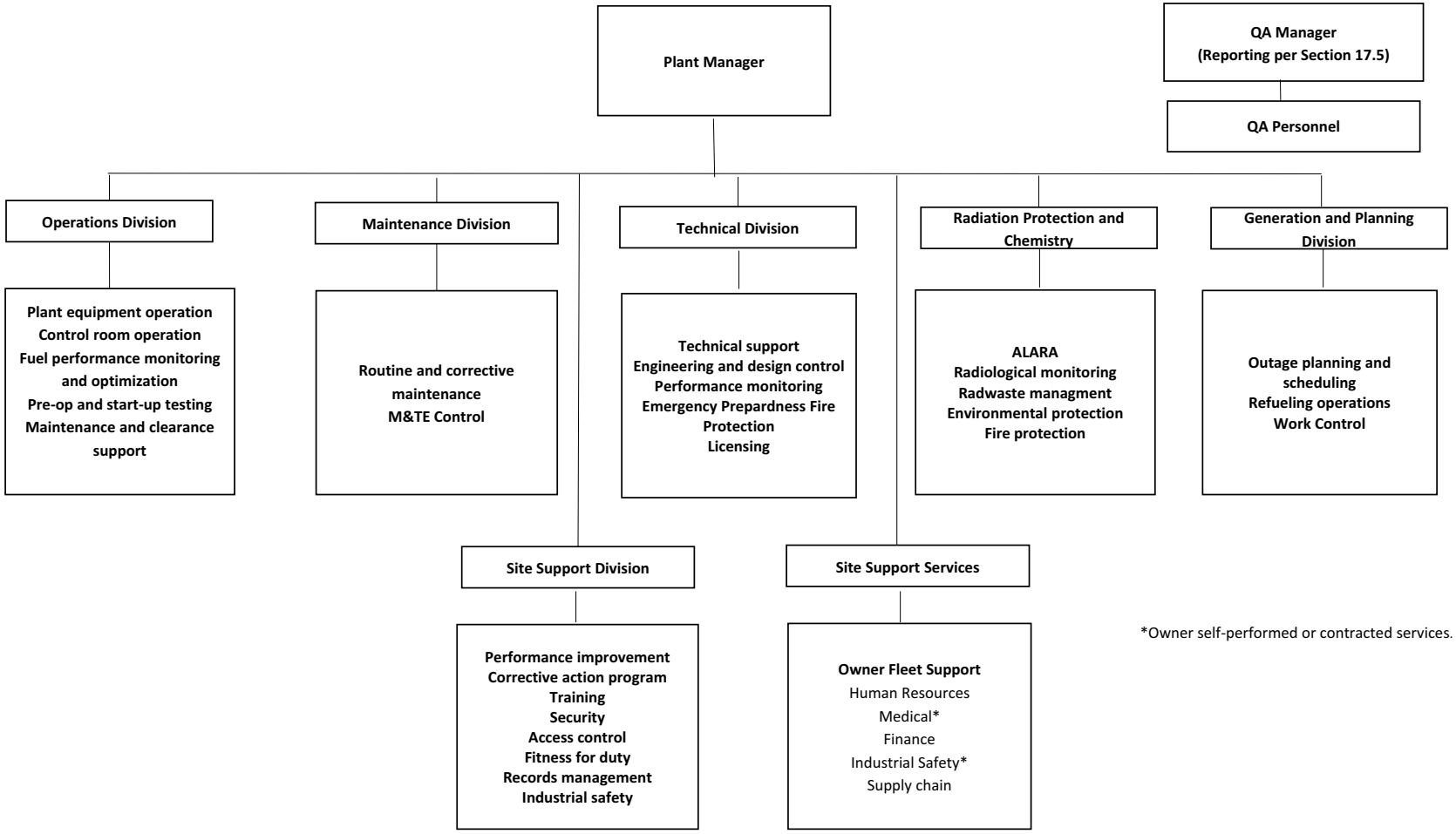

Figure 13.1-3 CRN-1 On-Site Functional Organization

13.2 Training

13.2.1 Plant Staff Training

The CRN-1 Training Program is developed to satisfy the applicable requirements in 10 CFR Part 50; 10 CFR Part 55; and U.S. Nuclear Regulatory Commission (USNRC) and industry guidance. The objectives of the training program are to assure that:

- Plant personnel are properly trained and qualified to perform their assigned tasks in a safe and efficient operation.
- Training is focused on the essential knowledge, skills and safety attributes required to meet the job requirements and nuclear safety specific needs throughout the lifecycle of the facility.
- Training requirements are applied in a manner commensurate with risk and may vary based on the safety significance and complexity of the work being performed.

Portions of the training program applicable to radioactive materials will be in place prior to receipt of byproduct, source, or special nuclear materials (excluding exempt sources) to satisfy the training requirements of licensing 10 CFR Parts 30, 40, and 70.

13.2.1.1 Program Description

The CRN-1 Training Program is developed, approved and maintained by the Training Division. The CRN-1 Training Program is developed using a Systems Approach to Training- based process consistent with 10 CFR 50.120. Concepts of operations of the BWRX-300 are incorporated into the CRN-1 Training Program, as applicable. Details of the concept of operations will be available prior to initial fuel load and will address, at a minimum:

- Human-systems interfaces
- Automation levels
- Operating experience

Details of the initial and requalification training programs for the operators will be provided to support the Operating License Application (OLA). The guidance in Regulatory Guide 1.149 “Nuclear Power Plant Simulation Facilities for Use in Operator Training, License Examinations, and Applicant Experience Requirements,” will be used for simulator training. The Training and Qualification Programs will be developed and implemented as identified in Table 13.4-1. The licensed operator training program will describe training and qualification requirements for hot and cold licenses that satisfy the 10 CFR Part 55 requirements and requalification requirements identified in Table 13.4-1. The guidance in NUREG 1021, “Operator Licensing Examination Standards for Power Reactors,” will be used for the licensed operator training program. The methods used to satisfy the requirements for hot and cold licenses and basis for using the selected methods will be described in the licensed operator training program.

Operating experience is incorporated into training and retraining programs. In addition, training programs are periodically reviewed for effectiveness. Revisions are made as appropriate. Training records are retained as described in Section 17.5.

13.2.1.2 Coordination with Preoperational Tests and Fuel Loading

The final training program to support plant operations will be described in the FSAR. The milestones for implementing the training program are shown in Table 13.4-1.

13.2.2 Requalification, Refresher and Replacement Training Programs

Continuing training and retraining are provided periodically to plant staff to maintain qualifications and enhance proficiencies. Table 13.4-1 identifies the implementation schedule for the requalification training program.

13.3 Emergency Preparedness

Section 13.3 of the referenced ESPA SSAR is incorporated by reference with the following supplements.

Replace ESPA SSAR Section 13.3 with the following:

Section 13.3 of the Early Site Permit Application (ESPA) Site Safety Analysis Report (SSAR), in conjunction with Part 5 Emergency Plan of the ESPA, describes emergency preparedness for a small modular reactor (SMR) facility at the Clinch River Nuclear (CRN) Site.

Part 5 of TVA's Early Site Permit Application (ESPA) contains the major features of two emergency plans addressing two distinct potential Plume Exposure Pathway (PEP) Emergency Planning Zones (EPZ) (Reference 13.3-2).

- Part 5A of the ESPA contains the major features of an emergency plan for a PEP EPZ at the site boundary of the CRN Site.
- Part 5B of the ESPA contains the major features of an emergency plan for a PEP EPZ that consists of an area approximately 2 miles in radius from the center point of the CRN Site.

Each major features emergency plan complies with 10 CFR 50.47(b) and Appendix E to 10 CFR Part 50, except where exemptions were requested, as described in Part 6, "Exemptions and Departures," of TVA's ESPA. The exemptions requested in TVA's ESPA were granted by the USNRC, as detailed in Paragraph 4, "Exemptions," of the Early Site Permit (ESP-006) (Reference 13.3-3). As discussed in Section 13.3 of the Final Safety Evaluation Report, the exemptions are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security. Also, special circumstances are present under 10 CFR 50.12(a)(2)(ii) because application of the regulations in the particular circumstances is not necessary to achieve the underlying purpose of the rule (Reference 13.3-2).

ESP-006 states that an applicant referencing the ESP may use the exemptions for either emergency plan contained in Part 5 of the ESPA in a future application only if:

- the condition in Section 3.F.(5) of ESP-006 is satisfied, and
- the applicant performs an analysis in accordance with the methodology in ESPA SSAR Section 13.3 that meets all criteria in ESPA SSAR Section 13.3 for a site boundary or for a 2-mile plume exposure pathway EPZ.

10 CFR 50.34(a)(10), "Contents of applications; technical information," requires that, in its PSAR, an application for a construction permit include a discussion of an applicant's preliminary plans for coping with emergencies.

Appendix E to 10 CFR 50, "Emergency Planning and Preparedness for Production and Utilization Facilities," establishes requirements for emergency plans to attain an acceptable state of emergency preparedness and to provide reasonable assurance that protective measures can

and will be taken to protect the health and safety of workers and the public. Specifically, Section II of Appendix E to 10 CFR 50 addresses emergency planning-related information required in a PSAR.

Appendix 13A provides the emergency planning-related information required in a PSAR by 10 CFR 50.34(a)(10) and Section II of Appendix E to 10 CFR Part 50.

13.3.1.1 Site Description

Replace the first sentence of the first paragraph of ESPA SSAR Section 13.3.1.1 with the following:

Chapter 2 of the PSAR provides a detailed description of the CRN Site and provides additional information regarding the physical characteristics of the proposed site, including the preliminary site layout and potential hazards in the site vicinity.

13.3.1.2 Area Population

Replace ESPA SSAR Subsection 13.3.1.2 with the following:

Permanent Resident Population

The permanent resident population has been estimated using U.S. Census Bureau (USCB) census block data obtained from the U.S. Census 2020 and is projected to 2030. The permanent resident population in the vicinity of the CRN Site is calculated using tabular and spatial 2020 census data downloaded from the USCB website. This data includes geographical state, county, and census block boundaries as well as 2020 population counts by census block. The ESPA considered a plant parameter envelope (PPE) that used a bounding approach to account for multiple reactor technologies and units at the CRN Site. This approach resulted in “analytical Exclusion Area Boundaries” encompassed by an ellipse with a short axis of 0.326 miles from the site center point and a long axis of 0.535 miles from the center point. To maintain consistency with the ESPA Environmental Report (ER) site center point, a radius equivalent to the long axis of the ellipse (0.535 mile) has been utilized to develop a “buffer” encompassing the CRN Site area evaluated in the ESPA. Distance measurements in relation to the CRN Site are made from the buffered limit of the CRN Site center point. For example, the calculations for the population within 10 miles of the CRN Site include those located within 10 miles of the bounding limits of the center point, or approximately 10.535 miles from the CRN Site center point.

According to the USCB 2020 data, projected to the year 2030, there are 1,294 permanent residents within 2.535 miles and approximately 90,168 permanent residents within 10.535 miles of the CRN Site.

Estimates of the resident population data obtained from the USCB are divided into distance bands and directional sectors established by 15 concentric bands at 0-0.3 mile, 0.3-1 mile,

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1-2 miles, 2-3 miles, 3-4 miles, 4-5 miles, 5-6 miles, 6-7 miles, 7-8 miles, 8-9 miles, 9-10 miles, 10-20 miles, 20-30 miles, 30-40 miles, and 40-50 miles from the buffered CRN Site center point in 22.5 degree segments centered on 16 compass points. Because the ESPA ER did not use a buffered center point in the previous demographic analyses, differences in permanent population are evident in these distance bands. The most prominent differences appear in the 0-0.3-mile distance band. The differences in permanent residents are not due to an increase in population in the immediate vicinity of the CRN Site, but rather the increase in land area within this band as a result of the buffer described above. With the inclusion of the 0.535-mile radius buffer, the 0.3-mile distance band for the current analysis is located 0.835 mile from the CRN Site center point, encompassing much of the area that was located in the 0.3-1-mile distance band in the original analysis. Thus, while the ESPA ER reported zero residents within the 0-0.3-mile distance for all directions, the updated estimate of 56 residents for the same distance in 2020 does not represent new residents or a change in the resident distribution; rather, they were presented as part of the 0.3-1-mile totals in the ESPA ER. At distance bands further from the CRN Site, the buffer still contributes to the inclusion of a greater total area, but the impacts are of a lesser magnitude.

Transient Resident Population

An evaluation of the transient population in the vicinity of the CRN Site was conducted to support the ESPA. Data for the transient populations were obtained from a variety of public sources regarding major places of employment, schools, medical facilities, lodging facilities, and recreation areas. Estimates of the transient population were then divided into the distance bands and directional sectors within a 30-mile radius. Because there is no reliable way to distinguish between recreational users who live in the 0- to 30-mile radius and those who do not, all users of recreational facilities in the 0- to 30-mile radius are considered as transients. This is a conservative assumption that slightly overestimates transient populations in the vicinity of the CRN Site. The number of transients potentially located within 10 miles of CRN Site is 12,700 people.

The evacuation time estimate (ETE) performed during development of the ESPA did not identify any physical characteristics unique to the CRN Site which pose significant impediments to the development of the Emergency Plan for the CRN Site. The roadway network is modeled in the ETE and is shown to be sufficient to handle the volume of traffic in the event of an emergency. The ETE performed during the ESPA serves as the preliminary analysis of the time required to evacuate the plume exposure pathway EPZ for transient and permanent populations.

13.3.3 Emergency Planning Zones

Replace the last sentence of ESPA SSAR Section 13.3.3 with the following:

The EPZs for the CRN Site are based on these requirements and the exemptions contained in ESP-006.

13.3.3.1.1 Environmental Protection Agency Protective Action Guides

Replace the last sentence of the second paragraph of ESPA SSAR Section 13.3.3.1.1 with the following:

For the BWRX-300 technology, the FSAR will confirm dose consequences outside the PEP EPZ do not exceed the EPA PAG limit of less than 1 rem TEDE for less severe BDBEs.

Replace the fifth paragraph of ESPA SSAR Section 13.3.3.1.1 with the following:

Step 1 - Accident Scenario Selection: DBA scenarios are identified in the postulated accidents defined in Chapter 15 of the FSAR.

Replace the seventh paragraph of ESPA SSAR Section 13.3.3.1.1 with the following:

Step 2 - Accident Source Term Evaluation: DBA accident source terms are defined for Postulated Accidents in Chapter 15 of the FSAR.

13.3.3.1.2 Substantial Reduction in Early Health Effects

Replace the last sentence of the third paragraph of ESPA SSAR Section 13.3.3.1.2 with the following:

For the BWRX-300 technology, the FSAR will confirm the conditional probability of exceeding 200 rem (whole body) outside the PEP EPZ < 1 x 10-3 per rx-yr.

Replace the first sentence of Step 3 of ESPA SSAR Section 13.3.3.1.2 with the following:

PEP EPZ boundary consequence calculations for less frequent, more severe BDBEs will be based on the methodology accepted for the BWRX-300 BDBA source terms.

13.3.3.1.3.1 Multiple Reactors at the CRN Site

Replace ESPA SSAR Subsection 13.3.3.1.3.1 with the following:

Not Applicable

13.3.3.1.3.2 Mitigation of Beyond-Design-Basis Events

Replace the fourth sentence of ESPA SSAR Section 13.3.3.1.3.2 with the following:

Additionally, compliance with 10 CFR 50.155 provides for significant additional confidence that such events would not result in impacts to public health and safety beyond the capabilities of the site to manage; this includes assessment of malevolent acts as well as prepositioning regional assets for mitigating Beyond-Design-Basis Events (BDBE).

Replace the fifth sentence of ESPA SSAR Section 13.3.3.1.3.2 with the following:

The FSAR will include Diverse and Flexible Coping Strategies that provide sufficient equipment and supporting staff to enable the core cooling, containment, and fuel pool cooling functions to be maintained or restored for CRN-1.

13.3.3.1.3.3 Comprehensive Emergency Management Plan

Replace the third paragraph of ESPA SSAR Section 13.3.3.1.3.3 with the following:

The NRC requires a level of emergency preparedness commensurate with the potential consequences to public health and safety and common defense and security at the CRN Site. The Emergency Plans presented in Part 5 consider the proposed exemptions contained in ESP-006. Implementation of the Site Boundary PEP EPZ as in Part 5A does not rely on specific offsite actions. Projected consequences are such that conditions exceeding pre-planned actions can be carried out with ad hoc protective actions. This approach is described in more detail below. As addressed in the FEMA's Comprehensive Preparedness Guide 101, Developing and Maintaining Emergency Operations Plans (CPG-101) (Reference 13.3-4), if determined appropriate, government officials may utilize a Comprehensive Emergency Management Plan (CEMP) approach to emergency planning to implement ad hoc protective actions to protect the public (Reference 13.3-4).

Replace the second sentence of the last paragraph of ESPA SSAR Section 13.3.3.1.3.3 with the following:

Finally, additional assurance exists via compliance with 10 CFR 50.155 that safety functions will be maintained in the highly unlikely event BDBEs.

Replace Reference 13.3-3 with Reference 13.3-4 in the third and fifth paragraphs of ESPA SSAR Section 13.3.3.1.3.3.

Rename ESPA SSAR Section 13.3.3.1.4 to Operating License Application and replace with the following:

13.3.3.1.4 Operating License Application

TVA intends to include one complete and integrated Emergency Plan in the FSAR. TVA intends to demonstrate that the design of the facility presented in the FSAR falls within the design parameters postulated in the ESPA.

If the dose consequences of the BWRX-300 technology do not exceed the EPA PAG at the Site Boundary and do not present a substantial risk that doses at which significant early health effects may occur, then TVA could elect to establish the PEP EPZ at the Site Boundary in the FSAR. In this scenario, TVA will coordinate with the OROs regarding establishment of the PEP EPZ boundary. The exemptions that are described in ESP-006, related to the Part 5A Emergency Plan, would not affect the authority that FEMA has under its regulations in 44 CFR Chapter I for overall emergency management and assistance to State and local response organizations, nor would it affect the responsibilities of State and local governments to establish and maintain Comprehensive Emergency Management Plans.

If the dose consequences of the BWRX-300 technology do not exceed the EPA PAG at a two-mile radius and do not present a substantial risk that doses at which significant early health effects may occur, then TVA could elect to establish the PEP EPZ at an approximate two-mile radius in the FSAR with consideration given to local emergency response needs and capabilities, including considerations of demography, topography, land characteristics, access routes, and jurisdictional boundaries. The exemptions that are described in ESP-006, related to the Part 5B Emergency Plan, enable the USNRC to base its finding of reasonable assurance on its review of the onsite emergency preparedness and would require a finding from FEMA on the adequacy of State and local radiological emergency response plans. 44 CFR 350(b) directs FEMA's Radiological Emergency Preparedness (REP) Program to review State and local radiological emergency plans and preparedness. Approved plans and preparedness must be determined to adequately protect the public health and safety by providing reasonable assurance that appropriate protective measures can be taken offsite in the event of a radiological emergency.

If the dose consequences of the BWRX-300 technology exceed the EPA PAG or present a substantial risk that doses at which significant early health effects may occur for the PEP EPZ boundary at a two-mile radius, then neither Emergency Plan included in Part 5 of this ESPA will be incorporated by reference in the FSAR and a new Emergency Plan will be included with the OLA for USNRC review.

The previous discussion regarding dose consequences and the selected EPZ that TVA could elect to describe in the FSAR are summarized below:

Summary Determination of EPZ Size

Criteria Satisfied at:	Site Boundary EPZ	Two-Mile EPZ	EPZ Described in FSAR
Distance < Site Boundary	X		
Distance > Site Boundary, but < 2 miles		X	
Distance > 2 miles			X

The FSAR will include the following detailed information of the BWRX-300 technology that is pertinent to the Emergency Plan:

- DBA analysis information from Chapter 15 of the FSAR
- PRA Level 1 information that defines and informs the selection of less severe and more severe accident sequences, including appropriate consideration of extreme external hazards
- Level 2 PRA information to define fission product releases for the selected less severe and more severe accident sequences
- Offsite dose calculations that determine the dose associated with the selected less severe and more severe accident sequences

The FSAR will apply the methodology in Subsection 13.3.3.1.1 for EPA PAG and Subsection 13.3.3.1.2 for Substantial Reduction in Early Health Effects to the BWRX-300 technology to determine an acceptable PEP EPZ.

The PRA-based evaluation to support the PEP EPZ included in the FSAR will be based on plant-specific Level 1 and Level 2 PRA information available at the FSAR . The PRA will address the following attributes:

- Applicable plant operating states (modes) including full power, low power and shutdown, and design-specific operating states unique to the BWRX-300 technology (e.g., refueling and/or concurrent power operation and refueling)
- Accident initiators appropriate for PRA evaluation, including internal and external hazards
- Applicable fuel handling accidents and fuel pool accidents

If necessary to support the FSAR, TVA will have an engineering assessment prepared for the BWRX-300 design in the FSAR (qualitative or quantitative, as appropriate) to show that risks assessed by the PRA are adequately treated inside the PRA. For risks not assessed by the PRA, TVA will confirm the existence, functionality, and capability of features and processes in the design, operation, accident management, and emergency response to address any such risks

and to provide confidence that the associated risk impact is acceptably low. The FSAR will also address, as needed, security, limited operating experience, co-location (if necessary), or risks for which industry standards may not be available.

The PRA methodologies used to support the Emergency Plan will be addressed in the FSAR.

13.3.3.2 Ingestion Exposure Pathway

Replace ESPA SSAR Section 13.3.3.3.2 with the following:

The ingestion exposure pathway EPZ for the CRN Site will be described in the FSAR.

13.3.5 Contacts and Arrangements

Replace ESPA SSAR Section 13.3.5 with the following:

TVA has engaged in discussions with State, county, and local agencies and emergency response organizations regarding the development of emergency plans to support operation of an SMR at the CRN Site. These discussions have been productive and indicative of broad support from these organizations in further development of emergency plans, as indicated by letters of support from the State of Tennessee, Anderson County, Roane County, and the City of Oak Ridge included in Appendix 13B.

Surrounding emergency response organizations currently support DOE's ORR. TVA will maintain arrangements with local medical facilities to receive patients from the CRN Site. A Letter of Agreement (LOA) for local medical facility services for the CRN Site will be obtained by TVA and provided in the FSAR. TVA has an existing agreement with the Radiation Emergency Assistance Center/Training Site in Oak Ridge, Tennessee, for radiological medical support (Reference 13.3-5).

TVA will maintain arrangements for offsite ambulance assistance to the CRN Site. A LOA for ambulance service for the CRN Site will be obtained by TVA and provided in the FSAR. TVA will maintain arrangements for local firefighting support. The senior offsite fire official responding to the CRN Site will coordinate response activities with the TVA Incident Commander.

The CRN Site is responsible for providing radiological protection and proper safety clearance in affected areas. A LOA for local firefighting support for the CRN Site will be obtained by TVA and provided in the FSAR.

TVA will maintain agreements with local law enforcement agencies to provide law enforcement support when necessary. A LOA for local law enforcement agency support for the CRN Site will be obtained by TVA and provided in the FSAR.

Add new ESPA SSAR Section 13.3.6

13.3.6 Permit Conditions and Action Items

Subsequent to USNRC issuance of ESP-006, TVA selected the GVH BWRX-300 reactor technology for further evaluation at the CRN Site. If TVA elects to pursue an Operating License for the BWRX-300 at the CRN Site, then TVA will update the emergency plan in the OLA based on the BWRX-300 reactor technology and address the applicable emergency planning-related Permit Conditions and Action Items in paragraph 3.F and Appendix C of ESP-006.

13.3.7 References

Renumber ESPA SSAR Section 13.3.6 to Section 13.3.7.

Replace the References listed in ESPA SSAR Section 13.3.6 with the following References.

13.3-1 EPA 400-R092-001, “Manual of Protective Action Guides and Protective Actions for Nuclear Incidents,” U.S. Environmental Protection Agency, Office of Radiation Programs, May 1992.

13.3-2 “Final Safety Evaluation Report for the Early Site Permit Application for the Clinch River Nuclear Site,” U.S. Nuclear Regulatory Commission, June 2019 (ML19162A157).

13.3-3 “Clinch River Nuclear Site Early Site Permit, ESP-006,” U.S. Nuclear Regulatory Commission, December 2019 (ML19352D868) and Final Safety Evaluation Report, June 2019 (ML19162A157).

13.3-4 FEMA, “Developing and Maintaining Emergency Operations Plans,” Comprehensive Preparedness Guide (CPG) 101, Ver. 2.0, November 2010.

13.3-5 Letter from Michele G. Branton, Contracting Officer’s Representative, Department of Energy, to John Parshall, Program Manager, State and Local Emergency Preparedness, TVA, “Letter of Agreement – Radiation Emergency Assistance Center/Training Site (REAC/TS) Support,” March 27, 2015.

13.4 Operational Programs

Operational programs are specific programs required by regulation. Table 13.4-1 identifies the operational programs for CRN-1.

Table 13.4-1 lists applicable operational programs, the regulatory source of the program; proposed implementation milestone for each program; and the section of the FSAR in which the operational program is described. Full details of the operational programs will be available to support the OLA.

Table 13.4-1 Operational Programs and Implementation Milestones
(Sheet 1 of 6)

Program Title	Source Requirement	Implementation Milestone	FSAR Section
Inservice Inspection Program	10 CFR 50.55a(g)	Initial Operations	5.2.4 6.6 3.8.1.7
Inservice Testing Program	10 CFR 50.55a(f) 10 CFR Part 50, Appendix A	Fuel Load	3.9.6
Environmental Qualification Program	10 CFR 50.49	Construction Permit application for conceptual approach OLA submittal actual program	3.11
Preservice Inspection Program	10 CFR 50.55a(g)	Fuel Load (Prior to initial plant start-up)	5.2.4 6.6 3.8.1.7
Preservice Testing Program	10 CFR 50.55a(f)) 10 CFR 50.34	Fuel Load	3.9.6
Reactor Vessel Material Surveillance Program	10 CFR 50.60 10 CFR 50, Appendix H	Fuel Load	5.3
Containment Leakage Rate Testing Program	10 CFR 50.54(o) 10 CFR 50, Appendix A 10 CFR 50, Appendix J	Fuel Load	6.2
Fire Protection Program	10 CFR 50.48 10 CFR 30.32 10 CFR 40.31 10 CFR 70.22	Fuel Load Material Receipt Material Receipt Material Receipt	9.5
Process and Effluent Monitoring and Sampling Program	10 CFR 20.1301 and 20.1302 10 CFR 50.34a 10 CFR 50.36a 10 CFR 50, Appendix I, Sections II, III and IV	Fuel Load	11.5
Offsite Dose Calculation Program	10 CFR 20.1301 and 20.1302 10 CFR 50.34a 10 CFR 50.36a 10 CFR 50, Appendix I, Sections II, III and IV	Fuel Load	11.5

Table 13.4-1 Operational Programs and Implementation Milestones
(Sheet 2 of 6)

Program Title	Source Requirement	Implementation Milestone	FSAR Section
Radiological Environmental Monitoring Program	10 CFR 20.1301 and 20.1302 10 CFR 50.34a 10 CFR 50.36a 10 CFR 50, Appendix I, Sections II, III and IV	Fuel Load	11.5
Process Control Program	10 CFR 20.1301 and 20.1302 10 CFR 50.34a 10 CFR 50.36a 10 CFR 50, Appendix I Sections II, III and IV 10 CFR 30.18	Fuel Load	11.4
Radiation Protection Program	10 CFR 20.1101 and 20.1406 10 CFR Part 30 10 CFR Part 40 10 CFR Part 70	Material Receipt Fuel Receipt Fuel Load (ALARA Program) Material Receipt Material Receipt Material Receipt	12.5
Non-licensed Plant Staff Training Program	10 CFR 50.120 10 CFR 30.32 10 CFR 40.31 10 CFR 70.22	18 months prior to fuel load Material Receipt Material Receipt Material Receipt	13.2
Reactor Operator Training Program	10 CFR 55.13 10 CFR 55.31 10 CFR 55.41 10 CFR 55.43 10 CFR 55.45 10 CFR 55.46	18 months prior to fuel load	13.2
Reactor Operator Requalification Program	10 CFR 50.34(b) 10 CFR 50.54(i) 10 CFR 50.54(j) 10 CFR 55.46 10 CFR 55.59	3 months after issuance of the Operating License	13.2

Table 13.4-1 Operational Programs and Implementation Milestones (Sheet 3 of 6)

Program Title	Source Requirement	Implementation Milestone	FSAR Section
Emergency Planning	10 CFR 50.34(a)(10) 10 CFR 50.47 10 CFR 50.54(q) 10 CFR 50, Appendix E	2 yr prior to issuance of full power license Onsite exercise conducted within 1 yr before issuance of full power license. Detailed implementing procedures submitted no less than 180 days prior to fuel load	13.3
Physical Security Program	10 CFR 50.34(c) 10 CFR 50.34(e) 10 CFR 50.54(p) 10 CFR 73.55 10 CFR 73.56 10 CFR 73.57	Prior to receipt of fuel onsite (protected area)	13.6
Physical Protection Program (SNM)	10 CFR 73.1 10 CFR 73.67	Prior to receipt of fuel onsite (protected area)	13.6
Safeguards Contingency Program	10 CFR 50.34(d) 10 CFR 73.55(c)(5) 10 CFR 73.55(k) 10 CFR 73, Appendix C	Prior to receipt of fuel onsite (protected area)	13.6
Security Guard Training and Qualification Program	10 CFR 50.34(c) 10 CFR 73.55(c)(4) 10 CFR 73.55(d)(3) 10 CFR 73, Appendix B	Prior to receipt of fuel onsite (protected area)	13.6
Cyber Security Plan	10 CFR 50.34(c) 10 CFR 73.54(b) 10 CFR 73.55(b)(8) 10 CFR 73.55(c)(6)	Prior to receipt of fuel onsite (protected area)	13.6

**Table 13.4-1 Operational Programs and Implementation Milestones
(Sheet 4 of 6)**

Program Title	Source Requirement	Implementation Milestone	FSAR Section
Security Program for aggregated category 1 or category 2 quantity of radioactive material (applicable to protection of aggregated quantity of radioactive material prior to fuel on site)	10 CFR 37.41(a)(1)	Prior to receipt of fuel onsite (protected area)	13.6
Fitness for Duty (FFD) Program for Construction (workers and first-line supervisors)	10 CFR 26.3(c) 10 CFR 26.4(f)	Prior to initiating 10 CFR Part 26 construction activities	13.7
Fitness for Duty (FFD) Program for Construction (management and oversight personnel)	10 CFR 26.3(c) 10 CFR 26.4(e)	Prior to initiating 10 CFR Part 26 construction activities	13.7
FFD Program for Security Personnel	10 CFR 26.4(e)(1)	Prior to initiating 10 CFR Part 26 construction activities	13.7
	10 CFR 26.4(a)(5)	Prior to the earlier of: A. Receipt of fuel onsite or B. Establishment of a protected area	
FFD Program for FFD Program personnel	10 CFR 26.3(c); 10 CFR 26.4(g)	Prior to initiating 10 CFR Part 26 construction activities	13.7

Table 13.4-1 Operational Programs and Implementation Milestones (Sheet 5 of 6)

Program Title	Source Requirement	Implementation Milestone	FSAR Section
FFD Program for persons required to physically report to the Technical Support Center (TSC) or Emergency Operations Facility (EOF)	10 CFR 26.4(c)	Prior to the conduct of the first full-participation emergency preparedness exercise under 10 CFR 50, Appendix E, Section F.2.a	13.7
FFD Program for Operation	10 CFR 26.4(a) and (b)	Prior to the earlier of: A. Receipt of fuel onsite or B. Establishment of a protected area	13.7
Quality Assurance Program – Design and Construction	10 CFR 50.34(a)(7) 10 CFR 50.54(a) 10 CFR 50.55(f) 10 CFR 50, Appendix A (GDC 1) 10 CFR 50, Appendix B 10 CFR 50.55a	Construction Permit	17.1
Quality Assurance Program – Operations	10 CFR 50.34(b)(6)(ii) 10 CFR 50.34(f)(3)(ii) 10 CFR 50.34(f)(3)(iii) 10 CFR 50.54(a) 10 CFR 50, Appendix A (GDC 1) 10 CFR 50, Appendix B	Fuel Load	17.5
Reliability Assurance Program (RAP)	10 CFR 50.34(h)	Fuel Load	17.4
Maintenance Rule	10 CFR 50.65	Fuel Load	17.6
Initial Test Program	10 CFR 50.34(b)(6)(iii)	OLA Submittal	14.2

**Table 13.4-1 Operational Programs and Implementation Milestones
(Sheet 6 of 6)**

Program Title	Source Requirement	Implementation Milestone	FSAR Section
Special Nuclear Material Control and Accounting Program	10 CFR 74, Subpart B (74.11 - 74.19, excl. 74.17)	Fuel Receipt	12.5 13.2 13.3 13.6 13.7

13.5 Plant Procedures

Design, construction, and plant operations activities are performed in accordance with written and approved procedures. Procedural controls for design and construction activities are described in Section 17.5. The guidance in Regulatory Guide 1.33 and ANSI/ANS-3.2, Appendix A (Reference 13.5-1) is used to develop a list of quality-related administrative control procedures. The level of detail in a procedure is dependent on the complexity of the task and considers the experience, education, and training of the users and the consequences of errors.

Procedures are prepared by the responsible organization or other knowledgeable staff personnel. The responsible manager ensures that required reviews and approvals are completed before authorizing the use of any site procedure. Expectations for procedure format, responsibilities, use, adherence, record retention and auditing are documented and communicated to facility personnel. Descriptions of the CRN-1 operational phase procedures will be provided with the OLA.

13.5.1 Administrative Procedures

Administrative control procedures address organizational responsibilities and implementation methods for operational programs. Administrative procedures administrative documents ensure that the requirements of regulatory and license commitments are implemented.

The administrative procedures that prescribe the general conduct of operation satisfy the requirements of 10 CFR 50.54(i-m) and ensure the proper manipulation of controls by licensed operators and senior operators. These procedures also provide for assignment of the duties and responsibilities of shift personnel, such as shift relief and turnover, limitations on working hours, independent verification of the correct performance of operating activities, feedback of operating experience, housekeeping, communications techniques, and control room access.

The operational experience and knowledge gained from the successful and safe startup of previous BWR plants and other reactor types have been factored into the design of BWRX-300 systems and equipment. In addition, prior industry experience is reviewed and evaluated during various project phases such as, planning, design, and project execution (e.g., construction, testing and startup) activities. Sources of industry experience for CRN-1 include, but are not limited to, TVA nuclear and non-nuclear plants, advanced reactors, and industry group (e.g., NEI, INPO, Construction Industry Institute). Industry experience is collected, screened, and documented.

Components that perform a safety function are listed in Table 3A-1. Operating procedures will describe the processes for ensuring that vendor technical information is updated in plant documentation to support plant operations, maintenance, and test activities. Administrative controls governing crane operations, including the qualification requirements for crane operators performing fuel handling activities over fuel pools are described in Section 9.1.

13.5.2 Operating and Maintenance Procedures

13.5.2.1 Operating and Emergency Operating Procedures

Plant operating and maintenance procedures ensure that routine operating, off-normal, and emergency operating activities are conducted in a safe manner.

Descriptions of CRN-1 procedures and milestone schedule for completion will be provided in the FSAR consistent with 10 CFR 50.34(b)(6)(vi).

13.5.2.2 Maintenance and Other Operating Procedures

Detailed descriptions of CRN-1 procedures and milestone schedule for completion will be provided in the FSAR consistent with 10 CFR 50.34(b)(6)(vi).

13.5.3 References

13.5-1 ANSI/ANS-3.2, “Managerial, Administrative, and Quality Assurance Controls for the Operational Phase of Nuclear Power Plants,” American National Standards Institute/American Nuclear Society, 2012.

13.6 Physical Security

Section 13.6 of the referenced ESPA SSAR is incorporated by reference with the following supplements:

Replace the first paragraph with the following:

In accordance with 10 CFR 100.21(f), the CRN Site characteristics are such that adequate security plans and measures can be developed pursuant to 10 CFR Part 73. 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.

Replace the first sentence in the seventh paragraph with the following:

The geological and seismology characteristics for the CRN Site have been determined to be suitable for the seismic design of a GVH BWRX-300 small modular reactor.

Replace the first and second sentence in the eighth paragraph with the following:

Planned culverts or unattended openings that could extend from the outside to the inside of the designated Protected Area will be evaluated for adequate intrusion protection on a case-by-case basis. The site-specific security assessment to be conducted for the OLA will identify and describe the specific security design features implemented to address planned culverts or unattended openings.

Add the following text after the eighth paragraph:

There will be a construction access control system implemented before construction and installation of security related Structures, Systems, and Components (SSCs) with increased measures following installation. The construction access program will provide for transition of

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procedures to operational authorization measures of 10 CFR 73.56 and 10 CFR 73.57. The program will require that a plan is developed to ensure full compliance with these regulations before a protected area is declared containing new fuel (e.g., consideration of background investigations for individuals needing unescorted access in accordance with 10 CFR 73.56(d), behavioral observation requirements of 10 CFR 73.56(f) are met, and requirements in 10 CFR 73.56(i) are met for maintaining unescorted access).

Prior to the initiation of onsite work related to installation of security-related SSCs, the construction organization will develop a construction security plan. The construction security plan will be reviewed and approved by TVA before work commences on onsite security-related SSCs.

The construction security plan will be implemented to protect security-related SSCs as well as measures for transitioning to the operational security requirements and the physical protection program under 10 CFR 73.55. The program will deter and detect malicious acts directed against security-related SSCs until all construction activities have ended and the transition to 10 CFR 73.55 requirements is complete.

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 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.

Milestones for implementation of the security plan (PSP, T&Q Plan, and Safeguards Contingency Plan) and the cyber security plan are described in Table 13.4-1.

13.7 Fitness for Duty

The Fitness for Duty (FFD) Program is implemented and maintained in multiple and progressive phases dependent on the activities, duties, or access afforded to certain individuals at the construction site. In general, two different phases of the FFD program will be implemented: a construction phase FFD program and an operating phase FFD program. TVA will maintain responsibility for the FFD program and has oversight of any duties that are delegated to selected contractors. The construction and operating phase programs are implemented as identified in Table 13.4-1 requirements.

The construction phase FFD program is consistent with NEI 06-06 (Reference 13.7-1), as endorsed in Regulatory Guide 5.84, and meets the applicable requirements of 10 CFR Part 26. The construction phase FFD program will meet or exceed the requirements in NEI 06-06, and any exceptions will be documented in Chapter 1. NEI 06-06 applies to persons constructing or directing the construction of SC1 and security-related SSCs performed onsite where the new reactor will be installed and operated.

The FFD program during the construction phase implements different requirements for different individuals by specific milestones, as required in regulation. This includes:

- Construction personnel (workers and first line supervisors as described in NEI 06-06) will be subject to 10 CFR Part 26, Subpart K.
- Personnel performing duties described in 10 CFR 26.4(e) will be subject to 10 CFR Part 26, Subparts A through H, N, and O.
- FFD program personnel performing duties described in 10 CFR 26.4(g) will be subject to 10 CFR Part 26, Subparts A, B, D-H, N & O.
- Upon the receipt of special nuclear material onsite in the form of fuel assemblies, security personnel will also be subject to 10 CFR Part 26, Subpart I.
- Upon the establishment of a protected area in accordance with 10 CFR 73.55 and receipt of fuel assemblies, the site will transition to the operating phase of the FFD program in accordance with the applicable requirements of 10 CFR Part 26.

The operating phase FFD program will be consistent with TVA's existing FFD program for operating nuclear power plants, which conforms with the applicable requirements of 10 CFR Part 26.

13.7.1 References

13.7-1 NEI 06-06, "Fitness for Duty Program Guidance for New Nuclear Power Plant Construction Sites," Nuclear Energy Institute, Revision 6, May 2012.

APPENDIX 13A DESCRIPTION OF THE EMERGENCY PLAN

Introduction

The regulations set forth in 10 CFR 50.34(a)(10), *Contents of applications; technical information*, requires that, in its Preliminary Safety Analysis Report (PSAR), an application for a construction permit include a discussion of an applicant's preliminary plans for coping with emergencies. Appendix E to 10 CFR Part 50, *Emergency Planning and Preparedness for Production and Utilization Facilities*, establishes requirements for emergency plans to attain an acceptable state of emergency preparedness and to provide reasonable assurance that protective measures can and will be taken to protect the health and safety of workers and the public. Specifically, Section II (The Preliminary Safety Analysis Report) of Appendix E (Emergency Planning and Preparedness for Production and Utilization Facilities) to 10 CFR Part 50 addresses emergency planning-related information required in a PSAR.

Part 5 of TVA's Early Site Permit Application (ESPA) contains the major features of two emergency plans addressing two distinct potential Plume Exposure Pathway (PEP) Emergency Planning Zones (EPZ).

- Part 5A of the ESPA contains the major features of an emergency plan for a PEP EPZ at the site boundary of the CRN Site.
- Part 5B of the ESPA contains the major features of an emergency plan for a PEP EPZ that consists of an area approximately 2 miles in radius from the center point of the Clinch River Nuclear (CRN) Site.

This appendix provides references to the ESPA Part 5A and Part 5B emergency plans, thereby providing preliminary emergency planning-related information required in a PSAR by 10 CFR 50.34(a)(10) and Section II of Appendix E to 10 CFR Part 50 (Appendix E.II).

In the OLA, TVA will:

- Provide detailed information showing the capability of the selected technology to meet the PEP EPZ, as described in ESPA, Part 2, Section 13.3.3, "Emergency Planning Zones"
- Identify the applicable major features emergency plan, based on the accident dose consequences of the selected technology
- Inform the offsite response organizations regarding establishment of the PEP EPZ
- Update the applicable major features emergency plan to reflect the selected technology
- Incorporate the applicable major features emergency plan into a complete and integrated emergency plan

A. Emergency Response Organization

Appendix E.II, Paragraph A: Onsite and offsite organizations for coping with emergencies and the means for notification, in the event of an emergency, of persons assigned to the emergency organizations.

TVA maintains an organization capable of responding to radiological emergencies and the means to notify the emergency response organization of an emergency declaration. Implementing procedures provide the details of the emergency response organization, including reporting relationships.

The normal facility organization is addressed in Section 3.2, “Onsite Organization,” and Appendix A, Section A.3.6, “Onsite Emergency Management Organization,” of the ESPA Part 5A and 5B emergency plans.

The offsite organization is addressed in Section 3.3, “Offsite Organization,” of the ESPA Part 5A and 5B emergency plans.

Emergency notification and activation of the emergency plan is addressed in Section 5.0, “Emergency Notification and Activation of Plan,” of the ESPA Part 5A and 5B emergency plans.

The emergency communications network is designed to provide initial notification of an emergency, and ongoing communications, and is addressed in Section 6.0, “Communications,” of the ESPA Part 5A and 5B emergency plans.

B. Authorities and Responsibilities of Governmental Agencies

Appendix E.II Paragraph B: Contacts and arrangements made and documented with local, State, and Federal governmental agencies with responsibility for coping with emergencies, including identification of the principal agencies.

The necessary arrangements with the local, State, and Federal governmental agencies with emergency planning responsibilities will be obtained, documented, and included in the application for an Operating License. These agreements will present a clear understanding of the emergency support responsibilities of each organization.

The extent of local, State, and Federal governmental agency involvement in an emergency response is dependent upon TVA satisfying the condition in Section 3.F.(5) of ESP-006 and successfully demonstrating that the criteria in SSAR Section 13.3.3 is met.

Agreement letters and contracts for services from outside organizations are addressed in Section 16.5 of the ESPA Part 5A and 5B emergency plans.

Support from Federal agencies is addressed in Section 3.3.25, “Federal Agency Support,” of the ESPA Part 5A and 5B emergency plans.

Support from State emergency response positions is addressed in Section 5.2, “Offsite,” of the ESPA Part 5A and 5B emergency plans.

Support from local services is addressed in Section 3.3.24, “Local Support,” and Appendix A Sections A.4.5.2.3, “Receiving Hospitals and Supplies;” A.4.5.2.5, “Ambulance Service;” and A.4.6, “Additional Local Support,” of the ESPA Part 5A and 5B emergency plans.

Support from the Radiation Emergency Assistance Center/Training Site (REAC/TS) and other local medical facilities is addressed in Section 12.3, “Interagency Assistance from REAC/TS,” of the ESPA Part 5A and 5B emergency plans.

C. Protective Measures

Appendix E.II, Paragraph C: Protective measures to be taken within the site boundary and within each EPZ to protect health and safety in the event of an accident; procedures by which these measures are to be carried out (e.g., in the case of an evacuation, who authorizes the evacuation, how the public is to be notified and instructed, how the evacuation is to be carried out); and the expected response of offsite agencies in the event of an emergency.

Onsite and offsite protective actions are addressed in Section 10.0, “Protective Response,” of the ESPA Part 5A and 5B emergency plans. Sections 10.1, “Onsite Protective Actions for Radiological Events,” and 10.2, “Onsite Protective Actions for Hostile Action Events,” address onsite protective actions. Section 10.3, “Offsite,” addresses offsite protective actions.

An alert and notification system and evacuation time estimate (ETE) would not be applicable to the ESPA Part 5A emergency plan.

The alert and notification system is addressed in Appendix A, Section A.6, “Alert and Notification System,” of the ESPA Part 5B emergency plan. An ETE for a potential 2-mile EPZ was provided as supplemental information to ESPA Part 5B. The analyses reflected in the ETE report did not identify physical characteristics unique to the CRN Site that could pose a significant impediment to the development of emergency plans.

D. First Aid, Decontamination, and Emergency Transportation

Appendix E.II, Paragraph D: Features of the facility to be provided for onsite emergency first aid and decontamination and for emergency transportation of onsite individuals to offsite treatment facilities.

Medical support, including onsite first aid, the classification (injured/contaminated/irradiated) of patients, decontamination, and transport and treatment of injured/contaminated personnel is addressed in Section 12.0, “Medical Support,” and in Appendix A, Section A.4.5.2, “First Aid and Medical Facilities,” of the ESPA Part 5A and 5B emergency plans.

E. Offsite Treatment

Appendix E.II, Paragraph E: Provisions to be made for emergency treatment at offsite facilities of individuals injured as a result of licensed activities.

Medical support, including the treatment of injured/contaminated personnel is addressed in Section 12.0, “Medical Support,” and in Appendix A, Section A.4.5.2, “First Aid and Medical Facilities,” of the ESPA Part 5A and 5B emergency plans.

F. Training

Appendix E.II, Paragraph F: Provisions for a training program for employees of the licensee, including those who are assigned specific authority and responsibility in the event of an emergency, and for other persons who are not employees of the licensee but whose assistance may be needed in the event of a radiological emergency.

Training, including training for onsite and offsite personnel with specific emergency response duties, is addressed in Section 15.0, “Training,” and in Appendix A, Section A.7, “Training and Drills,” of the ESPA Part 5A and 5B emergency plans.

G. Notification and Evacuation

Appendix E.II, Paragraph G: A preliminary analysis that projects the time and means to be employed in the notification of State and local governments and the public in the event of an emergency. A nuclear power plant applicant shall perform a preliminary analysis of the time required to evacuate various sectors and distances within the plume exposure pathway EPZ for transient and permanent populations, noting major impediments to the evacuation or taking of protective actions.

Implementing procedures provided to notify and activate TVA and State emergency staffs are addressed in Section 5.0, “Emergency Notification and Activation of Plan,” of the ESPA Part 5A and 5B emergency plans. Essential emergency positions are covered on a 24-hour-a-day basis by duty personnel.

The communications system used to communicate with Federal, state, and other supporting agencies, is addressed in Section 6.0, “Emergency Notification and Activation of Plan,” of the ESPA Part 5A and 5B emergency plans.

An alert and notification system and ETE would not be applicable to the ESPA Part 5A emergency plan.

The alert and notification system is addressed in Appendix A, Section A.6, Alert and Notification System,” of the Part 5B emergency plan. An ETE for a potential 2-mile EPZ was provided as supplemental information to ESPA Part 5B. The analyses reflected in the ETE report did not identify physical characteristics unique to the CRN Site that could pose a significant impediment to the development of emergency plans.

H. Emergency Equipment and Facilities

Appendix E.II, Paragraph H: A preliminary analysis reflecting the need to include facilities, systems, and methods for identifying the degree of seriousness and potential scope of radiological consequences of emergency situations within and outside the site boundary, including capabilities for dose projection using real-time meteorological information and for dispatch of radiological monitoring teams within the EPZs; and a preliminary analysis reflecting the role of the onsite technical support center and the emergency operations facility in assessing information, recommending protective action, and disseminating information to the public.

Emergency response facilities, equipment, and supplies necessary to respond to an emergency are addressed in Section 8.0, “Emergency Response Facilities, Equipment, and Supplies,” and in Appendix A, Section A.4, “Emergency Response Facilities, Equipment, and Supplies,” of the ESPA Part 5A and 5B emergency plans. Essential emergency positions are covered on a 24-hour-a-day basis by duty personnel.

Accident assessment, dose assessment, monitoring teams, and the ability to obtain meteorological information are addressed in Section 9.0, “Accident Assessment,” of the ESPA Part 5A and 5B emergency plans.

In the event of an EPZ coincident or within the CRN Site boundary (Part 5A emergency plan), there is no need for offsite monitoring teams. TVA would continue to maintain emergency response facilities required to effectively manage the emergency response.

TVA will be capable of performing timely notifications, assessing technical information, recommending protective actions, and disseminating information to the public, as required by regulation, considering any exemptions.

APPENDIX 13B LETTERS OF SUPPORT OF CLINCH RIVER NUCLEAR SITE SMALL MODULAR REACTOR EMERGENCY PREPAREDNESS PROGRAM

TN Department of Military | TEMA

Patrick C. Sheehan
Director

Major General
Warner A. Ross, II
The Adjutant General

May 8, 2024

Mr. Brian J. McDermott
Director, Licensing and Planning
Tennessee Valley Authority, New Nuclear Program
1101 Market St.
Chattanooga, TN 37405

Dear Mr. McDermott,

On behalf of the State of Tennessee, the Tennessee Emergency Management Agency (TEMA) expresses support for off-site emergency planning in conjunction with TVA's Construction Permit Application's (CPA) submittal to the NRC for the Clinch River Small Modular Reactor project (CRN) in Oak Ridge, Tennessee. TEMA intends to actively participate in all initial emergency planning and Radiological Emergency Preparedness activities. We will also ensure that this project meets or exceeds all off-site response standards required by 10 CFR 50.34(a) for preliminary planning, and 10 CFR Part 50, which requires TVA to document contacts and arrangements with state agencies for coping with emergencies.

While we understand that the standards in 10 CFR 50.160 provide optional emergency preparedness requirements for SMRs, TEMA will adapt to and implement any off-site standards established by the NRC. We will participate as requested during submission of the final emergency plan for CRN and final Operating License Application.

It is imperative to the nation and the world that the United States continues to lead the development of nuclear technologies and safer, always available energy, TEMA stands ready to support all required off-site response planning and jurisdictional coordination to promote this project. We fully support the TVA's efforts to develop advanced and new reactor concepts in nuclear power technology and look forward to the positive effects these developments will bring to the lives of Tennesseans.

If you have any further questions, please feel free to contact me at Patrick.Sheehan@tn.gov.

Respectfully,

Patrick C. Sheehan
Director, Tennessee Emergency Management Agency

PCS:th:cs

ANDERSON COUNTY GOVERNMENT

TERRY FRANK
COUNTY MAYOR

March 20, 2024

Brian McDermott
Director, Licensing and Planning
Tennessee Valley Authority
1101 Market Street
Chattanooga, TN 37402

Dear Mr. McDermott,

In December of 2015, I submitted a letter expressing support for the TVA's Early Site Permit application to the Nuclear Regulatory Commissioner for the Clinch River Small Modular Reactor (SMR) project.

As TVA moves into the next phase of the application process, Anderson County would once again like to submit a letter of support. This letter is authorized by a vote of County Commission, meeting in regular session on February 22, 2024.

While we recognize that Anderson County is not the site of the proposed SMR project, we continue to work closely with the Tennessee Department of Emergency Management (TEMA) in emergency planning preparedness exercises and evaluations. In support of TVA's SMR project, we will be a willing and active participant in the future development of emergency plans for the CRN Site, and emergency planning and Radiological Emergency Preparedness exercises and evaluations necessary to meet the needs of all off-site response standards required by regulations applicable to the Clinch River SMR project.

I was very happy to visit the SMR project site recently, along with several of our county commissioners, members of our county economic development board, and our Emergency Management Director. We received an informative and excellent briefing.

We are here to lend support, assistance, planning, and any cross-jurisdictional coordination needed in order to support this very important project.

My best regards,

Mrs. Terry Frank
Anderson County Mayor

100 NORTH MAIN STREET, SUITE 208 • CLINTON, TENNESSEE • 37716
PHONE: (865) 457-6200 •

March 12, 2024

Tennessee Emergency Management Agency
Office of the Deputy Commissioner
3041 Sidco Dr.
Nashville, Tennessee 37204-1502

Subject: Support of TVA's Clinch River Nuclear Site Small Modular Reactor Emergency Preparedness Program

Dear Deputy Commissioner:

The development of Small Modular Nuclear Reactors (SMR's) at TVA's Clinch River Nuclear (CRN) Site is of critical importance to our region's economy and to the future of energy production for our state and country. An essential component of the planning process for the development of SMR's at the CRN site is ensuring emergency preparedness and public safety related to the future operations of reactors on-site and its implications off-site.

Because of its 80-year history that includes the birth of the nuclear industry, Roane County is one of the most nuclear-friendly communities in the world. One of the key reasons for that is the importance placed on safety and preparedness by our local, state, and federal partners across the Oak Ridge Reservation. Looking into a future that includes a rebirth of new nuclear activity on the Reservation, ensuring that we remain prepared for any situation for the safety of our people will always be the highest priority.

With that in mind, please accept this letter as my official request for TEMA's direct involvement in any comprehensive emergency preparedness planning and cross-jurisdictional coordination as TVA continues to move forward with its SMR plans. Without TEMA's involvement and leadership, any planning effort and subsequent response to an incident would be lacking.

Thank you for your consideration,

Wade Creswell
Roane County Executive

Office of the County Executive

865.376. 5578
info@roanecountytn.gov
www.roanecountytn.gov
PO Box 643 Kingston, TN 37763

CITY OF OAK RIDGE TENNESSEE

OFFICE OF THE CITY MANAGER
(865) 425-3550

April 2, 2024

Tennessee Emergency Management Agency
Office of the Deputy Commissioner
3041 Sidco Dr.
Nashville, Tennessee 37204-1502

Subject: Support of TVA's Clinch River Nuclear Site Small Modular Reactor Emergency Preparedness Program

To Whom it May Concern:

On February 24, 2016, the City of Oak Ridge sent a letter of support for the Early Site Permit application for the TVA's proposed Small Modular Reactor in Oak Ridge. The City of Oak Ridge remains committed to our support for the project and TVA's planned submittal, a 10 CFR Part 50 construction permit application to the NRC for the construction of a SMR at the CRN Site.

The City could not be more pleased at this action to implement the latest in nuclear technology. The proposed site provides the convenience of proximity to the Oak Ridge National Laboratory and the efficiency of the Watts Bar nuclear power plant to provide additional resources in the vicinity.

The City of Oak Ridge is prepared to work with Tennessee Emergency Management Agency (TEMA) to support the site. The Oak Ridge Fire Department is one of six fire departments in the State of Tennessee with an ISO rating of 2, the highest of any department. Our ability to respond to hazardous material matters and public emergencies is the subject of continuous training and learning as all of our department members receive hazmat training. We look forward to being a willing partner in planning for the emergency programs that will be needed to support the small modular reactor. We look forward to being included with all exercises and evaluations associated with the Clinch River as chartered by Nureg-0654 and 44-CFR-350.

NRC regulations in 10 CFR 50.34(a) require TVA to include a discussion of preliminary plans for coping with emergencies. Additionally, NRC regulations in Appendix E to 10 CFR Part 50 require TVA to document contacts and arrangements made with State agencies with responsibility for coping with emergencies. Please consider this letter as confirmation of our contact on this project and our support for providing assistance in the area of emergency response if needed.

We look forward to being included as the adjacent local government and will assist as appropriate in any planning efforts to make this project successful.

Sincerely,

Randall W. Hemann
City Manager

Accepted by:

_____ _____
Signature Date

_____ _____
Print Name Title

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LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
CRN-1	Clinch River Nuclear Unit 1
FOAK	First-of-a-Kind
FSAR	Final Safety Analysis Report
GVH	GE Vernova Hitachi Nuclear Energy
ICS	Isolation Condenser System
ITP	Initial Test Program
RG	Regulatory Guide
SC1	Safety Class 1
SC2	Safety Class 2
SC3	Safety Class 3
SME	Subject Matter Expert
SSCs	Structures, Systems, and Components
TRACG	Transient Reactor Analysis Code General Electric
TVA	Tennessee Valley Authority
USNRC	U.S. Nuclear Regulatory Commission

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CHAPTER 14 INITIAL TEST PROGRAM

The purpose of this Chapter is to describe the test program that is performed during initial startup of Clinch River Nuclear Unit 1 (CRN-1). The program describes the manner in which the testing is performed, controlled, and documented for the 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) is composed of phases categorized as construction, preoperational and startup tests. Construction tests serve as prerequisites for preoperational tests. Preoperational tests are those tests conducted prior to 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 capability of the integrated plant to meet required performance requirements.

The ITP addresses Structures, Systems, and Components (SSCs) and design features for both the nuclear portion of the facility and the balance of plant. 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 classification is in Table 3A-1. The SSCs described in Table 3A-1 as SC1, SC2, or SC3, together with the ITP descriptions herein, provides a high-level description of the types of testing to be conducted by ITP. A graded approach is used in the development and implementation of the ITP to ensure greater attention is given to SSCs based on safety class. With this graded approach, documentation (such as procedures and records) associated with testing are commensurate with the importance to safety of the item being tested.

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 in a safe manner
- 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 also setting 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 downstream activities do not invalidate previous testing results.

Preoperational testing is conducted after construction testing and prior to fuel loading to demonstrate plant systems are capable of operating in a safe and efficient 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 the performance of equipment and systems
- Provide baseline test and operating data on equipment and systems 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 select off normal operations. Following the completion of necessary prerequisites, fuel loading commences, making the unit an operating power plant. Startup testing consists of five test plateaus, described below. Tests are commonly repeated throughout the plateaus to verify system performance or response at increasing temperatures/power levels:

- Open Vessel Testing – includes baseline data taken prior to insertion of fuel into the reactor vessel, the fuel load process through core fully loaded, and includes shutdown margin verifications.
- Heat-up Testing – begins after the reactor vessel is fully assembled and hydrostatically or leak tested, and includes reactor startup to power less than 5% for heat-up of the reactor to rated temperature and pressure, with testing, such as thermal expansion, temperature monitoring, and low-power nuclear instrument checks and process controller tuning checks performed at various points during the heat-up as well as system performance tests at various points along the heat-up and pressurization curve.
- Low-Power Testing – includes testing between approximately 5% and 25% power and includes putting the turbine generator online for the first time:
    - In addition to system performance testing and control system tuning, transient event testing is included in this test plateau.

- Mid-Power Testing – includes testing between 50% and 75% power of system performance and control system tuning and typically includes transient event testing.
- High-Power Testing – includes startup testing at (+0, -5%) 100% power 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. For instance, some tests take data at power levels that are between the test plateau definitions. These 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 for TVA New Nuclear 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 the preoperational and startup testing of a BWRX-300 plant. The startup administrative manual provides minimum requirements and recommendations for the BWRX-300 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 carry out the test program, methods and plans for providing the necessary staff, description of the staff responsibilities, authorities, and personnel qualifications for conducting the ITP.
- Delineate the development, review, and approval of test procedures per 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 the test procedures.
- Require the development of plant operating and emergency procedures prior to 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 to be used to control the resolution of test failures, deficiencies and oversights discovered in the ITP. Address the control of any plant modifications required to resolve these deficiencies.

RG 1.68 specifies the criteria for determining what SSCs and design features are required to be tested in accordance with the requirements therein.

14.1.1.4 Test Procedures

Testing during preoperational and startup testing is conducted using detailed, step-by-step written procedures to control the conduct of each test. These specifically include safety precautions and limits as needed for the test to supplement those in the normal operating procedure. Such test procedures:

- Specify testing prerequisites
- Describe desired initial conditions
- Include appropriate methods to direct and control test performance (including the sequencing of testing)
- Specify acceptance criteria by which the test is to be evaluated
- Provide for or specify the format by which data or observations are to be recorded

The procedures are developed and reviewed by qualified personnel with appropriate technical backgrounds and experience. This includes the participation of principal design organizations to establish test performance requirements and acceptance criteria. Consideration of test requirements is incorporated into the BWRX-300 design early in the design process. Preoperational and startup test specifications containing testing objectives and acceptance criteria are used to generate test procedures. Test specifications consider potentially hazardous conditions and various operating limitations during testing, such as the potential for large thermal gradients that may exceed stress limits or cause excessive fatigue usage due to 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 group conducts preoperational and startup testing in accordance with the startup administrative manual. This manual contains the administrative processes and requirements that govern the activities of the ITP group and their interface with other organizations. The startup administrative manual defines the specific format and content of preoperational and startup test procedures, as well as the review and approval process. The startup administrative manual also specifies the process for review and approval of test results and for resolution of failures to meet acceptance criteria and of other operational problems or design deficiencies noted. It also specifies that site-specific controls be in place to assure the as-tested status of each system is known and that track modifications, including retest requirements, deemed necessary for systems undergoing or already having completed specified testing. Additionally, the startup administrative manual requires that site-specific procedures be developed to govern the qualifications and responsibilities of the different positions within the ITP group.

14.1.1.6 Review, Evaluation, and Approval of Test Results

Individual preoperational and startup test results are evaluated and reviewed, including any test exceptions. Test results, including final resolutions, are then reviewed, and approved by designated ITP group supervisory personnel. Final approval is obtained as defined in the startup

administrative manual, including the 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 is specific to FOAK tests performed by the ITP group during the pre-operational and startup testing phases that are being reviewed for the first time by the U.S. Nuclear Regulatory Commission (USNRC). Methods of analysis (e.g. Transient Reactor Analysis Code General Electric (TRACG)) and validation of construction techniques (e.g. steel-plate composite containment vessel and reactor building structural design) are described elsewhere in the Preliminary Safety Analysis Report as required or in 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 Application, preoperational and startup tests may be added to this list.

14.1.3 Conformance of Test Programs with Regulatory Guides

The development of the ITP conforms to RG 1.68. Additional regulatory guides also apply to the ITP as described in RG 1.68 which are considered during test development. Conformance to these additional RGs is addressed in the Final Safety Analysis Report (FSAR) when the associated system test descriptions have been developed. Example additional RGs 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

TVA operates multiple nuclear reactors in its current Fleet: Browns Ferry, Watts Bar, and Sequoyah. Consequently, TVA has past experience in constructing, testing, and operating these nuclear reactors. Lessons learned may also be utilized from recent non-TVA facilities via

INPO 19-002 construction experience reporting (such as the AP1000 startup at Vogtle Units 3 & 4).

Operating and testing experience is utilized in the development and implementation of the BWRX-300 testing program. Lessons learned from previous plant designs, construction, testing, and operation are incorporated into test specification development as well as test procedures.

TVA intends to have a programmatic approach versus a one-time study due to new BWRX-300 construction and operating experience anticipated to become available. Such a program is to be developed and implemented prior to submittal of the FSAR. TVA and GE Vernova Hitachi Nuclear Energy (GVH) are to maintain communication as new construction and operating experience is learned from other possible BWRX-300 plants (either constructed before or concurrently with CRN-1). Ontario Power Generation already submitted a license to construct with the Canadian Nuclear Safety Commission regarding a future BWRX-300.

14.1.5 Test Program Schedule

The schedule, relative to the initial fuel load date, for conducting each major phase of the ITP is described and is dynamic in nature. The test schedule is still under development; if significantly shorter time periods than nine months for preoperational testing and three months for startup testing are used, a justification is to be provided. To allow for USNRC inspection, test procedure preparation is scheduled such that applicable approved procedures are available approximately 60 days prior to their intended use.

A detailed testing schedule is to be developed and made available for regulatory awareness. The schedule may be updated and continually optimized to reflect actual progress and subsequent revised projections. Implementation milestones for the ITP, along with plans for hiring and training of plant personnel and the development of plant operating and emergency procedures, are provided in Section 13.2, Section 13.5, and Table 13.4-1.

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 in the performance of tests. The use of these procedures provides the following:

- Prove 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 testing procedure may use a combination of references to operating, emergency, or abnormal procedures or repeat a series of steps from the procedure in order to accomplish the above goals while efficiently performing the specified testing.

14.1.7 Augmenting Staff During the Test Program

The organization and staffing of TVA's ITP is discussed in Subsection 14.1.1.2.

To support execution of the initial testing program, TVA requires various ITP technical staff and plant operating staff in place. TVA provides training as required for these positions. The overall ITP schedule and the schedules for development of test procedures and plant operating procedures are used to determine hiring needs, hiring dates, and any contracted support if needed by TVA. Milestones for development of associated programs (Preservice Inspection, Preservice Testing, etc.) are also discussed in Section 13.4.

Subject Matter Experts (SMEs) from outside organizations, such as from various equipment suppliers/vendors, are contracted by TVA as needed to support testing.

Augmenting staff from TVA's existing nuclear Fleet (Browns Ferry, Watts Bar, Sequoyah) to support preoperational and startup testing will be considered pending Fleet personnel availability and outage schedules. SMEs and other resources within the TVA Fleet are available for consultation as needed by the CRN-1 staff during preoperational and startup testing.

Table 14.1-1 Objectives of First-of-a-Kind Testing

System/Component Design Description	Test Objective
Isolation Condenser System – The ICS consists of three independent trains, each containing a heat exchanger or isolation condenser that is submerged in a pool of water and is connected to the reactor vessel by steam supply and condensate return piping. When in operation, the ICS removes heat from the reactor coolant and rejects it to the environment. This provides overpressure protection by maintaining the reactor vessel below design limits. The use of ICS for overpressure protection instead of traditional safety or relief valves is a new design feature that has not been previously demonstrated in an operating nuclear station.	Full scale prototype testing has been accomplished for the heat exchanger and return to the reactor vessel. During the startup phase, a FOAK test places the ICS in service and 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 ICS is in operation, the condensate return line to the reactor vessel is open. The subcooled condensate that is stored in the ICS during the standby state enters the reactor vessel chimney providing additional inventory while also condensing steam and increasing the fluid density in the chimney. The placement of the ICS condensate returns in the chimney above the core, instead of the downcomer as in previous designs, has 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 Transient Reactor Analysis Code General Electric (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 BWRX-300 core flow when ICS is operating based on the measured temperatures at the core inlet.

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LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
3D	Three-Dimensional
ABWR	Advanced Boiling Water Reactor
AC	Alternating Current
ACRW	All Control Rod Withdrawal at Power
ADAMS	Agencywide Documents Access and Management System
ALWR	Advanced Light Water Reactor
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrence
API	Application Programming Interface
ASME	American Society of Mechanical Engineers
AST	Alternative Source Term
ATLM	Automatic Thermal Limit Monitor
ATS	Anticipatory Trip System
ATWS	Anticipated Transient Without Scram
BDBA	Beyond Design Basis Accident
BE	Basic Event
BIS	Boron Injection System
BL-AOO	Baseline Abnormal Operational Occurrence
BL-DBA	Baseline Design Basis Accident
BL-DSA	Baseline Deterministic Safety Analysis
BOC	Beginning of Cycle
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
CEDE	Committed Effective Dose Equivalent
CEM	Containment Evaluation Method
CET	Containment Event Tree
CFR	Code of Federal Regulations
CIV	Containment Isolation Valve
CN-DBA	Conservative Design Basis Accident

LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
CN-DSA	Conservative Deterministic Safety Analysis
COLR	Core Operating Limits Report
CPR	Critical Power Ratio
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
CRDM	Control Rod Drive Mechanism
CRN	Clinch River Nuclear
CSAU	Code Scaling, Applicability, and Uncertainty
CUW	Reactor Water Cleanup System
DBA	Design Basis Accident
DC	Direct Current
DCF	Dose Conversion Factor
DCIS	Distributed Control and Information System
DDE	Deep Dose Equivalent
DEC	Design Extension Condition
DF	Decontamination Factor
D-in-D	Defense-in-Depth
DL	Defense Line
DR	Decay Ratio
DSA	Deterministic Safety Analysis
EAB	Exclusion Area Boundary
ECCS	Emergency Core Cooling System
EE	External Events
EFCV	Excess Flow Check Valve
EHE	External Hazard Evaluation
EOC	End of Cycle
EOP	Emergency Operating Procedure
EOR	End of Rated Power
EPRI	Electric Power Research Institute
ES	Event Sequence
ESBWR	Economic Simplified Boiling Water Reactor
ESF	Engineered Safety Feature
EX-DEC	Extended Design Extension Condition
EX-DSA	Extended Deterministic Safety Analysis
FDF	Fuel Damage Frequency
FFA	Functional Failure Analysis
FHA	Fuel Handling Accident

LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
FLEX	Diverse and Flexible Coping Strategy
FMCRD	Fine Motion Control Rod Drive
FMEA	Failure Modes and Effects Analysis
FSAR	Final Safety Analysis Report
FSF	Fundamental Safety Function
FTREX	Fault Tree Reliability Evaluation Expert
FV	Fussell-Vesely
FW	Feedwater
FWCIV	Feedwater Containment Isolation Valve
FWFI	Feedwater Flow Increase
FWLB	Feedwater Line Break
FWPT	Feedwater Pump Trip
FWRIV	Feedwater Reactor Isolation Valve
GDC	General Design Criteria
GNF	Global Nuclear Fuel
GVH	GE Vernova Hitachi Nuclear Energy
HCU	Hydraulic Control Unit
HEE	Human Error Event
HELB	High Energy Line Break
HEP	Human Error Probability
HEPA	High Efficiency Particulate Air
HFEA	Human Failure Event Analysis
HRA	Human Reliability Analysis
HVS	Heating, Ventilation, and Cooling System
I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency
IC	Isolation Condenser
ICLB	Isolation Condenser Line Break
ICRW	Inadvertent Control Rod Withdrawal at Power - Single Rod
ICS	Isolation Condenser System
IE	Initiating Event
IHE	Internal Hazard Evaluation
II	Inventory Increase
ILB	Instrument Line Break
IR	Inventory Reduction
ISLOCA	Interfacing System LOCA
LFWH	Loss of Feedwater Heating

LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
LHGR	Linear Heat Generation Rate
LOCA	Loss-of-Coolant Accident
LOCV	Loss of Condenser Vacuum
LOPP	Loss-of-Preferred Power
LPSD	Low-Power and Shutdown
LPZ	Low Population Zone
LRF	Large Release Frequency
LR-TT	Load Rejection - Turbine Trip
LWR	Light Water Reactor
MAAP	Modular Accident Analysis Program
MCC	Motor Control Center
MCPR	Minimum Critical Power Ratio
MCR	Main Control Room
MCRE	Main Control Room Envelope
MCS	Minimal Cutset
MOC	Middle of Cycle
MOV	Motor Operated Valve
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
NBR	Nuclear Boiler Rated
NBS	Nuclear Boiler System
NEA	Nuclear Energy Agency
NUREG	Nuclear Regulatory Report
OLMCPR	Operating Limit Minimum Critical Power Ratio
PAU	Physical Analysis Unit
PCCS	Passive Containment Cooling System
PCS	Primary Containment System
PCT	Peak Cladding Temperature
PCV	Primary Containment Vessel
PCW	Plant Cooling Water System
PDC	Principal Design Criteria
PDS	Plant Damage States
PI	Pressure Increase

LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
PIE	Postulated Initiating Event
PIRT	Phenomenon Identification and Ranking Table
POS	Plant Operating State
PPS	Plant Pneumatic System
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
PSAR	Preliminary Safety Analysis Report
PSF	Performance Shaping Factors
RAW	Risk Achievement Worth
RB	Reactor Building
RCPB	Reactor Coolant Pressure Boundary
RG	Regulatory Guide
RI	Reactivity Increase
RIV	Reactor Isolation Valve
RLC	Reactor Level Control
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
SC1	Safety Class 1
SC2	Safety Class 2
SC3	Safety Class 3
SCR	Secondary Control Room
SCRRI	Selected Control Rod Run-in
SDC	Shutdown Cooling System
SMR	Small Modular Reactor
SPSA	Seismic Probabilistic Safety Assessment
SRM	Staff Requirements Memoranda
SRP	Standard Review Plan
SSAR	Site Safety Analysis Report
SSC	Structures, Systems, and Components
STP	Simulated Thermal Power

LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
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
TS	Technical Specifications
TSV	Turbine Stop Valve
USNRC	U.S. Nuclear Regulatory Commission
ΔCPR/ICPR	Delta Critical Power Ratio Over Initial Critical Power Ratio

LIST OF EFFECTIVE SECTIONS

CHAPTER 15 SAFETY ANALYSES

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CHAPTER 15 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 U.S. 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 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)
- Source Term Analysis (Subsection 15.6.6)
- 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 (FFA)

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 Design Extension Conditions 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 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,” GE Vernova Hitachi Nuclear Energy Americas, LLC, Revision 2, 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 the 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,” GE Vernova Hitachi Nuclear Energy Americas, LLC, Revision 2, March 2026.

15.2-2 NEDC-33910P-A, “BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection,” GE Hitachi Nuclear Energy Americas, LLC, Revision 2, 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
(Sheet 1 of 4)

DSA Layer / Event Category	Event Sequence ID	Corresponding DSA Subsection 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	Subsection 15.5.3.2.1
	Closure of One MSRIV 1MSRIVC; PI-1MSRIVC_BL-AOO	Subsection 15.5.3.2.2
	Loss of Condenser Vacuum (LOCV) PI-LOCV_BL-AOO	Subsection 15.5.3.2.3
	Loss-of-Preferred Power (LOPP) PI-LOPP_BL-AOO	Subsection 15.5.3.2.4

Table 15.2-2 Bounding Events Transient (Non-LOCA) and LOCA
(Sheet 2 of 4)

DSA Layer / Event Category	Event Sequence ID	Corresponding DSA Subsection 15.5.3 Through Subsection 15.5.9 Event Summary Results
CN-DBA	**Generator Load Rejection or Turbine Trip** LR-TT, Passive CCF DL2 Technology Platform (CCF-DL2); PI-LR-TT_CCF-DL2_CN-DBA	Subsection 15.5.4.2.1
	Loss-of-Preferred Power LOPP, Passive CCF DL2 Technology Platform (CCF-DL2); PI-LOPP_CCF-DL2_CN-DBA	Subsection 15.5.4.2.2
	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	Subsection 15.5.4.2.3
	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 15.5.4.2.4
EX-DEC	**Closure of One MSRIV** PI-1MSRIVC_CCF-Hydraulic-Scram_EX-DEC	Subsection 15.5.5.2.1
	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	Subsection 15.5.5.2.2
	Loss of Condenser Vacuum LOCV, CCF-Hydraulic-Scram; PI-LOCV_CCF-Hydraulic-Scram_EX-DEC	Subsection 15.5.5.2.3
	Loss-of-Preferred Power LOPP, CCF-Hydraulic-Scram; PI-LOPP_CCF-Hydraulic-Scram_EX-DEC	Subsection 15.5.5.2.4

Table 15.2-2 Bounding Events Transient (Non-LOCA) and LOCA
(Sheet 3 of 4)

DSA Layer / Event Category	Event Sequence ID	Corresponding DSA Subsection 15.5.3 Through Subsection 15.5.9 Event Summary Results
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

Table 15.2-2 Bounding Events Transient (Non-LOCA) and LOCA
(Sheet 4 of 4)

DSA Layer / Event Category	Event Sequence ID	Corresponding DSA Subsection 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	Subsection 15.5.4.4.1
	Reactor Pressure Vessel Pressure Control Open CCF-RPCO, Passive CCF DL2 Technology Platform (CCF-DL2); IR-CCF-RPCO_CCF-DL2_CN-DBA	Subsection 15.5.4.4.2
EX-DEC	**Feedwater Isolation** FW Isolation (CCF-FWI-DL3); IR-CCF-FWI-DL3_EX-DEC	Subsection 15.5.5.4
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 CRN 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 USNRC “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,” GE Hitachi Nuclear Energy Americas, LLC, Revision 1, August 2017.

15.3-2 NEDE-33005P-A, “TRACG Application for Emergency Core Cooling Systems / Loss-of-Coolant-Accident Analyses for BWR/2-6,” GE Hitachi Nuclear Energy Americas, LLC, Revision 2, May 2018.

15.3-3 NEDE-33885P-A, “GNF CRDA Application Methodology,” Global Nuclear Fuel -Americas, LLC, Revision 1, March 2020.

Table 15.3-1 Anticipated Operational Occurrence Deterministic Safety Analysis Acceptance Criteria
(Sheet 1 of 2)

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.

**Table 15.3-1 Anticipated Operational Occurrence Deterministic Safety Analysis Acceptance Criteria
(Sheet 2 of 2)**

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 (Sheet 1 of 4)

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.

Table 15.3-2 Design Basis Accident Deterministic Safety Analysis Acceptance Criteria (Sheet 2 of 4)

Fission Product Barrier or Fundamental Safety Function	Qualitative Acceptance Criteria	Quantitative Acceptance Criteria
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.
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.

Table 15.3-2 Design Basis Accident Deterministic Safety Analysis Acceptance Criteria (Sheet 3 of 4)

Fission Product Barrier or Fundamental Safety Function	Qualitative Acceptance Criteria	Quantitative Acceptance 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.

Table 15.3-2 Design Basis Accident Deterministic Safety Analysis Acceptance Criteria (Sheet 4 of 4)

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.

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 Section 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 or 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,” code (Reference 15.5-8) 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 BWR 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 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 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 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 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 Table 15.5-3 through Table 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 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 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 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. 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 Figure 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 pressure increases 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 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 FWPT event is presented on Figure 15.5-34 through Figure 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 BWRX-300 non-LOCA and LOCA PIEs. Subsection 15.5.4.1 through Subsection 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 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 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 Table 15.5-3 through Table 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 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 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 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 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 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 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 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 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

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 prevent 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 IC 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, 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, 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 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 a Main Steam Line Break (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 Figure 15.5-161 through Figure 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

Table 15.5-24 and Table 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 GNF2 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 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 LOCV is presented on Figure 15.5-131 through Figure 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 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 loss of preferred power with CCF hydraulic scram failure is shown on Figure 15.5-137 through Figure 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 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 and MOC cycle exposure conditions. End of Cycle (EOD) exposure is not run because all control rods are fully withdrawn.

Results

The simulated All Control Rod Withdrawal at Power (ACRW) is presented on Figure 15.5-143 through Figure 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 control 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 Table 15.5-3 through Table 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, fuel grapple, mast, and head fall on top of the core or the fuel storage 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 fuel 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, fuel 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 $(0.01)/\upsilon$ 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:

$$E_1 = \frac{h_{drop} \times (W_{fuel} \times W_{mast})}{2}$$ 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 15.5-2 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 15.5-1) 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:

$$\frac{(50\%)(4799.2 \text{ kgf-m})(0.4997)}{37.503 \text{ kgf-m/rod}} = 31.97 \text{ rods or } 32 \text{ rods}$$

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:

$$E_2 = 50\% \times \left(h_{fuel} \times W_{mast} + \frac{1}{2} \times h_{fuel} \times W_{fuel} \right)$$ 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 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 Table 15.5-38a and Table 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 Clinch River Nuclear (CRN) Unit 1 site in the “Clinch River Nuclear Site Early Site Permit (ESP) Application Part 2, Site Safety Analysis Report,” (SSAR), (Reference 15.5-17) 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)

Location	0-2 hours	0-8 hours	8-24 hours	1-4 days	4-30 days
EAB	4.96E-03	-	-	-	-
LPZ	-	3.10E-04	2.26E-04	1.14E-04	4.30E-05

Transport to the Main Control Room Envelope (MCRE) from the BWRX-300 RB is simulated using onsite atmospheric dispersion factors (χ/Qs) calculated using CRN 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)

0-2 Hours	2-8 Hours	8-24 Hours	1-4 Days	4-30 Days
1.51E-03	1.06E-03	4.95E-04	4.68E-04	4.21E-04

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 CRN 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, etc.) 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 CRN Unit 1 site in the ESPA SSAR. The PAVAN code was applied along with the CRN site-specific data to develop offsite dispersion factors. After approval of the CRN Early Site Permit, RG 1.249 guidance became available endorsing use of the ARCON code for calculation of EAB and the outer boundary of the LPZ dispersion factors at distances less than 1,200 meters from release source structures. Although the application of the ARCON code would be expected to significantly reduce conservatism associated with the PAVAN code at near-field distances, the previously reviewed CRN Early Site Permit values for short-term offsite atmospheric dispersion factors (s/m^3) are applied in the analyses.

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

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 CRN Unit 1 site in the ESPA SSAR. The offsite atmospheric dispersion factors (s/m^3) used in the analysis are:

Offsite χ/Q (s/m^3)

Location	0-2 Hours	0-8 Hours	8-24 Hours	1-4 Days	4-30 Days
EAB	4.96E-03[1]				
LPZ	-	3.10E-04	2.26E-04	1.14E-04	4.30E-05

1. The 0-2 hour EAB χ/Q value is used for full duration of the event.

Transport to the MCRE from the TB is simulated using onsite atmospheric dispersion factors χ/Q values calculated using CRN Unit 1 site meteorological data collected from June 1, 2011, through May 31, 2013, at the CRN Site in accordance with RG 1.23. 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 area source release to the closest MCR normal intake:

Main Control Room Envelope χ/Q (s/m^3)

0-2 Hours	2-8 Hours	8-24 Hours	1-4 Days	4-30 Days
9.04E-04	6.59E-04	2.34E-04	2.54E-04	2.22E-04

Results

The results are listed in Table 15.7-5 for comparison to the radiological acceptance criteria.

The MSLB accident dose consequences satisfy the CRN 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

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 CRN Unit 1 site in the ESPA SSAR. The offsite atmospheric dispersion factors (s/m^3) used in the analysis are:

Offsite χ/Q (s/m^3)

Location	0-2 Hours	0-8 Hours	8-24 Hours	1-4 Days	4-30 Days
EAB	4.96E-03[1]				
LPZ	-	3.10E-04	2.26E-04	1.14E-04	4.30E-05

1. The 0-2 hour EAB χ/Q value is used for full duration of the event.

Transport to the MCRE from the TB is simulated using onsite atmospheric dispersion factors χ/Q values calculated using CRN site meteorological data collected from June 1, 2011, through May 31, 2013, at the Clinch River Nuclear Site in accordance with RG 1.23. 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)

0-2 Hours	2-8 Hours	8-24 Hours	1-4 Days	4-30 Days
9.04E-04	6.59E-04	2.34E-04	2.54E-04	2.22E-04

Results

The results are listed in Table 15.7-6 for comparison to the radiological acceptance criteria.

The FWLB accident dose consequences satisfy the CRN 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

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 CRN Unit 1 site in the ESPA SSAR. The offsite atmospheric dispersion factors (s/m^3) used in the analysis are:

Offsite χ/Q (s/m^3)

Location	0-2 Hours	0-8 Hours	8-24 Hours	1-4 Days	4-30 Days
EAB	4.96E-03[1]				
LPZ	-	3.10E-04	2.26E-04	1.14E-04	4.30E-05

1. The 0-2 hour EAB χ/Q value is used for full duration of the event.

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 CRN Unit 1 site meteorological data collected from June 1, 2011, through May 31, 2013, at the CRN Site in accordance with RG 1.23. 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)

0-2 Hours	2-8 Hours	8-24 Hours	1-4 Days	4-30 Days
6.99E-03	4.31E-03	1.71E-03	1.64E-03	1.28E-03

Results

The results are listed in Table 15.7-7 for comparison to the radiological acceptance criteria.

The ICLB accident dose consequences satisfy the CRN Unit 1 site DBA dose acceptance criteria with considerable margin as shown in Table 15.7-7.

15.5.9.1.5 Small Breaks Outside Containment

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 operation.

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 CRN Unit 1 site in the ESPA SSAR. The offsite atmospheric dispersion factors (s/m^3) used in the analysis are:

Offsite χ/Q (s/m^3)

Location	0-2 Hours	0-8 Hours	8-24 Hours	1-4 Days	4-30 Days
EAB	4.96E-03[1]				
LPZ	-	3.10E-04	2.26E-04	1.14E-04	4.30E-05

Note: The maximum EAB χ/Q value is used for full duration of the event to ensure the worst 2-hour dose consequence.

Transport to the MCRE from the BWRX-300 RB is simulated using onsite atmospheric dispersion factors (χ/Qs) calculated using CRN site meteorological data collected from June 1, 2011, through May 31, 2013, at the CRN Site in accordance with RG 1.23.

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)

0-2 Hours	2-8 Hours	8-24 Hours	1-4 Days	4-30 Days
1.51E-03	1.06E-03	4.95E-04	4.68E-04	4.21E-04

Results

The BWRX-300 ILB dose consequences satisfy the CRN Unit 1 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.

15.5.10 References

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15.5-16 Federal Guidance Report No. 15, “External Exposure to Radionuclides in Air, Water, and Soil,” U.S. Environmental Protection Agency, August 2019.

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15.5-19 NEDC-34270P, “BWRX-300 Stability Analysis,” GE Vernova Hitachi Nuclear Energy Americas, LLC, Revision 1, March 2026.

15.5-20 NEDC-33083P-A, “TRACG Application for ESBWR,” GE-Hitachi Nuclear Energy Americas, LLC, Revision 1, September 2010.

15.5-21 NEDC-33910P-A, “BWRX-300 Reactor Pressure Vessel Isolation and Overpressure Protection,” GE-Hitachi Nuclear Energy Americas, LLC, Revision 2, June 2021.

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 (Sheet 1 of 2)

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

Table 15.5-3 Input Parameters and Initial Conditions and Assumptions Used in Non-LOCA Analyses (Sheet 2 of 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
(Sheet 1 of 3)

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

Table 15.5-5 Defense Line Functions Used in Non-LOCA Analyses
(Sheet 2 of 3)

DL ID	Function Name	Implementation	Inputs	Setpoint / Delay Analytical Limits
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
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

Table 15.5-5 Defense Line Functions Used in Non-LOCA Analyses
(Sheet 3 of 3)

DL ID	Function Name	Implementation	Inputs	Setpoint / Delay Analytical Limits
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
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 High Flux after Scram	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 High Flux after Scram	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, fuel grapple, mast, and head fall on top of the core or fuel storage 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
(Sheet 1 of 3)

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

Table 15.5-38c Key Parameters for DBA FHA Analysis
(Sheet 2 of 3)

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	4.96E-03
2-8 hours	2.94E-03
8-24 hours	2.26E-03
1-4 days	1.28E-03
4-30 days	5.67E-04
LPZ	
0-8 hours	3.10E-04
8-24 hours	2.26E-04
1-4 days	1.14E-04
4-30 days	4.30E-05
MCRE	
0-2 hours	1.51E-03
2-8 hours	1.06E-03
8-24 hours	4.95E-04
1-4 days	4.68E-04
4-30 days	4.21E-04

Table 15.5-38c Key Parameters for DBA FHA Analysis
(Sheet 3 of 3)

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]
(Sheet 1 of 2)**

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
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

**Table 15.5-39 Main Steam Line Break Accident Airborne Release Source Term[1]
(Sheet 2 of 2)**

Isotope	Equilibrium Iodine		Pre-Incident Iodine Spike	
	MBq	Ci	MBq	Ci
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
(Sheet 1 of 2)

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
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

Table 15.5-40 FWLB Accident Airborne Release Source Term
(Sheet 2 of 2)

Isotope	Equilibrium Iodine		Pre-Incident Iodine Spike	
	Ci	MBq	Ci	MBq
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
(Sheet 1 of 2)

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
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

Table 15.5-41 ICS Line Break Accident Airborne Release Source Term
(Sheet 2 of 2)

Isotope	Equilibrium Iodine		Pre-Incident Iodine Spike	
	Ci	MBq	Ci	MBq
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
(Sheet 1 of 3)

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
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

Table 15.5-42a Instrument Line Break Accident Airborne Iodine Equilibrium Source Term
(Sheet 2 of 3)

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**
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
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

Table 15.5-42a Instrument Line Break Accident Airborne Iodine Equilibrium Source Term
(Sheet 3 of 3)

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**
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
(Sheet 1 of 3)

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
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

Table 15.5-42b Instrument Line Break Accident Iodine Spike Release Source Term
(Sheet 2 of 3)

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**
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
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

Table 15.5-42b Instrument Line Break Accident Iodine Spike Release Source Term
(Sheet 3 of 3)

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**
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 • 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 (Sheet 1 of 2)

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 DL2-14	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 DL2-14.

Table 15.5-45 DL2 and DL4a Functions Credited in DEC LOCA Analyses (Sheet 2 of 2)

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.

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.

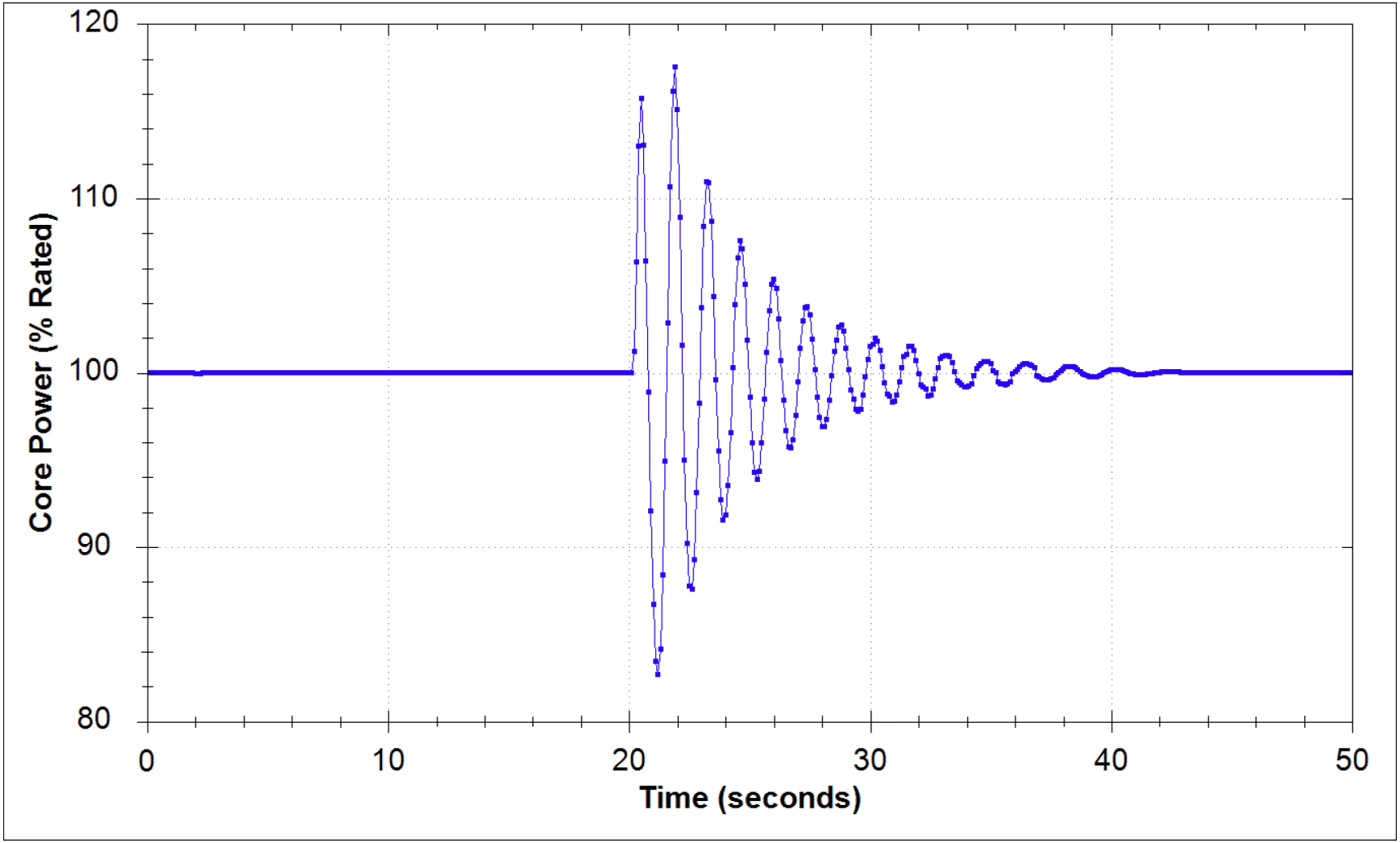

Figure 15.5-1 MOC Transient Results for Core-Wide Stability Analysis

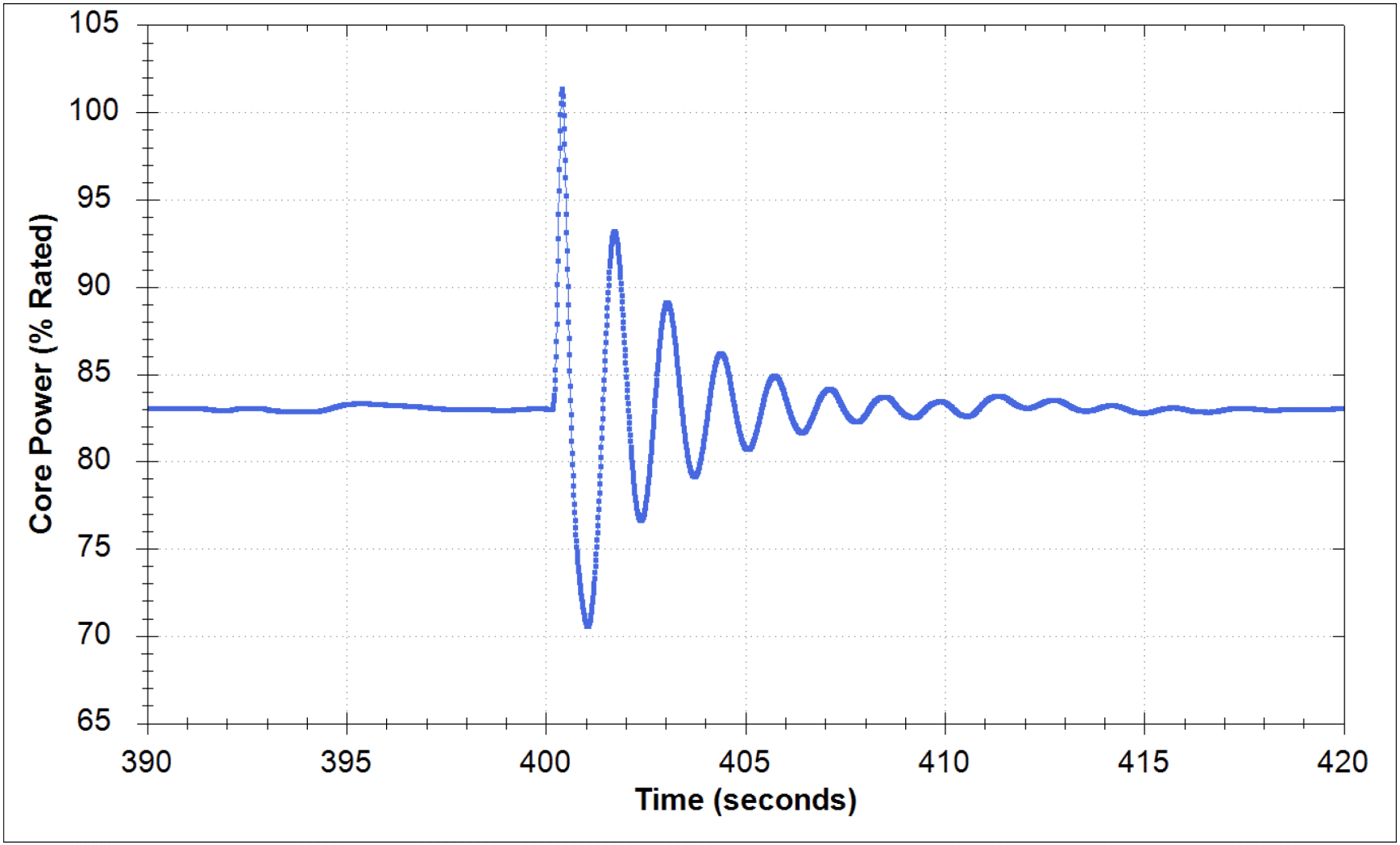

Figure 15.5-2 MOC Transient Results for LFWH with SCCRI Core-Wide Stability Analysis

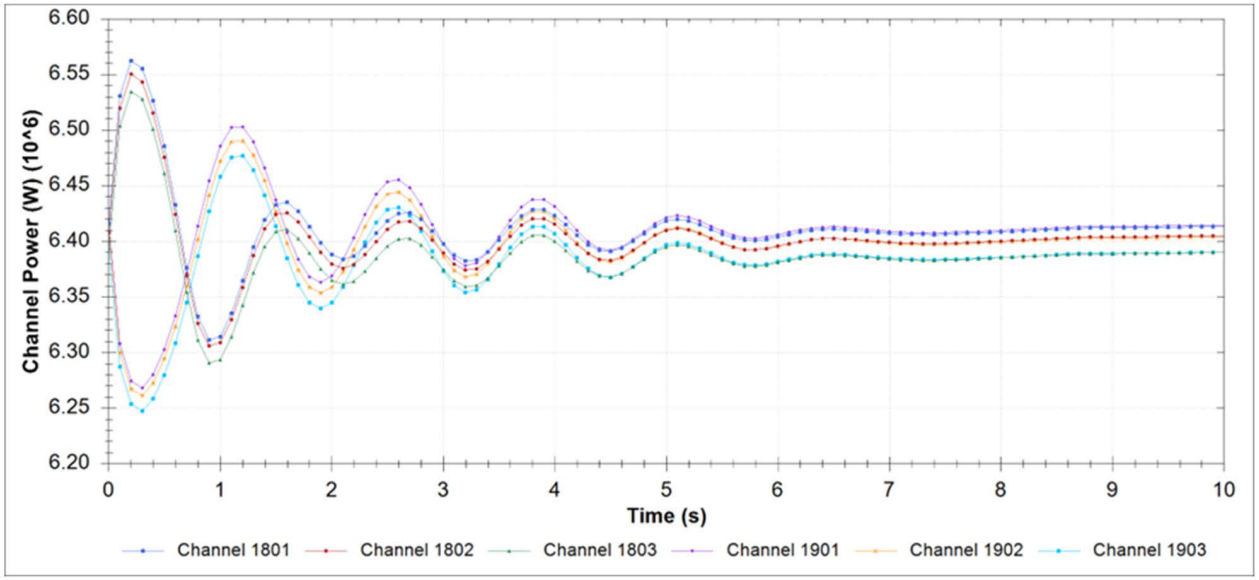

Figure 15.5-3 Regional Mode Stabilitiy Response at 115% Power

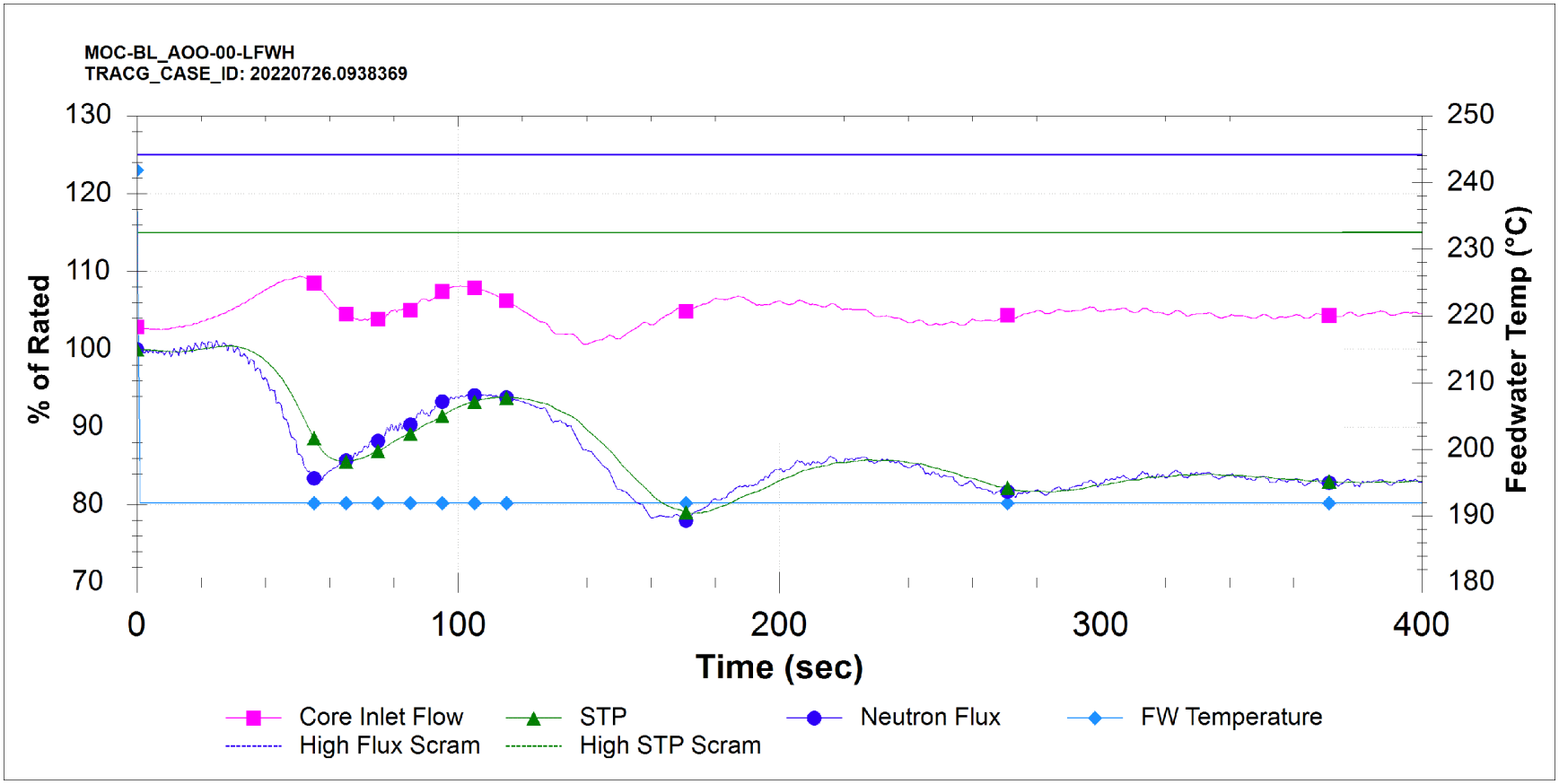

Figure 15.5-4 Loss of Feedwater Heating (AOO)

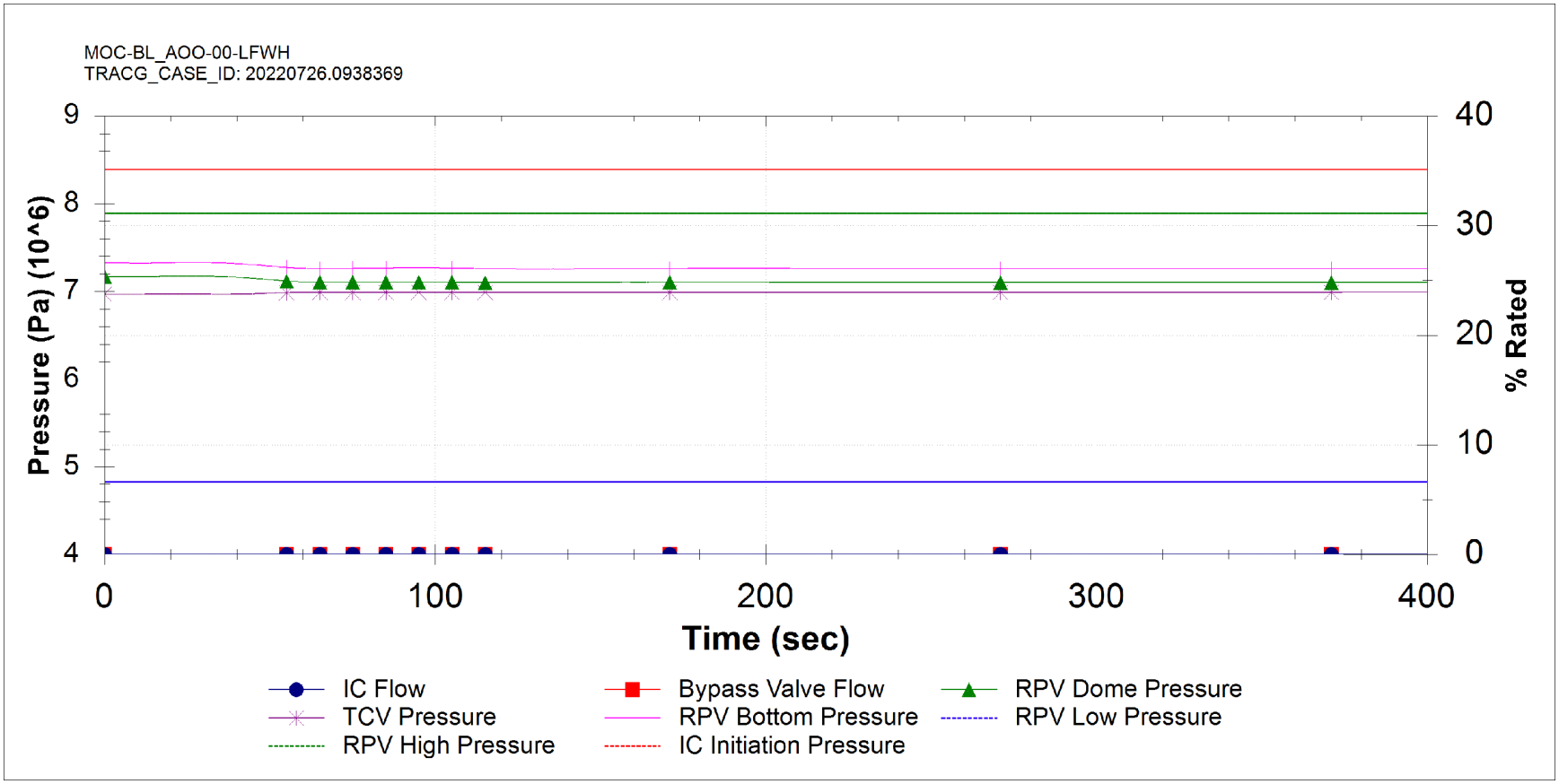

Figure 15.5-5 Loss of Feedwater Heating (AOO)

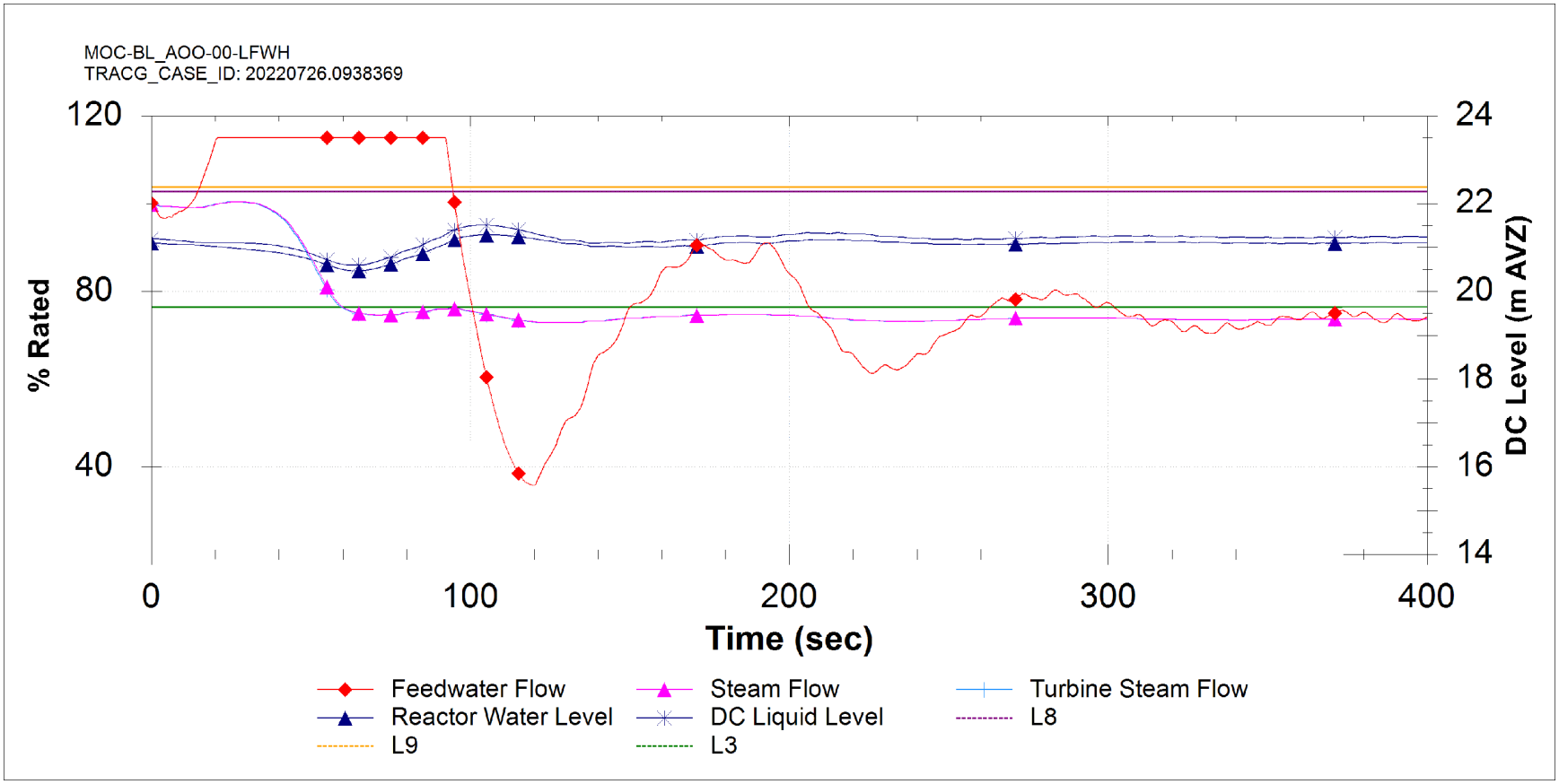

Figure 15.5-6 Loss of Feedwater Heating (AOO)

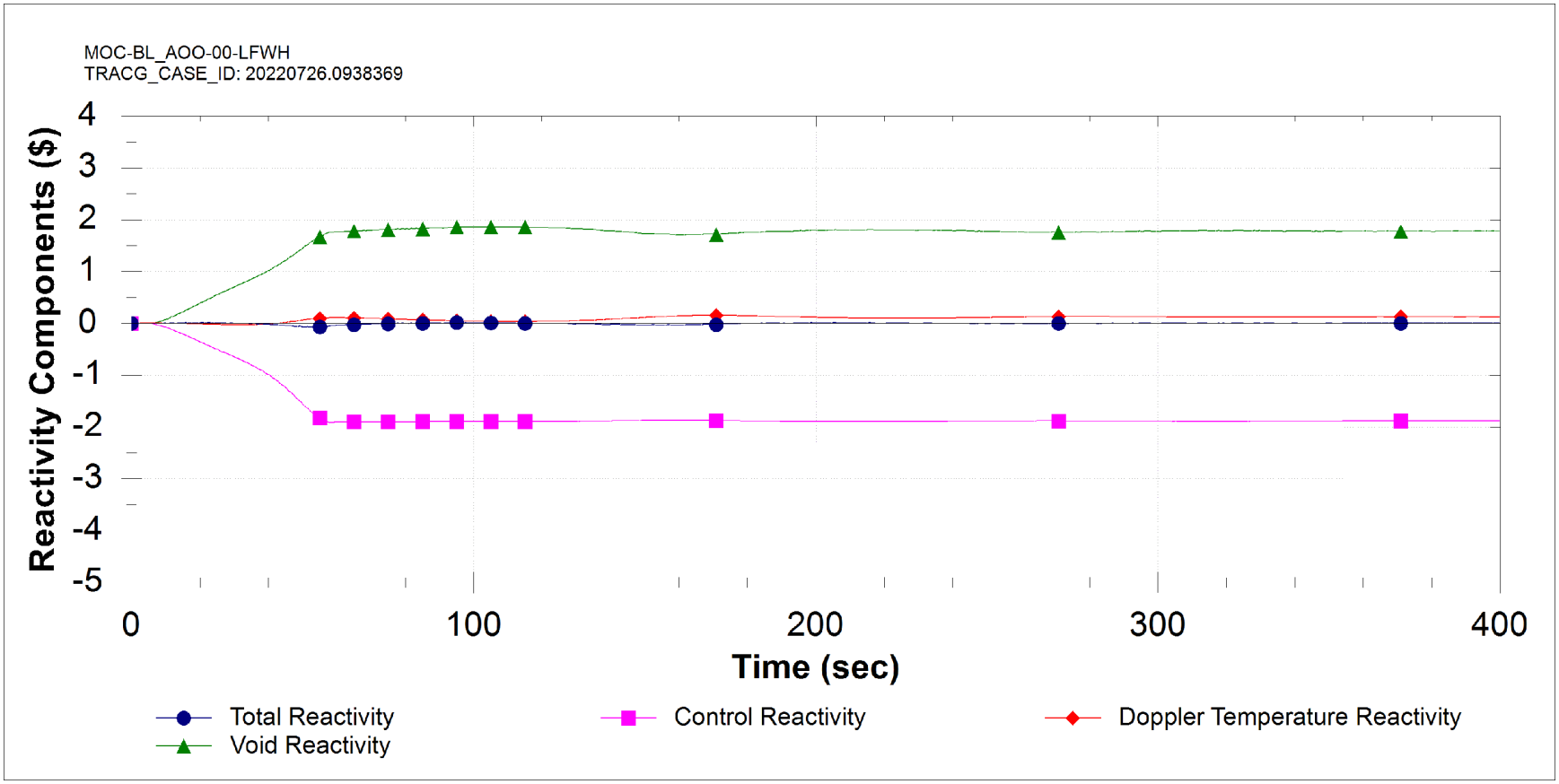

Figure 15.5-7 Loss of Feedwater Heating (AOO)

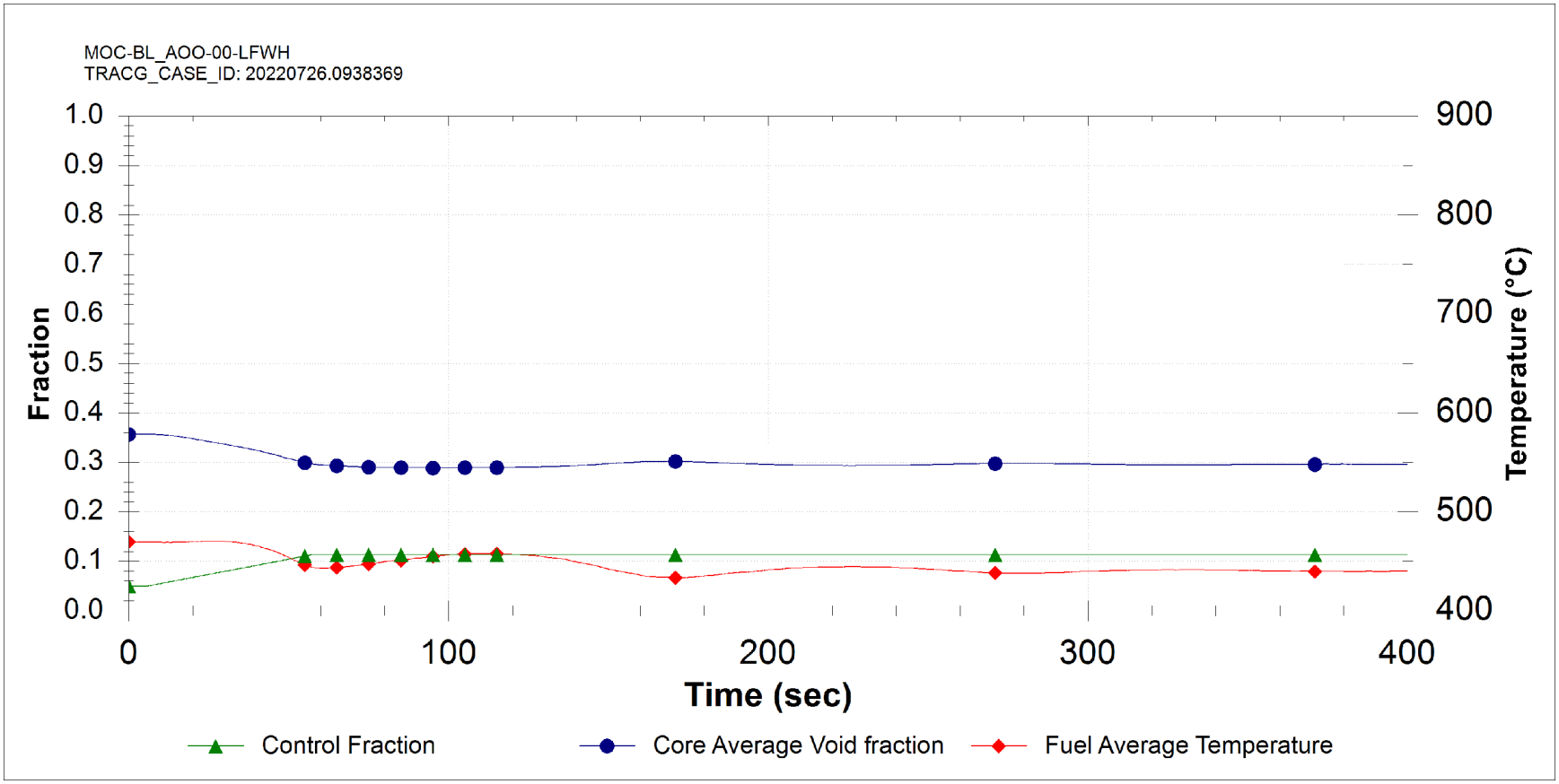

Figure 15.5-8 Loss of Feedwater Heating (AOO)

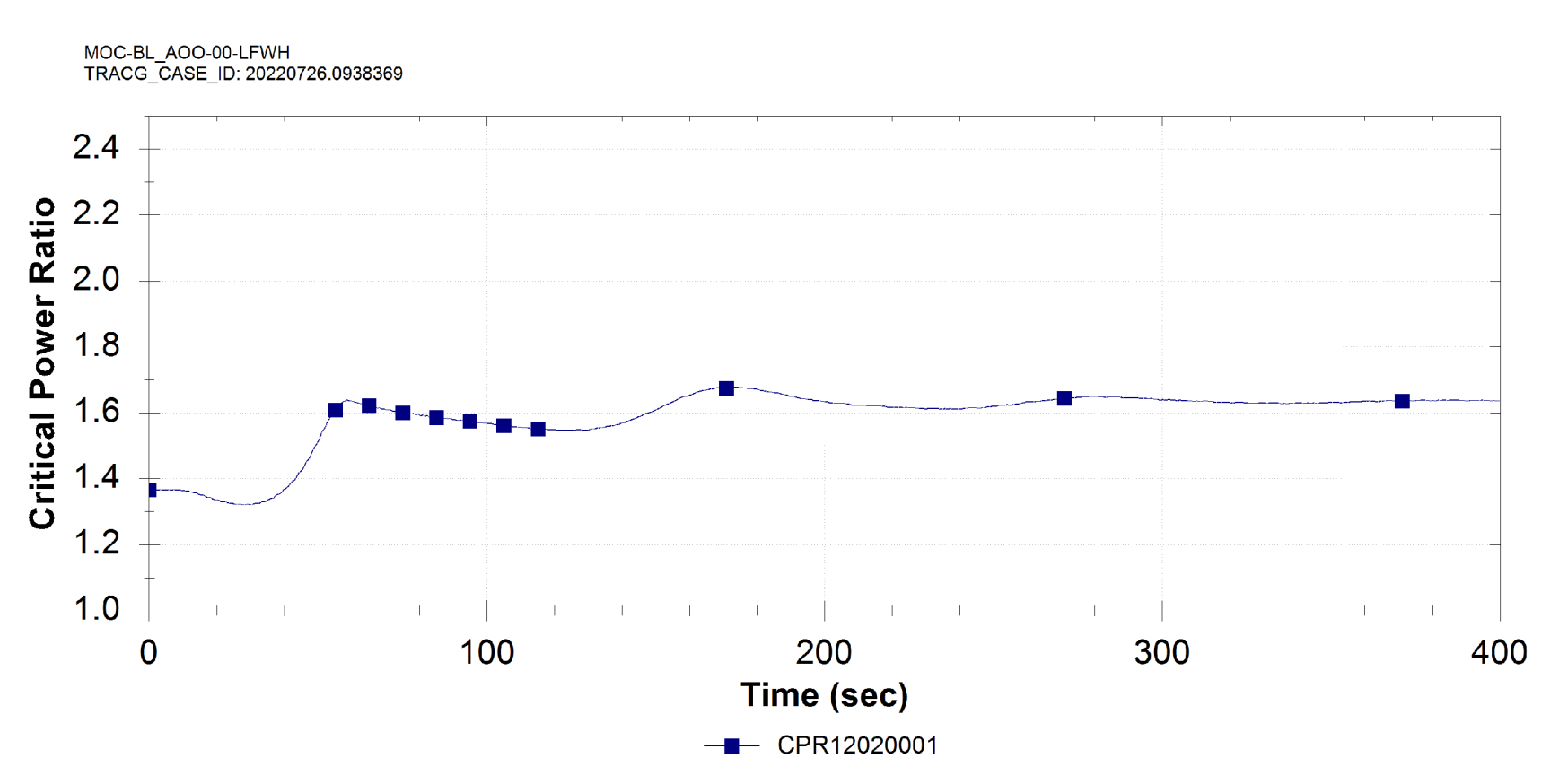

Figure 15.5-9 Loss of Feedwater Heating (AOO)

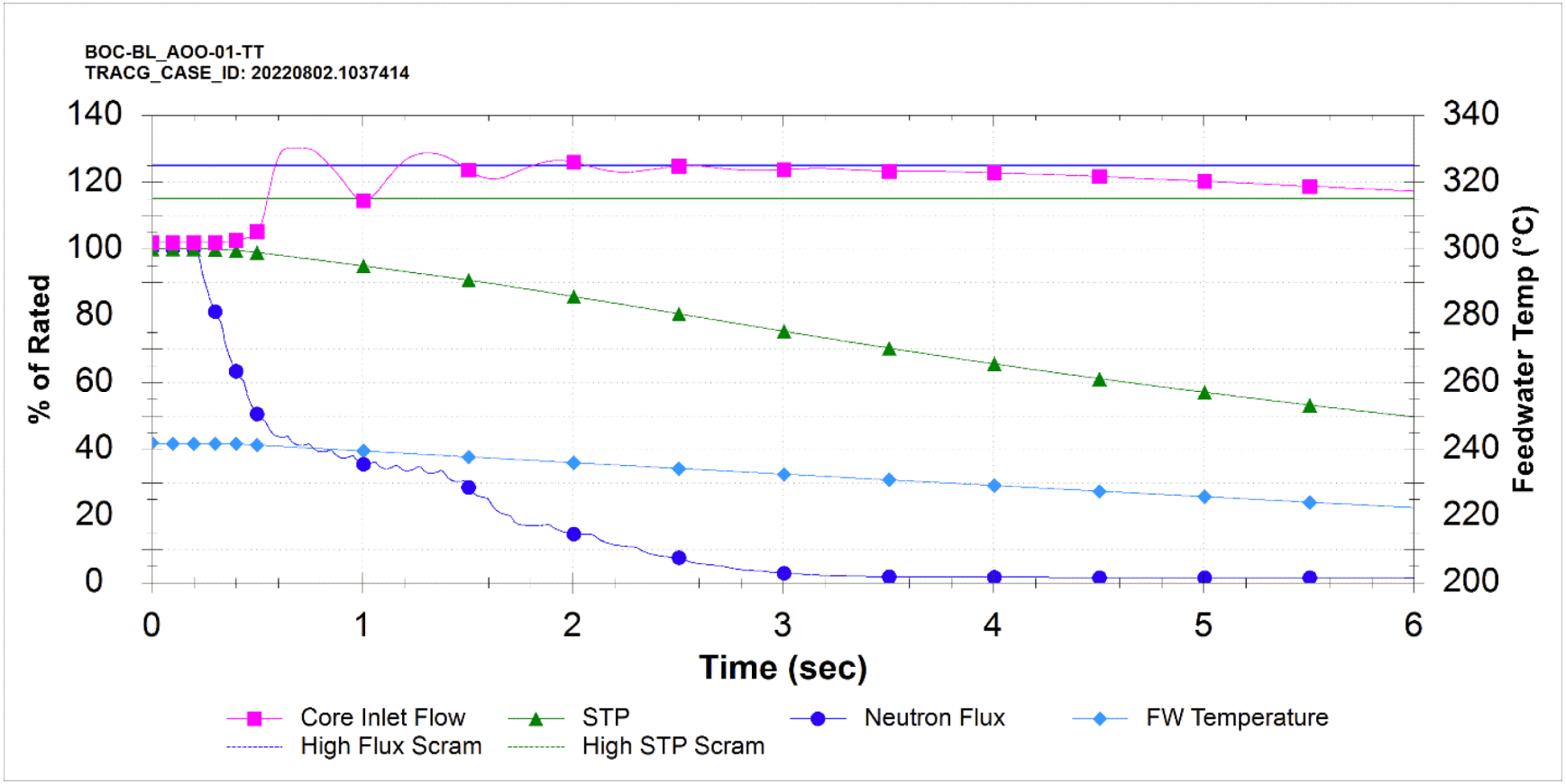

Figure 15.5-10 Turbine Trip (AOO)

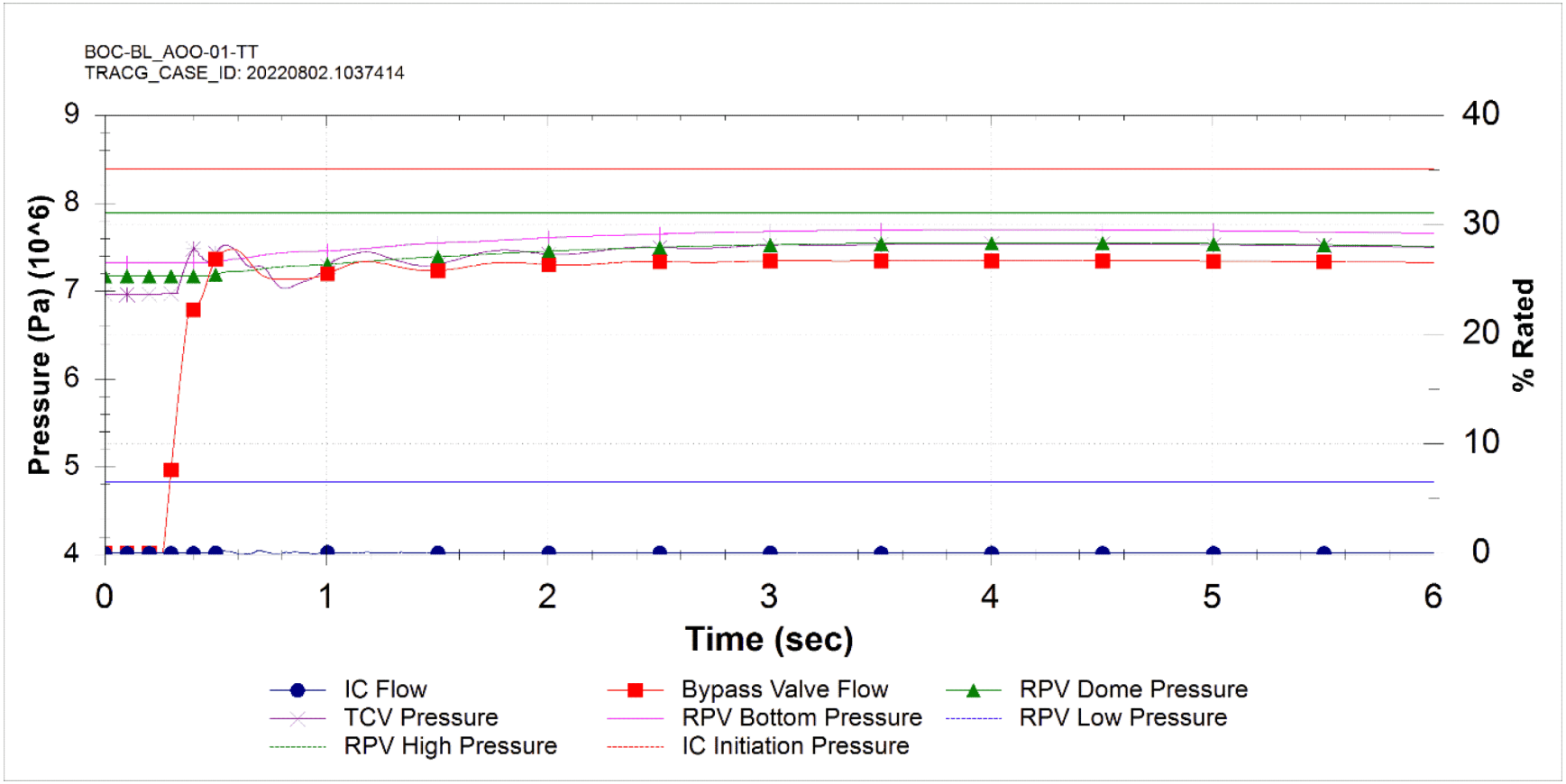

Figure 15.5-11 Turbine Trip (AOO)

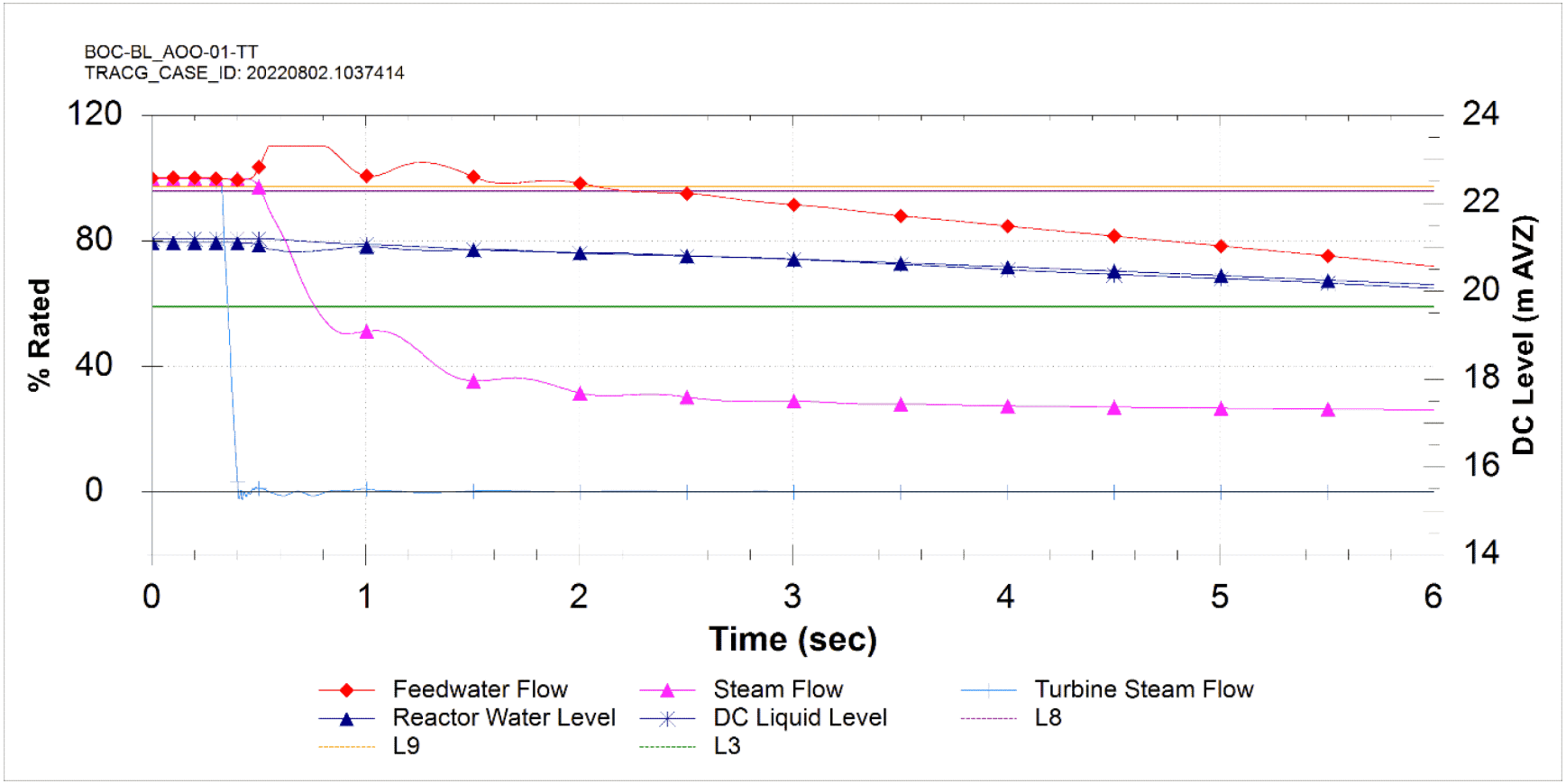

Figure 15.5-12 Turbine Trip (AOO)

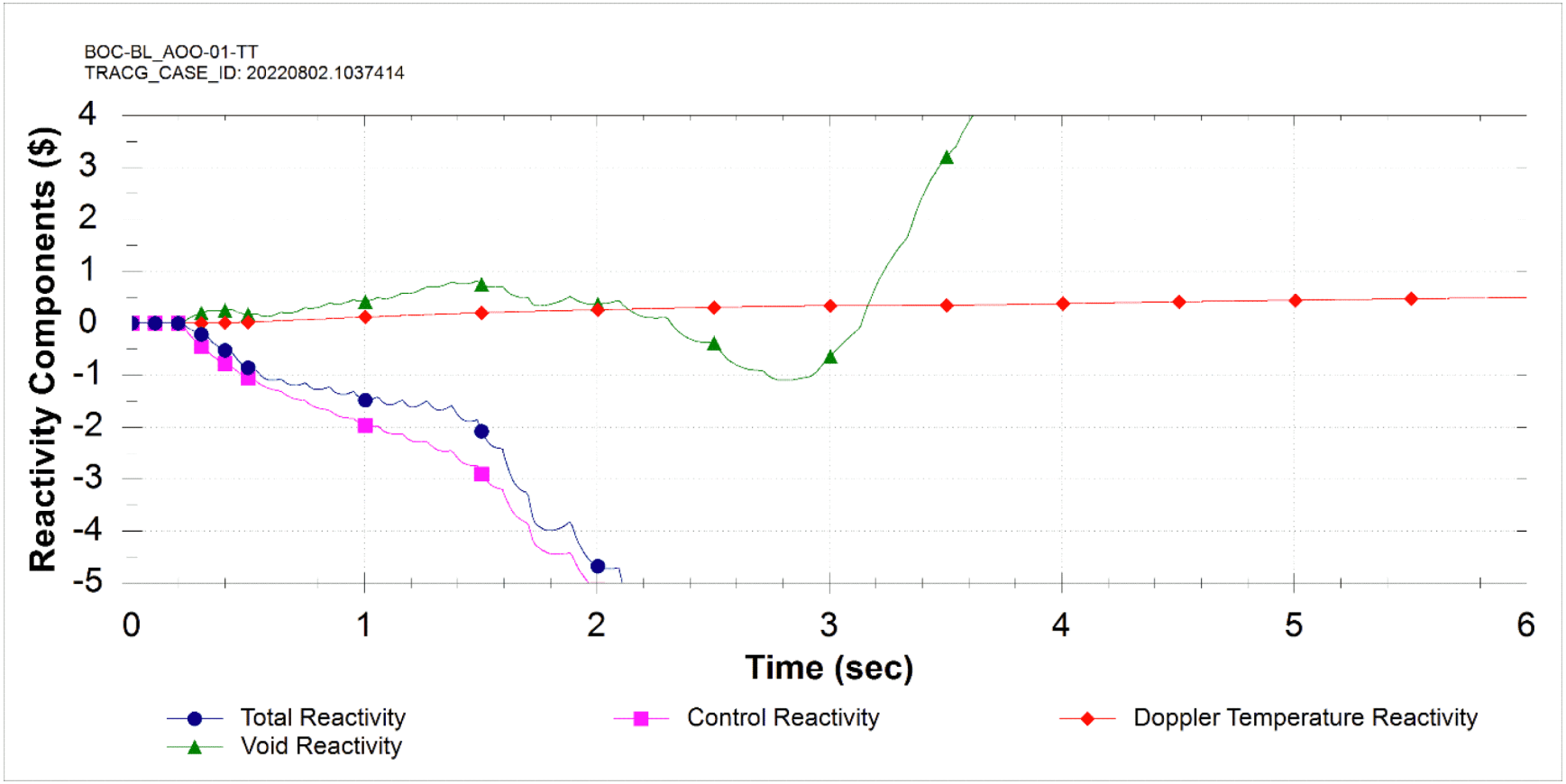

Figure 15.5-13 Turbine Trip (AOO)

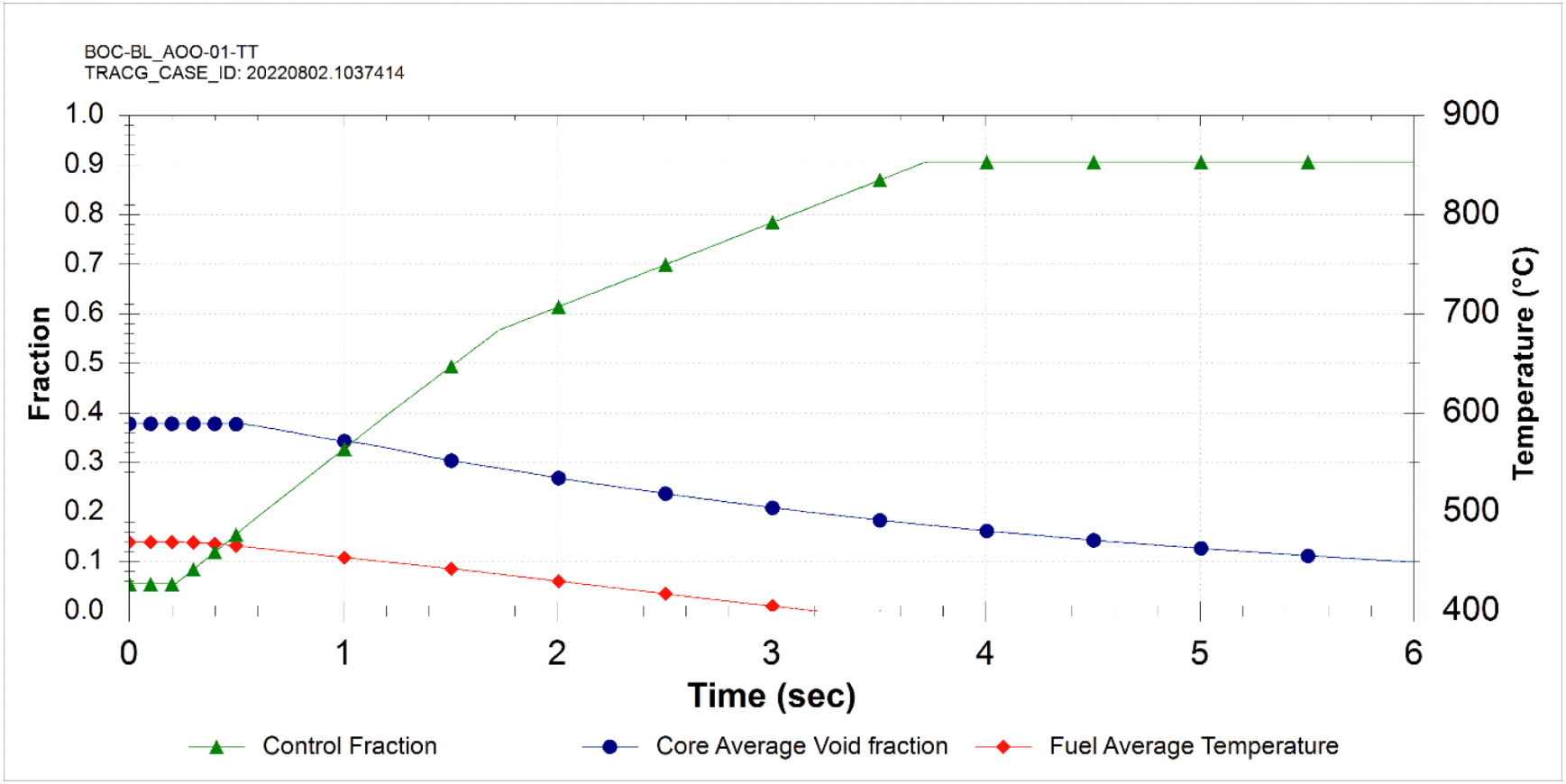

Figure 15.5-14 Turbine Trip (AOO)

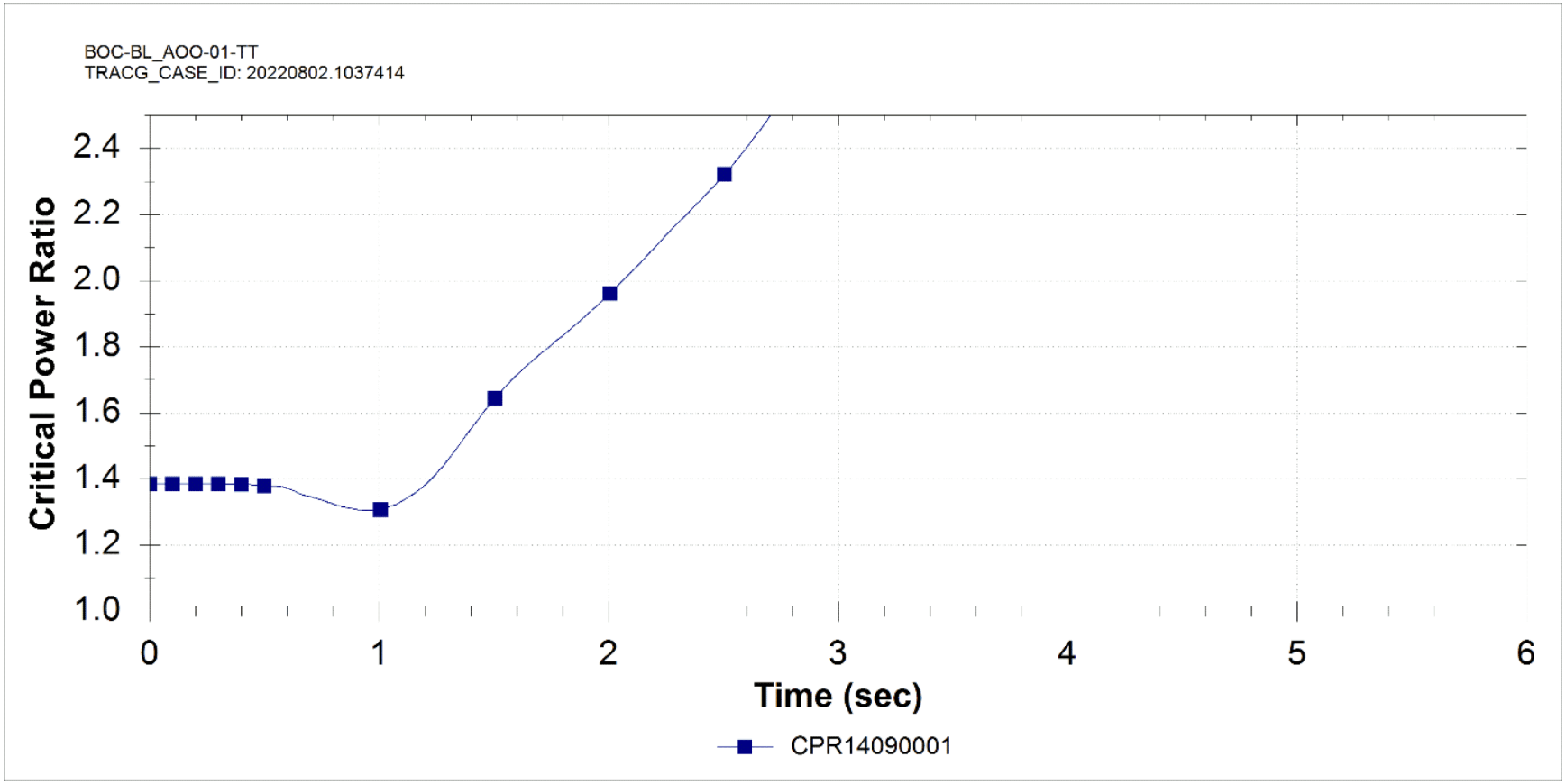

Figure 15.5-15 Turbine Trip (AOO)

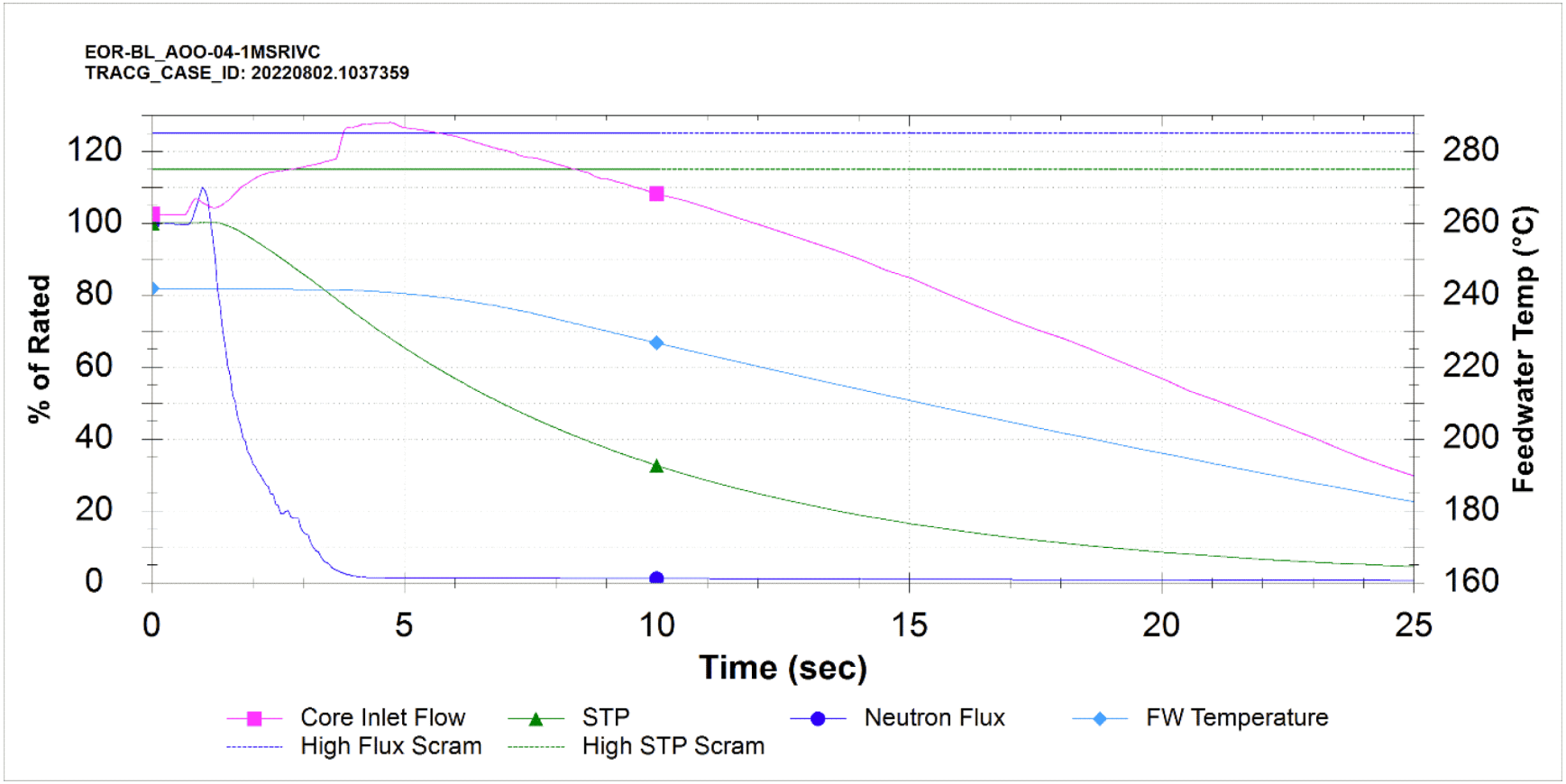

Figure 15.5-16 Closure of One Main Steam Reactor Isolation Valve (AOO)

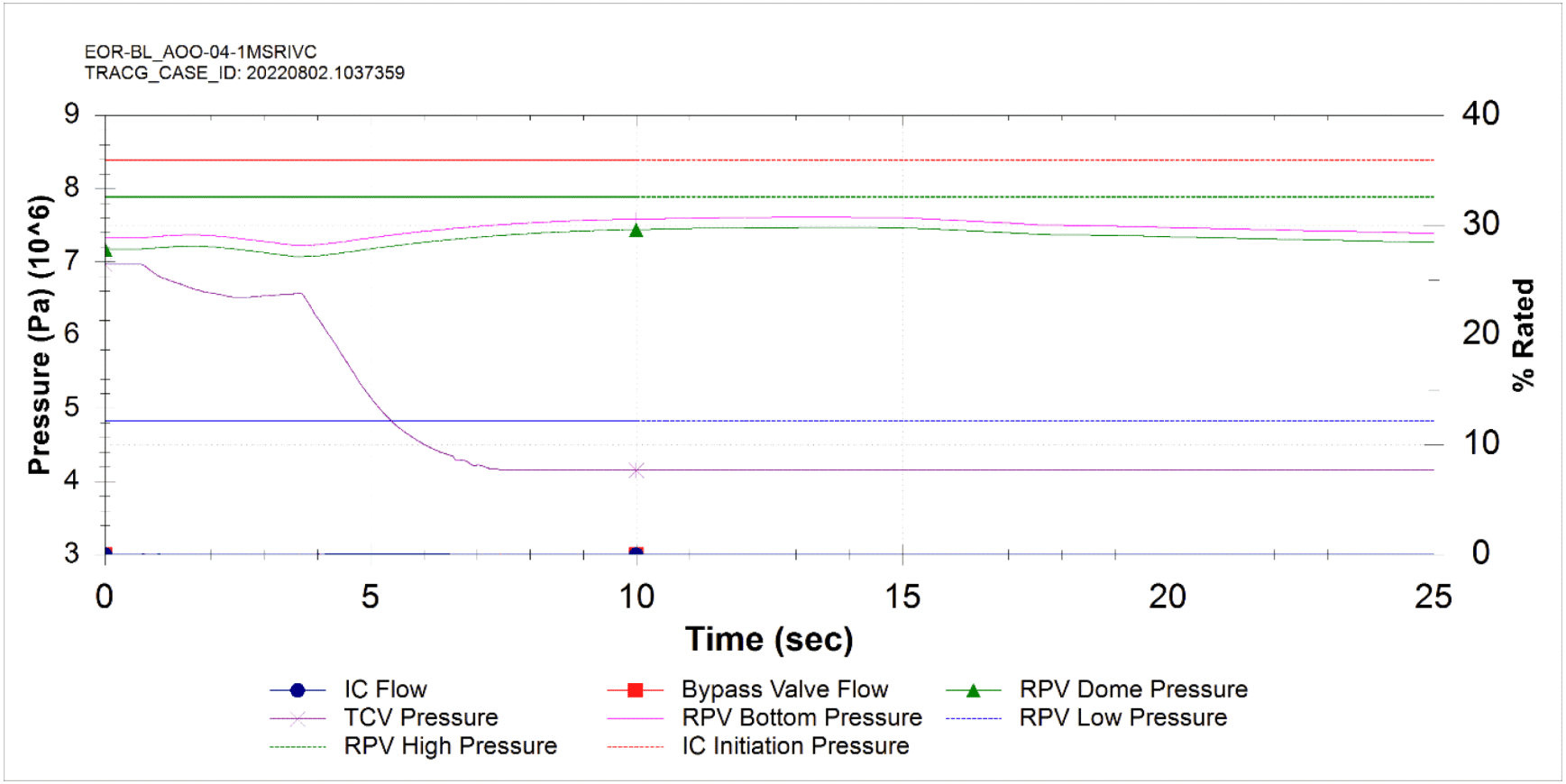

Figure 15.5-17 Closure of One Main Steam Reactor Isolation Valve (AOO)

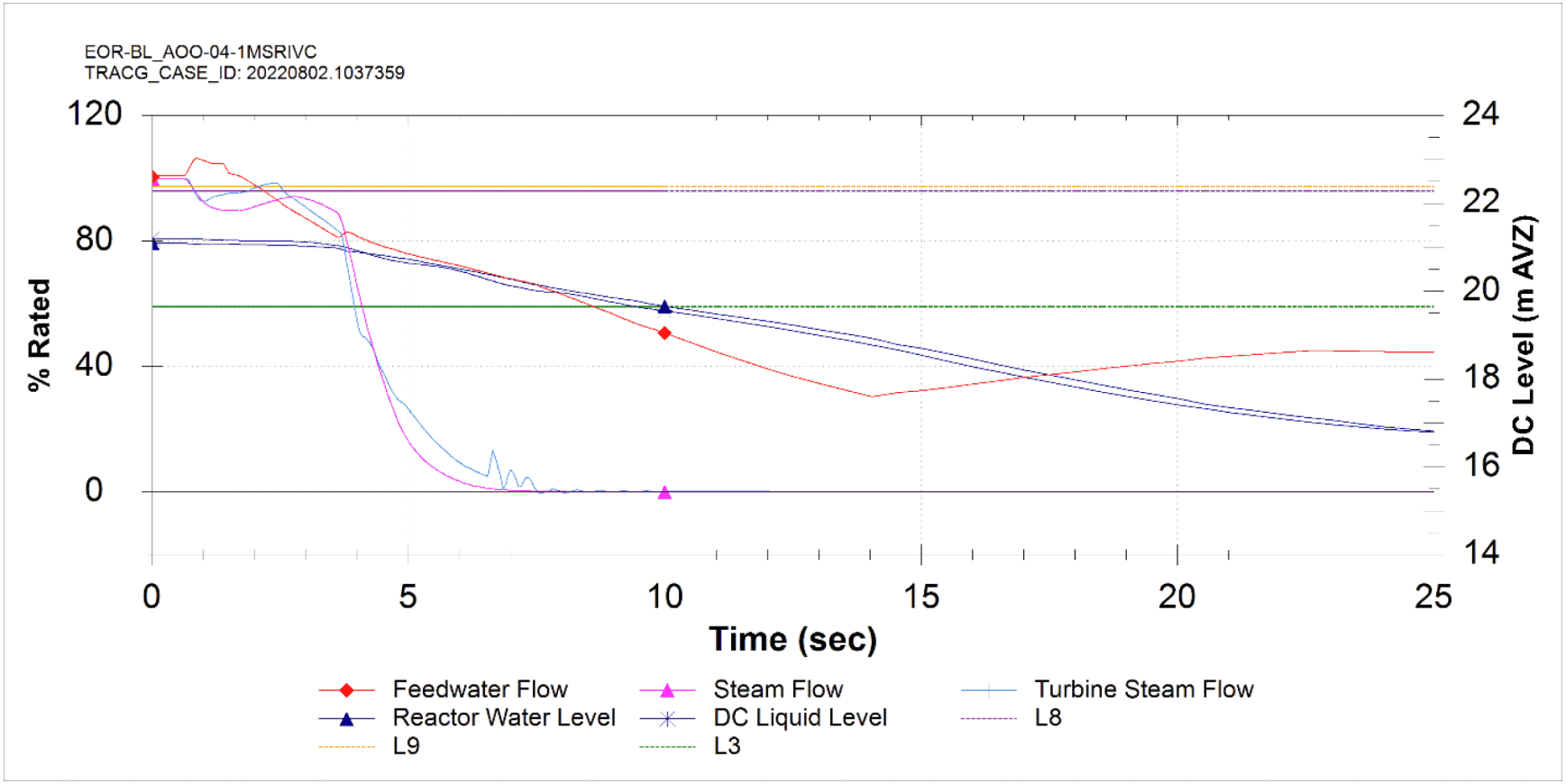

Figure 15.5-18 Closure of One Main Steam Reactor Isolation Valve (AOO)

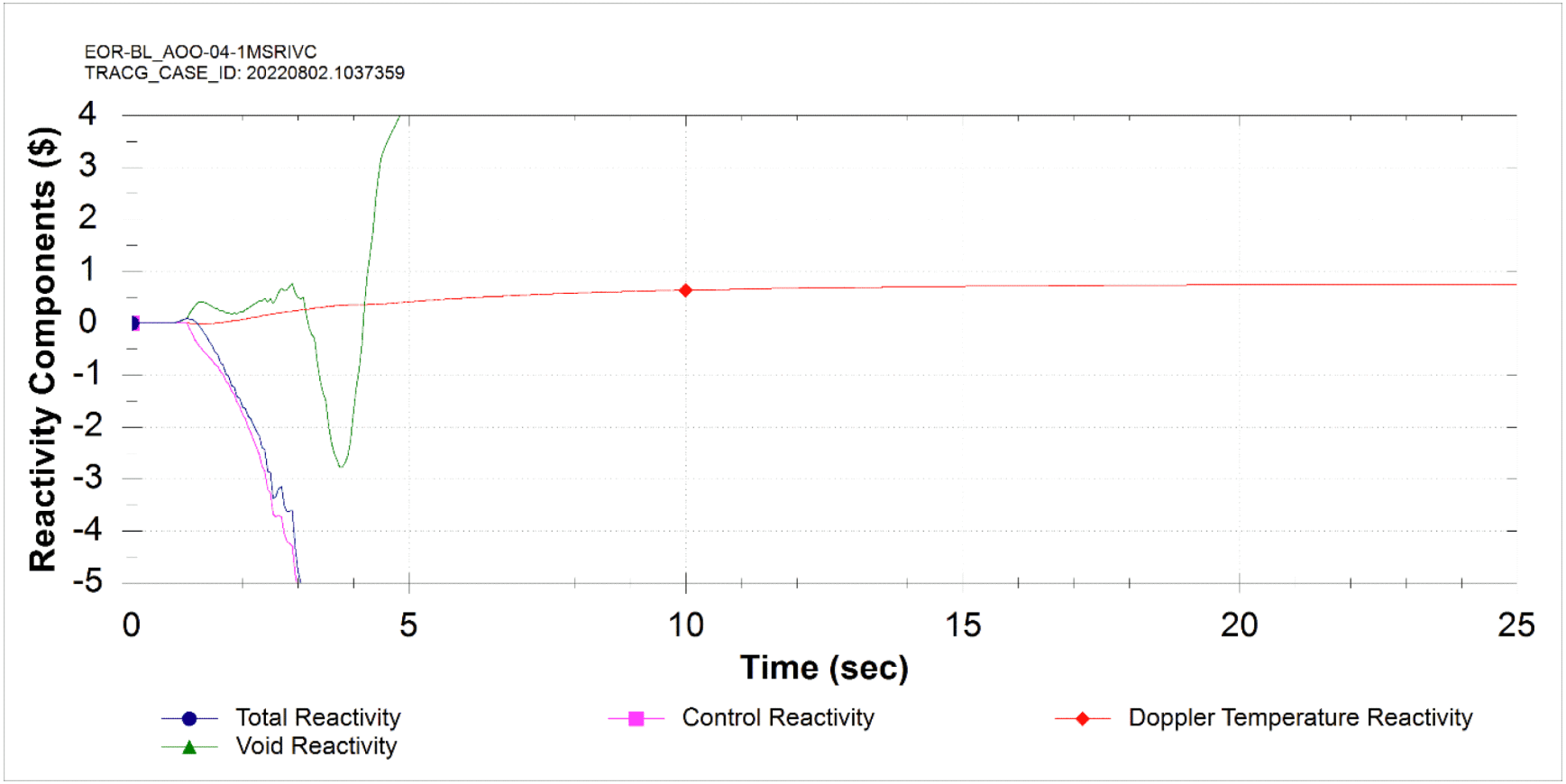

Figure 15.5-19 Closure of One Main Steam Reactor Isolation Valve (AOO)

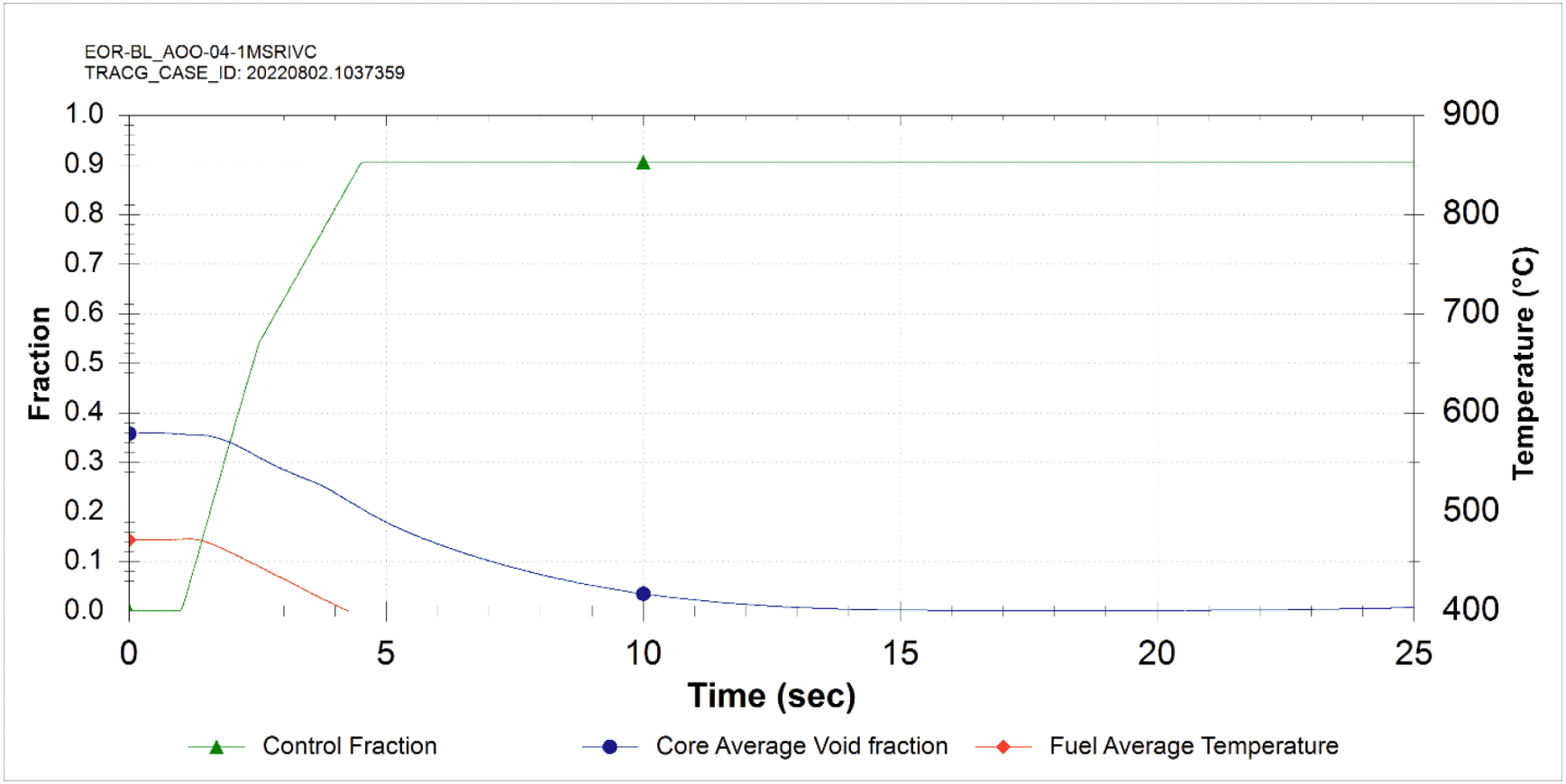

Figure 15.5-20 Closure of One Main Steam Reactor Isolation Valve (AOO)

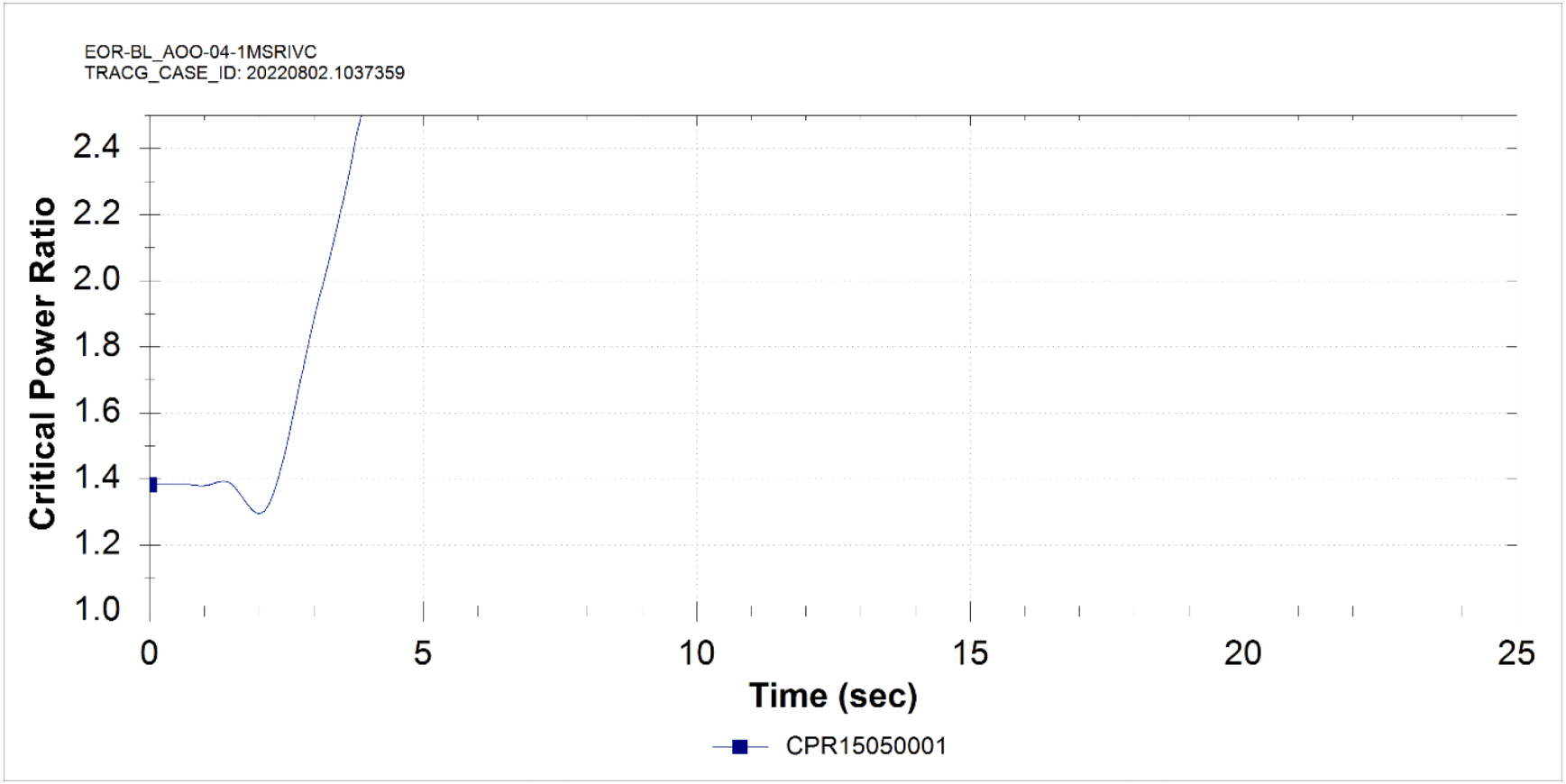

Figure 15.5-21 Closure of One Main Steam Reactor Isolation Valve (AOO)

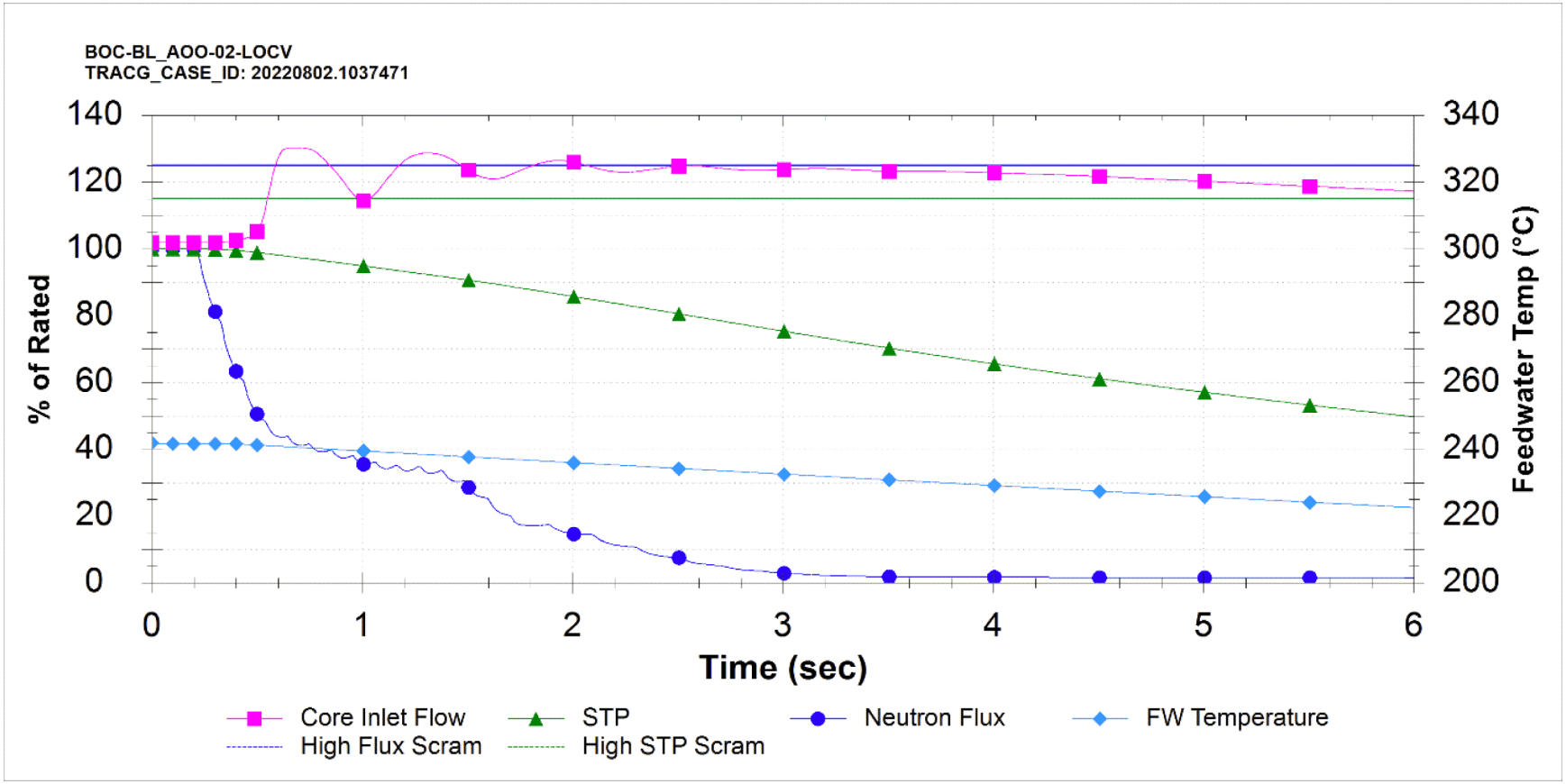

Figure 15.5-22 Loss of Condenser Vacuum (AOO)

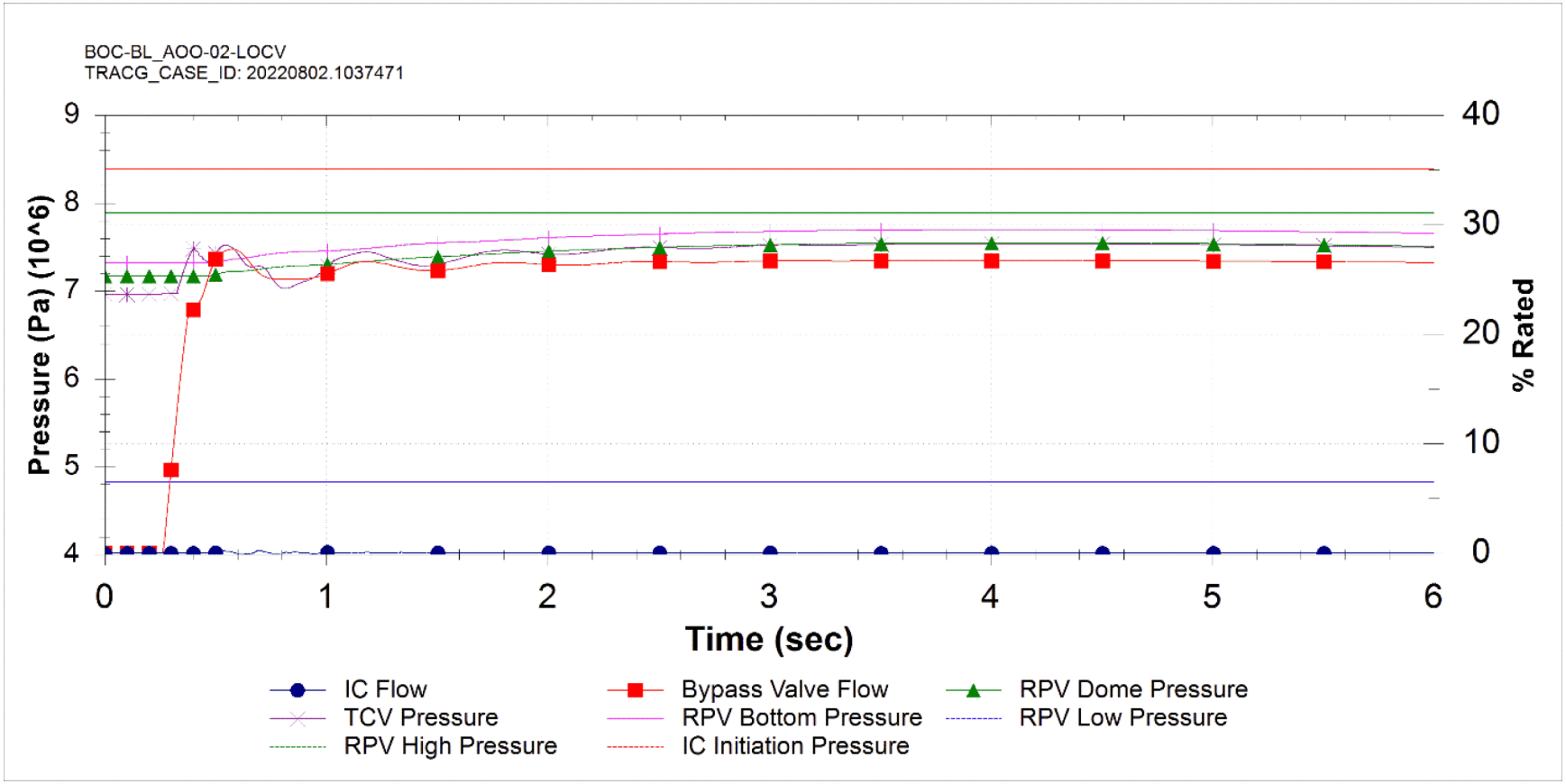

Figure 15.5-23 Loss of Condenser Vacuum (AOO)

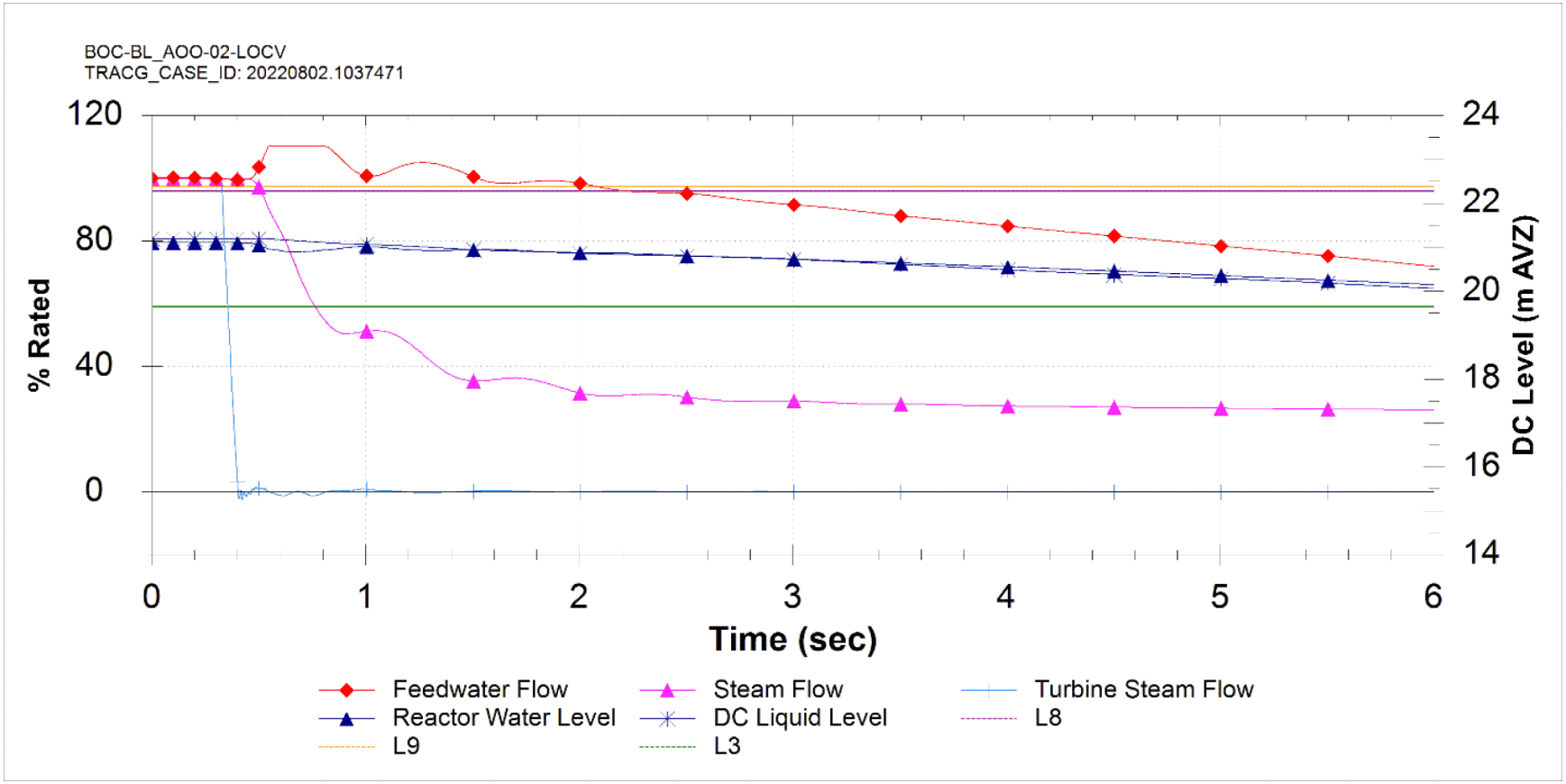

Figure 15.5-24 Loss of Condenser Vacuum (AOO)

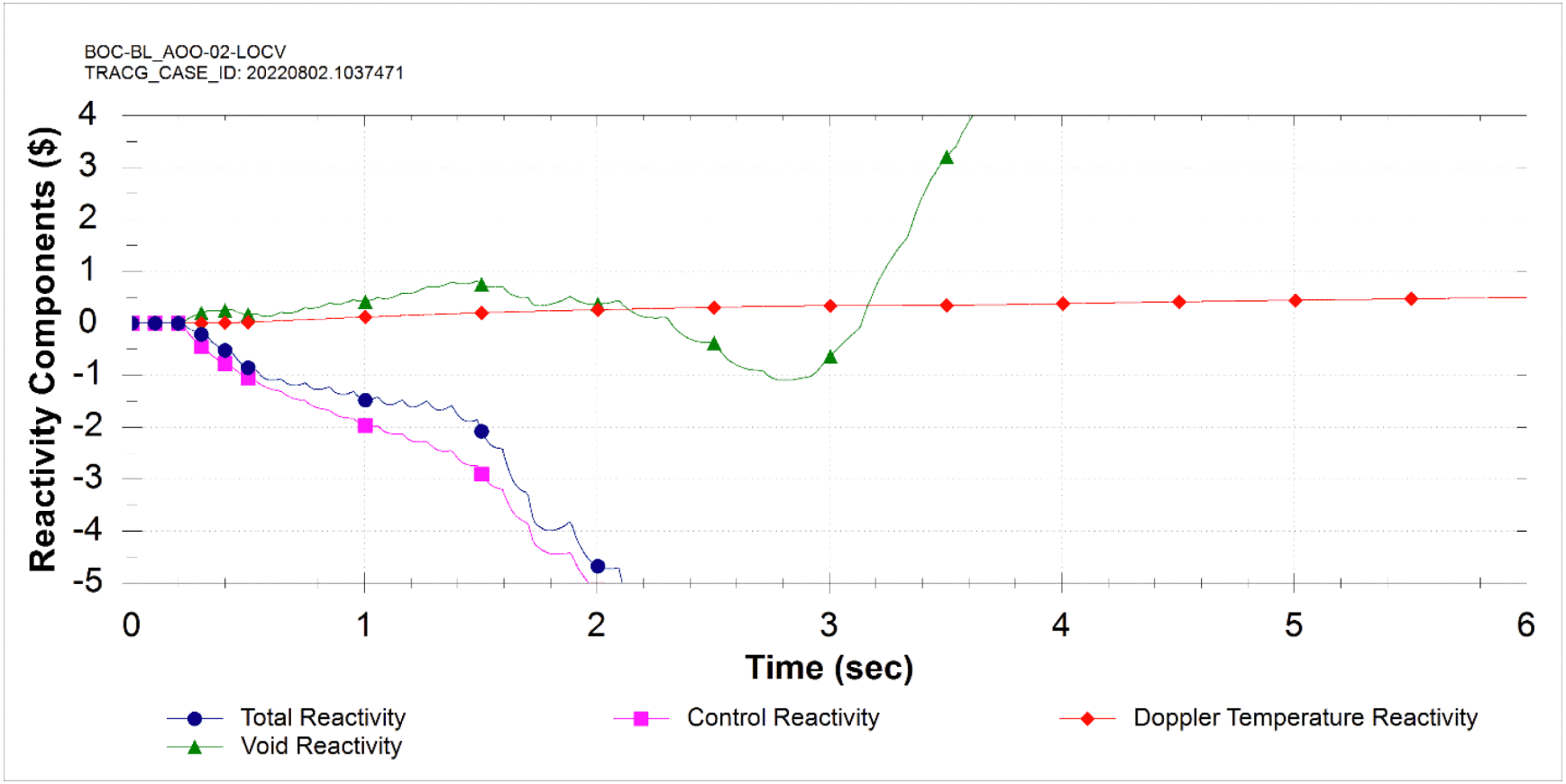

Figure 15.5-25 Loss of Condenser Vacuum (AOO)

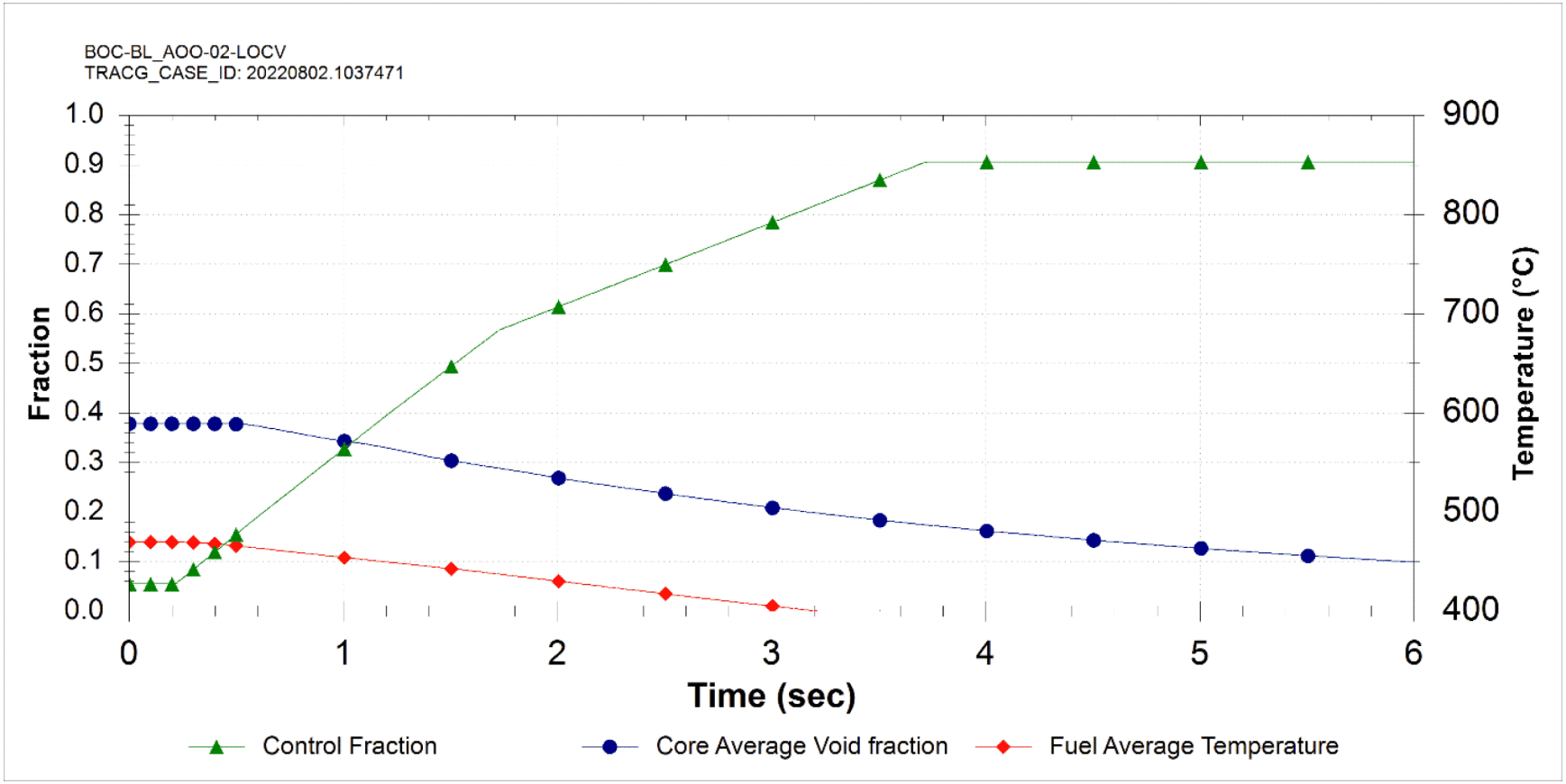

Figure 15.5-26 Loss of Condenser Vacuum (AOO)

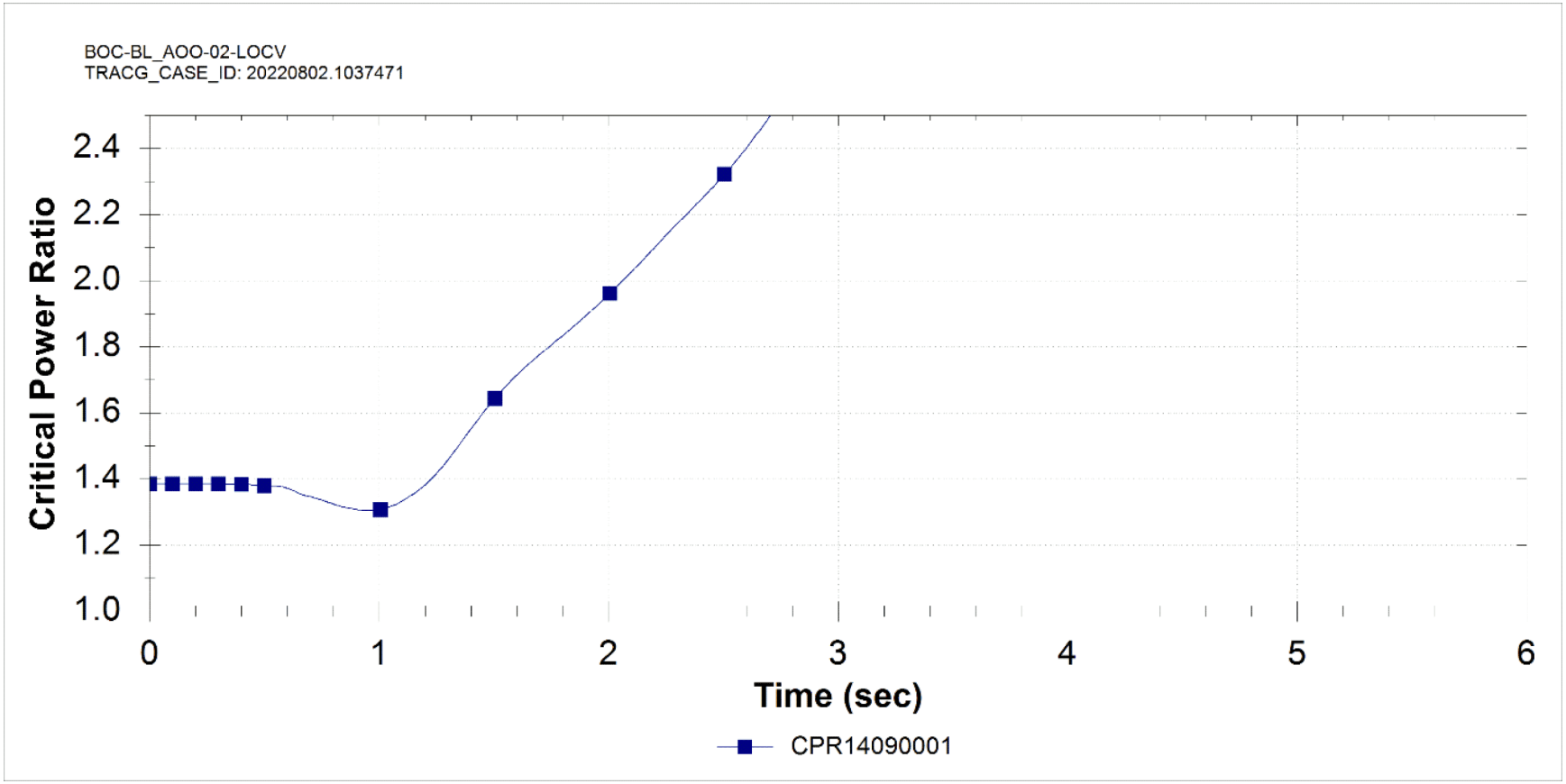

Figure 15.5-27 Loss of Condenser Vacuum (AOO)

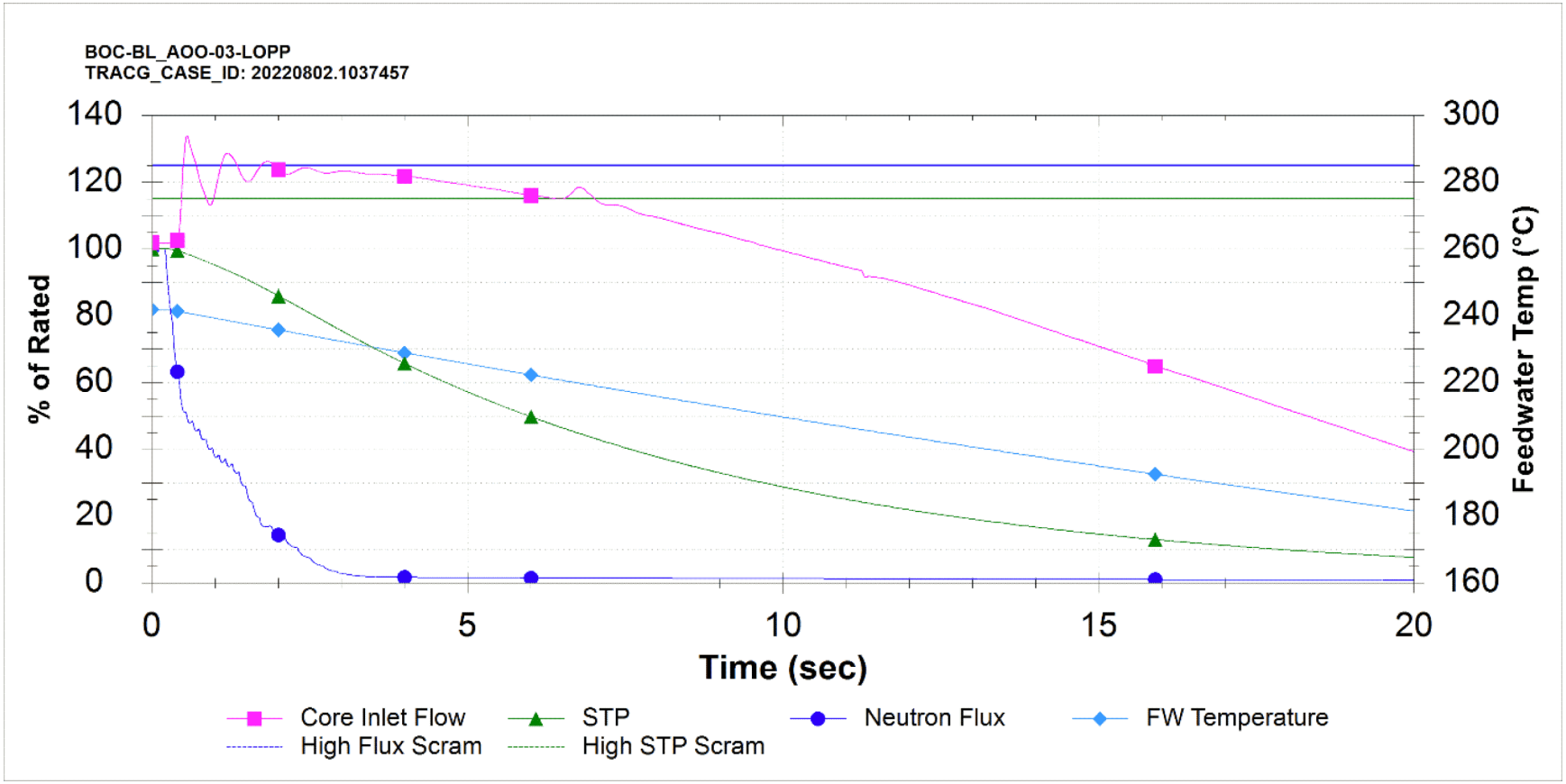

Figure 15.5-28 Loss of Preferred Power (AOO)

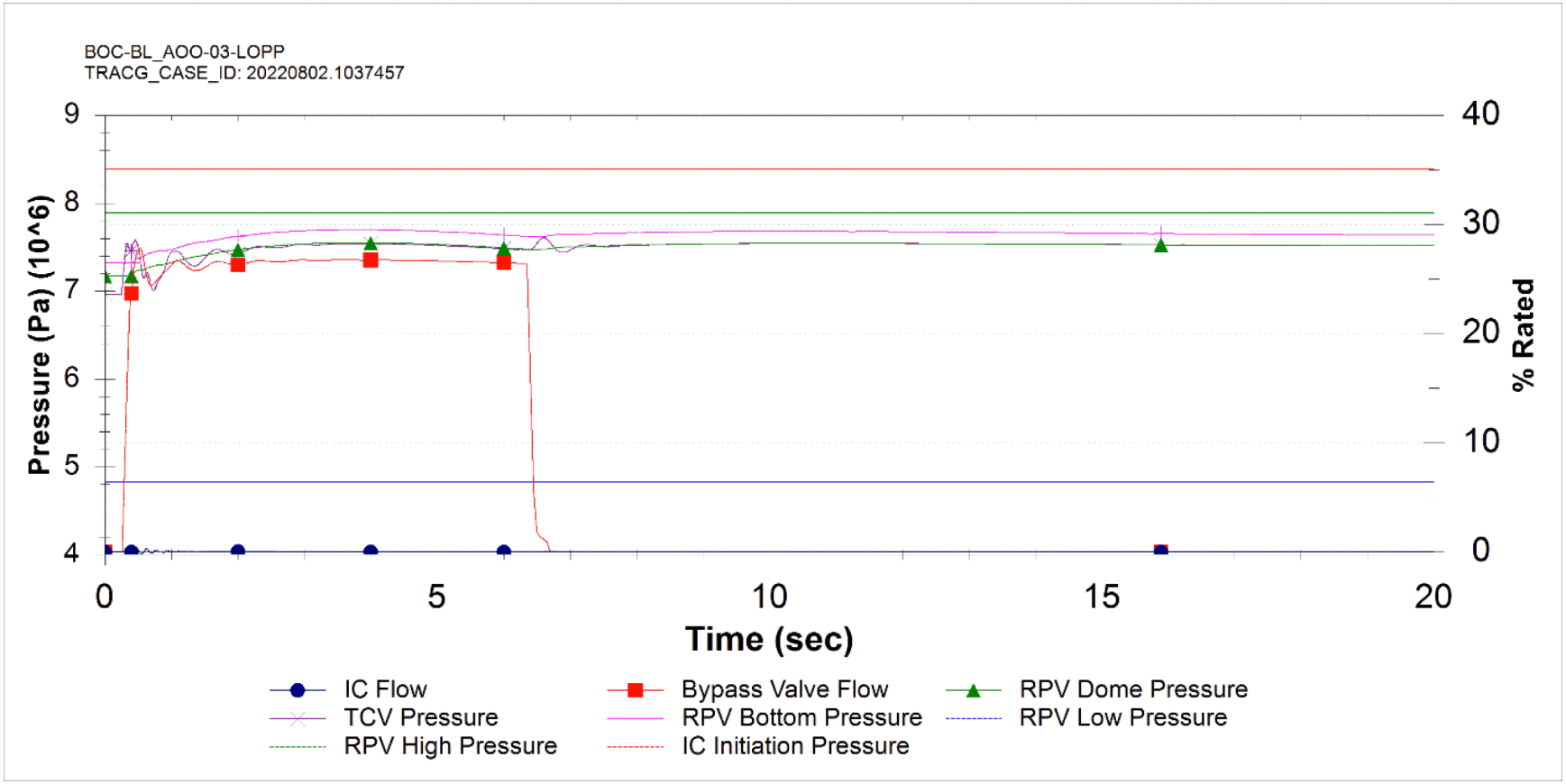

Figure 15.5-29 Loss of Preferred Power (AOO)

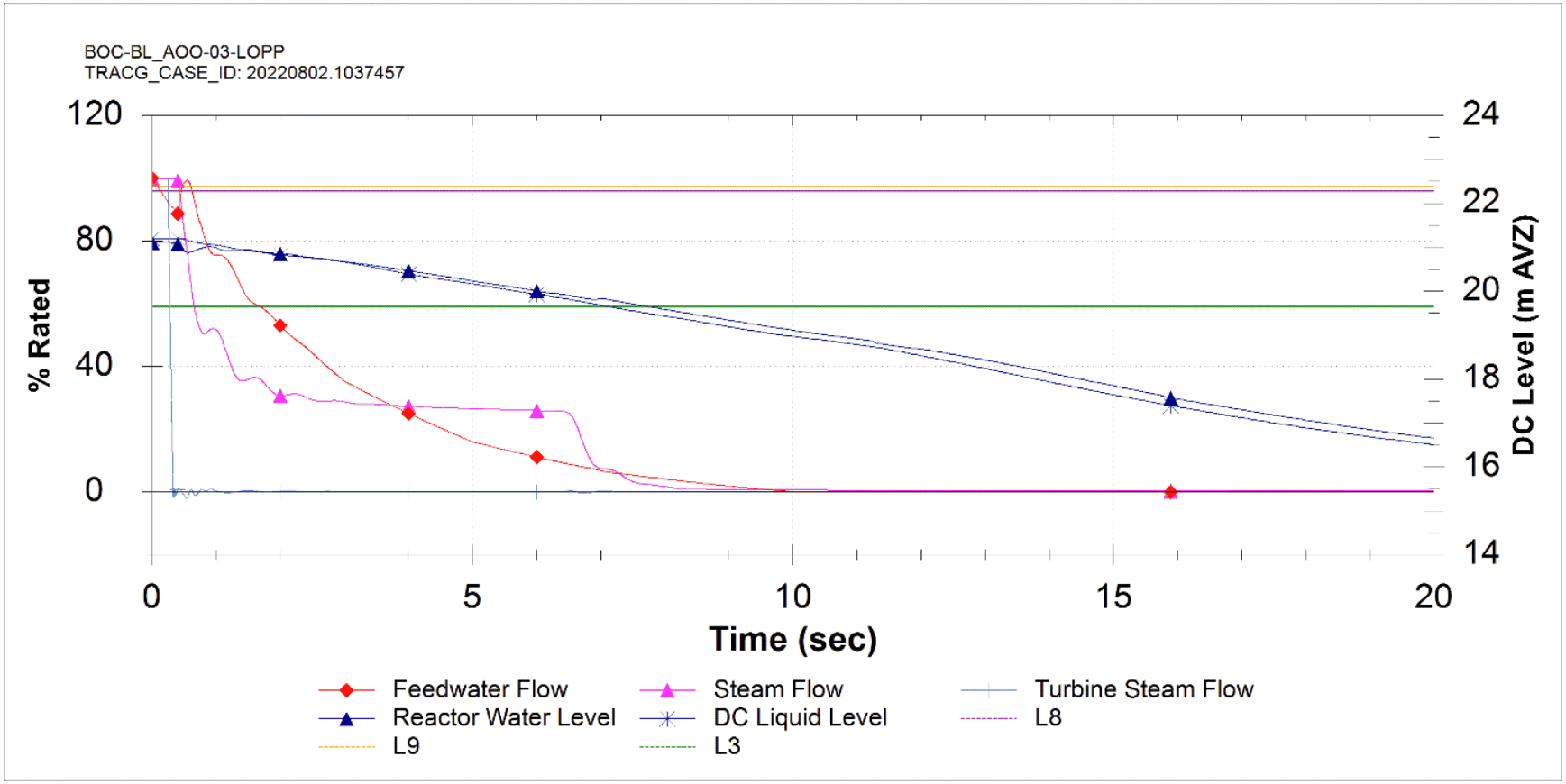

Figure 15.5-30 Loss of Preferred Power (AOO)

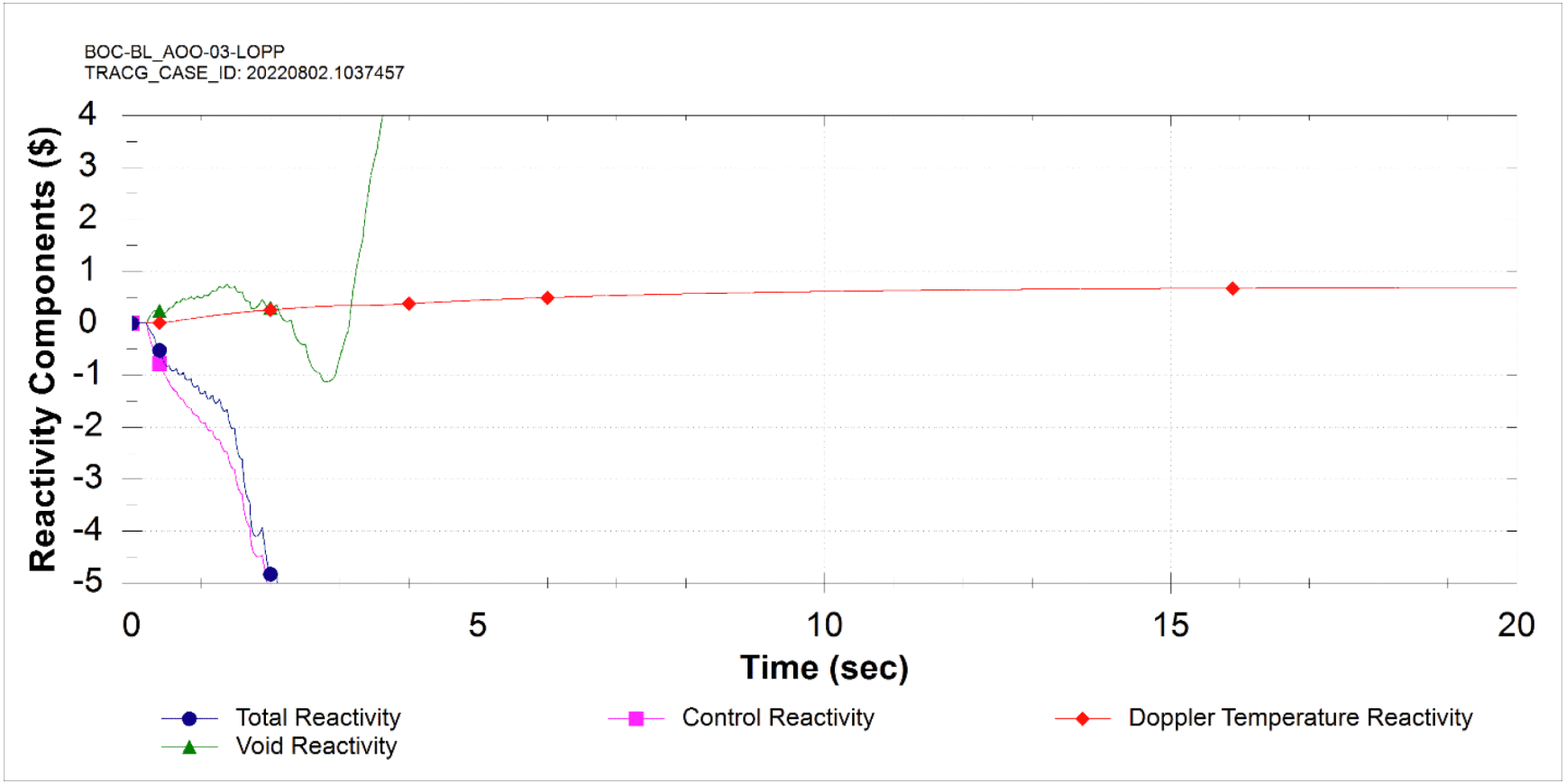

Figure 15.5-31 Loss of Preferred Power (AOO)

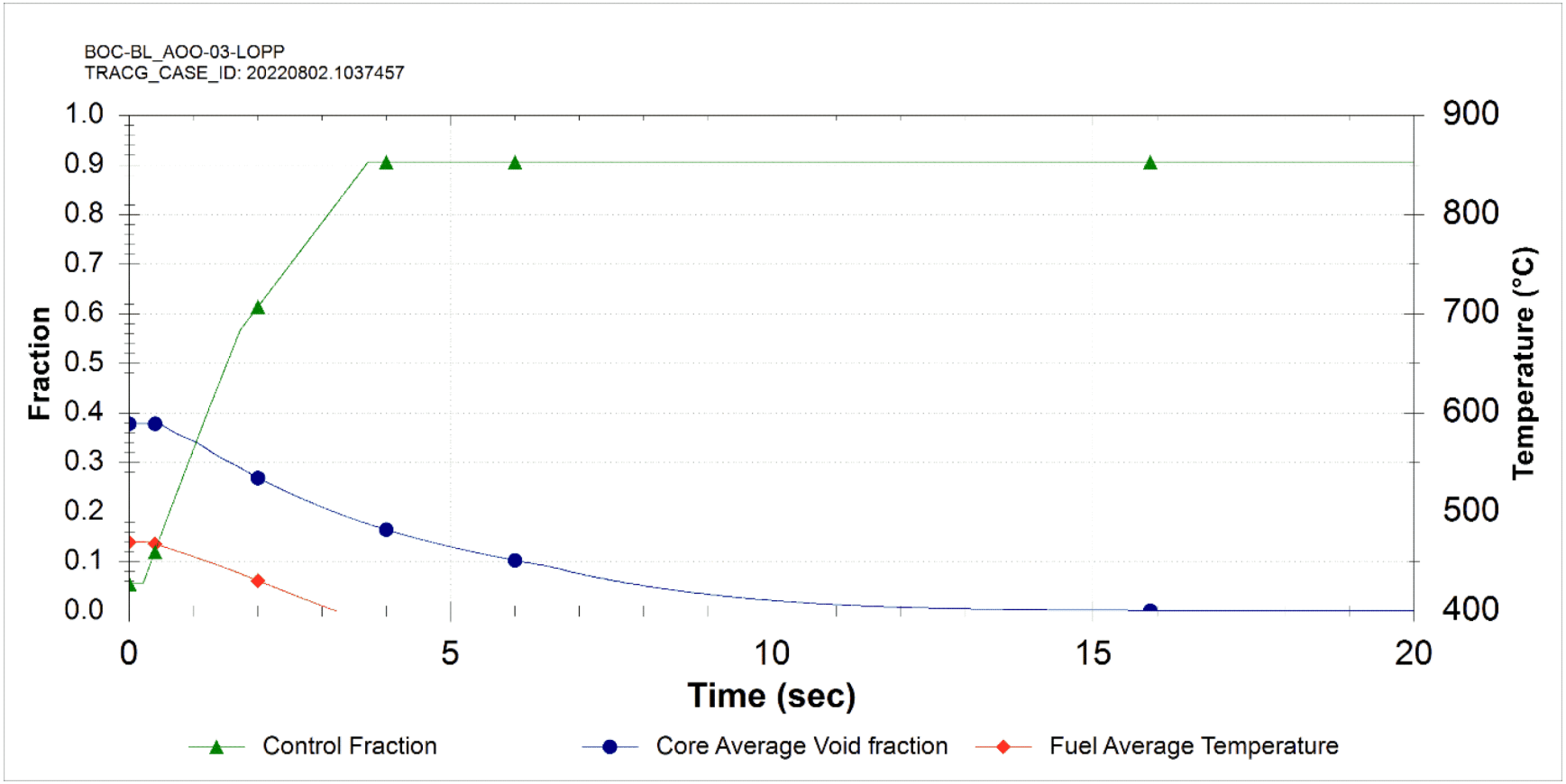

Figure 15.5-32 Loss of Preferred Power (AOO)

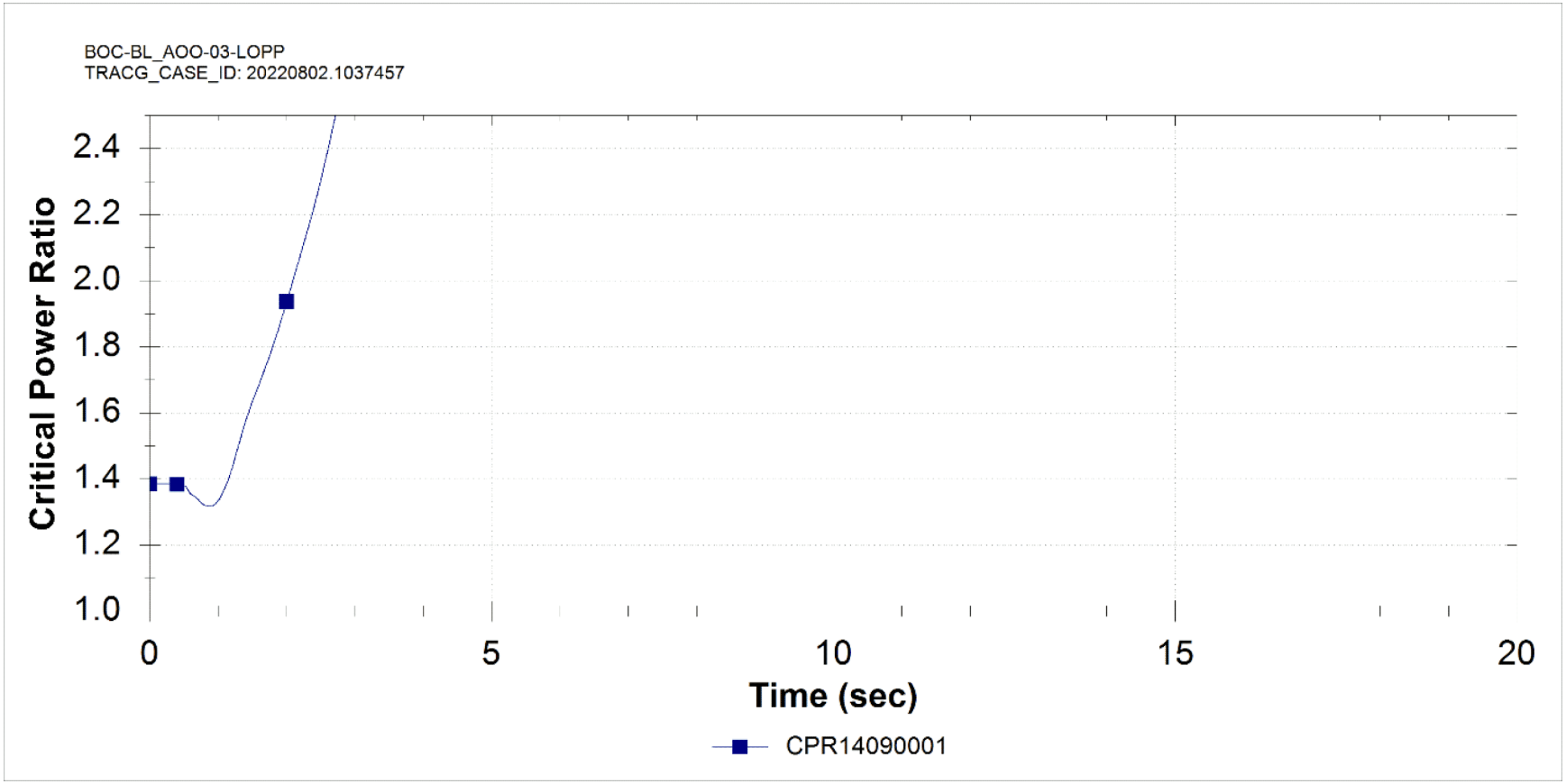

Figure 15.5-33 Loss of Preferred Power (AOO)

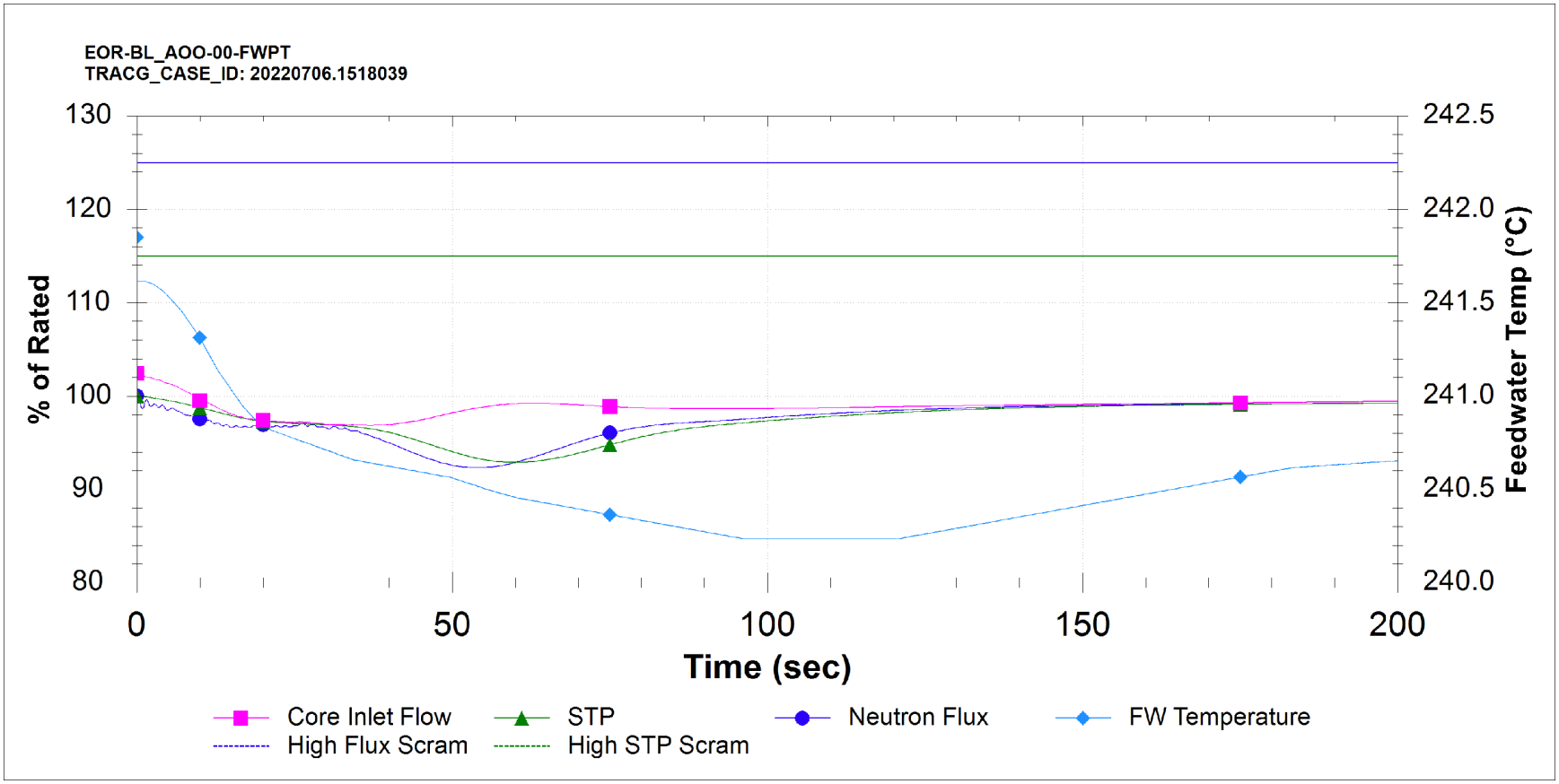

Figure 15.5-34 Feedwater Pump Trip – One Pump (AOO)

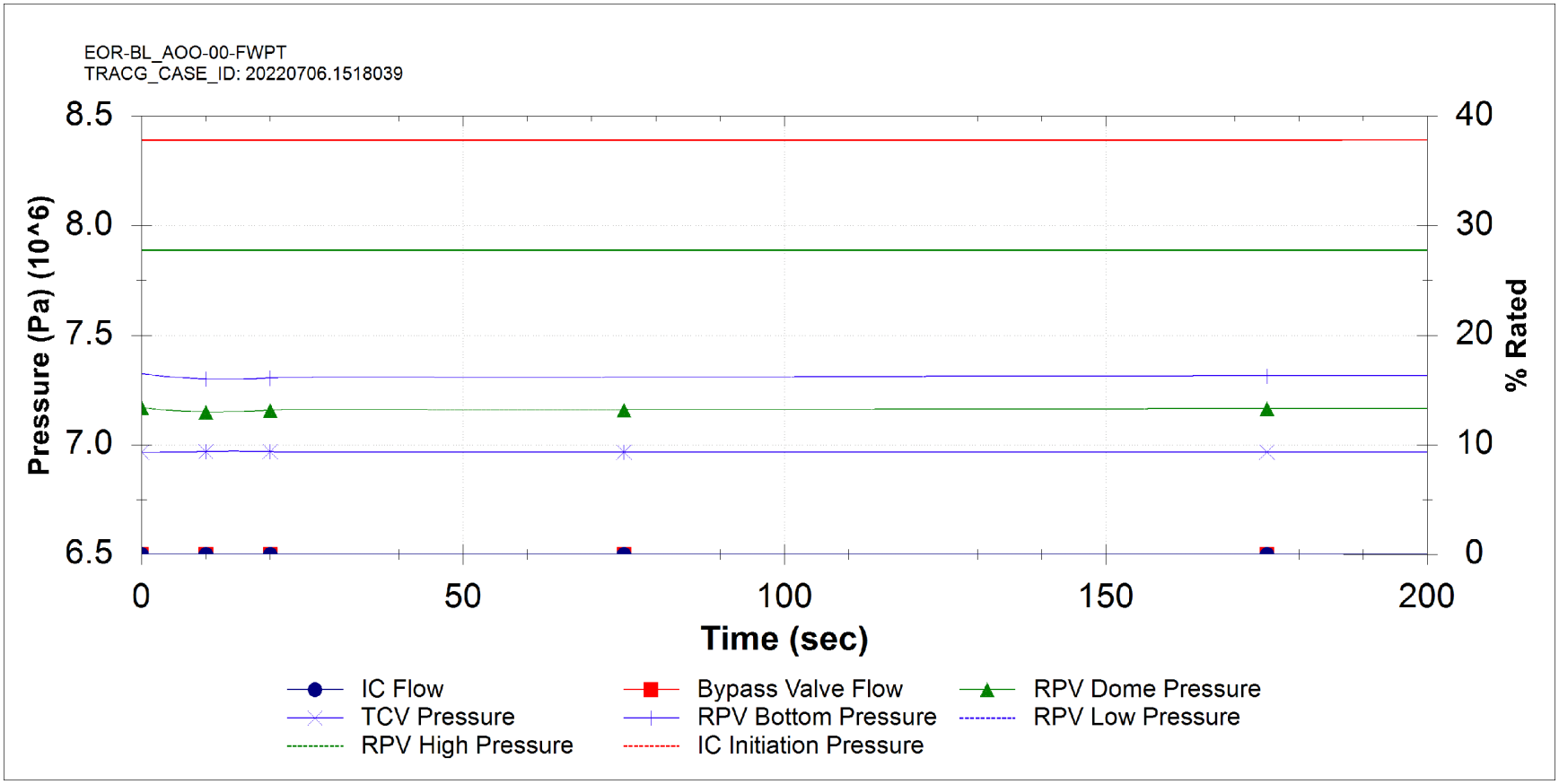

Figure 15.5-35 Feedwater Pump Trip – One Pump (AOO)

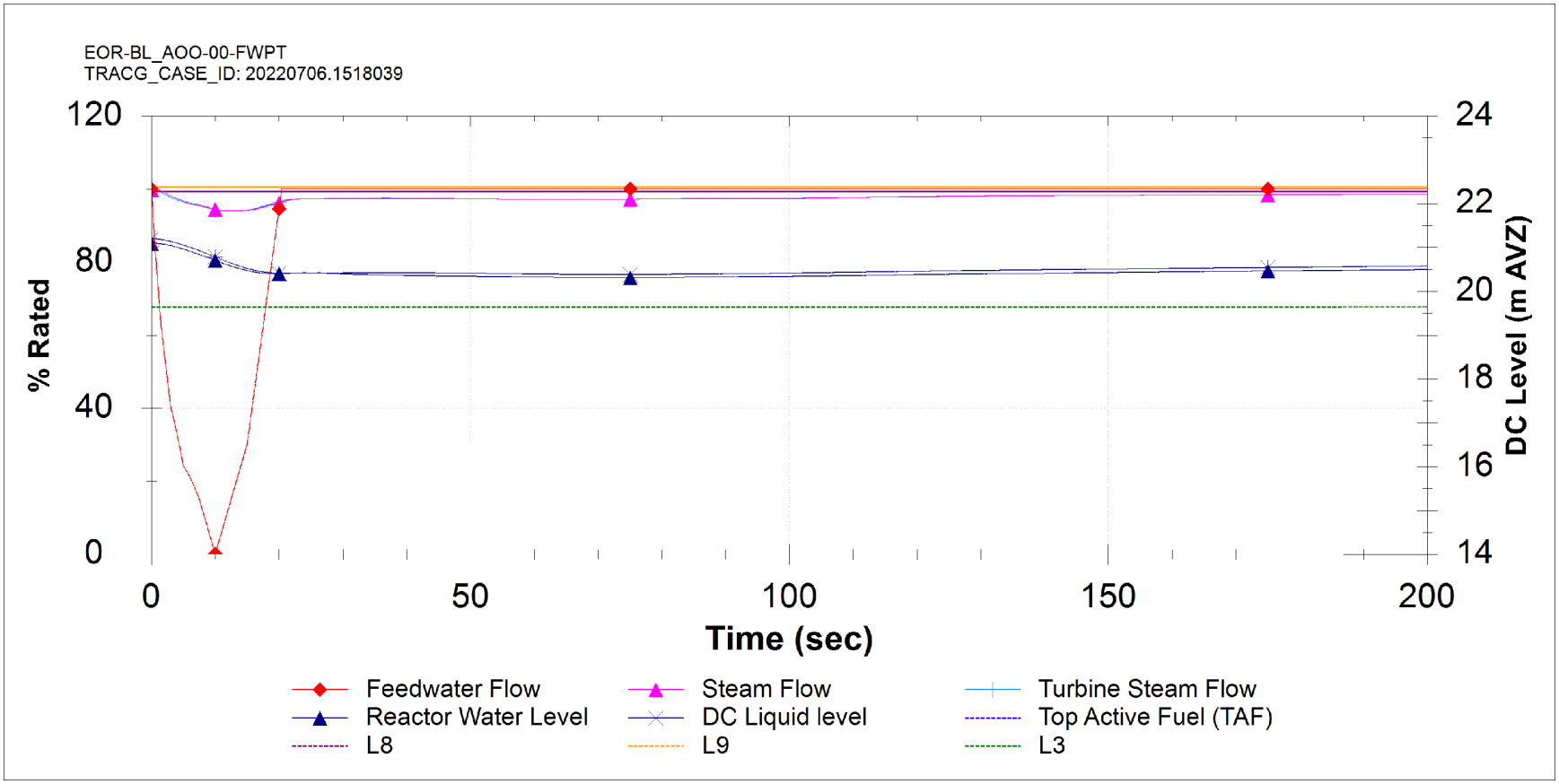

Figure 15.5-36 Feedwater Pump Trip – One Pump (AOO)

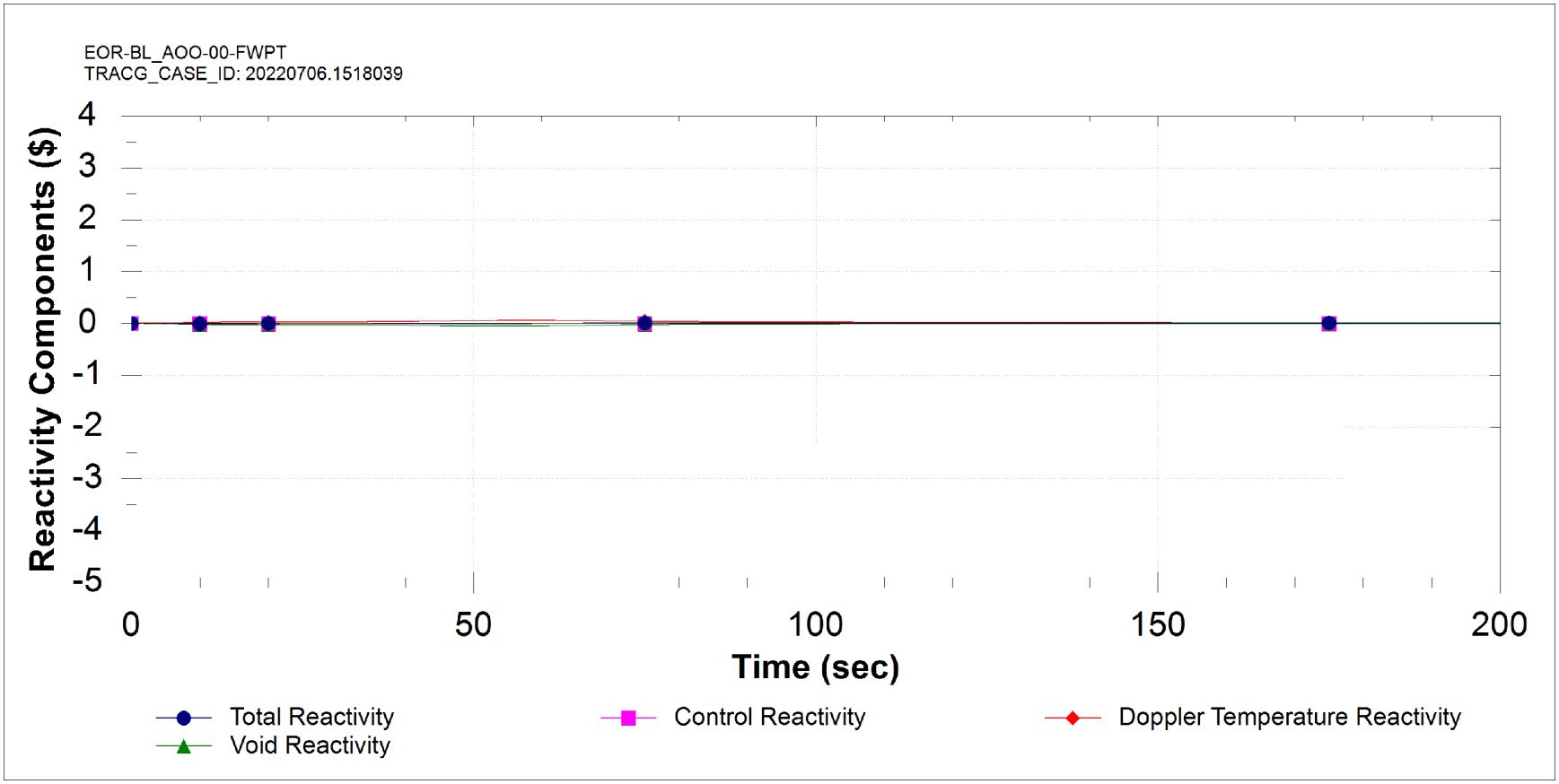

Figure 15.5-37 Feedwater Pump Trip – One Pump (AOO)

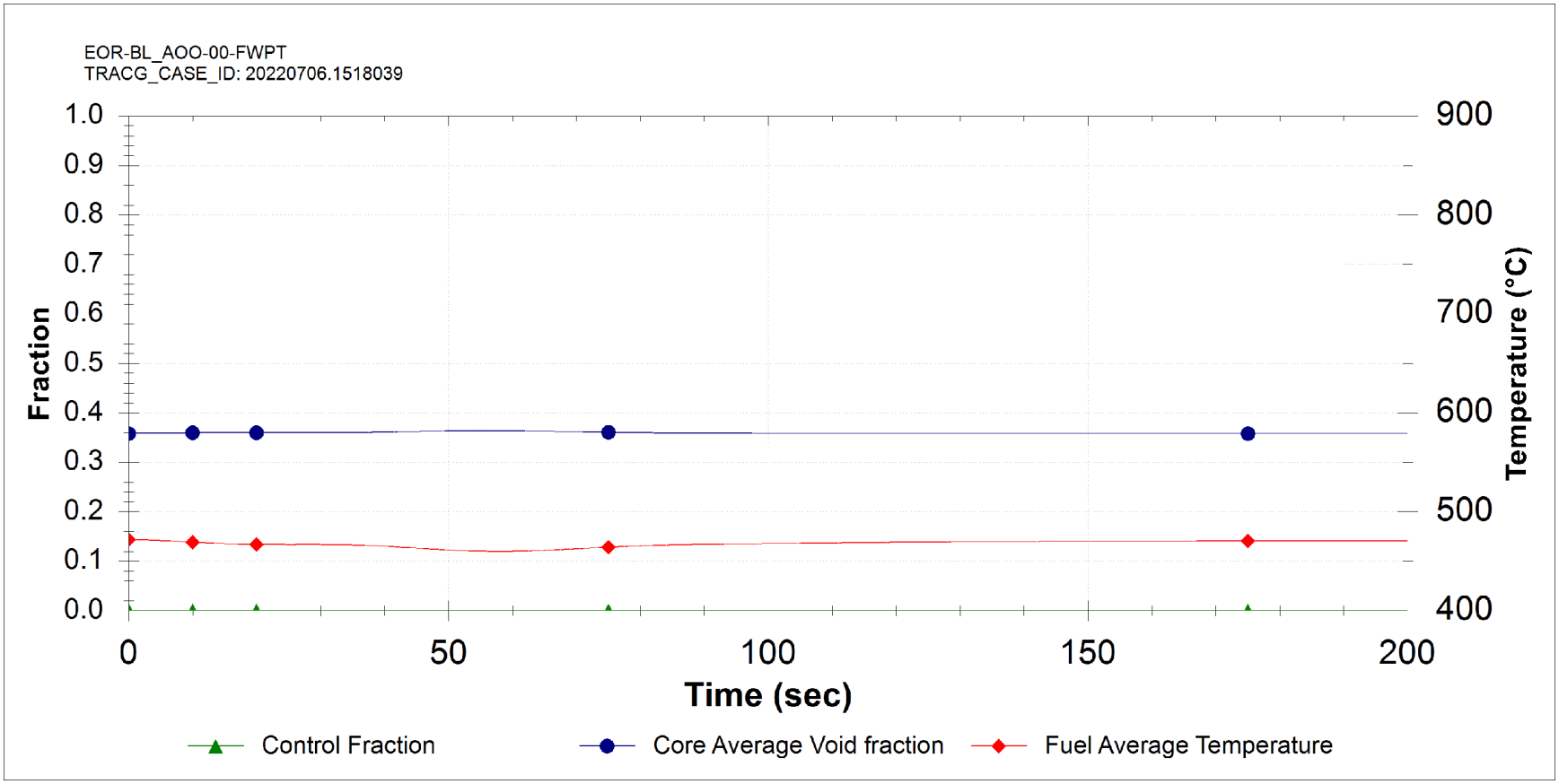

Figure 15.5-38 Feedwater Pump Trip – One Pump (AOO)

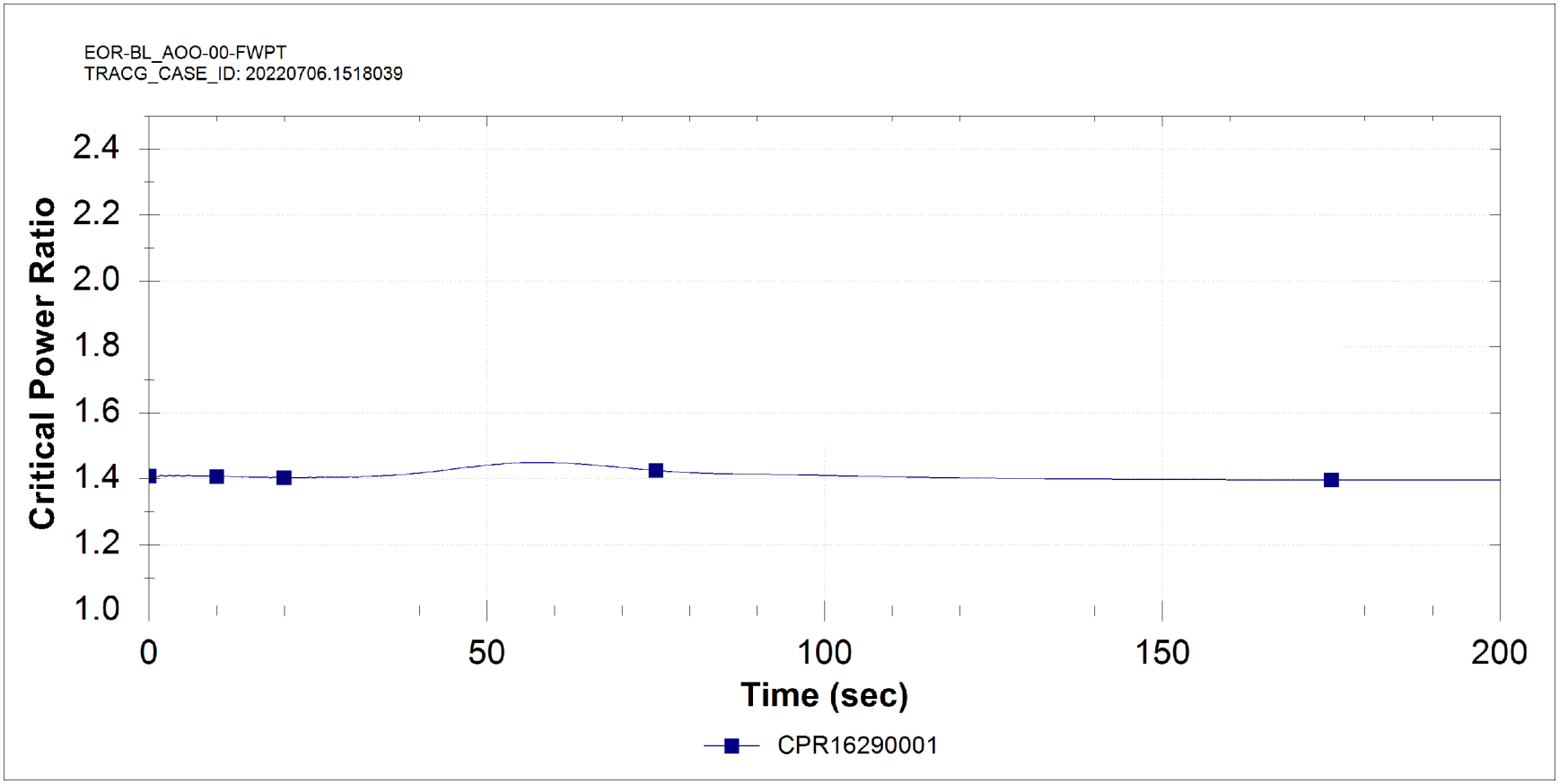

Figure 15.5-39 Feedwater Pump Trip – One Pump (AOO)

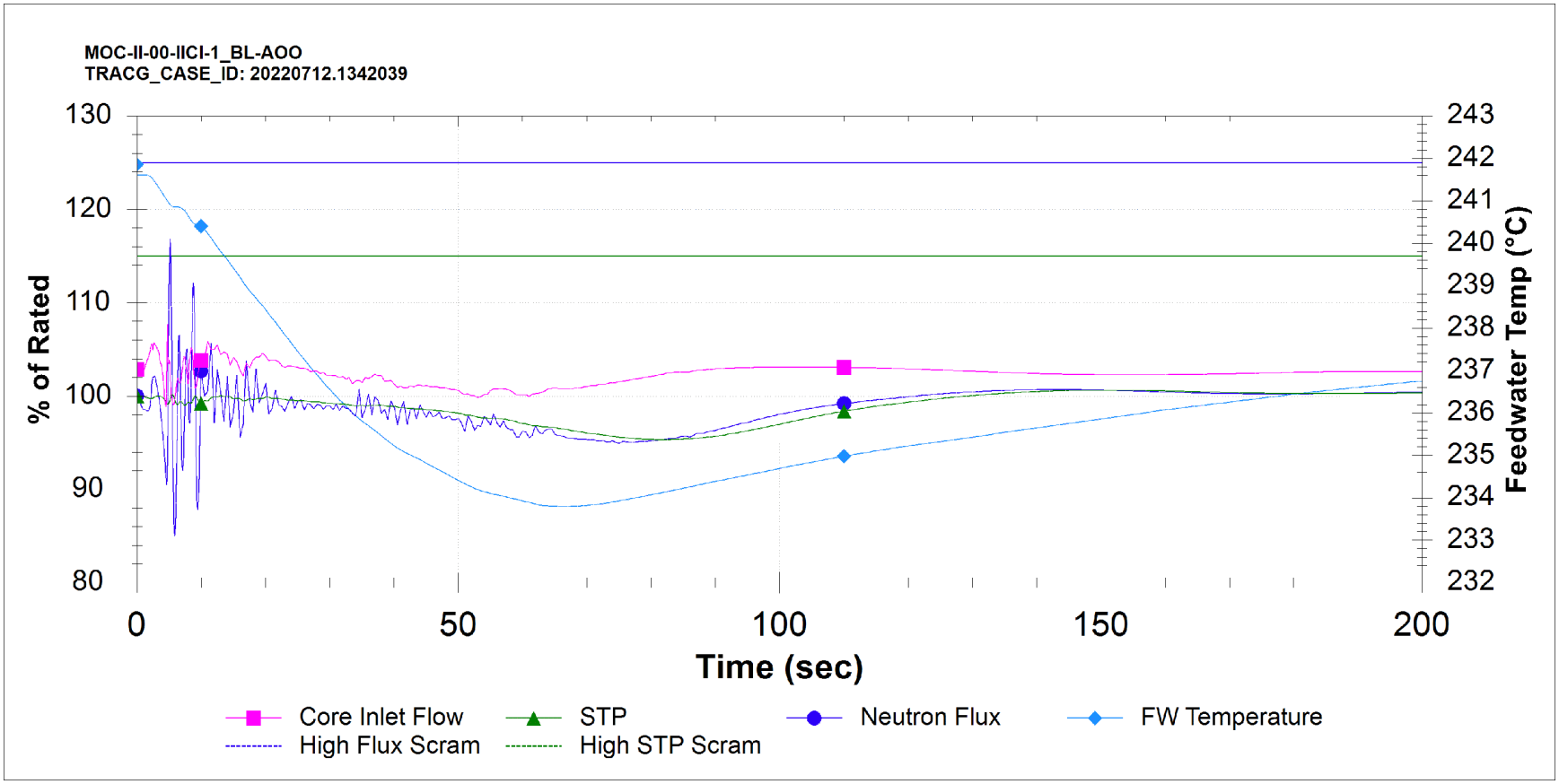

Figure 15.5-40 Inadvertent Isolation Condenser Initiation – One Train (AOO)

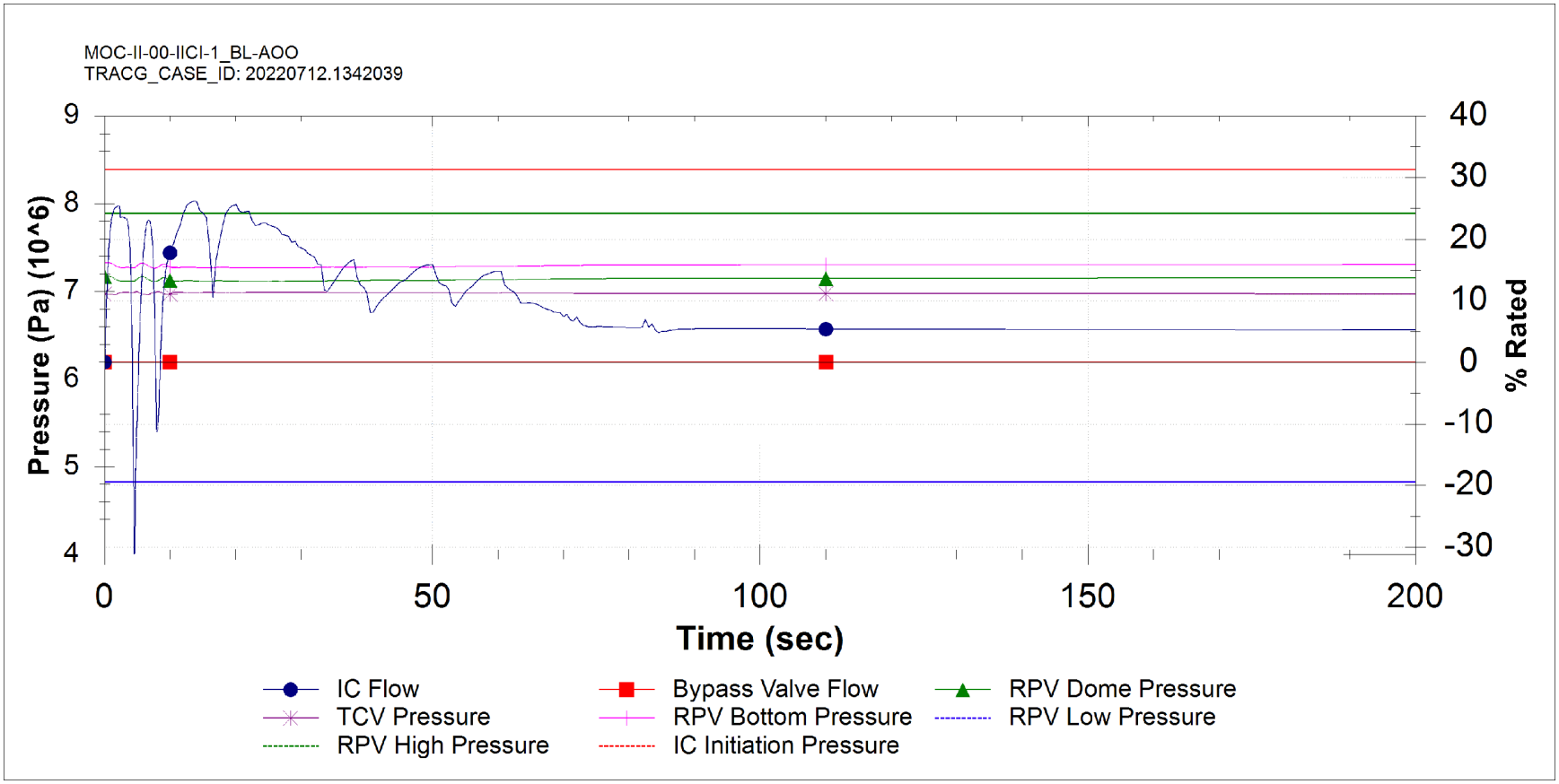

Figure 15.5-41 Inadvertent Isolation Condenser Initiation – One Train (AOO)

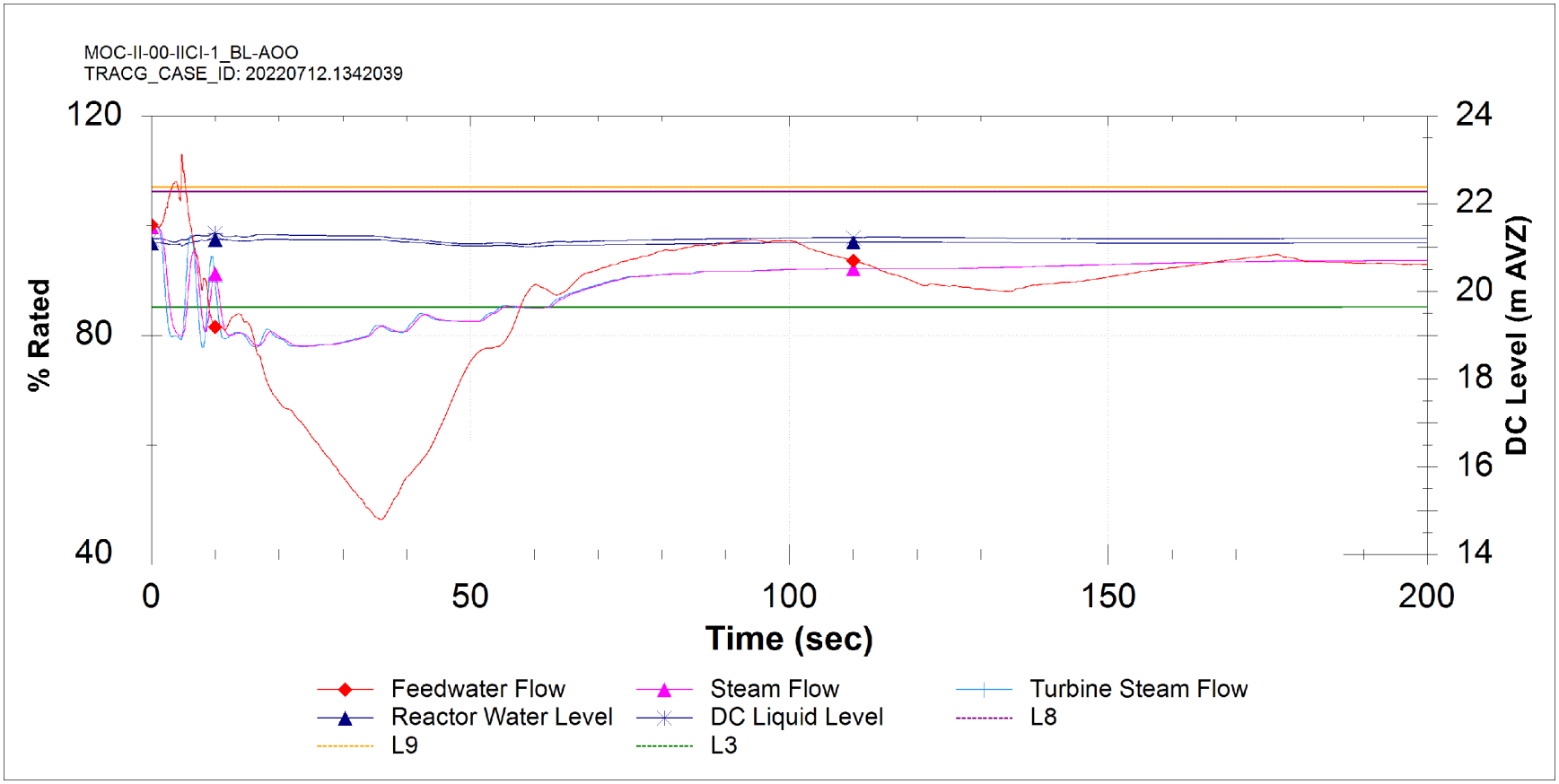

Figure 15.5-42 Inadvertent Isolation Condenser Initiation – One Train (AOO)

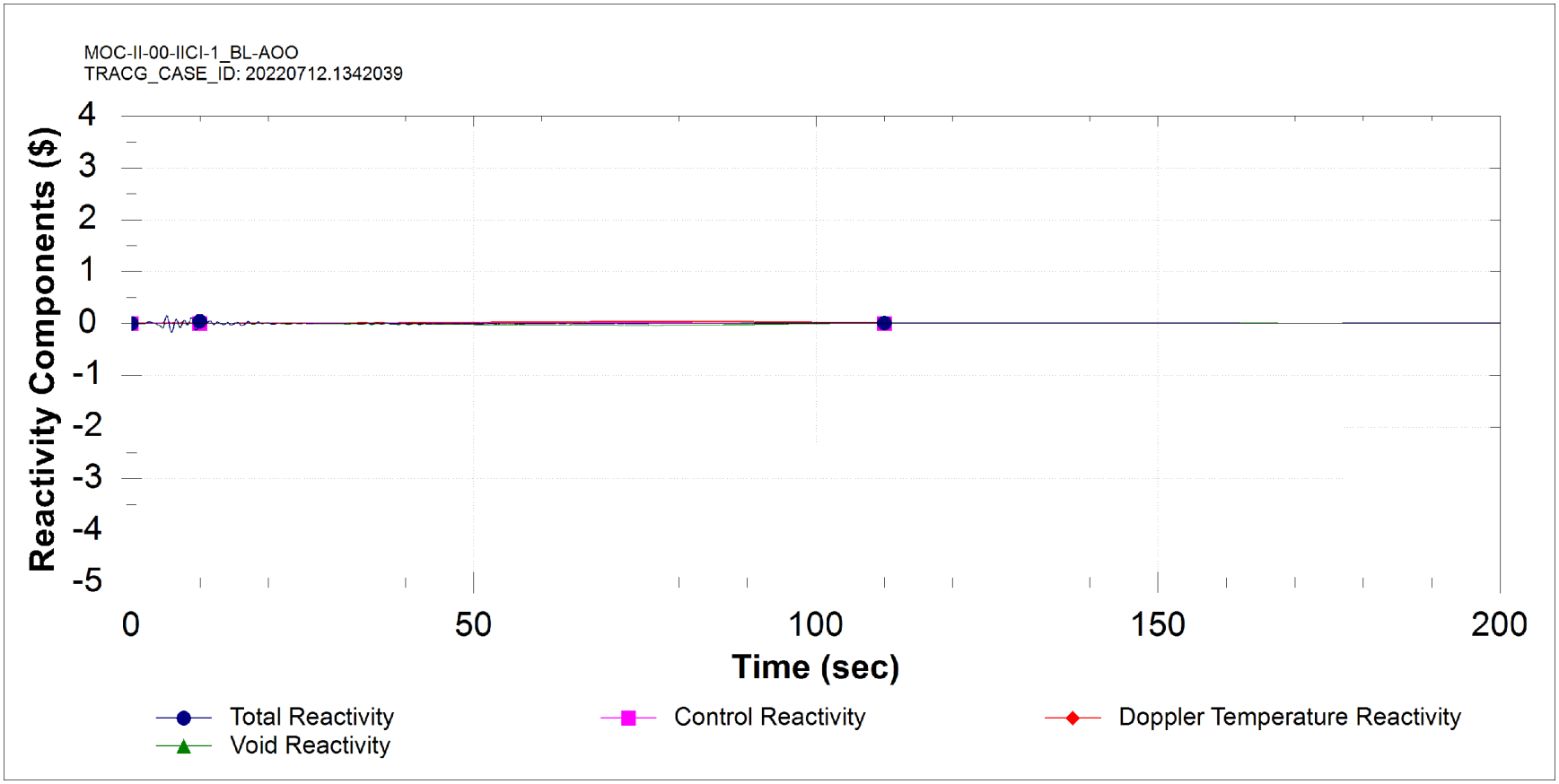

Figure 15.5-43 Inadvertent Isolation Condenser Initiation – One Train (AOO)

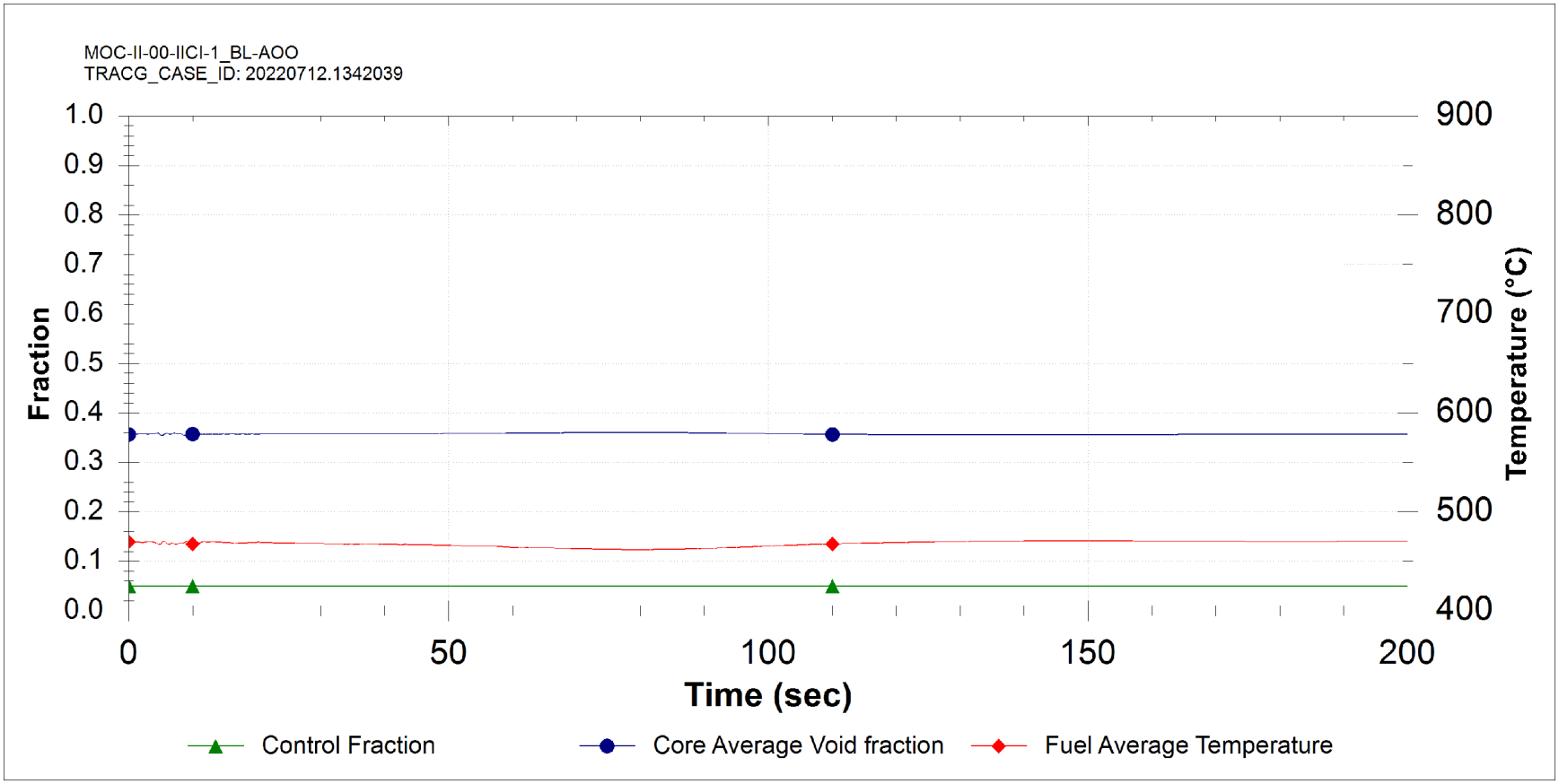

Figure 15.5-44 Inadvertent Isolation Condenser Initiation – One Train (AOO)

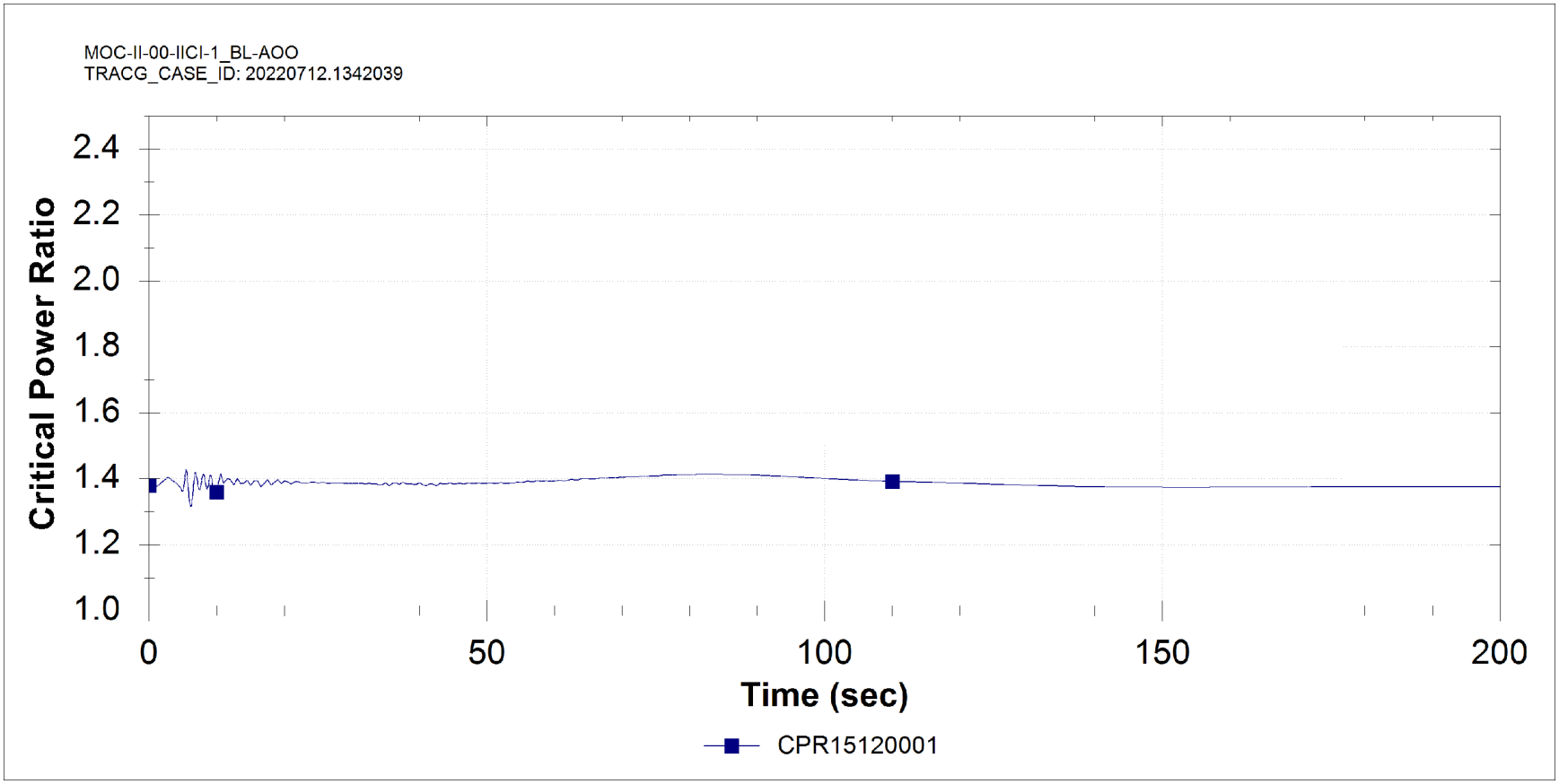

Figure 15.5-45 Inadvertent Isolation Condenser Initiation – One Train (AOO)

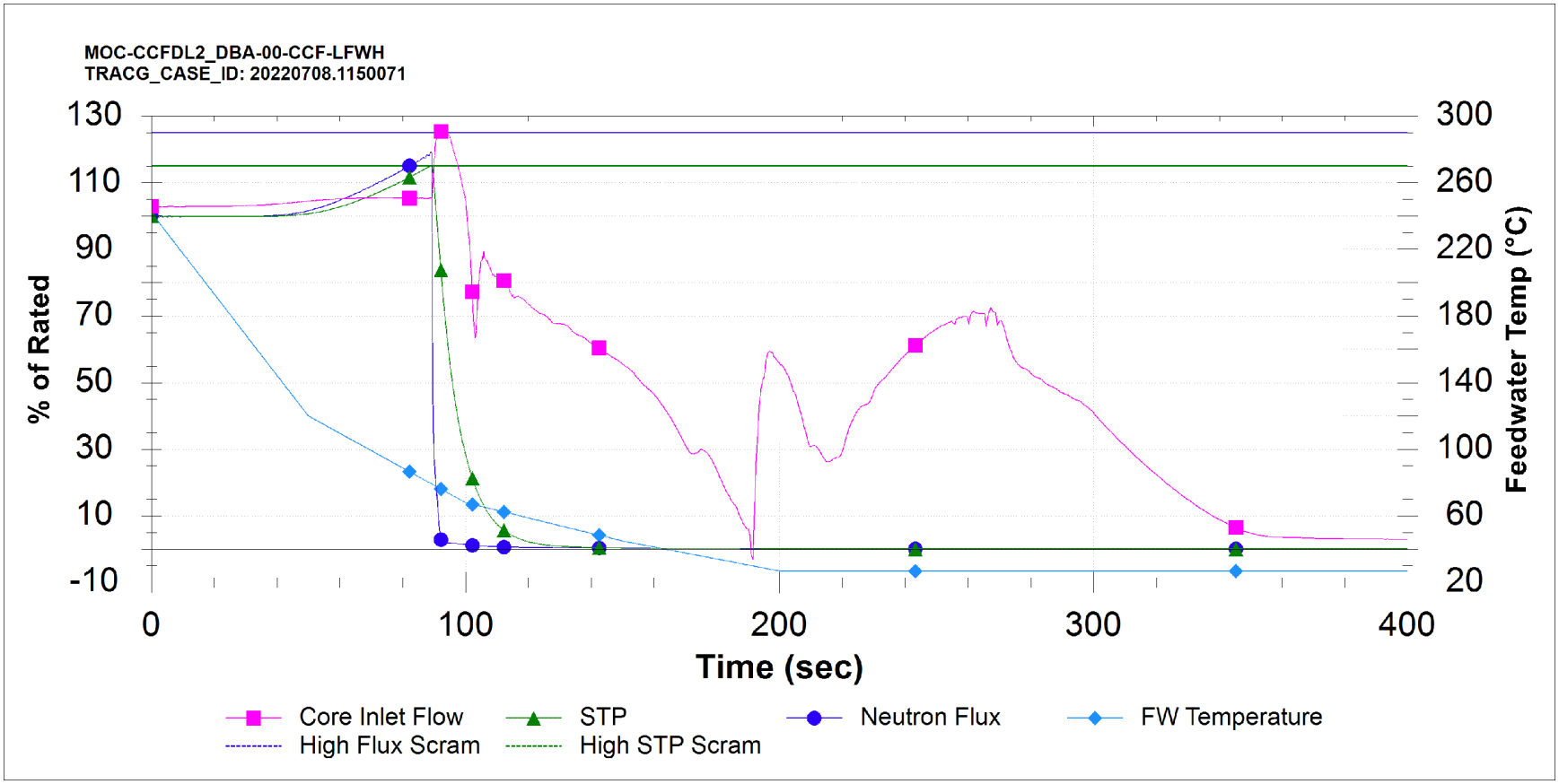

Figure 15.5-46 Loss of Feedwater Heating (DBA)

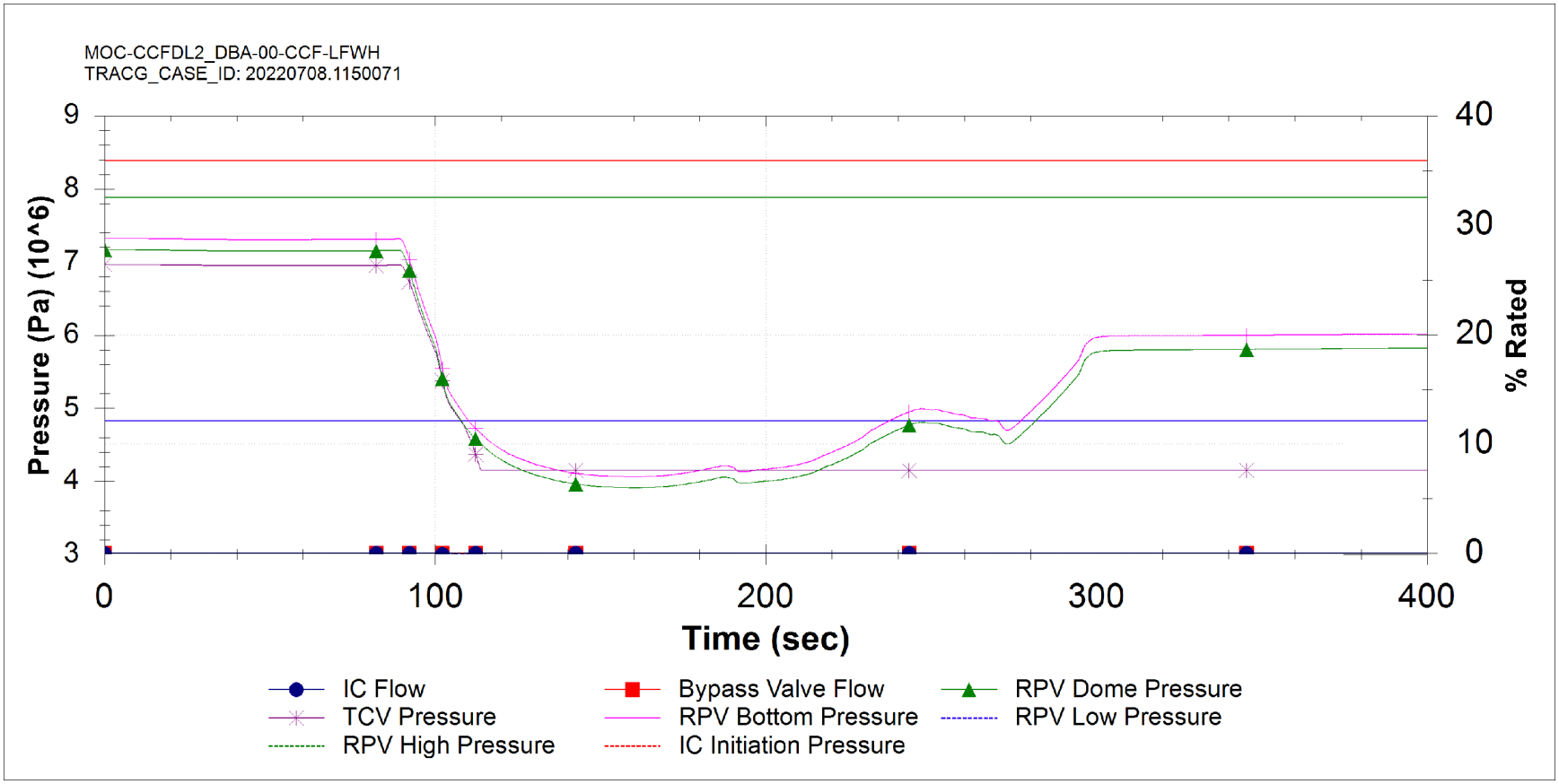

Figure 15.5-47 Loss of Feedwater Heating (DBA)

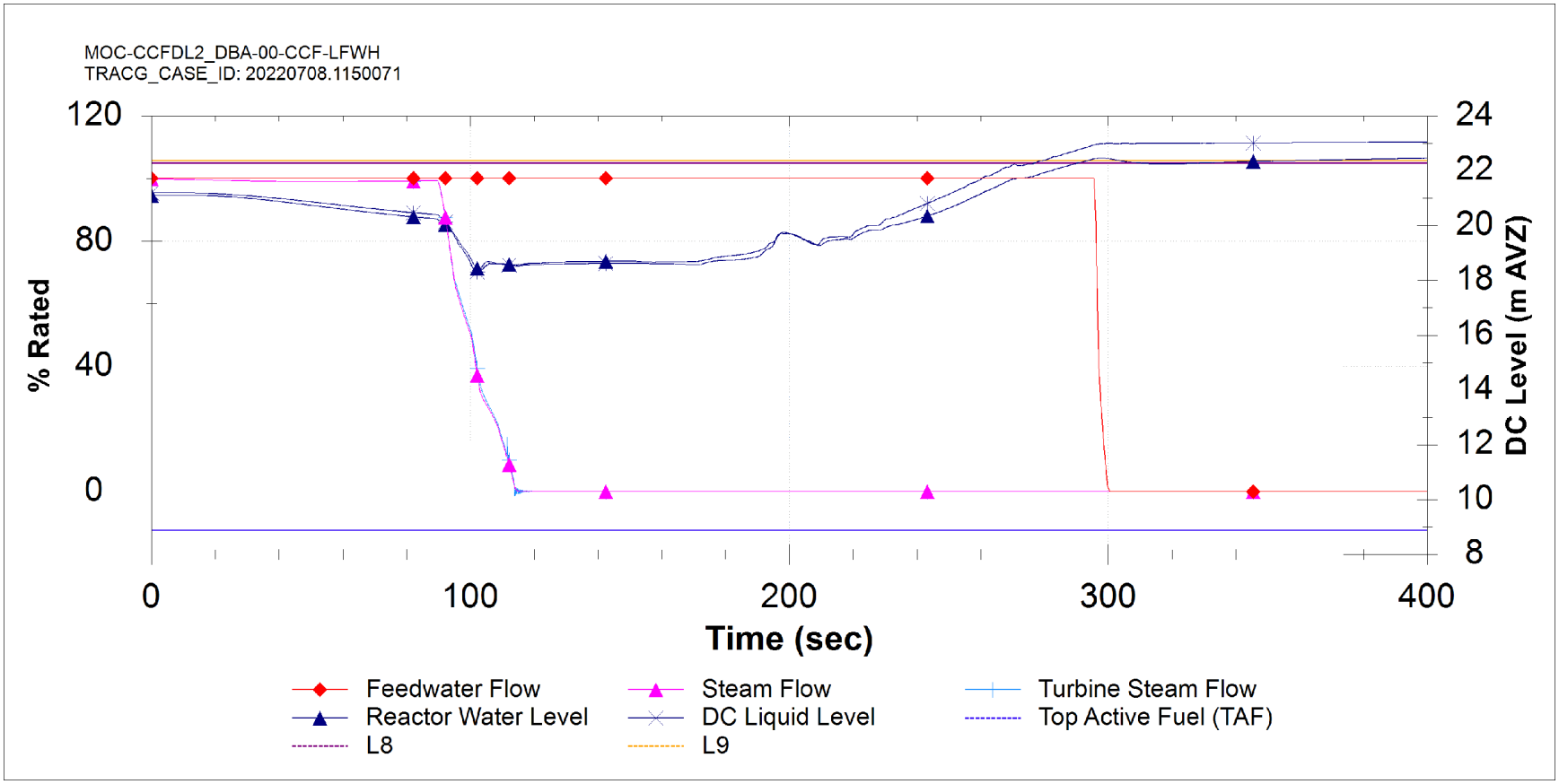

Figure 15.5-48 Loss of Feedwater Heating (DBA)

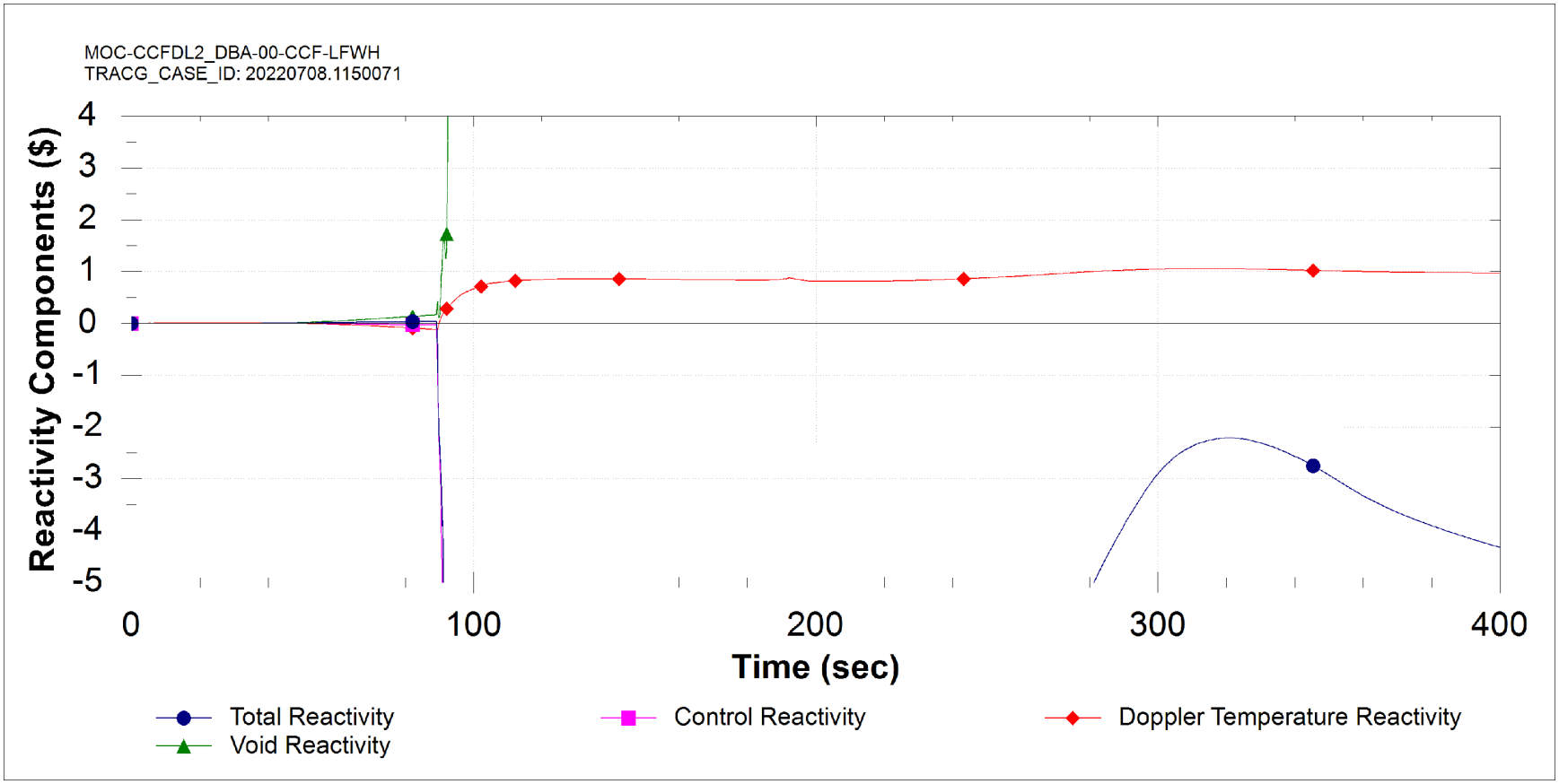

Figure 15.5-49 Loss of Feedwater Heating (DBA)

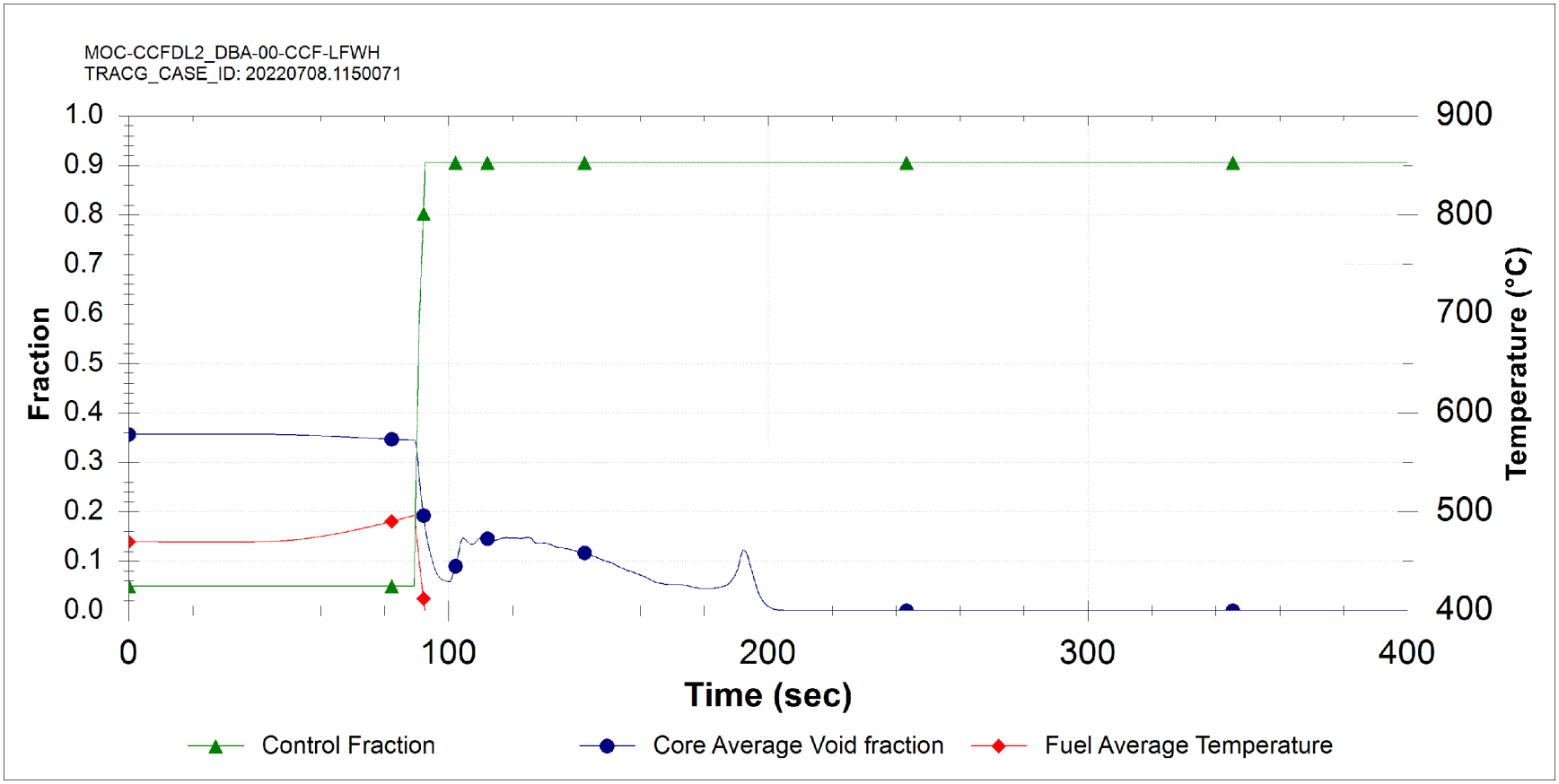

Figure 15.5-50 Loss of Feedwater Heating (DBA)

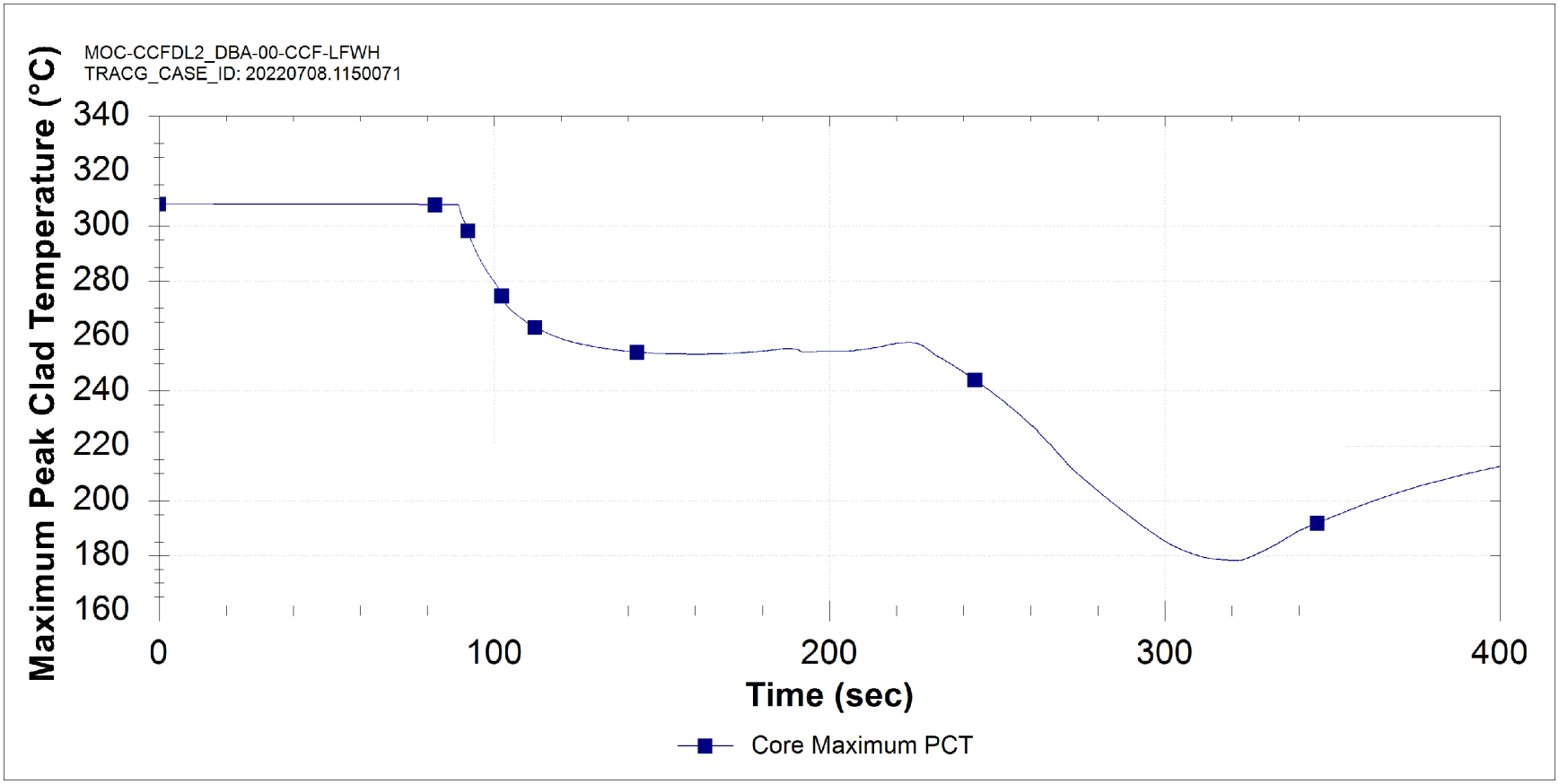

Figure 15.5-51 Loss of Feedwater Heating (DBA)

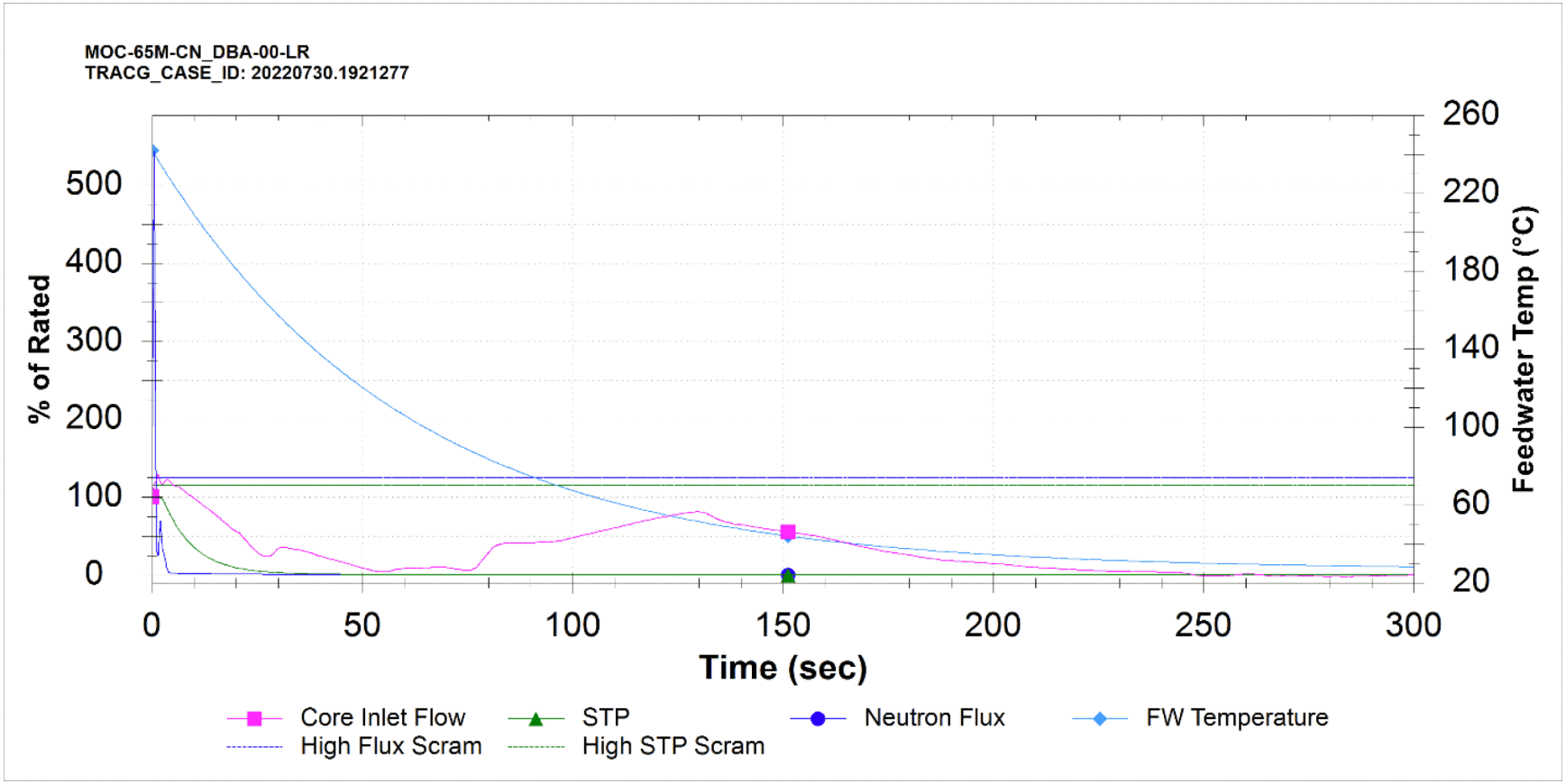

Figure 15.5-52 Generator Load Rejection (DBA)

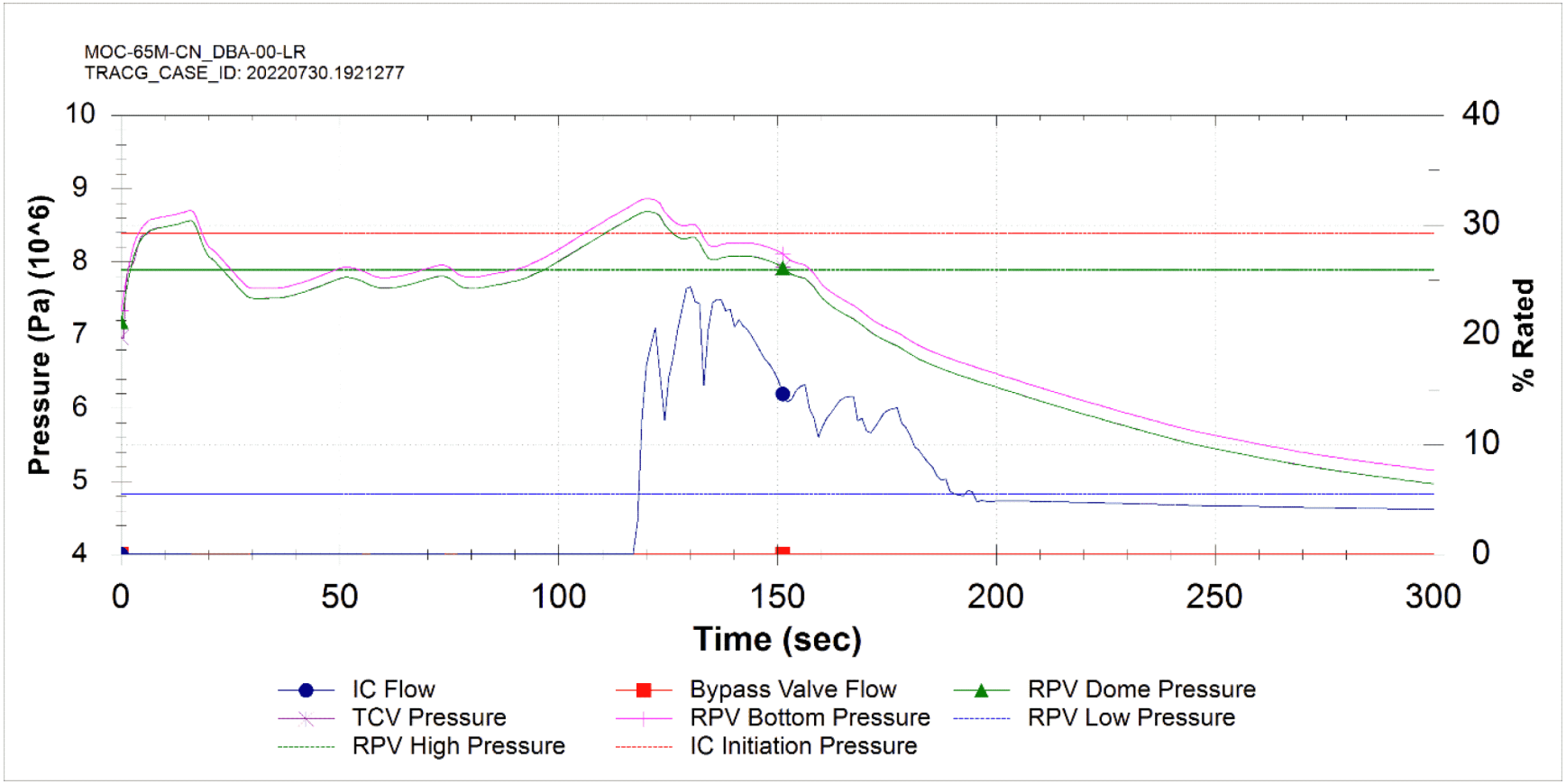

Figure 15.5-53 Generator Load Rejection (DBA)

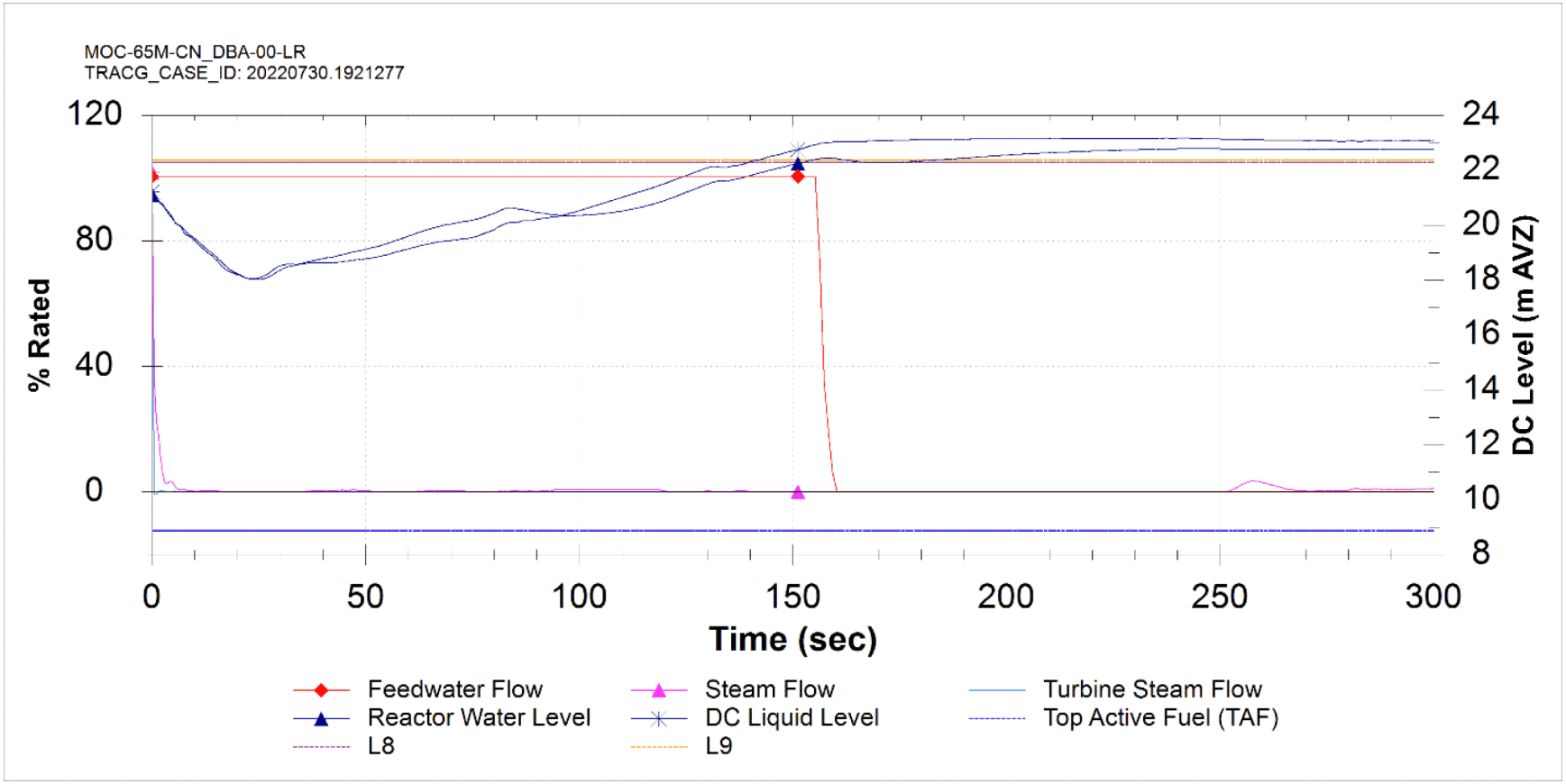

Figure 15.5-54 Generator Load Rejection (DBA)

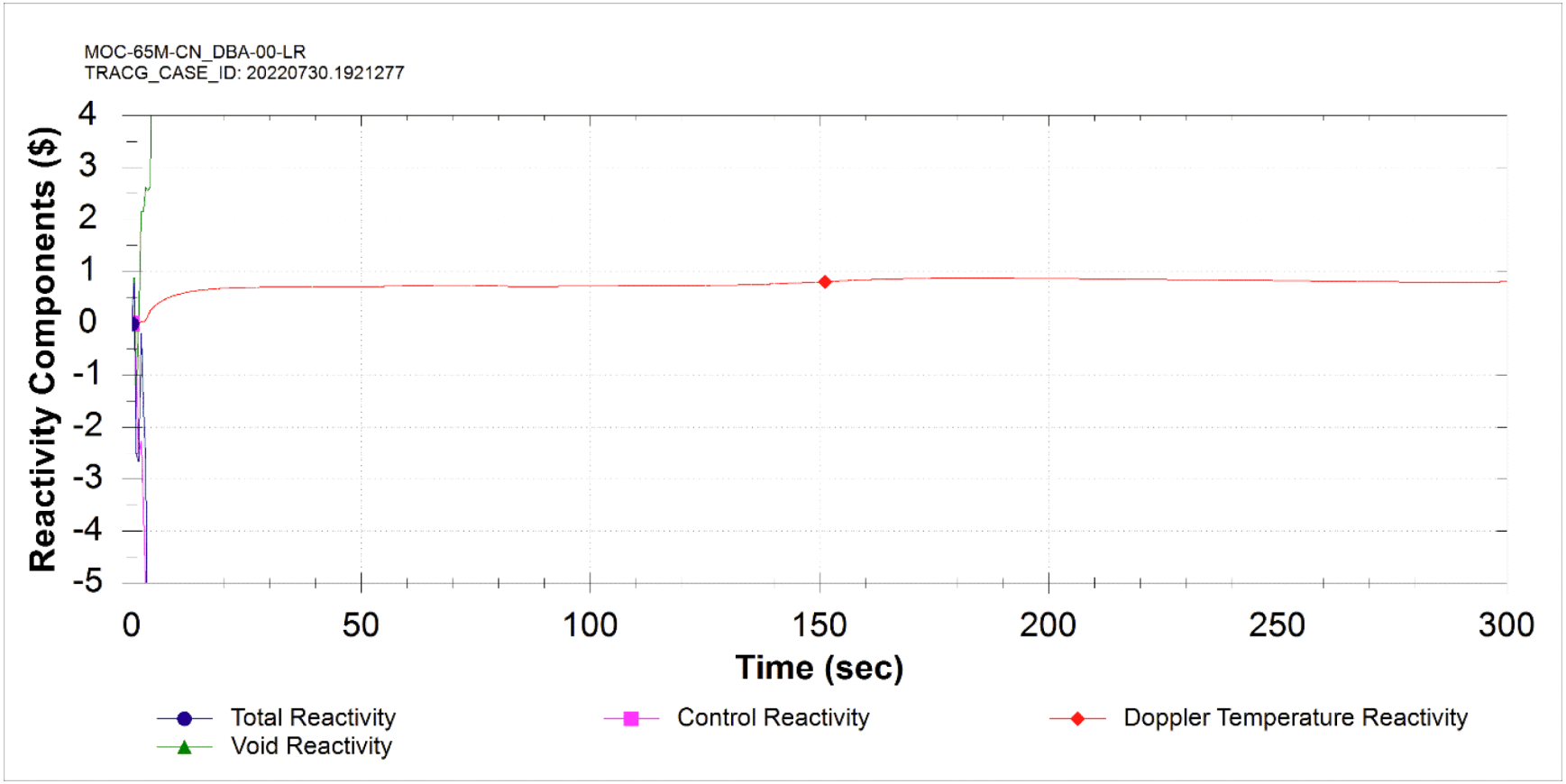

Figure 15.5-55 Generator Load Rejection (DBA)

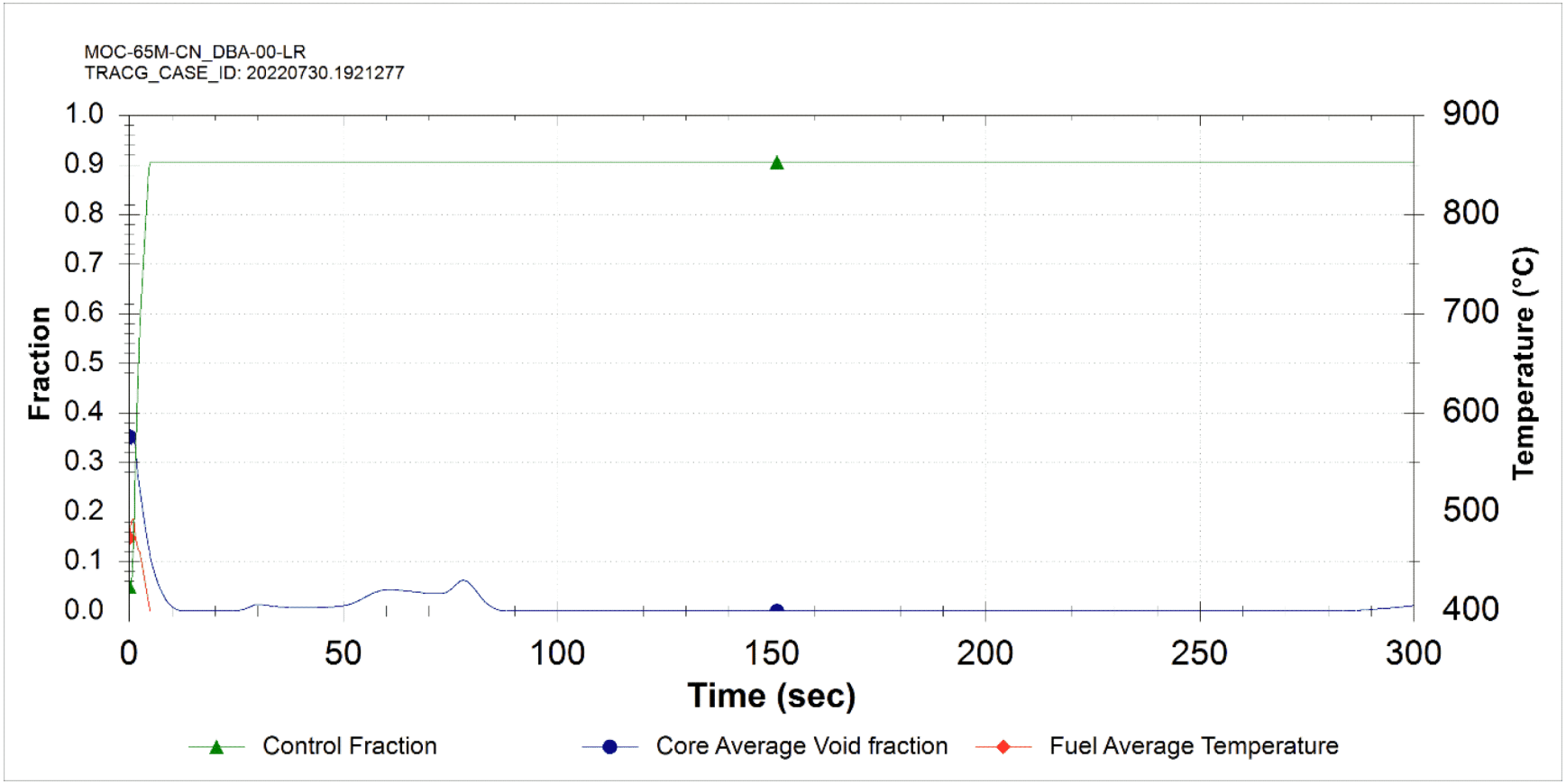

Figure 15.5-56 Generator Load Rejection (DBA)

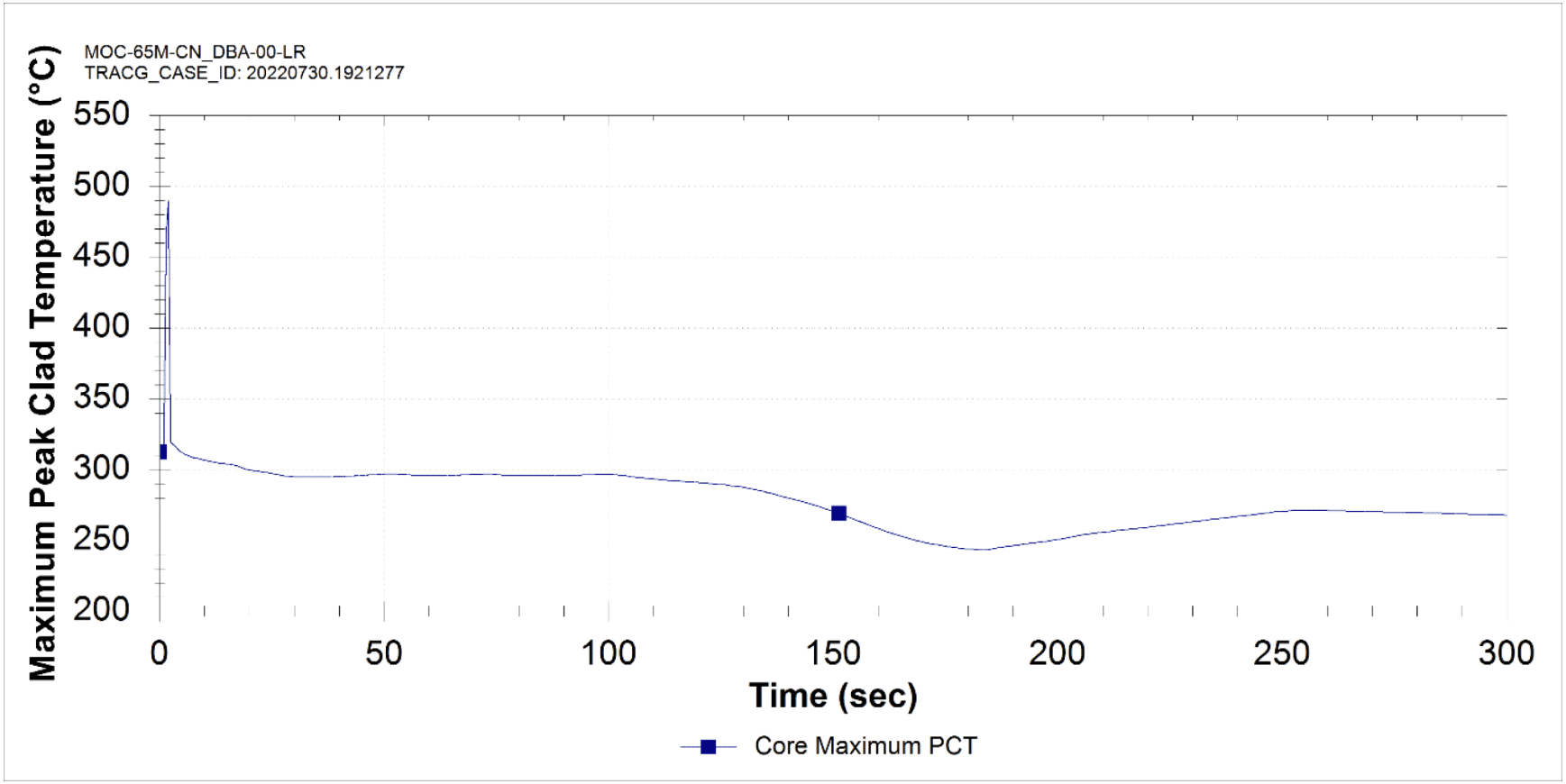

Figure 15.5-57 Generator Load Rejection (DBA)

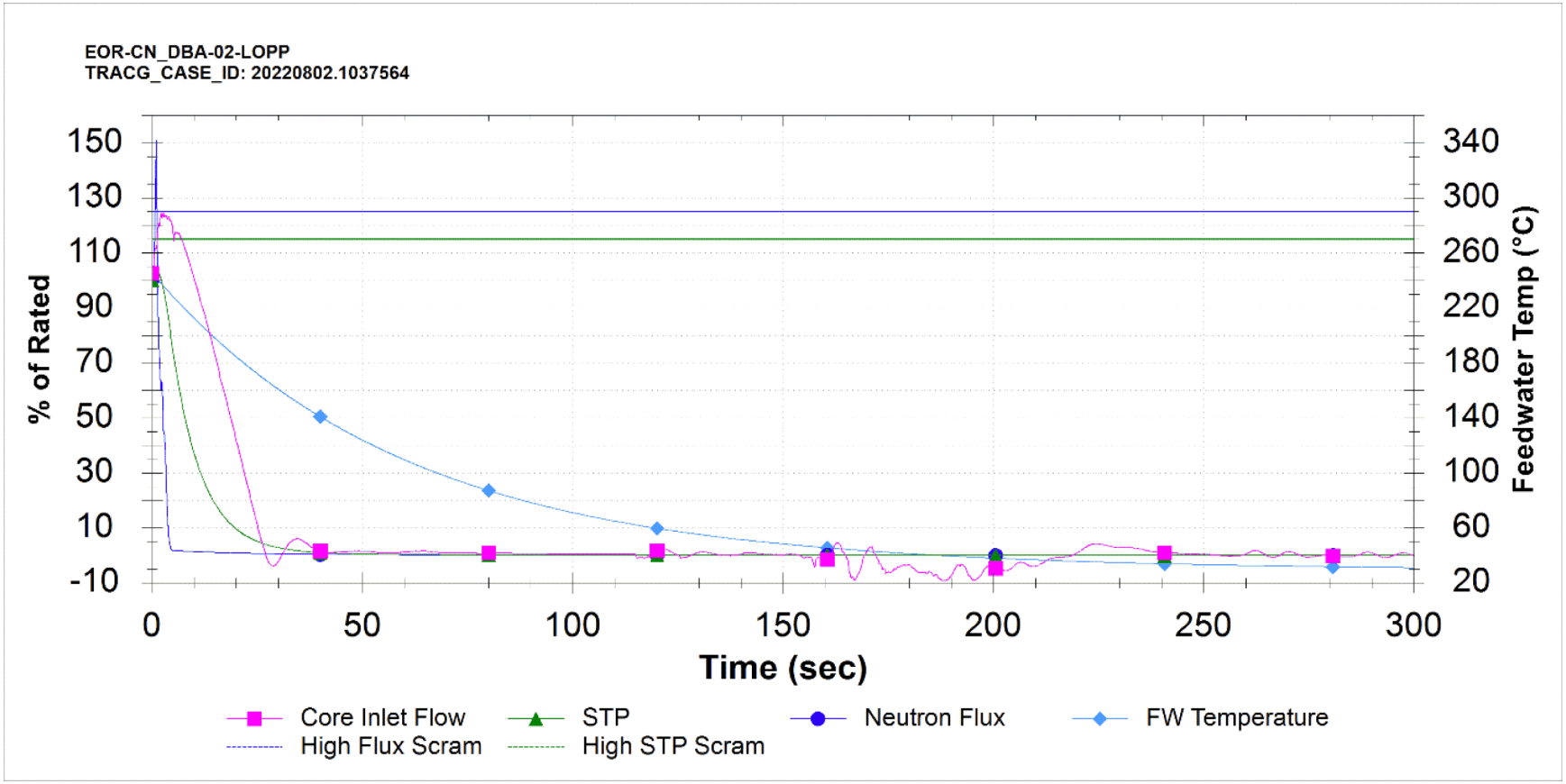

Figure 15.5-58 Loss of Preferred Power (DBA)

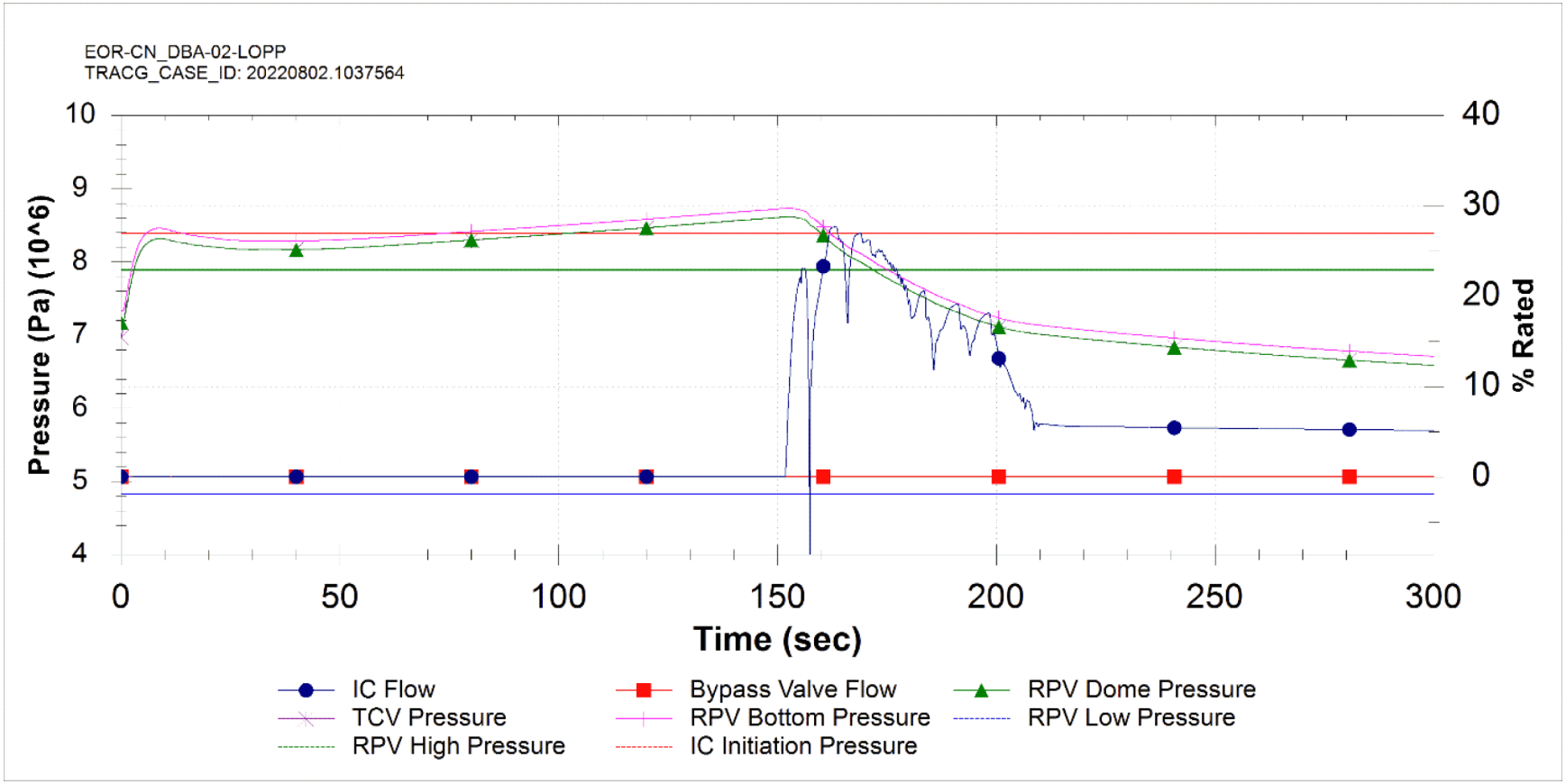

Figure 15.5-59 Loss of Preferred Power (DBA)

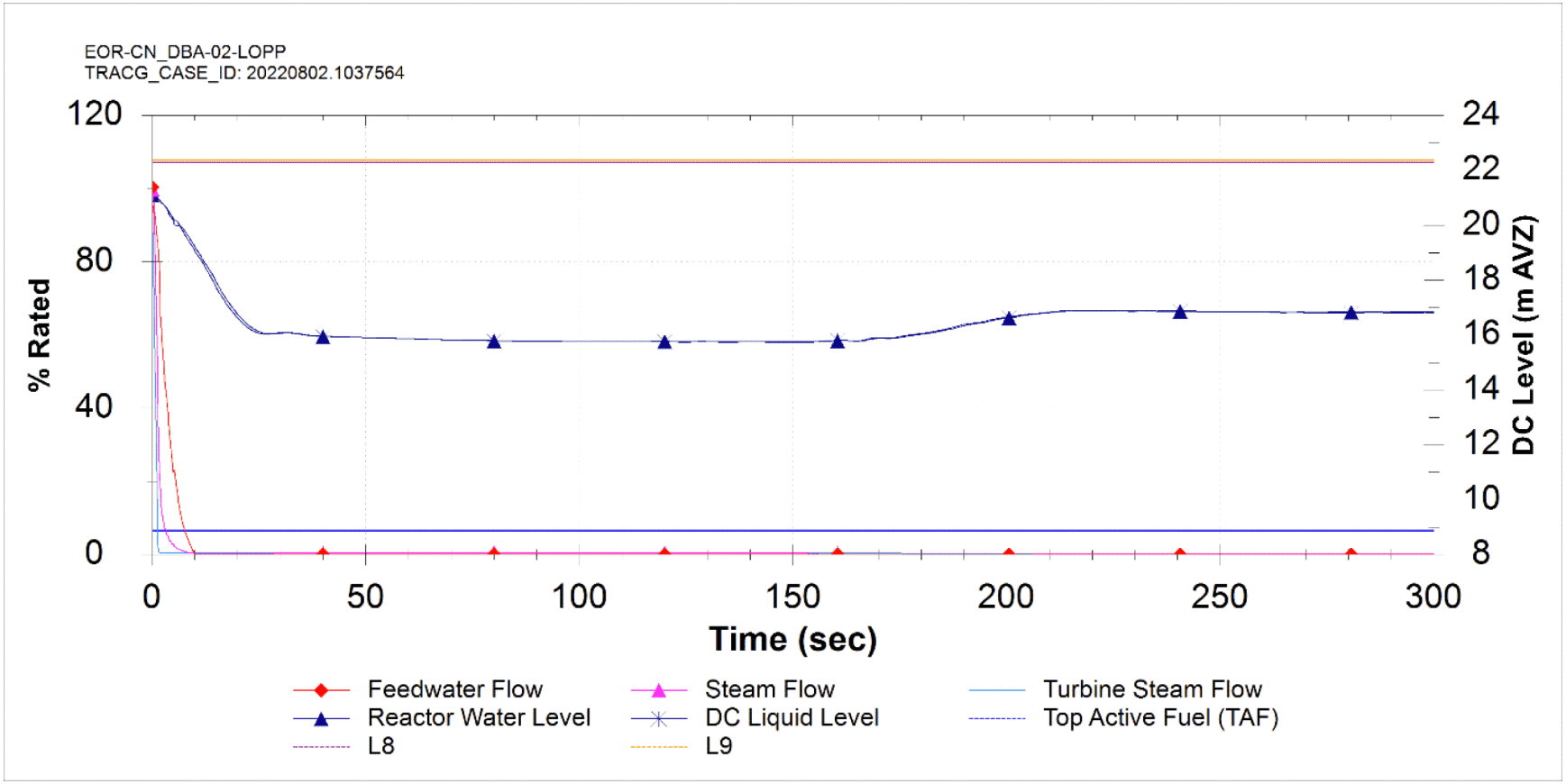

Figure 15.5-60 Loss of Preferred Power (DBA)

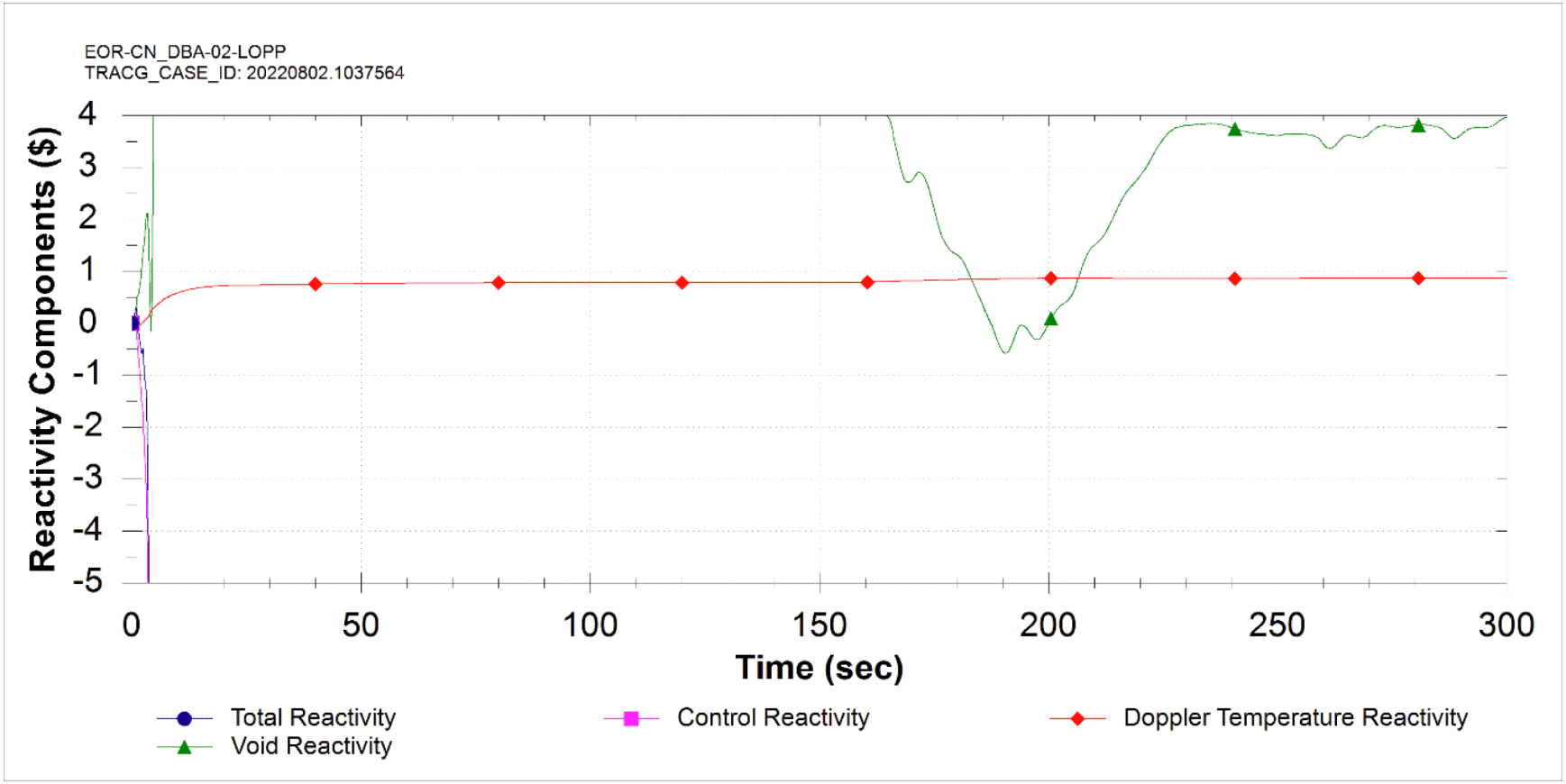

Figure 15.5-61 Loss of Preferred Power (DBA)

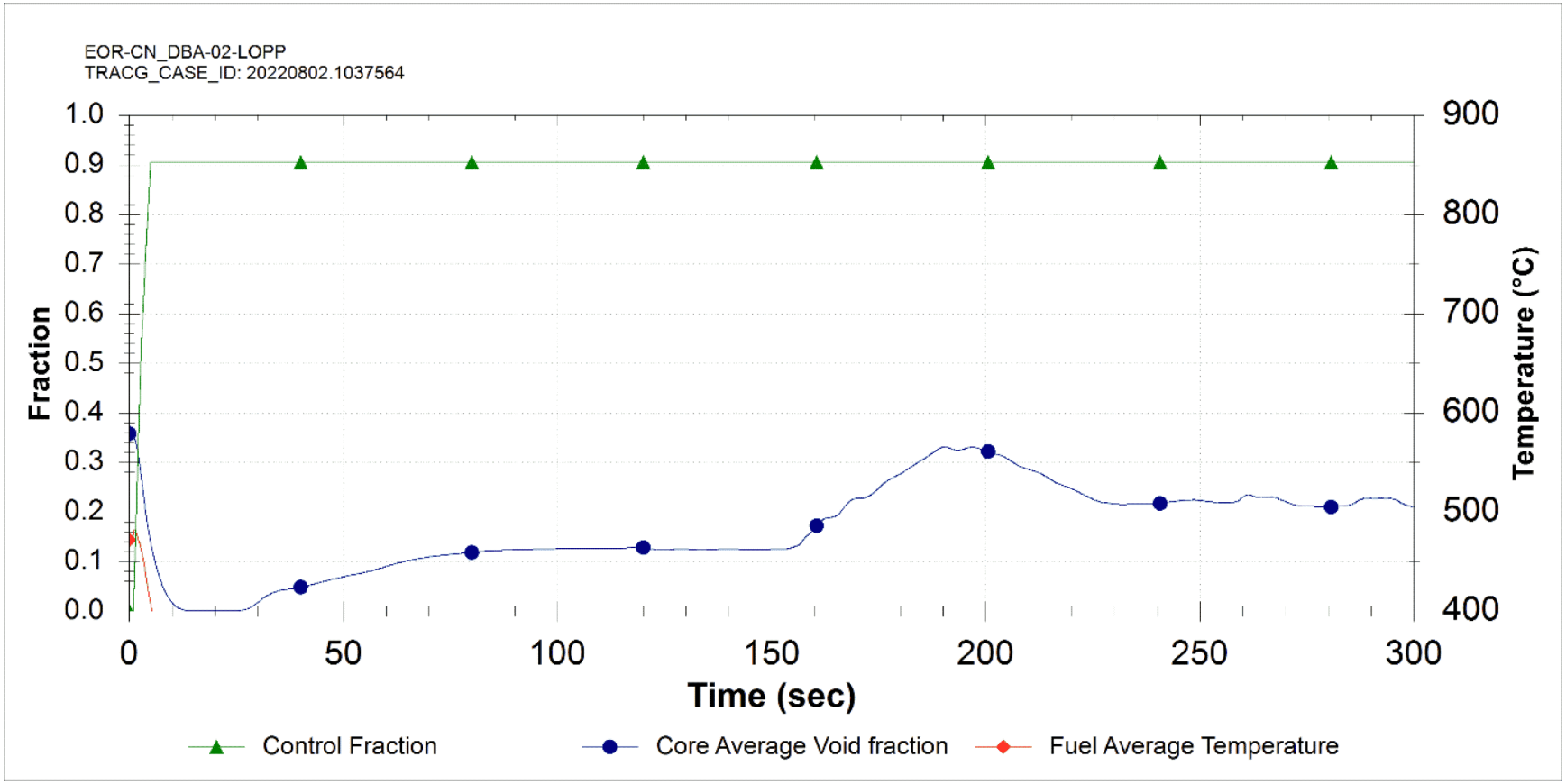

Figure 15.5-62 Loss of Preferred Power (DBA)

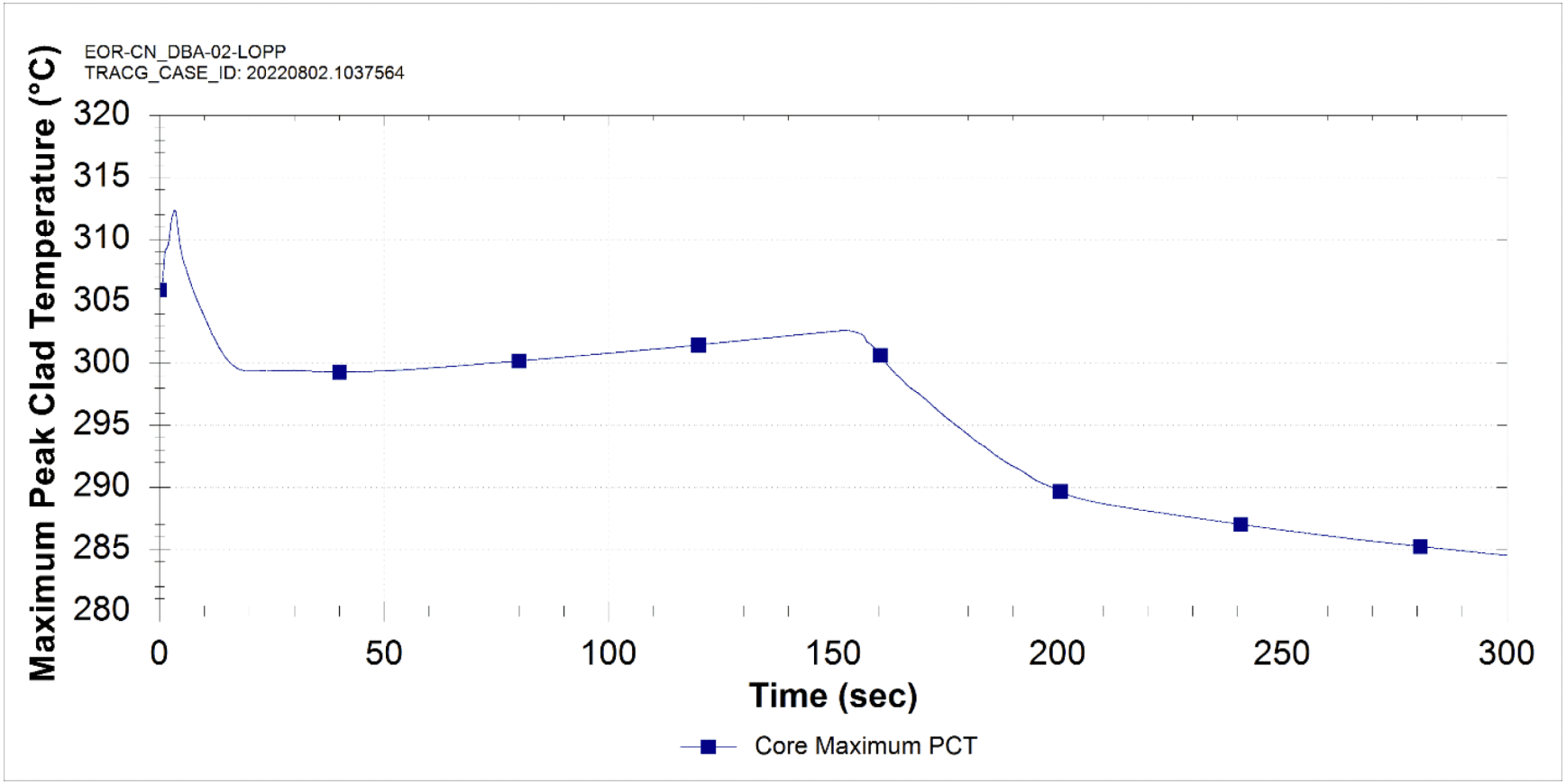

Figure 15.5-63 Loss of Preferred Power (DBA)

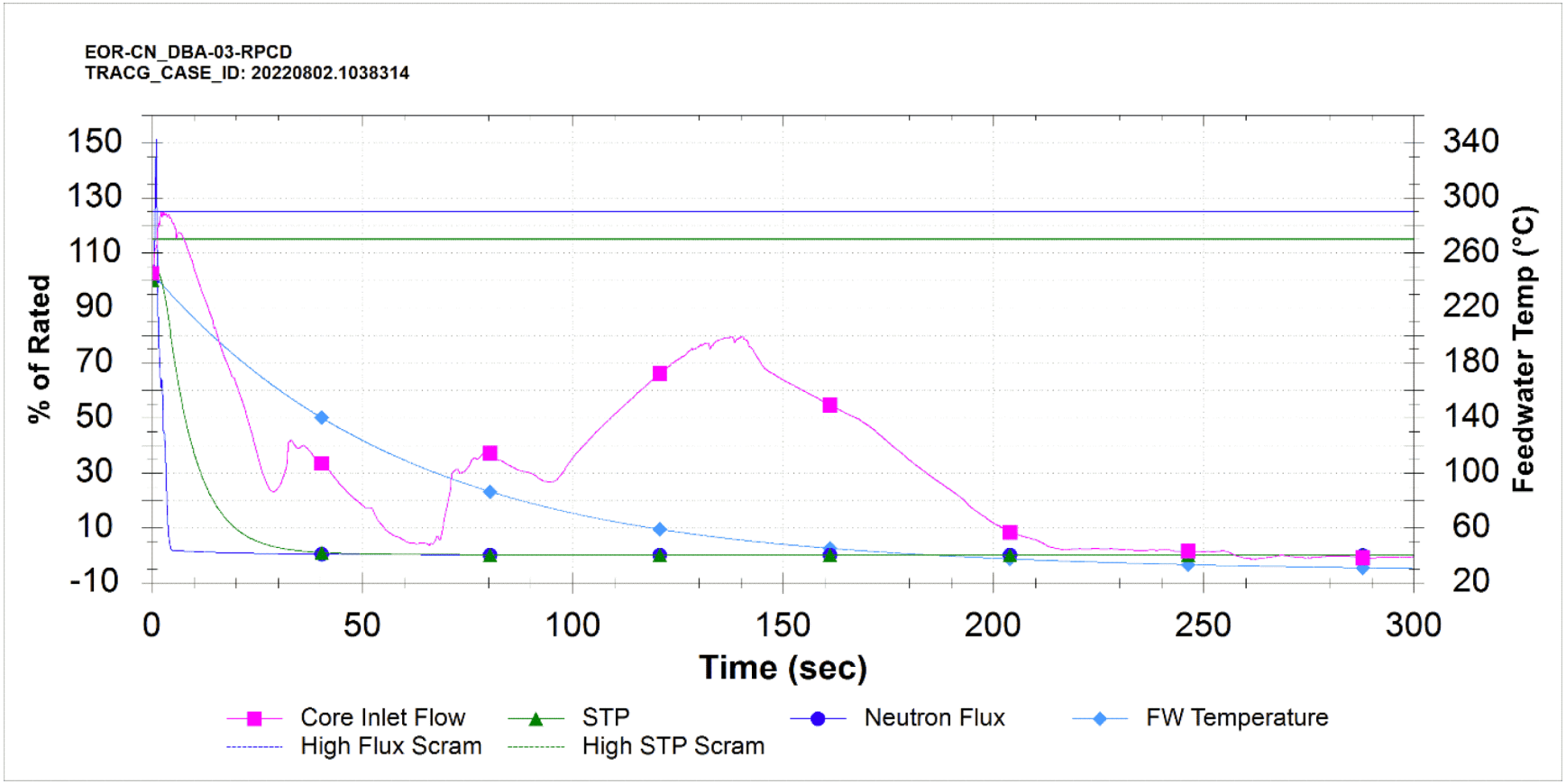

Figure 15.5-64 RPV Pressure Control Downscale (DBA)

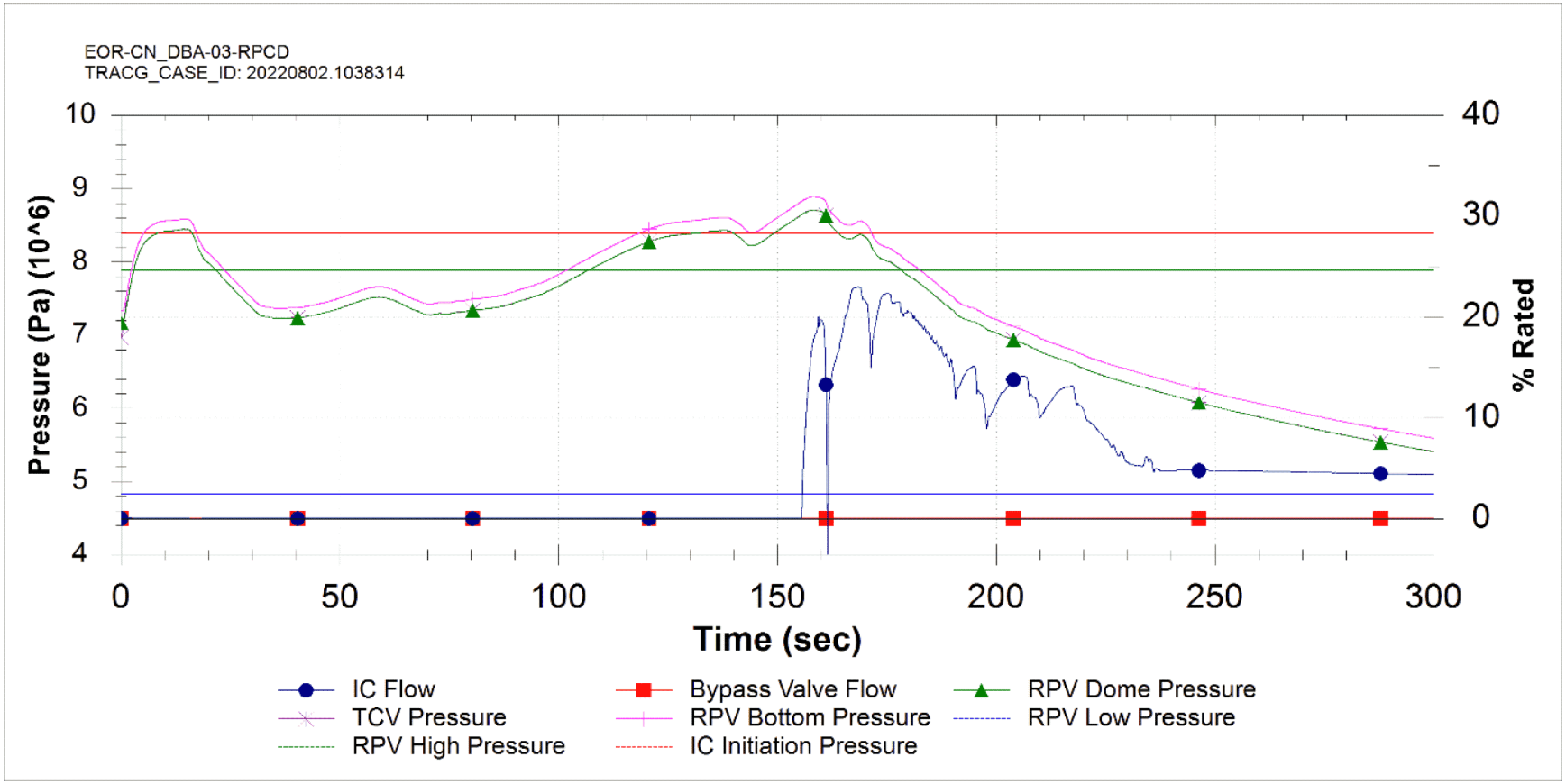

Figure 15.5-65 RPV Pressure Control Downscale (DBA)

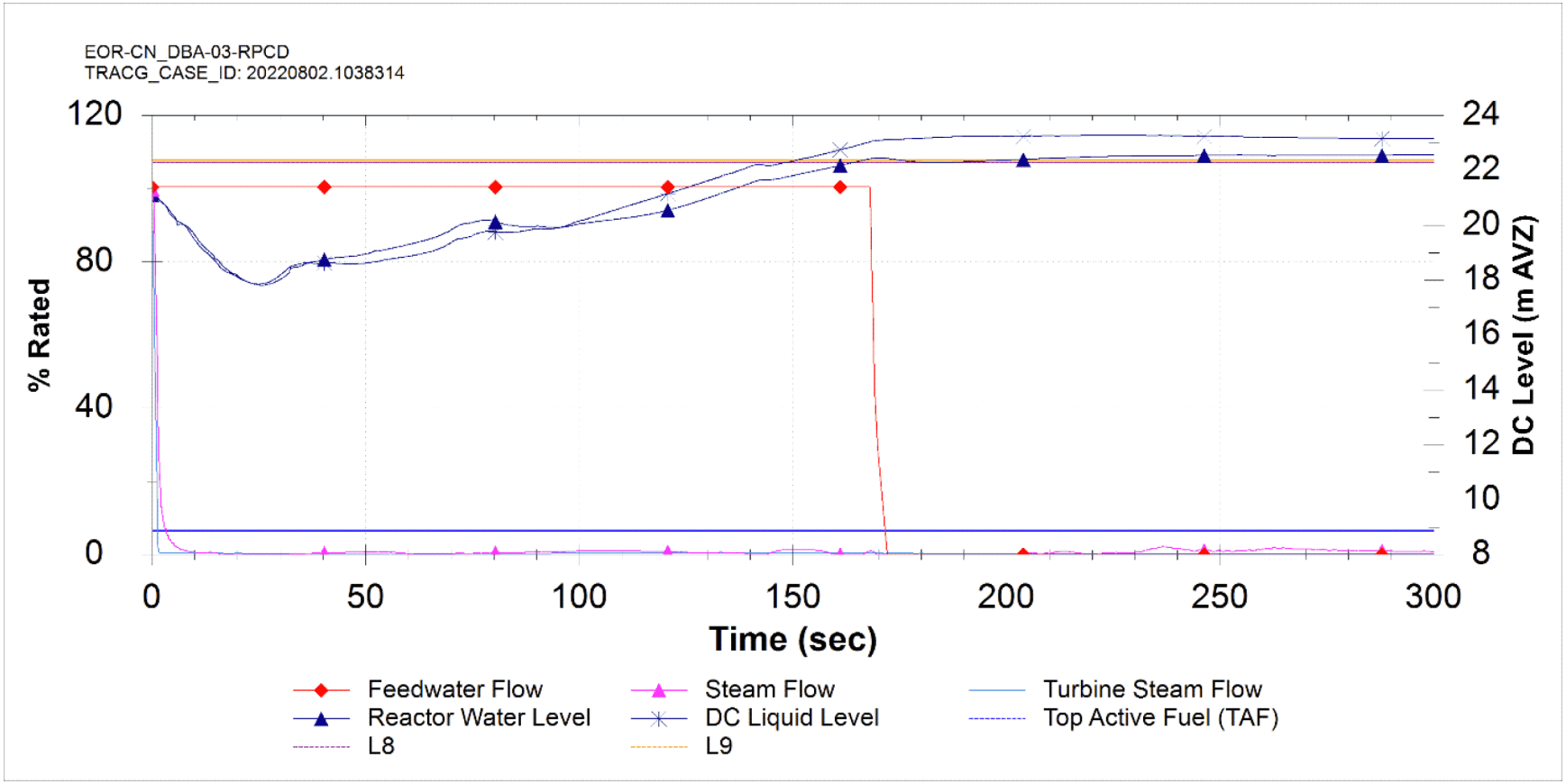

Figure 15.5-66 RPV Pressure Control Downscale (DBA)

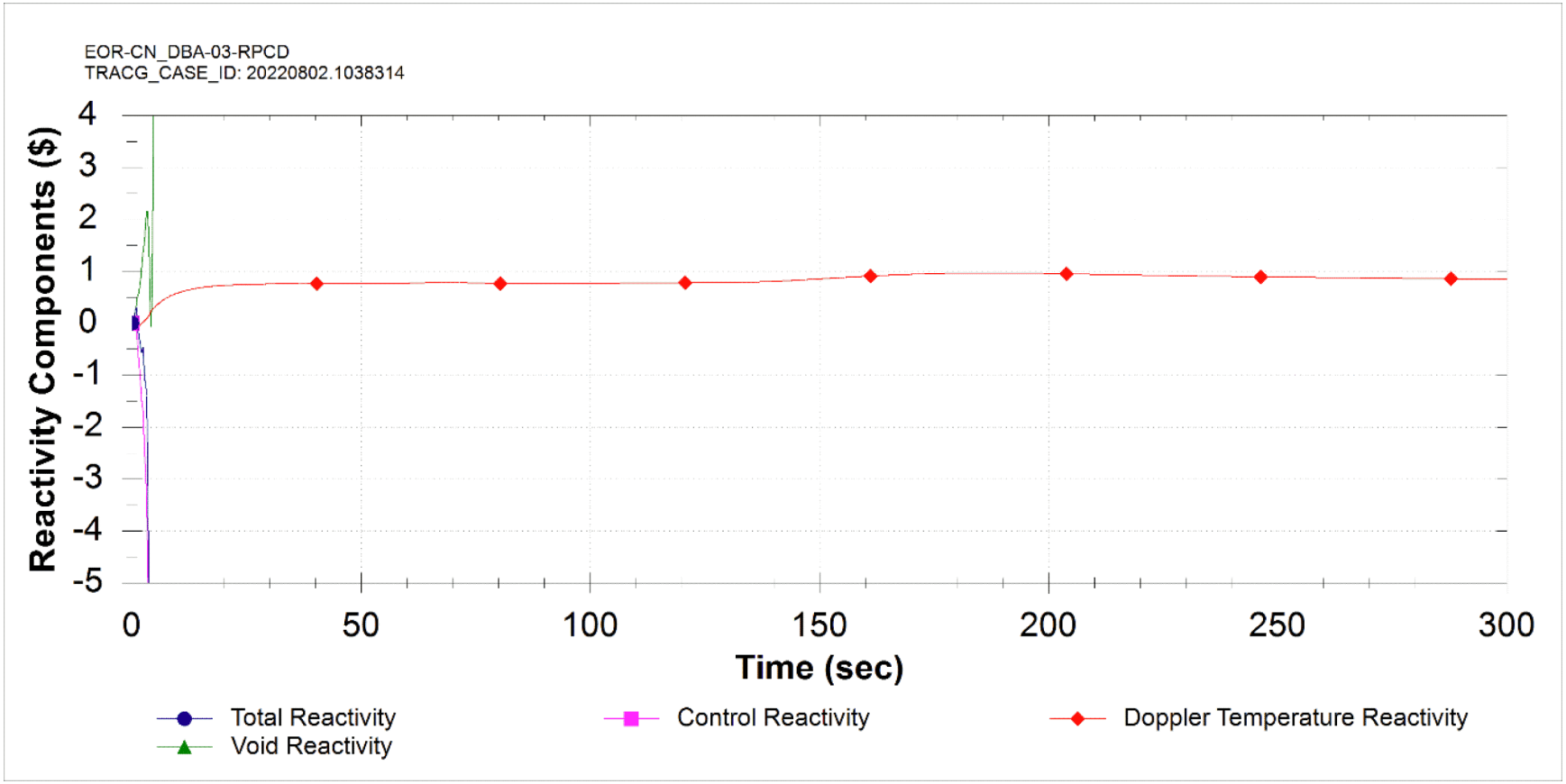

Figure 15.5-67 RPV Pressure Control Downscale (DBA)

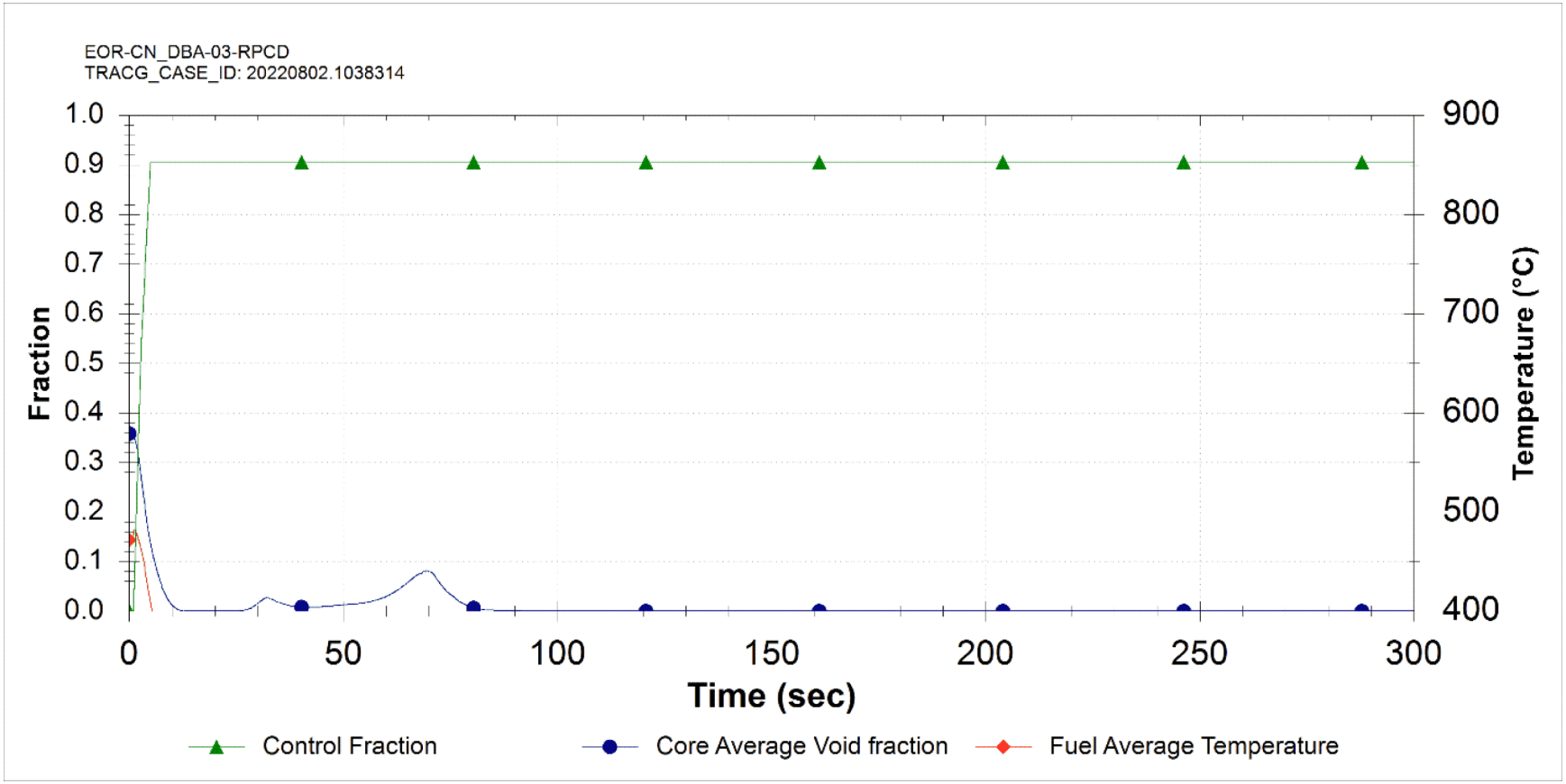

Figure 15.5-68 RPV Pressure Control Downscale (DBA)

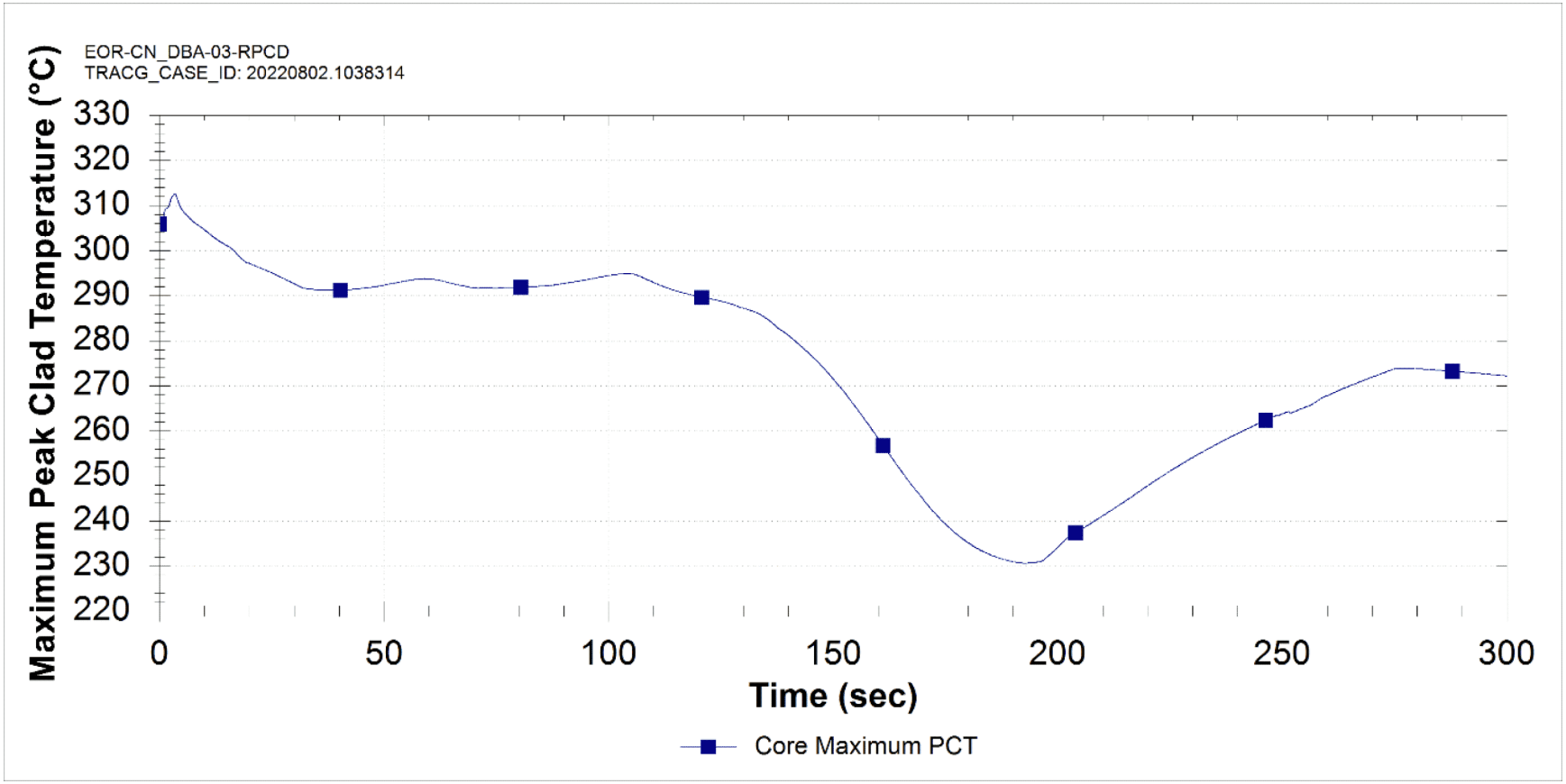

Figure 15.5-69 RPV Pressure Control Downscale (DBA)

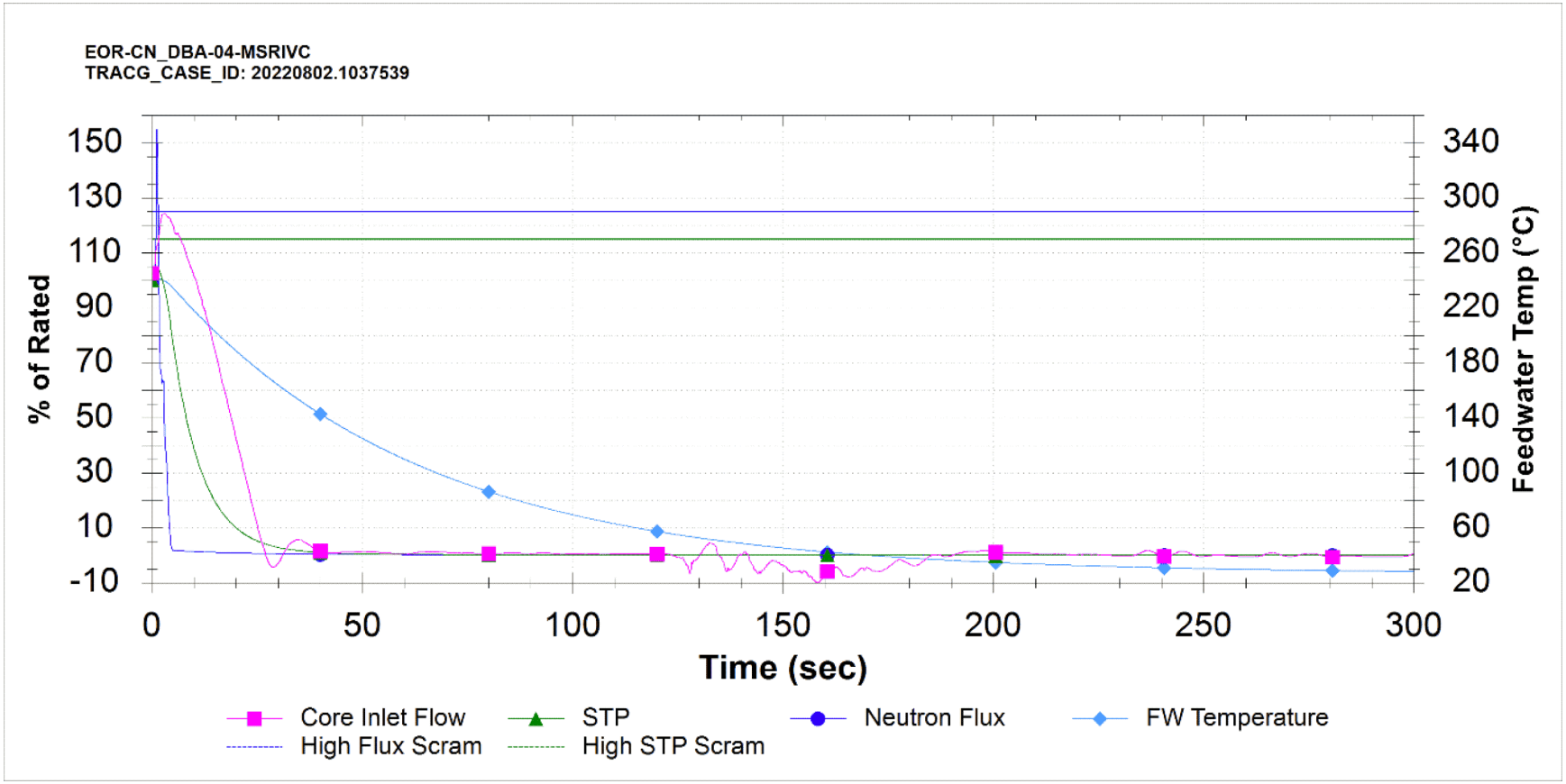

Figure 15.5-70 Closure of All MSRIVs and FW Isolation Valves (DBA)

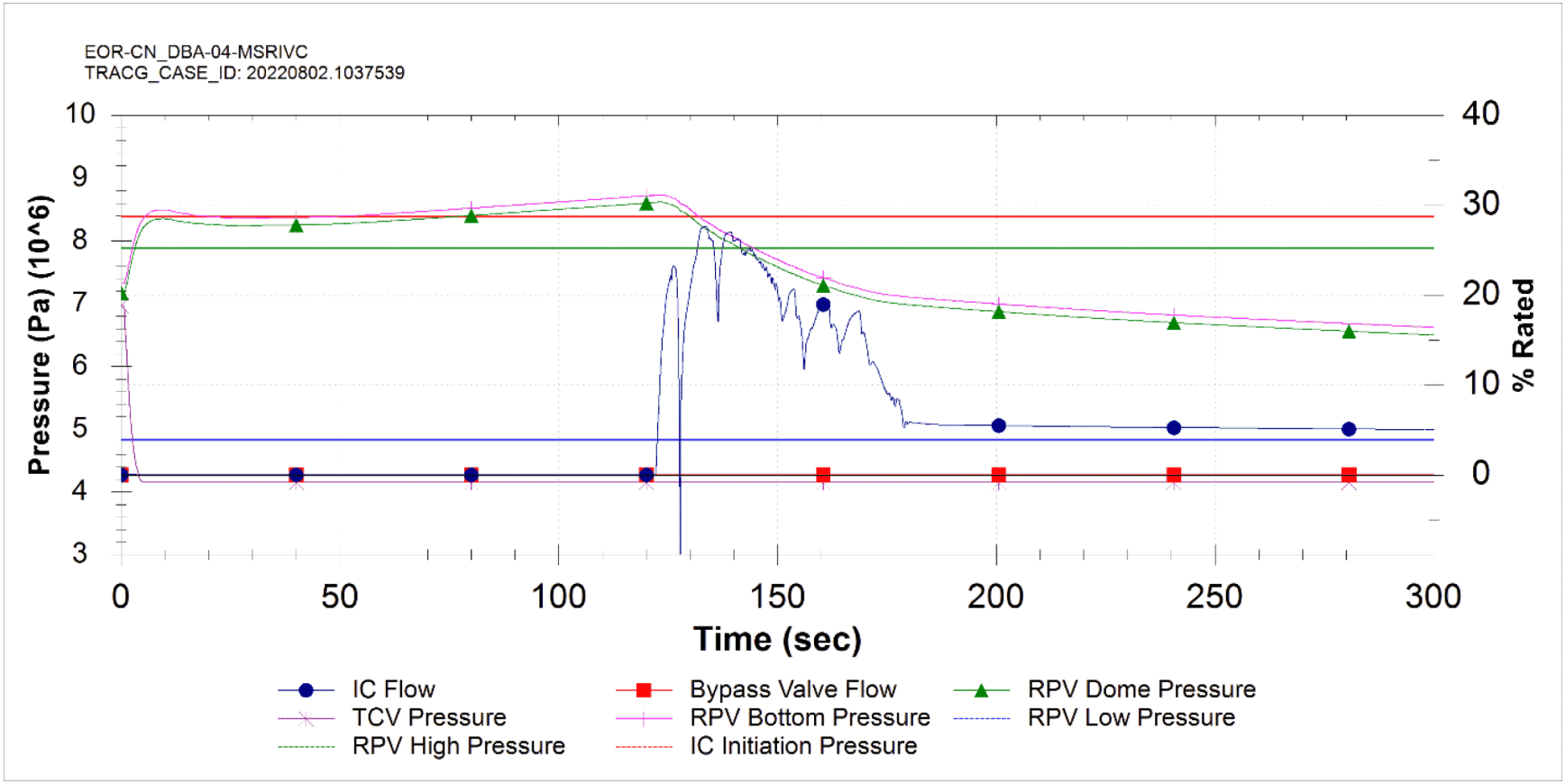

Figure 15.5-71 Closure of All MSRIVs and FW Isolation Valves (DBA)

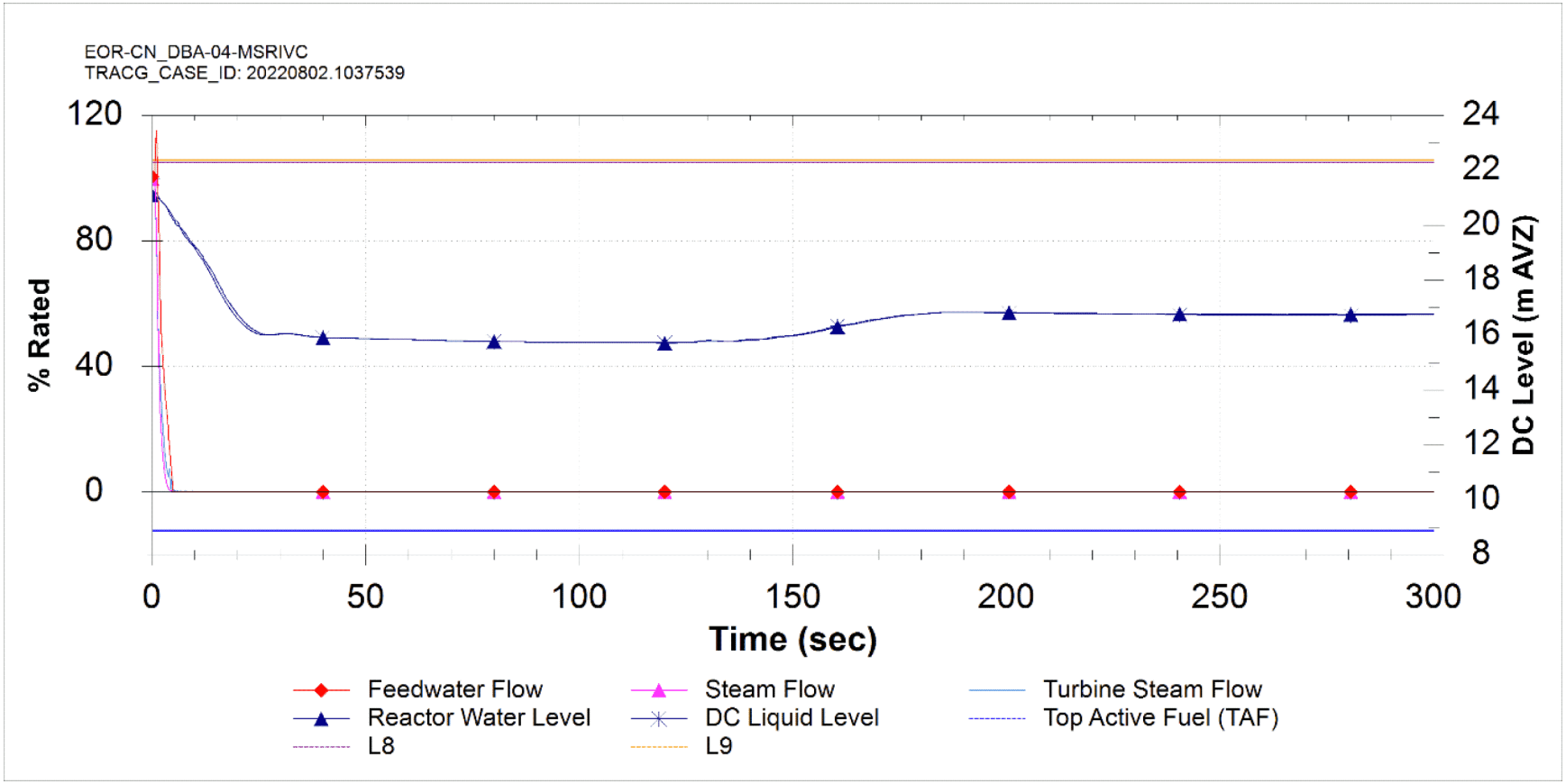

Figure 15.5-72 Closure of All MSRIVs and FW Isolation Valves (DBA)

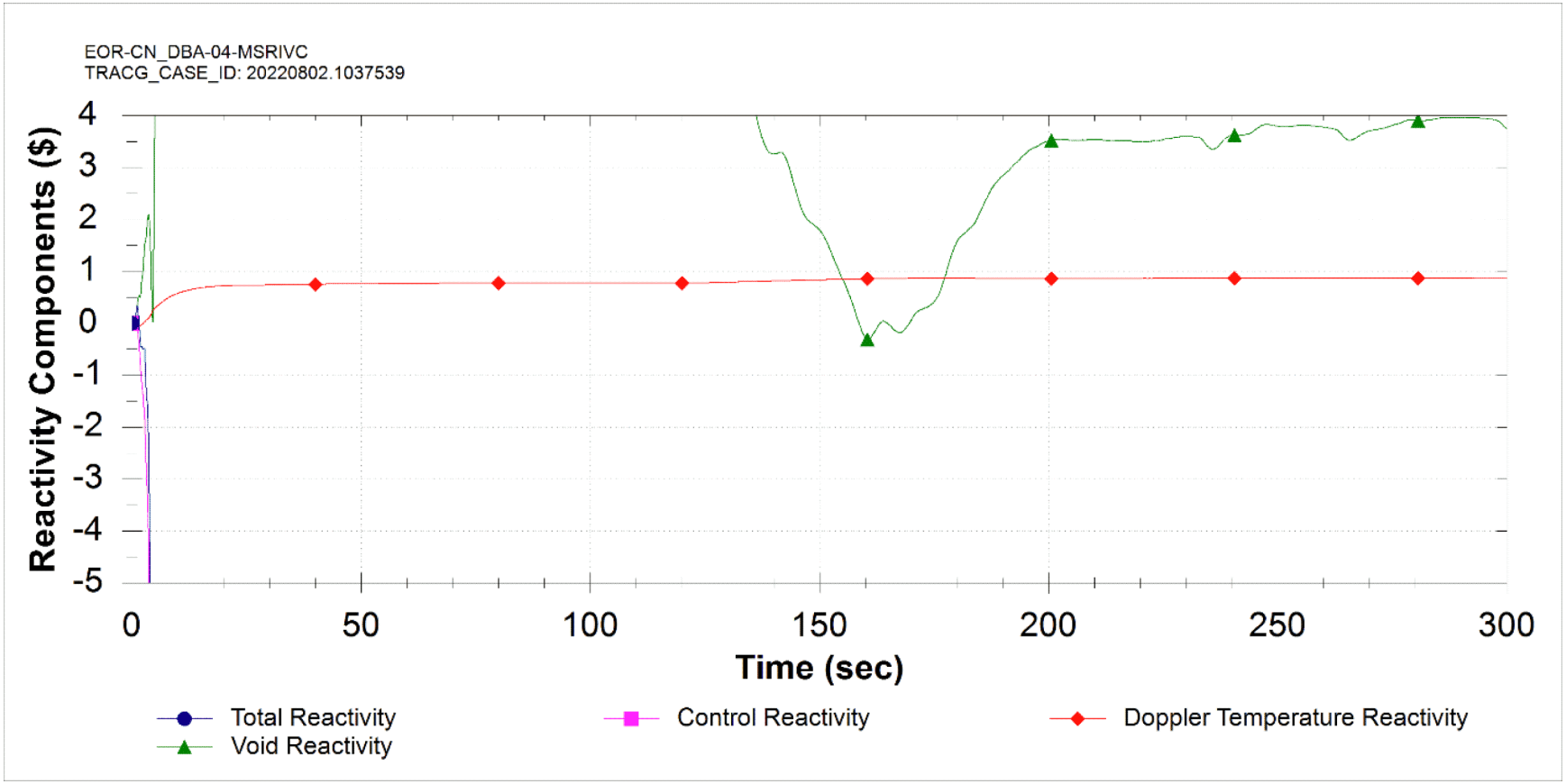

Figure 15.5-73 Closure of All MSRIVs and FW Isolation Valves (DBA)

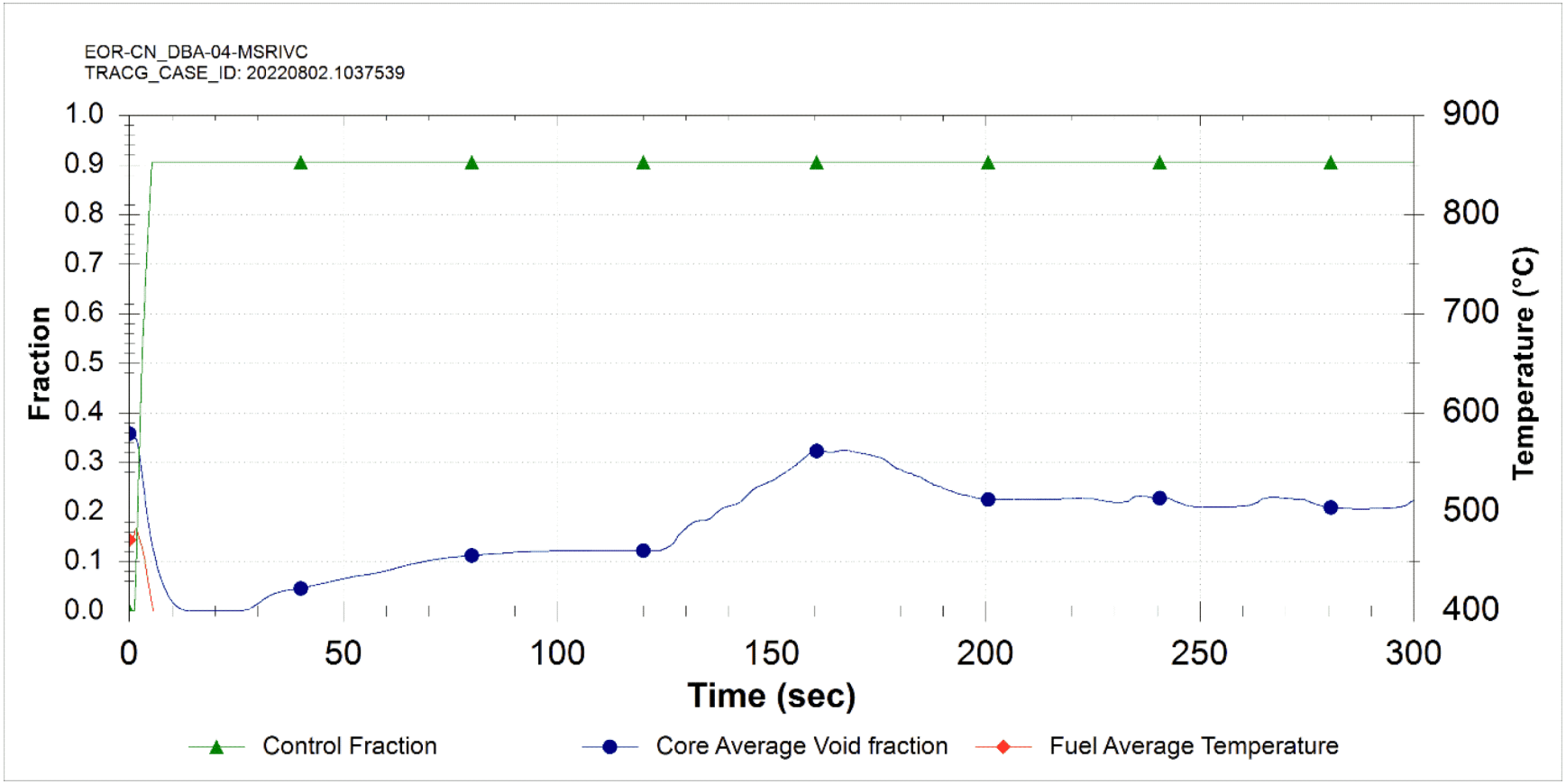

Figure 15.5-74 Closure of All MSRIVs and FW Isolation Valves (DBA)

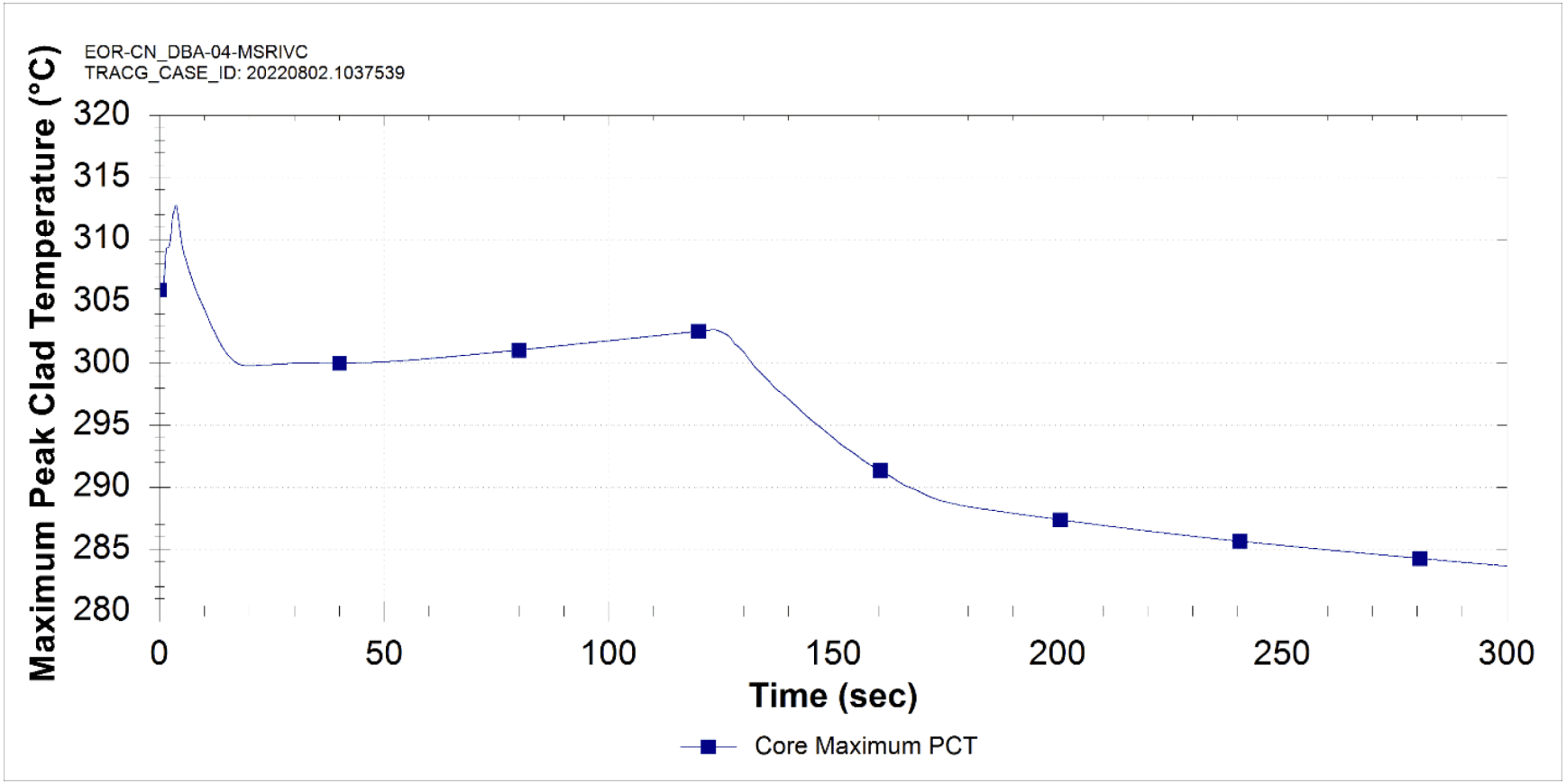

Figure 15.5-75 Closure of All MSRIVs and FW Isolation Valves (DBA)

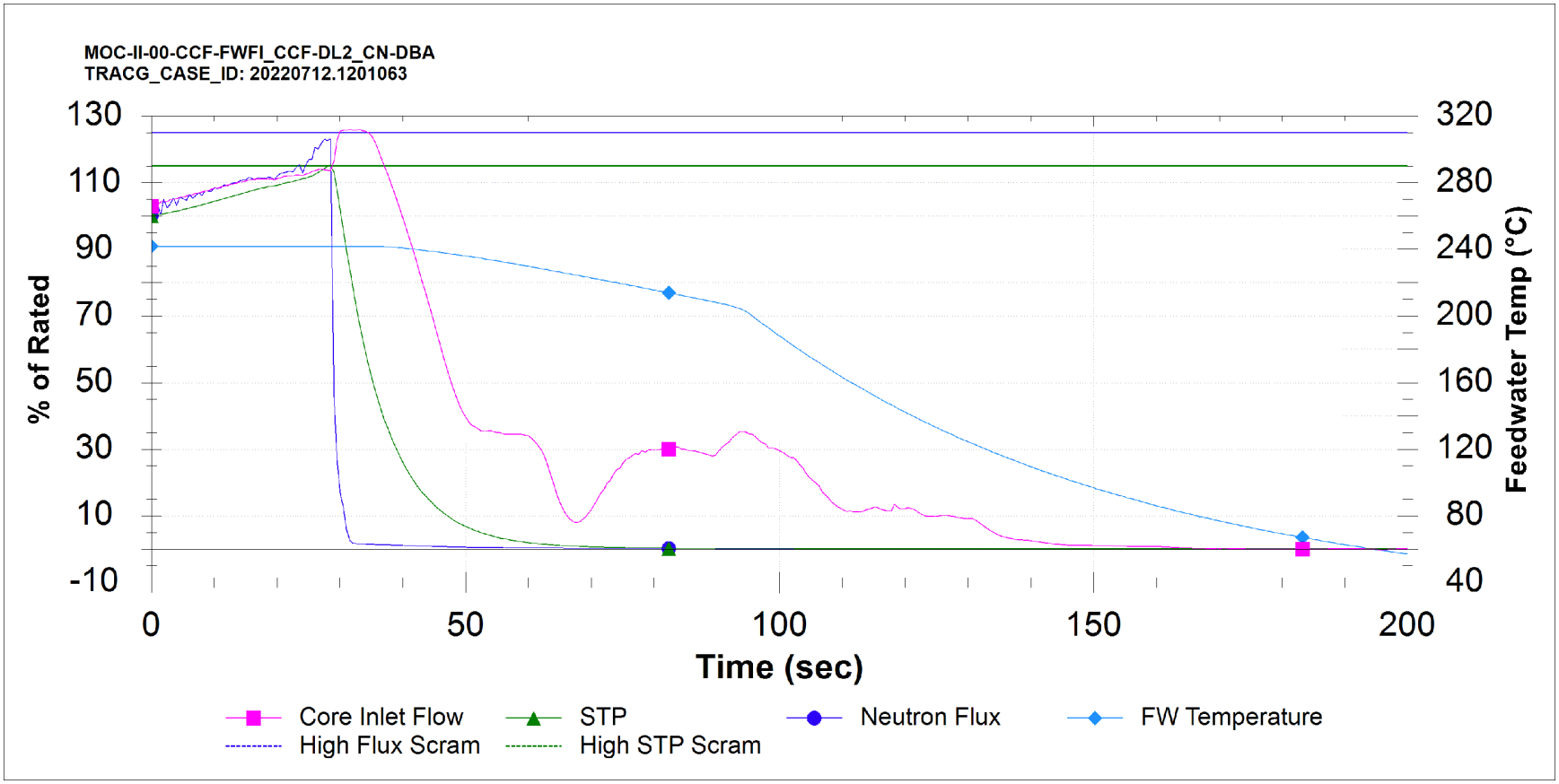

Figure 15.5-76 Feedwater Flow Increase (DBA)

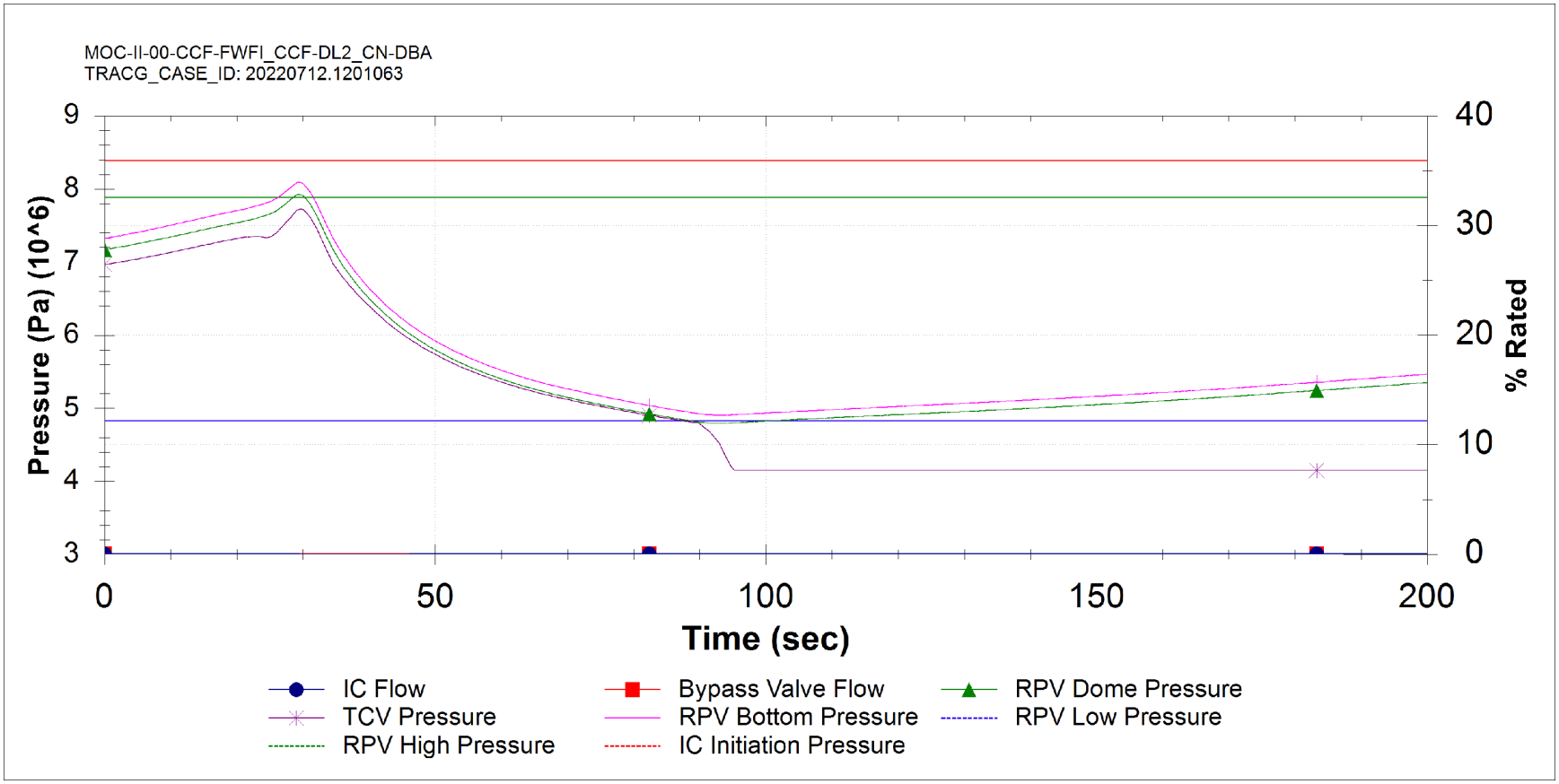

Figure 15.5-77 Feedwater Flow Increase (DBA)

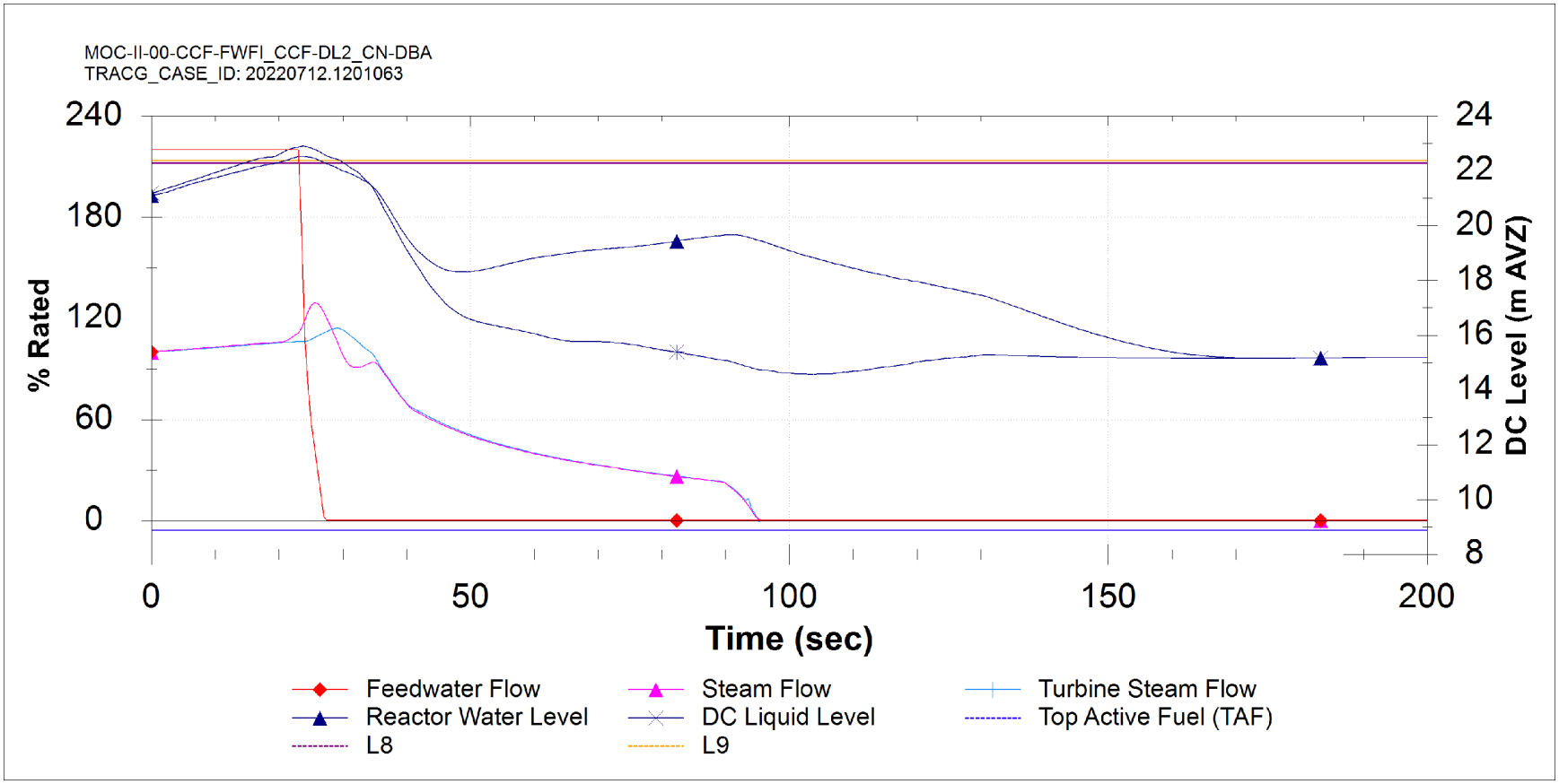

Figure 15.5-78 Feedwater Flow Increase (DBA)

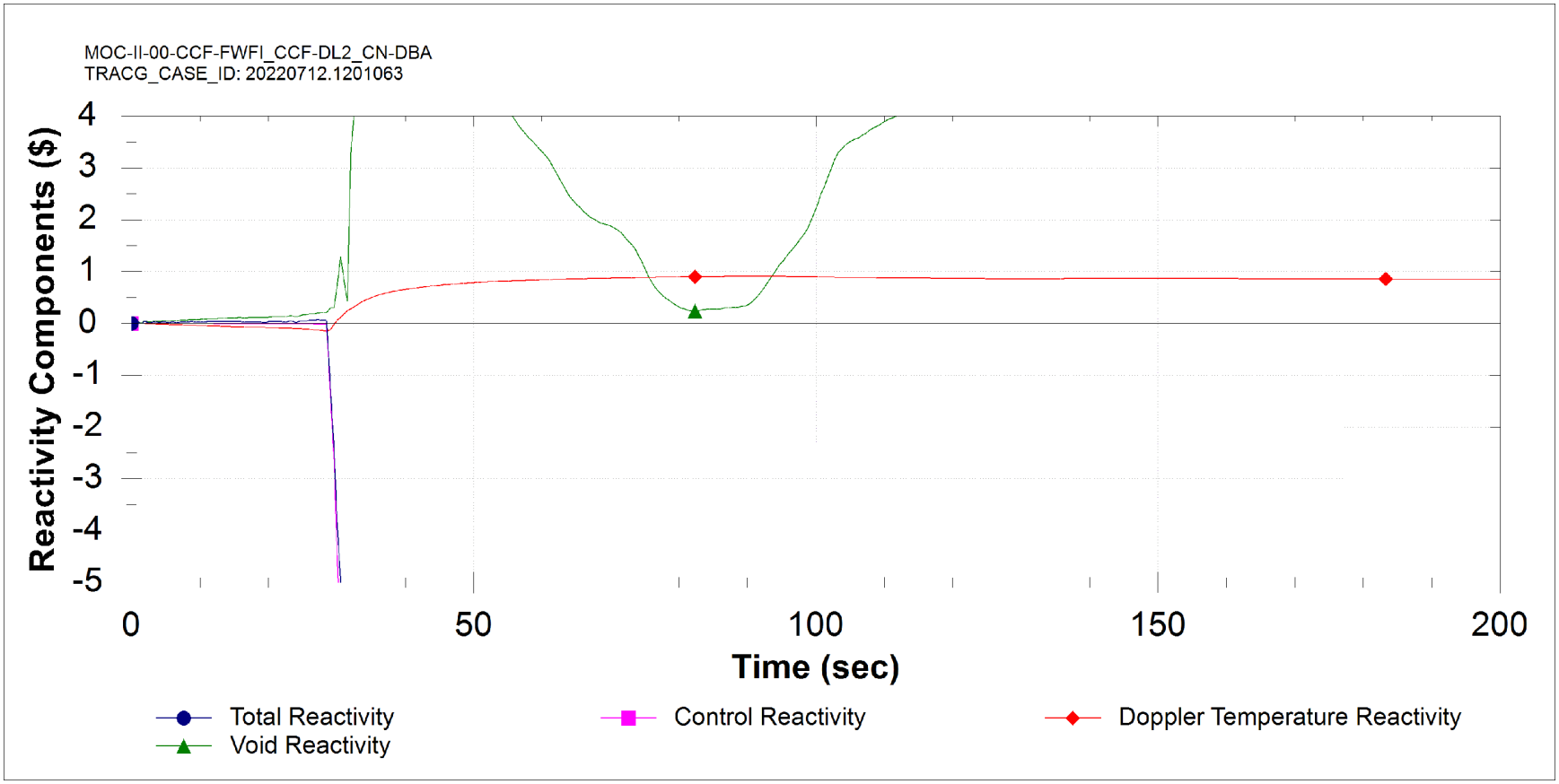

Figure 15.5-79 Feedwater Flow Increase (DBA)

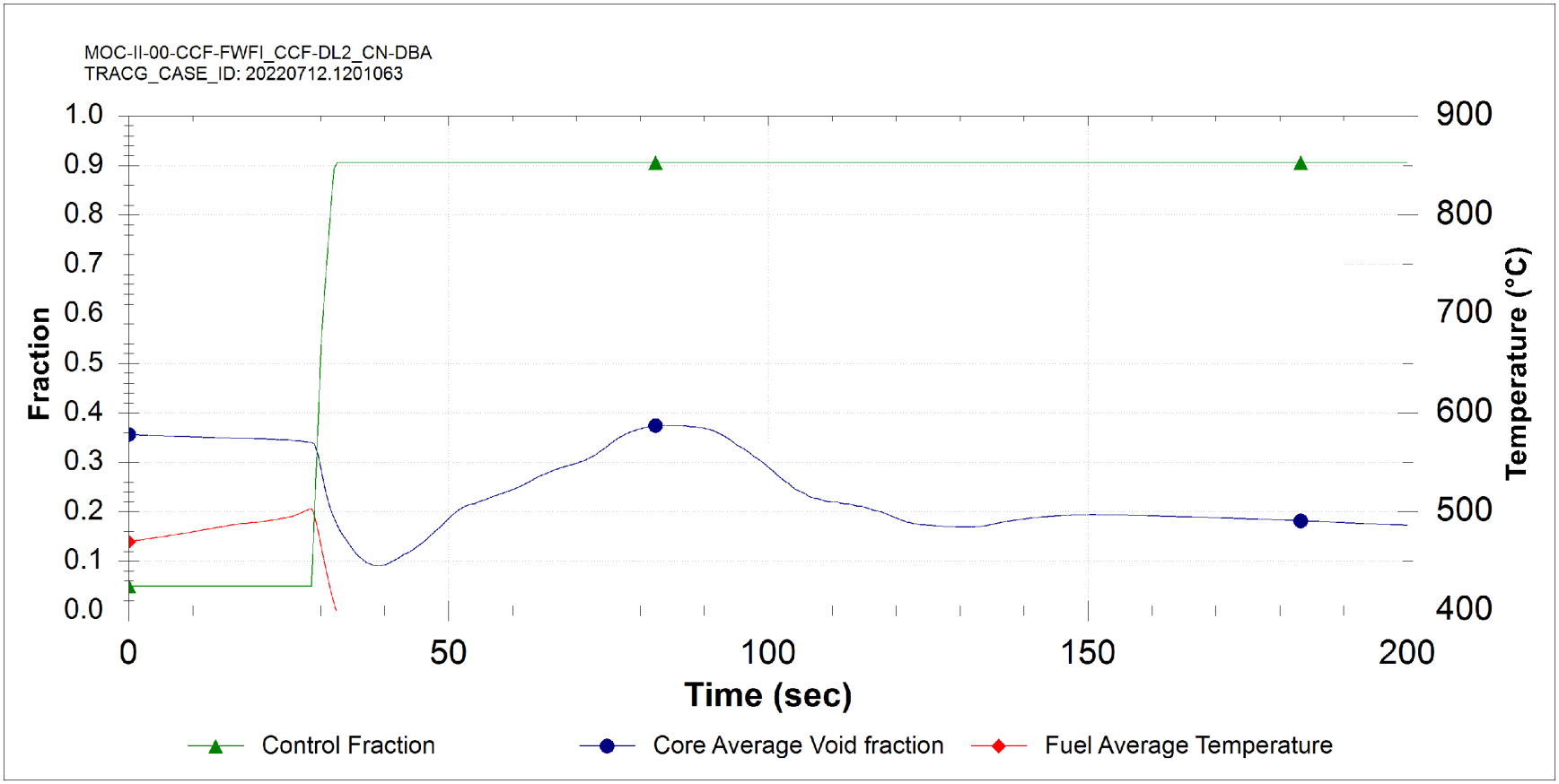

Figure 15.5-80 Feedwater Flow Increase (DBA)

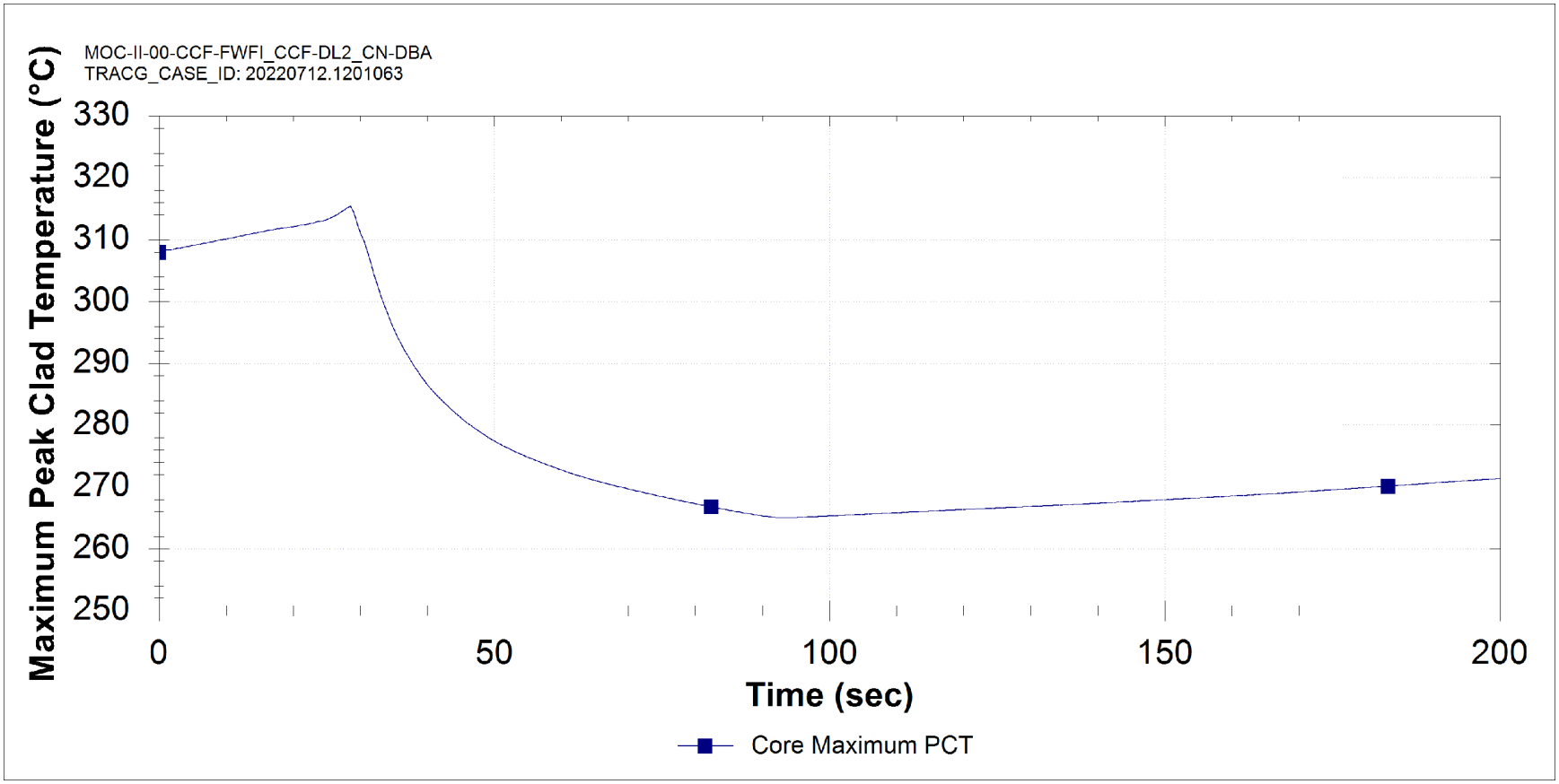

Figure 15.5-81 Feedwater Flow Increase (DBA)

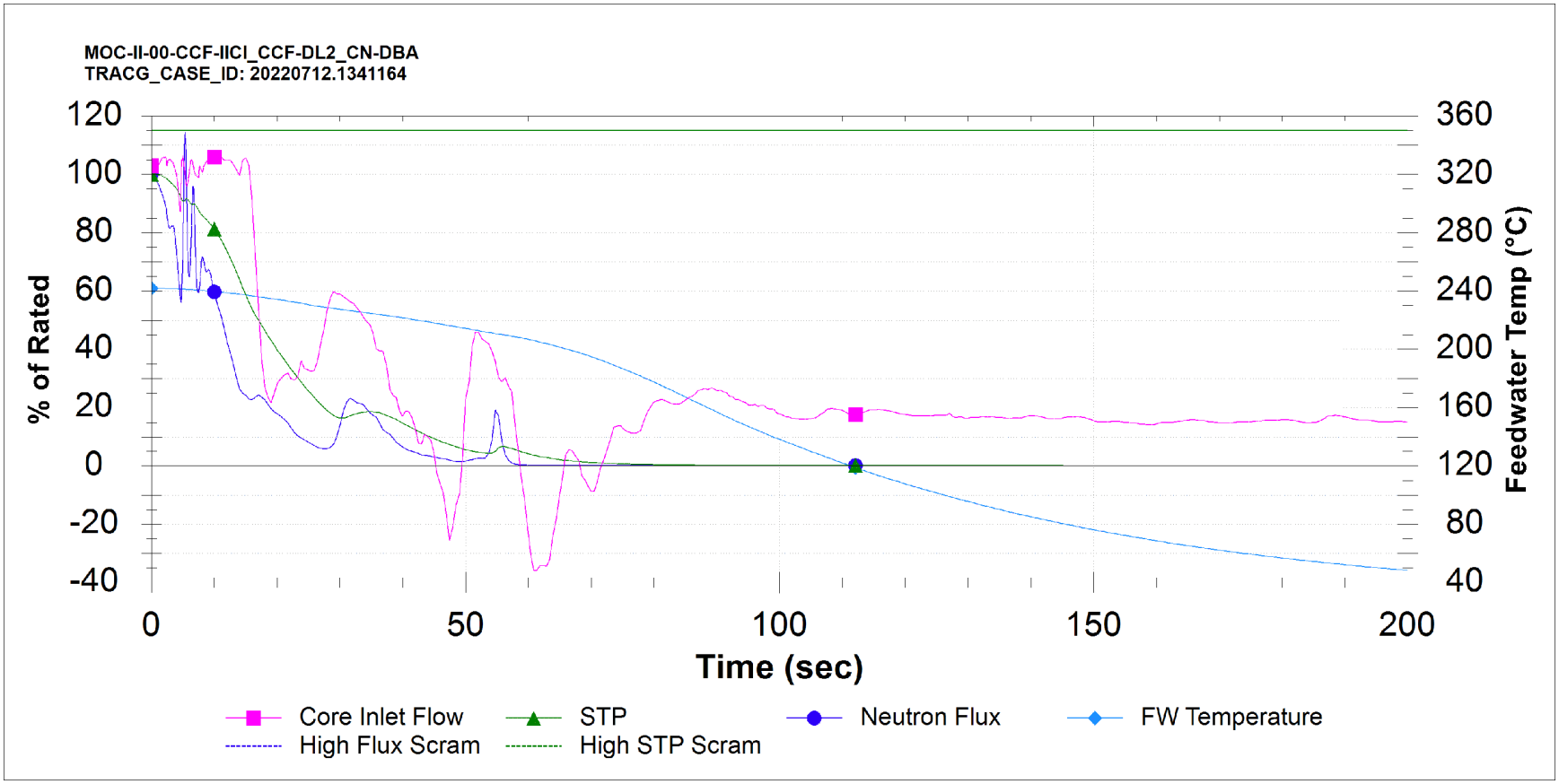

Figure 15.5-82 Inadvertent Isolation Condenser Initiation – All Trains (DBA)

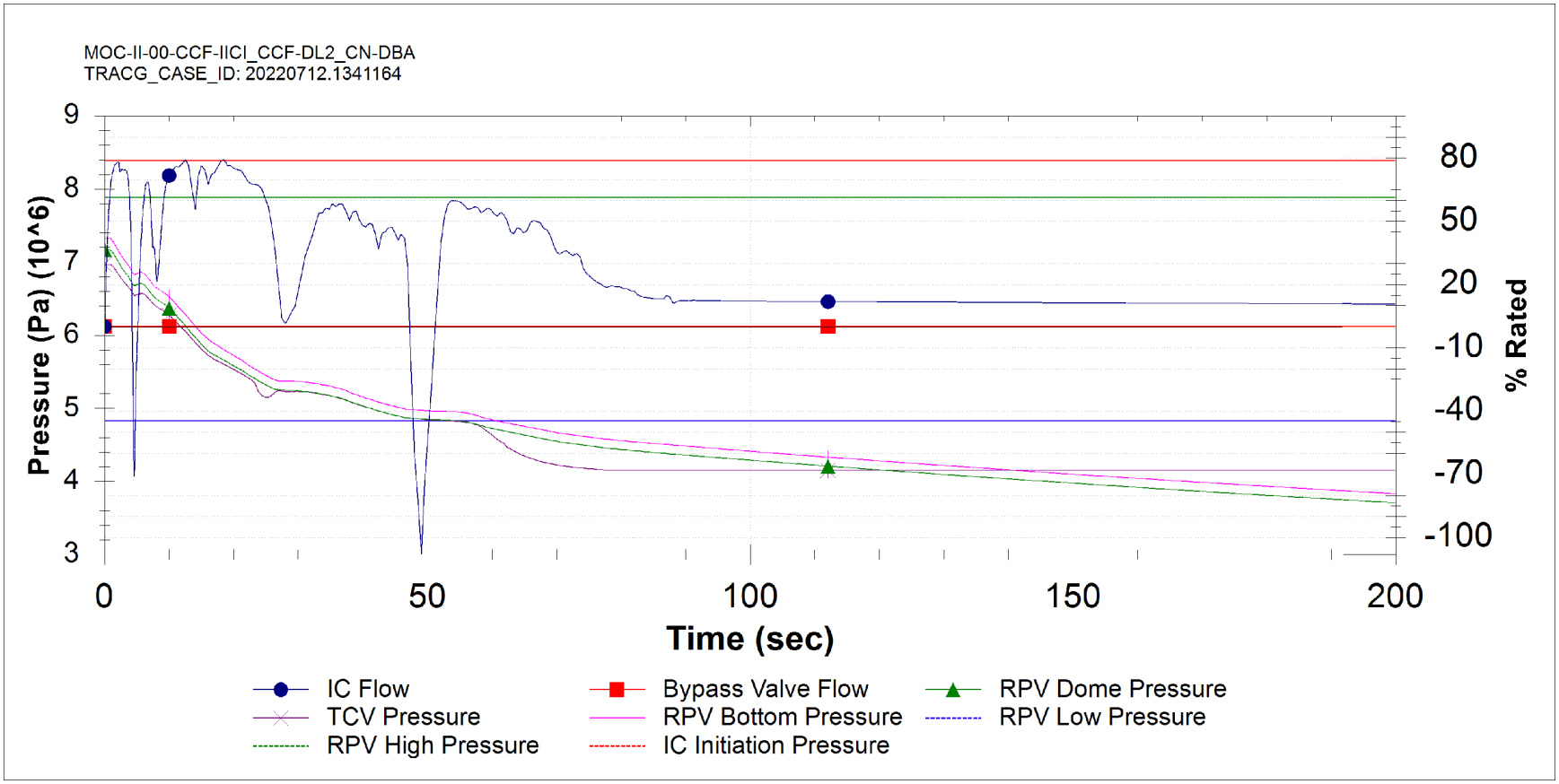

Figure 15.5-83 Inadvertent Isolation Condenser Initiation – All Trains (DBA)

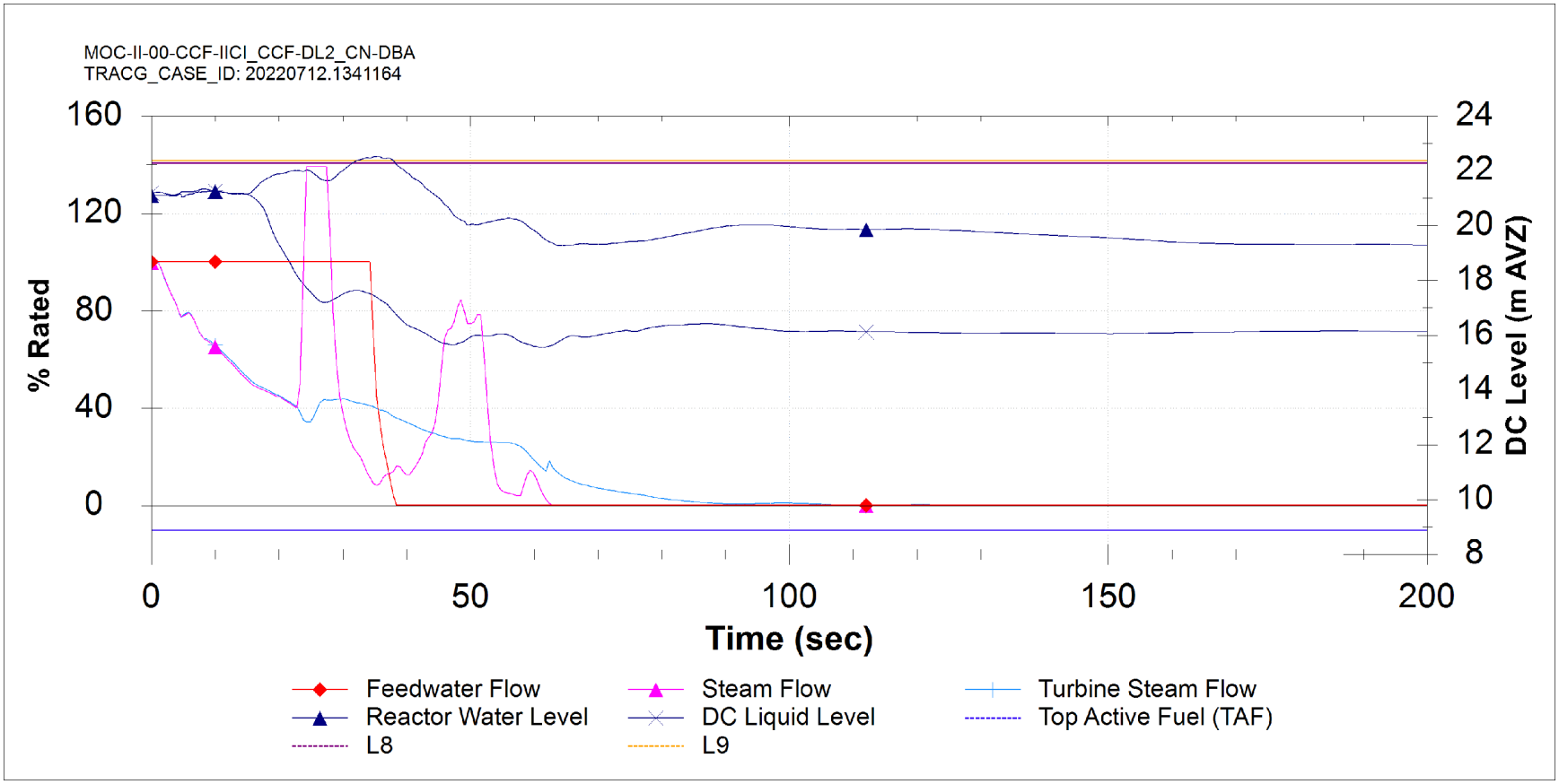

Figure 15.5-84 Inadvertent Isolation Condenser Initiation – All Trains (DBA)

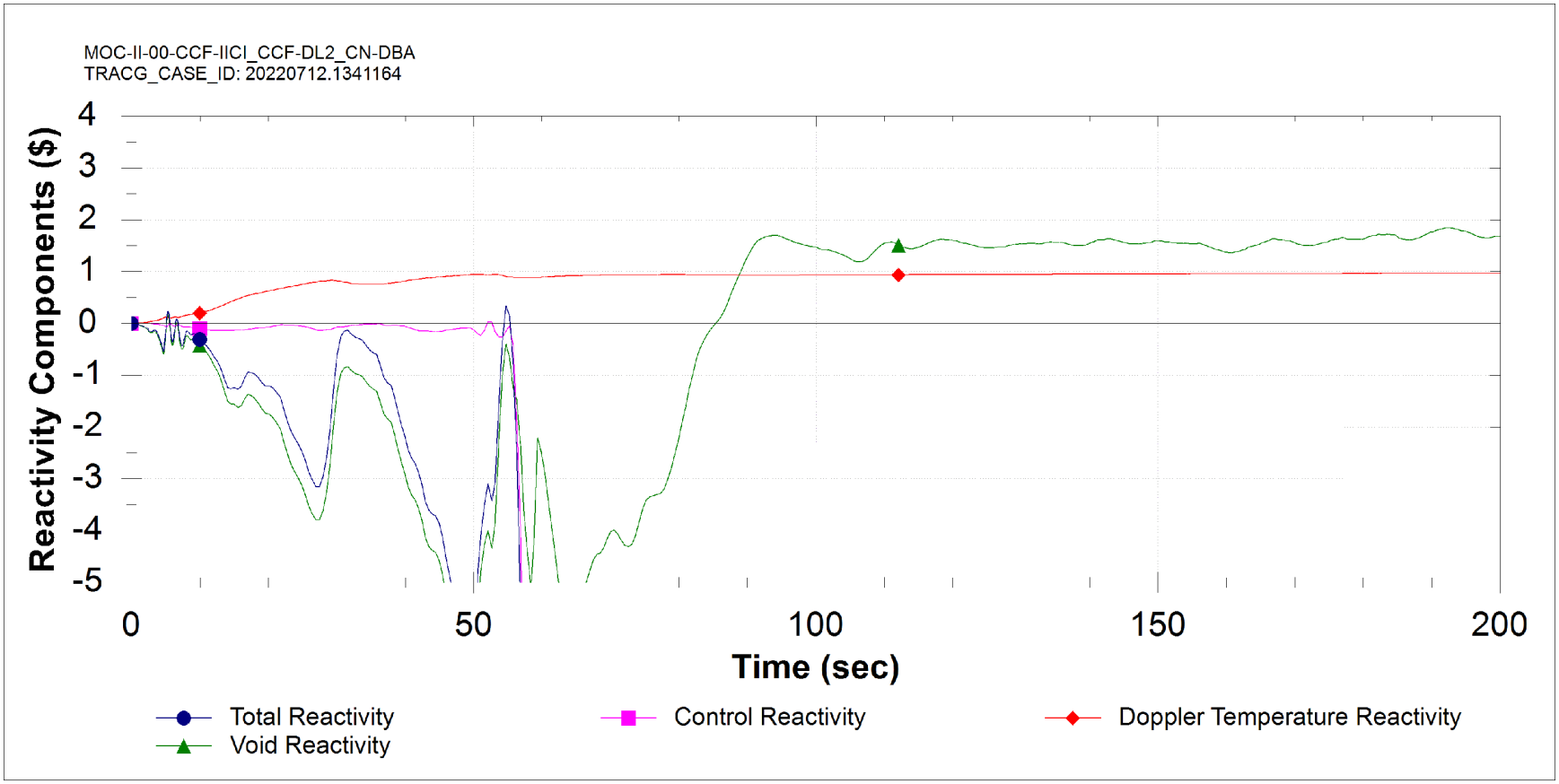

Figure 15.5-85 Inadvertent Isolation Condenser Initiation – All Trains (DBA)

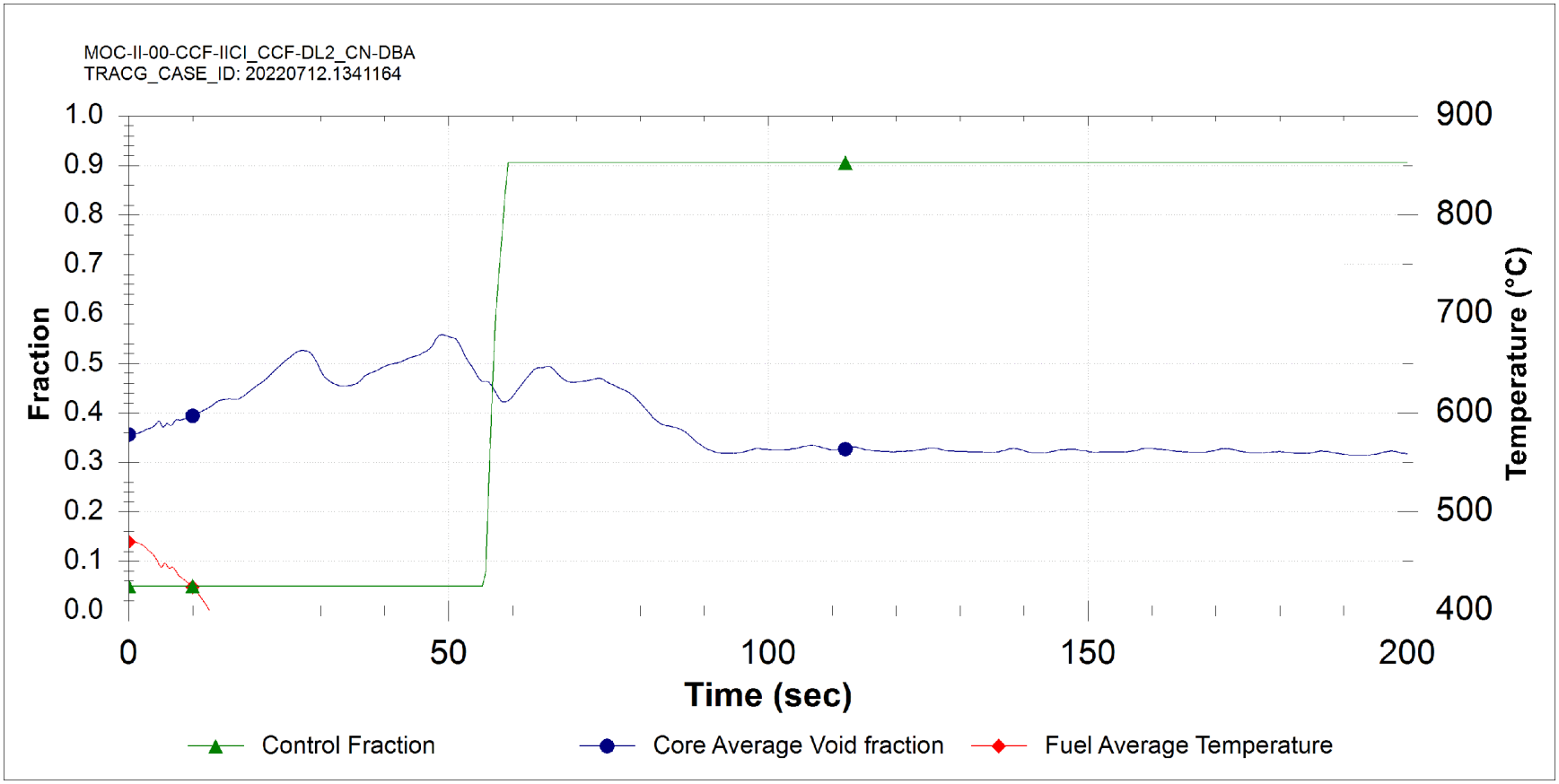

Figure 15.5-86 Inadvertent Isolation Condenser Initiation – All Trains (DBA)

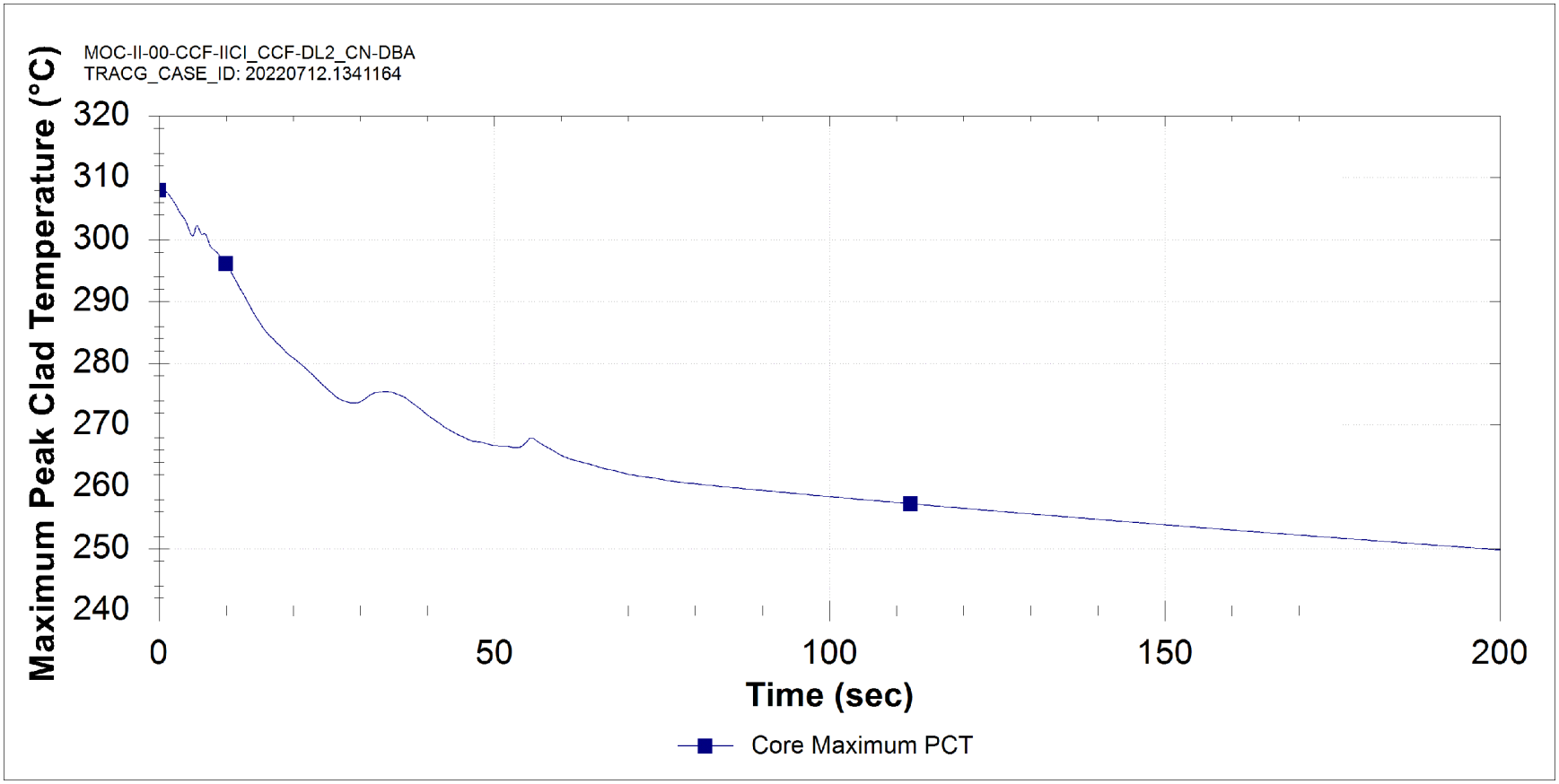

Figure 15.5-87 Inadvertent Isolation Condenser Initiation – All Trains (DBA)

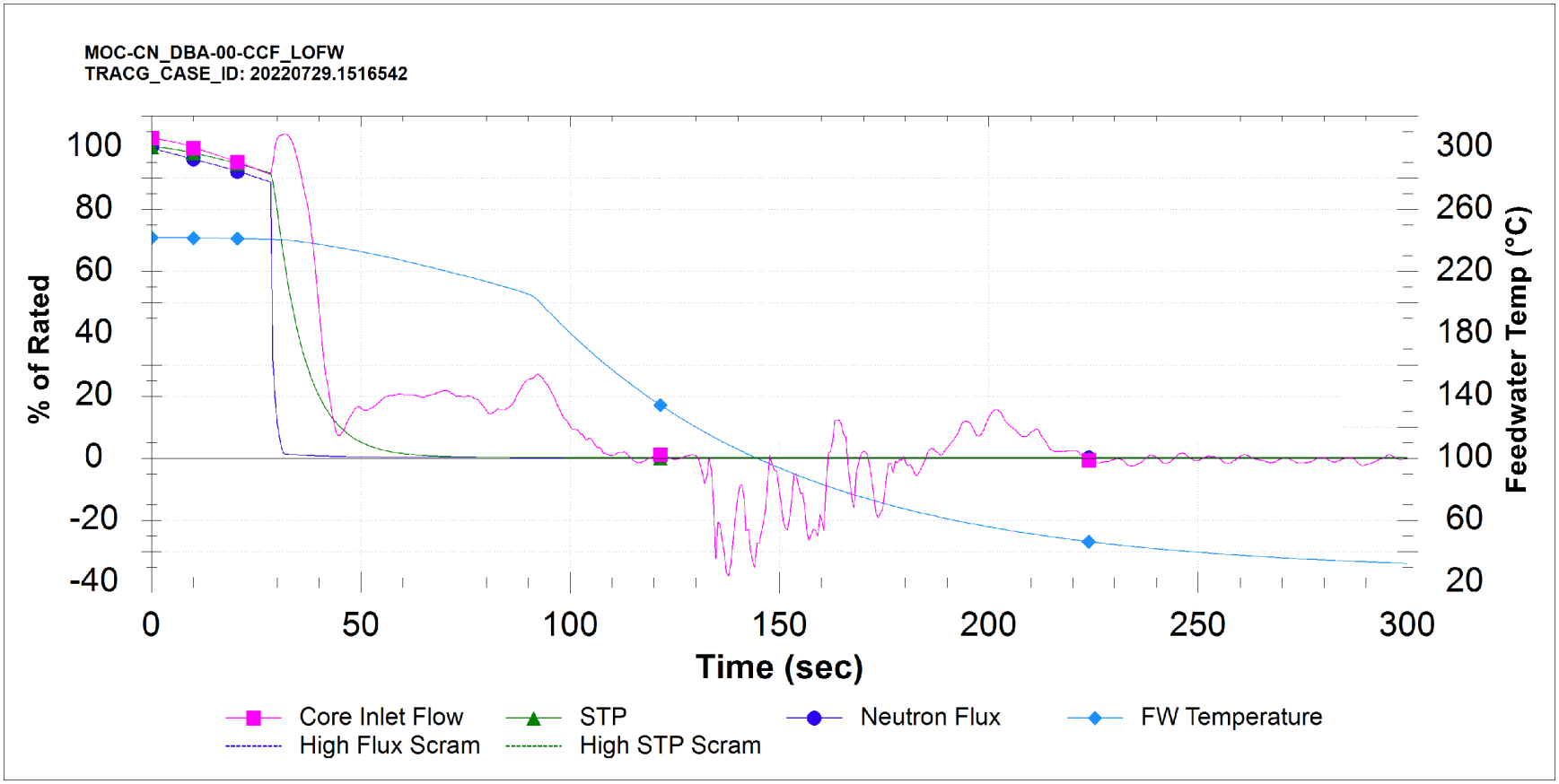

Figure 15.5-88 Loss of Feedwater Flow (DBA)

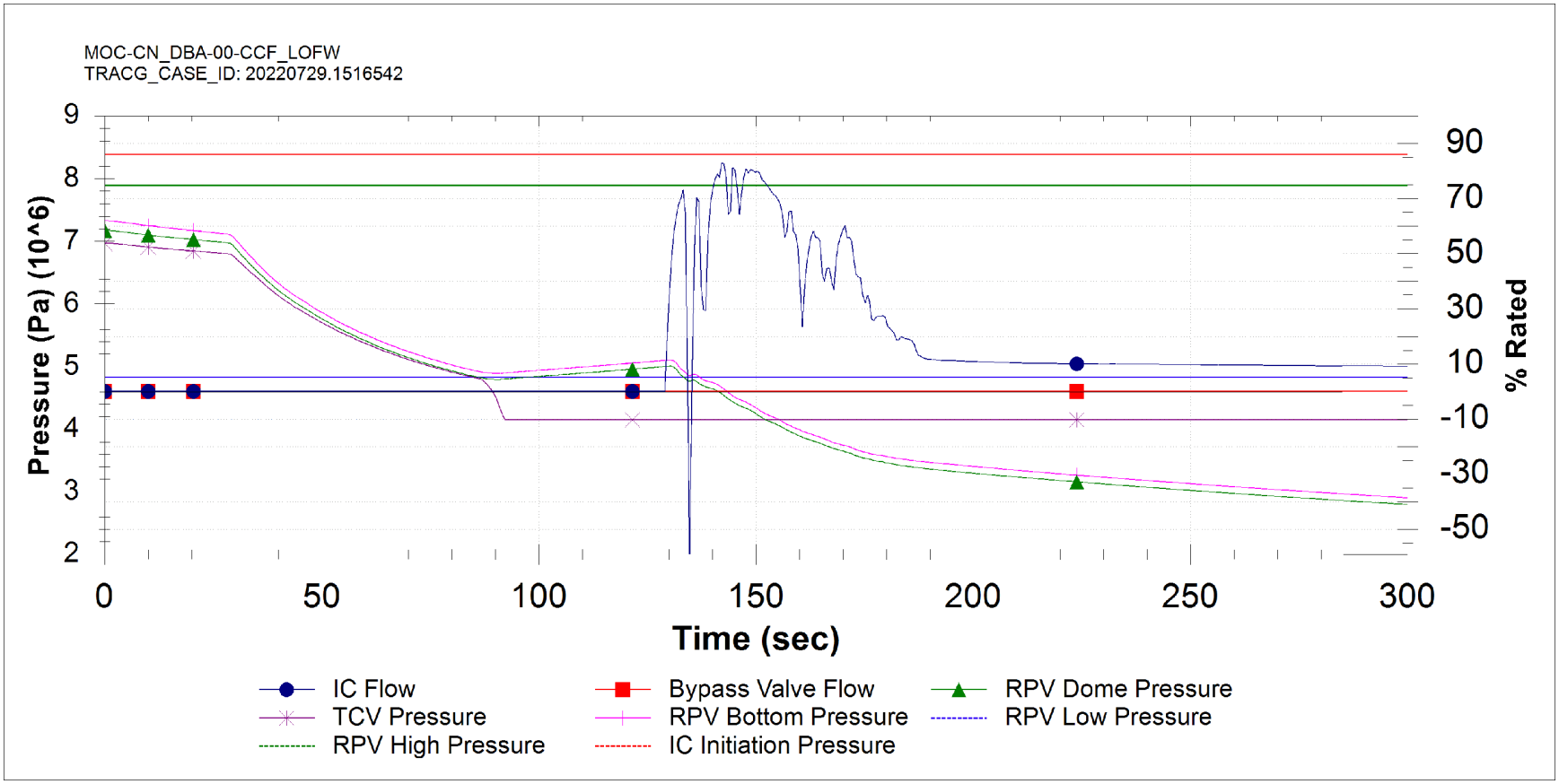

Figure 15.5-89 Loss of Feedwater Flow (DBA)

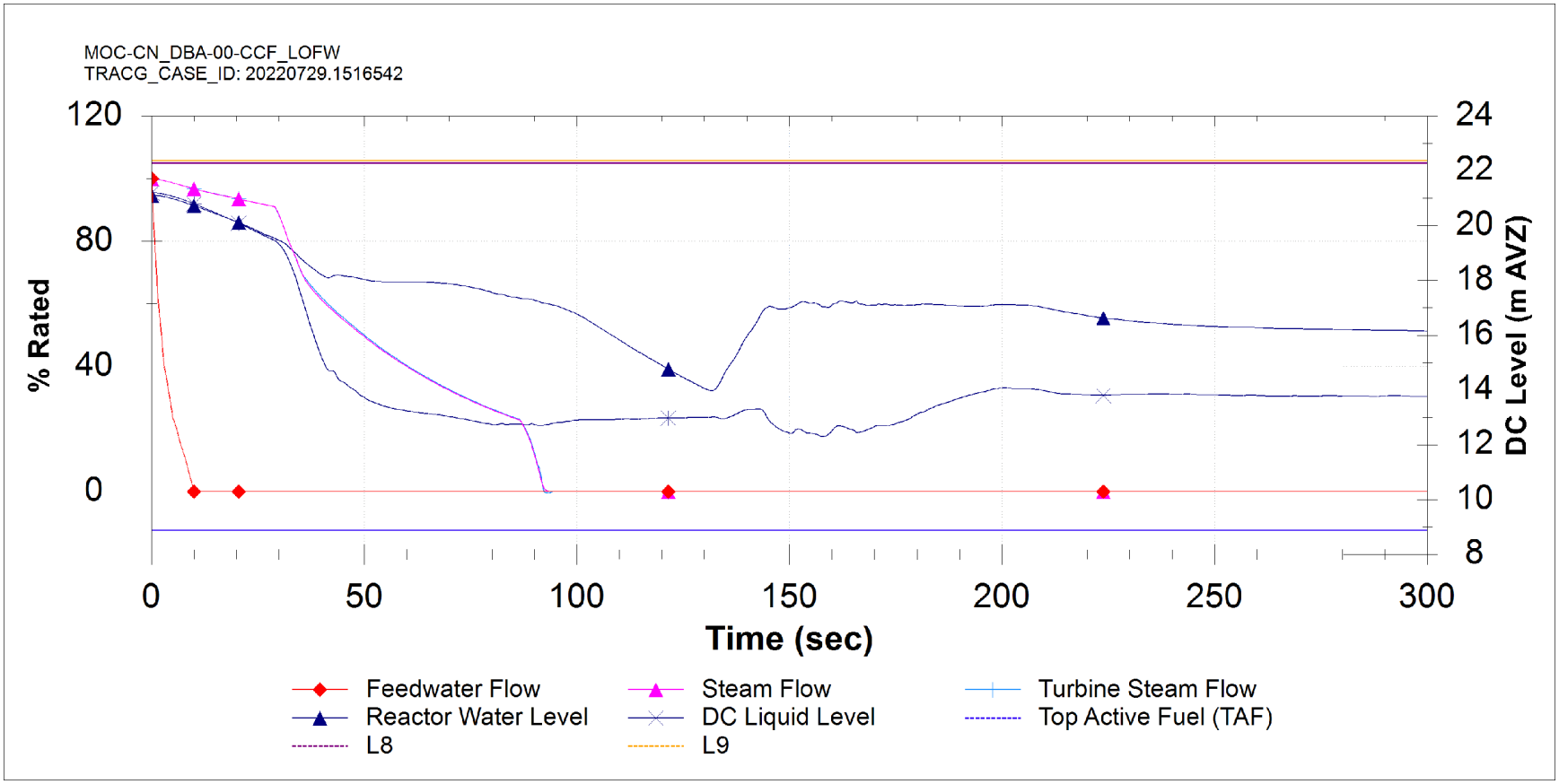

Figure 15.5-90 Loss of Feedwater Flow (DBA)

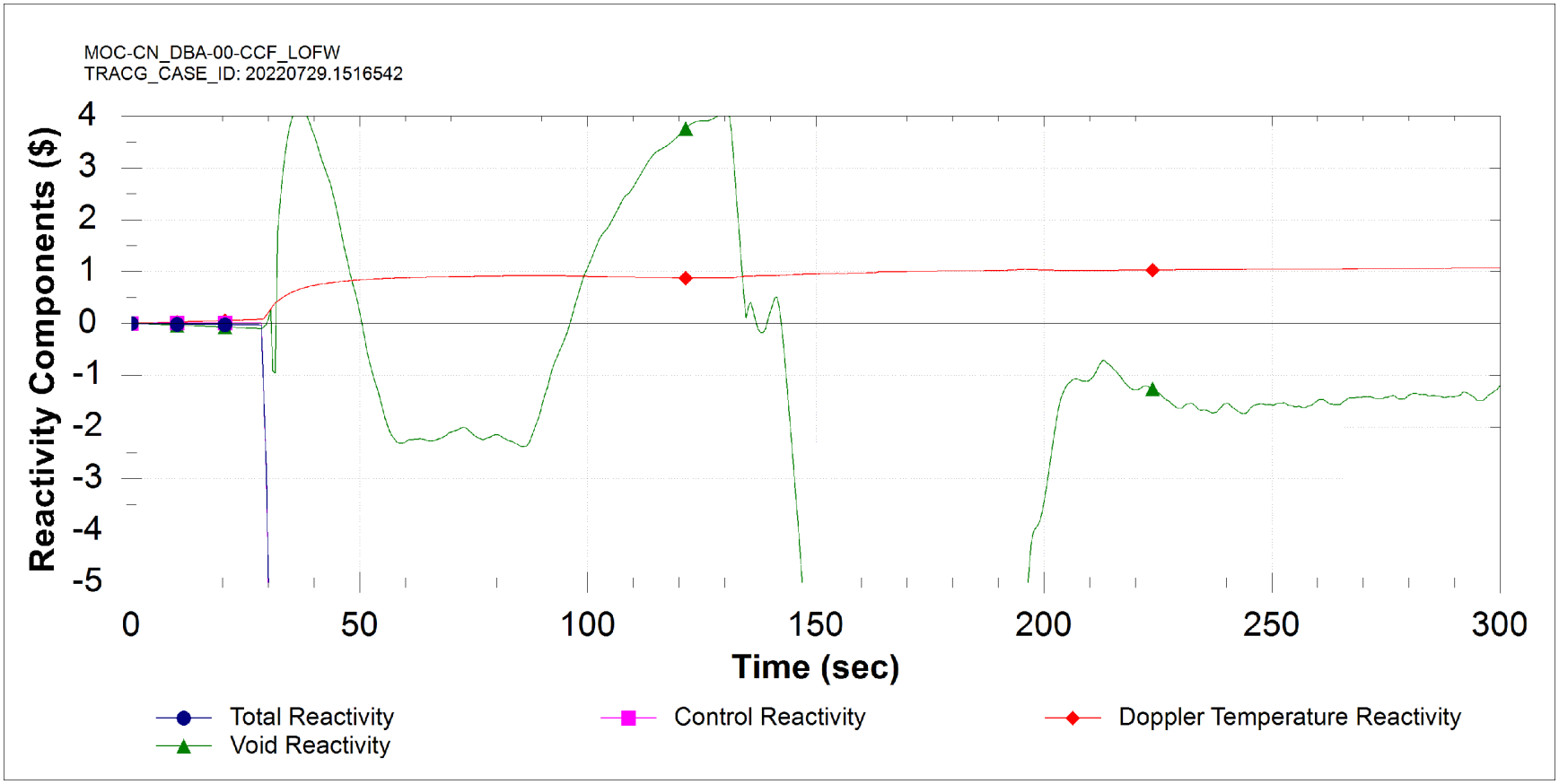

Figure 15.5-91 Loss of Feedwater Flow (DBA)

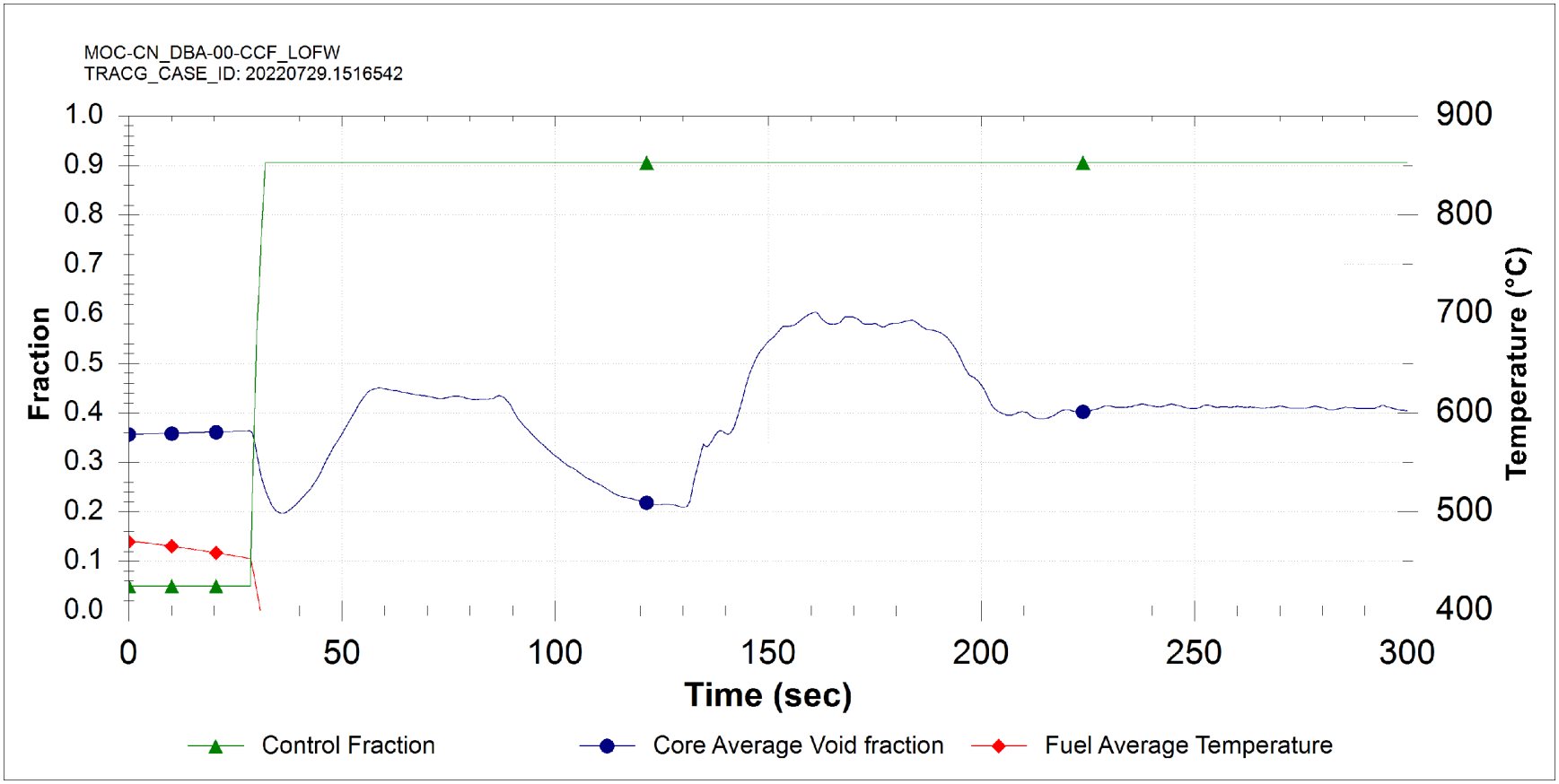

Figure 15.5-92 Loss of Feedwater Flow (DBA)

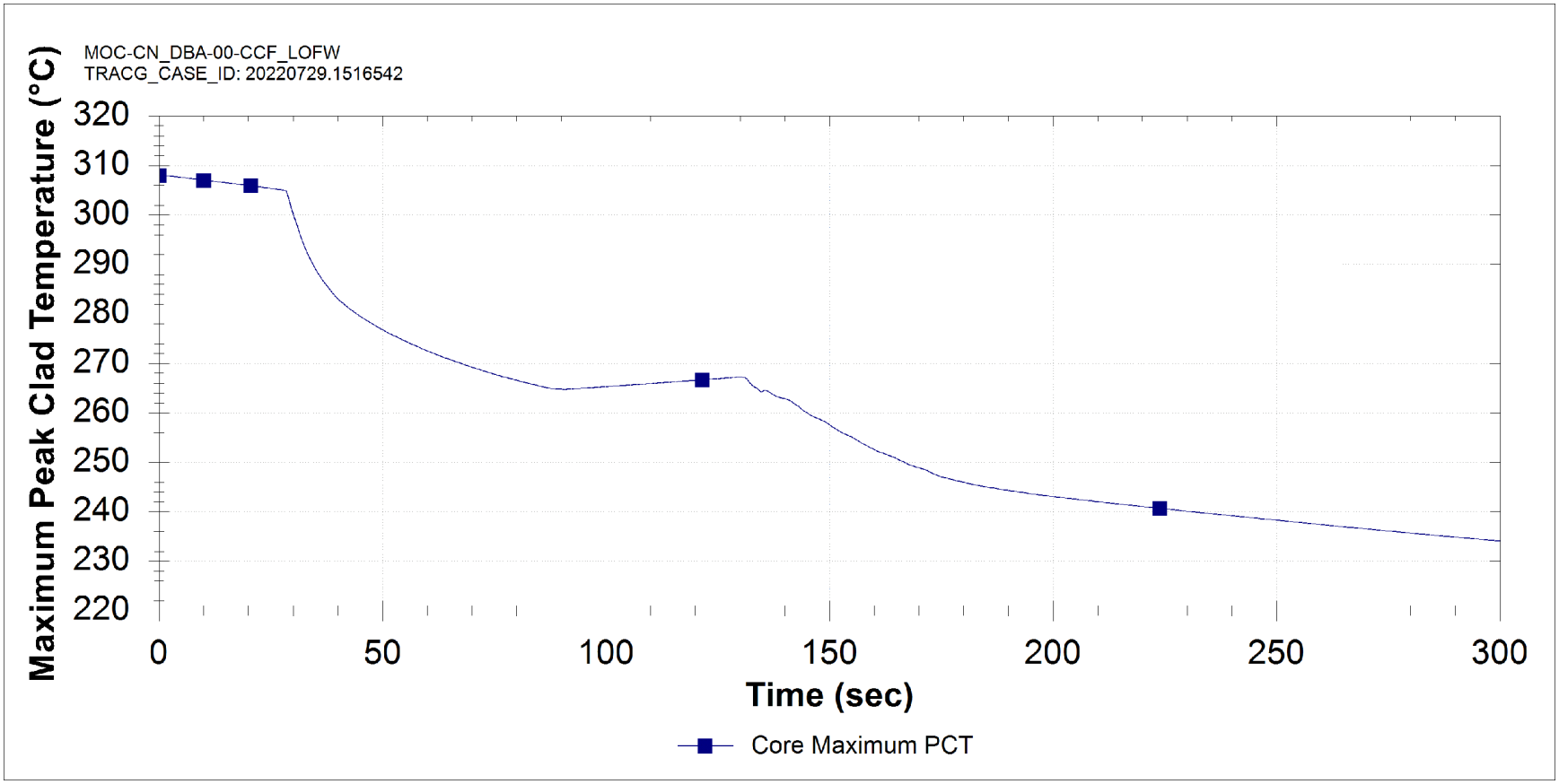

Figure 15.5-93 Loss of Feedwater Flow (DBA)

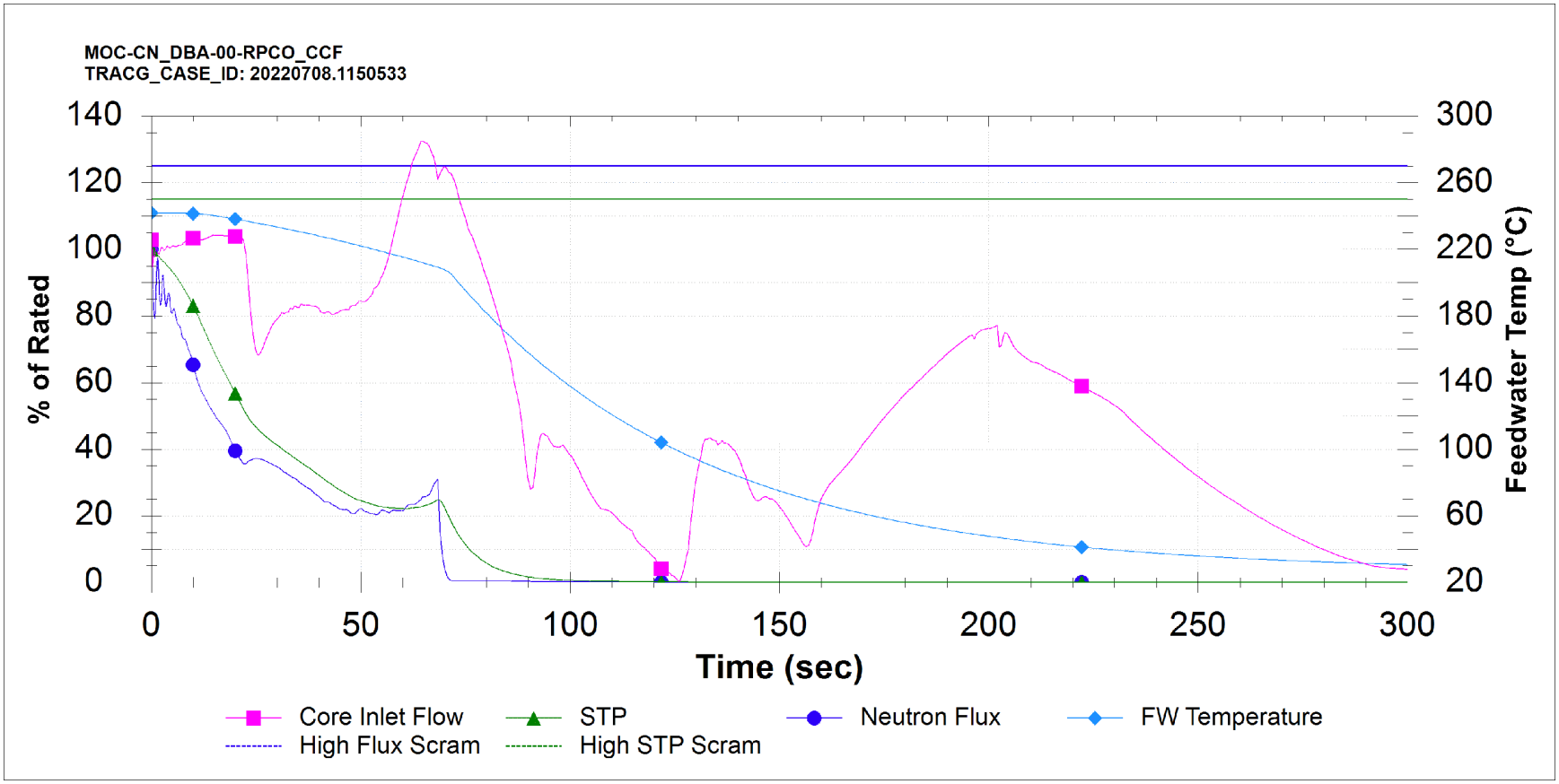

Figure 15.5-94 RPV Pressure Control Open (DBA)

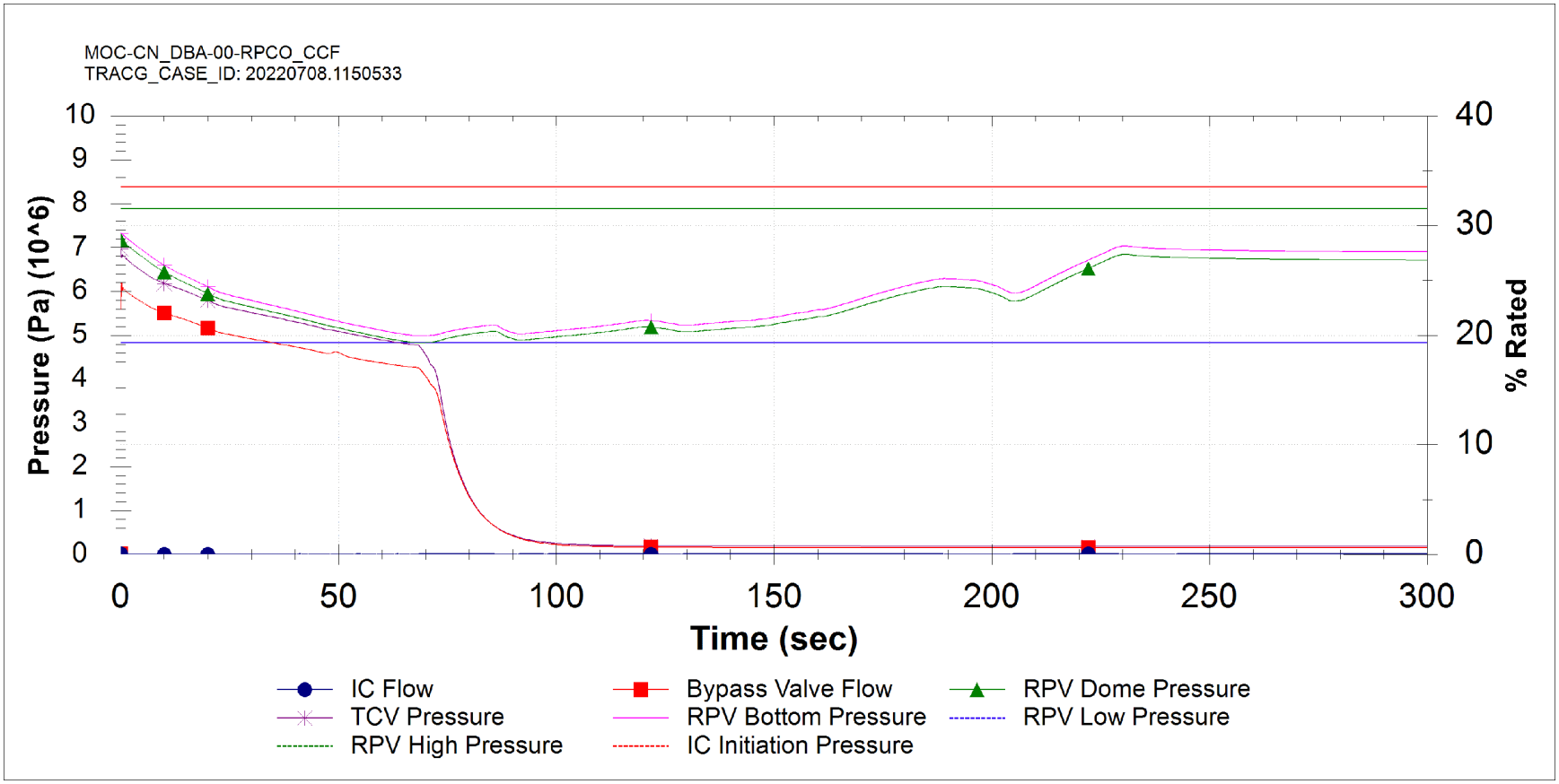

Figure 15.5-95 RPV Pressure Control Open (DBA)

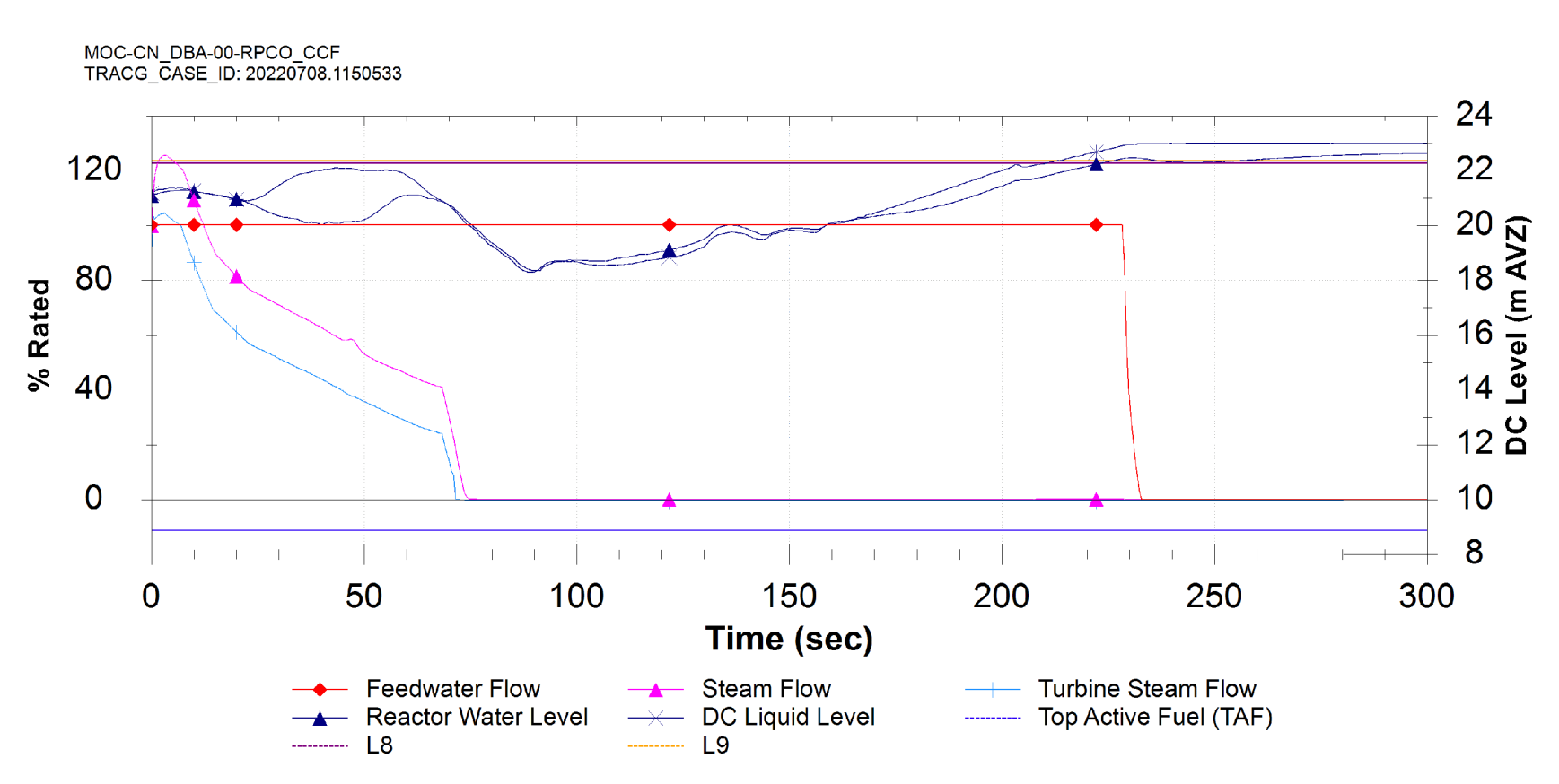

Figure 15.5-96 RPV Pressure Control Open (DBA)

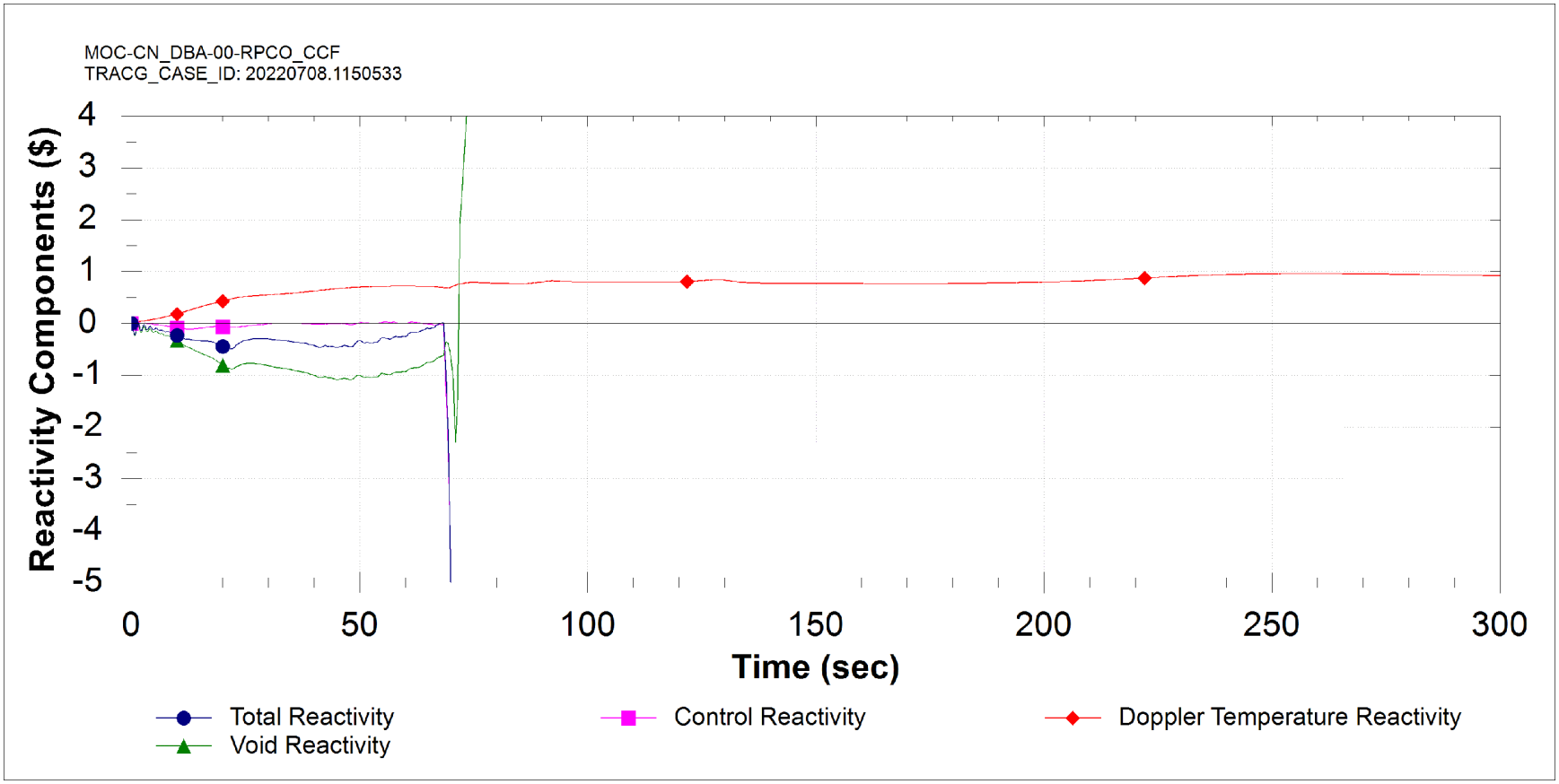

Figure 15.5-97 RPV Pressure Control Open (DBA)

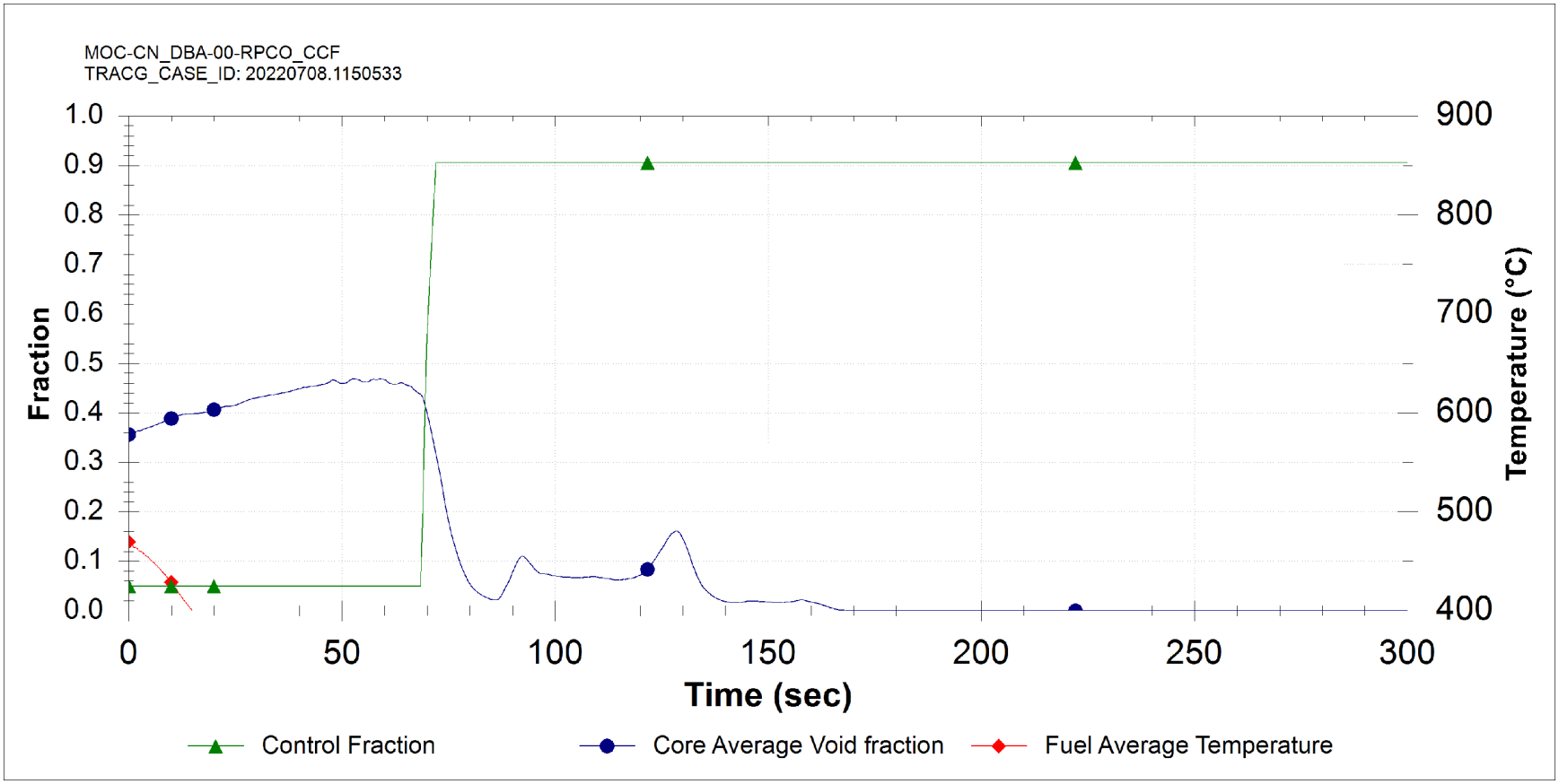

Figure 15.5-98 RPV Pressure Control Open (DBA)

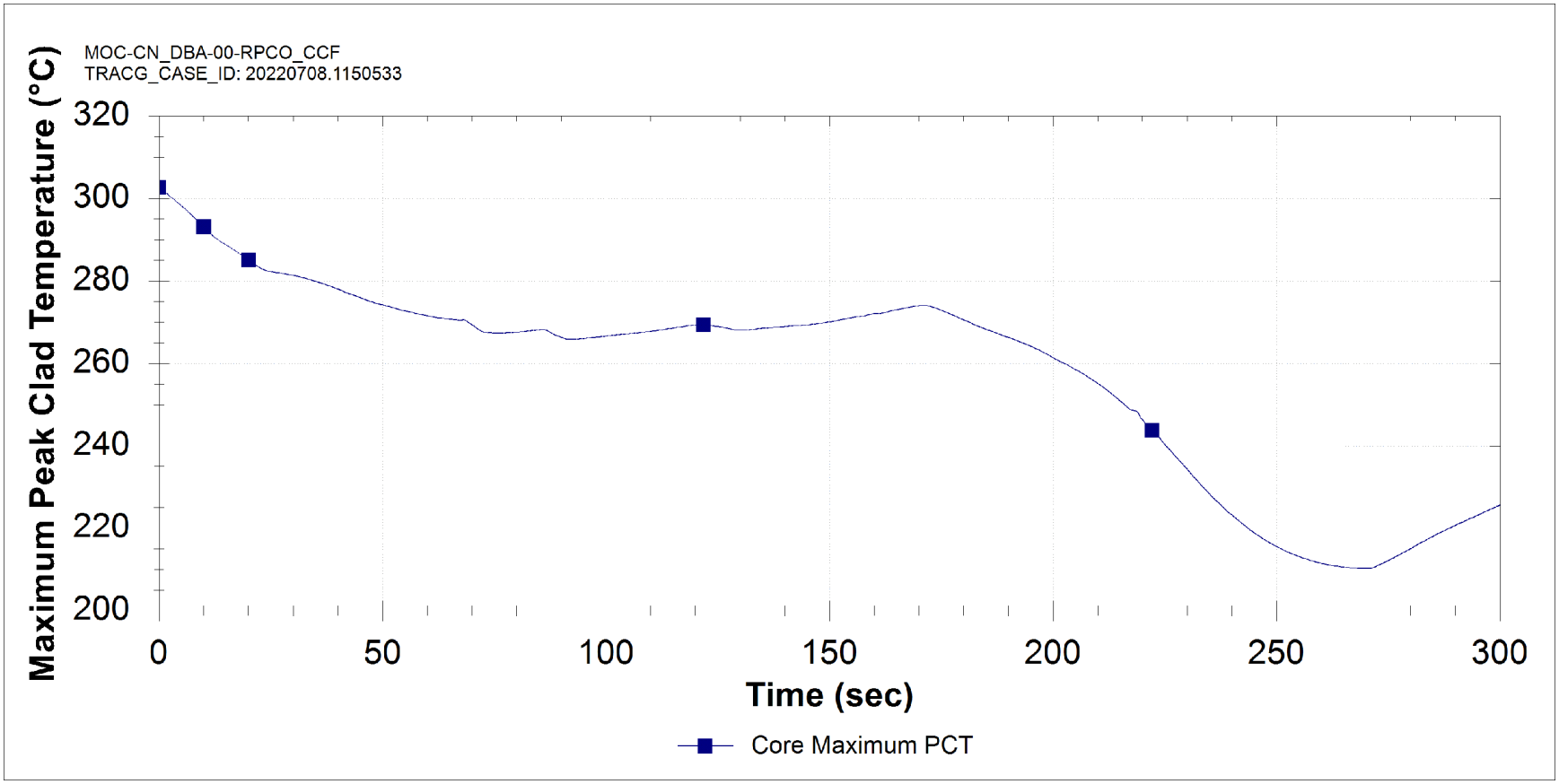

Figure 15.5-99 RPV Pressure Control Open (DBA)

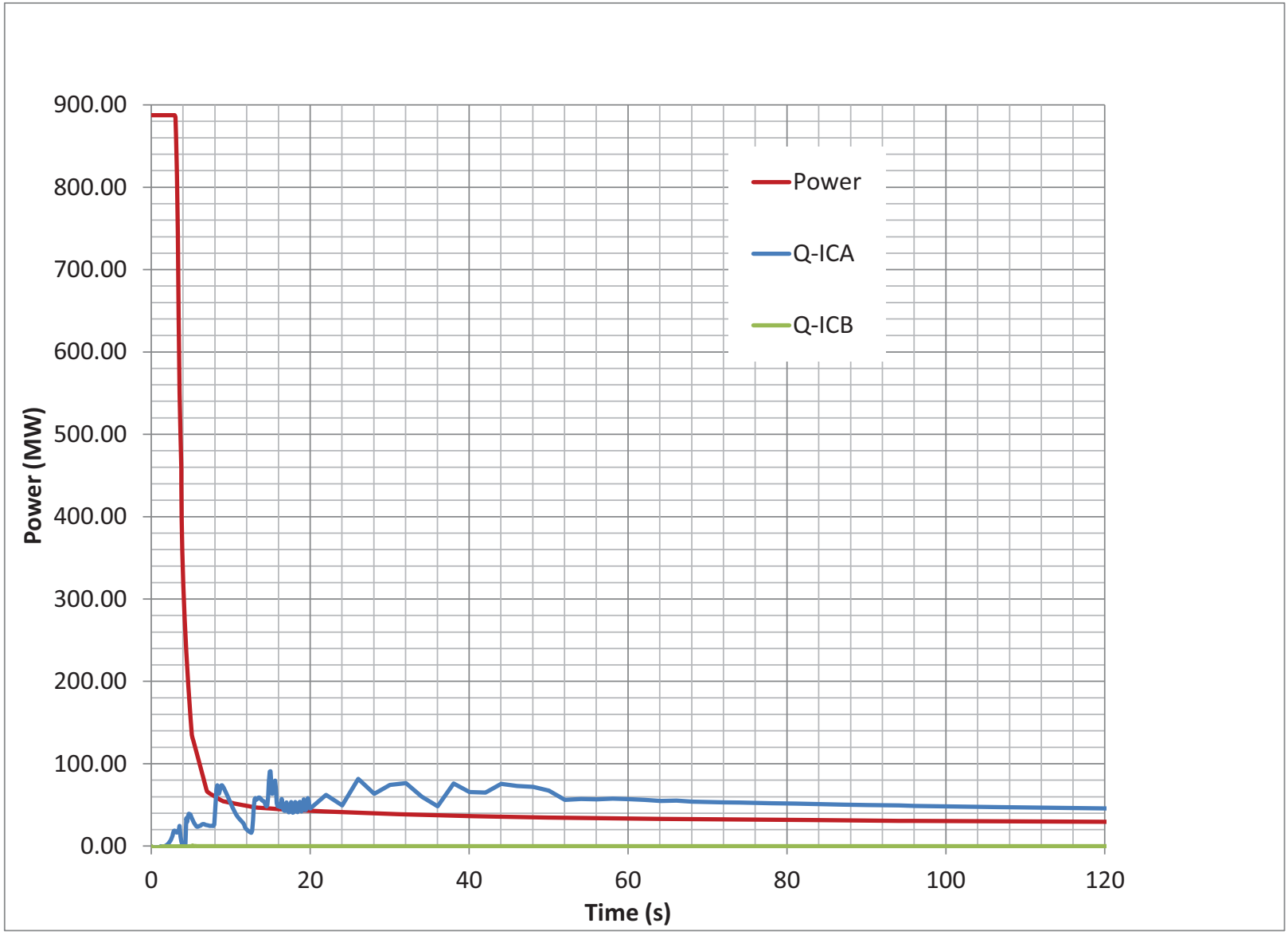

Figure 15.5-100 Reactor Power, Large Main Steam Pipe Break, Conservative Case

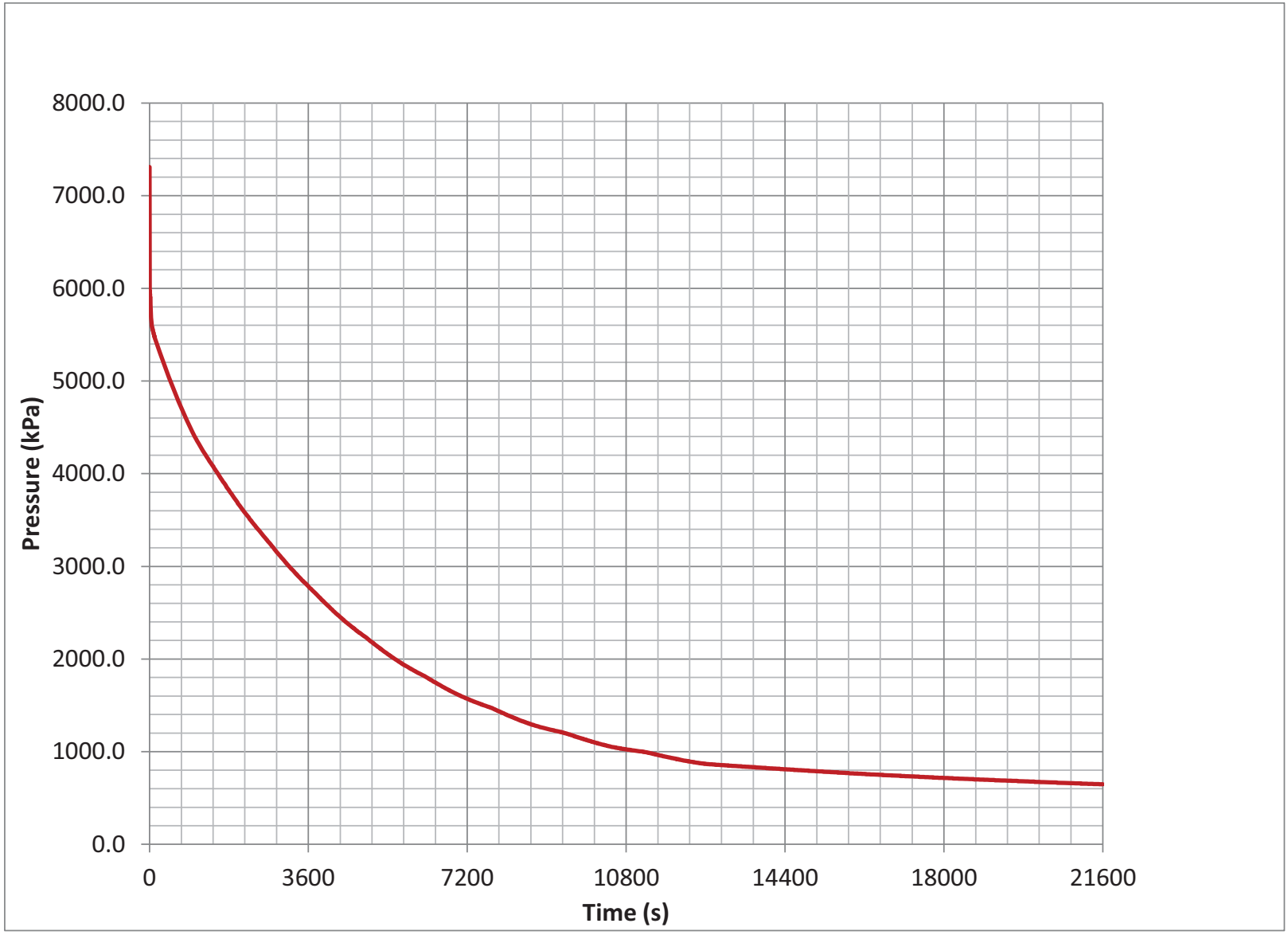

Figure 15.5-101 Reactor Pressure, Large Main Steam Pipe Break, Conservative Case

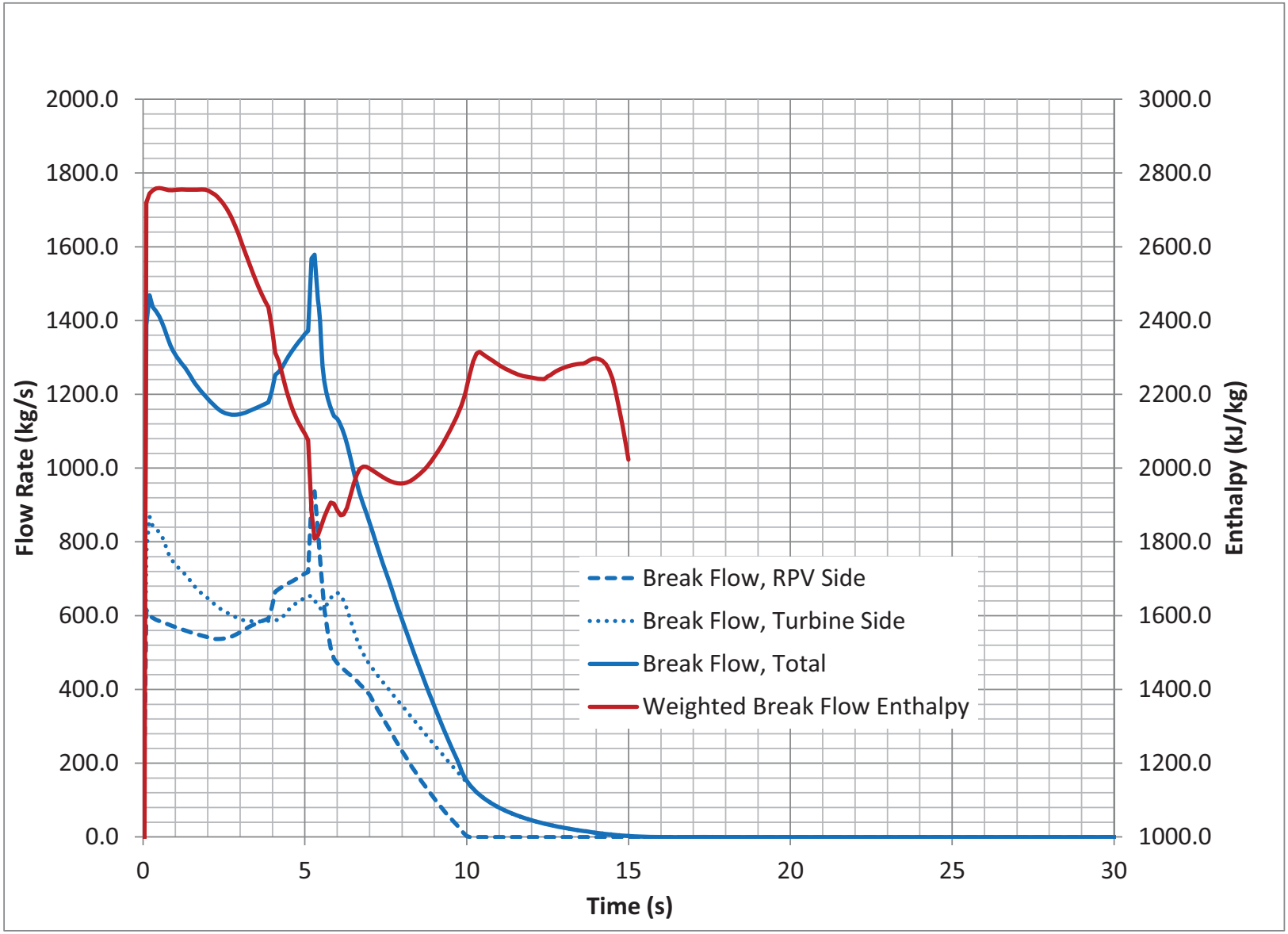

Figure 15.5-102 Break Flow Rate and Enthalpy, Large Main Steam Pipe Break, Conservative Case

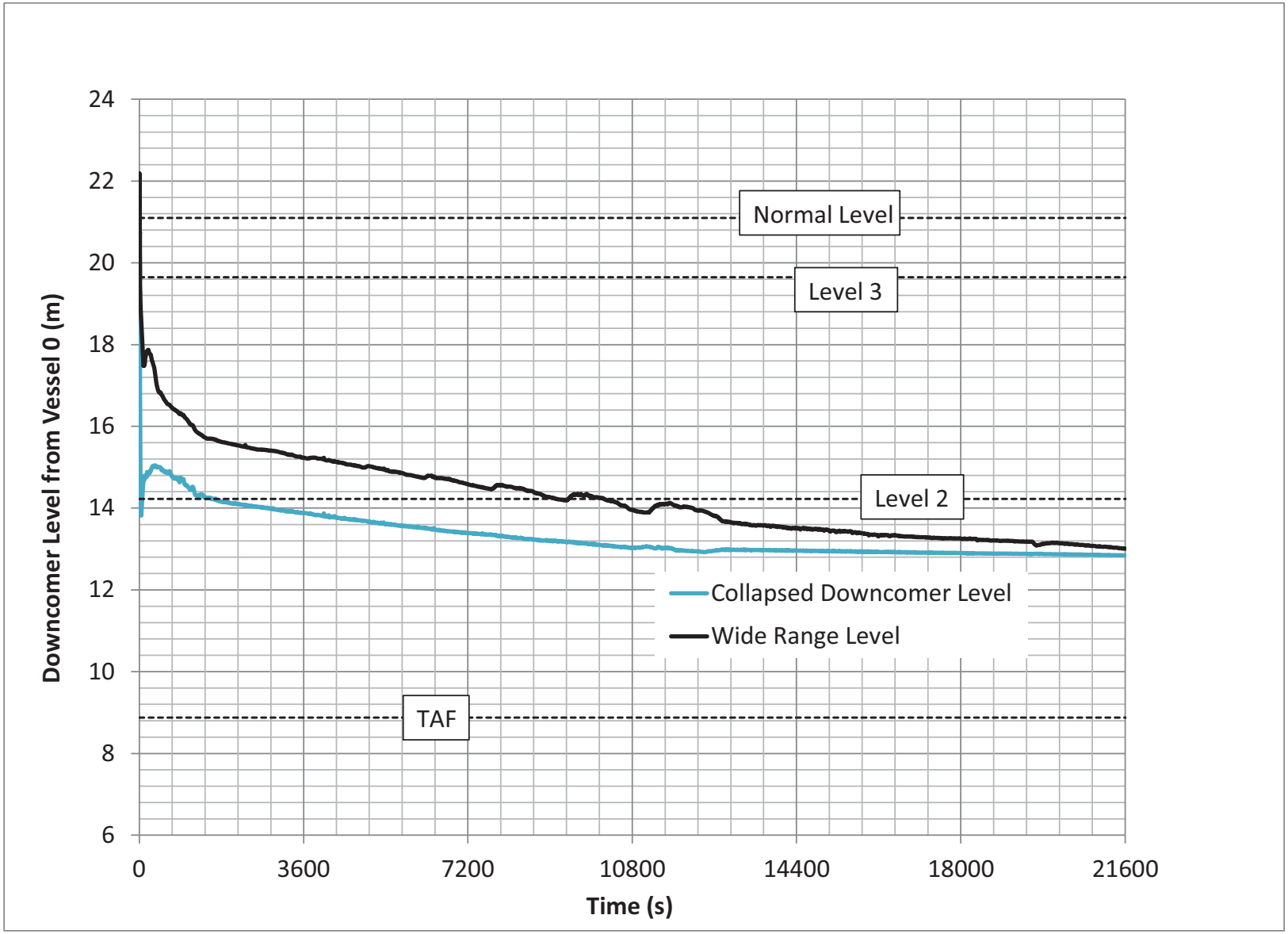

Figure 15.5-103 Reactor Water Level, Large Main Steam Pipe Break, Conservative Case

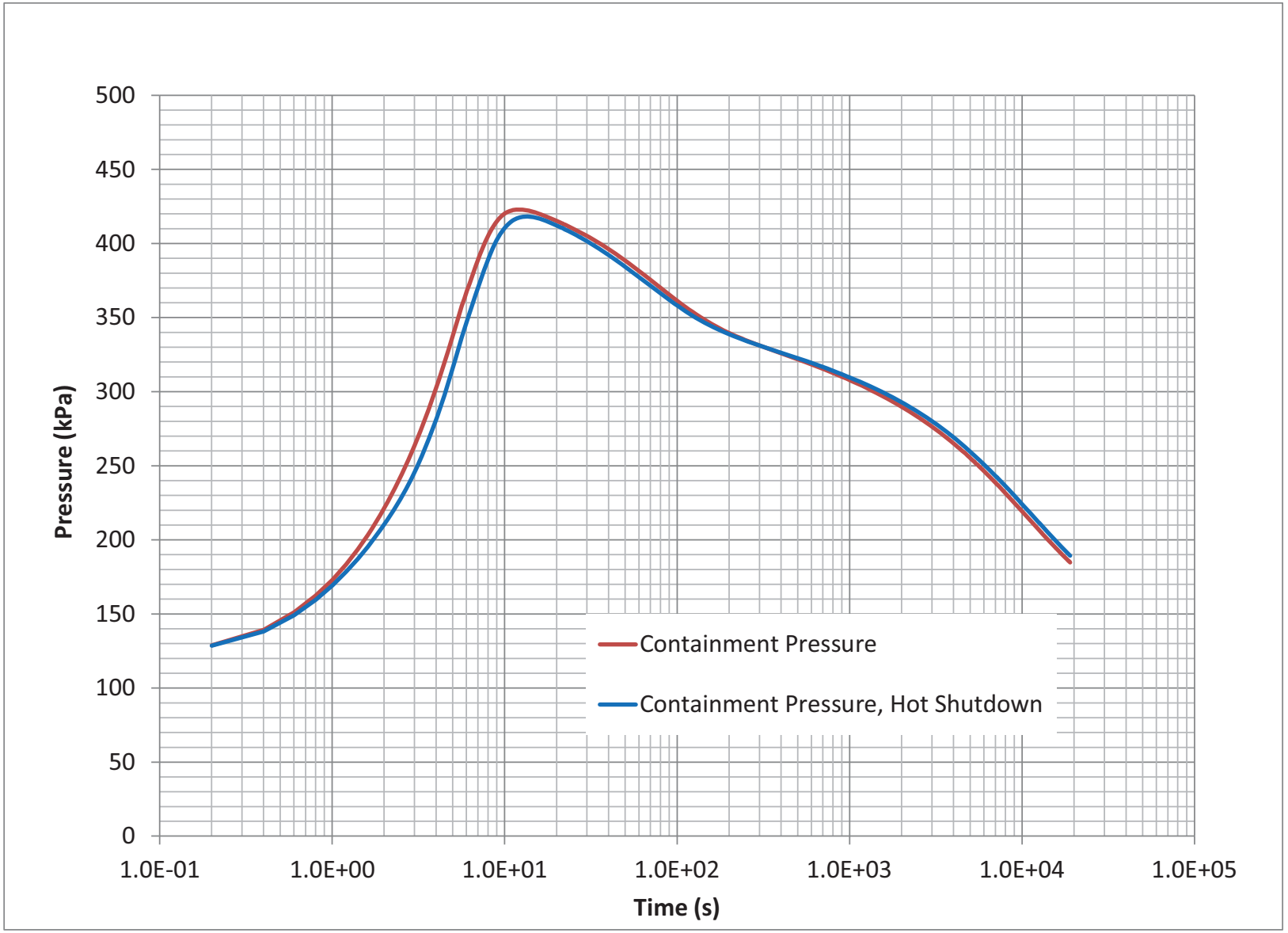

Figure 15.5-104 Containment Pressure, Large Main Steam Pipe Break, Conservative Case

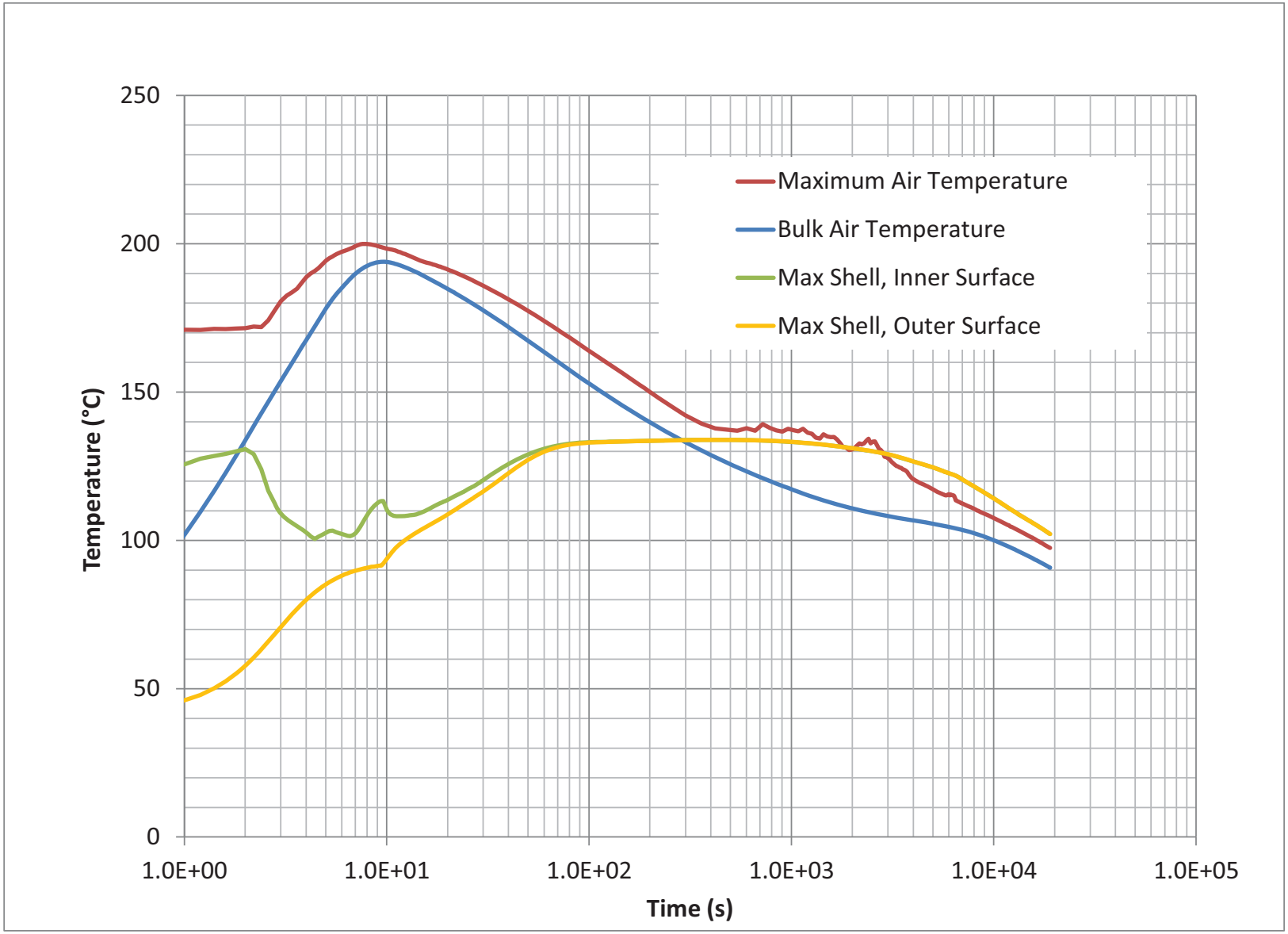

Figure 15.5-105 Containment Temperatures, Large Main Steam Pipe Break, Conservative Case

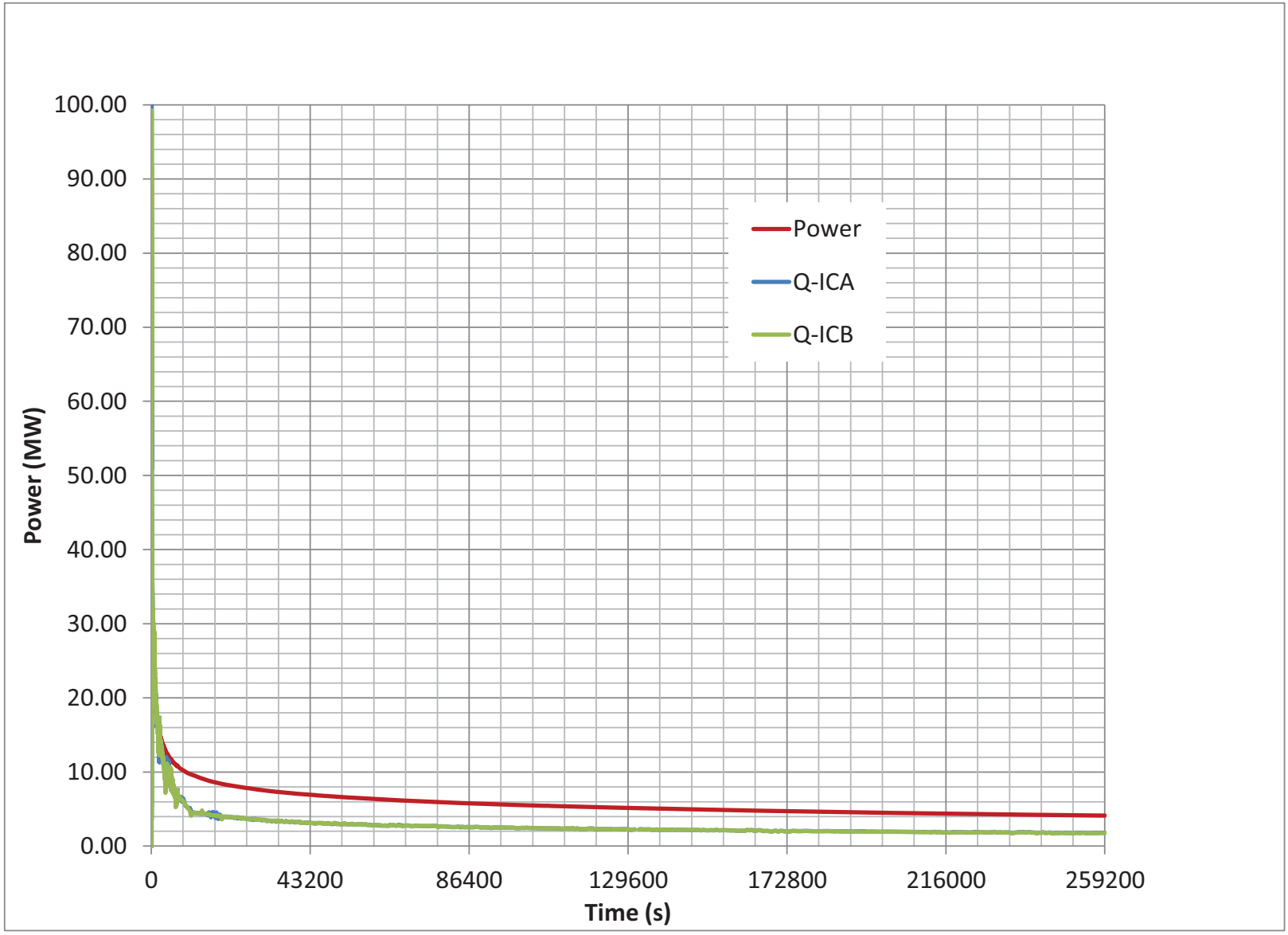

Figure 15.5-106 Reactor Power, Small Steam Break With LOPP, Conservative Case

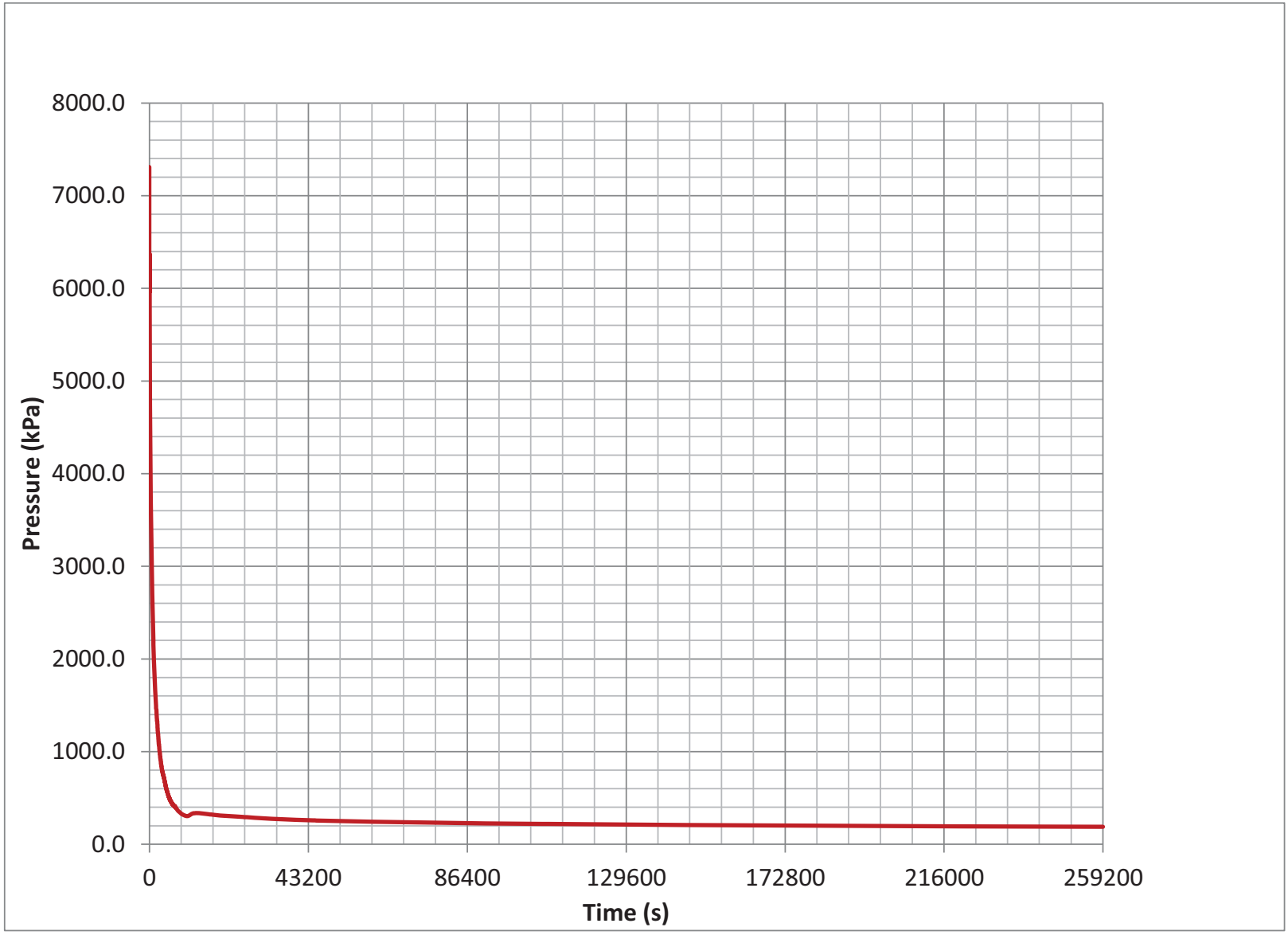

Figure 15.5-107 Reactor Pressure, Small Steam Pipe Break With LOPP, Conservative Case

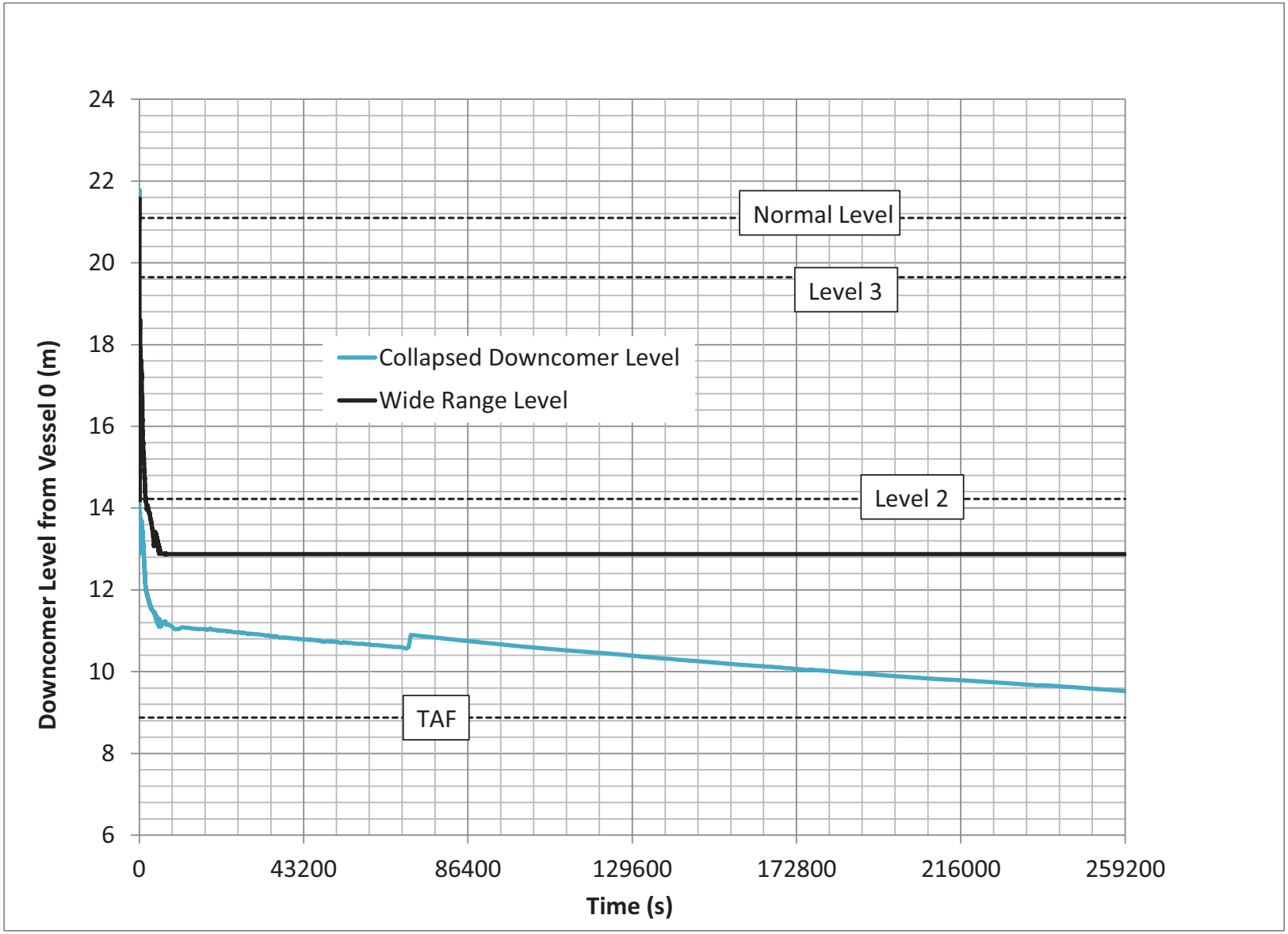

Figure 15.5-108 Reactor Water Level, Small Steam Pipe Break With LOPP, Conservative Case

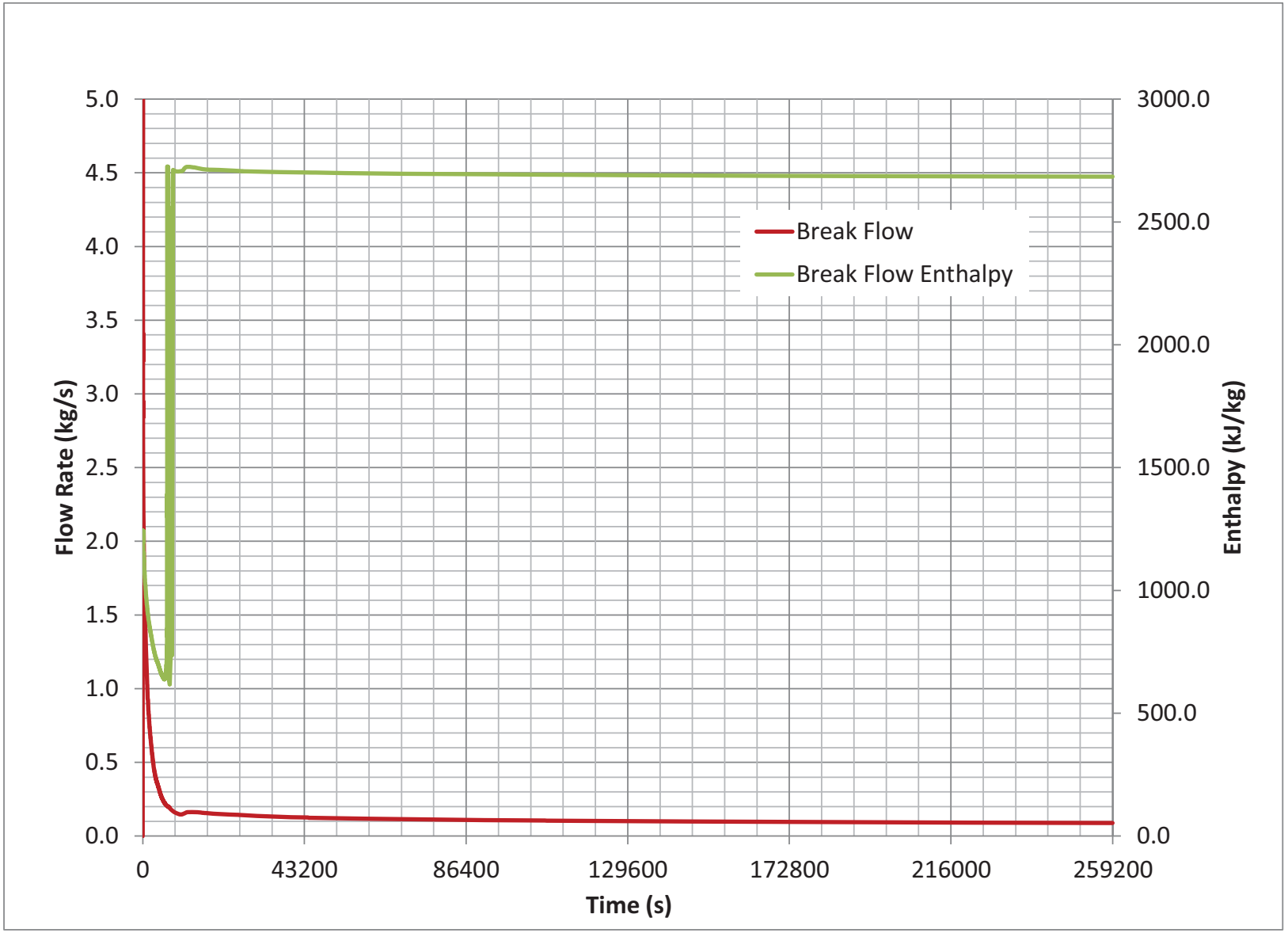

Figure 15.5-109 Break Flow Rate and Enthalpy, Small Steam Pipe Break With LOPP, Conservative Case

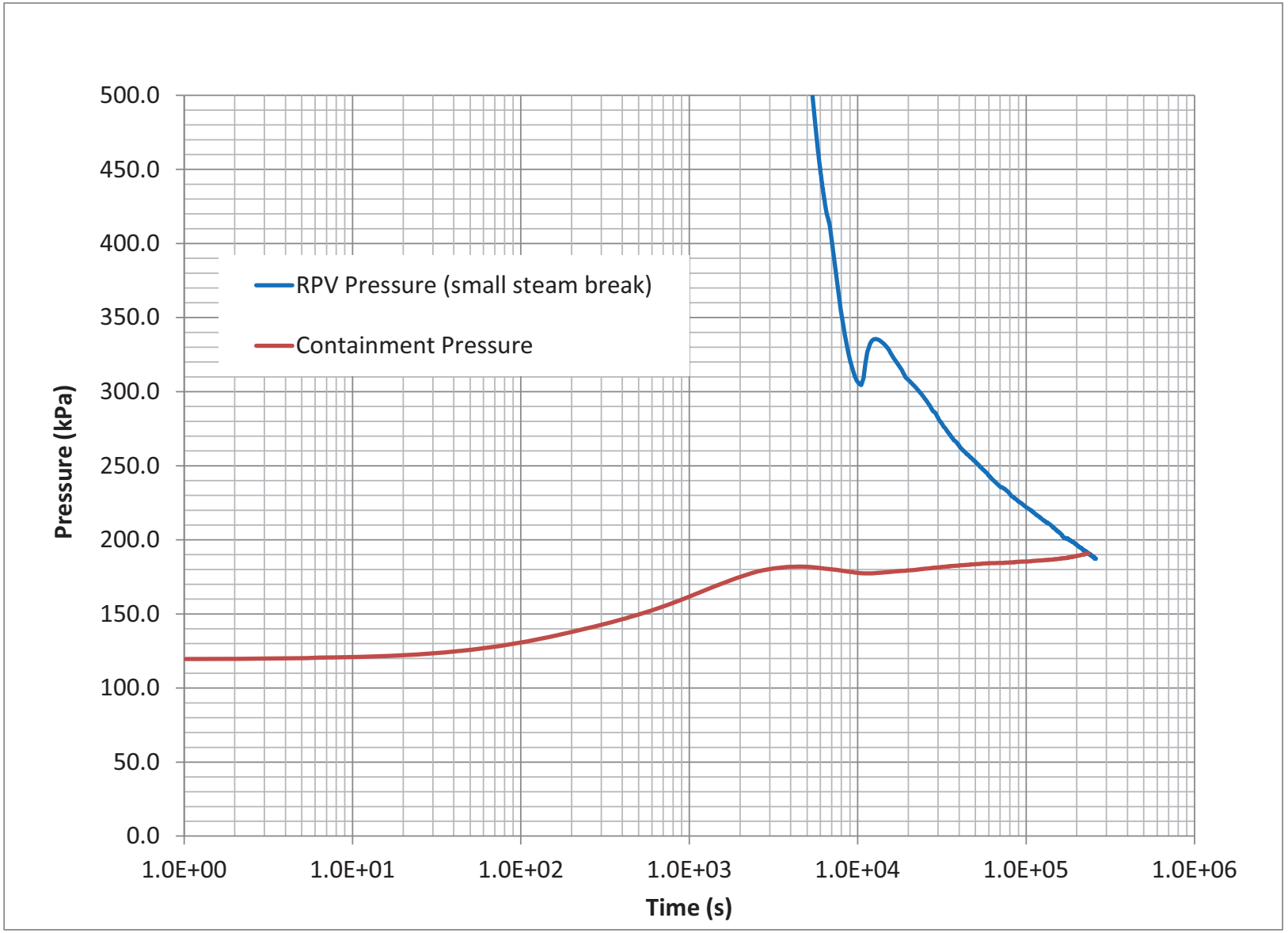

Figure 15.5-110 Containment Pressure, Small Steam Pipe Break With LOPP, Conservative Case

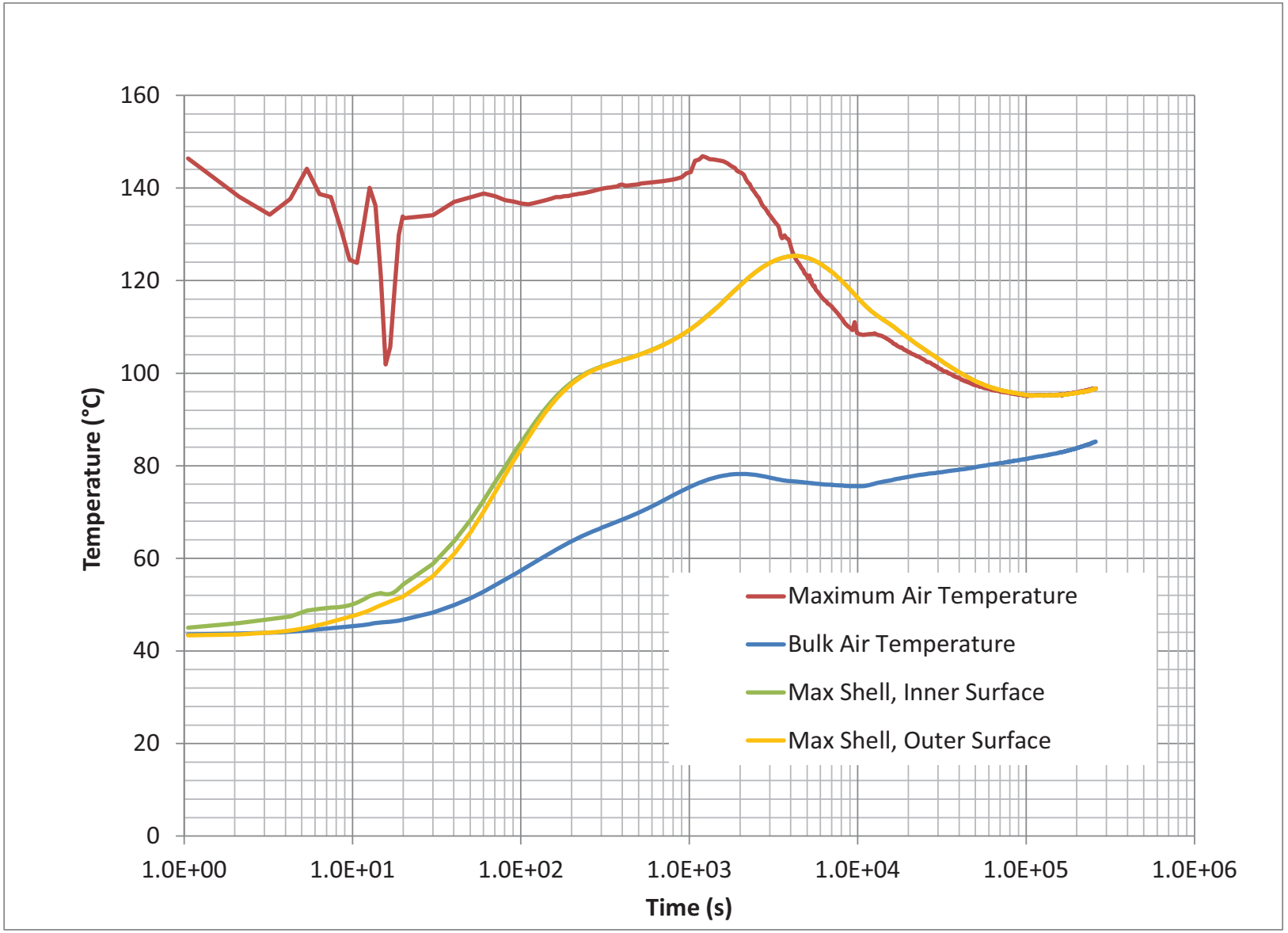

Figure 15.5-111 Containment Temperature, Small Steam Pipe Break with LOPP, Conservative Case

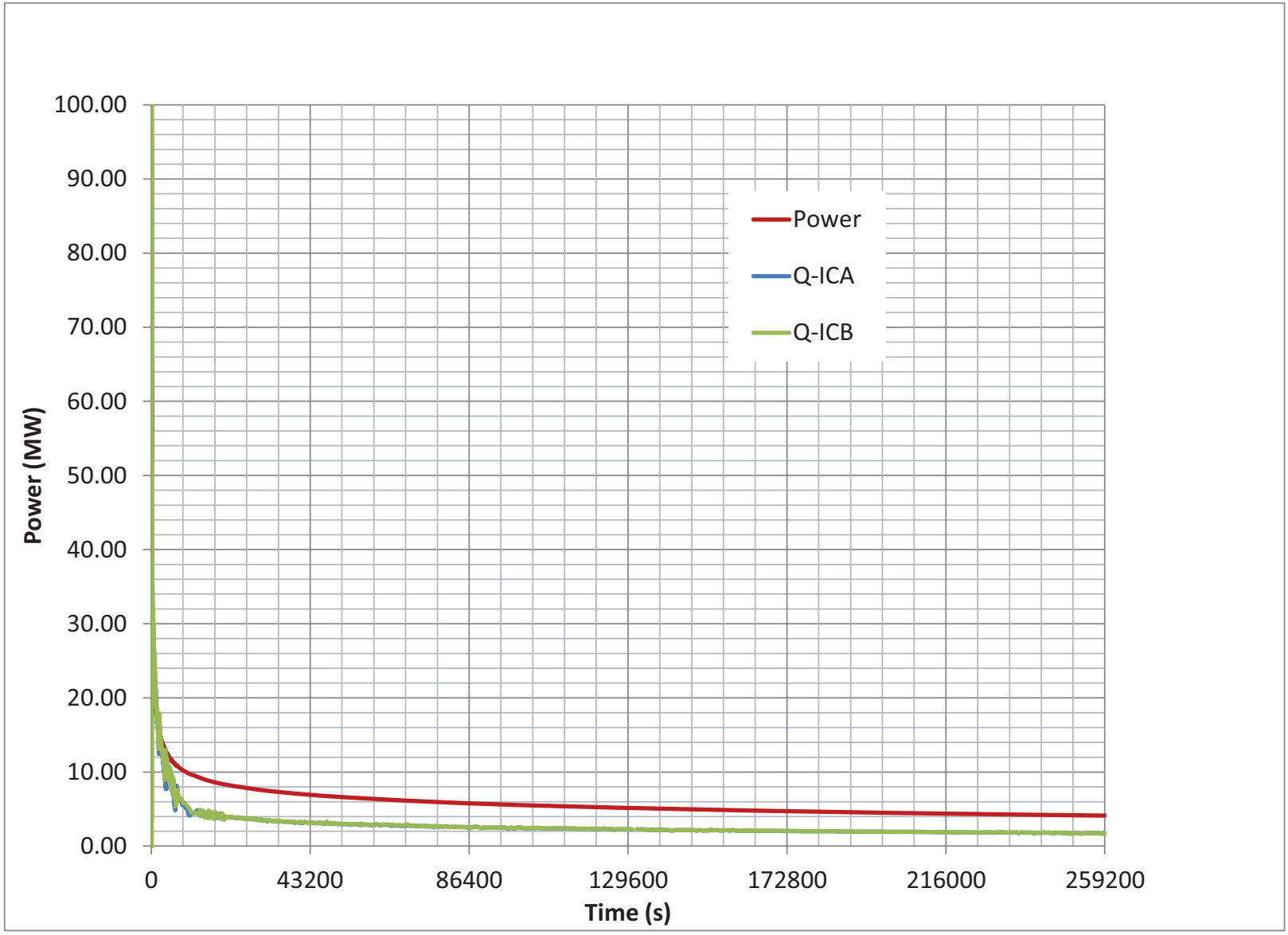

Figure 15.5-112 Reactor Power, Small Liquid Pipe Break, Conservative Case

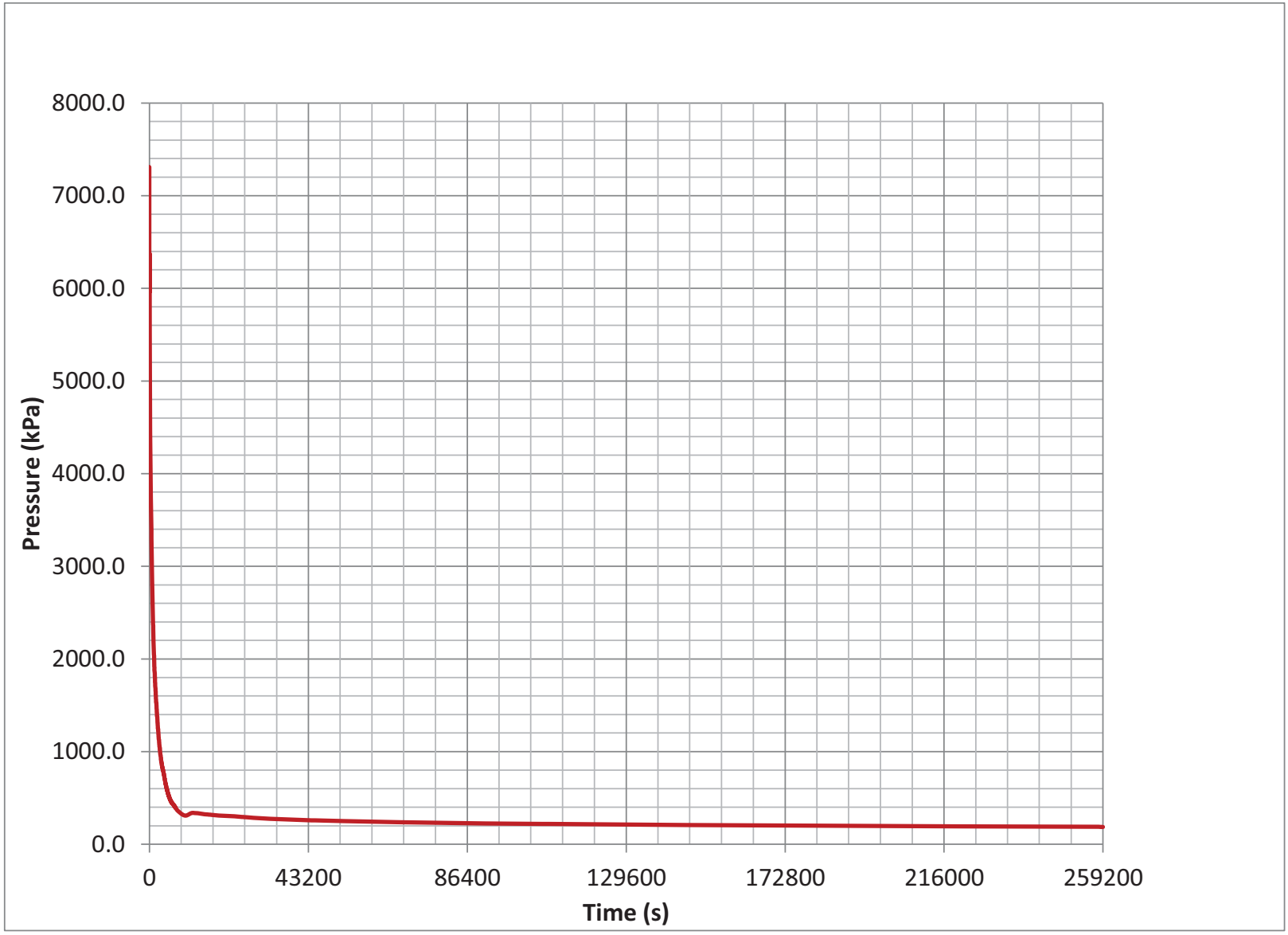

Figure 15.5-113 Reactor Pressure, Small Liquid Pipe Break, Conservative Case

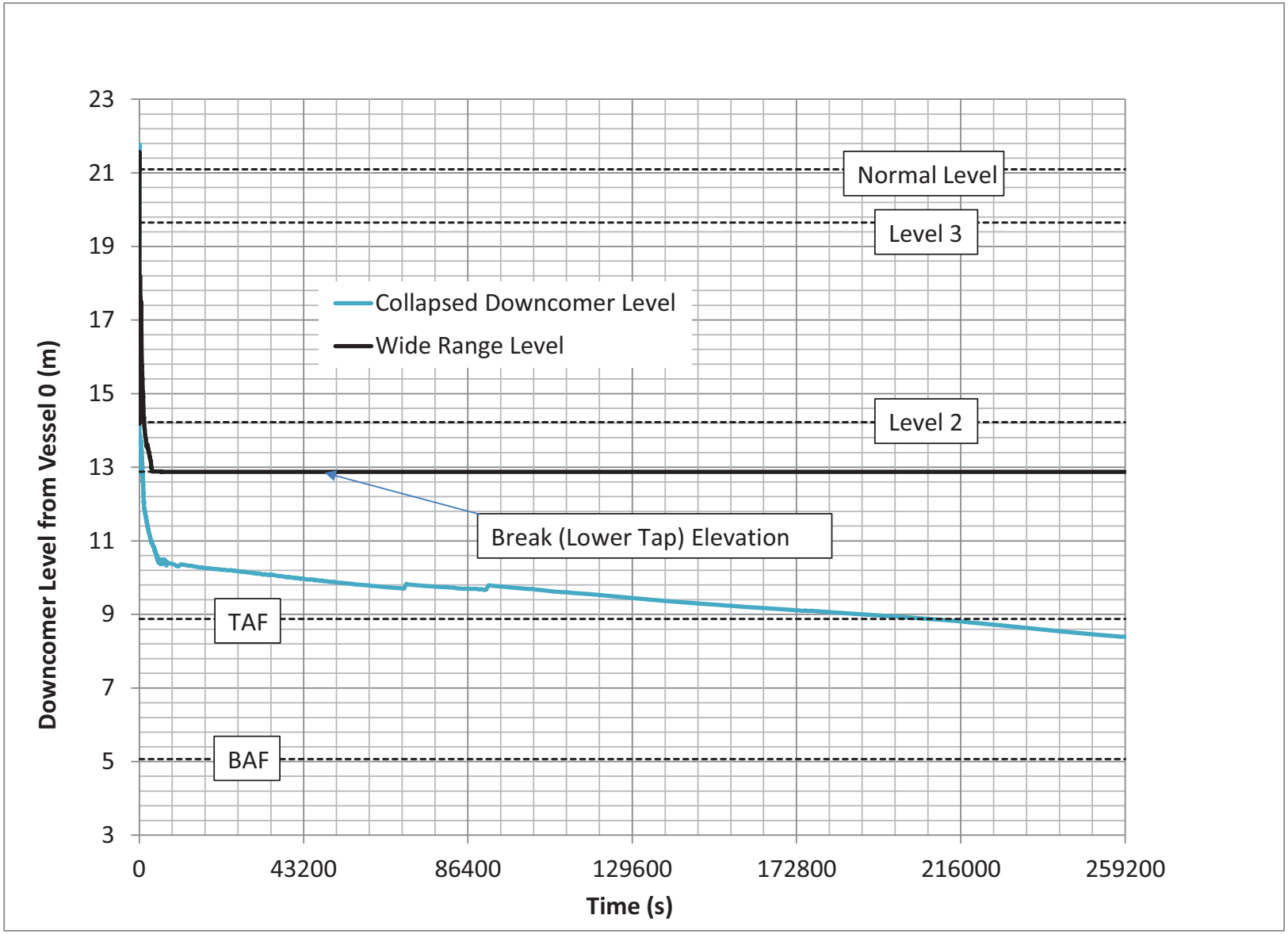

Figure 15.5-114 Reactor Water Level, Small Liquid Break, Conservative Case

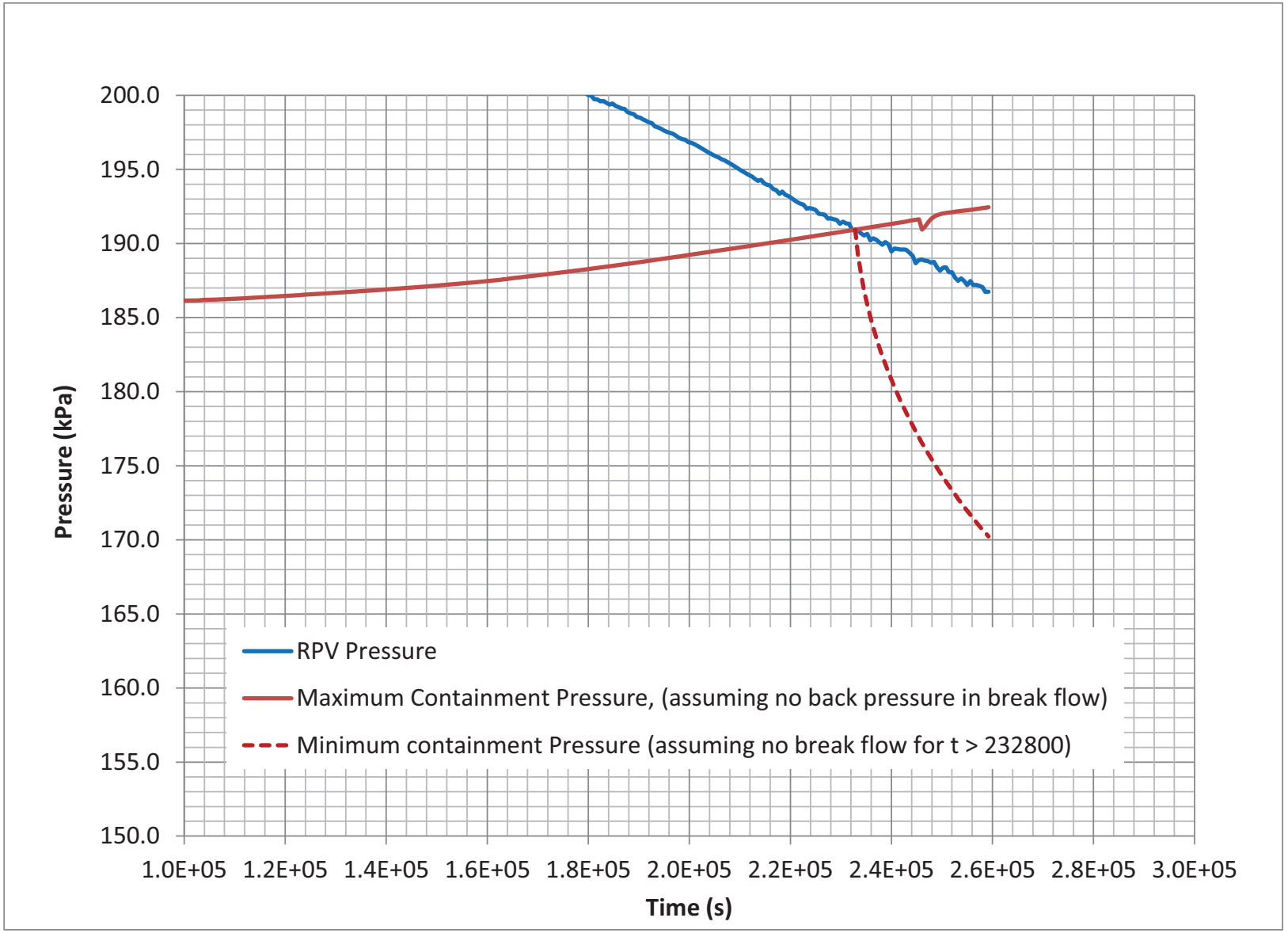

Figure 15.5-115 Containment Pressure After RPV Depressurization, Small Liquid Pipe Break, Conservative Case

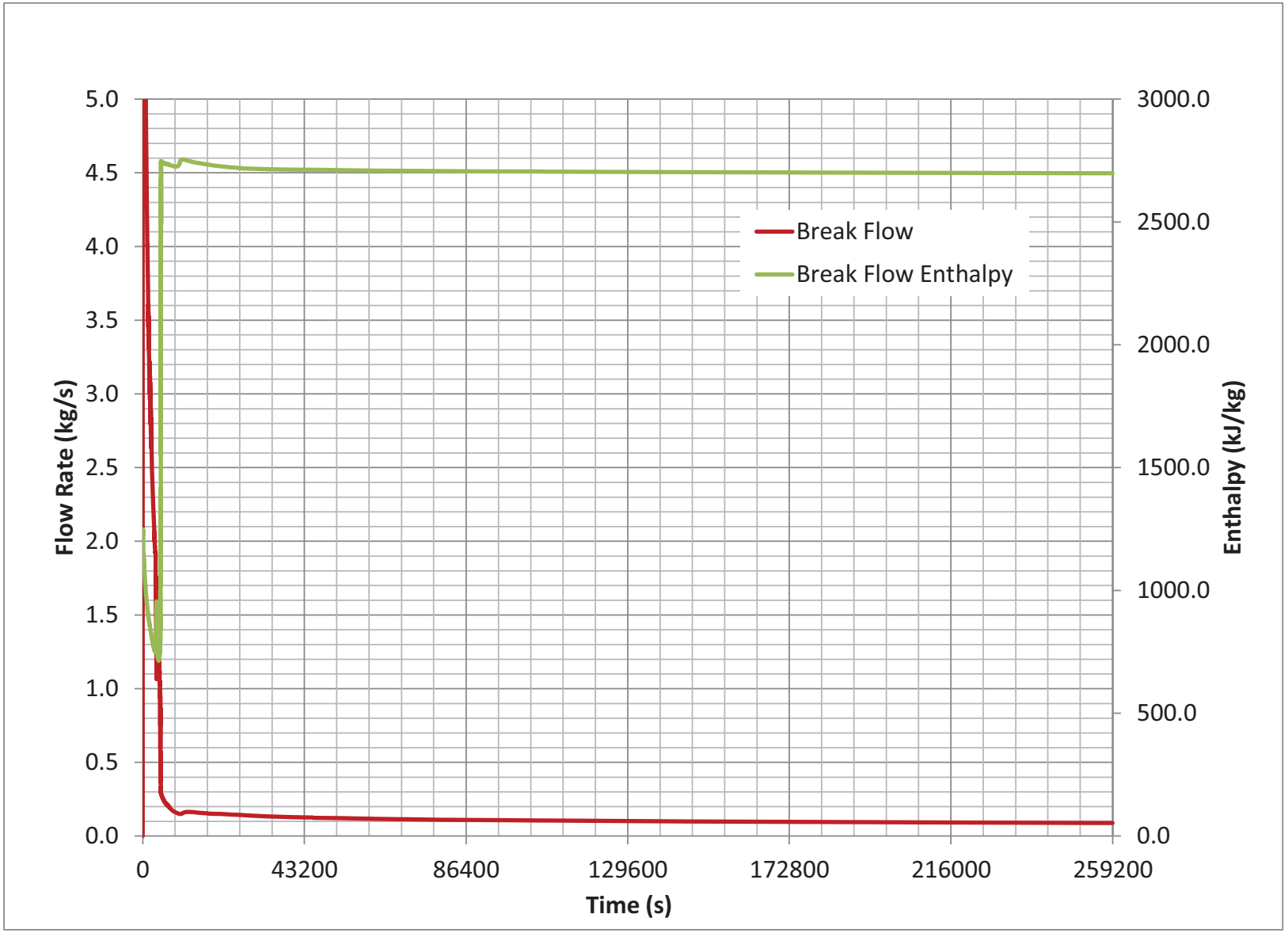

Figure 15.5-116 Break Flow Rate and Enthalpy, Small Liquid Pipe Break, Conservative Case

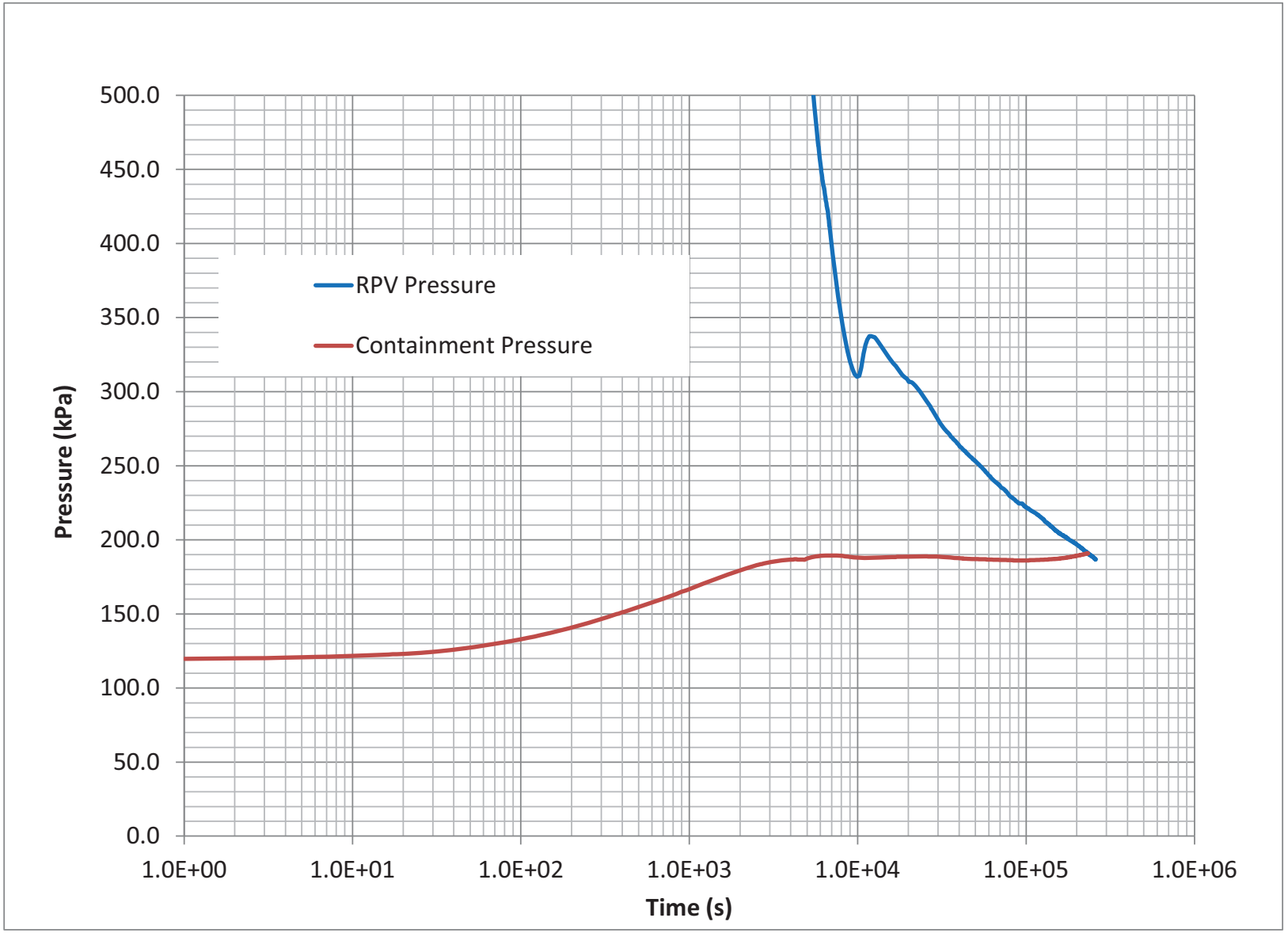

Figure 15.5-117 Containment Pressure, Small Liquid Pipe Break, Conservative Case

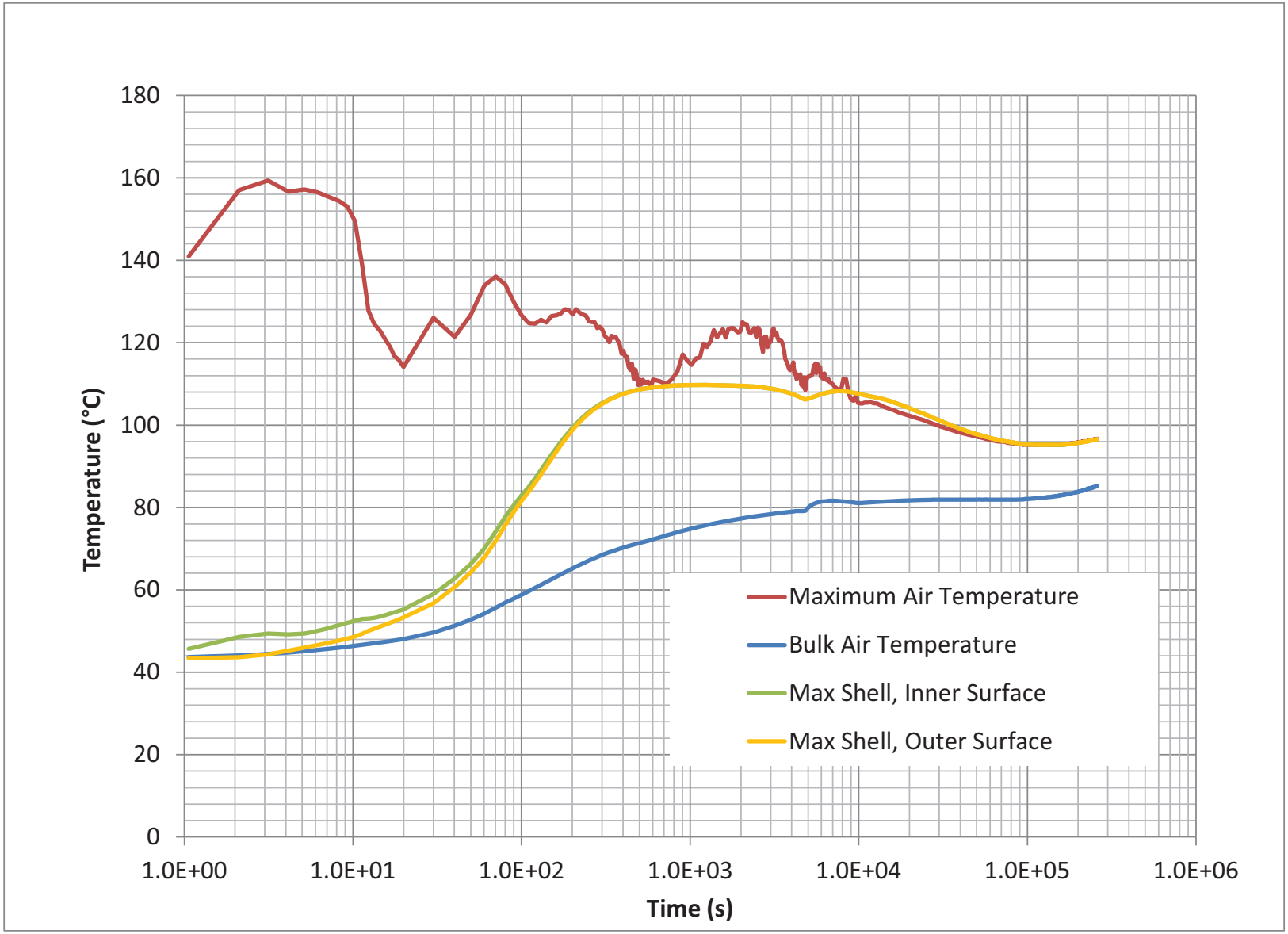

Figure 15.5-118 Containment Temperature, Small Liquid Pipe Break, Conservative Case

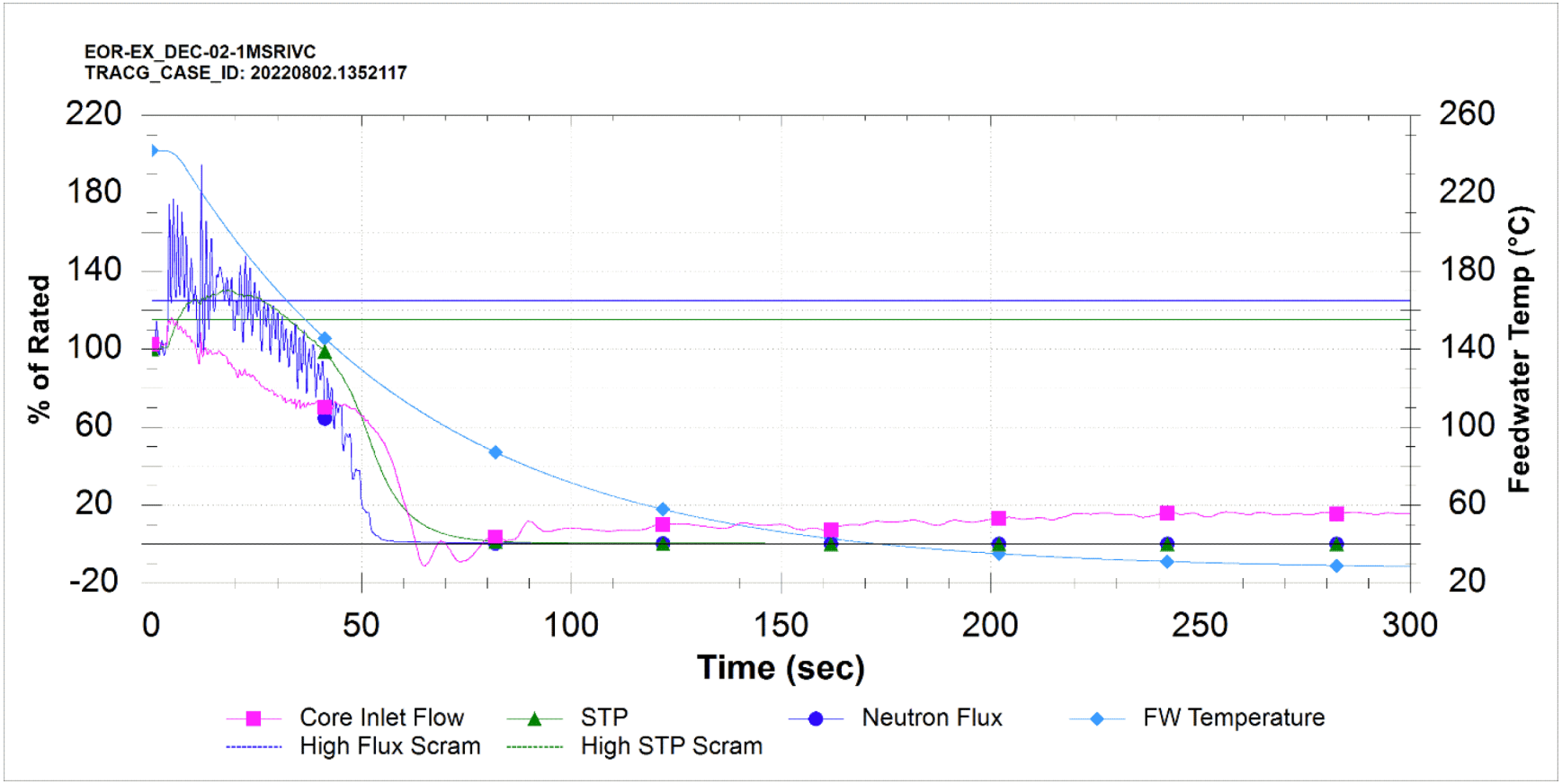

Figure 15.5-119 Closure of One Main Steam Reactor Isolation Valve with Failure to Scram (DEC)

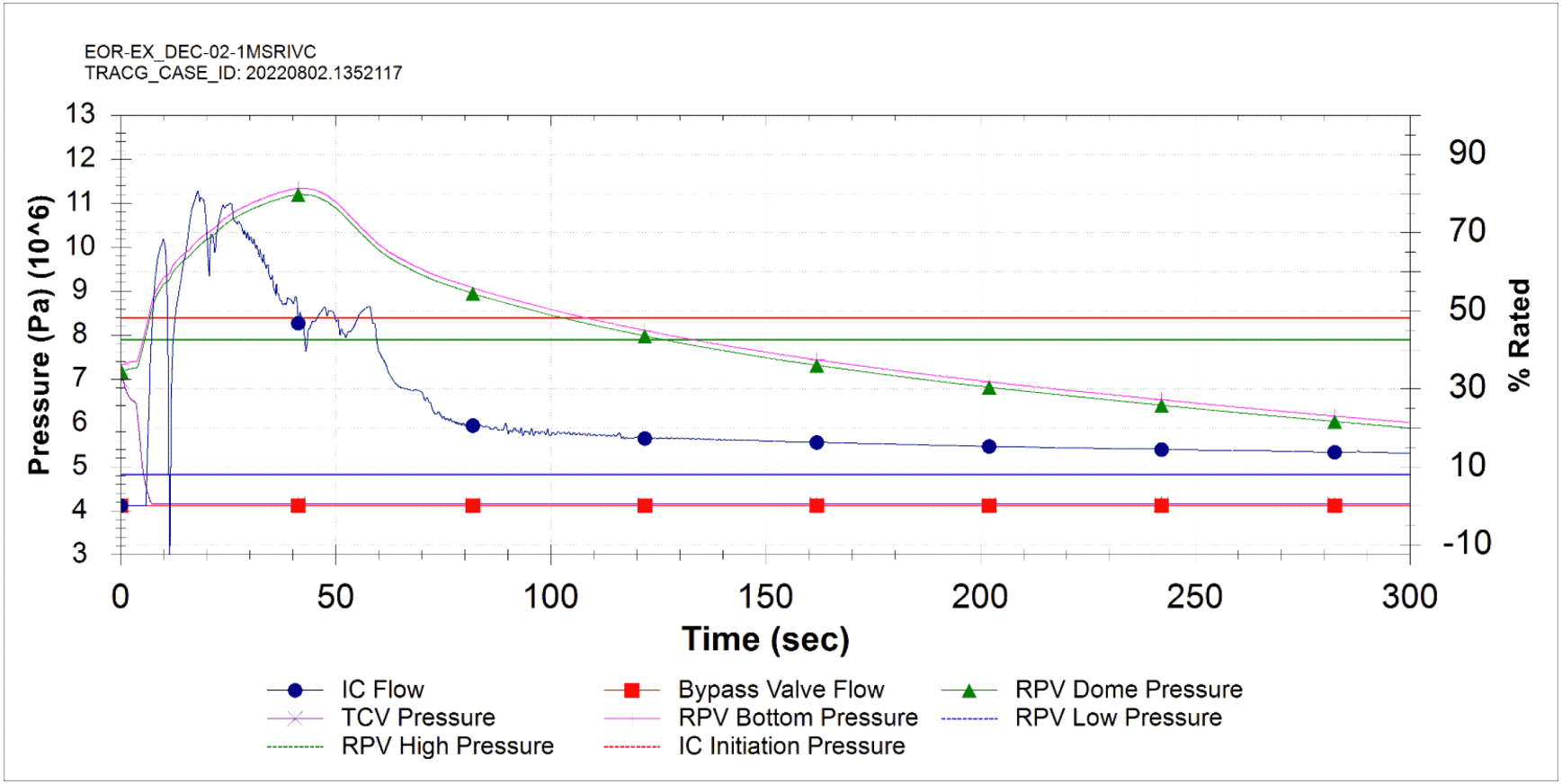

Figure 15.5-120 Closure of One Main Steam Reactor Isolation Valve with Failure to Scram (DEC)

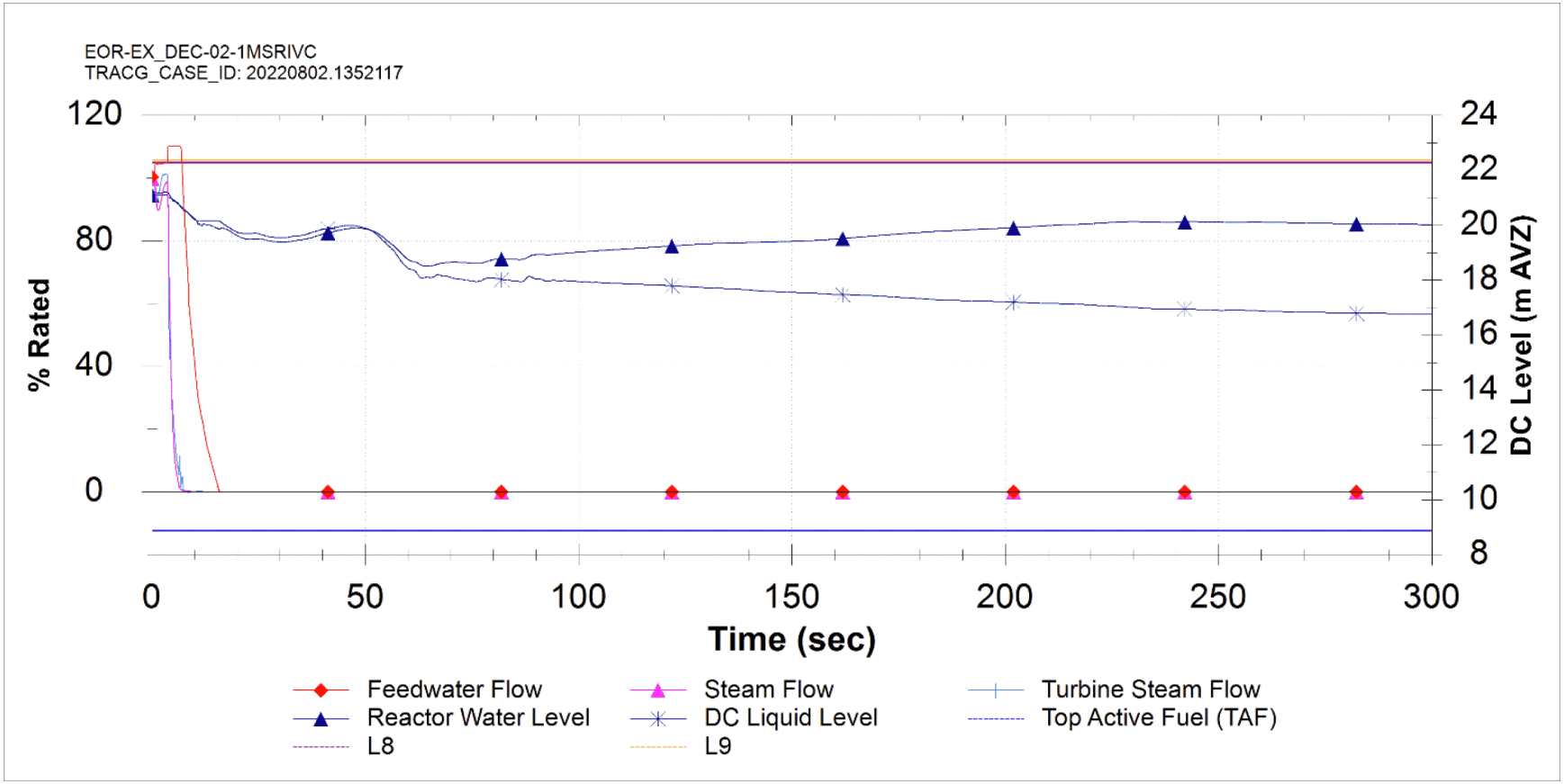

Figure 15.5-121 Closure of One Main Steam Reactor Isolation Valve with Failure to Scram (DEC)

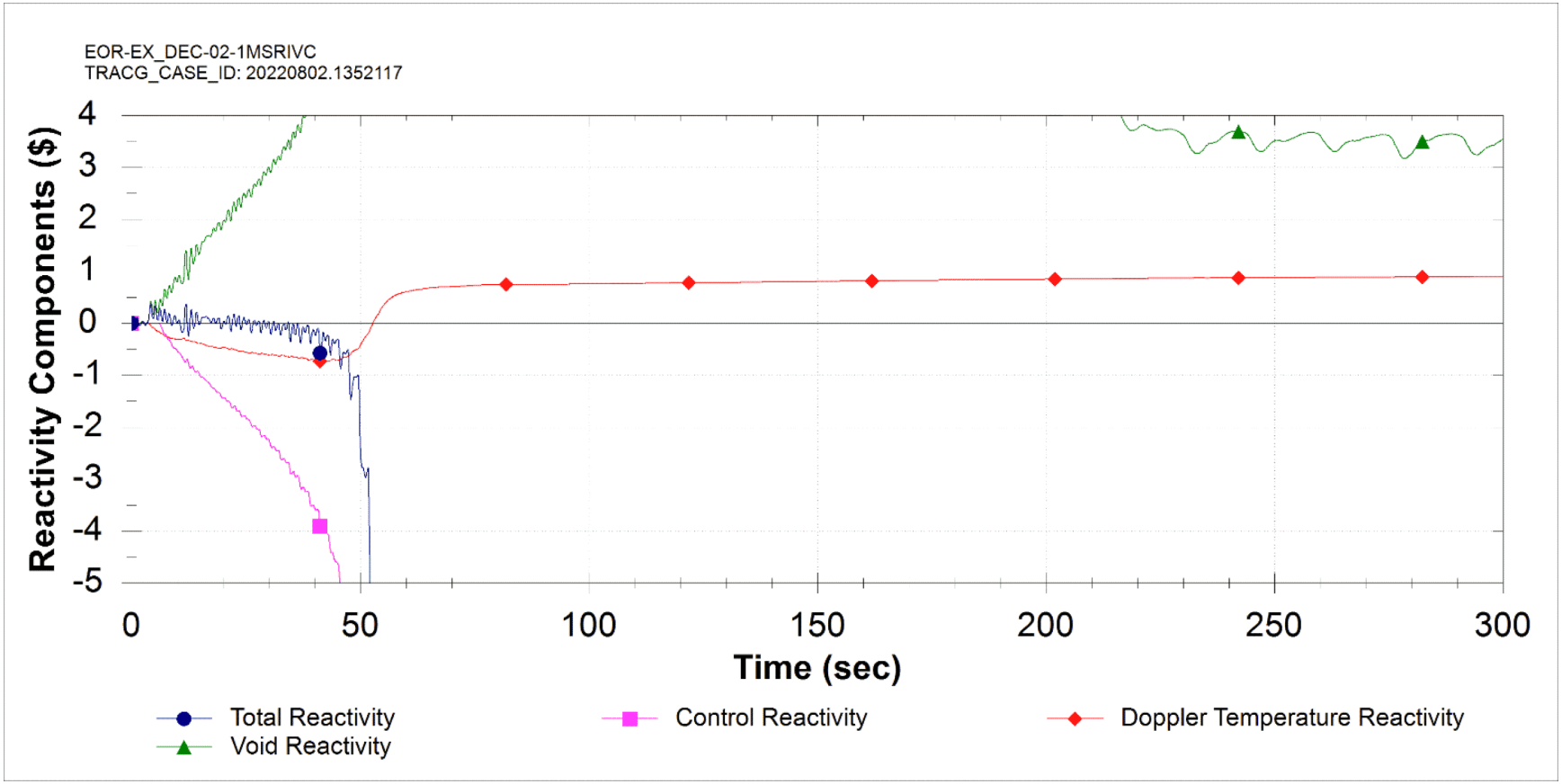

Figure 15.5-122 Closure of One Main Steam Reactor Isolation Valve with Failure to Scram (DEC)

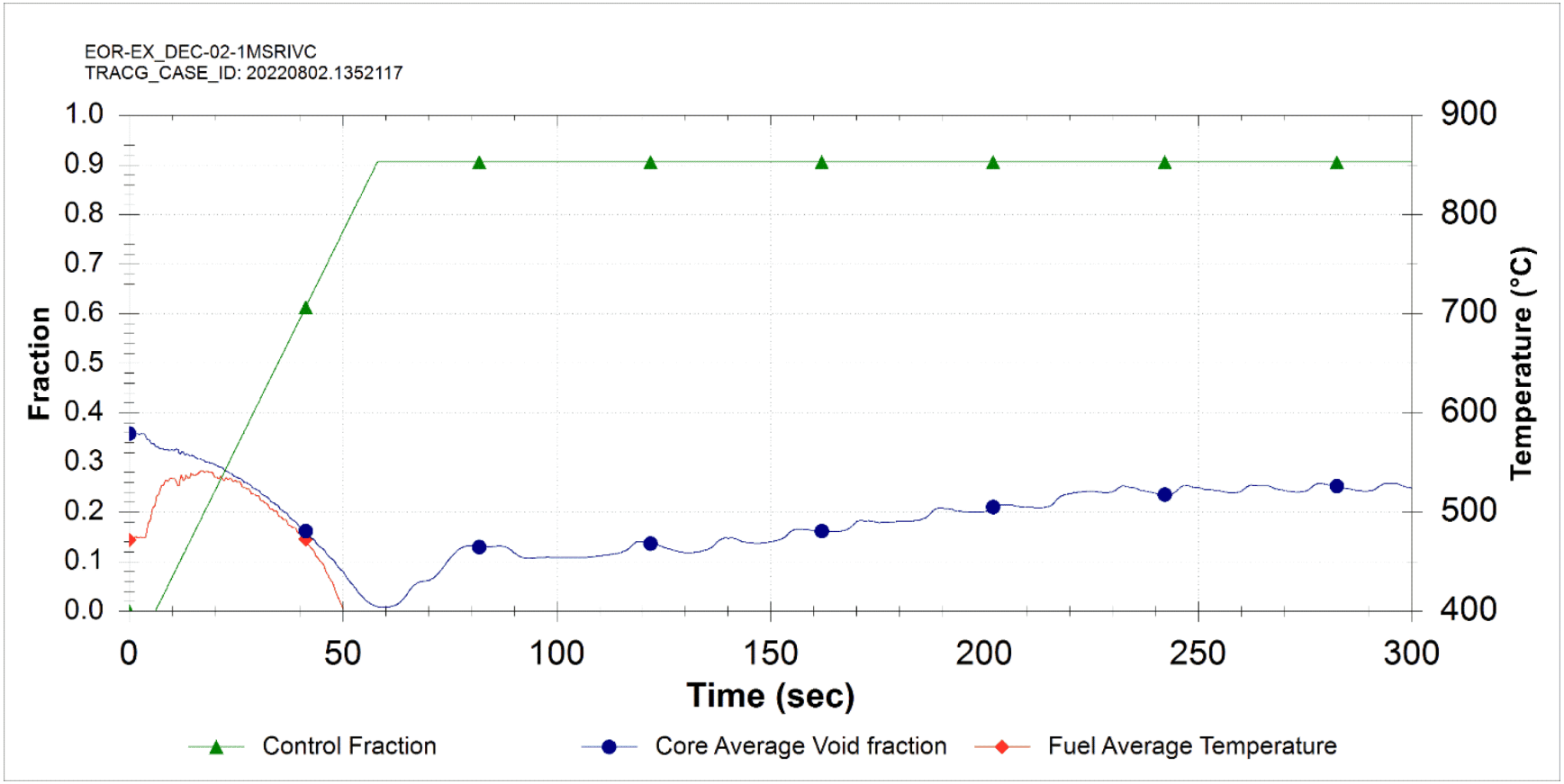

Figure 15.5-123 Closure of One Main Steam Reactor Isolation Valve with Failure to Scram (DEC)

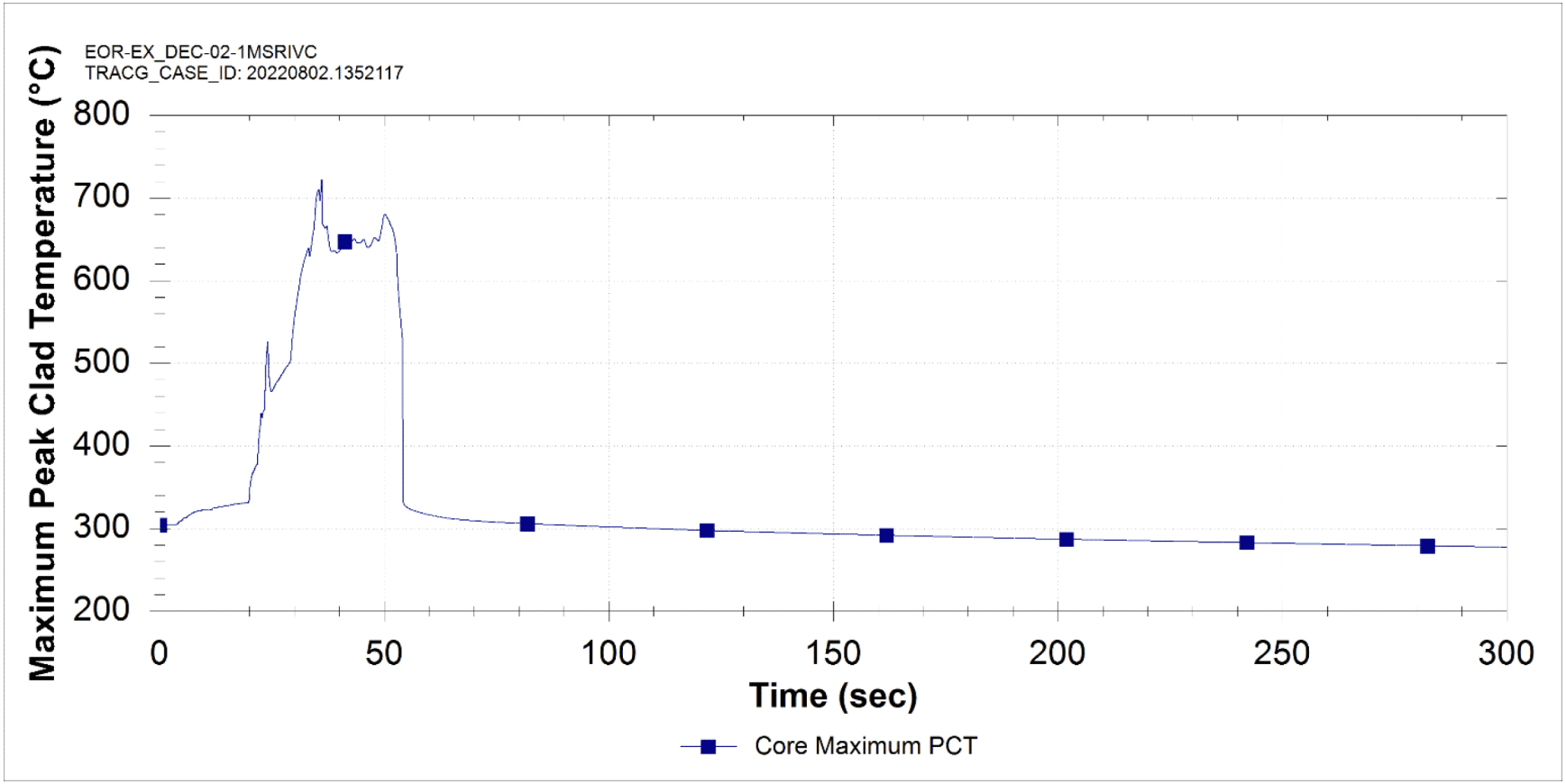

Figure 15.5-124 Closure of One Main Steam Reactor Isolation Valve with Failure to Scram (DEC)

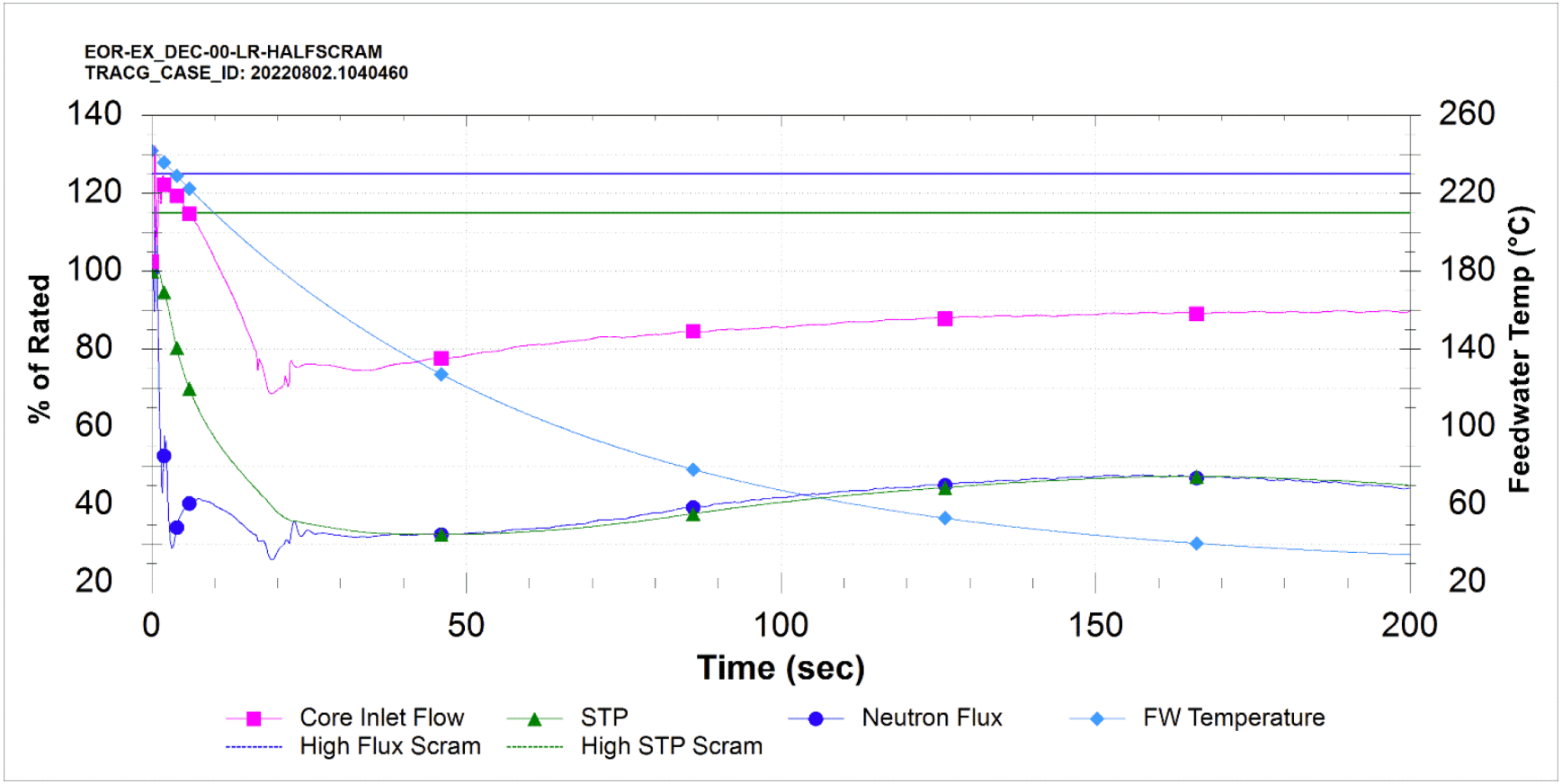

Figure 15.5-125 Complex Sequence Generator Load Rejection (DEC)

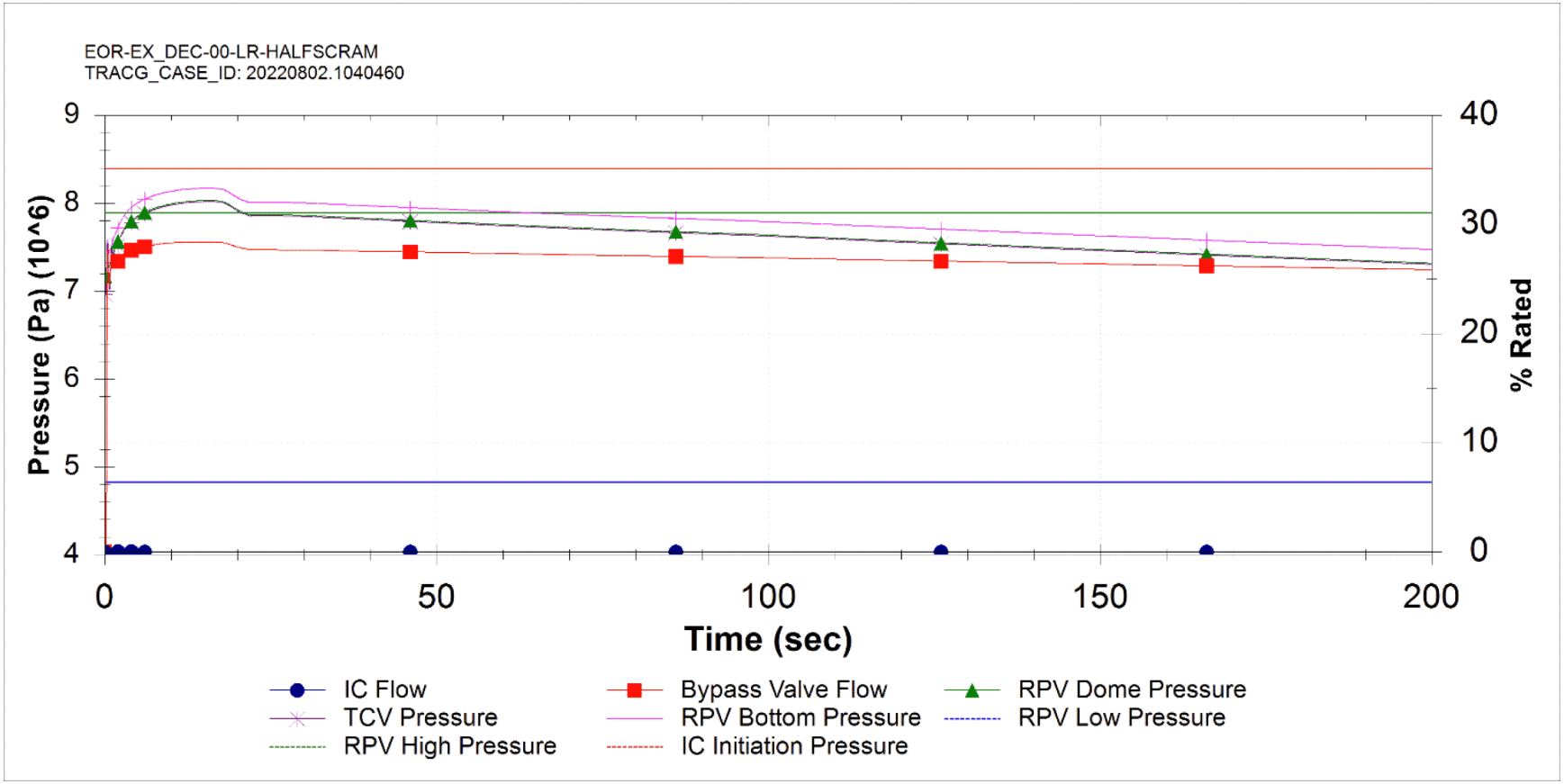

Figure 15.5-126 Complex Sequence Generator Load Rejection (DEC)

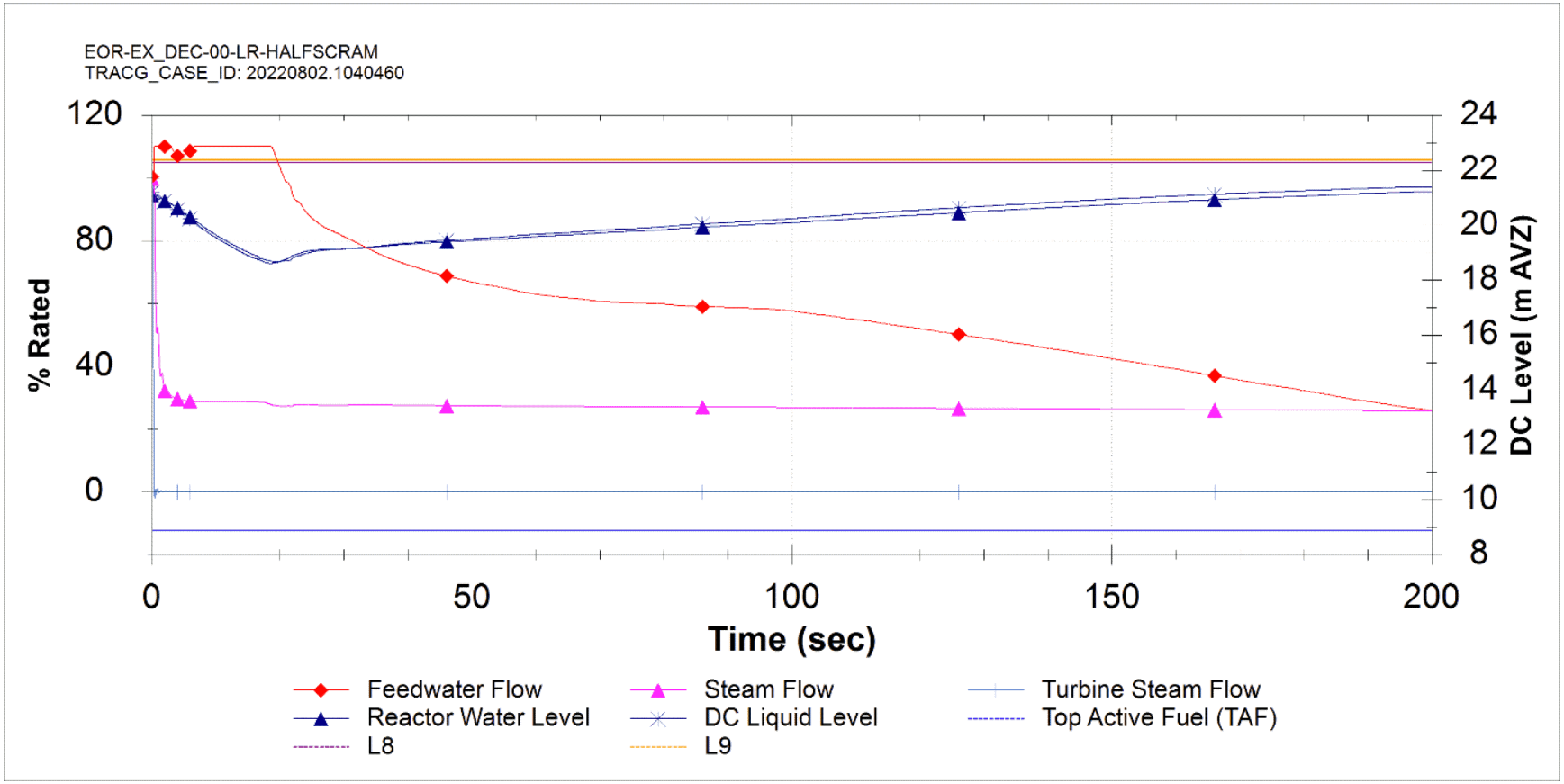

Figure 15.5-127 Complex Sequence Generator Load Rejection (DEC)

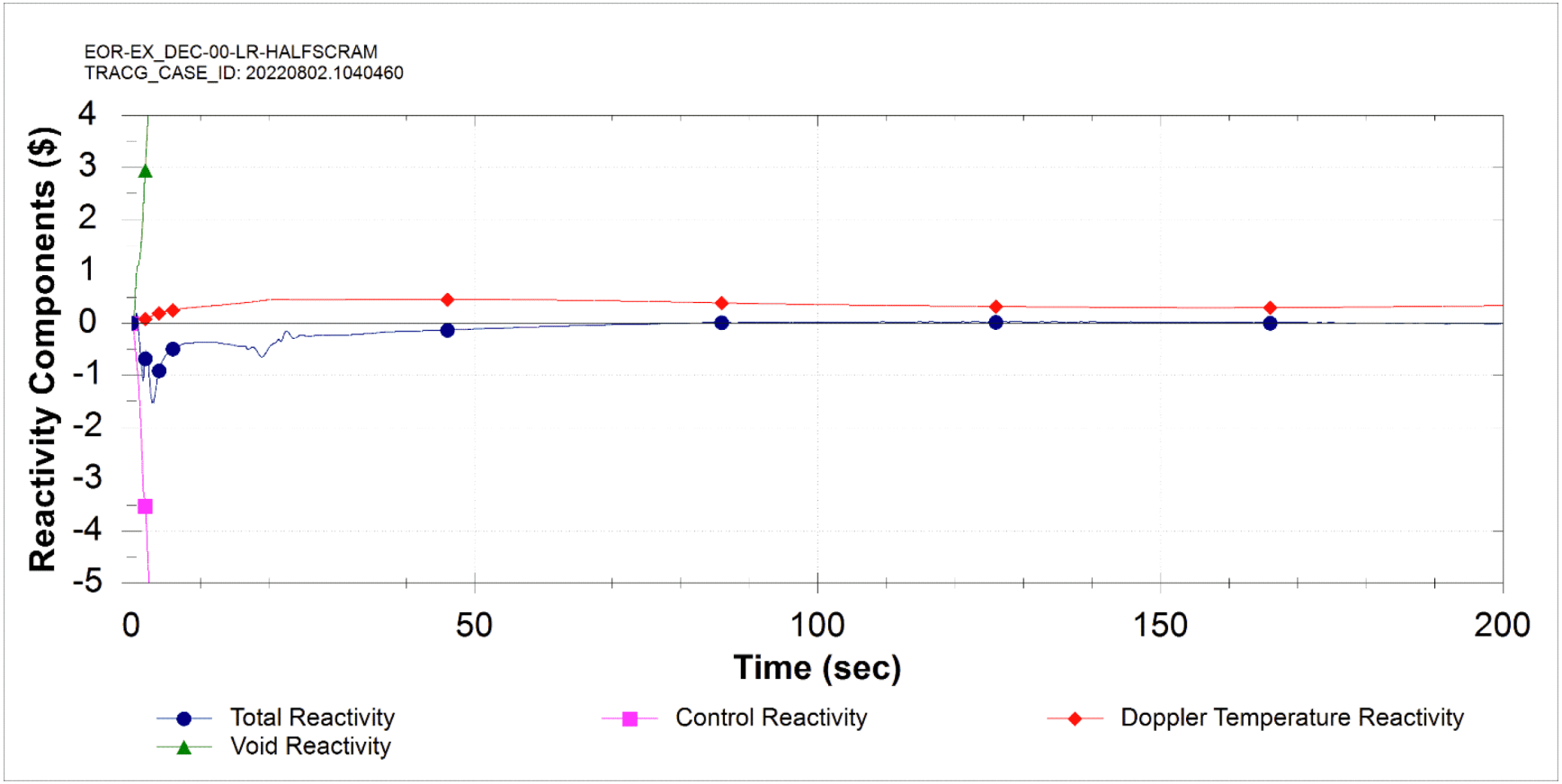

Figure 15.5-128 Complex Sequence Generator Load Rejection (DEC)

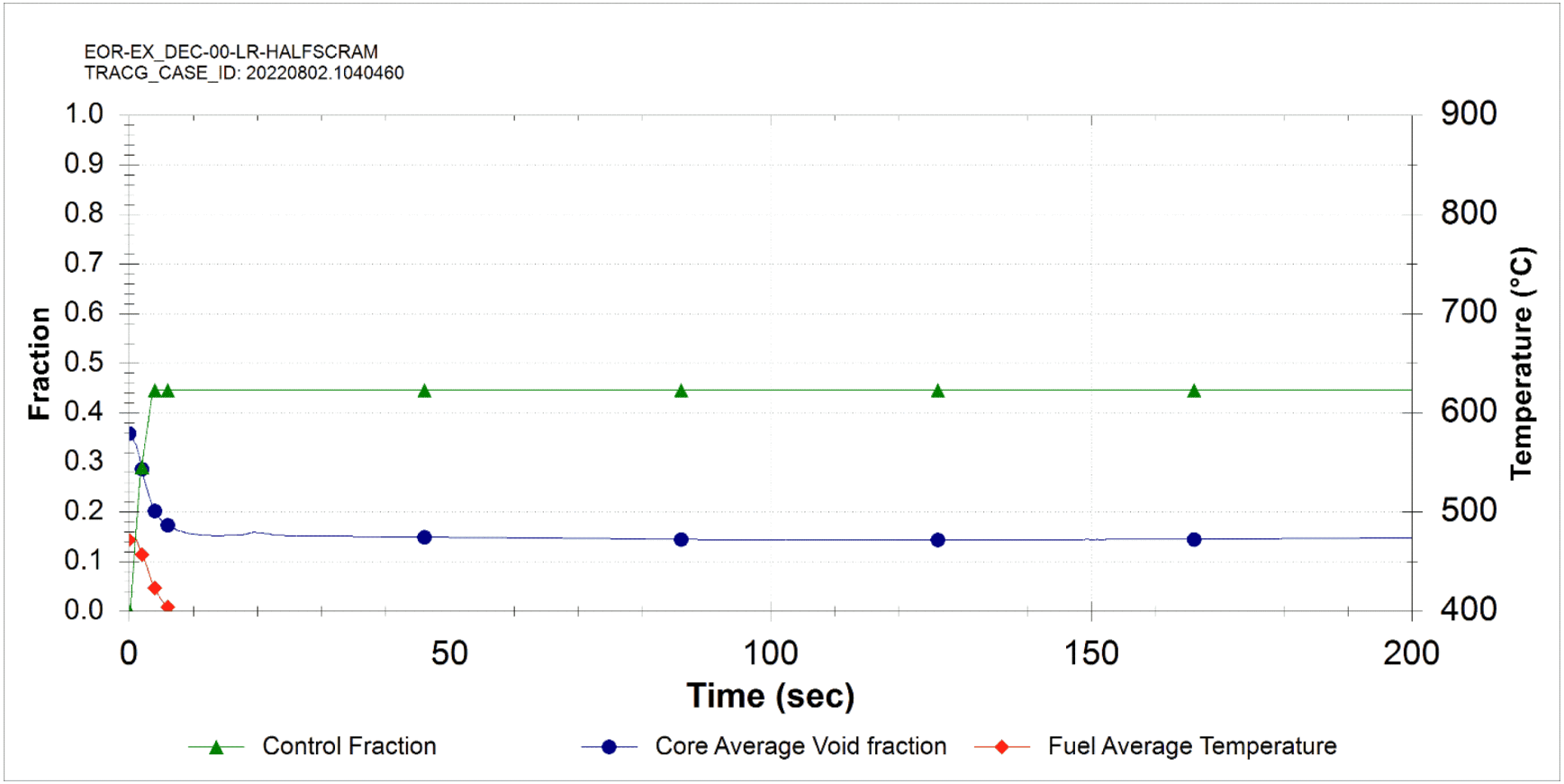

Figure 15.5-129 Complex Sequence Generator Load Rejection (DEC)

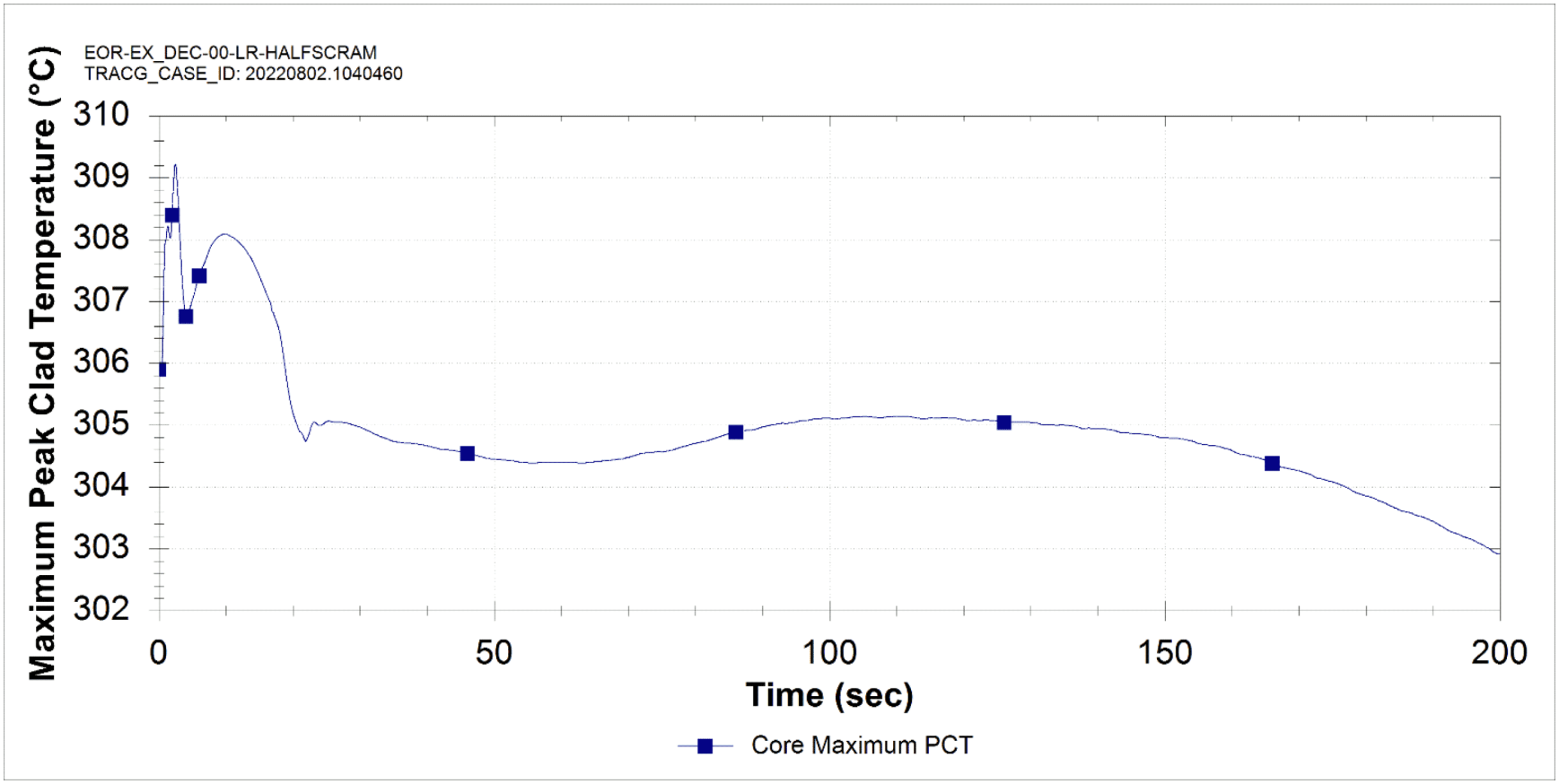

Figure 15.5-130 Complex Sequence Generator Load Rejection (DEC)

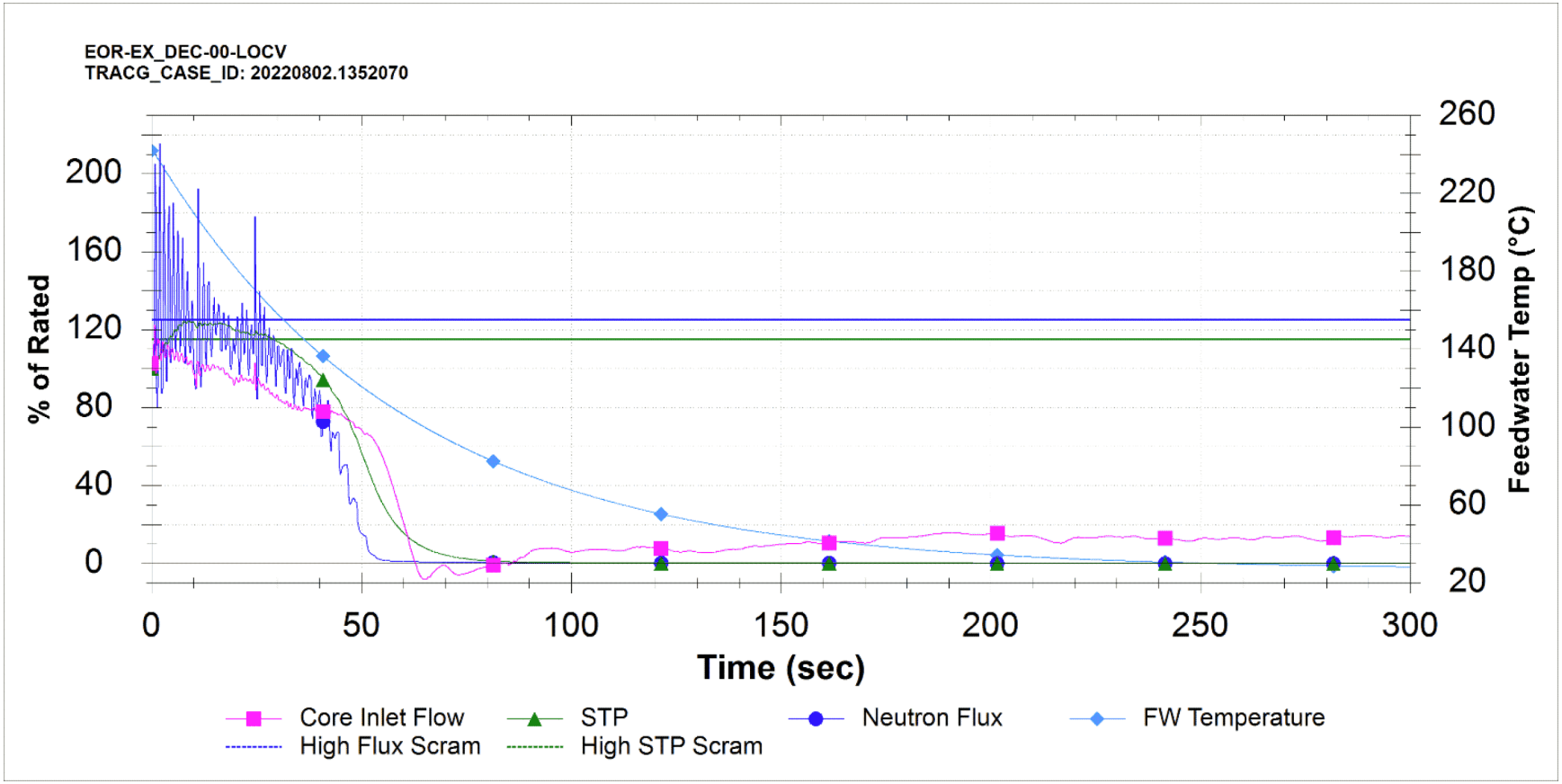

Figure 15.5-131 Loss of Condenser Vacuum (DEC)

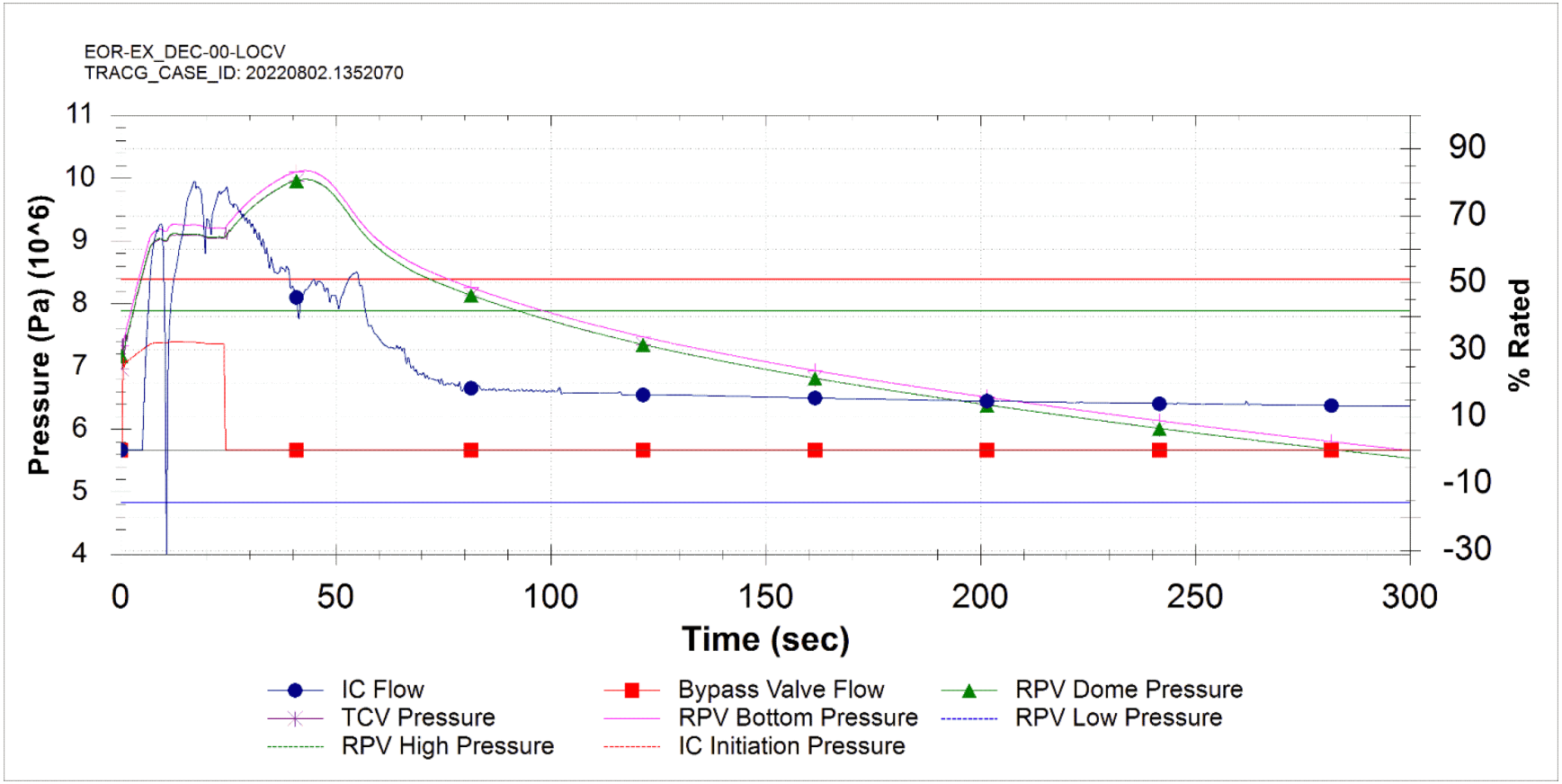

Figure 15.5-132 Loss of Condenser Vacuum (DEC)

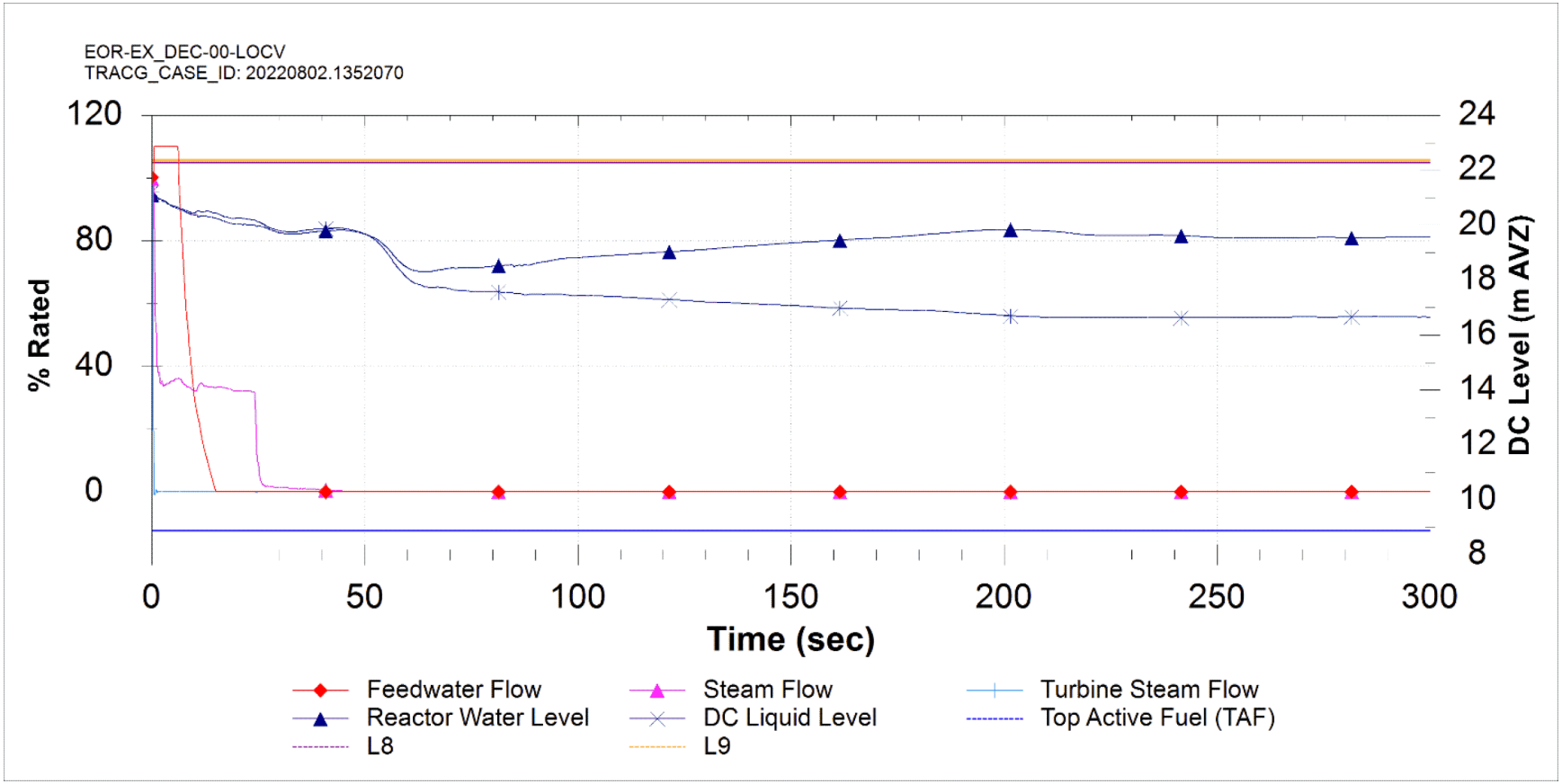

Figure 15.5-133 Loss of Condenser Vacuum (DEC)

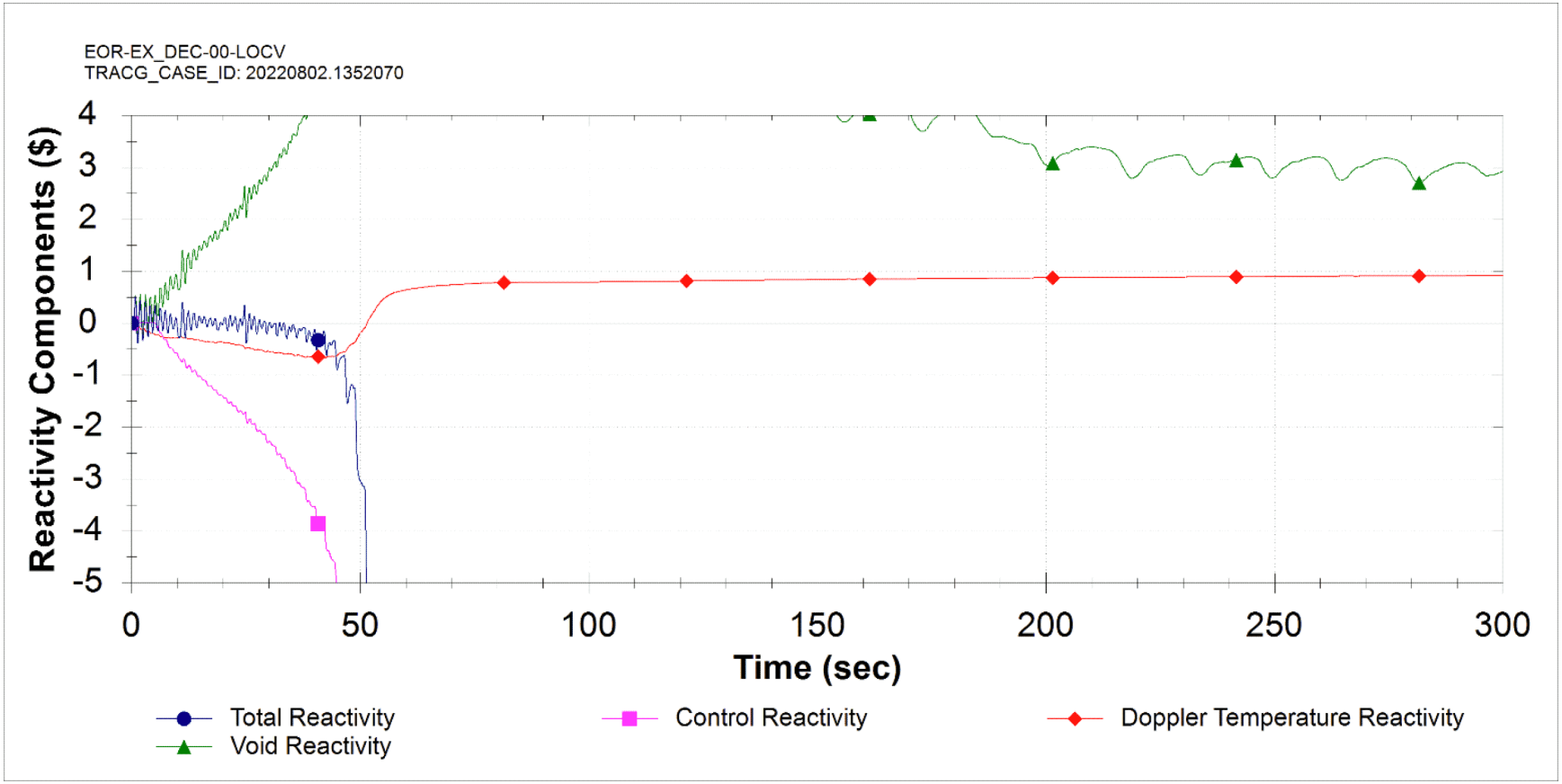

Figure 15.5-134 Loss of Condenser Vacuum (DEC)

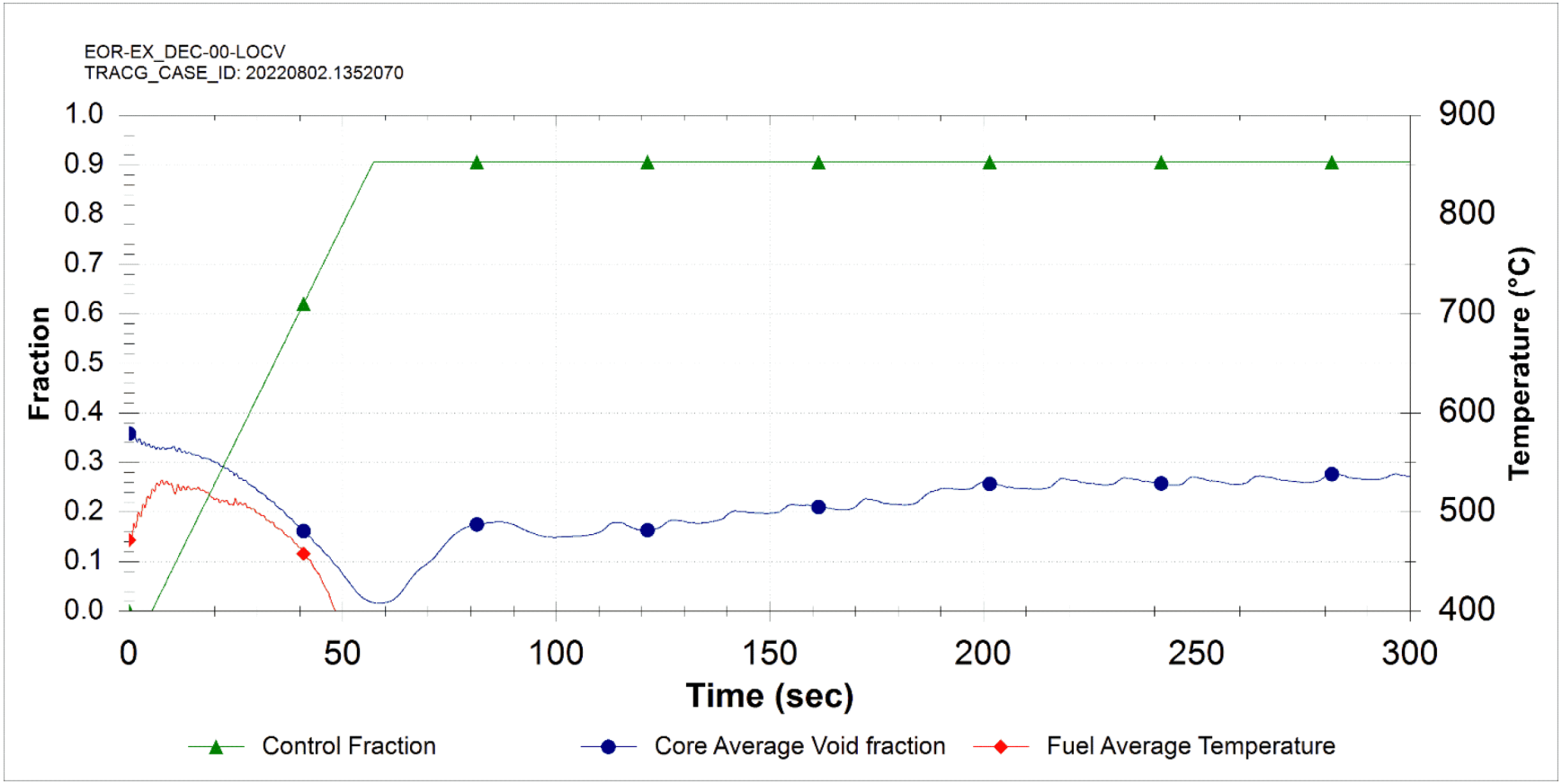

Figure 15.5-135 Loss of Condenser Vacuum (DEC)

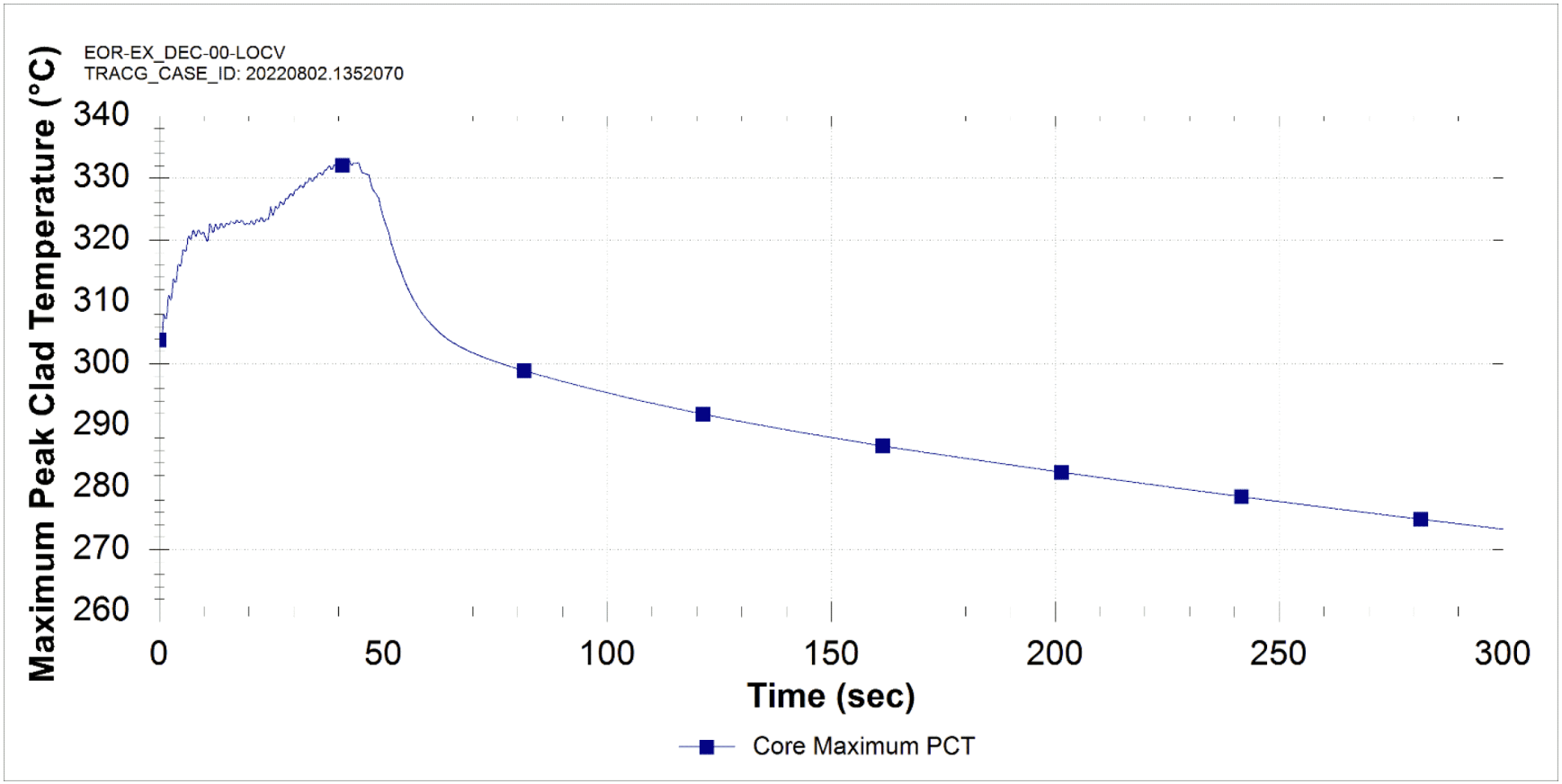

Figure 15.5-136 Loss of Condenser Vacuum (DEC)

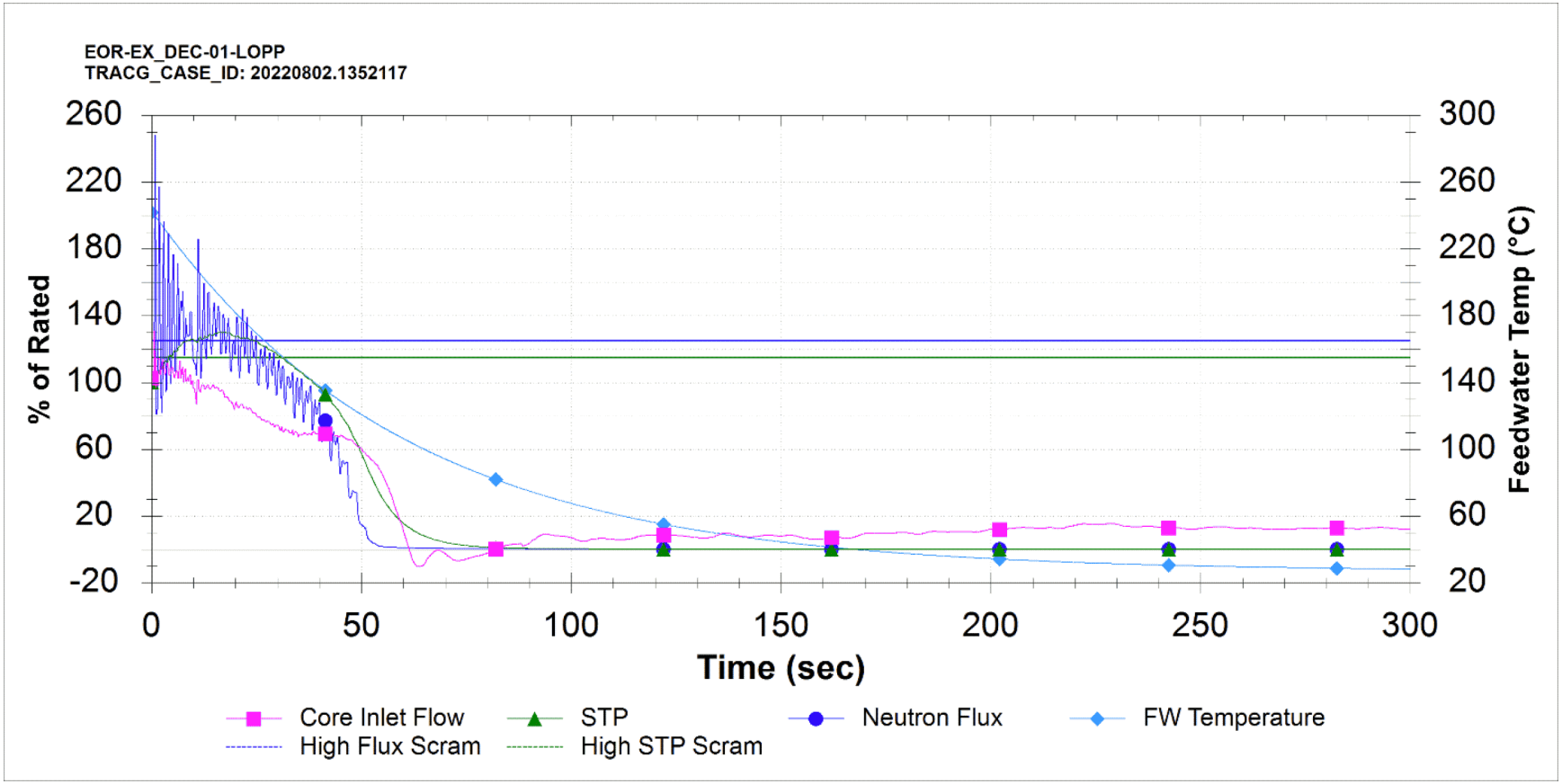

Figure 15.5-137 Loss of Preferred Power (DEC)

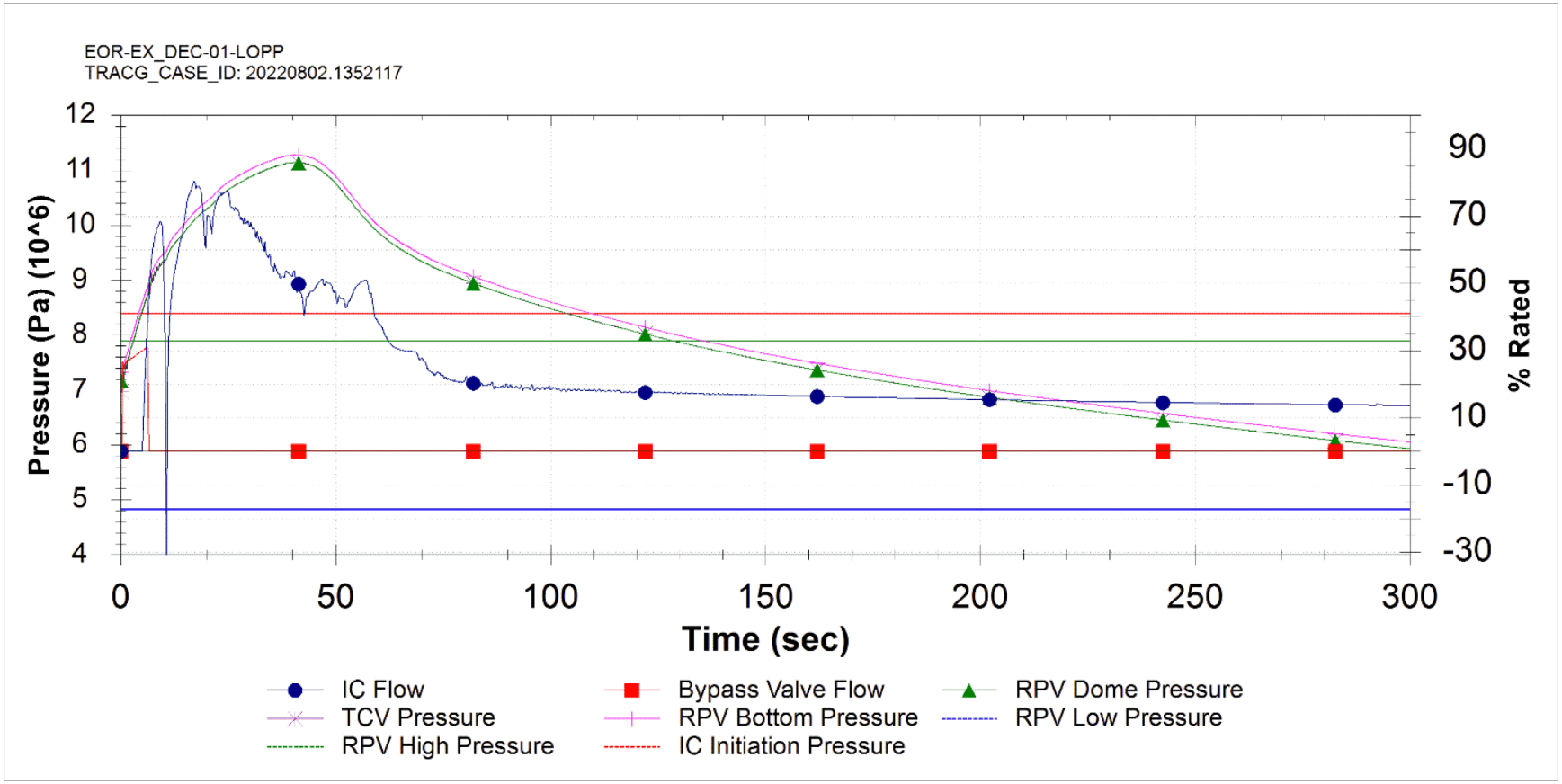

Figure 15.5-138 Loss of Preferred Power (DEC)

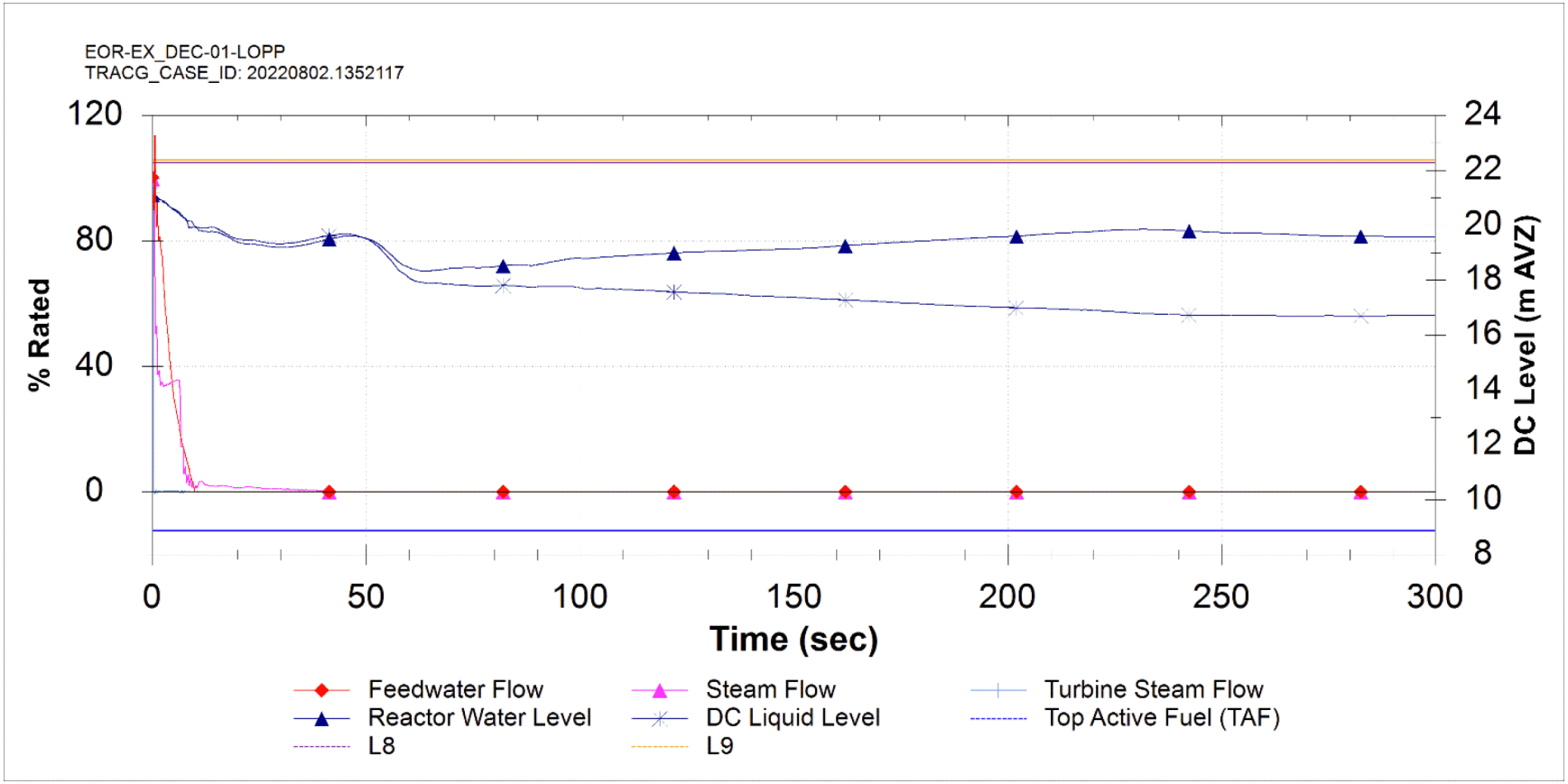

Figure 15.5-139 Loss of Preferred Power (DEC)

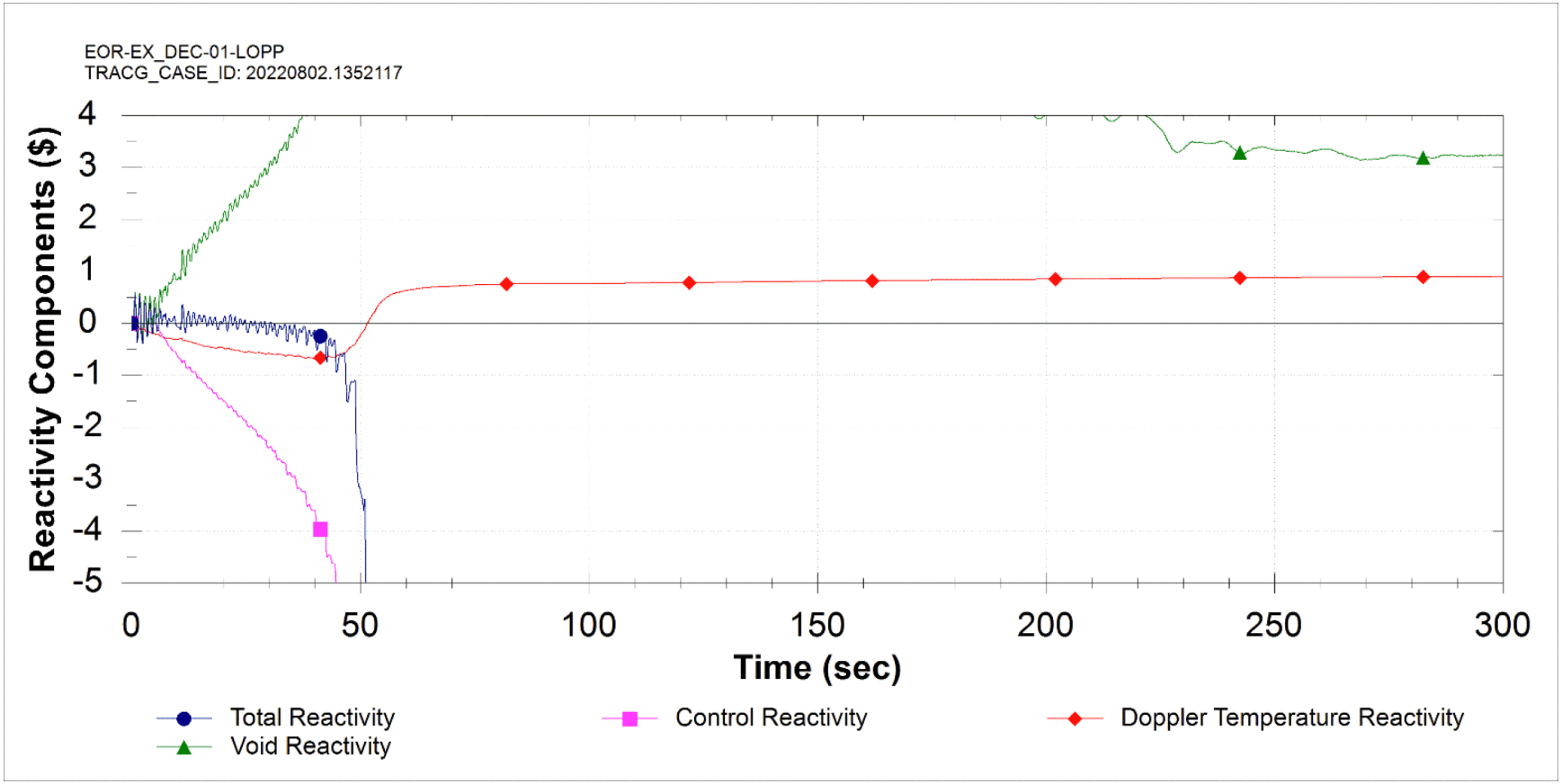

Figure 15.5-140 Loss of Preferred Power (DEC)

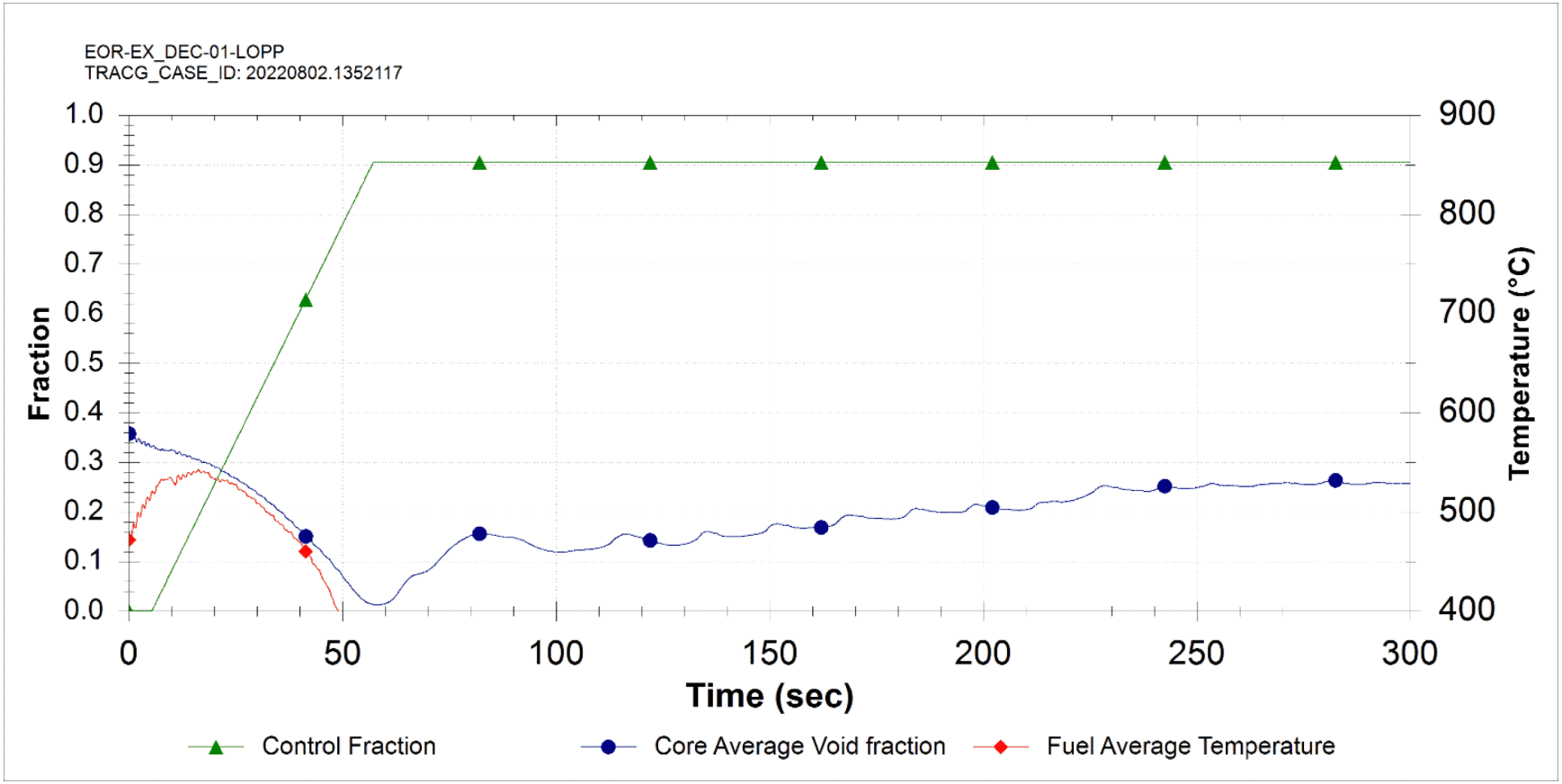

Figure 15.5-141 Loss of Preferred Power (DEC)

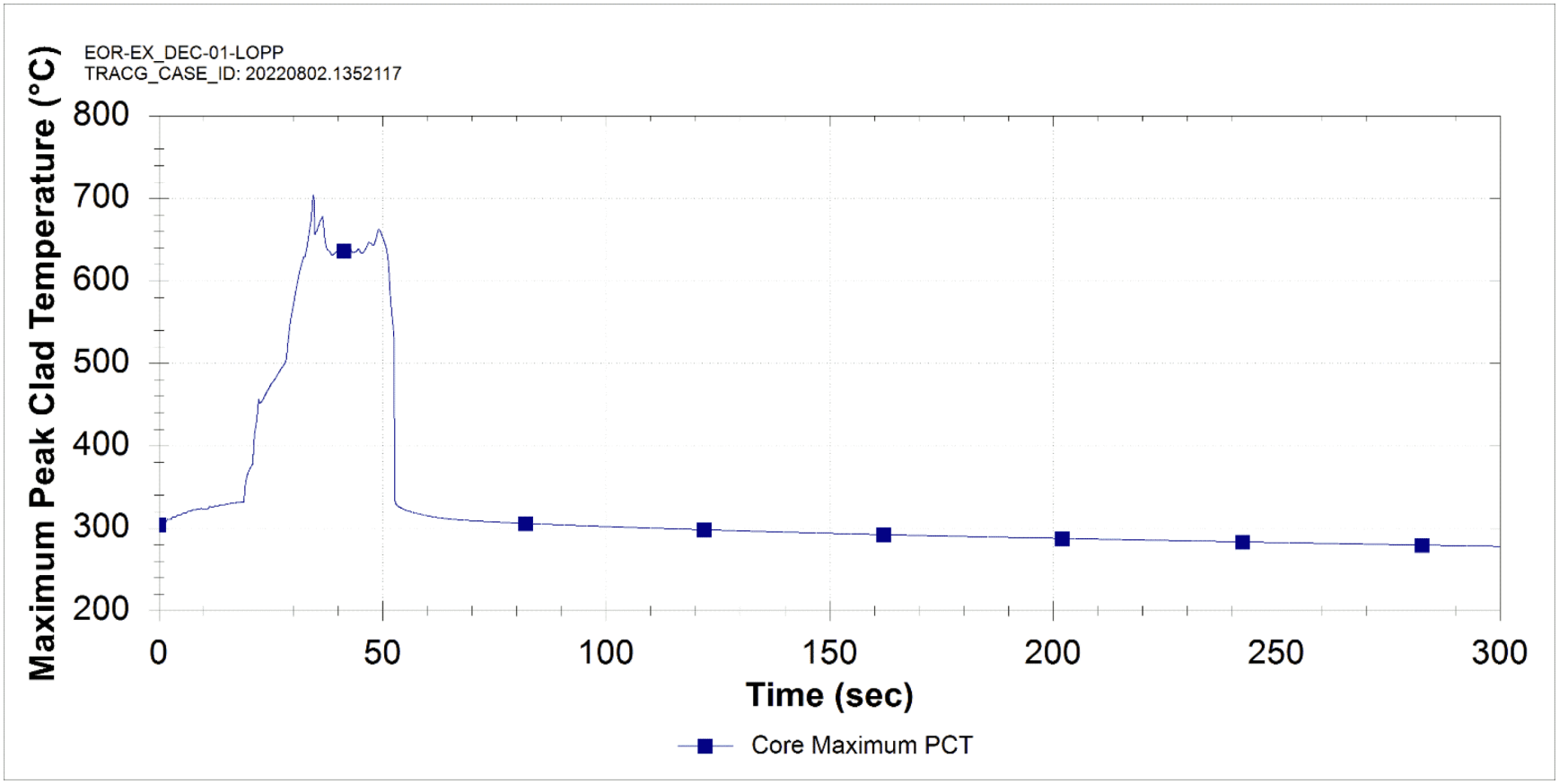

Figure 15.5-142 Loss of Preferred Power (DEC)

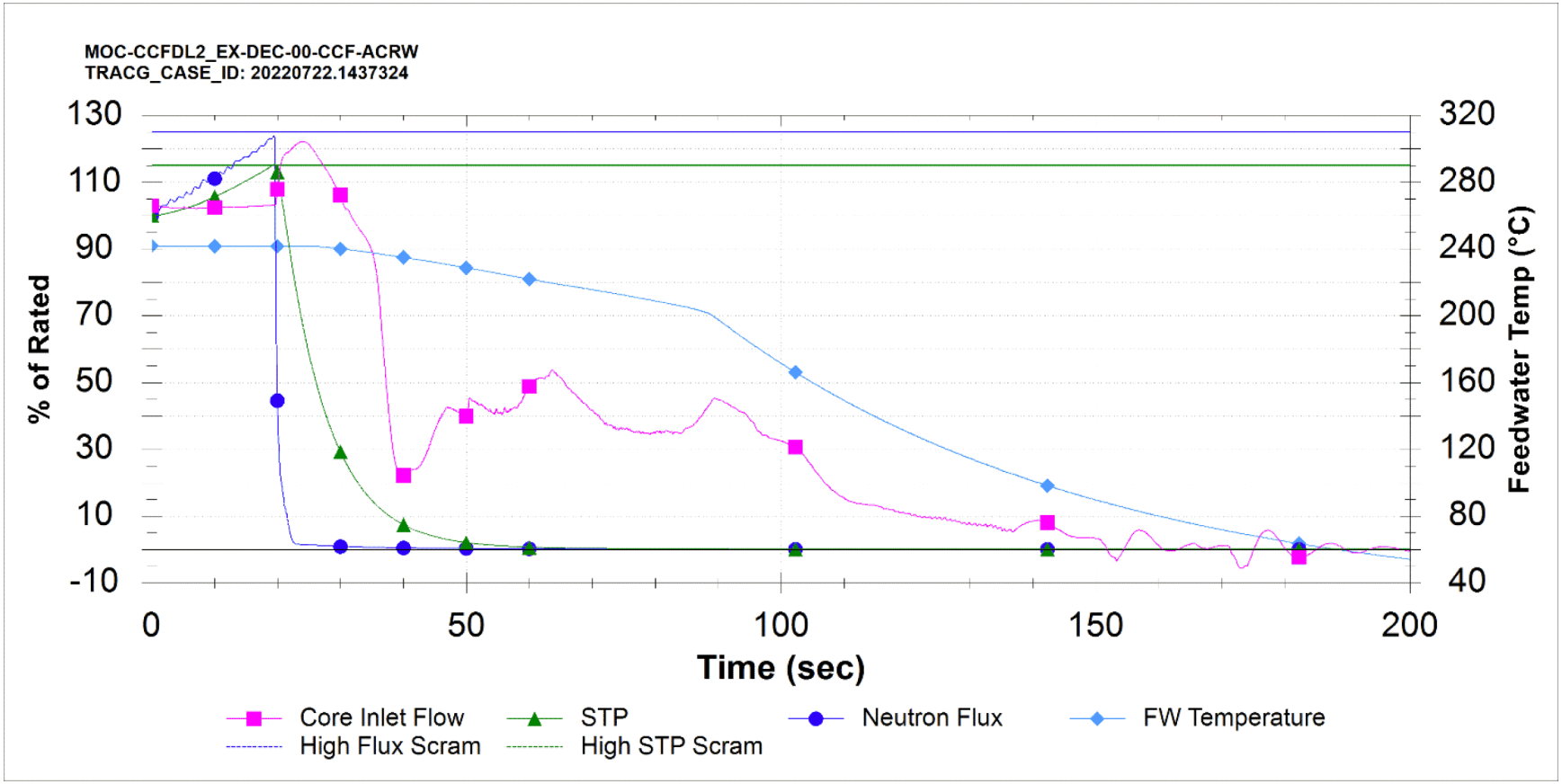

Figure 15.5-143 All Control Rod Withdrawal at Power (ACRW)

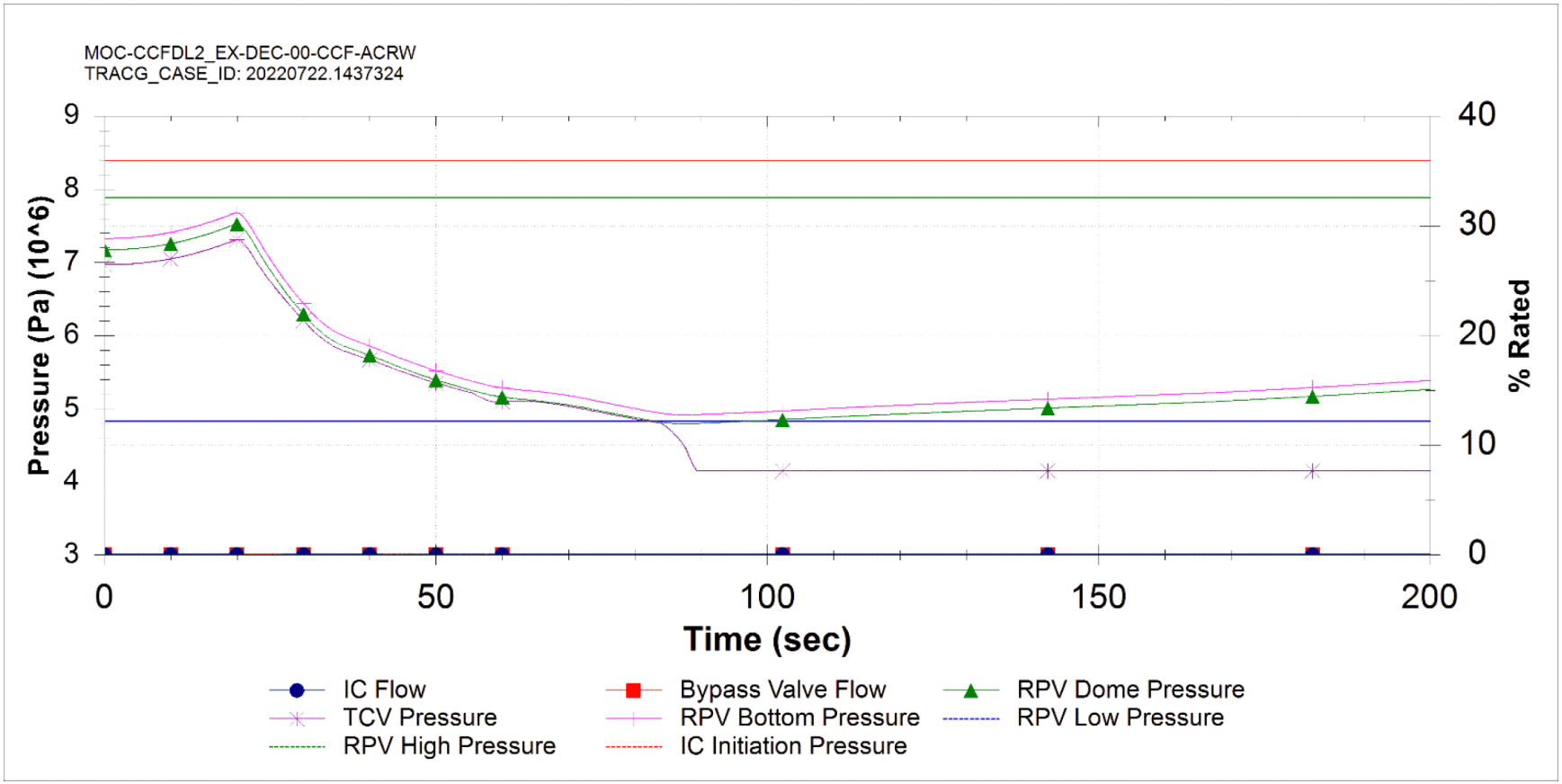

Figure 15.5-144 All Control Rod Withdrawal at Power (ACRW)

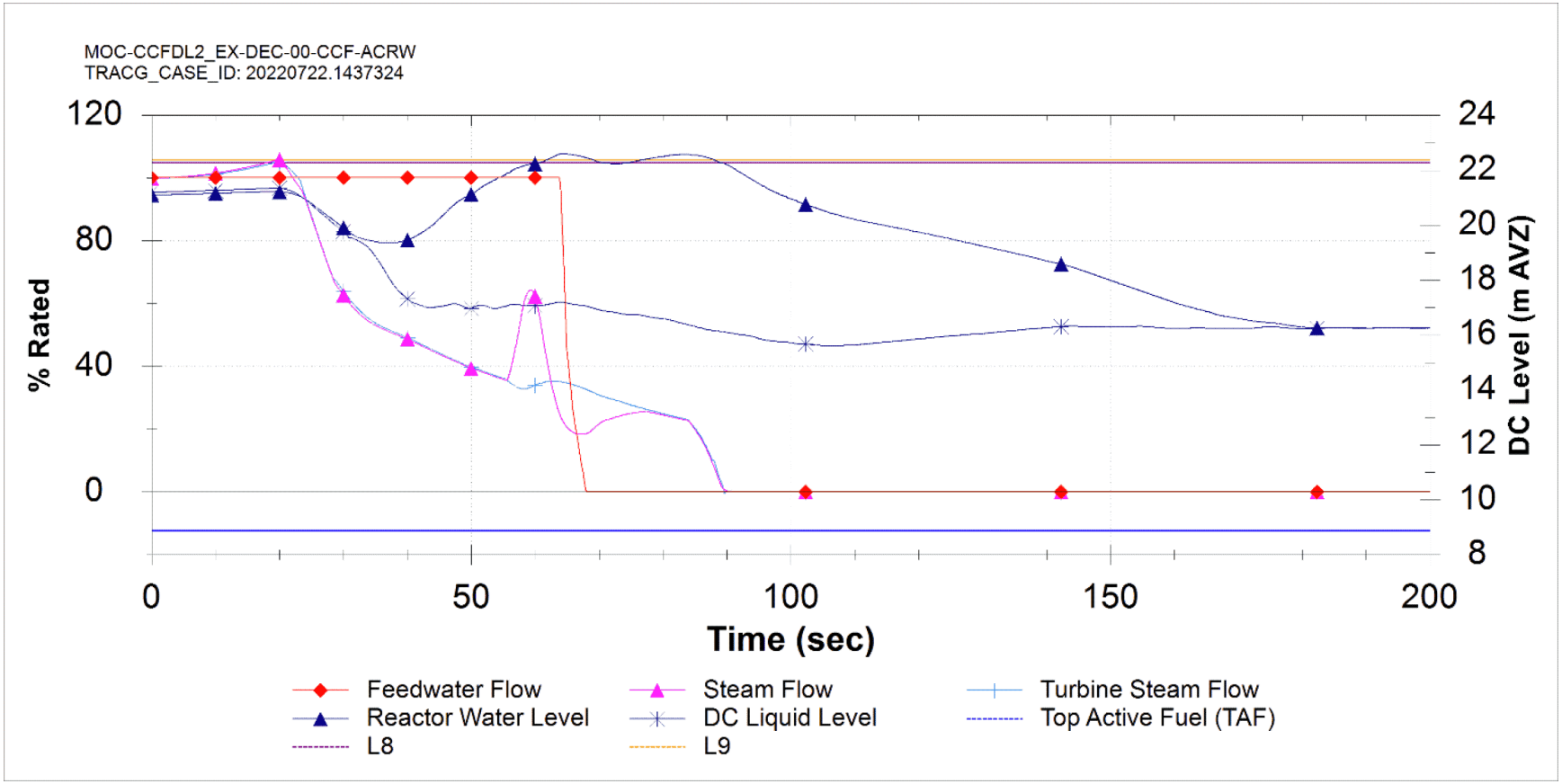

Figure 15.5-145 All Control Rod Withdrawal at Power (ACRW)

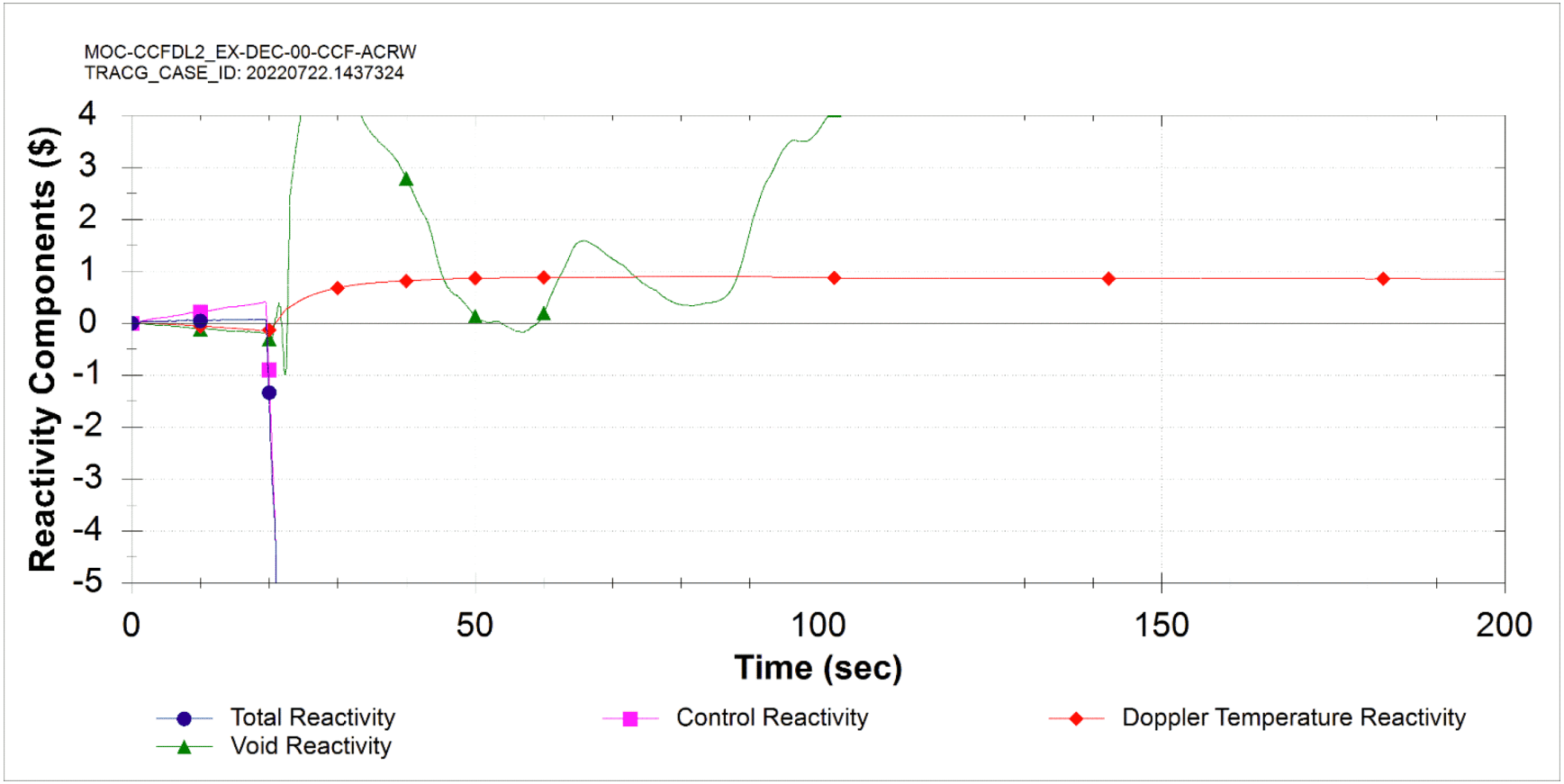

Figure 15.5-146 All Control Rod Withdrawal at Power (ACRW)

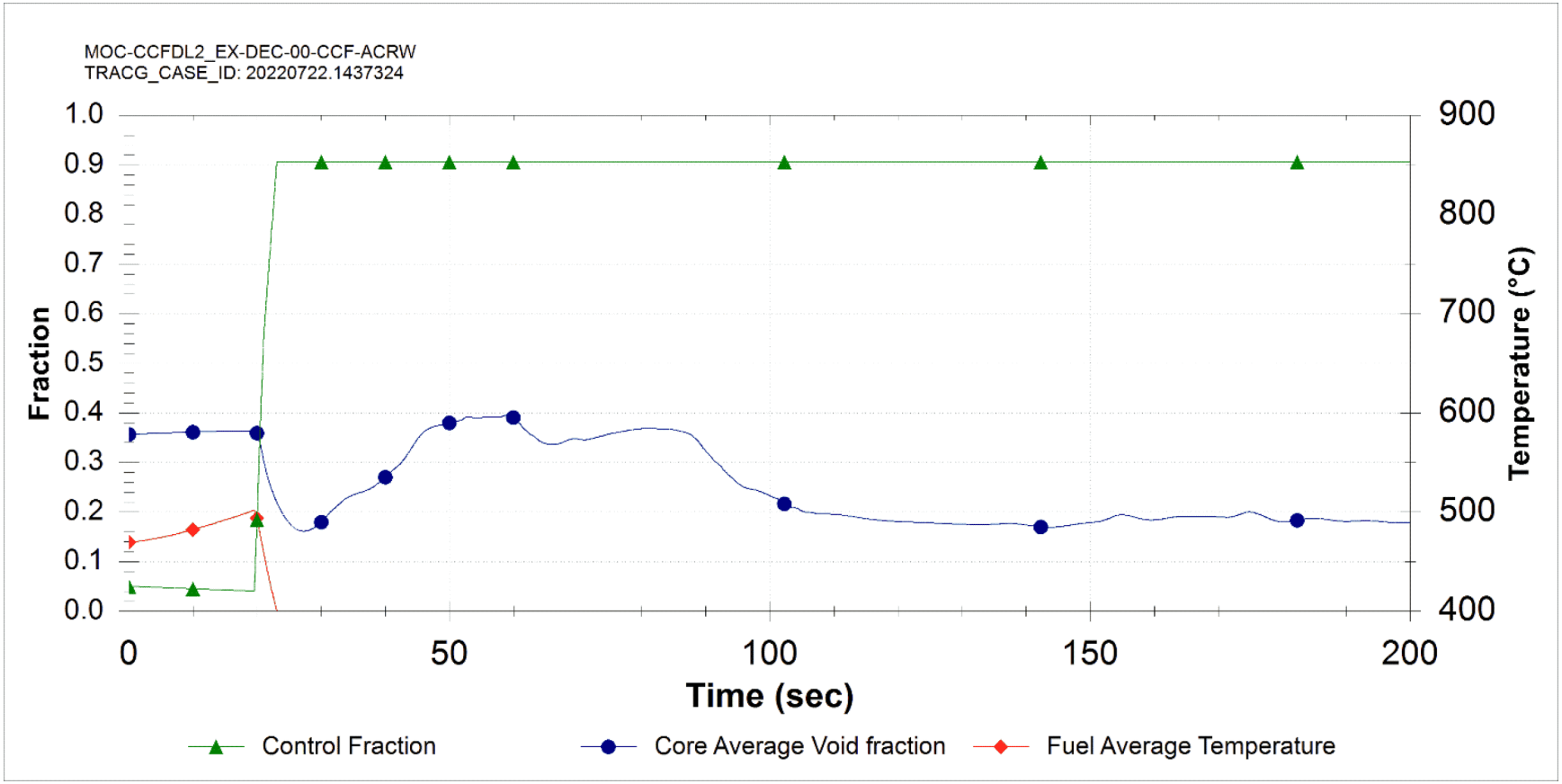

Figure 15.5-147 All Control Rod Withdrawal at Power (ACRW)

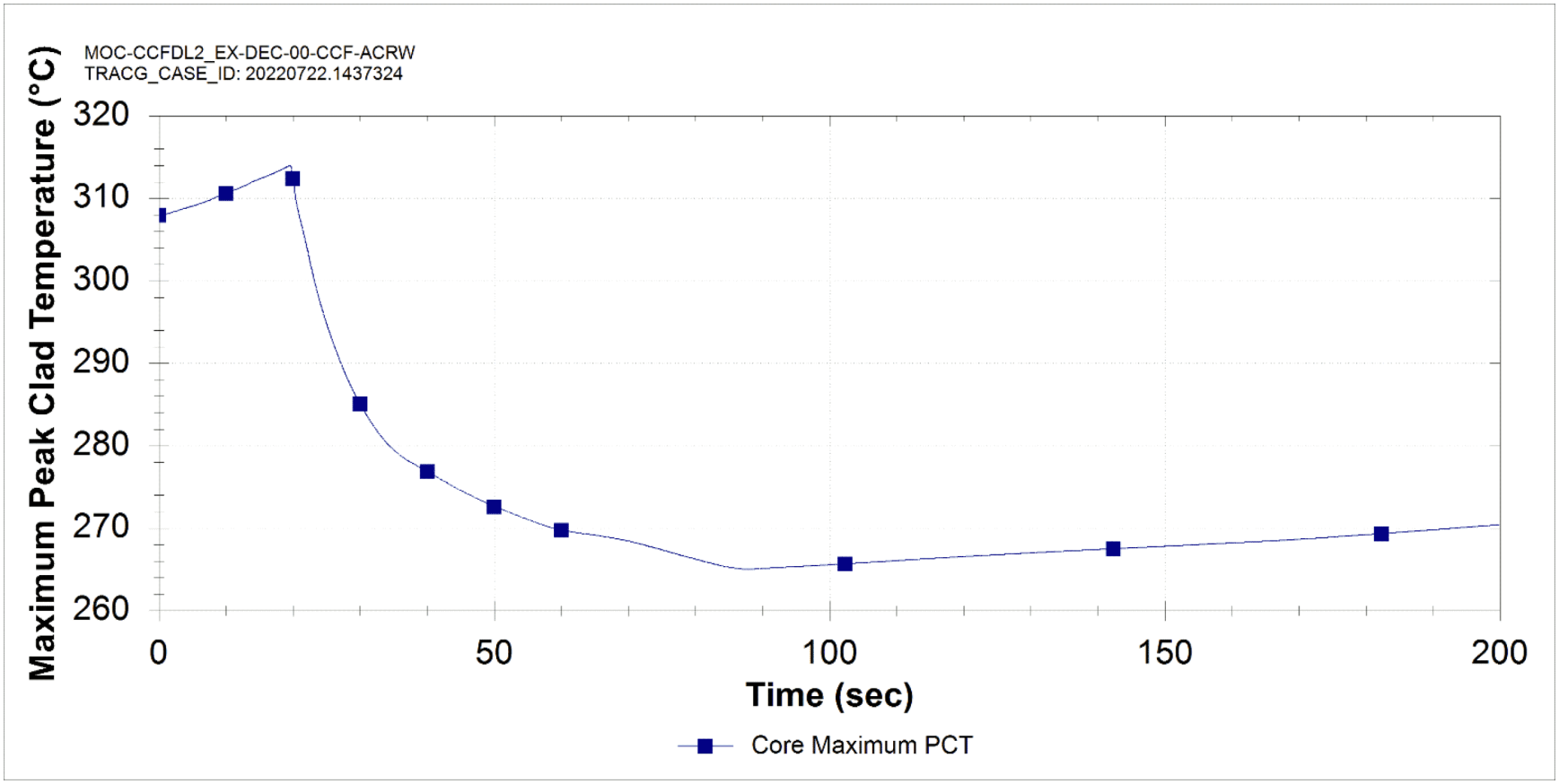

Figure 15.5-148 All Control Rod Withdrawal at Power (ACRW)

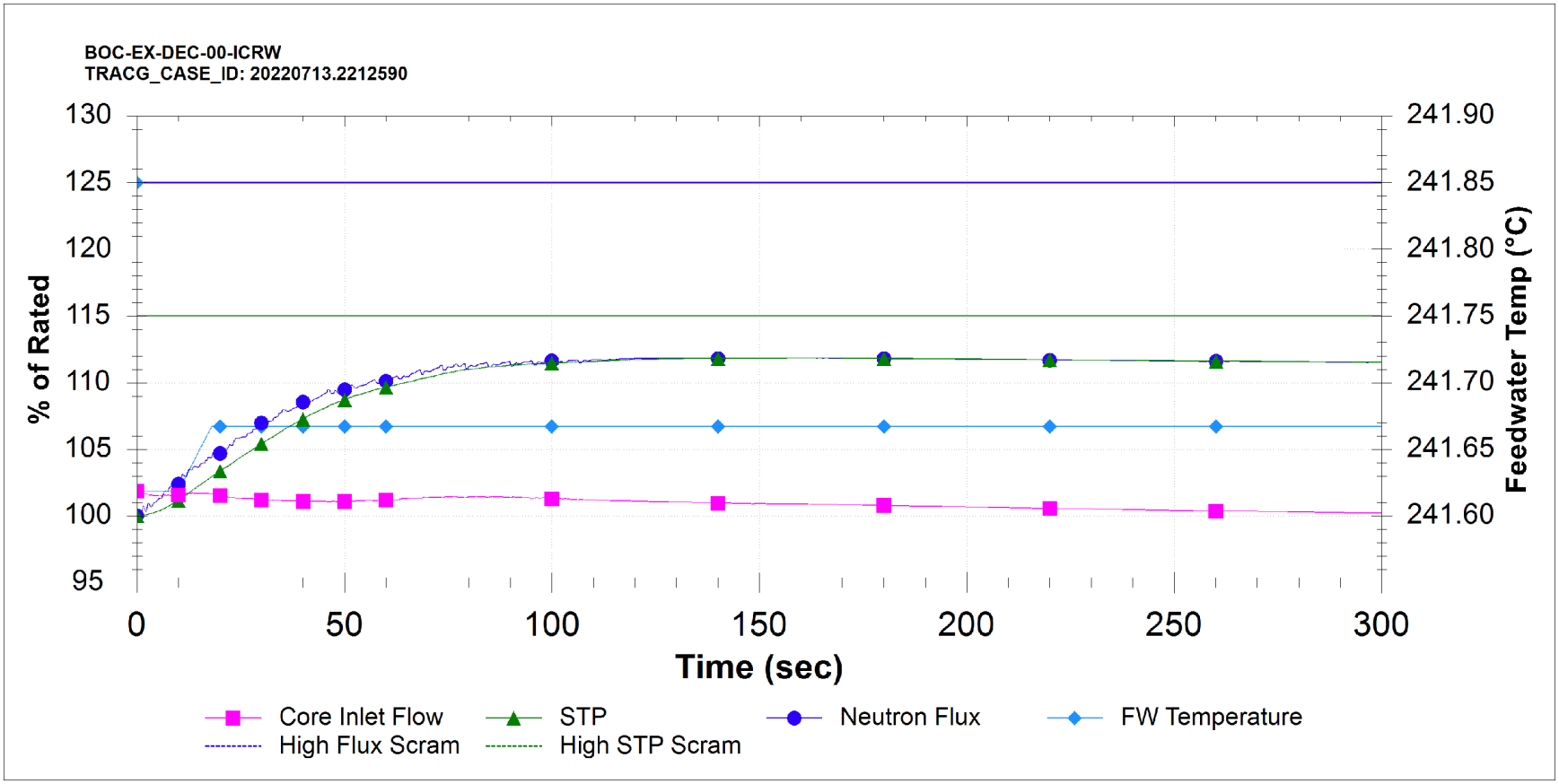

Figure 15.5-149 Inadvertent Control Rod Withdrawal at Power – Single Rod (ICRW)

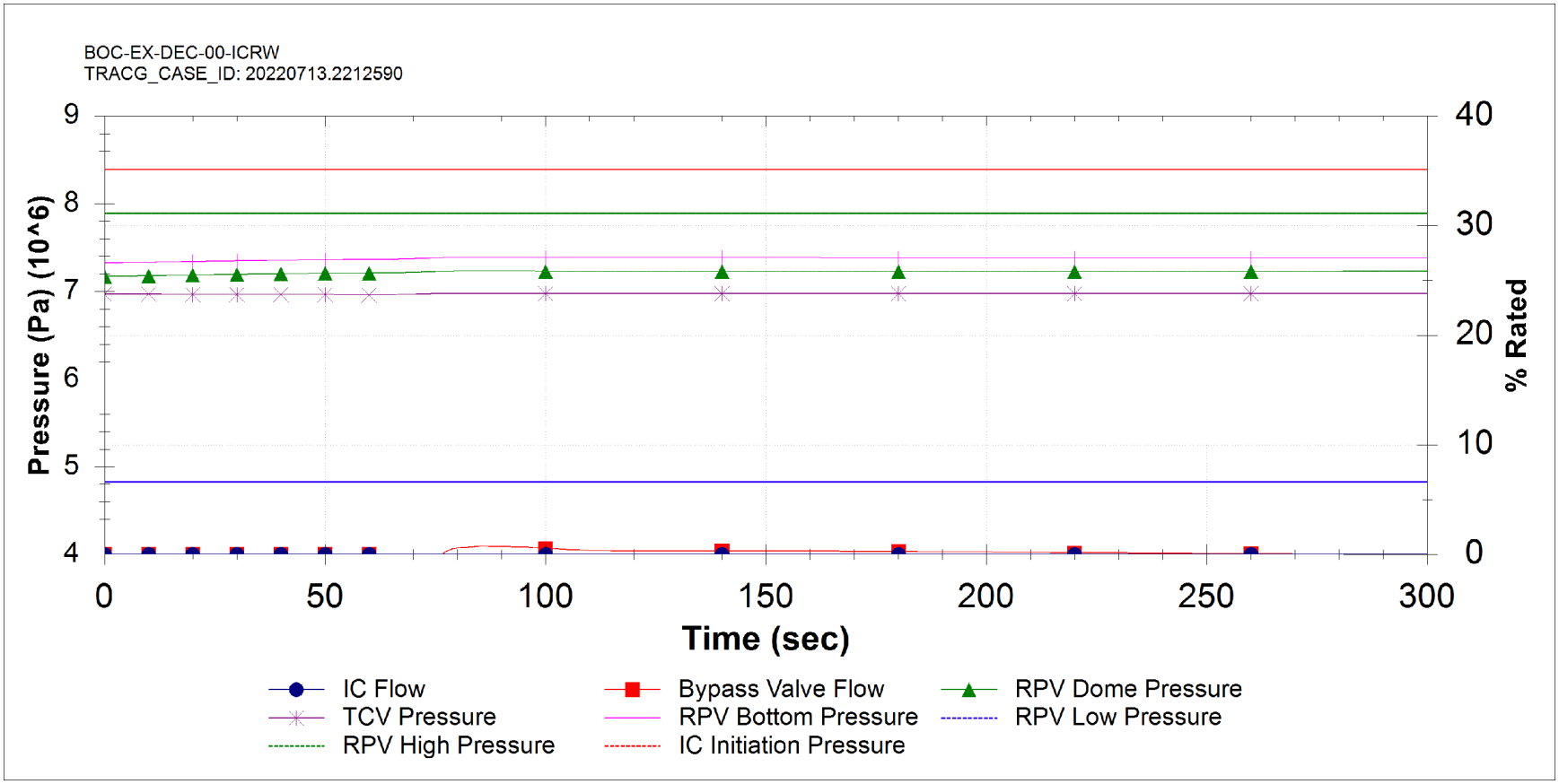

Figure 15.5-150 Inadvertent Control Rod Withdrawal at Power – Single Rod (ICRW)

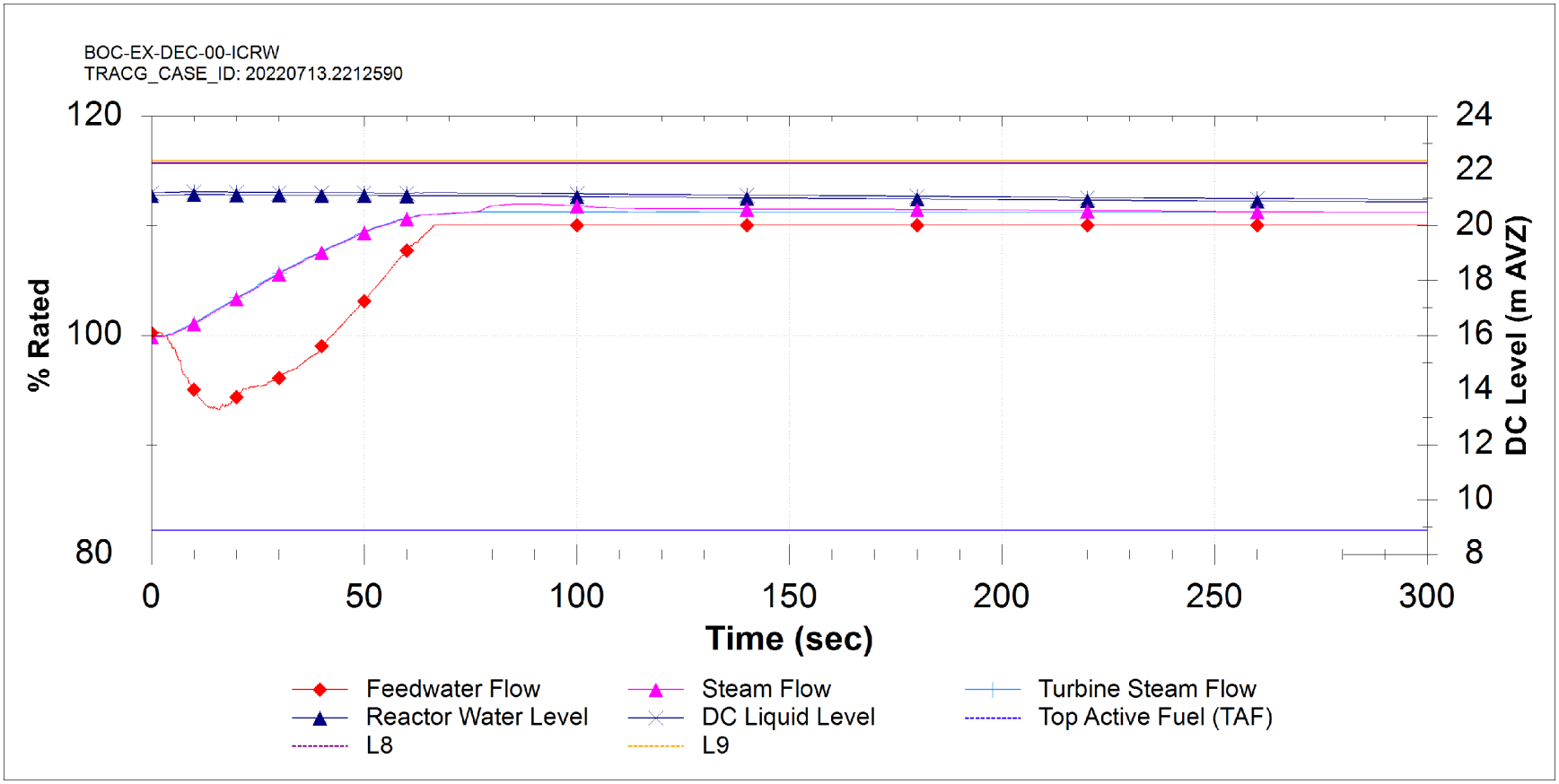

Figure 15.5-151 Inadvertent Control Rod Withdrawal at Power – Single Rod (ICRW)

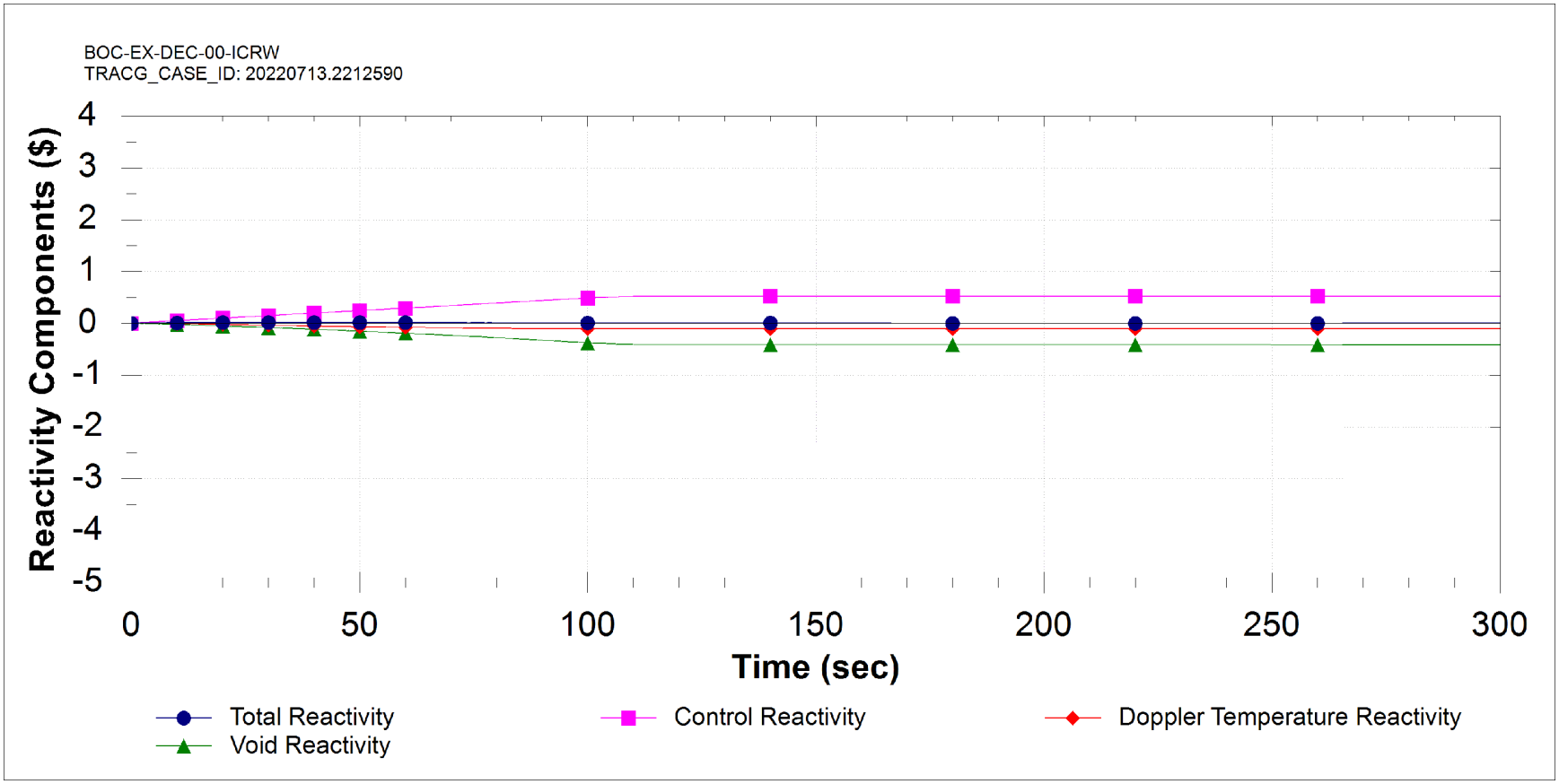

Figure 15.5-152 Inadvertent Control Rod Withdrawal at Power – Single Rod (ICRW)

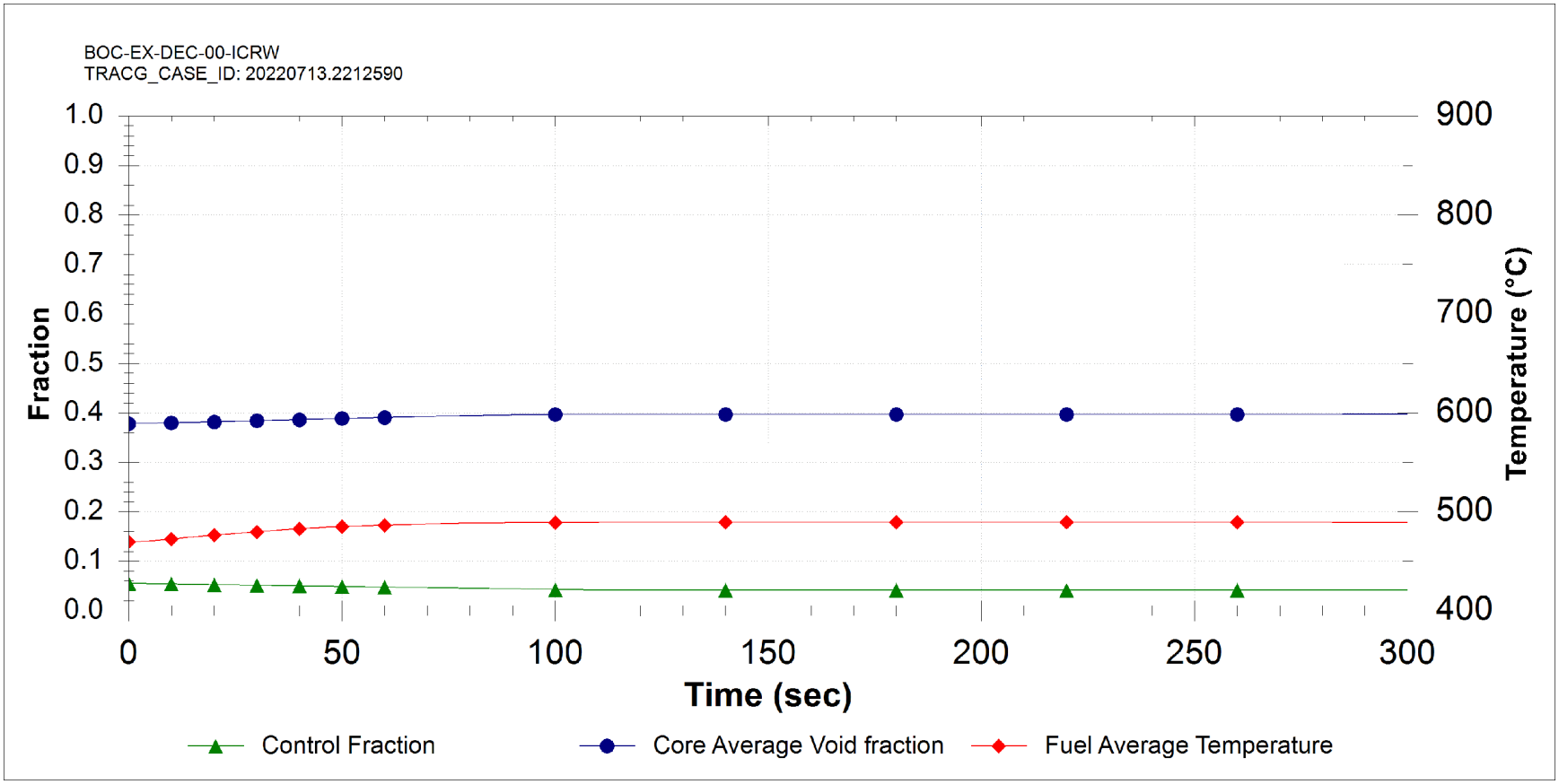

Figure 15.5-153 Inadvertent Control Rod Withdrawal at Power – Single Rod (ICRW)

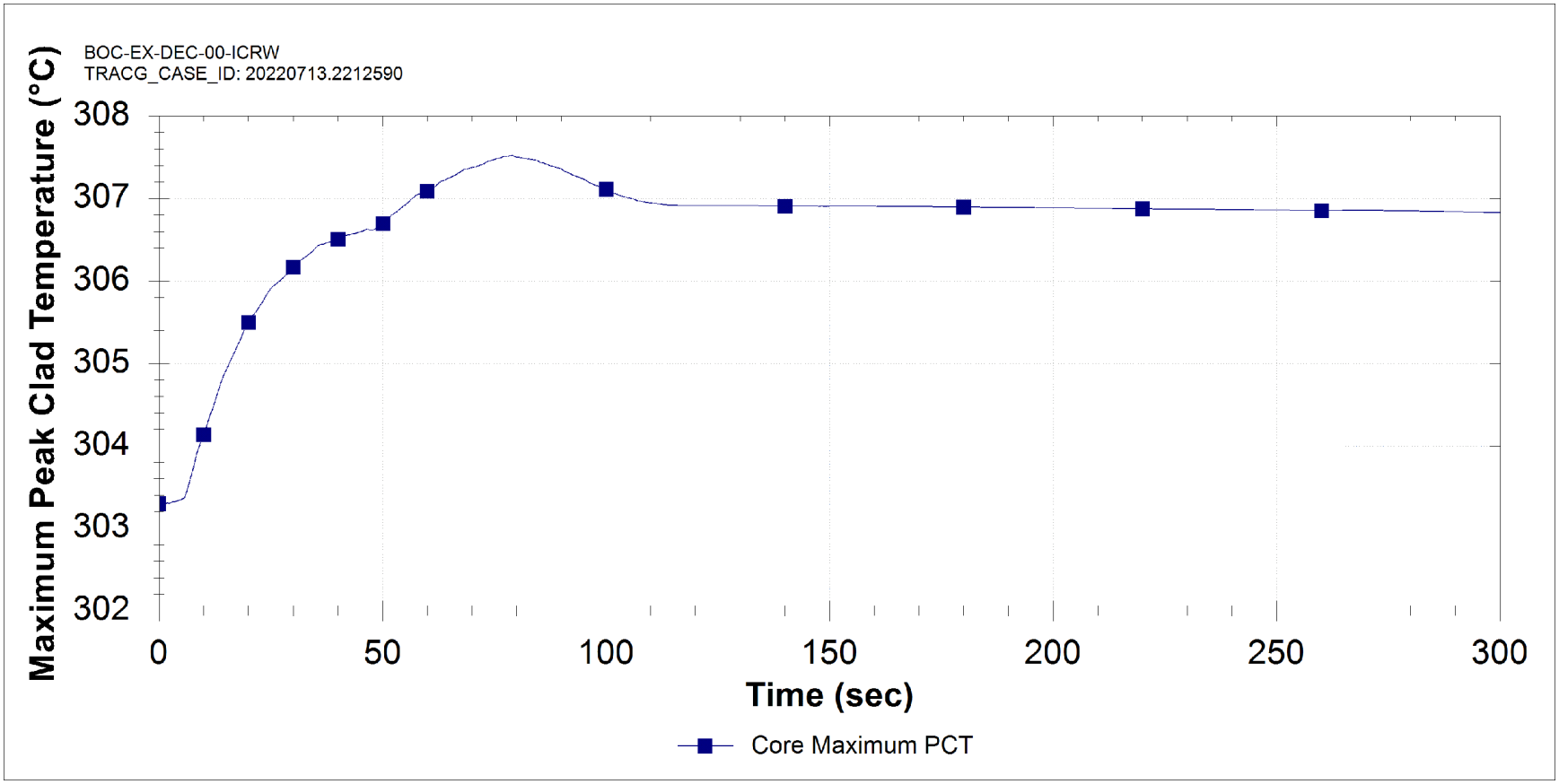

Figure 15.5-154 Inadvertent Control Rod Withdrawal at Power – Single Rod (ICRW)

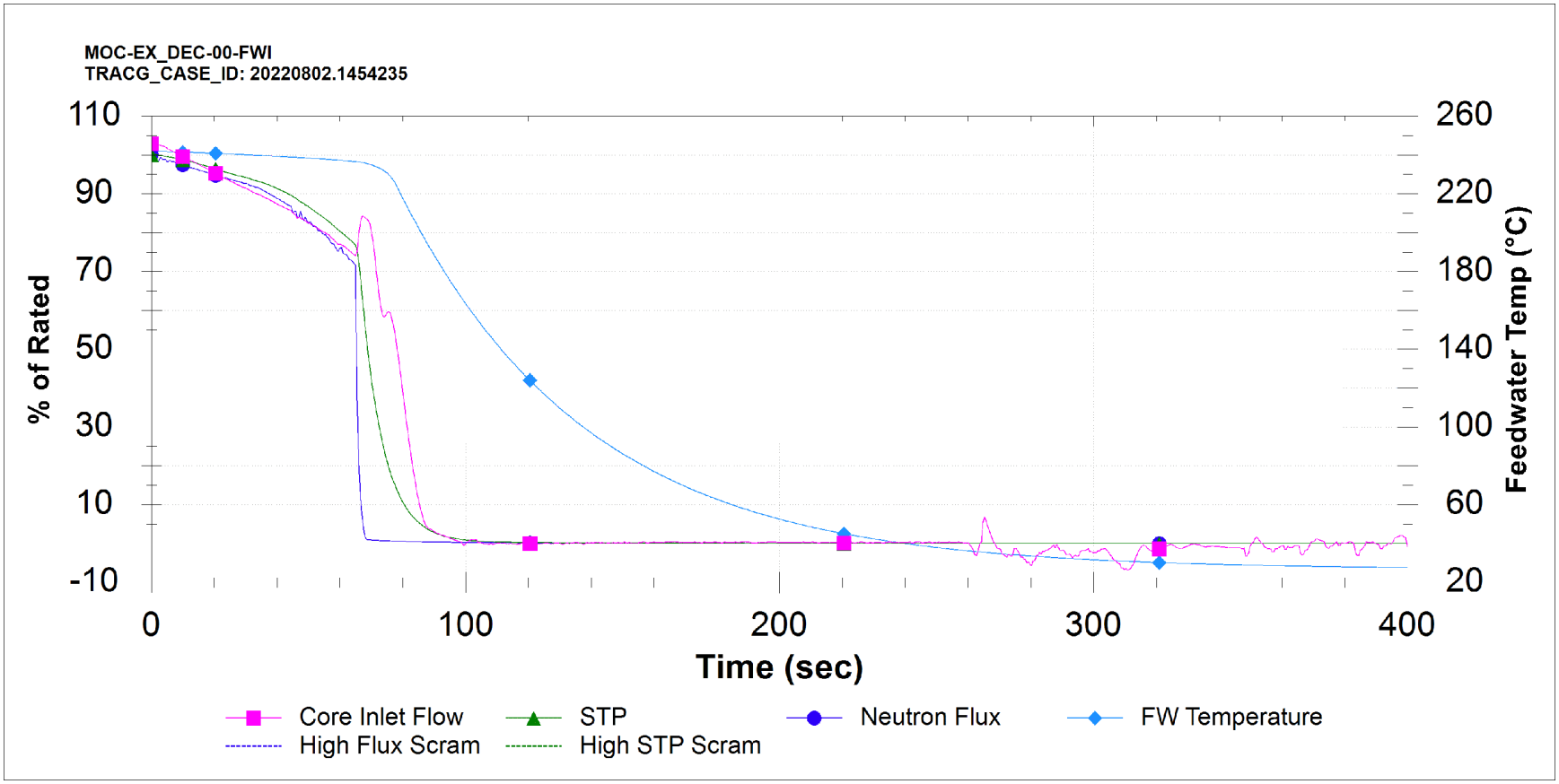

Figure 15.5-155 Feedwater Isolation (DEC)

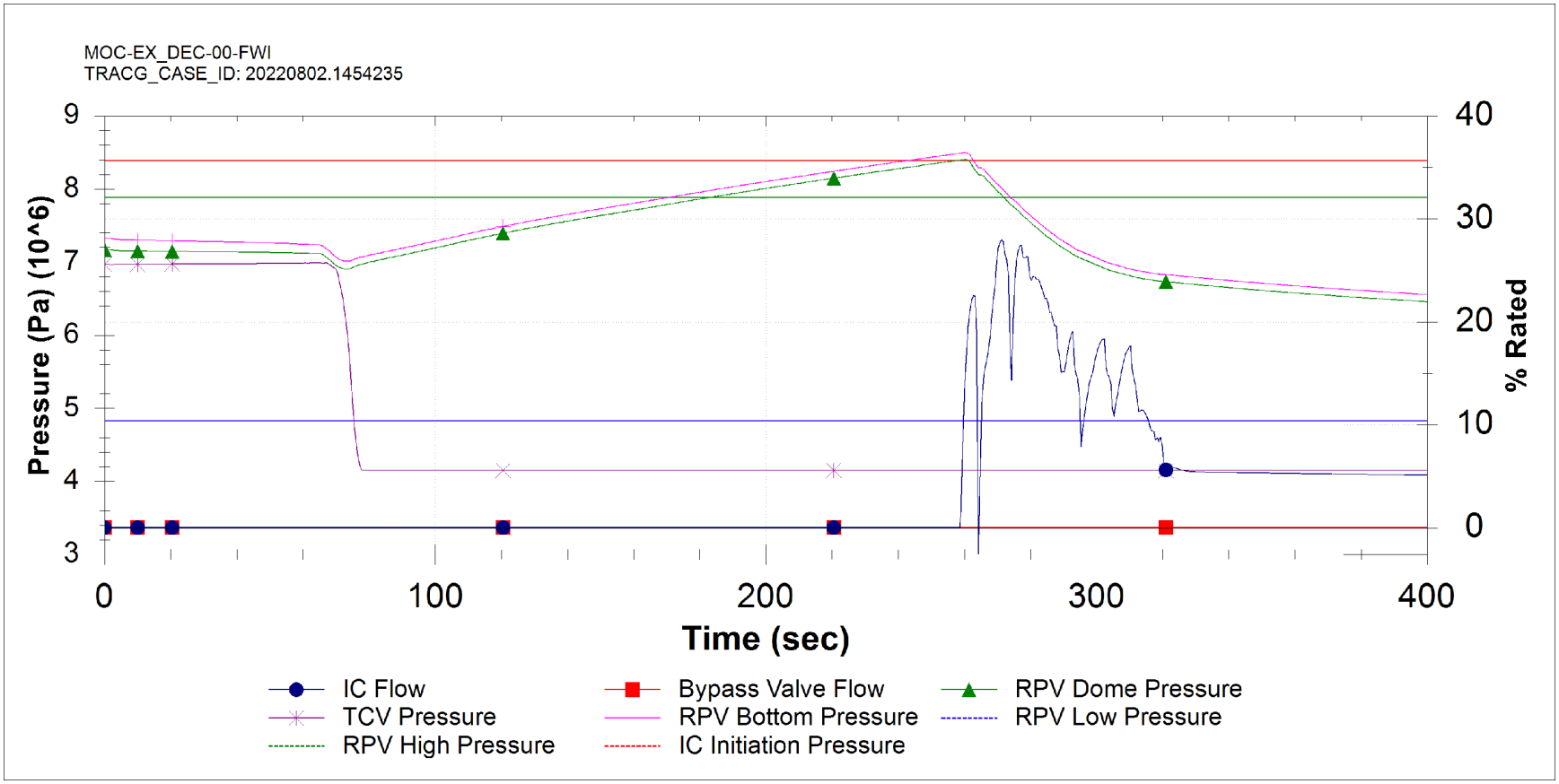

Figure 15.5-156 Feedwater Isolation (DEC)

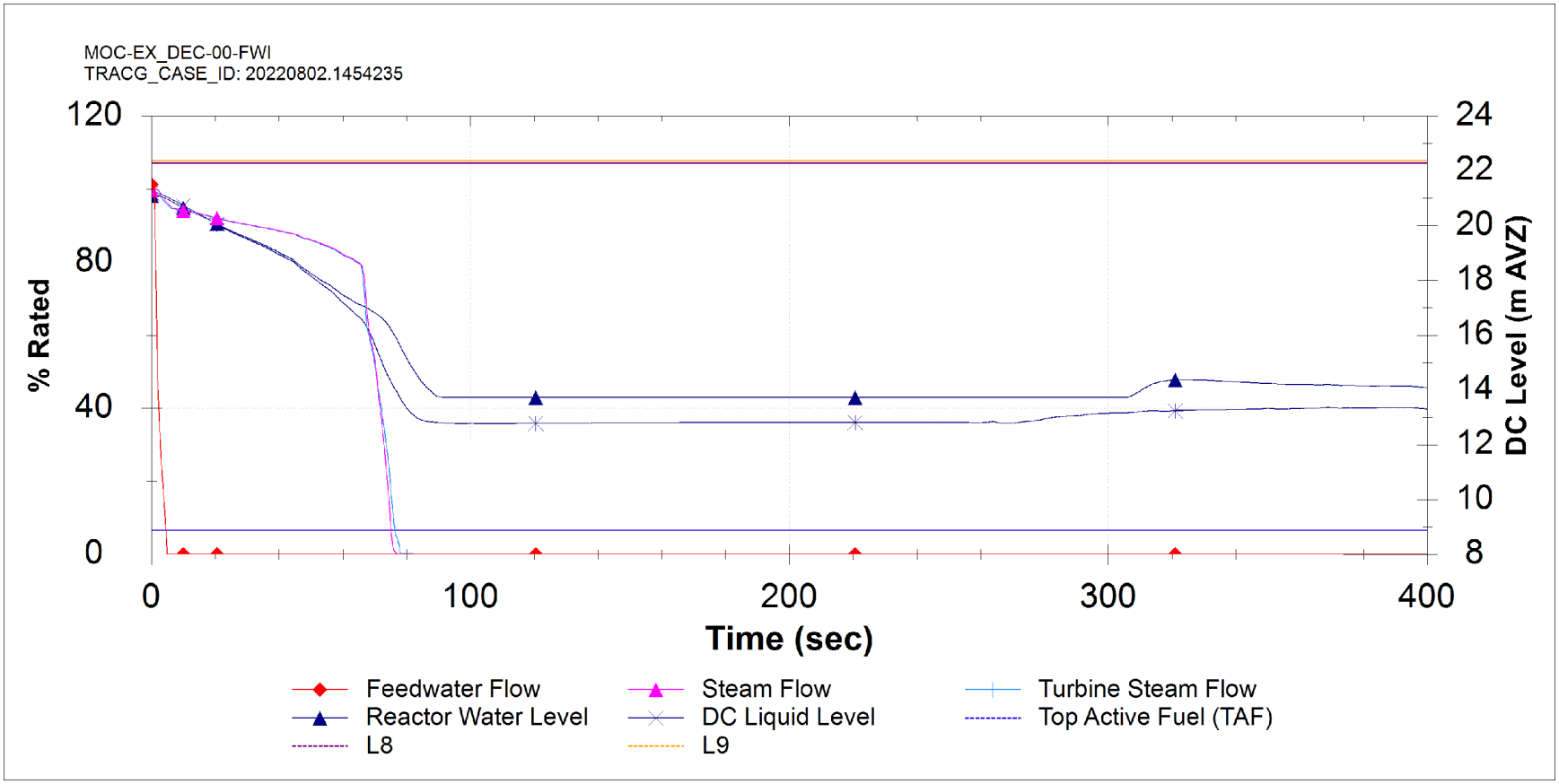

Figure 15.5-157 Feedwater Isolation (DEC)

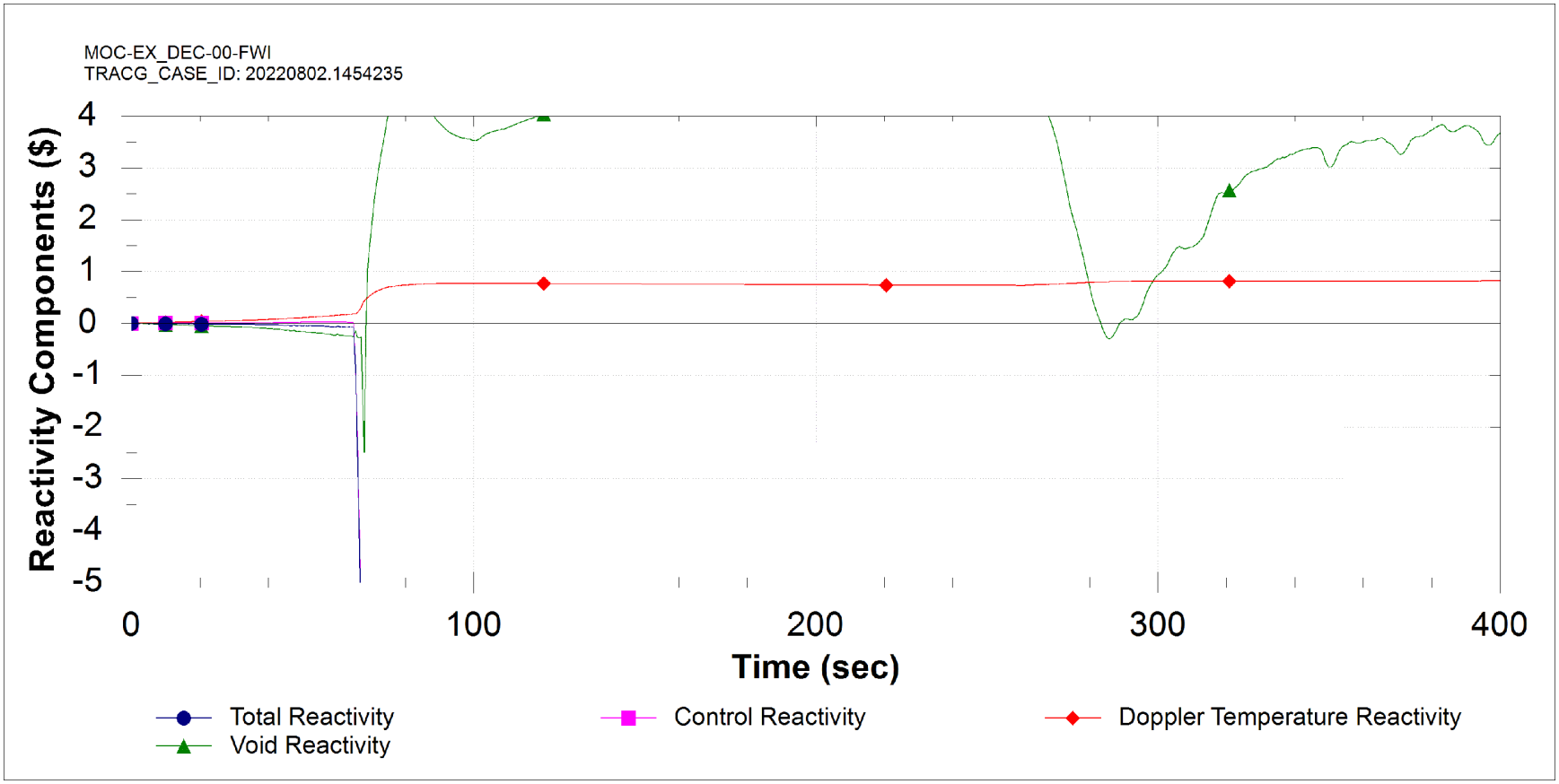

Figure 15.5-158 Feedwater Isolation (DEC)

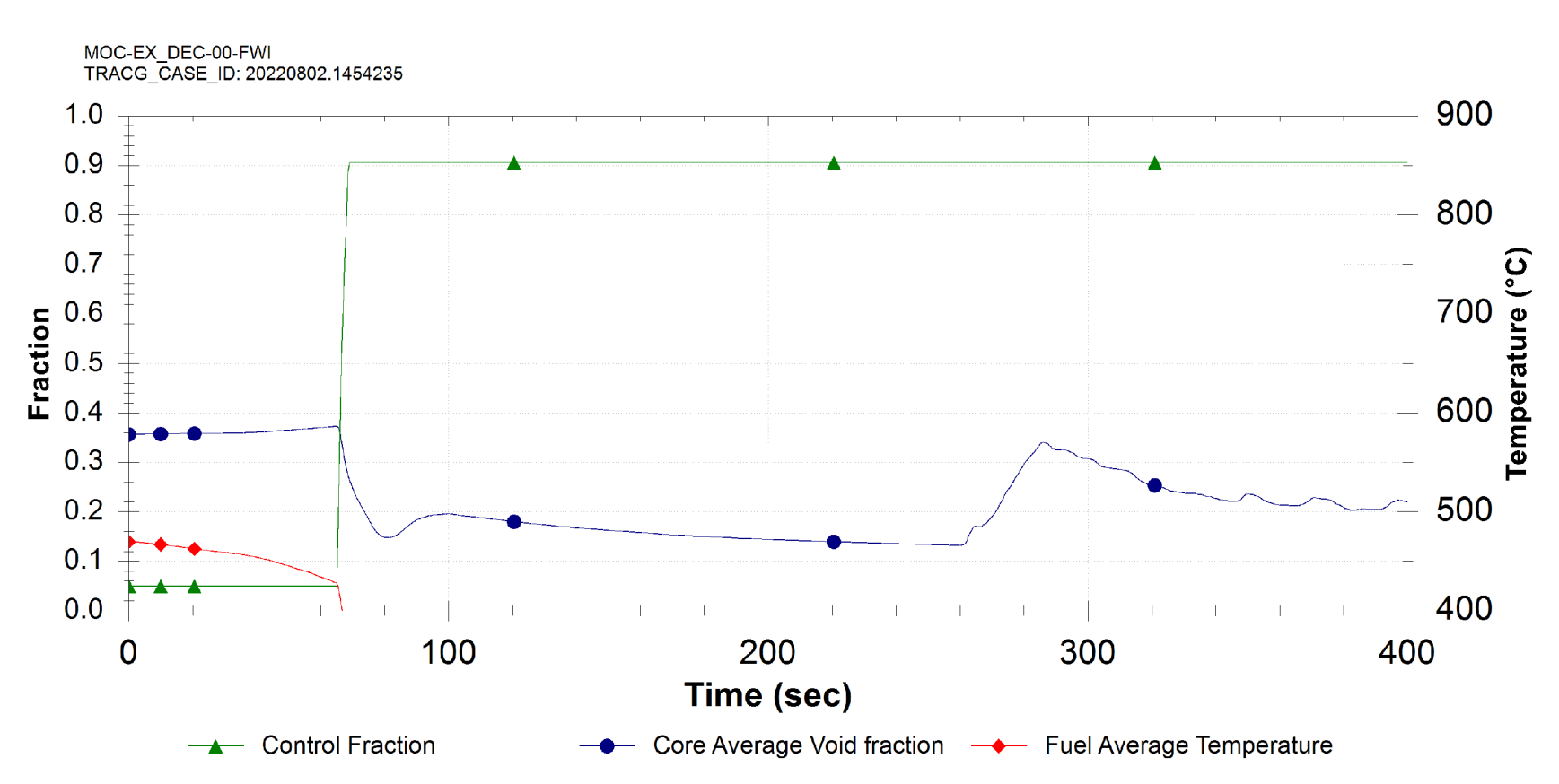

Figure 15.5-159 Feedwater Isolation (DEC)

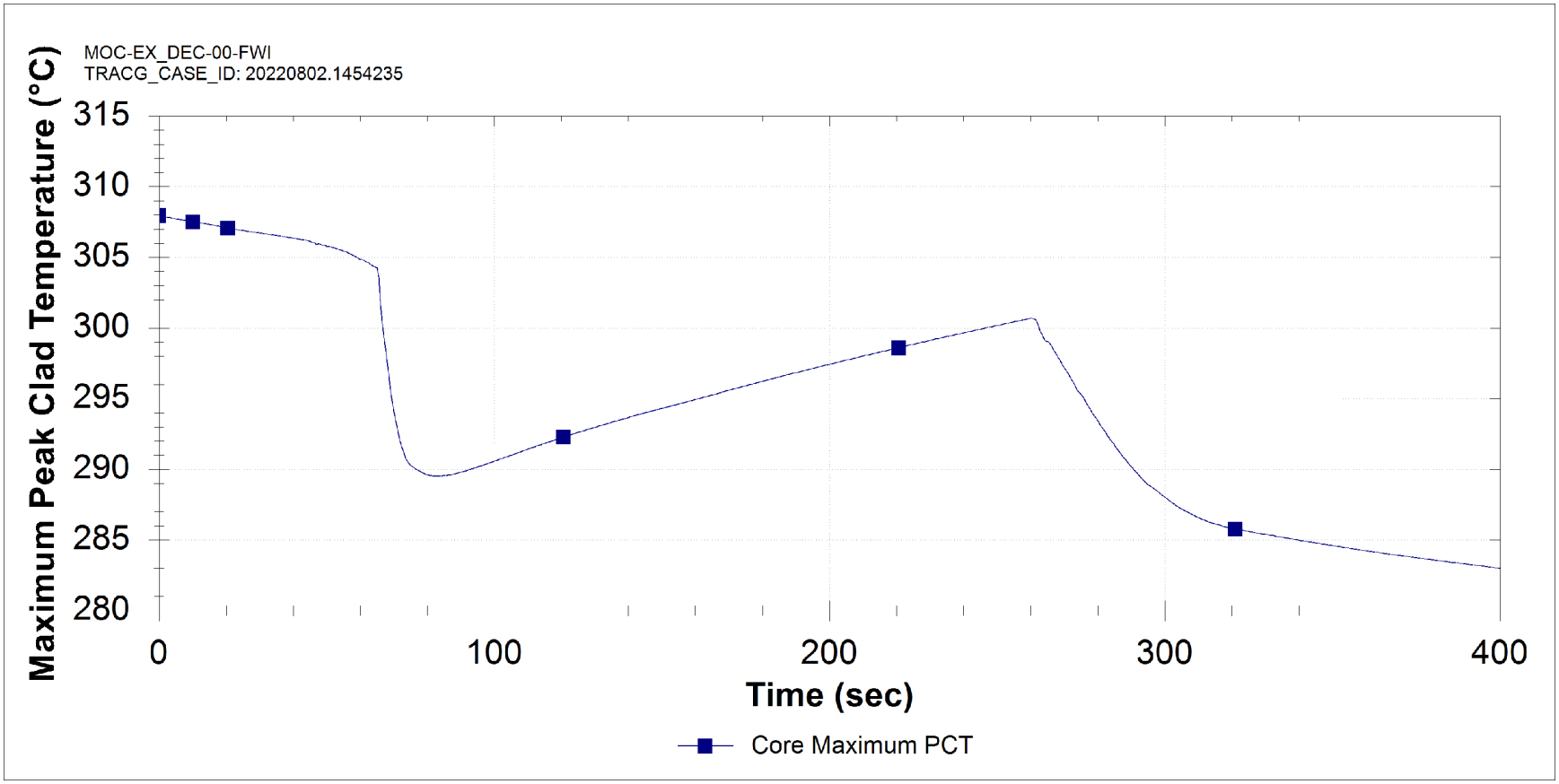

Figure 15.5-160 Feedwater Isolation (DEC)

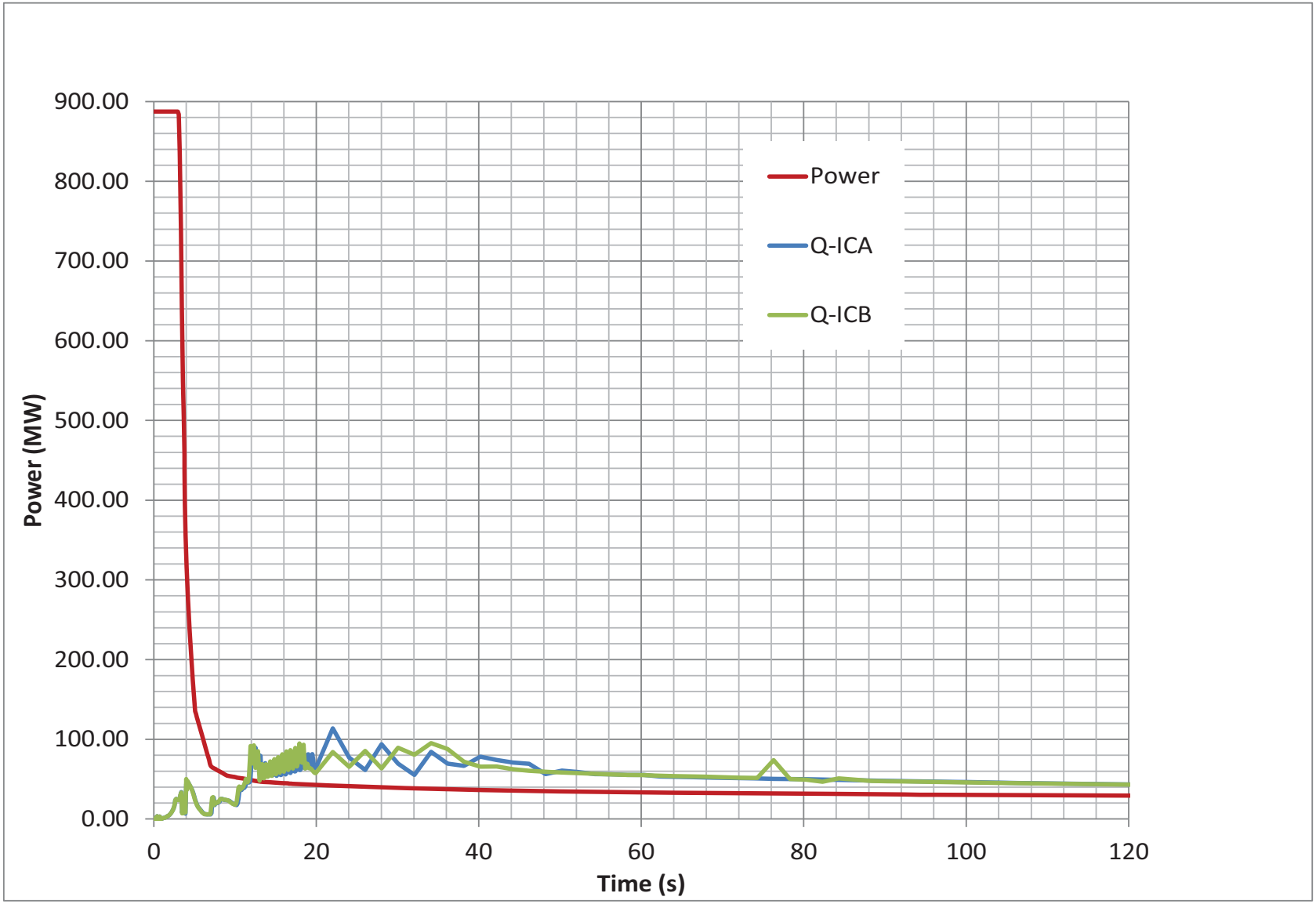

Figure 15.5-161 Reactor Power, Large FW Pipe Break, Conservative Case

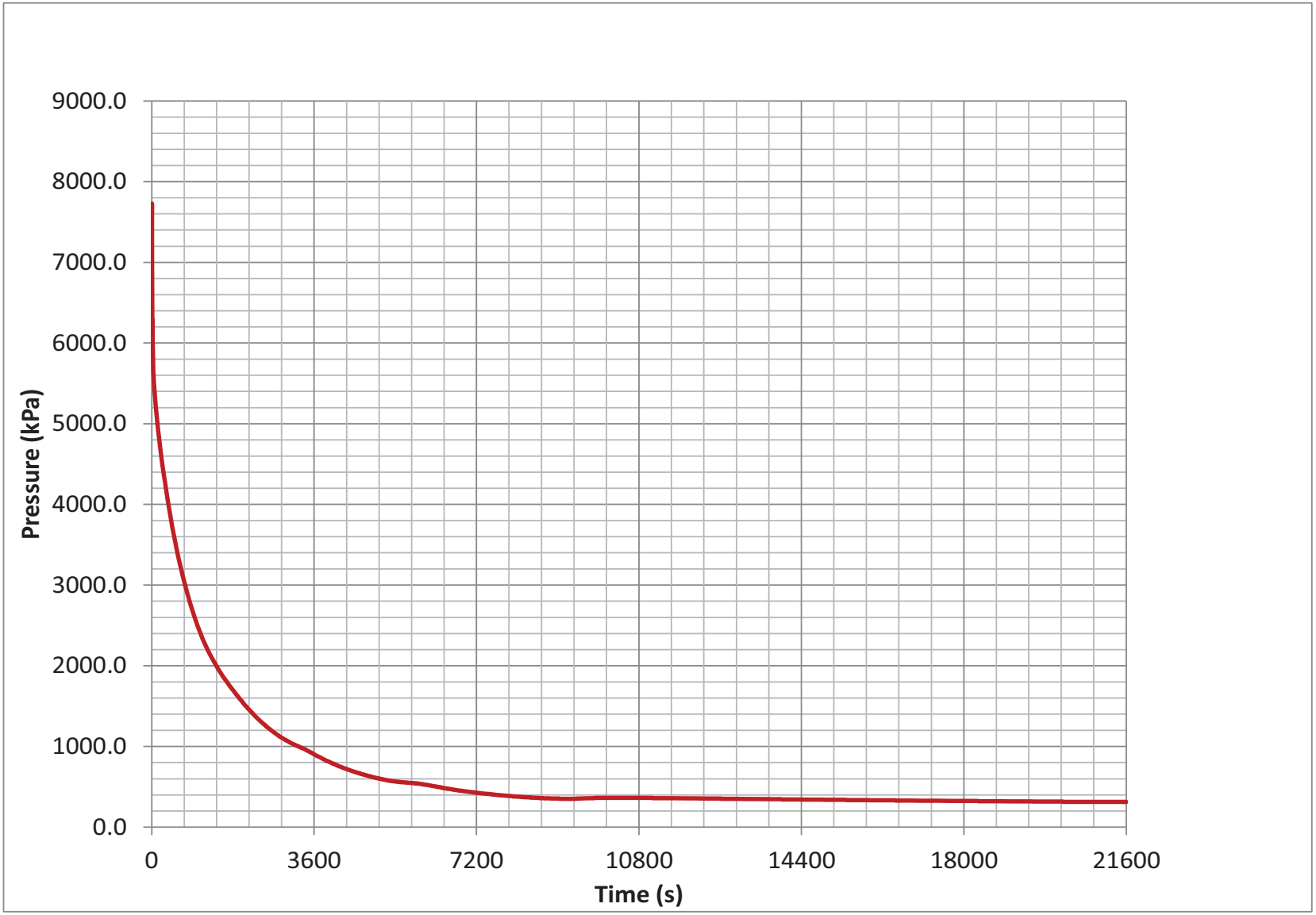

Figure 15.5-162 Reactor Pressure, Large FW Pipe Break, Conservative Case

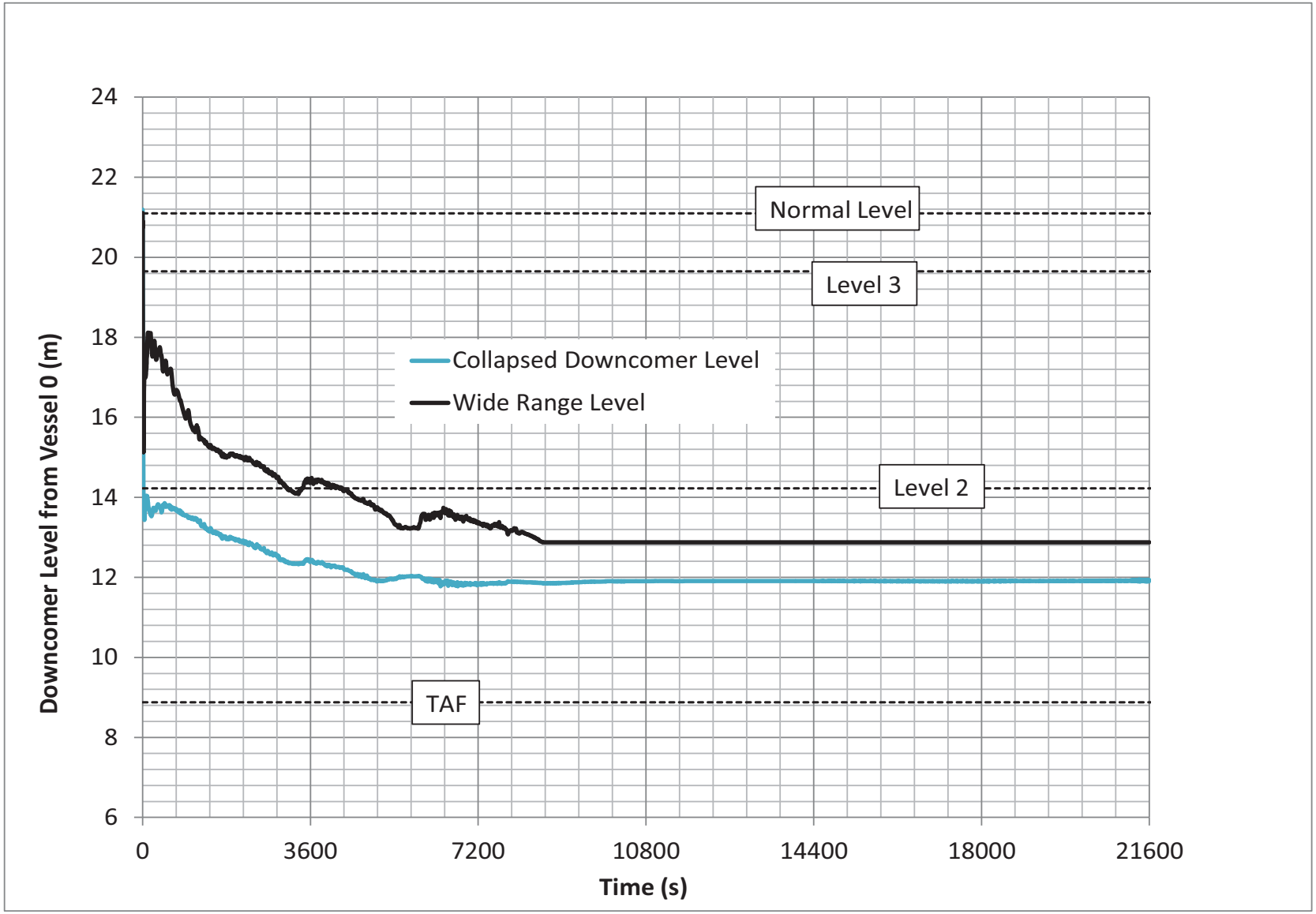

Figure 15.5-163 Reactor Water Level, Large FW Pipe Break, Conservative Case

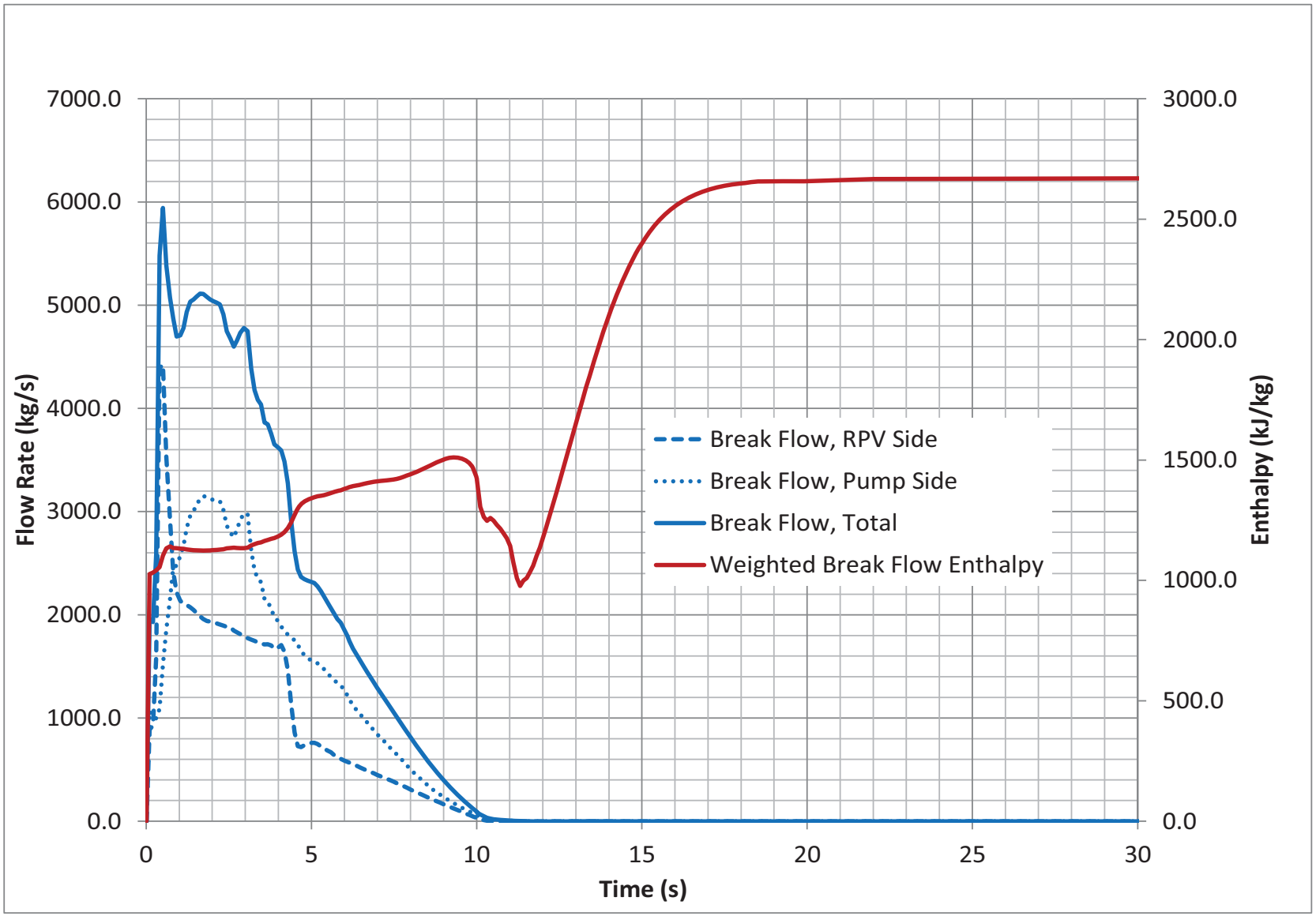

Figure 15.5-164 Break Flow Rate and Enthalpy, Large FW Pipe Break, Conservative Case

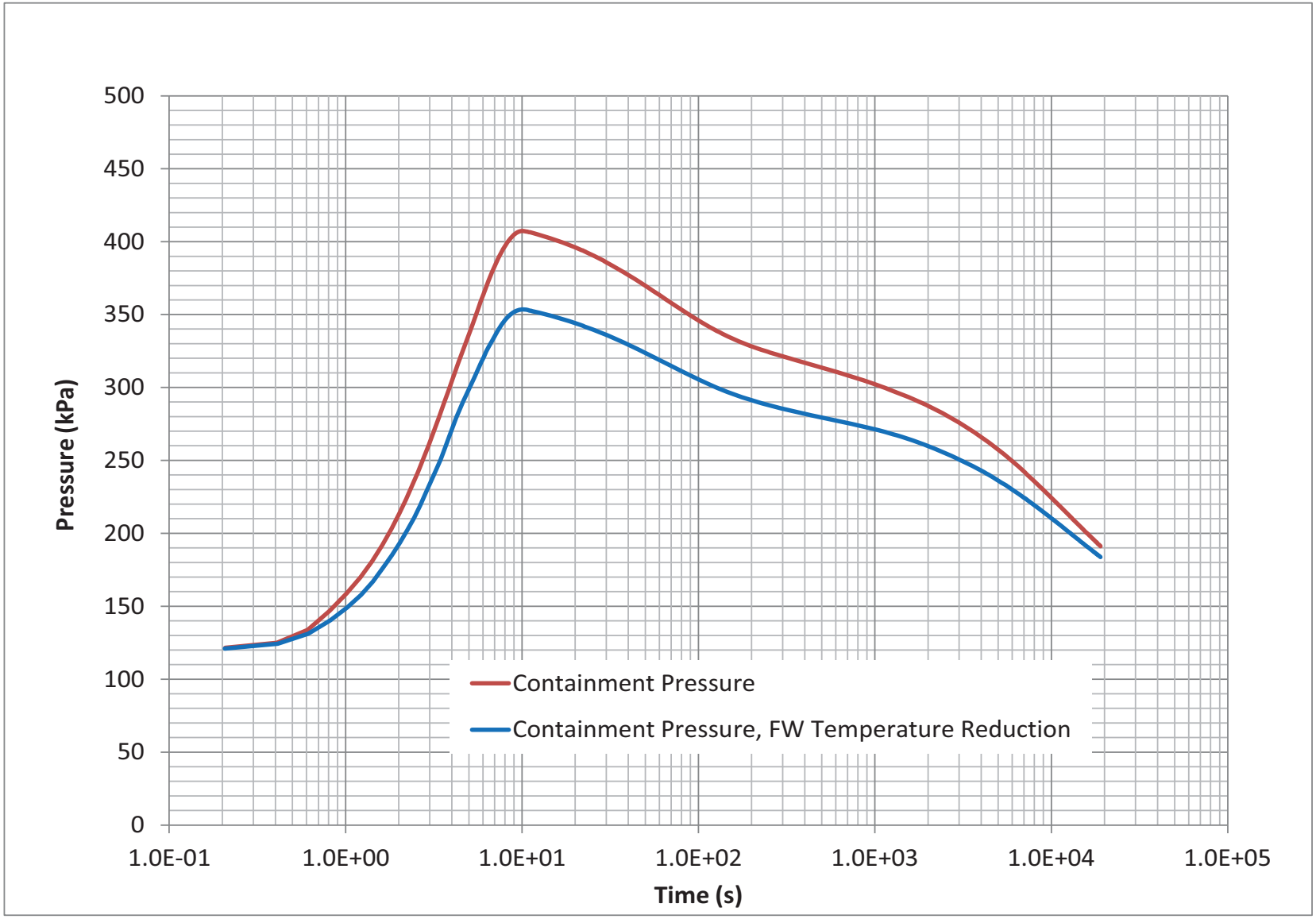

Figure 15.5-165 Containment Pressure, Large FW Pipe Break, Conservative Case

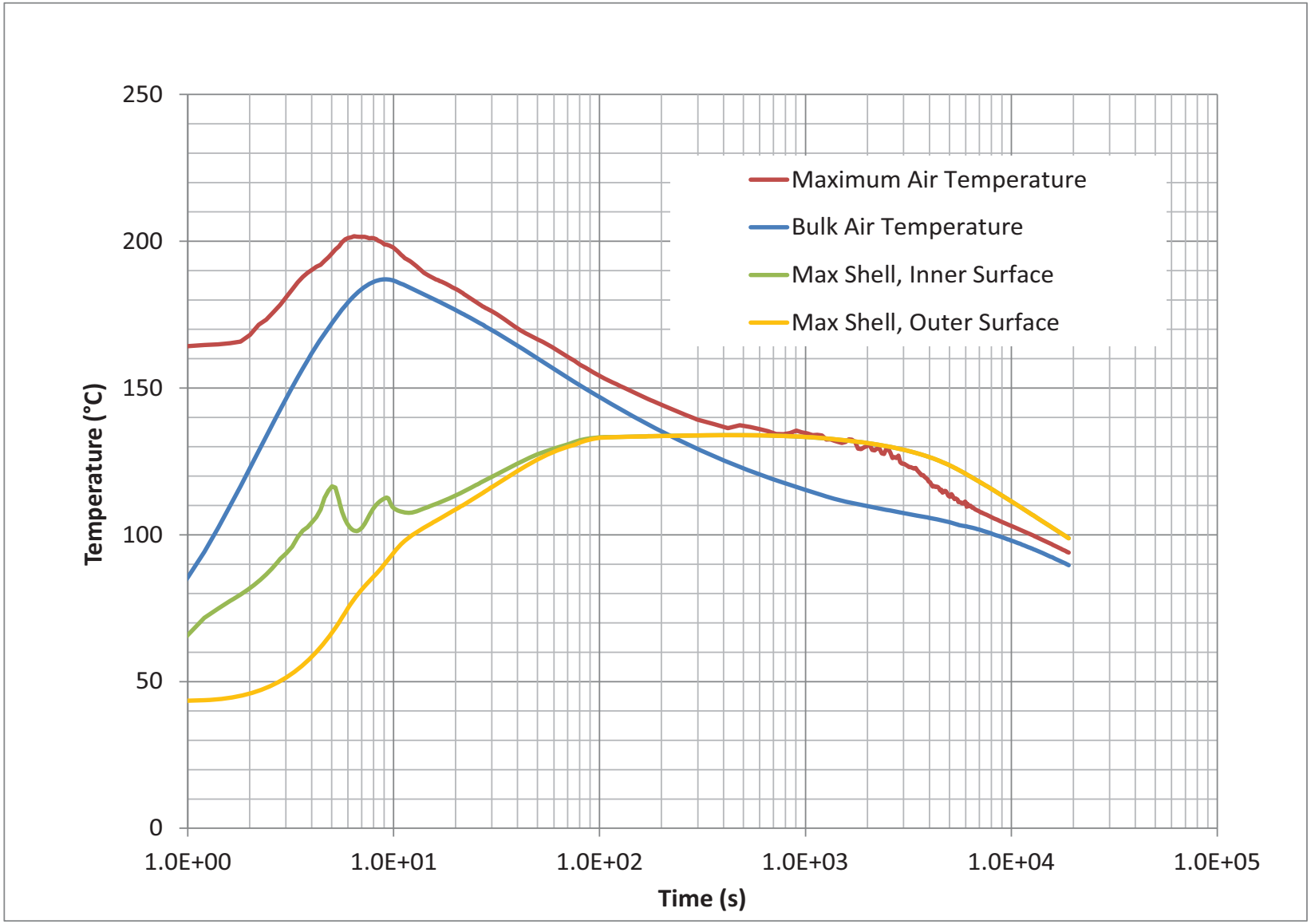

Figure 15.5-166 Containment Temperature, Large FW Pipe Break, Conservative Case

15.6 Probabilistic Safety Assessment

15.6.1 Introduction

An element of the BWRX-300 Safety Strategy design process provided in NEDC-33934P, “BWRX-300 Safety Strategy,” (Reference 15.6-1) is the development and results of a Probabilistic Safety Assessment (PSA) that provides an integrated review of the plant design, operational safety, and overall risk presented by the facility.

The BWRX-300 Safety Strategy implements a multi-faceted approach to safety assessment and determination of the overall plant risk profile. The approach makes use of hazard evaluations, DSA, and PSA, as well as specifically targeted analysis techniques that provide bases for a comprehensive set of “design-to-analysis” requirements. These requirements inform design development and modifications such that the design can be shown to effectively satisfy analysis acceptance criteria and the plant safety goals. The PSA supports risk-informed design development and, with the DSA, is used to understand the overall risk and any dominant contributors presented by the facility. The design is adjusted as necessary based upon the effectiveness of meeting component reliability targets and safety goals. The PSA also provides essential understanding of strengths and weaknesses of the design with complex systems and interdependencies.

As with the overall BWRX-300 Safety Strategy implementation process, the safety evaluations and analyses are performed iteratively as the design and licensing documentation are developed. These processes then implement required modifications and provide feedback to the next iteration of the evaluations and analyses.

The design phase PSA is updated as additional design and site-specific information becomes available for the Operating License Application and will be used to develop a site-specific PSA which will reflect the final design of the as-built to-be-operated plant. This is accomplished by assessing design vulnerabilities and optimizing the design using a graded approach that provides a strategy for meeting the Commissions “Safety Goals for the Operations of Nuclear Power Plants; Policy Statement,” (Reference 15.6-2). The results from the ongoing Level 1 and Level 2 design PSA for all modes and hazards and supporting analyses use the methodologies described herein and are available in separate supporting documents. The Final Safety Analysis Report (FSAR) will provide the results of the site-specific PSA.

This section describes the scope and methodology and design insights of the design-phase-specific PSA and Severe Accident (SA) evaluations performed for the BWRX-300. The risk assessment performed provides an understanding of outcomes, sensitivities, areas of importance, system interactions, and areas of uncertainty. Important accident scenarios can be identified from this assessment as recommended by U.S. Nuclear Regulatory Commission (USNRC) SRM-SECY 98-144 “White Paper on Risk-Informed and Performance-Based Regulation.”

Two PSA levels are applied that estimate the overall risk to the surrounding environment. Each level introduces a specific aspect of overall risk:

- Level 1 estimates the first measure of risk (core damage frequency)
- Level 2 estimates the second measure of risk (radioactivity release)

Severe Accident Analysis (SAA) is performed to support binning of the Level 2 PSA sequences. Further, SAA is used for radiological source term estimates for off-site dose consequence analysis. In SAA, the relevant phenomenological models are used to evaluate the plant response to an initiating event that leads to core damage. Level 1 and Level 2 PSA inform the overall sequence descriptions to be analyzed under the SAA domain.

For practical application, quantitative safety goals are established to achieve the intent of the qualitative safety goals. The two quantitative PSA safety goals are:

- Core Damage Frequency (CDF)
- Large Release Frequency (LRF)

PSA models are developed with proven and validated computer codes that produce appropriate results and are consistent with industry standards.

15.6.1.1 Probabilistic Safety Assessment Scope

The design stage PSA includes a Level 1 and Level 2 PSA that meets the performance requirements of the American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) RA-S-1.1-2024 (Reference 15.6-3) and RA-S-1.2-2024 (Reference 15.6-36), PSA standards for Light Water Reactors (LWRs).

The PSA model represents the standard plant BWRX-300 baseline design. This baseline PSA is subsequently used to develop the site-specific PSA. The PSA provided in the FSAR will present the results and consequences of PSA event scenarios that weigh the consequences by frequency to provide a measure of risk to the general public from the BWRX-300 compared to other non-nuclear risks.

The objectives of the PSA are listed in Table 15.6-1.

The BWRX-300 PSA is performed in an iterative manner with the design development used to evaluate and improve the risk aspects of the BWRX-300 design. The specific objectives of the PSA and SA evaluations are demonstrating the BWRX-300 has been designed with state-of-the-art safety features and incorporating highly reliable and available passive safety functions with significant redundancy and diversity that meets all established safety goals with margin.

The BWRX-300 design phase PSA uses current information available from the BWRX-300 plant design and procedures. Periodic updates of the PSA are required after the plant begins operation. Component failure data and Initiating Event (IE) frequencies are based on generic United States industry data considering the BWRX-300 design. This data and the assumptions

used to develop the plant-specific PSA are selected as realistically as practical. Based upon the state of the BWRX-300 plant design, when design, site, or programmatic information is not available, the PSA models the various elements in a conservative manner that contains conservative elements (e.g., pre-initiator and post-initiator human error probabilities; maintenance unavailabilities; component failure rates; flood and fire initiation, propagation, and effects; ground level release with no evacuation warning assumed in consequence analysis).

15.6.1.2 Methodology

The BWRX-300 design achieves a higher standard of SA safety performance than previous designs by considering guidance and goals for events that are beyond what is typically referred to as the design basis of the plant. For the BWRX-300, SA issues are addressed during the design stage to take full advantage of the insights gained from the PSA as well as from Operating Experience, SA research, and accident analysis. The insights are used to identify design features that reduce the likelihood that SAs will occur and to mitigate the consequences of SAs.

15.6.1.3 Risk Insights

The term “risk insights” used in this section is consistent with the term defined in SRM-SECY-98-144 that is results and findings that come from risk assessments. The end results of the assessments may relate directly to public health effects as in the USNRC’s Safety Goals for the Operations of Nuclear Power Plants. The results and findings include identifying dominant accident sequences, estimating CDF and LRF, as well as determining Structures, Systems, and Components (SSCs) importance measures.

15.6.1.4 Regulatory Compliance

Other important sources of PSA guidance for performing Level 1 and Level 2 PSAs are in RG 1.200, “Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities,” IAEA SSG-3, “Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plant,” (Reference 15.6-4) and IAEA SSG-4, “Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants,” (Reference 15.6-5).

The PSA methodology requires examination of systems and their interactions in an integrated and comprehensive manner that addresses a broad spectrum of Initiating Events (IEs) and considers their frequency. The PSA insights provide insights into the design that ameliorate public health effects and the USNRC Safety Goals.

15.6.1.4.1 Commission Policy Statements

The Commission acceptance criteria is delineated in the following policies:

1. Policy Statement, “Severe Reactor Accidents Regarding Future Designs and Existing Plants,” Volume 50 of the Federal Register, page 32138 (50 FR 32138; August 8, 1985).

"Regarding the decision process for certifying a new standard plant design (an approach the Commission strongly encourages for future plants) the Policy Statement affirms the Commission's belief that a new design for a nuclear power plant can be shown to be acceptable for SA concerns if it meets the following criteria and procedural requirements.

- Demonstration of compliance with the procedural requirements and the current Commission regulations, including the Three Mile Island requirements for new plants as reflected in the Commission Policy.
- Demonstration of technical resolution of applicable Unresolved Safety Issues and the medium and priority Generic Safety Issues, including special focus on assuring the reliability of decay heat removal and the reliability of both Alternating Current (AC), Direct Current (DC), and electrical supply systems.
- Completion of a PRA and consideration of the SA vulnerabilities the PRA exposes along with the insights that it may add to the assurance of no undue risk to public health and safety.
- Completion of a staff review with conclusive design safety acceptability using an approach stressing deterministic engineering analysis and judgement complemented by PRA."

2. Policy Statement, "Safety Goals for the Operations of Nuclear Power Plants" (51 FR 28044; August 4, 1986). Corrected and reprinted at Federal Register, Volume 51, No. 162, pp. 30028-30033, August 21, 1986, establishes the safety goals for nuclear power plants in the US.

   - The qualitative safety goals are:
     - Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.
     - Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.
     - The following quantitative objectives are to be used in determining achievement of the above safety goals:
       - The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.
       - The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.

     These Commission Safety Goals are the basis for the safety goals identified for the BWRX-300.

3. Policy Statement, “Regulation of Advanced Nuclear Power Plants” (59 FR 35461; July 12, 1994).

   - “Among the attributes that could assist in establishing the acceptability or licensability of a proposed advanced reactor design, and that therefore should be considered in advanced designs, are:
     - Highly reliable and less complex shutdown and decay heat removal systems. The use of inherent or passive means to accomplish this objective is encouraged (e.g., negative temperature coefficient, natural circulation).
     - Longer time constants and sufficient instrumentation to allow for more diagnosis and management before reaching safety systems challenge and/ or exposure of vital equipment to adverse conditions. Simplified safety systems that, where possible, reduce required operator actions, equipment subjected to severe environmental conditions, and components needed for maintaining safe shutdown conditions. Such simplified systems should facilitate operator comprehension, reliable system function, and more straightforward engineering analysis. Designs that minimize the potential for SAs and their consequences by providing sufficient inherent safety, reliability, redundancy, diversity, and independence in safety systems.
     - Designs that provide reliable equipment in the Balance of Plant (BOP) (or safety-system independence from BOP) to reduce the number of challenges to safety systems.
     - Designs that provide easily maintainable equipment and components. Designs that reduce potential radiation exposures to plant personnel.
     - Designs that incorporate defense-in depth philosophy by maintaining multiple barriers against radiation release, and by reducing the potential for and consequences of SAs.
     - Design features that can be proven by citation of existing technology or that can be satisfactorily established by commitment to a suitable technology development program.”

4. Policy Statement, “Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities” (60 FR 42622; August 16, 1995).

   > “The Commission believes that an overall policy on the use of PRA in nuclear regulatory activities should be established so that the many potential applications of PRA methodology can be implemented in a consistent and predictable manner that promotes regulatory stability and efficiency. In addition, the Commission believes that the use of PRA technology in USNRC regulatory activities should be increased to the extent supported by the state-of-the-art in PRA methods and data and deterministic approach.”

15.6.1.4.2 SECY Documents

Commission direction and staff guidance on PSAs relevant to licensing reviews are provided in the first five documents listed below. Excerpts are provided to highlight the appropriate course of action to address SAs and the use of PSA. The Staff Requirements Memoranda (SRM) relating to SECY-90-016, SECY-93-087, and SECY-96-128 provide Commission-approved guidance for

implementing features in new designs to prevent SAs and to mitigate their effects, should they occur.

Probabilistic risk calculations demonstrate that the BWRX-300 design meets USNRC, Canadian Nuclear Safety Commission, and IAEA guidelines for scram function reliability. The design has a less frequent Anticipated Transient Without Scram (ATWS) event than 1E-5/yr, has a conditional probability of scram failure less than 1E-5/demand, and an LRF contribution from fail-to-scram events less than 1E-7/yr.

1. SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” dated April 2, 1993.

   Specific Issues related to a risk informed approach are discussed as follows:

   I. SECY-90-016 Issues

      F. Intersystem Loss-of-Coolant Accident (LOCA)

      G. Hydrogen Control

      H. Core Debris Coolability

   III. Issues Limited to Passive Design

      A. Regulatory Treatment of Non-safety Systems in Passive Designs

      E. Control Room Habitability

2. SECY-94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs,” dated March 28, 1994, and its associated SRM for station blackout.

3. SECY-90-016, “Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements,” dated January 12, 1990, and the related SRM.

4. SECY-95-132, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-safety Systems in Passive Plant Designs,” dated May 22, 1995, and its associated SRM, dated June 28, 1995.

5. SECY-96-128, “Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design,” dated June 12, 1996.

   IV. Post-72 Hour actions

   “Therefore, staff recommends that the Commission approve the position that site be capable of sustaining all design basis events with onsite equipment and supplies for the long term. After 7 days, replenishment of consumables such as diesel fuel oil from offsite suppliers can

be credited. The equipment required after 72 hours need not be in automatic standby response mode but must be readily available for connection and be protected from natural phenomena including seismic events (per GDC 2)."

The BWRX-300 design meets the LRF goal of SECY-96-128 regarding the capability to sustain all DBEs with onsite equipment and supplies for 7 days.

6. SECY-94-219, "Proposed Agency-wide Implementation Plan for Probabilistic Risk Assessment," dated August 19, 1994.

   > "Therefore, the goal of the PRA Implementation Plan is to achieve these benefits by increasing the use of PRA in regulatory matters to the extent practical given the state-of-the-art in PRA methods and data available. An important aspect of the expanded use of PRA technology in reactor regulation will be a strengthening of USNRC's Defense-in-Depth (D-in-D) philosophy by allowing quantification of the levels of protection and by helping to identify and address weaknesses in the physical and functional barriers, should they exist."

   The BWRX-300 Safety Strategy utilizes the PSA to identify and address design weaknesses in protection of physical barriers.

   The PSA is used to measure the safety of plant design and operation in minimizing the risk of releasing radionuclides to the environment by assessing design vulnerabilities and optimizing the design using a graded approach as delineated in the Commission's PSA policy statement and implementation plan (60 FR 42622 and SECY-94-219).

15.6.1.4.3 Interim Staff Guidance

The PSA methodology implements the intent of DNRL-ISG-2022-01, "Safety Review of Light-Water Power Reactor Construction Permit Application," as follows:

The PSA identifies aspects of the design that contribute most to safety as well as those aspects that have low risk or safety significance.

The methodology integrates risk insights with traditional engineering approaches.

The methodology implements the direction in the Commission's Probabilistic Risk Assessment (PRA) Policy Statement (60 FR 42622; August 16, 1995) to use PRA technology to increase state-of-the-art in PRA methods and data and to complement the USNRC's deterministic approach and supports the USNRC's traditional Defense-in-Depth philosophy."

SRM-SECY-98-144 discusses the terms and concepts involved in the PSA policy statement and how these concepts are to be applied to USNRC rulemaking, licensing, inspection, assessment, enforcement, and other decision-making.

15.6.1.4.4 Regulatory Guides

Regulatory Guide 1.76 Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants

BWRX-300 SSCs are appropriately protected against the effects of tornado missiles as discussed in Section 3.5.1.4.

The following clarification was provided by the USNRC Staff of SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-safety Systems in Passive Plant Designs," in SECY-96-128, Item IV:

"During the staff review of AP600 post-72-hour design changes, there has been some misunderstanding of the SECY-96-128 policy regarding classification of the post-72-hour equipment. Because GDC 2 was parenthetically mentioned in SECY-96-128, it could be interpreted that SECY-96-128 required post-72-hour equipment conform to the safety-related application of GDC 2, that the added equipment be designated Seismic Category I, and that it be capable of remaining functional under tornado wind loads and tornado missiles. It was not intended that equipment or design changes made to comply with the post-72-hour actions policy of SECY-96-128 be safety-related. Rather, it was intended that the equipment be available on site (under the direct control of the licensee) and be protected from natural phenomena, including seismic events. Consequently, the staff has clarified the post-72-hour equipment classification and defined criteria for protecting post-72-hour equipment from the effects of natural phenomena based on the relative importance of the post-72-hour equipment to the overall safety of the AP600 design. The staff believes that this position is consistent with the positions taken in SECY-96-128, and subsequently approved by the Commission."

Regulatory Guide 1.200 Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities

The BWRX-300 PSA methodology complies with ASME/ANS RA-Sa-2009 and Addenda ASME/ANS RA-Sa-2008, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," (Reference 15.6-6) as endorsed by RG 1.200 and ASME/ANS RA-S-1.1-2024 when endorsed by the USNRC.

The BWRX-300 PSA methodology ensures the scope and technical content of the PSA is complete, correct, and accurate. The PSA produces insights with appropriate fidelity supporting decisions made during the maturation of the design. The PSA methodology meets the intent of RG 1.200 by providing a high quality, robust model of the plant that supports design decisions and provides insights that support the ongoing design process.

RG 1.200 notes that the following IAEA Safety Standards Series incorporate similar design and preoperational testing guidelines and are consistent with the basic safety principles considered in developing this RG:

- IAEA Safety Standards Series No. SSG-3, "Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants"

- IAEA Safety Standards Series No. SSG-4, “Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants”

The content of both SSG-3 and SSG-4 have been considered in the design of BWRX-300. Reference to these documents is only made to clarify how their content has been utilized in development of the BWRX-300 methodology.

15.6.1.4.5 NRC Staff Final Safety Evaluation Report Limits and Conditions

The USNRC Staff Final Safety Evaluation Report for NEDC-33912P, “BWRX-300 Reactivity Control,” (Reference 15.6-7) included the following Limitation and Condition:

> “Additionally, any applicant referencing NEDC-33912 must perform and document in an application the following:
>
> 5.1 Reliability analysis or testing, considering applicable operating experience and expected load follow conditions, of the BWRX-300 diverse scram features to demonstrate the probability of an ATWS is less than 1E-5 per reactor year.”

The BWRX-300 PSA includes the demonstration that the probability of an ATWS is less than 1E-5 per reactor year without dependence on boron injection or an alternate rod insertion system.

Subsection 15.6.3.2.2 discusses the modeling of ATWS events.

15.6.2 Probabilistic Safety Assessment Overview

The BWRX-300 PSA is performed in an iterative manner with the BWRX-300 design development to evaluate and improve the risk aspects of the BWRX-300 design. The specific objectives of the PSA and SA evaluations demonstrate that the BWRX-300 has been designed with state-of-the-art safety features, incorporating highly reliable and available passive safety functions with significant redundancy and diversity, meeting established safety goals with margin.

The design phase PSA for BWRX-300 covers all relevant IEs. The event types investigated include:

- Internal IEs (internal failures and disturbances, and human error related initiators)
- Loss-of-Preferred Power (LOPP) - plant centered, grid-related and switchyard-centered
- Internal hazards (e.g., fire, flooding, and lifting of heavy loads) and other possible relevant Common Cause Initiators (CCI)
- External hazards (e.g., adverse weather conditions, seismic events, and consequence of external human caused events excluding sabotage)

For the analysis of external hazards, possible combinations of external hazards are addressed. For example, high wind causing loss of the external grid are considered in the high wind analysis. Other site-specific combinations (such as snowstorm blocking the diesel generator combustion air intake and frazil ice blocking water cooling), are reviewed and included in BWRX-300 PSA models.

The design phase PSA covers both at-power (full and reduced power levels) and low-power and shutdown operations, which include outages and transitions between each plant operating state. The design phase PSA includes main systems and components (including reactor and turbine Instrumentation and Control (I&C) logic), main operator actions and the relevant event and system dependencies, interconnections, and Common Cause Failure (CCF) relationships.

The development of BWRX-300 PSA models is conducted with the EPRI Integrated Risk Technology suite of software which include FRANX used for Fire PSA model development and the “Phoenix Architect Module,” Version 2.0 (Reference 15.6-8), which includes the following:

- Computer Aided Fault Tree Analysis (CAFTA) for event tree, fault tree, and cutset development and viewing purposes
- Phoenix Application Programming Interface (API) for model integration and one-top development
- PRAQuant for quantification processes, as well as for sensitivity runs
- Human Reliability Analysis (HRA) Calculator for Human Error Probabilities (HEPs) and Operator Action Dependency Analysis
- UNCERT for Uncertainty Analysis (Reference 15.6-9)

The PSA model is integrated and quantified using the computer codes CAFTA, PRAQuant, and Fault Tree Reliability Evaluation Expert (FTREX) (Reference 15.6-10). These computer codes have been demonstrated throughout the industry to produce appropriate results. No method specific limitations have been identified regarding the software tools or the methodology implemented to quantify the model.

Using CAFTA, the BWRX-300 PSA model is developed by merging all model event trees, system fault trees, IEs, passive feature failure results, and associated basic event databases. A top logic fault tree is created using CAFTA. All system fault trees are merged with the top logic fault tree. The fault tree model top gates are quantified to generate cutsets and calculate total Core Damage and Large Release Frequencies, and individual event sequences are quantified to determine frequencies for the various accident sequence end states and release categories for the Level 2 PSA. Additionally, the linked/merged fault tree is quantified to generate minimal cutsets, which include identifiers to link them to the event tree sequences.

At a minimum, the following areas are addressed in the design phase PSA:

- BWRX-300 risk results compared to Quantitative Health Objectives, including:
  - Core Damage Frequency
  - Large Release Frequency

- IE selection, grouping and frequency for each analyzed plant operating state
- Event Sequence (ES) modeling and quantification for accident sequence end state and release category
- Detail level of system analysis
- Success criteria
- Component reliability data (including CCF data and modeling principles)
- Human Reliability Analysis (including operator action dependency analysis)
- Uncertainty and Sensitivity analysis
- Model documentation
- Structural analysis of the containment
- Mechanistic Source Term Analysis for each release category

The design phase PSA includes the PSA models as well as documentation, where the pertinent plant characteristics, modeling assumptions and techniques, model structure, data values, and sources used in the model and the analysis results are explained in a way that makes it possible to review and replicate the analyses.

Figure 15.6-3 highlights the integrated design process. The Reliability Assurance Program is described in Section 17.4. PSA model configuration control will be further described in the FSAR.

All work for the design phase PSA is performed in accordance with the requirements of the GVH quality assurance program.

Figure 15.6-1 shows the principal steps in the Level 1 PSA.

A full scope self-assessment and peer review of all internal and external events models is planned for the PSA model and documentation supporting the FSAR. The planned PSA peer review will assess the technical acceptability of the model and results against relevant PSA Standards. The PSA model, documentation, and peer review and self-assessments will comply with the industry and regulatory guidance such as NEI-00-02 and RG 1.200. The results of the peer review will be included in the licensing submittal, including gaps to meeting the Standard, limitations arising from the maturity of the design and operational details, as well as plans to address each identified issue.

RG 1.200, “Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities,” describes an acceptable means for conforming with SECY-04-0118, “Plan for the Implementation of the Commission’s Phased Approach to Probabilistic Risk Assessment Quality.” The requirement that the PSA be conducted under the management system or quality assurance program is established in the licensing basis. This section shows how PSA output (in concert with the DSA) provides input into the design that may affect the PSA models. Figure 15.6-3 highlights the configuration process for the integrated design. All work for the design phase PSA is performed in accordance with the requirements of the GVH quality assurance program. The description of the PSA model configuration process will be provided in the FSAR.

15.6.2.1 BWRX-300 Safety Goals

The safety goals for core damage and large release frequencies are listed in Table 15.3-3.

15.6.2.2 Principal Steps in the Probabilistic Safety Assessment Model Development

A design process that is “risk-informed” incorporates the results of risk evaluations and PSA where risk is evaluated for both the frequency and consequences of a possible event. In the nuclear industry, where design, operation, and maintenance are generally conducted using a prescriptive set of requirements, risk evaluations provide information on potential vulnerabilities of the design that may not be revealed by the deterministic design process. In using risk evaluations to assess a design, the risk results add to or supplement the design resulting in an increase in its overall reliability and safety. Using the PSA during the design process ensures that no particular feature or postulated IE makes a disproportionately large or significantly uncertain contribution to the overall risk. The application of risk information and PSA to supplement and improve a design during the design process risk-informs the design.

The design process uses the PSA results at each stage in the design to confirm the design adequacy and to evaluate alternatives being considered during the system engineering and design process. The PSA interfaces with the design requirements, the system engineering process, and D-in-D evaluations. The PSA process and use is described throughout this section.

The internal events at-power model is developed initially and covers these principal steps. The internal and external hazard models and the low-power and shutdown models are developed based on the internal events at-power model files.

15.6.2.3 Internal and External Events and Level 1 Probabilistic Safety Assessment

Internal and External Events (EE) and Level 1 PSA also include:

- Self-assessment results using NEI 17-01 guidance
- Basis for seismic PSA acceptability methodology for PSAR
- Basis for Chapter 15 event classifications as AOOs or Design Basis Accidents (DBA)

The internal and external hazards portion of the PSA analyzes radionuclide release accidents initiated during power and shutdown operation for the following hazards:

- Internal hazards
- Internal floods
- Internal fires
- Other internal events: PSA quantification is based on the results of screening analyses that identify potential significant contributors.

15.6.2.3.1 Internal Hazards

This section summarizes the screening for internal hazards for the BWRX-300 plant. Internal Hazards were screened for both consideration in the PSA and for inclusion in the fault list. In addition, potential non-reactor sources of radioactivity are screened in this section.

From IAEA SSG-64, "Protection against internal Hazards in the Design of Nuclear Power Plants," (Reference 15.6-11),

> "*Internal hazards are those hazards to the safety of the nuclear power plant that originate from within the site boundary and are associated with failures of facilities and activities that are under the control of the operating organization*."

This section documents the screening of internal hazards for the BWRX-300 Standard Plant Design. Because the BWRX-300 PSA is an all-hazards PSA, internal hazards are evaluated for potential PSA development. The BWRX-300 PSA includes guidance for screening external hazards and this screening approach can be adopted for screening internal hazards. The PSA Standard, ASME/ANS RA-S-1.1-2024, Sections 1 and 6, also provides several screening criteria.

The general approach screens internal hazards using the following steps:

1. Develop a list of candidate internal hazards.
2. Apply qualitative screening criteria to each of the internal hazards.
3. For those internal hazards remaining unscreened (i.e., pose a credible threat to nuclear safety) from Step 2, consider application of quantitative screening.
4. Any hazards left unscreened after Step 3 are retained as candidates for explicit treatment in the PSA.

In addition, on-site radioactivity sources are screened for potential treatment in the PSA.

This section on internal hazards is preliminary and will be updated as the conceptual design continues to be developed.

15.6.2.3.2 Scope

The internal hazard scope assessment applies screening criteria to all hazards originating within the site boundary. Several sources are consulted for developing the internal hazards list. A list of on-site radioactivity sources is considered for treatment in the PSA. This subsection does not assess external hazards, sabotage/terrorism, nor is it used in the screening of EEs.

15.6.2.3.3 Assumptions

1. Administrative controls are implemented to preclude compressed gas cylinders from becoming missiles in areas containing risk-significant mitigating equipment.

2. Valves are designed to prevent removable parts from becoming missiles in the event of failure in accordance with guidance in IAEA SSG-64.

3. Rotating equipment (excluding the main turbine) is designed such that potential failure-generated missiles are prevented from impacting risk-significant equipment through spatial or engineered means.

4. Administrative controls are placed to ensure stored combustibles are not collected in sufficient quantities to impact nuclear safety if ignited.

5. No risk-significant mitigating equipment resides in the Reactor Building (RB) Plant Services Area truck bay or the service bay area beneath the truck bay.

6. The on-site radwaste system does not contain radioactivity in sufficient form or quantity to pose a public health hazard to the level of a small or large release as defined by the PSA.

7. Dry casks containing spent fuel, when stored or handled outside the cask pit, are capable of passive cooling and are not vulnerable to potential hazards.

15.6.2.3.4 Hazard List

The preliminary list of BWRX-300 internal hazards is generated from industry guidelines, past studies, and a plant-specific review.

The internal hazards list of sources includes:

1. NEDC-33934P

2. ASME/ANS RA-S-1.1-2024. This source is an aggregation of hazards based on review of industry studies such as NUREG/CR-2300, NUREG-1407, IAEA SSG-3, NUREG/CR-5042, EPRI 1022997, EPRI 3002005287, and ASAMPSA_E List of External Hazards.

3. IAEA SSG-3 “Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants”

4. IAEA SSG-64 “Protection Against Internal Hazards in the Design of Nuclear Power Plants”

The initial hazards screening results are provided in Table 15.6-2 and Table 15.6-3.

15.6.2.3.5 Qualitative Screening

The following criteria are provided for qualitative screening of external hazards. Consideration of internal hazards is not fundamentally different than external hazards from a risk-impact

perspective; therefore, it is reasonable to apply these same criteria in the screening of internal hazards:

1. The event is of equal or lesser damage potential than events for which the plant is designed. This requires an evaluation of the plant design basis to estimate the resistance to a particular (internal) event.

2. The event has a significantly lower mean frequency of occurrence than similar events included in the PSA and could not result in worse consequences in those events.

3. The event cannot occur close enough to the plant to affect it.

4. The event is included in the definition of another event.

5. The event frequency is sufficiently low in compared with the probabilistic limits defined for the release category frequencies so that it does not need to be included in the PSA.

6. The event does not result in a plant trip (manual or automatic) or require a plant shutdown.

7. The event develops slowly (DSA, and PSA fault sequence figures show event progressions), and it is shown there is demonstrably conservative time margin available to eliminate the source or to provide adequate response.

The following events were assessed qualitatively:

- Heavy Load Drop
- Release of chemicals from on-site storage
- Turbine-generated missiles
- Other internally generated missiles from pressure vessels, valve failures, control rods
- High speed rating equipment (e.g., fan blades, turbines, pump impellers, fly wheels, coupling bolts)

Internal fires are treated in the same manner as EE for BWRX-300 and are thus “screened-in” as part of this analysis alongside external hazards.

Most explosion sources are within the Internal Fires PSA scope. Ignition of the following components with explosion potential is considered:

- Batteries
- Diesel generators
- Switchgear
- Hydrogen tanks
- Miscellaneous hydrogen sources
- Offgas/hydrogen recombiner
- Transformers
- Transient combustibles

- Boilers
- Turbine auxiliaries

For stored chemicals and combustibles, it is assumed that administrative controls are placed to ensure they cannot ignite or react in sufficient quantities to impact nuclear safety.

Explosions are qualitatively screened based on Criteria 1.

The qualitative assessment of the above hazards is summarized in Table 15.6-2 where collapse of structures, pipe whip, jet effects, and internal flooding were also assessed.

15.6.2.3.6 Quantitative Screening

The quantitative screening criteria for external hazards is also used for internal hazards.

To determine the CDF from turbine-generated missiles, the IE frequency, and Conditional Core Damage Probability (CCDP) are determined. An example of the process considers the following:

The CCDP was calculated by assuming an LOPP and failure of diesel generators and was estimated to be approximately 3.4E-7. Multiplying the CCDP by the IE frequency, 1E-4/yr, yields a conservative CDF of 3.4E-11/yr. The 1E-4/yr frequency is conservative because it does not credit the fractional probability of the turbine missile striking and then damaging the target equipment.

Turbine-generated missile hazards meet the screening criteria, so this hazard screens quantitatively.

Internal flooding hazards are treated explicitly in the PSA. No quantitative screening is performed for internal flooding.

The quantitative screening criteria from ASME/ANS RA-S-1.4-2021, “Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants,” (Reference 15.6-38) are applied. A hazard screens quantitatively if:

1. Based on absolute risk contribution, an ES family subject to screening does not exceed the selected risk significance criteria and has mean occurrence frequencies less than 1E-7/plant-year, as estimated using a bounding or demonstrably conservative analysis.

2. The total contribution of all screened out ES families does not exceed 1% of the cumulative risk targets included in the absolute risk significance criteria.

The results of the Internal Hazard and External Hazard screening analyses are presented, respectively, in Table 15.6-2 and Table 15.6-3.

15.6.2.3.7 Hazard Combinations

Combinations of External Hazards and Correlated Hazards

Occurrences of natural and human-induced EE and their credible combinations that could affect the safety of installation need to be identified and included in the safety assessment.

The combinations of external hazards or correlated hazards (referred to below as combinations of hazards) are analyzed for those hazards that have the potential to combine to have a larger impact than the individual hazards (e.g., heavy winds causing the loss of the external grid, snowstorm drifts blocking the diesel generator combustion air intakes and frazil ice blocking water cooling).

15.6.2.3.8 Categories of Hazard Combinations

An initial event is an external or internal hazard that results in at least one further external or internal hazard. Event chains of three or more hazards subsequent to each other are possible.

For the assessment of external hazards combinations, the initial event impacting a nuclear installation is either a natural hazard such as an earthquake or an external flooding, or a human-induced one, such as an accidental aircraft crash or an explosion pressure wave (blast) originating from outside the plant boundary.

The following hazard combinations involving external hazards are possible to occur:

1. Natural hazard and consequential external hazard covering:

    a. Natural hazard and consequential natural hazard (e.g., earthquake and consequential tsunami)

    b. Natural hazard and consequential human-induced hazard (e.g., extreme wind and consequential aircraft or helicopter crash)

    c. Natural hazard and consequential internal hazard (seismic and fire)

    d. Human-induced hazard and consequential internal hazard (aircraft crash and internal explosion)

2. Correlated Hazards: Combinations of correlated hazards are defined as:

    a. Two or more external and/or internal hazards occur as a result of a common cause initiator (CCI). The common cause can be any anticipated event including an external hazard or might be due to an unanticipated dependency. The two or more hazards correlated by this common cause could occur simultaneously.

3. The following hazard combinations involving external hazards may occur:

    a. A natural hazard or hazard source as common cause for other external and/or internal hazards and/or internal events. Typical examples include an earthquake as the common cause for landslide, external fire, internal fire, internal explosion, and station blackout, as potentially correlated events, or tsunami as the common cause for external flooding, internal flooding, and internal fire as potentially correlated events.

    b. A human-induced hazard (or hazard source) as common cause for other human-induced and/or internal hazards. Examples are an accidental aircraft crash of a large commercial passenger aircraft as the common cause for an external explosion (outside the plant boundary), a plant internal fire from an internal explosion, and a heavy load drop correlated by the aircraft crash, or an electromagnetic interference outside the plant boundary as the common cause for a plant internal electromagnetic interference and internal fire as the two correlated hazards.

4. Independent Hazards: Combinations of unrelated (or independent) events are defined as:

    a. An initial event, e.g., an external or internal hazard, occurs independently from, but simultaneously with another hazard without any common cause.

        The term ‘simultaneously’ in this case does not mean that the hazards occur exactly at the same time but rather that the second hazard occurs before the consequences of the previous hazards have been completely mitigated.

5. The following event combinations involving external hazards may occur:

    a. Natural hazards and independently occurring natural hazards. Typical examples include longer duration riverine flooding and independently occurring extreme weather conditions, such as extreme wind.

    b. Natural hazard and independently occurring human-induced hazard. Examples are a longer duration hydrological hazard and an independently occurring industry accident (e.g., an explosion or release of hazardous substances).

Identification and Screening of Hazard Combinations

Event combinations used to develop a list of potential combined hazards is initially based on the identified individual external hazards provided in Table 15.6-3. The number of individual external hazards identified to occur at a given plant location is already high. Building combinations of all possible individual external hazards result in a number of combined hazards, which is not manageable without applying a systematic approach.

Identification of consequential hazards begins with the unscreened single hazards and determining those hazards that may result from the initial hazard. This can be performed using a matrix approach, listing the hazards graphically and showing the combinations of two hazards (so-called first order combinations) where the possible combinations are identified. The

combined hazards can be further reviewed based on this initial identification to determine additional hazard combinations possible (second-order combinations) or additional identified combinations.

For combinations of correlated hazards, it is not directly possible to build a matrix for these combinations. The analyst starts the combined hazard screening by identifying potential common causes for those individual external (single) hazards not screened out for the plant/site and builds a tree-type structure (or similar) for all hazards correlated by such common causes representing the roots of the branches. With such a tree-type structure, higher order combinations of correlated hazards can be built and undergo a screening process.

For the third category of combinations of independent hazards, the same two-dimensional matrix structure is used. The possibility of an event combination is demonstrated by a color change or other designation in the corresponding matrix field. The matrix involves the identification of hazards that may impact the plant/site for a time period longer than 3 days. The likelihood is included in the matrix based on the duration of the first hazard and the likelihood of the second hazard occurring during that time period.

Combined events are identified by analyzing the correlations and the effects on the plant. The analysis of possible correlations (dependency) between events is made by assessing the physical bases of the phenomena, observed data, actual operating events, and general knowledge of local conditions.

Expert judgment and rough quantitative analysis are used for estimating correlations. The observed data for intensities relevant to EEs PSA is sufficient for order of magnitude correlation estimates.

For example, high wind often results in high water level and organic material in water (e.g., seaweeds, kelp, algae, Asiatic clams, mussels). The combination may simultaneously endanger offsite power and the diesel generator cooling water intake; therefore, it is analyzed as a combined event.

Extreme wind velocities are measured in winter and could be associated with snowfall resulting in possibility of simultaneous LOPP and loss of the diesel generators due to snowdrift blockage of combustion air intake.

Simultaneous high air and seawater temperature could also endanger plant equipment, (e.g., instrument room cooling at units with diverse cooling system heat sinks).

Identified hazard combinations are screened similar to single hazards using the process discussed previously.

Evaluation of Risks Associated with Unscreened External Hazards

For unscreened external hazards and identified combinations of external hazards, a similar model development and quantification processes described previously are adapted to evaluate external hazard risks. The process utilized for hazards combinations is discussed below.

Several external hazards from transportation and nearby facilities are evaluated. These external hazards are evaluated with bounding assumptions and can be modified with plant-specific inputs:

- Airports and airways hazards
- Industrial accidents
- Pipeline accidents
- Hydrogen storage failures
- Transportation accidents

Process for Selection, Screening and Analysis of Combinations of Hazards

Figure 15.6-2 provides an overall process for selecting, screening, and analyzing combinations of hazards. The quantification of the hazards is performed similar to the quantification of the individual hazards with consideration of the combined hazard on the SSCs fragilities, the likelihood of the combined hazards, and the inclusion of these factors in the PSA model.

The process for the specific modeling depends on the hazards, and the supporting single hazard PSA modeling. During the PSA quantification, the quantification steps may be slightly different if the combined hazards PSA results are calculated together with or separately from the single hazards PSA model quantification. For example, if a seismically induced flooding event PSA model is developed and quantified, it can be quantified as a part of the Seismic PSA or separately where only the accident sequences involving seismically induced flooding are quantified. There are no requirements in ASME/ANS RA-S-1.1-2024 to quantify single and combined hazards separately or together. There are advantages of long-term model maintenance in developing and analyzing the combined hazards using a single model. The quantification of each single and combined hazard avoids double counting the results. For example, if the CDF of a single hazard PSA model such as a seismic PSA provides a value of 1E-05 / reactor year, and the combined hazards PSA model for the combination of consequential hazards seismically induced internal flooding provides a CDF of 5E-06/reactor year, there is likely some double counting if the final results are equal to the sum of the two CDF values. In this case, a realistic assessment of the single hazard PSA is based on accident sequences where a flooding event does not occur.

The quantification of the combined hazard PSA is performed in a similar manner as the individual hazard PSA, accounting for the conditional probability/frequency of the combined hazard magnitudes. As previously discussed, this can be applied in a variety of approaches, but often specified magnitude ranges or bins are used. The analysis of hazard combinations magnitude ranges or bins include variation of the various hazards and the conditional probability of each hazard in combination with the others. The PSA modeling for low-power and shutdown states may also need to consider whether the reactor shuts down as a result of a slow-moving hazard or hazard combination, resulting in the plant being in a specific plant operational state when the hazard combination occurs (e.g., such as a high magnitude hurricane causing high wind and rain).

Example: Assuming a high wind PSA is performed, and it is determined that two buildings are potentially impacted by high winds, with building A failing at approximately 250 kph and building

B failing at 310 kph, based on the fragility analysis. Initial analysis shows high winds to be potentially significant and it should not be analyzed in a conservative manner. As a result, the following wind hazard analysis bins are created:

- Bin 1 < 235 kph
- Bin 2 = 235 kph to 265 kph
- Bin 3 = 265 kph to 295 kph
- Bin 4 = 295 kph to 325 kph
- Bin 5 > 325 kph
- (For mph conversion, multiply kph by 6.21 E-01)

When the analysis of a combined hazard including high wind and heavy rain (not defined for this example) is considered, the combined fragility for a high wind and heavy rain event (in this example) is shown that the building fragilities change to where damage can occur for each building at approximately 20% below the above estimates (e.g., 200 and 248 kph, respectively). As a result, the combined hazard analysis bins are modified by lowering the wind speeds above by approximately 20%. It may be found, however, that the frequency of the combined hazard is small enough that the number of analysis bins can be reduced. On the other hand, if the heavy rain is analyzed to have a range of impacts, the number of combined hazard analysis bins could increase from the above example, if an initial conservative approach results in risk-significant results.

Internal Hazard Combinations

Heavy load drop-induced floods and structure damage are addressed as part of the heavy loads analysis in the low-power/shutdown PSA evaluation.

Turbine-generator missile-induced fires/explosions are addressed in the fire PSA. Pipe whip and jet effects are addressed in the internal flooding PSA.

15.6.2.3.9 Other Considerations for External Hazards

Introduction

Site-specific external hazards are mapped and the design bases of the relevant SSCs are verified. This section describes other considerations for external hazards, which exclude those included in previous sections (i.e., high winds and seismic hazards).

AMSE/ANS RA-S-1.1-2024 includes a hazard screening section (Part 6) and Requirements for Other Hazards (Part 9). The enhanced standard, including the guidance in the non-mandatory appendix, provides requirements and guidance on performance of a detailed hazard screening and PSA performance. The changes in Parts 6 and 9 are significant changes in the PSA Standard and reflect the advancements in external hazards screening PSA in the last decade. Included in the standard requirements is the consideration of correlated hazards and hazard

effects, such as the potential for local flooding associated with high winds or the impact of high wind-driven rain. This standard is utilized for both the screening of potential hazards and the analysis of any unscreened hazards.

Methodology

External Hazards Screening

A questionnaire from the Nuclear Energy Agency Committee on the Safety of Nuclear Installations Nuclear Energy Agency (NEA)/CSNI/R, "Probabilistic Safety Analysis of Other External Events Other Than Earthquake," (Reference 15.6-12) has been sent to the Organization for Economic Cooperation (OECD)/NEA member countries for non-seismic EE. The objective of the task was to review the methods for risk analysis of offsite EE other than earthquake as well as the results and the insights developed in these analyses to present a basis for advances in the area.

In Appendix 1 of the NEA/CSNI/R, a table is provided for List and Characterization of EE. This list includes about 50 EE, which are comprehensive and consistent with the combined ANS/ASME PSA Standard and IAEA Level 1 PSA guideline. Therefore, this list is used for the identification of external hazards that are applicable to the plant-specific PSA. The following set of qualitative screening criteria are used to limit the number of external hazards to be analyzed:

- The hazard cannot physically impact the plant or plant operations (e.g., it cannot occur close enough to the plant to affect it).
- The hazard does not result in a plant trip (manual or automatic) or require a plant shutdown.
- The hazard is included in the definition of another hazard that is included in the PSA.
- The hazard could not result in worse effects to the plant or plant operations than another hazard that has a significantly higher frequency.
- The hazard is slow in developing, and it is shown that there is a demonstrably conservative estimate of time margin available to eliminate the source of the threat or to provide an adequate response.

15.6.3 Level 1 Probabilistic Safety Assessment

The Level 1 PSA assesses the CDF. ASME/ANS RA-S-1.1-2024 states requirements for a Level 1 PSA of internal and external hazards while At-Power for the evaluation of CDF.

Figure 15.6-1 shows the principal steps in the PSA. The BWRX-300 Level 1 internal events analysis includes the following steps shown on that figure:

- IEs Analysis
- Accident Sequence Analysis
- Success Criteria Analysis
- Systems Analysis
- Uncertainty Analysis

- Data Analysis
- Human Reliability Analysis
- Model Quantification

15.6.3.1 Initiating Events Analysis

One of the first and basic steps in a PSA is the identification and quantification of the IEs to be used in the sequence analysis. IEs have historically been broadly classified as either “internal” or “external” hazards. An IE may result from human causes, equipment failure from causes internal to the plant (e.g., hardware faults, floods, or fires) or external to the plant (e.g., earthquakes or high winds), or combinations thereof. The focus of this section is on internal IEs. The Internal IE PSA assesses events that are caused by systems or components located within the site structures. An exception of this is the LOPP which is analyzed within the internal events. Internal and external hazard induced IEs (e.g., seismic events, internal floods) and IEs during shutdown excluding sabotage are discussed in other sections in this report, including:

1. Fire PSA (internal fire events)
2. Internal Flooding PSA (internal flooding events)
3. High Wind Risk (including tornado events)
4. Seismic PSA (seismic events)
5. Shutdown Risk (shutdown IEs for internal events)
6. Spent Fuel Damage

Spent Fuel Damage evaluation is required for the plant-specific BWRX-300 PSA. For accidents, where the spent fuel is damaged outside of the reactor core, the term Spent Fuel Damage is applied. The associated risk metric is Spent Fuel Damage frequency. In particular, the Spent Fuel Damage frequency could be an important contributor to release risk. Traditionally, the term “fuel damage” represents damage to the fuel outside of the reactor vessel, while “core damage” is used for damage inside the vessel.

15.6.3.1.1 Methodology

Initiating Event Identification

To develop a comprehensive accounting of postulated IEs, a systematic approach is used to identify events that challenge normal plant operation and require successful mitigation to prevent radionuclide release. IAEA-TECDOC-1804 “Attributes of Full Scope Level 1 Probabilistic Safety Analysis,” (Reference 15.6-13) provides the following IE definition: “an event which could directly lead to core damage or challenges normal plant operation and requires successful mitigation to prevent core damage.”

A comprehensive identification of IEs includes a review of existing BWR PSAs, including the previous BWRs, and generic sources. Generic lists of IEs are included in multiple reference documents such as IAEA-TECDOC-719, NUREG/CR-5750, “Rates of Initiating Events at U.S. Nuclear Power Plants: 1987-1995,” (Reference 15.6-14), and NUREG/CR-6928 “Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants,” (Reference 15.6-15).

IEs identified typically include transients of various types, LOCAs, and support system initiators. IEs under low-power conditions are also identified.

All system failures that may bring disturbance in plant operation (e.g., front-line and support systems) are reviewed, except for those already identified as the source of IEs based on other analyses.

In addition, planned and unplanned manual shutdowns that seldom place demands on any standby safety equipment are treated as IEs because of their high frequency and because they represent changes in operating states that result in the available equipment demands to reach a safe shutdown condition.

Support system initiators are defined as unplanned normal shutdown events resulting from loss of a support system (with dependent failures).

Related to support system initiators are CCIs, which are equipment failures that cause an IE and simultaneously degrade equipment credited for mitigating the IE (e.g., failure of balance-of-plant equipment that could cause a plant trip and preclude use of the normal power conversion system). CCIs must be systematically identified and treated in the PSA.

Human error induced IEs are also considered for these analyses.

An example list of the general categories of internal IEs are shown in Table 15.6-4, which is taken from IAEA-TECDOC-719. Note that this list is supplemented with the other generic sources discussed above to develop a comprehensive list of IEs.

Screening of Initiating Events Candidates

The requirements for screening potential IE candidates from the PSA are discussed in IAEA-TECODC-1804 and the ASME/ANS RA-S-1.1-2024.

Based on the above references, the IE candidates identified are excluded from further consideration if they meet one of the following criteria.

- The event does not lead to the IE as defined in the PSA.
- The event does not correspond to the scope of the PSA.

- The frequency of the event is less than the truncation value related to the accident sequence frequency, and the event does not involve an Interfacing Systems LOCA (ISLOCA), containment bypass, or Reactor Pressure Vessel (RPV) rupture. For these events, the truncation value is at least one order of magnitude lower than the truncation value accepted in the PSA.
- The resulting reactor shutdown during At-Power Plant Operation State (POS) is not an immediate occurrence. That is, the event does not require the plant to transfer to shut down conditions until a defined amount of time has elapsed, the condition is detectable before plant systems are required to respond, and there is a high degree of certainty (based on supporting calculations), that the condition can be detected and corrected before normal plant operation is curtailed (either administratively or automatically).

Based on the above criteria, the IE screening is limited to those events that do not lead to an IE as defined in the PSA, and those events where a shutdown occurs prior to the conditions being corrected with certainty. Screening is carefully and conservatively applied, especially early in design where the impact for an event may not be fully understood without design details fully developed. Screening of external hazards is described separately in previous sections in this chapter.

Initiating Event Grouping

Each IE group is represented by the limiting IE or the hypothetical worst-case IE. For example, the IE group for water level decreasing events is represented by a hypothetical event with the feedwater system and main condenser initially not available.

Data Collection for Initiating Events Frequencies

In this step, generic data in the public domain is collected and reviewed for applicability and use for the grouped IEs. The IE data is reviewed for applicability for the BWRX-300. NUREG/CR-6928 contains U.S. data used for most BWR PSAs. The source data includes estimates for a range of events including rare events such as large LOCA and excessive LOCA. NUREG/CR-6928 also includes uncertainty estimates for each IE.

Because the BWRX-300 reactor vessel and attached piping are unique with respect to the existing BWR fleet (whose experience composes the basis for estimated LOCA frequencies), a detailed LOCA evaluation was performed for the BWRX-300 PSA.

Plant-specific IEs, such as those developed using Failure Modes and Effects Analysis (FMEAs) or fault tree IEs, are typically estimated using component-based data such as the component data in NUREG/CR-6928. For example, pipe-length based LOCA and Break Outside Containment IEs can be developed using generic pipe break frequencies and the design-specific pipe configurations. In the BWRX-300 design the Shutdown Cooling System (SDC) and Reactor Water Cleanup System (CUW) are designed to full RCS pressure, therefore an Interfacing System LOCA in these systems is considered unlikely. As a result, no Interfacing System LOCA pathways are included as IEs in the BWRX-300 PSA.

When analyzing IE frequencies for a specific site, site-specific data collection may be necessary. For example, a site-specific LOPP frequency and LOPP recovery curve is generally performed for sites where LOPP is risk significant.

Frequency Quantification

Most IE frequencies for the BWRX-300 are assigned based on NUREG/CR-6928 results, which are based on U.S. nuclear power plant operational experience. Where these sources do not provide information on a particular initiator, the source of the frequency for the initiator is provided in the appropriate section.

IE frequencies are calculated with units per reactor operating year. A recovery rule is applied that puts them in units per calendar year. When analyzing IEs for Low-Power and Shutdown (LPSD), the IE frequencies are input based on reactor “operating” year. Then, a recovery is appended that corrects for time in each POS based on plant availability factor.

15.6.3.2 Accident Sequence Analysis

The event trees developed for BWRX-300 are based on the IEs developed in the IE analysis. The event tree models include the set of safety functions needed to mitigate each IE. The objectives of accident sequence analysis are to ensure that the response of the plant’s systems and operators to an IE is reflected in the assessment of risk in such a way that:

- Significant operator actions, mitigation systems, and phenomena that influence or determine the course of sequences are appropriately included in the accident sequence model and sequence definition.
- Plant-specific dependencies due to IEs, human interfaces, functional dependencies, environmental, and spatial impact, and CCFs are reflected in the accident sequence structure.
- The individual function successes, mission times, and time windows for operator actions for each critical safety function modeled in the accident sequences reflects the success criteria evaluated in accordance with the attributes of Acceptance Criteria Methodology in Subsection 15.6.3.2.1.
- End states are clearly defined to be either core/fuel damage on one or more reactor units or successful prevention with the capability to support the interface between Level 1 and Level 2 PSA.
- The accident sequences are defined for the selected set of IEs, POSs, and times that a POS can occur.

The resulting accident sequence analysis addresses the following:

- General assumptions relating to all event tree development
- Sequence end states

- Definition of the type of models to be produced (e.g., small event trees and large fault trees) and level the event tree headings are to be defined (safety function, system, train):
  - For the BWRX-300, the event trees are generally small event trees developed using the CAFTA software. Event tree headings are largely based on top-level system functions, with functional fault trees added to connect the models to system and train failure events.
- Requirements for the development of ES diagrams
- Interface between the IE, success criteria, human reliability, system modeling and data analysis tasks

The accident sequence analysis also meets ASME/ANS RA-S-1.1-2024.

15.6.3.2.1 Methodology

Acceptance Criteria

Ahead of the function overall success criteria, a formal definition of core damage is defined and justified.

Core damage is defined in terms of physical processes and phenomena and failure mechanisms. The accident analyses and past experiences for similar plants are used to derive the definition of core damage. The physical plant parameters (e.g., highest node temperature) and associated acceptance criteria or limit values (e.g., percentage of cladding thickness oxidized) to be used in determining core damage are defined. The parameters are selected in such a way that the determination of core or fuel damage is as realistic as practical and consistent with current best practices and knowledge with a sufficient margin of the computer codes, sophistication of models, and uncertainties.

The success criteria supporting each event tree top event and safety function are the minimum requirements necessary for key safety functions to achieve safe, stable conditions, (i.e., to protect the fuel and prevent release of radionuclides to the environment). Safe conditions are determined by the ability to meet the following key safety function success criteria. Stable conditions are determined by the ability to maintain each key safety function for long-term operation:

- Reactivity Control: To achieve subcriticality and maintain the reactor in a subcritical state (combined with adequate core cooing and/or containment heat removal functions if appropriate) so that core damage preceding containment failure is avoided.
- Core Cooling: Adequate core cooling is provided to prevent core damage and the reactor is maintained in a safe-and-stable condition. Parameters for the overall success criteria of core cooling are peak cladding temperature and cladding oxidation.
- Containment System Integrity: Ensure that the containment pressure does not exceed the ultimate containment capacity, and containment isolation occurs when required.

15.6.3.2.2 Accident Sequence Development

The role of an event tree is graphical expression of IE groups, success criteria, and sequence end points. The basic concept is to prepare an event tree for each IE group defined in each IE analysis task.

IE group is studied and organized based on IAEA-TECDOC-1804. “For each IE group for internal events, internal hazards, and external hazards for each POS the accident progression for all sequences is identified and justified. For each IE group the accident sequence models are developed. Accident sequence models explicitly address realistic plant behavior in response to IE in terms of normal plant systems operation, operator actions, and mitigation systems that support the key safety functions necessary to achieve a stable safe state.”

Modeling accident sequence progression involves the following:

- Step 1: Analyze the progression of each accident sequence
- Step 2: Incorporate site-specific procedures into accident sequences
- Step 3: Develop accident sequence models

Based on success criteria analysis, each accident sequence is analyzed until one of the defined end states is reached. The end states are either non-successful (e.g., core damage) or a safe stable state, where long-term stability is ensured. Plant damage states, accident phenomena, and containment systems responses are used in the containment release analysis.

The accident progression analysis is performed for each sequence until a safe, stable, long-term condition is reached or until core/fuel damage occurs, to determine if there is a cliff-edge effect beyond the mission time used in the PSA. The safe stable state for each end state is defined to ensure:

- The core remains subcritical over the duration of the event
- Equilibrium conditions are obtained, and the conditions are trending in a safe direction
- Inventories are not lost within the defined mission time, or within a short time frame beyond the defined mission time
- Containment cooling is provided, with temperatures stabilized to the level where the heat removal rate and the decay heat are balanced

Possible long-term failure of a function may be recoverable and modeled in the PSA. The time available for recovery actions is extended to the degree that a wide variety of recovery actions are possible, and the recovery failure is considered negligible. When this is not the case (e.g., the recovery action does not have a negligible failure probability), then the recovery action is considered for the accident sequence analysis, and the mission time is extended to include the modeling of the recovery action timing.

Supporting analysis is performed to justify the accident progression for each accident sequence. However, if the accident sequence is not risk-significant, conservative assumptions can be used in place of supporting analysis (e.g., thermal-hydraulic, fractural mechanics, reactivity analysis). If risk-significant insights cannot be achieved with the use of conservative assumptions, more efforts need to be taken to remove conservatism with appropriate justification.

ATWS sequences are not grouped into a single event tree. Rather, each IE group has a sequence with scram failure with a transfer to a unique ATWS tree. Those ATWS sequences go either to the general transient ATWS tree, to the LOPP ATWS tree, or are assumed to result in core damage.

For each key safety function its dependence on the success or failure of preceding functions and the impact on accident progression are addressed. This is called a functional dependency that affects the availability of subsequent mitigation features.

To exclude the systems with dependent failures, detailed IE to front-line/support systems dependency matrices are developed. The examples of dependencies to be considered are:

- Detailed dependent failures from transients (e.g., Feedwater (FW) system disabled by loss of FW)
- Detailed dependent failure from support system initiators, all injection systems disabled by CCF of Class 1 DC power
- Adverse impacts on mitigation systems (e.g., loss of injection path due to location specific LOCAs, Interfacing System LOCA/Break Outside Containment induced flooding, Reactor Core Isolation System operability given loss of containment heat removal)

When accident progression and associated success criteria notably change after failure of specific functions, a linked event tree(s) is prepared. Some examples of possible link event trees include:

- Consequential LOPP link trees for transients with consequential LOPP
- Consequential LOCA link trees for transients with failure to open Safety/Relief Valves or failure to reclose multiple Safety/Relief Valves

Dependencies between event trees are inherently preserved by the underlying system fault trees that share common Basic Events (BE) and account for equipment that has already failed.

The front-line mitigating systems modeled to mitigate the IEs are listed in Subsection 15.6.9.1.1.

15.6.3.2.3 End States of Event Sequences

The event trees identify the potential sequences that can lead to radionuclide release. Many of the sequences have common characteristics with respect to the challenge on the containment fission product barrier. These sequences are grouped into damage classes that are analyzed in the release portion of the PSA. The end states of the ESs developed for the BWRX-300 PSA are defined to facilitate the Containment Performance Analysis and provide the link between core damage and a release category.

ESs were then qualitatively analyzed to determine which sequences lead to core damage end states. The core damage sequences are grouped together based upon the overall challenge to the containment barrier. They are defined as follows:

- OK: The core is successfully cooled, and the containment is intact. There is no core damage in these events.
- CD I: The containment is intact when core damage occurs and the RPV is at low (or controlled) pressure.
- CD II: The containment is breached, either due to over pressurization or venting, while the core is successfully cooled. Core damage results due to failure of long-term heat removal to maintain core cooling.
- CD III: The containment is intact when core damage occurs and there is high RPV pressure at the time of RPV failure.
- CD IV: Core damage results from an accident sequence with an initial failure of effective reactivity control (e.g., ATWS due to failure of Reactor Protection System (RPS), control rod binding). This has the potential to affect the containment in a more severe manner than the CD I and CD III because more energy is deposited into the containment prior to RPV failure. All CD IV end states could be treated as CD I or CD III (depending on the RPV pressure) without affecting the results of the containment analysis. This end state has been retained in this scoping analysis to more easily allow for sensitivity analyses related to reactivity control.
- CD V: The containment is bypassed at the time of core damage.
- CD VR: Core damage occurs due to RPV ruptures in the lower or mid-vessel regions.

Sequences where core damage is expected to occur very late are appended with an ‘L’ to signify it as late. An example of where this could occur is when Control Rod Drive (CRD) flow is being relied upon to avoid core damage, but eventually fails due to inventory depletion.

15.6.3.3 Success Criteria Analysis

The main objective of success criteria formulation is to determine for given IEs what represents a successful or unsuccessful plant response and to translate this information into detailed plant system and operator action success criteria accomplished for At Power or Low Power Shutdown operation. The BWRX-300 success criteria analysis is integrated with the accident sequence analysis.

Ahead of the function overall success criteria, a formal definition of core damage is defined and justified.

Success criteria are defined for the BWRX-300 on several different levels. The highest level of success criteria development is to determine what constitutes mitigation of any IE (i.e., prevention of core damage). This requires explicit qualitative description of core damage (e.g., gross fuel/cladding degradation due to failure to adequately remove fission and/or decay heat) and, from this qualitative description, a parametric description (e.g., peak core nodal temperature).

On a lower level, success criteria are developed to support the functions required to mitigate core damage. This requires an assessment of what front-line systems can provide which mitigating functions and determining their requirement for fulfilling those functions. Best-estimate analyses are employed using computer codes or other acceptable calculations to perform this task. Examples include determining how many trains of a certain system acceptably performs a function for a given sub-group of IEs.

Following from the functional success criteria, supporting success criteria are determined. This would include timing calculations for mission times and human actions, inventory availability determination, and any other system or train-level success criteria development.

15.6.3.3.1 Success Criteria Analysis Methodology

The definition of core or fuel damage is a key aspect of the determination of function success. Core damage is defined classically in terms of two key physical parameters:

- First, the peak cladding temperature reached during the accident sequence is used as a core damage parameter. An appropriate acceptance criterion (e.g., 1200 °C) is defined and justified in the analysis, as well as the appropriate credible margin to use in thermal-hydraulic calculations.
- Second, the total oxidation of the fuel cladding is also used to determine core damage. An appropriate acceptance criterion (e.g., pre-oxidation cladding thickness) is defined and justified. Like the peak cladding temperature criterion, an appropriate level of margin is also assigned for use in thermal-hydraulic calculations.

The above definition and criteria are consistent with the definition in ASME/ANS RA-S-1.1-2024 as “uncovery and heatup of the reactor core to the point at which prolonged oxidation and severe fuel damage are anticipated and involving enough of the core, if released, to result in offsite public health effects.”

The success criteria for a function define the minimum performance that differentiates the end state when that function succeeds from the end state reached when the function fails. With the parameters that determine core damage and decay heat levels defined, specific success criteria are added to define the necessary capability of SSCs and operator actions in accident sequences.

A mission time to achieve a stable end state after an IE for the accident sequences is determined. As a first approach, a general mission time of 24 hours is assumed for BEs. A longer mission time is used if needed for some accident sequences to achieve stable plant conditions.

Success criteria analysis also considers the available inventories needed to support each system or function. The available inventories of fuel, water or air in tanks are compared with those required to support each success criterion and the model reflects the results of this comparison.

Analysis models and computer codes are used that have sufficient capability to model the conditions and phenomena to determine success criteria. The plant model and parameters used for thermal-hydraulic analyses are established in a way that provides sufficient resolution and reflects the actual design and operational features of the plant.

In addition to supporting post-initiator success criteria, there are analyses required to support the grouping of the IEs themselves, namely for LOCA IEs. There are several examples of the types of analyses needed to group LOCA IEs:

- Analyses to determine whether a line is large enough to cause a large LOCA
- Analyses to determine whether a line is large enough to cause medium LOCA

Success criteria may utilize expert judgment to assess the conditions or response of systems, structures, and components in situations when there is a lack of available information, knowledge, or analytical methods upon which a prediction can be based if it can be demonstrated that the variability and uncertainty potentially inherent in the assessment does not significantly impact the PSA models and results. The use of expert judgment is discussed in ASME/ANS RA-S-1.1-2024.

For each accident sequence condition, the applicability ranges and conditions listed in the documentation for each analysis code are reviewed. An assessment is performed to ensure that all accident sequences are within the applicability of the codes that are used to determine success criteria. The use of all codes is justified based on this comparison.

Each success criterion in the BWRX-300 success criteria analysis is based on either a plant design parameter or a thermal-hydraulic calculation. When a success criterion is used to characterize a range of conditions, the limiting parameters are chosen to represent all cases. In addition, representative thermal-hydraulic analyses and offsite dose consequence analyses are run to validate the appropriateness of the success criteria.

15.6.3.4 System Analysis

The system analysis is performed for each plant system represented in the IE analysis and accident sequence analysis for each POS in such a way that:

1. Each safety function in accident sequence models and system models are developed that account for the success criteria.

2. System-level success criteria, mission times, time windows for operator actions, different initial system alignments and assumptions provide the basis for the system logic models reflected in the model. A reasonably complete set of system failure and unavailability modes for each system is represented.

3. Human errors and operator actions that influence the system unavailability or the system contribution to accident sequences are identified for development as part of the HRA element.

4. Intra-system dependencies and inter-system dependencies including functional, human, phenomenological, and CCFs that influence system unavailability or the system contribution to accident sequence frequencies are identified.

Other objectives include:

- Considering credible failure modes
- Modeling each failure mode impact on system performance
- Including support system failure modes in the front-line system fault trees
- Creating linked fault tree models that allow resolution
- Creating fault tree models where solutions (i.e., cutsets) are easily understood

The system analyst builds logic for each function identified within a system top event. The top event links with the sequence logic, or with other system logic.

15.6.3.4.1 Methodology

The BWRX-300 system analysis includes systems modeled in the BWRX-300 PSA, which correspond to the functional headings described in the event trees plus any support systems needed to accomplish those functions. The system modeling methodology follows the guidance in NUREG/CR-2300 “PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants,” (Reference 15.6-16), NUREG/CR-4550 “Analysis of Core Damage Frequency: Surry, Unit 1 Internal Events Appendices,” (Reference 15.6-17), and NUREG/CR-4551 “Evaluation of Severe Accident Risks: Methodology for the Containment, Source Term, Consequence, and Risk Integration Analyses,” (Reference 15.6-18), NUREG-0492, “Fault Tree Handbook,” (Reference 15.6-19) and the “Overview of the PSA Process and Basic PSA Techniques,” (Reference 15.6-20) provide additional guidance on fault tree modeling, current information for the system studies, and latest results.

The system descriptions contain the following typical items:

- Functional Description
- Assumptions
- System Description
- Hardware Configuration
- Equipment Alignments
- System Operation
- Component Location
- Automatic and Manual Control
- Safety Actuation System
- System Testing
- System Maintenance
- Common Cause Failures
- Fault Tree Analysis

- Top Event Definitions
- Fault Tree Description
- Human Interaction
- Special Events
- Results of Fault Tree Analysis
- PSA Insights and Design Requirements

Key to the system analysis:

- Guidance for the definition/identification of system boundaries and interfaces with other systems. This includes the identification of front-line/support system and of support/support system dependencies
- Process for determining when a fault tree model is developed for a system or whether the system is modeled by a single basic event or a small number of modules (for groups of components within the system)
- Approach to define system boundaries
- Identification of components and component failure modes relevant to the given system failure event (fault tree top events)
- Approach used to define component boundaries in the mechanical, I&C and electrical subsystems
- Approach applied for the inclusion of test and maintenance activities into the system model, and the way it is to be documented
- Way in which the human failure events and CCFs are to be modeled
- Generally applicable modeling assumptions (e.g., those related to inclusion or exclusion of passive components, criteria for inclusion or exclusion of diversion paths)
- Event naming scheme

For CCF analysis, this system modeling addresses the following:

- The identification of component groups susceptible to CCF
- Choice of modeling approach (e.g., beta-factor, alpha-factor)
- Method of inclusion of CCF BEs in the logic model (i.e., fault trees) (e.g., inclusion of all CCF terms, or inclusion of only the terms required to satisfy the failure criterion)
- Method for parameter estimation for both screening and detailed analyses; Data sources applicable
- Methods used, if any, to assess the efficiency of existing defensive measures against CCFs and to consider plant-specific experience

The analysis for each system begins by defining the system functions, system operation modes and alignments, establishing system-level success criteria (in conjunction with the event tree development that addresses operability and survivability), and defines the system boundaries. All system components and system connections that are modeled are described and shown in simplified drawings.

For systems that operate in several different modes or plant operating states, each mode and the functions performed, are explained. Only those functions that contribute to plant risk defined by the accident sequences are included. The interfaces and system connections of the system with other systems are described. This also includes the support systems and relevant system components required for the operation of the system. Any other relevant aspects of the system operation including system trips, interlocks and procedural restrictions during accident conditions are also addressed.

Necessary operator actions are identified for systems where manual initiation is necessary or manual backup is credited in the PSA. The post-initiator Human Failure Events (Type C) are identified based on operational procedure. Significant pre-initiator Human Failure Events (Type A) are identified by investigating test and maintenance procedures related to the SSCs modeled in fault trees and are identified as part of the system analysis.

The success criteria of front-line systems for each IE are identified in the success criteria analysis task. System success criteria of support systems are obtained by following success criteria for front-line systems. Based on these success criteria, necessary system functions are identified. Operability relative to trip setpoint limits and adverse environmental effects impacting all system components survivability are addressed. Any operator interventions that mitigate adverse environmental effects are proceduralized.

Systems models are developed that include all component failure modes and unavailability factors that lead to failing to achieve the system function defined by the system success criteria. Component failures that would be beneficial to system operation are not included in the model. Credit is taken for system or component operability beyond rated or design capabilities with adequate justification such as:

- Test or operational data
- Calculations
- Vendor input

No credit is given to repair (recovery) of hardware faults, unless the feasibility of repair is justified, such as long times are available before core/fuel damage, which is generally not true for At Power conditions.

Database for component failure probability has two general types of failures. One is a demand failure, and the other is a rate failure based on the amount of time a component needs to operate. The reliability models considered to support BEs are not discussed in detail here, but involve exposure-rate for demand failures, exposure-rate or periodic test models for rate failures, and models that account for monitoring and repair. The data that supports the parameters listed are discussed in the data analysis task.

System modeling includes the modeling of components shared between system or units. An important aspect to ensuring dependencies are accounted for in the model is the consistent naming of BEs in each system model and across units (if there is more than one unit).

The documentation of the system analysis is constructed to facilitate certification of the BWRX-300 design. It describes the processes used for modeling and quantification. It also includes the generic modeling assumptions. It is constructed with consideration of future updates to the BWRX-300 PSA and for risk-informed applications involving the BWRX-300.

15.6.3.4.2 Fault Trees

One fault tree for each system function is included in the PSA model. If multiple trains perform the function, then each train is modeled. All the logic for a particular system is contained in a single fault tree file.

The fault trees are constructed using the gate and basic event naming conventions. The fault tree database is stored in the CAFTA database file format.

Task Outputs and Preliminary Results

The following are key task outputs:

- System Information as documented in system design descriptions
- System fault trees
- System reliability data input to data analysis
- Assumptions for each system model
- A list of modeled BWRX-300 system models

Documentation requirements are prescribed in IAEA-TECDOC-1804 for Systems Analyses as follows:

- The processes that were followed to select, to model, and to quantify the system unavailability are documented.
- The following aspects of the systems analysis process are documented:
  - Revision history
  - Open issues (questions) and answers on previous issues
  - The coding system used for the PSA and PSA model
  - System function and operation under normal and accident conditions
  - System activation/blocking
  - Alternative system alignments
  - System boundary
  - System schematic illustrating all equipment and components necessary for system operation and components that are modeled
  - Dependency matrices on a component level
  - Information and calculations to support equipment operability considerations and assumptions
  - Actual operational history indicating any past problems in the system operation modification history

- System success criteria and relationship to accident sequence models
- Human actions necessary for operation of system
- Reference to system-related Test and Maintenance procedures
- System dependencies and shared component interface
- Component boundaries
- Component spatial information
- Assumptions or simplifications made in development of the system models
- The system's potential for being a CCI
- A list of all components and failure modes included in the model (the BEs) along with justification for any exclusion of components and failure modes
- A list of all human action failure modes included in the model
- A list of all Common Cause Component Groups (CCCGs) included in the model
- A list of all environmental or spatial dependencies
- A description of the modularization process (if used)
- Records of resolution of logic loops developed during fault tree linking (if used)
- Results of the system model evaluations
- Results of sensitivity studies (if used)
- The sources of the above information (e.g., completed checklist from walk downs, notes from discussions with plant personnel)
- Fault tree description including conditions for the model, top gate, fault tree layout, transfers, house events, attributes, failure modes, CCF modeling, Test and Maintenance modeling, operator actions, and signal modeling

- The sources of model uncertainty and related assumptions associated with the systems analysis are documented.
- The input and results of any screening process is stored for future re-analysis, which may be necessary for applications or update of the PSA.
- Storage of the systems analysis information:
  - The information from the systems analysis (e.g., FMEA and fault trees with gates and BEs) is part of the PSA model.
  - The PSA model is stored, and detailed background information is stored in a retrievable and accessible electronic form and format.

15.6.3.4.3 Data Analysis

The data analysis objective is providing estimates for the reliability parameters for the reliability models specified under the systems analysis. The reliability models are used in determining the BE probabilities of specific equipment failures and unavailability factors.

Data analysis is used in deriving the parameter values that estimate system failure probabilities and accident sequence frequencies. These BEs used in the PSA model are outputs from the IE analysis, accident sequence modeling, and system fault tree modeling discussed previously. The

PSA BEs include the IE frequencies, component unavailability, component/train unavailability factors (resulting from testing or maintenance), CCF events, and other types of events. At this stage in the design process, there is no plant-specific reliability data available other than that provided by operating experience at ABWR plants in Europe.

Generic data from the nuclear industry and associated uncertainty are used.

Where plant-specific reliability data are not available, mainly generic reliability data are used as discussed below under component reliability database.

In the absence of actual operating experience for the passive plants, EPRI TR-016780, "Advanced Light Water Reactor Utility Requirements Document," Volume 3, "ALWR Passive Plant," (Reference 15.6-21) recommends using failure data for components that are most similar to those used in passive plants. Additional adjustments to the generic data are introduced after analyzing the test and maintenance intervals and the environmental factors.

The data and basis for test and maintenance unavailabilities are based on bounding generic values and Operational Limits and Conditions allowed outage times. Because of limited operating experience and the lack of plant-specific data, the development of failure rates for equipment reflects appropriate characterizations of the associated uncertainties.

The level of redundancy in passive plant design and few critical support systems leads to increased focus on CCFs. CCFs are modeled for the components of the same type and size in the same operating and maintenance environment. The CCF data from the USNRC CCF database (Reference 15.6-22) is used in this analysis. Generic CCF factors are used when component-specific data are not available.

Methodology

Component Reliability Database

The component reliability data analysis addresses the following:

- It is essential that the parameters be estimated using data and other information that is compatible with the definitions of the associated BEs. Therefore, the task procedure explicitly contains the appropriate definitions of the events or does so by reference to the appropriate task procedures for systems analysis and IEs. The component boundaries and the interfaces among connected components are explicitly identified. Since a component may be required for different operating conditions, BEs representing different failure modes may be included in the system model.
- Guidance on how to choose an appropriate data source when generic data sources are used. Also, if several sources of generic data are to be combined, the choice of method needs to be stated and described.

Failure data is presented in both per demand units and per-hour units, depending on the component and data source used. NUREG/CR-6928 presents data for both running and standby components for many component types. To divide components into standby versus running

failure modes, the components are sorted by run hours. Components with less run time than 10% of a calendar year are placed in the standby category, while those with more run time are placed in the running/alternating category. The NUREG/CR-6928 leakage failure mode is subdivided into two modes: small, covering 1 to 50 gallons per minute (gpm) and large, covering > 50 gpm.

Both repair time and detection time can be derived from generic data sources. NUREG/CR-6928 generally does not include repair or detection times, although some are listed as assumptions to develop the reliability estimates provided. However, the other references such as the T-Book: Reliability Data of Components in Nordic Nuclear Power Plants, 6th Edition provides both repair and detection times for analyzed components.

If a component is normally used or detection of component failures leads to plant shutdown, these failures are considered in IE analysis and not modeled in a fault tree. If a unit of failure rate in data base is per demand in most of the case, it is directly used for a basic event.

For data analysis, if generic values are not available, then the time-related factors are determined based on the design features, including indications and alarms, and operational parameters such as time between testing.

Components are grouped into population groups for parameter estimation. The rationale for grouping components into a homogeneous population for parameter estimation considers the design, environmental, functional, and operational conditions of the components in the as-built and as-operated plant. For parameter estimation, components are grouped according to type and to the detailed usage characteristics.

In this analysis, generic reliability data based on NUREG/CR-6928 are used. When data from NUREG/CR-6928 is not available for a modeled failure mode, data is used from the following resources in this order:

- EPRI TR-016780, “Advanced Light Water Reactor Utility Requirements Document,” Volume 3, “ALWR Passive Plant”
- NUREG/CR-4550, “Analysis of Core Damage Frequency from Internal Events: Methodology Guidelines”
- NUREG/CR-2728, “Interim Reliability Evaluation Program Procedures Guide”
- NUREG/CR-1740, “Data Summaries of Licensee Event Reports of Selected Instrumentation and Control Components at U.S. Commercial Nuclear Power Plants”
- EGG-SSRE-8875, “Generic Component Failure Data Base for Light Water and Liquid Sodium Reactor PSAs”
- Engineering judgment

Dependent Events and Common Cause Failure Analysis

Dependent events challenge redundancy or diversity and ultimately increase the unavailability of a system. Four dependency types are analyzed in the PSA analysis:

1. Plant-level functional dependencies: Plant-level functional dependencies lead to functional failures as a direct result of a shared IE.

2. Intersystem dependencies: The dependencies of BWRX-300 systems are carefully analyzed in the PSA analysis. Physical and functional intersystem dependencies are incorporated into the system fault trees by using transfer gates. Physical dependencies include spatial interactions, influences for common environment, or impacts from adjacent SSCs. These physical dependencies are important when modeling the impacts from internal and external hazards (fire and flood).

3. Human action dependencies: Dependencies due to human actions (testing, maintenance, diagnostic errors, incorrect calibration of sensors or instrument), which could affect manual actions or actuations of redundant systems, are reviewed in the system PSA analysis, and modeled as separate events in the fault trees where necessary. Modeling of human action dependencies is discussed further in Section 15.4.

4. Intra-system dependencies: Intra-system dependencies CCFs can lead to multiple component unavailability's from shared causes. The remainder of this PSAR section discusses the analysis approach for this type of dependency.

Developing the Common Cause Failure Database

The general approach described in NUREG/CR-6268, "Common-Cause Failure Database and Analysis System: Event Data Collection, Classification, and Coding," (Reference 15.6-23) and by the CCF Parameter Estimations 2012 is the main reference used in the CCF analysis.

There are several models to estimate the CCF probabilities and failure rates. The CCF probability modeling is discussed in the next subsection.

CCFs are a subset of dependent failures (i.e., those failures that defeat the redundancy or diversity that is employed to improve the availability of plant safety functions). CCFs, like other dependent failures, are addressed in PSA by incorporating appropriate common cause BEs in the integrated plant logic model. The technique used for generating the data for the CCF BEs is described below.

The applicability of CCF grouping in system modeling is based on current design phase information regarding design, operating, maintenance, service, and testing information.

Running and standby components are assumed to belong to the same CCCG (e.g., redundant pumps), for the fail to run and fail to restart (following LOPP) failure modes. Components are placed into common cause groups according to certain factors (e.g., type, use, manufacturer, location, function, failure mode). Consideration is also given to intersystem CCFs. For significant

common cause contributions, system fault trees are reviewed to reassess the assumptions, or engineering and operational defenses that can be credited, and/or design changes that can be made to improve diversity.

In developing a system model, common mode and common cause dependent failures are included. An alpha-factor modeling approach was used to add CCF events to the model. This approach and the parameters that drive it are described in the data analysis task of this procedure.

CCFs are modeled for intrasystem failures of mechanical equipment, such as:

- Fail to Start/Run of Active Components (e.g., pumps, fans, diesel generators, compressors, and chillers)
- Fail to Open/Close of Valve
- Fail to Control of Valve
- Spurious Operation of Valves
- Failure of CRD
- Plugging of Static Components
- Air Dryer Unit Failure (stopping or bypassing flow)

CCFs are also modeled for intrasystem failures of electrical and electronic equipment, such as:

- Fail to Operate of Relay/Contactor
- Circuit Breakers
- Loss of Function of Battery/ Battery Charger
- Loss of Function of Electrical Equipment (e.g., Bus, Transformer)
- Fail to Operate/Higher Output/Lower Output of Transmitter, Sensor, and Transducer
- Fail to Operate of Switch
- Fail to Operate of Digital System Central Processing Unit /Field Programmable Gate Arrays

CCCGs are defined based on a process that considers similarity. Similarity is classified into categories described in NUREG/CR-6268, Table 8-1.

When identifying the CCCGs, the following guidelines were followed:

- Components within a CCF group have similar attributes and failure mechanisms and are functionally redundant with respect to each other.
- Only the failure to perform a specific function is modeled in each CCF group.
- Low/consequential failure modes (e.g., spurious failures) are compared and excluded from common cause consideration if a more consequential active common cause group is modeled.
- The specific origins of CCF (e.g., shock, high temperature, manufacturing defects) are not specifically defined.

The five main coupling factor categories that were used in the identification of the CCCGs were defined in NUREG/CR-6268, Table 8-1:

- Quality – Manufacturing quality
- Design – Component design
- Maintenance – Consideration of Test and Maintenance procedures, schedules, and staff
- Operation – Operational procedures
- Environment - External and internal environments

The analysis of the above factors for each component type needs to be considered in the common cause component group analysis. For example, the following parameters are considered for pumps:

- Pump type
- Operating pressure
- Operating temperature
- Medium (water type/air/boric acid/oil)
- Motive power
- Flow rate
- Total head
- Voltage source capacity
- Rotational Frequency

This analysis is used to aid decision making when assigning CCCGs in addition to consideration of the component function. Design and manufacturing information was not yet available for all components. As design information becomes available revisions to the CCCGs are developed.

The common cause groupings used in the PSA are generally limited within the same systems. This limitation is made for the following reasons:

- CCFs are considered for similar components (usually nearly identical) that operate in similar environments and have similar duty cycles. These characteristics are usually not present when considering components across system boundaries.
- It is generally found that the Test and Maintenance procedures for different systems are such that a CCF of similar components among different systems is all but precluded.

As the PSA progresses and becomes more detailed, intrasystem modeling is included based on the factors discussed above (similar design, operation, maintenance, and environment).

The CCCGs used in the PSA model for each component type are discussed below. For all components, internal and external leakage failure modes are not considered for CCFs. This is because of the low frequency compared to other failure modes.

The following are examples of mechanical equipment CCCGs:

- Active components (e.g., pumps, fans, diesel generators, compressors, and chillers) fail to start
- Active components (e.g., pumps, fans, diesel generators, compressors, and chillers) fail to run
- Valve fail to open/close
- Valve fails to control
- Valve spurious operation
- Dampers spurious closure (presently not modeled)
- Control rods - hydraulic scram function
- Static component plugging (e.g., strainers) or leakage (presently not modeled)

The following are examples of electrical and electronic Common Cause Component Groups:

- Circuit Breakers fail to operate
- Circuit Breakers spurious operation
- Battery charger loss of function (fail to operate)
- Battery loss of function
- Bus loss of function
- Transformer loss of function
- Transmitter, sensor, and transducer fail to operate/higher output/lower output
- Switch fail to operate
- Digital system processors fail to operate

Once the CCF BEs are identified, as discussed above, the model utilizes the CCF Tool in CAFTA to automatically identify all the CCF combinations for a given CCF group and place the appropriate CCF BEs into the fault trees. CAFTA is also used to automatically quantify each of the CCF BEs.

As recommended by the CAFTA developers, common cause groups containing more than four components are treated simplistically. CCF events are developed for all combinations of one, two, and three components within the group, with all remaining failure probability assigned to a CCF event of all components failing. This is a conservative representation of the event modeling all components failing, since the CCF probability of all components is overstated, but greatly simplifies the fault tree modeling.

When this treatment is too conservative, the calculations are completed outside of CAFTA and entered manually. In these cases, the CCF combinations are manually input to the fault trees using the parameters for the actual CCF group size and the CCF equations are directly input into the database file.

There are some CCFs for which the CCF tool cannot be used. In these cases, there is no specific data available and engineering judgment or other data sources are used to calculate the CCF events.

The alpha-factor method is used to quantify CCFs in the PSA. A detailed description of the alpha-factor method and its advantages over the older beta-factor and Multiple Greek letter methods for modeling CCFs is provided in the reports on Guidelines on Modeling CCFs in ASME/ANS RA-S-1.1-2024 and CCF Parameter Estimations (Reference 15.6-22). The alpha-factor method is preferred as it enables better modeling of higher CCCGs and more accurate modeling of uncertainty in the CCF parameters.

Task Outputs and Preliminary Results

The output of the data analysis task is the BWRX-300 reliability database that supports development of system analysis, HRA, and quantification tasks.

The risk insights of common cause failure analysis at the systems and the plant-level clarify the balance between the design and the reliability of SSCs. Furthermore, these insights help in the implementation of procedures and training to manage the balance between design configuration control and component reliability control.

Documentation requirements are prescribed in IAEA-TECDOC-1804 for the data analysis as follows:

- System and component boundaries used to establish component failure probabilities
- Grouping criteria for components
- The model used to evaluate each basic event probability
- Sources for generic parameter estimates
- The plant-specific and Plant Operating State-specific sources of data
- The time periods for which plant-specific data were gathered, and justification of any censoring of the data for specific Plant Operating State conditions
- Walkdown reports, including notes and photographs
- Assumptions made in the interpretation of data and the reasoning (based on engineering judgment, systems modeling, operations, and statistical knowledge) supporting its use in parameter estimation
- Justification for any exclusion of any data
- The basis for the estimates of common cause failure probabilities, including justification for screening or mapping of generic and plant-specific data
- The rationale for any distributions used as priors for Bayesian updates, where applicable
- Parameter estimates including the characterization of uncertainty as appropriate
- The rational for using generic parameter estimates for multiple Plant Operating States
- The results of the fire-induced circuit failure analyses and the associated conditional failure probability values
- The derivation of the parameter values is documented

- The information/database is documented and stored in a way which allows the reproduction of the data analysis task for example for reliability parameter updates
- In the PSA for hazards the documentation covers the following:
  - Description of the method of derivation of the probabilities of the failures due to the hazards
  - Description of the input information
  - Fragility and load curves if appropriate
  - Uncertainty estimations for each failure probability derived
  - Results in terms of probabilities and related uncertainties
- The sources of model uncertainty and related assumptions associated with the data analysis are documented
- For CCF Analysis:
  - Identification and definition of CCCGs
  - All CCCGs and components included
  - Treatment of intrasystem and intersystem CCFs
  - Models used
  - Special CCF events
  - CCF data
  - Description of how CCF is implemented in the overall PSA model

15.6.3.4.4 Human Reliability Analysis

The HRA is performed as part of the BWRX-300 PSA. Human actions play a key role in the risk assessment and may point to plant vulnerability early in the design phase that can be corrected before the design is finalized.

The HRA describes operator failure in a systematic and reproducible manner to facilitate an HEP that is applied to the PSA model. The methodology is described below. Individual failure events generate reports that are presented in appendices to the PSA model.

Scope

The HRA covers three types of human actions:

1. Those performed by maintenance or operations before an event
2. Those performed by operations or maintenance to cause an event
3. Those performed by operations after an event has occurred

The general documentation guidance of the HRA is to meet ASME/ANS RA-S-1.1-2024.

Inputs, Requirements, and Assumptions

The primary input to the current HRA comes from a review of the success criteria and ESs documents. These documents discuss the operator actions that are considered for the safe shutdown of the plant. The second resource is examination of the system fault trees to find actions that backup automatic functions. The third resource and the primary resource for pre-initiator actions comes from system engineers, operating and maintenance procedures, and any developed system maintenance schedules. Maintenance is not considered at this stage of the design in many of the systems and therefore pre-initiators are not included in the model.

Requirements

The requirements for the HRA analysis primarily come from ASME/ANS RA-S-1.1-2024. Due to the incompleteness of the design and the lack of procedures, this analysis is not expected to meet all the requirements of the standard at this time.

Assumptions

Generic assumptions used in the PSA:

1. All human actions are assumed to be driven by procedure. No actions are considered that are outside operations documentation (e.g., actions based on knowledge, or mitigating actions outside the procedure).

2. Errors of commission are screened out from the analysis because they have negligible probability compared to errors of omission.

3. It is assumed that all persons performing operations or maintenance are adequately trained by an accredited training program.

4. Post-initiator actions are all conducted from the control room unless otherwise specified in the human failure event report.

5. In the current phase of design, procedures for actions are assumed based on engineering judgement and experience.

6. The pre-initiator HRA assumes for a component that is unavailable during testing or maintenance, a restoration error is always possible.

7. The MCR displays and alarm annunciators are well-designed with critical alarms for all human failure events within the safety case identified and shown on the appropriate displays, with color-coding, lighting, and flashing to guide alarm management. Information is displayed in a manner appropriate to the type of display (e.g., large screen versus control desk versus local control panel).

8. Software-based display screens and control actuations are well-designed with consideration of human factors good practice principles including grouping, labeling, and spacing. In particular, potential selection errors either in screen navigation or in equipment icons on a particular screen are minimized through effective layout, screen item design, and group demarcations.

9. The time required to perform the steps in a post-initiator task has been estimated in most cases based on input from personnel that have BWR PSA human reliability analysis experience, have interacted with nuclear operators, and are familiar with BWRs operator actions. Wherever possible, worst-case plant conditions and competing task workload have been assumed in order to ensure suitably conservative estimates. In some cases, the times taken to carry out task steps that are part of a task but are not necessary for successfully completing the specific action claimed within the PSA are included in the overall task required time. This is done in those cases where members of the human reliability analysis team have indicated that such actions are likely to be attempted. Task timings are further examined and verified in later iterations of these analyses.

10. Assumptions related to individual dependencies include:

    - Although many actions occur over a long period of time, it is assumed that there are no intervening successes between actions.
    - The same crew is assumed to be performing the actions throughout the event.
    - Cognitive is assumed different except where the actions are similar (e.g., alignment of cooling).
    - Simultaneous actions are assumed when operators are working with multiple, immediate concerns (e.g., pump power and isolation).
    - The required resources to perform a task (e.g., operators, maintenance personnel, health physics) are estimated based on experience and engineering judgment.

11. Timing overrides are implemented as necessary to place actions into chronological order. In the current model, no overrides are necessary.

HRA Discussion

There are several types of human actions during normal plant operation:

- Those related with the failure to restore equipment to its normal condition following a test or maintenance action
- Those causing an IE

Human actions during an accident are:

- Those performed by operators following established procedures for mitigating the consequences of an accident

- Those performed in the mistaken belief that they are the appropriate actions as indicated in the procedures, but in reality, can worsen the conditions of the accident, thereby complicating the mitigation process
- Those that are not explicitly included in the procedures and are used to recover the operability of failed equipment or use alternative means

Adequate treatment of the human actions in a PSA is one of the keys to a realistic understanding of ESs and their relative importance to overall risk. To perform an adequate HRA, it is necessary to have a structured framework that allows the different parts of the analysis to be developed in an orderly fashion while considering the general boundary of the PSA.

There are three main types of human actions in the HRA:

Type A: Initiating Event Induced by Latent Human Actions

Before an IE, plant personnel affect availability of standby systems by inadvertently disabling equipment or failing to restore the correct position during the performance of operational activities in the plant. Example: maintenance staff incorrectly perform a valve lineup after maintenance on a pump. When the pump is needed during an event, it cannot run or is damaged.

Type B: Initiating Event Induced by Manifest Human Actions

Plant personnel can initiate an event through interactions with plant equipment. These typically occur due to a mispositioning or incorrect operation of equipment that inadvertently trips equipment or inserts false control signals. Example: One of three pumps providing service water to a condenser fails and operators do not align a backup in a timely manner. The plant trips on loss of service water.

Type C: Post-Initiating Event Human Actions

After an IE has occurred, plant operations attempt to place the plant in a safe, stable state using Emergency Operating Procedures (EOPs). There are two types of errors associated with Type C actions:

1. Commission Errors – By following procedures during the course of an accident, plant personnel operate standby equipment that aggravates the accident. This action, called a commission error, gives rise to a situation in which the operators place the plant in a worse condition than if they had done nothing (i.e., an omission error). The risk of commission errors is considered insignificant when the plant has EOPs that have a symptom-based orientation and the EOPs cover all the possible scenarios in the ESs analysis.

2. Omission Errors – Failure to perform a step in the procedure, or failure to act in a timely manner results in either equipment damage or core damage. It is imperative that operations perform the correct steps within the time frame before damage occurs.

15.6.3.5 Event Sequence Frequency Quantification

The BWRX-300 PSA model consists of event trees and fault trees that are quantified using a fault tree linking process.

The calculation of core/fuel damage and release category frequencies is performed as single top gate for each sequence and release category. The top gates include all sequences but use a sequence marker to identify the event tree sequences in the cutsets generated by the single top. The use of a sequence marker results in retaining all the minimal cutsets for that specific sequence but prevents subsuming of non-minimal cutsets among sequences. The contribution to risk from these non-minimal sequences is generally small, although this can affect the results for specific hazards, such as earthquakes (seismic). The sequence logic is also set up to exclude any combinations associated with the success branches in the specific sequence. The individual sequence results can then be combined as necessary for reporting, analyzing, or to be used as input for the Level 2 or Level 3 portions of the PSA.

15.6.3.5.1 Methodology

The purpose of the ES frequency quantification is to obtain the Boolean equation corresponding to the radionuclide release. The quantification is developed in terms of minimal cutsets representing the minimal combinations of events that result in radionuclide release.

The following key aspects characterize the release category frequency quantification process:

- Event trees model plant response to each group of IE
- Fault trees model the behavior of front-line and support systems
- Integration of event tree and fault tree structures into a single linked model
- Quantification of the linked Boolean model with the probabilistic database and boundary condition files (flag files)

Use of CAFTA for Solving the Probabilistic Safety Assessment

For the Level 1 BWRX-300 PSA model development and quantification, the EPRI Integrated Risk Technology suite of software packaged under the Phoenix Architect is used that includes the following:

- CAFTA for event tree, fault tree and cutset development and viewing purposes
- Phoenix Application Programming Interface (API) for model integration and one-top development
- PRAQuant for quantification processes, as well as for sensitivity runs
- FTREX as the quantification engine (cut set generation)

EPRI Integrated Risk Technologies User Group offers the suite of risk software tools (formerly known as a Risk and Reliability Workstation), which have been used for the Level 1 PSA model development and quantification. These risk software tools include CAFTA for event tree and fault tree construction and PRAQUANT for master model integration and quantification. Accident

sequence frequencies are derived from the quantification results of a single top master fault tree by post-processing using the sequence markers. The PSA code FTREX is the PSA model quantification engine.

CAFTA has worldwide users at more than 70 power plants and has a very large active user community ranging from power generation to communications, transportation, aviation, and space applications.

The FTREX code is used in generating Minimal Cutsets (MCSs) from fault tree logic models. FTREX uses a zero-suppressed binary decision diagram method for interpreting large fault tree models when developing MCSs. FTREX is built for PSA applications and is applicable to the BWRX-300 PSA without specific limitations.

Appendix 3M discusses the computer programs used in the PSA.

Quantification File Structure

The following files are generally associated with the PSA integration and quantification of the Level 1 internal events PSA. Other files may be developed for hazard quantification:

- Event Tree Files: These files contain the event tree structures for each group of IEs. The radionuclide release sequences of these models are converted into fault tree logic and integrated into the Master Fault Tree File along with the system fault tree logic.
- Fault Tree Files: A single fault tree file contains all the release sequences, with all the front-line and support systems linked. Each sequence subtree includes a top gate that combines the IE and the functional failures and successes of the sequence.
- Database File: The database contains the inputs, assumptions, probabilities, and frequencies of each of the events associated with the fault tree.
- Quantification File: This file contains the factors and conditions for quantifying the different ESs.
- “Flag” Files: “Flag” files contain boundary conditions (e.g., type of IE, assumed plant configuration) used in the quantification of the ESs.
- Recovery files: These files are used for post processing of cutsets.

The reliability databases supporting the BWRX-300 PSA include the IE frequencies, Component failure frequencies and probabilities, CCF data, component repair times, test intervals and durations, mission times, maintenance unavailability, unavailability due to testing, and human error data. Uncertainties for data values are also included. All the data sources are justified and documented.

Additional Quantification Steps

The quantification process used for the PSA is iterative and includes numerous stages where internal and external review is performed before final results are documented. Key steps performed during the quantification include the following:

1. Circular Logic: It is possible to generate a master fault tree that has circular logic. This is where a gate is used as input to a second gate that happens to also be an input somewhere in the tree of the first gate. The “Find Circular Logic” tool within CAFTA is used to identify any instances of circular logic. Breaking the logic involves revision to a system fault tree.

2. BE Consistency Check: The IE frequencies and the probabilities associated with BEs of the model should be consistent with the definitions of the events in the context of the logic model. As the PSA cutsets are reviewed, they are reviewed for consistency in terms of the applied assumptions, specific design, and operational experience. A simple example is to verify the assumed time windows for BEs within a cutset match the resulting cutset time window. Another example might be that the number of demands assumed for a safety relief valve lift is consistent with the success criteria thermal hydraulics analysis performed.

3. Mutually Exclusive Event Review: The model is solved with and without the mutually exclusive event file to verify the event combinations are removed.

4. Human Error Dependencies: One example of how human failure event dependencies are determined is summarized as follows: All post-initiator human failure event probabilities are set to one or a high value, the single top quantification is performed again, and the most critical human failure event dependencies are apparent in the cutset results. The PSA then uses the results generated in the cutsets to generate recovery rules that change the probability of these joint failures to a value that better reflects the dependency between these human failure events, taken from the HRA. The HRA Calculator is used to perform the HRA dependency analysis.

5. Recovery Actions: Generally, for a design level PSA, equipment recovery is not credited other than operation of backup systems which is addressed in the Type C operator actions. In the site-specific PSA, recovery is possible for longer term accident sequences where equipment recovery is possible. A specific cutset review is performed to identify equipment recovery cutsets and apply model changes where significant changes in the results can occur.

6. Truncation Justification: To justify that an appropriate truncation value is used, a test for CDF convergence is performed. To test for convergence, the CDF and Level 2 models are solved again with the truncation lowered by an order of magnitude (e.g., if the truncation was set to 1E-11, lower this value to 1E-12). If resulting CDF increases by more than 5%, the model has not converged. This process is repeated until the CDF increase for an extra decade of truncation is less than 5%. Additionally, to ensure ASME/ANS RA-S-1.1-2024 is met, the last two convergence runs are used to develop a list of risk-significant accident sequences (e.g., contributing greater than 1%) to ensure no new significant accident sequences are identified by lowering the truncation limit further.

7. Importance Analysis: This analysis documents the overall contribution of BEs to each phase of the PSA including the Level 1, and Level 2 PSA. Separate importance results are provided for BE types such as IEs, human errors, event class, and release categories. Importance measures are tabulated separately for internal events, internal hazards, and external hazards, and then again for the combined results given the combined (merged) results are developed.

8. Uncertainty and Sensitivity: Uncertainty analysis to assess statistical and state of knowledge uncertainties. Sensitivity analysis to determine what parameters have an impact on the results.

9. Review of the results: ASME/ANS RA-S-1.1-2024 and the IAEA-TECDOC-1804 require the review of the results in detail to ensure the results are consistent and justified. The steps include review of both risk-significant and non-risk-significant cutsets by knowledgeable staff, and comparison of the results with similar plants. For the BWRX-300, the comparison of the results with similar plants is not possible, other than a general comparison with other BWRs using similar components (e.g., Isolation Condenser System (ICS)).

10. Documentation: This step is key to the use the PSA for risk-informed applications including licensing. All of the steps discussed above, including the modeling integration and quantification process, are documented in a manner that facilitates external review as well as future upgrades and applications. This includes the clear documentation of assumptions.

Key Inputs

The key inputs are:

- Accident Sequence Event Trees
- System Fault trees and databases
- Reliability Database
- Event Tree Transfer Top Logic File

Task Outputs and Preliminary Results

Integration and quantification of the internal events At Power BWRX-300 PSA model results in the following key outputs:

- The final internal events At Power BWRX-300 PSA model files:
  - Single Top Sequence Logic File
  - Single top main fault tree
  - Final merged reliability database
  - Main Configuration Flag File
  - Recovery Rule File
- Core/fuel damage and release category frequencies

- Core/fuel damage and release category frequencies as a function of:
  - Internal Events
  - ESs
  - Event class
- Importance characterization of individual events, in terms of industry standard risk importance measures (e.g., Fussell-Vesely; Risk Achievement Worth) relative to the core damage and release category frequency

Documentation requirements are prescribed in IAEA-TECDOC-1804 for the integration and quantification analysis as follows:

- Integration process of all fault and event trees
- PSA model version
- General description of the quantification process
- Process and the results for establishing the truncation values
- Software version and quantification setup (e.g., truncation values)
- Process and the results of the quantification review
- Mean value and the uncertainty distribution of the total risk metric (e.g., CDF/Fuel Damage Frequency (FDF)), each Plant Operating State (POS), each class of IEs, and significant risk metric (e.g., core/fuel damage) sequences
- Accident sequences and their contributing cutsets
- Sources of model uncertainty and related assumptions associated with the model integration and risk metric quantification are documented

15.6.3.6 Internal Fire Hazard

15.6.3.6.1 Introduction

The probabilistic fire analysis is performed with simplifying assumptions because the specifics of cable routings, ignition sources, or target locations in each zone of the plant are still in the design phase. Because of this limitation, a simplified, conservative, and bounding approach is used in this analysis.

The aim of the analysis is to show that risks due to fire are not significant contributors to BWRX-300 risks.

The current scope of the analysis is for At Power accident scenarios.

15.6.3.6.2 Methodology

The BWRX-300 internal Fire PSA is performed according to the guidance in NUREG/CR-6850, “Fire PRA Methodology for Nuclear Power Facilities,” (Reference 15.6-24), as applicable. Other key references:

- NUREG/CR-6850, Supplement 1, “Fire Probabilistic Risk Assessment Methods Enhancements,” (Reference 15.6-25)
- NUREG-2169, “Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database,” (Reference 15.6-26)

For future updates, NUREG/CR-7150 Vol 1 “Joint Assessment of Cable Damage and Quantification of Effects of Fire,” (Reference 15.6-27), BNL-NUREG-98204-2012, EPRI 1026424, “Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE) Volume 1: Phenomena Identification and Ranking Table (PIRT) Exercise for Nuclear Power Plant Fire-Induced Electrical Circuit Failure,” and NUREG/CR-7150 Vol 2, BNL-NUREG-98204-2012, EPRI 3002001989, “Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE) Volume 2: Expert Elicitation Exercise for Nuclear Power Plant Fire-Induced Electrical Circuit Fire” are expected to be key references as well but are not utilized due to the bounding nature of this study as this stage of design.

NUREG/CR-6850 and NUREG/CR-6850 Supplement 1 document state-of-the-art methods, tools, and data for the conduct of a Fire PSA for a commercial nuclear power plant application. The methods have been developed under the fire risk requantification study. This study was conducted as a joint activity between the EPRI and the U.S. NRC Office of Nuclear Regulatory Research under the terms of a USNRC/EPRI Memorandum of Understanding and an accompanying Fire Research Addendum.

For the BWRX-300 Fire PSA model development, the following NUREG/CR-6850 and NUREG/CR-6850 Supplement 1, tasks are applicable:

- Task 1: Plant Boundary & Partitioning
- Task 2: Fire PSA Component Selection
- Task 3: Fire PSA Cable Selection
- Task 4: Qualitative Screening
- Task 5: Fire-Induced Risk Model
- Task 6: Fire Ignition Frequencies
- Task 7: Quantitative Screening
- Task 12: Post-Fire Human Reliability Analysis
- Task 14: Fire Risk Quantification
- Task 15: Uncertainty and Sensitivity Analyses
- Task 16: Fire PSA Documentation

Some of the above tasks are performed with simplification while others are omitted at this stage of the BWRX-300 Fire PSA due to the maturity of the plant design. Tasks not addressed in this study since the BWRX-300 plant is still in the design phase include: (a) Scoping Fire Modeling (Task 8); (b) Detailed Circuit Failure Analysis (Task 9); (c) Circuit Failure Mode Likelihood (Task 10); (d) Detailed Fire Modelling (Task 11) including multi-compartment analysis; (e) Seismic-Fire Interaction (Task 13); and Support Task A, Plant Walk Downs.

Support Task B, Fire PSA Database, is performed with an Access®-driven database (FRANX) that includes all tables that are necessary to develop a scoping-level Fire PSA model. Enhancements to the Fire PSA database are possible in future updates once more details are available for cable tray routes and their contents as well as more detailed ignition source locations.

Fire ignition frequencies for power operation at each plant area (Physical Analysis Unit (PAU)) are estimated using the NUREG/CR-6850, EPRI 1011989 and NUREG/CR-6850, EPRI 1019259, Supplement 1 methodology and data from NUREG-2169, EPRI 3002002936. Fire frequencies for shutdown conditions are not developed at this time.

For a postulated fire, a list of the impacted components is generated with the mapping defined in the Fire PSA database. A list of impacted components is also generated with the assumed cable routing. The cable routing is assumed based on the modeled PSA components, their supports, and the general room layout of the BWRX-300 design. Fires are conservatively assumed to propagate unmitigated in each PAU (no suppression is credited) and damage all functions in the PAU. The internal events PSA accident sequence structures and system fault trees and success criteria are used in the calculation of the fire CDF.

The BWRX-300 fire PSA employs the following EPRI software:

- FRANX, Version 4.4. This software uses the internal events model and adds fire initiators.
- Fault Tree Reliability Evaluation eXpert version 1.8. This software generates cutsets from the fault trees produced in CAFTA.
- CAFTA, version 6.0b. This software is used to build the logic model of the plant, producing all the fault trees and event trees.
- ACUBE 2.0, version 2.0. This software provides a more accurate solution to the cutsets than conventional solutions.

15.6.3.6.3 Key Inputs and Assumptions

Key Inputs

The following are the inputs for the BWRX-300 fire analysis:

- At-power internal events PSA model
- Low-power and shutdown internal events PSA model
- Fire compartment spatial information

General Assumptions

The fire risk analysis is performed using simplifying and conservative assumptions due, in part, to the stage of the design. The key conservative assumptions are summarized below:

1. A fire ignition in any fire area may grow into a fully developed fire.

2. The analysis does not take credit for any fire suppression (i.e., self-extinguishment, installed suppression systems, nor manual firefighting activities). Therefore, the analysis assumes that all fires disable all potentially affected equipment in the area.

3. Unless otherwise stated, a fire is assumed to cause failure of all fire-susceptible components in the subject fire area and detailed fire modeling is not performed in this revision of the BWRX-300 Fire PSA.

4. Recovery of the failed components or cables after the postulated fire is not credited.

5. Unless otherwise stated, the analysis does not take credit for the distance between fire sources and targets.

6. The analysis assumes that all fire-induced equipment damage occurs at a time of zero in the scenario progression.

7. Based on the plant general arrangement drawings with component locations, assumed cable routing is postulated for PSA purposes. Note that there are no details for cable routing in the BWRX-300 plant design at the time of this analysis beyond locations of some of the major electrical system components. A list of cables is generated that includes all modeled supports for PSA components included in the current At Power internal events PSA model. This list captures most cables, especially for expected risk-significant components.

8. Fires in the MCR and Secondary Control Room (SCR) are currently modeled as impacting components associated with any assumed cable routings that go through the rooms but do not terminate in the MCR or SCR (i.e., the assumed cable route for Bus B23 goes through the MCR between the Turbine Building (TB) and the B train Motor Control Center (MCC) room in the Control Building (CB)). The MCR and SCR are not assumed to impact components for which control or visualization cables may be in the rooms. This is likely a realistic treatment, as the plant design could be similar to other BWR designs, in which control room controls are connected to the Distributed Control and Information System (DCIS) rooms (which are unaffected by a MCR fire) via fiber cables. The loss, including melting, of the fibers or visual display units does not cause inadvertent actuations, nor affect the automatic actuations associated with safety class and non-safety class equipment. In addition, fires in the MCR and SCR are assumed to fail all modeled human failure events.

9. All finalized details of cable route information are assumed to be not yet available. Individual routes are assumed based on location of PSA modeled power/signal sources and the particular component.

10. The bins for cable fires, cable runs, and junction boxes are estimated with the cable routing information postulated in the cable selection task. NUREG/CR-6850 Table 6-1 provides the list of these bins and their respective generic mean frequencies (i.e., the mean value of the uncertainty distribution) in terms of number of events per reactor year.

11. The assumed cable routings attempt to use general separation criteria, though due to the number of different signal and power divisions, there are PAUs with more than one division, as well as power and control cables postulated. As examples, Buses A23 and B23 are located in separate rooms and their associated cables take routes from those separate rooms in the CB. Divisions I, II, and III of the DCIS and electrical switchgear are located in separate rooms within the RB and their cables take separate routes between the switchgear rooms and battery rooms. As the BWRX-300 design matures, these cable routings are subject to being refined, including incorporation of any credited fire barrier(s), within an overall PAU, that serve to separate divisions as appropriate.

12. The two standby diesel generators are assumed to be in a common room based on the current design information. Their associated cables are in the same room until entering the two separate divisional rooms of the TB switchgear rooms.

13. Fire ignition frequencies remain constant over time and are based on industry generic fire frequencies. Among the operating plants, total ignition frequency is the same between plants for the same equipment type, regardless of differences in the quantity and characteristics of the equipment type that may exist among the plants. This is likely conservative because the BWRX-300 design has significantly lower numbers of pumps, motors, and other active components than earlier plant designs from which the current generic ignition frequencies are based on.

14. It is assumed that all ignition source type bins are applicable to the BWRX-300 plant with the following exceptions:

    a. Bins 02 and 03 are not applicable since they are used for Pressurized Water Reactor plant designs.

    b. Bin 22 for RPS motor generator sets is not applicable to BWRX-300 plants since it is typically a Pressurized Water Reactor plant feature.

    c. Bin 04 is likely not applicable to the BWRX-300 MCR or SCR as the BWRX-300 design is completely digital as opposed to traditional electro-mechanical designs. However, to ensure a conservative analysis is produced, twenty-five percent (25%) of the traditional Bin 04 ignition frequency is assumed to be applicable and is assigned to the BWRX-300 MCR ignition frequency.

15. Although the BWRX-300 plant may be located with existing nuclear power plants, the plant location weighting factors are assumed to be one because the BWRX-300 plant is designed as a single-unit plant with no shared buildings.

16. Since the BWRX-300 plant is still in design phase, the count of components is performed with the modeled PSA components as well as preliminary design layout drawings.

17. Since the plant has not operated and no history of maintenance activity is available, the weighting factor evaluation is simplified. It is assumed that all compartment PAUs have the same transient fire influencing factors; a value of one is used for all influencing factors.

18. All fires are assumed to result in a manual reactor shutdown regardless of location or potential induced failures.

19. Full details of electrical cabinet locations are not yet available. Preliminary design information which notes the Division or Train of equipment is used in the development of this revision of the BWRX-300 Internal Fire PSA for electrical cabinet and high energy arcing fault counts. Note that some of these counts are assumptions with engineering judgment for typical nuclear plant layouts and functions.

20. The following PSA components are assumed to fail in a safe manner and to be train separated: ICS actuation logic, CRD system actuation logic for control rod insertion, CRD Hydraulic Control Units (HCUs), and RPS actuation logic. Therefore, cable routes are not required for these components because any cable impacts from a postulated fire actuates the component in its safe shutdown position. Note that cable routes are required for the CRD pumps and associated components as used for the injection logic model that is separate from the scram model logic in the internal events PSA model. Also note that cable routes are required for the ICS valves that are separate from the actuation software events in the internal events PSA model. Additionally, the current internal events PSA model represents the CRD HCUs by a single common cause event and that train specific HCU failures are not possible in that configuration.

21. The two modeled containment venting valves are both assumed to be in the refueling deck (PAU 1800) based on preliminary design information. Their cables are assumed to be train separated in locations other than PAU 1800 where both valves included in the internal events PSA model are located.

22. Diverse and Flexible Coping Strategy (FLEX) equipment is assumed to be located and stored in the general yard area (PAU YARD) and such equipment and human action is failed for a fire in that location. There may be multiple connection locations in the final BWRX-300 design that require refinement of this assumption.

23. Some components remain unlocated due to the preliminary design of the BWRX-300 plant. These components are assigned to a location zone of “UNL” (unknown location) in the analysis. The UNL zone is failed for all quantified scenarios, serving to fail any unlocated components for all scenarios. This is conservative since the components are impacted for many scenario locations that are not expected to impact the components once detailed location information and any applicable cable routing is finalized. The “UNL” zone is failed in the Fire PSA model database scenario list in that the zone names impacted for each scenario

are listed in the Fire PSA model database table “Zone to Raceway” that has the zone names and components that are impacted for the particular zone name. The component names are mapped to PSA model BEs which are failed during the Fire PSA model quantification.

24. The containment is assumed to always be inerted during power operation. The possibility exists that the containment could be de-inerted in preparation for an outage, or still de-inerted when coming out of an outage. Typically, this de-inerting occurs around two to three days before an outage, which can result in a fire during this time. This time window is not included in the current internal fire PSA but is subject to be included in the future shutdown operation internal fire PSA.

25. Loss of ventilation could affect PSA components. It is assumed that for the roof locations full room burnout is not realistic due to the general separation between components and open location. Failure of a single component is not modeled in this revision of the BWRX-300 Fire PSA. If future BWRX-300 plant design requires cooling during the PSA mission time, this assumption is subject to refinement.

26. The MCR fire is not expected to result in the LOPP or the loss of the diesel generator function.

27. Support systems, such as cooling (component and room), control and motive air or nitrogen, are not modeled in the Fire PSA, since they are not modeled in the internal events PSA.

28. Cables associated with display or visualization of component status or function that may be routed between a component or instrumentation location and the MCR or SCR are assumed to not impact any automatic function of the component.

29. Fires in the TB Main First Floor Area cannot cause a failure of both Train A and Train B cables, such that a complete LOPP and on-site AC power could result.

30. The PSA model used in this analysis assumes the modulating ICS condensate return valve opening logic requires two out of three high RPV pressure signals. This is the DL4a logic. This valve can also open from DL2 using only one of four available signals if the other three signals are lost. It is assumed that no single fire can damage all four signals, so only one signal is assumed failed in any fire to prevent the model from producing an overly conservative result with respect to this function.

15.6.3.6.4 Fire Ignition Frequencies

This section documents the calculations of the fire ignition frequency during At Power operations for each of the unscreened fire areas of this analysis. The NUREG/CR-6850 and NUREG/CR-6850 Supplement 1 methodology is used to calculate the full-power fire ignition frequencies using generic data from NUREG-2169.

This task was organized around the following eight steps:

- Step 1: Mapping plant ignition sources to generic sources
- Step 2: Plant fire event data collection and review

- Step 3: Plant-specific updates of generic ignition frequencies
- Step 4: Mapping plant-specific locations to generic locations
- Step 5: Location weighting factors
- Step 6: Fixed fire ignition source counts
- Step 7: Ignition source weighting factors
- Step 8: Ignition source and compartment fire frequency evaluation

Steps 2 and 3 are not applicable since the BWRX-300 plant is in the design phase. The location weighting factors are simplified as described in the assumptions.

15.6.3.6.5 Detailed Fire Modeling

Detailed Fire Modeling for Full Power Condition

Detailed fire modeling is not performed for this version of the BWRX-300 Fire PSA for At Power operation. Detailed fire modeling can be undertaken when the BWRX-300 design is more complete and additional details on equipment locations and cable tray routes are known with certainty.

In this version of the internal fire PSA, full room PAU burnout scenarios are used where any PSA-related component and cable within the particular PAU is failed at the total PAU ignition frequency during Fire PSA quantification.

15.6.3.6.6 Evaluation of Multi-Component Scenarios

Fire propagation cases (multi-compartment scenarios) that involve spread from one PAU into another PAU are not currently postulated for the fire areas listed in the At Power Fire PSA. As the BWRX-300 design matures with greater detail, this analysis will be undertaken.

15.6.3.6.7 Evaluation of Potential Smoke Damage

Detailed Heating Ventilation and Cooling System (HVS) design information is not yet available for the BWRX-300 plant design; therefore, consequential failures from potential smoke damage cannot be assessed. This is subject to being updated in the future updates of the internal Fire PSA as the plant design matures. The Fire PSA models full burnup of the PAU in which the fire originates and therefore implicitly addresses potential smoke damage that could occur in that PAU.

15.6.3.6.8 Detailed Fire Modeling for Shutdown Condition

The shutdown PSA model is not yet constructed but is expected to be based on the Level 1 internal events shutdown PSA model. Detailed fire modelling follows the same process as described in the previous section except the evaluation of the applicability of shutdown conditions to the Fire PSA model is included in future updates of the internal Fire PSA.

15.6.3.6.9 Task Outputs and Preliminary Results

The following are the key task outputs:

- At Power fire PSA model
- Summary reports for the Probabilistic Fire Analysis: Key PSA Assumptions and Insights from fire analysis

From IAEA SSG-3, the following documentation requirements for the internal fire PSA are used:

- Description of the fire protection features specific to the plant, including passive and active mitigation features of the plant as well as partitioning of the plant into fire compartments
- Description of the specific methods and data used for assessing the fire hazard
- Specific changes made in the Level 1 PSA model for internal IEs aimed to account for the effects of internal fire
- Characterization of fire compartments
- Justification for the screening of particular fire compartments from the analysis
- Results of the specific analyses for detailed fire scenarios, the MCR, the electrical component room, multi-compartment fire, and multiple hazards
- The final results of the Level 1 PSA for internal fire in terms of CDF as well as selected intermediate results
- Report of the plant walkdown in support of fire analysis when available (this requirement is not applicable in the design phase PSA)

15.6.3.7 Internal Flooding Hazard

15.6.3.7.1 Introduction

The objectives of the BWRX-300 internal probabilistic flood analysis are to identify and provide a quantitative assessment of the radionuclide release risk due to internal flood events. It models potential flood vulnerabilities, in conjunction with random failures modeled as part of the internal events PSA. Through this process, flood vulnerabilities that could jeopardize core integrity and containment integrity are identified.

Floods may be caused by large leaks due to rupture or cracking of pipes, piping components, or water/fluid containers such as storage tanks. Another possible flooding cause is the operation of fire protection equipment.

Flooding associated from external sources such as localized flooding events and intense weather events (normally considered under the External Flooding Hazard Analysis) are excluded from this analysis. These external flooding events are reasonably precluded from the BWRX-300 probabilistic flood analysis based on adherence to the design conditions set forth in the envelope of BWRX-300 standard plant site parameters.

15.6.3.7.2 Methodology

The internal PSA flooding analysis is conducted by identifying and classifying potential flooding sources and events. Locations of components for which a flood or spray impact would result in a reactor trip, or failure of SSCs relied on to mitigate the impact of a reactor trip, are identified to develop flooding scenarios. Component location and data is compiled to generate a frequency of occurrence to represent the effects associated with each of the potential flooding events. In addition, an evaluation is performed to identify, screen, and quantify specific plant effects/failures associated with each flooding event. Finally, the BWRX-300-specific flooding frequencies and plant effects are applied to the PSA model to obtain risk results. For the BWRX-300 Flooding PSA model development, the following tasks are performed:

- Identification of Flood Sources and Component Locations
- Development of Flooding Scenarios
- Development of Flooding Frequencies
- Analysis of Flooding Scenarios

The internal PSA flooding analysis is based on the design basis for the BWRX-300 SSCs. The location of these features and their interaction with other BWRX-300 SSCs and equipment is critical to the flooding analysis. The current list of system components and location of equipment is assumed based on the current design and plant layout drawings and rely in large part on the component location analysis collected for the fire PSA step (e.g., common spatial information).

The development of BWRX-300-specific flooding scenarios requires a detailed analysis of data including plant component location, system capacity, and potential failure mechanisms. Characteristic scenarios are selected as representative of flood areas and subject to quantitative analysis following the identification of potential flooding scenarios.

In order to develop the severity and effect of potential flooding scenarios, data is collected from industry sources (NUREG/CR-6928) for the BWRX-300 equipment and system components. Failure data for consideration in the flooding analysis includes piping runs, pumps, valves, tanks, heat exchangers, and circulating water expansion joints. These failure rates in combination with types and capacity of system components located within specific flood zones are used to develop the flooding frequencies and frequency uncertainties. Flooding frequencies for both large break and small leak scenarios are developed for each flooding scenario.

Finally, the flood scenarios for each flood area are quantified to calculate a probabilistic risk value and summed to provide an overall risk analysis for the BWRX-300. NUREG/CR-4639, “Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR)” provides additional guidance for flooding analysis. The requirements for a flooding PSA are discussed in ASME/ANS RA-S-1.1-2024, Part 4.

Walkdowns are a critical task for a mature flooding PSA and are performed and documented after the construction of the plant has completed.

The BWRX-300 Internal Flooding PSA employs the following EPRI software:

- FRANX, version 4.4. This software uses the internal events model and adds high wind initiators.
- Fault Tree Reliability Evaluation eXpert version 1.8. This software generates cutsets from the fault trees produced in CAFTA.
- CAFTA, version 6.0b. This software is used to build the logic model of the plant, producing all the fault trees and event trees.
- ACUBE 2.0, version 2.0. This software provides a more accurate solution to the cutsets than conventional solutions.

15.6.3.7.3 Key Inputs and Assumptions

Key Inputs

The following are the inputs for the BWRX-300 internal PSA flooding analysis:

- At Power internal events PSA model
- BWRX-300 design features for protection against flooding
- Flood zone spatial information

Key Assumptions

The assumptions listed below are used in support of the BWRX-300 flooding analysis:

1. Flooding resulting from component ruptures considered in this analysis.

2. For each tank rupture, the entire tank inventory is drained.

3. Non-qualified equipment (motors or solenoids for valves, control cabinets and circuitry) affected by submergence is assumed to fail if the water level in the flood zone reaches a level of 0.30 m (1 ft) above floor elevation or if sprayed.

4. If equipment is failed by the flooding event in the source location or along the propagation pathway, it is assumed the equipment is failed at the start of the flood.

5. The expected effect of flooding electrical equipment such as MCCs, electrical cabinets, and terminal boxes, is a short to ground, removing power from the loads served by the component. This analysis addresses all such failures of electrical equipment as ground shorts.

6. Motor Operated Valves (MOVs) require the application of current to the motor to change the valve position. Without power, the valve remains in its current position. Flooding and/or spraying of a MOV causes the valve to fail as-is.

7. Passive components, such as pipes, heat exchangers, and tanks are not considered to be vulnerable to flooding effects because of the passive nature of the components. Buoyancy or impactive force from the flooding source water is not analyzed or considered for potential failures of SSCs in this revision of the BWRX-300 Internal Flood PSA.

8. Flooding has no effect on Test and Maintenance unavailability. Prior to and during a flood, the unavailability of equipment from Test and Maintenance is unaffected by flooding. It is assumed that no credit taken to restore any equipment being worked on following plant notification of a plant internal flood event. This is independent of worker location with respect to the flood propagation path even if restoration of the equipment is possible.

9. Flooding has no effect on CCFs, except for the common mode of the flood itself.

10. Water collecting in a stairwell or propagating into a stairwell preferentially continues to travel down the stairwell as opposed to propagating under a door adjacent to the stairwell.

11. The mission time of the active equipment credited in the flooding risk analysis will be 24 hours, unless a shorter mission time is specified by analysis, or a longer-term mission time is needed to ensure the plant is safe-and-stable. This is the same as the internal events PSA.

12. Concurrent flooding events from different sources are not considered in the flooding analysis.

13. Insulation properties are not lost at any point of cable routing, and interaction with water can only occur at termination points.

14. The internal flooding analysis uses the same system success criteria as used in the internal events PSA.

15. The solenoid valve associated with an air operated valve is located in the vicinity of the air operated valve.

16. Concrete walls are considered flood barriers. Concrete walls are assumed to be capable of withstanding the expected maximum flood loading and are assumed to remain intact throughout a flooding event.

17. Electrical circuit fault protection has been designed to provide protection for plant electric circuits via protective relaying, circuit breakers, and fuses. Therefore, loss of a component due to flooding does not result in the loss of the bus that supplies power to the affected component.

18. Detail routing of the floor and equipment drain system is not yet available, and propagation of flooding through the drain system is not considered.

19. Doors that connect the RB with other buildings are watertight. For the flooding analysis, these RB watertight doors are normally closed At Power. Opening of these doors would generate an alarm in the control room and it is expected that operations staff will respond to the alarm and close the opened door.

20. Floor hatches are closed during power operation and are assumed to provide a watertight seal between floor elevations. Chances of an opened floor hatch are assumed to be very low and if discovered by plant staff during routine walkdowns or plant inspections, the floor hatch opening would be expected to be closed.

21. Fire doors are not watertight.

22. Credit for door opening direction is not taken for flood propagation due to the preliminary nature of the BWRX-300 design. Typically, a flood would be expected to fail a door that fails away from its frame at a lower accumulation height than a door that fails toward its frame.

23. While High Energy Line Break (HELB) will be present in several PAUs, no credit for potential watertight or flood-tight doors is taken other than those doors at the exterior of the RB as the current design of the BWRX-300 plant does not provide such details which are expected as the plant design matures.

24. Fire protection details are not yet available, as such, all fire protection piping included in the flooding analysis is assumed to be wet pipe rather than dry pipe with actuation devices.

25. Equipment located in the yard is not considered susceptible to internal flooding damage as any flooding sources which propagate out of a building and into the yard area would be expected to dissipate freely and not have significant accumulation over a large area.

26. Human-induced mechanisms such as overfilling tanks or diversion of flow created by maintenance are not modeled because operating and maintenance procedures, from which the frequency and duration of maintenance are determined, have not been developed. These will be included in the PSA updates when these inputs are created.

27. Other events which could result in a release into a flood area based on plant experience are not initially identified because there is no BWRX-300 plant operating experience; however, the events will be included in the PSA updates if identified.

28. The flood volume retained by the drain and sump system is not credited in the flooding analysis for large flooding scenarios. Therefore, the capacity of the drain and sump system has not been estimated.

29. The amount of water retained by sumps, berms, dikes, and curbs is assumed to be negligible. This is conservative because retained water provides additional time for flooding response.

30. Flooding initiator frequencies remain constant over time for the flooding source types (pipes and components) and are assessed over the normal operating timeframe of one year.

31. Among the plant buildings, flooding frequency is the same for the same equipment type, regardless of difference in the quantity and characteristics of the equipment type that exist within the plant.

32. The analysis does not take credit for any mitigation of the flooding event by operators or automatic system functions. Therefore, the analysis assumes that all floods disable all potentially affected equipment in the areas where the flood is postulated to propagate.

33. Detailed flooding depth and timing calculations are not performed. In addition, the analysis does not take credit for the distance between flood sources and targets.

34. Recovery of the failed components after the postulated flood is not credited.

35. Based on the plant general arrangement drawings with component locations, assumed pipe routing is postulated for PSA purposes. Note that there are no details for pipe routing in the BWRX-300 plant design for plant systems at the time of this analysis beyond locations of some of the major system components.

36. Assessment of spray impacts is not performed in this revision of the BWRX-300 Internal Flood PSA as whole PAU impacts are assumed for any source location or any postulated propagation location which bounds any localized spray failures within the source location or along the propagation pathway.

37. The following PSA components are assumed to fail in a safe manner and to be train separated:

    - ICS actuation logic
    - CRD system actuation logic for control rod insertion
    - CRD HCUs
    - RPS actuation logic
    - The CRD pumps and associated components as used for the reactor core injection logic model, which is separate from the scram model logic in the internal events PSA model, are not taken as a fail-safe design, and a flood in that location would cause the active component(s) to fail to provide injection flow. Additionally, the current internal events PSA model represents the CRD HCUs by a single common cause event, and train specific HCU failures are not possible in that configuration.

38. FLEX equipment is located and stored in the general yard area (PAU YARD). There may be multiple connection locations in the final BWRX-300 design which require refinement of this assumption.

39. Some components remain unlocated due to the preliminary design of the BWRX-300 plant. These components are assigned to a location zone of “UNL” in the analysis. The UNL zone is failed for all quantified scenarios, serving to fail any UNL components for all scenarios. This is conservative since the components are impacted for all scenario locations, many of which would not be expected to impact the components once detailed location information is finalized. The “UNL” zone is failed in the Internal Flood PSA model database scenario list in which the zone names impacted for each scenario are listed in the Internal Flood PSA model

database table “Zone to Raceway” which has the zone names and which components are impacted for the particular zone name. The component names are mapped to PSA model basic events which are failed during the Internal Flood PSA model quantification.

40. Flood events inside the primary containment are not analyzed in this flooding analysis as those events are modeled in the LOCA analysis.

41. Some support systems, such as cooling (component and room), control and motive air or nitrogen, are not modeled in the Internal Flood PSA, since they are not modeled in the internal events PSA.

42. Rupture of poured concrete tank structures such as the fuel pool is not expected to occur and is not assumed. Rupture of metal tanks is included in the Internal Flood PSA.

43. The Intake Building is expected to have open communication within its structure for flood propagation flow. This assumption can be revised as needed as the BWRX-300 design details mature.

44. The chilled water components for room cooling are assumed to be located in PAU 1661 in the RB, though no details of the design or location are available at the time of this Internal Flood PSA. Other buildings such as the Turbine, Rad Waste, and CBs have the chilled water components located on the roof based on the preliminary design information.

45. Plant Cooling Water System (PCW) is not currently divided into RB and TB cooling water for heat loads in the respective buildings. Based on the preliminary design information, only the PCW is defined at the time of the Internal Flood PSA.

46. The zone name to component name and component name to PSA model basic event name associations are used within the FRANX file as typically done in commercial nuclear Fire PSA studies. As piping characteristics are defined with more certainty, the typical flood-based features of the FRANX program can be utilized such as pipe diameter, flood rate, flood source volume, and potentially more limited propagation impacts.

15.6.3.7.4 Task Outputs and Preliminary Results

The following are the key task outputs:

- At Power flooding PSA models
- Summary reports for the Probabilistic Internal Flooding Analysis
- Key PSA Assumptions and Insights from internal flooding analysis

From IAEA SSG-3, the following are the documentation requirements for internal flooding PSA:

- Description of the specific methods and data used to assess the internal flooding hazard
- Specific changes made to the Level 1 PSA model for internal IEs aimed at accounting for the effects of internal flooding
- Justification for the screening of particular flooding scenarios from the analysis

- Results of the detailed analysis for flooding scenarios, including description of the scenarios, and significant assumptions made in the analysis

15.6.3.8 External Hazards

Other EEs, (e.g., heavy wind causing loss of external grid, snowstorm blocking diesel generator combustion air intakes, and frazil ice blocking plant intakes): PSA quantification is based on the results of screening analyses that identifies potential significant contributors.

The internal and external hazard analyses are initially bounding assessments that show significant design margin for these hazards. The frequencies of internal events are based on generic industry data or bounding site data and are applied in a bounding manner. The fault trees and event trees developed for the internal events evaluations are used in the internal and external hazards analyses to the extent possible, using logic flags or additional fault tree modeling that account for the failures induced by the internal or external hazard events.

15.6.3.8.1 High Wind Hazard

The BWRX-300 high wind analysis explicitly quantifies ESs and containment releases initiated by both tornado and straight-line winds. Straight winds are lesser velocity winds that pose minimal challenges to the plant design. Note that straight-line winds of lesser velocities are not included in the BWRX-300 high wind risk analysis. The consequences of these winds are addressed as part of a weather-related LOPP IE in the At Power internal events PSA model.

Methodology

The high wind risk analysis involves the following major steps:

1. Tornado high wind event frequency

2. Tornado-induced plant effects

3. Analysis of tornado-induced release category frequencies

4. Analysis of straight-line wind effects

The high wind PSA is conducted by identifying and classifying potential high wind events. The Standard Plant Design high wind hazard is being used for the CRN-1 high wind PSA. Data is then compiled to generate a frequency of occurrence to represent each of the potential high wind events. In addition, a qualitative evaluation of the BWRX-300 SSCs is performed to identify specific plant effects/failures associated with each high wind event. Finally, a BWRX-300-specific high wind event frequency and plant effects are applied to the at-power and Shutdown PSA model to obtain risk results. High wind hazards are to be characterized by their impacts (e.g., dynamic load from gusts, averaged loading, rotation velocity, pressure differential, tornado path, missile impact potential). The FSAR will provide results of the CRN-1 site-specific high wind PSA as appropriate.

The High Wind HRA addresses modeling of human failure events for the High Wind PSA. This HRA is performed to support quantification of high wind scenarios. For all human failure events in the high wind scenarios, the following scenario-specific impacts on Performance Shaping Factors (PSFs) for control room and ex-control room actions, as appropriate to the HRA methodology being used, are included:

- Additional workload and stress (above that for similar sequences not caused by high wind)
- Cue availability
- Effect of high wind on mitigation, required response, timing, and recovery activities (e.g., accessibility restrictions, possibility of physical harm)
- High wind-specific job aids and training (e.g., procedures, training exercises)

Both at-power and shutdown high wind-induced accident scenarios and releases are quantified and analyzed. The following references provide guidance for the high wind risk analysis:

- NUREG/CR-6890, Volume 1, “Reevaluation of Station Blackout Risk at Nuclear Power Plants Analysis of Loss of Offsite Power Events: 1986-2004”
- Enhanced Fujita Scale (EF-Scale), Wind Science and Engineering Center, October 10, 2006, Revision 2
- NUREG/CR-4461, “Tornado Climatology of the Contiguous United States,” Rev. 1
- EPRI, “High Wind Equipment List and Walkdown Guidance” 3002008092, December, (2016)
- NUREG/CR-7004, Technical Basis for Regulatory Guidance on Design-Basis Hurricane-Borne Missile Speeds for Nuclear Power Plants, 2011
- NUREG/CR-7005, Technical Basis for Regulatory Guidance on Design-Basis Hurricane Wind Speeds for Nuclear Power Plants, 2011

ASME/ANS RA-S-1.1-2024 includes a hazard screening section (Part 6) and a High Winds PSA section (Part 7). The enhanced standard, including the guidance in the non-mandatory appendix, provides requirements and guidance on performance of a detailed High Winds PSA. The changes in Part 7 are the most significant changes in the PSA Standard and reflect the advancements in High Winds PSA in the last decade. Included in the standard requirements is the consideration of correlated hazards and hazard effects, such as the potential for local flooding associated with high winds or the impact of high wind-driven rain. This standard is utilized for both the screening of potential hazards and the analysis of any unscreened hazards.

Key Inputs

Site-specific data are inputs for the external hazards PSA analyses. Site-specific wind hazard analysis generally requires the use of regional data. The size of the region requires judgment and depends on the regional climatology and type of wind hazard, the number of years accurate records are available, the extent and quality of the data, and the hazard’s spatial variability within the region. In the design phase, it is possible to select a bounding region for a high wind hazard; however, a bounding region for tornadoes may not be the bounding region for hurricanes.

The high winds equipment list is provided by the high winds PSA documentation and is evaluated based on the internal events PSA required functions and supporting SSCs. The internal events PSA is used to quantify the high winds PSA with modification of the damaged or potentially damaged SSCs.

Key Assumptions

The following assumptions were used in conducting the BWRX-300 high wind risk analysis:

- The winds classification used in the BWRX-300 high wind analysis is based on the Saffir-Simpson scale.
- The tornado winds classification for the BWRX-300 high wind analysis is based on the recommended wind speed ranges of the Enhanced Fujita scale (EF-scale).
- All at-power BWRX-300 high wind analyses include high winds assuming the plant is operating. This approach is conservative for the high wind analysis where sufficient advanced warning and procedures allow the plant to be placed into a safe condition (shutdown operations) prior to the high wind event.
- For the BWRX-300 high wind analysis, the frequency of occurrence data for tornadoes high wind events is compiled from sources based on the Enhanced Fujita scale (EF-scale) for classification of tornadoes. These data are correlated to the BWRX-300 EF-scale of tornado classification.

Task Outputs and Preliminary Results

The following are the key task outputs:

- At-power high wind PSA models
- Low-power and shutdown high wind PSA models
- Summary reports for the Probabilistic High Wind Analysis
- Key PSA Assumptions and Insights from high wind analysis

From SSG-3, the following are the documentation requirements for High Winds PSA:

- Description of the specific methods and data used for determining the hazard curves for high winds
- Specific changes made in the Level 1 PSA model to account for effects relating to high winds
- List of all SSCs considered in the analysis along with the justification for the SSCs that are screened out from the analysis
- Methodology and data used to derive wind fragilities for all SSCs modeled in the Level 1 PSA
- Final results of the Level 1 PSA in terms of core damage as well as useful intermediate results

15.6.3.8.2 Seismic Hazard

The BWRX-300 PSA develops a Seismic Probabilistic Safety Assessment (SPSA) for the scoping assessment. The CRN-1 site-specific hazard is being used for the Standard Plant Design Seismic PSA.

The seismic analysis integrates the seismic hazard analysis, the structural/mechanical dynamic response, the component fragility, and the plant systems response. The plant operational and emergency systems are represented by event and fault trees. The analysis gives insights into the dominant contributors for the seismic risk. The FSAR will provide results of the CRN-1 site-specific seismic PSA as appropriate.

For a more detailed seismic risk analysis, the scheme can usefully be thought of as consisting of a set of analyses at each of a sequence of ground motion input levels (e.g., 0.1g, 0.2g, 0.3g). The seismic hazard analysis provides the annual frequency of exceeding each level of peak horizontal ground acceleration. The plant components' behaviors at each level are evaluated to determine their failure probabilities. A fragility for each component, based on its design and its level in the plant is determined. HEPs are adjusted based on ground motion levels to account for increased difficulty due to seismic damage, as applicable.

In general terms the plant consists of normally operating and emergency standby systems and components. The failure during an earthquake (induced either directly by excessive inertial stresses or indirectly following the failure of some other item, e.g., a crane) of an operating component leads to a change in the state of the plant. In that case, various scenarios can follow depending on the "initiating" event and the status of other sub-systems. The analysis represents these possible chronological sequences by an event tree.

The event trees and the associated fault trees model the sub-systems down to the level of individual components (e.g., pumps, valves, tanks). These trees are expanded from those used in the scoping assessment and include items not found in non-Seismic PSA fault trees, such as buildings whose failure induce the seismic failure of several system components.

Simple combination of the frequency at each ground motion level and the two sets of probabilities (seismic input and core damage or release category frequency given the input) yields the final results.

Model Steps

The contribution of seismically induced events for the three ground motion input levels 0.1g, 0.2g, and 0.3g to the plant risk is estimated using the approach outlined in the following steps:

1. An estimate of seismicity of the site is characterized in terms of a seismic hazard curve.

2. Event tree models are constructed to represent possible responses of systems important to safe operation or shutdown of the plant under at-power and low-power and shutdown conditions following an earthquake, and to evaluate the contribution of the earthquake to plant risk.

3. Seismic fault trees are constructed for the systems and structures of interest. The fault trees identify and include the structures and components in each system that are subject to functional failure as a result of an earthquake.

4. Seismic fragilities are assessed for each component and structure included in the analysis.

5. The fault trees and event trees are evaluated including both seismically and randomly induced failures to determine ES and release category frequencies. Random failures during the ESs are determined from the internal events analysis.

6. Containment event trees are then developed. These event trees draw from the insights gained in the internal events analysis to determine the system and phenomenological issues of importance. In addition, special system considerations are given to unique failure modes identified from the seismic event trees. The results are then grouped into plant damage states.

7. Risk analyses are then performed using the frequencies and the consequences of each plant damage state to describe the core damage and release category frequency associated with each event tree sequence.

Quantification of Seismic Risks

The following seismic risk contributors are quantified:

- At-Power
- Seismic-Induced Fire
- Seismic-Induced Flood
- Shutdown

The risk analysis is performed using the EPRI software tool FRANX. The inputs to the program are the following:

- Seismic hazard function
- Seismic fault trees based on the internal events fault trees
- Seismic event trees
- Unavailability of systems and components that survive the seismic event (including seismic HEPs)
- Component and structural fragilities (probabilities)

Key Inputs and Assumptions

Key Inputs

- Site-specific data is requested as inputs for the external hazards PSA. Site-specific seismic hazard curves are used for the SPSA.
- Plant-specific internal events models for at-power and LPSD conditions

- Characteristics of all PSA-modeled SSCs that are used for the seismic fragility analyses

Assumptions

Similar to assumptions in previous GVH developed seismic PSAs, the following principal assumptions can be used:

General

- Identical equipment fails at the simultaneous acceleration in case of a seismic event. Identical equipment is defined as being of the same type and located at the same elevation. This means that the failure events for identical pieces of equipment are 100% correlated. There is no credit taken for redundant systems of the same type.
- No recovery of offsite power or diesel generators is assumed. This is a severe restriction on mitigation capability.
- No repair of mechanical failures is assumed.
- Because of extremely low fragility of ceramic insulators in the switchyard, it is assumed that seismic events always result in an LOPP. Conversely, if the offsite power is not lost, the earthquake is of relatively low intensity, and is not analyzed further.
- It is assumed that valves and dampers that are failed seismically fail “as-is”; (i.e., in the position they were in when the seismic event occurred).
- I&C components have a very high fragility compared to other components modeled in the PSA and can be excluded from the model without affecting the results.
- The components and systems that survive the earthquake have to function as designed to mitigate the accident. The probability of failure to function is estimated based on the internal events PSA model. The assumptions made in the internal event PSA are applicable for the Seismic PSA also. For instance, the failure to run probability is calculated as a 24-hour mission time for all components unless otherwise stated or longer-term operation is needed to support safe-and-stable operation.

Structure Fragility

- Building structural failure causes failure of all equipment in the building. In general, failure of key buildings (such as the RB) yields failure that leads to loss of all mitigation.
- The failure of the containment leads directly to release.

Others

- The random failure probability of a normally running system is negligible compared to the seismic failure of the system.
- Random failure probabilities of systems that survive the seismic event are calculated using the at-power PSA model, modified to reflect the available support systems. In these evaluations, the HEPs in the at-power PSA model are left unchanged.

Task Outputs and Preliminary Results

The outputs of the SPSA are:

- A Seismic Equipment List documenting the Seismic Capacity and Fragility evaluation of SSCs. In the scoping assessment, the fragility evaluation is by function level rather than by component.
- SPSA models for at-power and LPSD conditions, which also include the contributions from seismic-induced fires and floods.
- Documentation of seismically initiated accident sequences and risk metrics of core damage and release categories.
- Key risk insights and assumptions based on the SPSA model results.

From SSG-3, the following are documentation requirements for Seismic PSA:

- List of SSCs considered in the Level 1 PSA for seismic hazards
- Fragility characterization and the technical bases for them for each structure, system, and component
- Quantified probabilities of damage for the range of seismic hazards modeled in the Level 1 PSA
- Significant failure modes for SSCs and the location of each structure, system, and component
- Specific adaptations made in the Level 1 PSA model for internal IEs to account for the impact of the seismic events
- Comprehensive information on the dependencies (in particular, spatial interactions) modeled in the Level 1 PSA for seismic hazards, as well as any assumptions applied to eliminate or decrease the impact of the dependencies
- The basis for screening out any SSC is described fully
- The methodology and procedures used to quantify seismic fragilities should be documented, including the following aspects:
  - Seismic response analysis
  - Steps involved in screening
  - Plant walkdown
  - Review of design documents
  - Identification of critical failure modes for each structure, system, and component
  - Calculations of fragilities for each structure, system, and component
- The procedure for plant walkdowns, the compositions of walkdown teams, and the observations and conclusions made from the walkdown is fully documented.

15.6.3.9 Low Power and Shutdown Risks Probabilistic Safety Assessment

The LPSD PSA is developed following the requirements of the standard for LPSD, ANS/ASME 58.22-2014, “Requirements for Low Power and Shutdown Probabilistic Risk Assessment,” (Reference 15.6-28), which is issued for trial use and pilot application. After the trial use, feedback was provided to the PSA Standard committee and the LPSD standard is being revised.

The radionuclide release risk during shutdown is analyzed as part of the Level 2 PSA. Loss of the CUW/SDC, loss of PCW, and LOPP are all investigated. The approach for the LPSD PSA is similar to the Full-Power PSA, involving fault trees and event trees used in determining the shutdown risk for each IE analyzed.

The evaluation encompasses plant operation in hot shutdown, cold shutdown, and refueling. The plant is transitioned through several plant operating states that are distinguished in the LPSD PSA by different plant conditions and configurations. The LPSD PSA addresses plant operating states where there is fuel in the reactor vessel. It includes all aspects of the NSSS, the containment, and all systems that support operation of the NSSS and containment.

The ESs in the shutdown PSA are classified according to whether the core is damaged. All shutdown core damage sequences lead directly to radionuclides release to the environment (containment is assumed to be open at the time of the IE). In the final PSA, this assumption is revisited to accurately assess the PSA results.

The critical safety functions credited in the shutdown model are decay heat removal and inventory control. The containment function is credited for POSs where containment is not open. For POSs where the containment is open, containment integrity is not relevant to any modeled functions. Reactivity control is assumed to have no significant effect on the shutdown model. Electrical power availability, as well as the availability of other support systems, are modeled for the effect on decay heat removal. LOPP is evaluated as an IE, and power dependencies for systems are included in the model in a similar manner to the at-power PSA.

The LPSD PSA includes fuel and heavy load movements where the PSA analyzes heavy load drops that can cause fuel damage, core damage, or large releases. The PSA results for heavy load drops considers the following events:

- Fuel pool leak due to rupture At Power
- RPV leak due to rupture (below top of fuel) during Low Power and Shutdown
- RPV leak due to rupture (at feedwater nozzle) during Low Power and Shutdown
- Fuel pool leak due to rupture during Low Power and Shutdown
- Loss of ICS Train A to SDC during Low Power and Shutdown
- Loss of ICS Train B to SDC during Low Power and Shutdown

15.6.3.9.1 Methodology

The references provide guidance documents for the low-power and Shutdown PSA model development, including the LPSD PSA Standard American Society of Mechanical Engineers, “Requirements for Low Power and Shutdown Probabilistic Risk Assessment,” ASME 58.22-2014, (Reference 15.6-28).

15.6.3.9.2 Plant Configurations in Low Power and Shutdown Conditions

Differences between the Low-Power and Shutdown PSA and the Power Operation PSA are attributed to:

- Plant operating mode
- Plant operating state including configuration
- Time after shutdown
- Reactor vessel and containment status
- Vessel and core temperatures
- Fuel location
- Availability of required systems and support systems

This analysis addresses the BWRX-300 risk associated with a refueling outage. The systems modeled are evaluated based on anticipated activities associated with refueling operations.

To develop a suitable shutdown model, multiple bounding plant configurations/POSs are defined with similar characteristics in relation to the residual heat and the availability of systems.

The outage plant operating mode and POS are used to define the initial plant condition for individual ES quantification.

Once the outage POSs are defined, the duration of each is estimated to determine its contribution to the overall calculation of annual CDF. The duration is expressed in hours per refueling outage.

The Shutdown PSA considers outage plant configurations representative of the possible plant configurations during shutdown.

15.6.3.9.3 Initiating Events

The IEs that challenge normal operation including the critical safety functions (e.g., heat removal, inventory control) during shutdown operations are determined. A shutdown IE is defined as any event that challenges normal operation and requires action to prevent core damage.

15.6.3.9.4 Event Trees

The LPSD event tree construction considers the following aspects:

- Chronological order of system actuation

- Grouping of mitigating systems by safety functions

The ES analysis and end state nomenclature are the same as the at-power PSA. The associated success criteria analysis is performed similar to the at-power success criteria analysis. However, success criteria analysis for LPSD may be approximated by hand calculations.

15.6.3.9.5 System Analysis

The unavailability of a system to perform its safety function on demand is evaluated by fault tree analysis similar to the fault tree analysis approach.

The necessary fault trees are identified following construction of the event trees. These fault trees represent the nodes included in the event trees and any required support system fault trees.

Maximum use is made of the fault trees developed for the Full-Power PSA. Potential differences between the at-power and the shutdown fault tree models may result from differences in:

- Maintenance unavailabilities
- Success criteria between at-power and shutdown condition
- Initial system configuration between at-power and shutdown condition
- Human actions

Maintenance events used in the at-power model are not adjusted for the LPSD PSA, other than for POSs where some systems are entirely unavailable (for the POS). Additional maintenance events may be possible at LPSD that do not normally occur at full power. The evaluation of maintenance for each POS is justified and documented.

15.6.3.9.6 Quantification

The shutdown ES analysis models the effects on the following two critical safety functions during shutdown:

- Decay heat removal
- Containment inventory configuration changes in the containment boundary and reactor head during refueling

IEs and associated frequencies are identified that challenge the above critical safety functions. Event trees are developed specific to the shutdown configurations and the system fault tree analysis is based on at-power fault tree models adjusted to match shutdown conditions. The model development and quantification are performed using CAFTA, PRAQuant and FTREX in a similar manner to the at-power PSA.

15.6.3.10 Spent Fuel Probabilistic Safety Assessment

Spent fuel damage evaluation is required for the plant-specific BWRX-300 PSA because fuel damage frequency is an important contributor to release risk relative to other low risk contributors. For accidents where the spent fuel is damaged outside the reactor core, the term spent fuel damage is applied. In particular, the fuel damage frequency may be an important contributor to release risk. The term “fuel damage” represents damage to the fuel outside the reactor vessel, while “core damage” is used for damage inside the vessel.

15.6.3.10.1 Methodology

A separate PSA analysis investigates the ESs leading to spent fuel damage. The analysis estimates the related ESs and their frequencies, and it is documented separately as part of the PSA. The analysis covers both power operation and outages. Sabotage is explicitly excluded from the calculation of the spent fuel damage frequency.

The overall process of event tree analysis, success criteria analysis, and fault tree analysis, is the same process used in the LPSD PSA. Generally, the time available for responding to a fuel pool IE is much longer than the LPSD IE response.

ESs are developed and quantified that credit potential recovery actions taken by the operator.

Key Inputs

- Site-specific EE
- Site-specific spent fuel handling equipment design
- Plant procedure for heavy load lifting and spent fuel handling

Task Outputs and Preliminary Results

This task develops the Spent Fuel Damage PSA models. A number of IEs are expected to be screened with qualitative assessments. The outputs of this task include:

- A screening analysis for the identification of IEs applicable to Spent Fuel Damage
- Qualitative assessments supporting the screening analysis
- Spent Fuel Damage event trees for unscreened IEs
- Spent Fuel Damage system models
- Quantitative assessment for unscreened IEs and their associated event trees and system models
- Key risk insights and assumptions.

15.6.3.11 Start-up and Shutdown Transition Risks

In the BWRX-300 PSA shutdown models supporting the design, the start-up and shutdown transition risks are included in the at-power conditions.

Planned shutdown and start-up transition states are difficult to model with PSA for several reasons. The reactor is under continuous transition, e.g., main circulation flow. Temperature and pressure in the reactor vary significantly during the operation. Several manual operator actions and decisions are required. In addition, the operation may require the use of many systems and/or components that are not used during power operation. Identification of risks is also more challenging as it requires detailed understanding of the systems related to process and control. For these reasons, certain assumptions and simplifications are performed to create a usable and practical PSA model.

Key Inputs and Assumptions

- At-Power BWRX-300 PSA models (note EEs low-power and shutdown models are developed in the EE sections)
- Representative BWRX-300 Refueling Outage Plan Key Assumptions
- Containment is open during the outage

15.6.3.12 Systems Credited in the Level 1 Probabilistic Safety Assessment

The following design features are modeled in the PSA.

Isolation Condenser System

The ICS function modeled in the PSA is removing decay heat from the reactor by condensing steam in the ICS heat exchangers.

Reactor Isolation Function

The reactor isolation function mitigates the effects of large and medium sized pipe break LOCAs.

The term “RPV isolation system,” is used to refer to RIV closure. The system model is a surrogate for valves that are part of the Nuclear Boiler System (NBS).

Ultimate Pressure Regulation System

The RPV Ultimate Pressure Regulation system provides an emergency pressure relief path in the event of a severe pressure transient.

Control Rod Insertion System

The CRD system performs several insertion functions and consists of these design features: FMCRDs, HCUs, and the CRD hydraulic subsystem.

The PSA-credited function of the control rod insertion is to insert negative reactivity into the reactor core rapidly upon a scram signal. Subcriticality is achieved with the negative reactivity insertion and terminates fission heat generation.

The CRD system also provides an RPV inventory makeup function.

Feedwater Runback System

The feedwater runback system provides negative reactivity in an ATWS condition by reducing feedwater flow resulting in increased core voiding (the BWRX-300 has a negative reactivity void coefficient).

The FW runback system is part of the anticipatory trip system, which provides a signal to the operating FW pump to reduce the flow to the reactor. It is part of DL2 and is designed to insert negative reactivity in the event of a condition that could result in a scram signal.

Containment Isolation System

The containment isolation system provides isolation of the containment in the event of accidents or other conditions and prevents the unfiltered radioactive release before they exceed allowable limits.

The containment isolation system uses sensors that interface with the I&C and electrical penetrations required to route signals in and out of containment. This system is designed to automatically isolate the containment and prevent the release of radioactive contaminants into the environment in the event of a condition that could result in core damage.

Boron Injection System

The Boron Injection System (BIS) provides a separate, diverse means, D-in-D backup system to the CRD system for inserting negative reactivity into the reactor core for Beyond Design Basis Accidents (BDBAs). All equipment is located outside primary containment to allow access for testing and inspection activities during all plant operating conditions.

The BIS utilizes an aqueous solution of sodium pentaborate for reactivity control.

CRD Injection Function

The BWRX-300 CRD system consists of the FMCRDs, the HCUs, and the CRD hydraulic subsystem. The focus of this system for PSA is the CRD hydraulic subsystem that is used for inventory makeup and flow to the SDC pumps.

The CRD hydraulic subsystem provides clean, demineralized water that is regulated and distributed to provide charging to the scram accumulators and purge water flow to the FMCRDs during normal operation. The CRD hydraulic subsystem is also the source of purging water to the SDC system pumps and the NBS reactor water level reference leg instrument lines. Additionally, the CRD hydraulic subsystem provides high pressure injection to the reactor. This makeup water is supplied to the reactor via the CRDs.

Shutdown Cooling System

The SDC system provides long term decay heat removal during shutdown phases of operation. SDC consists of two independent trains designated as Train A and Train B. Each train suction is independently connected to an ICS condensate return line outside containment, downstream of the containment isolation valves.

The SDC return piping is connected to the FW lines outside the containment.

Power Conversion Function

The Power Conversion System, as modeled in the PSA, removes decay heat from the reactor by providing a pathway for FW from the condenser to the reactor and for steam from the reactor back to the condenser. Because the PSA model assumes a scram has occurred or is warranted, the full power function of providing steam to the turbine for power generation is not considered.

There is only one function for the power conversion function modeled in the PSA. This function is providing water from the condenser, via the condensate pumps, feedwater heaters and feedwater pumps, to the reactor where it is heated, and steam is produced. Steam is transferred to the condenser via the turbine bypass valves.

Cooling Water Systems

The PCW removes heat from loads in both the RB and TB, dividing PCW loads into two subsystems: Reactor Component Cooling Water Subsystem and Turbine Component Cooling Water Subsystem.

The PCW is a closed cooling water system supported by the Circulating Water System.

DC Power

The electrical distribution system is an integrated power supply and distribution system for the power plant. Three plant systems constitute the overall electrical system: Safety Class 1 Electrical Distribution System, Safety Class 2 and 3 Electrical Distribution System, and the Non-Safety Electrical Distribution System.

The electrical system powers automatic shutdown and decay heat removal functions. The Safety Class 1 portion of the electrical distribution system (includes DC) is limited to supplying power to Safety Class 1 SSCs within the RB rooms.

Safety Class 1 DC power three division (A, B, and C) arrangement supplies DC power to various loads. Each division has a DC battery and two redundant battery chargers powered from the Safety Class 2 and 3 Electrical Distribution System AC power system.

The primary load of the Safety Class 1 Electrical Distribution system is the DCIS.

AC Power

AC power modeled in the PSA, is providing medium and low voltage AC power to plant components, based on the BWRX-300 preliminary standard plant design.

Corium Shield

The corium shield prevents core melt from damaging the containment liner once the core has broken through the bottom of the RPV. The system is still in design and may not reflect the final system.

The corium shield is a risk reduction design feature that includes a refractory material below the RPV. After core damage, in cases where the RPV is assumed to be at a low enough pressure to preclude direct containment heating, the core debris eventually migrates from the core region to the RPV lower head and exits the through a breach (e.g., lower head failure or CRD housing failure). The refractory material below the RPV prevents molten core-concrete interaction and any potential ablation of the basemat and accompanying flammable gas generation.

Passive Containment Cooling System

The Passive Containment Cooling System (PCCS) rejects heat into the reactor cavity and/or equipment pool above containment. Supply and discharge connections from the pool are connected to closed loop piping within containment. Heat transfer occurs from the containment to the PCCS by natural convection and condensation.

Containment Filtered Venting

The Containment Pressure Relief System vents pressure in the containment because of LOCAs, RPV pressure relief, or core melt. By providing a vent to the containment, pressure and temperatures can be maintained. The release is filtered to reduce the amount of fission products. Components such as duct work and filters that are the normal path of containment do not have the ability to vent the pressures seen during an accident. Venting is assumed to be required when no other containment heat removal method succeeds and in certain special situations in which there is an excessive containment pressure load beyond the capacity of Passive Containment Cooling System. The system is simply comprised of a rupture disk, manual bypass, and an air operated isolation valve. Currently, the design has the effluent go to a pool of water to help limit radiological dispersion.

Heating Ventilation and Cooling System

The HVS provides normal room atmospheric temperature and ventilation control and room cooling during accident scenarios for locations determined to require such cooling.

Systems assumed to require the HVS for successful operation have placeholder transfer gates with an assumed failure point-estimate screening value used in PSA model quantifications.

Plant Pneumatic System

The Plant Pneumatic System (PPS) function provides motive and control air or nitrogen to various plant components. The pneumatic system supports multiple systems and valves.

There is no detailed PPS design information available at the time of this analysis, as such, no detailed system model or documentation notebook is developed.

Systems assumed to require PPS for successful operation have placeholder transfer gates with an assumed failure point-estimate screening value to use in PSA model quantifications.

15.6.4 Level 2 Probabilistic Safety Assessment

The tasks for the Level 2 PSA accident sequence analysis involve the following steps with some that are similar to the Level 1 PSA:

1. Develop an Interface between the Level 1 and Level 2 PSA
2. Identify and Model Safety Functions and Operator Actions
3. Containment Performance Analysis
4. Accident Progression Analysis and Development of Containment Event Trees
5. Release Category Analysis
6. Source Term Analysis – discussed below
7. Quantification and Interpretation of Results – discussed below

The steps not covered elsewhere supporting the Level 2 PSA accident sequence analysis are discussed in the following sections.

15.6.4.1 Identify and Model Safety Functions and Operator Actions

The engineered safety features and the operator actions which can influence the progression of SAs, the containment response, and the transport of radioactive materials are identified and highlighted. This exercise should include the RB, e.g., the treatment of hydrogen. The safety features and operator action should be identified in line with the site-specific Severe Accident Guideline developed prior to operation, available plant design information, and design requirements for the SA conditions.

Examples of the features that need to be identified are:

- The Primary Containment Vessel (PCV) design
- PCV isolation (Isolation Valves: Automatic/Manual)
- Prevention of re-criticality
- Prevention of hydrogen combustion in PCV before PCV venting

- Prevention of hydrogen combustion in PCV after PCV venting
- Prevention of hydrogen combustion in RB
- RPV depressurization
- RPV injection and RPV water level control
- PCV venting
- Accident management actions to recover the support systems
- Recovery actions such as offsite power recovery, if applicable

If the mitigation features and operator actions are credited, the survivability and operability in SA conditions (e.g., adverse environment, system limitations, interlock/trips, and degraded plant indication system) should be reviewed.

An HRA is performed for the identified operator actions. The HRA methodology is the same as that in the Level 1 PSA. The following are examples of the specific considerations for the HRA in the SA conditions:

- Dependency between the Level 1 PSA and the Level 2 PSA
- Stress for operators in SA conditions
- Environmental effect against the radiation, especially for the field operation
- Evacuation from MCR to an alternate location
- Instrument failure affecting human error probability

For HRA, a timeline analysis is performed including the time of operator cue, procedural guidance, and time available.

15.6.4.2 Containment Performance Analysis

Containment Performance Analysis includes the following key steps:

1. Identification of the Containment Failure Mode

2. Containment Structural Analysis

3. Analysis of Containment Bypass Events and Containment Isolation Failure

15.6.4.2.1 Identification of the Containment Failure Mode

The containment failure modes are analyzed to identify the accident conditions that might lead to the containment failure. This analysis begins with a Level 2 accident progression literature survey and screening of SA phenomena.

The examples of the literatures are shown below:

- NUREG/CR-4551 (Grand Gulf and Peach Bottom analyses)
- Recommended sensitivity analyses for an Individual Plant Examination
- EPRI ALWR requirements document

A list of key phenomena where accident conditions are considered is produced and then the evaluation methods for each phenomenon are studied. Among key SA phenomena, overpressure and overtemperature are important containment failure modes that are considered in all SA scenarios. Therefore, representative SAA are investigated to determine the range of containment pressures and containment temperatures during a SA.

15.6.4.2.2 Containment Structural Analysis

As a result of the above steps, key SA phenomena and their evaluation methods are identified. The primary objective of an assessment of containment performance is to develop a realistic characterization of the modes (mechanisms) of, and criteria for, containment leakage or failure under SA conditions. Design criteria for the containment are generally not adequate measures of capacity of the containment because of the safety factor built into such values. Actual values of the ultimate pressure capacity of the containment have sometimes been found to exceed design values by a factor of two to four. Further, containment design limits may not account for the harsh environmental conditions that can develop inside the containment during a SA, that often require consideration of entirely new failure modes.

Detailed information on the structural design of the containment and containment penetrations is required to generate a realistic assessment of containment performance limits.

15.6.4.2.3 Analysis of Containment Bypass Events

The containment isolation is one of the headings in the containment event tree and the containment isolation failure is captured by solving fault trees for the failure of this function. The containment bypass event and containment isolation failure lead to the loss of the containment function and the large fission product release.

15.6.4.2.4 Analysis of Containment Isolation Failure

The containment isolation is one of the headings in the containment event tree and the containment isolation failure is captured by solving fault trees for the failure of this function. This mode is the loss of containment isolation. The fission product is released from the RPV to the RB when the core damage occurs.

15.6.4.2.5 Accident Progression Analysis and Development of the Containment Event Trees

This task consists of following two steps:

- Step 1: Accident Progression Analysis
- Step 2: Accident Sequence Analysis (Development of containment event trees)

These two tasks are closely coupled since the phenomenological and probabilistic aspects of possible accident progressions are determined.

Accident Progression Analysis – This analysis involves three general steps discussed below:

Selection of Accident Sequence and Analysis Condition

There are many possible scenarios which lead to core damage. Therefore, some cases are selected as the representative cases of unmitigated analysis through a grouping process. The selection of IEs and availability of mitigation systems are determined based on the characteristics of plant responses and effect on fission product releases. Key items to select them are identical to those in the Plant Damage States (PDS) identification process.

Accident Progression Analysis

Accident progression analyses for each representative cases are performed in order to obtain the data which is necessary for the development of containment event trees such as plant thermal-hydraulic behavior, chronology of accident progression (the timing of the core damage and the containment failure), and containment loads from SA phenomena. Accident progression analyses include both cases in which mitigation systems fail or succeed.

The consequence analysis software (e.g., Modular Accident Analysis Program (MAAP) codes, MAAP (Reference 15.6-29) or MELCOR codes) can be used for the accident progression analysis in the Level 2 PSA of BWRX-300. This analysis includes models for the important accident phenomena that might occur in the RPV, in the PCV, and in the RB. The consequence analysis software MAAP and MELCOR calculate the progression of the postulated accident sequence, including the deposition of the fission products, from IEs to either a safe, stable state or to an impaired containment condition (due to overpressure or overtemperature). The software also calculates the amount of fission product released to the environment.

Chronological results of accident progression analyses are used as indication of available time for operator actions such as the coolant injection after the core damage, recovery actions.

Analysis on Occurrence Probability of Physical Phenomena

Occurrence probabilities of physical phenomena such as in-vessel / ex-vessel steam explosion, Direct Containment Heating, and Molten Core Concrete Interaction are evaluated. Each branch probability is estimated using information in NUREG/CR-4700 “Containment Event Analysis for Postulated Severe Accident: Grand Gulf Nuclear Station Unit 1,” (Reference 15.6-30) and is supplemented by plant-specific assessments, including insights developed from analytical approaches. The probability of the phenomenological events should be reviewed if the treatment is very conservative against the Level 2 PSA results.

15.6.4.2.6 Accident Sequence Analysis

Development of Containment Event Tree

Containment event trees are developed for each plant damage state. In the Level 2 PSA, a generic containment event tree is initially used. Tailoring of the containment event trees for each plant damage state are then developed to ensure that all appropriate conditions are assessed given the conditions at the time of core damage.

The top events or headings in a containment event tree address the events and physical processes that govern accident chronology, plant response to beyond design basis conditions, relevant challenges to barriers to radioactive material release, and the eventual magnitude of the release of radioactive material to the environment. Headings of the containment event tree also address issues and actions relating to SA management. Potential mechanisms that challenge containment integrity in an SA are identified in the Level 2 PSA.

The structure of each containment event tree and associated event probabilities including the system unavailability and the human error probability are determined considering SA conditions (equipment survivability, environment condition) evaluated by the accident progression analyses described above.

The containment event tree structure will be chronologically correct, properly account for interdependencies among events and/or phenomena, and reflect an appropriate level of detail to satisfy the objectives of the Level 2 PSA. Regarding chronology, the following phases are described in the containment event tree headings:

- T1: Immediate response of the plant to the plant damage state caused by the IE through the early period of in-vessel core damage
- T2: Late period of in-vessel core damage to immediately following RPV failure
- T3: Long-term response of the plant after RPV failure

The effect of the environmental conditions resulting from a SA on the survivability of components and systems credited within the Level 2 PSA model are also assessed and accounted for appropriately. Environmental impacts may include temperature, pressure, humidity, and radiation conditions, as well as effects derived from energetic events (e.g., short-term temperature and pressure spikes or impulse loadings from detonations or steam explosions).

Potential adverse effects of SA management actions are also considered as part of the event tree logic. For instance, injection of water into a degraded core may be able to arrest the progression of a SA. However, there is also the potential for energetic Fuel Coolant Interaction, fuel shattering, and additional releases of steam, hydrogen, and radioactive material.

Modeling Dependency

Dependency of SA phenomena on the success/failure of mitigation systems are modeled.

By linking the fault trees to containment event trees, dependencies regarding basic events including components, support systems, operator actions and success logic (“NOT” logic) are captured in the analysis. In addition, the containment event trees are connected to all relevant Level 1 core damage sequences. This enables treatment of dependency between the Level 1

PSA and the Level 2 PSA. Basically, system fault trees in the Level 2 PSA are developed in line with the steps for the Level 1 PSA discussed above. The following guidelines are used in developing or quantifying the Level 2 PSA:

- The same basic event is applied to both Level 1 and Level 2 analyses to ensure that complete dependency is captured by the fault tree linking (as explained above), (e.g., common cognition error for core cooling or system failure for RPV injection).
- For operator actions with the same cues, the dependency is treated as the recovery file to cutsets.

Treatment of High Probability Events

If there are high probability events (e.g., more than 0.1), success probability of these events is conservative in CAFTA. The success probabilities of high probability events are treated as described below.

Create a NAND gate and define a “success” event with a (1-p) probability under the NOT side of the NAND gate. The failure events in the accident sequence are modeled under the other input of the NAND gate.

Release Category Analysis

Containment event trees have a large number of end states, each representing a sequence of events that occurs following core damage. Many of these events have a significant influence on the release of radioactive material from the containment. Characteristics of such events include:

- The mode and time of failure of the containment
- The cooling mechanisms of the molten core material
- The retention mechanisms for radioactive material

Many of the end states of the containment event tree are identical or similar in terms of the phenomena that have occurred and the resulting release of radioactive material to the environment. Similar end states are grouped or binned together to reduce the number of distinct accident sequences that need deterministic source term analysis.

A table will be developed for the FSAR that shows the typical attributes used for the classification of release categories. These attributes will be used to specify the set of release categories used for the source term analysis in the Level 2 PSA. If this process generates a very large number of release categories, these should be further grouped into a manageable set that can be used in the source term analysis.

The end states of the containment event tree are grouped into the specified release categories. Because this involves the grouping of typically thousands of end states of the containment event tree into a small number of release categories, a systematic process is applied to this grouping process. The frequency of the release categories is calculated by summing the frequencies of all the end states of the containment event tree that are assigned to the group. It can involve post-processing of the end states of the cutsets.

Note: Release Categories are defined in the results of the PSA and will be provided in the FSAR.

15.6.5 Severe Accident Methodology

The PSA applies a methodology that identifies and analyzes accidents that can lead to significant core damage, and/or offsite releases of radioactive material. The SA conditions and radioactive releases are evaluated using the MAAP 5.06 computer code. MAAP5 (Reference 15.6-29) is a computer code used by nuclear utilities and research organizations to predict the progression of light water reactor accidents. MAAP was developed by the Electric Power Research Institute to support the industry probabilistic safety assessments. Such information is provided as input to the SAA, which is described in NEDC-33934P.

Quantitative risk analyses that show the extremely unlikely conditions can be performed in the PSA at a later design evolution stage. Information resulting from the quantitative risk analyses is provided as part of the site-specific PSA.

15.6.5.1 Acceptance Criteria

Table 15.3-3 provides the BWRX-300 plant safety goals, which are the acceptance criteria for DECs. DECs must meet these safety goals or be practically eliminated (see Appendix 15A). Risk reduction design features are provided to cope with DECs. The DEC risk reduction design features are described in Appendix 15B.

15.6.5.2 Severe Accident Sequence Selection

SAA is performed for sequences leading to core damage as informed through Level 1 PSA. The core damage states in Level 1 PSA are identified as unique core damage classes that signify the plant state when core damage occurs. These core damage class end states are traced via different accident sequence pathways that are represented in the Level 1 event trees. The Level 1 event trees leading to the identification of discrete core damage class end states are constructed based on the various accident initiator types as relevant to the design. The core damage class end states serve as the entry points for the Containment Event Trees (CET) that outline the post coredamage (i.e., Level 2) accident progression. This includes an assessment of the severe accident mitigating safety functions for the containment and the phenomena challenging the containment if the safety functions are unavailable or fail to achieve the required performance.

The end states of the CETs are the radiological release classes. The containment performance is assessed by the various temporally evolving accident sequence pathways within the CETs. The accident sequence pathways in the CETs along with the Level 1 PSA event trees provide the overall guidance for sequence selection and its analytical description in the SAA framework. Bounding arguments are made to chose the overall representative sequences under each core damage class that are analyzed as part of SAA.

15.6.6 Source Term Analysis

A source term is used to characterize the radionuclide release in each release category defined by the ES and success criteria analyses. The source term uses physical and chemical models to provide mechanistic treatment of radionuclide transport.

Release categories are used to group all sequences identified in the event sequence analysis that result in radionuclide release. The characteristics of each release are studied, and the release categories are defined in terms of release composition and quantities, timing, and thermal energy.

15.6.6.1 Methodology

15.6.6.1.1 Event Sequences Resulting in Radionuclide Release

Before release categories are defined, the sequences from the ES analysis that result in radionuclide release are identified. The characteristics of radionuclide release are studied for all instances where a sequence results in damage to the core within the reactor vessel or the fuel in the fuel pool. For completeness, this accounting includes sequences where the reactor vessel head, and containment are successfully performing their containment functions as radionuclide barriers. In this way, the release that results from core/fuel damage and the expected level of leakage from the vessel and containment are quantified for BWRX-300 for core/fuel damage.

15.6.6.1.2 Source Term Analysis for Each Release Category

The release categories used for the BWRX-300 PSA are a combination of the magnitude and timing of core damage and the specific release path taken.

Representative scenario/conditions for each release category are defined based on the definition of the release category. The consequence analysis software (e.g., MAAP) codes are used to determine the source term. The codes include models for almost all important accident phenomena that might occur in the RPV, in the containment, and in the RB. The codes calculate the progression of postulated accident sequences from a set of IEs to either a safe, stable state or to a degraded containment condition due to overpressure or overtemperature, and an amount of fission product releases to the environment.

For BWRX-300 SA and Source Term Analysis, the duration for the analysis period is justified. The source term cases are analyzed to a minimum of 72 hours, or >36 hours after core damage; whichever is longer. Justification of endpoint/termination time of the analysis would evaluate trends in results at the termination time and provide a technical basis for claims that the results and conclusions drawn from the calculation would not change if the termination time was extended. If the trends or source term trend cannot ensure the calculation results would only negligibly change, the source term analysis is extended up to 168 hours.

The radionuclide release paths in the Level 2 PSA are documented based on a detailed containment failure analysis from locations and timing affected by the debris location such as containment basemat, containment head, and personnel air lock. If there are some release paths with the near failure criteria and these characteristics are different, the failure location will be probabilistically considered.

Attributes affecting the source term, based on the release category attributes, are discussed below.

15.6.6.1.3 Radionuclide Release Composition and Quantities

The composition of the release is defined by the radionuclide inventory of the core. The radial and axial power distribution over the length of the fuel cycle is accounted for in the ES.

The release quantity for each release is given by an entire portion of the core for which any fuel cladding has been computed to fail. For example, in long-term sequences where passive Decay Heat Removal performance is degraded to the point that a portion of the core reaches failure temperatures, the inventory of each pin in that portion of the core is assumed to release into coolant.

15.6.6.1.4 Radionuclide Release Timing

The timing of the release in each release category is defined in terms of delay before release and the duration of the release. The delay before release is based on whether the release is the result of a short-term reactivity sequence or a long-term Decay Heat Removal sequence.

For short-term releases, no delay is credited as damage to fuel occurs early.

For long-term releases, a delay of 24 hours after shutdown is used as a lower bound delay time for damage temperatures to be reached inside the vessel. The analysis of LPSD or fuel pool releases may use longer delay times and account for the time since shutdown in determining the release.

The second dimension of the release timing is the duration of release. For release categories where the release is through leakage from the vessel or containment, the release duration could continue indefinitely as conservatively no mechanisms are modeled that remove heat from the vessel or containment. For these cases, a full 3-day release is conservatively assumed consistent with Decay Heat Removal incapacitation for 72 hours.

15.6.6.1.5 Radionuclide Release Energy

The thermal energy of the release is calculated using the release rate. The calculation is dominated primarily by the assumed release rate. These cases have not yet been completed for the BWRX-300 design.

15.6.6.1.6 Decontamination Factor

The Decontamination Factor (DF) can be estimated or calculated in numerous ways, and a full discussion of this analysis is not provided here. The general approach for PSA DF analysis is no different than that performed for other safety analysis or radiation protection analysis. The method below discusses the approach used within the consequence analysis software (i.e., MAAP), which is one possible approach.

The consequence analysis software, MAAP, includes an internal lookup table that calculates the DFs of the fission product aerosols. To reduce calculation time, the lookup table of DFs is created beforehand using a detailed calculation code developed by EPRI, which calculates DFs based on several dominant input parameters (e.g., gas injection mode, aerosol shape, thermal-hydraulic condition of gas and pool). For the gas injection mode, three modes (a sparger / quencher, a side vent, and a downcomer) are considered, and the consequence analysis software applies the sparger mode for evaluating the DF of the pool scrubbing by the water covering the corium.

The lookup table, created for the consequence analysis software, requires some input parameters such as aerosol particle size, steam ratio, pool water depth, subcooling of water, pressure, and gas composition. The consequence analysis software calculates each case of the DF by interpolation of the DF values which correspond to discrete input parameters prepared in the software. The software calculates the fission product behavior based on their weight; however, in case of the pool scrubbing, weight and diameter of the fission product aerosols are both normalized dimensionless. The DF for each particle diameter is calculated and then the averaged DF is calculated by using the aerosol diameter distribution.

For the fission product vapor, it is assumed that aerosols generated by condensation of vapors can be only trapped in water and vapors are never trapped in water as vapor phase. Therefore, the DF calculation for the fission product vapors is based on inertial impaction and theoretical consideration of thermophoresis of the generated aerosols in a vapor void.

15.6.6.1.7 Key Inputs

The key inputs are:

- Release Category ESs
- Release Category Frequency Results

15.6.6.1.8 Task Outputs and Preliminary Results

The outputs are:

- Source Term Characteristics
- Key Insights and Assumptions
- Underlying assumptions, exclusions, limitations, and features
- Any conclusions that can be drawn from analysis of uncertainties associated with phenomena, models, and databases and contributory analyses
- Results of sensitivity studies

- Any contribution from screening/truncation of Level 1 output
- Plant-specific design or operational vulnerabilities identified
- Key operator actions for mitigating SAs
- Potential benefits of various engineered safety systems
- Areas for possible improvement in operations or hardware for the plant and containment in particular
- Comparison with probabilistic safety criteria for Level 2 PSA
- The documentation should be divided into a summary report, main report, and appendices

15.6.7 Uncertainty and Sensitivity Analysis

15.6.7.1 Introduction

The Uncertainty and Sensitivity analysis performed for the BWRX-300 PSA model provides the uncertainty distribution for the overall risk calculated previously. The sensitivity analysis evaluates the sensitivity of the results to key model assumptions. The combined results of the two analyses describe the overall uncertainty of the PSA and identify those PSA attributes significantly affecting the PSA results.

The uncertainty results presenting the lower/upper bounds and the spread of the mean value of the risk (as measured by a Quantitative Health Objective) are considered a good representation of the aleatory uncertainties of the model, which are related to random processes. The sensitivity cases represent the epistemic (modeling) uncertainties, which are related to deficiencies in knowledge. Therefore, the sensitivity study results add valuable insights for the users of PSA results, future PSA model updates, and the identification of potential model deficiencies.

Various sensitivity analyses are conducted on the BWRX-300 PSA at-power, fire, flood, high wind, seismic, and shutdown models. Uncertainty analyses are performed on the Level 1 and Level 2 baseline internal events PSA models. The intent of these analyses is to evaluate the effects to the PSA models and to provide risk insights. The results of the ongoing analysis, including key assumptions, sources of uncertainty, and appropriate sensitivity evaluations, are available in separate supporting documents.

Sensitivities and uncertainties are identified from the following sources:

- Support for key assumptions
- Identified by system/PSA engineer

The analysis includes those uncertainties and assumptions affecting the Level 2 PSA, including SA phenomena. These are discussed below.

Whereas sensitivity analysis is used to measure the extent to which results change if alternative models, hypotheses or values of input parameters were selected (and thus provides an evaluation of uncertainty in respect of a particular issue or a particular group of related issues at a time), uncertainty analysis examines a range of alternative models or parameter values, assigns each model or value a probability and generates a distribution of the results, within which

the baseline results represent one possible outcome. Each result within the full distribution is accompanied by a (subjective) probability representing the degree of belief in that result. Cumulative probability levels for the results can be calculated (e.g., the 5th, 50th and 95th percentiles represent 5%, 50% and 95% probabilities, respectively, and the 'true' result is below the respective level for which each of these probabilities is stated).

15.6.7.2 Methodology

The methodology for conducting the sensitivity and uncertainty analysis was conducted in three phases: (1) selection/identification, (2) implementation/analyses and (3) results/benchmarking.

The first step was to evaluate the importance of the sensitivity itself. In some cases, sensitivities were identified, but upon further evaluation were discarded due to inherent model conservatisms or were delayed pending more detailed engineering.

15.6.7.2.1 Identification and Characterization of Uncertainties

As stated in the introduction section, the previous BWRX-300 Uncertainty and Sensitivity studies were mainly performed to support BWRX-300 design. For the PSA, the following additional guidance can be used to identify any additional modeling uncertainties:

- EPRI-1009652, "Guideline for the Treatment of Uncertainty in Risk-Informed Applications: Technical Basis Document," (Reference 15.6-31)
- EPRI-1013491, "Guideline for the Treatment of Uncertainty in Risk-Informed Applications: Applications Guide," (Reference 15.6-32)
- EPRI-1016737, "Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments," (Reference 15.6-33)
- NUREG-1855 Volume 1, "Guidance on the Treatment of Uncertainties Associated with PSAs in Risk-Informed Decision Making – Main Report," (Reference 15.6-34)
- EPRI-1026511, "Practical Guidance on the Use of Probabilistic Risk Assessment in Risk-Informed Applications with a Focus on the Treatment of Uncertainty," (Reference 15.6-35)

The above provide examples of generic uncertainties for each of the PSA tasks, that may or may not be applicable to the BWRX-300. The identified generic uncertainties are reviewed for applicability to the BWRX-300 and assessed if applicable. In addition, the PSA assumptions are reviewed for key assumptions affecting the results. Each key assumption is then assessed qualitatively or quantitatively through sensitivity analysis. When a quantitative assessment is not possible, or if the quantitative assessment is not valuable (i.e., the analysis is trivial), then a qualitative assessment is performed.

EPRI-1026511 provides a framework for addressing uncertainty in an SPSA. Table C-1 of that report lists various sources of model uncertainty and possible resolutions of these issues. Issue Characterization for Sources of Model Uncertainty can be used to identify and characterize any uncertainty (generic, plant-specific modeling and completeness uncertainty).

Issue characterization for sources of model uncertainty (seismic hazard group) for consequences of tornado damage to structures other than the RB assumes that damage to SSCs located within these structures could arise from damage of the structures due to tornado wind loading, tornado missile, or structural cladding failure combined with wind-driven precipitation. Diesel generator and AC switchgear availability is considered, and availability of operations crew is considered.

For tornado, the contents of all BWRX-300 structures other than the RB are modeled to be unavailable given a tornado event that strikes the site. No credit is taken for operator actions. SBO-ATWS cutsets are the dominant contributor to the tornado High Winds PSA results. The total tornado High Winds PSA CDF result is small, (1.3E-10/yr), and therefore, this approach and assumption are not a key source of uncertainty.

Key (Risk-significant) assumptions are judged based on guidance from the ASME standard. Factors that are considered include a determination if the assumption is conservative or optimistic and an investigation into the risk importance measures (i.e., Fussell-Vesely (FV) importance and Risk Achievement Worth (RAW)). If the importance of the assumption is not able to be confirmed based on a calculated importance measure, then the risk impact from the assumption is confirmed by sensitivity analysis.

The impact of all significant assumptions is assessed by performing sensitivity studies that apply different plausible assumptions. If assumptions affect the same aspects of the PSA model, then sensitivities are performed concurrently.

The focus of this document is epistemic uncertainty (i.e., uncertainties in the formulation of the PSA model). Therefore, guidance is given on identifying and describing the different types of sources of epistemic uncertainty and the different ways that they are treated.

The different types of epistemic uncertainty are:

- Parameter Uncertainty
- Model Uncertainty
- Completeness Uncertainty

15.6.7.2.2 Parametric Uncertainty Analysis

The mean values for the release category frequencies is obtained from uncertainty distributions.

UNCERT is used for parametric (statistical) uncertainty analysis, which involves the propagation of uncertainties through the PSA. UNCERT, which was developed by EPRI, is a program to perform the uncertainty analysis using a Monte Carlo method. UNCERT generates the random inputs based on the user specification, calculates all intermediate values such as component failure probabilities, and calculates the top event value from the cutsets. In addition, the state of knowledge correlation per parameter uncertainty is naturally considered as described in the user’s manual of UNCERT.

The following steps are followed to perform the uncertainty analysis:

- All basic events are tied to their most appropriate type codes, which include CCF basic events to satisfy the “state-of-the-knowledge” requirements.
- If recovery is used and some Human Error Events (HEEs) are set to a value of 1.0 at the very end of the rule file, these HEEs are set to one (1) in the reliability database in order to allow UNCERT to generate meaningful results.
- Run UNCERT with a seed value of 1.0 and use the Monte Carlo method (or other appropriate method) for different number of steps. Normally 30,000 to 40,000 steps provide suitably stable results.

15.6.7.2.3 Sensitivity Studies

Major model assumptions in the BWRX-300 system notebooks and the recent changes made to the BWRX-300 PSA models as a part of the update process are reviewed to identify the candidate sensitivity cases. Some of the important assumptions are related to the uncertainties caused by the lack of thermal-hydraulic analysis supporting PSA modeling and should be investigated in this task. The overall effect of the maintenance unavailability, the CCF probabilities, and HRA probabilities should be included in the analysis.

The sensitivities include the following:

- Plant Damage State Frequency Sensitivities
- Release Category Frequency Sensitivities
- Low-Power/Shutdown Sensitivities
- Transportation and Nearby Facilities Sensitivities
- Internal Fire Sensitivities
- Source Terms Sensitivities

15.6.7.2.4 Sensitivity Analyses to Address Cliff-Edge Effect

Cliff-edge effect is defined in IAEA Specific Safety Guide No. SSG-2, “Deterministic Safety Analysis for Nuclear Power Plants,” (Reference 15.6-37) as:

> “An instance of severely abnormal conditions caused by an abrupt transition from one status of a facility to another following a small deviation in a parameter or a small variation in an input value.”

Cliff-edge effects are reviewed through a systematic analysis of each PSA scope, starting with the sensitivity and uncertainty analysis. This includes a review of key assumptions, where small changes in the modeling assumptions have a significant change in the PSA results. Additional and specific steps are performed for key hazards, such as Seismic PSA and other hazards discussed below. This includes the review of hazards screened or qualitatively evaluated in the PSA that can potentially have a large consequence.

The BWRX-300 SPSA, for example, demonstrates that the BWRX-300 is inherently capable of safe shutdown in response to strong magnitude earthquakes beyond the Safe Shutdown Earthquake. A seismic event exceeding the maximum peak ground acceleration considered in the SPSA is assumed to result in core damage and large release. However, the seismic risk for such scenarios is anticipated to be extremely low.

BWRX-300 PSA specifically addresses the risks associated with exceptional weather such as blizzards and elevated sea level and other environmental conditions such as algae or oil. The external flooding including the elevated sea level are expected to be bounded by siting characteristics set for the BWRX-300 design. Blizzards and algae or oil that could fail multiple systems/functions are addressed but not be expected to affect risk insights.

A formal assessment of “cliff-edge” effects are developed. The following are the candidates for this assessment:

- Ambient temperatures
- External flooding based on the elevated water level based on site-specific inputs
- Passive cooling capabilities
- Other assessments as identified in the site-specific EE screening

15.6.7.2.5 Level 2 Uncertainty and Sensitivity Analysis

Uncertainty arises in a Level 2 PSA analysis as a result of several factors, including:

- Incompleteness uncertainty. The overall aim of a Level 2 PSA is to assess the possible scenarios (sequences of events) that can lead to releases of radionuclides, mainly those scenarios modeled in the Level 1 PSA. However, there is no guarantee that this process can ever be complete and that all possible scenarios have been identified and properly assessed. This potential lack of completeness introduces an uncertainty in the results and conclusions of the analysis that is difficult to assess or quantify. It is not possible to address this type of uncertainty explicitly. However, peer review can reduce this type of uncertainty.
- Loss of detail due to aggregation. Grouping accident sequences or cutsets from the Level 1 PSA into PDSs for input into the Level 2 PSA for practical reasons also introduces uncertainties due to the resulting loss of some modeling detail. Further, the process of ‘binning’ (or grouping) accident sequences introduces uncertainty through the possibility that the attributes used by the analyst to group ‘similar’ accident sequences are incomplete. These elements of uncertainty are also difficult or impossible to quantify, but diminish over time as increases in computing resources allow increasing levels of detail to be captured in the PSA.
- Modeling uncertainty. This arises due to a lack of complete knowledge concerning the appropriateness of the methods, models, assumptions, and approximations used in the individual analysis tasks that support a Level 2 PSA. Modeling uncertainties are addressed in the sensitivity analysis.

- Parameter uncertainty. This arises due to the uncertainties associated with the values of the fundamental parameters used in the quantification of the Level 2 PSA, such as equipment failure rates and IE sequences. This type of uncertainty is addressed similar to the Level 1 parameter uncertainty analysis.

In general, the process of quantification and propagation of uncertainties in the Level 1 and Level 2 PSA can be divided into the principal steps discussed below.

1. Characterization and/or evaluation of uncertainty issues:

   For parametric uncertainty, the first step is to identify the range of values of uncertain parameters. Each value within the range of values that the uncertain parameter can take on is associated with a probability, thereby creating a probability density function or probability distribution. In many cases, such density functions or probability distributions are determined in the assessment of probabilities for branch points in the containment event tree. Additional parameters that may also be characterized or evaluated by means of probability distributions may be, for example, source term calculation parameters not explicitly addressed in the containment event tree.

   Judgments reflected in the probability distributions for each parameter are supported by data, analyses, and consideration of the published literature.

2. Propagation of uncertainties:

   Because the information from the Level 1 core damage scenario is retained in the Level 2 sequences (through sequence markers and basic events), uncertainty propagation in the Level 2 analysis can be addressed through similar analysis performed for the Level 1 PSA. For example, sensitivities performed on the Level 1 PSA to examine the CDF impact from modeling uncertainties can be reproduced to examine the impact on LRF. Additionally, parametric uncertainty propagation is straight forward since the basic events contributing to core damage sequences are retained in the Level 2 end state cutsets. If the impact of non-minimal cutsets is found not to be insignificant, the sequence markers as described above are removed.

3. Display and interpretation of results:

   The results of the uncertainty analysis are carefully evaluated to strengthen the conclusions of the Level 2 PSA. The results are displayed using histograms, probability density functions, cumulative distribution functions, and tabular formats showing the various quantiles of the calculated uncertainties, together with the estimates of the mean and median of the probability distributions.

15.6.7.2.6 Key Inputs

The key inputs are:

- BWRX-300 PSA models (fault trees, cutsets, and associated databases)
- Insights and assumptions identified from other model elements that require sensitivity studies

- A specific list of exceptional weather conditions (e.g., strong winds, blizzards, elevated sea level) and any other harsh environmental conditions (e.g., algae, oil, explosions) at the site

15.6.7.2.7 Task Outputs and Preliminary Results

The outputs are:

- Mean values for risk results Quantitative Health Objectives
- Distributions for risk results Quantitative Health Objectives
- Risk-informed BWRX-300 design changes
- Key insights and assumptions confirmed by sensitivity studies

15.6.8 Model Integration and Quantification

The Level 2 PSA model is integrated with the Level 1 PSA model. The Level 1 accident sequences are placed under “collector gates” that group core damage sequences into classes.

The Level 2 ETs are constructed with the entry branch name identical to the applicable Level 1 Core Damage class collector gate. Functional node branches are given names identical to those in the functional fault tree file. When the event trees are converted into fault tree logic (creating accident sequence logic) and merged with functional fault tree logic, an integrated, quantifiable fault tree is developed.

The fault tree is quantified using the following PSA model development codes:

- CAFTA
- PRAQuant

In addition, FTREX is used as the quantification/cutset generation engine and SYSIMP is used to develop importance measures for systems, components, and human failure events.

Containment intact, small release, and large release frequencies are quantified at a truncation of 1E-15/yr. The cutsets are examined to ensure there was <5% top event frequency drop if the truncation limit is set to 1E-14/yr. This provides evidence that no risk-significant sequences were truncated.

15.6.9 Results of the Level 1 and Level 2 Probabilistic Safety Assessment

Preliminary dominant contributors to CDF and LRF are identified in Table 15.7-9 and Table 15.7-10. Final contributors to CDF and LRF will be maintained and updated as the design matures.

15.6.9.1 Initiating Events

The IEs were derived for internal events At Power and for low power shutdown states. The IEs will be tabulated with descriptions and frequencies provided in the FSAR.

15.6.9.1.1 Mitigating Systems Modeled

The following front line mitigating systems are modeled to mitigate the IEs. These mitigating systems or functions are:

- Shutdown Cooling System
- Isolation Condenser System
- Containment Isolation
- Ultimate Pressure Regulation
- Control Rod Insertion
- Boron Injection
- Control Rod Injection makeup
- Corium Shield
- Containment Venting
- Passive Containment Cooling System
- Cooling Water System
- DC Power
- Feedwater Runback
- Reactor Pressure Vessel Isolation
- AC Power
- Heating Ventilation and Cooling
- Plant Pneumatics
- Power Conversion System

15.6.9.1.2 Level 1 Event Tree Endpoints

The Level 1 PSA event tree end states end in one success or six core damage states. The six core damage states are described below.

ESs are qualitatively analyzed to determine which sequences lead to core damage end states. The core damage sequences are grouped together based upon the overall challenge to the containment barrier and defined as:

- OK: The core is successfully cooled, and the containment is intact. There is no core damage in these events.
- CD I: The containment is intact when core damage occurs and the RPV is at low (or controlled) pressure.
- CD II: The containment is breached, either from overpressurization or venting, while the core is successfully cooled. Core damage results from failure of maintaining long-term heat removal to maintain core cooling.
- CD III: The containment is intact when core damage occurs and there is high RPV pressure at the time of RPV failure.

- CD IV: Core damage results from an accident sequence with an initial failure of effective reactivity control (failure to scram due to failure of RPS, control rod binding, etc.). This has the potential to affect containment more severely than the CD I and CD III because more energy is deposited into containment prior to RPV failure. All CD IV end states could be treated as CD I or CD III (depending on the RPV pressure) without affecting the results of the containment analysis. This end state is retained in this scoping analysis to more readily allow for sensitivity analyses related to reactivity control.
- CD V: The containment is bypassed at the time of core damage.
- CD VR: Core damage occurs due to RPV ruptures in the lower or mid-vessel regions.

15.6.9.1.3 Event Tree Transition – Level 1 to Level 2 PSA

The core damage Level 2 PSA accident sequence analysis:

Class I/I-L

Core damage sequences occur with the RPV at low pressure and nominally have containment intact are Class I. There are core damage sequences that occur with Class I that occur “late,” and a Containment Event Tree (CET) is developed for both “early” and “late” Class I core damage.

Class II

Class II core damage (core damage resulting from containment failure due to overpressure). Class II core damage often results from injection without containment decay heat removal. The resultant overpressure of the containment leads to failure and loss of the injection due to loss of NPSH, adverse equipment environments, and disruption of injection lines similarly to the Level 1 PSA Core Damage Frequencies.

Class III/III-L

Core damage sequences that occur with the RPV at high pressure and that nominally have containment intact are Class III. There are core damage sequences that occur with Class III that occur “late,” and a CET is developed for both “early” and “late” Class III core damage.

Class IV

Class IV core damage scenarios are failure to scram and failure to depressurize scenarios. If this core damage scenario occurs, SA progression results in more severe RPV and containment conditions than those scenarios with successful negative reactivity insertion. The containment loading resulting from these scenarios is expected to exceed the containment capacity and gross failure of the containment is expected to occur.

Class V

Class V sequences are those with core damage and an initial containment bypass. Containment bypass provides a radionuclide release path. Any Class V sequences result in large early releases.

Class VR

Core damage occurs due to RPV ruptures in the lower or mid-vessel regions, the containment vent functions.

15.6.9.1.4 Risk Significance

The determination for the definition of risk significance for the BWRX-300 is established at each phase based on the overall plant risk. Risk Significance is defined for CDF and other PSA endpoints, such as large release frequency.

Risk significance is calculated and reported for significant basic events, significant accident sequences, significant key safety functions or systems, and significant model elements for internal and external hazards. These are discussed in the IAEA-TECDOC-1804.

15.6.10 Probabilistic Safety Assessment Insights and Applications

The BWRX-300 PSA activities are ongoing as the design progresses to maturity. The PSA scope includes Level 1 PSA internal events and external events (fire, flood, high winds, and seismic) at power and low power shutdown, and Level 2 PSA At Power. The following elements provide risk informed information used in the design as described in Subsection 15.2.1 and Subsection 15.2.4:

1. Provide a systematic analysis
2. Demonstrate a balanced design that no particular feature or postulated IE makes a disproportionate contribution to overall risk
3. Provide confidence of no cliff-edge effects
4. Provide assessments of quantitative safety goals
5. Provide site-specific assessments of external hazards
6. Identify facility vulnerabilities and systems where design improvements or modifications to operational procedures could reduce probabilities of SAs or mitigate consequences
7. Assess adequacy of emergency operating procedures
8. Provide insights into SA management

PSA objectives 1 to 6 are assessed at this stage of the design and objectives 7 and 8 are assessed once the detailed design and operational procedures are available.

Table 15.6-1 describes how these objectives are addressed in the PSA process. The BWRX-300 Safety Strategy, NEDC-33934P, describes how the PSA is used to demonstrate that the BWRX-300 design meets the overall plant safety goals. Section 15.2 describes how events are identified, categorized, and grouped into PIEs and bounding accident scenarios.

In some scenarios, small changes can lead to large differences in plant effects (i.e., cliff-edge effects are largely ameliorated by the demonstrated margin between plant risk metrics and the USNRC safety goals). Within the PSA accident sequence and success criteria analysis, no cliff edge effects were discovered where plant parameters approach critical thresholds where accident sequences are expected to follow a significantly more severe pathway. However, uncertainty in these analyses will be formally explored in more detail in future PSA work.

Table 15.7-9 and Table 15.7-10 show that the PSA CDF and LRF results to date meet the USNRC safety goals.

Internal and external hazards are identified and screened for PSA. Natural external hazards and human- induced external and internal hazards are assessed. There is limited assessment of external and internal hazards interaction. A more detailed hazard combination assessment is performed commensurate with the design progress.

Operating Manuals, EOPs, and Severe Accident Management (SAM) program are provided separately as the design matures and additional PSA work is completed.

PSA is further updated for the Operating License Application as the design is finalized and operational information matures.

Summary

The PSA CDF and LRF results to date meet the USNRC Safety Goals as shown in Tables 15.7-9 and 15.7-10.

Section 15.6 is a high-level summary of the Internal and External Event Hazard Screening, and Level 1 and Level 2 PSA status supporting the BWRX-300 PSAR.

The PSA scope:

- Level 1 PSA internal At Power and low power shutdown states
- Level 1 PSA for EE includes at-power, seismic, and high winds with site-specific results for CRN-1 shown in Table 15.7-9
- Level 2 PSA at-power

Internal and external hazards are identified and screened. Natural external hazards and human-induced external and internal hazards are assessed.

The PSA:

- Includes verification that EOPs are adequate during commissioning and future activities which will be provided in the FSAR
- Provides insights into the Severe Accident Management (SAM) program
- Describes how PSA is used during commissioning and future operation to identify design improvements and modifications to operational procedures to reduce probabilities of SAs and mitigate consequences

- Supports the BWRX-300 Safety Strategy

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15.6-35 EPRI TR-1026511, “Practical Guidance on the Use of Probabilistic Risk Assessment in Risk-Informed Applications with a Focus on the Treatment of Uncertainty,” Electric Power Research Institute, 2012.

15.6-36 RA-S-1.2-2024, “Severe Accident Progression and Radiological Release (Level 2) PRA Standard Nuclear Power Plant Applications For Light Water Reactors (LWRs),” American National Standards Institute, American Society of Mechanical Engineers/ American Nuclear Society (ASME/ANS), 2024.

15.6-37 IAEA SSG-2, “Deterministic Safety Analysis for Nuclear Power Plants,” International Atomic Energy Agency, 2019.

15.6-38 RA-S-1.4, 2021, “Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants,” American National Standards Institute, American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS), 2021.

Table 15.6-1 Probabilistic Safety Assessment Objectives

#	Objectives of the PSA	Comments
1	To provide a systematic analysis giving confidence that the reactor facility's design aligns with the fundamental safety objectives as established in IAEA No.SF-1, Fundamental Safety Principles, including to protect people and the environment from radiation and to support the BWRX-300 Safety Strategy	Overarching objective
2	To demonstrate that a balanced design has been achieved; this can be demonstrated as achieved if no particular feature or postulated IE makes a disproportionately large or significantly uncertain contribution to the overall risk	Is demonstrated in the PSA Summary Report
3	To provide confidence that small changes of conditions that may lead to a catastrophic increase in the severity of consequences (cliff-edge effects) are prevented	Is demonstrated in the PSA Summary Report and in the Sensitivity and Uncertainty Analysis
4	To provide assessments of the quantitative safety goals (the probabilities of occurrence for severe core damage states, and the assessments of the risks of radioactive releases to the environment) as defined in the ASME/ ANS PSA standards for Level 1 and Level 2 PSA, Design of Reactor Facilities, or as established in licensing basis for the facility	Is demonstrated in the results and discussion of Level 1 and 2 PSA. Both small and large releases are analyzed in the PSA.
5	To provide site-specific assessments of the probabilities of occurrence, and the consequences of external hazards	In the site-specific PSA
6	To identify facility vulnerabilities and systems for which design improvements or modifications to operational procedures could reduce the probabilities of SAs, or mitigate their consequences	Is demonstrated in the PSA Summary Report, and the sensitivity and uncertainty analysis
7	To assess the adequacy of EOP	Licensee activity using the PSA results and insights
8	To use risk insights and results to inform SA management decisions.	Licensee activity using the PSA results and insights

Table 15.6-2 Internal Hazard Qualitative Screening Analysis

Hazard	Screening Criterion (Refer to Subsection 15.6.2.3.5)
Heavy Load Drop	Not screened qualitatively
Release of Chemicals from On-Site Storage	1
Turbine-Generated Missiles	Not screened qualitatively
Other Internally-Generated Missiles	1, 2
Fires	Not screened qualitatively.
Explosions	Not screened qualitatively.
Collapse of Structures	4, 5
Pipe Whips	4
Jet Effects	4
Internal Flooding	Not subject to screening

Table 15.6-3 External Hazards Screening
(Sheet 1 of 21)

External Event	Description	BWRX-300 Screening	Hazard Screening Disposition	Cliff-Edge Effects	Fault List Disposition
Strong Winds	The event is defined as damage to the plants due to strong winds. It includes both direct damage from wind pressure and indirect damage from wind-carried missiles.	4 Addressed by BWRX-300 High Wind Scoping Assessment	The Strong Winds hazard was selected to proceed directly to detailed assessment	Addressed by detailed PSA. The full spectrum of strong wind speeds is evaluated.	Expected to occur greater than 1.0E-07/yr. Retained for Fault List.
Tornado	The event is defined as damage to the plants due to tornadoes. The event is separated from other strong winds due to its special characteristics both with respect to duration, wind speed, and frequency of occurrence.	4 Addressed by BWRX-300 High Wind Scoping Assessment	Tornado was selected to proceed directly to detailed assessment.	Addressed by detailed PSA. The full spectrum of strong wind speeds is evaluated.	Expected to occur greater than 1.0E-07/yr. Retained for Fault List.
High Air Temperature	The event is defined as plant impact caused by high air temperature.	Retained for Quantitative Screening	BWRX-300 SSCs directly exposed to ambient environmental conditions.	The evaluation of this hazard places no limit on the hazard intensity; any potential cliff-edge effects are implicitly addressed.	Retained in Fault List, based on Quantitative Screening

Table 15.6-3 External Hazards Screening
(Sheet 2 of 21)

External Event	Description	BWRX-300 Screening	Hazard Screening Disposition	Cliff-Edge Effects	Fault List Disposition
Low Air Temperature	The event is defined as plant impact from low air temperature.	Retained for Quantitative Screening	During periods of extreme cold, the potential exists for loss of grid due to high demand on the transmission lines and also failures of plant components due to low temperatures.	The evaluation of this hazard places no limit on the hazard intensity; any potential cliff-edge effects are implicitly addressed.	Retained in Fault List, based on Quantitative Screening
Extreme Air Pressure (high/low gradient)	Plant impact from high or low air pressure or from quick pressure changes.	4 Addressed by scoping assessment.	Addressed by scoping assessment.	Addressed by scoping assessment.	Expected to occur greater than 1.0E-07/yr. Retained for Fault List.
Extreme Rain	The event is defined as damage to the plants due to extreme rain. It includes both damage from rain load on structures and damage due to rain induced flooding.	1 Qualitative Based on Clinch River Site information external flooding screens out and external flooding PRA model is not required.	This hazard involves external flooding due to high precipitation events, runoff from nearby areas, and external flooding due to rivers or other area of bodies of water.	The evaluation of this hazard places no limit on the hazard intensity; any potential cliff-edge effects are implicitly addressed.	Expected to occur greater than 1.0E-07/yr. Retained for Fault List.

Table 15.6-3 External Hazards Screening
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External Event	Description	BWRX-300 Screening	Hazard Screening Disposition	Cliff-Edge Effects	Fault List Disposition
Extreme Snow	The event is defined as damage to the plants due to extreme snow, including snowstorms.	1 Qualitative	Extreme snow poses the potential to damage Non-Seismic Category I structures due to snow loading. Extreme snow is screened qualitatively.	The evaluation of this hazard places no limit on the hazard intensity; any potential cliff-edge effects are implicitly addressed.	Expected to occur greater than 1.0E-07/yr. Retained for Fault List.
Extreme Hail	The event is defined as damage to the plants due to extreme hail. It includes damage from hail load on structures	1 Qualitative	Clinch River structures are assumed to be designed to withstand accumulations of hail stones. Severe hail events were reported 43 and 81 times, respectively, since 1950.	Clinch River structures are assumed to be designed to withstand accumulations of hail stones.	Expected to occur greater than 1.0E-07/yr. Retained for Fault List.
Mist	The event is defined as plant impact due to mist.	1 Qualitative	Mist could impair visibility and lead to small airplane crash. The RB is designed to withstand large airplane crash. However, this hazard could impact Non-Seismic Category I structures. This impact is addressed by the assessment for Airplane Crash.	Addressed by the assessment of Airplane Crash.	Addressed by the assessment of Airplane Crash.

Table 15.6-3 External Hazards Screening
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External Event	Description	BWRX-300 Screening	Hazard Screening Disposition	Cliff-Edge Effects	Fault List Disposition
White Frost	The event is defined as plant impact due to white frost.	1 Qualitative	It is assumed that all outdoor structures of the CRN Site are designed to withstand white frost.	The evaluation of this hazard places no limit on the hazard intensity; any potential cliff-edge effects are implicitly addressed.	Expected to occur greater than 1.0E-07/yr. Retained for Fault List.
Drought	The event is defined as an extended drought period that lowers the water level of lakes, rivers, and open water basins.	7 Qualitative	This is a slow-developing event. If severe drought conditions exist that impact the normal heat sink, the plant is shut down prior to the loss of the heat sink.	The evaluation of this hazard places no limit on the hazard intensity; any potential cliff-edge effects are implicitly addressed.	Screened from inclusion in Fault List.
Salt Storm	The event is defined as a storm involving salt covering of plant structures.	1,7 Qualitative	The CRN-1 Site is located on a freshwater river and is not subject to saltwater corrosion. Therefore, this hazard screens qualitatively.	The evaluation of this hazard places no limit on the hazard intensity; any potential cliff-edge effects are implicitly addressed.	Screened from inclusion in Fault list.

Table 15.6-3 External Hazards Screening
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External Event	Description	BWRX-300 Screening	Hazard Screening Disposition	Cliff-Edge Effects	Fault List Disposition
Sandstorm	The event is defined as plant impact from a storm carrying sand.	7 Qualitative	Sand deposition on plant structures could impact the integrity of Non-Seismic Category I structures. This is a slow-developing event. It is assumed that the plant is shut down and in a safe, stable state prior to severe impact to plant structures. It is assumed there are no large sand bodies within close proximity to the CRN-1 site, thus this hazard is screened out.	The evaluation of this hazard places no limit on the hazard intensity; any potential cliff-edge effects are implicitly addressed.	Screened from inclusion in Fault list.
Lightning	The event is defined as plant damage from lightning. The impact may be direct, causing structural damage or LOPP events, or indirect due through the electromagnetic field or fire started by lightning.	1,4 Qualitative	The consequences of lightning strike consist of lightning-induced fires, as well as voltage transients, that interfere with electrical systems functions. Lightning-induced fires are addressed by the evaluation of External Fire.	Addressed by the assessment of External Fire, Weather-related LOPP events are included in the Internal Events PSA	Expected to occur greater than 1.0E-07/yr. Retained for Fault List.
Meteorite	The event is defined as plant damage due to meteorite impact.	Retained for Quantitative screening.	Meteorite IE frequencies are low enough that the event is screened out.	The evaluation of this hazard places no limit on the sizes of meteorites; any potential cliff- edge effects are implicitly addressed.	Screened from inclusion in Fault List.

Table 15.6-3 External Hazards Screening
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External Event	Description	BWRX-300 Screening	Hazard Screening Disposition	Cliff-Edge Effects	Fault List Disposition
Explosion within Plant	Not in scope of the external hazards screening assessment.				
Explosion Outside	The event covers damage to the plant due to explosions (deflagration or detonation) of solid substances or gas clouds outside the site. The damage may be due to pressure impact or impact from missiles	1 Qualitative	The ESPA SSAR identifies explosions from the following sources: • Railroad • Roadways • Pipelines • Nearby facilities	The CRN Site is assumed to be free from impacts from this hazard.	Screened out from Fault List.
Explosion after Transportation Accident	The event covers damage to the plants from ground transportation inside and outside the site or sea transportation accidents. The damage may be due to pressure impact or impact from missiles.	1 Qualitative	ESPA SSAR, Section 2.2.3.1.1.2 states “Therefore, an explosion from materials transported along nearby highways with the potential for explosion would not adversely affect the safe operation or shutdown of units located within the power block area at the CRN Site.”	The evaluation of this hazard places no limit on the hazard intensity; any potential cliff-edge effects are implicitly addressed.	Screened out from Fault List.
Explosion after Pipeline Accident	The event covers damage to the plants due to explosions (deflagration or detonation) after a pipeline accident. The damage may be due to pressure impact or impact from missiles.	1 Qualitative	Addressed in ESPA SSAR, Sections 2.2.3.1.1.3, and 2.2.3.3.1.1 Table 2.2-9 (TVA 2017-TN5387).	The CRN Site is assumed to be free from impacts from this hazard.	Screened out from Fault List.

Table 15.6-3 External Hazards Screening
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External Event	Description	BWRX-300 Screening	Hazard Screening Disposition	Cliff-Edge Effects	Fault List Disposition
Chemical Release Outside or Inside Site	The event includes toxic impact due to chemical release outside or inside the site. These releases may originate from process accidents inside or outside the plant or from leakages of substances stored inside or outside the plant.	1 Qualitative	It is assumed that the CRN Site is situated beyond the screening distances for these chemical releases.	The CRN Site is assumed to be free from impacts from chemical release. For dust release, the evaluation places no limit on the intensity of the hazard; any potential cliff-edge effects are implicitly addressed.	Screened from inclusion in Fault List
Chemical Release after Transportation Accident	The event includes toxic impact due to chemical release after ground transportation accidents inside or outside the site or due to river transportation accidents.	1 Qualitative	Chemical release could develop from a train, road transportation or marine Transportation Accident. The event is slow developing.	The evaluation of this hazard places no limit on the hazard intensity; any potential cliff-edge effects are implicitly addressed.	Screened from inclusion in Fault List.
Chemical Release After Pipeline Accident	The event includes toxic impact due to chemical release after a pipeline accident.	1 Qualitative	BWRX-300 does not rely on upon operators for safe shutdown and fuel cooling.	The CRN Site is free from this hazard.	Event is screened out from Fault List.

Table 15.6-3 External Hazards Screening
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External Event	Description	BWRX-300 Screening	Hazard Screening Disposition	Cliff-Edge Effects	Fault List Disposition
Magnetic Disturbance	The event includes impact from manufactured magnetic or electric fields. The main examples of such fields are fields from radar, radio, or from mobile phones.	1 Qualitative	BWRX-300 instrumentation and control systems are designed to prevent magnetic disturbance from significantly impacting them.	The BWRX-300 design is assumed to be free from impacts from this hazard	Expected to occur greater than 1.0E-07/yr. Retained for Fault List.
Satellite Crash	The event is defined as plant damage due to satellite impact.	Retained for Quantitative Screening	The frequency of satellite crash is based on Frequently Asked Question #13 Orbital Debris Program Office http://orbitaldebris.jsc.nasa.gov/faqs.html	The mass and kinetic energy of orbital debris impacting the site are bounded by the Airplane Crash hazard, Event Code A25.	Event is screened out from fault list.

Table 15.6-3 External Hazards Screening
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External Event	Description	BWRX-300 Screening	Hazard Screening Disposition	Cliff-Edge Effects	Fault List Disposition
Airplane Crash	The event includes damage to plant structures from an airplane crash within the site area. The airplane may be either commercial, private, or military.	Retained for Quantitative Screening	The RB is assumed to be designed to withstand large aircraft impact. The aircraft engines pose the most significant RB penetration hazard as these elements possess the greatest kinetic energy and effectively become missiles on impact. The RB utilizes DP-SC modules (Diaphragm Plate Steel-Plate Composite modules) construction, and it is assumed that design parameters for the DP-SC, such as steel thickness, steel ductility, and concrete thickness, are selected to withstand perforation of the RB due to aircraft and aircraft engine impact.	Airplane crash considers the impact from a large aircraft. There is no applicable cliff- edge effects.	Expected to occur greater than 1.0E-07/yr. Retained for Fault List.
Solar Flare	The event includes impact from electric fields produced from solar flares.	1 Qualitative	Solar flares can cause Geomagnetically-Induced currents that leads to a LOPP, Geomagnetically induced.	Weather related LOPP events are included in the Internal Events PSA IEs.	Expected to occur greater than 1.0E-07/yr. Retained for Fault List.

Table 15.6-3 External Hazards Screening
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External Event	Description	BWRX-300 Screening	Hazard Screening Disposition	Cliff-Edge Effects	Fault List Disposition
Land Rise/ Land Collapse	The event is defined as impact on the plant from land rise or land collapse including subsidence.	1 Qualitative	It is assumed that the CRN-1 Site selection includes an evaluation to exclude those sites that display historical evidence of land rise or subsidence also including clay swelling/shrinking. Site selection avoids areas of karst landforms subject to karst collapse, which are known to result in the dissolution of bedrock by erosion.	The CRN-1 Site is subject to further assessment for karst collapse.	Land rise/ collapse is screened from the Fault List.
Soil Frost	The event is defined as impact on the plant from soil frost.	1 Qualitative	It is assumed that BWRX-300 building foundations are designed to address potential soil frost and that pipes are buried below the frost line. There is no mention of soil frost in the ESPA SSAR.	The plant is assumed to be free from impacts from this hazard.	Expected to occur greater than 1.0E-07/yr. Retained for Fault List.
Animals	The event is defined as impact on the plant from animals.	1 Qualitative	BWRX-300 plant fencing and doors prevent intrusion into plant buildings. Animals in switchyard are implicitly included in the internal events LOPP IE.	The site is assumed to be free from impacts from this hazard, with the exception of a potential LOPP, which is addressed by internal events LOPP IE.	Expected to occur greater than 1.0E-07/yr. Retained for Fault List.

Table 15.6-3 External Hazards Screening
(Sheet 11 of 21)

External Event	Description	BWRX-300 Screening	Hazard Screening Disposition	Cliff-Edge Effects	Fault List Disposition
Volcanic Phenomena	The event is defined as impact on the plant from volcanic eruptions.	1 Qualitative	There are no volcanoes within close proximity to the CRN Site and thus this hazard is screened out.	The CRN Site is assumed to be free from impacts from this hazard.	Screened from inclusion in Fault List.
Avalanche	The event is defined as impact on the plant from avalanches.	1 Qualitative	No avalanche potential exists. There are no mountains with large snow accumulation close to site.	The CRN Site is assumed to be free from impacts from this hazard.	Screened from inclusion in Fault List.
Above-Water Landslide	The event is defined as impact on the plant from above-ground landslide.	1 Qualitative	ESPA SSAR, Section 2.4.5.3.1 states: “The SSAR also states that there is no evidence of landslide-induced seiches and that the slopes around the CRN Site are stable (TVA2017-TN5387).”	The CRN Site is assumed to be free from impacts from this hazard.	Screened from inclusion in Fault List.
External Fire	The event is defined as impact on the plant from fires originating from outside plants, inside or outside the site area.	1 Qualitative	The land coverage surrounding the CRN-1 Site is assumed to have minimal vegetation, limited to bushes, and mostly paved in the areas proximal to site buildings. It is assumed that within the site there are limited numbers of trees, and that the area is mostly paved and cleared to preclude the possibility of animals nesting, external fires, or tornado missiles, causing damage to equipment, or impacting control room.	The CRN-1 Site has analyzed external fires as part of ESP application as discussed in this report. Internal fires are analyzed by PSA.	External fires are screened from inclusion in Fault List.

Table 15.6-3 External Hazards Screening
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External Event	Description	BWRX-300 Screening	Hazard Screening Disposition	Cliff-Edge Effects	Fault List Disposition
Excavation Work	The event is defined as impact on the plant from excavation work inside or outside the site area.	1 Qualitative	It is assumed that plant procedures are in place to preclude damage to underground equipment, such as piping or tanks, from excavation work.	The plant is assumed to be free from impacts from this hazard.	Expected to occur greater than 1.0E-07/yr. Retained for Fault List.
Direct Impact from Heavy Transportation within Site	The event is defined as damage to the plant from direct impact from heavy transportation within site. This also includes the containment external maintenance platform.	1 Qualitative	It is assumed that plant procedures are in place to preclude damage to equipment, such as piping or tanks, from heavy transportation within the site.	The plant is assumed to be free from impacts from this hazard.	Expected to occur greater than 1.0E-07/yr. Retained for Fault List.
Missiles from Military Activity	The event is defined as impact on the plant from missiles from military activity.	4 Qualitative	Missile impacts from military activity are assumed to be bounded by the Airplane Crash hazard, Event Code A25.	Addressed by the assessment of Airplane Crash Event Code A25.	Addressed by the assessment of Airplane Crash, Event code A25.
Missiles from other Plant On-Site	The event includes damage from missiles generated at another plant on the site.	4 Qualitative	It is assumed there is no other nuclear power plant beside Clinch River BWRX-300. Irrespective, missiles from sources outside the BWRX-300 are addressed in BWRX-300 Conceptual High Wind Scoping Assessment. There are no other nuclear power plants beside Clinch River BWRX-300. Thus, this specific hazard is not applicable.	The CRN Site is assumed to be free from impacts from this hazard.	Missiles from other plants to site is screened from Fault List.

Table 15.6-3 External Hazards Screening
(Sheet 13 of 21)

External Event	Description	BWRX-300 Screening	Hazard Screening Disposition	Cliff-Edge Effects	Fault List Disposition
Internal Fire Spreading from Other Plant	The event is defined as impact on the plant from fires originating in another plant on the site.	1 Qualitative	It is assumed that the Clinch River BWRX-300 plant is not contiguous with a different plant.	The CRN Site is assumed to be free from impacts from this hazard.	Screened from inclusion in Fault List.
Contamination from Chemicals	Chemical releases pose the potential to spread toxic chemicals affecting main control room habitability.	4 Qualitative	Refer to CRN-1 ESP, Section 2.2.3.3.6, Permit Conditions, Second Bullet states; Permit Condition 2.2-3 (Permit Condition 2).	Addressed by Chemical Release Outside or Inside the Site hazard Event Code A20.	Addressed by the assessment of Chemical Release Outside or Inside the Site, Event Code, which is screened from inclusion in Fault List.
Erosion	The event includes damage to plant structures due to removal of surface materials (soil and rock).	1, 7 Qualitative	It is assumed that the CRN Site is mostly paved such that erosion of the ground near plant buildings is not possible. In addition, this is a slow-developing event. If conditions exist that impact the plant's operation, the plant is shut down prior to that occurrence.	The CRN Site is assumed to be free from impacts from this hazard.	Screened form inclusion in Fault List.

Table 15.6-3 External Hazards Screening
(Sheet 14 of 21)

External Event	Description	BWRX-300 Screening	Hazard Screening Disposition	Cliff-Edge Effects	Fault List Disposition
High Soil Temperature	The event is defined as impact on the plant from high soil temperature	1, 7 Qualitative	It is assumed that BWRX-300 building foundations are designed to address potential high soil temperatures and that pipes are buried below the frost line. The ultimate heat sink is not located underground, so soil temperature should not affect the ultimate heat sink. In addition, this is a slow developing event.	The CRN Site is assumed to be free from impacts from this hazard.	Screened from inclusion in Fault List.
Low Soil Temperature	The event is defined as impact on plant from low soil temperature.	1, 7 Qualitative	It is assumed that BWRX-300 building foundations are designed to address potential low soil temperatures and that pipes are buried below the frost line. The ultimate heat sink is not located underground, so soil temperature should not affect the ultimate heat sink. In addition, this is a slow developing event.	The CRN Site is assumed to be free from impacts from this hazard	Expected to occur greater than 1.0E-07/yr. Retained for Fault List.
Soil Liquefaction	The event includes damage to plant structures due to normally solid ground becoming a viscous liquid	4 Qualitative	This hazard is addressed by the Scoping Seismic PSA for the BWRX-300 as Soil Liquefaction occurs during seismic events.	Addressed by the Scoping Seismic PSA PS01. The Scoping Seismic PSA addresses potential cliff edge effects.	Retained for Fault List.

Table 15.6-3 External Hazards Screening
(Sheet 15 of 21)

External Event	Description	BWRX-300 Screening	Hazard Screening Disposition	Cliff-Edge Effects	Fault List Disposition
Strong Water Current (underwater erosion)	The event includes damage to plant structures due to strong water current.	1 Qualitative	Straight wind events can produce storm waves, posing the potential for the formation of rip currents and underwater erosion in the vicinity of the plant intake structure, and this hazard is modified by the BWRX-300 Conceptual High Wind Scoping Assessment. Note that the High Wind Assessment assumes the loss of the intake structure for high wind events.	The CRN Site is assumed to be free from impacts from this hazard in absence of high wind. The High Wind Assessment addressed potential cliff-edge effects from strong water currents caused by high wind.	Expected to occur greater than 1.0E-07/yr. Retained for Fault List.
Low Water Level	The event is defined as plant impact due to low water level	7 Qualitative	Low water is a slowly developing event. It is assumed that the plant is shut down and to reach a safe stable state prior to the loss of circulating water or PCW and the power conversion system. ICS success is not impacted by the loss of the circulating water or PCW and power conversion system. Severe geological shifting could accompany an earthquake and potential rapid loss of the circulating water or PCW and power conversion system. This potential will be addressed in Scoping Seismic PSA.	The evaluation of this hazard places no limit on the hazard intensity, any potential cliff-edge effects are implicitly addressed. The Scoping Seismic PSA addressed a potential sudden loss of the ultimate heat sink caused by geological shifting.	Expected to occur greater than 1.0E-07/yr. Retained for Fault List.

Table 15.6-3 External Hazards Screening
(Sheet 16 of 21)

External Event	Description	BWRX-300 Screening	Hazard Screening Disposition	Cliff-Edge Effects	Fault List Disposition
High Water Level	The event is defined as plant impact due to high water level. The high levels may be due to storm surges, waves, and seiches. They are also affected by variations due to tides.	1 Qualitative	Refer to ESPA SSAR, Sections 2.4.7.2, 2.4.4.2.3, and 2.4.4.5	The evaluation of this hazard places no limit on the hazard intensity, any potential cliff-edge effects are implicitly addressed.	Screened from Fault List due to qualitative input.
High Water Temperature	The event is defined as plant impact due to high water temperature.	7 Qualitative	This is a slow developing event. The plant power level can be adjusted to accommodate rises in cooling water temperature, if necessary.	The evaluation of this hazard places no limit on the hazard intensity, any potential cliff-edge effects are implicitly addressed.	Screened from inclusion in Fault List.
Low Water Temperature	The event is defined as plant impact due to low water temperature.	7 Qualitative	This is a slow developing event. PCW and circulating water flow can be adjusted to accommodate low cooling water temperature if necessary.	The evaluation of this hazard places no limit on the hazard intensity, any potential cliff-edge effects are implicitly addressed.	Screened from inclusion in Fault List.

Table 15.6-3 External Hazards Screening
(Sheet 17 of 21)

External Event	Description	BWRX-300 Screening	Hazard Screening Disposition	Cliff-Edge Effects	Fault List Disposition
Underwater Landside	The event is defined as plant impact due to underwater landslide. An underwater landslide may be due to above-water causes, such as prolonged intense precipitation.	3 Qualitative	There is no information on underwater landslide in the ESPA SSAR, submission. It is assumed underwater landslide is not expected to occur on river sites unless the river flows directly into the ocean.	The CRN Site is assumed to be free from impacts from this hazard.	Screened from inclusion in Fault List.
Surface Ice	The event is defined as plant impact due to thick surface ice.	1 Qualitative	Refer to ESPA SSAR, Section 2.4.7.3.	The CRN Site is designed to address surface ice.	Expected to occur greater than 1.0E-07/yr. Retained for Fault List.
Frazil Ice	The event is defined as plant impact due to formation of frazil ice in the cooling water intake.	1 Qualitative	Frazil ice forms in the presence of supercooled water. Supercooled water could form on the water body floor at the CRN-1 Site, where the cooling water intakes are assumed to be located. This could restrict cooling water intake. Refer to ESPA SSAR, Section 2.4.7.7.	The CRN Site is assumed to be free from impacts from this hazard.	The hazard is screened from the Fault List.

Table 15.6-3 External Hazards Screening
(Sheet 18 of 21)

External Event	Description	BWRX-300 Screening	Hazard Screening Disposition	Cliff-Edge Effects	Fault List Disposition
Ice Barriers	The event is defined as plant impact due to formation of ice barriers.	4 Addressed by the BWRX-300 Conceptual High-Wind Scoping Assessment	This hazard is signified as "Wind-Driven Ice" in the BWRX-300 Conceptual High Wind Scoping Assessment. ESPA SSAR, Section 2.4.7.3 states "As demonstrated above, ice could build up to a maximum thickness of approximately 11 in at the CRN Site and vicinity for the worst winter on record. Being located in a relatively mild climate region, the plant will not typically be subject to extreme ice conditions. Detailed design of the closed cycle cooling tower and intake structure was not performed to support the ESP; however, detailed design of the closed-cycle cooling tower and intake structure will consider ice effects, consistent with standard industry practice, to protect against unplanned interruption of the operation of the cooling system."	BWRX-300 Conceptual High Wind Scoping Assessment addressed potential cliff-edge effects.	Expected to occur greater than 1.0E-07/yr. Retained for Fault List.

Table 15.6-3 External Hazards Screening
(Sheet 19 of 21)

External Event	Description	BWRX-300 Screening	Hazard Screening Disposition	Cliff-Edge Effects	Fault List Disposition
Organic Material in Water	The event is defined as plant impact due to organic material in intake water. The material may be algae, seaweed, fish, mussels, jellyfish, or similar materials.	4 Qualitative	Algae, fish, and other marine life entrapment in the screen house could restrict cooling water intake. This potential is addressed by the internal events PSA loss of PCW and loss of condenser heat sink IEs.	Addressed by the internal events PSA loss of PCW and loss of condenser heat sink IEs.	Expected to occur greater than 1.0E-07/yr. Retained for Fault List.
Corrosion (from Salt Water)	The event is defined as impact due to corrosion.	3 Qualitative	The TVA CRN Site is located in Tennessee, not on a saltwater site. This hazard is not applicable.	Corrosion impacts are not applicable for the freshwater site.	The hazard is screened from the fault list.
Solid or (non-gaseous) Impurities from Ship Release	The event is defined as impact due to solid (non-gaseous) impurities released into the water from a ship.	1 Qualitative	ESPA SSAR, Section 2.2.2.4 states: “There are 802 stream miles in the lower Clinch River Watershed, which is located in East Tennessee.	There is inconsequential shipping on the river near the CRN-1 site.	Screened from inclusion in Fault List.
Chemical Release to Water	The event is defined as impact due to chemical releases to water. The focus is on reduction of water quality. The releases may be due to a ship accident by may also originate from land.	1 Qualitative	Refer to ESPA SSAR, Section 2.2.3.3.5	An accidental release of oil or liquids into Clinch River does not affect the safe operation or shutdown of units at the CRN Site.	Screened from inclusion in Fault List.

Table 15.6-3 External Hazards Screening
(Sheet 20 of 21)

External Event	Description	BWRX-300 Screening	Hazard Screening Disposition	Cliff-Edge Effects	Fault List Disposition
Direct Impact from Ship Collision	The event is defined as direct impact from a ship collision.	1, 4 Qualitative	Refer to ESPA SSAR, Section 2.2.3.1.5	The CRN Site fuel cooling is assumed to be free from impacts from this hazard. The BWRX-300 is designed to have fuel cooling with loss of intake structure.	Screened from inclusion in Fault List.
Sedimentation	The event is defined as plant impact due sediments in the water.	4 Qualitative	Collection of sediments could restrict cooling water intake. This potential is addressed by the internal events PSA loss of PCW and loss of condenser heat sink IEs.	Addressed by the internal events PSA loss of PCW and loss of condenser heat sink IEs.	Expected to occur greater than 1.0E-07/yr. Retained for Fault List.
Underwater Debris	The event is defined as plant impact due to non-organic material in intake water. The material may be trash or sediments or similar materials	4 Qualitative	Debris entrapment in the screen house could restrict cooling water intake. This potential is addressed by the internal events PSA loss of PCW and loss of condenser heat sink IEs.	Addressed by the internal events PSA loss of PCW and loss of condenser heat sink IEs.	Expected to occur greater than 1.0E-07/yr. Retained for Fault List.
Earthquake	Seismic events pose the potential to damage multiple plant SSCs due to ground motion.	Selected for PSA Development	The earthquake hazard was selected to proceed directly to detailed PSA.	The BWRX-300 Scoping Seismic Probabilistic Safety Assessment address potential cliff-edge effects.	Expected to occur greater than 1.0E-07/yr. Retained for Fault List.

Table 15.6-3 External Hazards Screening
(Sheet 21 of 21)

External Event	Description	BWRX-300 Screening	Hazard Screening Disposition	Cliff-Edge Effects	Fault List Disposition
Internal Fire	Internal fire events pose the potential to damage or fail multiple plant SSCs.	Selected for PSA Development	The Internal Fire hazard was selected to proceed directly to detailed BWRX-300 Fire PRA.	The BWRX-300 Fire PRA addresses potential cliff-edge effects.	Expected to occur greater than 1.0E-07/yr. Retained for Fault List.
Hurricane	The event is defined as damage to the plants due to hurricane. The event is separated from other strong winds due to its special characteristics both with respect to duration, wind speed, and frequency of occurrence.	4	Refer to ESPA SSAR, Section 2.3.1.3.3.5	Addressed by the detailed BWRX-300 High Wind PRA. The spectrum of hurricane wind speeds is evaluated.	Retained in the Fault List.
Dam Failure	Dam failures can occur due to a seismic event, or random dam failure by itself. The result is high water levels.	1 Qualitative	Refer to ESPA SSAR, Section 2.4.4.4, Section 2.4.4.3, Sections 2.4.4.2.1, 2.4.4.2.2, and 2.4.4.2.3	The CRN Site fuel cooling is assumed to be free from impacts from this hazard.	Screened from inclusion in Fault List.
Drones	Drones are increasing being used by utilities at nuclear sites for inspections	1 Qualitative	The impact of drones hitting Clinch River SSCs is expected to be bounded by small aircraft crash. Thus, drones are screened out.	The CRN-1 Site fuel cooling is assumed to be free from impacts from this hazard.	Screened from inclusion in Fault List.

Table 15.6-4 Generic List of Boiling Water Reactor Transient Initiating Events

No.	Initiating Event
1	Electric load rejection
2	Electric load rejection with Turbine Bypass Valve failure
3	Turbine trip
4	Turbine trip with Turbine Bypass Valve failure
5	Main Steam Isolation Valve closure
6	Inadvertent closure of one Main Steam Isolation Valve
7	Partial Main Steam Isolation Valve closure
8	Loss of normal condenser vacuum
9	Pressure regulator fails open
10	Pressure regulator fails closed
11	Inadvertent opening of a safety/relief valve (stuck)
12	Turbine bypass fails open
13	Turbine bypass or control valves cause increased pressure (closed)
14	Recirculation control failure - increasing flow
15	Recirculation control failure - decreasing flow
16	Trip of one recirculation pump
17	Trip of all recirculation pumps
18	Abnormal start up of idle recirculation pump
19	Recirculation pump seizure
20	FW - increasing flow At Power
21	Loss of FW heater
22	Loss of all FW flow
23	Trip of one FW pump (or condensate pump)
24	FW - low flow
25	Low FW flow during startup or shutdown
26	High FW flow during startup or shutdown
27	Rod withdrawal At Power
28	High flux due to rod withdrawal at startup
29	Inadvertent insertion of rod or rods
30	Detected fault in Reactor Protection System
31	LOPP
32	Loss of auxiliary power (loss of auxiliary transformer)
33	Inadvertent startup of High-Pressure Core Injection System/High Pressure Core Spray System
34	Scram due to plant occurrences
35	Spurious trip via instrumentation, RPS fault
36	Manual scram - no out-of-tolerance condition
37	Cause unknown

Table 15.6-5 Internal Hazard Quantitative Screening Analysis

Hazard	Quantitative Screening Result
Turbine-Generated Missiles	Screened from further PSA analysis
Internal Flooding	Covered by Internal Flooding PSA
Heavy Load Drop	To be covered by Internal Events PSA in the future

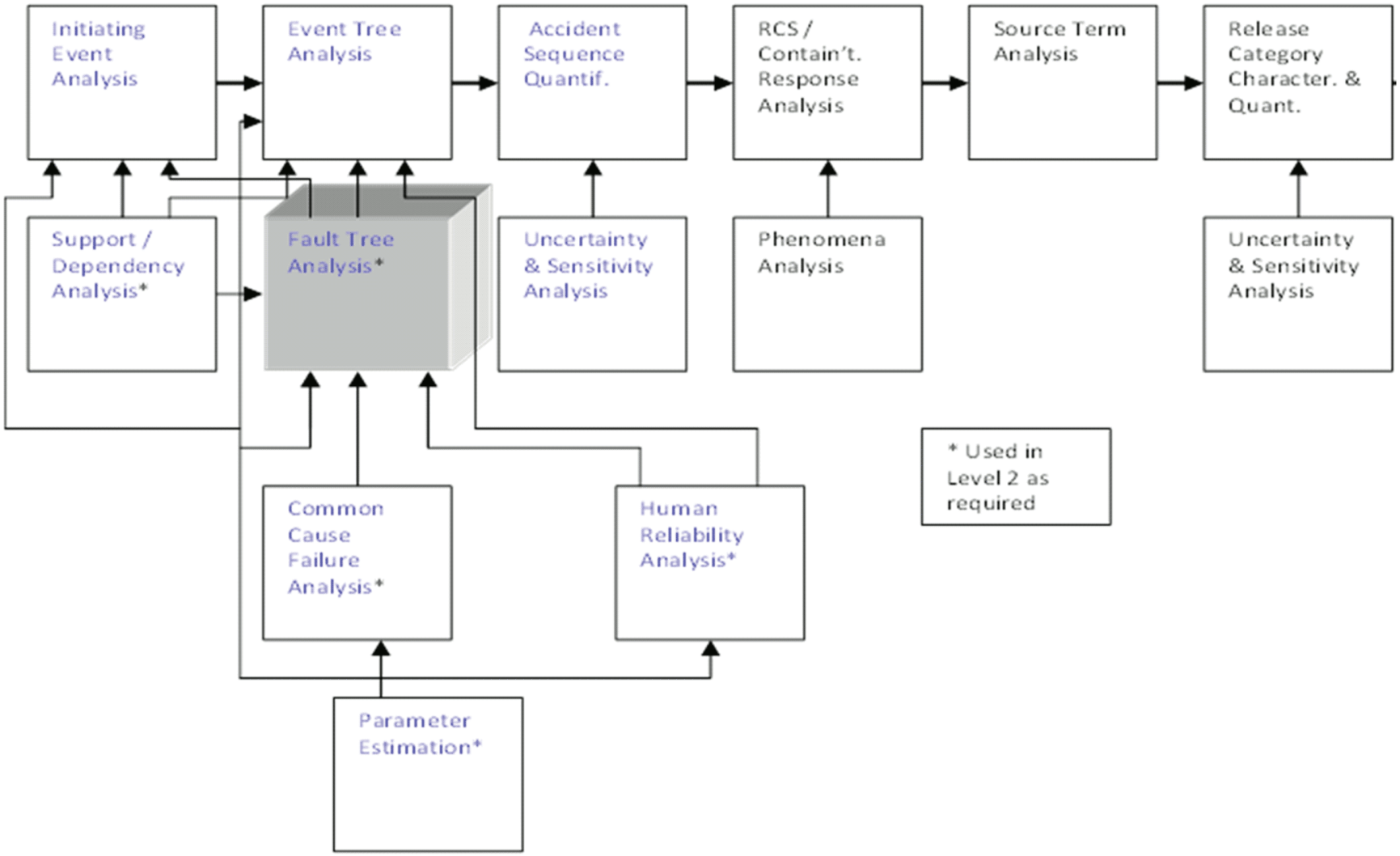

Figure 15.6-1 Principal Steps in Probabilistic Safety Assessment

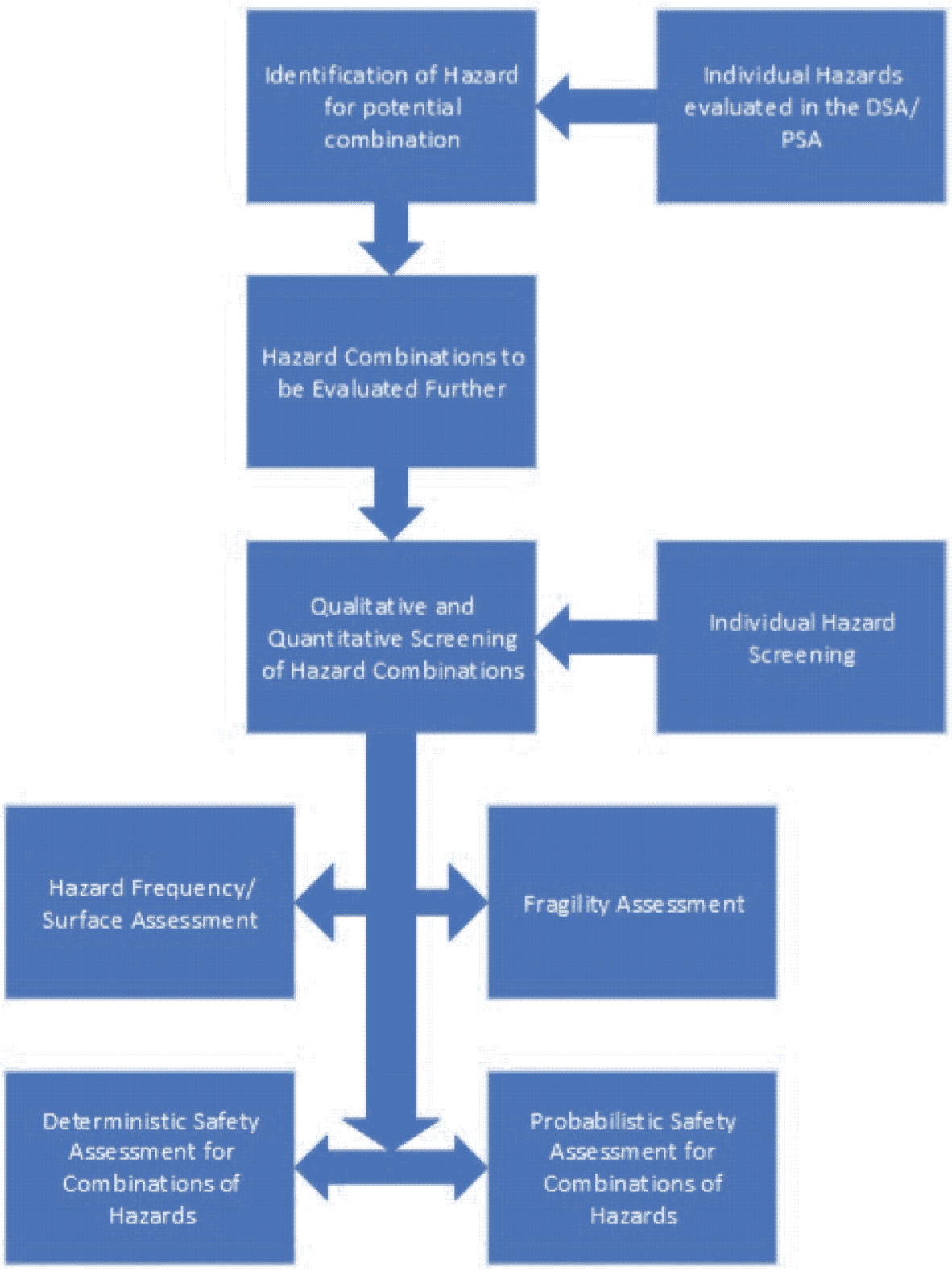

Figure 15.6-2 Overall Process for Selection, Screening and Analysis of Hazards

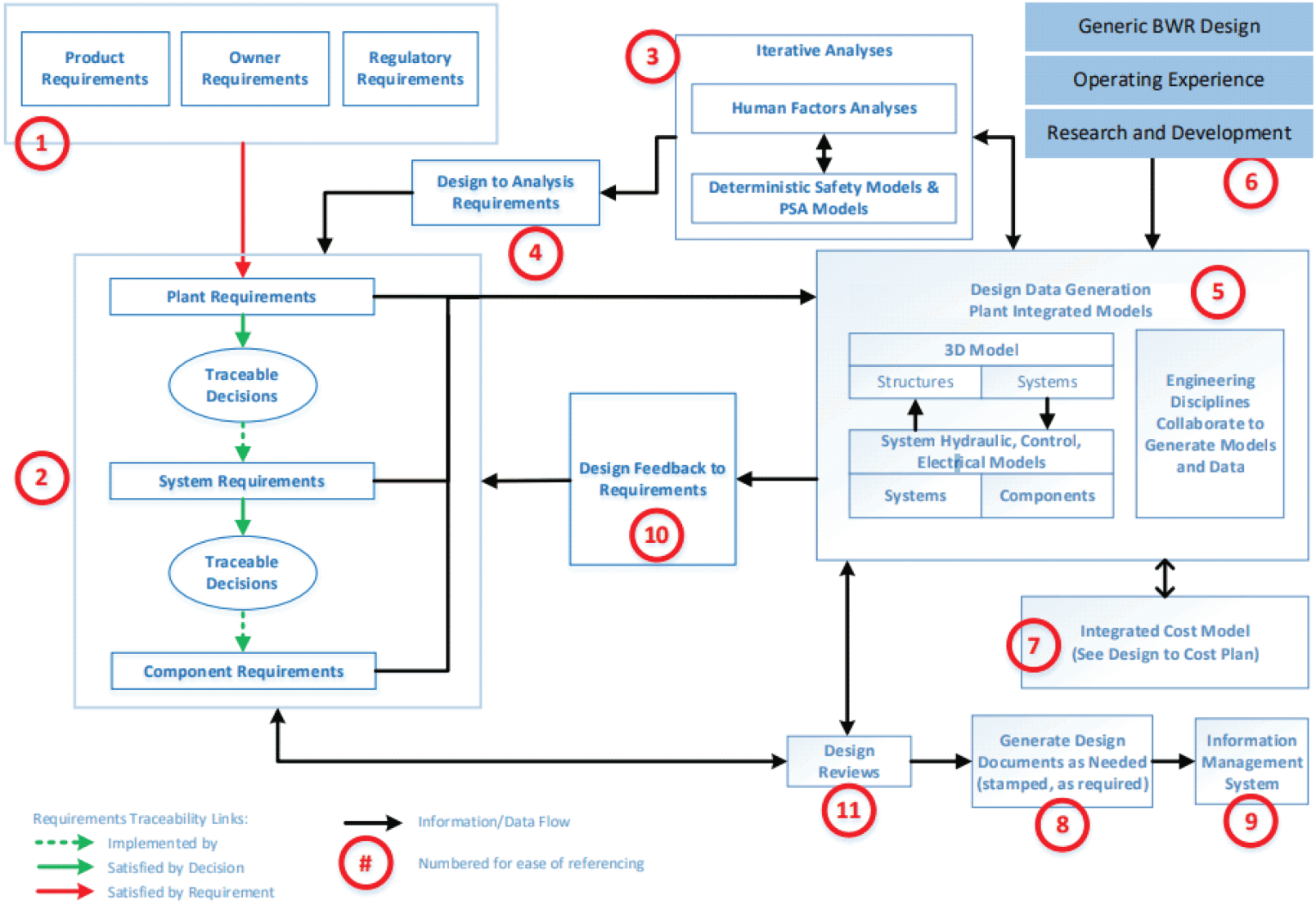

Figure 15.6-3 Integrated Design Engineering Process

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 CRN-1 Main Steam Line Break Outside Containment Accident Dose Consequences

Table 15.7-6 BWRX-300 CRN-1 Feedwater Line Break Outside Containment Accident Dose Consequences

Table 15.7-7 BWRX-300 CRN-1 Isolation Condenser System Line Break Outside Containment Accident Dose Consequences

Table 15.7-8 BWRX-300 CRN-1 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 BWR 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 FSAR.

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 FSAR.

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

Note:

1. NBR: Nuclear Boiler Rated

Table 15.7-2 Results Summary of the DBA and DEC Events – Non-LOCA (Sheet 1 of 2)

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)
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)

Table 15.7-2 Results Summary of the DBA and DEC Events – Non-LOCA (Sheet 2 of 2)

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)
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

Exposure Location and Time Period/Duration	Maximum Calculated TEDE, rem (Sv)	Acceptance Criterion TEDE, rem (Sv)
EAB for the worst 2 hours	2.4 (2.4E-02)	6.3 (0.063)
LPZ for 30 days	0.17 (1.7E-03)	6.3 (0.063)
MCR for 30 days	0.17 (1.7E-03)	5.0 (0.050)

Table 15.7-5 BWRX-300 CRN-1 Main Steam Line Break Outside Containment Accident Dose Consequences

Exposure Location and Time Period/Duration	Maximum Calculated TEDE, rem (Sv)	Acceptance Criteria TEDE, rem (Sv)
EAB for Any (Worst) 2-Hour Period		
Equilibrium Iodine Activity	1.9E-01 (1.9E-03)	2.5 (0.025)
Pre-Incident Spike	2.2E-00 (2.2E-02)	25.0 (0.25)
Outer Boundary of LPZ for the Duration of the Accident (30 days)		
Equilibrium Iodine Activity	1.2E-02 (1.2E-04)	2.5 (0.025)
Pre-Incident Spike	1.4E-01 (1.4E-03)	25.0 (0.25)
MCR Operator Dose for the Duration of the Accident (30 days)		
Equilibrium Iodine Activity	2.1E-02 (2.1E-04)	5.0 (0.05)
Pre-Incident Spike	2.4E-01 (2.4E-03)	5.0 (0.05)

Table 15.7-6 BWRX-300 CRN-1 Feedwater Line Break Outside Containment Accident Dose Consequences

Exposure Location and Time Period/Duration	Maximum Calculated TEDE, rem (Sv)	Acceptance Criteria TEDE, rem (Sv)
EAB for Any (Worst) 2-Hour Period		
Equilibrium Iodine Activity	2.3E-01 (2.3E-03)	2.5 (0.025)
Pre-Incident Spike	2.8E+00 (2.8E-02)	25.0 (0.25)
Outer Boundary of LPZ for the Duration of the Accident (30 days)		
Equilibrium Iodine Activity	1.4E-02 (1.4E-04)	2.5 (0.025)
Pre-Incident Spike	1.8E-01 (1.8E-03)	25.0 (0.25)
MCR Operator Dose for the Duration of the Accident (30 days)		
Equilibrium Iodine Activity	2.7E-02 (2.7E-04)	5.0 (0.05)
Pre-Incident Spike	3.2E-01 (3.2E-03)	5.0 (0.05)

Table 15.7-7 BWRX-300 CRN-1 Isolation Condenser System Line Break Outside Containment Accident Dose Consequences

Exposure Location and Time Period/Duration	Maximum Calculated TEDE, rem (Sv)	Acceptance Criteria TEDE, rem (Sv)
EAB for Any (Worst) 2-Hour Period		
Equilibrium Iodine Activity	1.4E-02 (1.4E-04)	2.5 (0.025)
Pre-Incident Spike	3.5E-02 (3.5E-04)	2.5 (0.025)
Outer Boundary of LPZ for the Duration of the Accident (30 days)		
Equilibrium Iodine Activity	8.4E-04 (8.4E-06)	2.5 (0.025)
Pre-Incident Spike	2.2E-03 (2.2E-05)	2.5 (0.025)
MCR Operator Dose for the Duration of the Accident (30 days)		
Equilibrium Iodine Activity	1.6E-03 (1.6E-05)	5.0 (0.05)
Pre-Incident Spike	2.0E-02 (2.0E-04)	5.0 (0.05)

Table 15.7-8 BWRX-300 CRN-1 Instrument Line Break Outside Containment Accident Dose Consequences

Exposure Location and Time Period/Duration	Maximum Calculated TEDE, rem (Sv)	Acceptance Criteria TEDE, rem (Sv)
EAB for Any (Worst) 2-Hour Period		
Equilibrium Iodine Activity	1.1E-01 (1.1E-03)	2.5 (0.025)
Pre-Incident Spike	1.4E+00 (1.4E-02)	2.5 (0.025)
Outer Boundary of LPZ for the Duration of the Accident (30 days)		
Equilibrium Iodine Activity	1.0E-01 (1.0E-03)	2.5 (0.025)
Pre-Incident Spike	1.3E+00 (1.3E-02)	2.5 (0.025)
MCR Operator Dose for the Duration of the Accident (30 days)		
Equilibrium Iodine Activity	2.3E-01 (2.3E-03)	5.0 (0.05)
Pre-Incident Spike	2.7E+00 (2.7E-02)	5.0 (0.05)

Table 15.7-9 Core Damage Frequency Preliminary Results
(Final Results Provided in FSAR)

PSA Events	Core Damage Frequency (CDF) (/yr)
Internal Events at Power	1.1E-08
Internal Events Low Power & Shutdown	7.0E-10
Seismic Events	2.8E-07
Internal Fire Events	1.3E-08
High Wind	4.3E-09 (Straight Wind) 1.3E-10 (Tornado)
Internal Flood	1.5E-09
Fuel and Heavy Load Movements	FDF At Power 2.3E-09 FDF at LPSD 1.8E-09 CDF at LPSD 1.6E-09
Fuel Pool Events	1.3E-08

Note:

1. Fuel Damage Frequency (FDF) is derived to differentiate fuel damage occurring outside of the reactor.

Table 15.7-10 Large Release Frequency Preliminary Results
(Final Results Provided in FSAR)

Level 2 PSA Scope	Large Release Frequency (LRF) Events per Year
Internal Events at Power	1.8E-09
Internal Events Low Power & Shutdown	7.0E-10
Seismic Events	2.7E-07
Internal Fire Events	1.3E-08
High Wind	4.3E-09 (Straight Wind) 1.3E-10 (Tornado)
Internal Flood	5.5E-10
Fuel and Heavy Load Movements	5.7E-09
Fuel Pool Events	1.3E-08

Note:

1. The Internal Fire Events and High Wind LRFs are assumed to be equal to the Internal Fire Events and High Wind CDFs.

APPENDIX 15A PRACTICAL ELIMINATION CLAIMS AND PROVISIONS

The BWRX-300 Safety Strategy, NEDC-33934P, outlines the concept of “Practical Elimination.” This concept is implemented in this section by capturing the supporting demonstration of engineering judgement, deterministic analyses, and probabilistic assessments. It should also be noted that some practical elimination claims are substantiated by applying an adequate amount of defense-in depth by means of defense line functions. The failures or conditions where a practical elimination claim is made are not included in the deterministic analyses described in NEDC-33934P, Section 3.2.6. They are included in the probabilistic analyses described in NEDC-33934P, Section 3.2.7, except where it is demonstrated that the failure or condition cannot, by the laws of nature, occur. Practical elimination claims and provisions to achieve practical elimination are provided in Table 15A-1, Practical Elimination Claims and Provisions. See Appendix 15B for discussion on complementary design features.

Table 15A-1 Practical Elimination Claims and Provisions
(Sheet 1 of 3)

Practically Eliminated Claim	Provisions to Achieve Practical Elimination
The possibility that a sudden mechanical failure of RPV, in which the failure eliminates the capability of holding and cooling the core, is practically eliminated.	DL1 design and supporting operating provisions for a strong RCPB for all operating states within the plant design envelope, including DECs, of the highest reliability and quality according to ASME Boiler and Pressure Vessel Code throughout the lifetime provides greater confidence that the likelihood of sudden RPV failure is extremely unlikely.
The possibility of a single control rod falling out of the core is practically eliminated with several features that make it extremely unlikely.	DL1 design provision that the control blades include a bayonet style coupling design provision that requires a 45-degree rotation to uncouple, thus making it extremely unlikely for the control rod blade to become uncoupled from the drive during reactor operation. DL2 dual separation detection devices that sense if the hollow piston is no longer on the ball nut block control rod withdrawal and do not allow a separation distance to occur between the FMCRD and the control rod, or from the ball nut to an unlatched hollow piston. This essentially limits possible separation such that it is not physically possible for a control rod drop accident involving a single control rod falling completely out of the core to occur. Section 4.6 discusses these design provisions for the FMCRD and the control rods.
The possibility of direct containment heating following a postulated vessel failure is practically eliminated by reducing the likelihood of RPV failure at high pressure and by providing a strong containment design with complementary design features and design and supporting operating provisions.	DL4a ICS actuation provides greater confidence that RPV failure occurring at high pressure is extremely unlikely. DL4b RPV ultimate pressure regulation complementary design feature that provides a diverse means to limit the increase in reactor vessel pressure such that there are no credible core damaging accident sequences involving the core without cooling while the reactor is also pressurized, making direct containment heating following a postulated vessel failure highly unlikely with a high degree of confidence. DL4b complementary design feature of a containment ultimate overpressure vent ensures containment does not overpressurize when RPV ultimate pressure regulation is actuated. DL1 design and supporting operating provisions for a strong containment for all states within the plant design envelope are of the highest reliability and quality according to ASME Boiler and Pressure Vessel Code throughout the lifetime.

Table 15A-1 Practical Elimination Claims and Provisions
(Sheet 2 of 3)

Practically Eliminated Claim	Provisions to Achieve Practical Elimination
The possibility of containment challenges from large steam explosions is practically eliminated by complementary design features and supporting operating provisions.	DL1 design provisions of a strong containment structure, components, and sufficiently large containment volume make containment load challenges from large steam explosions extremely unlikely. DL1 design and operating provisions ensure that inherent physical limitations associated with explosive interactions (e.g., no subcooled water pool) are established and maintained.
The possibility for containment challenges because of combustible gas detonation is practically eliminated by complementary design features and supporting operating provisions.	DL1 design provision for an inert containment atmosphere and supporting combustible gas management operating provisions limiting concentrations well below detonation levels make containment challenges from combustible gas detonation extremely unlikely.
The possibility of containment failure from molten core concrete interaction is practically eliminated by design and supporting operating provisions.	DL1 design provision for a corium shield/liner supported by operating provisions confine and prevent the spread of molten core, making containment failure from molten core concrete interaction extremely unlikely.
The conditional containment failure by quasi-static overpressurization from a long-term loss of containment heat removal is practically eliminated by complementary design feature to vent containment.	DL4b complementary design feature of a containment vent and operating provisions make containment failure by quasi-static overpressurization from a long-term loss of containment heat removal extremely unlikely.
The possibility of an interfacing system LOCA (ISLOCA) outside containment is practically eliminated by design provisions. An ISLOCA is defined in Subsection 15.6.3.1.1.	DL1 design provisions ensuring subsystems connected to the RCS are designed to an ultimate rupture strength at least equal to the RCS design pressure, making an ISLOCA extremely unlikely.
The condition of containment bypass consequential to Severe Accident (SA) progression is practically eliminated by design and operating provisions.	DL1 design and operating provisions for a strong RCPB for all states within the plant design envelope are the highest reliability and quality according to ASME Boiler and Pressure Vessel Code throughout the lifetime provides greater confidence that the likelihood of mechanical failure of the RIV internals due to severe accident conditions is extremely unlikely.

Table 15A-1 Practical Elimination Claims and Provisions
(Sheet 3 of 3)

Practically Eliminated Claim	Provisions to Achieve Practical Elimination
The possibility of an SA with an open containment is practically eliminated by design and supporting operating provisions.	DL1 design provisions of a large inventory of available water and operating provisions and ample time for several diverse means to provide makeup water and prevent fuel from being uncovered in the fuel pool or a shutdown of the reactor during a shutdown condition with the containment open provides greater confidence that the likelihood of an SA with an open containment is extremely unlikely.
The possibility a high pressure melt ejection with potential for debris dispersion such that debris is not retained in the lower containment is practically eliminated by complementary design features and design and supporting operating provisions.	DL4b ultimate pressure regulation complementary design feature that provides a diverse means to limit the increase in reactor vessel pressure ensures that there are no credible core damaging accident sequences involving the core without cooling while the reactor is also at high pressure, making containment failure from high pressure melt ejection extremely unlikely. DL4b complementary design feature of a containment vent and operating provisions make containment failure by quasi-static overpressurization from a high pressure melt ejection extremely unlikely. DL1 design provision for a corium shield/liner supported by operating provisions confine and prevent the spread of a molten core, making containment failure from molten core concrete interaction extremely unlikely.
Any credible possibility of an out of core criticality event with fuel handling is practically eliminated by design provisions and supporting operating provisions.	DL1 design provisions and supporting operating provisions provide design and administrative controls that effectively manage factors that influence system reactivity and the likelihood of criticality. These factors are enrichment, mass, moderation, geometry, reflection, interaction, and spacing. Plant procedures prohibit the handling and storage of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible in unborated water. The amount of water moderator available for internal flooding is limited by the design of the storage location structures thus making an out of core criticality event extremely unlikely.

APPENDIX 15B COMPLEMENTARY DEFENSE LINE 4 FUNCTIONS FOR MITIGATING DECs

Complementary design features are used to prevent accident progression or mitigate the consequences of DECs. Complementary design features mitigating functions are provided in Table 15B-1.

Complementary design features are finalized as the BDBA PSA Level 2 analysis progresses.

Table 15B-1 Complementary Design Features
(Sheet 1 of 2)

DEC Function	Complementary Design Feature
Reactivity Control	DL4a complementary design feature diverse protection system actuation logic diverse and independent from the DL3 Reactor Protection Systems actuation logic provides greater confidence that the condition is extremely unlikely to progress to a severe accident condition. DL4a complementary design feature of the fine motion control drive motor run-in function provides greater confidence that the condition is extremely unlikely to progress to a severe accident condition. DL4a complementary design feature of alternate control rod insertion to ensure that hydraulic control unit scram pressure is released to the fine motion control rod drives even in the unlikely event of failure of the scram solenoid pilot valves, provides greater confidence that the condition is extremely unlikely to progress to a severe accident condition. DL4b ultimate pressure regulation complementary design feature that provides a diverse means to limit the increase in reactor vessel pressure for the extremely unlikely condition, thus making failure propagation from the RPV to the containment extremely unlikely. DL4b boron injection system complementary design feature provides a means to place the plant in a stable, controlled configuration if control rods fail to insert. DL1 design provision that the control blades include a bayonet style coupling that requires a 45-degree rotation to uncouple, makes it extremely unlikely for the control rod blade to become uncoupled from the drive during reactor operation. DL2 dual separation detection devices that sense if the hollow piston is no longer on the ball nut block control rod withdrawal and do not allow a separation distance to occur between the fine motion control rod drive and the control rod or from the ball nut to an unlatched hollow piston. This essentially limits possible separation such that it is not physically possible for a control rod drop accident involving a single control rod falling completely out of the core to occur. DL1 design provisions and supporting operating provisions provide design and administrative controls that effectively manage factors that influence system reactivity and the likelihood of criticality. These factors are enrichment, mass, moderation, geometry, reflection, interaction and spacing. Plant procedures prohibit the handling and storage of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible in unborated water. The amount of water moderator available for internal flooding is limited by the design of the storage location structures thus making any credible possibility of an out of core criticality extremely unlikely.

Table 15B-1 Complementary Design Features
(Sheet 2 of 2)

DEC Function	Complementary Design Feature
RPV Depressurization	DL4b ultimate pressure regulation complementary design feature that provides a diverse means to limit the increase in reactor vessel pressure ensure that there are no credible core damaging accident sequences involving the core without cooling while the reactor is also pressurized, making direct containment heating following a postulated vessel failure or high pressure melt ejection extremely unlikely.
In-Vessel Core Cooling	None required because in-vessel retention is not considered for severe accidents.
Cooling of Corium Debris in the Containment Corium Shield/Liner	DL1 design provision for a corium shield/liner supported by operating provisions confine and prevent the spread of a molten core, making containment failure because of molten core concrete interaction extremely unlikely. The corium/shield liner contains the melt debris and provides a means of cooling the debris bed to prevent core concrete interaction and ultimate containment overpressurization. DL4b design provisions for corium shield cooling are provided. This is provided by a primary containment system piping connection routing water from the reactor cavity to the corium shield.
Cooling of High Pressure Melt Ejection Debris	Dispersal of debris material outside the corium shield/liner is limited by eliminating energetic phenomena like high pressure melt ejection so that it does not contribute to a significantly elevated containment temperature transient.
Containment Isolation	DL4a containment isolation actuation function provides greater confidence that containment isolation failure in the event of a CCF of DL3 isolation functions is extremely unlikely.
Containment Pressure Control: Heat Removal	DL1 design provision for passive containment cooling provides a means to remove decay heat.
Containment Pressure Control: Venting	DL4b complementary design feature of a containment vent and operating provisions make containment failure by quasi-static overpressurization from a long-term loss of containment heat removal is extremely unlikely.
Combustible Gas Control	DL1 design provision for an inert containment atmosphere and supporting combustible gas management operating provisions limiting concentrations well below detonation levels make containment challenges because of combustible gas detonation is extremely unlikely.
Post-Accident Monitoring	DL1 design provision for post-accident monitoring provides information to facilitate post-accident response and evaluation of RPV and containment conditions.

APPENDIX 15C EVALUATION OF RADIOLOGICAL CONSEQUENCES PER 10 CFR 50.34(A)(1)(ii)(D)

An evaluation assuming a fission product release from the core into containment is performed to satisfy the requirements of 10 CFR 50.34(a)(1)(ii)(D). The evaluation is based on a Maximum Hypothetical Accident (MHA) that results in core melt and is selected irrespective of an event sequence meeting a defined frequency threshold. Consistent with RG 1.183, the MHA is a conservative surrogate accident that is intended to challenge aspects of the facility design. As such, the evaluation provides insights into the mitigative capability of the containment and the classification of SSCs associated with mitigating the offsite consequences of such an event. Evaluation of events meeting the Deterministic Safety Analysis criteria is addressed in the main body of this chapter.

Consequences of the MHA are assessed in terms of the Total Effective Dose Equivalent (TEDE) at the Exclusion Area Boundary (EAB) and the boundary of the Low Population Zone (LPZ) to assess compliance with 10 CFR 50.34(a)(1)(ii)(D). The potential TEDE dose in the Main Control Room Envelope (MCRE) is also evaluated against the dose criterion of PDC 19 discussed in Subsection 3.1.2.10.

Section 15C.1 provides the bases of the analysis. Section 15C.2 discusses the methodology for analyzing the MHA and the results of the analysis.

15C.1 Bases

Subsections 15C.1.1 through 15C.1.5 summarize the bases for the evaluation.

15C.1.1 Maximum Hypothetical Accident

The MHA assumes a substantial meltdown of the core with a subsequent release into containment of appreciable quantities of fission products. The MHA is not derived from the fault evaluation process nor is it categorized by the BWRX-300 Safety Strategy, which is discussed in NEDC-33934P, “BWRX-300 Safety Strategy,” (Reference 15C-1). Rather, the MHA is evaluated as a licensing basis event and is a hypothetical major accident involving a release from the core to the containment in order to assess the performance of containment and systems that mitigate the potential release of fission products in accordance with 10 CFR 50.34(a)(1)(ii)(D).

Multiple active failures are assumed in order to hypothesize the MHA with substantial meltdown of the core and subsequent release into the containment of appreciable quantities of fission products. Additionally, the BWRX-300 design does not utilize active containment fission product removal systems to mitigate the consequences of offsite releases. Thus, an additional single active component failure that results in the most limiting radiological consequences as defined by RG 1.183 is not postulated.

15C.1.2 Core Inventory

The inventory of fission products in the reactor core available for release to the containment is a core-average inventory based on the maximum full-power operation of the core over a range fuel enrichments and exposures. Using an extended fuel burnup for an assumed 24-month refueling

cycle, and the maximum expected core thermal power maximizes fission product inventory, which is consistent with RG 1.183, Position 3.1. Table 15C-1 reports the BWRX-300 core inventory for the 60 isotopes used by the USNRC for Alternative Source Term (AST) dose consequences analysis as shown in NUREG/CR-6604, “RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation,” (Reference 15C-2). A core power measurement uncertainty factor of 1.02 is applied in the RADTRAD dose model consistent with RG 1.183.

15C.1.3 Radionuclide Release Pathways

Leakage from the Primary Containment System (PCS) is postulated through three types of pathways:

- Containment leakage into the Reactor Building (RB)
- Bypass leakage that bypasses the RB
- Engineered Safety Feature (ESF) leakage

Potential radionuclide release points associated with each of these pathways are discussed in Subsection 15C.2.3.

15C.1.4 Structures, Systems, and Component Performance

Integrity of the PCS is assumed with the potential dose evaluated based on the PCS design leakage rate (L_a) as defined in Table 6.2-1. The PCS is described in Section 6.2.

The pool of water that forms in the PCS sump, described in Subsection 6.2.1.2.1, is assumed to have a pH of 7 or greater to eliminate consideration of potential iodine re-evolution in the dose calculations. This assumption will be verified by design or analysis prior to the FSAR.

The MCRE is serviced by the Control Building (CB) Heating, Ventilation, and Cooling System (HVS). The CB HVS is credited for reducing airborne radiological exposure to personnel located in the MCRE, which encompasses the MCR and supporting rooms. The switch from normal to emergency filtration is assumed to occur 10 minutes after start of the MHA. The CB HVS is described in Subsection 9.4.1.

The RB, described in Subsection 3.8.4.1, is credited for dilution and holdup of containment leakage into the RB for calculating onsite dose. No credit for RB dilution or holdup is taken for calculating offsite dose. The RB HVS isolates automatically using isolation dampers. The RB HVS is described in Subsection 9.4.6.

15C.1.5 Operator Action

Operator action to mitigate the MHA is not credited.

15C.2 Methodology

The MHA dose consequences are conservatively calculated using the assumptions for an MHA in the guidance of RG 1.183 and relevant SSC information as summarized in Subsection 15C.1.4. Consistent with RG 1.183, L_a is defined at P_a in which P_a is determined by the design basis accident evaluation provided in Subsection 15.5.4. The evaluation demonstrates that the limiting value of P_a is associated with a large main steam line break and is illustrated in Figure 15.5-104 and Figure 15.5-105. After 24 hours, L_a is reduced by 50% consistent with RG 1.183.

Other elements of the analysis and supporting computer codes are:

- The ARCON code, NUREG/CR-6331, “Atmospheric Relative Concentrations in Building Wakes,” (Reference 15C-3) is used to model radionuclide transport to the onsite MCRE and offsite EAB and LPZ locations. Further description of the ARCON code is provided in Subsection 3M.1.15 and 15.5.1.2.7.
- The MAAP code, “MAAP 5.06 Applications Guidance,” (Reference 15C-4) has the capability to model the progression of an MHA and model fission product transport and natural deposition in the PCS. The MAAP code is used in this evaluation only to quantify fission product removal mechanisms for isotopes in aerosol form. Further description of the MAAP code is provided in Subsection 3M.1.6.
- The RADTRAD code, NUREG/CR-6604 is used to calculate EAB, LPZ, and MCRE doses. The code accounts for radionuclide removal mechanisms such as natural deposition in the containment and radionuclide deposition in piping. Decay and ingrowth of daughters are calculated over time as the material is transported. Further description of the RADTRAD code is provided in Subsection 3M.1.12 and Subsection 15.5.1.2.6.
- The MicroShield code,” MicroShield Version 7, User's Manual,” (Reference 15C-5) calculates the photon and energy fluxes at a receptor after shielding attenuation. MicroShield converts photon flux at the receptor to the exposure rate in air. MicroShield is used to calculate shine dose.

Subsections 15C.2.1 through 15C.2.6 discuss modeling of postulated radionuclide release from the fuel to potential dose receptors.

15C.2.1 Fraction and Timing of Core Inventory Released to Containment

The radioactivity released from the fuel in the MHA is assumed to mix instantaneously and homogeneously throughout the free air volume of the PCS as it is released. The free air volume accounts for the volume occupied by structures and components within the PCS.

Consistent with RG 1.183, there are two in-vessel release phases: (a) gap and (b) early in-vessel. The activity released from the core during each release phase is modeled as increasing in a linear fashion over the duration of the phase. This analysis does not analyze the coolant release phase (0 to 2 minutes) because it is conservatively assumed that the gap release

phase begins at the onset of the MHA (t = 0). The evaluation period is 30 days. The MHA timing is provided in RG 1.183, Table 5. The core inventory release fractions, by radionuclide groups for the release phases, are provided RG 1.183, Table 1.

The PCS sump pH is assumed to be 7 or greater for 30 days. Consistent with RG 1.183, Position A-1.1, the chemical form of radioiodine released from the reactor coolant system after the MHA to the PCS atmosphere is 95% as CsI (aerosol iodine), 4.85% as elemental iodine, and 0.15% as organic iodine.

15C.2.2 Fraction of Fission Product Removed from Containment Atmosphere

The radionuclide concentration in the containment atmosphere drives the potential releases. Removal phenomena credited in the PCS are (i) natural deposition on containment surfaces and (ii) retention in the pool of water that will accumulate at the bottom of containment as a result of the MHA. The removal of elemental iodine and aerosol fission products by deposition on containment surfaces is credited for all leakage pathways. Other forms of fission product removal are not credited within this analysis.

Organic Iodine

Organic iodine may deposit on PCS surfaces, but the phenomena involved are complex and the potential reduction is expected to be small. Therefore, removal of organic iodine in the PCS is conservatively neglected.

Elemental Iodine

The elemental iodine removal rate constant is calculated based on guidance found in NUREG/CR-0009, “Technical Bases for Models of Spray Washout of Airborne Containments in Containment Vessels,” (Reference 15C-6). Per NUREG/CR-0009, elemental iodine is the dominant iodine species in post-accident containment atmospheres and removed at an appreciable rate by spray drop absorption and by wall deposition. The BWRX-300 does not contain containment sprays, so only elemental iodine removal by wall deposition is credited. The elemental iodine removal rate constant is based on the Knudsen-Hilliard model described in NUREG/CR-0009. This model views deposition as a gas film transport process to vessel surfaces in which the bulk gas in the containment atmosphere is assumed to be well-mixed. As noted in SRP 6.5.2, this model represents the removal of elemental iodine by wall deposition.

The surface area applied in the iodine deposition model is calculated as the sum of the containment wall surface area, 50% of the floor area, and a conservative calculation of the Passive Containment Cooling System (PCCS) external pipe surface area. Although enhanced thermophoresis driven deposition is expected, the PCCS cooling system piping surfaces are only included as a wetted-surface in the Knudsen-Hilliard wall deposition model. The use of the NEDE-33279P, “ESBWR Containment Fission Product Removal Evaluation Model,” (Reference 15C-7) assumption that only 50% of the floor area is credited accounts for surfaces within containment in which deposition may not occur. The volume of containment is the minimum free air volume used in RADTRAD. The maximum credit for removal of elemental iodine by natural deposition is limited to a factor of 200.

Aerosol Fission Products

The fraction of aerosolized fission products removed from the containment atmosphere is determined using the MAAP code. The fission product release is modeled using MAAP's mechanistic correlations, which depend on fuel temperature, cladding oxidation, and fuel degradation state. Release fractions are calculated dynamically as fuel undergoes degradation and relocation, providing time-resolved source-terms for subsequent containment transport modeling. The MAAP model accounts for aerosolized radionuclide removal by natural mechanisms:

- Aerosol dynamics
- Thermophoresis and diffusiophoresis
- Impaction and inertial deposition
- Surface condensation and washout

The thermal hydraulic conditions for fission product deposition in containment for the MHA is based on a double-ended guillotine break in an Isolation Condenser System (ICS) return line; both Reactor Isolation Valves (RIVs) are assumed to fail, making the large break unisolated. Additionally, systems that could mitigate the event are assumed to be unavailable to prevent core melt. For the first 90 seconds after the break, coolant is discharged from the Reactor Pressure Vessel (RPV), after which, steam continues to be discharged from the break. The two-phase flow through the break leads to uncovering the core and core melt. As a result of the coolant discharge from the break, the RPV is depressurized, and a pool of water accumulates at the bottom of containment. Operator action to mitigate the event is not credited. The release into containment is terminated at the end of the Early-in-vessel (EIV) phase consistent with the guidance of RG 1.183.

Table 15C-2 provides aerosol removal rates from containment.

15C.2.3 Radionuclide Release Points to the Environment

As stated in Subsection 15C.1.3, there are three types of radionuclide release paths included in the dose models. Each type is discussed in the following subsections. Consistent with RG 1.183, radionuclide decay and daughtering are considered.

15C.2.3.1 Reactor Building Leakage

For radionuclides that may leak from PCS into the seismically qualified RB, the RB is modeled as a potential radionuclide release point. The RB is credited with mixing and holdup for onsite (MCRE) dose calculations. Holdup and mixing in the RB are not credited for offsite (EAB, LPZ) dose calculations. Leakage into the RB is conservatively modeled using only a portion of the RB that is above grade. Mixing and dilution of the radionuclide activity in that portion of the RB occur in the "free" air volume, which accounts for 20% volume of the RB volume being occupied by structures and equipment. Consistent with RG 1.183, mixing is limited to 50% of the free air volume in the above grade portion of the RB. Deposition on RB surfaces is not credited. The RB exfiltration rate is assumed to be 80 cfm (54.6 weight percent of the free air mixing volume per day).

For lines that terminate in the RB (e.g., Process Radiation and Environmental Monitoring System, ICS), holdup and deposition within these pathways are not credited. Leakage into the RB from lines terminating in the RB is accounted for in L_a, along with leakage from the personnel airlock, equipment hatch, and containment closure head.

15C.2.3.2 Bypass Leakage

Bypass leakage refers to leakage through Containment Isolation Valves (CIVs) in lines that bypass the RB. Such leakage may occur with systems that lead directly to the environment or to buildings other than the RB. For these pathways, holdup in the PCS is credited. Elemental Iodine and aerosol fission product removal within containment are credited for these pathways. For larger pipes, holdup in the piping is credited. Holdup is not credited for smaller lines because the potential reduction in activity would be small; those lines are grouped together as “generic bypass lines.”

Nuclear Boiler System (NBS): The NBS includes Main Steam Lines (MSLs) A and B, that extend from the RPV through the RIVs, through the RB, and into the Turbine Building (TB). Leakage from the MSL Train A outboard CIV is modeled as the sum of the MSL A, RPV Head Vent, and ICS Gas Purge Line A leakage rates. The ICS Gas Purge Lines B and C form a pathway from ICS Condenser B and C, respectively, to MSL Train B. The MSL Train B leakage is modeled as the sum of the MSL B, ICS Gas Purge Line B, and ICS Gas Purge Line C leakage rates.

Control Rod Drive system (CRD): The CRD system contains two CIVs associated with the purge and charging water lines for each of the Hydraulic Control Units (HCUs). Piping connections are provided to the Condensate and Feedwater Heating System (CFS) in the TB. There are 29 HCUs which are modeled with a conservative piping configuration.

Reactor Water Cleanup System (CUW): The CUW provides a leakage pathway to the TB. The CUW has two trains, A and B, from the RPV, with two sets of RIVs. The two lines join prior to the containment penetration. Therefore, the CUW is modeled as a single train.

Condensate and Feedwater Heating System (CFS): The CFS contains four Feedwater (FW) lines from the RPV, with two RIVs per line. The four lines join into two lines prior to the containment penetration. The FW leakage is modeled as two trains, each accounting for two FW lines.

Containment Inerting System (CIS): The CIS contains three lines with potential CIV leakage. The CIS supply, CIS exhaust, and CIS overpressure vent lines each provide a separate release point outside of the RB.

Generic Bypass Lines: Two generic bypass lines are modeled to account for different potential release points:

- Generic Train A represents releases on the RB/TB interface, e.g., the Plant Pneumatic System (PPS).

- Generic Train B represents releases into the Radwaste Building (RWB), including the Liquid Waste Management System (LWMS), Demineralized Water, and the Equipment and Floor Drain System (EFS).

15C.2.3.3 Engineered Safety Feature Leakage

Per RG 1.183, ESF systems that recirculate sump water outside of the primary containment are assumed to leak during their intended operation. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components. This release source may also include leakage through valves isolating interfacing systems.

The BWRX-300 containment does not include a suppression pool. The only contributor to ESF leakage is from the ICS, which recirculates reactor coolant outside of the primary containment. ESF leakage is assumed to occur from the ICS condensate return lines.

The total ESF liquid leakage is split between ESF seat and seal leakage. ESF seat leakage is included in L_a. ESF seal leakage is excluded from L_a and conservatively includes ESF assumptions in RG 1.183. ESF leakage is into the RB in which mixing and holdup are credited. All ESF leakage includes a correction for flashing and density to account for the leakage being defined as liquid leakage but modeled as steam leakage.

ESF leakage is modeled consistent with RG 1.183, Position A-4.

15C.2.4 Radionuclide Dispersion

Radionuclide transport from the potential sources to onsite and offsite locations is based on atmospheric dispersion factors (χ/Qs) calculated for the CRN Site. Consistent with RG 1.183, credit is not taken for depletion during transport to either onsite or offsite dose locations. Consistent with RG 1.183, Position 5.3, regarding the application of the χ/Qs, the worst 2-hour χ/Q is applied to the worst 2-hour dose in the analysis.

Atmospheric dispersion factors are calculated using the ARCON code described in Subsection 3M.1.15 and Subsection 15.5.1.2.7.

15C.2.4.1 Onsite Radionuclide Transport

Onsite radionuclide transport to the MCRE is simulated using χ/Qs that are calculated consistent with RG 1.194. The onsite dispersion analysis is sensitive to the distance from the source to the receptor at the MCRE ventilation intakes.Each leakage pathway has a different release (i.e., source) location and therefore a different set of χ/Qs. The MCRE χ/Q values are determined for initial averaging periods of 0 to 2 hours, 2 to 8 hours, 8 to 24 hours, 24 to 96 hours, and 96 to 720 hours.

The maximum MCRE χ/Q for any air intake location (i.e., normal or emergency ventilation) is used in the analysis. For leakage exiting the RB (i.e., RB leakage and ESF leakage), transport is modeled as a ground-level diffuse area source consistent with RG 1.194. Bypass leakage is

modeled as a ground level release directly to the environment from two release points. For leakage at the TB/RB boundary, the release point is at the steam tunnel. All other bypass leakage is modeled at the limiting RB/CB boundary.

To ensure a conservative dose analysis, consistent with Position 5.3 of RG 1.183, the period of the most adverse release of radioactive materials to the environment, with respect to doses, is assumed to occur coincident with the period of most unfavorable atmospheric dispersion.

Table 15C-3 provides the dispersion factors, which reflect different distances between radionuclide release point and a receptor in the MCRE.

15C.2.4.2 Offsite Radionuclide Transport

Offsite radionuclide transport to the EAB and LPZ is simulated using offsite χ/Qs that are calculated consistent with RG 1.249. RG 1.249 provides guidance for using the ARCON code to estimate χ/Qs out to distances of 1,200 meters from the nearest edge of a building within the nuclear island. Because the χ/Qs established for the CRN Site in the “Clinch River Nuclear Site Early Site Permit, ESP-006,” (Reference 15C-8) were determined prior to the issuance of RG 1.249, χ/Qs are recalculated at the EAB distance (335 m) and are applied to the LPZ. This is a conservative approach for dose calculations because χ/Qs decrease with distance. These χ/Qs represent general transport from the CRN Site to these offsite locations and are independent of the radionuclide pathway.

As is done for onsite radionuclide transport to the MCRE, the period of the most adverse release of radioactive materials to the environment, with respect to doses, is assumed to occur coincident with the period of most unfavorable atmospheric dispersion. Note that the worst 2-hour period for offsite doses may be different than for onsite dose because onsite dose credits the RB which affects the release profile.

Table 15C-4 provides the dispersion factors applicable to offsite radionuclide transport to an offsite receptor.

15C.2.5 Receptor Dose

Consistent with 10 CFR 50.34(a)(1)(ii)(D) and PDC 19, dose is evaluated in terms of TEDE. The TEDE is the summation of the Committed Effective Dose Equivalent (CEDE) from inhalation and the Effective Dose Equivalent (EDE) from external exposure.

The RADTRAD code is used to produce the time dependent receptor dose at the MCRE, EAB, and LPZ.

15C.2.5.1 Onsite (MCRE)

Per RG 1.183, the MCRE dose receptor is the hypothetical maximum exposed individual who is present in the control room for 100% of the time during the first 24 hours after the event, 60% of the time between 1 and 4 days, and 40% of the time from 4 days to 30 days. For the duration of the event, the breathing rate of this individual is assumed to be 3.5E-04 m^3/s.

Consistent with RG 1.183, the analysis considers potential radiation sources from:

1. Contamination by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility.

2. Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope. (The BWRX-300 CB that contains the MCRE is a separate structure maintained at a positive pressure, and there are several intervening structures such that the MCRE is not amenable to infiltration from adjacent structures.)

3. Radiation shine from the external radioactive plume released from the facility.

4. Radiation shine from radioactive material in the containment.

5. Radiation shine from radioactive materials in systems and components inside or external to the MCRE (e.g., radioactive material buildup on MCRE filters).

Items 3 through 5 are associated with shine dose and are evaluated for personnel in the MCRE. MicroShield is used to calculate shine doses given radionuclide activities from RADTRAD and occupancy factors.

15C.2.5.2 Offsite (EAB, LPZ)

Per RG 1.183, the first 8 hours, the breathing rate of a dose receptor at the EAB or LPZ is assumed to be 3.5E-04 m^3/s. From 8 to 24 hours following the accident, the breathing rate is assumed to be 1.8E-04 m^3/s. After that and until the end of the accident, the breathing rate is assumed to be 2.3E-04 m^3/s. Because the EAB dose is the dose over the worst 2 hours, the limiting breathing rate (the first 8 hours) is applied to the entire accident duration.

15C.2.6 Key Assumptions

Table 15C-5 summarizes key assumptions for this analysis.

15C.2.7 Results

Table 15C-6 provides evaluation results in terms of regulatory limits. Calculated onsite and offsite radiological consequences from a postulated MHA remain within applicable regulatory limits.

15C.3 References

15C-1 NEDC-33934P, “BWRX-300 Safety Strategy,” GE Vernova Hitachi Nuclear Energy Americas, LLC, Revision 2, March 2026.

15C-2 NUREG/CR-6604, “RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation,” U.S. Nuclear Regulatory Commission, April 1998.

15C-3 NUREG/CR-6331, “Atmospheric Relative Concentrations in Building Wakes,” U.S. Nuclear Regulatory Commission, Revision 1, May 1997.

15C-4 “Modular Accident Analysis Program 5 (MAAP5) Applications Guidance,” Electric Power Research Institute, September 2021.

15C-5 “MicroShield Version 7, User's Manual,” Grove Engineering Inc., 2009.

15C-6 NUREG/CR-0009, “Technical Bases for Models of Spray Washout of Airborne Containments in Containment Vessels,” U.S. Nuclear Regulatory Commission, October 1978.

15C-7 NEDE-33279P, “ESBWR Containment Fission Product Removal Evaluation Model,” GE Hitachi Nuclear Energy Americas, LLC, Revision 3, June 2009.

15C-8 “Clinch River Nuclear Site Early Site Permit, ESP-006,” U.S. Nuclear Regulatory Commission, December 2019 (ML19352D868) and Final Safety Evaluation Report, June 2019 (ML19162A157).

Table 15C-1 Core Inventory at Time of MHA Occurrence

Nuclide	Activity (Ci/MWt)	Nuclide	Activity (Ci/MWt)
Co-58	3.058E+02	Te-131m	4.031E+03
Co-60	6.851E+02	Te-132	3.883E+04
Kr-85	5.563E+02	I-131	2.751E+04
Kr-85m	1.015E+04	I-132	3.985E+04
Kr-87	2.040E+04	I-133	5.617E+04
Kr-88	2.884E+04	I-134	6.321E+04
Rb-86	7.996E+01	I-135	5.236E+04
Sr-89	2.870E+04	Xe-133	5.413E+04
Sr-90	4.626E+03	Xe-135	2.845E+04
Sr-91	4.657E+04	Cs-134	1.055E+04
Sr-92	4.794E+04	Cs-136	2.876E+03
Y-90	4.806E+03	Cs-137	6.335E+03
Y-91	3.251E+04	Ba-139	5.320E+04
Y-92	4.804E+04	Ba-140	5.098E+04
Y-93	5.221E+04	La-140	5.368E+04
Zr-95	4.665E+04	La-141	4.864E+04
Zr-97	5.167E+04	La-142	4.822E+04
Nb-95	4.680E+04	Ce-141	4.458E+04
Mo-99	5.171E+04	Ce-143	4.814E+04
Tc-99m	4.607E+04	Ce-144	3.763E+04
Ru-103	4.562E+04	Pr-143	4.612E+04
Ru-105	3.392E+04	Nd-147	1.891E+04
Ru-106	2.280E+04	Np-239	5.928E+05
Rh-105	3.124E+04	Pu-238	2.742E+02
Sb-127	3.169E+03	Pu-239	1.699E+01
Sb-129	9.218E+03	Pu-240	2.217E+01
Te-127	3.262E+03	Pu-241	9.787E+03
Te-127m	4.337E+02	Am-241	1.705E+01
Te-129	9.064E+03	Cm-242	3.986E+03
Te-129m	1.343E+03	Cm-244	3.271E+02

Table 15C-2 Containment Aerosol Removal Rates

Time (hour)	Removal Rate (per hour)
0.000	0.133
0.800	0.976
1.500	0.763
2.500	0.624
4.200	0.321
5.000	0.255
7.600	0.274
9.590	0.231
13.500	0.125
24.080	0.000

Table 15C-3 Main Control Room Envelope Dispersion Factors (χ/Q (s/m^3))

Time Period	Pathway(s)		
	RB (Including ESF) Diffuse Source Leakage	Bypass Leakage for Pipes in Steam Tunnel	All Other Bypass Leakage(1)
0 to 2 hours	1.51E-03	9.80E-04	6.99E-03
2 to 8 hours	1.06E-03	4.71E-04	4.31E-03
8 to 24 hours	4.95E-04	2.55E-04	1.71E-03
24 to 96 hours (1 to 4 days)	4.68E-04	2.22E-04	1.64E-03
96 to 720 hours (4 to 30 days)	4.21E-04	1.99E-04	1.28E-03

Note:

1. The most limiting location is at the RB/CB boundary.

Table 15C-4 EAB/LPZ Dispersion Factors (χ/Q (s/m^3))

Location	0 to 2 Hours	2 to 8 Hours	8 to 24 Hours	24 to 96 Hours (1 to 4 Days)	96 to 720 Hours (4 to 30 Days)
EAB	8.40E-05	-	-	-	-
LPZ	8.40E-05	7.25E-05	3.94E-05	3.99E-05	3.89E-05

Table 15C-5 Key Assumptions
(Sheet 1 of 2)

Assumption	Value	Basis
General		
Licensed power level	870 MWt	Table 1.2-1
Measurement uncertainty	2% (17.4 MW)	Subsection 15.5.1.1
Analysis power level	887.4 MWt	Table 15.5-1
MHA	Large pipe break inside PCS	RG 1.183
Containment (PCS)		
Release fraction to containment atmosphere	RG 1.183 (Table 1)	RG 1.183 (Table 1)
Containment Volume (ft^3)	2.46E+05	Table 6.2-1 Free air volume per RG 1.183 (A-2.1)
Containment (PCS) design leakage rate (L_a) (Initial)	0.35 weight percent per day(1)	Table 6.2-1
Containment (PCS) design leakage rate (24 hours)	50% of initial rate	RG 1.183 (A-2.7)
Containment Aerosol Removal Rates (per hour)	Table 15C-2	MAAP (See Subsection 15C.2.2)
Containment Elemental Iodine Removal	1.69 per hour for 9.59 hours	Knudsen-Hilliard wall deposition model (See Subsection 15C.2.2)
Containment sump pH	7 or greater	Confirm at FSAR
Reactor Building Function		
Isolation	Prior to onset of MHA	Subsection 15.5.4.5.1
RB volume (ft^3)	2.11E+05	50% of above grade free air volume per RG 1.183 (A-3.4)
Exfiltration rate	80 cfm (54.6 weight percent per day)	Conservative assumption
Control Building Function		
Control Building Emergency Filtration	10 minutes after start of MHA	Engineering judgment
Dispersion and Receptor Locations		
χ/Q	CRN Site	Site specific calculation
MCRE	CB	MCR and supporting rooms
EAB distance	335 meters (1100 feet)	Table 2.0-1R; Reference 15C-8 (Table 2.3.4-10)
LPZ distance	Same as EAB	EAB X/Qs are conservatively applied to the LPZ distance

Table 15C-5 Key Assumptions
(Sheet 2 of 2)

Assumption	Value	Basis
Dose Estimates		
Receptor characteristics	Location, breathing rate, exposure time	RG 1.183
Multiplier on calculated MCRE dose	10%	Conservative assumption

Note:

1. ESF seal leakage is modeled as an additional radionuclide release contribution above that associated with L_a.

Table 15C-6 Evaluation Results

Location	Time Period	Evaluation Results (Rem)	TEDE Dose Limit (Rem)	Reference
EAB	Worst 2-hours	2.01	25	10 CFR 50.34(a)(1)(ii)(D)(1)
LPZ	30 Days	10.67	25	10 CFR 50.34(a)(1)(ii)(D)(2)
MCRE	30 Days	4.88	5	PDC 19

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LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
CRN-1	Clinch River Nuclear Unit 1
ESBWR	Economic Simplified Boiling Water Reactor
FSAR	Final Safety Analysis Report
ICS	Isolation Condenser System
LCO	Limiting Condition for Operation
NUREG	Nuclear Regulatory Report
OL	Operating License
PSAR	Preliminary Safety Analysis Report
RCS	Reactor Coolant System
RG	Regulatory Guide
SL	Safety Limit
SR	Surveillance Requirement
STS	Standard Technical Specification
TS	Technical Specifications
TSTF	Technical Specifications Task Force
USNRC	U.S. Nuclear Regulatory Commission

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CHAPTER 16 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 Clinch River Nuclear Unit 1 (CRN-1) 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 CRN-1 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 CRN-1 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 STS Chapter 1, LCO 3.0, and Surveillance Requirement (SR) 3.0. The use of consistent rules has improved understanding of TS requirements by users, compliance during operations, and communications between the U.S. Nuclear Regulatory Commission (USNRC) staff, plant operators, and between facilities when sharing operating experience. The CRN-1 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 Nuclear Regulatory Report (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 CRN-1 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 CRN-1 TS.

16.3.4 References

16.3-1 TSTF-GG-05-01, “Writer's Guide for Plant-Specific Improved Technical Specifications,” Revision 1, June 2005.

16.3-2 NUREG-1434, “Standard Technical Specifications General Electric BWR/6 Plants,” U.S. Nuclear Regulatory Commission, Revision 5.0, September 2021.

16.3-3 NUREG-1433, “Standard Technical Specifications General Electric BWR/4 Plants,” U.S. Nuclear Regulatory Commission, Revision 5.0, 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 CRN-1 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. Providing layered protection against unacceptable releases of radiation is the basic premise of the BWRX-300 defense-in-depth concept by utilizing Defense Lines (DL). For each Anticipated Operational Occurrences (AOO) 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 defense-in-depth 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

DL2 and 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) 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 DSA (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 Technical Specifications.

SC3 SSCs performing DL2 functions are subject to Technical Specifications if the SSCs are determined to be risk-significant in accordance with LCO Criterion 4. Excluding SC3 SSCs that are not risk-significant from Technical Specifications 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 Technical Specifications and initial Availability Controls Manual are subject to USNRC review and approval prior to Operating License issuance. The primary difference between Availability Controls Manual and Technical Specifications 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 Technical Specifications require prior USNRC approval. The BWRX-300 Safety Strategy's treatment of SSCs ensures that Technical Specification 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 Technical Specifications:
  - 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 Technical Specifications 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 Technical Specifications, 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 Technical Specification 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 Technical Specification and identifying DL3 functions performed by SC1 SSCs as the primary success path for LCO criterion 3.

DBAs

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 Operating License (OL) application stage.

16.4.4 References

16.4-1 NUREG-1433, “Standard Technical Specifications General Electric BWR/4 Plants,” U.S. Nuclear Regulatory Commission, Revision 5.0, September 2021.

16.4-2 NUREG-1434, “Standard Technical Specifications General Electric BWR/6 Plants,” U.S. Nuclear Regulatory Commission, Revision 5.0, September 2021.

16.4-3 NEDC-33934P, “BWRX-300 Safety Strategy,” GE Vernova Hitachi Nuclear Energy Americas, LLC, Revision 2, March 2026.

16.5 Results of Selection Methodology Application

This section 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 CRN-1 TS, including the reason for inclusion or selection of that content.

16.5.1 Use and Application

TS 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 BWRX-300 design, analyses, and content of the balance of the TS.

- 1.2 Logical Connectors

  The Logical Connectors section generally aligns with industry practice and previously licensed BWR designs, however with modifications to reflect the CRN-1 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 CRN-1 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 CRN-1 TS content and usage.

16.5.2 Safety Limits

TS 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 CRN-1 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 Section 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

TS 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 in 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

TS 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, record keeping, 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 plant 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 plant organization including shift crew requirements. It contains information similar to the level of details in other TS.

- 5.3 Plant 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 USNRC and industry 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 USNRC and industry 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 Part 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, June 2005.

Table 16.5-1 Preliminary Table of Contents and Bases for Inclusion (Sheet 1 of 3)

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 (LCO) 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 15.2.4.6, Table 15.3-2
3.1.2	Reactivity Anomalies	2	Subsections 4.3.4.3, and 7.3.3
3.1.3	Control Rod Operability	3	Subsection 15.5.4, Table 15.5-4, Table 15.5-5, Table 15.5-44, and Section 4.6
3.1.4	Control Rod Scram Times	3	Subsection 15.5.4, Table 15.5-4, Table 15.5-5, Table 15.5-44
3.1.5	Control Rod Scram Accumulators	3	Subsection 15.5.4, Table 15.5-4, Table 15.5-5, Table 15.5-44, and Section 4.6
3.2.1	Linear Heat Generation Rate	2	Subsections 4.3.1.3.1, and 7.3.3
3.2.2	Minimum Critical Power Ratio	2	Subsections 4.3.1.3.2, and 7.3.3

Table 16.5-1 Preliminary Table of Contents and Bases for Inclusion (Sheet 2 of 3)

Specification	Title	LCO Selection Criteria	Bases for Inclusion
3.3.1 Function 1	Power Range Neutron Monitoring	3	Subsections 15.5.4.1 and 15.5.4.2, Table 15.5-5
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 Line 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, Subsection 15.5.4, and Table 15.5-3
3.4.5	RCS Leakage Detection Instrumentation	1	Subsection 5.2.5
3.5.1	Isolation Condenser System (ICS)	3	Section 6.3 and Subsection 15.5.4
3.5.2	IC Pool Complex	3	Subsections 6.3.2, 15.5.4, and Tables 15.5-1 and 15.5-3

Table 16.5-1 Preliminary Table of Contents and Bases for Inclusion (Sheet 3 of 3)

Specification	Title	LCO Selection Criteria	Bases for Inclusion
3.6.1	Containment	3	Section 6.2
3.6.2	Containment Air Locks	3	Section 6.2
3.6.3	Containment Isolation Valves and Reactor Isolation Valves	3	Subsection 6.2.4, Table 6.2-2, Section 5.4.3, and Subsection 15.5.4
3.6.4	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	Subsection 6.2.2 and Subsection 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	Subsection 9.1.2
3.8.1	Emergency Power System	3	Subsection 8.3.2.1
3.8.2	Battery Parameters	3	Subsection 8.3.2.1
3.9	Refueling Operations	2, 3, 4	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	Plant 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.

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LIST OF ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
ASME	American Society of Mechanical Engineers
BWR	Boiling Water Reactor
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
COL	Combined License
CP	Construction Permit
DC	Design Certification
D-RAP	Design Reliability Assurance Program
DSA	Deterministic Safety Analysis
ESP	Early Site Permit
FMEA	Failure Modes and Effects Analysis
FSAR	Final Safety Analysis Report
FV	Fussell-Vesely
GVH	GE Vernova Hitachi Nuclear Energy
IAEA	International Atomic Energy Agency
ITAAC	Inspections, Tests, Analyses, and Acceptance Criteria
LRF	Large Release Frequency
LWA	Limited Work Authorization
OE	Operating Experience
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
PSAI	Plant-Specific Action Item
QA	Quality Assurance
QAPD	Quality Assurance Program Description
RAW	Risk Achievement Worth
RG	Regulatory Guide
SC1	Safety Class 1
SDA	Standard Design Approval
SSCs	Structures, Systems, and Components
TVA	Tennessee Valley Authority
USNRC	U.S. Nuclear Regulatory Commission

LIST OF EFFECTIVE SECTIONS

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CHAPTER 17 QUALITY ASSURANCE

17.1 QUALITY ASSURANCE DURING THE DESIGN AND CONSTRUCTION PHASES

The QA Program for site design, construction, and operation activities controlled by TVA is described in Section 17.5. TVA owns responsibility for design activities related to the CRN site 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. TVA considers GVH a vendor under the oversight of the TVA Quality Assurance Program Description in Section 17.5.

17.2 Quality Assurance During Operations Phase

The Quality Assurance Program during the operations phase is described in Section 17.5.

17.3 Quality Assurance Program Description

The Quality Assurance Program Description is described in Section 17.5.

17.4 Reliability Assurance Program

The purpose of the reliability assurance program is to provide reasonable assurance of the following:

- The plant is designed, constructed, and operated in a manner that is consistent with the risk insights and key assumptions (e.g., SSCs design, reliability, and availability) from probabilistic and deterministic methods.
- SSCs within the scope of the reliability assurance program do not degrade to an unacceptable level of reliability, availability, or condition during plant operations.
- SSCs within the scope of the reliability assurance program function reliably when challenged.
- The frequency of transients that challenge SSCs within the scope of the reliability assurance program is minimized.

The reliability assurance program is implemented in two stages: a D-RAP that consists of design and construction activities that occur before initial fuel load and the operations phase of the reliability assurance program conducted during the operations phase of the plant's license.

The programmatic elements for developing and implementing the D-RAP for the BWRX-300 design and construction are described in this section.

The list of SSCs within the scope of the reliability assurance program and the operational phase of the reliability assurance program are described in the Final Safety Analysis Report (FSAR).

17.4.1 Scope

The D-RAP is utilized during the design and equipment specification phases to assure the risk insights and key assumptions of the Probabilistic Safety Assessment (PSA) and Deterministic Safety Analyses (DSA) are included in the plant design and construction. The key assumptions can pertain to design, reliability, or availability of the SSCs. The program identifies relevant aspects of risk-significant SSC operation, maintenance, and performance monitoring, for consideration, to assure safe performance of plant equipment and to limit risk to the public.

The D-RAP ensures that plant safety is considered as detailed design evolves through the implementation and procurement phases. During the early stages of the design, the PSA group works closely with the design group to maximize plant safety through the incorporation of risk analyses findings. Pertinent information is provided in the design documentation so that equipment reliability, as it affects plant safety, is maintained throughout operations and maintenance. For example, failure modes and assigned reliability targets are reflected in detailed design and procurement documentation.

A system can be defined as a group of related components that work together to carry out one or more specified functions. For systems with multiple functions, importance, reliability, and availability needs to be assessed at a functional level because the importance and mission time of each function may not contribute equally to system performance. Therefore, much of the D-RAP is focused on the performance of functions versus systems and includes requirements specific to risk-significant SSCs.

A comprehensive list of SSCs within the scope of D-RAP is developed by using a blended PSA and deterministic approach and an expert panel as specific information becomes available. The PSA and DSA for the BWRX 300, along with other sources such as historical records of Boiling Water Reactor (BWR) systems and components, are used to identify and prioritize those SSCs that are important to prevent or mitigate plant anticipated operational occurrences, design basis accidents, and design extension conditions that could present a risk to the public.

17.4.2 GVH Organization for Design Reliability Assurance Program

The GVH BWRX-300 engineering organization is an integrated design and engineering organization that is responsible for formulating and implementing the D-RAP. Responsibilities include:

- PSA and design engineers develop the design analysis and PSA engineering that is necessary to support the development of the D-RAP.
- PSA personnel are responsible for keeping the design staff cognizant of risk-significant items, program needs, and project status.
- Both PSA and design personnel participate in the design change control process, which includes providing D-RAP related inputs in the design process.
- Both PSA and design engineers are responsible for incorporating D-RAP related inputs into the design process.

The GVH BWRX-300 engineering organization follows guidance on the design process, control of design changes, and documentation storage and retrieval controls. The design control procedure defines the process for performing, documenting, and verifying design activities. This includes developing or modifying the design of systems, engineering evaluations, analyses, calculations, and documents, (e.g., specifications, drawings, reports).

17.4.3 Structures, Systems, and Components Identification

SSCs are classified using the processes described in NEDC-33934P, “BWRX-300 Safety Strategy,” (Reference 17.4-1) and the resulting SSC classification is listed in Appendix 3A. Following SSCs classification, the list of SSCs within the scope of D-RAP is developed by using a blended approach to identify and prioritize risk-significant SSCs. This approach combines the various PSA analytical results with Operating Experience (OE), deterministic evaluations, and an expert panel process to develop a comprehensive list of risk-significant SSCs. The process for selecting SSCs that are within the scope of the D-RAP is visually represented in Figure 17.4-1.

The following are inputs to an expert panel process that develops a list of SSCs within the scope of D-RAP:

- PSA Results - The Level 1 PSA is used to evaluate accident sequences from initiating events and failures of safety functions that lead to core damage. The Level 2 PSA is used to evaluate accident sequences from core damage events that lead to release. The PSA staff

will provide the list of risk-significant components, systems, and functions based on the risk-significant criteria described in Subsection 17.4.5. This preliminary list of risk-significant functions and SSCs will be reviewed and verified by the expert panel.

- DSA Results - The DSA performed for the BWRX-300 are described in NEDC-33934P, which implements a defense-in-depth approach, and include consideration of potential for SSCs to cause transients, SSCs role in severe accidents, as well as operational experience considerations. The results of the DSA are captured in the Fault List. Functions from the Fault List not included in the PSA and any SC1 SSCs that were not categorized as risk-significant are inputs for expert panel review. If the expert panel determines that such a function is risk-significant, the expert panel identifies the components that support the function as risk-significant.

Additional resources that are inputs to the expert panel for consideration include:

- Failure Modes and Effects Analyses Results - BWRX-300 PSA system models are supported by failure modes and effects analysis, which are detailed evaluations of system functions failures, supporting system failures, and potential component failure modes. This information is available to the expert panel for consideration in their determination of risk significance.
- PSA and Severe Accident Insights and Assumptions - PSA model assumptions and risk insights are qualitative attributes that describe important aspects of the PSA model and severe accident analysis. They are evaluated by the expert panel to identify failure modes of SSCs that might be implicit within the insight to inform risk-significant determination.
- Operating Experience - The expert panel also reviews operating experience to identify previous failures of components in similar applications, revealing situations where inappropriate human actions have led to functional failures of SSCs. The expert panel assesses component operating history and industry operating experience when it can be applied to assessing risk significance.
- Defense-in-Depth - As SSCs are considered for risk-significance, the expert panel will consider the role of an SSC, the functions it performs, and the effect on defense-in-depth and safety margins of a potential SSC failure.

The expert panel considers these inputs and develops the initial D-RAP risk-significant SSC list, which is documented in a report along with the basis for inclusion, reliability targets, and dominant failure modes. The list of risk-significant SSCs is updated as new information (e.g., operating experience, design changes, failure data, PSA model updates) is available. The basis for any revisions to the initial list of SSCs within the scope of the D-RAP is documented.

17.4.4 Expert Panel

The use of an expert panel compensates for limitations of PSA implementation approaches resulting from the PSA structure (e.g., model assumptions, treatment of support systems, level of definition of cut sets, cut set truncation, shadowing effect of very large (high frequency) cut sets, and inclusion of repair or restoration of failed equipment) and limitations in the meanings of the importance measures.

To compensate for potentially inconsistent risk ranking of SSCs in the PSA from the use of thresholds for the selected importance measures, the expert panel process is used to determine the final list of risk-significant components. This approach also requires the expert panel to consider issues related to design and OE to appropriately rank SSCs and avoid misclassifying risk-significant SSCs as non-risk-significant. To determine risk-significant SSCs, the expert panel reviews both PSA and non-PSA information, as described in Subsection 17.4.3, needed to integrate the risk significance determination through multi-disciplinary reviews of PSA importance measures, OE, engineering judgment, and qualitative risk analyses. The results of the expert panel decisions are documented following each meeting, including the expertise of the represented attendees and rationale for why SSCs were screened in or out of D-RAP.

The expert panel should have a collective experience in PSA, DSA, engineering, operations, and maintenance. It is recommended that each member have at least five or more years of experience in their areas of expertise. It is expected that panel members take design for reliability training prior to performing activities under D-RAP.

17.4.5 Risk-Significant Criteria

The PSA is described in Section 15.6 and is used to identify potentially risk-significant SSCs and functions. The PSA is constructed using basic events, which typically represent failure modes for a given component. Components may be represented in the PSA by several basic events representing different failure modes of the component (i.e., fails to open or fails to close). A system can be defined as a group of related components that work together to carry out one or more specified functions. For systems with multiple functions, importance, reliability, and availability must be assessed at a functional level because the importance and mission time of each function may not contribute equally to system performance.

The numerical values of risk significance are intended to be applied to baseline risk. Thus, the BWRX-300 risk-significant criteria are established to be consistent with International Atomic Energy Agency (IAEA) PSA safety goals: Core Damage Frequency (CDF) ≤ 10^{-5}/year and Large Release Frequency (LRF) ≤ 10^{-6}/year. To convert the CDF and LRF risk significance thresholds to a Risk Achievement Worth (RAW) risk significance threshold, the IAEA PSA safety goals are divided by the total plant target CDF, which is initially targeted to be ≤ 10^{-7}/year. A significance threshold of 20 percent is applied for Fussell-Vesely (FV) importance measure to ensure that any large risk contributors are identified. The resulting criteria are used to identify potentially risk-significant information: basic event/group with RAWCDF ≥ 100 or RAWLRF ≥ 10 or FV ≥ 0.2. Note: The FV and RAW values are subject to revision as the PSA evolves. For example, if the total plant target CDF changes, the resulting risk-significant criteria for RAW are impacted based on the way this criterion is determined.

Risk-significance of any given entity depends on the nominal availability of other entities in the PSA model. For this reason, the risk significance of entities of a collection of basic events (e.g., a system) cannot be determined by the risk significance of a constituent component or basic event. Because the risk significance of high-level entities cannot be derived from the significance of

lower-level ones, mapping is performed to understand the risk significance of components, systems, and functions. The risk significance criteria are applied at whatever level of entity being examined (i.e., basic event, component, system, or function).

17.4.6 Design Considerations

The predicted reliability of SSCs in the scope of D-RAP are evaluated at the detailed design stage by appropriate design reviews, PSA, and reliability analyses. If a proposed design change affects the PSA model, then the PSA is revised in accordance with the PSA update process, and the new results will be implemented into the D-RAP process.

17.4.7 Quality Controls Applicable to Design Reliability Assurance Program

The quality controls applicable to the D-RAP process during the design phase include the design control process, procedures used by D-RAP process, the corrective action program, records and audits.

Quality controls for non-SC1 SSCs that are risk-significant are consistent with those for "Non-safety related SSCs that are significant contributors to plant safety" described in NUREG-0800, "Standard Review Plan," Section 17.5, during the design phase of D-RAP. Quality controls for non-SC1 SSCs that are risk-significant are described in Section 17.5 during the construction phase.

17.4.8 References

17.4-1 NEDC-33934P, "BWRX-300 Safety Strategy," GE Vernova Hitachi Nuclear Energy Americas, LLC, Revision 2, March 2026.

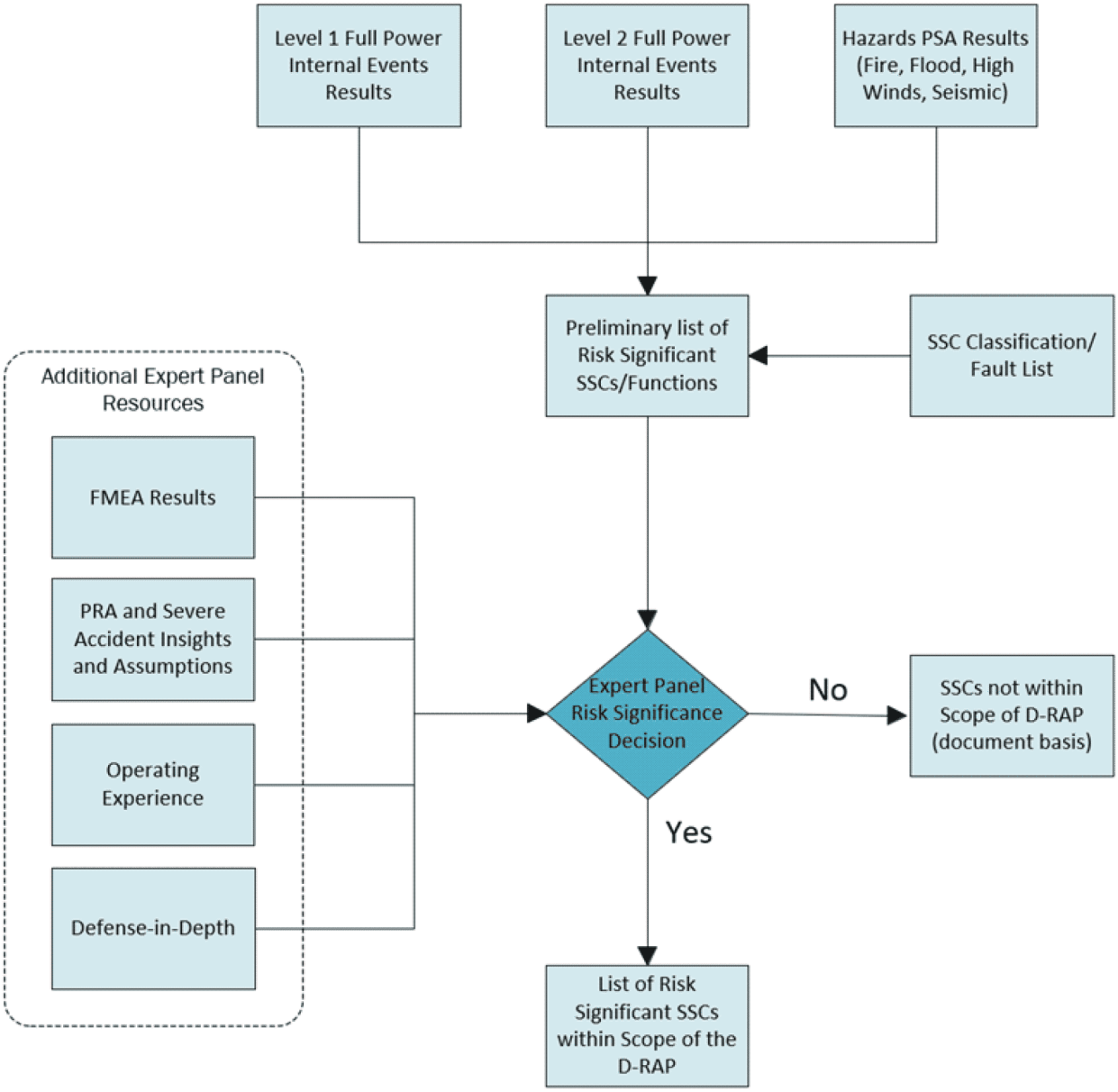

Figure 17.4-1 Process Flow for D-RAP SSCs Selection

17.5 Quality Assurance Program Description - Design Certification, Early Site Permit, and New License Applicants

The Quality Assurance Program Description (QAPD) for TVA New Nuclear is provided in the topical report NNP-TR-001-NP-A, “Quality Assurance Program Description for TVA New Nuclear,” (Reference 17.5-1). The classification of Structures, Systems, and Components (SSCs) for the BWRX-300 is described in Section 3.2. The TVA New Nuclear QAPD is applied to Safety Class 1 (SC1) SSCs relative to performance of their Safety Category 1 functions.

Revisions to NNP-TR-001-NP will be performed periodically in accordance with a defined process and transmitted to the NRC. The topical report is Incorporated by Reference with the following departures and/or supplements.

1. The last sentence in Part II, Section 1.2.1 should be read as follows:

   “The Senior Vice President, Engineering and Operations Support, is also responsible for governance, oversight, and support of existing nuclear procedures, processes, and personnel, which will be leveraged for the TVA New Nuclear Program and Projects.”

2. The final safety evaluation of NNP-TR-001-NP contains the following limitations and conditions.

   FINAL SAFETY EVALUTION BY THE OFFICE OF NUCLEAR REACTOR REGULATION REGARDING THE TOPICAL REPORT OF THE QUALITY ASSURANCE PROGRAM DESCRIPTION FOR THE TENNESSEE VALLEY AUTHORITY NEW NUCLEAR PROGRAM EPID NO. L-2022-TOP-0055 LIMITATIONS AND CONDITIONS

   An application for SDA, DC, ESP, COL, CP, OL, or LWA (as applicable) may reference the approved TVA New Nuclear QAPD provided that the application satisfies the following conditions and limitations, as applicable. The conditions and limitations are intended to address plant-specific aspects of TVA New Nuclear QA program that have not been described in the TVA New Nuclear QAPD. The TVA New Nuclear QAPD and the NRC staff's safety evaluation provides context and basis for the required additional information.

   The following plant-specific action items (PSAIs) must be satisfied by applications referencing the TVA New Nuclear QAPD:

   PSAI 4.1

   An application for SDA, DC, ESP, COL, CP, OL, or LWA (as applicable) that intends to credit the approved version of the TVA New Nuclear QAPD for activities where the regulations other than 10 CFR Part 50 and Part 52 established QA requirements must include an analysis that demonstrates the QA requirements set forth in these regulations are met by the TVA New Nuclear QAPD.

   Response: The CP application is submitted under 10 CFR Part 50.

PSAI 4.2

An application for DC, SDA, ESP, COL, CP, or OL that references the TVA New Nuclear QAPD must include an organizational diagram that aligns with the description of the roles and responsibilities of the Senior Vice President Engineering Operations Support in Part II, Sections 1.2.1 and 1.3.2.2.3 of the TVA New Nuclear QAPD.

Response: PSAI 4.2 is addressed in Section 17.5 as described above.

PSAI 4.3

An application for COL and OL that references the TVA New Nuclear QAPD must include (1) a description of site-specific positions required for operations, and (2) an organizational diagram that depicts these site-specific position group in the organizational structure.

Response: PSAI 4.3 is not applicable to a CP application.

PSAI 4.4

An application for a DC or SDA that references the TVA New Nuclear QAPD and intends to have A/E Suppliers for engineering services in support of the DC or SDA application must include a description of the management position that will oversee these A/E Suppliers.

Response: PSAI 4.4 is not applicable to a CP application.

PSAI 4.5

An application DC, SDA, ESP, COL, CP, or OL referencing the TVA New Nuclear QAPD must provide applicable standards or methodology for the selection and classification of SSCs.

Response: The methodology for the selection and classification of SSCs is described in Section 3.2.

PSAI 4.6

An application for a COL that references the TVA New Nuclear QAPD must specify that requisite design verification activities will be completed prior to relying on the design outputs to fulfill completion of ITAAC during construction.

Response: PSAI 4.6 is not applicable to a CP application.

PSAI 4.7

An application for an OL that references the TVA New Nuclear QAPD must specify that requisite design verification activities will be completed prior to relying on the design outputs in fulfilling the required information needed for OL issuance.

Response: PSAI 4.7 is not applicable to a CP application.

PSAI 4.8

An application for COL, CP, or OL that references the TVA New Nuclear QAPD must describe the qualification requirements for outside fire protection consultants who will be relied upon to conduct fire protection equipment and program inspections.

Response: No action required. Qualification requirements for fire protection staff are described in Section 1.6.1.a of RG 1.189, “Fire Protection for Nuclear Power Plants,” Revision 4. TVA New Nuclear will conform with these requirements for the selection of outside fire protection consultants. Section 1.9 addresses conformance to RG 1.189, Revision 4.

PSAI 4.9

An application for SDA, DC, ESP, COL, CP, or OL that references the TVA New Nuclear QAPD must provide information on conformances and exceptions to RG 1.8, Revision 4, RG 1.26, Revision 6, RG 1.28, Revision 5, RG 1.54, Revision 3, RG 1.164 Revision 0, RG 1.231, Revision 0, RG 1.234, Revision 0.

Response: Information on conformances and exceptions to Regulatory Guides is summarized in Section 1.9.

17.5.1 References

17.5-1 NNP-TR-001-NP-A, “Quality Assurance Program Description for TVA New Nuclear,” Tennessee Valley Authority, Revision 2, September 15, 2025 (ML25258A031).

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.