December 31, 2022 Docket No. 52-050

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Submittal of the NuScale Standard Design Approval Application Part 2 – Final Safety Analysis Report, Chapter 5, "Reactor Coolant System and Connecting Systems," Revision 0

REFERENCES:

1. NuScale letter to NRC, "NuScale Power, LLC Submittal of Planned Standard Design Approval Application Content," dated February 24, 2020 (ML20055E565)
2. NuScale letter to NRC, "NuScale Power, LLC Requests the NRC staff to conduct a pre-application readiness assessment of the draft, 'NuScale Standard Design Approval Application (SDAA),'" dated May 25, 2022 (ML22145A460)
3. NRC letter to NuScale, "Preapplication Readiness Assessment Report of the NuScale Power, LLC Standard Design Approval Draft Application," Office of Nuclear Reactor Regulation dated November 15, 2022 (ML22305A518)
4. NuScale letter to NRC, "NuScale Power, LLC Staged Submittal of Planned Standard Design Approval Application," dated November 21, 2022 (ML22325A349)

NuScale Power, LLC (NuScale) is pleased to submit Chapter 5 of the Standard Design Approval Application, "Reactor Coolant System and Connecting Systems," Revision 0. This chapter supports Part 2, "Final Safety Analysis Report," (FSAR) of the NuScale Standard Design Approval Application (SDAA), described in Reference 1. NuScale submits the chapter in accordance with requirements of 10 CFR 52 Subpart E, Standard Design Approvals. As described in Reference 4, the enclosure is part of a staged SDAA submittal. NuScale requests NRC review, approval, and granting of standard design approval for the US460 standard plant design.

From July 25, 2022 to October 26, 2022, the NRC performed a pre-application readiness assessment of available portions of the draft NuScale FSAR to determine the FSAR's readiness for submittal and for subsequent review by NRC staff (References 2 and 3). The NRC staff reviewed draft Chapter 5. NuScale is enclosing information in this submittal that: 1) closes gaps identified between the draft SDAA Chapter 5 and technical content generally expected by the NRC; and 2) resolves identified technical issues that may have adversely impacted acceptance, docketing, or technical review of the application. Section B of the enclosures provide NuScale's responses to Reference 3 for Chapter 5 observations.

Enclosure 1 contains SDAA Part 2 Chapter 5, "Reactor Coolant System and Connecting Systems," Revision 0, proprietary version. NuScale requests that the proprietary version (enclosure 1), be withheld from public disclosure in accordance with the requirements of 10

CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 1 has also been determined to contain Export Controlled Information. This information must be protected from disclosure per the requirements of 10 CFR § 810. Enclosure 2 contains the nonproprietary version.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Mark Shaver at 541-360-0630 or at mshaver@nuscalepower.com.

I declare under penalty of perjury that the foregoing is true and correct. Executed on December 31, 2022.

Sincerely,

Carrie Fosaaen
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Enclosure 1: SDAA Part 2 Chapter 5, “Reactor Coolant System and Connecting Systems,” Revision 0, (proprietary)
Enclosure 2: SDAA Part 2 Chapter 5, “Reactor Coolant System and Connecting Systems,” Revision 0, (nonproprietary)
Enclosure 3: Affidavit of Carrie Fosaaen AF-132199

Enclosure 1:

SDAA Part 2 Chapter 5, “Reactor Coolant System and Connecting Systems,” Revision 0, (proprietary)

Enclosure 2:

SDAA Part 2 Chapter 5 “Reactor Coolant System and Connecting Systems,” Revision 0, (nonproprietary)

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A	Chapter 5, “Reactor Coolant System and Connecting Systems,” Revision 0, nonproprietary
B	Readiness Assessment Review responses for Chapter 5
C	Technical Report(s)

Section A

NuScale US460 Plant
Standard Design Approval Application

Chapter Five
Reactor Coolant System and Connecting Systems

Final Safety Analysis Report

Revision 0

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CHAPTER 5 REACTOR COOLANT SYSTEM AND CONNECTING SYSTEMS

5.1 Summary Description

The reactor coolant system (RCS) provides for the circulation of the primary coolant. The US460 standard design relies on natural circulation flow for the reactor coolant and does not include reactor coolant pumps or an external piping system to drive coolant flow. The RCS is a subsystem of the NuScale Power Module (NPM). The RCS includes the reactor pressure vessel (RPV) and integral pressurizer (PZR), the reactor vessel internals (RVI), the reactor safety valves (RSVs), RCS piping inside the containment vessel (CNV) (RCS injection, RCS discharge, PZR spray supply, and RPV high-point degasification lines), the PZR control cabinet and the RCS instruments and cables.

Chapter 1, Introduction and General Description of the Plant, provides an overview of the plant design that includes up to six individual NPMs. The description in this chapter applies to each of the NPMs individually, unless otherwise stated.

5.1.1 Design Basis

The design bases of the RCS are:

- 10 CFR 50.55a. In accordance with Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, the design, fabrication, construction, testing, and inspection of the RPV and pressure retaining components associated with the reactor coolant pressure boundary (RCPB) meet the applicable conditions promulgated in 10 CFR 50.55a(b). Section 5.2.1, Compliance with Codes and Standards, provides additional details.
- General Design Criterion (GDC) 1 and GDC 30. RCS components design, fabrication, erection, and testing meet the highest quality standards practical. Sections 5.2.1 and 5.2.3 provide RCPB design details.
- GDC 4. Fabrication and design of the RPV and pressure retaining components associated with the RCPB are compatible with the environmental conditions of the reactor coolant and containment atmosphere. Section 5.2.3, RCPB Materials, provides additional details.
- GDC 14 and GDC 31. Design and fabrication of the RPV and pressure retaining components associated with the RCPB have sufficient margin to ensure the RCPB behaves in a non-brittle manner and minimize the probability of rapidly propagating fracture and gross rupture of the RCPB. The RCPB provides a barrier to the release of radionuclides. Section 5.2.3, RCPB Materials, provides additional details.
- GDC 15. Sufficient margin is in the RCS design to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences. Section 5.2.2, Overpressure Protection, provides additional details.
- GDC 32. The RCPB components design provides access to permit periodic inspection and testing of important areas and features to assess their structural

and leak-tight integrity. Section 5.2.4, RCPB Inservice Inspection and Testing, provides additional details.

5.1.2 System Description

The RCS is a subsystem of the NPM and is located inside the CNV, with the exception of the PZR control cabinet and associated cables.

The RCS interfacing systems include the chemical and volume control system via the containment system, the emergency core cooling system (ECCS) valves consisting of reactor vent valves (RVVs) and reactor recirculation valves (RRVs) and associated venturis, the CNV, control rod drive system, steam generator system (SGS), and the primary sampling system (through the chemical and volume control system). Instrumentation for the module protection system controls reactivity, maintains RCPB integrity, and performs post-accident monitoring functions. Chapter 7, Instrumentation and Controls, provides information on instrumentation.

A diagram of an individual NPM is in Figure 5.1-1 showing the RPV within the CNV and denoting major RCS components. A simplified RCS diagram is in Figure 5.1-2. Table 5.1-1 identifies the RCS volumes.

The RCS transfers approximately 250 MW of thermal power from the reactor core to the SGS during power operation. The RCS provides coolant to the reactor core such that specified acceptable fuel design limits are not violated during normal operation, including the effects of anticipated operational occurrences. The RCS removes decay heat from the reactor core during shutdown (including when the reactor is subcritical and during the initial phase of plant cooldown) and refueling operations via heat transfer to the decay heat removal system (DHRS) through the SGS and convection from the RPV, through the CNV, to the reactor pool. Temperature throughout the RCS stays below the RVI and RPV design limits during normal and Service Level A transient conditions.

The RCS maintains a uniform concentration of soluble boron during normal, transient, and accident conditions, and maintains adequate chemical and thermal mixing to ensure reactivity control. The RCS provides the core neutron moderator and when coupled with the reflector blocks (that reflect a portion of the neutrons that escape the fuel region), improves neutron economy in the core.

During normal operation, the RCS transports heat from the reactor core to the steam generators (SGs) by natural circulation. The motive force for the natural circulation reactor coolant flow is differences in coolant density between the hot coolant leaving the reactor core and the colder coolant leaving the SG annular space, and by the elevation difference between the reactor core (heat source) and the SG (heat sink). The reactor coolant is heated in the core, travels upward through the lower and upper riser assemblies, and at the top of the upper riser assembly turns downward by the PZR baffle plate. The heated coolant then flows into the annular space between the upper riser assembly and the vertical shell of the RPV. This annular space contains the SG helical tube bundles. As the reactor coolant flows downward across the outside of the SG tubes, it transfers heat to the secondary coolant inside the tubes. Heat transfer to the SG tubes lowers the temperature of the reactor coolant, increasing its density and causing the cooler, dense coolant to sink into the annular downcomer space between the lower riser assembly and core barrel and continue into the lower plenum near the bottom of the RPV where the reactor coolant returns to the reactor core. Figure 5.1-3 provides a schematic diagram of the RCS heat transfer flow loop during normal, steady-state, full-power operating conditions as described above.

The RCS uses a degasification line to remove noncondensible gases from the PZR volume at the top of the RPV during normal operation as necessary for chemistry control. The RVVs, which are opened to discharge the PZR steam space directly to the CNV as part of ECCS operation, also allow noncondensible gases to be removed from the PZR during emergency core cooling operation.

For planned shutdowns the SGS (in conjunction with the main steam and main feedwater systems) are used to remove decay heat from the RCS during the initial phase of module cooldown. When decay heat is sufficiently low and the RCS temperature is low enough to support filling the CNV, the CNV is filled to the PZR baffle plate by the containment flooding system. Nitrogen is introduced into the pressurizer, PZR spray is performed, and PZR heaters reduced and secured to collapse the PZR steam bubble, the RCS cooldown continues passively as decay heat transfers through the RPV to the flooded containment and through the CNV to the reactor pool. The DHRS actuation valves open to allow feedwater to recirculate through the DHRS heat exchangers, allowing establishment of secondary water chemistry conditions for wet layup. The SGs then isolate from the main steam and feedwater systems. The PZR water level is reduced using the chemical volume control system to match the water level in the CNV in preparation for opening the RVVs and RRVs. Nitrogen vents from the PZR through the RPV high point degasification line until PZR pressure matches containment pressure. When the PZR and containment pressure and water level match, the RVVs and RRVs open.

Control of the RCS water chemistry minimizes solid deposits on the reactor core and the SGs.

5.1.3 System Components

5.1.3.1 Reactor Pressure Vessel

The RPV is a metal vessel that forms part of the RCPB and is a barrier to the release of fission products. Most of the reactor coolant is in the RPV. The CNV supports the RPV both laterally and vertically. The RPV contains and supports the reactor core, RVI, SGs, and PZR. The RPV provides penetrations, support, and alignment for the control rod drive system. The RPV also provides penetrations and attachment locations for the RCS instruments, the PZR heaters, the ECCS valves (RVVs and RRVs), the RSVs, and RCS piping (RCS injection, RCS discharge, PZR spray supply, and RPV high point degasification lines). The RPV provides steam and feedwater plenums for the steam generators.

The RPV and appurtenances are further described in Section 5.3, Reactor Vessel.

5.1.3.2 Reactor Coolant System Piping

The RCS piping external to the RPV consists of the following lines:

- two PZR spray line branches from a common spray header
- RPV high point degasification line
- RCS injection line
- RCS discharge line

The RCS injection line has branch lines that connect to each of the ECCS valve reset valves.

Further description of the RCS piping is in Section 5.4, RCS Component and Subsystem Design.

5.1.3.3 Pressurizer

The PZR is integral to the RPV and occupies the volume inside the RPV above the PZR baffle plate. The RCS components in the PZR volume are the PZR spray nozzles and the PZR heater assemblies. The PZR provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions at a temperature greater than T_{Hot} for pressure control of the RCS during steady-state operations and transients. Maintaining the saturated conditions higher than T_{Hot} ensures the reactor coolant remains subcooled during normal operations. The PZR also serves as a surge volume. The PZR controls reactor coolant pressure within the permitted operating range for normal operating transients without actuating the RSVs.

Further description of the PZR is in Section 5.4.5, Pressurizer.

5.1.3.4 Reactor Vessel Internals

The RVI contain several sub-assemblies that provide support and alignment for the core, the control rod assemblies, the control rod drive shafts, and the instrument guide tubes. Additionally, the RVI channels reactor coolant flow from the reactor core to the SGs and back within the RPV.

The RVI sub-assemblies include the core support assembly, lower riser assembly, upper riser assembly, SG supports, instrumentation guide tubes, core support block assembly, and flow diverter.

Further description of the RVI are in Section 3.9, Mechanical Systems and Components, and Section 5.3, Reactor Vessel.

5.1.3.5 Reactor Safety Valves

Two direct spring operated RSVs connect to the top of the RPV upper head and discharge directly to the CNV. These valves are part of the RCPB and provide overpressure protection as required by the American Society of Mechanical Engineers Boiler and Pressure Vessel Code.

Further description of the RSVs are in Section 5.2.2, Overpressure Protection.

5.1.3.6 Emergency Core Cooling Valves

The ECCS valves consist of two RVVs and two RRVs. The RVVs connect to the upper head portion of the RPV and discharge the PZR steam space directly to the CNV. The RRVs connect to the RPV shell just above the main closure flange. When opened, they permit recirculation of water in the CNV back into the RPV and ultimately through the core. The ECCS valves are a part of the RCPB and function during emergency core cooling operation. The RVVs also provide overpressure protection during operations at low temperature conditions.

Further description of the RVVs and RRVs are in Section 6.3, Emergency Core Cooling System.

5.1.3.7 Steam Generators

The SG system is an integral part of the RPV comprised of the SG tubes, SG tube supports, steam and feedwater piping inside containment, SG tube inlet flow restrictors, feed plenums, and steam plenums. The design contains two independent, but intertwined, SGs located inside the RPV that facilitate heat transfer to the secondary coolant system and provide redundancy for the DHRS. The SGs are a once-through, helical-coil design, with primary-side reactor coolant outside the tubes and secondary-side fluid inside the tubes. On the primary side, the reactor coolant flows downward across the outside of the tubes, transferring heat to the fluid inside the SG tubes. On the secondary side, preheated feedwater enters the two feed plenums at the bottom of each SG, flows up the helical tubes where it is heated, boiled, and superheated, and exits the two steam plenums at the top of each SG. The DHRS connects to the steam and feed piping to permit use of the SG to remove decay heat from the primary coolant.

Further description of the SGs are in Section 5.4.1, Steam Generators.

5.1.4 System Evaluation

To optimize performance of the NPM over the range of power levels within the design basis, steady state values for primary and secondary side parameters are established as a function of reactor power. The following criteria are considered to define optimal performance: maximizing electrical generation, using support systems efficiently, and providing margin to analytical limits. The following parameters are determined: primary coolant temperatures, primary coolant flow rates, PZR water level, SG water level and mass, feedwater and steam flow rates, feedwater and steam pressure, and steam superheat.

Calculations establish a best-estimate flow, maximum flow, and minimum flow for the applicable design considerations, as well as the primary coolant temperatures at each flow condition. Table 5.1-2 contains the primary system temperatures and flow rates.

In establishing the range of primary coolant design flows, the calculations account for uncertainties in the component flow resistances and the amount of core bypass flow. Bounding uncertainties are determined based on testing data and design requirements.

The pressure losses through the RCS flow path are very low, to support the natural circulation design of the RCS. Because the pressure losses due to flow are small, the pressure at any location in the RCS flow path is primarily a function of the static head. The primary coolant flows over the outside of the helical SG tubes; therefore, SG tube plugging uncertainties are not applicable to the primary coolant natural circulation flow area determination.

Minimum design flow is the lowest expected value for the primary coolant flow rate and is calculated by biasing analytical uncertainties to minimize the flow rate. Minimum design primary coolant flow rate is used as a bounding parameter in certain design analyses.

Maximum design flow is the highest expected value for the primary coolant flow rate. Maximum design flow is calculated by biasing analytical uncertainties to maximize the primary coolant flow rate. Maximum design primary coolant flow rate is used as a bounding parameter in certain design analyses.

The module heatup system heats the RCS and provides primary coolant flow before the reactor is critical. Between hot zero power and 20 percent reactor power, the heat generated from the reactor core heats up the RCS to the normal operating average coolant temperature. Between 20 percent reactor power and full power operating conditions, the average primary coolant temperature is maintained at a constant value. The intent of the RCS temperature control scheme is to maintain a near constant average RCS density from 20 percent power to 100 percent power.

Discussion of the thermal-hydraulic design for reactor core coolant flow by natural circulation is in Section 4.4, Thermal and Hydraulic Design.

Table 5.1-1: Reactor Coolant System Volumes

RCS Region	Nominal Volume (ft^3)*
Hot Leg (lower riser, riser transition, upper riser, riser outlet)	637
Cold Leg [feedwater plenums, downcomer transition, downcomer (lower riser), core barrel, RPV bottom head]	622
Core Region (fuel assembly region and reflector cooling channels)	88
SG Region	643
PZR Region (main steam plenums, PZR, RPV top head)	583
PZR Region, cylindrical (main steam plenums and PZR)	500

*Volumes are rounded to the nearest cubic foot.

Table 5.1-2: Primary System Temperatures and Flow Rates

Reactor Power		Primary Flow		Primary Coolant Temperature			
%	MWt	%	(Kg/s)	Core dT(°F)[2]	T_{Cold} (°F)	T_{Avg} (°F)	T_{Hot} (°F)
Best-Estimate Flow							
0-20[1]	0-50.0	0-52.5	0-383.7	0-44.6	< 517.7	< 540.0	< 562.3
20	50.0	52.5	383.7	44.6	517.7	540.0	562.3
50	125.0	75.5	551.6	77.5	501.3	540.0	578.7
75	187.5	89.0	650.1	98.3	490.9	540.0	589.1
100	250.0	100.0	730.9	116.0	482.0	540.0	598.0
Minimum Design Flow							
100	250.0	88.5	647	132.0	474.0	540.0	606.0
Maximum Design Flow							
100	250.0	112.3	821	106.0	487.0	540.0	593.0

Notes
(1): A range of the steady state best-estimate flow rates are provided. During startup, the NPM is initially in single SG operation and transitions to two SG operation by 20 percent power. Based on these startup conditions, power, RCS (primary) flow rate, and RCS temperature vary.
(2): Core dT = T_{Hot} (measured at the riser outlet) - T_{Cold}

Figure 5.1-1: NuScale Power Module Major Components

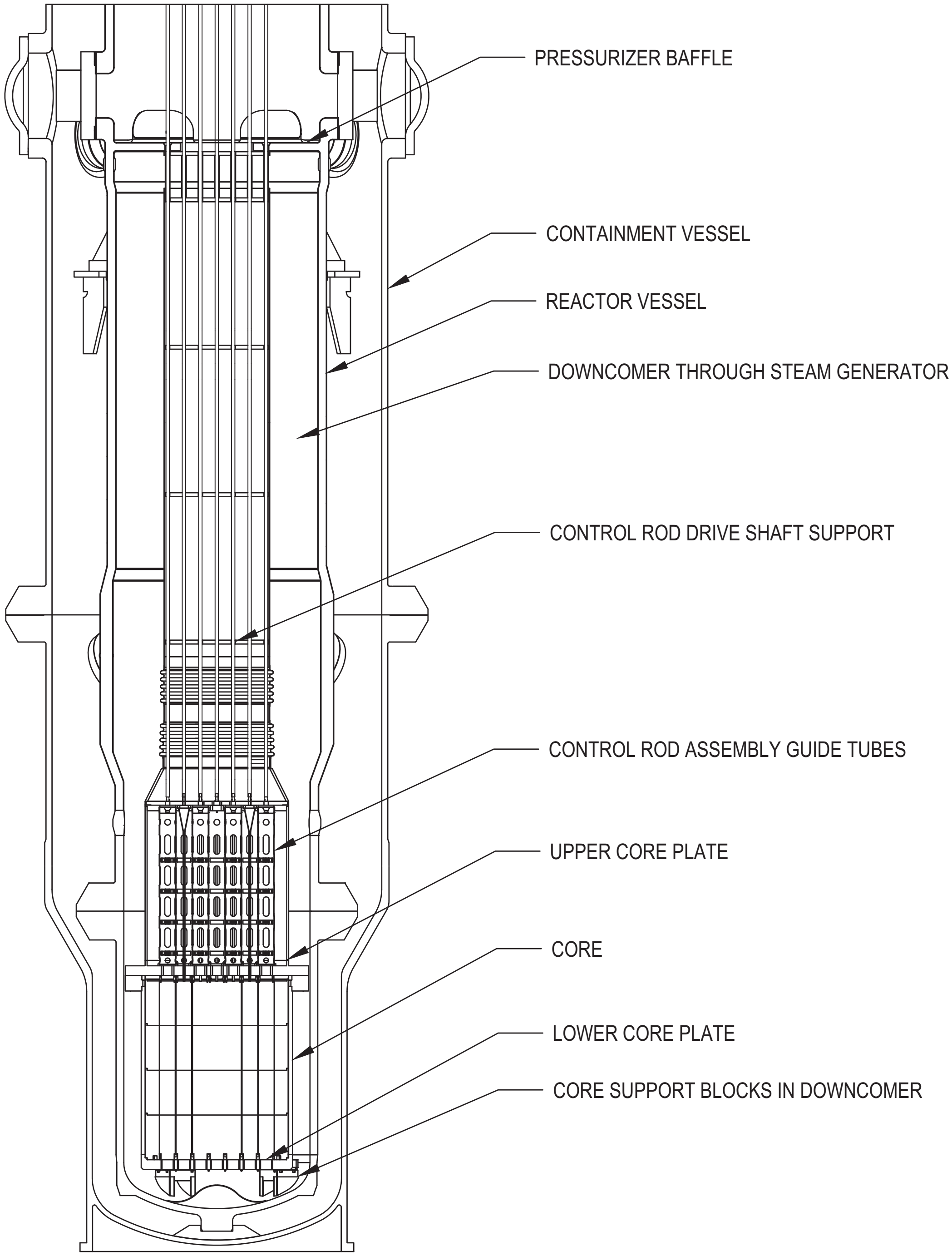


Figure 5.1-2: Reactor Coolant System Simplified Diagram

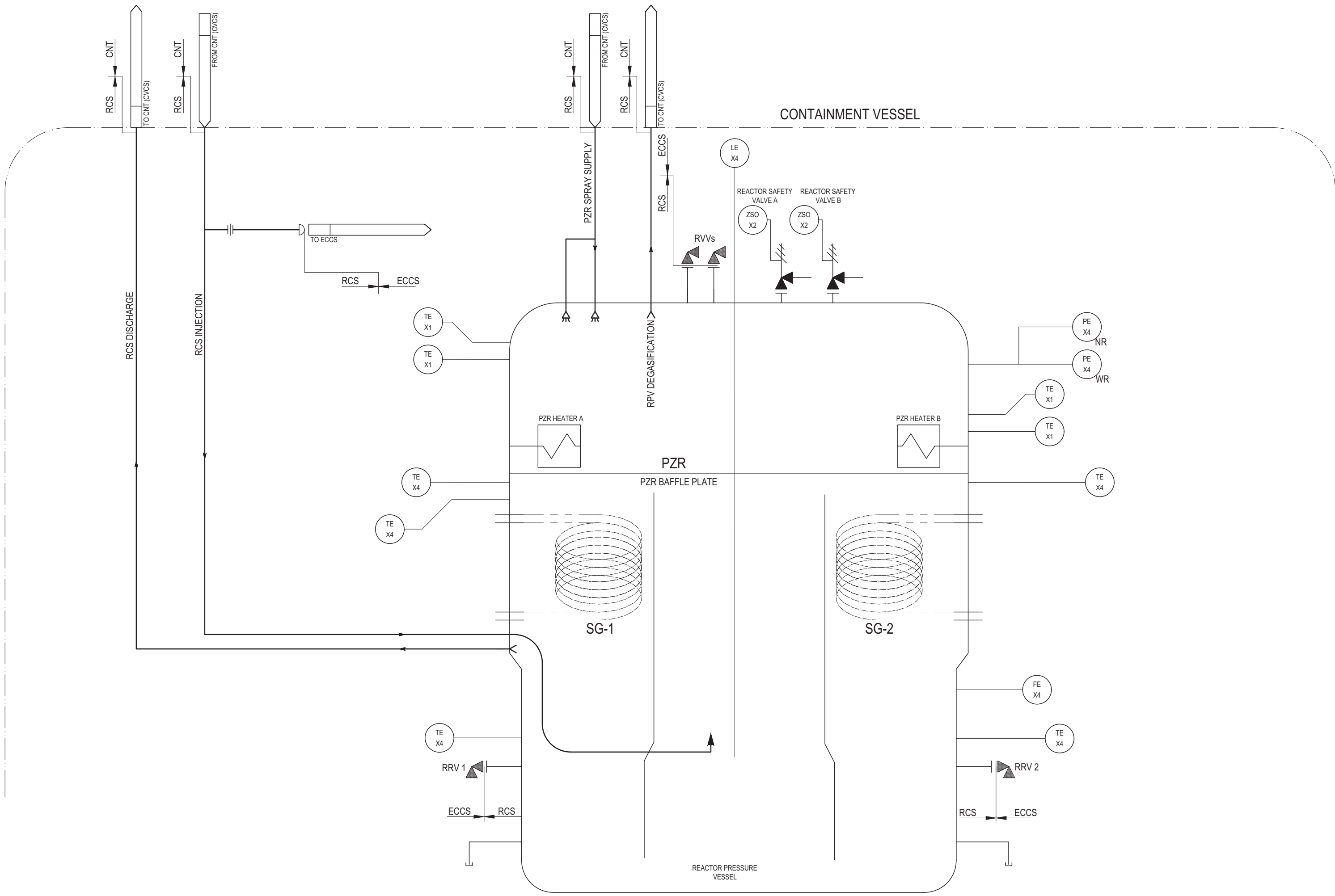
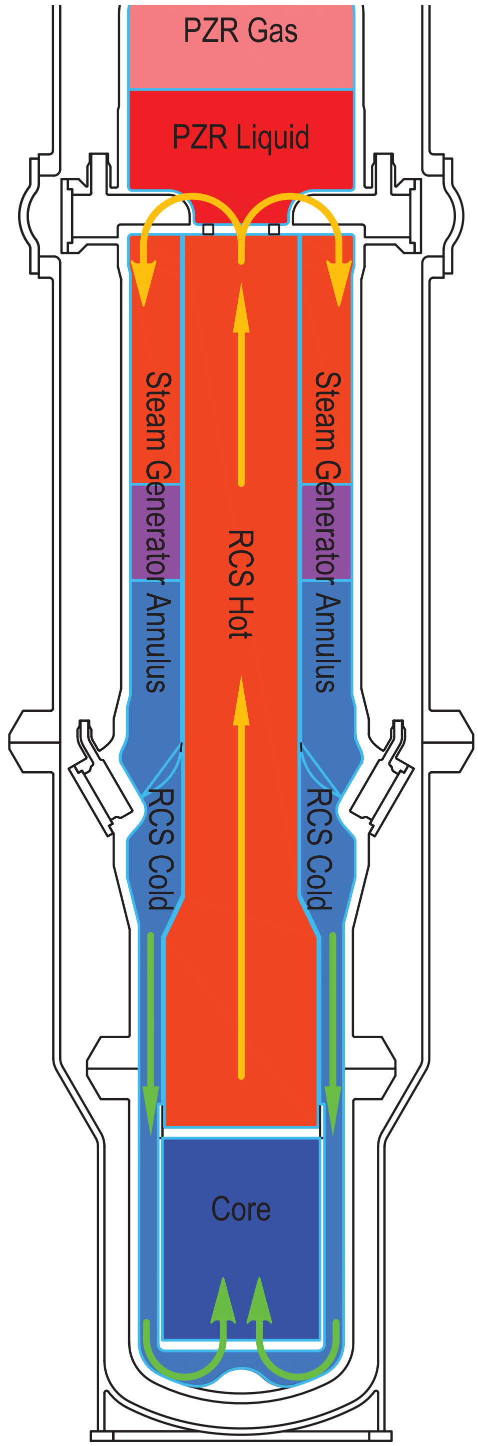


Figure 5.1-3: Reactor Coolant System Schematic Flow Diagram



5.2 Integrity of Reactor Coolant Boundary

The design features of each NuScale Power Module (NPM) maintain the integrity of the reactor coolant pressure boundary (RCPB) for the design life of the plant. The RCPB for the NPM meets the RCPB definition provided in 10 CFR 50.2 and includes the pressure retaining components that are part of the reactor coolant system (RCS) up to and including

- the outermost containment isolation valve (CIV) in system piping that penetrates the containment vessel (CNV).
- the reactor safety valves (RSVs).
- the emergency core cooling system (ECCS) reactor vent valves (RVVs) and reactor recirculation valves (RRVs).

The piping that is part of the NPM reactor coolant pressure boundary (RCPB) penetrates both the reactor pressure vessel (RPV) and the containment vessel (CNV) up to the outermost containment isolation valve (CIV). This piping does not terminate inside the CNV. The piping included in the RCPB is:

- RCS discharge line piping and CNTS nozzle and safe end (interior and exterior)
- RCS injection line piping and CNTS nozzle and safe end (interior and exterior)
- Pressurizer (PZR) spray line piping and CNTS nozzle and safe end (interior and exterior)
- RPV high point degasification line piping and CNTS nozzle and safe end (interior and exterior)

In addition to the RCPB piping, other portions of the RCPB include:

- RPV heads and shells
- Sensors (UT sensor, thermowells, and pressure sensors)
- Reactor Recirculation Valves
- SG tubes
- RCS discharge safe end
- RCS injection safe end (exterior)
- Integral main steam baffle plate in the main steam tube sheet region
- Integral steam plenum caps
- PZR heater bundle covers
- PZR spray safe ends
- RPV high point degasification safe end
- In-core instrumentation covers
- CRDM pressure housings
- Reactor safety valves
- Reactor vent valves

The RPV, described in Section 5.3, Reactor Vessel, is the primary component of the RCS and RCPB in the NPM. Section 3.9, Mechanical Systems and Components, describes the design transients, loading combinations, stress limits, and evaluation methods used in the design and fatigue analyses of RCPB components, and design information used to support the conclusion that the RCPB integrity is maintained. Design, construction, and maintenance, commensurate with quality standards, of the components of the RCPB ensure overpressure protection of the RCS.

The RCPB includes the RCS injection and discharge piping that interfaces with the CVCS up to the outermost CIV installed on the top of each NPM. A summary discussion of the containment system, including a discussion of the applicability of General Design Criteria (GDC) 55 and 57 to the RCPB, is in Section 6.2.4.

5.2.1 Compliance with Codes and Code Cases

5.2.1.1 Compliance with 10 CFR 50.55a

The NPM meets the relevant requirements of the following regulations.

- 10 CFR 50.55a. Design, fabrication, construction, testing, and inspection of the RPV and pressure retaining components associated with the RCPB are Class 1 in accordance with Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) (Reference 5.2-8) and meet the applicable conditions promulgated in 10 CFR 50.55a(b).
- 10 CFR 50, Appendix A, GDC 1 and GDC 30. Design, fabrication, and testing of the RPV and pressure retaining components associated with the RCPB are Class 1 and meet the highest quality standards in accordance with the Quality Assurance Program described in Chapter 17, Quality Assurance and Reliability Assurance.

The RCS injection and discharge piping that connects to the CVCS up to and including the associated isolation valves is Class 1 in accordance with the ASME BPVC Section III. The RCS piping interfacing with the CVCS from the outermost CIVs to the NPM flange connections is not part of the RCPB and is Class 3 in accordance with ASME BPVC Section III. Systems other than the CVCS that connect to the RCS require isolation and are not classified as part of the RCPB. A listing of the RCPB pressure retaining components and the quality group classification is in Table 3.2-2.

The ASME BPVC of record for the US460 standard design for the NPM is the ASME BPVC, 2017 Edition. The 2017 Edition of the ASME BPVC, as endorsed by the ASME and promulgated in the 2020 rulemaking proposing to amend 10 CFR 50.55a (85 FR 26576), meets the requirements of ASME BPVC editions specified in 10 CFR 50.55a(a)(1) and Regulatory Guides (RGs) 1.84 and 1.147.

The application of the ASME BPVC Section XI inservice inspection (ISI) requirements for Quality Group A systems and components (ASME BPVC Class 1) are in Section 5.2.4, RCPB Inservice Inspection and Testing. The RCPB does not include Quality Group B or C components. The ASME BPVC Section XI

ISI requirements for Quality Groups B and C systems and components (ASME BPVC Class 2 and 3) are in Section 6.6, ISI and Testing of Class 2 and 3 Systems and Components.

Operational and maintenance inservice testing codes, standards, and guides for the NPM design are in accordance with the ASME Operation and Maintenance (OM) Code OM-2017, "Operation and Maintenance of Nuclear Power Plants," as allowed by 10 CFR 50.55a(a)(1). ASME OM-2017, as endorsed by the ASME and promulgated in the 2020 rulemaking proposing to amend 10 CFR 50.55a (85 FR 26540), meets the requirements of ASME OM Code editions specified in 10 CFR 50.55a(a)(1) and RG 1.192. Use of an ASME OM Code edition or addenda dated later than ASME OM-2017 edition requires approval for incorporation by reference per 10 CFR 50.55a(a)(1) or authorization by the Nuclear Regulatory Commission (NRC) pursuant to 10 CFR 50.55a(a)(3) and subject to applicable provisions of 10 CFR 50.55a(b). An ASME OM Code edition or addenda not endorsed by the NRC may be used pursuant to the requirements of 10 CFR 50.55a(z).

Section 3.9, Mechanical Systems and Components, describes the Inservice Test (IST) Program and compliance with 10 CFR 50.55a(f)(3)(iii)(B) and 10 CFR 50.55a(f)(3)(iv)(B).

5.2.1.2 Compliance with Applicable Code Cases

ASME BPVC Section III code cases chosen for design, fabrication, and construction are from those listed in the applicable ASME BPVC Edition specified in 10 CFR 50.55a(a)(1)(i) or Tables 1 and 2 of RG 1.84 pursuant to 10 CFR 50.55a(a)(3)(i) and subject to the applicable provisions of 10 CFR 50.55a(b). Code cases used and listed in Table 2 of RG 1.84 also meet the conditions established in the RG.

Section 5.2.4, RCPB Inservice Inspection and Testing, and Section 6.6, Inservice Inspection and Testing of Class 2 and 3 Systems and Components, provide a summary discussion of preservice and ISI examinations and procedures. The ASME BPVC Section XI code cases used for preservice inspection and ISI listed in the applicable ASME BPVC Edition specified in 10 CFR 50.55a(a)(1)(ii) or Tables 1 and 2 of RG 1.147 pursuant to 10 CFR 50.55a(a)(3)(ii) and subject to the applicable provisions of 10 CFR 50.55a(b) are identified. Code cases used and listed in Table 2 of RG 1.147 also meet the conditions established in the RG.

The ASME OM code cases used for preservice testing and inservice testing from those listed in the applicable ASME OM Code Edition specified in 10 CFR 50.55a(a)(1)(iv) or Tables 1 and 2 of RG 1.192 pursuant to 10 CFR 50.55a(a)(3)(iii) and subject to the applicable provisions of 10 CFR 50.55a(b) are identified. Code cases used and listed in Table 2 of RG 1.192 also meet the conditions established in the RG.

Table 5.2-1, "ASME Code Cases," provides a list of the specific code cases used in the NPM design that are not addressed in ASME BPVC 2017 Edition. Conditionally acceptable ASME code cases listed in Table 5.2-1 are subject to the applicable conditions specified in Table 2 of RG 1.84 or Table 2 of RG 1.147. Other acceptable and conditionally acceptable ASME code cases listed in RGs 1.84, 1.147, and 1.192 in effect at the time of the application submittal and listed in RG revisions issued subsequent to the application submittal may be used for RCPB components. The ASME code cases listed in RG 1.193 are not used unless authorized by the NRC pursuant to the requirements of 10 CFR 50.55a(z).

5.2.2 Overpressure Protection

Each NPM has overpressure protection features to protect the RCPB, including the primary side of systems connected to the RCS and the secondary side of the SGs from overpressurization.

The RCPB integrated overpressure protection uses two ASME BPVC Section III safety valves during normal operations and anticipated operational occurrences (AOOs). The secondary side components with the same design pressure as the RCPB have integrated overpressure protection by a system design that does not exceed the ASME BPVC service limits during normal operations and AOOs. The low temperature overpressure protection (LTOP) system consists of the ECCS reactor vent valves and provides overpressure protection during low temperature conditions.

The RSVs are located above the PZR volume on the top of the RPV head to provide overpressure relief for the RCS. These valves relieve the RCS pressure directly into containment. Structural design and valve qualification information related to the RSVs is in Section 5.2.2.4.1, Reactor Safety Valves.

During NPM startup and shutdown conditions with the module at low temperature and with the RPV not vented, the ECCS reactor vent valves provide LTOP to prevent exceeding the pressure-temperature limits. Two RVVs connect to the RPV above the PZR volume and discharge steam directly into containment. Upon LTOP actuation by the module protection system logic, the ECCS reactor vent valves open to limit RCS pressure below the pressure-temperature limits.

Further description of the qualification, design, and operation of the ECCS reactor vent valves, including the actuators, is in Section 6.3, Emergency Core Cooling System.

5.2.2.1 Design Bases

Overpressure protection for the RCPB ensures that design pressure conditions are not exceeded during the normal range of operations, including AOOs, in accordance with the requirements of 10 CFR 50, Appendix A, GDC 15. The overpressure protection system has sufficient capacity to prevent the RCPB pressure from exceeding 110 percent of design pressure during normal operations and AOOs. The overpressure protection system performs its function assuming a single active failure and a concurrent loss of normal alternating current power. Application of GDC 15 to the overpressure protection system provides assurance that the RCPB has an extremely low probability of failure during transients.

The overpressure protection system for the secondary system ensures compliance with the ASME BPVC Section III, service limits during specified service conditions.

Overpressure protection provided by the RSVs is in accordance with the requirements of ASME BPVC Section III, Article NB-7000 for the RCPB and Subparagraph NC-7120(b) for the secondary system piping associated with the SGs and the decay heat removal system (DHRS) that extends from the RPV to the secondary main steam isolation valves (MSIVs) and the feedwater (FW) regulating valves.

The CVCS, which is normally connected to the RCS, isolates from the RCS following a containment isolation actuation as described in Section 6.2, Containment Systems and Section 7.1, Fundamental Design Principles. During an operational RCS pressure transient that does not result in isolation of the CVCS from the RCS, the RPV integral PZR, CVCS design, and relief valves provide CVCS overpressure protection, as described in Section 9.3.4, CVCS.

During low temperature conditions, overpressure protection for the RCPB is provided with sufficient margin to ensure the pressure boundary behaves in a non-brittle manner; the probability of a rapidly propagating fracture is minimized consistent with the requirements of 10 CFR 50, Appendix A, GDC 31.

The LTOP system design provides sufficient capacity to prevent RCPB pressure from exceeding the pressure-temperature limits, when below the LTOP system enable temperature such that RPV pressure is maintained below brittle fracture limits during operating, maintenance, testing, or postulated accident conditions.

During power operation, the PZR surge volume provides normal pressure control with sufficient capacity to preclude actuation of the RSVs during normal operational transients. During the following operating conditions, pressure protection is provided.

- The reactor is operating at the licensed core thermal power level.
- System and core parameters values are within normal operating range that produce the highest anticipated pressure.
- Components, instrumentation, and controls function normally.

The PZR heaters and spray control RCS pressure during normal power operations, but the NPM achieves safe shutdown conditions without reliance on pressure control by PZR heaters or PZR spray flow. Additionally, the thermal hydraulic analysis demonstrates that the PZR volume is adequate to accept the in-surge from a loss of load transient without liquid or two-phase flow reaching the RSVs. Further description of the PZR is in Section 5.4.5.

5.2.2.2 Design Evaluation

During power operations, the RSVs provide overpressure protection for the RPV; during operations at low-temperature conditions, the RVVs provide overpressure protection. The RSVs also provide external overpressure protection for the SG tubing, plena, and for piping external to the RPV that forms part of the RCPB (e.g., RCS injection, discharge, degasification, and PZR spray piping up to and including the outermost CIVs; ECCS valve pilot actuator lines; and several CNV nozzles and their safe ends).

A thermal relief valve provides overpressure protection for the control rod drive system cooling piping during a containment isolation event. Additional information regarding the control rod drive system is in Section 4.6, Functional Design of Control Rod Drive System.

Secondary systems with the same design pressure as the RCPB include the SG system, the DHRS, the MS and FW portions of the containment system, the portion of the condensate and feedwater system downstream of the feedwater regulating valves, and the portion of the MS system upstream of the secondary MSIVs. Description of the design and service limits for these systems is in Section 3.9, Mechanical Systems and Components. Under normal operating conditions and pressure transients, these systems do not exceed internal pressure limits because the design pressure is equivalent to the design pressure of the RCPB. Therefore, the overpressure protection for these secondary systems is provided by a system design that does not exceed the ASME BPVC service level limits during normal operation or during transients. This design is acceptable without the need for a pressure relieving device in accordance with ASME BPVC Section III, Subsubparagraph NB-7120(c).

Discussion of the overpressure protection for the portion of the MS system downstream of the secondary MSIVs is in Section 10.3, Main Steam System. In the event of an SG tube failure accident, the primary system RSVs provide overpressure protection for SG internal pressure.

During shutdown conditions thermal relief valves provide overpressure protection for the secondary side of the SGs, FW and MS piping in containment, and the DHRS when the secondary system is water solid during SG flush procedures. Additional information regarding the thermal relief valves is in Section 5.4.1, Steam Generators.

5.2.2.2.1 Overpressure Protection During Power Operations

Overpressure protection for the RPV is designed, fabricated, and constructed in accordance with the requirements of the ASME BPVC, Section III, Article NB-7000. 10 CFR 50, Appendix A, GDC 15, provides the main design function of the RSVs. The RSVs are part of the RCPB, bolted via flanges to the RPV head. The setpoint of each RSV actuates on a high RCS-to-containment pressure differential to allow flow directly to containment.

Table 3.2-2 lists the RSV design requirements.

The RSVs provide RCS overpressure protection during power operation or an AOO. The AOOs analyzed, which could lead to overpressure of the RCPB, include

- loss of load.
- turbine trip with bypass.
- turbine trip without bypass.
- loss of condenser vacuum.
- inadvertent MSIV closure.
- steam pressure regulator failure closed.
- loss of normal alternating current power.
- loss of FW.
- inadvertent operation of the DHRS.

A further description of these AOOs are in Chapter 15, Transient and Accident Analyses.

A turbine trip at full power without bypass capability is the most severe AOO and is the bounding event used in the determination of RSV capacity and the RPV overpressure analyses. Sizing of the RCS and the PZR steam space avoids an RSV lift during normal operational transients that produce the highest RPV pressure at full power conditions, with system and core parameters within normal operating range. In the event of a safety valve lift, the size of the PZR steam space is sufficient to preclude liquid discharge.

The analytical model used for the analysis of the overpressure protection system and the basis for its validity is in the NuScale Topical Reports "Non-Loss-of-Coolant Accident Analysis Methodology" (Reference 5.2-1) and "Loss-of-Coolant Accident Evaluation Model" (Reference 5.2-2).

5.2.2.2.2 Low Temperature Overpressure Protection System

The ECCS reactor vent valves, which are part of the RCPB, provide overpressure protection during low-temperature conditions in accordance with ASME BPVC Section III, Subsection NB.

An RCS overpressurization during low-temperature conditions could occur due to equipment malfunctions or operator error that results in excessive heat or inventory being added to the RCS, including inadvertent energization of the PZR heaters, inadvertent operation of the module heatup system, or excessive CVCS makeup. Isolation of the CVCS injection line on high PZR water level terminates increased RCS inventory events and inadvertent operation of the module heatup system, thereby precluding RCS inventory solid conditions from challenging the integrity of the RCPB at low-temperature conditions. The plant technical specifications provide operability and testing requirements associated with the automatic isolation of the RCS line piping on high PZR water level. The spurious actuation of the PZR heaters is the limiting RCS cold overpressurization event.

The RVVs have sufficient capacity to prevent RCPB pressure from exceeding the limiting pressure when below the LTOP enabling temperature such that an RPV is maintained below brittle fracture stress limits during operating, maintenance, testing or postulated accident conditions. The variable LTOP limit is based on the RCS cold temperature (i.e., temperature in the downcomer at the SG outlet). The selected LTOP pressure setpoint is a function of the cold temperature. Table 5.2-5 provides the LTOP pressure setpoint. Figure 5.2-3 provides a graph of the LTOP variable setpoint. The ECCS RVVs are part of the RCPB and are capable of opening during startup and shutdown discharging directly from the RCS to containment to provide the LTOP function.

Selection of the LTOP setpoint considers the worst case low temperature overpressure transient, which is the spurious actuation of the PZR heaters while below the LTOP enabling temperature. The LTOP analysis assumes a maximum PZR heater total power output of 800 kW, with additional heat input from core decay heat. An LTOP pressurization case demonstrates that the RVVs open before RCS pressure exceeds the low temperature pressure limit. This case assumes initial conditions that maximize the rate of PZR level increase as it approaches a water solid condition, thus maximizing the pressurization rate. The LTOP setpoint includes the following margin:

- pressure and temperature uncertainty
- the difference in elevation between the pressure sensing instrumentation and the bottom of the RPV
- the potential difference in temperature between downcomer regions
- the maximum delay in RVV opening
- delay in sensor response time
- module protection system processing time

As PZR level nears 100 percent, LTOP analysis shows pressure increasing and exceeding the LTOP pressure setpoint. During the valve opening delay, PZR pressure continues to increase, followed by opening of the RVV. The analysis results indicate the peak pressure remains below the brittle fracture stress limit.

COL Item 5.2-1: An applicant that references the NuScale Power Plant US460 standard design will provide a certified Overpressure Protection Report in compliance with American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section III, Subarticles NB-7200 and NC-7200 to demonstrate the reactor coolant pressure boundary and secondary system design contains adequate overpressure protection features, including low temperature overpressure protection features.

5.2.2.3 Piping and Instrumentation Diagram

Figure 5.1-2 provides the RCS piping and instrument diagram and illustrates the design configuration of the RSVs and RVVs, showing the number and location with respect to the RPV.

5.2.2.4 Equipment and Component Description

5.2.2.4.1 Reactor Safety Valves

The RSVs are safety-related components whose design requirements are specified in Table 3.2-2. Two RSVs are installed on the upper head of the RPV and provide overpressure protection to the RPV and RCS piping. A single RSV provides sufficient total relieving capacity for overpressure protection for the RPV. The function of the second RSV is to provide redundant relieving capacity and overpressure protection for the RPV. The set pressure of the second RSV is staggered to ensure that second RSV does not actuate in the event the first RSV lifts.

The RSV design information is in Table 5.2-2, and materials of the RSV components are in Table 6.1-3. The RSVs have a service life of 60 years. Each RSV is a spring operated safety relief valve designed in accordance with the requirements of ASME BPVC Section III, Article NB-7000. A simplified diagram of the RSV is in Figure 5.2-1. The RSVs are designed for 300 cycles over the service life. Environmental qualification information associated with the RSVs is in Section 3.11, Environmental Qualification of Mechanical and Electrical Equipment.

5.2.2.4.2 Reactor Vent Valves

The ECCS reactor vent valves are safety-related components, whose design requirements are specified in Table 3.2-2, and constructed in accordance with ASME BPVC Section III, Subsection NB, each designed with sufficient relief capacity to prevent brittle fracture stress limits from being exceeded on the RPV and pressure-retaining components associated with the RCPB when operating at low-temperatures conditions.

The trip and reset valves for the RVVs are solenoid pilot valves constructed in accordance with ASME BPVC Section III, Subsection NB. The pilot actuators are on the exterior of the CNV. Section 6.3, Emergency Core Cooling System, provides a detailed description of the RVVs and valve actuators.

Assuming a single active component failure, the RVVs, associated actuators, and controls maintain the LTOP function. The RVVs have sufficient pressure relief capacity to accommodate the most limiting single active failure assuming the most limiting allowable operating condition and system configuration.

Further description of the design and operation of the RVVs is covered in Section 6.3, Emergency Core Cooling System. Environmental qualification information associated with the RVVs is in Section 3.11, Environmental Qualification of Mechanical and Electrical Equipment.

5.2.2.5 Mounting of Pressure-Relief Devices

The RSVs attach via bolted flanges on the RPV head to allow for periodic removal for inspection and testing. A manway in the containment shell provides access to the RSVs.

The ECCS reactor vent valves attach via bolted flanges to the RPV. Further description of the design of the RVVs is in Section 6.3, Emergency Core Cooling System.

5.2.2.6 Applicable Codes and Classification

The RSVs and RVVs satisfy the overpressure protection criteria described in ASME BPVC Section III, Article NB-7000, and are designed in accordance with ASME BPVC Section III, Subarticle NB-3500. The applicable design code edition is in Section 5.2.1, Compliance with Codes and Code Cases, and Section 3.2, Classification of Structures, Systems, and Components, describes the classifications applicable to overpressurization equipment and components.

5.2.2.7 Material Specifications

Material specifications for the RSVs and the RVVs are in Section 6.1, Engineered Safety Feature Materials.

5.2.2.8 Process Instrumentation

The control room includes direct position indication for each RSV and RVV, pursuant to the requirements of 10 CFR 50.34(f)(2)(xi) promulgating Three Mile Island action plan recommendation Item II.D.3 of NUREG-0737. Due to the design of the NPM, classification of RCS leakage, including leakage from these valves, into the containment atmosphere is considered unidentified leakage.

Detection of leakage from the RPV to the CNV is in Section 9.3.6, Containment Evacuation System.

5.2.2.9 System Reliability

The RSVs and RVVs design, testing, and inspection standards conform to ASME BPVC Sections III and XI criteria. ASME BPVC safety and relief valves demonstrate a high degree of reliability over their many years of service in the nuclear industry. Functional qualification of the RSVs and RVVs is in accordance with ASME QME-1 as endorsed by RG 1.100. The inservice testing and inspection of the safety and vent valves provides reasonable assurance of continued reliability and conformance.

The RSVs are self-actuating devices that do not rely on external power or controls. The spring operated design is in accordance with ASME BPVC Section

III, Article NB-7000 requirements. An RSV actuates (opens) when the set pressure (Table 5.2-2) is exceeded by the pressure in the PZR region of the RPV. The RSVs have staggered set pressures. The first (lowest setpoint) RSV provides ASME overprotection to the RPV. The second RSV provides overpressure protection for the RPV in the event the first RSV fails to open. The set point of the second RSV is set higher to ensure that second RSV does not actuate should the first RSV lift.

The reliability of the RVVs is in Section 6.3, Emergency Core Cooling System.

5.2.2.10 Testing and Inspection

Testing and inspection of overpressure protection equipment is in accordance with accepted industry standards including Sections III and XI of the ASME BPVC, Mandatory Appendix I of the ASME OM Code, and the requirements of 10 CFR 50.34(f)(2)(x) promulgating Three Mile Island action plan recommendation Item II.D.1.

Technical specifications address overpressure protection surveillance testing requirements for normal and low temperature operating conditions.

The IST Program includes the RSVs. A position verification test is performed for each valve every 24 months during refueling conditions in accordance with ASME OM-2017, Division 1, Subarticle ISTC-3700. An exercise test is performed every five years on a staggered basis in accordance with ASME OM-2017, Division 1, Mandatory Appendix I, Subsubarticle I-3320.

Section 6.6, ISI and Testing of Class 2 and 3 Systems and Components; Section 14.2, Initial Plant Test Program; and Section 3.9, Mechanical Systems and Components, provide additional information on testing and inspection of the overpressure protection components.

5.2.3 Reactor Coolant Pressure Boundary Materials

The RCPB materials, including weld materials, conform to fabrication, construction, and testing requirements of ASME BPVC Section III, Subsection NB, requirements and the materials selected for fabrication of the RCPB comply with the requirements of ASME BPVC Section II. Details of the materials conformance for the RPV are in Section 5.3, Reactor Vessel.

The RCPB materials comply with the relevant requirements of the following regulations:

- 10 CFR 50, Appendix A.
  - GDC 1 and 30. The RPV materials and RCPB component materials are designed, fabricated and tested to Class 1 requirements; the highest quality standards in accordance with the Quality Assurance Program described in Chapter 17, Quality Assurance and Reliability Assurance.

– GDC 4. Design and fabrication of the RPV and pressure retaining components associated with the RCPB ensure compatibility with the environmental conditions of the reactor coolant and containment atmosphere.

– GDC 14 and 31. Design and fabrication of the RPV and pressure retaining components associated with the RCPB ensure sufficient margin such that the RCPB behaves in a non-brittle manner and minimizes the probability of rapidly propagating fracture and gross rupture of the RCPB (Reference 5.2-10).

- Criterion XIII of 10 CFR 50, Appendix B. The Quality Assurance Program requires procedures for the control of the on-site cleaning of RPV and the RCPB during construction.
- Appendix G to 10 CFR 50. The RPV ferritic pressure retaining and integrally attached materials meet applicable fracture toughness acceptance criteria (Reference 5.2-10). The design supports an exemption from the requirements of 10 CFR 50.60 which invokes compliance with 10 CFR 50, Appendix G. Section 5.3.1.6 provides further details.

5.2.3.1 Material Specifications

The materials for the Class 1 components and supports that comprise the RCPB, including the RPV and SGs, are in Table 5.2-3. Table 5.2-3 also includes materials and specifications associated with the RPV attachments and appurtenances. The table lists the grade or type, as applicable, of the ferritic low alloy steels, austenitic stainless steels, and nickel-based alloys specified for the RCPB. Except where noted in Table 5.2-3, the associated ASME BPVC material specification provides the final metallurgical condition. Further discussion of the materials associated with the RPV is in Section 5.3, Reactor Vessel.

The RCPB surface materials in contact with reactor coolant or in contact with pool water during refueling, including welds, are corrosion resistant alloys or clad with austenitic stainless steel or nickel-based alloy. The SG tubesheet bores are the exception, the SA-508 tubesheet bores do not have corrosion resistant cladded surface. The SG tubes expand into the tubesheet bore to provide corrosion protection in the crevice between the SG tube and tubesheet.

Processing and welding of unstablized American Iron and Steel Institute Type 3XX series austenitic stainless steels for pressure retaining components comply with RG 1.44 to prevent sensitization caused by chromium depletion at the grain boundaries during welding and heat treatment operations. For unstablized American Iron and Steel Institute Type 3XX series austenitic stainless steel subjected to sensitizing temperatures subsequent to solution heat treatment, the carbon content is no more than 0.03 weight percent.

Processing and welding of American Iron and Steel Institute Type 2XX series austenitic stainless steels for pressure retaining components comply with ASME BPVC paragraph NB-2433 and RG 1.31 for delta ferrite composition. The ferrite number are in the range of 5 FN to 16 FN. The carbon content of the weld filler materials is restricted to 0.04 percent maximum.

Nickel-based Alloy 690 is a base metal in RCPB components and structures along with Alloy 52/152 cladding and weld metals and similar alloys developed for improved weldability. Alloy 690 and 52/152 have a high resistance to general corrosion, high resistance to fast fracture, and superior tensile properties at elevated temperature. Steam generator tubes use Alloy 690 in the thermally treated condition. The RCPB design does not use Alloy 600 base metal or Alloy 82/182 cladding or weld metal.

5.2.3.2 Compatibility with Reactor Coolant

5.2.3.2.1 Reactor Coolant Chemistry

The RCS water chemistry is controlled to minimize corrosion of RCS surfaces and minimizes corrosion product transport during normal operation. These controls ensure the integrity of RCPB materials, the integrity of the fuel cladding, fuel performance by limiting cladding corrosion, and the minimization of radiation fields. Accordingly, the plant maintains alkaline-reducing water chemistry during power operation. Routine sampling and analysis of the coolant verifies its chemical composition.

The CVCS provides the means for chemical addition to the primary coolant via the RCS injection and spray lines and provides the means for removal of chemicals, suspended solids, and impurities by the CVCS purification systems via the RCS discharge line. Diluting the primary coolant with purified RCS injection flow reduces chemical concentrations and impurities.

For reactivity control, boric acid addition acts as a soluble neutron poison and is adjusted as needed for reactivity control to compensate for changes in fuel reactivity over each fuel cycle.

Makeup flow in the CVCS adds lithium hydroxide enriched with lithium-7 isotope to the reactor coolant to increase pH as required. The CVCS delithiating ion exchanger removes lithium from the RCS to maintain the pH level within the required range. Lithium hydroxide is compatible with boric acid, stainless steel, zirconium alloys, and nickel-base alloys. In accordance with the recommendations of the fuel vendor and the Electric Power Research Institute (EPRI) Pressurized Water Reactor Primary Water Chemistry Guidelines (Reference 5.2-3), the plant specific pH program maintains limits on primary coolant pH and lithium concentration.

Dissolved hydrogen added during operation maintains a reducing environment in the reactor coolant. Hydrogen use is compatible with the aqueous environment and is able to suppress oxygen generated by the radiolysis of water and oxygen introduced into the RCS with makeup water. Direct injection of high pressure gaseous hydrogen into the CVCS injection flow adds dissolved hydrogen to the reactor coolant. Added hydrazine scavenges dissolved oxygen at low temperature during startup.

Control of the quality of the chemicals and the makeup water added to the reactor coolant limits potential contamination. Reactor coolant chemistry parameters and impurity limitations monitored during power operations conform to the limits specified in the EPRI pressurized water reactor Primary Water Chemistry Guidelines, fuel vendor primary chemistry guidelines, and RG 1.44 limits as provided in Table 5.2-4. Zinc addition to the primary system reduces radiation levels in plant maintenance areas and reduces primary water stress-corrosion cracking (PWSCC) initiation rates.

Industry guidelines as described in EPRI Technical Report 3002000505, Pressurized Water Reactor Primary Water Chemistry Guidelines (Reference 5.2-3) inform the water chemistry program. The program includes periodic monitoring and control of chemical additives and reactor coolant impurities listed in Table 5.2-4. Detailed procedures implement the program requirements for sampling and analysis frequencies and corrective actions for control of reactor water chemistry.

The frequency of sampling water chemistry varies (e.g., continuous, daily, weekly, or as needed) based on plant operating conditions and the EPRI water chemistry guidelines. Remedial corrective actions are taken in response to adverse trends before a control parameter exceeds its normal range. When measured water chemistry parameters exceed the specified range, corrective actions bring the parameter back within the acceptable range and within the time period specified in the EPRI water chemistry guidelines. The actions are performed within specified time periods, based on the severity of the chemical condition. Chemistry procedures provide guidance for the sampling and monitoring of primary coolant properties.

Refueling operations require isolation, disconnection from the attached systems, and transportation of the NPM out of the bay for disassembly and refueling. The pool cooling and cleanup system purifies the pool water to ensure impurity levels in the pool water meet the impurity levels (i.e., chloride, fluoride, and sulfate) specified for RCS cold shutdown in the EPRI Pressurized Water Reactor Primary Water Chemistry Guidelines (Reference 5.2-3).

COL Item 5.2-2: An applicant that references the NuScale Power Plant US460 standard design will develop and implement a Strategic Water Chemistry Plan. The Strategic Water Chemistry Plan will provide the optimization strategy for maintaining primary coolant chemistry and provide the basis for requirements for sampling and analysis frequencies, and corrective actions for control of primary water chemistry consistent with the latest version of the Electric Power Research Institute Pressurized Water Reactor Primary Water Chemistry Guidelines.

COL Item 5.2-3: An applicant that references the NuScale Power Plant US460 standard design will develop and implement a Boric Acid Control Program that includes: inspection elements to ensure the integrity of the reactor coolant pressure boundary components for subsequent service, monitoring of the containment atmosphere for evidence of reactor coolant system leakage, the type of visual or other nondestructive inspections to be performed, and the required inspection frequency.

5.2.3.2.2 Compatibility of Construction Materials with Reactor Coolant

The RCPB ferritic low alloy steels used in pressure retaining applications have austenitic stainless steel cladding or Ni-Cr-Fe cladding on surfaces that are exposed to the reactor coolant. Low alloy steel forgings have an average grain size of Number 5 or finer in accordance with American Society for Testing and Materials standards. The cladding of ferritic type base material receives a post-weld heat treatment as required by ASME BPVC Section III, Subsubarticle NB-4622.

The inside and outside surfaces of carbon and low-alloy steels have austenitic stainless steel cladding, except for surfaces cladded with Ni-Cr-Fe, surfaces covered with stainless steel sleeves or inserts, or the inside surfaces of SG tubesheet bores. The final thickness of corrosion-resistant weld overlay is 0.125 inch minimum on both the inside and outside surfaces except for sealing surfaces or surfaces requiring additional weld-buildup. The Ni-Cr-Fe cladding is deposited with Alloy 52/152. Weld overlay cladding utilizes procedures qualified in accordance with the applicable requirements of ASME BPVC Section III, Subarticle NB-4300 and Section IX.

The integral steam plenum baffle plate contains holes to provide a path for the incore instrument and riser level sensor guide tubes, the control rod drive shafts, and reactor coolant to pass through the plate. Protection is provided to ensure that reactor coolant does not come in contact with the low-alloy steel at these holes.

The use of cobalt based alloys is minimized and limits are established to minimize cobalt intrusion into the reactor coolant. Hard surfacing and wear resistant parts in the CRDMs use cobalt based alloys. Section 4.5, Reactor Materials, contains additional details regarding the materials of the CRDMs. Low cobalt or cobalt-free alloys may be used for hardfacing and wear resistant parts in contact with the reactor coolant if their wear and corrosion resistance are qualified to meet design requirements.

5.2.3.3 Fabrication and Processing of Ferritic Materials

5.2.3.3.1 Fracture Toughness

The fracture toughness properties of the ferritic RCPB components comply with the requirements of 10 CFR 50, Appendix G, "Fracture toughness requirements," and ASME BPVC Section III, Subarticle NB-2300. Discussion of the fracture toughness requirements of the RPV materials is in Section 5.3, Reactor Vessel.

5.2.3.3.2 Welding Control - Ferritic Materials

Procedures qualified in accordance with the applicable requirements of ASME BPVC Section III, Subarticle NB-4300, and Section IX are used to conduct welding of ferritic materials used for components of the RCPB.

Stainless steel corrosion resistant weld overlay cladding of low alloy steel components conforms to the requirements of RG 1.43. Controls to limit underclad cracking of susceptible materials also conform to the requirements of RG 1.43.

Before cladding, the surfaces to be clad undergo examination using magnetic particle or liquid penetrant tests in accordance with ASME BPVC Section III, Article NB-2000.

Electroslag welding is not used, except for austenitic stainless steel cladding of low alloy steel.

Controls for preheating and interpass temperatures to support welding of carbon and low alloy steel in the RCPB, including preheat for weld deposited cladding, conform to the requirements of ASME BPVC Section III, Division 1, Non-mandatory Appendix D and are specified in the welding procedure specification as required by ASME BPVC Section IX, Article V. Control of the preheat temperature for low alloy steel forgings is in accordance with the requirements of RG 1.50.

Procedure qualification records and welding procedure specifications used to support welding of low alloy steel welds in the RCPB follow ASME BPVC Section III, Subarticle NB-4300 and Section IX. Welder and welding operator qualifications are in accordance with ASME BPVC Section III, Subarticle NB-4300 and ASME Section IX. Controls imposed on welding ferritic steels under conditions of limited accessibility are in accordance with the recommendations RG 1.71.

Post weld heat treatment temperature of the RPV low alloy steel material is 1125 degrees F ± 25 degrees F. Alternative post weld heat treatment time and temperatures specified in Subsubparagraph NB-4622.4(c) of ASME BPVC Section III, Subsection NB, are not used.

5.2.3.3.3 Nondestructive Examination of Ferritic Steel Tubular Products

The RCPB components do not contain ferritic steel tubular products. Nondestructive examination requirements associated with austenitic stainless steel tubular products are in Section 5.2.3.4.5, Nondestructive Examination for Austenitic Stainless Steel Tubular Products.

5.2.3.4 Fabrication and Processing of Austenitic Stainless Steels

5.2.3.4.1 Prevention of Sensitization and Intergranular Corrosion of Austenitic Stainless Steel

Intergranular corrosion is a problem for sensitized austenitic stainless steels in aggressive environments. Grain boundary carbide sensitization occurs when metal carbides precipitate on the grain boundaries when the material is heated in the temperature range of 800 degrees F to 1500 degrees F.

Compliance with RG 1.44 avoids sensitization and intergranular attack in unstablized Type 3XX austenitic stainless steels.

Austenitic stainless steel weld materials for RCPB are analyzed for delta ferrite content and limited to 5 FN to 16 FN that exceeds RG 1.31 and ASME BPVC Section III, Paragraph NB-2433 requirements.

The control of oxygen, chlorides, and fluorides in the reactor coolant during normal operation further minimizes the probability of stress corrosion cracking of unstabilized austenitic stainless steels. Description of the maintenance of the primary water chemistry is in Section 5.2.3.2.1, Reactor Coolant Chemistry. Additional information regarding the CVCS and the process for controlling RCS water chemistry is in Section 9.3.4, CVCS.

The use of hydrogen in the reactor coolant inhibits the presence of oxygen during operation. Gaseous argon may also be added to reactor coolant, if required, to support primary to secondary leakage controls. The effectiveness of these controls has been demonstrated by test and operating experience.

Precautions prevent the intrusion of contaminants into the system during fabrication, shipping, and storage.

Fabricators of RCPB components avoid, to the extent practicable, use of cold worked austenitic stainless steel. Fabricators of RCPB components do not use cold worked austenitic stainless steel with a material yield strength greater than 90,000 psi, as determined by the 0.2 percent offset method.

5.2.3.4.2 Cleaning and Contamination Protection Procedures

Cleaning of RCPB components complies with ASME NQA-1 requirements (Reference 5.2-4). The final cleanliness of the RCPB internal surfaces meets the requirements for "Class B" of Subpart 2.1. The final cleanliness of the RCPB external surfaces meets the requirements for “Class C” of Subpart 2.1, except for CRDM pressure housings. The CRDM pressure housings external surfaces meet the requirements for “Class B” of NQA-1 Subpart 2.1.

Handling, storage, and shipping of RCPB components comply with ASME Subparagraph NCA-4134.13 and meet the requirements for “Class C” items in accordance with ASME NQA-1, Subpart 2.2 (Reference 5.2-4).

Handling, protection, storage, and cleaning of austenitic stainless steel materials used in the fabrication, installation, and testing of nuclear steam supply components and systems comply with recognized and accepted methods designed to minimize contamination that could lead to stress corrosion cracking.

Procedures provide cleanliness controls during the various phases of manufacture and installation, including final flushing. The suppliers implement a written cleanliness control plan before and during manufacturing and assembly of components, which continues until components are sealed for shipment. The cleanliness control plan includes specific provisions for

- maintenance of cleanliness.
- controls to prevent foreign material from being introduced into the hardware.
- water purity control.
- controls to prevent detrimental material from contacting hardware.
- support system cleanliness and inspection.
- use of temporary plugs or seals to prevent entry of foreign material and objects and, as practical, prevent mechanical damage.
- use of stickers or other devices identifying cleanliness control requirements, affixed to temporary plugs and seals in such a manner that removal of the plug or seal cannot be accomplished without breaking the sticker.
- detection and removal of foreign objects.
- maintenance of cleanliness immediately before and during welding, brazing, and heat treating.
- tools and loose parts accountability.
- minimum exposure of hardware internal surfaces to shop atmosphere.
- periodic inspection of water transfer hoses.
- cleaning of surfaces immediately before assembly operations where surfaces that contact the fluid systems subsequently become inaccessible for inspection.

Controls minimize the introduction of potentially harmful contaminants including chlorides, fluorides, and low melting point alloys on the surface of austenitic stainless steel components. Removal of cleaning solutions, processing equipment, degreasing agents, and other foreign materials at any stage of processing before elevated temperature treatments is performed in accordance with RG 1.44. Acid pickling is avoided on stainless steel.

Minimal abrasive work avoids surface coldwork and contamination. Workers cannot use tools for abrasive work such as grinding, polishing, or wire brushing, that may be contaminated by previous usage on carbon or low alloy steels, or other non-corrosive resistant materials that could contribute to intergranular cracking or stress-corrosion cracking.

5.2.3.4.3 Compatibility of Construction Materials with External Reactor Coolant

The external surfaces of the upper RPV have austenitic stainless steel cladding. External surfaces of the RCPB have no exposed ferritic materials, maintain compatibility with a borated water environment, and are resistant to general corrosion.

5.2.3.4.4 Control of Welding - Austenitic Stainless Steel

Welding utilizes procedures qualified according to the rules of ASME BPVC Section III, Subarticle NB-4300, and ASME BPVC Section IX. Control of welding variables, as well as examination and testing during procedure qualification and production welding, is in accordance with ASME BPVC requirements.

Qualification of welders and welding operators is in accordance with ASME BPVC Section IX and RG 1.71.

5.2.3.4.5 Nondestructive Examination for Austenitic Stainless Steel Tubular Products

Preservice nondestructive examinations performed on Class 1 austenitic stainless steel tubular products to detect unacceptable defects comply with ASME BPVC Section III, Subsubarticle NB-5280, and ASME BPVC Section XI examination requirements. For Class 1 piping welds requiring an ultrasonic preservice examination, the welds meet the surface finish and marking requirements of ASME BPVC Section III, Subparagraph NB-4424.2.

5.2.3.5 Prevention of Primary Water Stress-Corrosion Cracking for Nickel-Based Alloys

Primary water stress-corrosion cracking is avoided in nickel-based alloy components in the RCS by

- using Alloy 690/152/52 in nickel-based alloy applications.
- controlling chemistry, mechanical properties, and thermo-mechanical processing requirements to produce an optimum microstructure for resistance to intergranular corrosion for nickel-based alloy base metal.
- limiting the sulfur content of nickel-based alloy base metal in contact with RCS primary fluid to maximum 0.02 weight percent.

The nickel-based alloy materials used in the RCPB, including weld materials, conform to the fabrication, construction, and testing requirements of ASME BPVC Section III. Material specifications comply with ASME BPVC Section II Parts B and C. Welding of nickel-base alloys in the RCPB complies with procedures qualified to the requirements of ASME BPVC Section III, Subarticle NB-4300 and ASME BPVC Section IX. Control of welding variables, as well as examination and testing during procedure qualification and production welding, conforms with ASME BPVC requirements. Qualification of welders and welding operators is in accordance with ASME BPVC Section IX and RG 1.71.

Chemistry, mechanical properties, and thermo mechanical processing requirements are controlled in nickel-based alloy base metal through use of solution annealing and thermal treatment to produce an optimum microstructure for resistance to intergranular corrosion.

Electric Power Research Institute Materials Reliability Program Reports MRP-111 (Reference 5.2-5) and MRP-258 (Reference 5.2-6) detail the Alloy 690, 52/52M, and 152 resistance to PWSCC. These documents conclude that Alloy 690 and its weld metals are highly corrosion resistant materials deemed acceptable for pressurized water reactor applications. There have been no signs of PWSCC in Alloy 690 materials in operating PWRs, and a wide variety of laboratory tests show that Alloy 690 resists PWSCC initiation.

The EPRI reports provide a comprehensive summary of Alloy 690 stress corrosion cracking laboratory test data from simulated primary water environments that provides reasonable assurance of the high resistance to PWSCC for Alloy 690 and its weld metals.

5.2.3.6 Threaded Fasteners

Threaded fasteners used in the RPV main closure flange, PZR heater bundle closures, RCS piping flanges, RVV flanges, RRV flanges, and RSV flanges are nickel-based Alloy 718. Threaded fastener materials conform to the applicable requirements of ASME BPVC Sections II and III, and are selected for their compatibility with the borated water environment in the RCS and reactor pool water.

Section 3.13, Threaded Fasteners, provides further description of the design of threaded fasteners for the RPV and pressure retaining components including design requirements for the use of Alloy 718 for the mitigation of SCC.

5.2.4 Reactor Coolant Pressure Boundary Inservice Inspection and Testing

Preservice inspection, ISI, and inservice testing of ASME BPVC Class 1 pressure-retaining components (including vessels, valves, bolting, and supports) within the RCPB are in accordance with ASME BPVC Section XI (Reference 5.2-9) pursuant to 10 CFR 50.55a(g), including ASME BPVC Section XI mandatory appendices.

The initial ISI Program incorporates the latest edition and addenda of the ASME BPVC approved in 10 CFR 50.55a(a) before initial fuel load, as specified in 10 CFR 50.55a, subject to the conditions listed in 10 CFR 50.55a(b). Inservice examination of components and system pressure tests conducted during successive 120-month inspection intervals must comply with the requirements of the latest edition and addenda of the ASME BPVC incorporated by reference in 10 CFR 50.55a(a), subject to the conditions listed in 10 CFR 50.55a(b).

The specific edition and addenda of the ASME BPVC used to determine the requirements for the inspection and testing plan for the initial and subsequent inspection intervals is provided in the inservice inspection program. The ASME BPVC includes requirements for system pressure tests and functional tests for active components. The requirements for system pressure tests are in Reference 5.2-9, Articles IWA-5000 and IWB-5000. These tests verify the pressure boundary integrity in conjunction with ISI. Section 6.6 discusses Class 2 and 3 component examinations.

5.2.4.1 Inservice Inspection and Testing Program

This section describes the process for inspection and testing of the ASME BPVC Class 1 components except for the SG tubes. Section 5.4.1, Steam Generators, describes the process for ISI requirements for the SG tubes.

The ISI and IST programs are composed of the following:

- the component inspection program, which includes non-destructive examination inspection of major components, piping system and support systems
- the valve IST program, which monitors and detects degradation of selected valves
- the hydrostatic testing program

The RCPB is accessible and permits periodic inspection and testing of important areas and features to assess their structural and leak-tight integrity pursuant to GDC 32. The design allows inspection, testing, and maintenance of the components located inside the RCPB of the NPM. Equipment that requires inspection or repair is in an accessible position to minimize time and radiation exposure during refueling and maintenance outages. Plant technicians access components without being placed at risk for excessive dose or situations where excessive plates, shields, covers, or piping must be moved or removed in order to access components.

The inspection requirements and conditions of 10 CFR 50.55a, as detailed in Section XI of the ASME BPVC, apply to Class 1 pressure-containing components and their supports. The RCPB components subject to inspection as Class 1 components are Quality Group A and comply with the ASME BPVC as described in Section 5.2.1, Compliance with Codes and Code Cases. Figure 6.6-1 shows the ASME BPVC Section III, Class 1 boundary for the RCS piping and SG system. Additionally, the ECCS valve actuators and actuator lines form a portion of the ASME BPVC Section III Class 1 boundary and are subject to ASME Section XI testing.

The inspection and testing program addresses the unique inspection and testing requirements for the NPM to ensure plant safety is maintained for the operating life.

The NPM inspection, testing, and maintenance strategy is (1) design the NPM components to anticipate required inspections, and (2) develop an ISI program to identify aspects such as interval and inspection frequencies, selection of components and welds for inspection, and expansion criteria.

Development of the inspection program consists of the following:

- identification of the appropriate ISI or IST requirements for the design (code version, overall inspections and tests required)

- identification of the structures, systems, and components (SSC), the subset inspections or test elements associated with SSC and those SSC that are subject to inspection and testing
- identification of appropriate ISI and IST requirements for each structure, system, and component
- assessment of each inspection and test element
- development of a comprehensive ISI and IST plan

The ISI schedule and requirements for Class 1 systems and components are in accordance with ASME BPVC Section XI.

The examination program for the ten-year inspection interval is defined in the ISI plan. The ISI plan for Class 1 systems and components is developed in accordance with Reference 5.2-9, Articles IWA-2400 and IWB-2400.

Examinations include liquid penetrant or magnetic particle techniques when performing surface examination; ultrasonic, radiographic, or eddy current techniques when performing volumetric examination; and visual inspection techniques when determining the surface condition of components and evidence of leakage for applicable components. Specific techniques, procedures and equipment, including any special techniques and equipment, are in accordance with the requirements of ASME BPVC Section XI and conform to the ISI program. Equivalent equipment and techniques support preservice inspection and subsequent ISI.

The visual, surface, and volumetric examination techniques and procedures conform with the requirements of articles IWA-2200, and applicable portions of Table IWB-2500-1 of Reference 5.2-9. The methods, procedures, and requirements for qualification of personnel performing ultrasonic examination conform to the requirements of Reference 5.2-9, IWA-2300. Qualification of personnel performing visual, liquid penetrant, magnetic particle, eddy current, or radiographic examinations as a part of the preservice inspection or ISI program are in accordance with the requirements of IWA-2300 of Reference 5.2-9.

The examination categories and requirements appropriate for each examination area follow the categories and requirements specified in applicable portions of Table IWB-2500-1 of Reference 5.2-9. The preservice inspection program includes the examination categories in accordance with Reference 5.2-9, IWB-2200.

Baseline examinations, collected in accordance with the related procedures, result in data contributing to a report with tabulated results. The report describes the scope of the inspection, the procedures utilized, the equipment utilized, names and qualifications of personnel, and the examination results including instrument calibration criteria in sufficient detail to provide reasonable assurance of repeatability for each examination.

Evaluation of examination results for Class 1 components is in accordance with IWA-3000 and IWB-3000 of Reference 5.2-9. Repair of unacceptable indications conforms to the requirements of IWA-4000 of Reference 5.2-9. Criteria for establishing need for repair or replacement are in accordance with IWB-3000 of Reference 5.2-9.

System leakage tests, followed by a VT-2 examination for the RPV Class 1 pressure retaining boundary, conform to the requirements specified in Reference 5.2-9, Table IWB-2500-1 (B-P) and Articles IWA-5000 and IWB-5000. Leakage monitoring continuously occurs from the Class 1 boundary into the CNV. This constitutes a VT-2 exam in accordance with Section XI IWA-5241 (c). Section 5.2.5, RCPB Leakage Detection contains further details.

The body-to-bonnet seals on the ECCS trip/reset actuator valve form a portion of the RCPB and require testing to RCS operating pressure before going into operation. Because this valve is located in the reactor pool, there is no means to perform the required ASME BPVC Section XI, Table IWB-2500-1 (B-P), VT-2 examination during the system pressure test. Therefore, a seal test is performed and meets the requirements of Reference 5.2-9, Table IWB-2500-1 (B-P).

The exterior nozzle-to-safe end welds and safe end-to-piping welds associated with the PZR spray lines, RPV high point degasification line, and CVCS injection and discharge lines require surface examination. The nozzle-to-safe end welds examination conform to the guidance in IWB-2500-1 Category B-F and safe end-to-piping welds examination conform to the guidance in IWB-2500-1 Category B-J.

The ASME Class 1 boundary valves (i.e., CIVs) are outside of the NPM. The reduced inspection requirements for the small primary system pipe welds associated with smaller than four inch nominal pipe size piping are not applied to the welds between the containment and the CIVs because a break at one of these weld locations would result in an RCPB leak outside the containment. Therefore, ASME Class 1 welds between the containment and the CIVs undergo a volumetric examination each interval in accordance with the requirements of Reference 5.2-9, Subarticle IWB-2500.

Flanges on the RPV have dual O-rings with a leak port tube between the O-rings to allow for leakage testing. Leakage testing is performed following installation of the O-rings each time they are removed to ensure the seals are seated as designed.

5.2.4.2 Preservice Inspection and Testing Program

Preservice examinations required by the design specification and preservice documentation are in accordance with Reference 5.2-8, Paragraph NB-5281. Volumetric and surface examinations conform to ASME BPVC Section III, Paragraph NB-5282. Components described in ASME BPVC Section III, Paragraph NB-5283, are exempt from preservice examination.

Surfaces of the RPV are suitable for examinations and conform to the applicable requirements of ASME BPVC Sections III and XI. For welds requiring an ultrasonic preservice examination, the surface finish meets the requirements of Reference 5.2-8, Subsubparagraph NB-4424.2(a).

Preservice examinations for ASME Code Class 1 pressure boundary and attachment welds conform with Reference 5.2-8, Paragraph NB-5280 and Reference 5.2-9, Subarticle IWB-2200. These preservice examinations include essentially 100 percent of the pressure boundary welds.

Preservice eddy current examinations for the SG tubing are in accordance with the applicable requirements of the EPRI Steam Generator Management Program guidelines (Reference 5.2-7) and Reference 5.2-9.

COL Item 5.2-4: An applicant that references the NuScale Power Plant US460 standard design will develop site-specific preservice examination, inservice inspection, and inservice testing program plans in accordance with Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code and the American Society of Mechanical Engineers Operations and Maintenance Code, and will establish implementation milestones. If applicable, an applicant that references the NuScale Power Plant US460 standard design will identify the implementation milestone for the augmented inservice inspection program. The applicant will identify the applicable edition of the American Society of Mechanical Engineers Code utilized in the program plans consistent with the requirements of 10 CFR 50.55a.

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection

The RCS of each NPM does not employ traditional light water reactor components with designed leakage rates, such as through pump seals or valve stem shafts.

The RCS leakage detection system withstands the effects of seismic events and other natural phenomena without losing the capability to perform its intended safety functions, thus meeting GDC 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 leak rate greater than the capability of the makeup system to replenish the coolant loss.

For each NPM, distinguishing between RCS identified and unidentified leakage inside the containment is not practicable with the installed instrumentation. Leakage into containment may originate from sources other than from the RCPB (e.g., leakage from reactor component cooling water). Expected leakage occurs from the RCS to containment through mechanical boundaries such as the RRVs, RVVs, and RSVs. There is a partial vacuum condition in the CNV during NPM startup and during reactor operation. As a result, reactor coolant leakage, whether from a known or unknown source, into containment quickly vaporizes and disperses within the containment atmosphere. Upon vaporization, there are no means to monitor separately the flow rates of identified and unidentified leakage from inside the containment. Therefore, containment leakage is treated as unidentified leakage until the source is known and quantifiable by other means. Performing an RCS inventory balance and comparing it to the total flow rate into the containment evacuation system (CES) determines the RCS leakage rate into the containment. The operational unidentified leakage limit is in plant technical specifications.

5.2.5.1 Leakage Detection and Monitoring

The CES satisfies GDC 30 requirements; the CES supports three methods for detecting and, to the extent practicable, identifying the source of leakage into the CNV. These leak-detection methods are

- containment vessel pressure monitoring.
- containment evacuation system sample tank level change monitoring.
- containment evacuation system vacuum pump discharge process radiation monitoring.

These leak detection methods satisfy the guidance in RG 1.45 for monitoring RCS leakage.

Regulatory Positions C.2.1 and C.2.2 of RG 1.45 are satisfied because leakage into the CNV from unidentified sources can be detected, monitored, and quantified for flow rates greater than or equal to 0.05 gpm using CNV pressure or CES sample tank level timing, and leakage detection response time (not including transport delay time) is less than one hour for a leakage rate greater than 1 gpm using CNV pressure or CES sample tank level timing. Radiation detectors in the CES condenser vent line and sample tank provide an early indication of RCS leakage. They provide the ability to discern changes in CES process radiation levels and assist the operator in assessing the source of leakage into the CNV. Section 11.5, Process and Effluent Radiation Monitoring Instrumentation and Sampling System, describes radiation monitoring for the CES.

Regulatory Position C.2.3 of RG 1.45 is satisfied because the technical specifications identify at least two independent and diverse methods for detection of leakage. Technical specification 3.4.7 identifies the three leak detection methods and their operability requirements.

Regulatory Position C.2.4 in RG 1.45 is satisfied because CNV pressure monitoring is performed by two redundant seismically qualified pressure sensors located on the suction line to the CES vacuum pumps. The attendant instrument and control platform for these transmitters is the module protection system, providing a seismically qualified interface to the main control room.

Regulatory Position C.2.5 in RG 1.45 is satisfied because each of the leakage monitoring systems have provisions that permit calibration and testing during plant operation. When an operating CNV vacuum is established, the equilibrium pressure in the CNV can be correlated directly to the total leakage into the CNV.

The CNV pressure monitoring detects and quantifies leakage, which is conservatively considered unidentified leakage unless a different method identifies the source and quantity. Factors that potentially would degrade CES performance result in conservative leak rate indication as they result in higher CNV pressure, overstating the leak rate into the CNV.

Section 9.3, Process Auxiliaries, provides a description of the CES. Figure 5.2-2 provides a containment pressure saturation curve as a function of reactor pool bulk temperature with an adjustment to account for containment pressure instrumentation uncertainty. When containment pressure is in the Not Acceptable region of Figure 5.2-2, condensation may exist inside the containment thus impacting the accuracy of the containment pressure monitoring and CES condensate monitoring systems.

5.2.5.2 Reactor Pressure Vessel Flange Leak-Off Monitoring

Bolted flanges and covers in the RCS are sealed by concentric O-rings. These flanges and covers include a leak-off port located between the two concentric O-ring grooves providing the capability to pressurize the space between the O-rings thereby confirming that the O-ring seal is leak tight prior to operation. The leak-off port is sized such that a break or leak within the leak-off connection would result in a leakage rate that is less than the normal makeup capacity of CVCS. There is no specific RPV flange leak-off monitoring.

5.2.5.3 Reactor Safety Valve and Emergency Core Cooling System Valve Leakage Monitoring

Leakage from the RSVs, ECCS valves, and actuators exhausts directly to the containment atmosphere; the total unidentified leakage into the containment includes this leakage. There is no specific leakage monitoring of the RSVs, ECCS valves, and pilot actuators.

5.2.5.4 Chemical and Volume Control System Intersystem Leakage Monitoring

Leakage from the CVCS outside the RCPB is classified as identified leakage. The CVCS leakage from pumps, valves or flanges that contain potentially radioactive liquid effluents from system vents, drains, and relief valves collects and drains to the reactor building equipment drain sump and flows to the low conductivity waste collection tanks. The liquid radioactive waste system provides the capability to monitor the level of the low conductivity waste collection tanks. An annunciation system alarms when a pre-set high leakage level in the tank is reached.

Normally open CIVs connect the CVCS to the RCPB. Intersystem leakage is considered for the following CVCS connected systems:

- boron addition system and demineralized water system
- reactor component cooling water system (RCCWS)
- process sampling system
- module heatup system heat exchangers
- letdown to the liquid radioactive waste system

Intersystem leakage is identified by

- increasing level, temperature, flow, or pressure.
- relief valve actuation.
- increasing radioactivity.

Section 9.3.3, Equipment and Floor Drain Systems, Section 9.3.4, CVCS, and Section 11.5, Process and Effluent Radiation Monitoring Instrumentation and Sampling System, contain further discussion related to the CVCS intersystem leakage detection and monitoring capabilities.

5.2.5.5 Reactor Component Cooling Water System Leakage Monitoring

Monitoring expansion tank level and an alarm in the control room provides leakage detection for the RCCWS. In the event of radioactivity in the RCCWS piping, radiation elements and transmitters located downstream of the non-regenerative heat exchanger, the process sampling system cooler lines, and the CES condenser for each NPM detect the radiation and alarm in the control room. Section 9.2.2, Reactor Component Cooling Water System, contains additional information on RCCWS.

5.2.5.6 Primary to Secondary Leakage Monitoring

The gaseous effluent from the condenser air removal system detects primary to secondary leakage. The MS lines condenser air removal system and turbine sealing steam system have radiation monitoring. There is the capability to obtain grab samples of MS and FW to analyze for indications of primary to secondary leakage. Additional detail of gaseous and liquid effluent radioactivity monitoring is in Section 11.5, Process and Effluent Radiation Monitoring Instrumentation and Sampling System.

COL Item 5.2-5: An applicant that references the NuScale Power Plant US460 standard design will establish plant-specific procedures that specify operator actions for identifying, monitoring, and trending reactor coolant system leakage in response to prolonged low leakage conditions that exist above normal leakage rates and below the technical specification limits. The objective of the methods of detecting and trending the reactor coolant pressure boundary leak will be to provide the operator sufficient time to take actions before the plant technical specification limits are reached.

5.2.6 References

5.2-1 NuScale Power, LLC, "Non-Loss-of-Coolant Accident Analysis Methodology," TR-0516-49416-P, Revision 4.

5.2-2 NuScale Power, LLC, "Loss-of-Coolant Accident Evaluation Model," TR-0516-49422-P, Revision 3.

5.2-3 Electric Power Research Institute, "Pressurized Water Reactor Primary Water Chemistry Guidelines," EPRI #3002000505, Revision 7, Palo Alto, CA, 2014.

5.2-4 American Society of Mechanical Engineers, "*Quality Assurance Requirements for Nuclear Facility Applications*," ASME NQA-1-2015, New York, NY.

5.2-5 Electric Power Research Institute, "Materials Reliability Program: Resistance to Primary Water Stress Corrosion Cracking of Alloys 690, 52, and 152 in Pressurized Water Reactors (MRP-111)," EPRI #1009801, Palo Alto, CA, 2004.

5.2-6 Electric Power Research Institute, "Materials Reliability Program: Resistance to Primary Water Stress Corrosion Cracking of Alloy 690 in Pressurized Water Reactors (MRP-258)," EPRI #1019086, Palo Alto, CA, 2009.

5.2-7 Electric Power Research Institute, "Steam Generator Management Program: Pressurized Water Reactor Steam Generator Examination Guidelines," EPRI #1013706, Revision 7, Palo Alto, CA, 2007.

5.2-8 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2017 Edition, Section III, Division 1, "Rules for Construction of Nuclear Facility Components," New York, NY.

5.2-9 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2017 Edition, Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components," New York, NY.

5.2-10 NuScale Power, LLC, "Use of Austenitic Stainless Steel for NPM Reactor Pressure Vessel," TR-130721-P, Revision 0.

Table 5.2-1: American Society of Mechanical Engineers Code Cases

Code Case Number	Title	Revision
N-4-13	Special Type 403 Modified Forgings or Bars, Section III, Division 1, Class 1 and CS	February 2008
N-60-6	Material for Core Support Structures, Section III, Division 1	December 2011
N-759-2	Alternative Rules for Determining Allowable External Pressure and Compressive Stresses for Cylinders, Cones, Spheres, and Formed Heads, Section III, Division 1	January 2008
N-774	Use of 13Cr-4Ni (Alloy UNS S41500) Grade F6NM Forgings Weighing in Excess of 10,000 lb (4,540 kg) and Otherwise Conforming to the Requirements of SA-336/SA-336M for Class 1, 2 and 3 Construction, Section III, Division 1	September 2008
N-782	Use of Code Editions, Addenda and Cases Section III. Division 1	January 2009
N-844	Alternatives to the Requirements of NB-4250(c) Section III, Division 1	February 2014
N-845-1	Qualification Requirements for Bolts and Studs, Section XI, Division 1	April 2016
N-849	In Situ VT-3 Examination of Removable Core Support Structure Without Removal, Section XI, Division 1	September 2014
N-883	Construction of Items Prior to the Establishment of a Section III, Division 1 Owner, Section III, Division 1	January 2018
N-885	Alternative Requirements for Table IWB-2500-1, Examination Category B-N-1, Interior of Reactor Vessel, Category B-N-2, Welded Core Support Structures and Interior Attachments to Reactor Vessels, Category B-N-3, Removable Core Support Structures, Section XI, Division 1	December 2018
N-890	Materials Exempted from G-2110(b) Requirement Section XI, Division 1	October 2018

Table 5.2-2: Reactor Safety Valves - Design Parameters

Parameter		Value
Quantity		2
Design Temperature		650°F
	Minimum Design Capacity per valve	84,100 lbm/hr saturated steam
Nominal set Pressure	First valve Second Valve	2200 psid 2290 psid
Operational set pressure tolerance		± 3%
Blowdown from set pressure		5%

Table 5.2-3: Reactor Coolant Pressure Boundary Component and Support Materials Including Reactor Vessel, Attachments, and Appurtenances

Component	Specification	Grade, Class, or Type
Reactor Pressure Vessel		
Lower Vessel (Lower Head, Shell and Flange)	SA-965	FXM-19[1]
Upper Vessel (Flange, Shells including Integral Steam Plenum Baffle Plate, Upper Head, Steam Plenum and Feed Plenum Access Ports)	SA-508	Grade 3 Class 2
SG Tubes	SB-163	UNS N06690
Nozzles (Thermowells, Level Sensors, Pressure Taps); Steam Plenum Caps	SB-166	UNS N06690
CNV-RPV Lateral Support Lugs; RPV-CNV Support Ledges	SB-168	UNS N06690
Safe-Ends	SA-182	F304[2]
Integral Steam Plenum Bore Sleeves	SA-240	Type 304[2]
Covers for Steam Plenum Access Ports Covers for Feed Plenum Access Ports	SA-182	F304[2] FXM-19
	SA-240	Type 304[2] FXM-19[1]
Leak Test Ports	SA-312	TP316 SMLS[2]
PZR Heater Bundle Flange	SB-168	UNS N06690
PZR Heater Element End Plug	SA-479	Type 304[2]
PZR Heater Element Sheath	SA-213	TP316[2]
Threaded Inserts; Pipe Reducers; PZR Spray Nozzles	SA-479	Type 304[2]
RPV Bolting		
Main Flange Closure	SB-637	UNS N07718[3]
Other Than Main Flange Closure	SA-193	Grade B8 Class 1
	SA-194	Grade 8
	SB-637	UNS N07718[3]
CRDM Support Structure		
Supports	SA-240	Type 304[2]
	SA-479	Type 304[2]
Bolting	SA-564	Type 630, H1100
CRDM Pressure Retaining Components		
CRDM Pressure Housing	SA-182	F304 or F304LN[2]
	SA-965	F304LN[2]
Top Plug Components	SA-479	Type 304[2] Type 410
Austenitic Stainless Steel Weld Filler Metals	SFA-5.4	E308[4]
	SFA-5.9	ER308[4]
Weld Filler Metals for RPV and CRDM Support Structure		
Low-Alloy Steel Weld Filler Metals	SFA-5.5	E90XX-X
	SFA-5.23	F9XX-EXX-XX or F10XX-EXX-XX
	SFA-5.28	ER90S-X
	SFA-5.29	E9XTX-XX

Table 5.2-3: Reactor Coolant Pressure Boundary Component and Support Materials Including Reactor Vessel, Attachments, and Appurtenances (Continued)

Component	Specification	Grade, Class, or Type
2XX Austenitic Stainless Steel Weld Filler Metals	SFA-5.4	E209, E240[1]
	SFA-5.9	ER209, ER240[1]
3XX Austenitic Stainless Steel Weld Filler Metals (include filler metals for weld-overlay cladding)	SFA-5.4	E308, E308L, E309, E309L, E316, E316L[4]
	SFA-5.9	ER308, ER308L, ER309, ER309L, ER316, ER316L, EQ308L, EQ309L[4]
	SFA-5.22	E308, E308L, E309, E309L, E316, E316L[4]
Nickel-Base Alloy Weld Filler Metals	SFA-5.11	ENiCrFe-7
	SFA-5.14	ERNiCrFe-7, ERNiCRFE-7A, EQNiCrFe-7, EQNiCrFe-7A
RCS Piping • RCS Injection Piping Assembly • RCS Discharge Piping Assembly • RCS PZR Spray Piping Assembly • RPV High Point Degasification Piping Assembly		
Pipe	SA-312	TP304 SMLS, TP316 SMLS[2]
Pipe Fittings	SA-182	F304. F316[2]
	SA-403	WP304 SMLS, WP316 SMLS[2]
	SA-479	Type 304, Type 304L, Type 316, Type 316L[2]
Piping Supports		
Supports	SA-240	Type 304 and Type 316[2] Type 405, Type 410S
	SA-479	Type 304, Type 316[2] Type 405, Type 410 Annealed or Class 1
	SA-312	TP304, TP304L, TP316, TP316L[2]
Bolting	SA-193	Grade B8, Grade B8M
	SA-194	Grade 8, Grade 8M
	SA-564	Type 630 H1100
Weld Filler Metals for Piping and Their Supports		
3XX Austenitic Stainless Steel Weld Filler Metals	SFA-5.4	E308, E308L, E316, E316L[4]
	SFA-5.9	ER308, ER308L, ER316, ER316L[4]
	SFA-5.30	IN308, IN308L, IN316, IN316L[4]
Nickel-Base Alloy Weld Filler Metals	SFA-5.11	ENiCrFe-7
	SFA-5.14	ERNiCrFe-7, ERNiCRFE-7A
SGs		
Piping	Table 5.4-3	
Piping Supports		
Bolting		
SG Supports		
SG Tube Supports		
Backing Strips		
RCPB Valves		

Table 5.2-3: Reactor Coolant Pressure Boundary Component and Support Materials Including Reactor Vessel, Attachments, and Appurtenances (Continued)

Component	Specification	Grade, Class, or Type
RSVs	Table 6.1-3	
RVVs		
RRVs		
RCS Injection and Discharge line Isolation Valves		
RCS PZR Spray Line Isolation Valves		
RPV High Point Degasification Line Isolation Valves		

Notes:

(1) 0.04 percent maximum carbon for FXM-19 and Type 2XX weld filler metals. Ferrite number in the range of 5 FN to 16 FN.

(2) 0.03 percent maximum carbon for unstabilized AISI Type 3XX base metals if welded or exposed to temperature range of 800 degrees F to 1500 degrees F subsequent to final solution anneal.

(3) SB-637 UNS N07718 solution treatment temperature range before precipitation hardening treatment restricted to 1800 degrees F to 1850 degrees F.

(4) 0.03 percent maximum carbon for unstabilized AISI Type 3XX weld filler metals; ferrite number in the range of 5 FN to 20 FN, except 5 FN to 16 FN for Type 316 and Type 316L.

Table 5.2-4: Reactor Coolant Water Chemistry Controls

Parameter (units)	Operating Range(1)	RG 1.44 Limit
Chloride (ppm)	≤ 0.15	0.15
Fluoride (ppm)	≤ 0.15	0.15
Dissolved oxygen (ppm)	≤ 0.10(2)	0.10
Sulfate (ppm)	≤ 0.15	-
Boron (ppm)	< 2000	-

Notes:

(1) The values include startup, shutdown and power operations.

(2) Applies only when RCS temperature is above 250F.

Table 5.2-5: Low Temperature Overpressure Protection Pressure Setpoint as Function of Cold Temperature

Cold Temperature (°F)	PZR Pressure (psia)
<146.0	420
146	1750
175	1750
210	2025
290	2025
>290	LTOP not enabled

Figure 5.2-1: Reactor Safety Valve Simplified Diagram

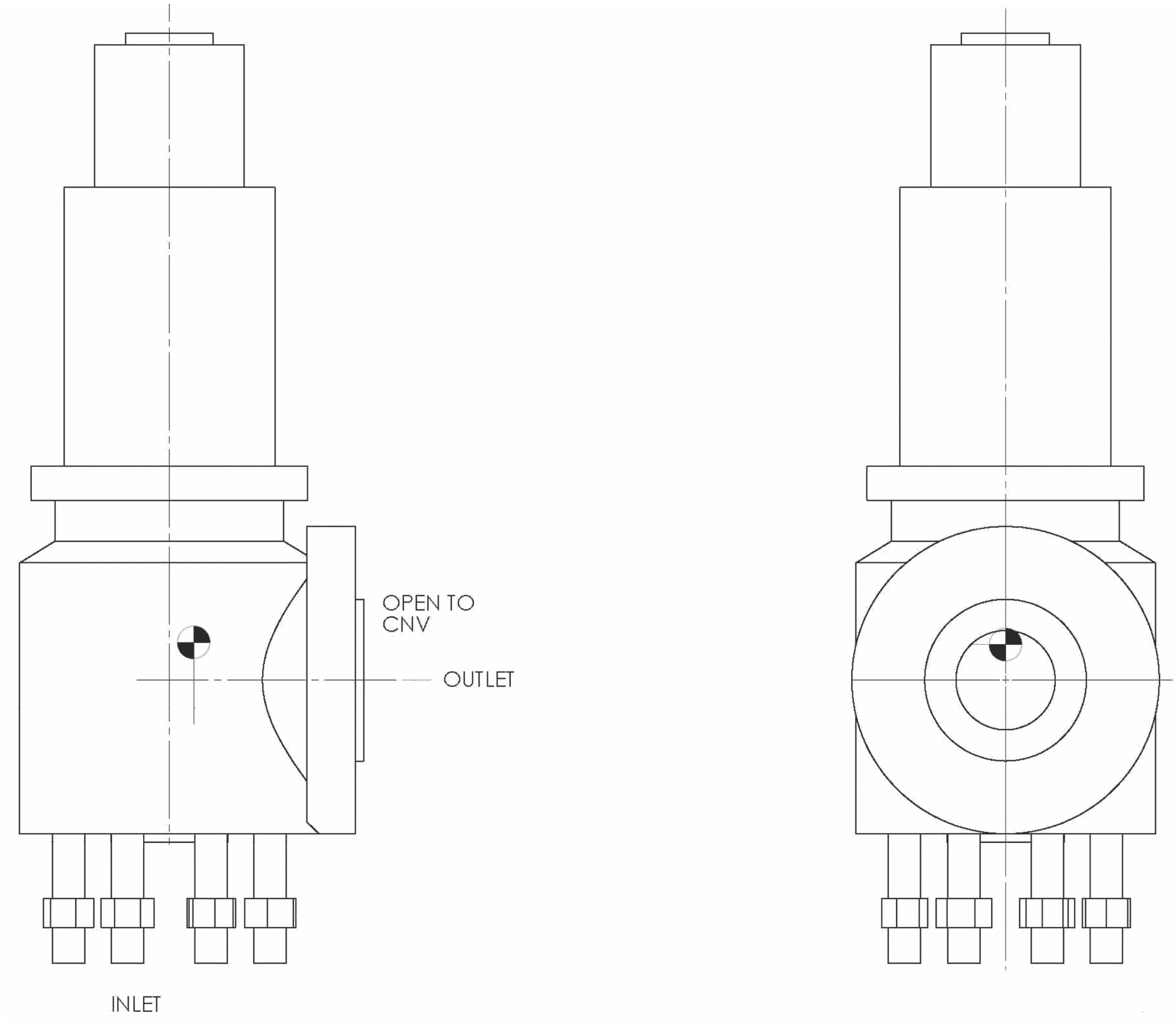


Figure 5.2-2: Containment Leakage Detection Acceptability

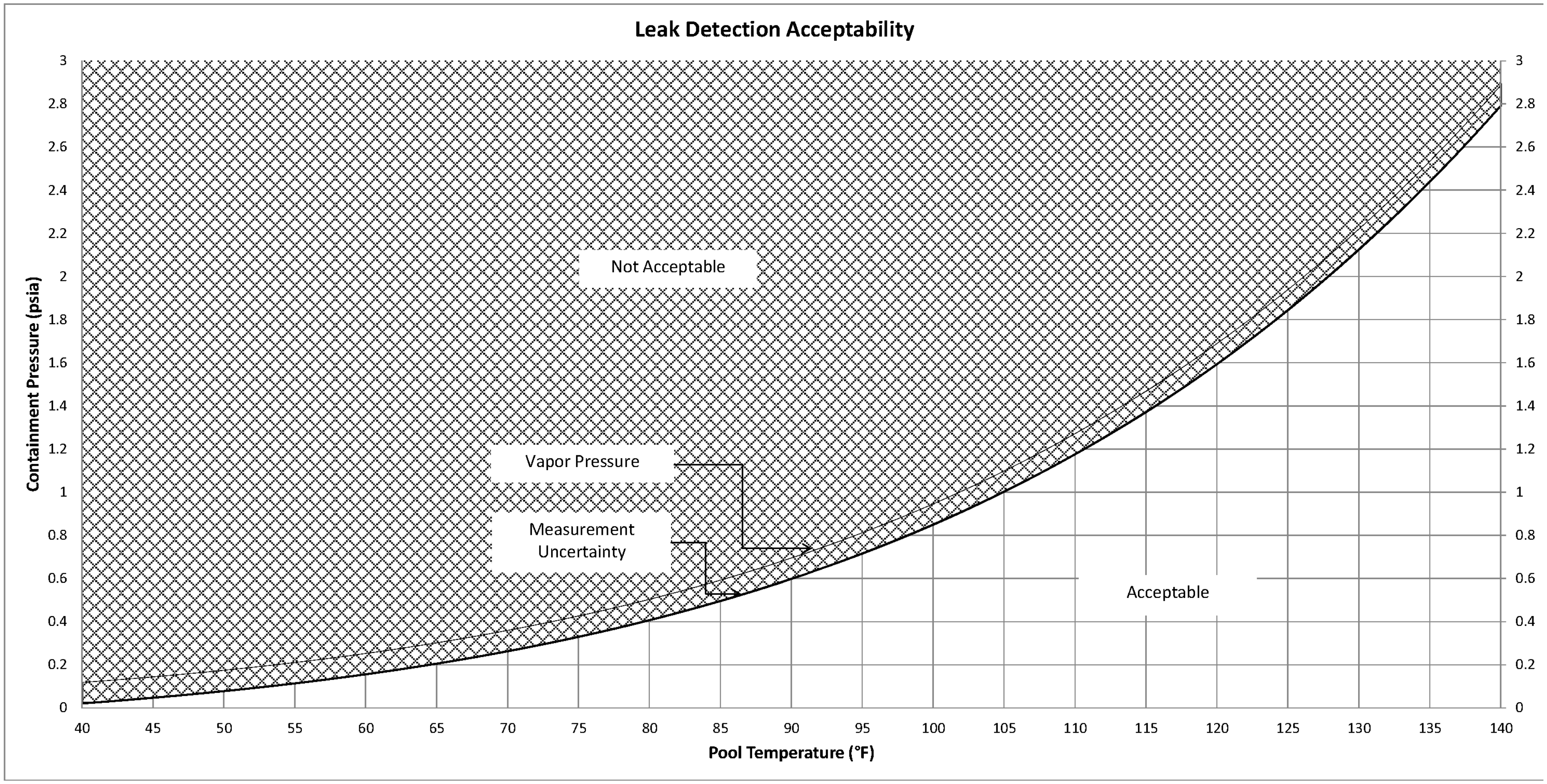
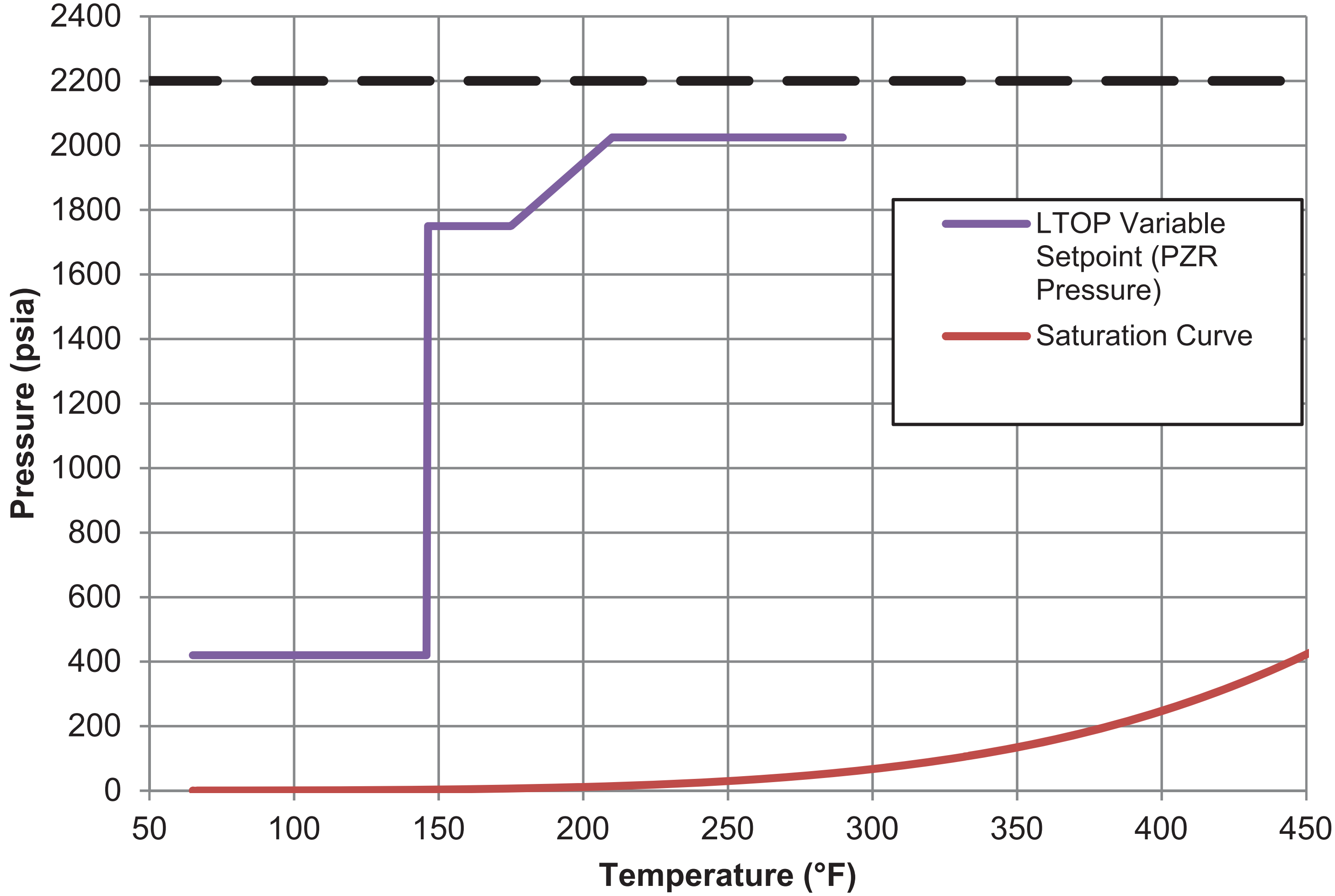


Figure 5.2-3: Variable Low Temperature Overpressure Protection Setpoint



Note: Black line is RPV Design Pressure

5.3 Reactor Vessel

A NuScale Power Module (NPM) consists of a reactor core, two steam generators (SGs), and a pressurizer all contained within a single reactor pressure vessel (RPV), with a containment vessel (CNV) that surrounds the RPV. The NPM includes the piping located between the RPV and the CNV.

The RPV is a pressure retaining vessel component of the reactor coolant system (RCS). Section 5.1 and Section 5.2 describe the RCS and reactor coolant pressure boundary (RCPB). The RPV metal vessel that forms part of the RCPB is a barrier to the release of fission products. The RPV contains the reactor core, reactor vessel internals (including the SGs and SG tube supports), pressurizer, and reactor coolant volume. The RPV is supported laterally and vertically by the CNV. The RPV provides support and attachment locations for the control rod drive mechanisms (CRDMs), the CRDM seismic support structure, pressurizer heater bundles, in-core instrumentation, SG system piping, RCS piping, reactor safety valves, reactor vent valves, and reactor recirculation valves. The RPV is certified and stamped in accordance with Article NCA-8000 of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) Section III. The reactor vessel is in Figure 5.3-1 and design parameters are in Table 5.3-1.

5.3.1 Reactor Vessel Materials

5.3.1.1 Material Specifications

The materials and applicable specifications used in the RPV and appurtenances are in Table 5.2-3.

Selection and fabrication of the RPV materials maintains RCPB integrity for the plant design lifetime. Selection of bolting materials, pressure retaining base materials, and weld filler materials are from the ASME BPVC Section II and comply with Article NB-2000 of ASME BPVC Section III (Reference 5.3-1). The austenitic stainless steel portion of the lower RPV has superior ductility and is less susceptible to the effects of neutron and thermal embrittlement, eliminating the need to calculate fracture toughness according to the requirements of 10 CFR 50, Appendix G. The ferritic low alloy steel of the upper RPV meets the fracture toughness requirements of 10 CFR 50, Appendix G. Reference 5.3-7 provides further details regarding the resistance to neutron and thermal embrittlement capability of the austenitic stainless steel material used in the lower RPV.

The RCPB materials comply with the relevant requirements of the following regulations:

- 10 CFR 50, Appendix A.
  - GDC 1 and 30. The RPV design, fabrication, and testing meets ASME BPVC Class 1 in accordance with the Quality Assurance Program described in Chapter 17, Quality Assurance and Reliability Assurance.
  - GDC 4. The RPV design and fabrication is compatible with environmental conditions of the reactor coolant and containment atmosphere (Reference 5.3-7).

  - GDC 14 and 31. The RPV design and fabrication has sufficient margin to assure the RCPB behaves in a non-brittle manner and minimizes the probability of rapidly propagating fracture and gross rupture of the RCPB (Reference 5.3-7).
  - GDC 15. The RPV design, fabrication, and testing meets ASME BPVC Class 1 requirements. Therefore, the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences. (Reference 5.3-7).
  - GDC 32. Inspection of the RCPB is in Section 5.2.4. Section 5.3.1.6 discusses that a material surveillance program for the RPV is not required. The design supports an exemption from the requirements of 10 CFR 50.60, which includes an exemption from 10 CFR 50, Appendix H (Reference 5.3-7).
- 10 CFR 50, Appendix G. The RPV materials meet applicable fracture toughness acceptance criteria. However, the design supports an exemption from the requirements of 10 CFR 50.60, which includes an exemption from 10 CFR 50, Appendix G (Reference 5.3-7). Section 5.3.1.5 provides further details.

The RPV fabrication is in accordance with the requirements of ASME BPVC Section III, NB-4000. The reactor vessel internals, including the SG supports and SG tube supports (shown in Figure 5.4-5), are fabricated in accordance with ASME BPVC Section III, NG-4000. The RPV supports and CRDM seismic support structure fabrication is in accordance with ASME BPVC Section III, NF-4000.

5.3.1.2 Special Processes Used for Manufacture and Fabrication of Components

Forged low alloy steel and austenitic stainless steel form the RPV assembly shells that surround the reactor core, pressurizer, and SGs. Forgings form the various required geometries with a minimum amount of welding.

Section 5.2.3, RCPB Materials, addresses the upper RPV cladding.

Measures are taken to prevent sensitization of austenitic stainless steel materials during component fabrication. Heat treatment parameters comply with ASME BPVC Section II. Water quenching cools the austenitic stainless steel materials to avoid carbide formation at the grain boundaries; alternatively, cooling through the sensitization temperature range occurs quickly enough to avoid carbide formation at the grain boundaries. When means other than water quenching are used, corrosion testing in accordance with Practice A or E of American Society for Testing and Materials (ASTM) A262 (Reference 5.3-3) verifies nonsensitization of the base material.

Due to necessary component welding, fabrication subjects the heat-affected zone within the austenitic stainless steel materials to the sensitizing temperature range (800 degrees F to 1500 degrees F). Control of welding practices and material composition manages the sensitization while the material is in this temperature range, and unstabilized Type 3XX austenitic stainless steels and corresponding austenitic stainless steel weld filler metals have a carbon content not exceeding 0.03 weight percent to prevent undue sensitization. In addition, where unstabilized Type 3XX austenitic stainless steels are subjected to sensitizing temperatures for greater than 60 minutes during a post-weld heat treatment, non-sensitization of the materials are verified by testing in accordance with ASTM A262 Practice A or E, as required by Regulatory Guide (RG) 1.44.

5.3.1.3 Special Methods for Nondestructive Examination

The RPV pressure retaining and integrally attached materials examinations meet the requirements specified in ASME BPVC Section III. The examination methods are in accordance with ASME BPVC Section V, except as modified by Section III and any additional requirements listed below.

Non-destructive examination of the RCPB is in Section 5.2.3, RCPB Materials.

Preservice examinations performed in accordance with subsubarticle NB-5280 of Section III and subarticle IWB-2200 of Section XI for ASME BPVC Class 1 pressure boundary and attachment welds use the examination methods in Section V, except as modified by paragraph NB-5111 of Section III. These preservice examinations include essentially 100 percent of the pressure boundary welds.

For ASME BPVC Class 2 pressure boundary items, preservice examinations are in accordance with subarticle IWC-2200 of Section XI.

5.3.1.4 Special Controls and Special Processes Used for Ferritic Steels and Austenitic Stainless Steels

Welding of ferritic steels for components in the RPV uses procedures qualified in accordance with the applicable requirements of ASME BPVC, Section III, subarticle NB-4300 and Section XI (Reference 5.3-4). Further information is in Section 5.2.3.3, Fabrication and Processing of Ferritic Materials.

Welding of austenitic stainless steel components in the RPV uses procedures qualified in accordance with the applicable requirements of ASME BPVC, Section III and Section IX. Further information is in Section 5.2.3.4, Fabrication and Processing of Austenititc Stainless Steels.

In addition, electroslag welding processes are not utilized for joining materials. Cladding low alloy steel allows electroslag welding processes and complies with RG 1.43 requirements.

Section 4.5.2, Reactor Internals and Core Support Structure Materials, addresses tools for abrasive work.

Section 4.5.1, Control Rod Drive - Materials Specifications, addresses the use of cold-worked austenitic stainless steel.

5.3.1.5 Fracture Toughness

The RPV design prevents non-ductile fracture in accordance with GDC 14, GDC 15, and GDC 31. However, the design supports an exemption from the requirements of 10 CFR 50.60, which includes an exemption from the requirements of 10 CFR 50, Appendix G. The materials used for the lower RPV are not applicable to the fracture toughness analyses required by 10 CFR 50, Appendix G, and the upper RPV does not meet the neutron fluence levels to be assessed for the effects of neutron embrittlement.

10 CFR 50, Appendix G, requirements apply to ferritic materials of pressure-retaining components of the RCPB of light water nuclear power reactors. The NPM uses austenitic stainless steel materials (Table 5.2-3) in the lower RPV shell. The requirements of 10 CFR 50, Appendix G, rely on impact testing data performed in accordance with ASME BPVC Section III, Paragraph NB-2331. However, NB-2311 does not require impact testing of austenitic stainless steel.

The fluence values for the upper RPV shell do not exceed 1.0E+17 n/cm^2 (E > 1 MeV), which is the peak neutron fluence at the end of the design life of a reactor vessel that requires an assessment of the effects of neutron embrittlement as specified in 10 CFR 50, Appendix H. Therefore, the upper RPV, which is made of ferritic steel, does not require an assessment for the effects of neutron embrittlement and, there are no considerations of adjustments for embrittlement necessary using the RG 1.99 methodology.

The neutron flux and fluence calculation methods follow the guidance of RG 1.190 with exceptions as described in the NuScale Technical Report "Fluence Calculation Methodology and Results" (Reference 5.3-7). Reference 5.3-7 provides further details regarding the fracture toughness capabilities of the austenitic stainless steel material used in the lower RPV. Reference 5.3-6 provides the methodology used for derivation of the pressure-temperature limits for the RPV.

5.3.1.6 Material Surveillance

10 CFR 50, Appendix H, “Reactor Vessel Material Surveillance Program Requirements,” applies to ferritic materials in the reactor vessel beltline region of light water nuclear power reactors. The RPV design prevents non-ductile fracture in accordance with GDC 14, GDC 15, and GDC 31. However, the design supports an exemption from the requirements of 10 CFR 50.60, which include an exemption form the requirements of 10 CFR 50, Appendix H. The NPM uses non-ferritic materials in the lower shell of the RPV. The material surveillance program required by 10 CFR 50, Appendix H is based on the nil-ductility reference temperature (RT_{NDT}) according to ASME Section III, NB-2331, for ferritic steels. Because RT_{NDT} cannot be established for the austenitic stainless steel used in the lower RPV according to NB-2331, 10 CFR 50, Appendix H, is not applicable to the lower RPV. In addition, the requirement for an appropriate material surveillance program in GDC 32 is not applicable. Reference 5.3-7 provides

further details regarding the resistance to the effects of neutron and thermal embrittlement of the austenitic stainless steel material used in the lower RPV.

10 CFR 50 Appendix H is not applicable to the upper RPV shell. The end of design life fluence value for the upper RPV shell does not exceed 1.0E+17 n/cm^2 (E > 1 MeV). Therefore, 10 CFR 50 Appendix H is not applicable to this area of the RPV.

5.3.1.7 Reactor Vessel Fasteners

The RPV closure studs, nuts and washers use the materials indicated in Table 5.2-3. Section 3.13.1, Threaded Fastener - Design Considerations, which provides details on threaded fastener design considerations.

The RPV threaded fasteners use threaded inserts except for the main RPV flange studs. The threaded inserts are externally and internally threaded into the associated base metal. After insertion into the base metal, a seal weld is applied at the clad flange face to prevent fluid from entering between the threaded insert and base metal. The seal weld is a non-structural weld and is not credited to carry any load. As such, the external threads on the inserts and internal threads in the flange bolt holes carry mechanical loads during normal and off-normal operations, including ECCS actuation. Table 5.2-3 contains threaded insert materials and applicable specifications. The fabrication inspections for threaded inserts follow ASME BPVC Section III (Reference 5.3-1), Subsubarticle NB-2580, using the outer diameter of the threaded insert for sizing requirements.

For the RPV flange connection, lock plates perform a tooling function to hold the RPV flange nut in place, on top of the flange, after flange stud removal or during flange stud installation. The lock plates are not part of the RCPB. The lock plates only resist the minor friction loads and forces that occur when inserting and threading the RPV flange studs into the nuts and do not resist the forces applied to tension the stud. The same is true for removing and detensioning the RPV flange studs.

Studs attached with a fillet weld to the top of the flange cladding hold the lock plates in place. The welded studs retaining the lock plates are nonstructural attachments as defined in ASME BPVC Section III, NB-1132.1(c)(2), similar to insulation supports. The lock plates are non-ASME, non-structural attachments to the RPV.

The welding of the stud to the cladding requires a cladding preservice liquid penetrant exam, per ASME BPVC Section III, paragraph NB-5272, Weld Metal Cladding. The welding of the stud to the cladding also complies with ASME BPVC Section III, paragraph NB-4435, Welding of Nonstructural Attachments.

There are no inservice exam requirements for the lock plate stud welds or the lock plates.

5.3.2 Pressure-Temperature Limits, Pressurized Thermal Shock, and Charpy Upper-Shelf Energy Data and Analyses

The information in this section describes the bases for setting operational limits on pressure and temperature for the RCPB. The RPV design prevents non-ductile fracture in accordance with GDC 14, GDC 15, and GDC 31. The design supports an exemption from the requirements of 10 CFR 50.60, which includes an exemption from the requirements of 10 CFR 50, Appendix G, and 10 CFR 50, Appendix H. The design supports an exemption from the requirements of 10 CFR 50.61. Reference 5.3-9 provides further details regarding austenitic stainless steel used in the lower RPV, which is resistant to the effects of neutron and thermal embrittlement.

5.3.2.1 Limit Curves

The calculation of a generic set of pressure-temperature limits at 57 EFPY uses the methodology provided in ASME BPVC Section XI, Appendix G, and the applicable limits provided in 10 CFR 50, Appendix G, as described below. Consideration of only the initial RT_{NDT} temperature is necessary because the lower portion of the RPV is not a ferritic material, and the peak fluence for the upper portion of the RPV shell is less than the 10 CFR 50, Appendix H, criteria (1.0E+17 n/cm^2(E > 1 MeV)). Therefore, no adjustment is necessary to account for fluence embrittlement effects (Reference 5.3-5). For conservatism, the 10 CFR 50, Appendix G, Table 1, limits have been applied to the final pressure-temperature limits.

The pressure-temperature limits for normal heatup and criticality conditions, normal cooldown, and inservice leak and hydrostatic (ISLH) tests including transient conditions are in Figure 5.3-2, Figure 5.3-3, and Figure 5.3-4, respectively. The corresponding numerical values are in Table 5.3-2 and Table 5.3-3. RCS pressure maintained below the limit of the pressure-temperature limit curves ensures protection against non-ductile failure. Acceptable pressure and temperature combinations for reactor vessel operation are below and to the right of the applicable pressure-temperature curves. These pressure-temperature curves include neither location correction nor instrument uncertainty. For the purpose of location correction, the allowable pressure in the pressure-temperature curves is the pressure at the RPV bottom. The reactor is not permitted to be critical until the pressure-temperature combinations are to the right of the criticality curve shown in Figure 5.3-2.

Further information on the methodology used to develop the limits is in the NuScale Technical Report, "Pressure and Temperature Limits Methodology" (Reference 5.3-6).

5.3.2.2 Operating Procedures

Section 13.5, Plant Procedures, addresses development of plant operating procedures that ensure pressure-temperature limit compliance. These procedures ensure compliance with the technical specifications during normal power operating conditions and anticipated transients.

COL Item 5.3-1: An applicant that references the NuScale Power Plant US460 standard design will develop operating procedures to ensure that transients will not be more severe than those for which the reactor design adequacy had been demonstrated. These procedures will be based on material properties of the as-built reactor vessels.

5.3.2.3 Pressurized Thermal Shock

The RPV design prevents non-ductile fracture in accordance with GDC 14, GDC 15, and GDC 31. The design supports an exemption from the requirements of 10 CFR 50.61. The methodology described in 10 CFR 50.61 determines RT_{PTS}, which is the RT_{NDT} evaluated for the end of design life peak fluence for each beltline material. Because the lower RPV material is austenitic stainless steel, this material is exempt from impact test requirements per ASME BPVC Section III, NB-2311. As a result, the PTS screening methodology in 10 CFR 50.61 is not applicable to RPV beltline materials. Further, 10 CFR 50.61 is not applicable to the upper RPV shell. The end of design life fluence value for the upper RPV shell does not exceed 1.0E+17 n/cm2 (E > 1 MeV). Therefore, 10 CFR 50.61 is not applicable to this area of the RPV.This fluence means that the entire upper RPV shell is outside the RPV beltline region per 10 CFR 50.61. Therefore, 10 CFR 50.61 PTS screening is not required for the upper RPV shell. Reference 5.3-7 provides further details regarding the effects of neutron and thermal embrittlement on the austenitic stainless steel material used in the lower RPV.

5.3.2.4 Upper-Shelf Energy

The evaluation of effects of neutron embrittlement on RPV materials uses Charpy Upper-Shelf Energy. A decrease in Charpy Upper-Shelf Energy level as defined in ASTM E 185-82 occurs based on fluence levels and copper content in the material. The design does not require this evaluation because the lower RPV shell is not a ferritic material, and the fluence levels for the upper RPV shell are less than the peak neutron fluence at the end of the design life of 1.0E+17 n/cm^2 (E>1 MeV) (Reference 5.3-5). Reference 5.3-7 provides further details regarding the resistance to the effects of neutron and thermal embrittlement of the austenitic stainless steel material used in the lower RPV.

5.3.3 Reactor Vessel Integrity

5.3.3.1 Design

Section 5.3.1, Reactor Vessel Materials describes the compatibility of the RPV design with established standards. Section 5.2.4, RCPB Inservice Inspection and Testing, and Section 5.3.1, Reactor Vessel Materials describes how the basic design of the RPV establishes compatibility with required inspections.

5.3.3.2 Materials of Construction

Section 5.2.3, RCPB Materials, and Section 5.3.1, Reactor Vessel Materials describe the reactor vessel materials of construction.

5.3.3.3 Fabrication Methods

Section 5.2.3, RCPB Materials, and Section 5.3.1, Reactor Vessel Materials describe the fabrication methods used in the construction of the reactor vessel.

5.3.3.4 Inspection Requirements

Section 5.3.1, Reactor Vessel Materials describes the nondestructive examinations performed.

5.3.3.5 Shipment and Installation

Section 5.2.3.4.2 describes the packaging, shipment, handling, and storage of the RPV.

A dry environment is maintained for RPV surfaces, both primary and secondary sides, by an installed non-chloride, non-corrosive desiccant. Humidity indicators covering a suitable range of moisture content are shipped with the RPV. Both the primary and secondary sides of the RPV ship under positive pressure. The internal atmosphere on both sides of the SG tubes are evacuated to eliminate residual moisture and filled with nitrogen having a dew point less than -20 degrees F.

In preparation for shipping the RPV, the fabricator takes appropriate foreign material exclusion measures.

There are cleanliness and contamination controls in place during handling, storage, shipping, and during installation of the RPV. Section 5.2.3.4.2, Cleaning and Contamination Protection Procedures, provides details of the cleanliness procedures.

5.3.3.6 Operating Conditions

Operating conditions as they relate to the integrity of the reactor vessel are in Section 5.2.2, Overpressure Protection, and Section 5.3.2, Pressure-Temperature Limits, and in the plant technical specifications.

5.3.3.7 Inservice Surveillance

Inservice surveillance of the RPV is in Section 5.2.4, RCPB Inservice Inspection and Testing, and Section 5.3.1, Reactor Vessel Materials.

5.3.3.8 Threaded Fasteners

Threaded fasteners are in Section 3.13, Threaded Fasteners, and Section 5.3.1, Reactor Vessel Materials.

5.3.4 References

5.3-1 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2017 Edition, Section III, "Rules for Construction of Nuclear Facility Components," New York, NY.

5.3-2 American Society of Mechanical Engineers, Quality Assurance Requirements for Nuclear Facility Applications, ASME NQA-1-2015, New York, NY.

5.3-3 ASTM International, "Standard Practices for Detecting Susceptibility to Intergranular Attack in Austenitic Stainless Steels," ASTM A262-15, West Conshohocken, PA.

5.3-4 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2017 Edition, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components," New York, NY.

5.3-5 NuScale Power, LLC, "Fluence Calculational Methodology and Results," TR-118976, Revision 0.

5.3-6 NuScale Power, LLC, "Pressure and Temperature Limits Methodology," TR-130877-P, Revision 0.

5.3-7 NuScale Power, LLC, "Use of Austenitic Stainless Steel for NPM Reactor Pressure Vessel," TR-130721-P, Revision 0.

Table 5.3-1: Reactor Vessel Parameters

Design Parameter	Value
Design pressure (psia)	2200
Design temperature (degrees F)	650
Approximate overall height of the upper RPV, from the upper RPV closure flange mating surface to the CRDM interface on the upper RPV head	528
Inside diameter of lower RPV section, cylindrical region (inches)	96
Outside diameter of lower RPV section, cylindrical region (inches)	104
Inside diameter of upper RPV section, cylindrical region (inches)	104
Outside diameter of upper RPV section, cylindrical region (inches)	113
Inside diameter of pressurizer, cylindrical region (inches)	106
Outside diameter of pressurizer, cylindrical region (inches)	115
RPV upper section minimum inner clad thickness (inches)	0.125
RPV upper section minimum outer clad thickness (inches)	0.125

Table 5.3-2: Pressure-Temperature Limits for Normal Heatup and Cooldown

Normal Combined Heatup and Power Ascent Transient (Core Not Critical)		Composite Normal (Core Critical with RPV Pressure < 20% Pressure = 535.3 psig)		Composite Normal (Core Critical with RPV Pressure > 20% Pressure = 535.3 psig)		Normal Combined Power Descent and Cooldown	
		(Minimum core critical temperature determined from the steady state and transient ISLH curves)					
Fluid Temperature °F	**Pressure psig**	**Fluid Temperature °F**	**Pressure psig**	**Fluid Temperature °F**	**Pressure psig**	**Fluid Temperature °F**	**Pressure psig**
65	535	Reactor is not permitted to be critical below 90°F if ISLH testing is performed at steady-state or HTS/PAC or PWD/RCD transient conditions.		Reactor is not permitted to be critical below 160°F if ISLH testing is performed at steady-state or HTS/PAC or PWD/RCD transient conditions		600	3260
120	535					220	3260
120	2230					210	2400
150	2230	90	0	160	0	150	1875
200	2285	90	535	160	1875	120	1875
300	2475	160	535	190	1875	120	535
600	2475	160	1875	240	2285	65	535
		190	1875	340	2475		
		240	2285	640	2475		
		340	2475				
		640	2475				

Table 5.3-3: Pressure-Temperature Limits for Inservice Leak and Hydrostatic Test

ISLH for Combined Heatup and Power Ascent Transient		ISLH for Combined Power Descent and Cooldown Transient		Transient ISLH (Bounding of HTS/PAC and PWD/ RCD)		Steady-State ISLH	
Fluid Temperature °F	Pressure psig	Fluid Temperature °F	Pressure psig	Fluid Temperature °F	Pressure psig	Fluid Temperature °F	Pressure psig
65	535	600	4350	65	535	65	535
90	535	220	4350	90	535	90	535
90	2980	210	3200	90	2500	90	3660
150	2980	150	2500	150	2500	95	3960
200	3050	90	2500	200	3050	100	4300
300	3300	90	535	300	3300	105	4610
600	3300	65	535	600	3300	600	4610

Figure 5.3-1: Reactor Vessel

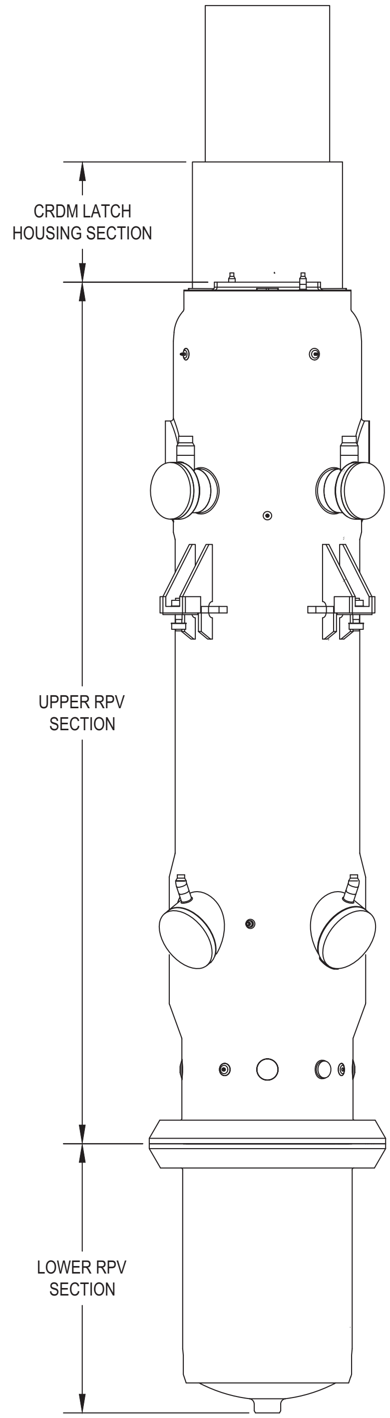
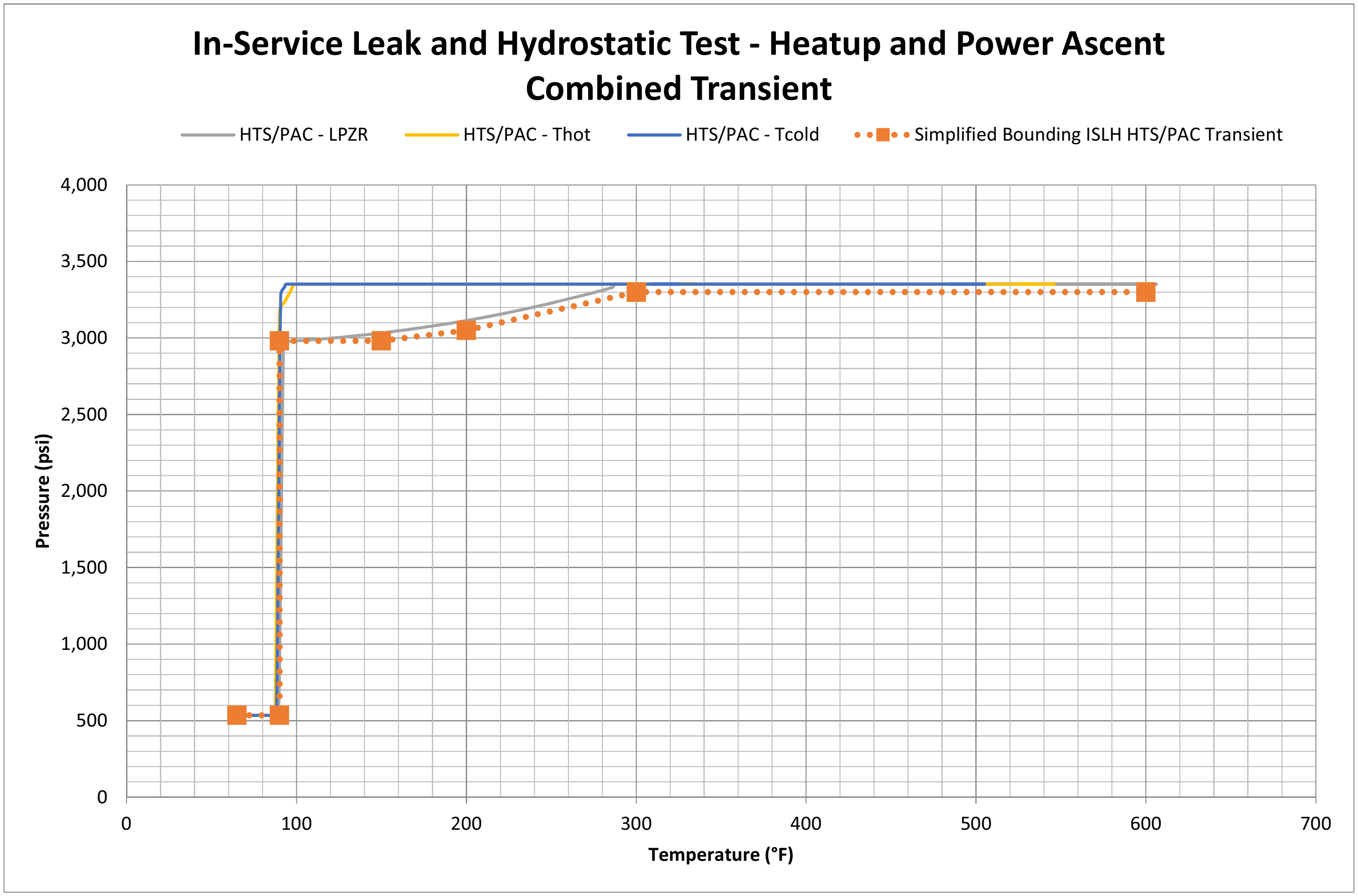


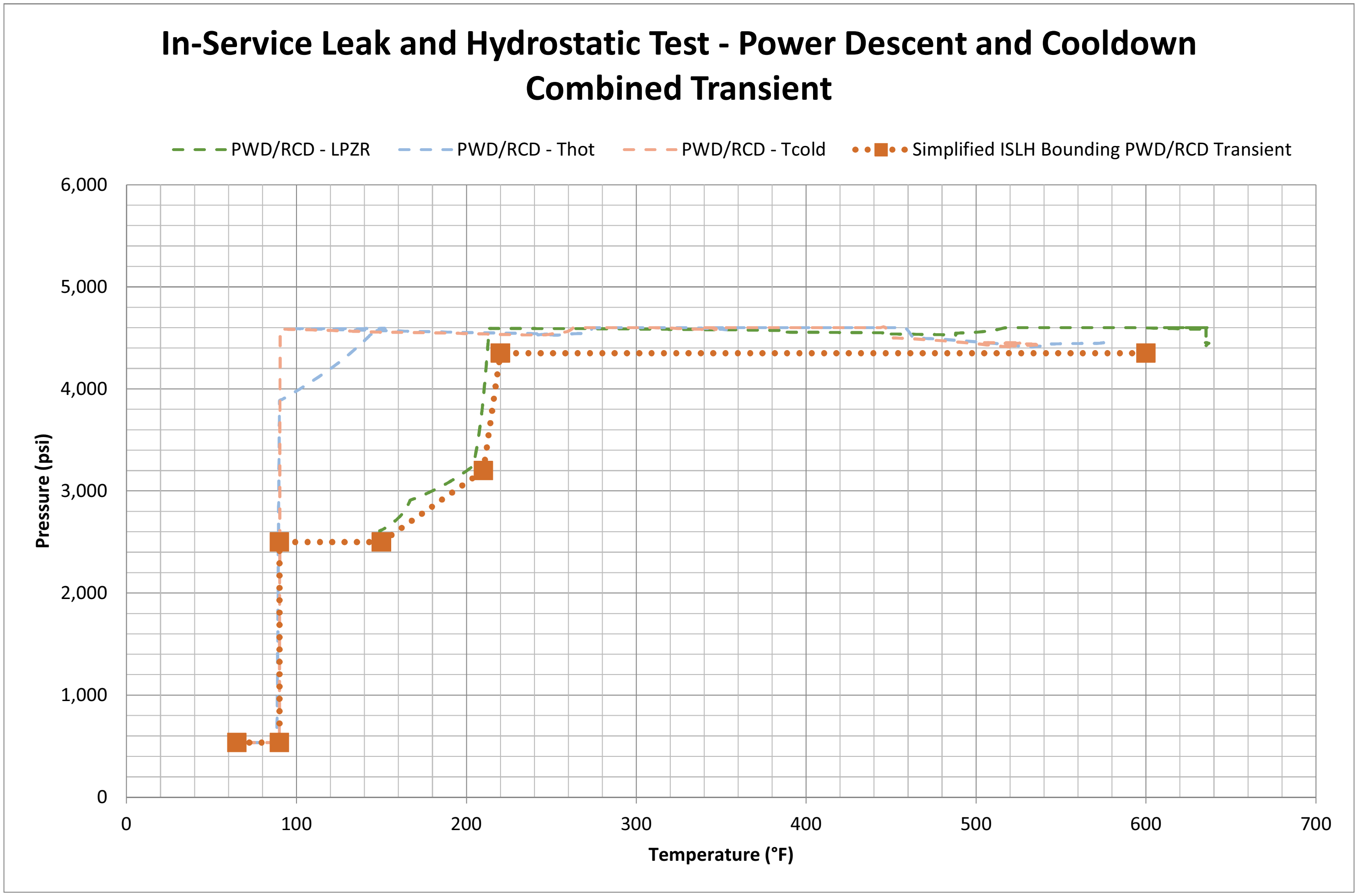
Figure 5.3-2: Pressure-Temperature Limits for Heatup and Power Ascent Combined Transient



Note: the following defines the nomenclature used in the above figure:

HTS: Heat-Up Transient
PAC: Power Ascent Transients
LPZR: Lower Pressurizer Region
ISLH: In-Service Leak and Hydrostatic Test

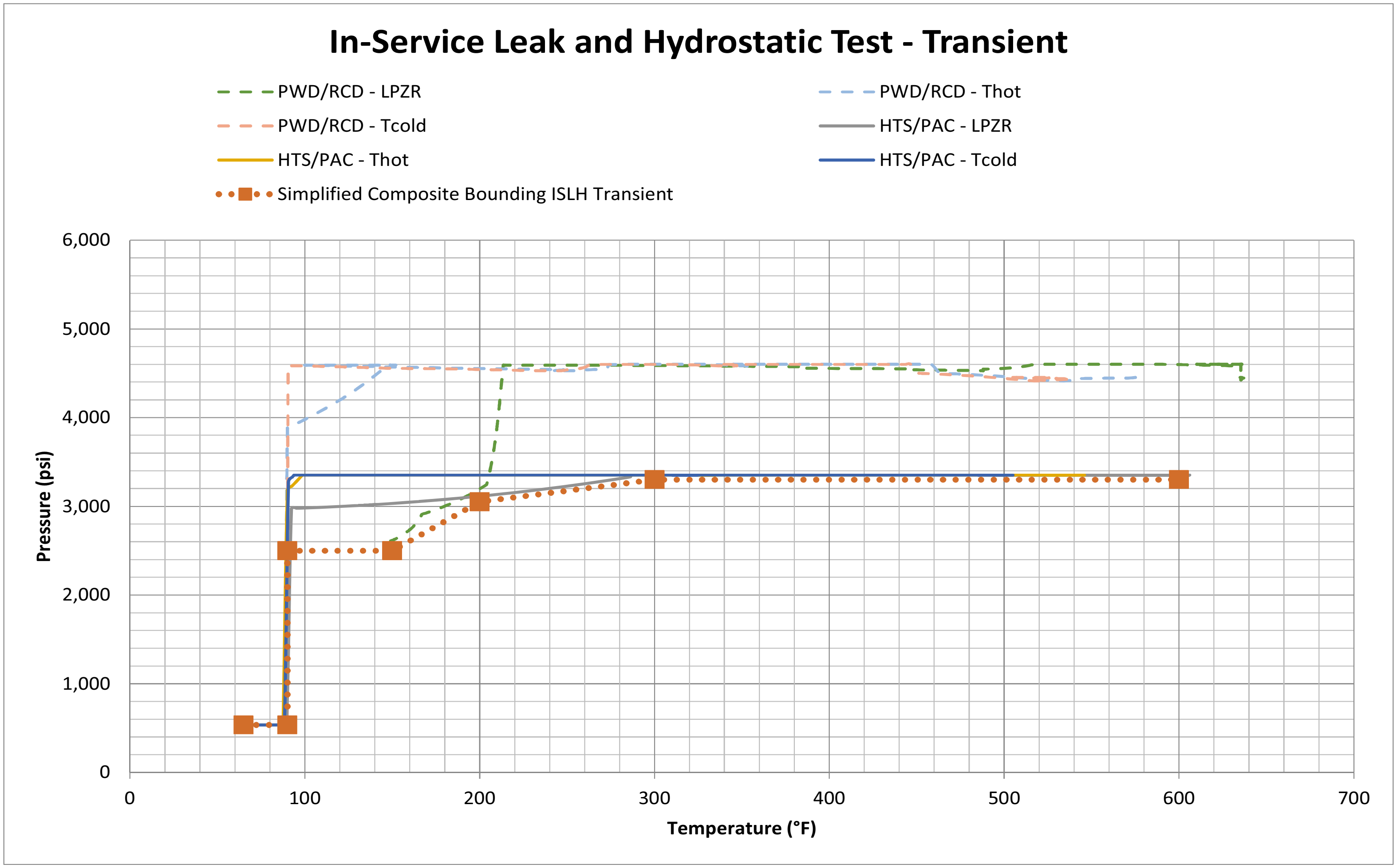
Figure 5.3-3: Pressure-Temperature Limits for Power Descent and Cooldown Combined Transient



Note: the following defines the nomenclature used in the above figure:

RCD: Cooldown Transient
PWD: Power Descent Transient
LPZR: Lower Pressurizer Region
ISLH: In-Service Leak and Hydrostatic Test

Figure 5.3-4: Pressure-Temperature Limits for Inservice Leak and Hydrostatic Tests



Note: the following defines the nomenclature used in the above figure:

HTS: Heat-Up Transient
PAC: Power Ascent Transient
RCD: Cooldown Transient
PWD: Power Descent Transient
LPZR: Lower Pressurizer Region
ISLH: In-Service Leak and Hydrostatic Test

5.4 Reactor Coolant System Component and Subsystem Design

The reactor coolant system (RCS) of the NuScale Power Module (NPM) contains the reactor pressure vessel (RPV) and reactor vessel internals; control rod drive mechanisms; a pressurizer (PZR); two steam generators (SGs); two reactor safety valves (RSVs); four emergency core cooling system (ECCS) valves; and reactor coolant system (RCS) injection, discharge, PZR spray, and high-point degasification vent lines. This section also discusses the decay heat removal system (DHRS) which is also part of the NPM.

The design basis and description of the reactor, reactor vessel internals, and control rod drive mechanisms are in Chapter 4. The design basis and description of the RSVs are in Section 5.2.2, Overpressure Protection, and the design basis and description of the ECCS valves (i.e., reactor vent valves (RVVs) and reactor recirculation valves (RRVs)) are in Section 6.3, Emergency Core Cooling System.

5.4.1 Steam Generators

The steam generator system (SGS) consists of: the feedwater (FW) piping from the containment system (CNTS) to the feed plenum access port; thermal relief valve; inlet flow restrictor; feed plenum access port and access cover; SGs tubes; steam plenum cap; steam plenum access port and access cover; and main steam (MS) piping from the steam plenum access port to the CNTS. The FW plenum is within the feed plenum access port with the tube sheet forming the boundary between the primary and secondary side. The MS plenum is within the RPV integral steam plenum shell with the steam plenum cap and RPV integral steam plenum baffle plate forming the boundary between the primary and secondary side. The FW piping, thermal relief valve, steam plenum access port, and MS piping of the SGS form the secondary side of the SGS.

The SGs in the NPM are integral to the RPV. The RPV forms the SG shell and provides the outer pressure boundary of the SGs. The SG tube, tube-to-tubesheet welds, and tubesheets provide part of the reactor coolant pressure boundary (RCPB). Section 5.2 and Section 5.3 describe the RPV and the RCPB.

5.4.1.1 Design Basis

The SGs transfer heat from the RCS to the secondary steam system and supply superheated steam to the steam and power conversion cycle as described in Chapter 10.

Table 5.4-1 provides a summary of the operating conditions for the thermal-hydraulic design of the SGs. The secondary plant parameters represent full-power steam flow conditions at best estimate primary coolant conditions.

The SGs provide sufficient stable flow on the secondary side of the tubes at operational power levels and mass flow rates to preclude reactor power oscillations that could result in exceeding specified acceptable fuel design limits.

The end of each SG tube in the FW plenum has a flow restriction device that creates the necessary secondary side pressure loss to produce stable, secondary fluid flow while operating in the nominal power generation range and to mitigate rapid temperature changes at the weld of the SG tube to the FW plenum.

Table 3.9-3 identifies load combinations on the RPV; which includes the SG tubes.

The SGs also provide two primary safety-related functions: they form a portion of the RCPB, and they transfer decay heat to the DHRS described in Section 5.4.3, Decay Heat Removal System.

The portions of the SGs that form a part of the RCPB provide one of the fission product barriers. In the event of fuel cladding failure, the barrier isolates radioactive material in the reactor coolant preventing release to the environment.

The SGs perform an integral part of the reactor residual and decay heat removal process when the DHRS is in operation. They transfer heat from the primary coolant to the naturally circulating DHRS closed loops that transfer decay heat to the reactor pool.

10 CFR 50.55a(g) requires the inservice inspection (ISI) program to meet the applicable inspection requirements of Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) (Reference 5.4-3). The SGS components allow performance of the ISI requirements of ASME BPVC, Section XI (Reference 5.4-5), including the preservice inspections specified by ASME Section III. Section 5.5.4 of the technical specifications describes an SG program and implements ASME Code Section III and XI for the SG tubes. The secondary sides of the SGs permit access for SG inspections. Integrity of SGs, integral steam plenum, integral steam plenum caps and FW plenum access ports that make up portions of the RCPB are in Section 5.2, Integrity of Reactor Coolant Boundary.

5.4.1.2 System Design

Each SG, located inside the RPV, has interlacing helical tube columns connecting to two feed and two steam plena. As shown in Figure 5.4-1 and Figure 5.4-2, the configuration of the helical tube columns of the two SGs form an intertwined bundle of tubes around the upper riser assembly with a total of four feed and four steam plena located 90 degrees apart around the RPV. Figure 5.4-3 and Figure 5.4-4 show cross-sectional views of an individual steam and feed plenum. The MS supply nozzles and the FW supply nozzles are also part of the SGS. Each SG has a pair of FW and MS supply nozzles. The MS supply nozzles are integral to the steam plenum access ports and the FW supply nozzles are integral to the feed plenum access ports as shown in Figure 5.4-3 and Figure 5.4-4, respectively. The primary reactor coolant circulates outside the SG tubes with steam formation occurring inside the SG tubes.

Each SG tube is a helix with bends at each end that transition from the helix to a straight configuration at the entry to the tubesheets as shown in Figure 5.4-1. The helical tubes are seamless with no intermediate welds. The helical tubes terminate at the feed and steam plenum tubesheets, where the tubes are secured to the tubesheet by expansion and are welded to the tubesheet on the secondary side. Crevices are minimized among the SG tubes, the tube supports, and tubesheets to limit the buildup of corrosion products. There are minimal quantities of corrosion products because the SG tube-to-tubesheet contact is within the primary coolant environment. Expansion of the tube within the tubesheet bore minimizes crevices depths and mitigates exposure of the low alloy steel tubesheet to corrosion products. Expansion of each tube is completed at both the steam and feed plenum tubesheets.

The SG has no secondary side crevices that could concentrate corrosion products or impurities accumulated during the steam generation process. In the once-through SG design there is no bulk reservoir of water at the inlet plena where the accumulation or concentration of corrosion products could occur. There is no SG blowdown to remove deposits in the once-through SG design based on the geometry of the design and flow characteristics that do not allow accumulation of corrosion products within a fluid reservoir. Therefore, a blowdown system would only serve to divert FW flow from the SG and would not remove corrosion products or impurities. Based on these factors, there is no SG blowdown system included in the NPM design.

Secondary coolant impurities and corrosion products may deposit directly on the interior tube surfaces as a scale or film, or be removed from the SG tubes by carryover. The concentration of corrosion products and impurities is low based on selection of materials for the condensate system and chemistry control requirements. Periodic cleaning performed during outage periods removes buildup of corrosion product films on the secondary surfaces of the SG tubes. Proven chemical or mechanical cleaning methods and techniques in use in the existing pressurized water reactor (PWR) fleet inform the buildup removal.

Secondary side SG surfaces are corrosion resistant, either nickel alloy, stainless steel, or stainless steel clad, which removes the concern for degradation of SG components by cleaning solutions. Connecting an appropriate system directly to the MS and FW disconnect flanges during an outage accomplishes cleaning of the SG tubes.

Heated primary coolant from the reactor core exits the riser and flows down the outer annulus across the SG tubes where heat is transferred to secondary coolant inside the SG tubes. Small flow paths in the upper and lower risers permit a small amount of reactor coolant to bypass the top of the riser and flow into the SG tube bundle region. These flow paths ensure sufficient boron mixing in the reactor coolant during DHRS-driven conditions where the riser is not submerged following non-loss-of-coolant accident (LOCA) transients. The primary coolant continues to flow down through the annular downcomer below the SG tubes into the lower reactor vessel plenum, where it reenters the reactor core. Further discussion of the RCS is in Section 5.1, RCS and Connecting Systems, and the RCS loop flow is in Figure 5.1-3.

The SGs deliver superheated steam with moisture content no greater than 0.10 percent by weight during full-power operating conditions.

Piping from the condensate and FW system located outside the Reactor Building (RXB) supplies FW to the SGs. The FW lines penetrate the containment vessel (CNV) wall and then into the FW plena. Feedwater flows from each feed plenum access port into the bottom of the SG tube columns, through the tubes, upward and around the outside of the upper riser assembly, where it converts to steam by the heat transferred from the reactor coolant flowing outside the SG tubes.

The steam plena collect steam from the top of the SG tube columns and direct the steam through the steam nozzles. Steam flows through the SG piping, through nozzles penetrating the containment, and then to the main steam system (MSS) and power conversion systems located outside the RXB.

The total SGS heat transfer area provided in Table 5.4-2 comprises the outer surface area of the full length of tubes from the primary face of the feed plenums to the primary face of the steam plenums. The total heat transfer area of each of the two independent SGs includes margin for tube plugging that reduces the heat transfer area by, at most, 10 percent.

Table 5.4-2 provides a fouling factor used for calculating end-of-life heat transfer performance.

The SG design data are in Table 5.4-2. Transient conditions applicable to the SGs are in Section 3.9.1, Special Topics for Mechanical Components; design stress limits, loads, and load combinations applicable to the SGs are in Section 3.9.3, ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures; and piping stress limits, loads, and load combinations are in Section 3.12, ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and Associated Supports.

Main steam isolation valves (MSIVs) and feedwater isolation valves (FWIVs) are outside the NPM on the MS and FW piping, respectively, on top of the CNV at the top support structure platform. A detailed discussion of the isolation functions of the valves is in Section 6.2.4, Containment Isolation System.

The DHRS forms a closed-loop connection between the steam lines and the FW lines inside the containment isolation boundary formed by the MSIVs and FWIVs. During normal operations, the DHRS is isolated from steam flow by the DHRS actuation valves (DHRSAV). A detailed description of the DHRS is in Section 5.4.3, Decay Heat Removal System.

The design of the SGs minimizes tube corrosion, minimizes tube vibration and wear, and enhances overall reliability. The design includes provisions to reduce the potential for tube damage due to loose parts.

The SG design permits periodic inspection and testing of critical areas and features to assess their structural and pressure boundary integrity when the NPM is disassembled for refueling as shown in Figure 5.4-2. The internal surface of SG tubes is accessible over their entire length for application of nondestructive examination methods and techniques that are capable of finding the types of degradation that may occur over the life of the tubes. Individual SG tubes may be plugged to prevent adverse interaction with non-plugged tubes. Access to the internal (secondary) sides of tubesheets affords opportunity for inspection, and for removal of foreign objects. Figure 5.4-3 and Figure 5.4-4 contain illustrations of the steam and feed plena inspection ports.

Classifications and Quality Group designations for design, fabrication, construction and testing of SGS components that form part of the RCPB are in Table 3.2-2. Chapter 3 provides detailed information regarding the design basis and qualification of structures, systems, and components based on these classifications and designations. Figure 6.6-1 shows the ASME BPVC Section III, Class 1 and 2 boundaries for the SGS.

Steam Generator Tube Supports and Steam Generator Supports

The seamless helical coil SG tubing is supported by a series of austenitic stainless steel tube support assemblies. The geometric design and materials utilized facilitate fluid flow while minimizing the potential for the generation of corrosive products and buildup. The material and geometry choice precludes two of the most significant historical contributors to tube degradation by the tube supports.

The tube support assemblies are between each column of helical tubes as shown in Figure 5.4-6. The tube support structure is within the primary coolant environment; therefore, no ingress path exists for general corrosion products from the secondary system to deposit on the primary side of the SG. Optimization of the circumferential spacing of the tube supports provides the minimum possible tube free span lengths while still accommodating the transition of the tubes to the steam and FW tube sheets.

The SG supports and SG tube supports provide support for vibration and seismic loads. As shown in Figure 5.4-5, the SG tube support assemblies attach to upper SG supports welded to the integral steam baffle plate and inner surface of the RPV, and also interface with lower SG supports welded to the inner surface of the RPV. The use of eight sets of tube support assemblies limit the unsupported tube lengths, which ensures the SG tubes do not experience unacceptable flow-induced vibration (FIV).

As shown in Figure 5.4-5, the lower SG supports permit thermal growth and provide lateral support of the tube supports.

Inlet Flow Restrictors

The SG inlet flow restrictors are installed in each SG tube at the FW plena locations. Each SG inlet flow restrictor is individually installed and seats against the secondary face of the FW plenum tubesheet and extends into a portion of the hydraulically expanded SG tube within the FW plenum tubesheet. A SG inlet flow restrictor consists of a mandrel, an expanding collet, a flanged sleeve, a locking plate and a hex nut. After the flow restrictor is inserted into the SG tube, the metallic collet on each SG inlet flow restrictor is expanded to seal with the inner diameter of the SG tube. The bearing contact resistance between the expanded collet and tube prevents bypass flow around the flow restrictor as well as the frictional interaction for securing the flow restrictor within the FW plenum.

Secondary side water flows from the FW plenum through a center-flow orifice in the mandrel. The flanged sleeve allows secondary side water from the feed plenum to enter into the space between the sleeve and SG tube to FW plenum tubesheet weld. This secondary side water provides a thermal barrier to the tube-to-tubesheet weld, helping mitigate rapid temperature changes at the weld. The devices permit in service tube inspections, cleaning, tube plugging, repairs and maintenance activities via installation and removal as needed.

Thermal Relief Valves

A single thermal relief valve is on each FW line upstream of the tee that supplies the feed plenums (Figure 5.4-7). The thermal relief valves provide overpressure protection during shutdown conditions for the secondary side of the SGs, FW and steam piping inside containment, and the DHRS when the secondary system is water solid for SG flushing operations and the containment isolation system is actuated. The trapped fluid is subject to heating by core decay heat. The thermal relief valves are spring operated relief valves that vent directly to containment. The thermal relief valves are classified as Seismic Category I and Quality Group B (ASME Class 2), and designed, fabricated, constructed, tested and inspected in accordance with Section III of the ASME BPVC. The pressure-retaining materials of thermal relief valves are in accordance with the materials identified in Table 6.1-3.

The thermal relief valves protect the secondary system components during off-normal conditions. The system design pressure and the RSVs provide overpressure protection during normal operation. Section 5.2.2, Overpressure Protection, contains details.

Feedwater Plenum Drain Valves

Manual valves allow draining the FW plenum before cover removal to facilitate outage maintenance and testing. The valves are for maintenance and are normally closed.

Compatibility of Steam Generator Tubing with Primary and Secondary Coolant

Control of the chemistry of the primary and secondary water is in accordance with industry guidelines suitably modified to address the unique NPM design and to ensure compatibility with the primary and secondary coolant. Section 5.2.3, RCPB Materials, describes the compatibility aspects of the reactor coolant chemistry that provide corrosion protection for stainless steels and nickel alloys, including SG components exposed to the reactor coolant. Section 6.1.1, Metallic Materials, describes the compatibility aspects of the secondary coolant chemistry that provide corrosion protection for stainless steels and nickel alloys, including the SG components exposed to the secondary system coolant, and Section 10.3.5, Water Chemistry, describes the secondary water quality control program. The SGs are flushed during NPM startup and shutdown to establish initial chemistry for power operations or refueling.

Section 11.1.2, Design Basis Secondary Coolant Activity, addresses estimated radioactivity design limits for the secondary side of the SGs during normal operation. The radiological effects associated with an SG tube failure are in Section 15.0.3, Design Basis Accident Radiological Consequences.

5.4.1.3 Performance Evaluation

The RCS natural circulation flow loop is entirely within the RPV, thereby eliminating distinct RCS piping loops and the associated potential for a large pipe break (i.e., large break LOCA) event. This design, combined with the intertwined SGs tube bundle configuration, eliminates the potential for asymmetric core cooling and temperatures as a result of a loss of a single SG function. Isolation or other loss-of-heat transfer capability by either of the two intertwined SGs does not introduce asymmetrical cooling in the reactor coolant system because the tube configuration of the remaining functional SG continues to provide symmetrical heat removal from the reactor coolant flowing in the downcomer of the reactor vessel.

The primary coolant system operates at a higher pressure than the secondary system, resulting in the SG tubes being in compression. This configuration reduces the likelihood of a tube failure and eliminates the potential for pipe whip due to tube-side jetting.

Feedwater enters the SG tubes at their lowest point. As it rises through the tubes, it undergoes a phase change and heats above saturation temperature before exiting the SG tubes as superheated steam. The configuration keeps the steam-water interface fluid, and the superheated steam at the top of the tubes separated from the subcooled liquid at their bottoms. This configuration minimizes the hydraulic instabilities that could introduce potential sources of water hammer.

<u>Stability Performance</u>

Flow instabilities, such as density wave oscillation, may arise in individual SG tubes because of fluid brought to boiling conditions as it travels up the tubes. Inlet flow restrictors at the FW inlet plenum interface provide the necessary pressure drop to preclude unacceptable secondary flow instabilities. Acceptable instabilities are tube mass flow fluctuations that do not cause reactor power oscillations that could exceed fuel design limits, and that result in applicable ASME BPVC criteria being met.

Stability analyses are documented in TR-0516-49417, Evaluation Methodology for Stability Analysis of the NuScale Power Module (Reference 5.4-9). The stability analysis documented in Appendix A of Reference 5.4-9 shows that the main effect of density waves in the tubes of the helical coil SGs is a small reduction in the effective heat transfer coefficient between the two sides of the SG. The unstable flow oscillations impact on heat transfer in individual tubes does not affect the overall heat transfer to the primary side because the flow oscillations in the tubes are not in-phase and thus their individual effects cancel out. Significant primary flow oscillations are not excited by the instabilities in the SG tubes

Analyses regarding the susceptibility of the NPM to develop DWO conditions use the approach documented in Appendix B of TR-131981-P, Methodology for the Determination of the Onset of Density Wave Oscillations (DWO), Reference 5.4-11. Results show that the combination of operating conditions and inlet flow restrictor design allow for margin to DWO onset at all nominal power levels from 20 percent to 100 percent power, which is the power generation range for turbine operation. While DWO may occur during limited operational times at low power levels, the SG and inlet flow restrictor design assures that DWO transient conditions are acceptable to meet applicable ASME BPVC criteria.

Comprehensive Vibration Assessment Program Performance

The results of the Comprehensive Vibration Assessment Program screening and performance analysis for the SG is in technical report TR-121353, "NuScale Comprehensive Vibration Assessment Program Analysis Technical Report," (Reference 5.4-10).

Section 17.4, Reliability Assurance Program, describes the reliability assurance plan used for SG reliability evaluation; the guidance in Chapter 19, Probabilistic Risk Assessment and Severe Accident Evaluation, describes the determination of SG risk significance.

5.4.1.3.1 Allowable Tube Wall Thinning under Accident Conditions

The SG tubes have a nominal wall thickness of 0.050 in. The design adds a lifetime degradation allowance of 0.010 in. to the calculated ASME BPVC minimum SG tube wall thickness per NB-3121 (Reference 5.4-3). This degradation allowance provides margin for potential in-service tube degradation mechanisms (e.g., general corrosion, erosion, wear). This degradation allowance also includes margin for SG tube wall thickness manufacturing tolerances, including wall thinning due to tube bending. The SG tubes construction meets the rules of ASME BPVC, Section III, Subsection NB.

5.4.1.4 Tests and Inspections

The SGs testing and inspection ensures conformance with the design requirements described in Section 5.2.4, RCPB ISI and Testing. Equipment requiring inspection or repair is in an accessible position to minimize time and radiation exposure during refueling and maintenance outages.

The SG tube inspections and testing meet requirements of the SG program. Performance of a preservice volumetric examination on the entire length of the SG tubing meets specifications in Table IWB-2500-1 (B-Q). A preservice eddy current test meets Electric Power Research Institute (EPRI) 1013706 (Reference 5.4-2).

Preservice examinations performed in accordance with the ASME BPVC, Section III, Subsubarticle NB-5280 and Section XI, Subarticle IWB-2200 (Reference 5.4-5) use examination methods of ASME BPVC Section V, except as modified by Section III, Paragraph NB-5111. These preservice examinations include essentially 100 percent of the pressure boundary welds.

A preservice volumetric, full-length preservice inspection of essentially 100 percent of the tubing in each SG is performed. The length of the tube extends from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet welds are not part of the tube. The preservice inspection is performed after tube installation and shop or field primary-side hydrostatic testing and before initial power operation to provide a definitive baseline record, against which future ISI can be compared. Tubes with flaws that exceed 40 percent of the nominal tube wall thickness are plugged. Tubes with flaws that could potentially compromise tube integrity before the performance of the first ISI, and tubes with indications that could affect future inspectability of the tube, are also plugged. The volumetric technique used for the preservice examination is capable of detecting the types of preservice flaws that may be present in the tubes and permits comparisons to the results of the ISI expected to be performed to satisfy the SG tube inspection requirements in accordance with the plant technical specifications.

As discussed above, the operational inservice testing and inspection programs described in Section 5.2.4, RCPB ISI and Testing, and the SG program described in Section 5.4.1.6, Steam Generator Program, provide testing and inspection requirements following initial plant startup. Inservice inspection and testing of the SGS steam and feedwater piping is described in Section 6.6.

5.4.1.5 Steam Generator Materials

Selection and fabrication of pressure boundary materials used in the SGs and associated components are in accordance with the requirements of ASME BPVC Section III and Section II as described in Section 5.2.3, RCPB Materials, and the materials used in the fabrication of the SGs are in Table 5.2-3.

The RCPB materials used in the SGS are Quality Group A and their design, fabrication, construction, tests, and inspections conform to Class 1 in accordance with the ASME BPVC and the applicable conditions promulgated in 10 CFR 50.55a(b). The SGS materials forming the RCPB, including weld materials, conform to fabrication, construction, and testing requirements of ASME BPVC, Section III, Subsection NB (Reference 5.4-3). The SG tubes are SB-163 Alloy 690 (UNS N06690) and all SGS materials forming the RCPB are in accordance with ASME BPVC Section II, and meet the requirements of Section III, Article NB-2000. Surfaces of pressure retaining parts of the SGs, including weld filler materials and bolting material, are corrosion-resistant materials, such as stainless steel or nickel-based alloy. The SGs use materials with a proven history in light water reactor environments.

The FW and MS piping from the CNTS to the plenum nozzles, including the thermal relief valves, are Quality Group B and their design, fabrication, construction, tests, and inspections conform to Class 2 in accordance with the ASME BPVC and the applicable conditions promulgated in 10 CFR 50.55a(d). The FW and MS piping, thermal relief valves, including weld materials, conform to fabrication, construction, and testing requirements of ASME BPVC, Section III, Subsection NC (Reference 5.4-3). The materials selected for fabrication conform to the applicable material specifications provided in ASME BPVC, Section II and meet the requirements of Section III, Article NC-2000.

The integral steam plenum, integral steam plenum caps, feed plenum access ports, and feed plenum access port covers are Quality Group A and their design, fabrication, construction, testing, and inspections conform to Class 1 in accordance with the ASME BPVC and the applicable conditions promulgated in 10 CFR 50.55a(b). The steam plenum access ports and steam plenum access port covers are Quality Group B, and inspection conforms to Class 2 in accordance with the applicable conditions promulgated in 10 CFR 50.55a(b). The Class 1 feed plenum components conform to fabrication, construction, and testing requirements of ASME BPVC, Section III, Subsection NB (Reference 5.4-5). The steam plenum components are classified as Class 2 but conform to fabrication, construction, and testing requirements of ASME BPVC, Section III, Subsection NB.

The materials and applicable specifications of the MS and FW piping, associated fittings, steam and feed plenum components, and fasteners are in Table 5.4-3.

Welding of the RCPB portions of the SGS with the steam access port components follows procedures qualified in accordance with the applicable requirements of ASME BPVC Section III, Subarticle NB-4300 and Section IX. Welding of the secondary side portions of the SGS constructed to Class 2 follows procedures qualified in accordance with the applicable requirements of ASME BPVC, Section III, Subarticle NC-4300 and Section IX.

The secondary side surfaces of the steam plenum tubesheet, and feed plenum tubesheet use alloy 52/152 cladding. The remaining inside and outside surfaces of the steam plenum and feedwater plenum are Alloy 690 material or low alloy steel clad with austenitic stainless steel.

The SG weld filler metals are in Table 5.4-3 and are in accordance with ASME BPVC Section II, Part C.

The SG supports and SG tube supports are designated as ASME BPVC, Section III, Subsection NG “Internal Structures.” The design, fabrication, construction, and testing of the SG supports and SG tube supports, including weld materials does not adversely affect the integrity of the core support structures.

The SG piping structural supports, including weld materials, conform to fabrication and construction requirements of ASME BPVC, Section III, Subsection NF. The SG piping structural support materials are in Table 5.4-3.

The SG inlet flow restrictors are non-structural attachments to the RPV. The SG inlet flow restrictors design, fabrication, construction, testing, and inspections conform with the ASME BPVC, Section III, Subsection NC.

Section 5.2.3, RCPB Materials, contains additional description of material compatibility, fabrication and process controls, and welding controls related to the ASME Class 1 components. Section 5.2.3.4.2, Cleaning and Contamination Protection Procedures, describes cleaning and cleanliness controls for the SGs. Section 6.1, Engineered Safety Feature Materials, has additional description of material compatibility, fabrication and process controls, and welding controls related to the ASME Class 2 components.

Section 3.13 describes threaded fasteners.

5.4.1.6 Steam Generator Program

The SG program monitors the performance and condition of the SGs to ensure they are capable of performing their intended functions. The program provides monitoring and management of tube degradation and degradation precursors that permit preventative and corrective actions to be taken in a timely manner, if needed. The SG program is based on NEI 97-06 (Reference 5.4-1) and Regulatory Guide (RG) 1.121 and is documented in the technical specifications. The program implements applicable portions of Section XI of the ASME BPVC and specifically addresses 10 CFR 50.55a(b)(2)(iii). Appendix B to 10 CFR 50 applies to implementation of the SG program.

Historically, significant SG tube degradation in the operating PWR SG fleet was due to various corrosion mechanisms, including wastage and both primary and secondary side stress corrosion cracking. These corrosion mechanisms relate to materials selection, plant chemistry control, and control of the ingress of impurities and corrosion products to the SGs. In the design, detrimental SG corrosion mitigation is achieved by use of SB-163 UNS N06690 SG tubing, application of EPRI primary and secondary plant chemistry control guidelines, and design of condensate systems (including extensive use of polishing resin beds and improved materials).

In addition to chemistry and materials considerations, there are two areas where the design reduces SG tube degradation risk. The SG tube wall thickness is thicker than existing designs (Table 5.4-2), based on incorporation of a substantial degradation allowance (additional tube wall thickness above minimum required for design) as discussed in Section 5.4.1.2, System Design. The NPM reactor coolant flowrates are also lower than the flowrates across the SG tubes in PWR recirculating SGs as discussed in Section 5.1, RCS and Connecting Systems. This low flow rate reduces the flow energy available to cause FIV wear degradation of SG tubes. Based on the additional tube wall margin and the additional margin against FIV turbulent buffeting wear (the most likely SG tube degradation mechanism), application of the existing PWR SG Program requirements to the design is appropriate.

For SGs in the PWR fleet with SB-163 UNS N06690 SG tubing, the only observed degradation has been wear as a result of FIV (tube-to-tube or tube-to-support plate) or wear due to foreign objects. With respect to the risk of introduction of foreign objects, the NPM is at no greater risk than existing designs; therefore, the design does not warrant deviations from existing SG program guidelines. From the standpoint of SG tube design, the two significant differences between the SG design and current large PWR designs is the helical shape of the SG tubing and the SG tube support structure. The helical shape of the SG tubing itself does not represent risk of degradation based on the minimum bend radius of the helical tubing being within the historical experience base of PWR SG designs. Prototypic testing of the SG tube supports validates acceptable performance (including wear) of the SG tube support design. Implementation of a typical SG program is appropriate based on evaluation of the design of the SG tube supports.

5.4.1.6.1 Degradation Assessment

A degradation assessment of the NPM SG identifies several potential degradation mechanisms. Wear is the most likely degradation mechanism, and there is the potential for several secondary side corrosion mechanisms, including under deposit pitting and intergranular attack based on the once-through design with secondary boiling occurring inside the tubes. The estimated growth rates for these potential defects is sufficiently low that the SG tube plugging criterion for the SG is a 40 percent through wall defect. Operational SG tube integrity is ensured by implementing tube plugging criteria, implementing elements of the SG program, and implementing the SG inspections.

COL Item 5.4-1: An applicant that references the NuScale Power Plant US460 standard design will develop and implement a Steam Generator Program for periodic monitoring of the degradation of steam generator components to ensure that steam generator tube integrity is maintained. The Steam Generator Program will be based on the latest revision of Nuclear Energy Institute NEI 97-06, "Steam Generator Program Guidelines," and applicable Electric Power Research Institute steam generator guidelines at the time of the application. The elements of the program will include: assessment of degradation, tube inspection requirements, tube integrity assessment, tube plugging, primary-to-secondary leakage monitoring, shell side integrity assessment, primary and secondary side water chemistry control, foreign material exclusion, loose parts management, contractor oversight, self-assessment, and reporting.

5.4.2 Reactor Coolant System Piping

5.4.2.1 Design Basis

Pressure-retaining portions of piping that penetrate the RCS form, in part, the RCPB as defined in 10 CFR 50.2 and include the PZR spray supply, RCS injection, RCS discharge, and RPV high-point degasification piping.

5.4.2.2 Design Description

Section 6.2, Containment Systems, describes how each of the RCS lines enter containment through nozzle safe ends on the containment upper head and contain two containment isolation valves mounted on the outside of the containment.

A single PZR spray supply line enters through the containment head. This line branches inside containment into two PZR spray supply lines, each welded to a nozzle safe end on the RPV upper head with a corresponding spray nozzle inside the RPV near the top of the PZR space.

The RPV high-point degasification line is a single line that connects the containment upper head to a nozzle safe end on the RPV upper head.

The RCS injection line connects the containment upper head to a nozzle safe end on the side of the RPV. Inside the RPV, the line continues from the RPV wall, through the lower portion of the upper riser assembly and terminates near the center of the riser. Reactor coolant injection flow enters in the central riser above the reactor core. The RCS injection line also contains a branch connection to the ECCS reset valves.

The RCS discharge line connects the containment upper head to a nozzle safe end on the side of the RPV at an elevation just below the SGs. This location is selected above the core to reduce the potential that the RPV water level drains below the top of the core in the event of a penetration failure. This line takes suction from the annular region between the RPV wall and the riser.

Class 1 lines larger than three-fourths in. nominal pipe size have no socket welds, and piping less than or equal to three-fourths in. nominal pipe size with socket welds conforms to 10 CFR 50.55a(b)(1)(ii). Socket weld fittings conform to ASME B16.11 (Reference 5.4-8).

Figure 6.6-1 depicts the RCS piping from the CNV upper head to the respective penetrations on the RPV.

5.4.2.3 Performance Evaluation

Section 3.9, Mechanical Systems and Components, Section 3.12, ASME Code Class 1, 2 and 3 Piping, and Section 5.2, Integrity of Reactor Coolant Boundary, provide information regarding the RCS piping criteria, methods, and materials, and include the design, fabrication, and operational provisions to control those factors that contribute to stress-corrosion cracking. The RCS piping supports the functional aspects of the chemical volume and control system (CVCS) as summarized in Section 9.3.4.

5.4.2.4 Tests and Inspections

Section 5.2.4, RCPB ISI and Testing, summarizes preservice and ISI requirements associated with ASME Class 1 components, which include the RCS piping.

5.4.2.5 Reactor Coolant System Piping Materials

Descriptions of the RCPB and materials associated with the RCS piping are in Section 5.2, Integrity of Reactor Coolant Boundary.

Section 5.2.3, RCPB Materials, and Section 5.2.4, RCPB ISI and Testing, have additional descriptions of material compatibility, fabrication and process controls, welding controls, and inspections related to the ASME Class 1 components.

5.4.3 Decay Heat Removal System

5.4.3.1 Design Bases

The DHRS provides cooling for design basis events when normal secondary-side cooling is unavailable or otherwise not utilized. The DHRS removes post-reactor trip residual and core decay heat from operating conditions and transitions the NPM to safe shutdown conditions without reliance on electrical power or operator action.

The safety-related DHRS function is an engineered safety feature of the NPM design. Evaluation of reliability of the DHRS uses the reliability assurance program described in Section 17.4, Reliability Assurance Program, and risk significance determination uses the guidance described in Chapter 19, Probabilistic Risk Assessment and Severe Accident Evaluation.

The DHRS design ensures that there is passive cooling of the RCS after an initiating event without challenging the RCPB integrity or uncovering the core.

The DHRS heat removal function does not rely on actuating the ECCS. Any ECCS actuation after a DHRS actuation allows continued residual heat removal by both systems from the reactor core as described in Section 5.4.3.3, DHRS Performance Evaluation.

Design Requirements

General Design Criteria (GDC) 1, 2, and 4: The DHRS is Quality Group B and Class 2; design, fabrication, construction, testing, and inspections are in accordance with Section III of the ASME BPVC and in accordance with the Quality Assurance Program described in Chapter 17. The DHRS withstands the effects of natural phenomena without loss of capability to perform its safety function. The DHRS accommodates the effects of, and is compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. The design of the RXB structure, NPM operating bays, and location of the NPM within the operating bays provides protection from possible sources of externally or internally generated missiles. Section 3.6.2, Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping, describes protection of the DHRS from the external dynamic effects of pipe breaks.

General Design Criterion 5: The DHRS does not share any active or passive components among individual NPMs necessary for performance of the DHRS safety functions. The NPMs share the reactor pool as the ultimate heat sink for removal of decay heat from the DHRS passive condensers. Chapters 1 and 3 describe the shared RXB and other structures, and Section 9.2.5, Ultimate Heat Sink, describes the reactor pool. The DHRS active components fail-safe on a loss of power. Therefore, shared power supplies among NPMs do not impact the capability of performing the DHRS safety functions.

General Design Criterion 14: The DHRS connects to the secondary system and does not directly interface with the RCPB. Section 5.4.1 describes the SGs, and Section 6.2.4, Containment Isolation System, describes the CNTS components coupling the DHRS to the SGs. There are no other interfaces or shared components between the DHRS and the RCPB.

Principal Design Criterion (PDC) 19: The DHRS initiates from the control room and is capable of safe shutdown of the reactor. The DHRS can also initiate from outside the main control room in the module protection system (MPS) equipment rooms within the RXB.

PDC 34 and PDC 44: The DHRS is a passive design that utilizes two-phase natural circulation flow from the SGs to dissipate residual and decay core heat to the reactor pool. The DHRS consists of two independent trains each capable of performing the system safety function in the event of a single failure. Stored energy devices cause the DHRSAVs to fail open (safety related position) when electrical power is interrupted to the valves. Therefore, system function does not require electrical power. The reactor pool performs the function of the ultimate heat sink by removing heat from systems, structures, and components under normal operating and accident conditions. Section 9.2.5, Ultimate Heat Sink, contains further details on the reactor pool. Section 3.1.4, Fluid Systems, contains a discussion of PDC 34 and PDC 44.

GDC 54 and GDC 57: The DHRS is a passive closed system connected directly to the CNV main steam safe-ends and FW piping between the MSS and FWS isolation valves and the RPV. The closed-loop piping of DHRS outside the containment connects directly to the closed-loop SGS within the RPV providing dual passive barriers between the RCS and the reactor pool outside the NPM. The DHRSAVs prevent system flow within the closed DHRS loop when the system is not in operation. Breaches of this piping system outside containment are not credible because the system is designed and constructed with a system design pressure and temperature equivalent to that of the RPV, designed to Class 2 requirements in accordance with ASME BPVC, Section III, and meets the applicable criteria of NRC Branch Technical Position 3-4, Revision 3, as described in Section 3.6.2, Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping. As a result, leakage detection and isolation capabilities of this piping system from containment are not important to safety. Section 3.1.5, Reactor Containment, and Section 6.2.4, Containment Isolation System, provide additional discussion regarding conformance with GDC 54. The design supports an exemption from GDC 57.

10 CFR 50.34(f)(2)(xxvi); NUREG-0737 Task Action Plan Item III.D.1.1: The DHRS has adequate leakage detection and control processes to minimize potential exposure to workers and the public and to provide reasonable assurance that the DHRS is available to perform its intended functions. The DHRS does not connect directly to the RCS. Interface with the RCS is via the SGs, which are subject to the design, inspection, and testing controls described in Section 5.2.4, RCPB ISI and Testing, and Section 5.4.1, Steam Generators. During normal operations, identification and resolution of leakage from the RCS into the SGs conforms to requirements limiting RCS leakage and the SG program. The requirements for primary-to-secondary leakage monitoring and the SG program are in the plant technical specifications.

10 CFR 50.62(c)(1): Anticipated transient without scram events do not impact DHRS functions. As discussed in Section 15.8, Anticipated Transients without Scram, the NuScale design supports an exemption from 10 CFR 50.62(c)(1). Section 19.2, Severe Accident Evaluation, provides additional information on the NuScale Power Plant response to anticipated transient without scram events.

10 CFR 50.63: Upon a loss of normal alternating current power with no backup power supply available, the DHRS removes decay heat at a rate sufficient to maintain adequate core cooling during the 72-hour station blackout coping duration. Discussion of the station blackout coping duration is in Section 8.4, Station Blackout.

10 CFR 20.1406: The DHRS is independent from the RCS. Therefore, radioactive contamination in the DHRS originates indirectly from the FWS and MSS. The system designs and programs that limit radioactive contamination of the facility from the FWS and MSS also minimize, to the extent practicable, the generation of liquid and gaseous radioactive waste in and by the DHRS. A welded design (with the exception of small diameter instrument connections) and provisions for leakage detection minimizes potential contamination by the DHRS.

A discussion of the facility design and procedures related to minimizing the generation of radioactive waste and the minimization of contamination to the facility and environment during operation and plant decommissioning is in Section 12.3.6, Minimization of Contamination and Radioactive Waste Generation.

5.4.3.2 System Design

One train of DHRS is aligned to one SGS train. The DHRS piping connects to the MS and FW lines specific to the associated SG. The DHRS steam inlet piping connects with the CNV main steam safe end upstream of the associated MSIV. The DHRS piping routes to two DHRSAVs arranged in parallel. Each train has an orifice located on the common line before the actuation valves to moderate flow during operation. The piping re-joins after the actuation valves and routes down the outside of the CNV to the train-specific DHRS passive condenser. The outlet of the DHRS passive condenser routes to the FW line supplying the associated SG, joining the FW line downstream of the FWIV. Figure 5.4-8 provides a simplified diagram of the DHRS illustrating the operational flowpath and major system components. Table 5.4-4 identifies the component materials used in the DHRS design. Table 5.4-5 provides a summary of DHRS design data.

Before power operations, the FW pumps fill the FW lines, SGs, and DHRS. Maintaining filled and pressurized DHRS passive condensers and piping occurs by connection to the FWS on the DHRS outlet line to the FW piping inside containment.

During normal power operations, the DHRS is in a standby configuration with each train of DHRS isolated from the associated MS lines by the closed DHRSAVs. These four valves, two in parallel on each train, remain closed.

Automatic actuation of the DHRS occurs using the MPS and has the capability for manual initiation from the main control room. The DHRS actuation signal opens the DHRSAVs for both trains of DHRS and closes the secondary system isolation valves (FWIV, feedwater regulating valve (FWRV), MSIV, and secondary MSIV). The MPS automatically actuates the DHRS. Manual controls for initiation of DHRS are also provided in the main control room. Details of the DHRS actuation design including redundancy, reliability, diversity, signals, interlocks, analytical limits, and functional logic are provided in Chapter 7.

Upon actuation, the MSIVs and FWIVs close, and the DHRSAVs open. The DHRSAVs open upon interruption of control power because of control system actuation or loss of power. The DHRSAVs use the same hydraulic system used for the CIVs. Section 6.2.4, Containment Isolation System, contains a discussion of hydraulic system operation. Actuation permits the water column in the DHRS piping to drain into the FWS piping and plenum, and steam to flow from the SG into the DHRS piping and the DHRS passive condenser. Steam condenses in the passive condenser by transfer of heat to the reactor pool. This process results in a flow of condensate from the passive condenser to the associated FW line and into the associated SG. Figure 5.4-7 depicts the system layout and interface with the MS and FW piping and SGs.

The DHRS function depends on the closure of the associated safety-related MSIVs and FWIVs. In the event an MSIV fails to close, the backup MSIV provides isolation for the DHRS loop. The FWRV provides isolation in the case where the FWIV fails to close. These closures isolate the SGs and associated DHRS loops from the MSS and FWS, ensuring adequate water inventory in the passive closed loop configuration. Section 6.2.4, Containment Isolation System, and Chapter 7, Instrumentation and Controls, describe the MSIV and FWIV functions, including their actuation. The SGs are described in Section 5.4.1. Chapter 10, Steam and Power Conversion System, describes the MS and FW piping.

Natural circulation resulting from the density differences between the steam and condensate portions of the DHRS and associated SG drive DHRS flow. The

DHRS passive condensers are at a higher elevation relative to the SGs to promote natural circulation flow to the SGs. The RCS temperature and pressure sensors provide indication of normal DHRS operation. The RCS temperature and pressure decrease following a reactor trip and DHRS actuation, providing an indication that the DHRS is working normally. Relative elevation differences are in Figure 1.2-6. The passive cooling and boron transport model used for DHRS evaluation is described in the Extended Passive Cooling and Reactivity Control Methodology Topical Report (Reference 5.4-6).

The DHRS function depends on the presence of the reactor pool to remove heat from the DHRS passive condensers. Section 9.2.5, Ultimate Heat Sink, describes the safety-related ultimate heat sink provided by the reactor pool.

The DHRS is not in direct contact with, nor does it utilize, the reactor coolant other than to depend on heat transfer from the reactor core to the SGs to perform its function.

Actuation of the DHRS function does not require reduction of the RCS pressure and temperature because the DHRS utilizes the normally operating SGs as the interface with the RCS. There is no potential for interfacing system loss of coolant to occur during DHRS operations because there is no direct flow path between the RCS and the DHRS. Section 5.2.2 describes overpressure protection for the DHRS via a system design that does not exceed the ASME BPVC service limits during normal operation or during design basis accidents and transients, thereby precluding the need for low-pressure system interlocks or pressure relieving devices on the DHRS. Under normal operating conditions and pressure transients, internal pressure limits on the DHRS are not exceeded. The RSVs provide overpressure protection for DHRS internal pressure in the event of a SG tube failure coincident with the RCS pressure exceeding the design pressure of the RCS.

During shutdown conditions, thermal relief valves provide overpressure protection for the DHRS when the secondary system is water solid and the containment is isolated. Section 5.4.1, Steam Generators, contains further discussion of secondary system thermal relief valves.

Upon cooling to stable shutdown conditions, cooldown to cold conditions and long term decay heat removal occurs via conduction and convection through a flooded containment and CNV shell to the reactor pool. When RCS pressure decreases, the RVVs and RRVs open and promote circulation between the RCS and flooded containment. Section 6.3, Emergency Core Cooling System, describes operation of the RVVs and RRVs.

During NPM movement to and from the refueling area and during refueling, the DHRS provides no decay heat removal function. Conduction through the RPV and containment shell with the RVVs and RRVs open or direct contact with the reactor pool water during refueling provides residual and core decay heat removal during shutdown conditions.

The DHRSAVs, piping, and passive condensers are Quality Group B, the design conforms to Class 2 in accordance with Section III of the ASME BPVC, and remain operable following a design basis seismic event. The DHRS condenser construction is in accordance with ASME BPVC, Section III, Subsection NC (Reference 5.4-3). The DHRS supports design and fabrication conform to Class 2 in accordance with ASME BPVC, Section III, Subsection NF. Details of the classification designations and the scope of their applicability are in Chapter 3.

Welding of the DHRS utilizes procedures qualified in accordance with the applicable requirements of ASME BPVC, Section III, Subarticle NC-4300 or NF-4300 and Section IX.

The DHRS condenser, actuation valves, and DHRS piping are Seismic Category I components, designed to Quality Group B (ASME Class 2) requirements. The RXB structure protects the DHRS from natural phenomena. Seismic qualification of the DHRS instrumentation and control components is in accordance with Institute of Electrical and Electronic Engineers (IEEE) 344-2004, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," (Reference 5.4-4) as modified by the NRC staff position in RG 1.100.

A portion of the DHRS is submerged in the reactor pool and protected from internally generated missiles by the NPM operating bay walls. There are no credible sources of internally generated missiles in the area above the NPM as there is no rotating equipment in proximity to the NPM. Section 3.6.2, Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping, provides additional information on DHRS protection from pipe whip and internally generated missiles.

Section 3.9, Mechanical Systems and Components, and Section 3.12, ASME Code Class 1, 2, and 3 Piping Systems have a discussion of stress analyses associated with the FW, DHRS, and MS piping inside the containment that connects the DHRS to the SGs.

5.4.3.2.1 Components

Actuation Valves

Each DHRSAV is an automatically actuated 4-in. ball valve located on the outside of the NPM between the steam line connection and the upper header of the DHRS condenser. Each train contains two parallel actuation valves.

The DHRSAVs fully open within 30 seconds from receipt of a DHRS actuation signal and fully close within 30 seconds from receipt of a close signal when differential pressure between the FWS and MSS is 50 psid or less.

The DHRSAVs are designed such that when they are in the safety-related position they remain in that position without force applied by the actuator. This means that forces applied by differential pressure or flow on the obturator do not cause the valve to move; the friction forces in the valve maintain the valve in the safety-related position.

Passive Condenser

Each NPM has two DHRS condensers mounted to the outside surface of the CNV, submerged in the reactor pool, providing the heat transfer area necessary to condense steam as part of the two-phase DHRS loop.

Each condenser consists of upper and lower headers connected to a series of tubes that provide the heat transfer surface and form the pressure boundary of the heat exchanger. The inside of the headers and the tubes contain secondary system fluid and the outside of the headers and tubes are exposed to reactor pool fluid. The condenser inlet and outlet headers consist of pipes that, collectively, constitute the upper and lower distribution manifolds, which are welded to the heat exchanger piping elements.

Restriction Orifice

An orifice installed on the common DHRS steam line upstream of the tee for the actuation valves restricts the mass flow rate through the DHRS loop. The DHRS thermal-hydraulic performance analysis confirms that the size of the restriction orifice is adequate to provide consistent heat transfer.

5.4.3.2.2 Instrumentation and Controls

The DHRS instrumentation and controls (I&C) described below have main control room indication.

Level

Accumulation of noncondensible gas in the DHRS steam lines degrades DHRS performance. Level sensors detect the accumulation of noncondensible gas in the steam lines below the actuation valves.

Each train has four level transmitters, two located on each steam pipe. Sensors are at an elevation near the DHRSAVs, ensuring maintenance of the noncondensible gas limit and performance of the safety function of the associated DHRS train.

A low-level alarm in the control room alerts the operators if the DHRS is not filled with liquid water and, during DHRS operation, confirms that the DHRS piping drains, which provides indication of a successful actuation.

Steam Pressure

Pressure indication is on the DHRS steam piping section between the actuation valves and the steam line, and represents MS pressure. This pressure instrumentation provides a safety-related signal used for reactor trip, DHRS actuation and post-accident monitoring.

There are eight total steam pressure sensors per NPM, four per train located on the DHRS supply line.

During DHRS operation, steam pressure indication yields a pressure close to the saturation pressure at the RCS temperature. Higher pressure may indicate a SG tube failure, and lower pressure may indicate a secondary side break or leak.

Actuation Valve Position

The actuation valves have position indication as a means for verifying that the valve position matches the demanded position. The valve position indication is a Type D accident monitoring variable in accordance with IEEE 497-2016 as endorsed by RG 1.97.

During DHRS operation, valve position indication confirms successful actuation of the system.

Condensate Temperature

Each train of the DHRS has two temperature sensors in the lower header of the condenser.

During DHRS operation, condensate line temperature indication increases above the reactor pool temperature indicating that condensation and DHRS flow are occurring. Condensate temperature approaching the saturation temperature may be an indication of reduced water level in the DHRS condenser. Condensate temperature approaching reactor pool temperature may be an indication of a lack of DHRS circulation or an overfilled DHRS condenser.

Condensate Pressure

Each train has three pressure sensors in the lower header of the condenser.

During DHRS operation, condensate line pressure indication yields a pressure close to the saturation pressure at the RCS temperature. Consistent with steam pressure, higher pressure may indicate a SG tube failure and a lower pressure may indicate a DHRS break or leak.

Decay Heat Removal System Controls

The DHRS control system is limited to an on-or-off signal to the actuation valves with no ability for modulation. The DHRS actuates from the MPS, as discussed in Chapter 7.

5.4.3.3 Performance Evaluation

The DHRS provides the passive, safety-related, single active failure-proof, and redundant capability to cool the reactor core and coolant to safe shutdown conditions. Both liquid and vapor water are in the DHRS on system actuation. The total water mass remains constant during system operation because the DHRS is a closed system.

Two independent trains of passive cooling loops ensure reliability of the DHRS. Table 5.4-8 provides the failure modes and effects analysis for the DHRS.

The DHRS piping and passive condensers are around the exterior of the CNV and separated to reduce the potential for a single condition to affect both trains. Submergence of the passive condensers in the reactor pool and the module bay walls located between operating NPMs, as shown in Figure 1.2-5, and Figure 1.2-6, provides protection from adverse interactions with other facility equipment. The RXB crane is used to move NPMs to and from the refueling area as discussed in Section 9.1, Fuel Storage and Handling, provides protection from adverse interaction with an NPM being moved to and from the refueling location.

Section 9.1.4, Fuel Handling Equipment, describes refueling and maintenance operations conducted in the refueling area. The DHRS is not functional or available during refueling operations.

5.4.3.3.1 Water Hammer

Loading conditions due to water hammer in the DHRS or surrounding systems are included in the analysis of the DHRS.The operating conditions for the main FWS, MSS, and DHRS lines are conducive to water hammer events caused by

- high pressure discharge.
- fast valve closure.
- pump trip transients.

The FW piping operates at a much higher pressure than atmospheric pressure and operates in such a way that prevents column rejoining from occurring.

During normal operation, the FW line carries liquid water, and the MS line carries superheated steam. The DHRS piping contains liquid water below the closed DHRSAVs and some combination of liquid water and steam above the valves. The FW line contains an FWRV, and the FWIV. The MS line contains the MSIV and the backup MSIV. Additionally, the DHRSAVs open as a result of a DHRS actuation signal. Water or steam hammer by valve actuation is possible in these lines. Condensation-induced water hammer is mitigated by maintaining a small slope in DHRS piping.

The DHRS contains high-pressure fluid in piping surrounded by low pressure regions. A pipe break results in a discharge to a low pressure environment creating a pressure wave.

A FW pump trip could cause a sudden drop in line source pressure. Similarly, a turbine trip event could cause a sudden reduction in MS line flow.

5.4.3.3.2 System Noncondensible Gas

The DHRS, SGs, and secondary system piping do not include safety-related high-point vent capability. During normal operation, noncondensible gases continuously vent via the MSS. Accumulation of noncondensible gas may occur in the DHRS steam piping below the closed actuation valves when DHRS is not in service. Level sensors located below the actuation valves detect the presence of noncondensible gas to limit the volume of gas that can accumulate in the DHRS piping. The DHRS performance analysis evaluates a conservative mass of noncondensible gas based on the internal volume of the piping below the DHRSAVs and above the DHRS level sensor and assumed gas conditions. The analysis concluded that the design provides reasonable assurance that the DHRS functions in the presence of a limiting amount of noncondensible gases.

5.4.3.3.3 Flow-Induced Vibration

Section 3.9, Mechanical Systems and Components, describes the Comprehensive Vibration Assessment Program for the NPM and includes an assessment the DHRS components exposed to secondary side flow.

5.4.3.3.4 Thermal-Hydraulic Performance

As a two phase natural circulation system, DHRS performance is dependent on the following factors:

- RCS temperature: A higher RCS temperature provides a larger driving temperature difference and increases DHRS heat transfer.
- water inventory: Water level is high enough to ensure the heat transfer surfaces are wetted, but low enough to ensure adequate surface area in contact with a two-phase mixture for boiling and condensation to be effective.
- noncondensible gas: Accumulation of noncondensible gas in the DHRS condenser has the potential to impede condensation heat transfer.
- reactor pool water temperature: Pool water temperature affects the mode of heat transfer on the exterior of the DHRS condenser tubes.
- pressure losses: A restriction orifice in the DHRS steam piping limits the mass flow rate and heat removal, and dominates the DHRS loop pressure losses.
- driving head: The elevation difference between the bottom of the DHRS condenser and the bottom of the SG provides the DHRS loop driving head.

A thermal-hydraulic analysis, performed with NRELAP5, determines the impact of these above factors on DHRS heat transfer rate using a series of steady state and transient cases. Steady state cases characterize the effect of a single parameter variation on DHRS heat removal. Nominal transient cases show the DHRS cooling capability for initiating events at typical initial conditions. Off-nominal transients are bounding evaluations of the combined impact of several factors on DHRS heat removal capability.

The factors impacting DHRS heat removal evaluated in the off-nominal cases include: core power uncertainty, reactor pool temperature, valve actuation delays, valve stroke times, noncondensible gas volume, system leakage, SG level, SG fouling, SG tube plugging, DHRS condenser fouling, and number of operational DHRS trains. If applicable, these parameters are biased to produce either a high or low DHRS loop inventory.

Decay Heat Removal System Performance Analysis

The analysis evaluates the DHRS capability of removing heat over a range of DHRS loop inventories with the steady state model. Sensitivity cases indicate that the DHRS is insensitive to valve coefficient and orifice loss coefficient. However, DHRS performance is sensitive to DHRS inventory. Low inventory greatly reduces the heat transfer rate. Similarly, with a high inventory there is also a decline in performance.

Fouling of the heat transfer surfaces and SG tube plugging has a moderate effect on DHRS performance, decreasing the peak heat removal capability, and the presence of noncondensible gas has an impact on DHRS performance.

The presence of non-condensible gas has a small effect on total system performance. The decrease in heat removal due to noncondensible gas increases as the DHRS pressure decreases, because of the same mass of noncondensible gas fills a larger fraction of the gas space.

Assessment of the likelihood of noncondensible gas accumulating down to the level sensors in the DHRS steam piping during the operating cycle concludes that reaching the noncondensible gas limit in the DHRS steam piping is unlikely, based on the allowed normal ranges and action levels specified in the secondary water chemistry control program.

Consideration of steam leakage through the closed MSIVs and water leakage through the closed FWIVs informs a bounding low inventory case because both types of leakage affect system performance in the same manner. In the high inventory case, any leakage from the DHRS loop improves system performance. Additionally, the loss of loop inventory mitigates secondary over-pressurization situations that would otherwise occur. Therefore, omission of loop leakage from high inventory cases creates the most conservative limiting heat transfer case.

SECY 94-084 states that the heat removal system must have sufficient capacity to reduce the RCS temperature to 420 degrees F (safe shutdown condition) within 36 hours and that cooling to 420 degrees F must still be possible in the event of a single active failure. Therefore, a longer cooling period is acceptable in the case of a pipe break (passive failure) that removes the functionality of an entire train. Cases are evaluated for single-train and two-train operation at nominal initial conditions, both of which show that the DHRS is capable of bringing the NPM to a passively-cooled safe shutdown condition.

Decay Heat Removal System Performance Results

The system performance analysis indicates the DHRS removes appreciable amounts of heat over a wide range of initial conditions.

Figure 5.4-9 shows RCS cooldown for 36 hours from full power conditions with one DHRS train in operation assuming nominal system conditions. Initially, the decay heat exceeds the combined heat removal of the DHRS. The decay heat power drops off quickly as the transient progresses, and the DHRS begins to remove more heat than is added. This imbalance cools the RCS. This case also demonstrates that a single train of DHRS can provide sufficient cooling of RCS using nominal system conditions.

Figure 5.4-10 shows RCS cooldown for 36 hours from full power conditions with two DHRS trains in operation assuming nominal system conditions. For this nominal two DHRS train case, RCS average temperature stabilizes below 300 degrees F within 36 hours.

Figure 5.4-11 shows an off-nominal DHRS actuation with high DHRS inventory and low DHRS heat transfer. The heat removal bias is lower because of the high fouling, high tube plugging, and a high volume of non-condensible gas. This case assumes 102 percent reactor power. For this off-nominal two DHRS train case, RCS average temperature stabilizes below 420 degrees F within 36 hours.

Figure 5.4-12 shows an off-nominal DHRS actuation with low DHRS inventory and low DHRS heat transfer. The heat removal bias is lower because of the high fouling, high tube plugging, and a high volume of non-condensible gas. This case also uses the presence of loop leakage to further bias results. This case assumes 102 percent reactor power. This event also considers the presence of decreasing inventory due to loop leakage. For this off-nominal two DHRS train case, RCS average temperature stabilizes below 420 degrees F within 36 hours.

The final results show that the DHRS is capable of removing appreciable amounts of heat over a relatively wide range of inventories. The analyses further show the ability to accommodate fouling, SG tube plugging, and the presence of noncondensible gas, thus precluding the need for high-point vent capability. The transient plots provided in Figure 5.4-11 and Figure 5.4-12 include these factors and show that even with the degraded heat transfer, the system meets its requirements. Under each of the off-nominal transients the DHRS provides continuous passive cooling of the RCS.

These results confirm that the DHRS is capable of bringing the NPM to a passively-cooled safe shutdown condition, within a reasonable period of time, and with no offsite power or operator action required.

5.4.3.4 Tests and Inspections

Preservice and ISI requirements of Section XI are applicable to the Class 2 components of the DHRS including the steam piping, actuation valves, condensers, and condensate piping.

The DHRS actuation valves are classified as Category B valves in accordance with ASME OM Code, Subparagraph ISTC-1300(b) because seat leakage in the closed position is inconsequential for fulfillment of the required function(s). Exercising the actuation valves while at power is not practicable. Therefore, the valves are full-stroke exercised during the equivalent of cold shutdown conditions as allowed by OM Code, Subparagraph ISTC-3521 (Reference 5.4-6). The DHRSAVs are subject to a fail safe test (loss of power) every 24 months in accordance with OM Code, Paragraph ISTC-3560. The valves are also subject to a position verification test every 24 months in accordance with OM Code, Paragraphs ISTC-3530 and ISTC-3700.

The DHRS automatic actuation testing and valve actuation testing, including position verification testing, is in accordance with plant technical specifications.

An in-situ test of the DHRS function to remove heat from the RCS is performed for the first installed reactor module. This one-time test uses the module heatup system to bring the RCS as close to normal operating conditions as practicable. Once test conditions are reached, the DHRSAVs open and containment isolation valves close via the MPS. The RCS bulk temperature observed during the duration of the test is compared to a test analysis using the code of record to verify the performance of the DHRS meets design basis requirements.

5.4.4 Reactor Coolant System High-Point Vents

5.4.4.1 Design Basis

10 CFR 52.47(a)(4) requires addressing the need for high-point vents following postulated LOCAs pursuant to 10 CFR 50.46a. 10 CFR 50.46a requires high-point vents for the RCS, reactor vessel head, and other systems required to maintain adequate core cooling if the accumulation of noncondensible gases cause a loss of function of these systems. 10 CFR 52.47(a)(8) requires demonstrating compliance with technically relevant portions of the Three Mile Island (TMI) requirements set forth in certain paragraphs of 10 CFR 50.34(f), including 10 CFR 50.34(f)(2)(vi). The RCS venting capability required by 10 CFR 50.34(f)(2)(vi) is similar to 10 CFR 50.46a requirements.

5.4.4.2 System Design

The RCS does not include safety-related high-point vent capability. The high-point degasification line connected to a nozzle on the upper head of the RPV, in the PZR region, permits venting the PZR to the liquid radioactive waste system (LRWS) via the CVCS during normal operations. The LRWS contains degasifiers to remove noncondensible gases from the high-point degasification flow via the CVCS. The gaseous radioactive waste system processes the noncondensible gases collected in the degasifiers. Figure 5.1-2 depicts the arrangement of the high-point degasification vent line. A description of the CVCS design is in Section 9.3.4 and a description of the design of the liquid and gaseous radioactive waste management systems is in Section 11.2 and Section 11.3, respectively.

The ECCS has two RVVs located on the top of the RPV that discharge to the CNV upon ECCS actuation, venting any noncondensible gases accumulated in the PZR space. Section 6.3 describes the ECCS, including the design, operation, and single failure capability of the RVVs.

The ECCS is a two-phase circulation system, and gas accumulation in the RPV cannot disrupt normal flow through the RVV because it is designed for gas flow. The ECCS accommodates the effects of noncondensible gases on heat transfer.

The DHRS is internally a two-phase natural circulation system that cannot have flow disrupted by gas accumulation. The design considers the heat transfer limiting effects of the maximum noncondensible gas accumulation as discussed in Section 5.4.3.3.2, System Noncondensible Gas.

The primary coolant is a single-phase natural circulation system during DHRS operation. The highpoint is the RPV head. Accumulation of noncondensible gas in the RPV head can increase the pressure of the system but cannot reduce the water level in the RPV because the liquid phase is incompressible. Accumulation of noncondensible gases does not affect primary system circulation during DHRS operation.

During startup, the high-point degasification line vents the nitrogen atmosphere and other noncondensible gases from the RPV as the RCS heats and transitions to saturation conditions. During operation, the high-point degasification line removes noncondensible gases as they accumulate in the PZR steam space. Pressurizer venting during reactor shutdown removes noncondensible gases and accelerates hydrogen removal from the RCS.

The SGs and secondary system do not include safety-related high-point vent capability.

5.4.4.3 Performance Evaluation

During normal operation, removal of noncondensible gases by the LRWS degasifiers using the high-point degasification line, as needed, via the CVCS minimizes accumulation of noncondensible gases in the RCS and the PZR steam space in the RPV. Additionally, there are no mechanisms for accumulation of noncondensible gases in the RPV during ECCS operation because the open RVVs provide a vent path directly from the RCS to the containment; thus, additional high-point venting is not required to maintain adequate core cooling and long-term cooling following a LOCA. Long-term cooling is not adversely impacted by noncondensible gases.

As described in Section 5.4.4.2, System Design, the NPM design does not require separate, safety-related, high-point venting in the RCS. Section 5.4.3.3.2, System

Noncondensible Gas, describes the noncondensible gas considerations in the DHRS performance analysis.

These reasons obviate the need for high-point vents, and the design supports an exemption from the requirements of 50.34(f)(2)(vi), as well as the substantively equivalent requirements of 10 CFR 50.46a.

Section 6.2.4, Containment Isolation System, describes the remote operation of the high-point degasification vent isolation valves from the control room.

5.4.4.4 Tests and Inspections

The ECCS valves form part of the RCPB during normal operations. Testing of the safety-related ECCS valves includes functional testing and RCPB testing and inspection. Section 3.9.6, Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints, Section 6.3.4, Tests and Inspections, Section 5.2.4, Reactor Coolant Pressure Boundary Inservice Inspection and Testing, and Section 3.6.2.7, Implementation of Criteria Dealing with Special Features, discuss ECCS valve testing and inspection.

The high-point degasification vent path forms part of the RCPB up to the isolation valves; testing and inspection occurs as a part of that boundary. Section 5.2.4 discusses RCPB testing and inspection.

5.4.5 Pressurizer

The PZR is an integral part of the reactor vessel and is the upper region of the reactor vessel, above the integral steam plenum baffle plate. The PZR region is shown in Figure 5.4-13.

The principle function of the PZR provides a surge volume of saturated water and steam that regulates RCS pressure by maintaining a saturated steam-water interface. Instruments permit continuous monitoring of PZR pressure and the steam-water interface level in the PZR.

Pressurizer pressure is controlled by the use of PZR heaters to increase pressure; PZR spray flow decreases pressure. Maintenance of a minimal spray flow during normal operation minimizes stresses from thermal transients on the spray line components.

5.4.5.1 Design Bases

Section 5.2, Integrity of Reactor Coolant Boundary, describes compliance with the ASME BPVC. Section 3.2, Classification of Structures, Systems, and Components, describes equipment classification, including seismic qualification. Section 3.9, Mechanical Systems and Components, describes loading conditions including design stress limits and design transients.

5.4.5.2 System Design

Table 5.4-6 provides a summary of PZR design data.

The PZR maintains RCS operating pressure so that operating transients do not result in a reactor trip or actuation of other safety systems when normal support systems are functional. Sufficient combined saturated water volume and steam expansion volume provide the desired pressure response to expected system volume changes without actuating safety systems.

The PZR accommodates surges resulting from operating transients without causing a reactor trip on RCS low or high pressure. It also provides sufficient steam volume to accept in-surge from a loss of load transient (most limiting) without liquid or two-phase flow reaching the RSVs.

The PZR location at the top of the reactor vessel allows a single location for venting of noncondensible gases. Section 5.4.4, Reactor Coolant System High-Point Vents, has a description of RCS high-point venting capabilities.

The integral steam baffle plate is a non-pressure retaining structural attachment of the reactor vessel that allows hydraulic communication between the PZR and SG regions of the RPV so the PZR performs its pressure control function. The integral steam plenum baffle plate has multiple holes that provide a pressure control function. The baffle plate design communicates RCS hydraulic load change responses to the PZR volume and provides thermal and chemical mixing of fluid entering the PZR.

Additionally, the integral steam plenum baffle plate limits heat transfer between the PZR region and the RCS coolant flowing from the reactor core to the SGs, and serves as the tubesheet for the upper termination of SG tube bundles into the integral steam plenums. The baffle plate provides penetrations for alignment and support of the control rod drive shafts and instrument guide tubes. The baffle plate supports the upper riser assembly.

The total PZR volume is approximately 23 percent of the total RCS volume. Pressurizer level during full-power operation is controlled to a nominal 60 percent with a minimum PZR level of approximately 40 percent below 15 percent power. These programmed values provide margin to the upper and lower PZR water level analytical limits of 80 percent and 35 percent.

The PZR provides a saturated steam-water interface at an elevated temperature such that the reactor coolant remains subcooled during normal operation. Section 5.2.2, Overpressure Protection, provides more information on overpressure protection for various states of operation.

Two sets of electrical heater bundles are in the lower portion of the PZR space. The heaters are horizontal and immersed in the PZR liquid. Table 5.4-7 provides PZR heater parameters. The PZR volume, in conjunction with the PZR level control band and the capabilities of the CVCS, prevent uncovering of the PZR heaters during anticipated operational transients. The PZR heaters are automatically de-energized by the MPS before decreasing water level to the top of the PZR heater elements. The PZR heater trip function is provided to protect the PZR heater elements and the integrity of the RCPB.

Heating the PZR fluid is required in order to maintain the PZR at an elevated temperature and saturated conditions. Under steady state conditions, the heater output makes up for continual heat losses to the containment and the RCS. In transient conditions involving increases in RCS volume, fluid from the hot region of the RCS enters the PZR and heats to saturated liquid conditions in order to maintain normal operating pressure. Similarly, for transients that involve decreases in RCS volume, the PZR liquid flows into the RCS hot region, and the PZR heaters produce additional steam to maintain normal operating pressure.

Control rod drive shafts and instrument lines occupy the central region of the PZR. Two spray nozzles located on opposite sides of the PZR ensure adequate coverage of the PZR spray to condense the steam.

The CVCS provides a small continuous flow to the PZR through the spray nozzles to maintain PZR region chemistry consistent with the balance of the RCS and to minimize stresses from thermal transients when full spray flow initiates. Instrumentation in the CVCS portion of the PZR spray line indicates the rate of spray flow in the control room.

Figure 5.4-13 shows two PZR heater bundles mounted through the side of the RPV located 180 degrees from each other around the integral steam plenum assembly. Two control groups of PZR heaters are in each bundle. A proportional integral controller controls group 1 heaters; they maintain nominal programmed RCS operating pressure when the reactor is at steady state power. The sizing includes a design consideration to maintain primary pressure considering the steady state heat losses, such as continuous PZR spray flow and heat transfer to the containment and the RCS. Group 2 backup heaters energize sequentially when RCS pressure drops below the nominal programmed pressure range and de-energize sequentially as pressure returns to the nominal operating range.

Natural circulation during normal operation and hot shutdown conditions due to the elevation difference and relative temperature difference between the reactor core and the SGs drives the RCS flow in the NPM. As a result, hot shutdown conditions do not require PZR heater operation to establish and maintain natural circulation, and the design supports an exemption from the PZR power supply and control power interface requirements of 10 CFR 50.34(f)(2)(xiii). In addition, the design does not include PZR relief valves or PZR block valves, and the power supply requirements for these valves in 10 CFR 50.34(f)(2)(xx) are not technically relevant.

Each of the two proportional heaters (A and B) from the low voltage alternating current electrical power system have non Class 1E electrical power supplies through two Class 1E circuit breakers that are part of the MPS, connected in series to the PZR control cabinet. The module control system (MCS) controls the PZR heaters via the PZR control cabinets. The safety-related function of PZR heater circuit breakers is isolation of the heaters from their power source to ensure the integrity of the RCPB if the heaters uncover. The MPS provides a trip function on lowering PZR level that removes power to the heaters before PZR level reaches the top of the PZR heaters. Section 7.1, Functional Design Principals, provides additional detail regarding the Class 1E breakers associated with the PZR heaters.

Pressurizer instrumentation measures the steam-water interface level and provides input to the safety-related MPS as described in Chapter 7, including the low water level protection of the PZR heaters. Chapter 7 describes the PZR level indication provided to the control room, to the operating staff, and to the MCS. As described in Table 1.9-5, the design supports an exemption from the power supply requirements for PZR level indication included in 10 CFR 50.34(f)(2)(xx).

Pressurizer pressure measurements provide input to the safety-related MPS as described in Chapter 7. Chapter 7 describes indications of RCS pressure in the control room to the operating staff and indications to the MCS.

Instrumentation indicates the spray flow into the PZR by the CVCS and PZR heater output, which are provided to the MCS. The MCS provides automated assistance to control level and pressure in the RCS.

5.4.5.3 Performance Evaluation

The CVCS controls PZR level. During normal reactor operation, the CVCS maintains the desired volume of coolant in the RCS as indicated by the PZR liquid level instrumentation. Operator permissive action or manual operator action maintains PZR level in its operating band by adding coolant inventory (makeup) or by reducing coolant inventory (letdown) by discharging fluid to the LRWS.

The nominal PZR water level is a function of reactor power level. Between hot zero power and 20 percent power, the reactor coolant experiences a heatup or cooldown and therefore a large change in volume. Changing PZR water level partially absorbs the change in volume; however, to maintain a sufficient steam volume for pressure control and margin to operating limits, makeup or letdown using the CVCS compensates for the large change in temperature by removing or adding reactor coolant mass from or to the RCS. Above 20 percent power, the changes in reactor coolant volume are much smaller. The nominal PZR water level accommodates the expected changes in RCS volume.

The PZR heaters add steam to the PZR steam bubble and PZR spray flow condenses steam from the PZR steam bubble to regulate PZR (and RCS) pressure. The RCS supports automatic control of pressure by providing the PZR control cabinet, PZR heaters and electrical cabling, PZR spray nozzles and supply piping from the CVCS, and PZR pressure measurement to the MPS and MCS.

<u>Startup Operations</u>

Nitrogen from the CVCS pressurizes the RCS. The CVCS adds water to the RCS to raise PZR water level to the normal operating level band. Nitrogen vents from the PZR as necessary. The CVCS regulates PZR level.

Pressurizer heaters energize to raise the temperature in the PZR and to draw a steam bubble. Pressurizer heater output adjusts to support pressurization commensurate with the RCS heat up rate to ensure RCS temperature and pressure remains within the specified limits.

During heatup, the module heatup system increases the RCS temperature to the desired temperature. During the temperature increase, the PZR increases the RCS pressure in order to provide subcooling at the module heatup system heater exit and to reach normal operating pressure.

The PZR heaters add energy to the RCS throughout heatup. The CVCS maintains PZR level and adjusts RCS chemistry and boron concentration.

Normal Operations

During normal full power operations, CVCS controls PZR level to a nominal 60 percent of the overall PZR level. The programmed PZR level increases as power is increased. The RCS pressure increases by increasing power to the PZR heaters, which generates steam that is added to the PZR steam bubble and raises pressure. The CVCS includes a PZR spray line with a control valve that provides flow to the PZR spray nozzles. Maintenance of a minimal spray flow during normal operation at power maintains the PZR chemistry in equilibrium with the RCS and minimizes thermal stresses to the spray line components. The PZR spray flow is at a lower temperature relative to the temperature of the saturated steam space. The RCS pressure decreases by initiating spray flow through the PZR spray nozzles into the PZR steam volume. Spray flow condenses steam, reducing pressure.

Effective mixing of fluid within the PZR volume occurs during normal operation because of thermal effects associated with cooling from the PZR walls and heating from the PZR heaters. Fluid that enters the RCS from the PZR effectively mixes with the rest of the reactor coolant as it flows down over the SG helical tube bundles, down the remainder of the downcomer, and into the reactor core. Therefore, the reactor coolant entering the RCS loop from the PZR has a uniform temperature and boron concentration.

Shutdown Operations

Pressurizer heaters de-energize and spray initiates as needed to reduce RCS pressure. The SG steaming continues cooldown in conjunction with the PZR pressure reduction to reduce the temperature of the RCS. When the RCS cools and pressure reduces, the PZR steam bubble may be replaced with a nitrogen bubble. The high-point degasification line introduces nitrogen to the PZR. Pressurizer spray is performed, and PZR heater power is reduced and then secured as the steam bubble collapses and is replaced by a nitrogen bubble.

The PZR heaters and PZR spray control RCS pressure during normal power operations, but the NPM achieves safe shutdown conditions without reliance on pressure control by PZR heaters or PZR spray flow.

5.4.5.4 Tests and Inspections

The RCPB portions of the PZR undergo testing and inspection as a part of the RPV testing and inspections. The PZR permits the required inspections. Section 5.2.4, RCPB ISI and Testing discusses RPV testing and inspections.

Portions of the integral steam plenum baffle plate that surround the steam plena are an ASME BPVC Section III, Class 1 component. The central portion of integral steam plenum baffle plate is a non-pressure retaining structural attachment. Based on this, the entire integral steam plenum baffle plate undergoes construction, inspection, and testing to ASME BPVC Section III, Subsection NB requirements.

Pressurizer heater monitoring and testing are in accordance with applicable ASME BPVC requirements as a part of the RCPB. Pressurizer heater testing verifies their heat addition functionality in accordance with vendor recommended acceptance criteria.

5.4.5.5 Pressurizer Materials

The PZR includes the top portion of the RPV upper shell, the RPV upper head, heater bundles, and spray nozzles. Section 5.2.3, RCPB Materials, describes the material of the RPV upper shell and upper head, integral steam plenum plate, the PZR spray nozzles, and PZR spray nozzle safe ends.

The materials for the heater bundle assemblies are in Table 5.2-3. These materials comply with ASME BPVC, Section II requirements.

5.4.6 References

5.4-1 Nuclear Energy Institute, "Steam Generator Program Guidelines," NEI 97-06, Revision 3, Washington, DC, January 2011.

5.4-2 Electric Power Research Institute, "Steam Generator Management Program: Pressurized Water Reactor Steam Generator Examination Guidelines," EPRI #1013706, EPRI, Palo Alto, CA, 2007.

5.4-3 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2017 Edition, Section III, Division 1, “Rules for Construction of Nuclear Facility Components,” New York, NY.

5.4-4 Institute of Electrical and Electronics Engineers, "Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," IEEE Standard 344-2004, Piscataway, NJ.

5.4-5 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2017 Edition, Section XI, Division 1, “Rules for Inservice Inspection of Nuclear Power Plant Components,” New York, NY.

5.4-6 NuScale Power, "Extended Passive Cooling and Reactivity Control Methodology Topical Report," TR-124587, Revision 0.

5.4-7 American Society of Mechanical Engineers, ASME OM-2017, “Operation and Maintenance of Nuclear Power Plants,” New York, NY.

5.4-8 American Society of Mechanical Engineers, "Forged Fittings, Socket-Welding and Threaded," ASME B16.11-2011, New York, NY.

5.4-9 NuScale Power, LLC, "Evaluation Methodology for Stability Analysis of the NuScale Power Module," TR-0516-49417-P-A, Revision 1.

5.4-10 NuScale Power, LLC, "NuScale Comprehensive Vibration Assessment Program Analysis Technical Report," TR-121353, Revision 0.

5.4-11 "Methodology for the Determination of the Onset of Density Wave Oscillations (DWO)," TR-131981-P, Revision 0.

Table 5.4-1: Steam Generator Full-Load Thermal-Hydraulic Operating Conditions (Best Estimate)(2)

Parameter	Value
(1)Total heat transfer (MW_t)	249.7
Steam pressure (psia)	475
Steam temperature (°F)	537
SG inlet temperature (°F)	250
Total SG flow (lbm/hr)	815,600

(1) Based on operation of both SGs, each SG is capable of providing half of the total heat transfer required.
(2) Based on beginning of life steady state operations.

Table 5.4-2: Steam Generator Design Data

Parameter	Value
Type	Helical, once-through
Total number of helical tubes per NPM	1380
Number of helical tube columns per NPM	21
Internal pressure - secondary (psia)	2200
External pressure - primary (psia)	2200
Internal temperature - secondary (°F)	650
External temperature - primary (°F)	650
External temperature - SG piping in containment (°F)	650
Tube wall outer diameter (inches)	0.625
Tube wall thickness (inches)	0.050
Steam tubesheet thickness, without clad (inches)	4.000
Feed tubesheet thickness, without clad (inches)	4.5
Steam and feed tubesheet clad thickness - secondary (inches)	0.250
Steam and feed tubesheet clad thickness - primary (inches)	0.500
Steam tubesheet thickness, with clad (inches)	4.750
Feed tubesheet thickness, with clad (inches)	5.25
Total heat transfer area (ft^2)	17928
Fouling factor (hr-ft^2-°F/BTU)	0.0001
Minimum SG tube transition bend radius (inches)	≥ 6.250

Table 5.4-3: Steam Generator System Component Materials

Component	Specification	Alloy Designation (Grade, Class, or Type)
SGS Piping • SGS Feedwater Piping Assembly • SGS Main Steam Piping Assembly		
Pipe	SA-312	TP304 SMLS, TP316 SMLS[1]
Pipe Fittings	SA-182	F304, F316 [1]
	SA-403	WP304 SMLS, WP316 SMLS [1]
Pressure-Retaining Bolting	SB-637	UNS N07718[2]
Piping Supports		
Supports	SA-240	Type 304, Type 316[1] Type 405, Type 410S
	SA-479	Type 304, Type 316[1] Type 405, Type 410 Annealed or Class 1
	SA-312	TP304, TP304L, TP316, TP316L[1]
Bolting	SA-193	Grade B8, Grade B8M
	SA-194	Grade 8, Grade 8M
	SA-564	Type 630 H1100
SG Supports		
SG Supports, SG Tube Supports SG Backing Strips	SA-240	Type 304
Weld Filler Metals for Piping, Piping Supports, and SG Supports		
3XX Austenitic Stainless Steel Weld Filler Metals	SFA-5.4	E308, E308L, E316, E316L[3]
	SFA-5.9	ER308, ER308L, ER316, ER316L[3]
	SFA-5.22	E308, E308L, E316, E316L [3,4]
	SFA-5.30	IN308, IN308L, IN316, IN316L[3]
Nickel-Base Alloy Weld Filler Metals	SFA-5.11	ENiCrFe-7
	SFA-5.14	ERNiCrFe-7, ERNiCrFe-7A, EQNiCrFe-7, EQNiCrFe-7A
Inlet Flow Restrictor Components		
Flow restrictor Hardware	SA-479	Type 316
Flow restrictor locking plate	SA-240	Type 304
Other SG Components		
SG Tubes Steam Plenum Access Port Steam Plenum Access Port Covers Steam Plenum Caps Feed Plenum Access Port Feed Plenum Access Port Covers	Table 5.2-3	

Notes:
(1) 0.03% maximum carbon if unstabilized Type 3XX base metals are welded or exposed to temperature range of 800°F to 1500°F subsequent to final solution anneal.
(2) SB-637 UNS N07718 solution treatment temperature range before precipitation hardening treatment restricted to 1800°F to 1850°F.
(3) 0.03% maximum carbon for unstabilized AISI Type 3XX weld filler metals; ferrite number in the range of 5FN to 20FN, except 5FN to 16FN for Type 316 and Type 316L.
(4) Applicable to reactor vessel internals components only.

Table 5.4-4: Decay Heat Removal System Component Materials

Component	Specification	Grade/Type/Class
DHRS Piping Assembly and DHRS Condenser Assembly		
Pipe	SA-312	TP316 SMLS, TP316L SMLS[1]
Pipe Fittings, Headers and Integral Orifice Flow Element	SA-182	F316[1]
	SA-403	WP316 SMLS[1]
	SA-479	Type 316, Type 316L[1]
DHRS Condenser Supports		
Supports	SA-240	Type 304, Type 316[1]
	SA-479	Type 304, Type 304L, Type 316, Type 316L[1]
	SA-312	TP304, TP304L, TP316, TP316L[1]
Bolting	SA-193	Grade B8, Grade B8M
	SA-194	Grade 8, Grade 8M
Piping Supports		
Supports	SA-240	Type 304, Type 316[1] Type 405, Type 410S
	SA-479	Type 304, Type 316[1] Type 405, Type 410 Annealed or Class 1
	SA-312	TP304, TP304L, TP316, TP316L[1]
Bolting	SA-193	Grade B8, Grade B8M
	SA-194	Grade 8, Grade 8M
	SA-564	Type 630 H1100
Weld Filler Metals for DHRS Piping, Condenser, and Their Supports		
3XX Austenitic Stainless Steel Weld Filler Metals	SFA-5.4	E308, E308L, E316, E316L[2]
	SFA-5.9	ER308, ER308L, ER316, ER316L[2]
	SFA-5.30	IN308, IN308L, IN316, IN316L[2]
Nickel-Base Alloy Weld Filler Metals	SFA-5.11	ENiCrFe-7
	SFA-5.14	ERNiCrFe-7, ERNiCrFe-7A

Notes:
(1) 0.03% maximum carbon if unstabilized Type 3XX base metals are welded or exposed to temperature range of 800°F to 1500°F subsequent to final solution anneal.
(2) 0.03% maximum carbon for unstabilized AISI Type 3XX weld filler metals; ferrite number in the range of 5FN to 20FN, except 5FN to 16FN for Type 316 and Type 316L.

Table 5.4-5: Decay Heat Removal System Design Data

Parameter	Value
Internal pressure (psia)	2200
Passive condenser operating pressure (psia)	550[1]
Temperature (°F)	650
Number of condensers	2
Total number of tubes per condenser	90
Tube wall outer diameter (inches)	1.315
Tube wall thickness (inches)	0.109
Tube external surface area per condenser (ft^2)	269.2
Fouling factor (hr-ft^2-F/BTU)	0.0005

Notes:
(1) Pressure can vary from 500 to 700 psia depending on operating conditions.

Table 5.4-6: Pressurizer Design Data

Parameter	Value
Internal pressure (psia)	2200
Temperature (°F)	650
PZR heater element temperature (°F)	800
Spray nozzles	2

Table 5.4-7: Pressurizer Heater Parameters

Parameter	Value
Voltage (Vac)	480, 3-phase
Frequency (Hz)	60
Heater bundles per PZR	2
Heater groups per bundle	2
Nominal total capacity per heater bundle (kW)	400

Table 5.4-8: Failure Modes and Effects Analysis - Decay Heat Removal System

Component Identification	Function	Failure Mode	Failure Mechanism	Effect on System	Method of Failure Detection
DHRSAV (normally closed, fail open)	1) Maintain DHRS in standby	A) Spurious opening	Mechanical Electrical/I&C	Affected DHRS condenser has open flow path to SG. Turbine must be isolated to prevent damage. Normal cooling is available through FW. Unaffected DHRS train remains available.	• Valve position indication • Steam pressure • Steam temperature • Passive condenser temperature
		B) Spurious DHRS actuation	Electrical/I&C Operator error	Both DHRS trains initiate operation, and the MSIVs and FWIVs close. The engineered safety features actuation system generates a reactor trip signal regardless of if the DHRS actuation signal is erroneous or not.	• PZR pressure • Steam pressure • Valve position indication • PZR level • Passive condenser temperature • Passive condenser level • Reactor trip
		C) Leakage (passive failure)	Mechanical	Minor leakage does not impact DHRS operation. DHRS inventory is maintained by the continuous makeup of FW to the SG. Major valve seat leakage causes both DHRS trains to actuate.	• Minor seat leakage: -none • Major seat leakage: -Passive condenser temperature -Steam pressure -Steam temperature • Minor valve bonnet leakage: -periodic inspections • Major valve bonnet leakage: -passive condenser temperature

Table 5.4-8: Failure Modes and Effects Analysis - Decay Heat Removal System (Continued)

Component Identification	Function	Failure Mode	Failure Mechanism	Effect on System	Method of Failure Detection
	2) Initiate DHRS cooling	A) Fail to open	Mechanical Electrical/I&C	Sufficient flow required for cooling of affected DHRS train is maintained by redundant DHRSAV. Other train operates normally.	• Valve position indication
		B) Partial opening	Mechanical		
		C) Slow opening (extended stroke time or delayed actuation)	Mechanical Electrical/I&C		
		D) Spurious closure	Electrical/I&C Operator error		
		E) Leakage (passive failure)	Mechanical	Leakage out of the system results in a reduction in cooling inventory for the affected DHRS train. DHRS operation is maintained by the unaffected train. Major system leakage is addressed under DHRS Loop Pressure Boundary failures.	• Minor valve bonnet leakage: -periodic inspection • Major valve bonnet leakage: -Steam pressure (valve bonnet failure) -Passive condenser temperature -RCS temperatures
		F) Flow blockage (passive failure)	Mechanical	Reduced or nonexistent flow through affected valve but adequate DHRS flow is maintained by redundant actuation valve. The other train operates normally.	None

Table 5.4-8: Failure Modes and Effects Analysis - Decay Heat Removal System (Continued)

Component Identification	Function	Failure Mode	Failure Mechanism	Effect on System	Method of Failure Detection
MSIV (normally open, fail closed)	1) Isolate MS line to establish closed DHRS cooling loop	A) Fail to close	Mechanical Electrical/I&C	The DHRS inventory in the train associated with the failed MSIV is maintained by the downstream secondary MSIV.	• Valve position indication • Steam pressure • Passive condenser temperature • RCS temperatures
		B) Partial closure	Mechanical		
		C) Slow closure (extended stroke time or delayed actuation)	Mechanical Electrical/I&C	The affected train functions normally. Secondary MSIV closure mitigates loss of inventory. System can function properly with a range of initial inventories so a loss of cooling capability in the affected train is not credible.	• Valve position indication • Steam pressure • Passive condenser temperature
		D) Spurious opening	Electrical/I&C Operator error	The DHRS inventory in the train associated with failed MSIV is maintained by the downstream secondary MSIV.	• Valve position indication • Steam pressure • Passive condenser temperature • RCS temperatures
		E) Leakage through seat or seal (passive failure)	Mechanical	The affected train functions normally. Secondary MSIV closure mitigates loss of inventory. System can function properly with a range of initial inventories so a loss of cooling capability in the affected train is not credible.	• Periodic testing & inspection • Steam pressure • Passive condenser temperature • RCS temperatures
MSIV Bypass Valve (normally closed, fail closed)	1) Maintain DHRS pressure boundary during DHRS operation	A) Spurious opening during DHRS operation	Operator error	The DHRS inventory in the train associated with failed MSIV bypass valve is maintained by the downstream secondary MSIV.	• Valve position indication • Steam pressure • Passive condenser temperature • RCS temperatures
		B) Leakage through seat or seal (passive failure)	Mechanical	MSIV failure mode 1E.	

Table 5.4-8: Failure Modes and Effects Analysis - Decay Heat Removal System (Continued)

Component Identification	Function	Failure Mode	Failure Mechanism	Effect on System	Method of Failure Detection
FWIV (normally open, fail closed)	1) Isolate FW line to establish closed DHRS cooling loop	A) Fail to close	Mechanical Electrical/I&C	Additional DHRS inventory may be added to the train associated with the failed FWIV until FW stops flowing into the SG. The FWRV closes simultaneously, providing redundant isolation. The system can function properly with a range of initial inventories so a loss of cooling capability due to overfill is not credible.	• Valve position indication • Passive condenser temperature • Steam pressure
		B) Partial closure	Mechanical		
		C) Slow closure (extended stroke time or delayed actuation)	Mechanical Electrical/I&C	In the affected train, more DHRS inventory than anticipated may be added while the FWIV and FWRV close but the system can function properly with a range of initial inventories so a loss of cooling capability is not credible.	• Valve position indication • Passive condenser temperature • Steam pressure
		D) Spurious opening	Electrical/I&C Operator error	The DHRS inventory in the train associated with failed FWIV is maintained by the FW check valves upstream of the FWIV and the closed FWRV.	• Valve position indication • Steam pressure
		E) Leakage through seat or seal (passive failure)	Mechanical	The affected train functions normally. Feedwater regulating valve closure mitigates loss of inventory. System can function properly with a range of initial inventories so a loss of cooling capability in the affected train is not credible.	• Periodic testing & inspection • Steam pressure • Passive condenser temperature • RCS temperatures
Safety-related FW check valve	1) Prevent backflow through FWIVs	A) Fail to close B) Partial closure C) Slow closure	Mechanical	The nonsafety-related check valve upstream of the safety-related check valve is credited to close during a FWLB. Thus, sufficient inventory is available for the affected train.	• Passive condenser temperature • Steam pressure • RCS temperatures

Table 5.4-8: Failure Modes and Effects Analysis - Decay Heat Removal System (Continued)

Component Identification	Function	Failure Mode	Failure Mechanism	Effect on System	Method of Failure Detection
SGS Thermal Relief Valve	1) Provide pressure boundary	A) Spurious opening of relief valve with failure to close (passive failure)	Mechanical	A spurious opening of the thermal relief valve with a failure to close causes the affected DHRS train to lose inventory and become inoperable. Safe shutdown without ECCS actuation is achieved with the remaining DHRS train.	• Steam pressure • DHRS level instrument switches • Containment leakage monitoring instrumentation (containment evacuation system)
	2) Provide overpressure protection when SGS is water solid	A) Failure of valve to lift at setpoint	Mechanical	A failure of the valve to lift during a water solid over pressure scenario could cause deformation or rupture of pressure boundary components. A rupture would cause the affected DHRS train to be inoperable. The module is in safe shutdown before this event, and cooling resumes with the unaffected DHRS train or through flooding the CNV with the containment flooding and drain system.	• Steam pressure • DHRS level instrument switches
DHRS Loop Pressure Boundary (Includes piping inside CNV, SG tubes)[1]	1) Provide a pressure boundary for the SGS and DHRS	A) Pipe rupture and loss of DHRS loop inventory (passive failure)	Mechanical	A pipe rupture of any pressure boundary piping that is part of the DHRS loop (SGS and DHRS piping within containment isolation valves) causes the affected DHRS train to lose inventory and become inoperable. Safe shutdown without ECCS actuation is achieved with the remaining DHRS train.	• Steam pressure • DHRS level instrument switches • Containment leakage monitoring instrumentation (containment evacuation system)

Notes:

(1) Pipe ruptures outside of the CNV are not postulated as the DHRS piping is specified to meet additional stress criteria specified in Branch Technical Position 3-4. Containment leakage monitoring instrumentation only identifies pressure boundary failures within the CNV.

Figure 5.4-1: Steam Generator Helical Tube Bundle

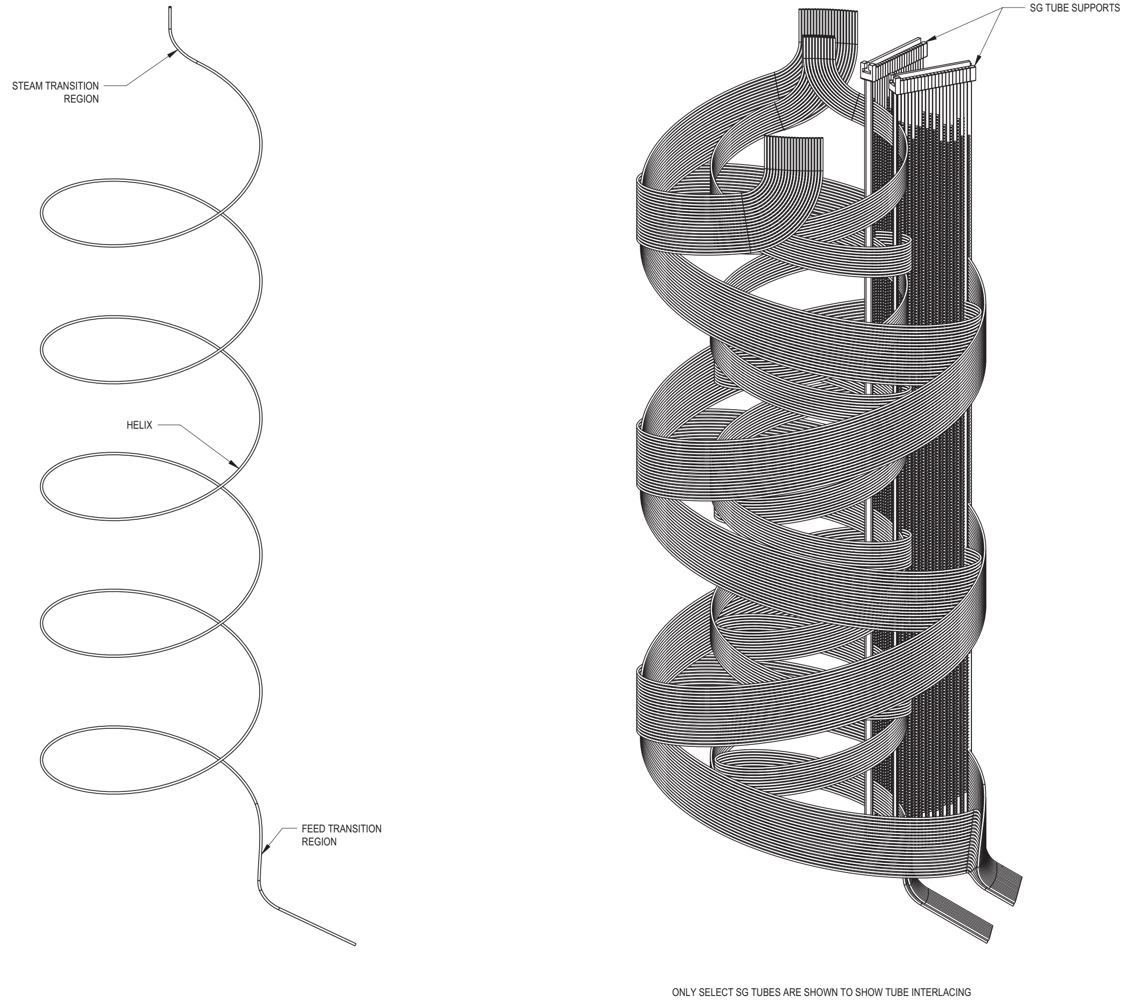


Figure 5.4-2: Configuration of Steam Generators in Upper Reactor Pressure Vessel Section

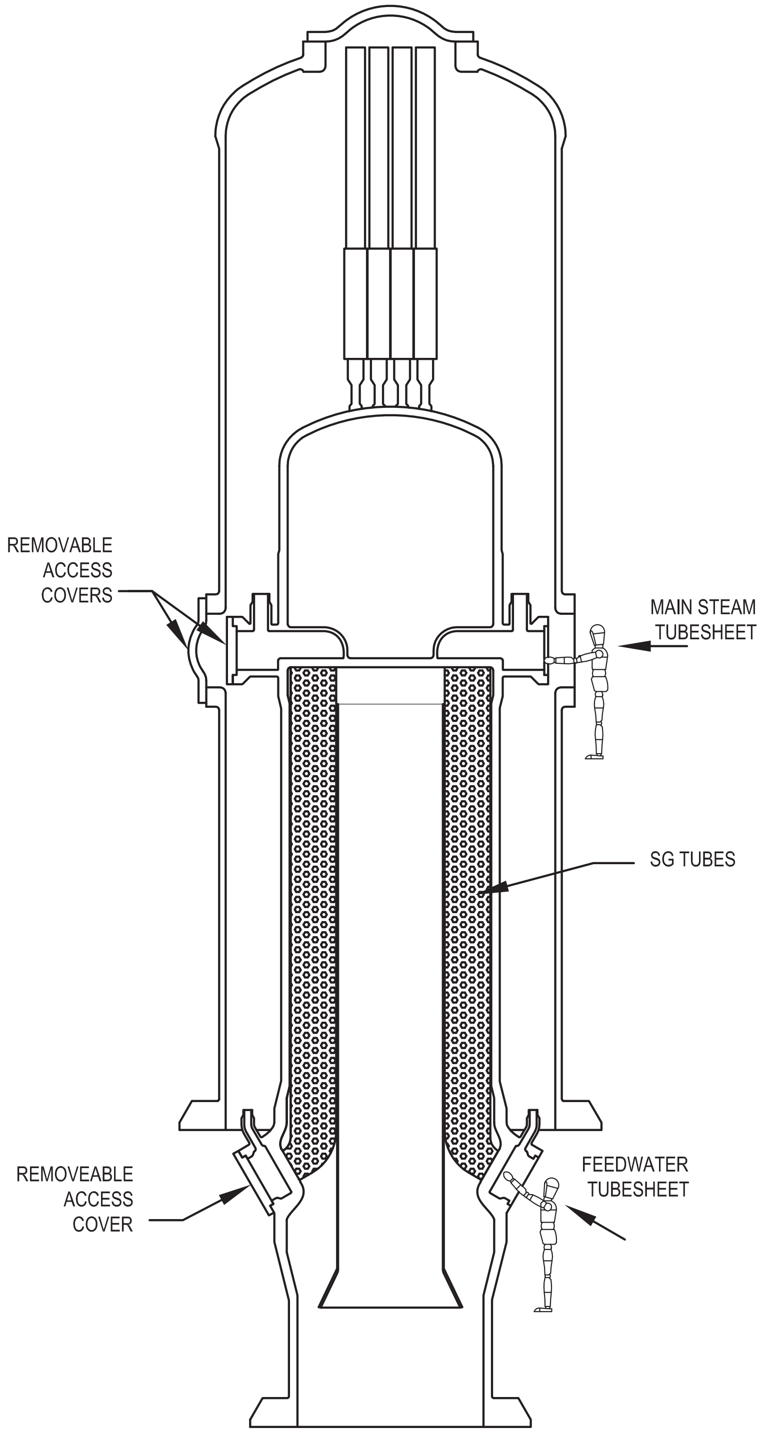


Figure 5.4-3: Integral Steam Plenum

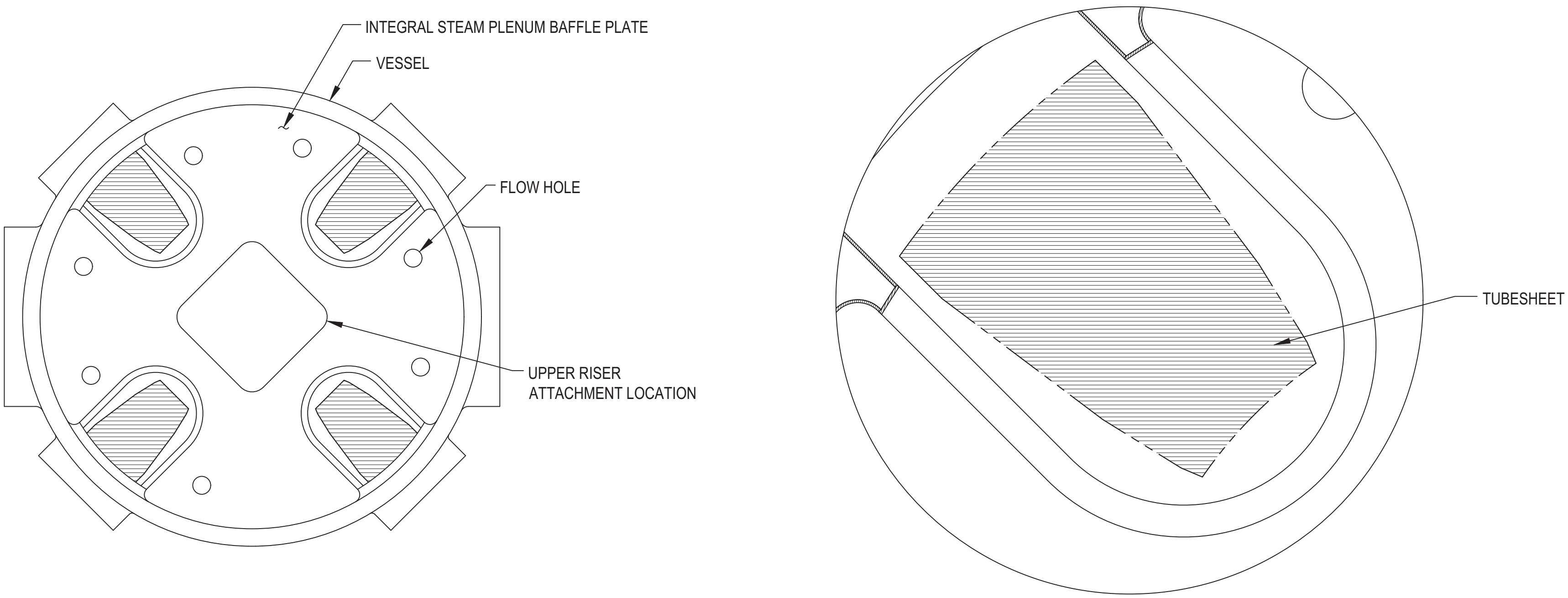


Figure 5.4-4: Feedwater Plenum Access Port

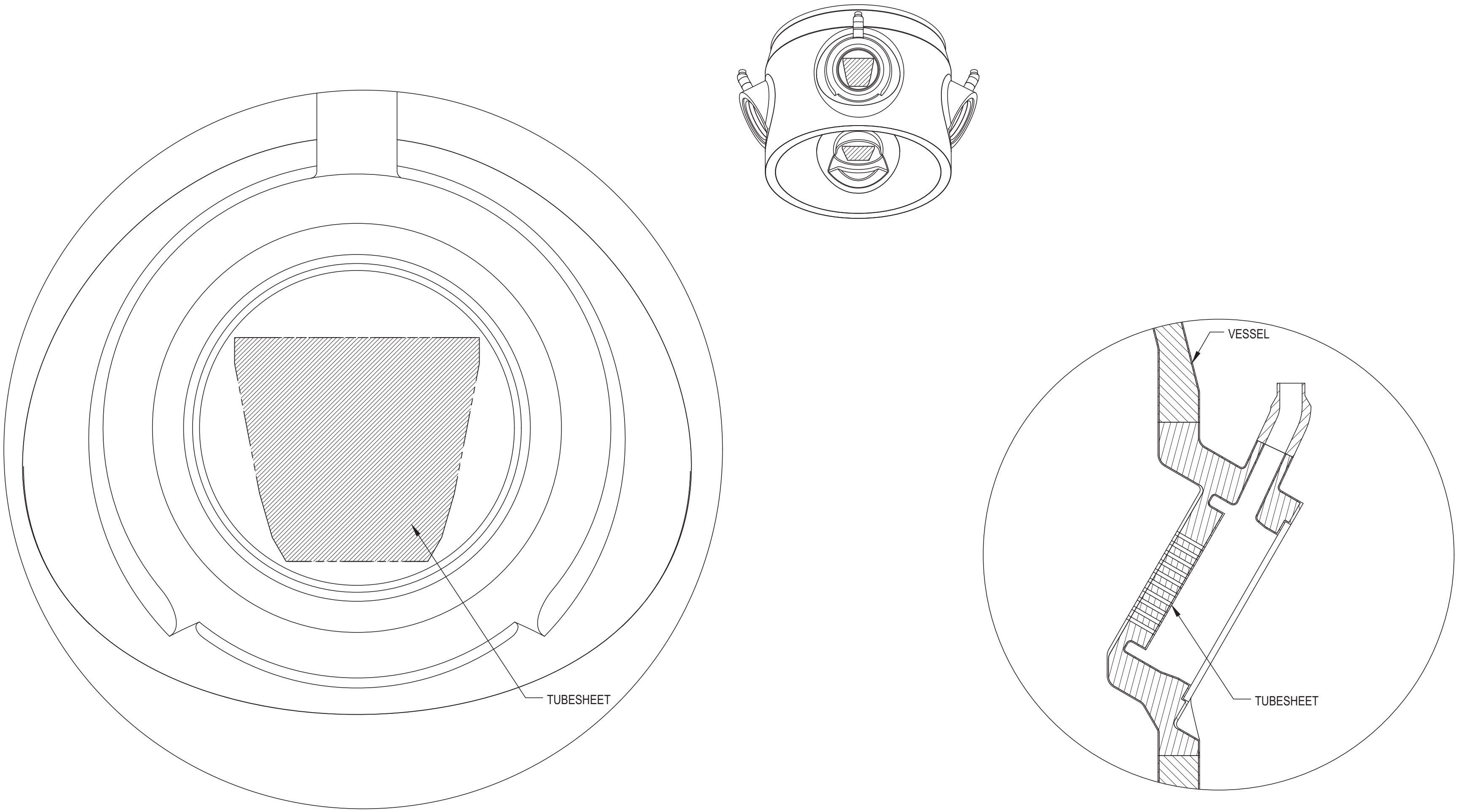


Figure 5.4-5: Steam Generator Supports and Steam Generator Tube Supports

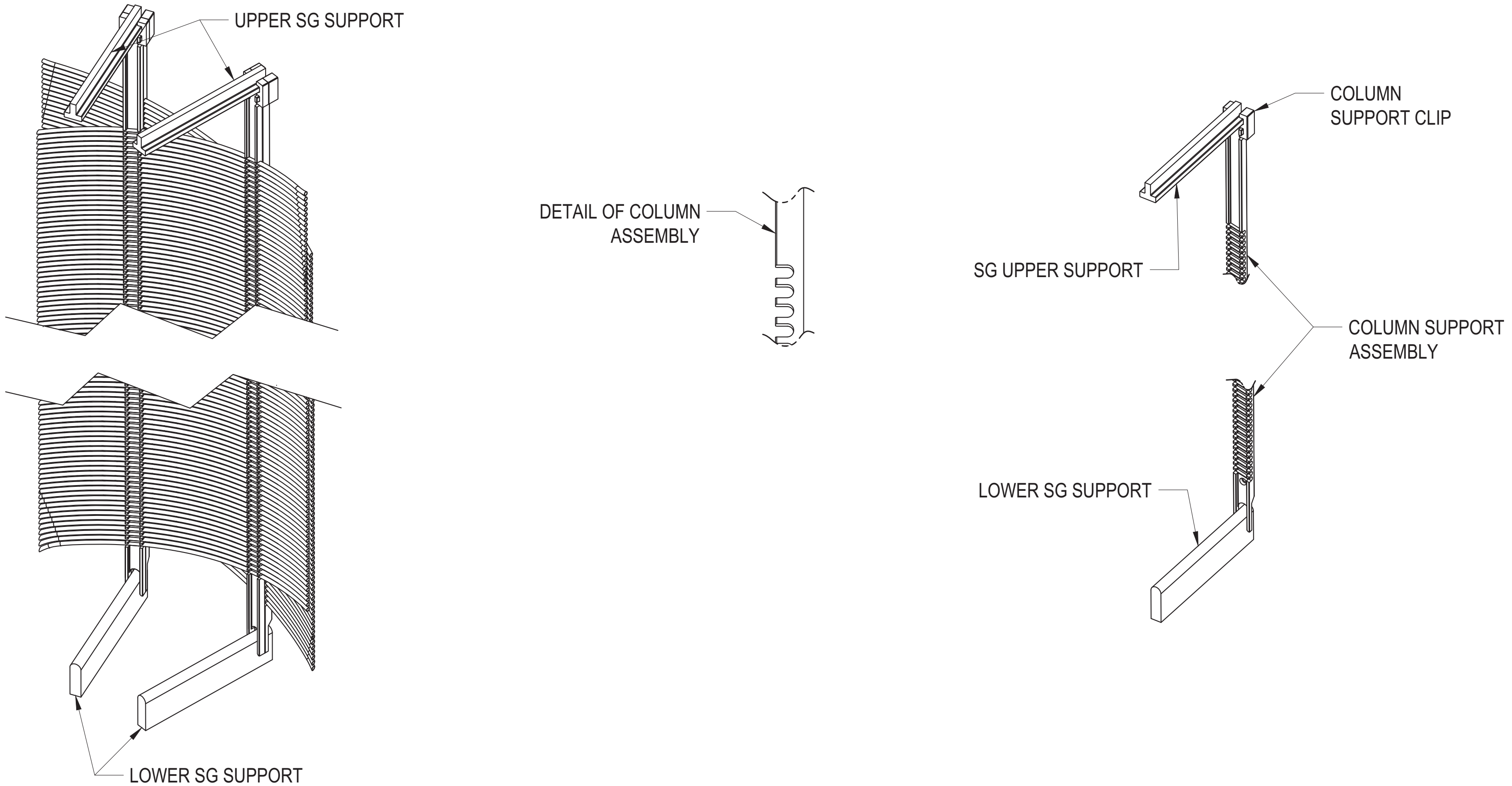


Figure 5.4-6: Steam Generator Tube Supports

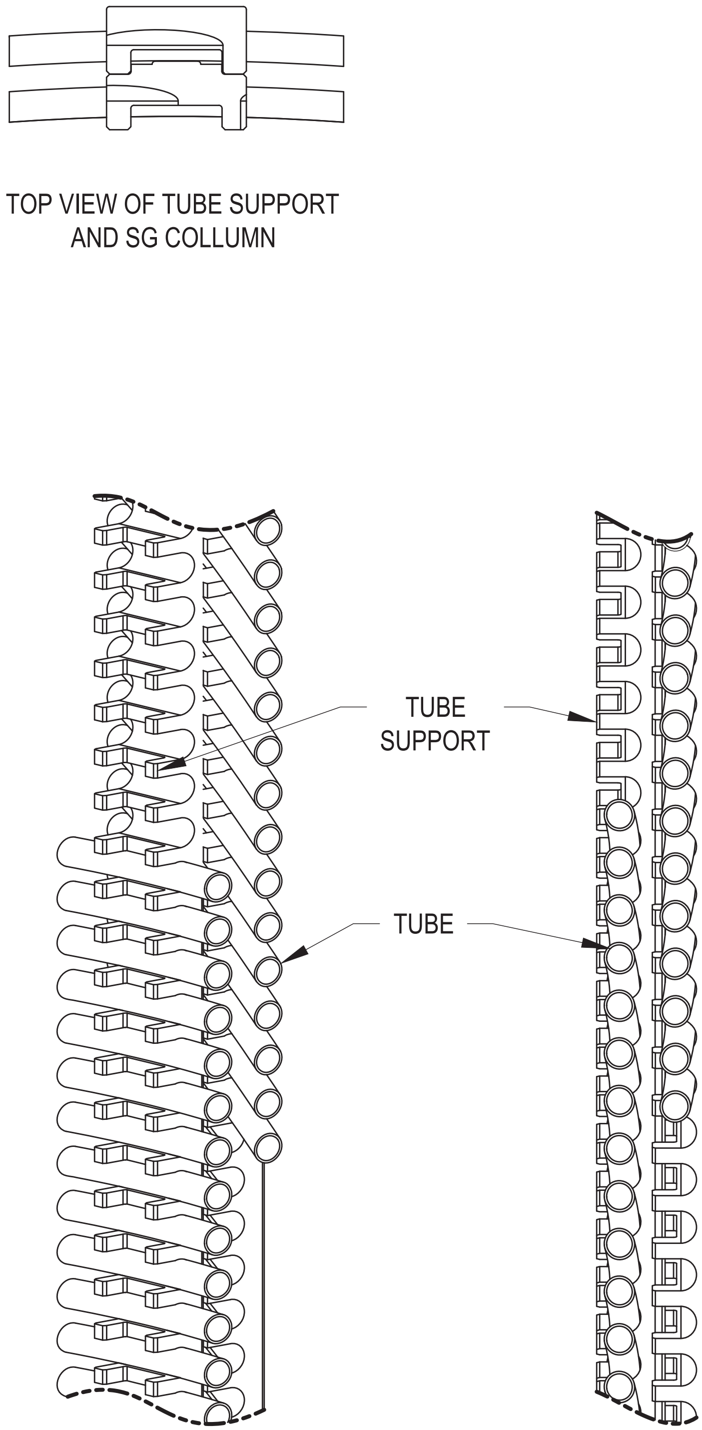


Figure 5.4-7: Steam Generator Simplified Diagram

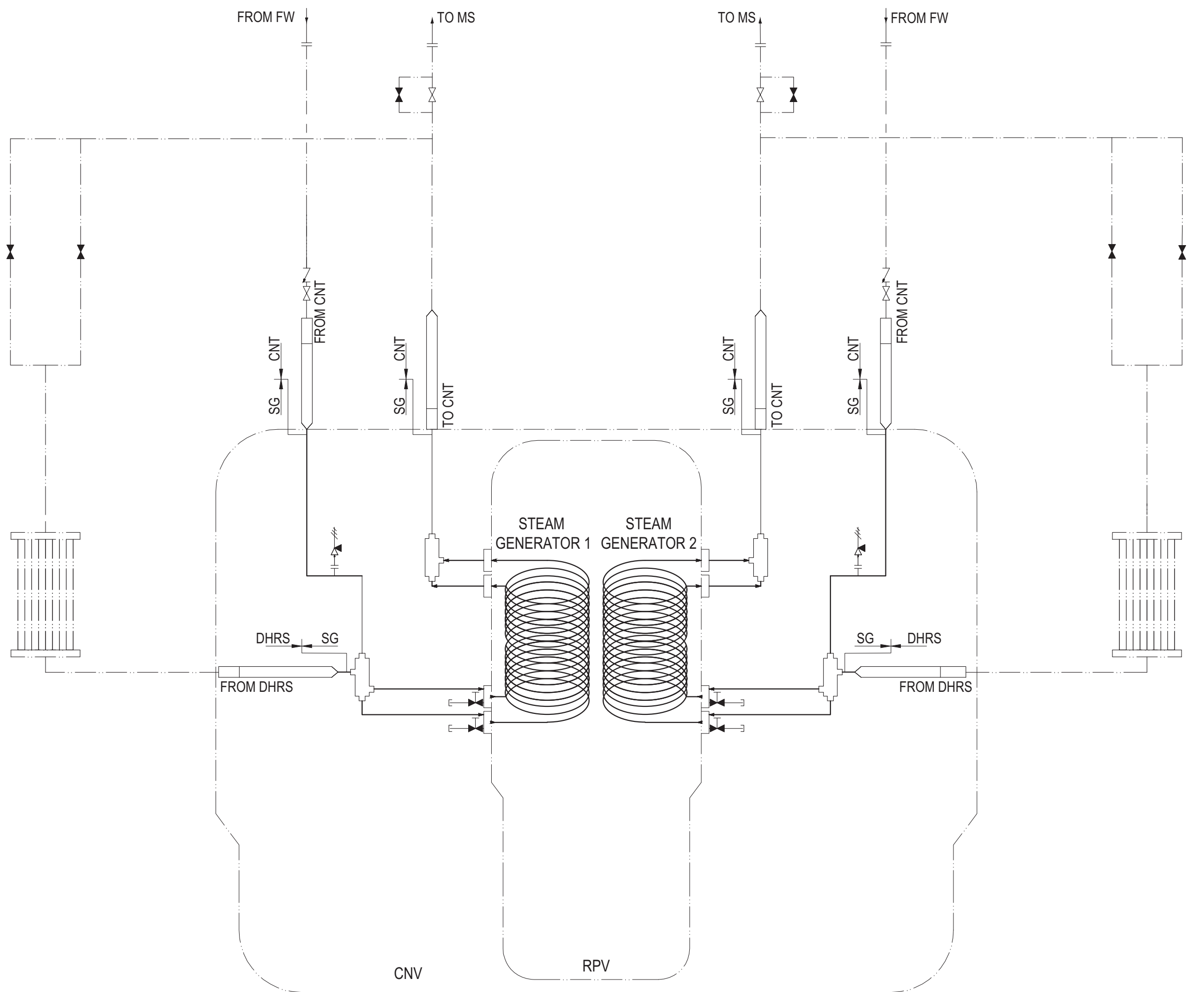


Figure 5.4-8: Decay Heat Removal System Simplified Diagram

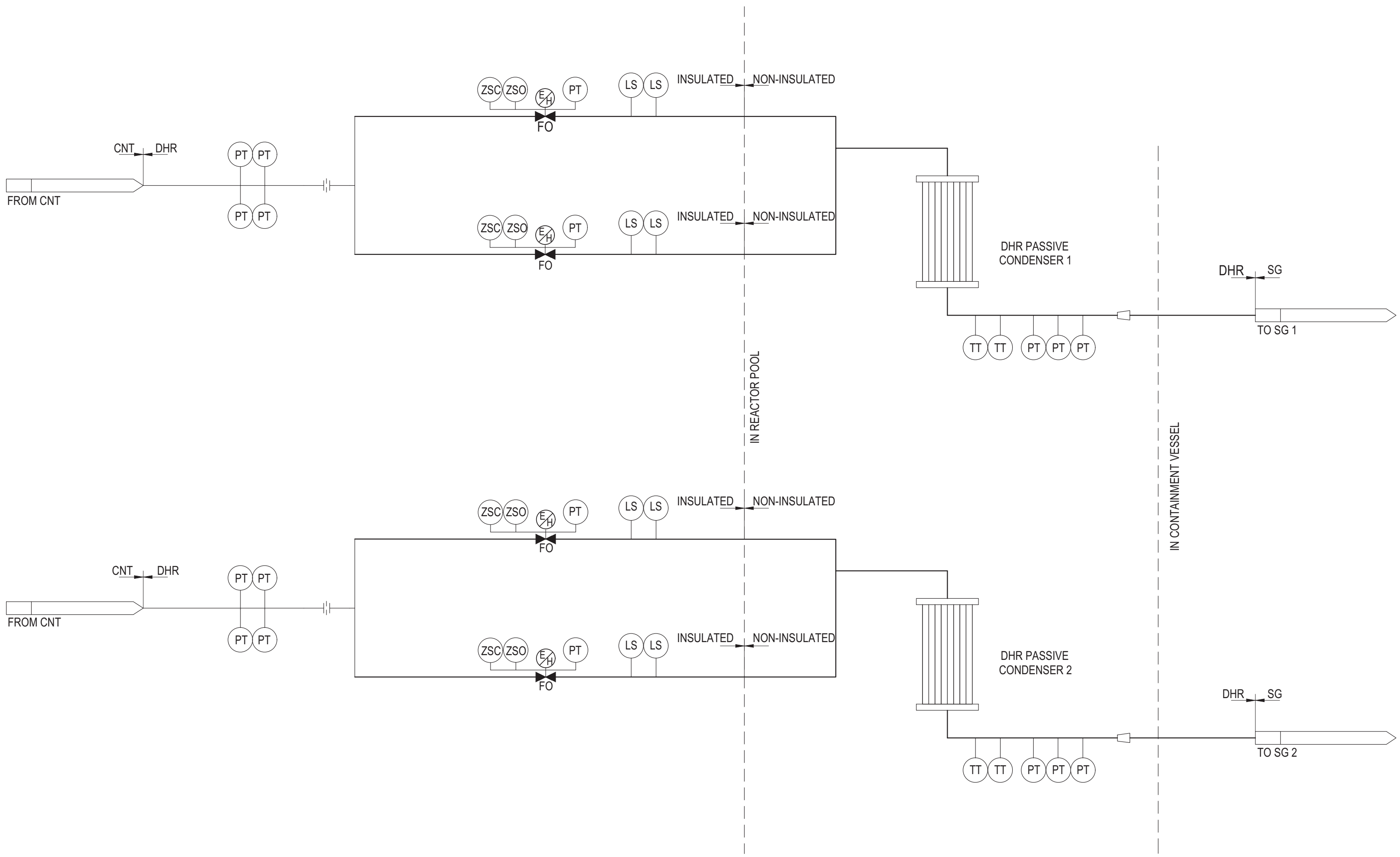


Figure 5.4-9: Primary Coolant Temperature with Decay Heat Removal System Single Train

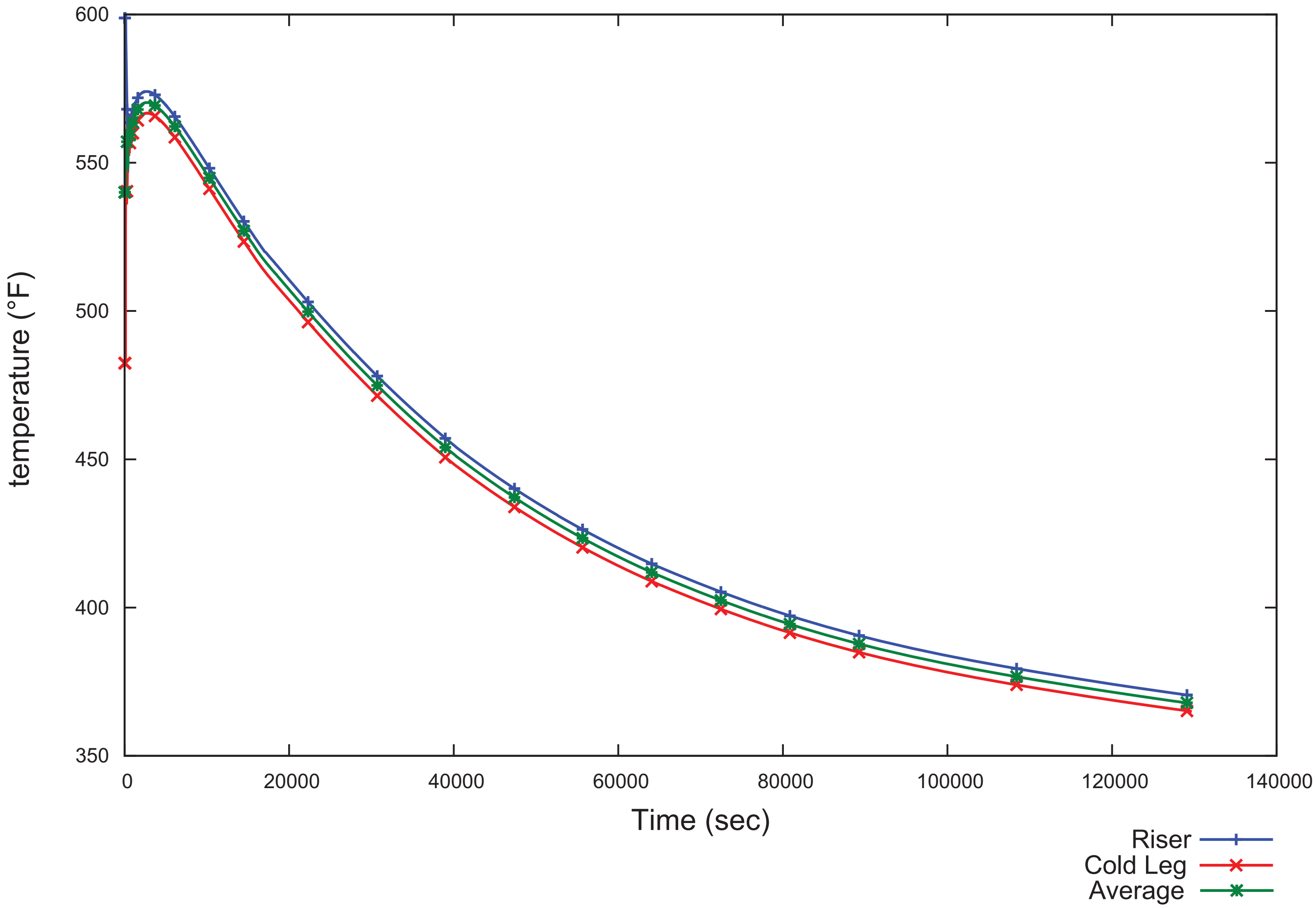


Figure 5.4-10: Primary Coolant Temperature with Decay Heat Removal System Two Trains

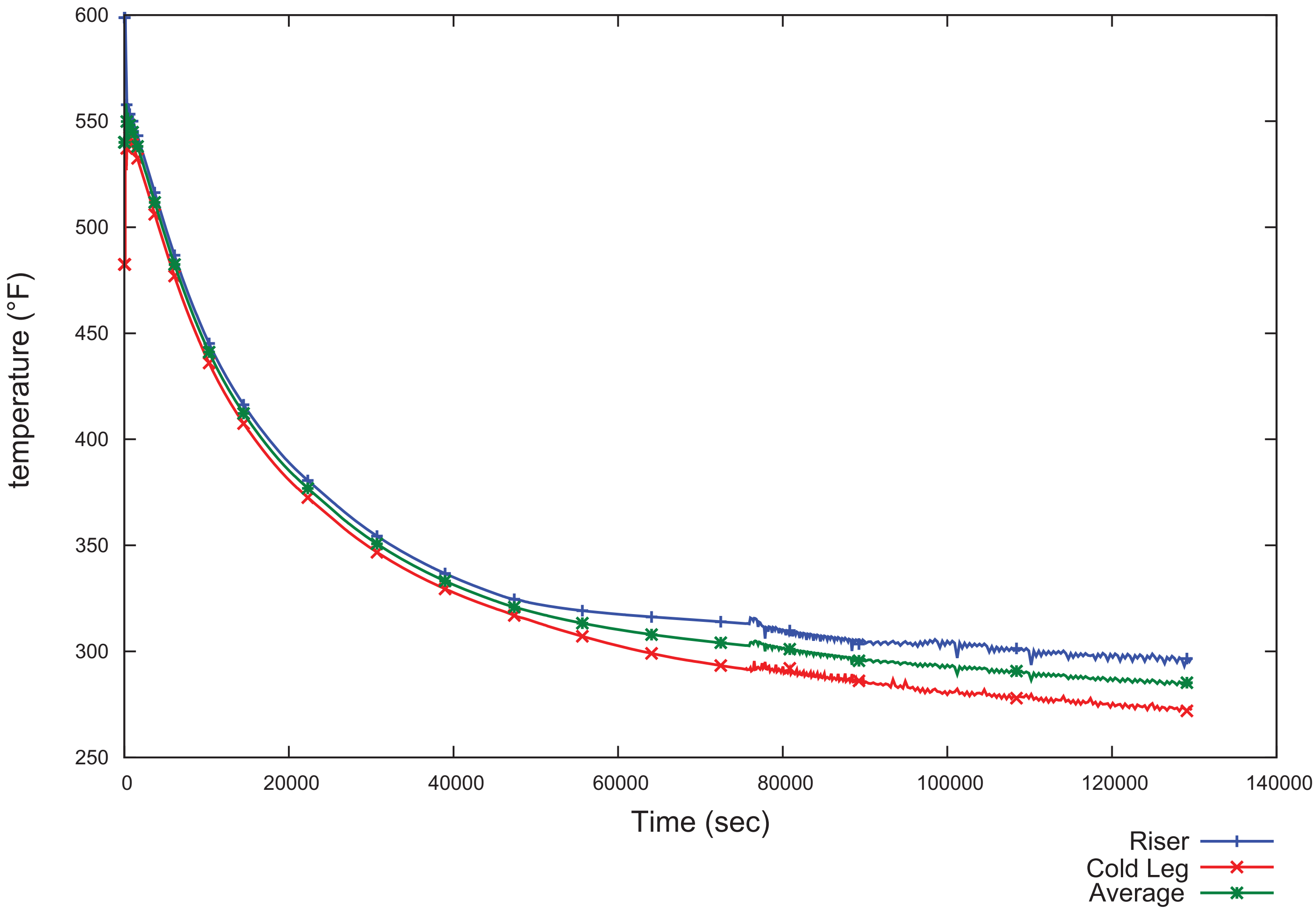


Figure 5.4-11: Primary Coolant Temperature with Decay Heat Removal System Two Trains High Inventory

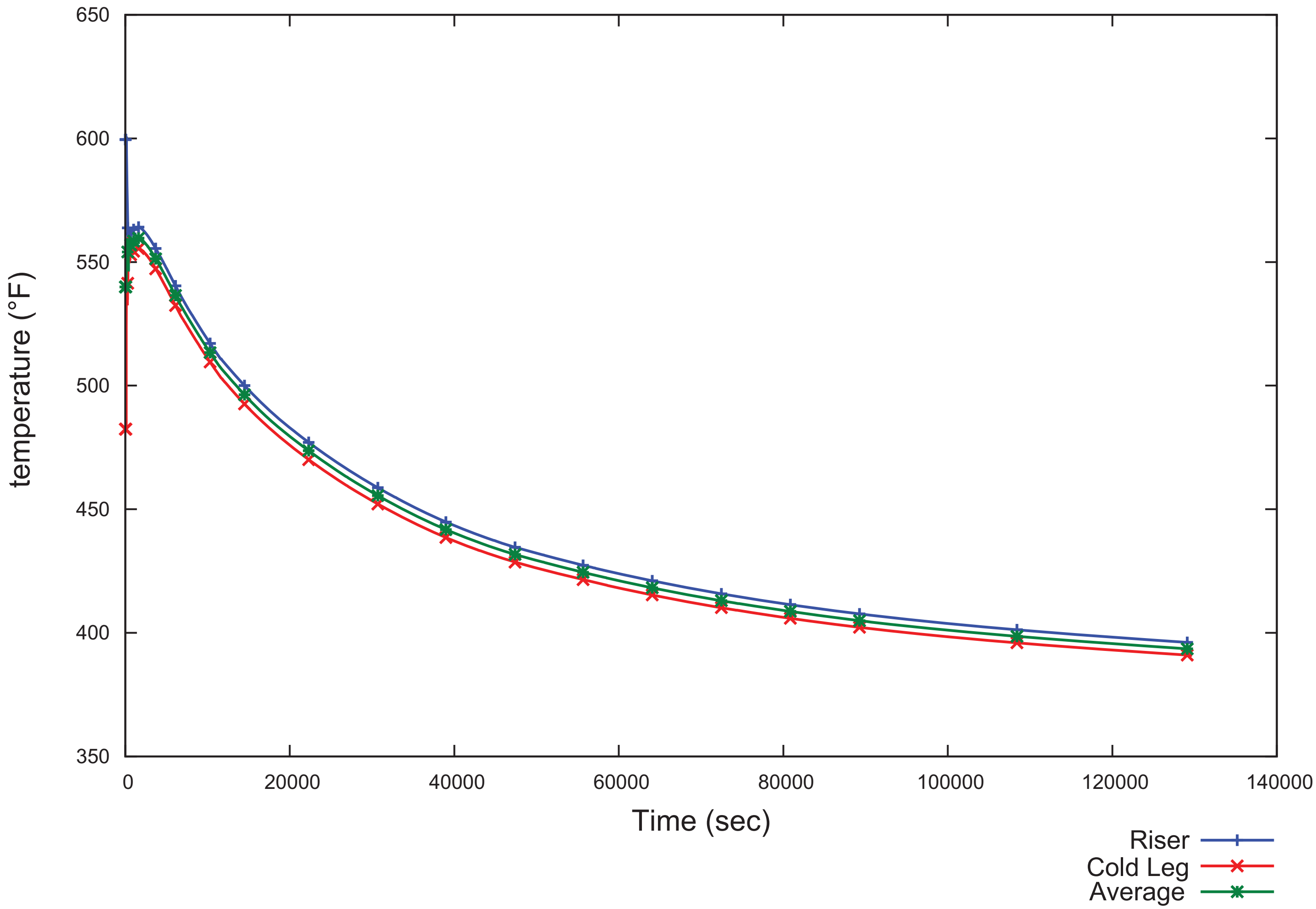


Figure 5.4-12: Primary Coolant Temperature with Decay Heat Removal System Two Trains Low Inventory

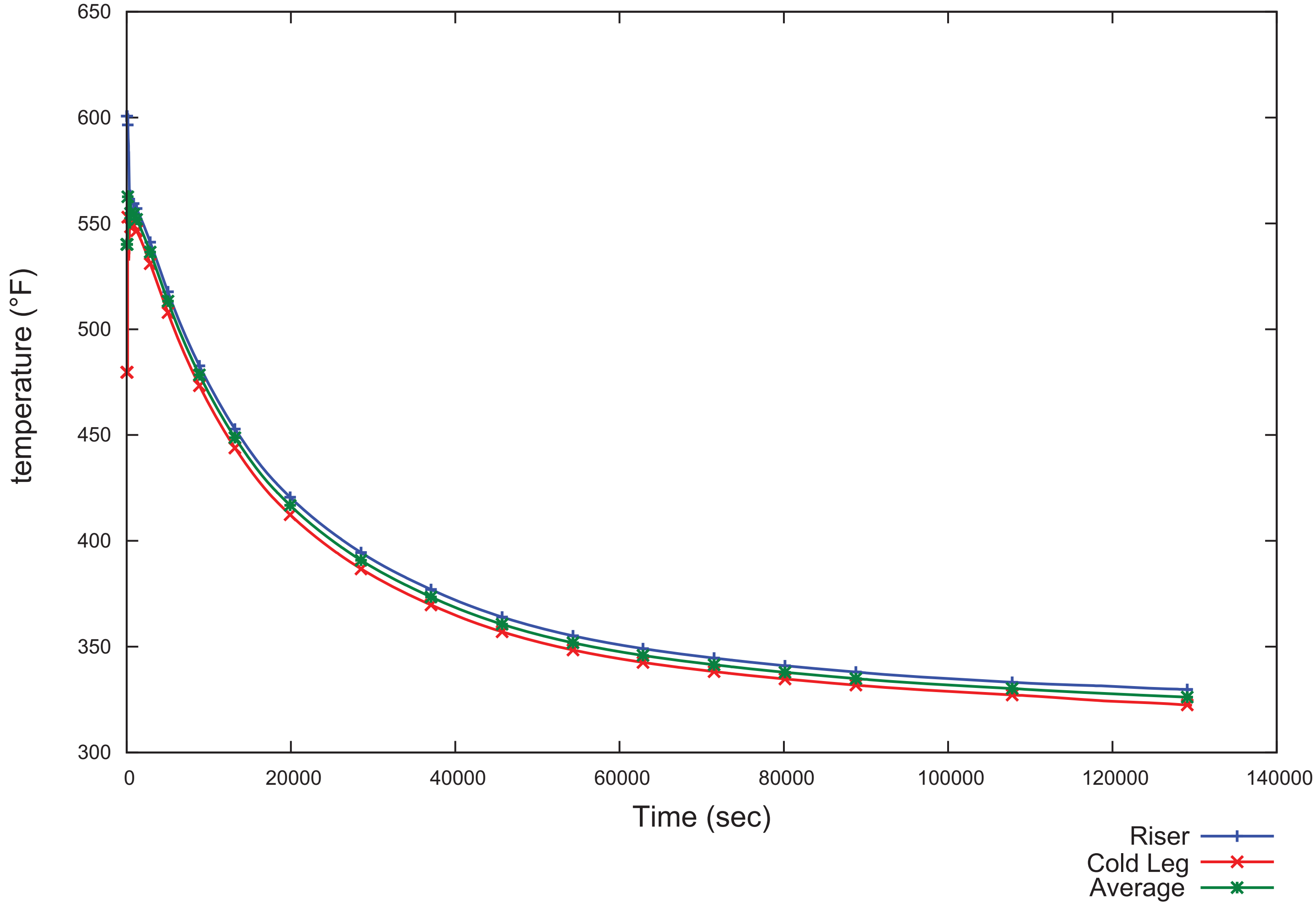
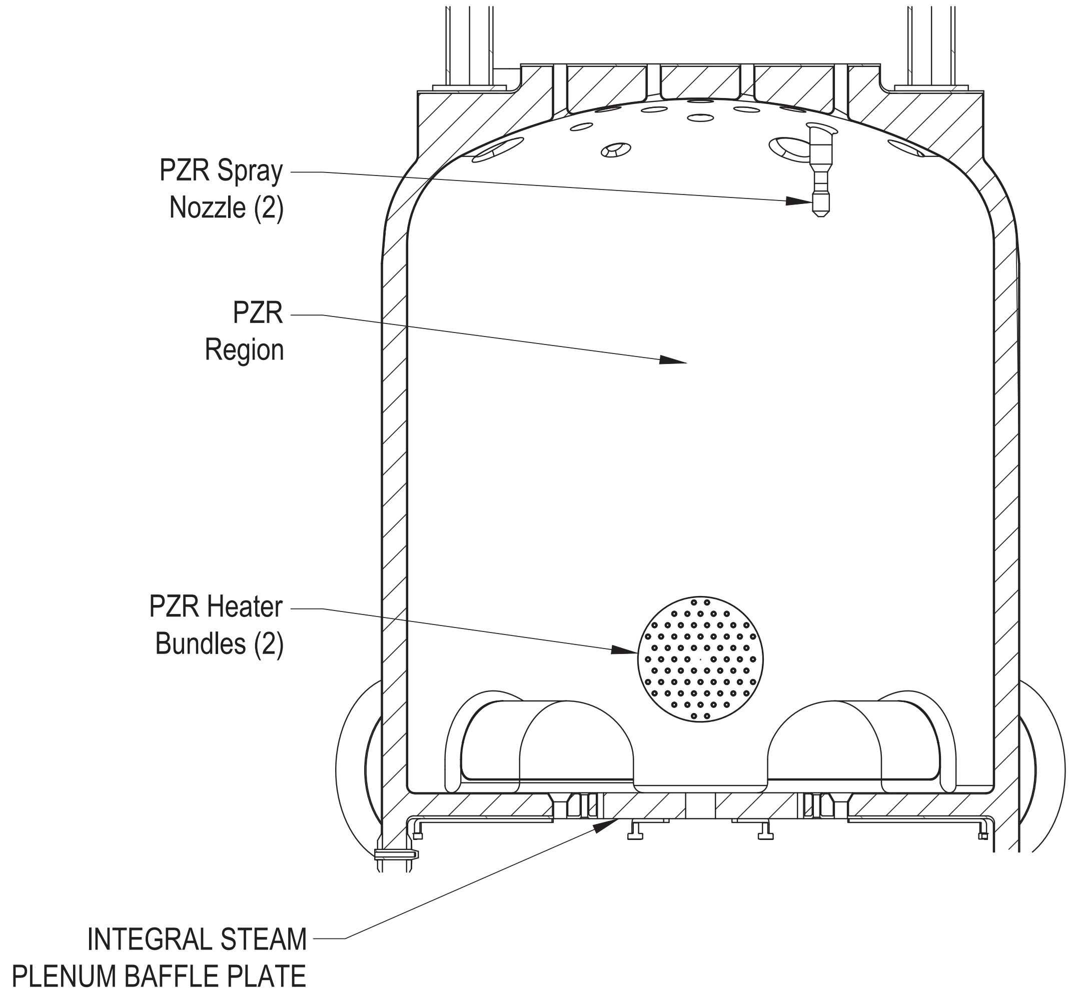


Figure 5.4-13: Pressurizer Region of Reactor Vessel



Section B

The table below provides the NuScale responses to each of the Nuclear Regulatory Commission readiness assessment observations on draft Chapter 5, “Reactor Coolant System and Connecting Systems” of the Standard Design Approval Application.

Section	Observation	Response
5.3	Stainless Steels (SSs) generally have superior ductility relative to low-alloy steels and are less subject to embrittlement effects at similar thermal and neutron flux conditions. Never-the-less, the lack of operating experience and materials data regarding use of SA-965, FXM-19 (and potentially other SS materials used for pressure boundary applications) for reactor vessel construction requires consideration. The staff will require a technical basis supporting that the requirements of General Design Criteria (GDC) 14, 15, and 31 will be met within the SDA that includes the proposed material selections.	NuScale surveyed data for austenitic stainless steels and assessed SA-965 FXM-19 to ensure that it meets General Design Criterion (GDC) 14, GDC 15, and GDC 31. The design supports an exemption from 10 CFR 50.60, which invokes compliance with 10 CFR 50, Appendix G and Appendix H, and from 10 CFR 50.61. The reactor pressure vessel (RPV) design prevents non-ductile fracture in accordance with GDC 14, GDC 15, and GDC 31. Technical Report, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel," TR-130721-P, Revision 0 is referenced in section 5.3 of the Standard Design Approval Application (SDAA) and addresses the technical basis for the material selections and GDC compliance.
5.3	10 CFR 50.60 – Acceptance Criteria for Fracture Prevention The rule does not specify application only to ferrite or low-alloy steel; however, reference is made to Appendices G and H. Alternatives may be granted to Appendices G and H through 50.60(b). As neither 10 CFR Appendix G nor H provide support for SA-965, FXM-19 material, exemptions and proposed alternative treatments will be necessary via 10 CFR 50.60(b). 10 CFR 50.61 – Fracture toughness requirements for protection from pressurized thermal shock events – The rule does not specify application only to ferritic or low-alloy steel; however, the embrittlement trend curves with the rule are fit to low-alloy data that does not apply to the proposed SA-965, FXM-19 material. Exemption would be needed, and a technical basis provided addressing the topics handled for ferritic steels in 10 CFR 50.61. Note – 50.61a may not be used due to 50.61a(b). Appendix G – Fracture Toughness Requirements Appendix G explicitly relates to ferritic materials and does not support determinations concerning fracture toughness in GDC. Stainless steels have superior ductility and are also subject to some thermal and neutron embrittlement effects. A technical basis considering thermal and embrittlement effect will be necessary for the staff to make determinations regarding whether the SDA meets GDC 14, 15, and 31.	The design supports an exemption from 10 CFR 50.60, which includes compliance with 10 CFR 50, Appendix G and Appendix H. The design also supports an exemption from 10 CFR 50.61. The fracture toughness analyses required in 10 CFR 50, Appendix G are not applicable to the materials used for the lower RPV. The RPV design prevents non-ductile fracture in accordance with GDC 14, GDC 15, and GDC 31. The material surveillance program required by 10 CFR 50, Appendix H, is based on nil-ductility reference temperature (RT_{NDT}) per American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) Section III, NB-2331, for ferritic materials. The nil-ductility reference temperature (RT_{NDT}) cannot be established for austenitic stainless steels; therefore, 10 CFR 50, Appendix H, is not applicable to the lower RPV. The available data for austenitic stainless steel supports the conclusion that austenitic stainless steel is highly ductile and less susceptible to the effects of neutron and thermal embrittlement than ferritic materials. Because the NuScale Power Module (NPM) lower RPV uses materials that increase the integrity and safety of the RCPB, the NPM lower RPV

Section	Observation	Response
	Appendix H – Reactor Vessel Materials Surveillance Program Requirements Appendix H explicitly relates to ferritic materials and does not support determinations made concerning material condition for stainless steels. An alternative should be provided. SSs, have superior ductility, and are subject to some thermal and neutron embrittlement effects. A technical basis considering thermal and embrittlement effect will be necessary for the staff to make determinations regarding whether the SDA meets GDC 14, 15, and 31.	design satisfies the intent of 10 CFR 50.60 without applying 10 CFR 50, Appendices G and H. Technical Report, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel," TR-130721-P, Revision 0, is referenced in section 5.3 of the SDAA and addresses the technical basis for the material selections and GDC compliance.
5.3.1	Section 5.3.1.5 contains no comparison of SSI properties and the low-alloy steel components. Discussion should indicate how SS portion of reactor pressure vessel (RPV) is appropriately bounded by (alternative?) Appendix G and low-alloy requirements. This is necessary to demonstrate compliance with GDCs.	The design supports an exemption from 10 CFR 50.60, which includes compliance with 10 CFR 50, Appendix G and Appendix H. The fracture toughness analyses required in 10 CFR 50, Appendix G are not applicable to the materials used for the lower RPV. The RPV design prevents non-ductile fracture in accordance with GDC 14, GDC 15, and GDC 31. Technical Report, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel," TR-130721-P, Revision 0, is referenced in section 5.3 of the SDA and addresses the technical basis for the material selections and GDC compliance.
5.3.1	Section 5.3.1.6 indicates that no surveillance program is necessary as NRC requirements only specify requirements for ferritic materials. This does not appear to be correct as 10 CFR 50.60 does not specify only ferritic materials, and GDCs cannot be met without some form of justification and an exemption. Operating experience concerning the aging (thermal and neutron induced) of SA-965 FXM-19, for example, is extremely limited.	The design supports an exemption from 10 CFR 50.60, which includes compliance with 10 CFR 50, Appendix G and Appendix H. The RPV design prevents non-ductile fracture in accordance with GDC 14, GDC 15, and GDC 31. The material surveillance program required by 10 CFR 50, Appendix H, is based on RT_{NDT} per ASME BPVC Section III, NB-2331, for ferritic materials. RT_{NDT} cannot be established for austenitic stainless steels; therefore, 10 CFR 50, Appendix H, is not applicable to the lower RPV. Technical Report, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel," TR-130721-P, Revision 0, is referenced in section 5.3 of the SDA and addresses the technical basis for the material selections and GDC compliance.

Section	Observation	Response
5.3.2	Section 5.3.2 only addresses requirements for ferritic materials and makes no justification that these will adequately bound the SS portions and any potential thermal aging.	The design supports an exemption from 10 CFR 50.60, which includes compliance with 10 CFR 50, Appendix G and Appendix H. The design also supports an exemption from 10 CFR 50.61. The fracture toughness analyses required in 10 CFR 50, Appendix G are not applicable to the materials used for the lower RPV, and the peak neutron fluence for the upper RPV is below the minimum neutron fluence level required for the assessment of the effects of neutron embrittlement. The RPV design prevents non-ductile fracture in accordance with GDC 14, GDC 15, and GDC 31. The material surveillance program required by 10 CFR 50, Appendix H, is based on RT_{NDT} per ASME BPVC Section III, NB-2331, for ferritic materials. RT_{NDT} cannot be established for austenitic stainless steels; therefore, 10 CFR 50, Appendix H, is not applicable to the lower RPV. Technical Report, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel," TR-130721-P, Revision 0, is referenced in section 5.3 of the SDAA and addresses the technical basis for the material selections and GDC compliance.
5.3.2	Section 5.3.2.3 contains no comparison of SS properties and the low-alloy steel components. Discussion should indicate how SS portion of RPV is appropriately bounded by 10 CFR 50.61 ferritic requirements and whether thermal aging considerations should apply or have been considered. This is necessary to demonstrate compliance with GDCs.	The design supports an exemption from 10 CFR 50.61. The fracture toughness analyses required in 10 CFR 50.61 are not applicable to the materials used for the lower RPV. The RPV design prevents non-ductile fracture in accordance with GDC 14, GDC 15, and GDC 31. The fracture toughness requirements against pressurized thermal shock events required by 10 CFR 50.61 is based on RT_{NDT} per ASME BPVC Section III, NB-2331, for ferritic materials. RT_{NDT} and subsequently reference temperature for pressurized thermal shock (RT_{PTS}) cannot be established for austenitic stainless steels; therefore, 10 CFR 50.61 is not applicable to the lower RPV. Technical Report, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel," TR-130721-P, Revision 0, is referenced in section 5.3 of the SDAA and addresses the technical basis for the material selections and GDC compliance.

Section	Observation	Response
5.3.2	Section 5.3.2.4 does not contain any discussion of thermal effects of aging on components and provides no basis to establish the conclusions therein for SS components. A basis is necessary to demonstrate compliance with the GDCs.	The RPV design prevents non-ductile fracture in accordance with GDC 14, GDC 15, and GDC 31. Technical Report, "Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel," TR-130721-P, Revision 0, is part of the SDAA documentation and addresses the technical basis for the material selections and GDC compliance.

Section C

TR Number	TR Title
TR-118976-NP Revision 0	Fluence Calculation Methodology and Results
TR-130877-NP Revision 0	Pressure and Temperature Limits Methodology
TR-130721-NP Revision 0	Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel

Licensing Technical Report

Fluence Calculation Methodology and Results

December 2022

Revision 0

Docket: 52-050

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Abstract

This Technical Report provides the methodology developed by NuScale Power, LLC, to calculate the neutron fluence for the NuScale Power Module reactor pressure vessel (RPV) and containment vessel (CNV). Estimations of the bias and uncertainty associated with the fluence calculations, derived from benchmarking and sensitivity studies, are presented along with associated end-of-life fluence predictions for the RPV, CNV, and other locations.

NuScale's fluence methodology uses the Monte Carlo N-Particle Transport Code 6 and is based on the guidance found in Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence.". Alternatives to particular Regulatory Guide 1.190 regulatory positions are described and justified. Measured data from the Vulcain Experimental Nuclear Study 3 pressure vessel simulator benchmark are used to validate the NuScale methodology.

Executive Summary

This report provides the methodology for predicting the end-of-life fluence for the NuScale reactor pressure vessel (RPV) and containment vessel (CNV).

A best-estimate neutron fluence calculation for the Nucale Power Module (NPM) is performed using the Monte Carlo N-Particle Transport Code 6 (MCNP6) version 1.0 based on Nuclear Regulatory Commission Regulatory Guide 1.190. Alternatives to particular Regulatory Guide 1.190 regulatory positions are provided. Biases and uncertainties associated with the MCNP6 best-estimate neutron fluence model are also reported. These biases and uncertainties are established through benchmarking against the Vulcain Experimental Nuclear Study 3 experiment and NPM-specific sensitivity studies associated with key MCNP6 modeling simplifications and inputs.

The peak RPV beltline surface and CNV beltline at ¼-T fluence over a 60-year NPM operating life (assumed 95 percent capacity factor) is calculated and provides acceptable results. Neutron fluence estimates provided in this report are acceptable for supporting Final Safety Analysis Report Section 4.3 and Section 5.3 for the US460 standard design, and meet the regulatory guidance and requirements discussed in this report.

1.0 Introduction

1.1 Purpose

This report describes the methodology used to calculate the neutron fluence for the NuScale Power Module (NPM) reactor pressure vessel (RPV) and containment vessel (CNV). It also provides estimations of biases and uncertainties associated with these fluence calculations, derived from benchmarking and sensitivity studies, along with associated end-of-life fluence predictions for the RPV, CNV, and other locations.

1.2 Scope

This report provides the methodology for predicting the end-of-life fluence for the NuScale RPV and NuScale CNV as well as the associated results of applying the methodology to support the Final Safety Analysis Report (FSAR) Section 4.3 and Section 5.3 for the US460 standard design. The testing program associated with confirming these fluence predictions in the operating plant, the methodology for adjusting best-estimate fluence predictions throughout an NPM's operating life, and the effects on material properties caused by the fluence are outside the scope of this report.

1.3 Abbreviations and Definitions

Table 1-1 Abbreviations

Term	Definition
CMS	core management software
CNV	containment vessel
LCP	lower core plate
MeV	megaelectron volt
NPM	NuScale Power Module
RG	Regulatory Guide
RPV	reactor pressure vessel
UCP	upper core plate
VENUS-3	Vulcain Experimental Nuclear Study 3

Table 1-2 Definitions

Term	Definition
Fluence	In the context of this report, the term "fluence" is taken to mean the fast neutron fluence, which is the time-integrated flux of neutrons with an energy greater than 1 megaelectron volt (MeV).

2.0 Background

Neutron fluence is known to affect the material properties of RPV materials. The extent of the effect is influenced by the magnitude of the fluence, among other factors.

Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," (Reference 7.1) provides guidance for calculating pressure vessel neutron fluence. NuScale's fluence calculation methodology is based on RG 1.190. Descriptions of, and justifications for, alternatives to portions of RG 1.190 regulatory positions are provided in Appendix C.

The NuScale CNV is in close proximity to the RPV compared to a typical large light water reactor and the same methodology used to calculate RPV fluence is taken to be directly applicable to calculating CNV fluence.

2.1 Regulatory Requirements

The regulatory requirements pertaining to vessel fluence analysis are:

- 10 CFR Part 50 Appendix A, General Design Criterion 14 as it relates to ensuring an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture of the reactor coolant pressure boundary, in part, insofar as it considers calculations of neutron fluence
- General Design Criterion 31 as it relates to ensuring the reactor coolant pressure boundary behaves in a nonbrittle manner and the probability of rapidly propagating fracture is minimized, in part, insofar as it considers calculations of fluence
- 10 CFR Part 50, Appendix G, as it relates to RPV material fracture toughness requirements, in part, insofar as it considers calculations of neutron fluence
- 10 CFR Part 50, Appendix H, as it relates to RPV material surveillance program requirements, in part, insofar as it considers calculations of neutron fluence
- 10 CFR 50.61 as it relates to fracture toughness criteria for pressurized water reactors relevant to pressurized thermal shock events, in part, insofar as it considers calculations of neutron fluence

The following applicable NRC acceptance criteria are listed for the vessel fluence analysis methodology:

- There is reasonable assurance that the proposed design limits can be met for the expected range of reactor operation, taking into account analysis uncertainties.
- There is reasonable assurance that during normal operation the design limits are not exceeded.
- The acceptance criteria of RG 1.190 (Reference 7.1)
- The acceptance criteria of RG 1.99 (Reference 7.2)

3.0 Analysis

3.1 Approach/Methodology

NuScale's fluence calculation methodology uses Monte Carlo N-Particle Transport Code 6 version 1.0 (MCNP6), which was released in 2013 by Los Alamos National Laboratory and merges MCNP5 and MCNPX functions. The MCNP6 is a general-purpose Monte Carlo method code used for neutron, photon, electron, or coupled neutron/photon/electron transport (Reference 7.5). The code treats an arbitrary three-dimensional configuration of materials in geometric cells. The Monte Carlo method has the advantage of allowing an exact representation of the reactor's three-dimensional geometry. In addition, the Monte Carlo method allows a continuous energy description of the nuclear cross-sections and flux solution.

NuScale calculates three-dimensional exposure and power distribution data for each fuel assembly using core management software (CMS) codes CASMO5 and SIMULATE5. CASMO5 is a lattice physics code that characterizes reactor fuel assembly designs. SIMULATE5 is a three-dimensional core simulator code for core design and core load calculations. Information from CASMO5 and SIMULATE5 is used as inputs to the MCNP6 based fluence calculation.

The variance reduction scheme used in NuScale's fluence calculation methodology is the mesh based weight window produced by Automated Variance Reduction Generator (ADVANTG) software (Reference 7.4), which is developed, maintained, and distributed by Oak Ridge National Laboratory.

3.2 Geometry

Calculations are performed using a three-dimensional MCNP6 model.

An illustration of the vertical cross-sectional view of the lower section of the NPM is shown in Figure 3-1. The vertical cross-sectional view of the MCNP6 NuScale best-estimate fluence model is presented in Figure 3-2 and the horizontal cross-sectional view is presented in Figure 3-3.

The NuScale best-estimate fluence model is representative of the US460 standard NPM design with the following general exceptions and modeling simplifications.

- The geometry is specified using cold dimensions, and thermal expansion is not modeled. Thermal expansion for hot full power dimensions is accounted for in NuScale's Studsvik Scandpower CMS codes (SIMULATE5 and CASMO5), whose outputs are used as inputs to establish the neutron source distribution in the MCNP6 model. The effect of this modeling simplification and the effect of this difference between MCNP6 and CMS treatment of cold dimensions on the fluence estimate is provided in Section B.1.3 and Section B.1.4.

- The NuScale best-estimate fluence model contains an axially homogenized representation of the active fuel region of the fuel assemblies. This modeling simplification is implemented for consistency because fuel assembly power information is taken from NuScale's SIMULATE5 model output, which is a homogenized model. A sensitivity study comparing this homogenized treatment to an MCNP6 model that explicitly models the fuel across {{ }}2(a),(c) is provided in Section B.1.1.
- Each fuel assembly consists of {{ }}2(a),(c). The active fuel pin region consists of a {{ }}2(a),(c). On the basis of engineering judgment, the impact of this modeling simplification on the fluence estimates is negligible.
- The top nozzle skirt and upper core plate are modeled explicitly as part of the fuel assembly for assemblies that do not contain control rod assemblies. On the basis of engineering judgment, the impact of this modeling simplification on the fluence estimates is negligible.
- The NuScale best-estimate fluence model accurately represents the NPM reactor pressure vessel and CNV bottom head designs, as can be seen by comparing Figure 3-1 and Figure 3-2.
- The RPV bottom core support block is not explicitly modeled. The RPV beltline region is the main region of interest for the vessel fluence estimation. On the basis of engineering judgment, the impact of these modeling simplification on the RPV beltline region fluence estimates is negligible.
- All water densities in the NuScale best estimate fluence model are {{ }}2(a),(c). The effect of this modeling simplification on the fluence estimate is provided in Section B.1.12.
- All temperatures of components in the NuScale best-estimate fluence model are {{ }}2(a),(c). On the basis of engineering judgment, the impact of this modeling simplification on the fluence estimates is small relative to the effect of using a single water coolant density for the primary coolant.
- There are existing negligible differences between the calculated time weighted exposure power profiles presented in both Table 3-1 and Figure 3-5, compared with fission neutron generation probabilities entered in MCNP input files. The impact of this modeling differences on the fluence estimates is negligible.

Figure 3-1 Vertical Cross-Sectional View of the Lower Section of the NuScale Power Module

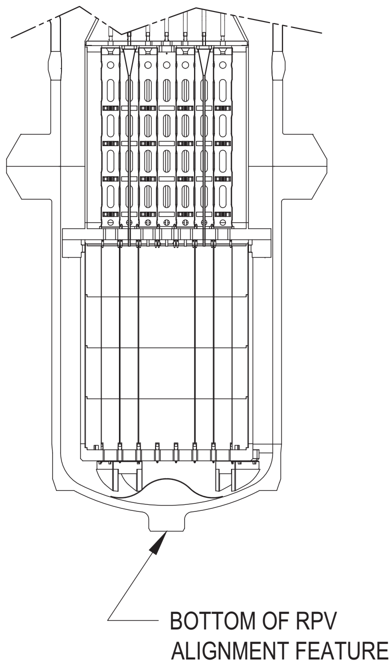


Figure 3-2 Vertical Cross-Sectional View of the Monte Carlo N-Particle Transport Code 6 Fluence Homogenized Model

{{

}}2(a),(c)

Figure 3-3 Horizontal Cross-Sectional View of the Monte Carlo N-Particle Transport Code 6 Fluence Homogenized Model

{{

}}2(a),(c)

3.3 Material Compositions

The material composition information used in the MCNP6 NuScale best-estimate fluence model is based on the typical isotopic contents associated with the materials associated with the NPM design. Cold dimensions are used and thermal expansion is not taken into account in the determination of material densities. The effect of this modeling simplification on the fluence estimate is discussed in Section B.1.3 and Section B.1.4.

The core composition of the MCNP6 base model is based on the core composition of the SIMULATE5 base model core design. The NuScale best-estimate fluence model does not contain ^{239}Pu because it is based on a fresh core (beginning of Cycle 1). A bias and uncertainty to account for the contribution of ^{239}Pu buildup to fluence is derived in Section B.1.2.

The material composition of the homogenized active fuel comprises fuel at an averaged 3.5 percent enrichment, fuel cladding, borated water, and guide tubes.

3.4 Cross-Sections

NuScale's MCNP6 based fluence calculation methodology uses the ENDF/B-VII.1 nuclear data for continuous energy cross-section libraries.

A .92c file extension is used to represent isotopic cross-section data with a temperature at {{ }}$^{2(a),(c)}$. The ENDF/B-VII.1 data libraries have cross-sections processed at selected temperatures {{ }}$^{2(a),(c)}$. The MAKXSF code is used to derive the {{ }}$^{2(a),(c)}$ library from {{ }}$^{2(a),(c)}$ and {{ }}$^{2(a),(c)}$ libraries. The {{ }}$^{2(a),(c)}$ file extension is also copied into the new data library and used for pool water at {{ }}$^{2(a),(c)}$, which has a negligible impact to vessel component fluence.

The temperature card "TMP" is used in MCNP6 to provide the time-dependent cell thermal temperatures necessary for the free-gas thermal treatment of low-energy neutron transport at the correct material temperatures. The temperature card "TMP" requires inputs to be in units of megaelectronvolts (MeV), so a conversion is performed. For example, NuScale uses {{ }}$^{2(a),(c)}$ as the averaged temperature of moderator and this temperature in K is converted to MeVs as shown in Equation 3-1.

{{

Equation 3-1

}}$^{2(a),(c)}$

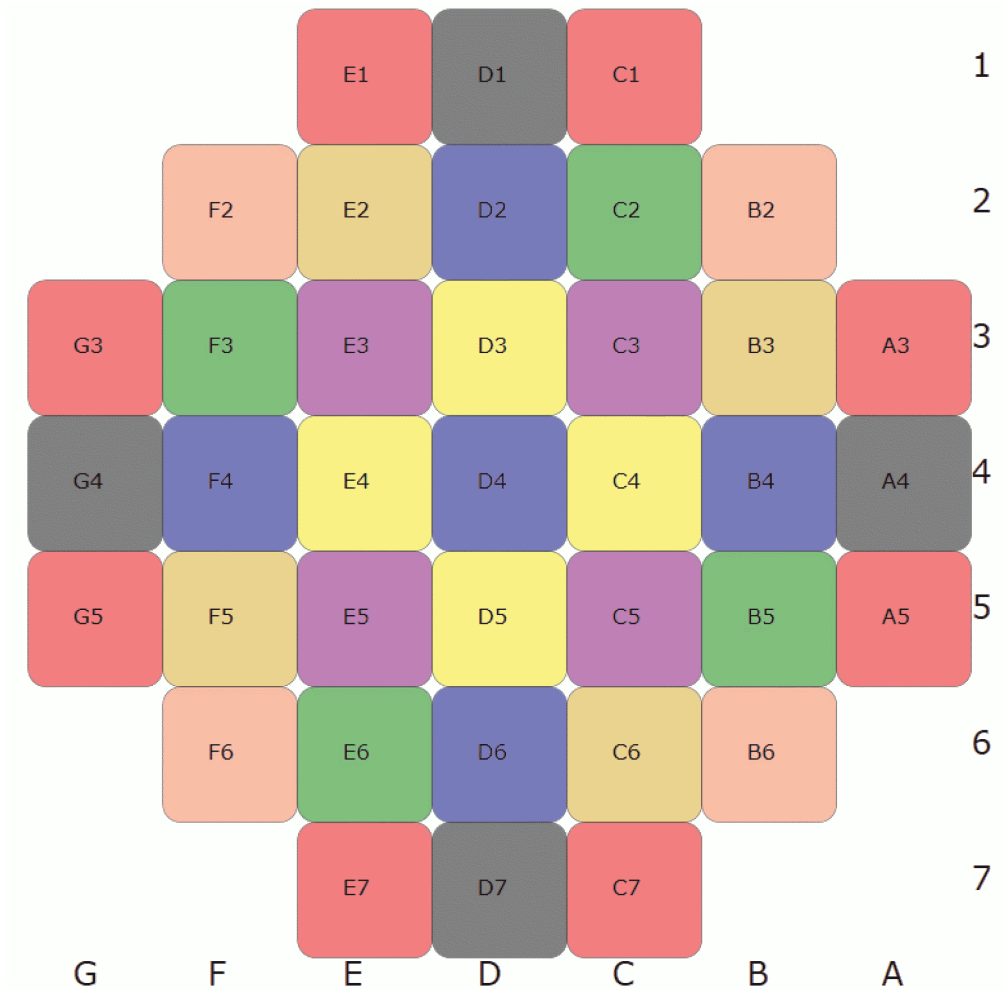
3.5 Neutron Source

For the NuScale best-estimate fluence model, the energy spectrum of the fission neutrons emitted from the fuel assemblies is taken as the Watt fission spectrum for ^{235}U. Sensitivity studies on the effect of ^{239}Pu buildup are presented in Section B.1.2.

There are no delayed neutrons separately modeled because the fission modeling is turned off by using the "NONU" card in MCNP6 input decks for neutron transport. For the purpose of the NuScale best estimate of fast neutron fluence, the delayed neutron contribution to fast neutron fluence is negligible.

For the purposes of this report, the fuel assemblies are referred to according to the naming index shown in Figure 3-4.

Figure 3-4 Fuel Assembly Naming Index



SIMULATE5 is used to calculate the core average axial power profile associated with each cycle in a lifetime refueling scheme for {{ }}$^{2(a),(c)}$. The axial power profiles associated with each cycle are averaged to produce a lifetime exposure averaged axial power profile shown in Table 3-1. Table 3-1 is used to establish the vertical sampling of the neutron source used in the MCNP6 NuScale best-estimate fluence model. SIMULATE5 is used to calculate the assembly averaged radial power profile associated with each cycle in an 8-cycle refueling scheme. The assembly averaged radial power profile associated with each cycle are averaged to produce a lifetime exposure averaged radial power profile shown in Figure 3-5. The radial sampling of the neutron source used in the MCNP6 NuScale best-estimate fluence model is based on Figure 3-5.

Table 3-1 Lifetime Exposure Averaged Core Axial Power Profile

{{

}}2(a),(c),ECI

Figure 3-5 Lifetime Exposure Averaged Assembly Averaged Radial Power Profile

{{

}}2(a),(b),(c),ECI

MCNP6 produces flux results that are on a "per source particle" basis and part of converting to final reported results involves establishing the source intensity. The total fission neutron source intensity S (neutrons/second) in the NPM at a given power is determined by Equation 3-2:

$$S = \frac{\upsilon P \times 10^{6}\left(\frac{W}{MW}\right)}{1.602 \times 10^{-13}\left(\frac{J}{MeV}\right) K_{eff} Q_{ave}}$$ Equation 3-2

where,

υ = Average number of neutrons produced per fission in NPM (neutrons/fission); calculated from results in the MCNP6 output file to be υ=2.46 at initial cycle for a fresh core with 3.5 percent ^{235}U enrichment at hot zero power,

P = Fission power (MW); taken to be 250 MW based on NPM's thermal power rating,

K_{eff} = Effective multiplication factor; taken to be 1.000 for critical light water reactor, and

Q_{ave} = The average recoverable energy per fission for all fissionable materials (MeV/fission); taken to be 198 MeV/fission as a best estimate based on other low enriched uranium systems.

The calculated fission neutron intensity for the NPM is estimated as:

$$S = \frac{2.46\frac{neutrons}{fission} * 250MW \times 10^{6}\left[\frac{W}{MW}\right]}{1.602 \times 10^{-13}\left(\frac{J}{MeV}\right) * 1.000 * 198\frac{MeV}{fission}} = 1.94 \times 10^{19}\frac{\text{neutrons}}{\text{second}} \qquad \text{Equation 3-3}$$

A factor of 1.8 x 10^9 seconds (57 effective full-power years) is then used to convert from flux to fluence based on a 60-year operating life with a 95 percent power capacity factor.

3.6 Other Modeling Considerations

There is no upper limit placed on the neutron source energy, and neutrons are treated with implicit capture in the NuScale best-estimate fluence model. A lower cut off energy of 0.9 MeV is used. Because there are no processes modeled that would result in a higher energy neutron, the implementation of the 0.9 MeV lower cut off energy makes no difference to the >1 MeV neutron fluence results.

A series of cylindrical mesh tallies are used to specify the locations of interest where fluence is calculated throughout the MCNP6 model.

Example illustrations of mesh tallies used in the calculation of RPV and CNV fluence are shown in Figure 3-6 and Figure 3-7, including naming and numbering conventions for the axial and azimuthal segments. The effect of the tally region volume impact on final fluence results is discussed in Section B.1.14.

{{

}}2(a),(c)

3.7 Variance Reduction Scheme

{{

}}2(a),(c)

{{

}}2(a),(c)

Figure 3-6 Horizontal Cross-Sectional View of the Reactor Pressure Vessel Mesh Tally

{{

}}2(a),(c)

Figure 3-7 Horizontal Cross-Sectional View of the Containment Vessel Mesh Tally

{{

}}2(a),(c)

Figure 3-8 Y-Z Plot of the Mesh-Based Weight Window Structure

{{

}}2(a),(c)

Figure 3-9 Example of X-Y Plot of ADVANTG Generated Mesh-Based Weight Window

{{

}}2(a),(c)

Figure 3-10 Y-Z Plot of the Global Fast Neutron Fluence

{{

}}2(a),(c)

Figure 3-11 X-Y Plot of the Global Statistic Check on the Fast Neutron Fluence Relative Error

{{

}}2(a),(c)

Figure 3-12 Y-Z Plot of the Global Statistic Check on the Fast Neutron Fluence Relative Error

{{

}}2(a),(c)

4.0 Bias and Uncertainty

4.1 Quantified Biases and Uncertainties

Appendix A describes the NuScale best-estimate fluence prediction benchmarking work. Appendix B describes sensitivity analysis associated with the best-estimate fluence calculation. A summary of the relevant results associated with the NuScale best-estimate fluence bias and uncertainty, and a reference to the applicable report section, are provided in Table 4-1.

Table 4-1 List of Quantified Systematic Biases and Random Uncertainties

{{

}}2(a),(c)

4.2 Combination of Biases

The analytical bias (also known as B_a^c per RG 1.190) is composed of known uncertainties that are biased in a certain direction compared to the best-estimate fluence calculation. For the NuScale best-estimate fluence calculation, B_a^c is calculated as the algebraic summation of systematic biases presented in Table 4-1, excluding B_b^c, as shown in Equation 4-1.

$$B_a^c = B_{homo} + B_{Pu} + B_{Pin} + B_{ax} \qquad \text{Equation 4-1}$$

A tendency for NuScale's MCNP6 based-fluence calculation methodology to {{

}}2(a),(c).

The total bias (B_T) of the best estimate fluence calculation is quantified as shown in Equation 4-2:

{{

Equation 4-2

}}2(a),(c)

4.3 Combination of Uncertainties

Independent random uncertainties have no specific direction associated with them with respect to their effect on the final fluence estimate. The overall uncertainty (σ^c) is established per Equation 4-3 for the NuScale best-estimate fluence MCNP6 model.

$$\sigma^c = \sqrt{\left(\sigma_a^c\right)^2} \qquad \text{Equation 4-3}$$

$$\sigma_a^c = \sqrt{\left(\sigma_b^c\right)^2 + \sigma_{Pin}^2 + \sigma_{Pu}^2 + \sigma_{water}^2 + \sigma_m^2 + \sigma_g^2 + \sigma_{ap}^2 + \sigma_{pa}^2 + \sigma_{pr}^2 + \sigma_{Boron}^2 + \sigma_{tally}^2 + \sigma_{mt}^2} \qquad \text{Equation 4-4}$$

{{

Equation 4-5

}}2(a),(c)

Where σ_{mt} is the relative error associated with the particular location's reported result from MCNP6 output and σ_a^c is the square root of the sum of the squares of random uncertainties in Table 4-1, as shown in Equation 4-4.

Substituting the value established for σ_a^c back into Equation 4-4 gives Equation 4-5. Equation 4-5 is used to establish overall uncertainties given in Equation 4-6.
{{

Equation 4-6

}}2(a),(c)

A single {{

}}2(a),(c). Section B.1.11 contains more details.

5.0 Results

5.1 NuScale Power Module Fluence Prediction Results

Table 5-1 presents the results of the best estimate fluence analysis. {{

}}2(a),(c) established in Section 4.2, to the "MCNP Calculated Neutron Fluence."

Table 5-1 Best Estimate of Fluence Expected in Various NuScale Power Module Components and Locations

{{

}}2(a),(c)

Table 5-1 Best Estimate of Fluence Expected in Various NuScale Power Module Components and Locations (Continued)

{{

}}2(a),(c)

Table 5-1 Best Estimate of Fluence Expected in Various NuScale Power Module Components and Locations (Continued)

{{

}}2(a),(c)

6.0 Summary and Conclusions

A best-estimate neutron fluence calculation for the NPM is performed using of the MCNP6 code based on RG 1.190. Alternatives to particular RG 1.190 regulatory positions are provided in Appendix C. Biases and uncertainties associated with the MCNP6 best-estimate neutron fluence model are reported in Table 4-1, which are established through benchmarking against the VENUS-3 experiment and NPM-specific sensitivity studies associated with key MCNP6 modeling simplifications and inputs.

The peak RPV beltline surface and CNV beltline at ¼-T fluence over a 60-year NPM operating life (assumed 95 percent capacity factor) is calculated to be {{

}}[2(a),(c)], as reported in Table 5-1. Neutron fluence estimates provided in this report are acceptable for supporting Final Safety Analysis Report Section 4.3 for the US460 standard design and meet the regulatory guidance and requirements discussed in Section 2.1 of this report.

7.0 References

7.1 U.S. Nuclear Regulatory Commission, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," Regulatory Guide 1.190, Revision 0, March 2001.

7.2 U.S. Nuclear Regulatory Commission, "Radiation Embrittlement of Reactor Vessel Materials," Regulatory Guide 1.99, Revision 2, May 1988.

7.3 Los Alamos National Laboratory, Trellure, H.R. and Poston, D.I., "User's Manual, Version 2.0 for Monteburns, Version 5B," LA-UR-99-4999, Los Alamos, NM, September 1999.

7.4 Oak Ridge National Laboratory, “ADVANTG-An Automated Variance Reduction Parameter Generator" ORNL/TM2013/416, Rev. 1, Oak Ridge, TN, August 2015.

7.5 Los Alamos National Laboratory, DB Pelowitz, "Monte Carlo N-Particle Transport Code 6 Users Manual, Version 1.0," LA-CP-13-00634, Rev. 0, Los Alamos, NM, May 2013.

7.6 Oak Ridge National Laboratory, Radiation Safety Information Computational Center, "Shielding Integral Benchmark and Database," DCL-237, SINBAD-2013.12, Oak Ridge, TN, December 2013.

7.7 Organisation for Economic Co-operation and Development, Nuclear Energy Agency, Nuclear Science Committee, "Prediction of Neutron Embrittlement in the Reactor Pressure Vessel: VENUS-1 and VENUS-3 Benchmarks," OECD, 2000.

8.0 Appendices

The following Appendices are included in this report:

- Appendix A - Benchmarking Monte Carlo N-Particle Transport Code 6 for Fluence Applications
- Appendix B - NuScale Power Module Fluence Prediction Sensitivity Studies and Uncertainty Analysis
- Appendix C - Alternative Approaches to Regulatory Guide 1.190 Regulatory Positions

Appendix A Benchmarking Monte Carlo N-Particle Transport Code 6 for Fluence Applications

A.1 Vulcain Experimental Nuclear Study 3 Benchmark

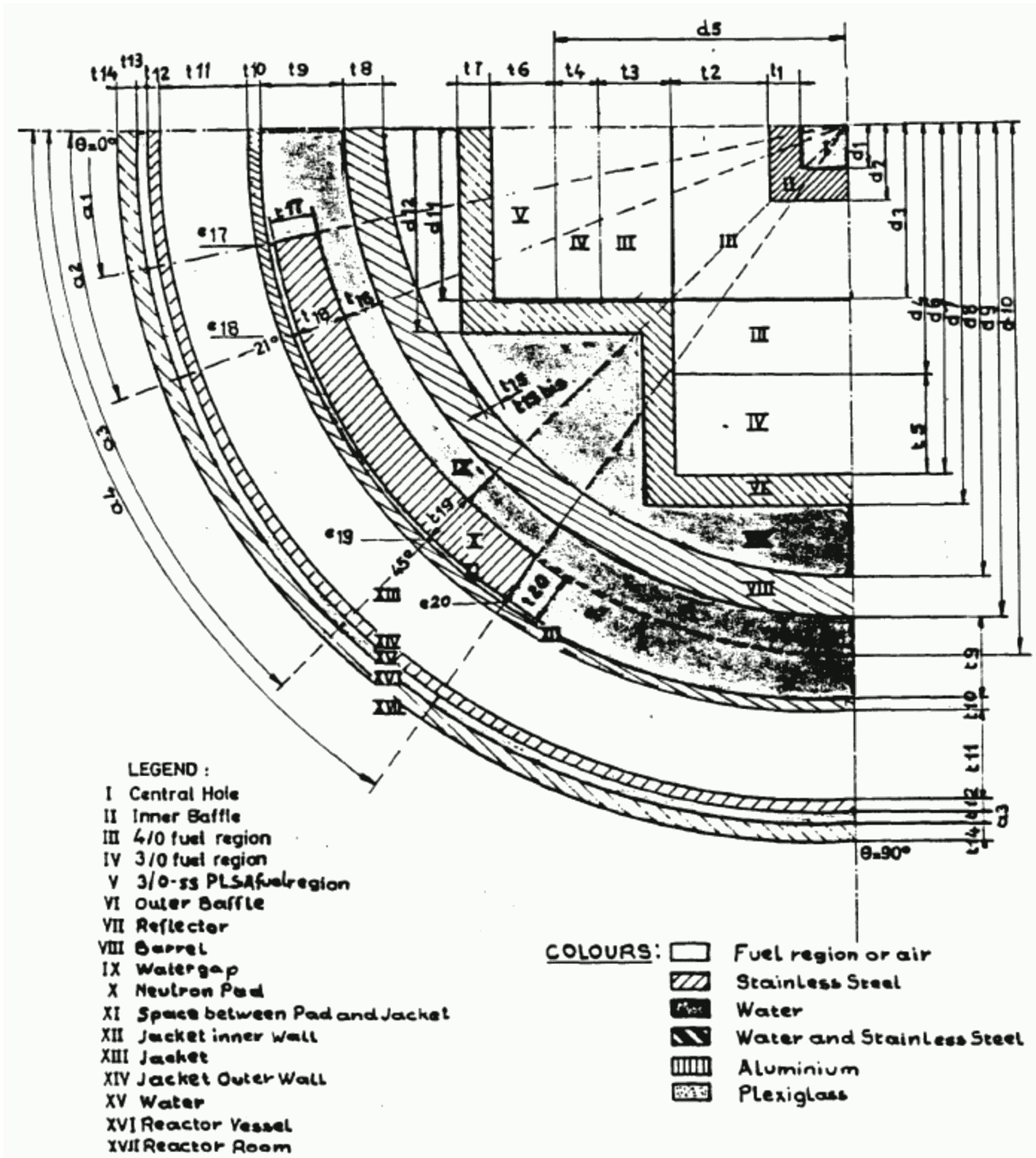
This appendix presents a description of benchmarking work performed to demonstrate that MCNP6 can perform neutron flux determinations that compare favorably with expected or experimental results. The benchmarking work shown in this appendix is also used to establish the bias and uncertainty stemming from use of the MCNP6 transport code and associated cross section data.

A.1.1 Modeling

MCNP6 code version 1.0 is used to create a model of the third configuration in the Vulcain Experimental Nuclear Study, commonly known as "VENUS-3." The VENUS-3 pressure vessel fluence benchmark is based on documentation from the Shielding Integral Benchmark Archive and Database from the Radiation Safety Information Computational Center (Reference 7.6). The VENUS-3 benchmark provides reaction rates associated with various detector types for the core barrel of an experimental reactor setup. The VENUS-3 benchmark is considered to be generally applicable to the NPM.

The basic configuration of the VENUS-3 benchmark is shown in Figure A-1.

Figure A-1 Horizontal Cross-Sectional View of the Vulcain Experimental Nuclear Study 3 Benchmark Geometry



The MCNP6 model is based on the MCNP model supplied as part of the VENUS-3 benchmark collection in Reference 7.6, which used an earlier version of MCNP. This model is reviewed for correctness and updated as needed for use with the current MCNP version MCNP6.

The ENDF/B-VII.1 libraries associated with 293.6 degrees K (.80c extension) are used for all materials. In addition, a light water S(α,β) library based on the ENDF/B VII.1, lwtr.20t, is used for those materials containing water. The benchmark used a ^{235}U Watt fission spectrum.

Portions of the NuScale MCNP6 model of the VENUS-3 benchmark are shown in Figure A-2 and Figure A-3.

Figure A-2 Vertical Cross-Sectional View of the Monte Carlo N-Particle Transport Code 6 Model of the Vulcain Experimental Nuclear Study 3 Benchmark

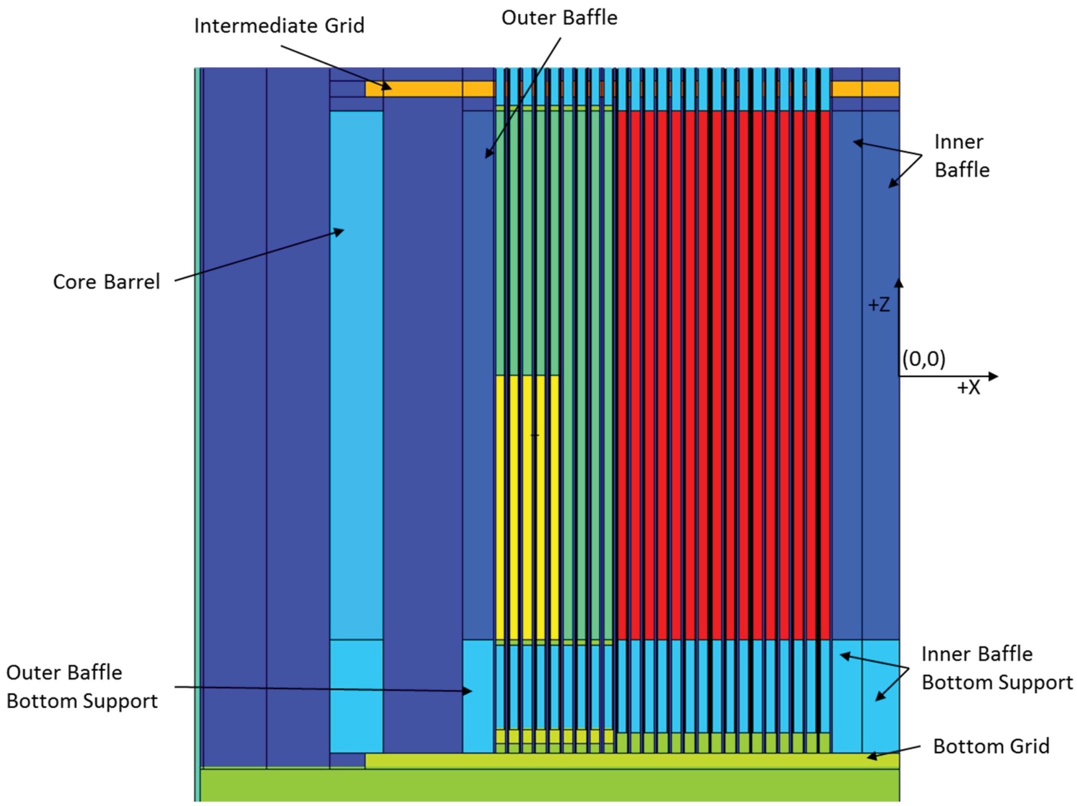
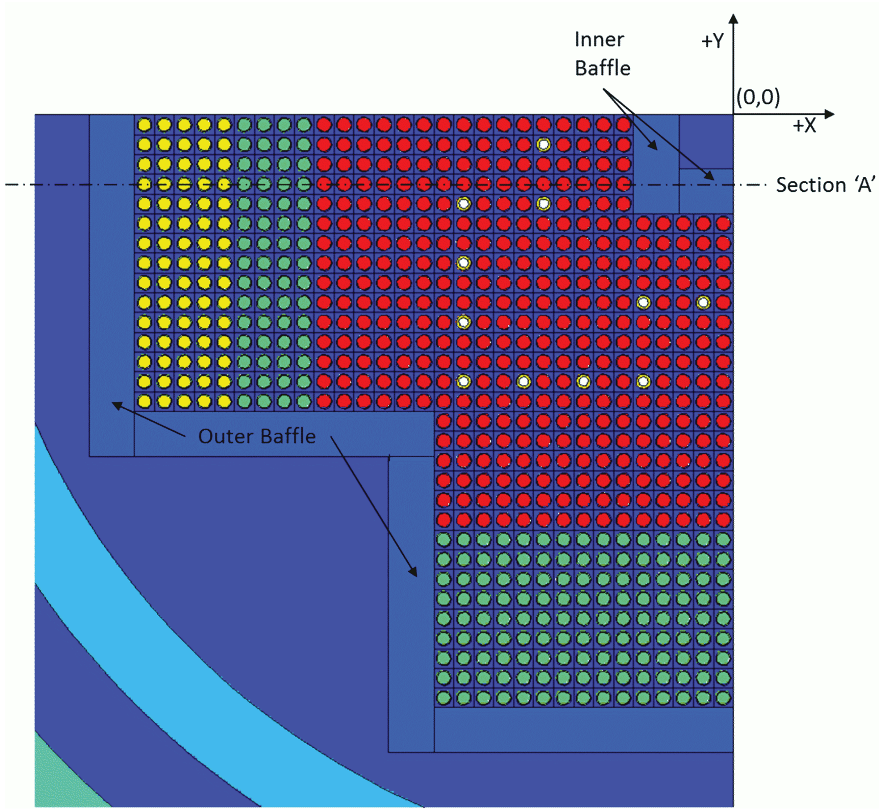


Figure A-3 Horizontal Cross-Sectional View of the Inner and Outer Baffle of the Monte Carlo N-Particle Transport Code 6 Model of the Vulcain Experimental Nuclear Study 3 Benchmark



A variety of experimental results are provided as part of the VENUS-3 collection of data, but the results of specific interest to this benchmark are the results associated with the core barrel only. These results are based on nickel, indium, and aluminum reaction rates ^{58}Ni(n,p), ^{115}In(n,n'), and ^{27}Al(n,), respectively.

Based on the energy thresholds associated with the reaction rates, the ^{115}In(n,n') reaction rates are associated with the neutron flux greater than 1 MeV, the ^{58}Ni(n,p) reaction rates are associated with neutron fluxes greater than 3 MeV, and the ^{27}Al(n,α) reaction rates are associated with neutron fluxes greater than 8 MeV. The relative experimental uncertainties for the reaction rates in the core barrel for the VENUS-3 data are reported to be 9 percent for ^{58}Ni(n,p), 7 percent for ^{115}In(n,n'), and 14 percent for ^{27}Al(n,α) in Section 6.1 of Reference 7.7.

The relative difference between the reported experimental (Exp) values for these reaction rates and the MCNP6 calculated values (Calc) is established for each data point provided in the VENUS-3 benchmark, relative to the experimental value, using

Equation A-1.The average relative difference of experimental versus calculated values and standard deviations are reported in Table A-1.

$$\text{Relative difference (\%)} = \frac{Exp - Calc}{Exp} \times 100\% \qquad \text{Equation A-1}$$

The ^{115}In(n,n') reaction rate comparisons are judged to provide the best comparison to the overall neutron flux because it has the lowest threshold energy of ~1 MeV. The ^{58}Ni(n,p) and ^{27}Al(n,) reaction rates have higher thresholds, 3 MeV and 8 MeV, respectively. The ^{115}In(n,n') results also have the lowest experimental uncertainty associated with them. Further, the ^{115}In(n,n') results are the only results from the NuScale VENUS-3 benchmark that indicate MCNP6 has a tendency to {{

}}2(a),(c) compared to incorporating the ^{58}Ni(n,p) or ^{27}Al(n,α) based benchmark results.

{{

}}2(a),(c).

{{

Equation A-2

}}2(a),(c)

Table A-1 Vulcain Experimental Nuclear Study 3 Experimental Versus Calculated Results

{{

}}2(a),(c)

{{

}}2(a),(c).

The results of this benchmark demonstrate that MCNP6 can perform neutron flux determinations that compare favorably with expected or experimental results. The results show good agreement between MCNP6 and the benchmark results.

Appendix B NuScale Power Module Fluence Prediction Sensitivity Studies and Uncertainty Analysis

This appendix presents sensitivity studies and an uncertainty analysis associated with the NPM fluence prediction calculations. Appendix B results are combined with Appendix A findings in Section 4.0 of this report in order to properly present results with total uncertainty in Section 5.0 of this report.

B.1 Sensitivity Studies

B.1.1 Homogenized Fuel Model vs Explicit Fuel Model

The best-estimate fluence predictions presented in Table 5-1 are based on a homogenized fuel model. {{

}}$^{2(a),(c)}$.

B.1.2 Contribution of ^{239}Pu to Neutron Source

As discussed in Section 3.3, the MCNP6 NuScale best-estimate fluence model does not contain plutonium because it is based on a fresh core. {{

}}$^{2(a),(c)}$

{{

Equation B-1

}}2(a),(c)

{{

}}2(a),(c)

{{

Equation B-2

}}2(a),(c)

{{

}}2(a),(c)

{{ }}2(a),(c)

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Equation B-3

}}2(a),(c)

{{

}}2(a),(c)

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Equation B-4

}}2(a),(c)

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}}2(a),(c)

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}}2(a),(c)

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Equation B-5

}}2(a),(c)

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}}2(a),(c)

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Equation B-6

}}2(a),(c)

{{ }}2(a),(c)

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Equation B-7

}}2(a),(c)

{{

}}2(a),(c)

{{

Equation B-8

}}2(a),(c)

{{

Equation B-9

}}2(a),(c)

{{

}}2(a),(c)

B.1.3 Material Composition

The uncertainty in fluence estimates associated with differences between the as built and operating NPM material chemical compositions and densities compared to how these characteristics are modeled in the NuScale best-estimate fluence model is assumed to be {{

}}2(a),(c).

B.1.4 Geometrical Tolerances

The uncertainty in fluence estimates associated with differences between as built and operating NPM dimensions and dimensions modeled in the NuScale best-estimate fluence model is assumed to be {{

}}2(a),(c)

{{

}}2(a),(c).

B.1.5 Assembly Averaged Neutron Source Bias and Uncertainty

The MCNP6 NuScale best-estimate fluence model uses an assembly averaged pin power profile instead of an explicit pin-wise power profile.

{{

}}2(a),(c)

{{

}}2(a),(c)

{{

}}2(a),(c)

Table B-1 Averaged Fast Neutron Flux in Pin Lattice of Fuel Assembly G4 from SIMULATE5, Cycle 8

{{

}}2(a),(c),ECI

Table B-2 Averaged Fast Neutron Flux in Pin Lattice of Fuel Assembly G5 from SIMULATE5, Cycle 8

{{

}}2(a),(c),ECI

Table B-3 Averaged Fast Neutron Flux in Pin Lattice of Fuel Assembly F6 from SIMULATE5, Cycle 8

{{

}}2(a),(c),ECI

Table B-4 Averaged Fast Neutron Flux in Pin Lattice of Fuel Assembly E7 from SIMULATE5, Cycle 8

{{

}}2(a),(c),ECI

Table B-5 Averaged Fast Neutron Flux in Pin Lattice of Fuel Assembly G4 from MCNP6, Cycle 8

{{

}}2(a),(c),ECI

Table B-6 Averaged Fast Neutron Flux in Pin Lattice of Fuel Assembly G5 from MCNP6, Cycle 8

{{

}}2(a),(c),ECI

Table B-7 Averaged Fast Neutron Flux in Pin Lattice of Fuel Assembly F6 from MCNP6, Cycle 8

{{

}}2(a),(c),ECI

Table B-8 Averaged Fast Neutron Flux in Pin Lattice of Fuel Assembly E7 from MCNP6, Cycle 8

{{

}}2(a),(c),ECI

B.1.6 Core Power

The uncertainty of the core power level is directly proportional to the uncertainty of the fluence estimates. {{

}}2(a),(c).

B.1.7 Radial Power Profile

Uncertainty in the radial power profile is directly proportional to the uncertainty of the fluence estimates. The radial power profile uncertainty (σ_{pr}) is estimated by

{{

}}2(a),(c)

{{

}}2(a),(c)

Figure B-1 Time-Weighted Averages and Weighted Standard Deviations for Radial Power Profile

{{

}}2(a),(c),ECI

B.1.8 Axial Power Profile

A single, time-averaged axial profile is utilized in the MCNP6 NuScale best-estimate fluence model. Variations in the axial power profile could impact fluence estimates. {{

}}2(a),(c)

{{

Equation B-10

}}2(a),(c)

{{

}}2(a),(c)

{{

Equation B-11

}}2(a),(c)

{{

}}2(a),(c)

Table B-9 Average Axial Power Profiles

{{

}}2(a),(c),ECI

Table B-10 Variance and Weighted Standard Deviation for the Axial Power Profiles

{{

}}2(a),(c),ECI

B.1.9 Boron Concentration

The best estimate fluence prediction MCNP6 model assumed a boron concentration of {{

}}2(a),(c).

The concentration of soluble boron in the primary coolant varies over the course of the fuel cycle, in a range {{

}}2(a),(c)

B.1.10 Nuclear Cross-Section Data and Transport Code

There is uncertainty associated with the various cross sections taken from the ENDF/B-VII.1 nuclear data library and there is uncertainty associated with the use of the transport code MCNP6. {{

}}2(a),(c)

B.1.11 Monte Carlo Method

In Monte Carlo analysis, a calculational uncertainty (σ_{mt}) is introduced as a result of the finite number of particle histories sampled. The relative error (standard deviation/mean) associated with the MCNP6 results is taken to account for this uncertainty. {{

}}2(a),(c)

B.1.12 Water Density

{{

}}2(a),(c)

B.1.13 Axial Coolant Density Bias

The coolant in the MCNP6 NuScale best-estimate fluence model is modeled as {{

}}2(a),(c)

Table B-11 Coolant Water Axial Density Variations

{{

}}2(a),(c)

Table B-12 Peak Fluence Results for Axially Varied Coolant Density

{{

}}2(a),(c)

B.1.14 Tally Mesh Size

This section presents the results of the determination of the uncertainty, σ_{tally}.

{{

}}2(a),(c)

Table B-13 Peak Fluence Results for Axially-Varied Coolant Density

{{

}}2(a),(c)

Appendix C Alternative Approaches to Regulatory Guide 1.190 Regulatory Positions

RG 1.190 (Reference 7.1) provides guidance for calculating pressure vessel neutron fluence. The NuScale fluence calculation methodology described in this report used some alternative approaches to those recommended in RG 1.190. This appendix describes and justifies these alternatives in Table C-1.

The descriptions in Table C-1 are summaries or excerpts of specific portions of regulatory positions in RG 1.190.

Table C-1 Alternative Approaches to Regulatory Guide 1.190 Regulatory Positions

RG 1.190 Regulatory Position	Description of Regulatory Position	Description of Alternative and Justification
1.1.1	Regional temperatures should be included in the input data.	All materials in the NuScale best-estimate fluence model are taken to be at {{ }}2(a),(c). The effect of the latter is accounted for in Section B.1.13.
1.1.1 and 1.4.1	In the absence of plant-specific information, conservative estimates of the variations in the material compositions and dimensions should be made and accounted for in the determination of the fluence uncertainty.	Uncertainty between the “as built and operating” and “as modeled” design is accounted for {{ }}2(a),(c) estimates as discussed in Section B.1.3 and Section B.1.4.
1.1.1	The input data should account for axial and radial variations in water density.	{{ }}2(a),(c) The effect of this modeling simplification is accounted for in Section B.1.13.
1.2	The peripheral assemblies, which contribute the most to the vessel fluence, have strong radial power gradients, and these gradients should not be neglected. Peripheral assembly pin-wise neutron source distributions obtained from core depletion calculations should be used.	Assembly-averaged power profiles obtained from core depletion calculations are used in the MCNP6 NuScale best-estimate fluence model. A sensitivity study to establish the effect of this modeling simplification on the NuScale fluence estimates is discussed in Section B.1.5.

Table C-1 Alternative Approaches to Regulatory Guide 1.190 Regulatory Positions

RG 1.190 Regulatory Position	Description of Regulatory Position	Description of Alternative and Justification
1.3.2	The bias introduced by the neutron energy cutoff technique should be estimated by comparison with an unbiased calculation.	The MCNP6 NuScale best-estimate fluence model implements a cutoff energy threshold of 0.9 MeV. An additional study involving an MCNP6 model without a cutoff energy threshold is unnecessary. Because there are no processes modeled that would result in a higher energy neutron, the use of a 0.9 MeV cutoff energy threshold makes no difference to the >1 MeV fluence results.
1.3.2	Statement of 10 statistic tests provided by Monte Carlo code	{{ }}2(a),(c) discussed in Section 3.7.
1.3.3	The capsule fluence is extremely sensitive to the representation of the capsule geometry and internal water region (if present), and the adequacy of the capsule representation and mesh must be demonstrated using sensitivity calculations.	{{ }}2(a),(c)
1.4.2	The fluence calculation methods must be validated against (1) operating reactor measurements or both, (2) a pressure vessel simulator benchmark, and (3) the fluence calculation benchmark.	The pressure vessel simulator benchmark VENUS-3 is used to validate the NuScale fluence calculation methodology (Appendix A). The VENUS-3 benchmark results are adequate to validate the NuScale fluence calculation methodology.

Licensing Technical Report

Pressure and Temperature Limits Methodology

December 2022

Revision 0

Docket: 52-050

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Abstract

This report describes the methodology used to develop the pressure-temperature (P-T) limits and the low temperature overpressure protection (LTOP) setpoint for the NuScale Power, LLC, NuScale Power Module (NPM). Plant operation within these limits protects the reactor coolant pressure boundary (RCPB) from non-ductile fracture.

This report bases its requirements and methodology for developing P-T limits on the requirements and the methodologies in Title 10 of the Code of Federal Regulations (CFR) Part 50 (10 CFR 50), Appendix G, and the American Society of Mechanical Engineering (ASME) Boiler and Pressure Vessel Code (BPVC) Section XI, Appendix G; the P-T limits in the reactor pressure vessel (RPV) account for vessel embrittlement due to neutron fluence in accordance with Regulatory Guide (RG) 1.99. Representative P-T limits for the NPM are in tables and figures displaying maximum allowable reactor coolant system (RCS) pressure as a function of RCS temperature.

The NPM reactor vessel uses an LTOP system to provide protection against non-ductile failure due to LTOP events during reactor start-up and shutdown operation. The basis of the LTOP methodology in this report is ASME BPVC Section XI, Appendix G. The LTOP setpoints account for the effects of neutron embrittlement.

The basis for representative limits in this report is the projected 57 effective full-power years (EFPY) neutron fluence over the 60-year design life of the module. The P-T limits and LTOP setpoints applicable to operating modules are module-specific based on material properties of as-built reactor vessels. Plant licensees provide these limits, based on the methods provided in this report.

10 CFR 50.61 requires pressurized thermal shock (PTS) screening for the RPV beltline region of pressurized water reactors (PWRs). A PTS event is an event or transient in PWRs causing severe overcooling (thermal shock) concurrent with or followed by significant pressure within the RPV.

Executive Summary

There are a number of Nuclear Regulatory Commission (NRC) regulations related to reactor coolant pressure boundary (RCPB) integrity, including General Design Criterion (GDC) 31; GDC 32; Title 10 of the Code of Federal Regulations (CFR) Part 50.60 (10 CFR 50.60); 10 CFR 50.61; 10 CFR 50, Appendix G; and 10 CFR 50, Appendix H. Collectively, these regulations require a licensee to

- ensure that the RCPB has sufficient margin to prevent non-ductile failure during all phases of operation, including postulated accident conditions, accounting for material changes due to neutron fluence and temperature history over the life of the RCPB.
- develop reactor vessel pressure-temperature (P-T) limits for the reactor pressure vessel (RPV), which are limitations on reactor operating pressure as a function of reactor coolant temperature for various operating conditions.
- develop and maintain an appropriate surveillance program to monitor reduction in material toughness in ferritic materials over the life of the reactor vessel.

This report presents the methodologies used to demonstrate that the regulatory requirements identified above are met or are not applicable to the NuScale Power Module (NPM) reactor vessel. Historically, P-T limits were in the plant's technical specifications. The NRC guidance in Generic Letter (GL) 96-03 provides a means of relocating the P-T limits to a pressure-temperature limits report (PTLR), which facilitates modifications to P-T limits as needed over the life of the plant. Moving the P-T limits from the technical specifications to the PTLR requires the licensee to develop methods and programs to address each of the following aspects:

- neutron fluence calculation method
- adjusted reference temperature (ART) calculation method to account for the effects of neutron embrittlement
- minimum temperature requirements for the reactor vessel during various operational and testing modes
- reactor vessel surveillance program (RVSP) for ferritic steel
- the low temperature overpressure protection (LTOP) setpoint calculation method

This report addresses each of these topics as applicable to the NPM design. A licensee may use the methods found in this report to develop a PTLR rather than maintaining P-T limits in the plant's technical specifications. This report also includes the pressurized thermal shock (PTS) screening results.

1.0 Introduction

1.1 Purpose

This report describes the methodology used to develop the NuScale Power Module (NPM) heatup and cooldown curves (pressure-temperature (P-T) curves) and low temperature overpressure protection (LTOP) setpoints. Operation within these limits protects the reactor vessel from brittle fracture. This report also provides an embrittlement analysis in accordance with Regulatory Guide (RG) 1.99 (Reference 6.1.1) and outlines whether the design requires a reactor vessel surveillance program (RVSP). This report includes the pressurized thermal shock (PTS) screening results.

1.2 Scope

This report provides a methodology for development of P-T limits for the NPM reactor coolant pressure boundary (RCPB) including

- heatup and cooldown curves and P-T limits for normal operation.
- the P-T limits for in-service leak and hydrostatic tests.
- the LTOP setpoints.

In addition, this report provides values for each of these items based on assumed material properties at an exposure of 57 effective full-power years (EFPY) fluence, which represents the end-of-design-life neutron exposure based on a 60-year design life of the module with an assumed 95 percent capacity factor. This report does not provide P-T limits for use in an as-built NPM; the P-T limits must be created on a module-specific basis with consideration of the material properties of the as-built reactor pressure vessel (RPV). Licensees may reference the methods contained in this report to develop their module-specific pressure-temperature limits report (PTLR), or they may choose to develop an alternative methodology.

This report includes the PTS screening results.

In accordance with Generic Letter (GL) 96-03 (Reference 6.1.2), this report addresses the following five methodology aspects:

- neutron fluence calculation method
- the adjusted reference temperature (ART) calculation method to account for the effects of neutron embrittlement, in accordance with Reference 6.1.1
- minimum temperature requirements for the reactor vessel during various operational and testing modes based on Appendix G of Reference 6.1.3
- the RVSP for ferritic steel
- the LTOP setpoint calculation method

Table 1-1 Abbreviations

Term	Definition
ART	adjusted reference temperature
ASME	American Society of Mechanical Engineers
BPVC	Boiler and Pressure Vessel Code
CNV	containment vessel
EFPY	effective full-power years
GDC	General Design Criterion
ISLH	inservice leak and hydrostatic testing
LTOP	low temperature overpressure protection
NPM	NuScale Power Module
NRC	Nuclear Regulatory Commission
P-T	pressure and temperature
PTLR	pressure and temperature limits report
PTS	pressurized thermal shock
RCPB	reactor coolant pressure boundary
RG	Regulatory Guide
RPV	reactor pressure vessel
RT_{NDT}	nil-ductility reference temperature
RVSP	reactor vessel surveillance program
RVV	reactor vent valve
SIF	stress intensity factor

2.0 Background

This report outlines the P-T limits methodology and the LTOP setpoints methodology that can be used by a licensee to create a module-specific PTLR for an NPM. In addition, this report outlines the neutron fluence calculation method, ART calculation method, minimum P-T curves, RVSP recommendations, LTOP setpoint calculation method, and PTS screening results.

2.1 Regulatory Requirements and Recommendations

2.1.1 General Design Criterion 31 - Fracture Prevention of Reactor Coolant Pressure Boundary

General Design Criterion (GDC) 31 requires that the RCPB have sufficient margin to ensure that when stressed under operating, maintenance, testing, and postulated accident conditions, the boundary behaves in a non-brittle manner, and there is minimal probability of rapidly propagating fracture.

Changes in material properties must account for service temperatures and other conditions of the pressure boundary material under operating, maintenance, testing, and postulated accident conditions, as well as the uncertainties in determining

- material properties.
- the effects of irradiation on material properties.
- residual, steady state, and transient stresses.
- size of flaws.

2.1.2 General Design Criterion 32 - Inspection of Reactor Coolant Pressure Boundary

General Design Criterion 32 requires that the RCPB be designed to permit periodic inspection and testing of important areas and an appropriate material surveillance program for the RPV.

2.1.3 10 CFR 50.60 - Acceptance Criteria for Fracture Prevention Measures for Light Water Nuclear Power Reactors for Normal Operation

Regulation 10 CFR 50.60 (Reference 6.1.3) requires that light water reactors meet the fracture toughness and material surveillance program requirements set forth in Appendix G and Appendix H of Reference 6.1.3. Proposed alternatives to the requirements described in Appendix G and Appendix H of Reference 6.1.3 or portions thereof are allowed when the NRC grants an exemption under 10 CFR 50.12. The NPM design supports an exemption to 10 CFR 50.60 due to the absence of ferritic material in the RPV beltline region. The NPM design uses austenitic stainless steel in the lower RPV, which has superior ductility and is less susceptible to the effects of neutron and thermal embrittlement than ferritic materials, which increases the integrity and safety of the RCPB.

2.1.4 10 CFR 50, Appendix G - Fracture Toughness Requirements

Appendix G of Reference 6.1.3 specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the RCPB of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation. Conditions of normal operation include anticipated operational occurrences and system hydrostatic tests to which the pressure boundary may be subjected over its service lifetime. The NPM design supports an exemption to 10 CFR 50.60 due to the absence of ferritic material in the RPV beltline region. The NPM design uses austenitic stainless steel in the lower RPV, which has superior ductility and is less susceptible to the effects of neutron and thermal embrittlement than ferritic materials, which increases the integrity and safety of the RCPB.

2.1.5 10 CFR 50, Appendix H - Reactor Vessel Material Surveillance Program Requirements

Appendix H of Reference 6.1.3 establishes the necessary material surveillance program to satisfy GDC 32 for light water reactors. Appendix H of Reference 6.1.3 requires that licensees establish and maintain a material surveillance program to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors. The materials in the reactor vessel beltline region undergo exposure to neutron irradiation and to the thermal environment. The NPM design supports an exemption to 10 CFR 50.60 due to the absence of ferritic material in the RPV beltline region. The NPM design uses austenitic stainless steel in the lower RPV, which has superior ductility and is less susceptible to the effects of neutron and thermal embrittlement than ferritic materials, which increases the integrity and safety of the RCPB. Upper RPV ferritic materials, which are outside the beltline, do not exceed the Appendix H of Reference 6.1.3 threshold for requiring an RVSP.

2.1.6 Generic Letter 96-03 - Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits

Reference 6.1.2 provides information that describes the methodology that licensees may use to create PTLRs.

2.1.7 Regulatory Guide 1.99 - Radiation Embrittlement of Reactor Vessel Materials

Reference 6.1.1 provides general procedures that calculate the effects of neutron embrittlement of low-alloy steels used in light water reactor vessels.

2.1.8 10 CFR 50.61 - Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events

Regulation 10 CFR 50.61 requires PTS screening for the RPV beltline region of pressurized water reactors (PWRs). A PTS event is an event or transient in PWRs causing severe overcooling (thermal shock) concurrent with or followed by significant pressure within the RPV. The NPM design supports an exemption to 10 CFR 50.61

due to the absence of ferritic material in the RPV beltline region. The NPM design uses austenitic stainless steel in the lower RPV, which has superior ductility and is less susceptible to the effects of neutron and thermal embrittlement than ferritic materials, which increases the integrity and safety of the RCPB.

3.0 Analysis

3.1 Materials

In accordance with Appendix G of Reference 6.1.3, the calculations in this report apply to the pressure-retaining components of the RCPB. Because Appendix G of Reference 6.1.3 only contains data and methods applicable to ferritic materials, and because the NPM lower RPV is not made of ferritic materials, this report also evaluates ferritic materials in the upper RPV (i.e., the region above the upper flange).

Table 3-1 lists the materials in the RPV. Table 3-2 lists the materials in the containment vessel (CNV). Figure 3-1, Figure 3-2, and Figure 3-3 show the material distribution model for the RPV and the CNV.

Table 3-1 Reactor Pressure Vessel Material Distribution

Component	Material
Lower Seismic Cap	SA-693, Type 630, Condition H1100
Lower Head	SA-965, Grade FXM-19
Lower Flange Core Region Shell	SA-965, Grade FXM-19
Upper Flange Shell	SA-508, Grade 3 Class 2
Upper Feed Plenum Shell	SA-508, Grade 3 Class 2
Upper Steam Generator Shell	SA-508, Grade 3 Class 2
Upper Support Ledge Shell	SA-508, Grade 3 Class 2
Upper Support Ledge Shell Cladding	Alloy 690
RPV - CNV Support Ledge	SB-168, Alloy 690
Upper Steam Plenum Shell	SA-508, Grade 3 Class 2
Upper Pressurizer Shell	SA-508, Grade 3 Class 2
Upper Head	SA-508, Grade 3 Class 2
Interior and exterior cladding, except for the RPV upper support ledge exterior cladding	308L/309L

Table 3-2 Containment Vessel Material Distribution

Component	Material
Lower Seismic Support Pads	SA-479, Type 304
Lower Support Skirt	SA-182, Grade F304
Lower Head	SA-965, Grade FXM-19
Lower Core Region Shell	SA-965, Grade FXM-19
Lower Transition Shell	SA-965, Grade FXM-19
Buttering and Weld between the Lower Shell and Lower Transition Shell	Alloy 52/152
Lower Shell	SA-336, Grade F6NM
Lower Flange	SA-336, Grade F6NM
Upper Flange	SA-336, Grade F6NM
Upper Support Ledge Shell	SA-336, Grade F6NM
Upper Steam Generator Access Shell	SA-336, Grade F6NM
Upper Intermediate Shell	SA-336, Grade F6NM
Upper Manway Access Shell	SA-336, Grade F6NM
Upper Seismic Support Shell	SA-336, Grade F6NM
Upper Head	SA-336, Grade F6NM
Control Rod Drive Mechanism Top Head Cover	SA-182, Grade F6NM

Figure 3-1 Two Dimensional Model Material Distribution

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}}2(a),(c),ECI

Figure 3-2 Two Dimensional Model Containment Vessel Material Distribution

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}}2(a),(c),ECI

Figure 3-3 Two Dimensional Model Reactor Pressure Vessel Material Distribution

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}}2(a),(c),ECI

3.1.1 Neutron Fluence and Ferritic Materials

Per Appendix G of Reference 6.1.3, the calculations in this report apply to the pressure-retaining components of the RCPB. The lower RPV (i.e., the region below the upper flange) undergoes exposure to higher neutron fluence than other portions of the RCPB; however, the NPM lower RPV is made of austenitic stainless steel rather than ferritic materials (Table 3-1). Despite the higher neutron fluence in the lower RPV region, the use of austenitic stainless steel ensures safety of the RCPB because austenitic stainless steel has superior ductility and is less susceptible to the effects of neutron and thermal embrittlement than ferritic materials.

Appendix G of Reference 6.1.3, provides the following definition of the RPV beltline.

> *Beltline or Beltline region of reactor vessel means the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage.*

The NPM design does not require testing Charpy upper shelf energy per Appendix G of Reference 6.1.3, for the following reason.

- Appendix G of Reference 6.1.3 applies to ferritic materials, while the portion of the NPM in the beltline region is austenitic stainless steel. The ASME BPVC Section III, NB-2311, does not require impact testing for austenitic stainless steel because these materials do not undergo ductile-to-brittle transition temperature and have higher toughness than ferritic materials used for ASME BPVC Section III Class 1 pressure-retaining components. Because impact testing is not required for austenitic stainless steel, the nil-ductility reference temperature (RT_{NDT}) cannot be calculated. The NRC endorsed ASME BPVC Section III in 10 CFR 50.55a.

Appendix H of Reference 6.1.3 specifically applies to ferritic steel because the requirements for an RVSP were developed for ferritic materials and there is no guidance for an RVSP for austenitic stainless steel. Furthermore, austenitic stainless steel has superior ductility and is less susceptible to the effects of neutron and thermal embrittlement than ferritic materials, which increases the integrity and safety of the RCPB. Because the lower RPV is austenitic stainless steel, the ferritic portion of the RPV that experiences the highest fluence is evaluated against the Appendix H of Reference 6.1.3 criteria requiring an RVSP. The upper RPV lower flange has a design life peak fluence less than 1E+17 n/cm^2, E > 1 MeV, Section III.A of Appendix H of Reference 6.1.3 does not require an RVSP.

3.2 Adjusted Reference Temperature

There is no ART for the NPM because there is no need to adjust the RT_{NDT} for fluence because the peak neutron fluence at the top of the lower flange of the RPV is

{{ }}2(a),(c),ECI, which is less than the 1E+17 n/cm^2, E > 1 MeV regulatory limit.

Appendix H of Reference 6.1.3 requires beltline material surveillance if the portions of the RPV experience a maximum fluence greater than 1.0E+17 n/cm^2, E > 1 MeV; however, the portion of the RPV experiencing the highest neutron fluence is the lower RPV, which is made of austenitic stainless steel. The ASME BPVC Section III, NB-2311, does not require impact testing for austenitic stainless steels because they do not undergo ductile-to-brittle transition temperature and have higher toughness than ferritic materials used for ASME BPVC Section III Class 1 pressure-retaining components. Without impact testing, RT_{NDT} cannot be calculated for austenitic stainless steel, and thus ART is not applicable. The NRC endorsed ASME BPVC Section III in 10 CFR 50.55a. Since the upper RPV is the only part of the RPV made of ferritic materials, an evaluation of the upper RPV experiencing the highest design life peak fluence indicates that the upper RPV neutron fluence would have to increase by a factor of {{ }}2(a),(c),ECI to experience a fluence greater than 1.0E+17n/cm^2, E > 1 MeV; therefore, there is no need to adjust the reference temperatures.

3.3 Scope of Pressure-Temperature Limits Analysis

In order to develop a P-T limits methodology for the NPM, this report calculates minimum P-T limits for the NPM upper RPV design based on the requirements of Appendix G of Reference 6.1.3 and based on the methodologies in ASME BPVC Section XI, Appendix G (Reference 6.1.5). Finite element models simulate thermal transient stress and analyze fracture mechanics.

3.3.1 Thermal Transients

Thermal transients, in the context of this evaluation, include two heat transfer mechanisms: convection and radiation.

Convection is considered on the following surfaces:

- internal surfaces of the RPV (free and forced convection)
- the annulus between the RPV and CNV when flooded during the heatup and cooldown transients (free convection)
- the outside of the CNV for locations submerged in the pool (free convection)

Radiation is considered in the following regions:

- between the RPV outer surface and the CNV inner surface
- between the lower and upper RPV in the gap in the RPV flange
- between the lower and upper CNV in the gap in the CNV flange
- between the upper CNV and the control rod drive mechanism access cover at the closure surface

Convection driven by condensation in the upper pressurizer is also a driving heat transfer mechanism that occurs during these transients when the pressurizer wall temperature dips below saturation temperature. This occurrence can increase the convective film coefficients.

The four thermal transients considered in this evaluation include

- heatup.
- power ascent.
- power descent.
- cooldown.

This report creates P-T limit curves for the following transient conditions:

- heatup, including power ascent. The heatup transient begins with the annulus between the RPV and CNV flooded with water.
- cooldown, starting with power descent. The cooldown transient includes the annulus between the RPV and the CNV flooded with water.
- inservice leak and hydrostatic testing (ISLH). The ISLH considers both steady state and heatup/cooldown transient conditions.

3.3.1.1 Heatup Transient

Figure 3-4 shows the heatup transient.

Figure 3-4 Transient Temperature for Heatup

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}}2(a),(c),ECI

3.3.1.2 Power Ascent Transient

Figure 3-5 shows the power ascent transient.

Figure 3-5 Power Ascent Transient Definition - Temperatures and Convection Coefficients

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}}2(a),(c),ECI

3.3.1.3 Power Descent Transient

Figure 3-6 shows the power descent transient.

Figure 3-6 Power Descent Transient Definition - Temperatures and Convection Coefficients

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}}2(a),(c),ECI

3.3.1.4 Cooldown Transient

Figure 3-7 shows the cooldown transient.

Figure 3-7 Cooldown Transient Definition - Temperatures and Convection Coefficients

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}}2(a),(c),ECI

3.3.2 Fracture Mechanics

This report analyzes axial and circumferential flaw locations at the most limiting thermal and pressure stress locations. Fracture mechanics analyses consider postulated flaws as follows:

- axial flaws: one-fourth thickness from the inner surface and one-fourth thickness from the outer surface.
- circumferential flaws: one-fourth thickness from the inner surface and one-fourth thickness from the outer surface.

3.3.3 Pressure and Temperature Limit Methodology

3.3.3.1 Pressure Boundary Components

In accordance with Appendix G of Reference 6.1.3, the calculations in this report bound the pressure-retaining components of the RCPB. The lower RPV is austenitic stainless steel (Table 3-1). Section 3.1 discusses the material distribution for the RPV and CNV. This evaluation considers the upper RPV because it contains ferritic materials.

3.3.3.2 Maximum Postulated Cracks

The methods of Appendix G, Article G-2214.1, of Reference 6.1.5 postulate the existence of a sharp surface crack in the RPV that is normal to the direction of the maximum stress. As specified in paragraph G-2120 of Reference 6.1.5, the crack depth is one-fourth of the RPV wall thickness, and the crack length is 1.5 times the wall thickness. This report considers both inside and outside surface cracks in axial and circumferential directions individually. For crack evaluations, a single crack is present in the RPV.

Figure 3-8 and Figure 3-9 show representations of circumferential and axial cracks, respectively.

Figure 3-8 Representation of Postulated Semi-Elliptical Circumferential Cracks in Reactor Pressure Vessel Wall

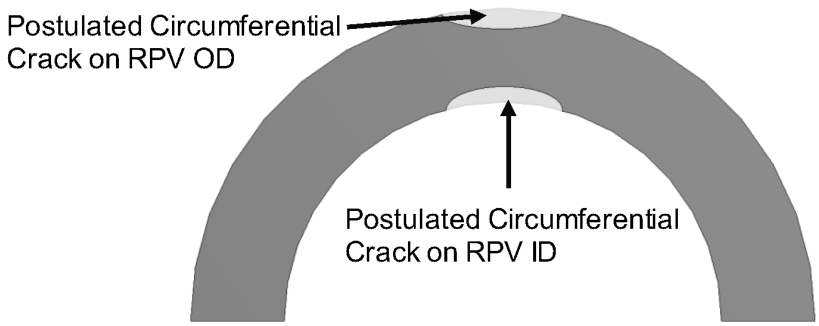
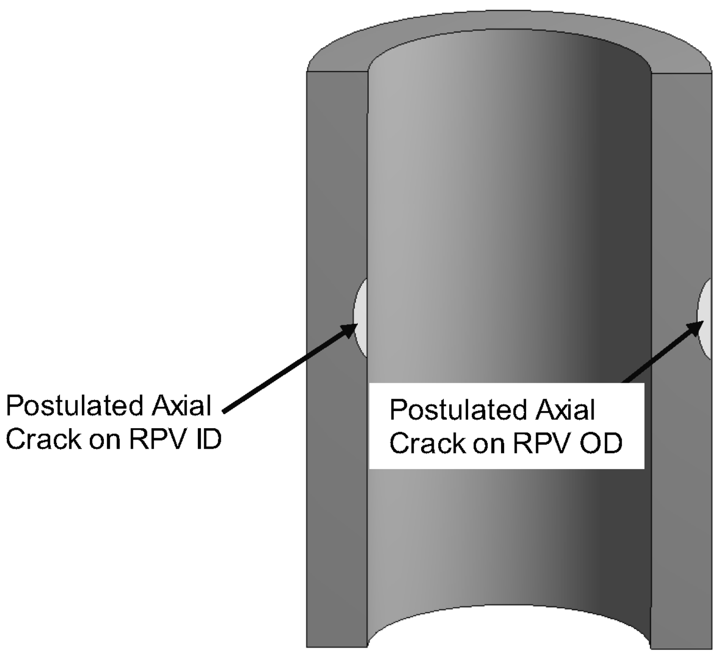


Figure 3-9 Representation of Postulated Semi-Elliptical Axial Cracks in Reactor Pressure Vessel Wall



3.3.3.3 Fracture Toughness

Appendix G, Article G-2110, of Reference 6.1.5 requires use of the critical stress intensity factor (SIF), K_{IC}, defined by Equation 3-1, in P-T limits calculations.

$$K_{IC} = 33.2 + 20.734\exp[0.02(T - RT_{NDT})] \qquad \text{Eq. 3-1}$$

Where:

K_{IC} = Critical SIF measuring fracture toughness (ksi in $^{0.5}$).

T = Temperature at crack tip (degrees F).

RT_{NDT} = Reference temperature for nil-ductility transition (degrees F).

The conservative limit on upper shelf fracture toughness, K_{IC} from Equation 3-1, has an upper bound value of $200\ \mathrm{ksi} \cdot \mathrm{in}^{0.5}$, which is slightly lower than the upper cutoff of lower bound K_{IC} in Appendix G, Article G-2212 of Reference 6.1.5. The crack-tip temperatures needed for these fracture toughness calculations are from transient thermal analysis.

3.3.3.4 Fracture Mechanics Analysis

3.3.3.4.1 Calculation of Stress Intensity Factors due to Internal Pressure

Appendix G, Article G-2214.1, of Reference 6.1.5 provides a method to calculate K_{Im} corresponding to membrane tension for postulated axial and circumferential cracks. This method applies to locations away from geometric discontinuity where calculation of hoop stress and axial stress occurs directly through an influence coefficient M_m (M_{m_axial} for axial cracks and M_{m_circ} for circumferential cracks).

For postulated axial cracks:

$$K_{Im\_axial} = M_{m\_axial}(pR_i/t) \qquad \text{Eq. 3-2}$$

Where:

p = internal pressure (ksi).

R_i = vessel inner radius (inches).

t = vessel wall thickness (inches).

On the inside surface:

$$M_{m\_axial} = \begin{cases} 1.85 & for\ t < 4\ in \\ 0.926\sqrt{t} & for\ 4\ in \le t \le 12\ in \\ 3.21 & for\ t > 12\ in \end{cases} \quad \text{Eq. 3-3}$$

On the outside surface:

$$M_{m\_axial} = \begin{cases} 1.77 & for\ t < 4\ in \\ 0.893\sqrt{t} & for\ 4\ in \le t \le 12\ in \\ 3.09 & for\ t > 12\ in \end{cases} \quad \text{Eq. 3-4}$$

And for postulated circumferential cracks on the inside or outside surface:

$$K_{Im\_circ} = M_{m\_circ}(pR_i/t) \quad \text{Eq. 3-5}$$

$$M_{m\_circ} = \begin{cases} 0.89 & for\ t < 4\ in \\ 0.443\sqrt{t} & for\ 4\ in \le t \le 12\ in \\ 1.53 & for\ t > 12\ in \end{cases} \quad \text{Eq. 3-6}$$

Equation 3-2 through Equation 3-6 are not valid for cracks postulated at locations with a geometric discontinuity. A finite element analysis crack model calculates the SIFs due to pressure for all locations. A unit pressure (1 psig) is applied to the RPV inner surface. The SIFs for the crack tip node at the deepest point are calculated for five contours. The maximum value from contours two through five for the deepest point is the maximum SIF (K_{Im}) for this evaluation. The first contour is not used because it is not accurate due to numerical inaccuracies in the stresses and strains at the crack tip.

3.3.3.4.2 Calculation of Stress Intensity Factors due to Thermal Stress

The hoop and axial thermal stresses are curve-fit to third order polynomial functions, which calculate thermal stress intensity factors K_{IT}. The format of the polynomial function is:

$$\sigma = c_0 + c_1\left(\frac{x}{a}\right) + c_2\left(\frac{x}{a}\right)^2 + c_3\left(\frac{x}{a}\right)^3 \quad \text{Eq. 3-7}$$

Where c_0, c_1, c_2, and c_3 are coefficients.

σ =hoop stress or axial stress used to calculate SIF for postulated axial or circumferential crack (psi).

a =crack depth (inches).

x =distance from the appropriate (i.e., inside or outside) surface with $x = a$ at the deepest crack tip (inches).

Appendix G, Article G-2214.3(b) of Reference 6.1.5 provides generic equations to calculate K_{IT} for radial thermal gradient for any thermal stress distribution. For postulated axial and circumferential cracks away from geometry discontinuity, the following equations calculate SIFs.

For an inside surface crack during a cooldown transient:

$$K_{IT} = (1.0359c_0 + 0.6322c_1 + 0.4753c_2 + 0.3855c_3)\sqrt{\pi a} \qquad \text{Eq. 3-8}$$

For an outside surface crack during a heat up transient:

$$K_{IT} = (1.043c_0 + 0.630c_1 + 0.481c_2 + 0.401c_3)\sqrt{\pi a} \qquad \text{Eq. 3-9}$$

Where *a* is the crack depth (inches), and c_0, c_1, c_2 and c_3 are coefficients of the third order polynomial equation for hoop or axial thermal stresses.

Equation 3-8 and Equation 3-9 are not accurate for cracks postulated at locations with a geometric discontinuity. A finite element analysis crack model calculates the SIFs due to transient thermal stresses by the superposition principle. To do so, a unit pressure (1psig) is applied to the crack top face and crack bottom face in four separate steps.

1. Constant unit pressure, set $c_0 = 1$, $c_1 = 0$, $c_2 = 0$ and $c_3 = 0$ in Equation 3-7. The calculated SIF is $K_{It\_c_0}$.
2. Linear pressure along the crack depth direction, set $c_0 = 0$, $c_1 = 1$, $c_2 = 0$ and $c_3 = 0$ in Equation 3-7. The calculated SIF is $K_{It\_c_1}$.
3. Quadratic pressure along the crack depth direction, set $c_0 = 0$, $c_1 = 0$, $c_2 = 1$ and $c_3 = 0$ in Equation 3-7. The calculated SIF is $K_{It\_c_2}$.
4. Cubic pressure along the crack depth direction, set $c_0 = 0$, $c_1 = 0$, $c_2 = 0$ and $c_3 = 1$ in Equation 3-7. The calculated SIF is $K_{It\_c_3}$.

The SIFs for the crack tip node at the deepest point are calculated for five contours. The maximum value from the integrals of contour paths two through five is the maximum SIF. The proposed crack-specific equation to calculate SIFs for any axial/circumferential inside/outside surface cracks is:

$$K_{IT} = c_0 K_{It\_c_0} + c_1 K_{It\_c_1} + c_2 K_{It\_c_2} + c_3 K_{It\_c_3} \qquad \text{Eq. 3-10}$$

Where c_0, c_1, c_2, and c_3 are the actual coefficients of the 3rd order polynomial equation. If K_{IT} is negative, the allowable pressure calculation uses a zero value.

3.3.3.5 American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section XI, Appendix G, Limits

This section documents the Appendix G of Reference 6.1.5 methodology for calculating the RPV allowable pressure for preservice hydrostatic test, normal heatup and cooldown transients, and ISLH conditions. This report documents development of a representative set of P-T calculations.

The ASME BPVC allowable pressure is part of the Appendix G of Reference 6.1.3 requirements. Except for the preservice hydrostatic test, the requirement of Appendix G of Reference 6.1.3 is that the test temperature must be greater than 50 degrees F.

The fundamental equation that is used to calculate P-T limits with a required safety margin is given by:

$$K_{I\,applied} = K_{IC} \qquad \text{Eq. 3-11}$$

Where K_{IC} is the lower bound crack initiation fracture toughness factor for the material as represented in Equation 3-1, and $K_{I\,applied}$ is the stress intensity factor due to pressure and thermal gradient loads at the tip of the one-fourth T postulated cracks.

$$K_{I\,applied} = SF \cdot M_m \cdot (pR_i / t) + K_{IT} \qquad \text{Eq. 3-12}$$

Where SF is the required structural factor applied to the pressure loading, and dependent on which P-T limits curve is being evaluated, M_m is the influence coefficient from Section 3.3.3.4.1, and K_{IT} is calculated using the Section 3.3.3.4.2 methodology.

The allowable pressure associated with a specified temperature along a P-T limits curve is:

$$P = \frac{(K_{IC} - K_{IT})t}{SF \cdot M_m \cdot R_i} = \frac{K_{IC} - K_{IT}}{SF \cdot K_{Im}} \quad \text{Eq. 3-13}$$

The appropriate K_{IT} and SF values used for various conditions are:

- For preservice hydrostatic tests, a steady-state condition ($K_{IT} = 0$) is applied, and the required structural factor $SF = 1$.

$$P = \frac{K_{IC}t}{M_m \cdot R_i} = \frac{K_{IC}}{K_{Im}} \quad \text{Eq. 3-14}$$

Performance of the allowable pressure calculation occurs for the crack with highest M_m that bounds other cracks. The basis for the preservice limiting pressure is NUREG-0800, Section 5.3.2 (Reference 6.1.4).

For the heat up and cooldown transients, the thermal SIF K_{IT} calculation occurs at selected time points, and the required structural factor $SF = 2$.

$$P = \frac{(K_{IC} - K_{IT})t}{2M_m \cdot R_i} = \frac{K_{IC} - K_{IT}}{2K_{Im}} \quad \text{Eq. 3-15}$$

For ISLH, the SIF K_{IT} from heat up and cooldown transients conservatively apply to the most limiting crack, and the required structural factor $SF = 1.5$.

$$P = \frac{(K_{IC} - K_{IT})t}{1.5M_m \cdot R_i} = \frac{K_{IC} - K_{IT}}{1.5K_{Im}} \quad \text{Eq. 3-16}$$

3.3.3.6 10 CFR 50, Appendix G, Pressure and Temperature Limits

Appendix G of Reference 6.1.3 requires that the P-T limits are at least as conservative as limits obtained by following the Appendix G of Reference 6.1.5, methods presented in Section 3.3.3.5. Additionally, Table 1 of Appendix G of Reference 6.1.3 requires further limitations (Table 3-3).

Table 3-3 Pressure and Temperature Requirements for the Reactor Pressure Vessel

Operating Condition	Vessel Pressure(1)	Requirements for Pressure-Temperature Limits	Minimum Temperature Requirements
Hydrostatic Pressure and Leak Tests (core is not critical)			
Fuel in the Vessel	≤ 20%	ASME BPVC § XI App. G Limits	(2)
Fuel in the Vessel	> 20%	ASME BPVC § XI App. G Limits	(2) + 90 degrees F(5)
No Fuel in the Vessel (preservice hydrostatic test)	all	Not Applicable	(3) + 60 degrees F
Normal Operation (including heatup and cooldown), Including Anticipated Operational Occurrences			
Core Not Critical	≤ 20%	ASME BPVC § XI App. G Limits	(2)
Core Not Critical	> 20%	ASME BPVC § XI App. G Limits	(2) + 120 degrees F(5)
Core Critical	≤ 20%	ASME BPVC § XI App. G Limits + 40 degrees F	maximum of (4) or ((2) + 40 degrees F)
Core Critical	> 20%	ASME BPVC § XI App. G Limits + 40 degrees F	maximum of (4) or ((2) + 160 degrees F)

Notes:
1. Percent of the preservice system hydrostatic test pressure.
2. The highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload.
3. The highest reference temperature of the vessel.
4. The minimum permissible temperature for the in-service system hydrostatic pressure test.
5. Lower temperatures are permissible if they can be justified by showing that the margins of safety of the controlling region are equivalent to those required for the beltline when it is controlling.

3.4 Reactor Vessel Surveillance Program Consideration

Appendix H of Reference 6.1.3 states:

> *The purpose of the material surveillance program required by this appendix is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors which result from exposure of these materials to neutron irradiation and the thermal environment.*

No material surveillance program is required for reactor vessels for which it can be conservatively demonstrated by analytical methods applied to experimental data and tests performed on comparable vessels, making appropriate allowances for uncertainties in the measurements, that the peak neutron fluence at the end of the design life of the vessel does not exceed 1E+17 n/cm^2, E > 1 MeV.

Because Appendix H of Reference 6.1.3 applies to ferritic materials with peak neutron fluence at the end of the design life above 1E+17 n/cm^2, E > 1 MeV, the NPM reactor pressure vessel has no RVSP requirement in accordance with Appendix H of Reference 6.1.3 because the lower RPV is made of austenitic stainless steel and the maximum design life peak fluence of the ferritic portion of the RPV is below 1E+17 n/cm^2, E > 1 MeV. The NPM design supports an exemption to 10 CFR 50.60 due to the absence of ferritic materials in the RPV beltline region.

3.5 Low Temperature Overpressure Protection

The NPM reactor vessel uses LTOP systems for protection against failure during reactor start-up and shutdown operation due to LTOP events classified as service level A or B events. Per Appendix G, paragraph G-2215, of Reference 6.1.5, LTOP systems must be effective at coolant temperatures less than 200 degrees F or at coolant temperatures corresponding to a reactor vessel metal temperature less than RT_{NDT} + 50 degrees F, whichever is greater. {{

}}2(a),(c),ECI

3.6 Pressurized Thermal Shock

Regulation 10 CFR 50.61 requires the PTS screening for the RPV beltline region of PWRs. A PTS event means an event or transient in PWRs causing severe overcooling (thermal shock) concurrent with, or followed by, significant pressure within the RPV. The 10 CFR 50.61 definition of beltline is:

> *(The) RPV beltline means the region of the reactor vessel (shell material including welds, heat affected zones and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage.*

Regulation 10 CFR 50.61 (Reference 6.1.3) does not define significant radiation damage. However, Appendix H of Reference 6.1.3 requires the monitoring of ferritic RPV beltline materials with peak neutron fluence at the end of the design life exceeding 1E+17 n/cm^2, E > 1 MeV.

The 10 CFR 50.61 (Reference 6.1.3) PTS screening methodology is based on calculating the reference temperature for PTS (RT_{PTS}). The RT_{PTS} means RT_{NDT} evaluated for the end of design life peak fluence for each of the vessel beltline materials using the 10 CFR 50.61 (Reference 6.1.3) procedures per the following 10 CFR 50.61 equation:

$$RT_{PTS} = RT_{NDT(u)} + \Delta RT_{NDT} + Margin \qquad \text{Eq. 3-17}$$

The $RT_{NDT(U)}$ is the reference temperature RT_{NDT} before service (unirradiated condition) established by impact testing per NB-2311 of Reference 6.1.6.

The 10 CFR 50.61(b)(2) (Reference 6.1.3) acceptance criteria for passing the PTS screening are: RT_{PTS} not to exceed 270 degrees F for plates, forgings, and axial welds, and RT_{PTS} not to exceed 300 degrees F for circumferential welds.

4.0 Results

4.1 Adjusted Reference Temperature

There is no ART because there is no need to adjust RT_{NDT} for fluence because the peak fluence at the top of the lower flange of the RPV is {{ }}2(a),(c),ECI, which is less than 1E+17n/cm^2, E > 1 MeV.

Appendix H of Reference 6.1.3 requires surveillance of ferritic materials that experience a maximum fluence greater than 1.0E+17 n/cm^2, E > 1 MeV. The upper RPV fluence would have to increase by a factor of {{ }}2(a),(c),ECI to experience a fluence greater than 1.0E+17n/cm^2, E > 1 MeV; therefore, there is no need to adjust the reference temperatures.

4.2 Pressure Temperature Limits

The Appendix G of Reference 6.1.3 P-T limits are based on the requirements presented in Table 3-3. {{

}}2(a),(c),ECI Table 4-1 presents the application of Appendix G, Table 1 of Reference 6.1.3 to the NPM reactor pressure vessel. Figure 4-1 through Figure 4-7 present the uncorrected P-T limits curves.

Table 4-1 Pressure-Temperature Limits for NuScale Power Module Reactor Pressure Vessel per 10 CFR 50, Appendix G

Operating Condition	Vessel Pressure(1)	Requirements for Pressure-Temperature Limits	Minimum Temperature Requirements
Hydrostatic Pressure and Leak Tests (core is not critical)			
Fuel in the Vessel	535.3 psig	ASME BPVC § XI App. G Limits	0 degrees F
Fuel in the Vessel	535.5 psig	ASME BPVC § XI App. G Limits	90 degrees F
No Fuel in the Vessel (preservice hydrostatic test)	535.5 psig	Not Applicable	60 degrees F
Normal Operation (including heatup and cooldown), Including Anticipated Operational Occurrences			
Core Not Critical	535.5 psig	ASME BPVC § XI App. G Limits	0 degrees F
Core Not Critical	535.5 psig	ASME BPVC § XI App. G Limits	120 degrees F
Core Critical	535.5 psig	ASME BPVC § XI App. G Limits + 40degrees F	90 degrees F
Core Critical	535.5 psig	ASME BPVC § XI App. G Limits + 40degrees F	160 degrees F

Notes:
1. Percent of the preservice system hydrostatic test pressure.

Table 4-2 Summary of Pressure-Temperature Limits - Normal

Normal Combined Heatup and Power Ascent Transient (Core Not Critical)		Composite Normal (Core Critical with RPV Pressure = 20 Percent Pressure = 535.3 psig) (Minimum core critical temperature determined from the steady state and transient ISLH curves)		Composite Normal (Core Critical with RPV Pressure > 20 Percent Pressure = 535.3 psig)		Normal Combined Power Descent and Cooldown	
Fluid Temperature degrees F	Pressure psig	Fluid Temperature degrees F	Pressure psig	Fluid Temperature degrees F	Pressure psig	Fluid Temperature degrees F	Pressure psig
65	535	Reactor is not permitted to be critical below 90°F if ISLH testing is performed at steady-state or transient conditions.		Reactor is not permitted to be critical below 160°F if ISLH testing is performed at steady-state or transient conditions.		600	3260
120	535					220	3260
120	2230					210	2400
150	2230	90	0	160	0	150	1875
200	2285	90	535	160	1875	120	1875
300	2475	160	535	190	1875	120	535
600	2475	160	1875	240	2285	65	535
		190	1875	340	2475		
		240	2285	640	2475		
		340	2475				
		640	2475				

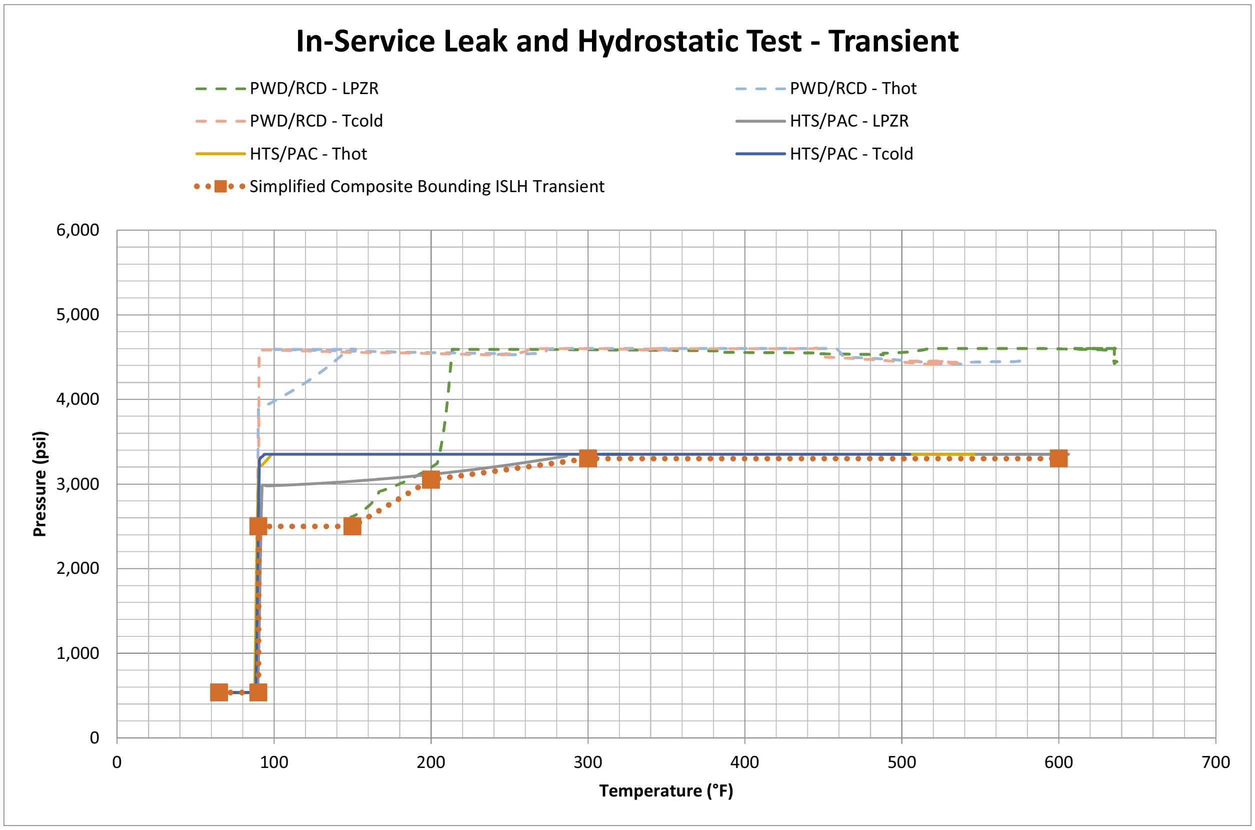
Note: Linear interpolation can be used to calculate the allowable pressures for the temperatures not listed in the table.

Table 4-3 Summary of Pressure-Temperature Limits - Inservice Leak and Hydrostatic Testing

ISLH for Combined Heatup and Power Ascent Transient		ISLH for Combined Power Descent and Cooldown Transient		Transient ISLH (Bounding)		Steady-State ISLH	
Fluid Temperature degrees F	Pressure psig	Fluid Temperature degrees F	Pressure psig	Fluid Temperature degrees F	Pressure psig	Fluid Temperature degrees F	Pressure psig
65	535	600	4350	65	535	65	535
90	535	220	4350	90	535	90	535
90	2980	210	3200	90	2500	90	3660
150	2980	150	2500	150	2500	95	3960
200	3050	90	2500	200	3050	100	4300
300	3300	90	535	300	3300	105	4610
600	3300	65	535	600	3300	600	4610

Note: Linear interpolation can be used to calculate the allowable pressures for the temperatures not listed in the table.

Figure 4-1 Pressure-Temperature Limits for Transient Inservice Leak and Hydrostatic Testing (Composite of Transients)



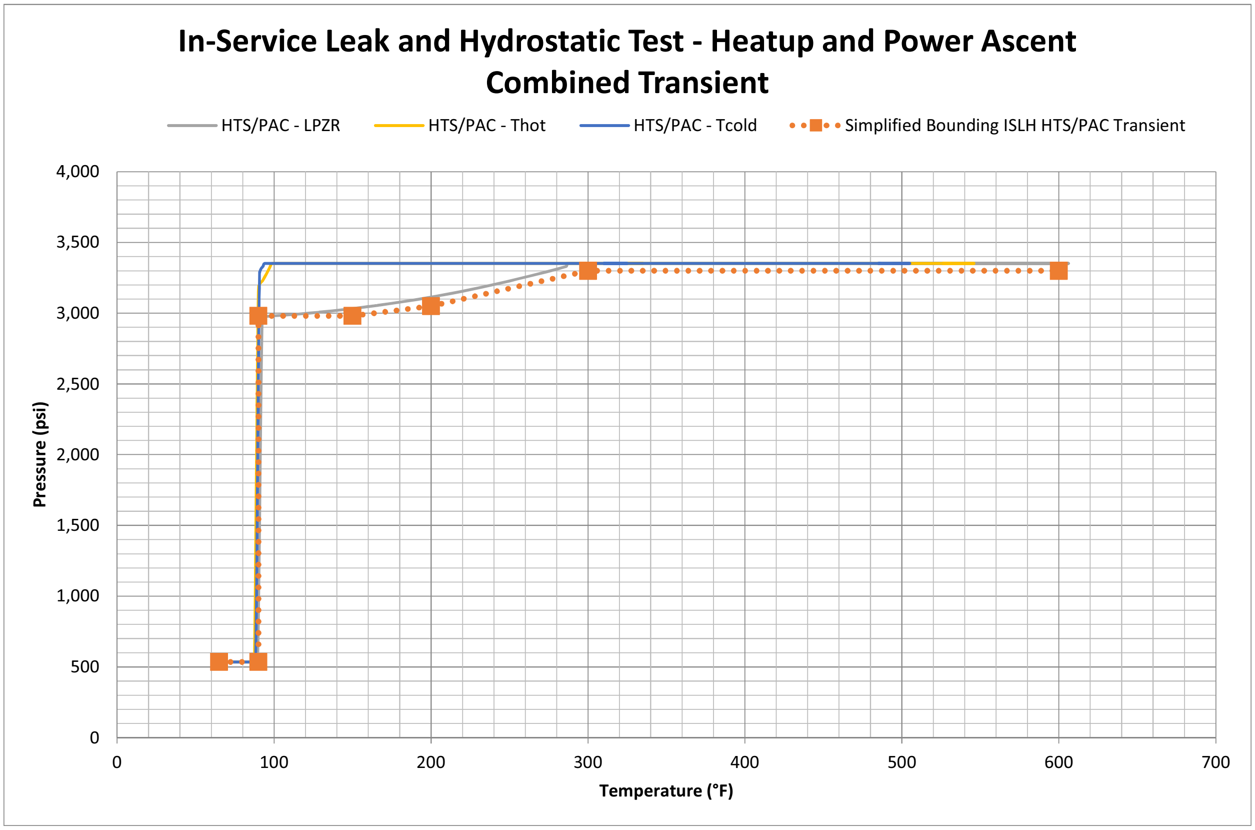
Notes:
1. This image is intended to be viewed in color.
2. The following are abbreviations used in the figure above:
   a. PWD: power descent
   b. RCD: cooldown
   c. LPZR: lower pressurizer region
   d. Thot: reactor coolant system hot temperature
   e. Tcold: reactor coolant system cold temperature
   f. HTS: heatup transient
   g. PAC: power ascent

Figure 4-2 Pressure-Temperature Limits for Steady-State Inservice Leak and Hydrostatic Testing

{{

}}2(a),(c),ECI

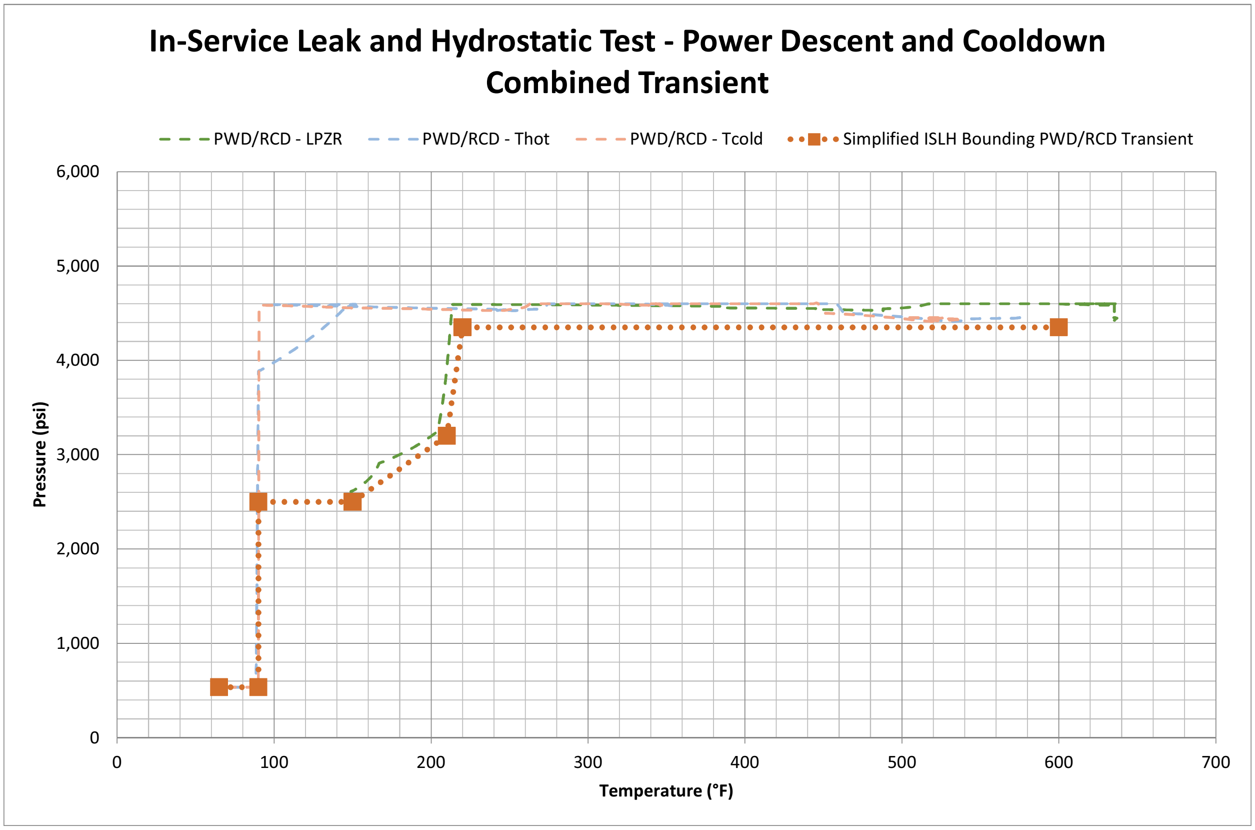
Figure 4-3 Pressure-Temperature Limits for Bounding Heatup and Power Ascent Transient Inservice Leak and Hydrostatic Testing



Notes:

1. This image is intended to be viewed in color.
2. The following are abbreviations used in the figure above:
   a. HTS: heatup
   b. PAC: power ascent
   c. LPZR: lower pressurizer region
   d. Thot: reactor coolant system hot temperature
   e. Tcold: reactor coolant system cold temperature

Figure 4-4 Pressure-Temperature Limits for Bounding Power Descent and Cooldown Transient Inservice Leak and Hydrostatic Testing



Notes:
1. This image is intended to be viewed in color.
2. The following are abbreviations used in the figure above:
    a. PWD: power descent
    b. RCD: cooldown
    c. LPZR: lower pressurizer region
    d. Thot: reactor coolant system hot temperature
    e. Tcold: reactor coolant system cold temperature

Figure 4-5 Pressure-Temperature Limits for Bounding Normal Heatup and Power Ascent Transient

{{

}}2(a),(c),ECI

Figure 4-6 Pressure-Temperature Limits for Bounding Normal Power Descent and Cooldown Transient

{{

}}2(a),(c),ECI

Figure 4-7 Pressure-Temperature Limits for Core Critical Heatup/Power Ascent and Power Descent/Cooldown Transients

{{

}}2(a),(c),ECI

4.3 Low Temperature Overpressure Protection Setpoint Limits

The LTOP setpoint limits the maximum pressure in the reactor vessel to less than the pressure limit curves in Figure 4-8. It uses the minimum pressure from the heatup and cooldown curves. Overpressure protection occurs by opening the two reactor vent valves (RVVs) located on the head of the reactor vessel when exceeding the LTOP pressure setpoint. For a given cold temperature, a pressurizer pressure above the LTOP setpoint causes the module protection system to send a RVV open signal. Above {{ }}2(a),(c),ECI, the reactor safety valves provide overpressure protection. The LTOP logic and components can continue to perform their function in the event of a single active failure.

The reactor safety valves do not lift when LTOP is enabled. This calculation accounts for pressure and temperature measurement uncertainties, the static pressure difference between the pressure measurement and the bottom of the RPV, the maximum delay in the RVV opening, and the delay in sensor response and module protection system processing time.

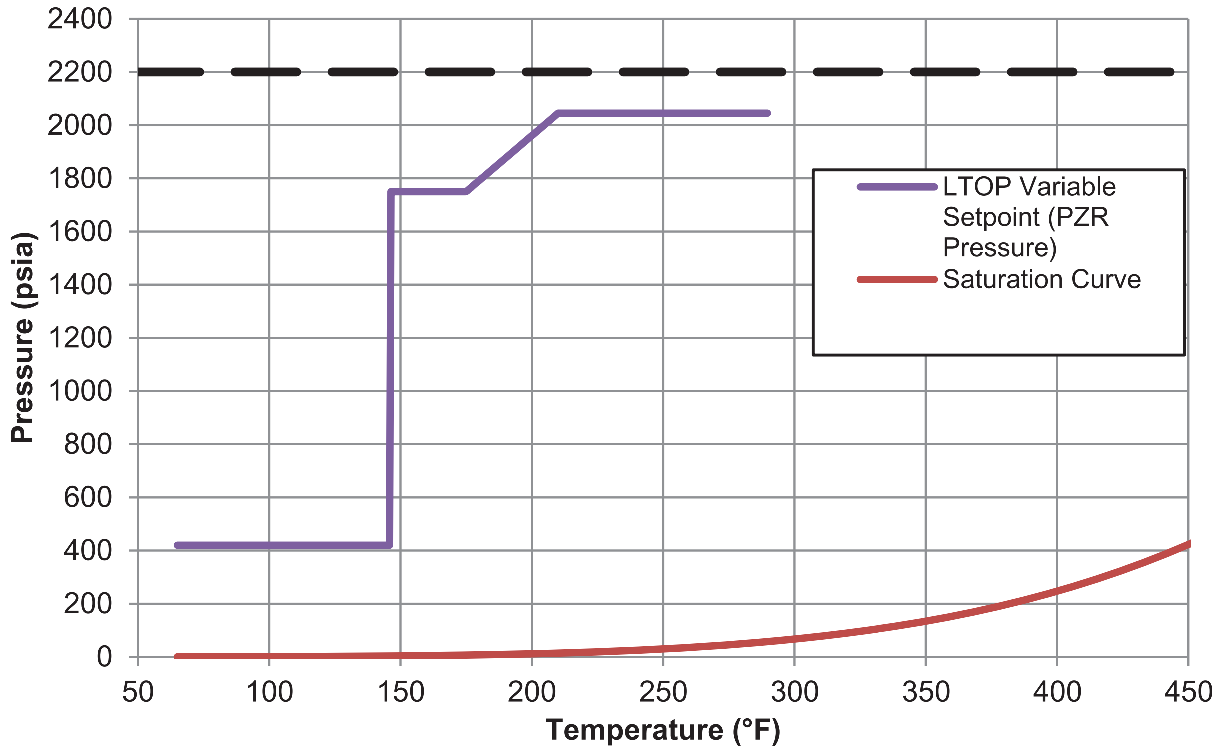
The pressurizer pressure determines the recommended LTOP setpoint; the LTOP setpoint has a conservative bias for the elevation head to the bottom of the RPV. Table 4-4 shows the recommended LTOP pressure setpoint as a function of reactor coolant system (RCS) cold temperature.

Table 4-4 Recommended Low Temperature Overpressure Protection Pressure Setpoint as a Function of Cold Temperature

T_{cold} (degrees F)	Pressurizer Pressure (psia)
<146.0	420.0
146.0	1750.0
175.0	1750.0
210.0	2025.0
290.0	2025.0
>290.0	LTOP not enabled

Figure 4-8 shows the recommended LTOP setpoint along with the saturation pressure curve.

Figure 4-8 Recommended Low Temperature Overpressure Protection Setpoint



4.3.1 Pressurized Thermal Shock Screening

The $RT_{NDT(U)}$ is the RT_{NDT} before service (unirradiated condition) established by impact testing per NB-2331 of Reference 6.1.6.

The 10 CFR 50.61(b)(2) (Reference 6.1.3) acceptance criteria for passing the PTS screening are as follows: RT_{PTS} not to exceed 270 degrees F for plates, forgings, and axial welds; and RT_{PTS} not to exceed 300 degrees F for circumferential welds.

Per NB-2311 of Reference 6.1.6, austenitic stainless steels are exempt from impact test requirements and therefore are exempt from RT_{NDT} requirements of NB-2331 of Reference 6.1.6. While 10 CFR 50.61 (Reference 6.1.3) does not specifically state that it applies only to ferritic materials, the chemistry factors in Table 1 and Table 2 of 10 CFR 50.61 (Reference 6.1.3) were derived for ferritic materials. Therefore, the PTS screening requirements in 10 CFR 50.61 (Reference 6.1.3) do not apply to the austenitic stainless steel used in the lower RPV of the NPM (Table 3-1). While there are ferritic materials in the upper RPV of the NPM, the 57 EFPY design life peak fluence for the top surface of the lower RPV flange is {{ }}[2(a),(c),ECI]. Hence, the design life peak fluence for the upper RPV shell is below the Appendix H of Reference 6.1.3 threshold value of 1E+17 n/cm^2, E > 1 MeV for the RPV beltline. Therefore, PTS screening of the upper RPV is not required, and PTS screening does not apply to the lower RPV shell.

5.0 Summary and Conclusions

This report contains methodology based on Appendix G of Reference 6.1.3 and Appendix G of Reference 6.1.5 for the RCPB and P-T limits applicable to the NPM. An example set of P-T curves applicable to the NPM included in this report use these methods. These limits account for the effects of neutron-induced embrittlement up to an exposure of 57 EFPY fluence. Curves developed include

- transient ISLH.
- steady-state ISLH.
- bounding heatup and power ascent transient ISLH.
- bounding power descent and cooldown transient ISLH.
- bounding normal heatup and power ascent.
- bounding normal power descent and cooldown.
- core critical.

The NPM design does not necessitate an RVSP to ensure adequate fracture toughness.

This report contains the LTOP limits and methodology for the NPM.

Using the material properties of an as-built reactor vessel, the licensee may use the methods developed in this report to develop their P-T limits and LTOP setpoints.

The PTS screening requirement of 10 CFR 50.61 (Reference 6.1.3) does not apply to the NPM reactor pressure vessel.

6.0 References

6.1 Source Documents

6.1.1 U.S. Nuclear Regulatory Commission, “Radiation Embrittlement of Reactor Vessel Materials,” Regulatory Guide 1.99, Revision 2, May 1988.

6.1.2 U.S. Nuclear Regulatory Commission, “Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits,” Generic Letter 96-03, January 1996.

6.1.3 U.S. Code of Federal Regulations, “Domestic Licensing of Production and Utilization Facilities,” Part 50, Chapter I, Title 10, “Energy,” (10 CFR 50).

6.1.4 U.S. Nuclear Regulatory Commission, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” NUREG-0800, Revision 2, June 1987.

6.1.5 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2017 Edition, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, New York, NY.

6.1.6 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2017 Edition, Section III, Rules for Construction of Nuclear Facility Components, New York, NY.

Licensing Technical Report

Use of Austenitic Stainless Steel for NPM Lower Reactor Pressure Vessel

December 2022

Revision 0

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Licensing Technical Report

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Abstract

This report describes the acceptability of SA-965 Grade FXM-19 austenitic stainless steel base metal and E/ER209 or E/ER240 weld filler metal for use in the NuScale Power Module (NPM) lower reactor pressure vessel (RPV).

The RPVs in operating pressurized water reactors (PWRs) in the United States are made of ferritic materials. The Nuclear Regulatory Commission (NRC) regulations in Title 10 of the Code of Federal Regulations (CFR) Section 50.60 (10 CFR 50.60); 10 CFR 50.61; 10 CFR 50, Appendix G; and 10 CFR 50, Appendix H, either refer specifically to or utilize data for ferritic materials only. These regulations support compliance with General Design Criterion (GDC) 14, GDC 15, GDC 31, and GDC 32.

Since there are no regulatory data or guidance available for austenitic stainless steel RPVs, the NPM RPV beltline cannot be evaluated using the current regulations.

This report summarizes the known data relating to FXM-19 base metal and E/ER209 or E/ER240 weld filler metal. The results of the literature review support exemptions from 10 CFR 50.60 and 10 CFR 50.61. This report also summarizes NuScale's position on the reactor vessel surveillance program (RVSP) requirements in 10 CFR 50, Appendix H, and GDC 32.

Executive Summary

This report describes the acceptability of SA-965 Grade FXM-19 austenitic stainless steel base metal and E/ER209 or E/ER240 weld filler metal for use in the NuScale Power Module (NPM) lower reactor pressure vessel (RPV).

The RPVs in operating pressurized water reactors (PWRs) in the United States are made of ferritic materials. The Nuclear Regulatory Commission (NRC) regulations in Title 10 of the Code of Federal Regulations (CFR) Section 50.60 (10 CFR 50.60); 10 CFR 50.61; 10 CFR 50, Appendix G; and 10 CFR 50, Appendix H, either refer specifically to or utilize data for ferritic materials only. These regulations support compliance with General Design Criterion (GDC) 14, GDC 15, GDC 31, and GDC 32.

NuScale evaluated data on austenitic stainless steel, including SA-965 Grade FXM-19 base metal, E/ER209 or E/ER240 weld filler metal, and Type 3XX austenitic stainless steel for comparison purposes. The data show that austenitic stainless steels have superior ductility and are less susceptible to the effects of neutron and thermal embrittlement than ferritic materials. In addition, the data and methodology in 10 CFR 50.61; 10 CFR 50, Appendix G; and 10 CFR 50, Appendix H, are not applicable to non-ferritic materials.

In addition to assessing material properties and the safety of austenitic stainless steel in the lower RPV, NuScale assessed the beltline of the RPV. Since the lower RPV is austenitic stainless steel and thus less susceptible to the effects of neutron and thermal embrittlement than ferritic materials, it should not be considered the RPV beltline. The upper RPV, which is made of ferritic steel, is not within the RPV beltline; the 57 effective full-power year (EFPY) peak design fluence for the upper RPV is less than the 10 CFR 50, Appendix H, reactor vessel surveillance program (RVSP) threshold of 1E+17 n/cm^2, E > 1 MeV, and thus is not subject to supplementary fracture toughness and reactor vessel surveillance program (RVSP) requirements to address the effects of neutron embrittlement.

Though the NPM design does not use the methodology in 10 CFR 50.61; 10 CFR 50, Appendix G; and 10 CFR 50, Appendix H, it does meet the requirements of GDC 14, GDC 15, GDC 31, and GDC 32. The requirements of GDC 14, GDC 15, and GDC 31 are met by ensuring that the NPM lower RPV is constructed of a material that has superior ductility and is less susceptible to the effects of neutron and thermal embrittlement compared to ferritic materials, which increases the integrity and safety of the RCPB. Because the austenitic stainless steel used in the lower RPV is less susceptible to the effects of neutron and thermal embrittlement compared to ferritic materials, and because the ferritic materials in the RPV are below the 10 CFR 50, Appendix H, RVSP threshold, the RPV does not need an RVSP to ensure adequate fracture toughness; therefore, the portion of GDC 32 requiring an "appropriate" material surveillance program is satisfied without an RVSP.

The US600 design used austenitic stainless steel for the lower containment vessel (CNV) because its material properties are less susceptible to the effects of neutron and thermal embrittlement than ferritic materials.

In Section 6.1.1.4.2 of the US600 design final safety evaluation report, the NRC stated:

> *The staff finds the use of SA-965, Grade FXM-19, and its associated weld filler metals acceptable for use in the lower portion of the CNV, as the calculated fluence to the CNV is lower than what is expected to cause embrittlement, and the selection of SA-965, Grade FXM-19, an austenitic stainless steel, is resistant to radiation embrittlement.*

In Section 6.2.7.4 of the final safety evaluation report, the NRC stated:

> *Within the ASME Code, detailed fracture toughness requirements are placed on ferritic materials, as nonferritic materials exhibit sufficient inherent fracture toughness that additional requirements are deemed unnecessary. For example, the austenitic stainless steel used for the CNV lower shell, SA-965, FXM-19, was explicitly chosen specifically for its superior fracture toughness and resistance to neutron embrittlement.*

The results of this report confirm that austenitic stainless steels and compatible weld filler metals are likewise acceptable for use in the lower RPV without additional fracture toughness requirements because they have superior ductility and are less susceptible to the effects neutron and thermal embrittlement than ferritic materials.

1.0 Introduction

1.1 Purpose

The NuScale Power Module (NPM) lower reactor pressure vessel (RPV) is made of SA-965 Grade FXM-19 austenitic stainless steel base metal and uses E/ER209 or E/ER240 weld filler metal. The use of austenitic stainless steel benefits overall plant safety because it has superior ductility and is less susceptible to the effects of neutron and thermal embrittlement than ferritic materials.

Nuclear Regulatory Commission (NRC) regulations in Title 10 of the Code of Federal Regulations (CFR) Section 50.60 (10 CFR 50.60); 10 CFR 50.61; 10 CFR 50, Appendix G; and 10 CFR 50, Appendix H, while applicable to all materials, were developed for and only include data and methods for ferritic materials. Four known RPVs used austenitic stainless steel RPVs, and reactor vessel internals (RVIs) in operating light water reactors are made of austenitic stainless steel. NuScale assessed available literature for FXM-19 base metal and bounding materials for the weld filler metals to evaluate the safety of using austenitic stainless steel in the lower RPV. NuScale identified four RPVs and RVIs made of austenitic stainless steel. The available literature demonstrates the acceptability of austentic stainless steel for the lower RPV of the NPM, without additional fracture toughness or material surveillance requirements. Therefore, the US460 standard design supports exemptions from 10 CFR 50.60 and 10 CFR 50.61.

The US460 standard design complies with General Design Criterion (GDC) 14, GDC 15, GDC 31, and GDC 32.

1.2 Scope

This report applies to the US460 standard design with an austenitic stainless steel lower RPV and supports the Standard Design Approval Application (SDAA).

1.3 Abbreviations

Table 1-1 Abbreviations

Term	Definition
ASME	American Society of Mechanical Engineers
ATR	Advanced Test Reactor
BPVC	Boiler Pressure Vessel Code
CASS	cast austenitic stainless steel
CFR	Code of Federal Regulations
dpa	displacements per atom
EFPY	effective full-power years
EPRI	Electric Power Research Institute
FN	ferrite number
GDC	General Design Criterion
INL	Idaho National Laboratory
LWR	light water reactor
NPM	NuScale Power Module
NRC	Nuclear Regulatory Commission
PTS	pressurized thermal shock
PWR	pressurized water reactor
RCPB	reactor coolant pressure boundary
RPV	reactor pressure vessel
RT_{NDT}	nil-ductility reference temperature
$RT_{NDT(u)}$	unirradiated nil-ductility reference temperature
RT_{PTS}	reference temperature for pressurized thermal shock
RVI	reactor vessel internals
RVSP	reactor vessel surveillance program

2.0 Background

The US460 standard design uses austenitic stainless steel and compatible weld filler metals in the lower RPV because austenitic stainless steels have superior ductility and lower susceptibility to the effects of neutron and thermal embrittlement compared to ferritic materials. This increases the integrity and safety of the reactor coolant pressure boundary (RCPB). By NRC regulations, the beltline is the region of the RPV that directly surrounds the effective height of the active core and is predicted to experience sufficient neutron radiation such that it is the most limiting material with regard to radiation damage. For the US460 standard design, the lower RPV contains the beltline; however, the lower RPV is made of austenitic stainless steel, which can withstand the most severe radiation damage in the RPV better than ferritic materials. Therefore, use of austenitic stainless steel and compatible weld filler metals increases the overall safety of the RPV.

NuScale conducted a literature review of data related to SA-965 Grade FXM-19 base metal and to E/ER209 or E/ER240 weld filler metals, as well as a comparison to 3XX austenitic stainless steel properties.

2.1 Regulatory Requirements

2.1.1 General Design Criterion 14 - Reactor Coolant Pressure Boundary

General Design Criterion 14 (Reference 5.1.5) requires:

> *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.*

2.1.2 General Design Criterion 15 - Reactor Coolant System Design

General Design Criterion 15 (Reference 5.1.6) requires:

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.

2.1.3 General Design Criterion 31 - Fracture Prevention of Reactor Coolant Pressure Boundary

General Design Criterion 31 (Reference 5.1.7) requires:

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.

2.1.4 General Design Criterion 32 - Inspection of Reactor Coolant Pressure Boundary

General Design Criterion 32 (Reference 5.1.8) requires:

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.

2.1.5 10 CFR 50.60 - Acceptance Criteria for Fracture Prevention Measures for Light Water Nuclear Power Reactors for Normal Operation

10 CFR 50.60 (Reference 5.1.1) requires that licensed light water reactors (LWRs) meet the fracture toughness and material surveillance program requirements for ferritic materials in the RCPB set forth in 10 CFR 50, Appendix G (Reference 5.1.3), and in 10 CFR 50, Appendix H (Reference 5.1.4). Proposed alternatives to the requirements described in Reference 5.1.3 or Reference 5.1.4 or portions thereof may be used when the NRC grants an exemption under 10 CFR 50.12.

2.1.6 10 CFR 50, Appendix G - Fracture Toughness Requirements

10 CFR 50, Appendix G (Reference 5.1.3), specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the RCPB of LWRs to provide adequate margins of safety during any condition of normal operation.

Conditions of normal operation include anticipated operational occurrences and system hydrostatic tests to which the pressure boundary may be subjected over its service lifetime. 10 CFR 50, Appendix G, requires that beltline materials be tested in accordance with 10 CFR, Appendix H, the results of which are utilized in establishing the fracture toughness requirements for those materials.

2.1.7 10 CFR 50, Appendix H - Reactor Vessel Material Surveillance Program Requirements

10 CFR 50, Appendix H (Reference 5.1.4), requires that licensees establish and maintain a material surveillance program to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of light water nuclear power reactors. The materials in the reactor vessel beltline region undergo exposure to neutron irradiation and to the thermal environment.

2.1.8 10 CFR 50.61 - Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events

10 CFR 50.61 (Reference 5.1.2) requires the pressurized thermal shock (PTS) event screening for the RPV beltline region of pressurized water reactors (PWRs). Pressurized thermal shock events are events or transients in PWRs causing severe overcooling (thermal shock) concurrent with or followed by significant pressure within the RPV.

3.0 Evaluation of Austenitic Stainless Steel Properties

3.1 NPM Lower Reactor Pressure Vessel and Beltline

The NPM lower RPV is made of SA-965 Grade FXM-19 austenitic stainless steel and contains two SA-965 Grade FXM-19 forgings joined by one circumferential weld. The permitted weld filler metal for the lower RPV circumferential weld is SFA 5.4, E209 or SA5.4, E240 and SA 5.9 E209 or SA 5.9 E240 (E/ER209 or E/ER240). Figure 3-1 shows the configuration of the NPM lower RPV.

Figure 3-1 NPM Lower Reactor Pressure Vessel Pressure-Retaining Materials

{{

}}2(a),(c),ECI

3.2 NPM Lower Reactor Pressure Vessel Austenitic Stainless Steel Characteristics

SA-965 Grade FXM-19 is a nitrogen-strengthened austenitic stainless steel with a nominal composition of 22Cr-13Ni-5Mn. The unified numbering system designation is S20910. This material is known as XM-19 or Nitronic 50 in literature.

Table 3-1 compares the chemical composition of SA-965 Grade FXM-19 base metal and E/ER209 or E/ER240 weld filler metal for the lower RPV with Type 3XX austenitic stainless steels. Table 3-2 shows the tensile requirements and that SA-965 Grade FXM-19 and E/ER209 or E/ER240 are stronger than Type 3XX austenitic stainless steels due to elevated levels of manganese and nitrogen.

Table 3-1 Austenitic Stainless Steel Chemical Compositions

Material	Chemical Composition Requirement (weight percent)(1)										
	C	Mn	P	S	Si	Ni	Cr	Mo	NB+Ta	N	V	Cu
FXM-19	0.06(2)	4.0-6.0	0.045	0.03	1.0	11.5-13.5	20.5-23.5	1.5-3.0	0.10-0.30	0.20-0.40	0.10-0.30	--
SA-965, F304	0.08	2.0	0.045	0.03	1.0	8.0-11.0	18.0-20.0	--	--	--	--	--
SA-965, F316	0.08	2.0	0.045	0.03	1.0	10.0-14.0	16.0-18.0	2.0-3.0	--	--	--	--
SFA 5.4, E209	0.06(2)	4.0-7.0	0.04	0.03	1.0	9.5-12.0	20.5-24.0	1.5-3.0	--	0.10-0.30	0.10-0.30	0.75
SFA 5.9, ER209	0.05(2)	4.0-7.0	0.03	0.03	0.9	9.5-12.0	20.5-24.0	1.5-3.0	--	0.10-0.30	0.10-0.30	0.75
SFA 5.4, E240	0.06(2)	10.5-13.5	0.04	0.03	1.0	4.0-6.0	17.0-19.0	0.75	--	0.10-0.30	--	0.75
SFA 5.9, ER240	0.05(2)	10.5-13.5	0.03	0.03	1.0	4.0-6.0	17.0-19.0	0.75	--	0.10-0.30	--	0.75
SFA 5.4, E308	0.08	0.5-2.5	0.04	0.03	1.00	9.0-11.0	18.0-21.0	0.75	--	--	--	0.75
SFA 5.9, ER308	0.08	1.0-2.5	0.03	0.03	0.30-0.65	9.0-11.0	19.5-22.0	0.75	--	--	--	0.75

(1) Values are maximum unless there is a range.

(2) For the lower RPV, the maximum carbon content is limited to 0.04 percent for the base metal and for the E/ER209 or E/ER240 weld filler metal.

Table 3-2 Austenitic Stainless Steel Room Temperature Tensile Requirements

Material	Minimum Yield Strength		Minimum Tensile Strength		Minimum Total Elongation
	MPa	ksi	MPa	ksi	percent
SA-965 FXM-19	380	55	690	100	30
SA-965 F304 and F316	205	30	485	70	30
SFA 5.4 E209 and E240; SFA 5.9 ER209 and ER240	--	--	690	100	15
SFA 5.4 E308; SFA 5.9 ER308	--	--	550	80	30

3.3 Austenitic Stainless Steel Literature Search Results

NuScale searched available literature for data on austenitic stainless steel properties to support the statement that austenitic stainless steel has superior ductility and is less susceptible to the effects of neutron and thermal embrittlement than ferritic materials.

Two studies contain tensile properties of irradiated FXM-19 (Reference 5.2.1 and Reference 5.2.2). Reference 5.2.1 contains data from FXM-19 irradiated in the RBT6 reactor, and Reference 5.2.2 contains data from the Idaho National Laboratory (INL) Advanced Test Reactor (ATR); the ATR study reports fracture toughness properties of irradiated FXM-19 (Reference 5.2.2).

While there are studies of irradiated FXM-19, there are no studies that assess FXM-19 at fluence levels relevant to the NPM lower RPV; however, there is extensive data for irradiated Type 3XX austenitic stainless steels because Type 3XX base metal, weld filler metal, and equivalent casting are used as structural materials in RVIs in operating LWRs. The Electric Power Research Institute (EPRI) Material Reliability Program reviews fracture toughness data on behalf of PWR owners and reports fracture toughness data at different fluences and temperatures (Reference 5.2.3). In addition, Argonne National Laboratory, on behalf of PWR owners and the NRC, reviewed irradiated fracture toughness data for Type 3XX austenitic stainless steels and summarized the results in NUREG/CR-7027 (Reference 5.2.4).

Thermal embrittlement or thermal aging embrittlement is a time- and temperature-dependent process whereby a material undergoes microstructural changes leading to decreased ductility and degradation of toughness and impact properties. According to Reference 5.2.3, wrought austenitic stainless steels are not subject to thermal embrittlement at PWR operating temperatures; however, cast austenitic stainless steel (CASS) and austenitic stainless steel welds are potentially susceptible because they contain residual delta ferrite.

Four LWRs not regulated by the NRC had or have RPVs made from austenitic stainless steel, as shown in Table 3-3. Information pertinent to RPV neutron embrittlement was found for the MH-1A and for the ATR.

Table 3-3 Light Water Reactors with Reactor Pressure Vessels Made from Type 3XX Austenitic Stainless Steel

Reactor Name	Reactor Type	Operator	Years Active	Capacity	RPV Material
PM-1(1)	PWR(1)	US Air Force(1)	1962 - 1968(1)	1.25 MWe(1)	Type 347(2)
PM-3A(3)	PWR(3)	US Navy(3)	1962 - 1972(3)	1.75 MWe(3)	Type 347(2)
MH-1A(4)	PWR(4)	US Army(4)	1967 - 1977(4)	10 MWe(4)	Type 316(2)
ATR(5)	PWR(5)	Department of Energy (INL)(5)	1967 - present(5)	250 MWt(5)	Type 304(6)

(1) Reference 5.2.5
(2) Reference 5.2.6
(3) Reference 5.2.7
(4) Reference 5.2.8
(5) Reference 5.2.9
(6) Reference 5.2.10

3.4 Evaluation of Data on Austenitic Stainless Steel to Address Regulations

3.4.1 10 CFR 50.60

10 CFR 50.60 (Reference 5.1.1) requires all LWRs to meet the fracture toughness and material surveillance program requirements for the RCPB in 10 CFR 50, Appendix G (Reference 5.1.3), and 10 CFR 50, Appendix H (Reference 5.1.4).

Current operating LWRs regulated by the NRC have RPVs made of carbon and low-alloy steels (ferritic materials), and the regulations and guidance contain data and procedures pertaining only to ferritic materials. Reference 5.1.1 requires compliance with the fracture toughness requirements in 10 CFR 50, Appendix G (Reference 5.1.3), and with the RVSP requirements in 10 CFR 50, Appendix H (Reference 5.1.4). Since the data and requirements for Reference 5.1.4 and Reference 5.1.4 only apply to ferritic materials, the NPM lower RPV must comply with the intent of the regulations in order to demonstrate compliance with GDC 14, GDC 15, and GDC 31.

A survey of data on austenitic stainless steel and its tensile strength and fracture toughness after irradiation, as well as the effects of neutron and thermal embrittlement on austenitic stainless steel, is in Section 3.4.1.1 and Section 3.4.1.2.

From the regulations, the RPV beltline is the region of the RPV that directly surrounds the effective height of the active core and is predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage. 10 CFR 50, Appendix H (Reference 5.1.4), requires monitoring for neutron and thermal embrittlement of ferritic materials in the RPV, including base metal, weld metal, and the heat-affected zone of the RCPB beltline during the RPV design life. Reference 5.1.4 requires a RVSP for the ferritic materials whose design life peak fluence exceeds 1E+17 n/cm^2, E > 1 MeV. The 57 effective full-power year (EFPY) peak fluence for the NPM lower

RPV beltline base metal (inside surface) is {{ }}$^{2(a),(c),ECI}$; the 57 EFPY peak fluence for the NPM lower RPV beltline weld filler metal (inside surface) is {{ }}$^{2(a),(c),ECI}$, E > 1 MeV. Since the beltline is in the lower RPV, which is made of austenitic stainless steel, the region of the RPV containing ferritic materials that experiences the highest fluence is evaluated for an RVSP. Therefore, the top surface of the upper RPV lower flange is evaluated. The 57 EFPY peak fluence for the top surface of the lower flange is {{ }}$^{2(a),(c),ECI}$, E > 1 MeV. Therefore, the design life peak fluence for the upper RPV is less than the threshold value of 1E+17 n/cm^2, E > 1 MeV.

3.4.1.1 Fracture Toughness Evaluation

Because Type 304 heavy reflectors surround the NPM fuel assemblies in the RVIs, the 57 EFPY peak fluence for the NPM lower RPV beltline base metal (inside surface) is {{ }}$^{2(a),(c),ECI}$, E > 1 MeV, and the 57 EFPY peak fluence for the NPM lower RPV beltline weld filler metal (inside surface) is {{ }}$^{2(a),(c),ECI}$, E > 1 MeV. The 57 EFPY peak fluence values convert to {{ }}$^{2(a),(c),ECI}$ for the NPM lower RPV base metal and {{ }}$^{2(a),(c),ECI}$ for the NPM lower RPV weld filler metal. Figure 3-2 and Figure 3-3 show the effect of neutron irradiation on the tensile properties of FXM-19 that was irradiated in light water moderated research reactors or test reactors.

Reference 5.2.1 reflects data from solution-annealed FXM-19 specimens irradiated and tested at 572 degrees F to 0.0007 dpa, 0.007 dpa, and 0.05 dpa. The uniform elongation declined slightly at 0.05 dpa but remained high (greater than 40 percent), and the total elongation did not change. Therefore, neutron embrittlement of solution-annealed FXM-19 was minor after irradiation to 0.05 dpa, which bounds the 57 EFPY peak fluence for the NPM lower RPV.

Reference 5.2.2 reports data from mill-annealed FXM-19 specimens irradiated at 550 degrees F and tested at 572 degrees F; however, no unirradiated specimens were tested at ATR, so the unirradiated data in Figure 3-2 is used for comparison. The ATR specimens were irradiated to 0.076 dpa. The uniform elongation increased by about 28 percent, and the total elongation increased by about 10 percent. Uniform elongation remained very high (greater than 30 percent) at 0.076 dpa. Therefore, neutron embrittlement of solution-annealed FXM-19 was minor after irradiation to 0.076 dpa, which bounds the 57 EFPY peak fluence for the NPM lower RPV.

Figure 3-2 and Figure 3-3 show the effect of neutron irradiation on the tensile properties of FXM-19 that was irradiated in light water moderated research reactors or test reactors. Table 3-4 summarizes the fracture toughness test results of the mill-annealed FXM-19 irradiated at the ATR, corresponding to Figure 3-3. The irradiated and unirradiated fracture toughness specimens were in the L-T orientation with respect to the original major working direction and were tested at 550 degrees F. Irradiation to 0.08 dpa caused only 4 percent reduction in average plane-strain fracture-toughness values (K_{JQ} or K_{Jc}) at 550 degrees F. Therefore, neutron embrittlement of mill-annealed FXM-19 after irradiation to 0.08 dpa was insignificant. It is noted that 0.08 dpa is much greater than the 57 EFPY peak neutron dose of the lower RPV.

Although a neutron dose above 0.08 dpa is far beyond the lower RPV design life fluence, the mill-annealed FXM-19 still possessed high plane-strain fracture toughness exceeding 200 MPa√m after irradiation to 1.4 dpa, which is greater than the 57 EFPY peak neutron dose of the lower RPV.

Figure 3-2 Typical Stress-Strain Curves of Solution-Annealed FXM-19 Irradiated in RBT6

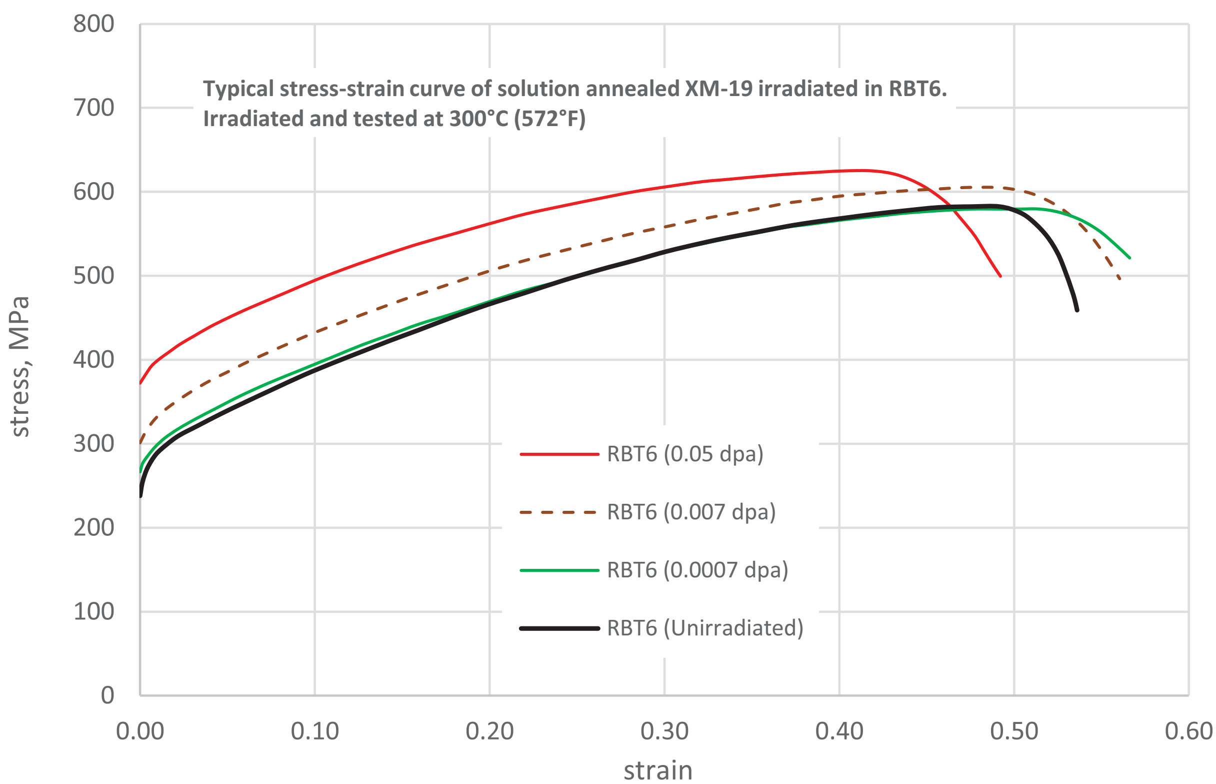


Figure 3-3 Typical Stress-Strain Curves of Mill-Annealed FXM-19 Irradiated in the Advanced Test Reactor

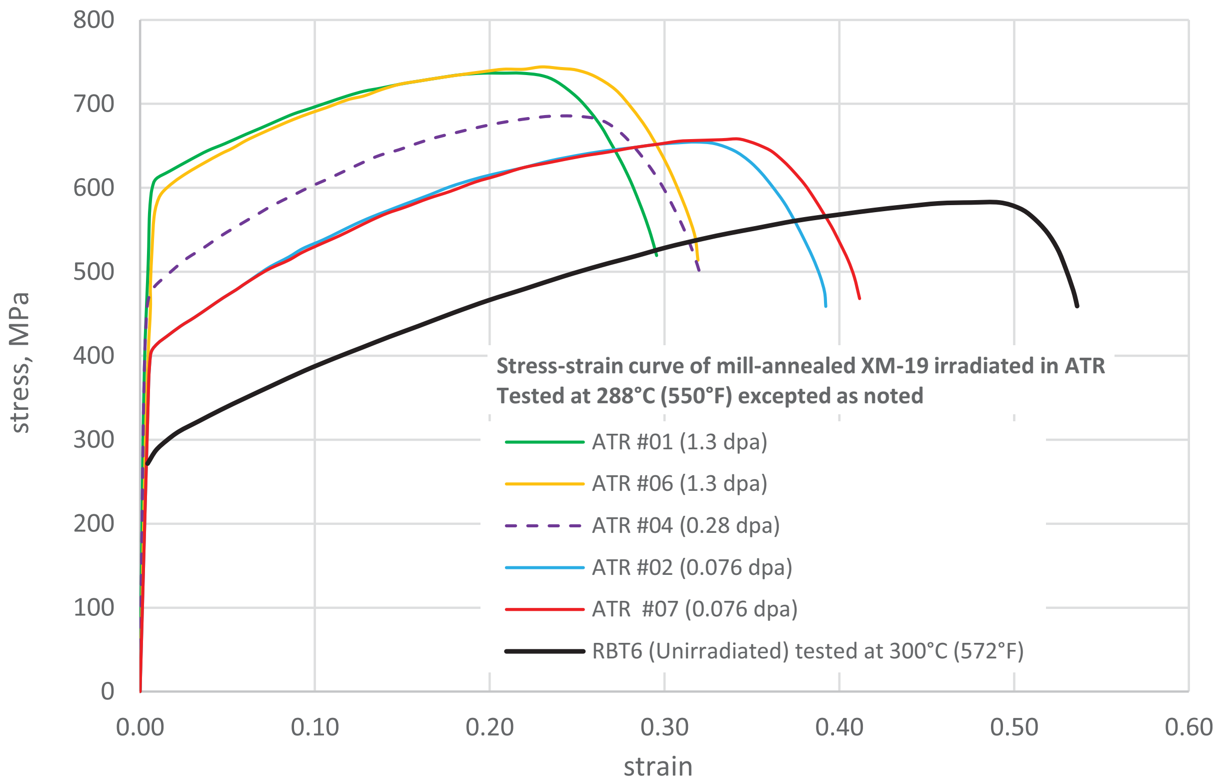


Table 3-4 Fracture Toughness of Unirradiated and Irradiated Mill-Annealed FXM-19

FXM-19 Fracture Toughness Specimen[(1)]	Specimen Size	Average Irradiation Temperature (degrees F)	Fluence	J_Q or J_{Ic}	K_{JQ} or K_{Ic}		
			dpa	kJ/m²	MPa√m	ksi√in[(2)]	Percent Change
Unirradiated C747	1T-CT	Unirradiated	0	388	296	269	--
Unirradiated C748	1T-CT	Unirradiated	0	354	283	258	--
Average of two unirradiated specimens				371	290	263	--
Irradiated in ATR 10A0001A02	0.4T-CT	621	0.08	312	265	241	--
Irradiated in ATR 10A0001A07	0.4T-CT	642	0.08	377	291	265	--
Average of two specimens irradiated to 0.08 dpa				345	278	253	-4
Irradiated in ATR 10A0001B01	0.4T-CT	624	0.29	231	230	209	--
Irradiated in ATR 10A0001B02	0.4T-CT	662	0.29	310	266	242	--
Average of two specimens irradiated to 0.29 dpa				271	248	226	-14
Irradiated in ATR 10A0001D05	0.4T-CT	631 / 507[(3)]	1.47	303	262	238	--
Irradiated in ATR 10A0001D01	0.4T-CT	626 / 502[(3)]	1.43	203	215	196	--
Irradiated in ATR 10A0001D04	0.4T-CT	574 / 507[(3)]	1.41	251	237	216	--
Average of three specimens irradiated from 1.41 dpa to 1.47 dpa				252	238	217	-18

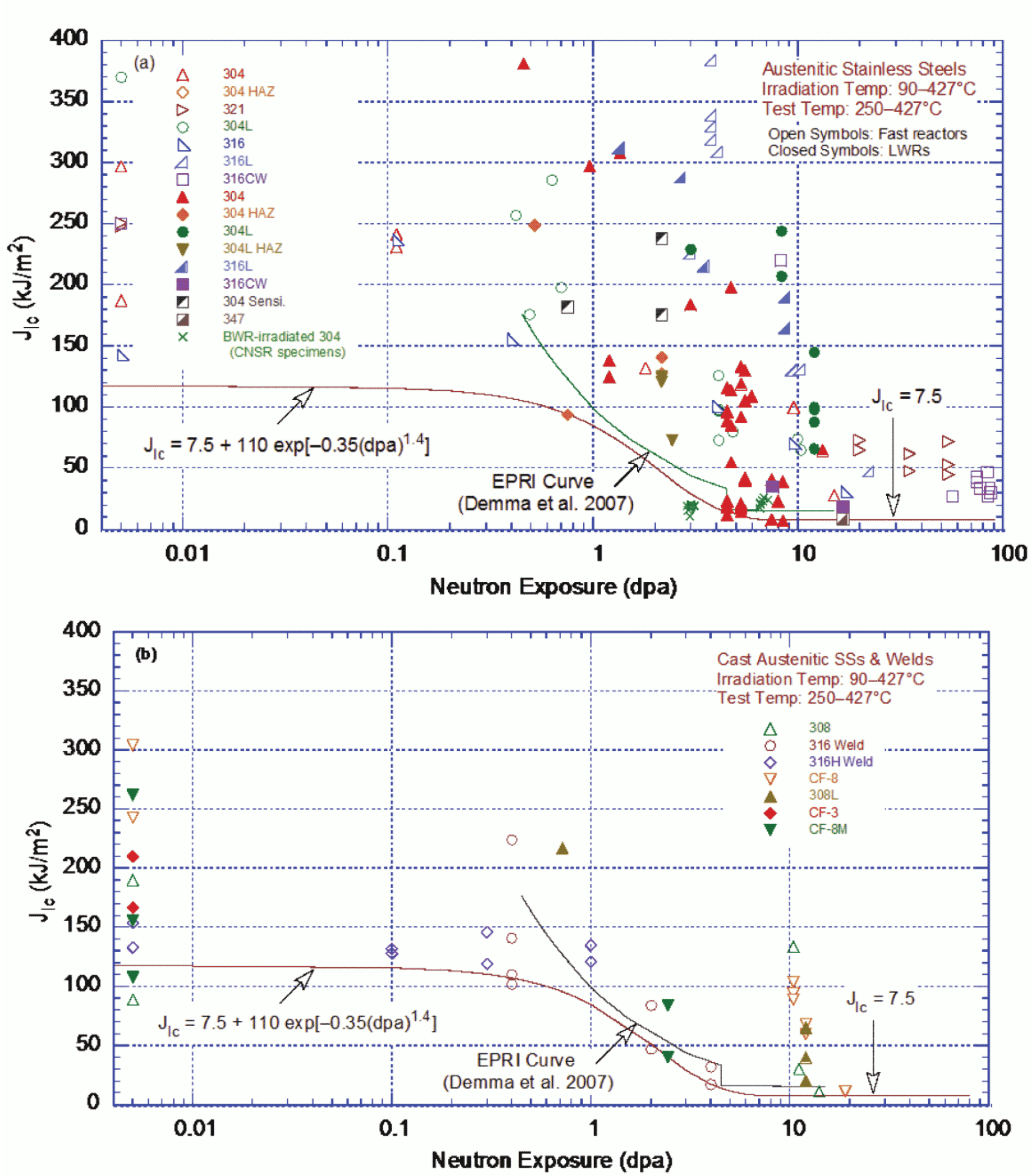
(1) Irradiated values are from Table 4-6 of Reference 5.2.2. Unirradiated values of the same FXM-19 heat is from Table 1 of Reference 5.2.11. Values listed are from specimens tested in the L-T orientation.

(2)1 MPa√m = 0.910 ksi√in

(3) First cycle irradiation temperature and second cycle irradiation temperature, respectively.

Argonne National Laboratory reviewed irradiated fracture toughness data of Type 3XX austenitic stainless steels and summarized the results in Reference 5.2.4. Table 3-4 shows the fracture toughness as a function of fluence level with an irradiation temperature and test temperature range of 482 degrees F to 800 degrees F. Based on the data in Table 3-4, Reference 5.2.4 proposes a 0.5 dpa threshold fluence for austenitic stainless steel base metal and a 0.3 dpa threshold fluence for austenitic stainless steel weld filler metal; the threshold fluence is the fluence below which irradiation has little to no effect on fracture toughness.

Figure 3-4 Fracture Toughness of Irradiated Type 3XX Austenitic Stainless Steel



FXM-19 typically contains 0.3 percent nitrogen, which is higher than the nitrogen level of Type 3XX austenitic stainless steels. Solid solution strengthening with nitrogen is also used for Type 3XX grades with a nitrogen range of 0.10 percent to 0.16 percent. There is no evidence in the literature to suggest that higher manganese and nitrogen content in FXM-19 compared to Type 3XX austenitic stainless steels contributes to heightened effects of neutron embrittlement.

Table 3-5 compares the fracture toughness of FXM-19 from Reference 5.2.2 (shown in Table 3-4) with the Reference 5.2.4 fracture toughness of Type 3XX austenitic stainless steel. The irradiated FXM-19 values from Reference 5.2.2 are well above the lower bound of the Reference 5.2.4 data. Figure 3-5 also shows that the FXM-19 fracture toughness of mill-annealed FXM-19 responds to neutron exposure similarly to Type 3XX austenitic stainless steels.

Finally, for the NPM, the comparison of 57 EFPY peak fluence values with the threshold values from Reference 5.2.4 is shown in Table 3-5. Table 3-5 supports the conclusion that neutron embrittlement is not a concern for the NPM lower RPV made of austenitic stainless steel.

Figure 3-5 Comparison of FXM-19 Fracture Toughness with Type 3XX Austenitic Stainless Steel Fracture Toughness

{{

}}2(a),(c),ECI

Table 3-5 Comparison of Design Life Peak Fluence with Threshold Fluence

{{

}}2(a),(c),ECI

3.4.1.2 Thermal Embrittlement Evaluation

According to Reference 5.2.3, wrought austenitic stainless steels are not subject to thermal embrittlement at PWR operating temperatures. However, CASS and austenitic stainless steel weld filler metals are potentially susceptible to thermal embrittlement because they contain some residual delta ferrite. The NPM lower RPV does not use CASS but does contain austenitic stainless steel weld filler metal (E/ER209 or E/ER240).

Based on the thermal embrittlement data for Type 3XX austenitic stainless steel, Table 3-2 of Reference 5.2.3 lists criteria to help evaluate when there is a potential for synergistic effects between thermal embrittlement and neutron embrittlement of CASS and austenitic stainless steel weld filler metal. The criteria are based on molybdenum content, which is known to increase thermal embrittlement of austenitic stainless steel welds; delta ferrite content, which is known to increase thermal embrittlement of austenitic stainless steel welds; and neutron dose. The following are the criteria for austenitic stainless steel weld filler metal where thermal embrittlement is of concern.

1. end of life neutron dose greater than 0.5 dpa with any molybdenum and delta ferrite content
2. molybdenum content less than 0.50 percent along with greater than 20 percent delta ferrite
3. molybdenum content greater than 0.50 percent along with greater than 14 percent delta ferrite

The 57 EFPY peak life fluence for the NPM lower RPV is well below the end of life neutron dose criterion of 0.5 dpa, so the first criterion is not met.

Welds using E/ER209 filler metal contain 1.5 percent to 3.0 percent molybdenum, and welds using E/ER240 filler metal contain up to 0.75 percent molybdenum, which corresponds to item three above. Based on the revised DeLong diagram in Reference 5.2.12, 14 percent delta ferrite is approximately equivalent to 16 ferrite number (FN). Therefore, the NPM design limits the delta ferrite in the lower RPV weld filler metal (E/ER209 or E/ER240) to 16 FN in order to avoid synergistic effects between thermal embrittlement and neutron embrittlement in the weld filler metal.

3.4.1.3 Material Surveillance Program Evaluation

Austenitic stainless steels do not undergo ductile-to-brittle transition temperature and have higher toughness than ferritic materials used for ASME BPVC Section III Class 1 pressure-retaining components. These factors are why Reference 5.1.9, NB-2311, does not have impact test requirements for austenitic stainless steels, including SA-965 Grade FXM-19 used in the NPM lower RPV.

The requirements for establishing a material surveillance program rely on the nil-ductility reference temperature (RT_{NDT}) calculation in Reference 5.1.9, NB-2331. However, Reference 5.1.9, NB-2331, is not applicable to austenitic stainless steels because there are no impact test requirements. The drop-weight test required by Reference 5.1.9, NB-2331, to establish RT_{NDT} is limited to ferritic materials. Because RT_{NDT} cannot be calculated for the SA-965 Grade FXM-19 lower RPV, the requirements in Reference 5.1.1, Reference 5.1.3, and Reference 5.1.4 do not apply to the NPM lower RPV. The NRC endorsed the ASME BPVC in 10 CFR 50.55a.

The US600 (Reference 5.2.14) design lower RPV is designed with SA-508 Grade 3 Class 1 ferritic steel and compatible weld filler metals. Since SA-965 Grade FXM-19 austenitic stainless steel and E/ER209 or E/ER240 weld filler metals have superior ductility and are less susceptible to the effects of neutron and thermal embrittlement than SA-508 Grade 3 Class 1 ferritic steel and its compatible weld filler metals, the probability of rapid failure in the NPM lower RPV is even lower than that of the lower RPV of the approved US600 design. In addition, there have been no reports of rapidly propagating failures of ferritic steel RPVs in LWRs actors worldwide, after accumulating over 16,000 reactor-years of operation as of June 2020 (Reference 5.2.13).

3.4.2 10 CFR 50.61

10 CFR 50.61 (Reference 5.1.2) requires protection against PTS events. Pressurized thermal shock events are events or transients in PWRs causing severe overcooling (thermal shock) concurrent with or followed by significant pressure within the RPV.

The PTS screening criterion (RT_{PTS}) calculation uses the acceptance criteria in 10 CFR 50.61(b)(2). The PTS screening methodology in Reference 5.1.2 is based on calculating RT_{PTS}, which is the RT_{NDT} evaluated for end of design life peak fluence for each of the RPV beltline materials using the Reference 5.1.2 procedures. The Reference 5.1.2 procedures require the use of $RT_{NDT(u)}$, which is the unirradiated reference temperature established by impact testing according to Reference 5.1.9, NB-2331. The NRC endorsed the ASME BPVC in 10 CFR 50.55a. Impact testing is not required for austenitic stainless steels because they do not undergo ductile-to-brittle transition temperature and have higher toughness than ferritic materials. Because RT_{NDT} cannot be calculated for the austenitic stainless steels in the lower RPV, the requirements in Reference 5.1.2 do not apply to the NPM lower RPV. Furthermore, the chemistry factors in Reference 5.1.2 assume use of carbon or low-alloy steel because Equation 4 of Reference 5.1.2 uses a chemistry factor that corresponds to copper and nickel content of ferritic materials. There is no chemistry factor for austenitic stainless steel in Reference 5.1.2.

As noted in Section 3.4.1, the beltline of the NPM lower RPV is not made of ferritic steel; therefore, the region of the RPV containing ferritic materials that experiences the highest fluence is the top surface of the upper RPV lower flange. The upper RPV 57 EFPY peak fluence is {{ }}[2(a),(c),ECI], E > 1 MeV. Consequently, the NPM upper RPV does not require PTS screening since the design life peak fluence is less than 1E+17 n/cm^2, E > 1 MeV.

4.0 Summary and Conclusions

The US460 standard design meets the requirements in GDC 14, GDC 15, GDC 31, and GDC 32. While the requirements in 10 CFR 50.60 and 10 CFR 50.61 cannot be used for the NPM lower RPV because it is made of austenitic stainless steel, the design satisfies the requirements of the GDCs.

The US460 standard design meets GDC 14 by using austenitic stainless steel in the RPV beltline, which has superior ductility and is less susceptible to the effects of neutron and thermal embrittlement than ferritic materials, which increases the integrity and safety of the RCPB.

The US460 standard design meets GDC 15 by using austenitic stainless steel in the RPV beltline, which has superior ductility and is less susceptible to the effects of neutron and thermal embrittlement than ferritic materials, which increases the integrity and safety of the RCPB. The US460 standard design ensures that the RCPB limits are not exceeded during operation.

The US460 standard design meets GDC 31 by using austenitic stainless steel in the RPV beltline, which has superior ductility and is less susceptible to the effects of neutron and thermal embrittlement than ferritic materials, which increases the integrity and safety of the RCPB.

The US460 design meets item (2) of GDC 32 by using austenitic stainless steel in the RPV beltline, which has superior ductility and is less susceptible to the effects of neutron and thermal embrittlement than ferritic materials, which increases the integrity and safety of the RCPB. Item 2 of GDC 32 requires an appropriate material surveillance program. An RVSP is not necessary to ensure the safety of the US460 standard design because the austenitic stainless steel used in the lower RPV is less susceptible to the effects of neutron and thermal embrittlement compared to ferritic materials. Therefore, the design satisfies Item 2 of GDC 32 without an RVSP.

The US600 design (Reference 5.2.14) used austenitic stainless steel for the lower containment vessel (CNV) because its material properties are less susceptible to the effects of neutron and thermal embrittlement than ferritic materials. In Section 6.1.1.4.2 of the US600 design final safety evaluation report (Reference 5.2.15), the NRC stated:

> *The staff finds the use of SA-965, Grade FXM-19, and its associated weld filler metals acceptable for use in the lower portion of the CNV, as the calculated fluence to the CNV is lower than what is expected to cause embrittlement, and the selection of SA-965, Grade FXM-19, an austenitic stainless steel, is resistant to radiation embrittlement.*

In Section 6.2.7.4 of the final safety evaluation report (Reference 5.2.15), the NRC stated:

> *Within the ASME Code, detailed fracture toughness requirements are placed on ferritic materials, as nonferritic materials exhibit sufficient inherent fracture toughness that additional requirements are deemed unnecessary. For example, the austenitic*

stainless steel used for the CNV lower shell, SA-965, FXM-19, was explicitly chosen specifically for its superior fracture toughness and resistance to neutron embrittlement.

The results of this report confirm that austenitic stainless steels and compatible weld filler metals are likewise acceptable for use in the lower RPV without additional fracture toughness requirements because they have superior ductility and are less susceptible to the effects of neutron and thermal embrittlement than ferritic materials.

5.0 References

5.1 Source Documents

5.1.1 U.S. Code of Federal Regulations, "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation," Section 50.60, Part 50, Chapter I, Title 10, "Energy," (10 CFR 50.60).

5.1.2 *U.S. Code of Federal Regulations*, Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events," Section 50.61, Part 50, Chapter I, Title 10, Energy," (10 CFR 50.61).

5.1.3 *U.S. Code of Federal Regulations*, Fracture Toughness Requirements," Appendix G, Part 50, Chapter I, Title 10, Energy," (10 CFR 50, Appendix G).

5.1.4 *U.S. Code of Federal Regulations*, Reactor Vessel Material Surveillance Program Requirements," Appendix H, Part 50, Chapter I, Title 10, Energy," (10 CFR 50, Appendix H).

5.1.5 *U.S. Code of Federal Regulations*, Reactor Coolant Pressure Boundary," Criterion 14, Appendix A, Part 50, Chapter I, Title 10, Energy," (10 CFR 50, Appendix A).

5.1.6 *U.S. Code of Federal Regulations*, Reactor Coolant System Design," Criterion 15, Appendix A, Part 50, Chapter I, Title 10, Energy," (10 CFR 50, Appendix A).

5.1.7 *U.S. Code of Federal Regulations*, Fracture Prevention of Reactor Coolant Pressure Boundary," Criterion 31, Appendix A, Part 50, Chapter I, Title 10, Energy," (10 CFR 50, Appendix A).

5.1.8 *U.S. Code of Federal Regulations*, Inspection of Reactor Coolant Pressure Boundary," Criterion 32, Appendix A, Part 50, Chapter I, Title 10, Energy," (10 CFR 50, Appendix A).

5.1.9 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2017 Edition, Section III, Subsection NB, "Rules for Construction of Nuclear Facility Components," New York, NY.

5.2 Referenced Documents

5.2.1 Pokrovsky, A.S., et al., Effect of Neutron Irradiation on Tensile Properties of Austenitic Steel XM-19 for the ITER Application," *Journal of Nuclear Materials* (2011): 417:874-877.

5.2.2 Idaho National Laboratory INL/EXT-20-58432, Revision 1, "Irradiation and PIE of Alloys X-750 and XM-19 (EPRI Phase III)," Jackson, J.H., et al., July 2020.

5.2.3 Electric Power Research Institute MRP-175, “Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values,” Palo Alto, CA, 2005: 1012081.

5.2.4 U.S. Nuclear Regulatory Commission, “Degradation of LWR Core Internal Materials due to Neutron Irradiation,” NUREG/CR-7027, December 2010.

5.2.5 Characteristics of PM-1 (Sundance),” *Nucleonics*, 1962. (from Wikipedia)

5.2.6 Karnoski, Jr., P.J., et al., Stainless Steel Reactor Pressure Vessels,” *Nuclear Engineering and Design*, (1970): Volume 11, Issue 3: 347-167.

5.2.7 Adams Atomic Engines, Inc., “PM-3A Design and Construction,” October 1996. (from Wikipedia)

5.2.8 Floating Nuclear Plant Sturgis Dismantled,” *The Maritime Executive*, March 16, 2019. (from Wikipedia)

5.2.9 Idaho National Laboratory, “Advanced Test Reactor User Guide.”

5.2.10 Phillips Petroleum Company, “Reactor Vessel,” Safety Analysis Report, Advanced Test Reactor, Volume 1 of 2, IDO-17021, April 1965: 93-108.

5.2.11 Andresen, P., et al., SCC and Fracture Toughness of XM-19,” *Proceedings of the 18th International Conference on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors*.

5.2.12 Long, C.J. and DeLong, W.T., The Ferrite Content of Austenitic Stainless Steel Weld Metal,” *Welding Research Supplement, Welding Journal*, July 1973.

5.2.13 International Atomic Energy Agency, “Operating Experience with Nuclear Power Stations in Member States (2020 Edition),” Vienna, Austria, August 2020.

5.2.14 NuScale Power, LLC, NuScale Standard Plant Design Certification Application, Rev. 5, July 29, 2020 (ML20225A044), Portland, OR.

5.2.15 U.S. Nuclear Regulatory Commission, “NuScale Design Certification Final Safety Evaluation Report, FSER Chapter 6 - Engineered Safety Features,” July 23, 2020 (ML20205L406).

Enclosure 3:

Affidavit of Carrie Fosaaen, AF-132199

NuScale Power, LLC

AFFIDAVIT of Carrie Fosaaen

I, Carrie Fosaaen, state as follows:

(1) I am the Senior Director of Regulatory Affairs of NuScale Power, LLC (NuScale), and as such, I have been specifically delegated the function of reviewing the information described in this Affidavit that NuScale seeks to have withheld from public disclosure, and am authorized to apply for its withholding on behalf of NuScale

(2) I am knowledgeable of the criteria and procedures used by NuScale in designating information as a trade secret, privileged, or as confidential commercial or financial information. This request to withhold information from public disclosure is driven by one or more of the following:

(a) The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by NuScale competitors, without a license from NuScale, would constitute a competitive economic disadvantage to NuScale.

(b) The information requested to be withheld consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage, as described more fully in paragraph 3 of this Affidavit.

(c) Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

(d) The information requested to be withheld reveals cost or price information, production capabilities, budget levels, or commercial strategies of NuScale.

(e) The information requested to be withheld consists of patentable ideas.

(3) Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale's competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying report reveals distinguishing aspects about the system by which NuScale develops its Reactor Coolant System and Connecting Systems.

NuScale has performed significant research and evaluation to develop a basis for this system and has invested significant resources, including the expenditure of a considerable sum of money.

The precise financial value of the information is difficult to quantify, but it is a key element of the design basis for a NuScale plant and, therefore, has substantial value to NuScale.

If the information were disclosed to the public, NuScale's competitors would have access to the information without purchasing the right to use it or having been required to undertake a similar expenditure of resources. Such disclosure would constitute a misappropriation of NuScale's intellectual property, and would deprive NuScale of the opportunity to exercise its competitive advantage to seek an adequate return on its investment.

(4) The information sought to be withheld is in the enclosed report entitled Reactor Coolant System and Connecting Systems. The enclosure contains the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.

(5) The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC §

552(b)(4), as well as exemptions applicable to the NRC under 10 CFR §§ 2.390(a)(4) and 9.17(a)(4).

(6) Pursuant to the provisions set forth in 10 CFR § 2.390(b)(4), the following is provided for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld:

(a) The information sought to be withheld is owned and has been held in confidence by NuScale.

(b) The information is of a sort customarily held in confidence by NuScale and, to the best of my knowledge and belief, consistently has been held in confidence by NuScale. The procedure for approval of external release of such information typically requires review by the staff manager, project manager, chief technology officer or other equivalent authority, or the manager of the cognizant marketing function (or his delegate), for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside NuScale are limited to regulatory bodies, customers and potential customers and their agents, suppliers, licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or contractual agreements to maintain confidentiality.

(c) The information is being transmitted to and received by the NRC in confidence.

(d) No public disclosure of the information has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or contractual agreements that provide for maintenance of the information in confidence.

(e) Public disclosure of the information is likely to cause substantial harm to the competitive position of NuScale, taking into account the value of the information to NuScale, the amount of effort and money expended by NuScale in developing the information, and the difficulty others would have in acquiring or duplicating the information. The information sought to be withheld is part of NuScale's technology that provides NuScale with a competitive advantage over other firms in the industry. NuScale has invested significant human and financial capital in developing this technology and NuScale believes it would be difficult for others to duplicate the technology without access to the information sought to be withheld.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 12/31/2022.

Carrie Fosaaen