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 6, “Engineered Safety Features,” 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 6 of the Standard Design Approval Application, “Engineered Safety Features,” 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 6. NuScale is enclosing information in this submittal that: 1) closes gaps identified between the draft SDAA Chapter 6 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 6 observations.

Enclosure 1 contains SDAA Part 2 Chapter 6, “Engineered Safety Features,” 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
Senior Director, Regulatory Affairs
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Enclosure 1: SDAA Part 2 Chapter 6, “Engineered Safety Features,” Revision 0 (proprietary)
Enclosure 2: SDAA Part 2 Chapter 6, “Engineered Safety Features,” Revision 0 (nonproprietary)
Enclosure 3: Affidavit of Carrie Fosaaen, AF-132201

Enclosure 1:

SDAA Part 2 Chapter 6, “Engineered Safety Features,” Revision 0, (proprietary)

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Enclosure 2:

SDAA Part 2 Chapter 6, “Engineered Safety Features,” Revision 0 (nonproprietary)

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Section A

NuScale US460 Plant
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CHAPTER 6 ENGINEERED SAFETY FEATURES

6.1 Engineered Safety Feature Materials

6.1.1 Metallic Materials

This section provides information on engineered safety feature (ESF) component material selection and fabrication methods and discusses compatibility with fluids that the component may be exposed to during normal, accident, maintenance, and testing conditions.

Design, fabrication, erection, and testing of ESF system components conforms to quality standards commensurate with the importance of their intended safety functions (General Design Criterion [GDC 1]).

The ESF systems include the containment system (CNTS), emergency core cooling system (ECCS), and decay heat removal system (DHRS). Section 6.3 and Section 6.2 describe details of the ECCS and CNTS, respectively. Section 5.4.3 describes the DHRS.

Material selection and fabrication methods ensure ESF component compatibility with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs). In addition, ESF pressure-retaining component materials have a low probability of abnormal leakage, rapidly propagating failure, or gross rupture (non-brittle) considering the design operational parameters (GDC 4, GDC 14, GDC 16, GDC 31, principal design criteria 35 and 41, 10 CFR 50.55a, and Appendix B to 10 CFR Part 50, Criteria IX and XIII). Chapter 3 provides further details of code and regulatory applicability.

This section also provides information on materials within the containment vessel (CNV) that are associated with non-ESF systems. Figure 6.2-3 depicts the reactor coolant system (RCS), steam generator system (SGS) and control rod drive system (CRDS) have lines inside the CNV. The materials for these systems ensure compatibility with the environmental conditions associated with normal operation and postulated accidents within the CNV, including those that expose the components to reactor coolant water chemistry.

6.1.1.1 Material Selection and Fabrication

The ESF components have a 60-year design life and use materials permitted by American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section III, Division 1. Table 6.1-1, Material Specifications for ESF Components, provides a list of the components, their material grade, and material type. The ESF pressure boundary materials, including weld materials, and associated supports conform to the fabrication, construction, and testing requirements of ASME BPVC, Sections II and III and meet the requirements of NB-2000, NC-2000 or NF-2000, as applicable. Selection of these materials ensures compatibility with the coolant system fluids and their selection is consistent with Parts A, B, and C of ASME BPVC, Section II and Appendix I to ASME BPVC, Section III.

The design, fabrication, and materials of construction of the CNV includes sufficient margin that provides reasonable assurance that the CNV pressure boundary avoids brittle fracture. The margin minimizes the probability of a rapidly propagating fracture under operating, maintenance, testing, and postulated accident conditions for the 60-year design life. Code Case N-759-2 is used for the CNV (Section 5.2).

Selection of materials used in regions of the CNV subjected to neutron irradiation ensures their resistance to neutron embrittlement over the design life of the plant. Austenitic stainless steel demonstrates good resistance to neutron embrittlement when exposed to neutron fluence levels below 1E19 n/cm^2 for energies greater than 1 MeV.

Austenitic stainless steel, which has a 60-year design life peak fluence of less than 1E19 n/cm^2 (>1MeV), comprises the core region of the CNV. The region of the CNV that is fabricated from martinsitic stainless steel has a peak fluence of less than 1E17 n/cm^2. Section 6.2.1 provides additional CNV design detail.

The fracture toughness properties of the ferritic pressure retaining ESF components and associated supports comply with the requirements of ASME BPVC, Section III, Subsections NB-2300, NC-2300 and NF-2300, as applicable.

The stainless steels and nickel-base alloys in Table 6.1-1 and Table 6.1-2 that may be exposed to reactor coolant have been used by the operating nuclear power plants. Operating plant experience shows that there is minimal degradation concern for these materials.

The lower CNV shell above the RPV flange elevation, the lower flange, upper flange, upper shell, and top head are fabricated from martensitic stainless steel. The use of F6NM is consistent with applicable regulations cited in SRP 6.1. RG 1.84, Rev. 39, which is incorporated by reference into 10 CFR 50.55a, includes Code Case N-774, which permits the use of F6NM materials for forgings in excess of 10,000 lbm. Therefore, the use of SA-336 Gr F6NM for the CNV meets the requirements of 10 CFR 50.55a. The lower shell, bottom head and associated supports are fabricated from solution annealed austenitic stainless steel forgings.

The allowable CNV materials including weld filler metals are listed in Table 6.1-1. The allowable materials for ESF valves are listed in Table 6.1-3. When welding together different material types there may be a need for a dissimilar metal weld (DMW) weld joint. When a DMW is necessary, a suitable weld material will be used that is of adequate strength and composition for the design and operating conditions. The design and fabrication of the weld meet the applicable Code sections as required.

Implementation of Regulatory Guide (RG) 1.44 guidelines minimizes the potential for stainless steel intergranular stress corrosion cracking. Before fabrication, unstabilized austenitic stainless steel of the American Iron and Steel Institute (AISI) Type 3XX series undergoes solution treatment per the guidance of RG 1.44, which describes acceptable criteria for preventing intergranular corrosion of stainless steel components. During post weld heat treatment, when austenitic stainless steel materials undergo sensitizing temperatures for greater than 60 minutes, testing in accordance with American Society for Testing and Materials (ASTM) A262, Practice A or E verifies non-sensitization of the materials. ESF components do not use furnace-sensitized austenitic stainless steel.

Delta ferrite content of stainless steel weld filler material conforms to the guidelines stipulated in ASME BPVC, Section III, Subsections NB-2433, NC-2433 or NF-2433 and RG 1.31, ensuring sufficient ferrite content to avoid microfissures in welds, offset dilution, and reduce thermal aging. The delta ferrite content in stainless steel weld metal is between ferrite number 5 and 20. Alloy 52/152/52M filler metals are used for welding Alloy 690 to provide a high level of corrosion resistance.

Information regarding pressure retaining bolting material is in Section 3.13.

Minimal cold-working of austenitic stainless steel surfaces from abrasive work, such as grinding or wire brushing, reduces the potential for stress corrosion cracking. When abrasive work occurs, control of the tools ensures that there are no ferritic carbon steel contaminants. The design avoids the use of cold worked austenitic stainless steel. If cold work occurs, the limit for yield strength as determined by the 0.2 percent offset method is 90 ksi maximum.

There is no thermal insulation (metallic or non-metallic) inside the CNV. Any insulation on the CNV is above the reactor pool level and uses reflective metallic insulation. There are no fibrous insulation materials.

The qualification of welders for making welds in areas with limited access, and the methods for monitoring and certifying such welds, are in accordance with RG 1.71.

Cleanliness controls for fabrication, pre-operational and during the operational phases meet and satisfy the applicable requirements of ASME NQA-1.

Controls established for special processes such as welding, heat treating and non-destructive testing of the CNV and ESF materials satisfy the applicable requirements of 10 CFR 50, Appendix B Criterion IX.

Controls established for the handling, storage, shipping, cleaning and preservation of CNV and ESF materials and equipment to prevent damage or deterioration meet the applicable requirements of 10 CFR 50, Appendix B Criterion XIII. Regulatory Guide 1.28 provides quality assurance criteria for cleaning fluid systems and associated components that comply with 10 CFR 50, Appendix B. The design for threaded fasteners meets the cleaning criteria in RG 1.28.

6.1.1.2 Composition and Compatibility of Core Cooling Coolants

The CNTS includes the CNV and containment isolation valves. This section addresses the compatibility of the CNV and pressure retaining portion of the containment isolation valves with their environment in supporting the ESF function and also the portions of the additional non-ESF functional systems (containment flooding and drain system, chemical and volume control system, main stream, feedwater and reactor component cooling water system) that penetrate the CNV boundary and become part of the RCS, CNTS, SGS, and CRDS inside the CNV.

The ECCS includes two reactor vent valves (RVVs) attached to the reactor vessel head and two reactor recirculation valves (RRVs) that are attached to the reactor vessel shell with associated remote solenoid trip and reset actuators (connected by hydraulic line) attached to the exterior of the CNV upper shell. The RCS or reactor pool water covers the RRVs and their hydraulic lines during periodic shutdown, cooldown, and refueling operations and during ECCS operation. The RVVs are subjected to high temperature RCS steam during ECCS actuation. The actuator assemblies for the RVVs and RRVs are normally immersed in the reactor pool. The ECCS reactor vent valves, RRVs, their actuators, and the connecting hydraulic lines are compatible with the RCS chemistry under LOCA conditions or intermittent exposure to reactor pool water. Section 6.3 provides a more detailed discussion and description of the ECCS.

The DHRS consists of two redundant trains, each including a passive condenser with piping. With the exception of some portions of the steam side piping, the DHRS is submerged in the reactor pool. The DHRS piping including the portion that penetrates the CNV boundary meets ASME Class 2 criteria. Compatibility with the secondary fluid in contact with the DHRS components and with borated water present in the RCS and the reactor pool informs the selection of the DHRS piping material internal and external to the CNV. Section 5.4 provides a more detailed discussion and description of the DHRS.

Exposure to borated RCS or reactor pool water occurs to the interior and exterior surfaces of ESF components, with the exception of the CNV head exterior, and the non-ESF piping and components within the CNV over the life of the plant. The CNV is partially immersed and DHRS condensers as well as the ECCS valve actuator assemblies are submerged in the reactor pool.

There are no socket welds on lines larger than 3/4 inch NPS for the Class 1 lines in Table 6.1-1. Socket welds used on piping less than 3/4 inch NPS conform to 10 CFR 50.55a(b)(1)(ii) and ASME B16.11. There are no socket welds on piping in Table 6.1-2, including piping of NPS 2 or less in size.

During normal power operations the interior environment of the CNV is dry, at a partial vacuum. Reactor pool water partially floods the CNV during cooldown before the movement of a NuScale Power Module for refueling operations.

Reactor coolant discharging into the CNV facilitates emergency core cooling for the NuScale Power Module. Reactor coolant chemistry is consistent with the guidance found in the Electric Power Research Institute Pressurized Water

Reactor Primary Water Chemistry Guidelines. As a result, during transients or accidents that result in reactor coolant discharge into the CNV, interior components are exposed to the same chemistry controlled coolant that is used in day- to-day operations. ESF component materials that undergo exposure to primary reactor coolant (internally or externally) are compatible with reactor coolant chemistry. The design prohibits the use of materials within the CNV that alters post-accident coolant chemistry. Section 5.2.3 contains additional information on reactor coolant water chemistry.

The materials for ESF components that are partially immersed within the reactor pool are compatible with the reactor pool chemistry conditions in the pool. Based on the purity of the reactor pool water outside of the CNV and the material selection of the CNV, there is no expectation of significant corrosion. Section 9.1.3 describes operation of the pool cleanup system that maintains reactor pool water chemistry within the expected range of values shown on Table 9.1.3-4. There is no corrosion allowance for ESF materials exposed to process fluids or reactor pool chemistry.

Piping, supports, and components associated with the containment flooding and drain system and located in the CNV interior but defined as part of the CNTS are compatible with the reactor coolant chemistry that occurs under operation of ECCS conditions. The CNTS piping, fittings, pipe supports and components are austenitic stainless steel with a carbon content not exceeding 0.03 percent to mitigate intergranular attack; if they are exposed to temperature range 800 to 1500 degree F after final solution anneal. Table 6.1-2 lists non-ESF components in the CNV.

Piping, supports, and components associated with the functional systems that communicate through the CNV boundary and defined as part of the RCS or SGS are compatible with the reactor coolant chemistry present under ECCS operation conditions. Table 6.1-2 lists the applicable sections that describe materials used for these systems.

No materials, paint, or coatings in the CNV contribute to corrosion-related hydrogen production or alter post-LOCA coolant chemistry that would enhance stress corrosion cracking of austenitic stainless steel.

6.1.2 Organic Materials

Protective coatings are not permitted on the inside or outside surface of the CNV, or on any other ESF or non-ESF system components located within the CNV.

Cabling that runs through the CNV is unpainted corrosion resistant, seamless construction, type 304L, stainless steel jacketed, mineral (silicon dioxide) insulated cabling. The cable material is free of organic material in the insulation and sheath.

Table 6.1-1: Material Specifications for ESF Components

Component	Specification	Grade/Type/Class
Containment Vessel (CNV)		
Lower Vessel Support Skirt	SA-182	F304 (Note 1)
Lower Vessel (Lower Head, Core Region Shell, Transition Shell)	SA-965	FXM-19 (Note 2)
Lower Vessel (Flange, Lower Shell); Upper Vessel (Flange, Shells, Upper Head, and Manway Nozzles)	Code Case N-774 (SA-336)	F6NM
CNV Top Head Cover; Covers for CNV Manways and Access Ports	SA-182	F6NM
CNV Safe-Ends and Fittings	SA-182	F304 and F316 (Note 1)
CNV Thermowells	SA-479	Type 304 (Note 1)
CNV Seismic Support Lugs; CNV-RPV Supports (Gussets, Plates, and Lugs)	SB-168	UNS N06690
Leak Detection Ports	SA-312	TP316L SMLS (Note 1)
CRDM Support Frame	SA-240	Type 304 (Note 1)
	SA-479	Type 304 (Note 1)
CNV Bolting		
Main Flange Closure	SB-637	UNS N07718 (Note 3)
Other Than Main Flange Closure	SA-193	Grade B8 Class 1
	SA-194	Grade 8
	SA-564	Type 630, H1100
	SB-637	UNS N07718 (Note 3)
CNV Top Support Structure		
Support	SA-182	F6NM
	SA-240	Type 304 (Note 1)
	SA-312	TP 304 (Note 1)
	SA-479	Type 304 (Note 1)
Bolting	SA-193	Grade B8 Class 1
	SA-194	Grade 8
	SA-564	Type 630, H1100
	SB-637	UNS N07718 (Note 3)
Weld Filler Metals for CNV and CNV Top Support Structure		
2XX Austenitic Stainless Steel Weld Filler Metals	SFA-5.4	E209, E240 (Note 2)
	SFA-5.9	ER209, ER240 (Note 2)
3XX Austenitic Stainless Steel Weld Filler Metals	SFA-5.4	E308, E308L, E309, E309L, E316, E316L (Note 4)
	SFA-5.9	ER308, ER308L, ER309, ER309L, ER316, ER316L (Note 4)
	SFA-5.22	E308, E308L, E309, E309L, E316, E316L (Note 4)
4XX Materialistic Stainless Steel Weld Filler Metals	SFA-5.4	E410NiMo
	SFA-5.9	ER410NiMo
Nickel-Base Alloy Weld Filler Metals	SFA-5.11	ENiCrFe-7
	SFA-5.14	ERNiCrFe-7, ERNiCRFE-7A, EQNiCrFe-7, EQNiCrFe-7A
Containment System (CNTS) Piping • CNTS Chemical and Volume Control Injection Piping Assembly • CNTS Chemical and Volume Control Discharge Piping Assembly • CNTS Pressurizer Spray Piping Assembly • CNTS Reactor Pressure Vessel High Point Degasification Piping Assembly		
Pipe	SA-312	TP304 SMLS (Note 1)

Table 6.1-1: Material Specifications for ESF Components (Continued)

Component	Specification	Grade/Type/Class
Pipe Fitting	SA-182	F304 (Note 1)
	SA-403	WP304 SMLS (Note 1)
Emergency Core Cooling System (ECCS) Piping Assembly		
Tube	SA-213	TP316 (Note 1)
Tube Fitting	SA-182	F316 (Note 1)
	SA-479	Type 316 (Note 1)

Note:

1) 0.03 percent maximum carbon if unstablized Type 3XX base metals are welded or exposed to temperature range of 800 °F to 1500 °F subsequent to final solution anneal.
2) 0.04 percent maximum carbon for FXM-19 and Type 2XX weld filler metals.
3) SB-637 UNS N07718 solution treatment temperature of range before precipitation hardening treatment restricted to "1800 °F to 1850 °F.”
4) 0.03 percent maximum carbon for unstablized 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 6.1-2: Material Specifications for Containment Vessel Related non-Engineered Safety Feature Components

Component	Material/Grade/Type
CNTS outside containment	
Piping Materials	SA-312 Gr TP304[2] SA-182 Gr 304[2] SA-479 TP316[2]/TP304L[2] SA-403 Gr WP304-S[2]
CRDS Piping Materials	SA-479 TP 316/316L SA-182, Gr F316/F304[2] SA-312 Gr TP304[2] SA-213, Gr TP316 SA-403 Gr WP304[2] SA-193 Gr B8MN Class 2 SA-194 Gr 8M ASTM A-240 UNS S41000
Steam Generator System	Table 5.2-3
Weld Filler Materials (Stainless steel to be compatible with base material)	Table 5.2-3
Alignment Pins	SA-564, Type 630, Condition H1100

Note:

1) When the material is designated as Type or Grade 304/304L, this refers to dual certified stainless steel material.
2) 0.03 percent max carbon content

Table 6.1-3: Pressure Retaining Materials for Reactor Coolant Pressure Boundary and Engineered Safety Feature Valves

Component	Specification	Grade/Type/Class
Body and Bonnet	SA-182	F304, F316, F316L (Note 1)
	SA-351	CF3, CF8 (Note 2)
	SA-479	Type 304, Type 316 (Note 1)
Ball and Disc	SA-182	F304, F316 (Note 1)
	SA-479	Type 304, Type 316 (Note 1)
	SA-564	Type 630 H1100 or H1150
	SA-638	Grade 660
	SA-637	UNS N07718
Seat	A-182 or SA-182	F304, F316 (Note 1)
	A-276 or SA-276	UNS S21800
	A-479 or SA-479	Type 304, Type 316 (Note 1) UNS S21800
	B-637 or SB-637	UNS N07718
Stem	A-276 or SA-276	UNS S21800
	A-479 or SA-479	Type 316 Strain-Hardened Level 1 or Level 2 XM-19 Annealed or Hot-Rolled or Strain-Hardened UNS S21800
	B-637 or SB-637	UNS N07718
	SA-564	Type 630 H1100 or H1150
Stud	SA-193	Grade B8 Class 1, Grade 8M Class 1
	SB-637	UNS N07718 (Note 3)
Nut	SA-194	Grade 6, Grade 8
	SB-637	UNS N07718 (Note 3)
Austenitic Stainless Steel Weld Filler Metals	SFA-5.4	E308, E308L, E309, E309L, E316, E316L (Note 4)
	SFA-5.9	ER308, ER308L, ER309, ER309L, ER316, ER316L (Note 4)

Note:

1) 0.03 percent maximum carbon if unstablized 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 percent maximum carbon, and 20 percent maximum delta ferrite for Grade CF3 and Grade CF8 austenitic stainless steel casting.
3) SB-637 UNS N07718 solution treatment temperature of range before precipitation hardening treatment restricted to "1800°F to 1850°F.”
4) 0.03 percent maximum carbon for unstablized 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.

6.2 Containment Systems

6.2.1 Containment Functional Design

The containment is an integral part of the NuScale Power Module (NPM) and provides primary containment for the reactor coolant system (RCS). The containment system (CNTS) includes the containment vessel (CNV), CNV supports, containment isolation valves (CIVs), passive containment isolation barriers, and containment instruments (Figure 6.2-1).

6.2.1.1 Containment Structure

6.2.1.1.1 Design Bases

The CNV is an evacuated pressure vessel fabricated with a combination of martensitic and austenitic stainless steel that houses, supports, and protects the reactor pressure vessel (RPV) from external hazards and provides a barrier to the release of fission products. The CNV is partially immersed in a below grade, borated-water filled, stainless steel lined, reinforced concrete pool to facilitate heat removal. The CNV is an American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) Class MC (steel) containment whose design, analysis, fabrication, inspection, testing, and stamping conform to ASME BPVC Class 1 pressure vessel requirements in accordance with Section III, Subsection NB as permitted by NCA-2134(c). Overpressure protection is provided in accordance with ASME BPVC, Section III, Article NE-7000.

The CNTS provides a barrier that accommodates, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident (LOCA) (General Design Criterion [GDC] 50). As a minimum, pressure retaining components that comprise the CNTS have a design pressure and temperature as indicated in Table 6.2-1, which bounds the calculated pressure and temperature conditions for any design-basis event (DBE). In concert with the CIVs and passive containment isolation barriers discussed in Section 6.2.4, the CNV serves as a final barrier to the release of radioactivity and radiological contaminants to the environment (GDC 16).

The supporting analyses methodology is in the Loss-of-Coolant Accident Evaluation Model (Reference 6.2-1). Limiting analysis results are summarized in Section 6.2.1.1.3.

Section 6.2.5 describes combustible gas control for the CNV. The CNV design includes no internal subcompartments, which eliminates the potential for localized collection of combustible gases and differential pressures resulting from postulated primary release events within containment.

The structural and pressure retaining components of the CNV consist of the closure flanges and bolting, vessel shells, vessel top and bottom heads, nozzles and penetrations for piping and instrumentation, access and inspection ports, CNV support skirt, CNV support lugs, bolting for the RPV upper support ledge, CRDM support frame, emergency core cooling system (ECCS) trip and reset valve assemblies, and the NPM top support structure mounting assemblies. Section 3.8.2 provides additional design detail that includes a physical description of the geometry of the CNV and supports, plan views, and design criteria relating to construction techniques, static loads, and seismic loads. Table 6.1-1 describes the materials used for fabrication of the CNV and associated components.

Instrumentation monitors containment parameters for normal operation, anticipated operational occurrences, and accidents to include temperature, pressure, isolation valve position, and liquid level (GDC 13 and 64). Section 7.1 discusses the monitored containment parameters.

The integrated design of the RPV and CNV ensures that RCS leakage collects within the CNV. In the event of primary system releases (e.g., LOCAs or valve opening events), the CNV retains adequate reactor coolant inventory to prevent core uncovery or loss of core cooling. Section 6.3 describes that the reactor coolant water collecting in the CNV returns passively to the reactor vessel by natural circulation via the ECCS.

Under these conditions, the CNV transfers the sensible and core decay heat through its walls to the ultimate heat sink (UHS) and provides effective passive, natural circulation emergency core cooling flow. The containment reduces CNV pressure and temperature and maintains them at acceptably low levels following postulated mass and energy releases, including LOCA, into containment (principal design criterion [PDC] 38, Section 3.1.4). In postulated events, containment pressure reduces to less than 50 percent of the peak calculated pressure in less than 24 hours. Section 6.2.2 describes the containment heat removal function.

The containment and associated systems are designed to establish an essentially leak-tight barrier against an uncontrolled release of radioactivity to the environment and ensure that conditions important to safety are not exceeded for as long as the postulated accident conditions require (GDC 16).

The CNV is designed to accommodate the effects of and be compatible with the internal and external environmental conditions associated with normal operation, maintenance, testing, and postulated accidents and to protect against the dynamic effects that may result from external events (GDC 4). A discussion of these design considerations is provided in Chapter 3.

Chapter 3 includes additional information discussing protection from natural phenomena, environmental and dynamic effects, effects of missile impact, sharing of safety related structures, testing and inspection, adequacy of mechanical components and environmental qualification associated with the CNV functional design-basis considerations.

Chapter 15 provides information discussing offsite and control room dose consequences associated with the containment functional design, and

Chapter 19 provides a shutdown risk assessment, including containment analysis.

The evaluation methodology used to determine CNV peak pressure and peak temperature is in Reference 6.2-1. Table 6.2-2 presents the results of the base case and limiting CNV pressure and wall temperature analyses for primary release (LOCA and valve opening events), and limiting secondary system break scenarios.

Regarding 10 CFR 50.34(f)(3)(iv), the CNV does not include one or more dedicated containment penetrations, equivalent in size to a single 3-foot diameter opening, to accommodate future installation of systems to prevent containment failure. As discussed in this section, the calculated peak containment pressures for DBEs remain less than the CNV internal design pressure. As discussed in Section 19.2.3, peak containment pressures do not challenge vessel integrity for any analyzed severe accident progression. Therefore, 10 CFR 50.34(f)(3)(iv) is not technically relevant to the NuScale Power Plant design.

6.2.1.1.2 Design Features

The CNTS includes the CNV, top support structure, CNV supports, control rod drive mechanism (CRDM) support, CIVs, and containment instruments.

The CNTS design features support

- enclosure of the RPV, RCS, and associated components.
- containment of fission product releases from the reactor coolant pressure boundary (RCPB).
- containment of the postulated mass and energy releases (LOCA and non-LOCA) inside containment.
- operation of the ECCS by the retention of reactor coolant and the transfer of sensible and core heat to the UHS.

The CNV is a compact, steel pressure vessel that consists of an upright cylinder with top and bottom head closures. A below-grade reactor pool partially immerses the CNV, which provides a passive heat sink and has no internal sumps or subcompartments that entrap water or gases. The CNV and the reactor pool are within the Reactor Building (RXB).

The RXB provides vertical and lateral support for the CNV via a support skirt at the bottom of the vessel. The reactor bay pool walls provide lateral support for the CNV through the lateral support lugs on the upper CNV shell. The CNV houses and supports the RPV and associated piping systems and valves.

Table 6.2-1 provides a list of design and operating parameters relevant to the CNV. Figure 6.2-1 depicts the containment general arrangement.

During normal operation, the CNV is in a partially evacuated, dry condition. However, there are specific operational conditions that involve the presence of water in the CNV (e.g., primary and secondary system leakage, ECCS actuation, component cooling system leakage, or module disassembly and refueling).

Maintaining the containment at a vacuum has benefits for both normal operation and post mass and energy release events. A vacuum precludes the need for thermal insulation inside containment because there is minimal convective heat transfer from the reactor vessel during normal reactor operation. The containment evacuation system (CES) can detect leakage when the CNV vacuum is below saturation pressure of the CNV wall temperature.

The NPM moves via the reactor building crane to and from the refueling area without loss of reactor coolant inventory, and is refueled in a partially flooded condition, precluding operation with reduced inventory conditions. Section 9.2.5 discusses core decay heat removal during this process.

In the event of a mass and energy release into CNV, a process of condensation and retention within the CNV facilitates the transfer of the energy to the UHS.

Reactor coolant released from the RPV, from a chemical and volume control system (CVCS) pipe break, reactor safety valves (RSVs), or through the ECCS valves, and main steam (MS) or feedwater (FW) pipe break released from the secondary system condenses on the relatively cool inner surface of the CNV wall. The resulting condensate flows down the inner CNV wall and collects in the bottom of the CNV shell. The vapor condensation and heat removal from containment is accomplished passively by transferring the energy through the CNV wall to the reactor pool.

For releases from the RPV with ECCS actuation, when RPV and CNV pressures approach equilibrium and the accumulated level in the CNV shell reaches a level where sufficient driving head is available, coolant flow from the CNV returns to the RPV through the ECCS recirculation valves for core cooling. Opening of the reactor vent valves (RVVs) and reactor recirculation valves (RRVs) establishes the CNV shell as the outer boundary of the coolant circulation flow path. Section 6.2.2 describes this method of passive coolant circulation and heat removal.

The CNTS design provides for the isolation of process systems that penetrate the CNV. The design allows for the normal or emergency passage of fluids, vapor or gases through the containment boundary while preserving the ability of the boundary to prevent or limit the escape of fission products in postulated events. Section 6.2.4 describes the CIVs.

The design of the CNV components and appurtenances ensure pressure boundary integrity for the life of the plant when considering fatigue, corrosion and wear. The CNV pressure boundary components include the CNV, penetration assemblies or appurtenances attached to the CNV, piping, and valves attached to the CNV or to penetration assemblies out to and including the pressure boundary materials of valves required to isolate the system and provide a pressure boundary for the containment function. The CNV components and penetrations (piping, electrical and instrumentation and controls (I&C)) are designed and tested to harsh environment conditions (temperature, pressure, radiation, and submergence). Chapter 3 contains additional component design detail.

The CNV inspection and testing of its appurtenances ensure maintenance of leak tightness and functional capability under calculated design-basis conditions. The CNV supports the leakage testing requirements of 10 CFR 50 Appendix J, with the exception of Type A testing as discussed in Section 3.1.5, Section 6.2.6, and Technical Report TR-123952. Access allows for inspections, testing and maintenance of components contained within the CNTS that are within the scope of the inservice inspection (ISI) program and the inservice testing (IST) program during the life of the plant.

The CNV design allows for the inspection, testing, and maintenance of equipment and structural features inside containment (control rod drives, ECCS valves (including their venturis), RSVs, pressurizer heaters, instruments, electrical connections, welds, supports, and piping inside containment).

The CNTS components, and their associated supports, facilitate the ASME BPVC, Section XI (Reference 6.2-5) inspection requirements for Class 1, Class 2, and Class MC, including the preservice inspection requirements.

Section 3.8.2 provides the CNV design information addressing structural loads and loading combinations.

6.2.1.1.3 Design Evaluation

Design specific analyses demonstrate the functional capabilities of the containment to provide

- a leak-tight barrier that can withstand worst-case accident conditions for the duration of postulated accidents.
- a heat removal capability sufficient to maintain peak calculated containment pressures and temperatures following a postulated mass and energy release inside containment to less than design values and without exceeding the containment's design leak rate.
- sufficient heat removal capability to rapidly reduce containment pressure following postulated mass and energy releases to less than 50 percent of the peak calculated pressure for the worst-case event within 24 hours and maintain at acceptably low levels.
- the capability to withstand the maximum expected external pressure.
- the capability to preclude combustible gas mixtures.

- instrumentation capable of operating in a post-accident environment to monitor the containment such that automatic actions can be monitored or the appropriate manual action(s) can be taken.

Section 3.8.2 addresses the applicable codes, standards, and guides that apply to the containment design. Table 6.2-1 lists the CNV design and operating parameters, and a general arrangement drawing of the CNV is provided by Figure 6.2-1. Section 3.6 discusses consideration in the structural design of the dynamic effects of mass and energy releases into containment resulting from primary system release events are considered in the structural design.

Evaluation of the pressure and temperature response of the CNV is accomplished through the analysis of a variety of primary system and secondary system release events that bound all LOCA events, other primary system release events, and secondary system pipe break events. The postulated primary and secondary release events that are considered include the following:

- pipe breaks (LOCAs)
- RSVs opening
- inadvertent ECCS (RVV or RRV) valve opening
- control rod drive housing failure
- inadvertent ECCS actuation
- secondary system pipe breaks inside containment with postulated loss of augmented direct current power system (EDAS)

The peak containment pressure and temperature resulting from a mass and energy release in containment depends on the nature, size, and location of the postulated breach. The CNV contains the energy discharged from a worst-case event.

The NRELAP5 code, based on RELAP5-3D with the application of conservative initial and boundary conditions that reflect the design, predicts the mass and energy releases and the bounding CNV pressures and temperatures. The evaluation methodology addresses applicable regulatory guidance contained in Design Specific Review Standard Section 6.2.1 as discussed by Reference 6.2-1. The results are within the design pressure and design temperature of the CNV. Figure 6.2-7 through Figure 6.2-12 shows graphical results for the limiting CNV pressure, temperature, and mass and energy release rates. Table 6.2-6 provides the sequence of events for the CNV peak pressure and peak temperature cases, respectively.

The mass and energy release and containment pressure and temperature response methodology analyzes a spectrum of possible break locations, sizes, and types of mass and energy releases. The spectrum of mass and energy release events analyzed to determine the limiting results for the design includes the following:

- RCS discharge line break
- RCS injection line break
- RPV high point vent degasification supply line break
- inadvertent opening of a single RVV or RRV
- inadvertent ECCS actuation
- steam line break with a postulated loss of EDAS supply
- feedwater line break (FWLB) with a postulated loss of EDAS supply

The above spectrum of postulated release events bound the primary and secondary release events for the NPM.

Reference 6.2-1 describes the selection process used to determine initial conditions and boundary condition assumptions reflecting the design that are used for evaluation of containment response to postulated primary and secondary system mass and energy releases into containment. These initial conditions and assumptions are based on the range of normal operating conditions with consideration given to maximizing the calculated peak containment pressure and temperature.

Reference 6.2-1 describes analysis of each mass and energy release event, including consideration of the worst-case single active failure as identified by sensitivity cases and a determination of how the availability of normal alternating current (AC) and EDAS power affects the results.

The overall limiting peak calculated containment pressure and temperature, based on the mass and energy release spectrum analyses, occurs as the result of the RCS discharge line break event. The analysis models an expansion of the RCS fluid into the CNV volume and includes all relevant energy input from RCS, secondary, and fuel stored energy sources, along with conservatively modeled core power and decay heat. The limiting containment peak pressure is more than ten percent below the design pressure. Additional assumptions applied in the limiting peak containment pressure case include

- loss of normal AC power at event initiation.
- EDAS power supply is available.
- no single active failure.
- ECCS actuation on low RPV riser level, biased to high end of range.

The limiting peak containment temperature case also assumes that normal AC power is lost at event initiation; EDAS power supply is available, and.

- no signal active failure.
- ECCS actuation on low RPV riser level is biased to high end of the range.

Table 6.2-2 lists the peak calculated pressure and temperature for each event. Figure 6.2-7 through Figure 6.2-12 depict graphical results for the limiting CNV pressure, temperature and mass and energy releases. Table 6.2-6 provides the sequence of events for the CNV peak pressure and peak temperature cases, respectively. The CNV design pressure and temperature provide margin to the peak calculated temperature and pressure for the limiting event.

A double-ended steam line break inside containment results in the peak calculated containment pressure and temperature for secondary side release events. Table 6.2-2 lists the peak calculated pressures and temperatures.

The primary system mass and energy release events bound the secondary system mass and energy release event results.

The basis for the CNV external design pressure listed in Table 6.2-1 is an internal pressure of 0 psia and an external pressure resulting from pool water hydrostatic pressure for the normal operating pool level in Table 9.2.5-1.

Section 3.11 discusses the environmental qualification of mechanical and electrical equipment exposed to the containment environment following a primary or secondary system mass and energy release inside containment.

Chapter 7 and Section 3.11 discuss the CNV instrumentation that monitors and records the required containment parameters and the capability to operate in post-accident environments.

6.2.1.2 Containment Subcompartments

A subcompartment design-basis and supporting analysis for mass and energy release is not relevant to the CNV because the CNV has no interior subcompartments.

6.2.1.3 Mass and Energy Release Analyses for Primary System Release Events

The containment receives the primary system mass and energy released following a postulated rupture of piping containing reactor coolant or opening of an ECCS valve or RSV. The CNV response analysis methodology (Reference 6.2-1) is an extension of the NuScale LOCA evaluation model that was developed in accordance with the guidance of Regulatory Guide (RG) 1.203, "Transient and Accident Analysis Methods," December 2005. Reference 6.2-1 discusses development of the containment response methodology from the LOCA evaluation model.

The containment response analysis methodology provides conservative modeling of the heat transfer to and from the CNV inside diameter, and from the CNV outside diameter to the reactor pool, to ensure a bounding peak CNV pressure and temperature response following a LOCA. The methodology includes the following elements:

- high initial CNV pressure, maximum non-condensable gas concentration within the CNV, and maximum non-condensable gas release from the RPV to the CNV during a LOCA or RRV opening event
- heat transfer from RPV outside diameter including convection and boiling heat transfer to the fluid in the CNV
- condensation on CNV inside diameter including the effects of non-condensable gas
- conservative reactor pool level
- conservative reactor pool temperature
- conservative low CNV free volume assumption, which accounts for RCS thermal expansion and includes an allowance for piping, valves, cabling and miscellaneous components, such as platforms and ladders

Reference 6.2-1 shows the heat transfer correlations and models for of the processes that could impact the CNV peak pressure and temperature response.

The CNV modeling in the LOCA containment response analysis model starts with the LOCA evaluation model (Reference 6.2-1), with assumptions that maximize the mass and energy release and consequential containment pressure and temperature. The CNV and reactor pool models for secondary system pipe break containment response analysis methodology match the modeling for primary release events.

Reference 6.2-1 provides simplified diagrams of the nodalization used in the containment response analysis methodology.

The LOCA evaluation model (Reference 6.2-1) divided the NuScale Power Module LOCA scenarios into two phases for phenomena identification: LOCA blowdown phase (Phase 1a) and ECCS actuation (Phase 1b).

For primary system mass release events, the blowdown phase begins at break initiation or valve opening. Reactor coolant released into the containment volume pressurizes the containment volume and depressurizes the RPV. Pressurization of the containment and the decreased inventory within the RPV results in reactor trip and closure of the CIVs. The blowdown phase ends when the ECCS actuates the RVVs and the RRVs.

The ECCS actuation occurs as a result of a module protection system (MPS) signal as discussed in Chapter 7 or a postulated loss of EDAS power.

The RRVs open under the following conditions.

- If the pressure differential across the RRVs is greater than the inadvertent actuation block (IAB) threshold when the ECCS signal actuates, then the

RRVs stay closed until the pressure differential decreases to below the IAB release pressure.

- If the pressure differential across the RRVs decreases to below the IAB threshold pressure when the ECCS signal actuates, then the RRVs open at that time.
- If the pressure differential across the valves is less than the valve opening spring force (approximately 15 psid), then the valves open even without an ECCS actuation signal.

The RVVs open immediately upon removal of power to the ECCS valve actuator solenoid valves. The RVVs do not have an IAB.

Opening of the RVVs increases the depressurization rate, and the primary system and CNV pressures approach equalization. As the pressures equalize, the break and valve flow decreases. With pressure equalization and the increase in the CNV pool level, flow through the RRVs into the reactor vessel starts to provide long-term cooling (LTC) via recirculation. Pressure equalization terminates the reactor vessel level decrease before core uncovery. Heat transfer to the CNV wall and to the reactor pool eventually exceeds the energy addition from the break flow and the RVV flow. When this occurs, it completes the period of peak containment pressure and temperature, and a gradual depressurization and cooling phase begins.

Sensitivity cases determine the effect of loss of power (AC or DC) scenarios, as well as postulated single failures, on the primary system mass and energy release scenarios considered by the containment response analysis methodology. Reference 6.2-1 discusses insights from the results of the sensitivity cases; these insights determine the limiting cases for CNV pressure and temperature.

6.2.1.3.1 Mass and Energy Release Data - Primary System Release Events

Conservatively modeling the mass and energy release and minimizing the performance of the containment heat removal function of containment determines the maximum containment peak pressure and peak temperature scenarios.

Reference 6.2-1 provides representative results of analyses of the spectrum of the primary system mass and energy release scenarios for the NPM. Applying the Reference 6.2-1 methodology, the limiting primary system release event CNV pressure, temperature, and mass and energy release rates are determined and are depicted by Figure 6.2-7 through Figure 6.2-12. The limiting peak pressure and temperature results are below the CNV design pressure and temperature.

6.2.1.3.2 Energy Sources - Primary System Release Events

The containment response analysis methodology (Reference 6.2-1) models available energy sources identified by 10 CFR Part 50, Appendix K, paragraph I.A, with the exception of energy associated with fuel clad metal-water reaction, because calculated cladding temperatures for design-basis LOCAs remain below the threshold for cladding oxidation. Energy sources addressed in the containment response analysis analyses include

- core power initialized at 102 percent of rated thermal power.
- decay heat modeled using the 1979 ANS standard decay heat model with a 1.2 multiplier.
- RCS stored energy based on conservative initial conditions of pressure, average RCS temperature, and pressurizer level that consider the normal operating range including instrumentation uncertainties and deadband.
- stored energy in vessel internal structures.
- RCS piping inside containment.
- stored fuel energy.
- stored secondary energy (steam generator [SG] tubes, MS, and FW piping inside containment) based on conservative initial conditions of steam pressure and FW temperature that consider the normal operating range including instrumentation uncertainties and deadband.

6.2.1.3.3 Description of the Blowdown Model - Primary System Release Events

The previous sections describe the maximum containment peak pressure and temperature scenarios that result from modeling the mass and energy release and minimizing the heat removal of containment.

Engineering information, drawings, and associated reference documents inform development of a thermal-hydraulic simulation model that calculates the mass and energy released from the RCS during blowdown.

The containment response analysis methodology (Reference 6.2-1) assumes an initial power level of 1.02 times the licensed power level. The initial RCS volume and mass are consistent with that power level.

The mass and energy release determined by the containment response analysis methodology is from the NRELAP5 computer code, and the modeling approach is very similar to the LOCA evaluation model that complies with the applicable portions of 10 CFR 50 Appendix K. Reference 6.2-1 describes specific changes to the LOCA evaluation model required to model primary system mass release events. The model applies a discharge coefficient of 1.0 to the applicable critical flow correlation. Reference 6.2-1 demonstrates the adequacy of the LOCA evaluation model two-phase and single-phase choked and unchoked flow models for predictions of mass and energy release based on assessments of comparisons of NRELAP5 mass flow predictions to experimental data.

The LOCA evaluation model report demonstrates these correlations are applicable to the NPM design (Reference 6.2-1).

6.2.1.3.4 Description of the Emergency Core Cooling System Actuation Model

The containment response analysis methodology (Reference 6.2-1) models the applicable phenomena that contribute to maximizing the mass and energy release into containment and the resulting pressure and temperature during the ECCS recirculation phase.

The methodology applied during the ECCS actuation phase is the same as previously described for the blowdown phase.

Section 6.3 further discusses operation of the ECCS.

6.2.1.3.5 Description of the Long-Term Cooling Model

The containment response analyses demonstrate that the CNV pressure and temperature rapidly reduce and remain at acceptably low levels following postulated mass and energy releases, including LOCA, into containment (PDC 38 [Section 3.1.4]). The previously described methodology and model serve this purpose. Section 15.0.5 discusses long-term cooling analyses that demonstrate ECCS recirculation and CNV heat removal for at least 72 hours. This demonstrates adequate long term containment heat removal.

6.2.1.3.6 Single-Failure Analysis

This containment response analysis considers potential single failures, and when including the worst-case single failure, the results of each case meet the necessary safety function. Reference 6.2-1 discusses insights obtained from the sensitivity studies that determine single failures that create a bounding set of assumptions that result in limiting CNV peak temperature and pressure for primary release events. The sensitivity results demonstrate that in some scenarios the consideration of no single failure provides a more limiting result.

6.2.1.3.7 Metal-Water Reaction

Because of the absence of significant post-LOCA cladding heat-up, the methodology does not model additional energy resulting from cladding metal-water reaction (Reference 6.2-1).

6.2.1.3.8 Energy Inventories - Loss-of-Coolant Accident

Figure 6.2-8, Figure 6.2-9, Figure 6.2-11, and Figure 6.2-12 provide the integrated mass and energy release rates for the primary system release pressure and temperature limiting events.

6.2.1.3.9 Additional Information Required for Confirmatory Analysis

Table 6.2-1 and Figure 6.2-8 through Figure 6.2-12 provide information supporting confirmatory analysis.

Reference 6.2-1 provides tabulated mass and energy data for representative results.

6.2.1.4 Mass and Energy Release Analysis for Secondary System Pipe Ruptures Inside Containment

Main steam and feedwater line breaks are not required to be postulated for dynamic effects in Section 3.9.1. They are required to be postulated to determine limiting peak temperature and pressure. The containment receives the secondary system mass and energy released following a postulated MSLB or FWLB. The containment response analysis methodology (Reference 6.2-1) is developed in accordance with the guidance of RG 1.203, "Transient and Accident Analysis Methods," December 2005.

The limiting MSLB event and FWLB event are double-ended ruptures of the largest MS line and FW line pipes. The limiting MSLB and FWLB events include postulated loss of EDAS power supply, which results in ECCS actuation, adding primary side mass and energy release into containment during the secondary side mass and energy release.

Secondary system mass and energy releases consist of the MSLB and FWLB events with the asymmetric responses in SGs included. The affected SG blows down into the CNV, and the FW supply and MS lines isolate.

Conservative modeling of secondary system mass and energy release scenarios ensures a bounding analysis. All breaks consider a maximum break size at each location.

The containment response analysis methodology (Reference 6.2-1) uses the heat transfer correlation package in the NRELAP5 computer code for secondary system pipe break analysis. The LOCA and non-LOCA transient analysis methodology reports demonstrate these correlations are applicable to the NPM design (Reference 6.2-1 and Reference 6.2-2). The local fluid conditions and the local heat structure surface temperatures determine the heat transfer mode. Nucleate boiling heat transfer is in the code and applies if the local conditions are appropriate. For the helical coil SG, other heat transfer modes exist as the coolant enters as subcooled liquid and exits as superheated steam. Initial and boundary conditions maximize containment pressure and temperature response.

Reference 6.2-1 provides a description of each postulated secondary system mass and energy release event. Table 6.2-2 provides results of the limiting analyses. The primary system limiting events bound the secondary system mass and energy analyses.

6.2.1.4.1 Mass and Energy Release Data - Secondary System

Similar to primary system mass and energy release scenarios, conservative modeling of the mass and energy release and minimizing the performance of the heat removal function of containment, determine the maximum containment peak pressure and peak temperature scenarios for secondary system releases into containment (Reference 6.2-1).

6.2.1.4.2 Single-Failure Analysis - Secondary System

The containment response analysis methodology considers potential single failures. Due to the simplicity of the NPM design, there are few candidate single failures for the secondary system mass and energy release scenarios. The scenario where ECCS valves fail to open reduces the mass and energy release, which removes the scenario from consideration. Sensitivity studies consider failures of main steam isolation valves (MSIVs) or feedwater isolation valves (FWIVs) to close.

6.2.1.4.3 Initial and Boundary Conditions - Secondary System

Initial conditions for secondary system line break containment response analyses, as described in Reference 6.2-1, ensure a conservative CNV peak pressure and peak temperature result. Selection of the initial conditions follows applicable design specific review standard guidance. The selection process ensures maximization of energy sources and minimization of energy sinks. Initial and boundary conditions associated with primary side parameters for MSLB and FWLB analyses are similar to those described for the primary mass and energy release events, with exceptions noted by Reference 6.2-1. Boundary conditions for secondary system line break containment response analyses ensure a conservative CNV peak pressure and peak temperature result. The process of selecting the boundary conditions is consistent with applicable design specific review standard guidance. The selection process ensures maximization of energy sources and minimization of energy sinks. Boundary conditions assumed by MSLB and FWLB analyses are the same as those used in primary release event analyses except for those listed in Reference 6.2-1.

6.2.1.4.4 Description of Blowdown Model - Secondary System

Modeling of the MSLB considers a double-ended break of an MS line inside the CNV that depressurizes the secondary system and pressurizes the CNV. Cross-connected MS lines downstream of the MS isolation results in both SGs discharging to containment until the steam lines isolate because of a secondary side isolation signal or containment isolation signal. A low steam line pressure signal or high containment pressure signal results in closure of the MSIVs and FWIVs, and reactor trip. Subsequently, the decay heat removal system (DHRS) actuates after FW isolation. Closure of the FW regulating valve mitigates a single failure of the FWIV to close on the affected SG. After the initiation of the break, there are two potential limiting events depending on the evolution of the scenario: a scenario that assumes continued AC power or one that assumes a loss of normal AC and EDAS power. Analysis of the two above scenarios determined that the case with loss of normal AC and EDAS power results in the peak CNV pressure and peak CNV temperature results.

The FWLB is a double-ended break of the largest FW pipe inside containment that results in a depressurization of the affected SG and pressurization of the CNV. A high containment pressure signal results in closure of the MSIVs and FWIVs and reactor trip. The DHRS actuation occurs subsequently after FW isolation. Actuation of DHRS establishes long-term decay heat removal using the unaffected SG and the DHRS. A single failure of the MSIV to close on the affected SG allows more high energy steam to be discharged out of the FWLB before secondary side isolation than would occur if the associated FWIV failed to close. The limiting case, also assumes a loss of normal AC and EDAS power at event initiation, which results in ECCS actuation and blowdown from the RPV into the CNV. The maximum CNV pressure and temperature occurs after the ECCS valves open.

6.2.1.4.5 Energy Inventories - Secondary System

The energy inventories in the secondary system match those evaluated for the primary system mass and energy releases with the exception of the additional conservatisms applied in the initial and boundary condition assumptions applied to the secondary system components, as described in Section 6.2.1.4.3.

6.2.1.4.6 Additional Information Required for Confirmatory Analyses - Secondary System

Information supporting confirmatory analysis for representative secondary side break event progressions is provided in Reference 6.2-1.

6.2.1.5 Minimum Containment Pressure Analysis for Performance Capability Studies of the Emergency Core Cooling System

The ECCS operation directly connects the RPV and CNV volumes and relies on the equalization of pressures within the two volumes. The ECCS flow consists of the RCS vapor flow through the vent valves, that condenses and collects within the CNV volume returning to the RPV. The driving force for the condensate flow back to the RPV through the recirculation valves is the hydrostatic head of coolant in the CNV that collects above the ECCS reactor recirculation valves. In the event of large coolant leaks (e.g., LOCAs, valve opening events), the CNV provides for the retention of adequate reactor coolant inventory to prevent core uncovery or loss of core cooling. The ECCS long-term cooling analyses performed in accordance with Reference 6.2-3, consider a range of boundary conditions that affect CNV heat removal, RPV and CNV internal pressure, and ECCS performance. Chapter 15 presents the ECCS long-term cooling results.

6.2.1.6 Testing and Inspection

The ISI and IST programs identify the required inspections, tests, frequencies, and acceptance criteria for the applicable components and systems.

Section 3.8.2.7 addresses the testing and ISI requirements with respect to compliance with the ASME BPVC for fabrication and preservice examinations used to inspect and test the steel CNV and the other components relied on for containment integrity. Additional information is in Section 14.2 describing the test programs that control initial plant testing (pre-operational and startup) conducted on the CNV and associated structures, systems, and components.

As described in Section 3.8.2.7, fabrication and preservice testing and inspection of the CNV meets ASME BPVC Section III and Section XI requirements for a metallic containment.

Section 3.8.2.7 describes hydrostatic testing of the CNV.

Based on the high pressure and the safety functions of the CNV, enhanced inspection requirements are provided for it that are beyond the Class MC requirements of Reference 6.2-5 Subsection IWE. Specifically, rather than just a visual examination for an ASME BPVC Class MC containment, the upper head-to-shell, lower head-to-shell, and lower shell-to-lower transition shell welds of the CNV require a volumetric examination performed per Reference 6.2-5, Article IWB-2000. Reference 6.2-4 contains further information on CNV inspection requirements.

Periodic inservice inspection of the containment heat removal surfaces ensures compliance with GDC 39 to assess for surface fouling or degradation that could potentially impede heat transfer from the CNV.

Section 6.6 provides a description of the ISI requirements for Class 2 and 3 components.

6.2.1.7 Instrumentation Requirements

Instrumentation monitors the conditions inside the containment and actuates the appropriate engineered safety features (ESFs), should those conditions exceed predetermined levels. Instruments measure containment pressure, temperature, and water level. Section 5.2.5 describes instrumentation that monitors RCS leakage into containment and compliance with RG 1.45.

Containment pressure instrumentation provides control room indication to monitor containment pressure boundary integrity, RCS pressure boundary integrity, ECCS performance, and to support the actuation of critical safety functions as further discussed in Chapter 7.

Four narrow range safety-related instruments and two wide range nonsafety-related instruments measure and monitor containment pressure. The narrow range sensors (transducer/transmitter type) are inside the CNV wall enclosure near the top of containment. There are four independent channels of narrow range CNV pressure instrumentation. The wide range sensors (transducer/transmitter type) are inside the CNV wall enclosure near the top of containment. There are two independent channels of wide range CNV pressure instrumentation.

Containment water level instrumentation provides control room indication to monitor containment pressure boundary integrity, RCS pressure boundary integrity and ECCS performance.

Four instruments that measure and monitor containment water level (digital type) are at the reactor pressure boundary interface. There are four independent channels of CNV water level instrumentation.

Containment air temperature instrumentation provides control room indication to monitor the environment in containment.

Two nonsafety-related instruments measure and monitor containment air temperature. The sensors (resistance temperature detector type) are inside the CNV near the top of the CNV head. There are two independent channels of CNV air temperature instrumentation.

Chapter 7 provides additional containment instrumentation design detail addressing power supplies, actuation logic, and initiation signals for the engineering safety feature functions.

6.2.1.8 Containment Vessel Bolted Closures

The CNV closure studs, nuts, and washers use the material indicated in Table 6.1-1. Section 3.13.1.1 contains details on threaded fastener design considerations.

For the CNV main flange connection, lock plates perform a tooling function holding the CNV main 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 resist the minor friction loads and forces resulting from inserting and threading the studs into the nuts. The lock plates do not resist the forces applied to tension the stud or for the removing and detensioning the studs.

Studs are threaded into the top flange of the CNV. These studs are non structural per ASME BPVC Section NB-1132.1(c)(2), similar to insulation supports.

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

6.2.2 Containment Heat Removal

The CNV material selection and the physical configuration of the NuScale Power Plant design forms the basis for containment heat removal for accident conditions. The reactor pool partially immerses the steel CNV and heat transfers to the water from the outer surfaces of the CNV in contact with the water. The continuous presence of cooling water on the outside of the CNV ensures an immediate, effective, and passive means for containment heat removal. The large inventory of water in the UHS ensures a supply sufficient for long-term containment heat removal.

Under normal operating conditions, the interior of the CNV is in a dry condition under vacuum (<1 psia). The primary method of heat transfer from the outer surfaces of the RPV through the containment volume to the inner containment wall is via radiation. The radiated heat energy conducts through the CNV wall to the exterior surface where it transfers via convection into the reactor pool water or into ambient air above the pool. Table 9.2.5-1 lists the water level in the pool during normal operations. Most of the CNV is in contact with the pool water and removes containment heat during operations. Section 9.1.3 describes the removal of the heat from the pool water by the active reactor pool cooling system.

Section 9.3.6 discusses how the CES maintains containment vacuum during normal operation. Maintaining the containment under a vacuum during normal operation minimizes heat transfer from the RPV to the CNV and minimizes the associated loss of efficiency and is used as a method to measure leakage.

During postulated primary and secondary release events into containment, the CNV collects and accumulates released inventory. Actuation of the ECCS (opening of the RVVs and RRVs) and containment isolation provides for a natural circulation coolant pathway that circulates reactor coolant inventory through the containment volume back to the RPV and through the reactor core.

In the event of a postulated MSLB or FWLB inside containment, the mass and energy released into the containment consists of the inventory present in the train with the break, including the content of the attached DHRS. The total inventory released considers the automatic isolation of the MS and FW lines and single failures. The inventory released into containment flashes, condenses, and accumulates within the CNV.

The steel wall of the CNV provides for the direct (passive) transfer of containment heat (normal, transient, or accident conditions) to the UHS. There is no reliance on active components or electrical power. The design configuration provides the ability to reliably remove containment heat immediately in an accident and for at least 30 days as described in Section 9.2.5. Section 3.1.1 and Section 9.2.5 address conformance with GDC 5.

A description of the design and operation of the ECCS is in Section 6.3. A description of the UHS design and operation is in Section 9.2.5.

6.2.2.1 Design Bases

The CNV is partially immersed which facilitates removal of thermal energy from the containment for accident conditions. Following a DBE that results in containment pressurization, containment pressure rapidly reduces and remains below the design value without operator action as discussed in Reference 6.2-1. For the postulated DBEs described in Chapter 15, containment pressure reduces to less than 50 percent of the peak calculated pressure in less than 24 hours and meets the requirement in PDC 38 (Section 3.1.4) for reducing pressure rapidly after an accident.

The CNTS and UHS also provide LTC that removes sensible and decay heat from the reactor core and containment atmosphere following DBEs. With the exception of the backup secondary isolation valves, the containment heat removal function does not require use of nonsafety-related systems or operator support for long periods of time following an event. The containment volume predominantly operates at less than two percent of CNV design pressure during LTC, which limits the cumulative leakage of coolant inventory from the pressure boundary of the CNV.

The RXB, which is designed to withstand the effects of natural phenomena such as earthquakes, winds, tornadoes, or floods, protects the NPMs and the UHS. The CNTS structures, systems, and components design and construction meet Seismic Category I classification guidance per RG 1.29.

The components of the CNTS that perform containment heat removal allow inspection in accordance with the intervals specified in Reference 6.2-5. The CNV design meets GDC 39 and permits appropriate periodic examinations that ensure continuing integrity and capability for heat transfer (i.e., the design allows inspection of the interior and exterior surfaces for fouling or degradation that could potentially impede heat transfer from the CNV).

The passive cooling of a CNV does not include or require active components to perform the containment heat removal function. The CNV provides a large heat transfer surface with no active components needed for heat removal to the UHS water. Reference 6.2-1 and Reference 6.2-3 describe testing of the passive containment heat removal function for LOCA conditions. The design supports an exemption from GDC 40. Section 3.1.4 addresses the exemption from GDC 40.

6.2.2.2 System Design

Direct communication between two safety-related systems, the CNTS and the UHS, accomplishes passive containment cooling. The passive containment cooling function uses the steel CNV and the UHS water that surrounds most of the CNV except at the top. Containment heat removal consists of heat flow through the CNV wall to the water in the UHS. For normal operations, the UHS transfers the heat energy to the active reactor pool cooling system as described in Section 9.1.3. For accident conditions, the CNV transfers heat to the UHS, and the UHS transfers the heat to the environment as described in Section 9.2.5. For CNV cooling during accident conditions, the UHS includes the inventory of water in the reactor pool and the refueling pool and the water in the spent fuel pool above the top of the weir. Without active cooling, the water in these pools heats and evaporates into the RXB air space before release through the RXB heating, ventilation, and air conditioning system. Heat removal by the water in the UHS during passive containment cooling does not credit heat transfer to the RXB atmosphere or to the heat sinks within the RXB, such as the reactor pool liner or pool concrete. As described in Section 6.2.1, the transfer of heat energy prevents the containment from exceeding its design pressure and temperature following the postulated design-basis accidents (DBAs) identified in Chapter 15.

Section 3.8.2 describes the design features of the CNV including the dimensions, materials, penetrations, and attachments.

Protective coatings are not used or allowed in the CNV, therefore, the effects of post-accident debris generated by coatings are precluded in the design. Section 6.3.2.4 describes conformance with RG 1.82 and the approach used to address Generic Safety Issue 191, Assessment of Debris Accumulation on Pressurized Water Reactor Sump Performance.

6.2.2.3 Design Evaluation

The safety-related systems that provide passive containment cooling are the CNTS and the UHS. These systems are in the RXB, which is designed to withstand the effects of natural phenomena hazards such as earthquakes, winds, tornadoes, or floods while protecting the systems inside. In addition, the CNV and UHS withstand the safe shutdown earthquake.

Removal of heat by the CNV for an accident in an NPM occurs as the accident progresses with no operator actions required. The inventory of water in the UHS pool during power operations ensures sufficient containment heat removal capability at the start of an event. Table 9.2.5-1 provides the normal operating level range for the water in the UHS pools. Table 9.2.5-2 shows that the pool water can be lower than the normal level and provides more than 72 hours of water coverage over the DHRS condensers and supports ECCS operation for a minimum of 30 days. These results indicate the conservatism in the initial inventory of water available in the UHS pools to support containment heat removal.

The Extended Passive Cooling and Reactivity Control methodology report (Reference 6.2-3) describes the LTC Evaluation Model, developed using the evaluation model development and assessment process guidelines given in RG 1.203, to evaluate long-term NPM response during ECCS operation. The LTC analyses results demonstrate that all postulated conditions that could affect the ability of the ECCS to provide adequate LTC are satisfied with respect to collapsed liquid levels, temperatures, and pressures. The results of all cases within these analyses demonstrate that the collapsed liquid level remains above the active fuel at all times, the core remains subcritical, and coolable geometry is maintained. These results demonstrate sufficient LTC capability for 72 hours, without operator action, following a DBE with ECCS operation with or without power available. Reference 6.2-3 and Chapter 15 contains further detail.

Sensitivity analysis results also demonstrate that release of non-condensable gases contained within the pressurizer does not adversely impact pool heat transfer and long term cooling conditions.

Section 6.3.2.4 describes the generation of post-accident debris, debris transport, and downstream effects considered in the design.

Additional design evaluation detail addressing the UHS capability is in Section 9.2.5.

6.2.2.4 Testing and Inspection

Section 6.2.1.6 provides information on the test programs for the CNV for preservice and initial plant testing (preoperational and startup). After startup, inspection in accordance with GDC 39 ensures operability and performance of the passive containment heat removal function. Periodic inservice inspection of the containment heat removal surfaces assesses surface fouling or degradation that could potentially impede heat transfer from the CNV. The inspections ensure the continuing operability of the containment surfaces that perform the heat removal function.

Inspection of the CNV heat removal surfaces is in Section 6.2.1.6.

Testing and inspection of the UHS are in Section 9.2.5.4.

6.2.2.5 Instrumentation Requirements

Section 6.2.1.7 addresses instrumentation for the CNV; Section 9.2.5.4 addresses instrumentation for the UHS.

6.2.3 Secondary Containment Functional Design

The NuScale Power Plant is not a dual containment design (i.e., no secondary containment function requirement) facility. Section 3.8.2, Section 6.1, and Section 6.2.1 describes the design of the CNV.

6.2.4 Containment Isolation System

The RXB has up to six NPMs, and each NPM has a CNTS with a containment boundary that prevents or limits the release of radioactive materials under postulated accident conditions. The CNV and the CIVs form the containment boundary and the passive containment isolation barriers prevent releases through the penetrations in the CNV. Although there is no "containment isolation system", the CIVs for an NPM are similar to such a system for existing light water reactor plants.

Figure 6.2-1 has a cutaway view of an NPM. Figures 6.2-2a and 6.2-2b show the CNV top head and side penetrations, along with showing and listing the penetrations by nozzle number. Figure 6.2-6 is a hydraulic schematic depicting the major components of the CIV actuators. Section 6.2.4.2.2 describes the actuators in more detail.

There are 43 penetration openings in the CNV. There are the following types of penetrations:

- mechanical penetrations for process fluids or gases - 18 total with 12 on the top head for process flows and six on the side with four penetrations for ECCS valve actuator assemblies and two penetrations for DHRS process flows
- electrical penetrations for power supply: four total on the side
- I&C penetrations for signals: 12 total: four on the top and eight on the side

- access and inspection port penetrations: nine total with one on the top head and eight on the side

Note that in Table 6.2-3, there is no CNV penetration number 21; the opening of the CNV at the main flange is not a penetration opening.

Each of the CNV penetrations has the following type of component(s) forming the containment boundary.

- Eight mechanical penetrations through the CNV top head have two primary system containment isolation valves (PSCIVs) in series outside of the CNV that stop the flow through a process flow path when the valves are closed. These lines connect to the RCPB or open to the atmosphere inside of a CNV. The reactor cooling component water lines in containment are a closed loop but not designed as a containment barrier.
- Six mechanical penetrations through the CNV top head have a single secondary system containment isolation valve (SSCIV) on lines outside of the CNV because the piping inside of the CNV is a closed piping system and does not connect to the RCPB or the atmosphere inside of the CNV. These closed piping systems inside of the CNV are part of the containment boundary inside of the CNV, while the CIVs are part of the containment boundary outside of the CNV when the valves are closed.
- The piping outside of the CNV welded to the two CNV nozzle safe-ends for the main steam system (MSS) lines is part of the containment boundary outside of the CNV. This piping is from the safe-end to an MSIV, an MSIBV, and the two parallel DHRS branch lines.
- Four of the mechanical penetrations through the side of the CNV have a containment boundary formed by the ECCS valve actuator assembly that includes a sealed valve bonnet for each trip and reset valve.
- Two of the mechanical penetrations through the side of the CNV have closed DHRS piping outside of the CNV. The piping inside of the CNV is a closed piping system and does not connect to the RCPB or the atmosphere inside of the CNV. These closed piping systems inside of the CNV are part of the containment boundary inside of the CNV, while the closed DHRS piping is part of the containment boundary outside of the CNV.
- Twenty-five electrical, I&C, and access penetrations have a flange opening with a bolted connection and use O-ring seals to prevent penetration leakage.

The CIVs provide for the passage of fluids and gases through the CNV penetrations used for process flows while preserving the integrity of the containment boundary and preventing or limiting the release of fission products under postulated accident conditions. Table 6.1-3 lists materials that may be used in fabrication of the CIVs.

The CIVs are in the CNTS and are not part of the various process systems that penetrate the CNV upper head. The valves have labels with the interfacing system acronym for easier identification (e.g., CES, containment flooding and drain system [CFDS], CVCS, reactor component cooling water system (RCCW), feedwater system [FWS], and MSS). For example, Figure 9.3.4-1 shows that the CIVs on the CVCS injection line are not part of CVCS. Figure 6.2-3 shows the same system boundary change between the CVCS and CNTS outside of containment, as well as the system boundary changes for the other process systems with lines penetrating the CNV upper head.

Figure 6.2-3 also shows the systems to which the process lines connect inside of containment (e.g., CFDS, RCS, RCCW, and the SGs). Note that the figure shows the DHRS has lines that connect to the MSS lines outside of containment and upstream of the CIVs on the MSS lines. The DHRS also has penetrations through the side of the CNV without CIVs on the line outside of the CNV.

The passive containment isolation barriers are

- the closed piping in the MSS, FWS, and DHRS inside of the CNV.
- the piping between a CNV safe-end and an MSIV, and main steam isolation bypass valve (MSIBV).
- the ECCS valve actuator assemblies.
- the DHRS closed piping outside of containment.
- CNV flange connections.
- electrical penetration assemblies (EPAs) and instrumentation seal assemblies.
- CITF covers.

Note that some of these components are not part of the CNTS, but are part of the systems that penetrate the CNV.

6.2.4.1 Design Bases

The CNTS protects against the release of radioactive material to the environment as a result of an accident. The CNTS accomplishes this based in part on leakage rates being within acceptance criteria for the CIVs and passive containment isolation barriers. In addition, there is automatic actuation and closure of the CIVs when specific defined limits for process variables are exceeded. Periodic inservice inspection and testing of these containment isolation components maintains both of these capabilities. Table 6.2-3 provides a list of the containment penetrations and shows which penetrations are used for which process system fluids or gases. Table 6.2-3 lists the CIV open or closed position for normal and accident conditions.

Table 6.2-4 provides a list of the CIVs. The CIVs consist of PSCIVs and SSCIVs. Table 6.2-4 shows the valves in each group. The PSCIVs have two valves in series on each line through containment, and the pair of valves meets the intent of GDC 55 or GDC 56. The SSCIVs have a single valve on each line (and bypass line) and meet GDC 57. The bases for meeting the GDC, or the intent of the GDC, are below, as are the exemptions needed from the GDC and their justifications. Table 7.1-4 provides the conditions under which containment isolation becomes mandatory.

Consistent with GDC 1, design, fabrication, erection, and testing of the CIVs and the passive containment isolation barriers meet the quality standards commensurate with the importance of the required safety functions. Consistent with GDC 2, these containment isolation components are in the RXB, which is designed to withstand the effects of natural phenomena hazards such as earthquakes, winds, tornadoes, and floods. The RXB protects these components from natural phenomena hazards and meet Seismic Category I requirements. A summary discussion of compliance with GDC 1 and GDC 2 is in Section 3.1.1.

Consistent with GDC 4, CIVs and barriers accommodate the effects of and are compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents (including LOCAs). These CIVs and barriers protect the containment isolation components from the dynamic effects of missiles, pipe whip, and discharging fluids that result from in-plant equipment failure or from events and conditions external to the facility. Additional information addressing design criteria for preventing dynamic effects on CIVs is in Section 3.6.2.1.2. A summary discussion of compliance with GDC 4 is in Section 3.1.1.

Consistent with GDC 5, the CIVs and barriers are independent to one NPM and do not function for the other NPMs at a NuScale Power Plant. A summary discussion of compliance with GDC 5 is in Section 3.1.1.

Consistent with GDC 16, the containment isolation components (valves and barriers) isolate the penetrations and the associated fluid systems, together with the CNV, establishing an essentially leak-tight barrier against the uncontrolled release of radioactive material to the environment for as long as accident conditions require. A summary discussion of compliance with GDC 16 is in Section 3.1.2.

Consistent with GDC 54, the piping systems that penetrate the CNV have leak detection, isolation, and containment capabilities that are redundant and perform reliably considering the requirements for the type of piping described in GDC 55, GDC 56, or GDC 57, and the exemptions for these GDCs that are justified, as described below for each. A summary discussion of compliance with GDC 54 is in Section 3.1.5.

Consistent with GDC 55 except for the location of isolation valves, each line that penetrates the containment boundary and is part of the RCPB has two isolation valves in series. While GDC 55 provides the alternative of an automatic isolation valve inside and an automatic isolation valve outside containment, both automatic CIVs on these lines are outside of the CNV. A summary discussion of compliance with GDC 55 is in Section 3.1.5.

Consistent with GDC 56 except for the location of isolation valves, each line that penetrates the containment boundary and connects directly to the containment atmosphere has two isolation valves in series. While GDC 56 provides the alternative of an automatic isolation valve inside and an automatic isolation valve outside containment, both automatic CIVs on these lines are outside of the CNV. A summary discussion of compliance with GDC 56 is in Section 3.1.5.

Consistent with GDC 57 except for the DHRS piping, the lines that penetrate the CNV pressure boundary and are neither part of the RCPB nor connected directly to the containment atmosphere have an automatic isolation valve located outside of containment. While GDC 57 requires at least one CIV outside of containment along with closed piping inside of containment, the DHRS containment penetrations have closed loops of MSS, FWS, and DHRS piping inside of the CNV and closed loops of DHRS piping outside of the CNV. The closed piping systems inside of the CNV are part of containment boundary inside of the CNV. The closed DHRS piping outside of the CNV is part of the containment boundary outside of the CNV and performs this function without the need for valve actuation. A summary discussion of compliance with GDC 57 is in Section 3.1.5.

With the exception of 10 CFR 50.34(f)(2)(xiv)(E) (Section 9.3.6.1 contains the CES exemption), the CIVs, including associated controls, comply with 10 CFR 50.34(f)(2)(xiv). The design conforms to the requirements of RG 1.141 through adherence to ANS N271-1976. The provisions of ANSI/ANS 56.2-1984, Section 3.6.5 apply to penetrations with both CIVs outside containment that serve non-ESF process systems (Table 6.2-3). The design conforms to RG 1.155 station blackout requirements for CIV closure and valve position indication with respect to establishing and maintaining containment integrity and identifying valve closure status. Additional information addressing conformance with the applicable requirements of 10 CFR 50.34 and conformance with RG 1.141 is in Section 1.9.

The CIVs operate hydraulically and fail to the closed position, which is the safe position, on loss of power or loss of hydraulic pressure. Each PSCIV, FWIV and MSIV actuator is equipped with a stored energy device that applies a constant force to close the valve. This force is overcome by hydraulic power supplied by a hydraulic control skid to open the CIV as required for normal power operations. The actuators require hydraulic fluid pressure to open the valves. On a containment isolation signal, a loss of power, or a manual close signal from the main control room, the CHPU relieves hydraulic pressure, allowing the valve to close. Upon reaching the fail-safe position, the hydraulically operated valves remain in this fail-safe position without power from the actuator until the pressurized hydraulic fluid is restored to the actuator (as depicted by Figure 6.2-6).

The actuators, position indicators, and safety-related devices required to vent the hydraulic pressure are qualified to IEEE-323, IEEE-344, and IEEE-382

When the PSCIVs are closed, the fluid between the two valves could have heat added from the containment or external environment. Overpressure protection for this condition is incorporated in the dual valve design. The valve closest to the CNV is a single seat uni-directional design such that high pressure between the two valves is passively vented back into the CNV as shown in Figure 6.2-4.

6.2.4.2 System Design

6.2.4.2.1 General Description

The containment pressure boundary includes the steel CNV (described in Section 3.8.2 and Section 6.1) and the CIVs and barriers that close the penetrations in the CNV. A schematic of CNV penetrations identified by penetration number is provided by Figures 6.2-2a and 6.2-2b.

The passive containment isolation barriers use the following design features. The vertically oriented PSCIVs weld directly to their respective containment isolation test fixture (CITF) valves. All PSCIVs close upon de-energizing to isolate the CNV. The PSCIVs are a single assembly design consisting of two valves (fully independent discs, seats and actuators) within a single valve assembly.

Between a CNV and a MSIV of the MS lines, there is a tee to the DHRS piping. Unlike the other CIVs, the MSIVs and bypass valves do not attach via weld to a safe-end on the CNV top head. The tees are leak-tight based on the welded design (as described in Section 3.6.2.2 and as shown in Figure 6.2-5a).

The ECCS valve actuator assemblies attach via weld on the outside of the CNV to a nozzle safe-end. The containment boundary is the body of the valve actuator assembly (valve manifold) and a sealed valve bonnet. Each trip and reset valve assembly has double metal O-ring seals with a port between the seals for periodic seal leak rate testing.

The DHRS closed piping outside of the containment is leak-tight based on the welded design described in Section 5.4.3. The piping design meets ASME BPVC, Section III, Class 2, Subsection NC requirements, and the applicable criteria of Nuclear Regulatory Commission Branch Technical Position 3-4, Revision 2, as described in Section 3.6.2.

The flange connection closures on the CNV are the covers for the access and inspection ports, manways, and electrical penetrations assemblies (EPAs and instrument seal assemblies (ISA)). The closure flanges have double metal O-ring seals with a port between the seals for periodic testing of the seal leakage rate.

The containment isolation components with moving parts are the CIVs, which function to provide a means of isolating process flow paths that are not required for safe shutdown or accident mitigation and that pass through containment penetrations. These valves minimize the release of fission products to the environment during DBEs while allowing process flows into and out of the CNV during normal operations.

The CNTS includes instruments that provide signals, along with the instruments in the various process systems, to the MPS, which has the control logics necessary to generate an actuation signal to isolate the appropriate

CIVs in the process lines. Instrumentation and control (monitoring and actuation logic) for the CIVs is in Section 7.1.

Two barriers provide containment isolation of process piping lines. Lines that communicate directly with the RCPB or with the containment atmosphere have two redundant isolation valves. Except for the DHRS, the lines penetrating the containment boundary that are not part of the RCPB or connected directly to the containment atmosphere have one isolation valve outside of the CNV with the closed system piping inside of containment functioning as the inner containment boundary. For the DHRS piping system, the closed loops inside and outside of containment function as the two containment boundaries.

The CNTS components and barriers classification is by safety, quality group, and seismic category. The CIVs, their valve actuators, and their penetration safe-ends are safety-related and Seismic Category I. The CIVs connected to lines that directly contact the reactor coolant are Quality Group A. The other CIVs (i.e., those connected to lines that are open to the containment atmosphere or that form a closed loop inside containment) are SSCIVs and are Quality Group B, as shown in Table 6.2-3. With the exception of the DHRS, neither accident mitigation nor safe shutdown requires the process systems that penetrate the containment pressure boundary. The tertiary isolation valves on the chemical and volume control (CVC) lines outside of the chemical and volume control CIVs are nonsafety-related and Seismic Category I, and are Quality Group C. The main control room can open the CFDS and CVC injection CIVs during an active containment isolation signal for inventory makeup following a beyond design-basis event (BDBE) as discussed in Chapter 19.

Actuation of the isolation function does not require electrical power. The valves close upon de-energization to perform the containment isolation function. In addition, as shown in Table 7.1-4, the MPS actuates the CIVs upon detection of low AC voltage to the battery chargers.

The CIVs are attached to the top head of the CNV and are external to the CNV. The CIVs have position indication available in the main control room.

The PSCIVs consist of the isolation valves that meet the intent of GDC 55 or 56 requirements and justify an exemption from these GDCs. The PSCIVs on the RCS injection (CVCS makeup), RCS (CVCS) pressurizer spray, RCS discharge (CVCS letdown), and RCS (CVCS) high point degasification lines satisfy the intent of GDC 55 for isolation of lines penetrating the containment that are part of the RCPB. Although GDC 55 specifies as an acceptable alternative that one isolation valve needs to be inside and one outside of containment, the design provides two isolation valves in a single valve body outside of containment welded to a CITF, which is welded to a CNV nozzle safe-end. The benefits of this approach includes minimizing piping between valves and between the vessel and the valve. This minimizes RCPB welds outside of containment and precludes a pipe break or piping leakage between the vessel and a valve. Also, this approach removes a hydraulically-operated valve from inside of containment and keeps the valve out of the post-accident atmosphere in a CNV. These benefits justify an exemption from GDC 55. The vertically oriented PSCIVs are welded directly to their respective containment isolation test fixture (CITF) valves. All PSCIVs close upon de-energizing to isolate the CNV. The PSCIVs are a single assembly design consisting of two valves (fully independent discs, seats and actuators) within a single valve assembly.

The PSCIVs on the CES and CFDS lines also justify an exemption and satisfy the intent of GDC 56 for isolation lines penetrating the containment that are connected directly to the containment atmosphere. Although GDC 56 specifies as an acceptable alternative that one isolation valve needs to be inside and one outside of containment, the design provides two isolation valves in a single valve body outside of containment welded to a CITF which is welded to a CNV nozzle safe-end. The benefits of this approach includes minimizing piping between valves and between the vessel and the valve. This minimizes welds outside of containment and precludes a pipe break or piping leakage between the vessel and a valve. Also, this approach removes a hydraulically-operated valve from inside of containment and keeps the valve out of the post-accident atmosphere in a CNV. These benefits justify an exemption from GDC 56.

The PSCIVs are on the RCCW lines because these lines are not credited as a containment boundary even though the lines form a closed loop inside containment and do not connect to the RCPB and do not open to containment atmosphere. While a single CIV is sufficient for these lines to meet GDC 57, two CIVs in series are provided.

The design configuration for the DHRS satisfies GDC 57 with the exception that closed loop piping provides an isolation barrier outside of containment rather than an isolation valve. The design configuration for DHRS piping does not comply with the GDC 57 criteria for an isolation valve for each containment penetration with a closed piping loop inside of containment. An exemption from GDC 57 is based on the design of the DHRS piping. GDC 57 requires at least one CIV outside of containment along with closed piping inside of containment. For the DHRS containment penetrations, the closed loops of piping inside of containment and closed loops of piping outside of containment justify an exemption from GDC 57.

The SSCIVs consist of the MSIVs, the MSIBVs, and the FWIVs. These CIVs satisfy GDC 57 for isolation of lines penetrating the containment that are not part of the RCPB and are not open to the containment atmosphere.

As shown on Figure 6.2-5a, each line with an MSIV has an MSIBV in a parallel flow arrangement to bypass an MSIV. The MSIBV is a normally open valve that is in a parallel flow path with the larger diameter MSIV. The MSIBV introduce steam for secondary system startup before opening the MSIVs. General Design Criterion 57 requires closed system isolation valves to be as close to containment as practical. The MSIVs are approximately four feet from the CNV, which meets this requirement.

As shown on Figure 6.2-5b, a FWIV and FW nozzle check valves are in the same valve body. These valves weld to a CITF which is welded to the CNV nozzle safe-end. The purpose of the FWIV is for containment isolation and to provide a DHRS pressure boundary. The feedwater nozzle check valve closes more rapidly (<1 second) than the FWIV during an FWLB outside containment. The safety function of the check valve is to close quickly to preserve DHRS inventory.

The CNTS design does not include instrument lines containing process fluids penetrating the containment boundary. Pressure, temperature, level, and flow sensors that monitor processes within an NPM are inside the containment pressure boundary with digital or analog signals coupled to equipment outside of the CNV for processing. The design conforms to GDC 55 and 56 and to the guidance of RG 1.11 by restricting the instrumentation process lines to within the containment pressure boundary.

6.2.4.2.2 Component Description

The valves used to isolate process lines penetrating the containment are of two basic designs. One design consists of a configuration of two valves (obturators) contained within a single valve body used for the PSCIVs. The second design is a single valve design used for SSCIVs.

As shown in Figure 6.2-4, the dual-valve, single-body PSCIV design consists of two valves (fully independent balls, seats, and actuators) within a single valve body welded to a CIFT which is welded to a CNV nozzle safe-end.

In accordance with the regulatory requirements for redundancy, each PSCIV assembly contains two independent balls and actuators. The single body design welded to a CITF places the inboard CIV as close to the CNV as possible and eliminates the potential for a line break between the two valves in series. Independent divisions of the MPS provide the I&C for each valve within a PSCIV assembly.

The PSCIVs connected to lines that directly contact the reactor coolant (i.e., GDC 55 lines – RCS injection, RCS discharge, pressurizer PZR spray, and RPV high-point degasification) during normal operation are Quality Group A components with design, fabrication, construction, testing, and inspection in accordance with the ASME BPVC, Section III, Class 1, Subsection NB and Seismic Category I criteria.

The PSCIVs on the RCCW, CES, and CFDS lines connect to lines that directly contact the containment atmosphere (i.e., GDC 56 lines) or to lines that form a closed loop inside the containment (i.e., GDC 57 lines) and are Quality Group B components. These PSCIVs are ASME BPVC Class 2.

As shown in Figure 6.2-5a and Figure 6.2-5b, the single valve design consists of one ball-type valve. The SSCIVs have a ball, seat, and seals that allow for maintenance, repair, or replacement.

The SSCIVs, design, fabrication, construction, testing, and inspection conform with the ASME BPVC, Section III, Class 2, Subsection NC, Quality Group B, and Seismic Category I criteria. The FWIVs, FW check valves, MSIVs, and MSIBVs satisfy these criteria. The MSIVs and MSIBVs attach via weld directly to a CNV top head nozzle safe-end. The FWIVs and FW check valves are in a common valve body and weld to a CITF which is welded to a CNV top head nozzle safe-end. The SSCIV designs include provisions to preclude thermal binding and pressure locking.

When the FWIVs actuate to close, the FW check valves close first. The FWIVs then close and fluid between the FWIVs and FW check valves could heat up. The dual valve design considers design overpressure for this condition. As the fluid heats up and expands, the pressure relieves passively through a small port in the check valve disk. Section 3.9.6.3.2 addresses check valve exercise testing requirements for this passive overpressure feature. The closed exercise test verifies the valve is within a flow range that verifies that the valve is closed, and that the disk port is clear and is passing fluid.

The PSCIV materials, including weld materials, conform to fabrication, construction, and testing requirements of ASME BPVC, Section III, Subsection NB or NC for the respective class valves. The materials selected for fabrication conform to the applicable material specifications provided in ASME BPVC, Section II and meet the requirements of ASME BPVC, Section III, Article NB-2000 or NC-2000 for the respective class valves. The SSCIV and MS piping materials, including weld materials, conform to fabrication, construction, and testing requirements of ASME BPVC, Section III, Subsection NC. The materials selected for fabrication conform to the applicable material specifications provided in ASME BPVC, Section II and meet the requirements of ASME BPVC, Section III, Article NC-2000. The PSCIVs, SSCIVs, and MS piping are constructed of materials with a proven history in light water reactor environments. Surfaces of pressure retaining parts of the valves, including weld filler materials and bolting material, are corrosion resistant materials such as stainless steel or nickel-based alloy.

Attachment welding of the PSCIVs uses procedures qualified in accordance with the applicable requirements of ASME BPVC, Section III, Subarticle NB-4300 or NC-4300 for the respective class valve and Section IX. Welding of the SSCIVs and MSIV and MSIBV valve assemblies use procedures qualified in accordance with the applicable requirements of ASME BPVC, Section III, Subarticle NC-4300 and Section IX.

When the PSCIVs close, the fluid between the two valves could heat up. The dual valve design considers design overpressure for this condition is addressed in the valve design. In this manner the PSCIV design includes provisions to preclude pressure locking. Hydraulic-operated valve performance assessment testing in Section 3.9.6 addresses testing requirements for the PSCIV subcompartment relief devices. Section 3.9.6 addresses testing of skid mounted subcomponents, other than relief devices.

The PSCIV design provides provisions for Appendix J, Type C testing via the use of the CITFs that allow the pressurization of each of the two valves (Figure 6.2-4).

Both PSCIV and SSCIV designs use hydraulic actuators. Each actuator has a hydraulic cylinder applying opening force to the valve and a stored energy device (pneumatic bottle or mechanical spring) applying closing force to the valve. To open the valve, hydraulic pressure increases, exceeding the force applied by the passive stored energy device. To close the valve, pilot valves vent hydraulic fluid, allowing the stored energy device force to exceed the hydraulic pressure force. The PSCIV and SSCIV designs and qualifications require that torque closure provide a sealing that prevents reopening and unseating of each ball valve for the extended time period for the design basis and beyond design basis functions assumed for each individual ball valve. The hydraulic cylinders are pressurized by a hydraulic skid and vented by redundant safety related pilot valves. The hydraulic skid is safety-related; however, the only safety-related components contained on the skid are the safety-related pilot valves and associated vent path. Testing requirements for hydraulic skid-mounted components are in Section 3.9.6.

The CIV actuators interface with a central hydraulic power unit (CHPU). A simplified hydraulic schematic of the actuator to CHPU interface is provided by Figure 6.2-6. Non-safety related electrical equipment does not prevent the CIVs from opening.

The CIV actuators subassembly contain a separate stored energy device that provides the required motive force to close the valves in the required stroke time, when the hydraulic pressure is removed. Hydraulic pressure is removed either a loss of power to the CHPU solenoids or hydraulic fluid loss such as a hydraulic line break. Electrical power to the actuator is not required to close the CIVs.

The MPS controls the hydraulic pressure venting as described in Chapter 7.

6.2.4.2.2.1 Piping Systems Connected to the Reactor Coolant Pressure Boundary

A dual valve, single body PSCIV provides isolation of each containment penetration that is part of the RCPB (RCS injection, RCS discharge, RCS high point degasification and pressurizer spray).

When closed, the PSCIVs isolate the reactor coolant in the primary systems that penetrate the CNV from the outside environment. In accordance with regulatory requirements for redundancy, two separate valves are in each single valve body.

Each PSCIV operates remotely from the main control room with valve position indicated and each valve automatically closes under accident conditions that require containment isolation. These valves also close under loss of power.

The CNTS components also include reverse flow valves on the CVC influent lines. The CVC discharge line contains an air operated isolation valve, located downstream of the CIVs. The CVC injection line and the pressurizer spray line each contain a reverse flow check valve upstream of the CIVs. The RPV high point vent degasification line contains a solenoid valve downstream of the CIVs.

6.2.4.2.2.2 Piping Systems Open to the Containment

The design includes two systems that penetrate the containment boundary and are open to the containment atmosphere: CES and CFDS.

Isolation of the CES and CFDS lines use the same type of dual valve, single body PSCIV. The PSCIVs on the CES and CFDS lines do not connect to the RCPB; these PSCIVs are ASME BPVC Class 2 valves.

When closed, the PSCIVs isolate the containment atmosphere from the CES and CFDS lines outside of the CNV. In accordance with regulatory requirements for redundancy, two separate valves are in a single body.

Independent divisions of the MPS provide the instrumentation and control for each valve within a dual valve, single body PSCIV. Each PSCIV remotely operates from the main control room and automatically closes upon an isolation signal or loss of power.

6.2.4.2.2.3 Piping Systems Closed to Containment and not Connected to the Reactor Coolant Pressure Boundary

Each closed piping loop inside of the CNV for a system penetrating the containment boundary that is neither part of the RCPB nor connected directly to the containment atmosphere has an SSCIV that is a single CIV outside of containment or has a closed loop of piping outside of containment. The closed piping loop inside containment serves as one of the two containment boundaries necessary to meet the containment isolation design requirements.

Each of the following systems is a closed piping loop inside of the CNV that penetrates the containment boundary:

- MSS lines (single SSCIV)
- FWS lines (single SSCIV)
- DHRS piping (closed piping loop outside of containment)

When closed, the SSCIVs isolate the MSS and FWS flow paths within the containment from the lines outside of containment while establishing the flow path for the DHRS.

Each MSIV and MSIBV welds directly to a CNV top head nozzle safe-end. The closed loop section of the piping inside containment conforms to ASME BPVC, Section III, Class 2, Subsection NC criteria.

The SSCIVs on the MS and FW lines are Seismic Category I, Quality Group B components capable of remote operation from the control room. The SSCIVs meet the requirements of ASME BPVC, Section III, Class 2, Subsection NC criteria.

Each SSCIV remotely operates from the main control room and automatically closes under accident conditions that require containment isolation. Each valve has remote position indication in the main control room. The valves also close under loss of power.

The RCCW cooling water lines have CIVs that are of the dual valve, single body design. These valves are Quality Group B, ASME BPVC, Section III, Class 2, Subsection NC. The RCCW piping and pressure retaining components inside containment form a closed loop and have design, fabrication, construction, testing, and inspection to ASME BPVC, B31.1 and are part of the CRDS. The CIVs for these RCCW lines are ASME BPVC Class 2 components. The isolation valves weld directly to the CITF which is welded to the CNV top head nozzle safe-end.

When closed, the PSCIVs on the RCCW lines isolate reactor component cooling water system piping to and from the CRDMs.

Each PSCIV on a RCCW line operates remotely from the main control room and automatically closes under accident conditions that require containment isolation. The main control room provides valve position. The valves close under loss of power.

The DHRS is a passive engineered safety system that relies on natural circulation to remove heat from the RCS through the SG and reject heat to the reactor pool. Containment isolation uses closed loop Seismic Category I, Quality Group B, ASME BPVC Class 2 piping inside and outside of the containment boundary.

The DHRS steam side piping branches off each CNV safe-end upstream of an MSIV. Each DHRS steam supply line contains two actuation valves in parallel that are normally closed during operation.

The FWIV assembly includes a safety-related check valve housed in the same valve body as the SSCIV. The FWIV check valve closes more rapidly than the FWIV in the event of an FWLB outside containment to preserve DHRS inventory. Figure 6.2-5b depicts the FWIV assembly.

Section 5.4.3 describes the DHRS actuation valves.

The DHRS piping outside of containment performs the isolation function and establishes the containment boundary outside of containment without the need for valve actuation. Section 5.4.3 provides a description of the DHRS.

6.2.4.2.3 System Operation

The components forming the containment boundary for the containment penetrations in these piping systems provide two independent means of isolating fission products from beyond the outer containment boundary. The designs for the PSCIVs and SSCIVs allow for normal operation of the systems shown in Figure 6.2-3 and automatic isolation of the valves for accident conditions that require containment isolation. Each CIV is fully closed within 10 seconds after power is removed from the actuator. The designs of the CIVs permit periodic operability and leak testing. Section 6.2.4.4 and Section 6.2.6 further discuss testing details.

6.2.4.3 Design Evaluation

The containment isolation function prevents the release of radioactive materials with an essentially leak-tight barrier. The integrity of the CNV along with isolation of the penetrations for the CNV accomplishes this purpose with consideration of containment isolation actuation and control logic, valve actuation and control features, and valve closure times.

The seismic classification of CNTS components is given in Section 3.7. The CIVs conform to ASME BPVC, Section III and meet Group A or Group B quality standards, as defined in RG 1.26. Components that serve as part of the RCPB meet Group A quality standards. Section 1.9.1 discusses conformance with RG 1.26.

Section 3.2 describes quality standards applicable to the respective quality groups.

The placement of CIVs within the RXB structure protects them from external hazard effects. Section 3.4 (flooding), Section 3.5 (missiles), Section 3.6 (pipe rupture), Section 3.7 (earthquake), and Section 9.5 (fire) addresses protection from internal hazards. Section 3.11 addresses environmental qualification of CNTS components.

There is redundant containment isolation of the lines penetrating containment. Two barriers in series provide isolation of each piping penetration; the barriers consist of either two redundant CIVs outside of containment, a CIV outside of containment combined with a closed piping system inside of containment, or a closed piping system outside of containment and a closed piping system inside of containment. With this arrangement, no single active failure prevents containment isolation of a piping penetration. There are redundant containment boundary valves and barrier components; each isolation valve and barrier arrangement provides for backup in the event of accident conditions. A failure modes and effects analysis evaluating the effect of postulated CIV failures is in Table 6.2-5. The isolation valve and barrier arrangements satisfy the intent of GDC 54, GDC 55, GDC 56, and GDC 57 and conform to applicable portions of RG 1.141

through adherence with ANS N271-1976. The provisions of ANSI/ANS 56.2-1984, Section 3.6.5 apply to penetrations with two redundant CIVs outside containment that serve non-ESF process systems. Section 1.9.1 addresses conformance with RG 1.141.

Where a CIV provides the isolation boundary, the valve fails to the safe (isolate) position on loss of hydraulic pressure, loss of signal, or loss of power to the actuator.

Redundant safety-related divisions of MPS power CIV actuator vent solenoid valves. Section 7.1 discusses redundancy in the I&C system for the containment isolation system (CIS).

The fail-safe feature of the CIVs and passive containment cooling design ensures maintenance of containment integrity in the event of a station blackout. The containment isolation function initiates upon loss of power, and the isolation valves remain closed for the duration of DBEs. The passive cooling design of the NPM provides a reliable coping capability that achieves safe shutdown and maintains core cooling and containment integrity for an indefinite duration, independent of AC power sources.

The closed piping systems credited for containment isolation boundaries within the containment withstand CNV design conditions for external design pressure and temperature and the environmental effects resulting from a LOCA. There are protections for the closed piping systems against LOCA effect missiles, pipe whip, and jet forces. The qualification of the closed piping inside containment also considers differential pressure, maximum humidity, a steam-laden atmosphere, and the presence of chemical additives in the atmosphere.

The closed DHRS piping loops outside of containment that do not include an isolation valve, designed to withstand reactor vessel design pressure and temperature. The piping forms a passive containment isolation barrier located outside of containment and functions under the most adverse anticipated environmental conditions to which the piping may be exposed.

The piping systems that penetrate containment have leak detection, isolation, and containment capability having redundancy, reliability, and performance capabilities commensurate with the containment isolation function. Section 6.2.6 addresses leakage detection capability and the leakage detection test program. Section 3.9.6 discusses the CIV operability tests. Section 7.1 discusses redundancy and reliability of the CIV actuation system. Taken together, these programs establish the overall reliability of the CIVs.

The containment penetrations associated with systems not required for safe shutdown or accident mitigation isolate automatically upon a containment isolation actuation signal. The isolation signal is an engineered safety features actuation system (ESFAS) signal received from the MPS. The MPS consists of four independent measurement channels that monitor plant parameters and activate the operation of the ESF systems, including the initiation of the containment isolation function. Section 7.1 discusses the design-basis of the MPS.

The control systems are able to reset an isolation signal without automatically reopening the valves. Deliberate operator action performs an override control function after actuation signal reset. Reset does not automatically cause any isolation valve to change position. The design precludes the reopening of more than one CIV at a time. Reopening of isolation valves is valve by valve or line by line.

Actuation of the containment isolation function consists of the closure of the CIVs on the MSS, FWS, RCCW system, CVCS, CES, and CFDS lines. As shown in Table 7.1-4, an ESFAS containment system isolation actuation signal initiates on high containment pressure, low pressurizer level, low-low pressurizer level, low AC voltage to the battery chargers, or high under-the-bioshield temperature.

Discussion of the redundancy and diversity of the instrumentation relied upon to initiate the containment isolation function is in Section 7.1.

Closure times for CIVs minimize the release of containment atmosphere to outside of the containment boundary and mitigate offsite radiological consequences by providing a rapid response to a closure signal, in accordance with GDC 54. CIVs close within ten seconds from the time when the power is removed from the actuator. Closure times for the isolation valves are independent of containment back-pressure considerations. Valve closure times consider the instrument delay time, actuation signal setpoint, and time to be in the fully-closed position.

The PSCIVs and SSCIVs actuate to the fully closed position within ten seconds from the time of de-energization.

The PSCIVs stop fully developed pipe break flow for both steam and liquid conditions within the ten second total valve closure time. The basis for these flow conditions bounds the expected range of flows.

The MSIVs stop fully developed pipe break flow for steam conditions within the ten second total valve closure time. These flow conditions bound the expected range of flows. The MSIBV closes within ten seconds of receipt of a closure signal or loss of power.

The FWIVs stop fully developed pipe break flow in both the forward and reverse directions within the ten second total valve closure time.

The main control room provides the open or closed status of CIVs.

Chapter 15 discusses the effects and consequences of events that require the containment isolation function. The containment pressure and temperature response following mass and energy releases inside containment (e.g., LOCA, MSLBs, or FWLBs) is in Section 6.2.1.

The MPS conforms with 10 CFR 50.34(f)(2)(xiv) in that it provides signal diversity for the containment isolation function.

The PSCIVs, SSCIVs, and safety-related instrumentation that support the containment isolation function ensure that no single failure can result in loss of the protective function.

Section 3.9.6.3 and Section 6.2.6 detail that the design provides the capability to periodically test the CIVs for leakage and functionality.

6.2.4.4 Tests and Inspections

The CIVs and barrier components permit the required inspections and performance of tests to ensure that functional capability of the components is maintained under DBA conditions.

The PSCIVs connected to lines that directly contact reactor coolant undergo inspection as Class 1 components, and those that directly contact the containment atmosphere or form a closed loop inside containment undergo inspection as Class 2 components. The SSCIVs undergo classification and inspection as Class 2 components. Preservice and ISI requirements associated with the PSCIVs ASME BPVC Class 1 components are in Section 5.2. Preservice and ISI requirements associated with the SSCIVs and MS piping meet the applicable inspection requirements of Reference 6.2-5. The SSCIVs and MS piping components design ensures performance of the ISI requirements of Reference 6.2-5, including the preservice inspections of ASME BPVC Section III.

The periodic inspections and testing programs meet ASME BPVC and Operations and Management (OM) Codes in accordance with 10 CFR 50.55a.

6.2.4.4.1 Initial Functional Testing

A description of initial test programs, including tests for the CIVs and barriers, is in Chapter 14.

CIVs allow testing through the entire sequence initiated by a containment isolation signal. The CIVs close within ten seconds from the time when the power is removed from the actuator.

6.2.4.4.2 Periodic Operability Testing

Section 6.2.6 describes the establishment and verification of the leak-tight integrity of the CNV and the isolation valves and barriers. Additional information addressing the IST of containment isolation components is in Section 3.9.6.3.

With exception to the MSIV and MSIBV, which utilize a test connection, each CIV has a CITF that is used to facilitate Appendix J, Type C, leak rate testing. Section 6.2.6 and Section 6.6 contain details on testing features that provide for containment leak rate tests according to 10 CFR 50, Appendix J, for the CNV penetrations.

Plant technical specifications specify the periodic leak rate testing and inspection and surveillance testing requirements for the CNV and the CIVs and barriers.

6.2.4.5 Instrumentation and Control

Instrumentation and controls covering the anticipated range of normal operation, anticipated operational occurrences, and accident conditions for the variables that could affect containment isolation and associated systems ensure adequate safety. The protection system senses operating conditions and automatically initiates the actuation of the CIVs and other components or systems needed for accident mitigation. Section 6.2.1.7 provides details on containment instrumentation and their functions. The I&C system provides manual control capability to mitigate the consequences of faulted conditions at the division level.

The design integrates the safety system parameters and capabilities into the control room design and displays. The CIV position indication is in the main control room. Two redundant MPS divisions that transmit the system information to the safety display and indication system displays provide control room indication. The safety display and indication system provides the continuous indication of system status and supports manual initiation of protective actions, if required.

To satisfy single failure requirements, containment isolation signals actuate by either of two redundant hardwired switches located in the main control room for each module, one per ESFAS signal division. Using two-out-of-four logic, each division actuates one of a pair of redundant isolation valves. The ESFAS signal generates a containment isolation signal on low and low-low pressurizer level, high narrow range containment pressure, low AC voltage, and high under-the-bioshield temperature.

When plant conditions allow, the operator can manually reset the division level actuation signal and reposition the isolation valves. Operator action occurs only after the equipment has fully actuated to the required position by the ESF safety function.

Section 7.1 contains further details on the fundamental design principles of I&C, and Section 7.2.12 for details on automatic and manual control.

6.2.5 Combustible Gas Control in the Containment Vessel

The NPM design controls combustible gases to prevent hydrogen combustion inside containment following a severe accident. The combustible gas control requirements for future water-cooled reactor designs that have a potential for the production of combustible gases comparable to the light water reactor designs licensed as of October 16, 2003 are in 10 CFR 50.44(c).

6.2.5.1 Design Bases

In compliance with 10 CFR 50.44(c)(1), the CNV maintains a mixed containment atmosphere during design-basis and significant BDBE. Adequate mixing of the CNV occurs by virtue of temperature differences between the annular and head regions of the CNV and its partially immersed design with no sub-compartments that could facilitate separation, coupled with the dynamic nature of events associated with RCS discharge to the CNV (e.g., LOCA or inadvertent ECCS valve opening events).

The design includes a passive autocatalytic recombiner (PAR) that is non-safety related, seismic Class 2 with augmented requirements. The PAR is designed to survive severe accident conditions and the environment in which the PAR is relied upon to function. The PAR is sized to limit oxygen concentrations to a level that does not support combustion (less than four percent). This results in an inert containment atmosphere, thereby satisfying 10 CFR 50.44(c)(2) and 10 CFR 50.44(c)(3).

The design supports an exemption from the 10 CFR 50.44(c)(4) requirements for monitoring combustible gases during an accident.

The NPM relies on a PAR to maintain the containment atmosphere inert through the continuous consumption of oxygen generated post-accident.

Following a BDBA, the containment is oxygen-limited. The sources of oxygen are from the initial quantities in the reactor coolant system controlled by the Primary Coolant Chemistry Program and through radiolytic decomposition of water. Inerting is accomplished solely by the PAR recombining oxygen; no inert gas is added to the containment during operations or post-accident. The PAR has adequate capacity to maintain the containment oxygen concentration below four percent by volume.

The design does not rely on hydrogen monitoring to assess core damage. The radiation monitors under the bioshield and core exit thermocouples provide the ability to assess core damage.

The design relies on the PAR to maintain an inert containment atmosphere following a severe accident, therefore an analysis of the effects of combustion on containment integrity is not necessary. The PAR is a reliable passive device that self-actuates to recombine oxygen and hydrogen present in the surrounding environment. The NPM is not susceptible to de-inerting. The PAR is designed to function in the severe accident environment for which it is intended. Section 19.2 evaluates a bounding BDBE case that produces more hydrogen than the 100 percent clad water reaction would and determines that the CNV does not exceed its design pressure. Therefore, the design conforms to the requirements of 10 CFR 50.44(c)(5).

The design does not require compliance with 10 CFR 50.34(f)(3)(v)(A)(1). 10 CFR 50.34 states that applicants for design approval under Part 52 need not demonstrate compliance with paragraph (f)(3)(v).

The PAR maintains the containment inert post-accident. The systems and components within the CNV that establish and maintain safe shutdown or support containment structural integrity remain capable of performing their required functions after BDBEs.

Section 6.3 addresses hydrogen generation criteria associated with the ECCS performance criteria requirements of 10 CFR 50.46.

Consistent with GDC 5, the design relies on passive control of combustible gases that does not involve sharing between NPMs.

The PAR maintains the containment inert post-accident. Implementation of the requirements of 10 CFR 50.44, as modified by an exemption, meets the requirement of PDC 41 to provide systems to control, as necessary, the concentration of hydrogen and oxygen to ensure containment integrity.

Section 1.9 addresses compliance with guidance in RG 1.7.

6.2.5.2 System Design

The CNV is a metal containment, Class MC pressure vessel that undergoes design, analysis, fabrication, inspection, testing, and stamping as an ASME BPVC Class 1 pressure vessel maintained partially immersed in a reactor pool common to other NPMs.

The CNV meets 10 CFR 50.44(c) by safely accommodating the hydrogen generated by the equivalent of up to a 100 percent fuel-cladding metal water reaction. This type of accident is a BDBE in which hydrogen generation could exceed the flammability limits. The CNV is a passive design that relies on a PAR to maintain a containment atmosphere that does not support combustion following a significant BDBE for combustible gas control.

The CES establishes a partial vacuum in the CNV before NPM startup that continues during reactor operation. The initial CNV pressure contributes to calculations that result in the initial combustible gas composition in the CNV based on the initial CNV pressure. Section 9.3.6 addresses the CES.

When RCS discharge to the containment occurs, the dynamic nature of the event creates a mixed atmosphere because of the induced high turbulent condition. As turbulence subsides later in the event, continued mixing occurs through convection and molecular diffusion. There are no partitions or subcompartments to impede these natural mixing forces. Relevant events ensure convective mixing due to decay heat. Section 6.2.5.3 discusses turbulence in the CNV. The analysis shows that turbulent convective mixing exists in the CNV throughout the first 72 hours of a DBE or BDBE.

The CNV design utilizes a PAR to limit oxygen concentrations to a level that maintains an inerted containment atmosphere following a BDBE that releases an equivalent amount of hydrogen generated from a 100 percent fuel clad-coolant reaction, uniformly distributed. The configuration of the containment coupled with the dynamics of the LOCA and mitigating components ensures adequate mixing within the containment volume during and following events that generate and release combustible gases to containment. Section 6.2.5.3 discusses potential methods of gas accumulation. The limited-oxygen environment and mixed atmosphere maintains an inerted containment atmosphere, thereby precluding combustion that could challenge containment structural integrity.

As described in Section 6.2.5.3, there is margin to the containment pressure capacity limit such that there is no need for containment overpressure protection.

Section 6.2.5.5 addresses combustible gas monitoring.

6.2.5.3 Design Evaluation

The partially immersed design with no sub-compartments that could facilitate separation, coupled with the dynamic nature of events associated with RCS discharge to the CNV (e.g., LOCA or inadvertent ECCS valve opening events) ensure adequate mixing of the CNV. To demonstrate compliance with the 10 CFR 50.44(c) requirement for a well mixed containment, CNV conditions at 72 hours are evaluated. Conditions earlier than 72 hours are generally more turbulent than conditions afterward. The nondimensional Rayleigh (Ra) number, which represents whether the fluid heat transfer is primarily conductive or convective, evaluates mixing. A transition to bulk turbulent conditions occurs in a tall vertical cavity with a hot surface and a cool surface (in air) somewhere between Ra = 10,000 and Ra = 100,000. At 72 hours in the CNV, post-accident Ra exceeds this transition regime by at least one order of magnitude, thereby demonstrating a well mixed volume.

Safety analyses show that the core does not uncover during a design-basis LOCA and as a result there is no fuel damage or fuel clad-coolant reaction that would result in an associated production and release of hydrogen or fission products. The risk-informed revision of 10 CFR 50.44 (68 FR 54125) eliminates the design-basis LOCA hydrogen release from the combustible gas control requirements of 10 CFR 50.44.

An evaluation for the potential for combustible gas (hydrogen and oxygen) accumulation in the containment during and following postulated BDBEs was performed. The evaluation considered those BDBEs an intact containment boundary and resulting in varying degrees of core damage. One example of this type of BDBE is a LOCA inside containment with an ECCS failure that prevents the recirculation of coolant from the CNV back into the RPV. This scenario results in uncovering the reactor core with resulting fuel damage. Uncovering the reactor core can result in the production of a significant amount of hydrogen due to high temperature cladding-fuel interaction with additional amounts of hydrogen and oxygen produced from radiolytic decomposition of the reactor coolant that accumulates within the CNV. The sources of hydrogen in containment following a BDBE are limited to

- oxidation of zirconium in the fuel cladding.
- radiolysis of water (reactor coolant).

- initial amount of dissolved hydrogen in the RCS.
- the amount of hydrogen accumulated in the upper region of the RPV (i.e., the pressurizer).

Within the CNV, the design restricts materials that have the potential to yield hydrogen gas because of contact with liquid contents in the CNV (upon ECCS actuation or other condition involving liquid in containment). Section 6.1 identifies any such materials.

Following a BDBE that releases an equivalent amount of hydrogen as would be generated from a 100 percent fuel clad-coolant reaction, the PAR is sized to maintain oxygen at a level (less than four percent) that does not support hydrogen combustion. Therefore, there is no hydrogen combustion, ensuring CNV integrity.

6.2.5.4 Inspection and Testing

Section 3.8.2.7, Section 6.2.1, Section 6.2.2, Section 6.2.4, Section 6.2.6, Section 6.2.7, Section 6.6, and Section 14.2 describes inspection and testing of the CNV and its components.

6.2.5.5 Instrumentation

Hydrogen and oxygen analyzers are within the containment sampling system portion of the process sampling system. During normal operation, the containment gas discharge from the CES vacuum pumps routes to the containment sampling system sample panel for online analysis of hydrogen and oxygen concentrations with indication in the main control room.

The CES isolates during DBAs and BDBEs. Because the design precludes a combustible atmosphere, monitoring of hydrogen concentration following a significant BDBE is not necessary for successful combustible gas control in the CNV.

6.2.6 Containment Leakage Testing

Provisions for containment leakage rate testing (CLRT) permit verification of the leak-tight integrity of the reactor containment. The CIVs on CNV piping penetrations and the passive containment isolation barriers permit the periodic leakage testing described in GDC 53 and GDC 54 ensure that leakage through the CNTS and components does not exceed the allowable leakage rate specified in Technical Specifications. Section 3.1 further describes compliance with GDC 52, GDC 53, and GDC 54. The design supports an exemption from the GDC 52 requirement to design the containment for integrated leak rate testing, as well as from the 10 CFR 50 Appendix J requirement for preoperational and periodic Type A integrated leak rate testing. Reference 6.2-4 provides further details.

The design of containment penetrations supports performance of local leak rate tests (Type B and Type C tests) in accordance with the guidance provided in ANSI/ANS 56.8, RG 1.163, and NEI 94-01, Rev. 3-A. The CLRT is specific to each

NPM. The design, in conformance with 10 CFR 50.54(o), accommodates the 10 CFR 50 Appendix J, test frequencies of Option A or Option B, with the exception of a containment integrated leak rate test (Type A test as defined by 10 CFR 50 Appendix J, Sections II.F and III.A).

Type B and C tests occur before initial entry into Mode 4. Subsequent periodic Type B and C tests occur at a baseline frequency of at least once per 30 months until establishment of acceptable performance in accordance with NEI 94-01, Rev 3-A. Under Option B of 10 CFR 50 Appendix J, extended test intervals are possible for Type B and C penetrations beyond the baseline frequency following the completion of two consecutive periodic as-found Type B tests, if the results of each test are within allowable administrative limits. Elapsed time between the first and last tests in a series of consecutive satisfactory performance tests shall be 24 months or the nominal test interval (e.g., refueling cycle) for the component before implementing Option B of 10 CFR Part 50, Appendix J. Test intervals may increase from greater than once per 30 months (the baseline frequency) up to a maximum of once per 120 months for Type B penetrations and once per 75 months for Type C penetrations, subject to the requirements of NEI 94-01.

COL Item 6.2-1: An applicant that references the NuScale Power Plant US460 standard design will verify that the final design of the containment vessel meets the design-basis requirement to maintain flange contact pressure at accident temperature, concurrent with peak accident pressure.

As discussed by Section 6.2.6.5.2, a preservice design pressure leakage test, Type B tests and Type C tests verify the leak tightness of the reactor containment before initial operations. Type B and Type C tests occur periodically thereafter, ensuring that leakage rates through the containment, and the systems or components, that penetrate containment, do not exceed the maximum allowable leak rate.

Section 6.2.6.2 and Section 6.2.6.3 discuss Type B and Type C LLRTs. Penetrations are either ASME BPVC Class 1 flanged joints capable of Type B testing or ASME BPVC Class 1 welded nozzles with isolation valves capable of Type C testing. An additional Type B test occurs at the main CNV flange. The pipe penetrations with CIVs, with the exception of the GDC 57 MS and FW lines, are subject to Type C testing (Reference 6.2-4).

Section 6.2.6.5 specifies details on the performance of the CNV preservice design pressure leakage test. Flange preload verifications ensure that flange bolting preloading conforms to design requirements.

Table 6.5-1 lists the specified maximum allowable containment leak rate, L_a, at the calculated peak accident pressure, P_a, identified in Section 6.2.1. Containment leak rate testing verifies that leakage from containment remains within the prescribed Technical Specification limits.

6.2.6.1 Containment Integrated Leakage Rate Test

The design supports an exemption from the GDC 52 requirement to design the containment for integrated leak rate testing, as well as from the 10 CFR 50 Appendix J requirement for preoperational and periodic Type A integrated leak rate testing.

The "NuScale Containment Leakage Integrity Assurance," Technical Report (Reference 6.2-4) provides the technical basis for the GDC 52 exemption. Reference 6.2-4 contains details of the design, testing, and inspection requirements that provide reasonable assurance of continued containment leakage integrity.

6.2.6.2 Containment Penetration Leakage Rate Test

The CNV supports Type B pneumatic tests (local penetration leak tests) for detecting and measuring leakage across pressure-retaining, leakage-limiting boundaries.

Preoperational and periodic Type B leakage rate testing is in accordance with 10 CFR 50, Appendix J, NEI 94-01, ANSI-56.8, and the plant Technical Specifications within the defined test intervals. The containment penetrations subject to Type B tests are in Reference 6.2-4.

The following containment penetrations are subject to preoperational and periodic Type B leakage rate tests:

- flange openings with bolted connections
- main CNV flange
- EPAs
- ISAs
- ECCS trip/reset valve body-to-bonnet connections

CNV bolted closures have dual seals and a testing port between the seals.

CNV flange openings with bolted connections conform to ASME BPVC Class 1. These openings have identical double seals with a test port to facilitate Type B testing by pressurizing between the seals. The main CNV flange has a similar double seal and test port arrangement. Flanges that are underwater during normal operation shall have the capability to evacuate water between the flange seals before performing any Type B test. Flanges have no excess water between the flange seals before startup.

A containment flange bolting calculation demonstrates that each containment flange, at design bolting preload, maintains flange contact pressure at accident temperature, concurrent with peak accident pressure. Maintaining flange contact pressure is defined as a condition where the flange surfaces of the bolted connection have no separation of the flanges from the inboard seal to the containment inner surface when peak accident pressure is applied to the inside of the CNV. Flange contact provides reasonable assurance that the seal would exhibit similar flange gaps at peak accident pressure as would be shown during the Type B test. Therefore, the leakage rate measured during the Type B test would be representative of leakage at peak accident pressure.

The EPAs use an established glass-to-metal sealing technology that is not vulnerable to thermal or radiation aging, do not require periodic maintenance, and will achieve a less than minimum detectable leak rate. These EPAs are in a CNV penetration that includes a testable flange connection. Installed EPAs comply with local leak rate test acceptance criteria. The design includes the ability to test the double O-ring seals by pressurizing between the seals. An EPA would only be disassembled for modification or if leakage was indicated. The ISA is mechanical seal device for the incore instrumentation (ICI) stringer assembly. The ICI stringer assembly is a stainless steel tube containing several mineral insulated cables with imbedded detectors. When fully inserted and installed within the NPM for normal power operations, the ICI stringer assembly will penetrate both the CNV and RPV pressure boundaries. ISA compression seal fittings are used integral with the ICI stringer assembly. The ISA is a Type B testable boundary and complies with local leak rate test acceptance criteria.

There are four ECCS main actuation valves supported by twelve trip and reset valves for actuation. Each actuation valve has redundant testable seals between the valve body and bonnet. A test port between the seals facilitates Type B testing. Because the actuator valve is both a containment and RCS pressure boundary, there will be a seal test to RCS pressure proving that on the ECCS trip/reset actuator valve seals. Section 5.2.4.1 contains the details of the seal test that is performed.

Type B tests involve local pressurization at containment peak accident pressure, P_a, using either the pressure-decay or flowmeter method of detection. For the pressure-decay method, a test volume pressurizes with air or nitrogen to at least P_a. The monitored rate of decay of pressure in the known test volume allows calculation of a leakage rate using the pressure-decay method. For the flowmeter method, makeup air or nitrogen through a calibrated flowmeter maintains the required test pressure in the test volume. The flowmeter fluid flow rate is the leakage rate from the test volume.

In accordance with 10 CFR 50 Appendix J, Type B tests occur during each reactor shutdown for refueling, or other convenient intervals in accordance with the Technical Specification Containment Leakage Rate Testing Program.

6.2.6.3 Containment Isolation Valve Leakage Rate Test

The CNV and CIVs support Type C pneumatic tests. Preoperational and periodic Type C leakage rate testing of CIVs is in accordance with the 10 CFR 50 Appendix J requirements, the Technical Specifications, and ANSI-56.8.

The CIVs subject to Type C tests are in Reference 6.2-4.

Isolation valve tests use either the pressure decay or flowmeter method. For the pressure decay method, the test volume pressurizes with air or nitrogen. The monitored rate of decay of pressure in the known volume facilitates calculation of the leak rate using the pressure decay method. For the flowmeter method, air or nitrogen supplied through a calibrated flowmeter maintains constant test pressure in the test volume. The measured makeup flow rate is the isolation valve leak rate.

Applied pressure to the CIV is in the same direction as pressure applied when the valve performs its safety function.

The CIVs have a similar wedged, quarter-turn ball type design and do not use a seal system.

Each CIV welds directly to a containment isolation test fixture (CITF) valve. The main functions of the CITF valves are to provide isolation to facilitate testing of their respective CIVs (except for the MSIV and MSIBV) for Appendix J and IST testing and to provide an uninterrupted flow path for their respective process fluid during normal operation.

The CITF valves are single top-entry floating ball valves welded directly to the exterior of the CNV penetration nozzles for each of the fluid process piping lines. These are manual valves which are locked open during normal operation and closed to allow testing of their respective CIVs. The CITF valves are the same ASME BPVC class as their respective CIVs. The CITF cover seal is a passive isolation barrier that requires an Appendix J Type B leakage test (Reference 6.2-4).

Each CIV subject to Type C testing closes by normal means without any preliminary exercising or adjustment. Piping is drained and vented as needed, and a test volume, when pressurized, produces a differential pressure across the valve. Preparation of the valve for testing is by removing the cover of the upstream CITF and closing the ball valve, then reinstalling the cover to lock the valve in place. The test port on the CITF establishes test pressure in the same direction as the pressure applied when the valve perform its safety function (i.e., upstream from CNV). Test equipment is on the CITF test port, and system valves align to establish a vent path downstream of the tested valve. The valve undergoes testing via air or nitrogen using either the pressure decay or flow makeup method as specified in the CLRT program.

When testing of the first CIV in the penetration completes, the test equipment vents and the valves realign. The first CIV opens, and the second CIV closes to establish the test alignment for the latter. The test equipment re-pressurizes, and then there is testing of the second valve.

Isolation valves with seat exposure to the containment atmosphere during a LOCA undergo air or nitrogen testing at a pressure not less than P_a.

6.2.6.4 Scheduling and Reporting of Periodic Tests

Schedules for performance of the periodic Type B and C leak rate tests are in the plant Technical Specifications (Section 5.5.9). Provisions for reporting test results are in the CLRT program.

Type B and C tests may be conducted at any time that plant conditions permit, provided that the time between tests for any individual penetration or valve is less than the maximum allowable interval specified in the CLRT program.

Completion of a Type B or C test results in an update of the overall total leakage rate for required Type B and C tests to reflect the most recent test results. Thus, there is a reliable current summary of containment leak tightness. In accordance with 10 CFR 50 Appendix J, the combined leakage rate limit for penetrations and valves subject to Type B and C tests is less than 0.60 L_a.

6.2.6.5 Special Testing Requirements

6.2.6.5.1 Testing Following Major Component Modification or Replacement

Major modifications or replacement of components that are part of the containment boundary performed after preoperational leakage rate testing are followed by a Type B or Type C test, as applicable, for the area affected by the modification.

6.2.6.5.2 Preservice Design Pressure Leakage Test

Each CNV undergoes a preservice design pressure leakage test that tests CNV bolted flange connections under design preload at CNV design pressure. Reference 6.2-4 describes the Preservice Design Pressure Leakage test.

Each ECCS trip valve and reset valve contains a body-to-bonnet joint that is also subject to Type B test requirements. The body-to-bonnet seal is designed for RCS design pressure. This seal is tested to meet both Type B and RCPB criteria every refueling outage. These body-to-bonnet seals are not considered to be a flanged connection and are not included in the containment flange bolting calculation or preservice design pressure leakage test.

6.2.7 Fracture Prevention of Containment Vessel

The CNTS encloses the RPV, providing a CNV pressure boundary and an essentially leak-tight final barrier against the release of radioactive fission products resulting from postulated accidents. Section 3.8.2 addresses design of the steel containment. The upper CNV shell and a portion of the lower shell above the RPV closure flange is composed of a martensitic (ferritic steel), while the remainder of the lower CNV shell is composed of austenitic steels. Table 6.1-1 contains CNV material details.

The CNV system meets the relevant requirements of GDC 1 (Section 3.1.1) with regards to quality standards for design and fabrication, GDC 16 (Section 3.1.2) related to the release of radioactivity, and GDC 51 (Section 3.1.5) related to sufficient design margin for fracture prevention. In meeting the requirements of GDC 1, 16 and 51, the ferritic containment pressure boundary materials satisfy the fracture toughness criteria for ASME BPVC Section III, Subsection NB.

The design, fabrication, and materials of construction for the CNV system includes margin to provide reasonable assurance that the CNV pressure boundary will not undergo brittle fracture, which minimizes the probability of rapidly propagating fracture under operating, maintenance, testing, and postulated accident conditions occurring over its 60-year design life. Section 6.2.1 provides additional detail.

For ferritic materials classified as pressure-retaining components of the RCPB, the requirements of Reference 6.2-5, Appendix G apply.

The ferritic containment pressure boundary conforms to ASME BPVC, Section II material specifications and meets the fracture toughness criteria and testing requirements identified in ASME BPVC Section III, Division 1, NB-2300.

Portions of the lower CNV have 60-year design fluence in excess of 1E+17 neutrons/cm^2, E > 1 MeV, with the peak fluence in the lower CNV not exceeding 2.5E+18 neutrons/cm^2, E > 1 MeV. The portions of the lower CNV with peak neutron fluence greater that 1E+17 neutrons/cm^2, E > 1 MeV, are composed of austenitic stainless steel. Austenitic stainless steels have superior ductility and are less susceptible to the effects of neutron embrittlement than ferritic materials. The peak neutron fluence for the ferritic portion of the CNV is less than the regulatory limit of 1E+17 neutrons/cm^2, E > 1 MeV. The material selection for the CNV pressure boundary ensures facture prevention.

6.2.8 References

6.2-1 NuScale Power, LLC, "Loss-of-Coolant Accident Evaluation Model," TR-0516-49422-P, Revision 3.

6.2-2 NuScale Power, LLC, "Non-Loss-of-Coolant Accident Analysis Methodology Report," TR-0516-49416-P-A, Revision 3.

6.2-3 NuScale Power LLC, “Extended Passive Cooling and Reactivity Control Methodology Topical Report” TR-124587, Revision 0.

6.2-4 NuScale Power, LLC, "NuScale Containment Leakage Integrity Assurance," TR-123952-P, Rev. 0.

6.2-5 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2017 edition, Section XI Division 1, "Rules for Inservice Inspection of Nuclear Components," New York, NY.

Table 6.2-1: Containment Design and Operating Parameters

Containment Parameter	Value
Design Conditions	
• Internal Design Pressure	1200 psia*
• External Design Pressure	50 psia
• Design Temperature	600 °F
• Design Maximum Containment Leakage	Table 6.5-1
• UHS Pool Water Temperature (Max Analysis)	212 °F (boiling)
Normal Operating Conditions (nominal)	
• Internal CNV Pressure	less than 1 psia
• External CNV Pressure	< 45 psia
• UHS Pool Water Level (Figure 9.1.3-2)	Table 9.2.5-1
• UHS Pool Water Volume - from normal operating level to top of weir (values listed in Table 9.2.5-1) as shown in Figure 9.1.3-2	Section 9.2.5.3
• UHS Pool Water (Avg) Temperature	Table 9.2.5-1
• Minimum allowable operating Temperature	65°F

* Hydrostatically tested at 1500 psig (at highest point of test boundary)/70°F (minimum) to 140°F (maximum)

** Additional RXB design temperature information is provided in Section 9.4

Table 6.2-2: Containment Response Analysis Results

Event Description	Case Description	CNV Pressure (psia)	CNV Wall Temperature (°F)
RCS Discharge Line Break	Base Case	910	531
RCS Discharge Line Break	Limiting Sensitivity Case Results	937[1]	533[2]
RCS Injection Line Break	Base Case	890	529
RCS Injection Line Break	Limiting Sensitivity Case Results	909	531
RPV High Point Vent Degasification Line Break	Base Case	807	517
RPV High Point Vent Degasification Line Break	Limiting Sensitivity Case Results	877	525
Inadvertent RVV Actuation	Base Case	822	519
Inadvertent RVV Actuation	Limiting Sensitivity Case Results	876	525
Inadvertent RRV Actuation	Base Case	906	530
Inadvertent RRV Actuation	Limiting Sensitivity Case Results	925	533[2]
Inadvertent ECCS Actuation	Base Case	852	522
Inadvertent ECCS Actuation	Limiting Sensitivity Case Results	889	527
MSLB	Limiting Results	900	530
FWLB	Limiting Results	886	526

[1] Limiting NPM primary/secondary release event peak pressure.

[2] Limiting NPM primary/secondary release event peak temperature.

Table 6.2-3: Containment Penetrations

Containment Penetration	Description	System Quality Group	Appendix J Type B/C Test	Piping Penetration Regulatory Reference/ Compliance
CNV1	FW Line 1	B	TS, DHRS operability, Note 4	Meets GDC 57
CNV2	FW Line 2	B	TS, DHRS operability, Note 4	Meets GDC 57
CNV3	MS Line 1	B	TS, DHRS operability	Meets intent of GDC 57
CNV4	MS Line 2	B	TS, DHRS operability	Meets intent of GDC 57
CNV5	RCCW Return	B	B, C, Note 3	Meets the intent of GDC 56 Exemption justified - Note 2
CNV6	RCS Injection	A	B, C, Note 3	Meets the intent of GDC 55 Exemption justified - Note 2
CNV7	Pressurizer Spray	A	B, C, Note 3	Meets the intent of GDC 55 Exemption justified - Note 2
CNV8	I&C Division 1	B	B	N/A
CNV9	I&C Division 2	B	B	N/A
CNV10	CES	B	B, C, Note 3	Meets the intent of GDC 56 Exemption justified - Note 2
CNV11	Containment Flood and Drain System	B	B, C, Note 3	Meets the intent of GDC 56 Exemption justified - Note 2
CNV12	RCCW Supply	B	B, C, Note 3	Meets the intent of GDC 56 Exemption justified - Note 2
CNV13	CNTS CVC Discharge	A	B, C, Note 3	Meets the intent of GDC 55 Exemption justified - Note 2
CNV14	RPV High Point Degasification	A	B, C, Note 3	Meets the intent of GDC 55 Exemption justified - Note 2
CNV15	Pressurizer Heater Power (Elect-1)	B	B	N/A
CNV16	Pressurizer Heater Power (Elect-2)	B	B	N/A
CNV17	ICI Channel A	B	B	N/A
CNV18	ICI Channel C	B	B	N/A
CNV19	ICI Channel B	B	B	N/A
CNV20	ICI Channel D	B	B	N/A
CNV22	Decay Heat Removal 1	B	N/A	Meets the intent of GDC 57 Exemption justified - Note 1
CNV23	Decay Heat Removal 2	B	N/A	Meets the intent of GDC 57 Exemption justified - Note 1
CNV24	CNV Manway Access Port 1	B	B	N/A
CNV25	CRDM Access Opening	B	B	N/A
CNV26	CNV Manway Access Port 2	B	B	N/A
CNV27	SG Access Port 1	B	B	N/A

Table 6.2-3: Containment Penetrations (Continued)

Containment Penetration	Description	System Quality Group	Appendix J Type B/C Test	Piping Penetration Regulatory Reference/ Compliance
CNV28	SG Access Port 2	B	B	N/A
CNV29	SG Access Port 3	B	B	N/A
CNV30	SG Access Port 4	B	B	N/A
CNV31	Pressurizer Heater Access Port 1	B	B	N/A
CNV32	Pressurizer Heater Access Port 2	B	B	N/A
CNV33	RVV Trip/Reset A	B	B	N/A
CNV34	RVV Trip/Reset B	B	B	N/A
CNV35	RRV Trip/Reset A	B	B	N/A
CNV36	RRV Trip/Reset B	B	B	N/A
CNV37	CRDM Power 1	B	B	N/A
CNV38	RPI Group 1	B	B	N/A
CNV39	RPI Group 2	B	B	N/A
CNV40	I&C Separation Group A	B	B	NA
CNV41	I&C Separation Group B	B	B	NA
CNV42	I&C separation Group C	B	B	NA
CNV43	I&C separation Group D	B	B	NA
CNV44	CRDM Power 2	B	B	N/A
CNV45	Decay Heat Removal 1	B	N/A	Meets the intent of GDC 57 Exemption justified - Note 1
CNV46	Decay Heat Removal 2	B	N/A	Meets the intent of GDC 57 Exemption justified - Note 1
-	Main CNV Flange	B	B	N/A

Notes:
1. Exemption allows the use of a closed piping system (DHRS) outside of the containment vessel rather than providing an isolation valve.
2. Exemption allows placement of both CIVs outside of the containment boundary.
3. The CITF attached to each PSCIV is a passive isolation barrier that requires a 10 CFR Part 50, Appendix J Type B leakage test.
4. The CITF attached to the FWIV is exempt from TS, DHRS operability.

Table 6.2-4: Containment Isolation Valve Information

Containment Isolation Valve[2, 3, 4]	Valve Location Inside/Outside Containment	Penetration	Primary Actuation Mode	Secondary Actuation Mode	Valve Position Norm/ Shutdown	Valve Position Post-Accident/ Power Failure	Power Failure Position Motive/Control
CVC-HOV-0331	Outside Note 1	CNV6	Auto	Remote Manual	Open/Closed	Closed/Closed	Open/Closed
CVC-HOV-0330	Outside Note 1	CNV6	Auto	Remote Manual	Open/Closed	Closed/Closed	Open/Closed
CVC-HOV-0325	Outside Note 1	CNV7	Auto	Remote Manual	Open/Closed	Closed/Closed	Open/Closed
CVC-HOV-0324	Outside Note 1	CNV7	Auto	Remote Manual	Open/Closed	Closed/Closed	Open/Closed
CVC-HOV-0335	Outside Note 1	CNV13	Auto	Remote Manual	Open/Closed	Closed/Closed	Open/Closed
CVC-HOV-0334	Outside Note 1	CNV13	Auto	Remote Manual	Open/Closed	Closed/Closed	Open/Closed
CVC-HOV-0402	Outside Note 1	CNV14	Auto	Remote Manual	Closed/Closed	Closed/Closed	Open/Closed
CVC-HOV-0401	Outside Note 1	CNV14	Auto	Remote Manual	Closed/Closed	Closed/Closed	Open/Closed
CE-HOV-0001	Outside Note 1	CNV10	Auto	Remote Manual	Open/Closed	Closed/Closed	Open/Closed
CE-HOV-0002	Outside Note 1	CNV10	Auto	Remote Manual	Open/Closed	Closed/Closed	Open/Closed
CFD-HOV-0022	Outside Note 1	CNV11	Auto	Remote Manual	Closed/Closed	Closed/Closed	Open/Closed
CFD-HOV-0021	Outside Note 1	CNV11	Auto	Remote Manual	Closed/Closed	Closed/Closed	Open/Closed
RCCW-HOV-0185	Outside Note 1	CNV12	Auto	Remote Manual	Open/Closed	Closed/Closed	Open/Closed
RCCW-HOV-0184	Outside Note 1	CNV12	Auto	Remote Manual	Open/Closed	Closed/Closed	Open/Closed
RCCW-HOV-0190	Outside Note 1	CNV5	Auto	Remote Manual	Open/Closed	Closed/Closed	Open/Closed
RCCW-HOV-0191	Outside Note 1	CNV5	Auto	Remote Manual	Open/Closed	Closed/Closed	Open/Closed
FW-HOV-0137	Outside	CNV1	Auto	Remote Manual	Open/Closed	Closed/Closed	Open/Closed
FW-HOV-0237	Outside	CNV2	Auto	Remote Manual	Open/Closed	Closed/Closed	Open/Closed
MS-HOV-0101	Outside	CNV3	Auto	Remote Manual	Open/Closed	Closed/Closed	Open/Closed
MS-HOV-0103	Outside	CNV3	Remote Manual	Auto	Closed/Closed	Closed/Closed	Open/Closed
MS-HOV-0201	Outside	CNV4	Auto	Remote Manual	Open/Closed	Closed/Closed	Open/Closed
MS-HOV-0203	Outside	CNV4	Remote Manual	Auto	Closed/Closed	Closed/Closed	Open/Closed

Notes:
1. Exemption as discussed by Section 6.2.4.1 allows placement of both CIVs outside of the containment boundary.
2. Each ball valve CIV closes using a stored energy device within 10 seconds. The closure time is the total required isolation time from receipt of a closure signal generated by the MPS to the time when the valve is fully closed. Isolation signal delays are listed by Table 7.1-6.
3. Each CIV closes on a containment isolation system signal. Table 7.1-4 identifies the parameters monitored by the MPS to generate a containment isolation system signal.
4. All with the exception of the CE CIVs have an internal design pressure of at least 2100 psia and a design temperature of 650 °F, the CE CIVs have an internal design pressure of at least 1200 psia and a design temperature of 600 °F.

Table 6.2-5: Failure Modes and Effects Analysis Containment System

Component Identification	Function	Failure Mode	Failure Mechanism	Effect on CNTS Effect on Interfacing System	Method of Failure Detection	Remarks
Isolation Valves: RCS Discharge CVC-HOV-0334 CVC-HOV-0335 RCS Injection CVC-HOV-0331 CVC-HOV-0330 PZR Spray Supply CVC-HOV-0325 CVC-HOV-0324 RPV High Point Degasification CVC-HOV-0401 CVC-HOV-0402	1) Isolate the CVCS lines from the CNV to maintain containment integrity	A) Spurious closure	Electrical/I&C Operator error Mechanical	No adverse effect on the CNTS	CVCS pump trip Low flow in affected line(s) Valve position indication	Valve open signals are routed from a nonsafety control system through the MPS to a single valve at a time. Nonsafety control must first be enabled before an operator can remotely reopen a CIV, and nonsafety control is blocked under certain conditions (e.g., if conditions requiring containment isolation are reestablished), reducing the likelihood an operator will erroneously open a CIV.
		B) Spurious opening when containment is isolated C) Failure to close or partial closure D) Valve seat leakage	Electrical/I&C Operator error Mechanical	No adverse effect on the CNTS - redundant CIV is closed.	Valve position indication Leak rate testing	
CES Isolation Valves CE-HOV-0001 CE-HOV-0002	1) Isolate the CES lines from the CNV to maintain containment integrity	A) Spurious closure	Electrical/I&C Operator error Mechanical	No adverse effect on the CNTS.	Valve position indication Containment pressure	
		B) Spurious opening when containment is isolated C) Failure to close or partial closure D) Valve seat leakage	Electrical/I&C Operator error Mechanical	No adverse effect on the CNTS - redundant CIV is closed.	Valve position indication Leak rate testing	Valve open signals are routed from a nonsafety control system through the MPS to a single valve at a time. Nonsafety control must first be enabled before an operator can remotely reopen a CIV, and nonsafety control is blocked under certain conditions (e.g., if conditions requiring containment isolation are reestablished), reducing the likelihood an operator will erroneously open a CIV.
CFD Isolation Valves CFD-HOV-0021 CFD-HOV-0022	1) Isolate the CFDS lines from the CNV to maintain containment integrity	A) Spurious closure	Electrical/I&C Operator error Mechanical	No adverse effect on the CNTS.	Valve position indication CFDS line pressure or flow indication	Valve open signals are routed from a nonsafety control system through the MPS to a single valve at a time. Nonsafety control must first be enabled before an operator can remotely reopen a CIV, and nonsafety control is blocked under certain conditions (e.g., if conditions requiring containment isolation are reestablished), reducing the likelihood an operator will erroneously open a CIV.
		B) Spurious opening when containment is isolated C) Failure to close or partial closure D) Valve seat leakage	Electrical/I&C Operator error Mechanical	No adverse effect on the CNTS - redundant CIV is closed.	Valve position indication Leak rate testing	
RCCWS Supply Isolation Valves RCCW-HOV-0184 RCCW-HOV-0185 RCCWS Return Isolation Valves RCCW-HOV-0190 RCCW-HOV-0191	1) Isolate the RCCWS lines from the CNV to maintain containment integrity	A) Spurious closure	Electrical/I&C Operator error Mechanical	No adverse effect on the CNTS.	Valve position indication RCCWS line flow indication CRDM temperature indication	Valve open signals are routed from a nonsafety control system through the MPS to a single valve at a time. Nonsafety control must first be enabled before an operator can remotely reopen a CIV, and nonsafety control is blocked under certain conditions (e.g., if conditions requiring containment isolation are reestablished), reducing the likelihood an operator will erroneously open a CIV.
		B) Spurious opening when containment is isolated C) Failure to close or partial closure D) Valve seat leakage	Electrical/I&C Operator error Mechanical	No adverse effect on the CNTS - redundant CIV is closed.	Valve position indication Leak rate testing	

Table 6.2-5: Failure Modes and Effects Analysis Containment System (Continued)

Component Identification	Function	Failure Mode	Failure Mechanism	Effect on CNTS Effect on Interfacing System	Method of Failure Detection	Remarks
FWS Isolation Valves FW-HOV-0137 FW-HOV-0237	1) Isolate the FWS lines from the CNV to maintain containment integrity	A) Spurious closure	Electrical/I&C Operator error Mechanical	No adverse effect on the CNTS.	Valve position indication FWS, MSS, or RCS pressure or flow indication	Nonsafety backup feed water isolation valves are located in the RXB near the CNV. One secondary FWIV is provided for each of the two feed water lines exiting the CNV and downstream of the FWIV. The secondary isolation valves are nonsafety-related. The actuators for the secondary FWIVs close on a loss of motive force to the actuator or a loss of the control signal. The secondary FWIVs are remotely operated valves that serve to isolate the portion of the FWS outside the CNV.
		B) Spurious opening when containment is isolated C) Failure to close or partial closure D) Valve seat leakage	Electrical/I&C Operator error Mechanical	No adverse effect on the CNTS - The pressure boundary of the secondary side maintains containment integrity.	Valve position indication Leak rate testing	
MSS Isolation Valves MS-HOV-0101 MS-HOV-0201	1) Isolate the MSS lines from the CNV to maintain containment integrity	A) Spurious closure	Electrical/I&C Operator error Mechanical	No adverse effect on the CNTS.	Valve position indication FWS, MSS, or RCS pressure or flow indication	Nonsafety secondary MSIVs are located in the RXB near the CNV. One secondary MSIV is provided for each of the two steam lines exiting the CNV and downstream of the MSIV. The secondary isolation valves are nonsafety-related, however, they will serve as backup if the MSIVs fail. The actuators for the secondary MSIVs close on a loss of motive force to the actuator or a loss of the control signal. The secondary MSIVs are remotely operated valves that isolate the portion of the MSS outside the CNV.
		B) Spurious opening when containment is isolated C) Failure to close or partial closure D) Valve seat leakage	Electrical/I&C Operator error Mechanical	No adverse effect on the CNTS - The pressure boundary of the secondary side maintains containment integrity. No adverse effect on the DHRS- redundant train of DHRS and SG performs cooling function.	Valve position indication Leak rate testing	
MSS Bypass Valves MS-HOV-0103 MS-HOV-0203	1) Isolate the MSS lines from the CNV to maintain containment integrity	A) Spurious closure	Electrical/I&C Operator error	No adverse effect on the CNTS.	Valve position indication FWS, MSS, or RCS pressure or flow indication	
		B) Spurious opening when containment is isolated C) Failure to close or partial closure D) Valve seat leakage	Electrical/I&C Operator error Mechanical	No adverse effect on the CNTS - The pressure boundary of the secondary side maintains containment integrity.	Valve position indication Leak rate testing	

Table 6.2-6: Sequence of Events - Peak Containment Vessel Pressure and Temperature Case

Event	Time (s)
Transient initiated by a RCS discharge line break into the CNV	0
High CNV pressure signal is reached	0.7
Reactor trip DHRS actuation and CNV isolation	2.7
Low RCS level signal is reached	87
ECCS valve opening	147
Peak CNV wall temperature is reached (for peak pressure case)	168
Peak CNV internal pressure is reached (937 psia)	170
CNV pressure drops to less than 50% of peak pressure	Approximately 3200

Figure 6.2-1: Containment System

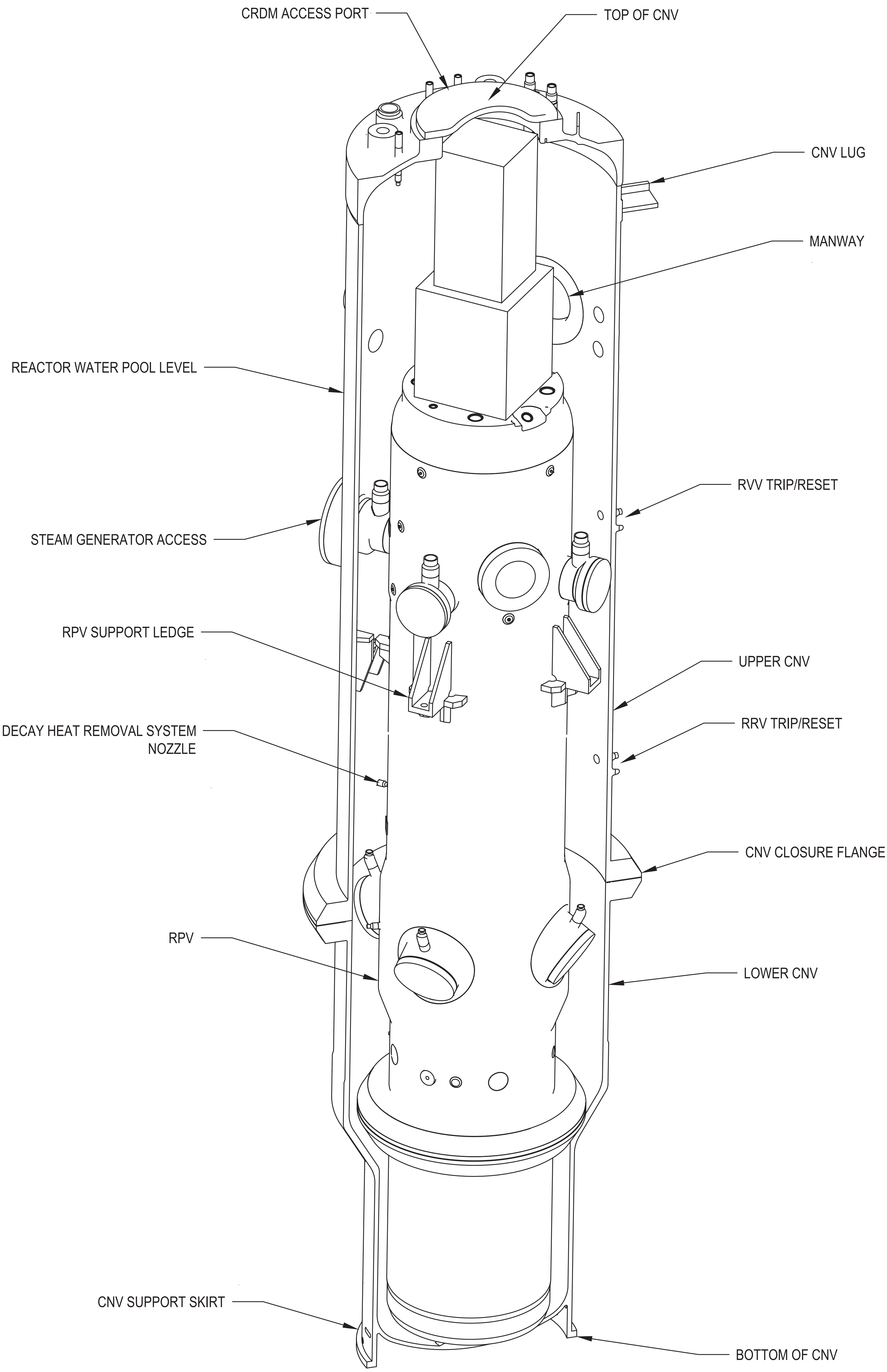
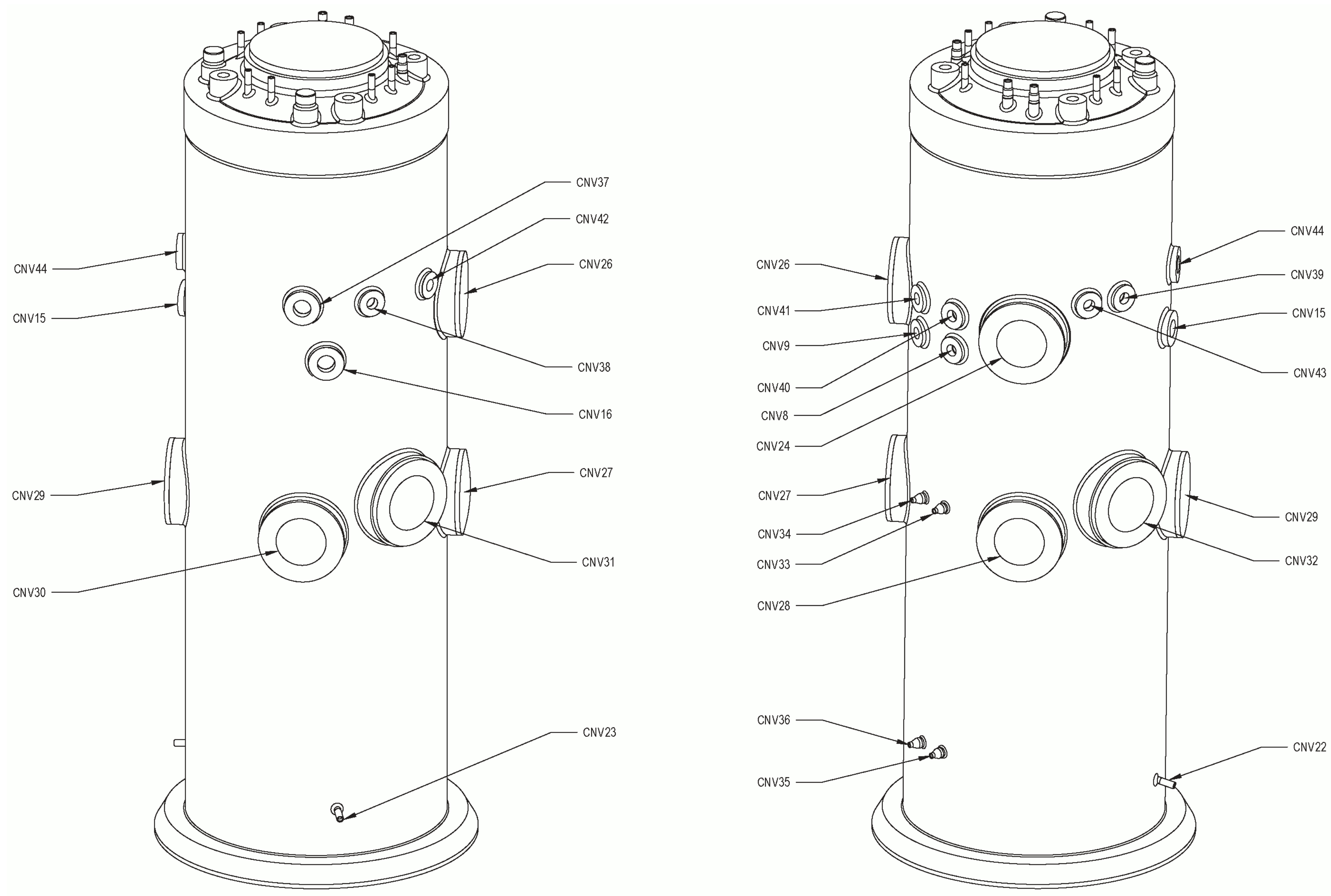
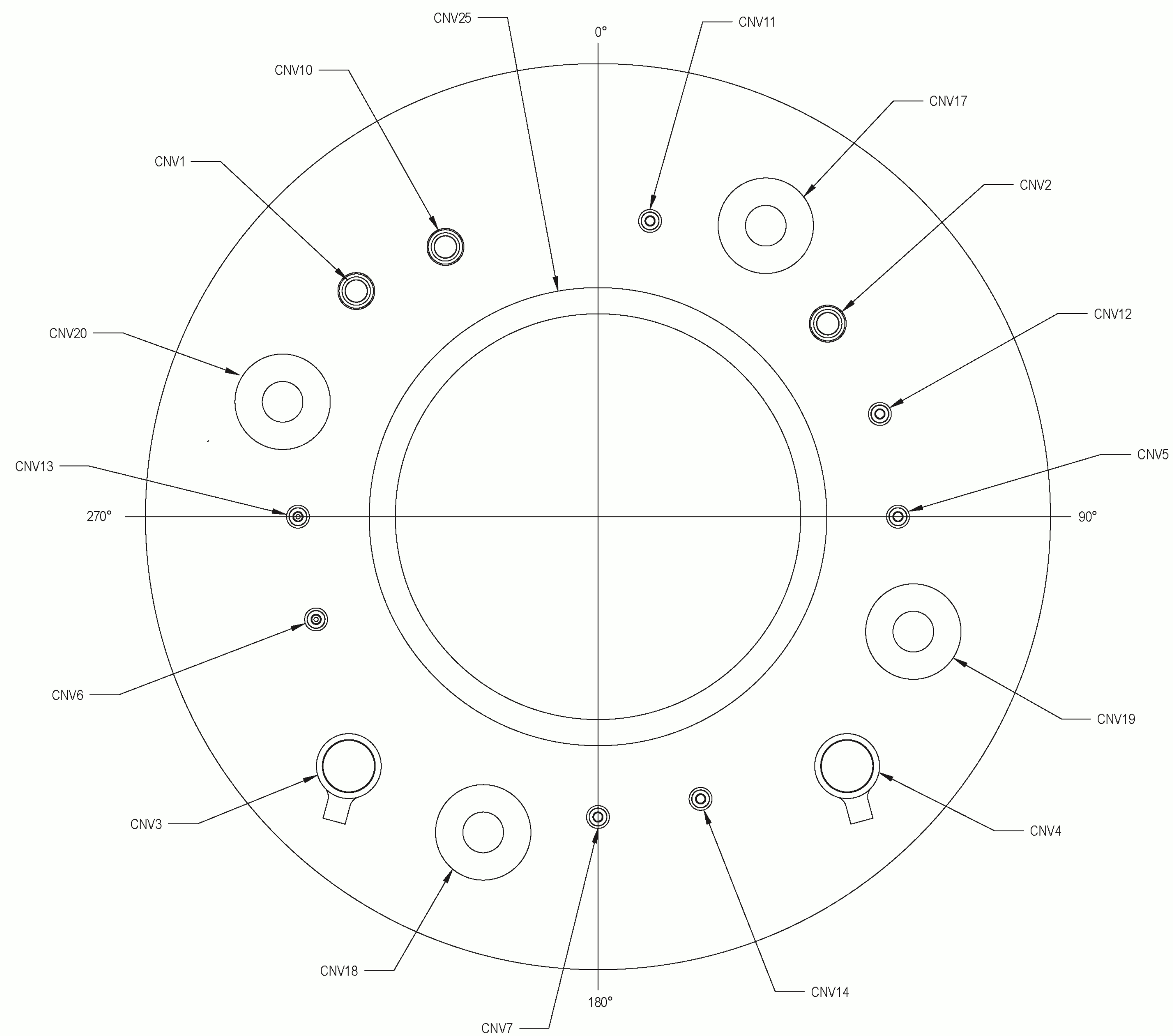


Figure 6.2-2a: Containment Vessel Assembly



CNV NOZZLE No.	DESCRIPTION
CNV8	I&C - DIVISION 1
CNV9	I&C - DIVISION 2
CNV15	PZR HEATER POWER - 1
CNV16	PZR HEATER POWER - 2
CNV22	DHRS RETURN LINE 1
CNV23	DHRS RETURN LINE 2
CNV24	CNV MANWAY ACCESS PORT 1
CNV26	CNV MANWAY ACCESS PORT 2
CNV27	SG ACCESS PORT 1
CNV28	SG ACCESS PORT 2
CNV29	SG ACCESS PORT 3
CNV30	SG ACCESS PORT 4
CNV31	PZR HEATER ACCESS PORT 1
CNV32	PZR HEATER ACCESS PORT 2
CNV33	TRIP/RESET ASSEMBLY - RVV A
CNV34	TRIP/RESET ASSEMBLY - RVV B
CNV35	TRIP/RESET ASSEMBLY - RRV A
CNV36	TRIP/RESET ASSEMBLY - RRV B
CNV37	CRDM POWER 1
CNV38	RPI GROUP 1
CNV39	RPI GROUP 2
CNV40	I&C SEPARATION GROUP A
CNV41	I&C SEPARATION GROUP B
CNV42	I&C SEPARATION GROUP C
CNV43	I&C SEPARATION GROUP D
CNV44	CRDM POWER 2

Figure 6.2-2b: Containment Vessel Assembly



CNV NOZZLE No.	DESCRIPTION
CNV1	FEEDWATER LINE 1
CNV2	FEEDWATER LINE 2
CNV3	MAIN STEAM 1
CNV4	MAIN STEAM 2
CNV5	RCCW RETURN
CNV6	CVC INJECTION
CNV7	CVC PRESSURIZER SPRAY
CNV10	CONTAINMENT EVACUATION
CNV11	CONTAINMENT FLOODING & DRAINING
CNV12	RCCW SUPPLY
CNV13	CVC DISCHARGE
CNV14	RPV HIGH POINT DEGASIFICATION
CNV17	ICI - CHANNEL A
CNV18	ICI - CHANNEL B
CNV19	ICI - CHANNEL C
CNV20	ICI - CHANNEL D
CNV25	CRDM ACCESS PORT

Figure 6.2-3: Containment System Piping and Instrumentation Diagram

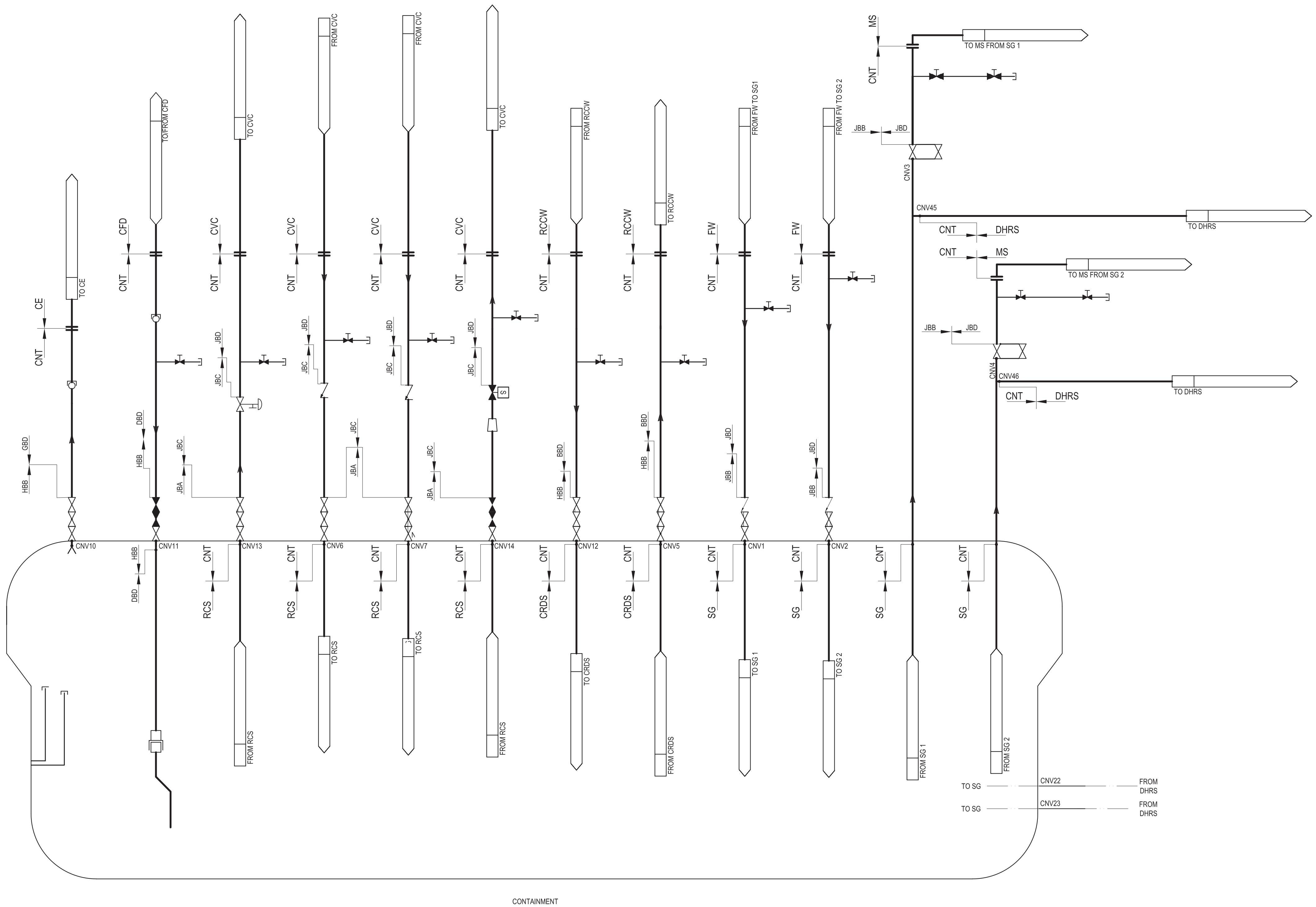


Figure 6.2-4: Primary System Containment Isolation Valves Dual Valve, Single Body Design

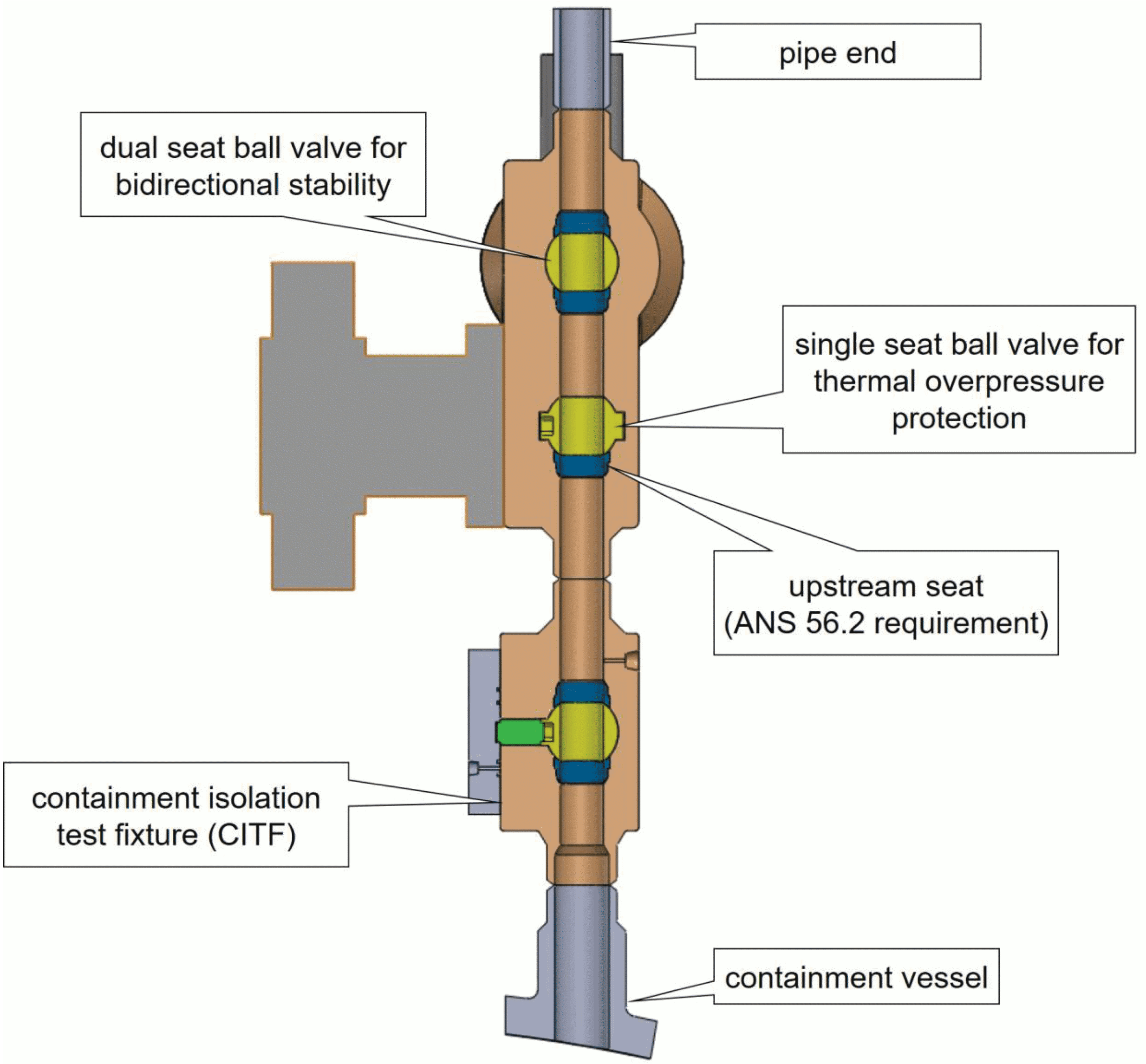


Figure 6.2-5a: Main Steam Isolation Valve with Bypass Valve and Actuator Assembly

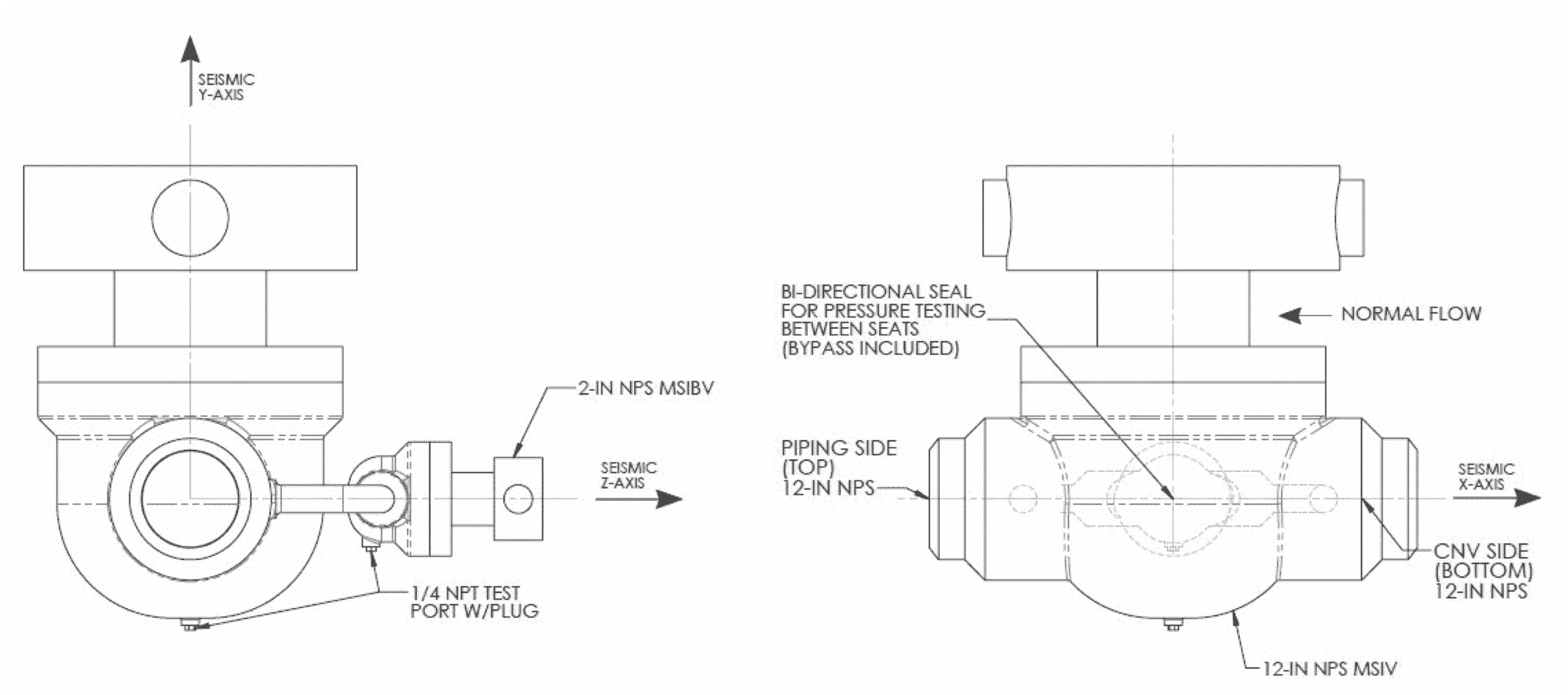


Figure 6.2-5b: Feedwater Isolation Valve with Nozzle Check Valve and Actuator Assembly

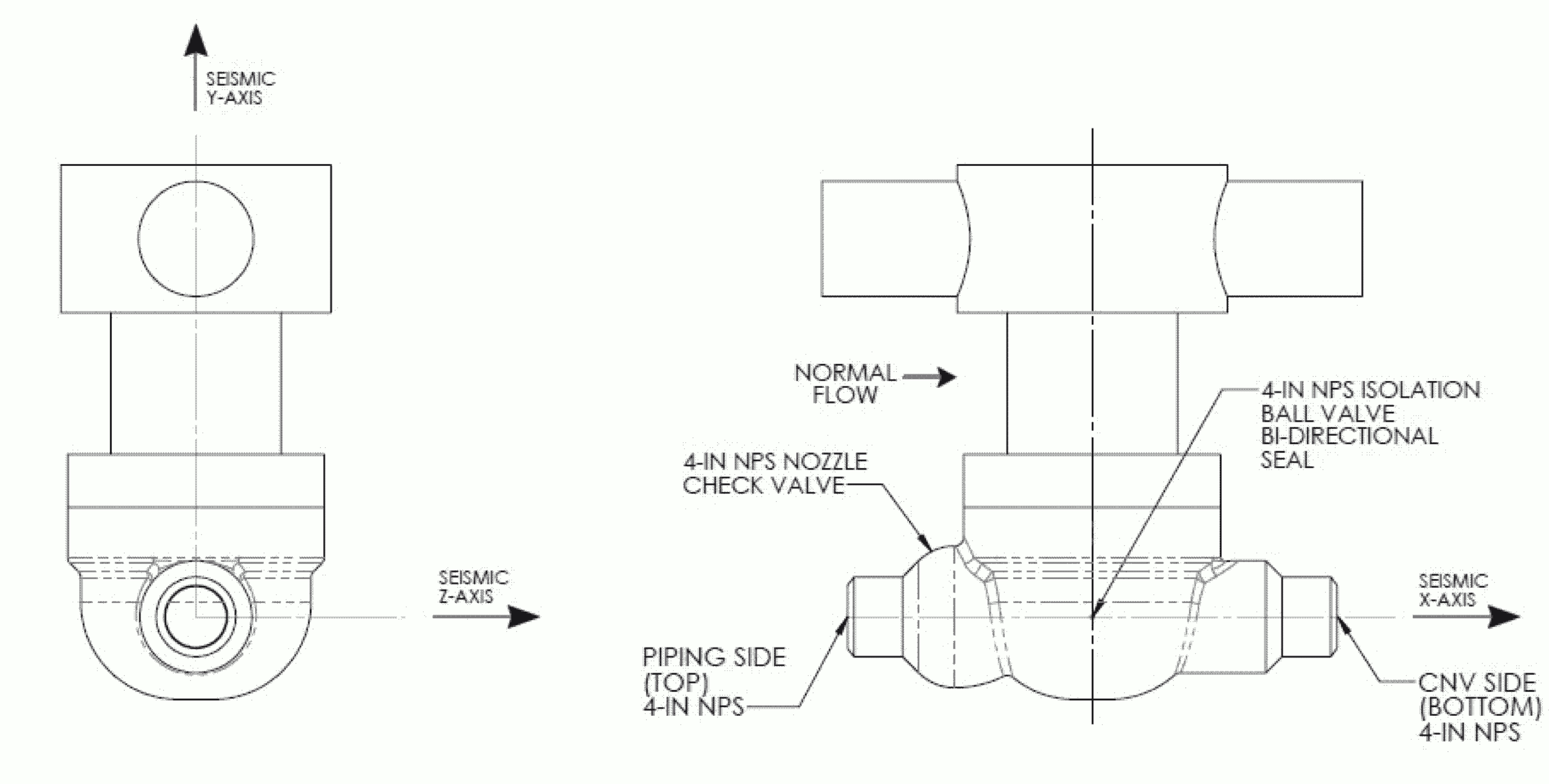


Figure 6.2-6: Containment Isolation Valve Actuator Hydraulic Schematic

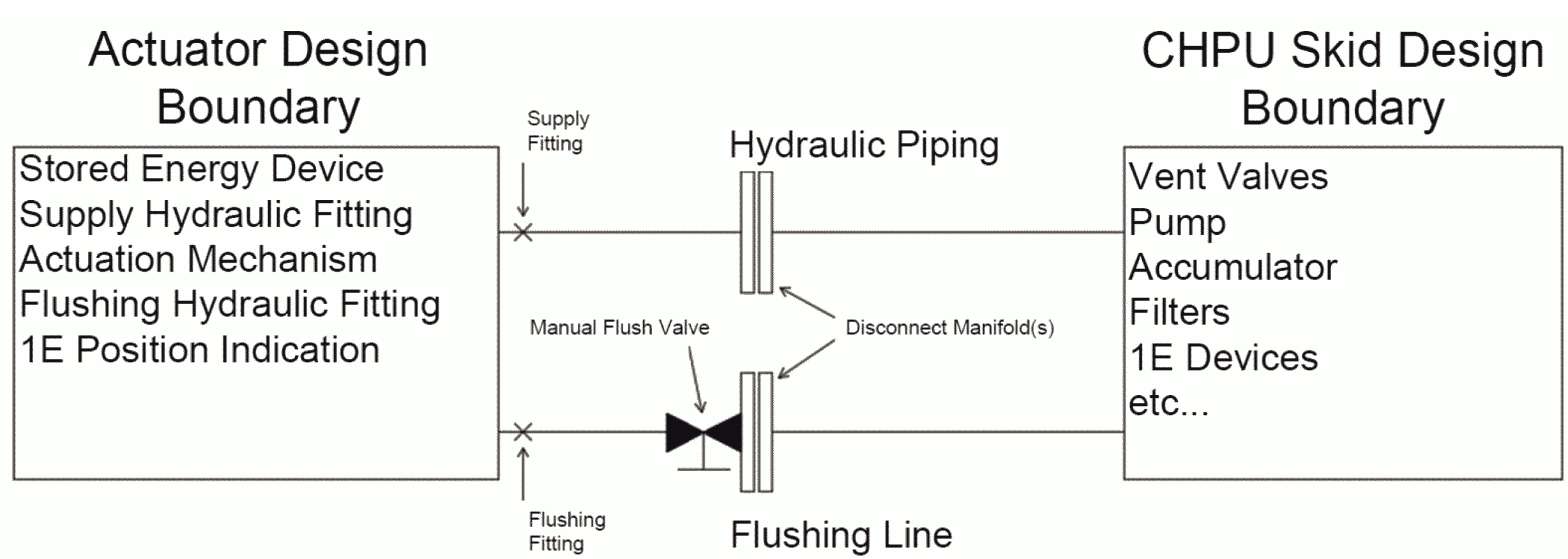


Figure 6.2-7: Maximum Containment Internal Pressure - Peak Containment Vessel Pressure Case

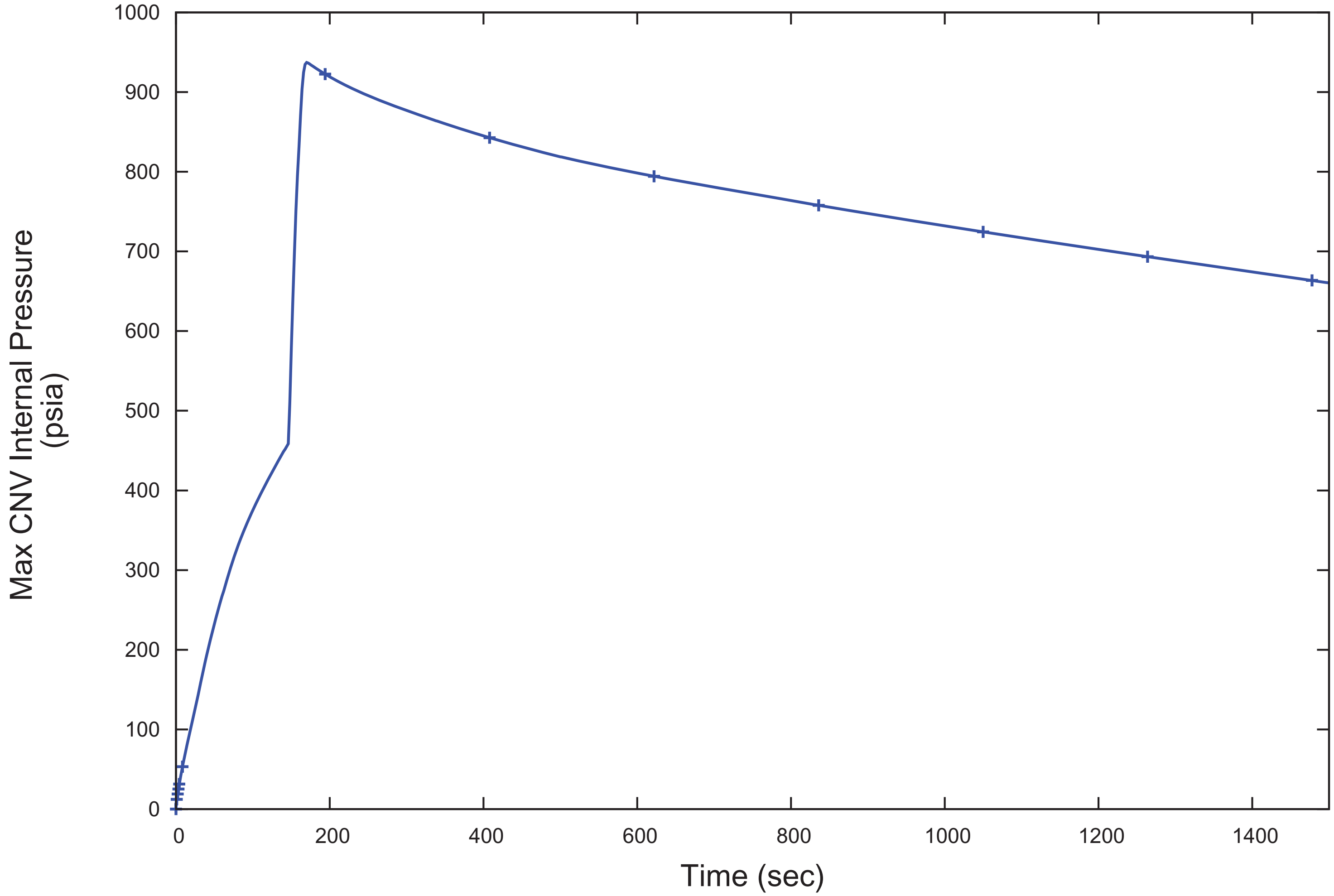


Figure 6.2-8: Integrated Break and Emergency Core Cooling System Mass Release Rate - Peak Containment Vessel Pressure Case

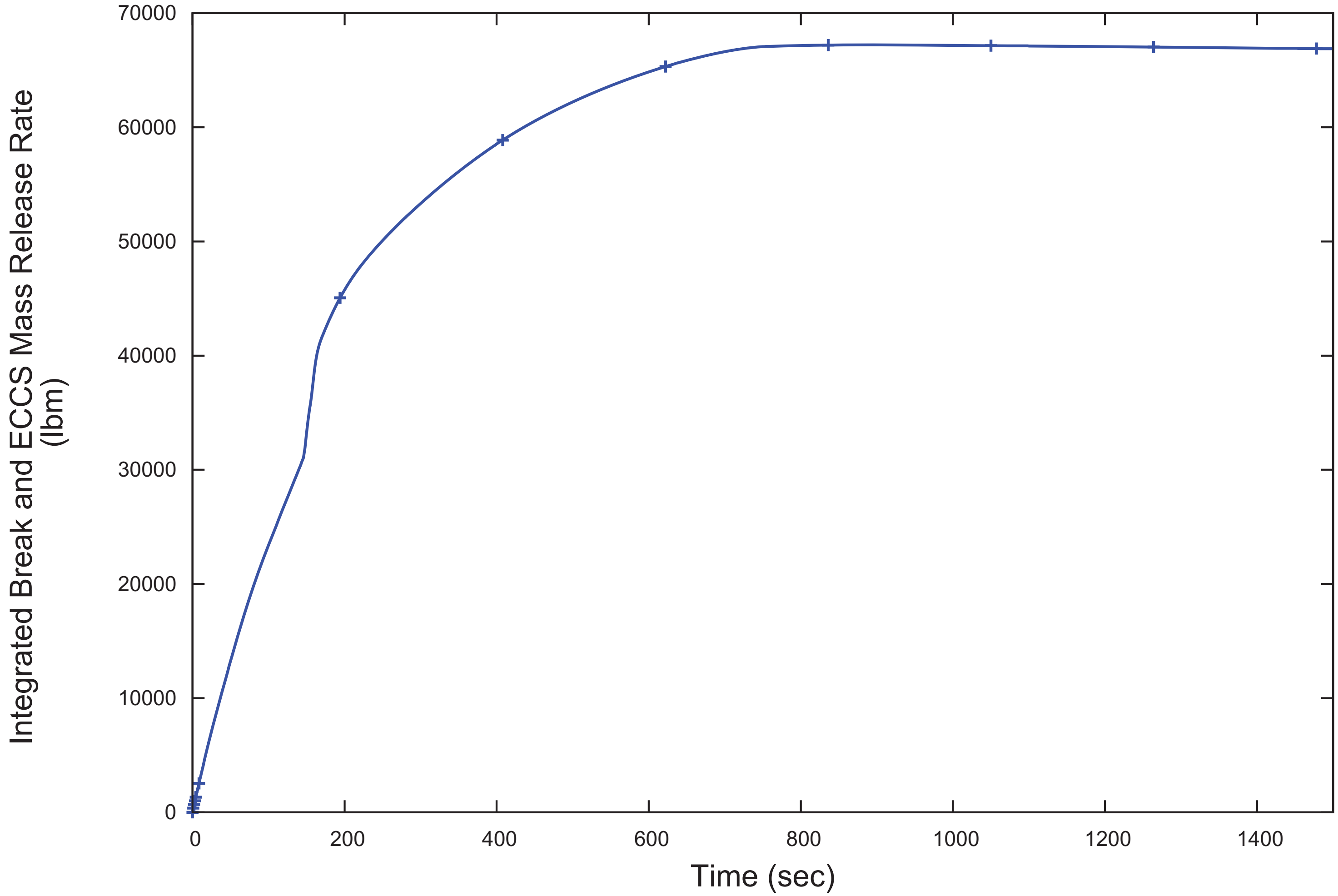


Figure 6.2-9: Integrated Break and Emergency Core Cooling System Energy Release Rate - Peak Containment Vessel Pressure Case

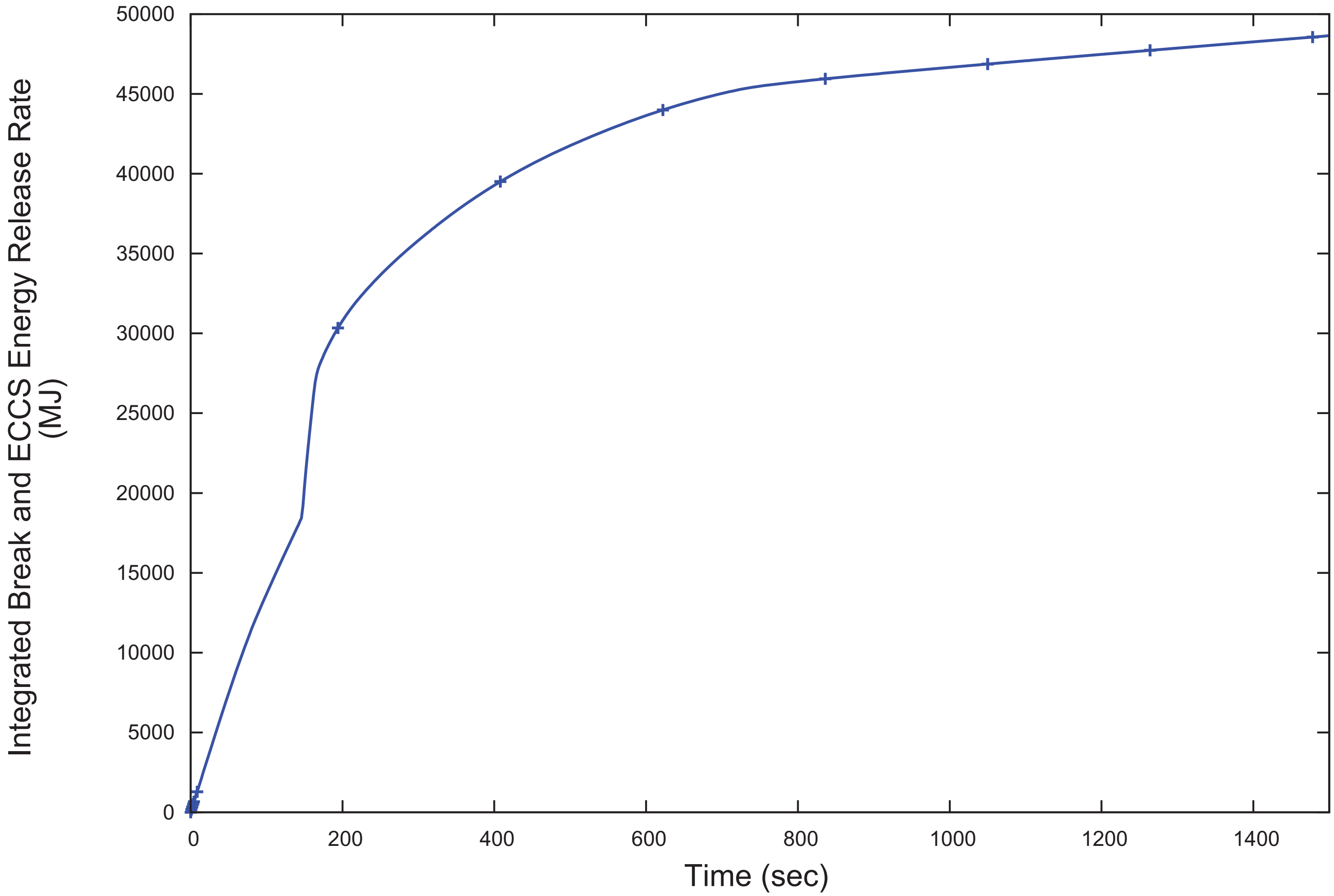


Figure 6.2-10: Maximum Containment Wall Temperature - Peak Containment Vessel Temperature Case

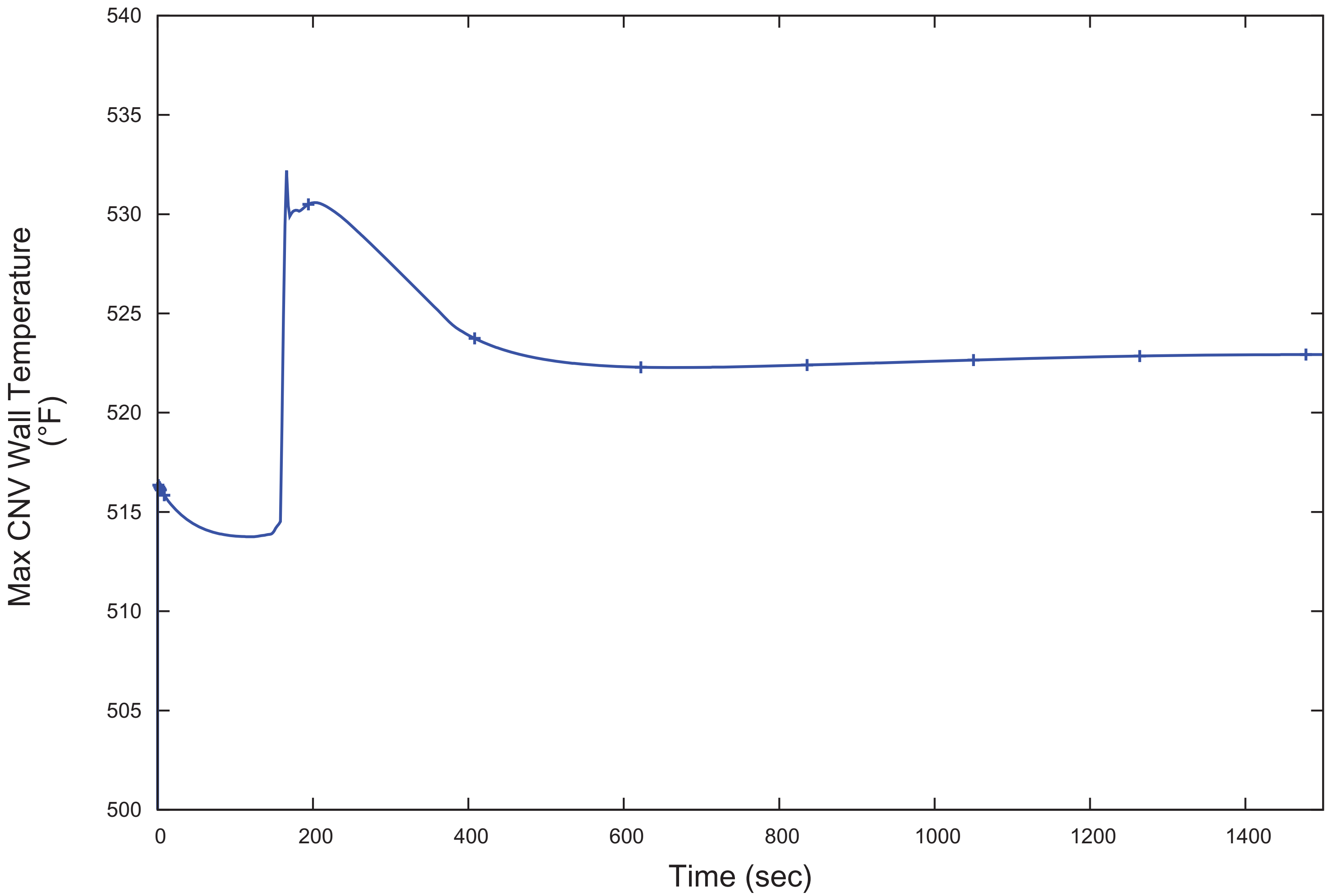


Figure 6.2-11: Integrated Break and Emergency Core Cooling System Mass Release Rate - Peak Containment Vessel Temperature Case

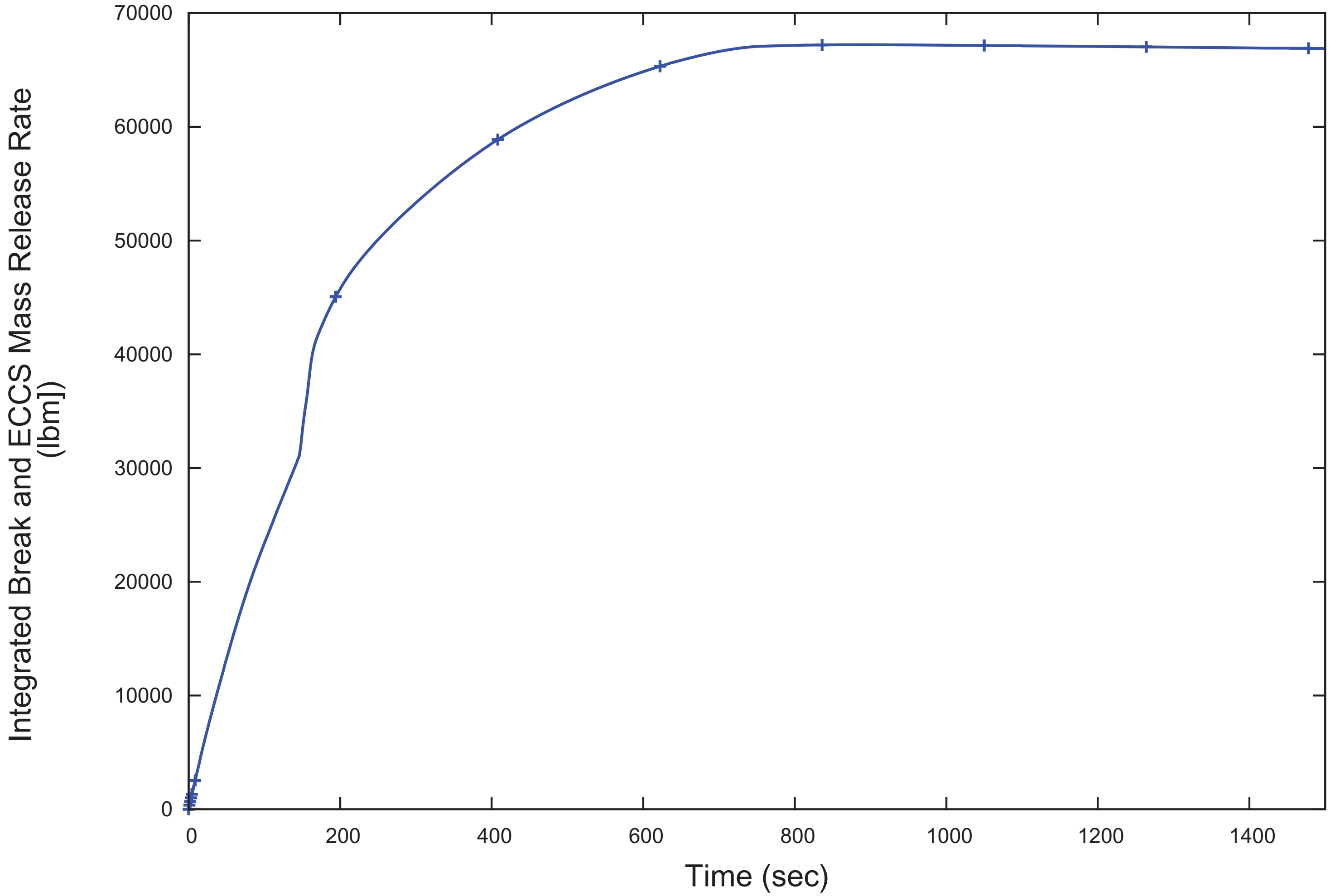
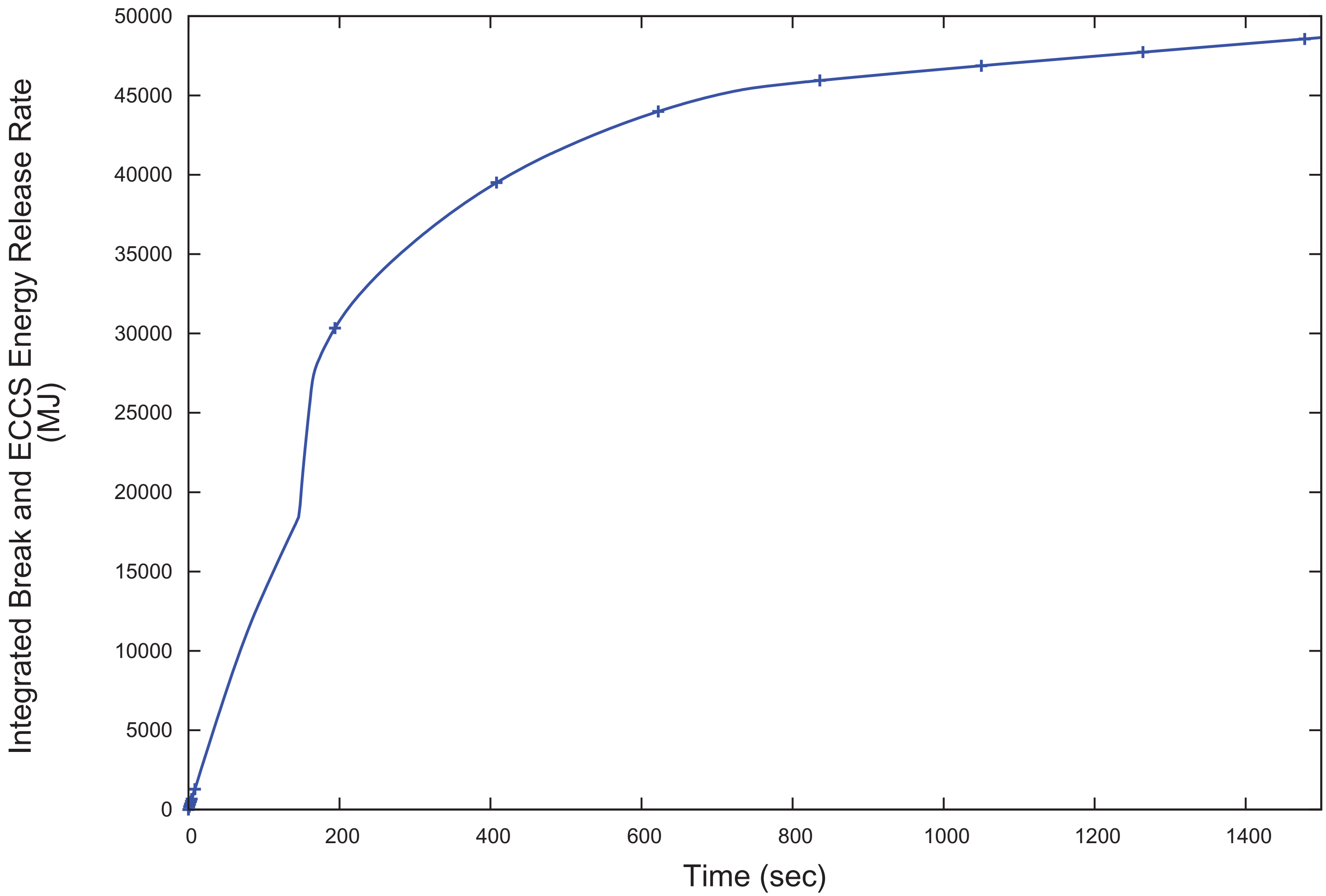


Figure 6.2-12: Integrated Break and Emergency Core Cooling System Energy Release Rate - Peak Containment Vessel Temperature Case



6.3 Emergency Core Cooling System

The emergency core cooling system (ECCS) provides core cooling during and after anticipated operational occurrences and postulated accidents, including loss-of-coolant accidents (LOCAs). The ECCS is an important NuScale Power Plant safety system in its safety-related response to LOCAs and as a component of both the reactor coolant and containment vessel (CNV) pressure boundaries. In conjunction with the containment heat removal function of containment, the ECCS provides core decay heat removal in the event of a LOCA. The ECCS does not provide replacement or addition of primary inventory from an external source.

The ECCS consists of two reactor vent valves (RVVs) mounted on the upper head of the reactor pressure vessel (RPV), two reactor recirculation valves (RRVs) mounted on the side of the RPV, and associated actuators located on the upper CNV as shown in Figure 6.3-1. The RVVs and RRVs are closed during normal plant operation and open to actuate the system during applicable accident conditions. When actuated, the RVVs vent steam from the RPV into the CNV, where the steam condenses and liquid condensate collects in the bottom of the containment. The RRVs allow the accumulated coolant to reenter the RPV for recirculation and cooling of the reactor core. Placement of the RRV penetrations on the side of the RPV is such that when the system actuates, the coolant level in the RPV remains above the core and the fuel remains covered. The cooling function of the ECCS is entirely passive, with heat conducted through the CNV wall to the reactor pool.

The ECCS is a passive system that does not include long lengths of piping or holding tanks. The system is made up of the valves described above, which allow recirculation of the reactor coolant between the RPV and the CNV. The valves stay in the closed position during normal plant operation and receive an actuation signal upon predetermined event conditions (listed in Table 6.3-1) to open, depressurizing the RPV and allowing flow of reactor coolant between the CNV and the RPV. In events that result in rapid equalization of pressure between the reactor coolant system (RCS) and the CNV before reaching the low riser level ECCS actuation setpoint, such as an inadvertent RVV opening, the ECCS valves open on low differential pressure before an ECCS actuation signal is received. The ECCS includes a passive ECCS supplemental boron (ESB) feature. The ESB dissolvers contain solid boron oxide that dissolves into condensate during ECCS operation for recirculation into the core to maintain subcriticality. Mixing tube components in the lower containment vessel transport condensate to mix with liquid in the bottom of containment.

Passive design provisions ensure retention of sufficient coolant inventory in the module to maintain a covered and cooled core following a LOCA and valve opening event. Makeup (i.e., addition) of reactor coolant inventory is not necessary or relied upon to protect against the effects of breaks. Reactor coolant inventory released from the reactor vessel during an in-containment unisolable LOCA collects and remains within the CNV. Actuating the ECCS ensures that the core remains covered and that RCS temperature and pressure decrease for all design-basis LOCAs. The analyzed loss-of-coolant events do not result in periods of refill or reflood, and the design does not require forced circulation. Actuation and operation of the ECCS passively establishes a natural circulation path whereby coolant heated in the core leaves as steam or vapor through the RVVs and condenses in the CNV, where it is collects and remains for return to the RPV.

The RRV penetrations are on opposite sides of the RPV shell in a horizontal orientation above the top of the reactor core. Figure 6.3-2, a schematic depicting ECCS conditions (accident conditions) shows the locations of the ECCS valves.

The ECCS valves actuate by stored energy and have no reliance on power or nonsafety-related support systems for actuation (i.e., opening the valves). After actuation, the valves do not require a subsequent change of state nor availability of power to maintain the intended safety function position.

6.3.1 Design Basis

The ECCS serves three fundamental purposes. The system is normally in a standby state in which the RVVs and RRVs are closed and function as part of the reactor coolant pressure boundary (RCPB). The principal function of the ECCS is to cool the reactor core in situations when it cannot be cooled by other means, such as during a LOCA inside the CNV. In addition, the ECCS provides low temperature overpressure protection (LTOP) for the RPV.

The ECCS design provides fuel protection during postulated LOCAs. The system provides core cooling following the LOCA at a rate such that clad-metal water reactions are limited to negligible amounts thereby preventing fuel and cladding damage that could interfere with long-term effective core cooling. The ECCS provides adequate core cooling with one RVV and one RRV in the open position. The cooling and associated depressurization functions use safety-related equipment.

The postulated LOCA conditions satisfy the functional requirements and system performance requirements of 10 CFR 50.46. Section 6.3.3 describes the 10 CFR 50.46 requirements in more detail.

No credible single failure event prevents the ECCS from performing its safety function including electrical power (normal alternating current [AC] or augmented direct current power system [EDAS] power availability, busses, electrical and mechanical parts, cabinets, and wiring), initiation logic, and single active or passive component failure.

Section 5.2 discusses that LTOP, provided by the ECCS, does not exceed the RPV pressure-temperature limits.

Section 6.2 discusses long-term cooling requirements that call for the removal of decay heat via the passive containment heat removal function. The ECCS is designed as a passive system and is able to perform its safety function for at least seven days after a design basis event without support from a non-safety system.

The ECCS meets General Design Criterion (GDC) 2 requirements related to the seismic design of structures, systems, and components (SSC) whose failure could cause an unacceptable reduction in the capability of the system to perform its design safety function. Seismic design information is in Section 3.7, Section 1.9 addresses conformance with Regulatory Guide (RG) 1.29, Revision 6. The ECCS withstands the effects of natural phenomena and is structurally robust, ensuring a preserved safety function in the event of a safe shutdown earthquake. Installation inside the containment pressure boundary protects components of the ECCS are protected from physical damage due to pipe whip and internal missiles. Section 3.5 and Section 3.6 address missile protection and the effects of pipe whip, respectively.

The ECCS meets GDC 3 requirements. Their physical location within the CNV protects the RVVs and RRVs from fire. The valve actuators are on the exterior surface of the CNV shell (underwater). Neither environment is susceptible to fires, being protected by the reactor pool or the CNV.

The ECCS meets GDC 4 with respect to the environmental and dynamic effects associated with the normal operation, maintenance, testing, and postulated accidents. All ECCS components are designed against the loads caused by the dynamic effects resulting from valve discharge or equipment failures inside containment. The physical location within the containment pressure boundary provides protection from the dynamic effects of events and conditions outside the NuScale Power Module (NPM). Environmental qualifications of the ECCS components are discussed in Section 3.11.

The ECCS meets the regulatory requirements of GDC 5 because ECCS components are not shared between NPMs. The ultimate heat sink is common to all NPMs and Section 9.2 describes its compliance with GDC 5.

The ECCS meets the regulatory requirements of GDC 14 in that design, fabrication, and testing of those portions of the ECCS that are part of the RCPB ensures an extremely low probability of abnormal leakage, rapidly propagating failure, or gross failure. Chapter 3 discusses compliance with GDC 14.

The ECCS design does not require AC or EDAS power to effectively cool the core. The ECCS is a safety-related passive system designed to maintain core cooling and containment integrity independent of AC or EDAS power sources, by requiring structures, systems, and components to transition to their safety state upon loss of control power or motive power. Because the electric power systems do not perform a safety function in the facility design, the design supports an exemption from GDC 17 (Section 3.1.2.8).

Section 6.3.2 describes that the ECCS meets the requirements of GDC 27 with the ECCS supplemental boron function.

The ECCS meets the regulatory requirements of GDC 30 and 31 for those portions of the ECCS that are part of the RCPB in that design, fabrication and testing meet the highest quality standards practicable, and components are designed with sufficient margin to ensure that the RCPB behaves in a non-brittle manner. Therefore, the design minimizes the probability of a rapidly propagating failure. Section 6.3.2 discusses component descriptions with the applicable design codes and classifications.

The design supports an exemption from GDC 33 and Section 3.1.4 addresses this exemption.

The module protection system (MPS) and containment system isolates postulated leaks that occur outside the containment, thereby preserving the remaining inventory in the containment. This inventory maintains the coolant level above the top of active fuel and establishes cooling using the ECCS. The reactor coolant system or the ECCS supplemental boron function assure that reactivity control is maintained prior to and after ECCS actuation.

The ECCS setpoints ensure automatic actuation of ECCS valves in response to design-basis LOCA events or 24 hours after a loss of AC power, LTOP function, or if needed to maintain subcriticality during extended passive cooling. ECCS valves also automatically actuate at a RCS pressure or RCS temperature conditions that could occur during beyond-design-basis event conditions, to provide defense in depth RPV and CNV over-pressure protection. Table 7.1-4 provides analytical limits used in analyses for ECCS actuation. The RPV and CNV design, in conjunction with the passive design and operation of ECCS and containment isolation, ensure that the core remains covered and ensures maintenance of adequate core cooling if a break occurs in the RCPB.

There is no safety-related coolant makeup system for coolant for protection against small breaks in the RCPB. The CVCS provides reactor coolant makeup during normal operation for small leaks in the RCPB, but is not relied upon during a design-basis event. The RPV and CNV design retain sufficient coolant inventory that, in conjunction with safety actuation setpoints to isolate CVCS from the RCS and operation of ECCS, adequate cooling is maintained and the SAFDLs are not exceeded in the event of a small break in the RCPB.Therefore, the ECCS design does not require a reactor makeup system and satisfies the underlying purpose of GDC 33.

Facility design meets the regulatory requirements of principal design criterion 35, and GDC 36 and GDC 37 as they relate to the ECCS providing sufficient core cooling to transfer heat from the core at a rate such that fuel and cladding damage does not interfere with or prevent long-term core cooling, permit appropriate periodic inspection of important components, and provide for appropriate periodic testing. Redundancy of ECCS components, features, and capabilities ensures the system safety functions assuming the single failure criteria.

Section 7.2.15 discusses that the MPS provides the capability to perform periodic functional testing of the ECCS, which ensures operability and performance of system components.

The ECCS meets the regulatory requirements of GDC 50, GDC 51, GDC 52, and GDC 53 for those portions of the ECCS that are part of the containment pressure boundary in that the components can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions that result from any LOCA. The ECCS component design reflects consideration of service temperatures and the uncertainties in material properties, stresses (residual, steady state and transient), and flaw size. The components accommodate the required containment integrated leakage rate and periodic inspection, surveillance, and penetration testing requirements. Section 6.2.5 discusses compliance with 10CFR 50.44.

The ECCS meets 10 CFR 50.34(f)(1)(vii) (Three Mile Island Action Plan NUREG 0737, Item II.K.3.118) in that ECCS automatically initiates without requiring manual operator action to ensure adequate core cooling for any design basis event. The ECCS does not require operator action or nonsafety-related system support for operation although manual actuation is possible from the control room.

Section 3.11 addresses environmental qualification information for the ECCS.

The ECCS passive and simple design satisfies the minimization of contamination requirements of 10 CFR 20.1406. The system does not extend beyond the CNV boundary, which precludes the possibility of contaminating the facility or environment. By design, limiting operation of ECCS to the RPV and CNV components minimizes the generation of radioactive waste facilitating facility decommissioning.

Facility design meets 10 CFR 50.34(f)(2)(xi) (Three Mile Island Action Plan NUREG 0737, Item II.D.3) in that the control room provides valve position indication for the ECCS valves and trip and reset actuator valves. In addition, the control room provides solenoid power indication for the ECCS trip and reset valves.

Facility design meets 10 CFR 50.34(f)(2)(xviii) (Three Mile Island Action Plan II.NUREG 0737, Item.II.D.3). Adequate core cooling is ensured by RPV design considerations that limit the location of vessel penetrations to above the top of active fuel in the reactor core, which eliminates the possibility of process system failures that could potentially drain vessel coolant inventory to levels below the top of fuel. Control room indication for a diverse selection of monitored parameters ensures unambiguous identification of conditions indicative of inadequate core cooling. Parameters monitored include core inlet and exit temperatures, RPV water level and pressure, and degree of subcooling. Section 7.2 describes control room indication and instrumentation.

The LOCA Evaluation Model (EM) (Reference 6.3-1) satisfies the requirements of 10 CFR 50, Appendix K. The ECCS performance analysis results demonstrate that the ECCS meets the 10 CFR 50.46 acceptance criteria (Table 15.0-4). The NPM design complies with 10 CFR 50.46 in that cooling performance is in accordance with an acceptable EM. The EM satisfies 10 CFR 50 Appendix K requirements.

6.3.2 System Design

6.3.2.1 Schematic Piping and Instrumentation Diagrams

Figure 6.3-1 shows a schematic of the ECCS configuration.

6.3.2.2 Equipment and Component Descriptions

The ECCS consists of independent RVVs and RRVs with associated actuator assemblies, instrumentation, and controls (I&C). Figure 6.3-3 shows a typical RVV. Figure 6.3-4 shows a typical RRV. The flanged RVVs and RRVs are bolted to the RPV by mating bolted penetrations which are integral to the vessel shell. Each main valve is a solenoid pilot-operated relief valve that is hydraulically closed, spring-assist to open, normally closed, and fails open. Table 6.3-2 provides the ECCS valve design and normal operating information.

The ECCS main valves are part of the RCPB and allow flow between the RPV and CNV when opened. The valves actuate automatically by a safety function signal from the MPS, by operator action that de-energizes the actuator trip valve solenoids, or by loss of power from the EDAS.

During power operations, the ECCS maintains a standby condition with the main valve disks held closed against spring force by a pressurized control chamber. The main valve initially closes (resets) by operator action that supplies CVCS water to pressurize the control chamber and to close the valve against spring pressure. Once closed, the valve maintains closure by a pressurized control chamber. Reactor pressure ported through an internal orifice located in the body of the main valve disk maintains control chamber pressure during normal operation.

The main valve opens by depressurizing the control chamber (venting to containment). Depressurizing the control chamber allows spring pressure, assisted by reactor coolant pressure, to reposition the main valve disk open.

Each main valve has an associated ECCS actuator assembly that contains three pilot valves: two trip valves and a reset valve, each with an associated solenoid as shown in Figure 6.3-3 and Figure 6.3-4. Each ECCS actuator assembly attaches to the exterior of the containment pressure vessel shell with a nozzle and safe end configuration. The body of the actuator assembly serves as both a CNV pressure boundary and the RCPB. Valve bonnet seals on each pilot valve establish the pressure boundaries internal to the valve assembly body. The ECCS trip/reset assemblies are submerged in the reactor pool.

Each reset pilot valve controls a hydraulic line that supplies coolant from the CVCS to its associated main valve control chamber. The reset pilot valve is an energize-to-open, normally closed, fail closed direct current solenoid-operated valve. Energizing the reset pilot valve solenoid ports CVCS (RCS coolant) to the main valve to pressurize the control chamber and to close the main valve against spring pressure. When reactor pressure is sufficient to maintain control chamber pressure against main valve spring pressure, the pilot valve solenoid de-energizes to close the reset valve.

Each of the trip pilot valves controls a hydraulic line that vents the main valve control chamber into the CNV, allowing spring force and RCS pressure to open the main valves. The trip pilot valves are de-energize-to-open, normally closed, fail open direct current solenoid-operated valves. A loss of direct current power to the actuator solenoid results in the opening of the actuator trip valves.

Each RRV main valve includes an inadvertent actuation block (IAB) feature designed to reduce the frequency of inadvertent operation (opening) of the main valve during power operations. The IAB also delays the RRVs from opening immediately with the RVVs until a lower differential pressure is reached between the RCS and containment. The IAB is in the path from the RRV control chamber to the trip and reset pilot valves. The IAB consists of a block valve with a spring-loaded disc that functions to block venting of the main valve control chamber when the RPV to CNV differential pressure is above a predetermined threshold. When differential pressure across the block valve lowers to below the IAB release pressure, the spring retracts the block valve to open the control chamber vent path.

The threshold pressure for operation of the IAB to prevent spurious opening of the RRV is 900 psid. Therefore, the IAB prevents RRV opening for all reactor pressures 900 psid and greater with respect to containment. Given an initial IAB block, the IAB releases at 450 psid +/- 50 psi once reactor pressure reduces. The IAB does not prevent RRV opening for initial pressures of 400 psid and below.

If the RRV trip valves inadvertently open (vented to containment) while the reactor is at normal operating pressure, the IAB valve seats to prevent the RRV main valve control chamber from depressurizing and the main valve from opening.

The RVVs are nominal pipe size 5 solenoid pilot-operated relief valves attached to the reactor vessel head and connected directly to the pressurizer steam space of the RPV.

The RRVs are nominal pipe size 2, solenoid pilot-operated relief valves attached to the upper shell section of the RPV, above the main closure flange above the top of the reactor core. The valves connect directly to the downcomer space of the RPV and maintain a minimum flow coefficient of 55. For ECCS demands, where reactor pressures are below the IAB functional range, the RRVs fully open within 10 seconds after trip valve solenoid power removal.

The ECCS valves perform their safety function under postulated events in compliance with GDC 27. Table 6.1-1 provides additional information describing the ECCS main valve, tubing, and actuator valve materials of construction.

Stainless steel bolt-on flow diffusers are mounted on the discharge of the RVVs to diffuse the high pressure steam and water flow discharged to the CNV. RRVs do not require diffusers because they are smaller and more distant from equipment requiring protection. The RVV and diffuser, as a combined unit, have a minimum flow coefficient of 375 and minimum terminal pressure drop ratio (Xt) of 0.62. For ECCS demands, the RVVs fully open within 10 seconds after trip valve solenoid power removal.

A venturi is in the inlet of each RVV and RRV between the RPV and the RRV and the RVV and is inserted internal to the main valve body. Each venturi throat diameter is sufficiently small to limit (choked) blowdown flow during postulated inadvertent reactor valve actuation events when there is a high differential pressure between the RPV and CNV to slow the depressurization rate. The venturi size and orientation maintains sufficient flow capacity through the RVV and RRV when there is lower differential pressure for recirculation conditions (unchoked flow) and long-term cooling and maintains margins for precluding the potential for flow blockage due to debris.

The containment shell provides passive heat removal by transferring decay and sensible heat to the reactor pool. The accumulated discharge of coolant into the CNV provides conductive and convective heat transfer to the reactor pool. Section 6.2 describes the CNV with additional information on the heat removal function in Section 6.2.2.

The capability for containment heat removal through ECCS operation occurs without operator action for at least 7 days. Section 9.2.5 describes the reactor pool (ultimate heat sink).

Upon a sensed loss of AC power to the EDAS power system battery chargers, the MPS initiates reactor trip, decay heat removal actuation, demineralized water system isolation, and containment isolation to reduce battery load. In addition, three 24-hour digital timers in each division of the MPS start. If AC power cannot be restored within 24 hours, the timers initiate the ECCS by de-energizing the engineered safety features actuation system (ESFAS) MPS divisions.

This ECCS hold mode maintains energized ECCS trip valve solenoids without an actuation signal, but sheds the load at 24 hours ensuring sufficient battery power for post-accident monitoring for at least 72 hours. The ECCS immediately initiates upon receipt of an ECCS actuation signal as listed in Table 6.3-1 during the 24-hour timing period. An automatic ECCS actuation after an automatic or manual reactor trip allows the ECCS supplemental boron to recirculate into the reactor core region before xenon decays from the core, to passively ensure subcriticality without requiring operator actions. Operators may manually bypass the actuation upon confirmation of subcriticality at cold conditions.

6.3.2.2.1 ECCS Core Cooling System Supplemental Boron

Upon actuation of ECCS, an ECCS supplemental boron (ESB) feature provides additional boron concentration to ensure that the reactor remains subcritical for at least 72 hours following an event. Thus for DBEs, the combined reactivity of the control rod assemblies and ESB ensures reactivty is controlled in accordance with GDC 27, as demonstrated in Section 15.0.5. The ESB provides sufficient boron to ensure core subcriticality and that the reactor core boron concentration remains below precipitation limits. The ESB and its components are not part of the RCPB and accordingly are not required to be designed to Quality Group A requirements. They are designed to remain operable following a design basis seismic event.

The two ESB dissolvers add boron to the ECCS recirculating coolant for reactivity control to maintain subcriticality following some design basis events (Figure 6.3-2). The dissolvers maintain subcriticality when the highest-worth control rod is stuck in a fully withdrawn position during long term cooling to prevent an overcooling return to power. Although the boron added by the dissolvers is not necessary during all design basis events to maintain subcriticality, the dissolvers are passive and respond to design basis accidents and transients that result in ECCS actuation where condensate forms on the inner containment surfaces. The dissolvers are fed by hoppers during the startup process and do not require personnel in the area to perform dissolver loading. This activity is performed remotely.

The dissolvers are located inside the CNV. The dissolvers contain solid boron oxide that is loaded during startup and remains unused during normal plant operations, without ECCS actuation. Two condensate channels extend outwards and upwards along the CNV inner wall. The dissolvers are located below the reactor pool level to ensure that sufficient condensate is generated above the channels. The condensate that is generated on the CNV wall during ECCS actuation is directed towards the dissolvers which dissolve the boron oxide creating a boric acid solution. This solution subsequently exits the dissolver and is added to the ECCS recirculating coolant in the CNV and enters the RPV through the RRVs.

Two lower mixing tubes are located on the inside wall of containment below the CNV flange. These mixing tubes are designed to force condensate flow to the bottom of containment. During ECCS actuation a portion of condensate generated on the CNV inner wall is captured by condensate channels and redirected through the lower mixing tubes. This collected condensate creates a water column in the lower mixing tubes, which is used to passively mix the collected condensate with the colder or more highly borated liquid in the bottom of containment. This mixing ensures the colder and potentially higher boron concentration coolant is not sequestered below the RPV flange and is sufficiently mixed and recirculated into the core through the RRVs.

6.3.2.3 Applicable Codes and Classifications

The pressure boundary components of the ECCS (valves, hydraulic lines, and actuator assemblies) are Quality Group A, Seismic Category I components constructed to the requirements of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section III, Subsection NB, 2017 Edition. Safety related components of the ECCS Supplemental Boron feature are Seismic Category I and accordingly designed to maintain structural integrity to perform their safety function during the SSE. The safety related ESB components are constructed of ASME BPVC Section II austenitic stainless steel materials, which are compatible with the coolant and boric oxide chemistry during normal operation and post-accident conditions. The ESB components are nonstructural attachments inside the CNV. Additional information addressing compliance with the applicable codes and classification of ECCS components is in Section 1.9 and Section 3.2. Valve position indication for the ECCS is qualified for seismic loads in accordance with Institute of Electronics and Electrical Engineers Std. 344. Section 7.1 addresses the Institute of Electronics and Electrical Engineers standards applicable to the controls and power supplies.

6.3.2.4 System Reliability

Redundant valves are remotely and separately actuated by two divisions of the ESFAS function of the MPS to provide reliability of the ECCS. Separate divisions for instrumentation and electric power are used for the ECCS valves and actuators.

The ECCS main valves do not rely on power or nonsafety-related systems for actuation (i.e., opening the valves) because they are capable of actuation with stored energy. Following actuation, the valves do not require a subsequent change of state or the availability of power to continue to perform their intended safety functions. Two trip valves are provided in series to reduce the probability of inadvertent opening of the main valve.

The safety related components of the ESB do not rely on power or non-safety systems to perform their function. The safety related components of the ESB perform their function passively by collecting and diverting condensate already present during ECCS operation. A change of state or the availability of power is not needed to continue to perform their intended safety functions.

No single active or passive failure prevents ECCS initiation or the capability of the system from performing its core cooling or LTOP safety function. Section 6.3.3 provides a performance evaluation of the ECCS, and Table 6.3-3 provides the failure modes and effects analysis.

The ECCS main valves, which are attached to the RPV, are not susceptible to water hammer. The system design includes no pumps or piping, which precludes the susceptibility to water hammer mechanisms. The ECCS actuator lines and trip reset valves are also not susceptible to water hammer. The design and operation of the ECCS valve actuator lines and trip reset valves precludes susceptibility to water hammer mechanisms. The ECCS venturis are also not susceptible to any known form of water hammer, nor do they add susceptibility to any surrounding ECCS component.

The design partially complies with the regulatory positions of RG 1.82. The design complies with the RG 1.82 regulatory positions that address the design criteria, performance standards, and analysis methods related to water sources for long-term cooling.

The design does not comply with regulatory position C1.1, with the exception that the design does comply with the following regulatory positions.

- Position C1.1.1.9 (assessment of the possibility of downstream clogging), and position C1.1.1.10 (buildup of debris and chemical reaction products downstream).
- Position C.1.1.2 (minimization of debris source term, cleanliness programs, monitoring/sampling for latent debris, insulation selection, restriction on coatings and cladding of carbon steel).
- Positions C1.1.3 and C1.1.4 are not applicable because the design does not rely on operator action to mitigate the consequences of debris accumulation and does not include active devices or systems to prevent debris accumulation.

The design does not comply with regulatory position C.1.2 (alternative water sources for inoperable strainers).

The design complies with regulatory position C.1.3 (evaluation of long term recirculation capability as applicable to the design), with the exception of the following:

- Position C.1.3.1 (net positive suction head).
- Portions of position C.1.3.2 that are not consistent with the design.

The design does not comply with regulatory positions C1.3.7 (upstream effects) or C.1.3.9 (strainer structural integrity).

The design does not comply with regulatory position C1.3.12 (prototypical head loss testing).

The design does not comply with regulatory position C.2, with the exception of meeting chemical reaction effects (position 2.2).

The design does not comply with regulatory position C.3.

Analyses evaluate the impact on ECCS and long-term core cooling operation of the generation of post-LOCA or high-energy line break debris and the presence of latent debris. Based upon the design and cleanliness requirements, there is minimal debris generation and accumulation expected, and it does not adversely impact the ability of ECCS to perform its required functions. Debris limits address generic safety issue GSI-191. The results of an evaluation of the effects of fibrous, particulate, and chemical debris in the reactor coolant on the long-term cooling capability demonstrate that long-term core cooling is not adversely impacted. Evaluations assess the debris impact on ECCS components, the fuel, and the core. Section 6.3.3.1 provides further information.

Latent debris is the basis for evaluating debris because insulation, paint, and coatings used in typical pressurized water reactors are not used or allowed in the CNV. Latent debris (defined as unintended dirt, dust, paint chips, fibers, pieces of paper, plastic, tape, etc.) is the expected source of debris in the CNV and has both fibrous and particulate constituents.

Protective coatings are not used or allowed within the CNV. Susceptible components (e.g., cables) withstand the fluid jets and conditions associated with a LOCA or high-energy line break. The ECCS design does not include pumps, piping, trash racks, debris interceptors, or sump screens.

The impact areas for potential debris effects include the RRVs and reactor core. The amount of latent debris (debris limits provided in Section 6.3.3.1) does not adversely affect the ability of the ECCS to provide adequate core cooling. The RRVs are not prone to flow blockage as a result of latent debris within the ECCS flowpath.

COL Item 6.3-1: An applicant that references the NuScale Power Plant US460 standard design will describe a containment cleanliness program that limits debris within containment. The program should contain the following elements:

- Foreign material exclusion controls to limit the introduction of foreign material and debris sources into containment.
- Maintenance activity controls, including temporary changes, that confirm the emergency core cooling system function is not reduced by changes to analytical inputs or assumptions or other activities that could introduce debris or potential debris sources into containment.
- Controls that prohibit the introduction of coating materials into containment.
- An inspection program to confirm containment vessel cleanliness before closing for normal power operation.

6.3.2.5 Protection Provisions

The design provides protection against missiles, pipe whip, flooding, thermal stresses, LOCA loading, and seismic effects. The reactor building, NPM bio-shield, and CNV wall protect the ECCS main valves from external missiles. There is no rotating equipment beneath the bio-shields or in the CNV that could malfunction and generate a missile.

Section 3.6.1 describes protection of the ECCS against the effects of missiles, pipe whip (that might result from piping failures), jet impingement, and environment changes (wetting and temperature change). Section 3.6.1 describes protection from the direct dynamic effects of pipe breaks and from loads applied by the impinging fluid released for safety-related structures, systems, and components, including the ECCS. Protection against environmental changes (wetting and temperature) are in the design requirements of the individual components. Section 3.6.2 provides additional discussion of pipe break location selection and dynamic effects.

Section 3.10 contains seismic qualification of equipment, Section 3.5 for missile protection, Section 3.6 for protection against dynamic effects, Section 9.5 for fire protection, Section 3.11 for environmental qualifications, and Section 3.9 for thermal and displacement stresses.

Flow diffusers are on the outlets of the RVVs to protect plant cables and other components from jet impingement when the RVVs open.

6.3.2.6 Provisions for Performance Testing and Inspection

The general installation and design of the ECCS provides accessibility for testing and inspection. There is no insulation, so all surfaces and welds are normally accessible for testing and inspection. The ECCS valves accommodate the preservice and inservice testing and inspection requirements of IWC-2500 and ISTC-3100 of the ASME BPVC with the valves in place. Section 3.9.6 provides additional discussion.

No maintenance, inspection, or testing of ECCS components occurs during normal operations. Inspection and maintenance of the ECCS main valves occur during NPM outages. Because the CNV interior is inaccessible during normal operation, the required maintenance and inspections occur during outages.

The ECCS valves are power-actuated relief valves (ASME Operations and Maintenance Code Category B/C) with an open/close safety function. The main valve and associated pilot actuators are exercised and fail-safe tested as a single valve in accordance with ISTC-3521, -3560 and -3630 and I-3320 criteria. Valve position testing is in accordance with ISTC-3700 criteria.

Functional testing of ECCS valves under conditions similar to design conditions is only possible with a differential pressure established between the RPV and the CNV because the main valve control chamber must vent to the CNV. These tests occur under conditions that are colder than would exist for a required actuation of the ECCS valves and at a lower differential pressure.

Leak tightness and leakage rate of the containment boundary testing is in accordance with ISTC-3630 and Appendix J, Type B criteria. The requirements of Section XI, Subsection IWB of the ASME BPVC apply to all pressure-retaining (RCPB and containment pressure boundary) ECCS components.

Section 3.9 describes inservice testing of ECCS valves.

6.3.2.7 Manual Actions

There is possible manual actuation of the ECCS from the control room, but operator action is not credited or required for design basis events, including those that involve active failures of the ECCS. If a valve fails to change to its safety position on command, there is an operator alert. An alert generates when the position instrumentation on an ECCS valve differs from the MPS demand position.

There is no operator action requirement because suitable redundancy is in the design of the ECCS to accommodate such a failure. To restore capability, the operator can change the position of the malfunctioning valve to its demand position.

The ESFAS design prohibits manual override or interruption of an ECCS actuation while a valid safety system armed or active signal exists. The MPS seals in the ECCS actuation signal, ensuring completion of ECCS actuation (Section 7.2).

6.3.3 Performance Evaluations

The ECCS performance evaluation under accident conditions is in Chapter 15. The conditions that result in actuation of the passive ECCS discussed in Chapters 15 and 19 include the range of LOCAs, inadvertent RPV valve opening events, inadvertent ECCS actuation event, or other design basis event where loss of DC power supply or ECCS actuation occurs.

Performance of the ECCS uses an evaluation of system response to a postulated breach in the RCPB. The analyses demonstrate the adequacy of ECCS performance for the spectrum of postulated break sizes. The LOCA EM (Reference 6.3-1) is based on the guidelines in RG 1.203.

The LOCA analyses demonstrate that no fuel clad melting occurs for any design-basis LOCAs. Peak cladding temperature (PCT) is maintained within the regulatory acceptance criterion of 1,204 degrees C (2,200 degrees F). Critical heat flux does not occur for any of the cases assessed in the break spectrum. LOCA analysis results are in Section 15.6.

The LOCA EM evaluates a break spectrum as specified by Appendix K to 10 CFR 50 to provide identification of the most severe postulated LOCA. The LOCA analyses are in Section 15.6.

The long-term cooling capability following ECCS operation is analyzed using the extended passive cooling EM (Reference 6.3-2). Results of long-term ECCS cooling are in Section 15.0.

Table 15.0-4 shows the applicable ECCS performance acceptance criteria taken from 10 CFR 50.46.

The technical specifications define the minimum ECCS operational capabilities with respect to inoperable components and maximum allowable time period for components to be out of service.

The fuel parameters conform to the 10 CFR 50.46 acceptance criteria for postulated accidents. The acceptance criteria for local oxidation, metal-water reaction (hydrogen generation), and core geometry are inferred by demonstration that water level remains above the top of the active core in all cases.

Calculation results as described in Section 15.0 demonstrate that the ESB feature of the ECCS provides sufficient dissolution and recirculation of boron into the core region to maintain subcriticality during long-term ECCS cooling, accounting for the highest worth control rod stuck out of the core. Calculations demonstrate that maximum boron concentrations in the core region following actuation of the ECCS do not result in boron precipitation to the extent that there is an interruption of natural circulation flow in the core and long-term cooling capability. The calculation established a boron concentration maximum in the core region and an associated boric acid solubility temperature (Reference 6.3-2). As discussed in Section 15.0, the maximum boron concentration in the core region remains below the boric acid solubility limit for the coolant temperature.

The design provides containment cooling by natural convection on the exterior of the CNV, with the reactor pool serving as the heat sink. Containment heat removal provides for heat transfer from the CNV to the reactor pool. The system is completely passive and does not include any means of forced circulation.

Containment cooling provides sufficient capacity to reduce pressure to less than 50 percent of design pressure within 24 hours after a design basis event.

Containment heat removal in conjunction with ECCS operation rapidly provides the required reduction in containment pressure.

Section 15.0 shows that the ultimate heat sink provides sufficient cooling in the event of an accident in one NPM and permits the simultaneous safe shutdown and cooldown of the remaining NPMs, then maintains them in a safe shutdown condition.

Active systems are not necessary in order to show continued containment heat removal. The reactor pool volume at normal pool level and temperature provides adequate heat transfer from the spent fuel in the spent fuel pool and from each of the shutdown NPMs for more than 72 hours of post-accident cooling.

6.3.3.1 Debris Generation and Impact Evaluation

An evaluation of debris accumulation and effects on long-term ECCS capability has been performed. Specifically, there is an evaluation of the effects of fibrous, particulate, and chemical species or precipitates on long-term core cooling, including the potential for debris accumulation at the RRVs, core inlet, and in the heated core region.

Separately applied acceptance criteria determine the associated fiber and particulate limits with each debris type evaluated individually at each RPV location. The minimum acceptable value for each debris type defines the maximum allowable debris limit for that debris type. The following acceptance criteria, which include the acceptance criteria in approved WCAP-16793-NP-A, Revision 2, are applied:

- Fuel rod scale thickness.
- Peak cladding temperature.
- Excessive pressure drop due to debris.

A conservative limit on debris deposition on any fuel rod that precludes two adjacent fuel rods from touching and reducing convective heat removal from the rods is established. The limit includes clad oxide, crud layer, and debris deposition. The fuel assembly dimensions within the fuel design informs the calculation of the minimum clearance between adjacent fuel rods. The calculated pin-to-pin distance that is determined provides a conservative margin to the debris deposition limit.

Peak cladding temperature is limited to 800 degrees F to preclude occurrences of rapid nodular corrosion and accelerated hydrogen pickup rates that can reduce cladding performance.

Analysis assumptions include:

- Chemical species take the form of aluminum oxyhydroxide. and precipitate out of solution.
- The decay heat level during the initial portion of the transient applies for the entire event to maximize energy removal requirements.

- Power at 255 MWt includes a two percent measurement uncertainty.
- 120 percent of the American Nuclear Society 1971 decay heat standard.

The design minimizes debris generation by restricting the use of insulation, paint, and coatings within containment. Restrictions on the use of debris-generating materials (e.g., insulation, paint, coatings, etc.) and components within containment that can withstand accident conditions limit debris generation to less than the amount needed to block or restrict long-term cooling flowpaths.

An evaluation of the latent debris data in current operating plants informed an estimate of the amount of debris present in a NPM. The use of operating plant information as the basis for a latent-debris source term establishes a conservative estimate because the design does not have the large containment with machinery, insulation, and personnel traffic during outages typical of a traditional operating plant.

The fiber content of the latent debris consists of a particulate and fibrous component. The evaluation assumes the maximum fractions of the particulate and fibrous components. The assumption of maximum fractions of fiber and particulate adds additional conservatism to further overestimate the debris source term.

Given the potential challenge to core cooling, the total amount of chemical species is also evaluated. Aluminum is not present in significant quantities in the NPM, nevertheless this is the chemical species evaluated as there are no other detrimental species present in significant levels to evaluate.

Chemical species are not expected to form in the design. Boron controls reactivity and buffering agents are not added to containment. Stainless steel fabricated or stainless steel clad form containment and components within containment, which precludes the production of corrosion components. In addition, the maintenance of a rigorous cleanliness program minimizes the collection of material that could react with boric acid to form other chemical species.

Conditions associated with an inadvertent opening of a RRV form the basis for the evaluated debris accumulation. The inadvertent opening results in discharge of RCS coolant into the CNV that collects in the bottom of the CNV. Discharged reactor coolant that flashes to steam condenses on the CNV walls and washes down the CNV walls, collecting latent debris as it flows to the bottom of the CNV. The event results in conditions that initiate the ECCS which opens the remaining ECCS valves. The ECCS actuation results in the release of additional steam for condensation and collection of additional latent debris retained in the condensed coolant solution. When sufficient condensation raises containment coolant level to above the RRVs, fluid from the CNV re-enters the RPV for recirculation through the core and subsequent boil off for discharge as steam through the RVVs.

Debris transport into and through the core depends on the nature of the debris and the flow field in the containment and RPV. The evaluation assumes the earliest arrival of debris to the core. The fiber and particulate collected in the containment coolant are well mixed within a homogeneous solution, and debris transport is proportional to the ECCS flow rate into the RPV. Fiber is neutrally buoyant and all fiber approaches the core inlet in proportion to the ECCS flow.

The flow rate through the RPV and the CNV liquid volume determines the time it takes to introduce all debris from containment into the RPV. Once the debris enters the RPV, it concentrates because only steam exits through the RVVs. The evaluation conservatively assumes approximately half the debris mass is delivered to the core initially and 99 percent is delivered in less than five hours.

The evaluation assumes that debris that approaches the core inlet and ends up on the lower end fittings or structural grid is a single debris bed. As fibrous debris transports to the fuel, the potential exists for fiber to collect and reduce the effective opening size. The potential effect builds on itself and, if enough fibers are present, a thin fiber bed forms with openings small enough to begin to collect particulate debris. A fiber bed must accumulate to capture the particulate because particulates and precipitates are too small for capture without fibrous debris accumulating first.

Fuel assembly testing demonstrates that fiber loads of up to 7.5g/FA generate acceptable head loss. The results conclude that an accumulation of up to as much as 7.5g/FA of fiber at the core inlet would have no effect on long-term cooling capability. This amount of fiber (7.5g/FA) is not sufficient to establish an effective filtering bed, so the limiting mass of particulates and chemical species cannot be defined by accumulation at the core inlet. The limiting mass of particulate and chemical species is therefore defined by the amount that can be tolerated within the heated core.

Because of the size and nature of debris that reaches the RPV, it may not accumulate at the core inlet. Instead, some or all of the debris may pass through the core inlet and reach the heated core region. While energy from the boiling process precludes debris buildup at the leading edge of a spacer grid, it may force debris into internal grid locations and form localized blockages. Although complete blockage of a fluid sub-channel around a single rod resulting from debris buildup is improbable, the evaluation demonstrates that the requirements of 10 CFR 50.46 are met under this circumstance. The debris plug is completely insulated with heat removal only by axial conduction through the cladding. The peak cladding temperature is below the acceptance criteria of 800 degrees F.

Because of the natural heat generation decrease over time, the assumed decay heat is conservatively based on what would be generated early in the event. In an actual event, the decay heat load and temperature at the center of the plug decreases as time progresses.

The particulate and chemical precipitate calculation demonstrates that the analysis limit amount of particulate and chemical species can be tolerated in the core without affecting core heat transfer. The total debris mass estimate used in the evaluation provides for a conservative design limit.

Debris that reaches the heated core region may deposit on the fuel rods and form a layer that blocks the fuel sub-channel or hinders core heat removal. The EPR deposition analysis model examined the associated effects. The model incorporates the deposition and heat transfer calculations to determine the effect of fibrous, particulate, and chemical debris that deposit on the fuel rods. The model assumes that oxide and crud layers exist on the fuel surfaces before the event with deposition occurring as impurities transport into the crud deposit through large pores. Small particulate and formed precipitates draw into and merge with the growing scale. The analysis shows that the analysis assumed amount of fiber and of particulate in addition to the analysis assumed amount of aluminum deposited on the fuel rods meets the acceptance criteria with additional margin.

Analyses demonstrate adequate design margin with respect to the defined acceptance criteria. Adequate core cooling is ensured at debris levels of up to 7.5gm/FA (fiber), 30 lbm (particulate) and 194.5 lbm (aluminum). The following debris limits are established: 7.5 gm/FA (fiber), 30 lbm (particulate) and 194.5 lbm (aluminum).

6.3.4 Tests and Inspections

Preoperational testing of the ECCS function ensures that the specified design functions are met during any condition of normal operation, AOOs, or postulated accident conditions. Section 14.2 provides a description of initial plant testing requirements. The applicable guidance of RG 1.79, Revision 2, is in the preoperational testing described in Section 14.2.

Section 3.9.6 and Section 6.6 describe the preservice and inservice testing and inspection programs. The ECCS operational surveillance requirements are addressed in technical specifications.

Chapter 14.3 provides the methodology for the ECCS-related inspections, tests, analyses and acceptance criteria.

6.3.5 Instrumentation Requirements

The MPS provides for the control of the valves and monitoring instruments required for ECCS actuation. Post-accident monitoring information is in the main control room through the safety display indication system and the module control system that includes ECCS valve position. Section 7.2 addresses the ECCS-related instrumentation.

The ECCS has automatic actuation signals for emergency core cooling and LTOP from the ESFAS portion of the MPS, which also provides for manual actuation of the RVVs and RRVs by manual actuation switches in the main control room.

Automatic actuation signals for the ECCS are from independent and redundant sensors. The ECCS automatically actuates and requires no operator action during the first 72 hours following event initiation. Table 6.3-1 lists ECCS actuation values.

The ESFAS uses four redundant sensors (channels) to monitor ECCS-associated actuation parameters listed in Table 6.3-1 processed through MPS separation groups. The separation groups supply signals to two independent divisions of ESFAS that use two-out-of-four voting so that a single failure of an initiation signal cannot prevent a valid actuation or initiate an invalid actuation.

The EDAS supplies power to the actuators for the ECCS solenoid valves and ECCS valve position indications. This power may not necessarily be available during an accident, and valve closure is not required during an accident. Position indication cabling conforms with Institute of Electronics and Electrical Engineers Std. 323-1974 for the design conditions (temperature, humidity, submergence, pressure, radiation) of containment.

The MPS instrumentation accomplishes the ECCS performance monitoring for RPV riser and CNV water level, temperature, and pressure; reactor pool temperature and level; and valve positions for the ECCS valves, actuators, and containment isolation valves.

The MPS monitors wide range RCS cold temperature and wide range RCS pressure parameters that provide the signal to initiate LTOP (opening of the RVVs). The MCS provides for the control and monitoring instruments of the ECCS Supplemental Boron components.

6.3.6 References

6.3-1 NuScale Power, LLC, "Loss-of-Coolant Accident Evaluation Model," TR-0516-49422-P, Revision 3.

6.3-2 NuScale Power, LLC, "Extended Passive Cooling and Reactivity Control Methodology," TR-124587-P, Revision 0.

Table 6.3-1: Emergency Core Cooling System Actuation Values

Parameter(2)	**Value**(1)
Low RPV Riser Level	552 inches[3]
Low low Riser Level	460 inches[3]
RPV low temperature & high pressure (LTOP) actuation	The LTOP pressure setpoint is a function of the RCS cold temperature (Table 5.2-5 and Figure 5.2-3).
Low AC Voltage timer	Table 7.1-4
ECCS actuation delay after reactor trip	Table 7.1-4
High high RCS pressure[4]	Table 7.1-4
High high RCS average temperature[4]	Table 7.1-4

Note 1: Additional information for ECCS actuation values is provided in Table 7.1-4.
Note 2: Interlocks for these signals are described in FSAR Table 7.1-5.
Note 3: Reference to the bottom of the pool.
Note 4: ECCS actuation provides defense in depth.

Table 6.3-2: Emergency Core Cooling System Valve and Actuator Design and Operating Parameters

Service Condition	Parameter	RRV	RVV	Valve Actuators
Design conditions	Internal design pressure	2200 psia	2200 psia	2200 psia
	External design pressure	1200 psia	1200 psia	34 psia
	Design temperature	650°F	650°F	650°F
Normal operating conditions	Internal pressure	2000 psia	2000 psia	2000 psia
	External pressure	<1 psia	<1 psia	14.7 - 28 psia
	Fluid temperature	486°F	636°F	40 to 160°F
	Valve external temperature	Note 1	Note 1	100°F
Accident conditions	Design temperature	<650°F	<650°F	<650°F
	Internal pressure	<2200 psia	<2200 psia	<2200 psia

Note 1: Not temperature controlled. At vacuum.

Table 6.3-3: Emergency Core Cooling System Failure Modes and Effects

Component Identification	Function	Failure Mode	Failure Mechanism	Effect on System	Method of Failure Detection
Reactor Vent Valve (normally closed, fail open)	1) Maintain Closed Position	A) Spurious opening	Mechanical Electrical/I&C	The function of maintaining a closed position has failed, and RPV coolant is released into the CNV. The rest of the ECCS is actuated when the RPV water level reaches the ECCS actuation setpoint.	• Reactor trip (if critical at the time of failure) • Valve position indication • Pressurizer pressure monitoring • Pressurizer level monitoring • RPV riser level monitoring • RPV pressure monitoring • CNV pressure monitoring • RPV temperature monitoring • CNV temperature monitoring
		B) Leakage in excess of technical specification limits (passive failure)	Mechanical	There is little short-term effect on the system for minor leakage (on the order of the technical specification leakage rate limit) unless leakage is sufficient to depressurize the chamber holding the main valve closed, which could result in a spurious valve opening (Spurious Opening above).	• Containment pressure monitoring • Pressurizer level monitoring • Monitoring of leakage removed from CNV by containment evacuation system (CES) • Monitoring of radioactive gaseous discharge removed from CNV by containment evacuation system (CES) • Valve inspection and testing during outages
	2) Vent steam to CNV for emergency core cooling	A) Fail to open	Mechanical Electrical/I&C	Venting limited to redundant RVVs to ensure the core cooling function of ECCS is maintained. A mechanical failure may cause a single valve to fail to open when actuated. A divisional electrical I&C failure may cause one RVV (and one RRV) to fail to open.	• Valve position indication • Valve inspection and testing during outages
		B) Slow opening (extended stroke time or delayed actuation)	Mechanical Electrical/I&C	Greater portion of ECCS vent flow rate is through the redundant RVV until affected valve opens completely. In the meantime, the redundant RVV ensures the core cooling safety function of the ECCS is not compromised.	• Valve position indication • Valve inspection and testing during outages

Table 6.3-3: Emergency Core Cooling System Failure Modes and Effects (Continued)

Component Identification	Function	Failure Mode	Failure Mechanism	Effect on System	Method of Failure Detection
		C) Spurious closure	Mechanical Electrical/I&C Operator error	ECCS vented steam flow is exclusively through the redundant RVV. Redundant RVVs ensures adequate venting and thus successful completion of the ECCS safety function of core cooling. Spurious closure may be an indication of a systemic issue through Electrical/I&C failure or Operator error and may lead to both RVVs being closed and the ECCS system not providing a flow path for recirculation. Spurious closure of both valves is not considered credible due to MPS design.	• Valve position indication • RPV temperature monitoring (systemic only) • RPV pressure monitoring (systemic only) • Valve inspection and testing during outages
		D) Flow blockage (passive failure)	Mechanical	Reduced or nonexistent flow through affected valve(s). Redundant RVVs ensures adequate venting and thus successful completion of the ECCS safety function of core cooling. Flow blockage may be an indication of a systemic issue and may lead to both valves being blocked. Flow blockage would lead to the ECCS system not providing a flow path for recirculation. Systemic flow blockage of both valves is not consideration credible, Section 6.3.3.1.	• RPV temperature monitoring (systemic only) • RPV pressure monitoring (systemic only) • Valve inspection and testing during outages
	3) Depressurize RPV for low temperature overpressure protection (LTOP)	A) Fail to open	Mechanical Electrical/I&C	Reduced or nonexistent flow through affected valve but adequate depressurization rate is maintained by redundant RVV, thus ensuring protection of the RPV.	• Valve position indication • Valve inspection and testing during outages
		B) Slow opening (extended stroke time or delayed actuation)	Mechanical Electrical/I&C		

Table 6.3-3: Emergency Core Cooling System Failure Modes and Effects (Continued)

Component Identification	Function	Failure Mode	Failure Mechanism	Effect on System	Method of Failure Detection
		C) Spurious closure	Mechanical Electrical/I&C Operator error	Reduced or nonexistent flow through affected valve(s). Redundant RVVs ensures adequate depressurization rate is maintained by redundant RVV, thus ensuring protection of the RPV. Spurious closure may be an indication of a systemic issue through Electrical/I&C failure or Operator error and may lead to both valves being closed and the ECCS system not providing a flow path for recirculation. Spurious closure of both valves is not considered credible due to MPS design.	• Valve position indication • RPV temperature monitoring (systemic only) • RPV pressure monitoring (systemic only) • Valve inspection and testing during outages
		D) Flow blockage (passive failure)	Mechanical	Reduced or nonexistent flow through affected valve(s). Redundant RVVs ensures adequate venting and thus successful completion of the ECCS safety function of core cooling. Flow blockage may be an indication of a systemic issue and may lead to both valves being blocked. Flow blockage would lead to the ECCS system not providing LTOP relief. Systemic flow blockage of both valves is not consideration credible, Section 6.3.3.1.	• RPV temperature monitoring (systemic only) • RPV pressure monitoring (systemic only) • Valve inspection and testing during outages

Table 6.3-3: Emergency Core Cooling System Failure Modes and Effects (Continued)

Component Identification	Function	Failure Mode	Failure Mechanism	Effect on System	Method of Failure Detection
Reactor Recirculation Valve (normally closed, fail open)	1) Maintain Closed Position	A) Spurious opening	Mechanical Electrical/I&C	The function of maintaining a closed position has failed, and RPV coolant is released into the CNV. The rest of the ECCS is actuated when the RPV water level reaches the ECCS actuation setpoint.	• Reactor trip (if critical at the time of failure) • Valve position indication • Pressurizer pressure monitoring • Pressurizer level monitoring • RPV riser level monitoring • RPV pressure monitoring • CNV pressure monitoring • RPV temperature monitoring • CNV temperature monitoring
		B) Leakage in excess of technical specification limits (passive failure)	Mechanical	There is little short-term effect on the system for minor leakage (on the order of the technical specification leakage rate limit) unless leakage is sufficient to depressurize the chamber holding the main valve closed, which could result in a spurious valve opening (Spurious Opening above).	• Containment pressure monitoring • Pressurizer level monitoring • Monitoring of leakage removed from CNV by containment evacuation system • Monitoring of radioactive gaseous discharge removed from CNV by containment evacuation system • Valve inspection and testing during outages

Table 6.3-3: Emergency Core Cooling System Failure Modes and Effects (Continued)

Component Identification	Function	Failure Mode	Failure Mechanism	Effect on System	Method of Failure Detection
	2) Provide flow path for condensed steam to reenter RPV from CNV	A) Fail to open	Mechanical Electrical/I&C	Coolant return flow to the RPV is exclusively through the redundant RRV. Core cooling function of ECCS is not compromised. A mechanical failure may cause a single valve to fail to open when actuated. A divisional electrical/I&C failure may cause one RRV (and one RVV) to fail to open.	• Valve position indication • Valve inspection and testing during outages
		B) Slow opening (extended stroke time or delayed actuation)	Mechanical Electrical/I&C	Greater portion of ECCS recirculated flow is through redundant RRV until affected valve opens completely. In the meantime, the redundant RRV ensures the core cooling safety function of the ECCS is not compromised.	
		C) Spurious closure	Mechanical Electrical/I&C Operator error	ECCS recirculation flow is exclusively through redundant RRV. In the meantime, the redundant RRV ensures the core cooling safety function of the ECCS is not compromised. Spurious closure may be an indication of a systemic issue through Electrical/I&C failure or Operator error and may lead to both valves being closed and the ECCS system not providing a flow path for recirculation. Spurious closure of both valves is not considered credible due to MPS design.	• Valve position indication • RPV temperature monitoring (systemic only) • Valve inspection and testing during outages
		D) Flow blockage (passive failure)	Mechanical	Reduced or nonexistent flow through affected valve(s). In the meantime, the redundant RRV ensures the core cooling safety function of the ECCS is not compromised. Flow blockage may be an indication of a systemic issue and may lead to both valves being blocked. Flow blockage would lead to the ECCS system not providing a flow path for recirculation. Systemic flow blockage of both valves is not consideration credible, Section 6.3.3.1.	• RPV temperature monitoring (systemic only) • Valve inspection and testing during outages

Table 6.3-3: Emergency Core Cooling System Failure Modes and Effects (Continued)

Component Identification	Function	Failure Mode	Failure Mechanism	Effect on System	Method of Failure Detection
ECCS valves controlled by one ESFAS division One RVV and one RRV controlled by the same division. (normally closed, fail open)	1) Maintain Closed Position	A) Spurious opening	Electrical/I&C Operator error	The core cooling safety function is maintained. The affected RVV would open immediately, but the RRV is blocked from opening by the IAB. This failure then proceeds similarly to the spurious opening of a single RVV. The IAB keeps the RRV closed until the differential pressure between the RPV and CNV drops below the IAB release pressure setting. If a second failure occurs and the IAB fails to block the spurious opening of the RRV through additional mechanical failure, then the RRV opens immediately. The rest of the ECCS is actuated when the RPV water level reaches the ECCS actuation setpoint.	• Reactor trip (if critical at the time of failure) • Valve position indication • Pressurizer pressure monitoring • Pressurizer level monitoring • RPV riser level monitoring • RPV pressure monitoring • CNV pressure monitoring • RPV temperature monitoring • CNV temperature monitoring
ECCS valves with failure of shared RCS injection line All RVVs and all RRVs (normally closed, fail open)	1) Maintain Closed Position	A) Spurious opening	Mechanical	The core cooling safety function is maintained. The affected RVVs would open immediately, but the RRVs are blocked from opening by the IAB. This failure then proceeds with a spurious opening of both RVVs. The IAB keeps the RRVs closed until the differential pressure between the RPV and CNV drops below the IAB release pressure setting. If a second failure occurs and the IAB fails to block the spurious opening of an RRV through additional mechanical failure, then the affected RRV is opens immediately.	• Reactor trip (if critical at the time of failure) • Valve position indication • Pressurizer pressure monitoring • Pressurizer level monitoring • RPV riser level monitoring • RPV pressure monitoring • CNV pressure monitoring • RPV temperature monitoring • CNV temperature monitoring

Table 6.3-3: Emergency Core Cooling System Failure Modes and Effects (Continued)

Component Identification	Function	Failure Mode	Failure Mechanism	Effect on System	Method of Failure Detection
Lower Mixing Tube	1) Collect and transport condensate to the bottom of containment for recirculating coolant mixing during ECCS actuation.	A) Flow blockage (passive)	Mechanical	CNV mixing tube does not transport condensate to the bottom of containment for mixing. Colder borated coolant remains in the lower containment and is not recirculated through the RRV for reactivity control. Failure to maintain subcriticality during long term cooling following some design basis events. Blockage and structural failure of either lower mixing tube is not considered credible with passive design of the lower mixing tube.	• ICI neutron flux monitoring • ICI temperature monitoring • NMS excore neutron flux monitoring • RPV temperature monitoring • RPV pressure monitoring
		B) Breach in fluid boundary (passive)	Mechanical		
Supplemental Boron Dissolver	1) Dissolve boric oxide for boron addition into ECCS recirculating coolant	A) Flow blockage (passive)	Mechanical	Full blockage of flow into the dissolver prevents dissolution and boron addition. Partial blockage of flow may not allow dissolution to meet the needed boron concentration increase within the time required. Failure to increase the boron concentration in the ECCS recirculating coolant within the time required to maintain subcriticality following some design basis events. Blockage of either supplemental boron dissolver is not considered credible with passive design of the supplemental boron dissolver.	• ICI neutron monitoring • ICI core temperature monitoring • NMS excore neutron monitoring • RPV temperature monitoring • RPV pressure monitoring
		B) Breach in fluid boundary (passive)	Mechanical	Breach in fluid boundary prevents condensate flow from performing dissolution to meet the needed boron concentration increase within the time required. Failure to increase the boron concentration in the ECCS recirculating coolant within the time required to maintain subcriticality during long term cooling following some design basis events. Structural failure of either supplemental boron dissolver is not considered credible with passive design of the supplemental boron dissolver.	

Figure 6.3-1: Emergency Core Cooling System

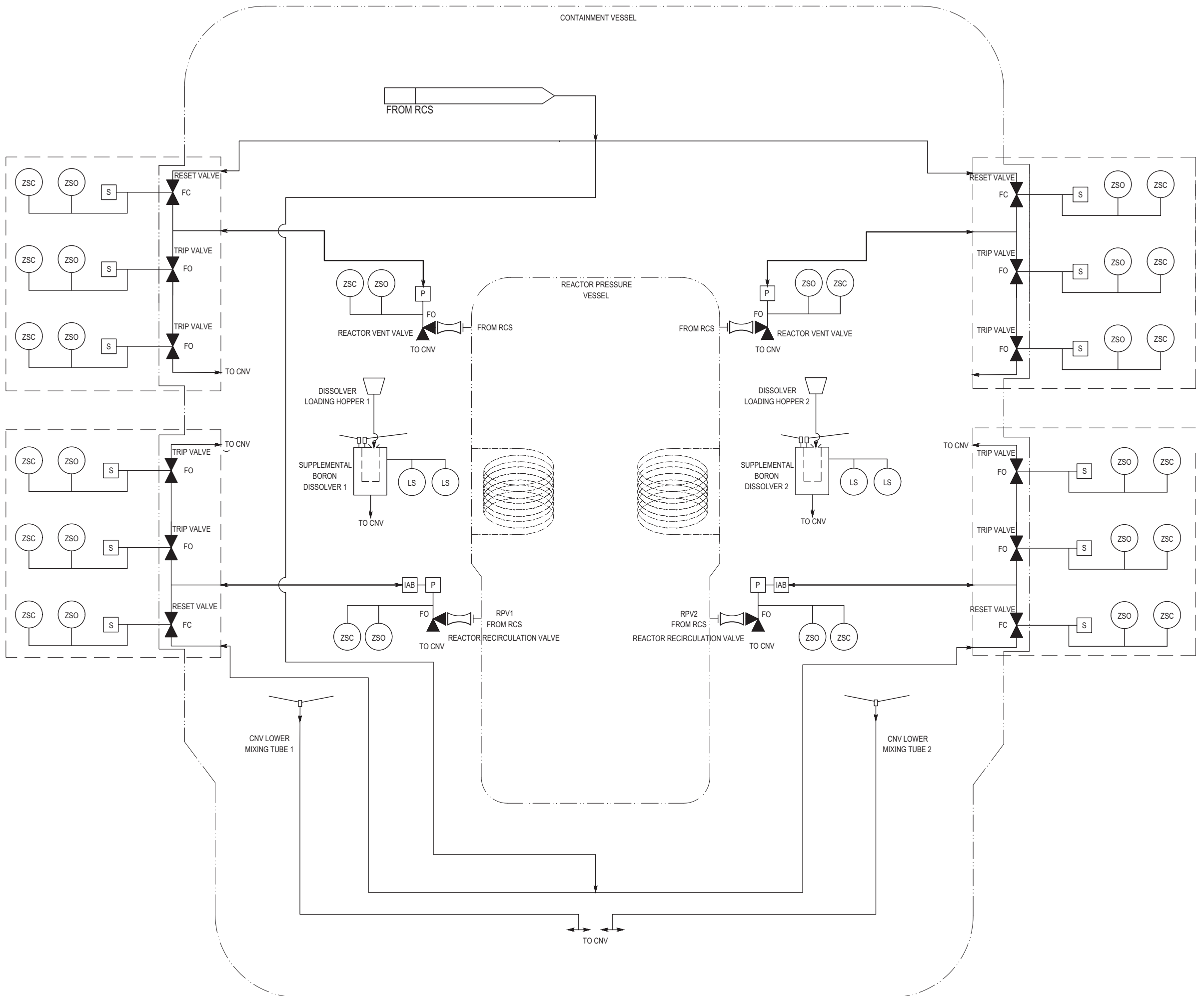


Figure 6.3-2: Emergency Core Cooling System Operation

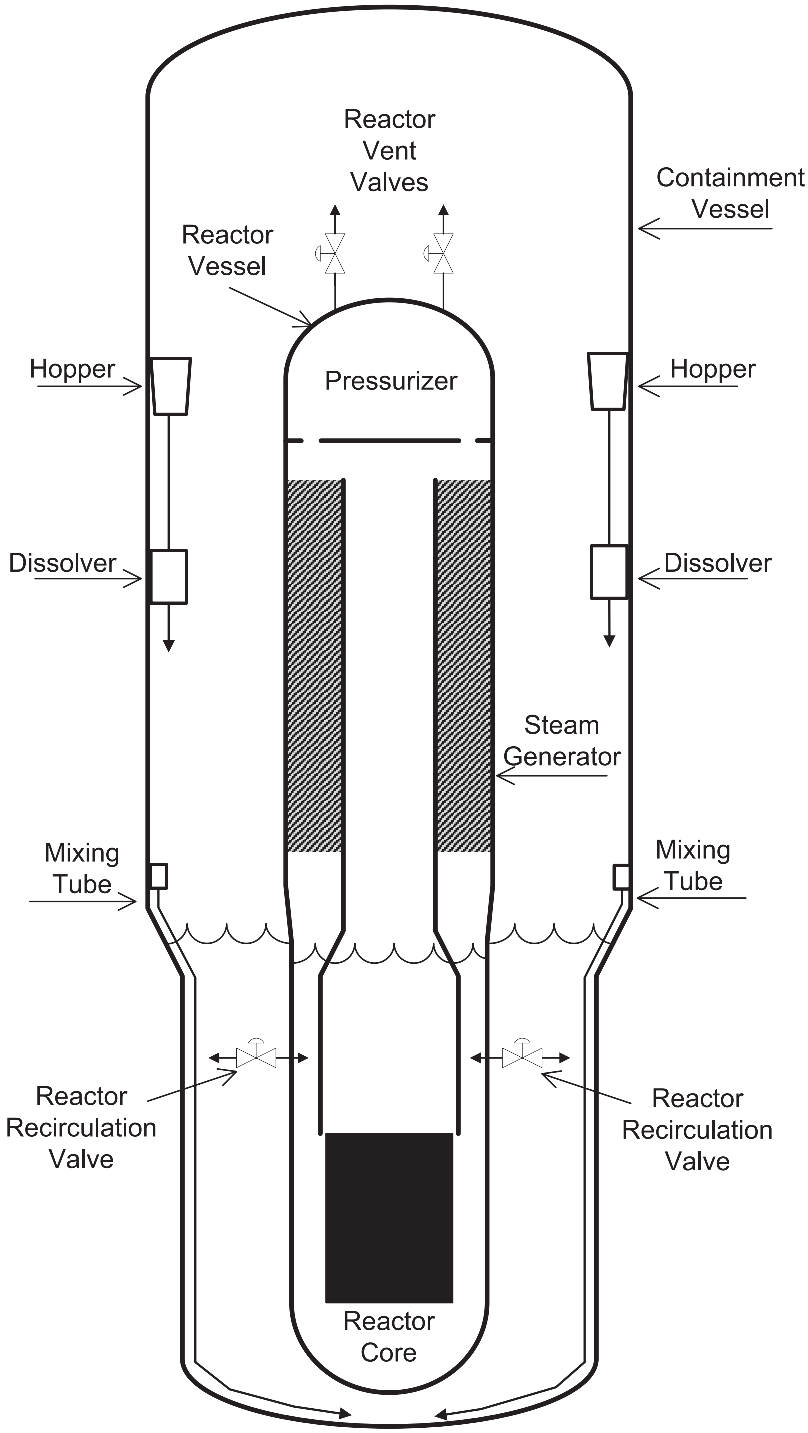


Figure 6.3-3: Simplified Reactor Vent Valve Diagram

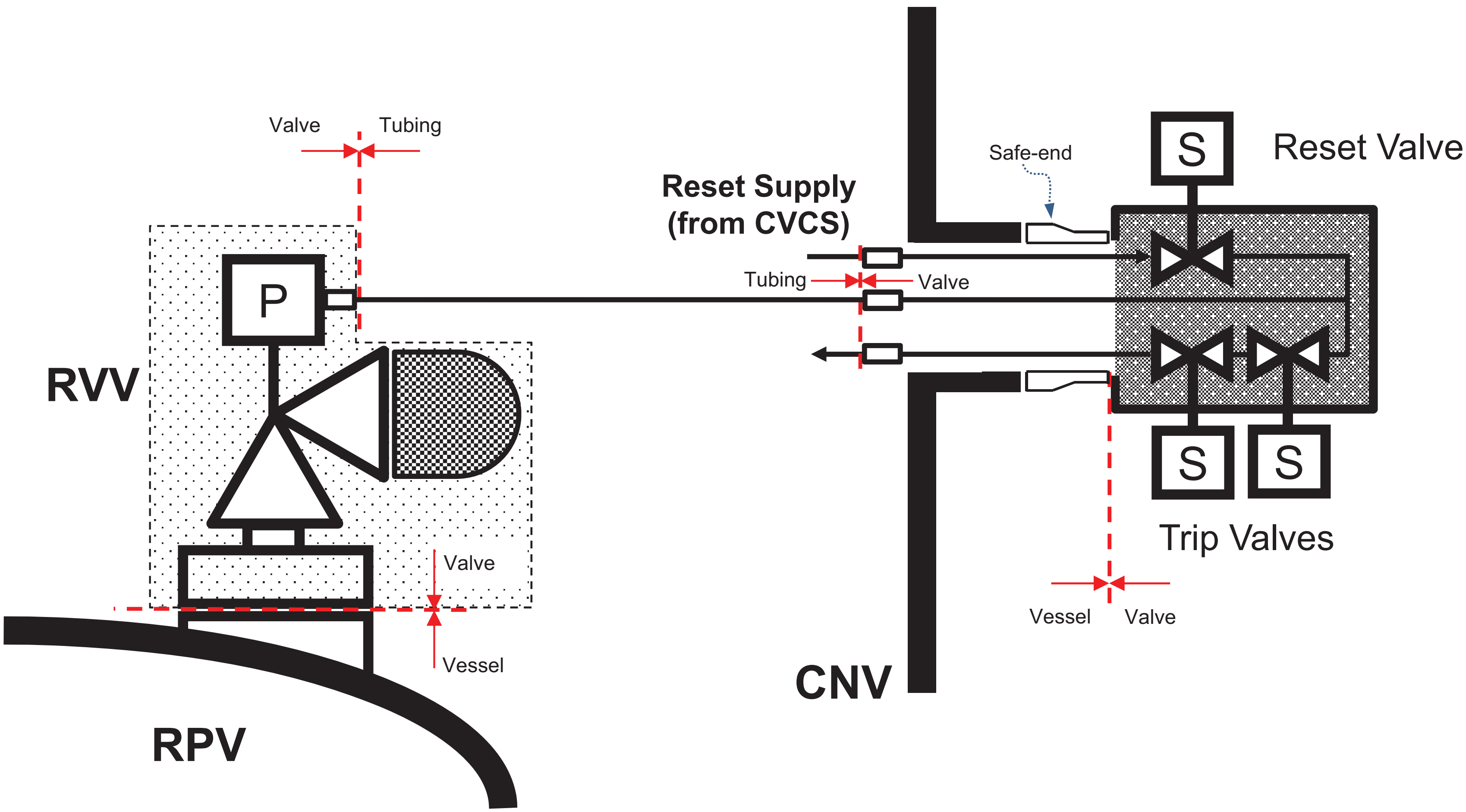
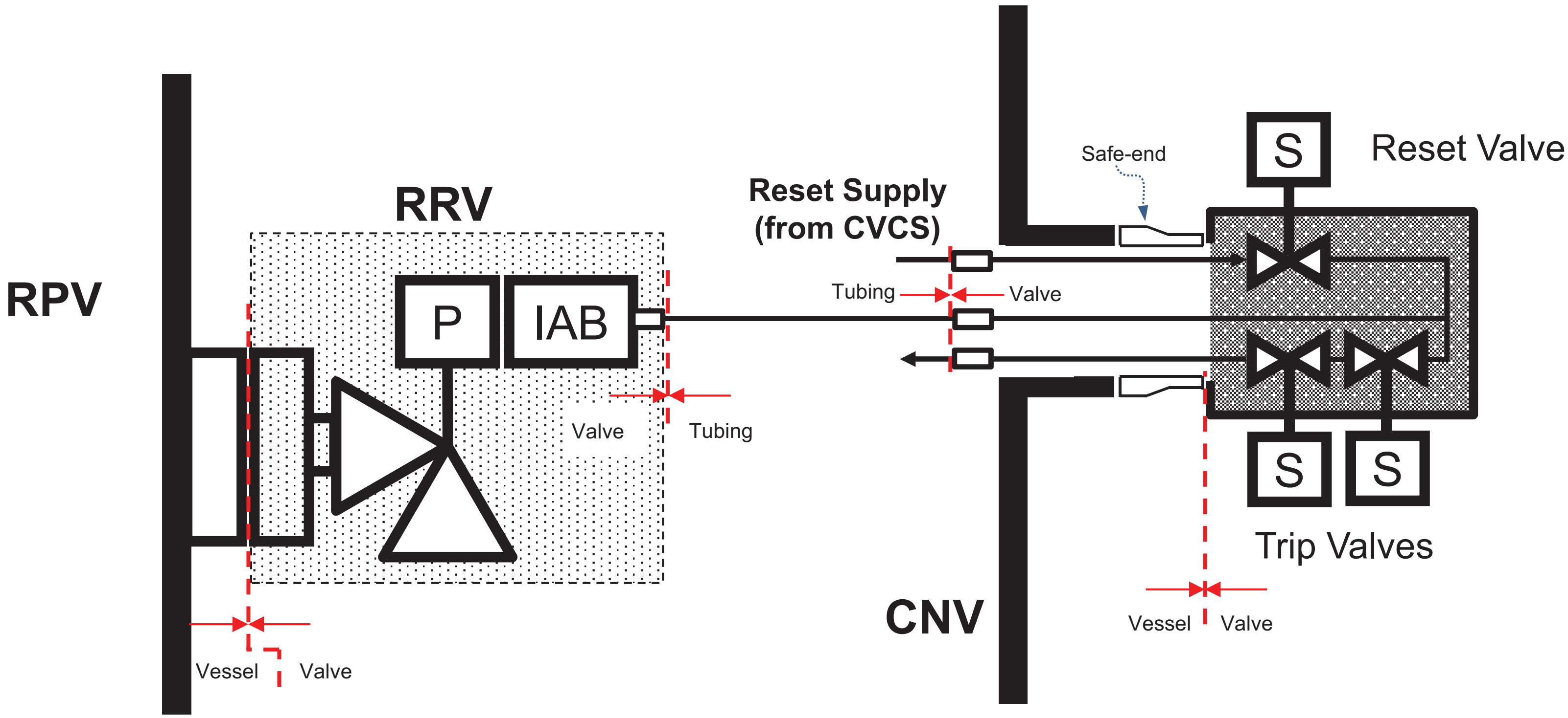


Figure 6.3-4: Simplified Reactor Recirculation Valve Diagram



6.4 Control Room Habitability

Control room habitability refers to the conditions required for life support and safe, effective operation of the plant during normal conditions and following an accident. These conditions include adequate lighting, food, water, air, and climate control. Systems and equipment provide habitability functions to protect the operators against postulated releases such as radioactive materials and smoke. Control room habitability functions include:

- missile protection (Section 3.5)
- radiation shielding (Chapter 12, Radiation Protection)
- normal pressurization, air filtration, and air conditioning (Section 9.4.1)
- fire protection (Section 9.5.1)
- radiation monitoring (Section 9.4.1 and Section 11.5)
- smoke detection (Section 9.4.1)
- lighting (Section 9.5.3)

This section describes the control room habitability system (CRHS), which provides breathable air to the control room for 72 hours without reliance on electrical power if the normal control room HVAC system (CRVS) is unavailable. After 72 hours, the CRVS, if restored, provides filtered heating, ventilation, and air conditioning (HVAC) service to the control building (CRB) for the remainder of an event recovery period (the CRVS is described in Section 9.4.1).

6.4.1 Design Bases

The CRHS is a nonsafety-related system that provides emergency breathing air to the control room envelope (CRE) and maintains a positive control room pressure for habitability and control of radioactivity when conditions prohibit the CRVS from fulfilling these functions.

General Design Criterion 4 is considered in the design of the CRHS. The CRHS is designed to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. The CRHS components are located in the CRB, which provides protection from potential adverse environmental conditions. The CRHS air bottle racks are designed to ensure that the air bottles do not become missiles within the CRB.

General Design Criterion 5 is considered in the design of the CRHS. The CRHS services the control room that contains the controls for up to six NuScale Power Modules and is designed such that a failure of one portion of the system does not impair the ability to perform its regulatory required functions including, in the event of an accident in one module, an orderly shutdown and cooldown of the remaining module(s).

The CRHS, in conjunction with the CRVS, provides compliance with Principal Design Criterion (PDC) 19, as it relates to maintaining the control room in a safe condition under accident conditions and providing adequate radiation protection. The CRVS has radiation monitors and smoke detectors located in the outside air intake as described in Section 9.4.1. Upon detection of smoke in the outside air duct, the plant control system closes the outside air isolation dampers to isolate the CRB from the environment. The CRB is not pressurized under these conditions.

In conjunction with the CRVS, the design of the CRHS satisfies 10 CFR 50.34(f)(2)(xxviii), in that it provides assurance that, in the event of an accident, radiation doses to operators do not exceed acceptable limits and do not prevent operators from performing control functions.

6.4.2 System Design

The CRE includes the main control room (MCR), reference room, shift manager's office, shift turnover room, office space, and other areas to support MCR operation. These areas are either frequently or continuously occupied. The CRE includes air locks for ingress and egress.

The CRVS provides normal HVAC service to the CRE. The CRVS includes redundant isolation dampers that close to isolate the CRE. The supply of breathing air from the CRHS limits the concentration of carbon dioxide in the CRE to the value given in Table 6.4-1 for 72 hours after the CRE is isolated.

The major components of the CRHS include:

- high pressure air compressor
- high pressure air storage bottles
- air bottle racks
- eductor
- silencers
- piping, valves, and instrumentation

The CRHS design includes an external air supply connection so that the air bottles can be replenished from an offsite source in the event that the compressor is unavailable or local air is contaminated.

6.4.2.1 High Pressure Air Compressor

An air compressor charges the emergency air storage bottles. The compressor uses intake filtration to provide breathing quality air that meets the standards of Compressed Gas Association G-7.1 Level D (Reference 6.4-1).

6.4.2.2 Air Bottles

The CRHS air storage bottles are located on an elevation below the CRE. Table 6.4-1 provides the design parameter for the number of personnel assumed in the CRE. The CRHS includes a sufficient number of bottles to maintain the inventory of breathing air required for 72 hours of operation with 25 percent of the bottles out of service for maintenance.

6.4.2.3 Valves

Compressed air supply lines contain pressure regulating valves located downstream of the common supply header. The outlet pressure of the regulating valves is set to function with a downstream orifice to maintain pressure within the CRE.

Two normally closed air supply isolation valves in parallel in the main supply line maintain the pressure boundary of the common supply header. These solenoid operated valves operate remotely and are located within the CRE pressure boundary downstream of the pressure regulating valves and orifice. These valves initiate breathing air flow to the CRE upon receipt of a signal to open.

The alternate air delivery flowpath contains a normally closed, manually operated valve, located within the CRE pressure boundary. This valve provides a means of manually activating the alternate air delivery flowpath in the event the main air delivery flowpath is inoperable. This alternative flowpath has separate pressure regulating equipment and is capable of supplying air to the CRE at the required flow rate.

Two redundant pressure relief solenoid valves in parallel allow air to exit the CRE during CRHS operation to maintain the CRE at the desired pressure. A manual balancing valve is provided downstream of the solenoid valves.

6.4.3 System Operation

6.4.3.1 Normal Operation

The CRHS is in standby mode during normal plant operation, with the air bottles pressurized and available to provide breathing air and CRE pressurization when actuated.

6.4.3.2 Off-Normal Operation

The plant protection system generates a signal to actuate the CRHS on the following conditions:

- high radiation detected in the CRE supply duct
- loss of power to both CRVS air handling units, after a ten-minute delay
- loss of power to the common augmented direct current (DC) power system battery chargers, after a ten-minute delay

When one of the conditions listed above occurs, the following actions occur:

- The isolation dampers in the CRVS ducts that penetrate the CRE close, isolating the CRE from its surroundings.

- The CRHS isolation valves open, providing the CRE with air from the emergency air storage bottles.
- The CRHS pressure relief isolation valves open, allowing air to discharge from the CRE to the surroundings.
- The CRVS outside air isolation dampers close and the operating supply air handling unit and general exhaust fan stop.

These actions provide CRE occupants with clean breathing air under conditions in which the normal air supply may be contaminated. The ten-minute delay allows operators time to restore power if possible and prevent an unnecessary initiation of the CRHS.

A loss of DC power from the common augmented DC power system to either division of the plant protection system actuates the CRHS.

Operators in the MCR can monitor the differential pressure between the CRE and the surrounding area to ensure that a positive pressure is maintained in the control room with respect to its surroundings. Alarms in the MCR alert the operators to both low and high CRE differential pressure conditions. Operators can manually adjust the CRE balancing valves as needed to restore the desired pressure.

6.4.4 Design Evaluation

As noted in Section 15.0 (Transient and Accident Analyses), no operator actions are required or credited to mitigate the consequences of design basis events. As such, the operators perform no safety-related functions, as defined in 10 CFR 50.2. Therefore, although a habitable control room is provided for the operators to perform other important nonsafety-related functions, the CRE and supporting habitability systems and components, including the CRHS, are not safety-related.

The air bottles and racks, supply piping and associated valves, and pressure relief solenoid valves are designated as Seismic Category I per Regulatory Guide (RG) 1.29. The air compressor and piping up to the isolation valves between the compressor and the air bottles are designed to Seismic Category III standards.

The CRHS does not serve the technical support center (TSC), and therefore does not provide pressurization air to the TSC in the event the CRVS is unavailable. If the CRVS is not able to provide air of acceptable quality for pressurization of the TSC, the TSC is considered uninhabitable and is evacuated. The TSC function is then transferred to another location in accordance with the emergency plan.

6.4.4.1 Radiological Protection

In the presence of significant airborne radiation in the CRE supply duct, the CRVS radiation monitors generate a signal that results in isolation of the CRE, securing CRVS operation, and initiating CRHS operation. The integrated design of the CRE, the CRVS, and the CRHS prevents radioactive materials from entering the CRE that would result in an operator dose exceeding the PDC 19 limit. The CRHS does not interface with other systems that would provide a potential pathway for radioactive materials.

With the CRHS in operation, the CRE is maintained at a positive pressure of at least 1/8-inch water column with respect to its surroundings. Table 6.4-1 summarizes the flow requirements to maintain this pressurization, as well as the flow requirements to maintain the level of carbon dioxide in the CRE below the acceptable limit.

Section 15.0, Transient and Accident Analyses, discusses the radiological consequences of postulated accidents. Analysis shows that the sum of radiation doses to control room personnel from all sources is less than 5 rem for the duration of a postulated accident. These results demonstrate compliance with 10 CFR 50.34(f)(2)(xxviii).

6.4.4.2 Toxic Gas Protection

COL Item 6.4-1: An applicant that references the NuScale Power Plant US460 standard design will comply with Regulatory Guide 1.78 Revision 2, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release."

6.4.4.3 Other Habitability Considerations

When normal HVAC service from the CRVS is not available, the thermal mass of the CRB and its contents limit the temperature increase as shown in Table 6.4-2. The analysis uses the conservative assumption that control room equipment powered by the normal DC power system remains powered for 3 hours. After 72 hours, the CRVS, if available, provides cooling to the CRE.

The MCR has adequate lighting for safe operation of the plant during accidents and all modes of operation. Section 9.5.3 provides information on the plant lighting system.

The CRE includes access to procedures, drawings, and other technical resources useful for mitigating accident conditions. The CRE includes lavatories, kitchen facilities, and break areas.

6.4.5 Testing and Inspection

Section 14.2 provides information regarding preoperational testing.

Inservice testing includes demonstration of the integrity of the CRE in accordance with RG 1.197 and evaluation of the CRHS in accordance with RG 1.196 as specified in Table 6.4-3.

Controls over the availability and reliability of the CRHS and the CRE will be included in the owner-controlled requirements manual.

6.4.6 Reference

6.4-1 Compressed Gas Association Inc., “Commodity Specification for Air,” CGA G-7.1-2018, Sixth Edition, Chantilly, VA.

Table 6.4-1: Control Room Habitability System Design Parameters

Parameter	Value
Personnel assumed present in control room envelope	20 people
Duration of supply air for breathing and pressurization	72 hours
Pressurization	At least 1/8-inch water gauge with respect to adjacent areas. This requires at least 60 scfm based on estimated CRE leakage.
Breathing air supply flow rate	At least 89.4 scfm in order to maintain acceptable levels of CO_2, but not greater than 107 scfm in order to ensure a 72 hour supply of air.
Carbon dioxide level in main control room	5000 ppm maximum
Control room envelope unfiltered inleakage with the CRHS pressurizing the control room envelope to at least 1/8-inch water gauge with respect to surrounding areas	10* cfm plus 5 cfm assumed for air lock operation.

*Because the CRE is pressurized, the unfiltered inleakage is expected to be 0 cfm. However, the dose analysis conservatively assumes 10 cfm of unfiltered inleakage.

Table 6.4-2: Main Control Room Temperature under Passive Cooling Conditions

Time after loss of normal cooling (hours)	Dry Bulb Temperature (ºF)	Relative Humidity (%)	Wet Bulb Globe Temperature (ºF)
0	78.0	60.0	67.3
1	97.8	39.7	76.1
2	102	34.1	76.3
3	104	29.9	75.5
4	99.9	31.7	73.7
8	99.7	28.6	71.9
24	100	24.4	70.1
48	102	23.0	70.2
72	103	22.2	70.5

Table 6.4-3: CRHS Testing

Parameter	Acceptance Criteria
CRE pressure with CRHS in service	≥ 1/8 inch water gauge
CRHS flow rate from storage bottles to CRE via actuated flow path	≥ 89.4 scfm and ≤ 107 scfm
CRHS flow rate from storage bottles to CRE via manual flow path	≥ 89.4 scfm and ≤ 107 scfm
CRE isolation dampers required for CRHS operation	close on CRHS actuation signal
CRHS stored air quality (oxygen, carbon dioxide, carbon monoxide, oil, dew point) supplied by the CRHS air compressor	CGA G-7.1 Level D
CRHS available air inventory	37,900 lb_m
CRHS supply actuation valves and CRE pressure relief valves operation	Stroke open on CRHS actuation signal
CRHS pressure regulating valves operation	Within pressure specifications

6.5 Fission Product Removal and Control Systems

6.5.1 Engineered Safety Features Filter Systems

The design does not use engineered safety feature (ESF) filter systems or ESF ventilation systems to mitigate the consequences of a design basis accident (DBA). The design incorporates the nonsafety-related reactor building heating ventilation and air conditioning system that includes filtering, which is not credited in the dose analysis.

6.5.2 Containment Spray Systems

The design does not use ESF containment spray systems to mitigate the consequences of a DBA.

6.5.3 Fission Product Control Systems

There are no design basis events that result in significant core damage in the design. Core damage occurs only as a result of multiple coincident postulated failures. The only ESF fission product control systems credited to mitigate the consequences of a DBA or a beyond-design-basis core damage event (described in Chapter 15) in the design are the containment vessel (CNV) in conjunction with the containment isolation system. Meeting regulatory requirements does not require active fission product removal systems in the design. The containment design includes inherent natural aerosol removal mechanisms, such as thermophoresis, diffusiophoresis, hygroscopicity, and sedimentation. Chapter 15 discusses these passive removal processes, which deplete elemental iodine and particulates in the containment atmosphere. The limited containment leakage and these natural fission product control mechanisms result in offsite doses that are less than regulatory limits, thus satisfying the fission product control requirement of principal design criterion 41. General Design Criterion (GDC) 42 and GDC 43 do not apply to the design because the design does not include containment cleanup systems that would be subject to those GDCs.

6.5.3.1 Primary Containment

An ESF function of the containment provides a barrier to fission product migration to the environment in the event of a release of core fission products to the containment atmosphere. Table 6.5-1 provides the CNV key attributes. Chapter 15 addresses the fission product control by the CNV and containment isolation system.

6.5.3.2 Secondary Containments

The design does not use a secondary containment to mitigate the consequences of a DBA. The Reactor Building delays releases to the environment and allows fission products to deposit on its surfaces in the event of a severe accident; however, this is not credited in the dose analysis associated with the design.

6.5.4 Ice Condenser as a Fission Product Cleanup System

The design does not use an ice condenser to mitigate the consequences of a DBA.

6.5.5 Pressure Suppression Pool as a Fission Product Cleanup System

The design does not utilize a suppression pool to mitigate the consequences of a DBA.

Table 6.5-1: Containment Vessel Key Attributes

Parameter	Value
Design-basis containment leak rate (L_a)	0.2 wt% per day
Design-basis containment leak rate after 24 hours	0.1 wt% per day
Containment minimum free volume	6000 cu ft

6.6 Inservice Inspection and Testing of Class 2 and 3 Systems and Components

In accordance with General Design Criterion 1, Class 2 and 3 components and systems undergo inspection and testing to quality standards commensurate with their safety functions. Generally recognized codes and standards undergo evaluation for applicability and adequacy. If necessary, these requirements undergo modification to assure quality commensurate with the required safety function. The description of the Quality Assurance Program, which governs the quality standards for the inservice inspection (ISI) program and supporting documentation, is in Chapter 17.

The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) Section XI defines the ISI Program requirements for Class 2 and Class 3 systems and components.

The basis for the ISI program for Class 2 and 3 components is 10 CFR 50.55a(g)(3), which requires that ASME BPVC Class components allow for the inspections detailed in Reference 6.6-1.

The initial ISI program incorporates an edition of the ASME BPVC approved in 10 CFR 50.55a(b). The ISI of components and system pressure tests conducted during successive 120-month inspection intervals comply with the requirements of the of the ASME BPVC incorporated by reference in 10 CFR 50.55a(b) subject to the conditions listed in 10 CFR 50.55a(b). In addition, the optional ASME BPVC cases listed in Regulatory Guide (RG) 1.147 may be used. The ASME BPVC includes requirements for system leakage tests for active components. The ASME BPVC Section XI, Article IWC-5220 for Class 2 pressure retaining components defines the requirements for system leakage tests and ASME BPVC Section XI, Article IWD-5220 for Class 3 pressure retaining components defines the requirements for system leakage tests. These tests verify the pressure boundary integrity in conjunction with ISI.

The Preservice Inspection Program (non-destructive baseline examination) includes the selection of areas subject to inspection, non-destructive examination methods, and the extent of preservice inspections and frequency of inspections.

There are no exemptions to the ISI requirements for Class 2 and 3 systems, structures, and components. There are no relief requests necessary for preservice inspection and first interval ISI examinations for Class 2 and 3 components.

The inservice testing program meets the requirements set forth in Reference 6.6-3, as stipulated in 10 CFR 50.55a(f). Section 3.9.6 provides details of the Inservice Testing Program.

Section 3.13.2 describes the ISI requirements for Class 1, 2, and 3 threaded fasteners.

Technical Specifications provide the administrative controls and surveillance requirements specified by 10 CFR 50.36.

COL Item 6.6-1: An applicant that references the NuScale Power Plant US460 standard design will develop Preservice Inspection and Inservice Inspection Program plans in accordance with Section XI of the American Society of Mechanical Engineers

Boiler and Pressure Vessel Code, and will establish the implementation milestones for the program. The applicant will identify the applicable edition of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code used in the program plan consistent with the requirements of 10 CFR 50.55a. The applicant will, if needed, address the use of a single Inservice Inspection Program for multiple NuScale Power Modules, including any alternative to the code that may be necessary to implement such an Inservice Inspection Program.

6.6.1 Components Subject to Examination

The ASME BPVC Class 2 and 3 components undergo preservice inspection and ISI in accordance with Reference 6.6-1. These components are RG 1.26, Quality Group B and C components, respectively. Section 3.2.2 describes the ASME BPVC Class 2 boundaries, based on RG 1.26 for Quality Group B. Section 3.2.2 describes the ASME Class 3 boundaries, based on RG 1.26 for Quality Group C. ASME Class boundaries for piping penetrating the CNV are depicted by Figure 6.6-1.

The ASME BPVC Class 2 and 3 pressure-retaining components and supports subject to examination include pressure vessels, piping, valves, and their bolting.

Section 3.13.2 describes the preservice inspection and ISI of threaded fasteners, in accordance with the requirements and the criteria of ASME BPVC, Section XI for bolting and mechanical joints used in ASME BPVC Class 2 systems.

Examination of the ASME BPVC Class 2 pressure retaining components and integral attachments is in accordance with the requirements of ASME BPVC Section XI Article IWC-2500.

Examination of the ASME BPVC Class 3 pressure retaining components and integral attachments is in accordance with the requirements of ASME BPVC Section XI, Article IWD-2500.

6.6.2 Accessibility

In accordance with 10 CFR 50.55a(g)(3), Class 2 and Class 3 systems and components (including supports) have access to enable the performance of inservice examinations.

The design and layout of the Code Class 2 and 3 systems allows for the performance of the ISI requirements contained in ASME BPVC Section XI, Articles IWC-2000 and IWD-2000, and as defined in the ISI program.

Design, materials, and geometry do not restrict inspections required by ASME BPVC Section XI. The piping arrangement allows for adequate separation of piping welds so that there is space available to perform ISI. Sections of straight pipe of sufficient length to conduct inspections of separate adjacent welds are provided. The location of welds in piping that passes through walls is away from the wall as required by ASME BPVC Section XI. Component nozzles, tees, elbows, valves, branch connections, and other fittings remain disconnected unless they have an extended tangent length adjacent to the weld to permit weld examination.

In accordance with ASME BPVC Section XI, Article IWA-1500(d), there is space for examinations alternative to those specified, in the event that there are structural defects or modifications requiring alternative examinations. In accordance with ASME BPVC Section XI, Article IWA-1500(e), there is space for necessary operations associated with repair or replacement activities.

Piping and pipe support locations, insulation, hangers, and stops do not interfere with the inspection equipment and personnel. Where this cannot be done, the components are removable.

The Nuscale Power Plant ensures accessibility of welds and other areas requiring periodic inspection. Reinforcing pads, supports, piping, and equipment do not obstruct welds. Removable insulating materials above the containment vessel provide accessibility for ISI.

Areas requiring inspection and servicing of valves provide working platforms. Temporary or permanent working platforms, walkways, scaffolding, and ladders facilitate access to piping and component welds. The components and welds requiring ISI allow for the application of the required ISI methods. Such design features include sufficient clearances for personnel and equipment, maximized examination surface distances, access, favorable materials, weld-joint simplicity, elimination of geometrical interferences, and proper weld surface preparation.

Some ASME BPVC Class 2 and 3 components are in modules fabricated offsite and shipped to the site. The modules' design and engineering provides access for preservice inspections, ISI, and maintenance activities.

6.6.3 Examination Techniques and Procedures

The ISI examination techniques include visual, surface, and volumetric examination methods. The examination procedures describe the examination equipment, inspection techniques, operator qualifications, calibration standards, flaw evaluation methods, and records. The techniques and procedures meet the requirements of ASME BPVC Section XI, Articles IWA-2000, IWC-2000 and IWD-2000. Preservice inspection and subsequent ISI use equivalent equipment and techniques.

Preservice inspections occur once, in accordance with ASME BPVC, Section XI, Article IWC-2000 and Article IWD-2000.

Qualification of ultrasonic examination equipment, and procedures is in accordance with ASME BPVC, Section XI.

The ASME BPVC Case N-849, "In Situ VT-3 Examination of Removable Core Support Structure Without Removal" meets the conditions in NRC Regulatory Guide 1.147.

Alternate examination methods are not used.

6.6.4 Inspection Intervals

For inside containment piping, the inspection intervals for Class 2 examinations in the ISI Program follow ASME BPVC Section XI, Articles IWA-2400 and IWC-2400. ASME BPVC Section XI, Article IWC-2500, Examination Category C-F-1 requires volumetric and surface examination of Class 2 piping welds larger than nominal pipe size (NPS) 4 and equal to or greater than 3/8 inch thick, and for piping welds NPS 2 or larger greater than 1/5 inch thick. However, ASME BPVC Section XI, Article IWC-1220 exempts most Class 2 piping NPS 4 and smaller from ISI. The exceptions are high pressure injection systems and auxiliary feedwater systems, which do not exist in the design. For pipe break exclusion zone considerations, welds in the main steam, feedwater, and decay heat removal system piping undergo volumetric examination.

For outside containment pipe break exclusion considerations, welds from the containment vessel nozzles to the isolation valves undergo volumetric examinations. This includes pressurizer (PZR) spray, chemical and volume control (CVC) injection and discharge, reactor pressure vessel (RPV) high point degasification, and main steam and feedwater piping. These piping systems range from NPS 2 to NPS 12.

Piping beyond the containment isolation valves is ASME BPVC Section III, Class 3, or ASME BPVC B31.1. The ASME BPVC Section XI, Class 3 piping requires a VT-2 examination during system leakage testing. Welded attachments that perform a component support function require a visual/VT-1 examination. Piping constructed to ASME BPVC B31.1 is outside the scope of ASME BPVC Section XI.

The inspection intervals for Class 3 examinations in the ISI Program are in accordance with ASME BPVC Section XI, Article IWD-2400 and Article IWF-2400. Piping beyond the containment isolation valves is ASME BPVC Section III, Class 3, or ASME BPVC B31.1. The ASME BPVC Section XI, Class 3 piping requires a VT-2 examination during system leakage testing. Welded attachments that perform a component support function require a visual/VT-1 examination. Piping constructed to ASME BPVC B31.1 is outside the scope of ASME BPVC Section XI.

The Examination Program for the 120-month inspection interval is described in the ISI Program. The Initial Inservice Inspection Program incorporates the ASME BPVC approved in 10 CFR 50.55a(b) prior to initial fuel load, as specified in 10 CFR 50.55a. The ISI of components and system pressure tests occur during successive 120-month inspection intervals and comply with the requirements of the ASME BPVC incorporated by reference in 10 CFR 50.55a(b), subject to the conditions listed in 10 CFR 50.55a(b).

6.6.5 Examination Categories and Requirements

Examination categories and requirements for Class 2 systems and components are in accordance with ASME BPVC Section XI, Article IWC-2500 and Table IWC-2500-1.

For Class 3 systems and components, the examination categories and requirements are in conformance with ASME BPVC Section XI, Article IWD-2500 and Table IWD-2500-1.

The examinations and alternative tests for fixed structural supports of Class 2 and 3 systems and components are in accordance with ASME BPVC Section XI, Article IWF-1200.

Examination categories for ASME BPVC Class 2 pressure retaining components include C-F-1 (pressure retaining welds in austenitic stainless steel or high alloy piping) and F-A (component supports).

Examination categories for ASME BPVC Class 3 pressure retaining components include D-A (welded attachments for vessels, piping, pumps, and valves) and F-A (component supports).

6.6.6 Evaluation of Examination Results

Performance of evaluations of Class 2 examination results are in accordance with ASME BPVC Section XI, Articles IWA-3000 and IWC-3000.

Performance of evaluations of Class 3 examination results are in accordance with ASME BPVC Section XI, Articles IWA-3000 and IWD-3000.

Documentation of the results of the examinations and evaluations are in accordance with ASME BPVC Section XI, Article IWA-6000.

The procedures for repair and replacement of Class 2 and 3 components are in accordance with ASME BPVC Section XI, Article IWA-4000.

6.6.7 System Pressure Tests

Performance of system pressure tests of Class 2 systems are in accordance with ASME BPVC Section XI, Articles IWA-5000 and IWC-5000 and Table IWC-2500-1.

Performance of system pressure tests of Class 3 systems are in accordance with ASME BPVC Section XI, Articles IWA-5000 and IWD-5000 and Table IWD-2500-1.

6.6.8 Augmented Inservice Inspection to Protect Against Postulated Piping Failures

The Class 2 and Class 3 ISI program includes augmented ISI to protect against postulated piping failures. These inspections provide reasonable assurance of the structural integrity of cold-worked austenitic stainless steel components (Section 6.1.1).

Section 3.6.2 defines high-energy piping systems as fluid systems that, during normal plant conditions, are either in operation or maintain pressurization under conditions where either the maximum operating temperature exceeds 200 degrees or the maximum operating pressure exceeds 275 psig.

The examination areas include the high-energy fluid piping systems described in Section 3.6.1 and Section 3.6.2.

Augmented ISI meets the requirements of BTP 3-4 for areas designated as containment penetration areas (Section 3.6). The augmented requirements include vessel safe-end and nozzle welds connected to piping or containment isolation valves within containment penetration areas.

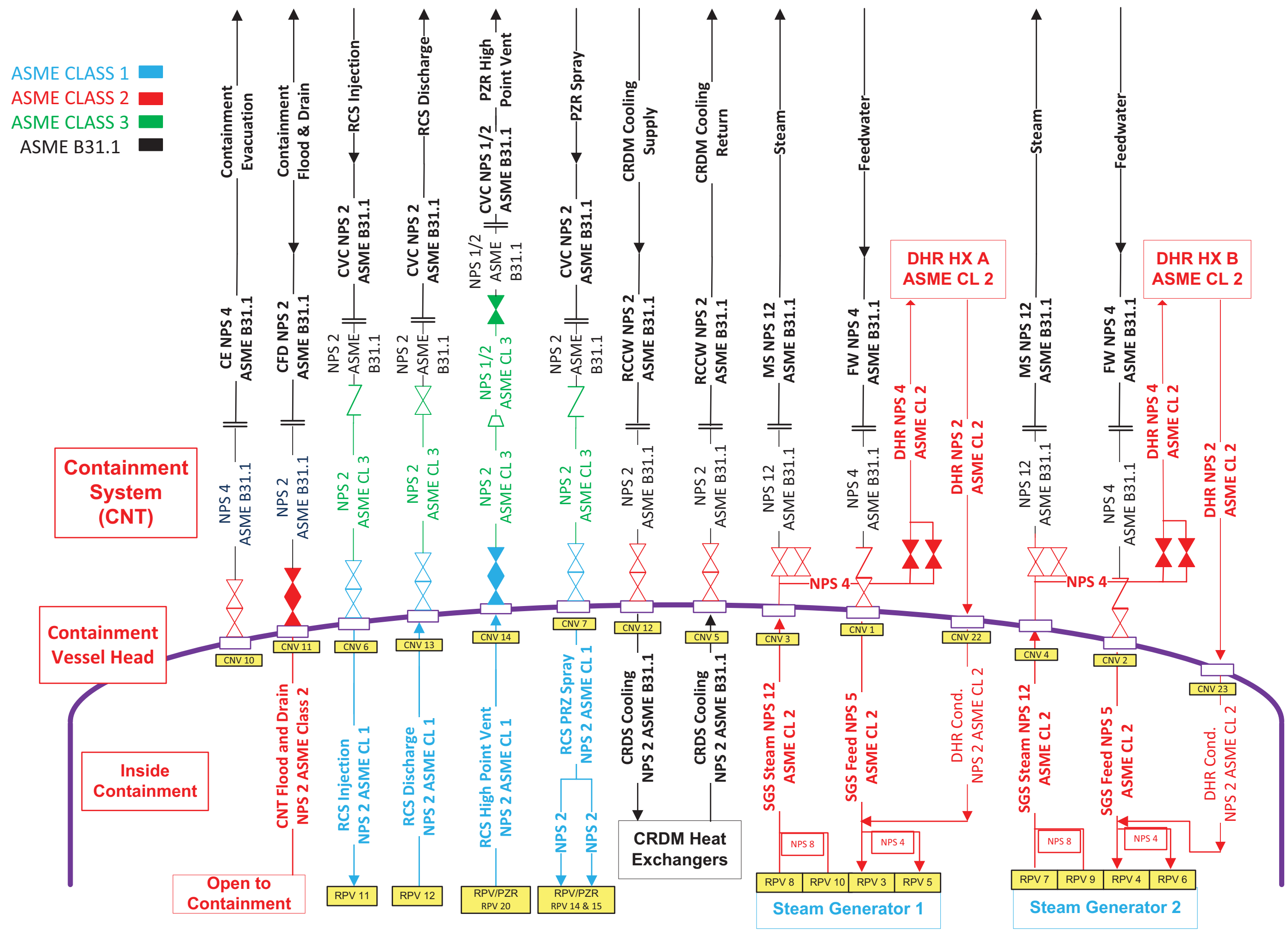
6.6.9 References

6.6-1 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2017 edition, Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Components," New York, NY.

6.6-2 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.

6.6-3 American Society of Mechanical Engineers, Operation and Maintenance of Nuclear Power Plants, ASME OM-2017, New York, NY.

Figure 6.6-1: ASME Class Boundaries for NuScale Power Module Piping Systems



6.7 Main Steamline Isolation Valve Leakage Control System (BWR)

The main steamline isolation valve leakage control system is applicable to boiling water reactor designs. The NuScale Power Plant design is a pressurized water reactor. Therefore, this section is not applicable to the NuScale Power Plant application for design certification.

Section B

The table below provides the NuScale responses to each of the Nuclear Regulatory Commission readiness assessment observations on draft Chapter 6, “Engineered Safety Features” of the Standard Design Approval Application.

Section	Observation	Response
6.1.1	The NuScale SDA US460 Final Safety Analysis Report (FSAR) Section 6.1.1 indicates that F6NM (Type 415) Martensitic SS is used to fabricate the lower CNV shell above the RPV flange elevation, the lower flange, upper flange, upper shell, and top head. This material is permitted by way of ASME Code Case N-774 Use of 13Cr-4Ni (Alloy UNS S41500) Grade F6NM Forgings Weighing in Excess of 10,000 lb (4540 kg) and Otherwise Conforming to the Requirements of SA-336/ SA-336M for Class 1, 2 and 3 Construction Section III, Division 1. Code Case N-774 is listed in RG 1.84, Design, Fabrication, and Materials Code Case Acceptability, ASME Section III, Revision 39, December 2021, as acceptable without conditions. This is a substantial change from low-alloy steel used in the previously approved NuScale design. The NRC staff is unaware of any pressure vessels fabricated from martensitic SS being used in any nuclear power plant in the United States or abroad, in safety-related or nonsafety-related systems. In addition, this material is not typically used to fabricate large pressure vessels in any industry although there may be certain applications that the NRC staff is not aware of. There is no nuclear operating experience with this material in its intended application and environment. The draft FSAR Section 6.1.1 does not provide any description of the acceptability of this material for its intended application. In addition, due to the potential difficulty in fabricating pressure components using this material, the staff is concerned about through thickness impact properties of weld heat affected zones. The use of this material would represent manufacturing challenges as it can be difficult to weld and is highly sensitive to hydrogen cracking and thus special controls during welding will be needed (Appendix B to 10 CFR Part 50). The applicant needs to provide information regarding the justification for the use of this material in its intended application, a description of the special controls that will be used during welding that will ensure that excessive repairs will not be required which could results in potential degradation issues during the life of the CVN, and a description of how through thickness impact properties will be assured. The NRC staff notes that there are no exemptions for post-weld heat treatment, by way of using preheat, regardless of the thickness of the weld. As a result, ALL	The Final Safety Analysis Report (FSAR) demonstrates the NuScale Power Module (NPM) design meets these requirements. Section 6.1.1.1 of the FSAR indicates that Engineered Safety Features (ESF) components (i.e. containment vessel [CNV]) are constructed using materials permitted by American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section III, and Division 1 and conform to the fabrication, construction and testing requirements of ASME BPVC, Sections II and III and meet the requirements of NB-2000, NC-2000 and NF-2000. This is consistent with the acceptance basis in Design Certification Application Safety Evaluation Report Section 6.1. SA-336, Gr F6NM material is included as an allowable material by RG 1.84 by NRC acceptance without condition Code Case N-774 (Ref: ML14008A332). RG 1.84, Rev. 39 which is incorporated by reference into 10 CFR 50.55a, explicitly lists Code Case N-774 which permits the use of F6NM materials for forgings in excess of 10,000 lb. Therefore, the use of SA-336 Gr F6NM for the CNV meets the requirements of 10 CFR 50.55a. This provides a basis for acceptance of F6NM consistent with applicable regulations cited in SRP 6.1. Compliance with ASME BPVC Section II and III as above also demonstrates the fabrication requirements imposed on construction of the CNV (including welding) are consistent with 10 CFR 50.55a and GDC 1. As discussed during a pre-application meeting concerning F6NM material held in October 2022, fracture mechanics testing of F6NM material is ongoing and is due to complete in 2023 and thermal embrittlement testing will complete in

Section	Observation	Response
	welds regardless of size require a preheat and postweld heat treatment to be performed.	2024. The results of this testing can be made available to the staff during a review audit.
6.3	Supporting information for the Emergency Supplemental Boron (ESB) addition system related to performance and operability is missing from this section. Boron dissolution testing and operating requirements related to the ESB should be presented or referenced in this section. Section 6.3 is missing a detailed description of the ESB system design (not just the ESB function description) or reference to the system design description.	Dissolution testing and the dissolution rate are documented in the Extended Passive Cooling Topical Report (TR-124587). This topical report is part of the final SDAA submittal. Section 6.3 provides a detailed description of the ESB system.
6.3	Section 6.3 is missing discussion or reference to information related to the operability of ESB considering potential dose to operators during outages and impact of dose on the system. The boron pellets of ESB need to be reloaded during each outage when the containment water is drained below the basket elevation. The drainage of water in the containment could result in higher dose rate than that with a flooded containment.	Section 6.3 describes the remote operation that loads boron oxide pellets. The boron oxide pellets will be loaded into a hopper that feeds the dissolver. The hopper is in the containment near the reactor head and will be filled when the module is moved to the operating bay around the time of in-core instrumentation (ICI) insertion. At this point in time, the CNV is flooded. The dissolvers are fed by hoppers once the CNV is closed and the CNV is drained below the dissolvers and do not require personnel in the area during dissolver loading. Section 6.3 indicates: The dissolvers are fed by hoppers at the CNV work platform and do not require personnel in the area during dissolver loading. Therefore, the design and loading sequence limits the dose concern to operators during the boron oxide pellet loading process.
6.3	The ECCS description provided in SDA Section 6.3 is missing inclusion of the decay heat removal system (DHRS) being part of the ECCS, and associated portions of the secondary side, although it is relied upon to mitigate the consequences of a loss-of-coolant accident (LOCA) through removing decay heat and maintaining core cooling. FSAR Table 3.2-2 of the SDA has not been updated to classify DHRS and relevant secondary side steam system as ASME Code Class 1 components consistent with all other ECCS components. If the DHRS system function is as described above, it needs to meet General Design Criteria 2, 14, 27, 30, 31, 50, 51, 52, and 53. The relevant information would need to be added in Sections 3.2.2, 3.5, 3.6, 3.7, and 4.6.2.	Regulatory Guide (RG) 1.26 provides a quality classification system that may be used to determine quality standards acceptable to the NRC staff for "Satisfying General Design Criterion 1 of Appendix A" and its applicability extends to applicants for standard design approval under 10 CFR Part 52. The RG indicates that Quality Group A is "limited to components that are part of the reactor coolant pressure boundary" and that Quality Group B applies to "systems or portions of systems important to safety that are designed for (1) reactor shutdown or (2) residual heat removal." The DHRS in the NPM design serves a residual heat removal function and is not part of or hydraulically connected to the

Section	Observation	Response
		reactor coolant pressure boundary. Therefore, the DHRS is correctly classified as Quality Group B (Class 2) and does not meet to be classified Quality Group A (Class 1). No changes to the FSAR are necessary.
6.2.1/6.2.2	Even though the draft SDAA identifies the design-basis events (DBEs) conditions, the modeling details of the limiting events are referenced to the LOCA methodology topical report cited as Reference 6.2-2. Now, Tables 6.2-6 and 6.2-7 have credited DHRS actuation in the sequence of both the peak containment pressure and wall temperature events, but no further information is provided on the DHRS T/H now credited to the revised safety analyses. The staff expects the relevant DHRS modeling and testing details to be available in the future licensing topical reports, as appropriately revised for the SDAA. The staff will also review the potential deterioration of DHRS performance due to the transport of radiolytic gases from the core and accumulation in the DHRS, and its adverse impact on in-tube condensation.	DHRS modeling and assessment information are in the non-LOCA topical report (TR-0516-49416) as discussed in pre-application engagements with the staff. The secondary side fluid in the SG/DHRS loop is shielded by the primary system water from effects of radiolysis and therefore there is not an additional source term of noncondensable gases in the DHRS. The sequence of events for the maximum temperature and pressure cases are identical because they result from the same event - the discharge line break. Table 6.2-8 has been deleted and the text now only refers to Table 6.2-7 for both sequence of events.
6.2.1	Table 6.2-1 has documented the revision of the containment internal design pressure from 1050 to 1200 psia, and the containment design temperature from 550 F to 600 F. Due to the revision of these most important containment design parameters by the containment material change to Code Case N-774 SA-336, Gr F6NM, a reference needs to be included in Chapter 6 for the [technical/topical] report where the technical justifications will be provided for revising these containment design parameters. The staff would also need the material and thermophysical properties of the revised material, as used by the applicant in the updated design and safety analyses.	This US460 Standard Design Approval Application (SDAA) is a standalone application and justification for differences from previous applications are not given in this SDAA. This information is documented in NuScale calculations and analyses. These can be made available for the staff to review in an audit.
6.2.1	Section 6.2.1.3.5 has a brief description of the long-term cooling (LTC) model. It states, “The previously described methodology and model are utilized for this purpose. This demonstrates adequate long-term containment removal.” The applicant needs to include either a justification or a [technical/topical] report reference in Chapter 6 for making this conclusion.	Extended Passive Cooling Topical Report (TR-124587) describes the long term cooling methodology and model This LTR is part of the SDAA submittal.

Section	Observation	Response
6.2.1/6.2.2	The SDAA CNV is a passive design that relies on a passive autocatalytic recombiner (PAR) to preclude combustible gas mixtures by limiting the oxygen concentration in the containment atmosphere to a level (<4% by volume) that does not support combustion for at least 72 hours. The prescribed 4% oxygen limit would also correspond to an 8% volume concentration limit for radiolytic hydrogen, which means an overall 12% non-condensable presence by volume in the containment during the event. The staff recognizes that oxygen and hydrogen generation from radiolysis is a slow process that may not impact short term CNV peak pressure calculations since CNV peak pressure occurs early in the event. However, NuScale would need to demonstrate the LTC capacity of the containment in the presence of at least 12% non-condensables due to radiolytic gases.	Extended Passive Cooling Topical Report (TR-124587) addresses the impact of non-condensable gases on long term cooling, which is addressed by the SDAA submittal.
6.2.1/6.2.2	The staff understands that Containment Response Analysis Methodology will now be merged into the LOCA topical report. Section 6.2.1.3 recognizes the distinct nature of containment conservatism for safety analysis and identifies its various elements for maximizing the mass and energy release and minimizing the performance of the containment heat removal systems. The staff expects the consolidated LOCA topical report to appropriately capture the different levels of conservatisms for the containment and RPV safety analyses, which would result into two different sets of NRELAP5 decks. The staff also expects NuScale to have justifications for its proposed approach in the draft NuScale SDAA Chapter 6.	The LOCA topical report (TR-0516-49422) addresses the assumption differences between the LOCA model and the containment response model. The internal calculations that document these assumptions can be available for NRC audit during SDAA review.
6.1.1	Section 6.1.1 does not provide any detail regarding dissimilar metal welds. Dissimilar metal welds (DMWs) are more susceptible to degradation inservice than similar metal welds. The staff would most likely request additional information on this subject if the information is not included in the FSAR, particularly for DMWs involving F6NM martensitic SS. The staff would expect description of all DMWs.	The following is added to Section 6.1.1: When welding together different material types there may be a need for a dissimilar metal weld (DMW) weld joint. When a DMW is necessary for fabrication or design a suitable weld material will be used that is of adequate strength and composition for the design and operating conditions. The design and fabrication of the weld will meet the applicable code sections as required.
6.2.7	Section 6.2.7 includes discussion presuming that the upper CNV is ferritic, however F6NM as proposed in Section 6.1.1 is a martensitic specification. Consequently, text in Section 6.2.7 does not appear to have been updated consistent with Section 6.1.1. Consequently, the staff have no basis to provide substantive comments on Section 6.2.7.	The containment vessel is composed of two materials; SA-336, Grade F6NM martensitic steel and SA-965 FXM-19 austinetic stainless steel. SA-336 Grade F6NM is a ferritic material that meets the requirements of NB-2300 of the ASME Boiler and Pressure Vessel Code Section III and is designed to protect against non-ductile fracture in

Section	Observation	Response
		accordance with ASME Boiler and Pressure Vessel Code Section XI Nonmandatory Appendix G. Within the ASME BPVC, 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, is explicitly chosen specifically for its superior fracture toughness and resistance to neutron embrittlement. Section 6.2.7 describes the CNV materials.
6.2.7	Reference is made to Section 3.1.5 in Section 6.2.7 regarding adequacy of fracture toughness margin; however, Section 3.1.5 was not made available to the staff for the preapplication review. Consequently, the staff have no alternative basis to provide substantive comments on Section 6.2.7.	FSAR Section 3.1.5 is provided as part of the FSAR. The discussion of compliance with the fracture prevention requirements of GDC 51 for the CNV pressure boundary are not updated for the FSAR. The CNV is constructed of two materials; SA-965, Gr FXM-19 austenitic stainless steel and SA-336 Grade F6NM martensitic steel. The SA-336, Grade F6NM martensitic steel is a ferritic pressure retaining material that is used for the upper CNV pressure boundary. The CNV components are designed to protect against non-ductile fracture in accordance with ASME BPVC Section XI Nonmandatory Appendix G. The SDAA FSAR Section 6.2.7 indicates that the ferritic containment pressure boundary materials satisfy the fracture toughness criteria for ASME BPVC Section III, Subsection NB. This is also stated in FSAR Section 6.1.1, which indicates that the "design, fabrication, and materials of construction of the CNV includes sufficient margin that provides reasonable assurance that the CNV pressure boundary will not undergo brittle fracture." This demonstrates that the ferritic martensitic SA-336, Grade F6NM material is protected against brittle fracture. As indicated by Section 6.2.7.4 of the Design Certification safety evaluation report within the ASME Code, "detailed fracture toughness requirements are placed on non-ferritic materials, as non-ferritic materials exhibit sufficient inherent fracture toughness that additional requirements are deemed

Section	Observation	Response
		unnecessary." Accordingly, FSAR Section 6.2.7 was updated by adding the following: The design meets the requirements of GDC 51. The design, fabrication, and construction materials for the CNV system includes sufficient margin to provide assurance that the containment pressure boundary will not undergo brittle fracture and the probability of rapidly propagating fracture will be minimized under operating, maintenance, and postulated accident conditions. The ferritic containment pressure boundary materials satisfy the fracture toughness criteria for ASME BPVC Section III Division 1, Class 1 and 2 components.
6.2.1.1.3/6.2.2	Section 6.2.1.1.3 states that both containment limiting peak pressure and temperature calculations assumed “no single active failure.” However, Section 6.2.2 states that the evaluation of mass and energy released into the containment following a postulated main steam-line break or feedwater line break considered “the automatic isolation of the main steam and feed water lines and single failures.” Please clarify considering that Principal Design Criterion 35 requires “Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that the system safety function can be accomplished, assuming a single failure.”	The containment response analysis considered failure of a Main Steam Isolation Valve or Feedwater Isolation Valve and determined that the cases assuming these failures were not limiting. Therefore, the limiting cases did not assume any single failure. The FSAR Section 6.2.1.3.6 states that consideration of no single failure produces a more limiting result. Therefore, since consideration of no single failure produces limiting results there is no inconsistency between sections.
6.3	The SDAA should include drawings of the new key design features (e.g., ECCS flow restricting venturis, and ESB feature) depicting details for location, layout and configuration.	Figure 6.3-2 provides a simplified flow schematic for the entire Emergency Core Cooling System (ECCS), including ECCS Supplemental Boron. ECCS P&ID Figure 6.3-1 provides a flow schematic of the entire ECCS that includes the venturis and ECCS components.
6.3	SDAA Section 19.3 states that the passive safety functions of the nuclear power modules (NPMs) are relied on for 7 days following a design-basis event as justification to not need regulatory treatment of nonsafety systems (RTNSS) B SSCs. Accordingly, this 7-day design feature of the ECCS should be defined and described within SDAA Section 6.3.	The ECCS is designed to perform its safety function beyond 7 days following a design basis event without support from a non-safety system. SDAA Section 6.3.1 includes this design function.

Section C

TR Number	TR Title
TR-123952-NP	NuScale Containment Leakage Integrity Assurance

Licensing Technical Report

NuScale Containment Leakage Integrity Assurance

December 2022

Revision 0

Docket: 52-050

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Licensing Technical Report

Revision History		
Revision	**Date**	**Notes**
0	12/2022	Initial Issuance

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Abstract

This technical report describes the NuScale Power, LLC (NuScale) Containment Leakage Integrity Program (CLIP). This program provides assurance that leakage integrity of containment is maintained and that containment leakage does not exceed allowable leakage rate values. The CLIP is a consolidation of programs described in the US460 Standard Design Final Safety Analysis Report (FSAR). CLIP elements are implemented under other programs as described in this report and the FSAR. The requirements of Title 10 of the Code of Federal Regulations (CFR) Part 50, Appendix A, General Design Criterion 52 (GDC 52) state that containments shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure. The requirements of 10 CFR 50, Appendix J, Type A tests, include test specifications directly related to GDC 52 design requirements. The CLIP integrates

- containment vessel (CNV) flange design, which remains sealed at design pressure.
- preservice leak test at design pressure performed for all CNVs.
- initial (first-of-a-kind) containment vessel preservice leak test at design pressure performed with the vessel fully assembled with flanges in place.
- preservice 10 CFR 50, Appendix J, Type B testing.
- preservice 10 CFR 50, Appendix J, Type C testing.
- post-installation and repair inspection and testing.
- inservice inspection and examination.
- periodic 10 CFR 50, Appendix J, Type B testing.
- periodic 10 CFR 50, Appendix J, Type C testing.

This report provides relevant details of the CNV and containment systems (CNTS) designs, which support the CLIP in assuring containment leakage integrity. The CLIP provides leakage integrity assurance equivalent to the containment leakage testing requirements of 10 CFR 50, Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors.” This report provides supplemental information designed to inform the Nuclear Regulatory Commission evaluation of NuScale Final Safety Analysis Report, Section 6.2.6.

Executive Summary

This technical report describes NuScale's Containment Leakage Integrity Program (CLIP). The CLIP, supported by the NuScale containment vessel (CNV) and containment system (CNTS) design, provides leakage integrity assurance for the NuScale containment. As discussed in the NuScale US460 Standard Design Final Safety Analysis Report (FSAR), Section 6.2.6, the design supports an exemption from the requirements of Title 10 of the Code of Federal Regulations (CFR) Part 50, Appendix A, General Design Criterion (GDC) 52 and 10 CFR 50, Appendix J, which specify the design for and performance of preoperational and periodic integrated leak rate testing at containment design pressure.

The CLIP, supported by the design and analysis of the CNV and CNTS, provides leakage integrity assurance for the containment. The CLIP is a consolidation of programs described in the FSAR. CLIP elements are implemented under other programs as described in this report and the US460 design. Each element of the CLIP is consistent with a corresponding element of an approved program for reactor pressure vessels or large light water reactor containments. The primary CLIP elements that provide leakage integrity assurance are

- containment vessel flanges that are designed to remain sealed at design pressure.
- factory inspection and testing, including preservice leak testing at design pressure with zero visible leakage, to ensure initial containment leakage integrity in accordance with Inspections, Tests, Analyses, and Acceptance Criteria.
- preservice and periodic Type B and C testing to ensure overall containment leakage does not exceed allowable leakage rate values (i.e., to quantify overall containment leak rates).
- American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) Section III (Reference 9.1), and ASME Operation and Maintenance (OM) and ASME Section XI to ensure continued leakage integrity (i.e., ensures no unknown leak pathways develop over time).
- Type B and Type C testing, inspections, and administrative controls (e.g., configuration management and procedural requirements for system restoration) to ensure leakage integrity associated with activity-based failure mechanisms (i.e., ensures CNV flanges and containment isolation valves [CIVs] remain within allowable leakage rate values after system and component modifications or maintenance).

While the CLIP described in this report does not conform to GDC 52 or Type A testing requirements, the advanced design and CLIP provide more comprehensive leakage integrity assurance than was considered when the subject regulations were adopted. This report provides a detailed overview of the key aspects of the testing, inspection, and design that ensures containment leakage integrity is maintained, including:

- the overall containment leakage rate testing program, including the scope of the Type B and Type C testing to ensure adequate margin against design-basis leak rates
- Type B testing adequacy as ensured by the following.
  - Containment vessel flanges are designed to remain sealed at design pressure.
  - Preservice design pressure leakage test of the CNV with bolted flanges in place using as-designed flange covers installed with the design bolting materials, design bolting

assembly preloads, and design seals installed to demonstrate no visible leakage at design pressure. The test is performed at design pressure and a minimum temperature of 70 degrees F and a maximum temperature of 140 degrees F to minimize the possibility of brittle fracture.

    - The upper and lower halves of the CNV are assembled for the test of the first NuScale Power Module (NPM) of the initial plant.
    - After successful testing, the upper and lower halves of other CNVs may be tested separately.

  - Covers with electrical and instrumentation penetrations may be substituted with blank covers having the same sealing design.
  - Flange assembly uses positive verification to ensure proper flange loading from each stud.
  - The test configuration may use blanked-off pipe ends in place of the CIVs.
  - The acceptance criterion is no observed leakage from seals at examination pressure.
  - The emergency core cooling system trip valve and reset valve body-to-bonnet joint seals are not considered to be a flanged connection and are not included in the containment flange bolting calculation or preservice design pressure leakage test.

- the CNTS design as it applies to the containment function
- the ISI Program as it applies to the CNV
- the IST Program as it applies to CIVs
- materials selection and aging degradation assessment

As described in this report, the containment design and CLIP ensure leakage integrity of containment is maintained and containment leakage does not exceed allowable leakage rate values. This report provides supplemental information designed to inform the Nuclear Regulatory Commission evaluation of NuScale Final Safety Analysis Report, Section 6.2.6 and SDAA, Part 7.

1.0 Introduction

1.1 Purpose

The purpose of this technical report is to describe the Containment Leakage Integrity Program (CLIP) as well as the containment vessel (CNV) and containment system (CNTS) design elements that ensure leakage integrity. This report evaluates the plant design and the CLIP against the requirements in Title 10 of the Code of Federal Regulations (CFR) Part 50, Appendix J as incorporated in DesignSpecific Review Standard, Section 6.2.6. This evaluation includes an assessment of the capability of the containment design to meet specific testing requirements in 10 CFR 50, Appendix J. This report identifies Type A requirements that are not to be applied because certain functions are achieved differently compared to existing large light water reactors (LLWRs) for which 10 CFR 50, Appendix J requirements were developed and written.

This report describes the approach to Type B and Type C testing through an evaluation of the containment design. This report provides supplemental information designed to inform the Nuclear Regulatory Commission evaluation of NuScale Final Safety Analysis Report (FSAR) Section 6.2.6 . As shown in the table below, each element of CLIP is consistent with a corresponding element of an approved program for reactor pressure vessels (RPVs) or LLWR containments, which have been incorporated within the FSAR. This report provides a consolidated description of inspection, testing, and examination elements from several programs described in the FSAR related to containment leakage integrity. This report does not describe any elements that are not described in the FSAR.

Table 1-1 Containment Leakage Integrity Program Elements

CLIP Element	Licensing Requirement
CNV flange design	FSAR COL Item 6.2-2
Preservice inspection (Section 4.0)	ASME BPVC Class Section III (FSAR 6.2.6)
Fabrication structural integrity testing (Section 4.0)	ASME BPVC Class Section III (FSAR 6.2.6)
Preservice leakage testing	FSAR 6.2.6, Chapter 14
Preservice Type B and Type C local leak rate test (LLRT) (Section 4.0)	Technical Specifications (Section 5.5.9)
Preservice Type B and Type C LLRT (Section 4.0)	Initial Test Program (FSAR Table 14.2-43)
Post-installation/repair inspection & testing (Section 5.0)	ASME BPVC Class Section III / XI (FSAR 6.2.6)
Post-installation/repair inspection & testing (Section 5.0)	TS (, Section 5.5.9)
Inservice inspection (ISI) and examination (Section 5.0)	ASME BPVC Class Section XI (FSAR 6.2)
Periodic Type B and Type C LLRT (Section 5.0)	TS (, Section 5.5.9)

1.2 Scope

This report describes the CLIP for the design and evaluates the CLIP against 10 CFR 50, Appendix J. This report describes

- the overall containment leakage rate testing (CLRT) program, including the scope and frequency of Type B and Type C testing of CNV penetrations.

- the CNTS design as it applies to CNV design and the containment function.
- materials selection and aging degradation as it applies to the containment pressure boundary.
- the ISI Program as it applies to the CNV.
- the inservice testing (IST) program as it applies containment isolation valves (CIVs).
- Type A integrated leak rate testing (ILRT) challenges.

1.3 Background

The design supports an exemption from GDC 52 to design the containment for ILRT and supports a licensee's exemption from 10 CFR 50, Appendix J, for the performance of Type A preoperational and periodic integrated leakage rate testing.

This technical report describes the containment testing, inspection, and design criteria that ensure leakage integrity of containment is maintained and containment leakage does not exceed allowable leakage rate values, thereby satisfying the underlying purpose of GDC 52 and Type A testing.

1.4 Containment Leakage Integrity Assurance

The CLIP provides containment leakage integrity by

- demonstrating the containment design can use LLRT to adequately ensure containment leakage integrity.
  - Containment vessel flanges are designed to remain sealed at design pressure.
  - Preservice design pressure leakage test of the CNV with CNV bolted flanges in place utilizing as-designed flange covers installed with the design bolting materials, design bolting assembly preloads, and design seals installed to demonstrate no leakage at design pressure.
    - The upper and lower halves of the CNV are assembled for the test of the first NPM of the initial NuScale plant.
    - After successful testing, the upper and lower halves of other CNVs may be tested separately.
    - Covers with electrical and instrumentation penetrations may be substituted with blank covers having the same sealing design.
  - Flange assembly requires positive verification to ensure proper flange loading from each stud.
- ensuring no unknown leakage pathways exist.
- quantifying overall containment leak rates by LLRTs that provide accurate results for every potential leak path.
- ensuring no unknown leak paths develop over time due to degradation.
- ensuring no unknown leak paths develop due to activity-based failure mechanisms.

1.5 Regulatory Requirements

10 CFR 52.137(a) states in part:

The application must contain a final safety analysis report that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole, and must include the following information:

(3) The design of the facility including:

(i) The principal design criteria for the facility. Appendix A to 10 CFR part 50, general design criteria (GDC), establishes minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have previously been issued by the Commission and provides guidance to applicants in establishing principal design criteria for other types of nuclear power units;

(ii) The design bases and the relation of the design bases to the principal design criteria

The introduction to 10 CFR 50, Appendix A, states in part:

Also, there may be water-cooled nuclear power units for which fulfillment of some of the General Design Criteria may not be necessary or appropriate. For plants such as these, departures from the General Design Criteria must be identified and justified.

10 CFR 50, Appendix A, GDC 52 states:

Criterion 52 - Capability for containment leakage rate testing. The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

10 CFR 50.54(o) states in part:

Primary reactor containments for water cooled power reactors shall be subject to the requirements set forth in appendix J to this part.

Appendix J to 10 CFR 50, Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors, states, in part:

One of the conditions of all operating licenses under this part and combined licenses under part 52 of this chapter for water-cooled power reactors as specified in 50.54(o) is that primary reactor containments shall meet the containment leakage test requirements set forth in this appendix. These test requirements provide for preoperational and periodic verification by tests of the leak-tight integrity of the primary reactor containment, and systems and components which penetrate *containment of water-cooled power reactors, and establish the acceptance criteria for these tests. The purposes of the tests are to assure that (a) leakage through the primary reactor containment and systems and components penetrating primary containment shall not exceed allowable leakage rate values as specified in the technical specifications or associated bases; and (b) periodic surveillance of reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment, and systems and components penetrating primary containment. These test requirements may also be used for guidance in establishing appropriate containment leakage test requirements in technical specifications or associated bases for other types of nuclear power reactors.*

1.6 Abbreviations and Definitions

Table 1-2 Abbreviations

Term	Definition
ANS	American Nuclear Society
ANSI	American National Standard Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BPVC	Boiler and Pressure Vessel Code
CES	containment evacuation system
CFR	Code of Federal Regulations
CITF	containment isolation test fixture
CIV	containment isolation valve
CLIP	Containment Leakage Integrity Program
CLRT	containment leakage rate testing
CNTS	containment system
CNV	containment vessel
CRDM	control rod drive mechanism
CRDS	control rod drive system
CVCS	chemical and volume control system
DHRS	decay heat removal system
ECCS	emergency core cooling system
EFPY	effective full-power year
EPA	electrical penetration assembly
EPRI	Electric Power Research Institute
FSAR	Final Safety Analysis Report
FWIV	feedwater isolation valve
FWS	feedwater system
GDC	general design criteria
HRC	Rockwell C hardness number
HV	Vickers hardness number
I&C	instrumentation and controls
ICI	incore instrumentation
IGSCC	intergranular stress corrosion-cracking
ILRT	integrated leak rate testing

Table 1-2 Abbreviations (Continued)

Term	Definition
ISA	instrument seal assembly
ISI	inservice inspection
IST	inservice testing
ITAAC	Inspections, Tests, Analyses, and Acceptance Criteria
LLRT	local leak rate test
LLWR	large light water reactor
MeV	million electron volt
MSIBV	main steam isolation bypass valve
MSIV	main steam isolation valve
MSS	main steam system
NEI	Nuclear Energy Institute
NPM	NuScale Power Module
NPS	nominal pipe size (ASME B36.10M)
NRC	Nuclear Regulatory Commission
PSCIV	primary system containment isolation valve
PWHT	post weld heat treatment
PWR	pressurized water reactor
PWSCC	primary water stress corrosion-cracking
PZR	pressurizer
RAI	request for additional information
RCCWS	reactor component cooling water system
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RPV	reactor pressure vessel
RVV	reactor vent valve
SCC	stress corrosion-cracking
scfh	standard cubic foot per hour
scfm	standard cubic foot per minute
SDAA	Standard Design Approval Application
SG	steam generator
SGS	steam generator system
SSC	structures, systems, and components
SSCIV	secondary system containment isolation valve
TS	technical specification
UHS	ultimate heat sink

Table 1-3 Definitions

Term	Definition
GDC 55 penetration	*Reactor coolant pressure boundary penetrating containment.* This type of penetration requires two NRC Quality Group A, ASME BPVC Class1 CIVs at each penetration.
GDC 56 penetration	*Containment boundary.* This type of penetration requires two NRC Quality Group A, CIVs at each penetration.

Table 1-3 Definitions (Continued)

Term	Definition
GDC 57 penetration	*Closed system lines* that penetrate reactor containment and are neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere. This type of penetration requires one NRC Quality Group B, CIV at each penetration, or a second closed loop.
L_a	Maximum allowable containment leakage rate at pressure P_a
P_a	Peak CNV accident pressure

1.7 Containment Leakage Integrity Assurance Overview

The CLIP testing, inspection, and examination, as supported by the design and analysis of the CNV and CNTS, ensure leakage integrity is maintained for the containment. The CLRT, in combination with other CLIP elements, verifies the leakage integrity of the reactor containment by testing that the actual containment leakage rates do not exceed the values assumed in the applicable safety analysis calculations for design basis events. The preoperational and periodic CLRT requirements and acceptance criteria that demonstrate leakage integrity of the CNTS and associated components are performed in accordance with 10 CFR 50, Appendix J and implemented through the licensee's CLRT program described in Section 5.5.9 of the TS. The maximum allowable containment leakage rate is referred to as L_a, which corresponds to the peak accident pressure inside containment (P_a) (these terms are defined in 10 CFR 50, Appendix J). The containment penetrations and containment isolation barriers are designed to permit the periodic leakage testing described in GDC 53 and GDC 54 to verify leakage through the containment penetrations does not exceed the allowable leakage rate.

The design of the containment penetrations support performance of Type B and Type C testing in accordance with the guidance provided in American National Standards Institute/American Nuclear Society (ANSI/ANS) 56.8 (Reference 9.4) and Nuclear Energy Institute (NEI) 94-01 Rev. 3-A (Reference 9.6). The CNTS design accommodates the test method frequencies permitted by 10 CFR 50, Appendix J; Option B, Performance-Based Requirements. Applicants that reference the NuScale power plant US460 standard design approval will develop a "Containment Leakage Rate Testing Program" that will identify Option B to be implemented under 10 CFR 50, Appendix J. It is expected that the licensee will ultimately adopt Option B extended interval testing pursuant to the guidance in NEI 94-01 Rev. 3-A. Considerations for Option B extended interval testing are outlined in Section 7.5.

The CNTS is designed for flanged joints to remain sealed at design pressure. The containment is initially inspected and tested at the factory, including American Society of Mechanical Engineers (ASME) BPVC Section III hydrostatic testing with an acceptance criterion of zero visible leakage, to verify no unknown leak pathways exist. Additionally, a CNV preservice design pressure leakage test is performed that loads CNV bolted flange connections to containment design pressure and confirms no observed leakage under these conditions. Because potential leakage pathways are known and testable, preservice and periodic Type B and C testing quantify the overall containment leakage rate to verify maximum allowable leakage is not exceeded. Thus, the design and configuration of potential leak pathways, including CNV flanges and CIVs, accommodate LLRT requirements and acceptance criteria. Periodic inspection and testing verifies no unknown leakage pathways develop over time. Thus, any potential through-wall degradation is be precluded as a credible mechanism for containment leakage. Post-maintenance inspection and testing, including Type B and C testing and administrative controls, verify that no unknown leakage pathways develop due to activity-based failure mechanisms during maintenance or modifications.

2.0 NuScale Containment Vessel Structure

The CNV design ensures leakage integrity through design, inspection and testing, other than ILRT, as required by GDC 52 and 10 CFR 50, Appendix J. NEI 94-01 Rev. 3-A describes the purpose of 10 CFR 50, Appendix J, for traditional large containment structures:

> *The purpose of Type A testing is to verify the leakage integrity of the containment structure. The primary performance objective of the Type A test is not to quantify an overall containment system leakage rate. The Type A testing methodology as described in ANSI/ANS-56.8-2002 serves to ensure continued leakage integrity of the containment structure. Type B and Type C testing ensures individual leakage rates support the leakage tightness of primary containment by minimizing potential leakage paths.*

Continued leakage integrity of the CNV structure is ensured by precluding through-wall degradation as a credible leakage mechanism. The CNV is a welded-metal vessel design, in contrast to existing pressurized water reactors (PWRs) that incorporate large containment building structures. The containment is designed for flanged joints to remain sealed at design pressure. Manufacturing acceptance tests and inspections are similar to RPV tests and inspections and are performed in a factory environment. Comprehensive ISI applying ASME Boiler and Pressure Vessel Code (BPVC) Class 1 criteria also ensures no new leakage paths develop over the life of the plant due to degradation. Surface areas and welds are accessible for inspection. Additionally, a separate preservice design pressure leakage test is required for CNVs with CNV bolted flange connections in place to demonstrate no observed leakage using as-designed flange covers installed with the design bolting materials, design bolting assembly preloads, and design seals. This leakage test is required by an Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC). The first CNV of the initial plant shall be tested with the upper and lower halves of the CNV assembled. Penetration pathways are tested to Type B or Type C criteria at peak containment accident pressure. These features ensure continued leakage integrity of the CNTS is maintained without the need for Type A testing.

The NuScale CNV design is different from traditional containments in several fundamental aspects. These design differences provide reasons that the design does not need to conform with GDC 52 and 10 CFR 50, Appendix J, and necessitate alternative means of ensuring the leakage integrity of the NuScale containment. The major containment functional differences are:

- The CNV is a high-pressure vessel with no internal subcompartments, classified as an ASME BPVC Class MC component, and constructed to ASME Code Class 1 vessel rules, constructed of stainless materials.
- Penetrations are ASME Code Class 1 flanged joints capable of Type B testing, ASME Code Class 1 welded nozzles with isolation valves capable of Type C testing, or part of a closed system inside containment. Flanged joints are designed to remain in contact at accident temperature, concurrent with peak accident pressure.

- During refueling, the NPM, including the CNTS, is physically moved by a crane to the refueling area. The upper and lower CNV shells are separated during outages for refueling, maintenance, and inspection. The CNV is designed to accommodate comprehensive inspections of welds, including volumetric and surface inspections. Welds are accessible, and there are no areas that cannot be inspected. The CNV design allows for visual inspection of the entire inner and outer surfaces. Through-wall degradation can be identified before development of potential leak paths precluding this as a credible leakage mechanism.
- During reassembly, positive verification is used to verify proper stud elongation to ensure proper loading on each flange seal.
- During normal operation, the CNV is partially submerged in borated water with its internal environment under near-vacuum conditions. Automatic engineered safety feature actuation systems initiate on high containment pressure. Containment vacuum pressure and leakage into the CNV is constantly monitored during normal operation. In comparison to traditional PWR designs, the small containment volume and evacuated operating conditions allows wide-ranging detection capabilities for liquid or vapor in-leakage, providing an additional layer of leakage integrity assurance.

The NuScale CNV design is described in detail in Section 3.0.

2.1 NuScale Containment Vessel Penetrations

The CNTS design supports leakage integrity assurance through inspection and testing other than ILRT. When compared to traditional LLWR containments, the NuScale CNTS design is simplified. The CNV has a low number of penetrations, which are either ASME BPVC Class 1 flanged joints capable of Type B testing, ASME BPVC Class 1 welded nozzles with isolation valves capable of Type C testing, or part of a closed system inside containment (i.e., steam generator system piping). The CNV has no penetrations equipped with resilient seals. No instrument tubing penetrates containment; therefore, there are no small diameter fluid lines without isolation capability that are not subject to Type B or Type C LLRT. There are no air locks, flexible sleeves, or nonmetallic boundaries. The simplicity of design provides for alternate means of assuring containment leakage integrity. Leakage integrity assurance is primarily achieved by ensuring no unknown leak paths via ISI and accurately measuring the leakage rate of potential leak pathways via LLRT. Key features that ensure CNTS leakage integrity is maintained are:

- Containment vessel flanges are designed to remain in contact at accident temperature, concurrent with peak accident pressure.
- As described in Section 2.1, the CNV is an ASME Code Class 1 pressure vessel with a relatively low volume and no internal subcompartments. The NPM's comparatively simple design (compared to existing LLWR designs) allows identification of potential leakage pathways.
- The CNV pressure vessel preservice test and inspections are equivalent to RPV requirements, including hydrostatic testing requirements. These tests and inspections verify no unknown leakage pathways exist.

- Preservice design pressure leakage test of the CNV with CNV bolted flanges in place using as-designed flange covers installed with the design bolting materials, design bolting assembly preloads, and design seals to demonstrate no observed leakage at design pressure.
- The upper and lower halves of the CNV are assembled for the first NPM of the initial NuScale plant.
- After successful testing, the upper and lower halves of other CNVs may be tested separately.
- Covers with electrical and instrumentation penetrations can be substituted with blank covers having the same sealing design.
- The limited number of CNV penetrations have similar seal designs that are tested by Type B or Type C LLRT. The limited number of penetrations, and other aspects of the penetration design, allows accurate quantification of the overall leakage rate by LLRT.
- The ISI Program and planned CNV examinations meet ASME Code Class 1 criteria. This program ensures no new unidentified leakage pathways develop over time.
- Disassembly and reassembly procedures and controls of the CNV are similar to the RPV. Positive verification is used to verify proper loading on each flange seal. This verification ensures these potential activity-based failure mechanisms do not degrade CNTS leakage integrity.

The CNV is an ASME BPVC Section III, Subsection NE, Class MC containment designed, fabricated, and stamped as a Subsection NB, Class 1 pressure vessel, with overpressure protection provided in accordance with NE-7000.

The CNV is made of corrosion-resistant materials, has a low number of penetrations, and features no penetrations with resilient seals. The use of welded nozzles and testable flange seals at the containment penetrations ensure Type B and Type C testing provide an accurate assessment of overall containment leakage rate.

The unique CNV and CNTS design allows testing and inspection options not suitable to current LLWR containment designs. Based on the containment vessel ASME pressure vessel design and its function, more alternate methods of testing and inspection are available. Each element of the CLIP is consistent with a corresponding element of an approved program for RPVs or LLWR containments.

The CNTS design is described in detail in Section 3.0. Table 2-1 compares elements of the CLIP with testing performed on the containment, reactor coolant pressure boundary (RCPB), and traditional containments. The table demonstrates the testing is commensurate with the design and safety function of the containment.

Table 2-1 NuScale Containment Leak Rate Test Comparison

<table>
<tr><th>CLIP Program Element to Ensure Essentially Leak-Tight Barrier</th><th>NuScale Containment</th><th>Reactor Coolant Pressure Boundary Testing for NuScale and Other Licensed Facilities</th><th>Traditional Containment</th></tr>
<tr><td>Initial verification of structural integrity</td><td>Hydrostatic testing per ASME BPVC Section III</td><td>Hydrostatic testing per ASME BPVC Section III</td><td>Preservice ILRT and structural integrity test</td></tr>
<tr><td rowspan="2">Initial verification of leakage integrity</td><td><u>Factory</u> - hydrostatic testing per ASME BPVC Section III Containment preservice leakage test (ITAAC) (no visible leakage allowed)</td><td rowspan="2">Hydrostatic testing per ASME BPVC Section III</td><td rowspan="2">Preservice ILRT (leakage allowed below prescribed limit)</td></tr>
<tr><td><u>On-site</u> - preservice LLRT</td></tr>
<tr><td>Prevention of leakage from activity-based failure mechanisms (degradation due to system and/or component modifications or maintenance)</td><td>Administrative controls such as configuration management and procedural requirements for system restoration that ensure integrity is not degraded by plant modifications or maintenance activities</td><td>Administrative controls such as configuration management and procedural requirements for system restoration that ensure integrity is not degraded by plant modifications or maintenance activities</td><td>Administrative controls such as configuration management and procedural requirements for system restoration that ensure integrity is not degraded by plant modifications or maintenance activities</td></tr>
<tr><td>Detection of leakage from activity-based failure mechanisms</td><td>LLRT</td><td>Reactor coolant system (RCS) leak test - operational pressure</td><td>LLRT</td></tr>
<tr><td>Prevention of leakage from age-based failure mechanisms (age-related degradation)</td><td rowspan="2">Design and construction requirements for CNV, inspections/ examinations performed in accordance with ASME BPVC Section XI, the maintenance rule and regulatory commitments
NuScale CNV design allows for comprehensive ISI surface and weld examination</td><td rowspan="2">Design and construction requirements for RCS, inspections/examinations performed in accordance with ASME BPVC Section XI, the maintenance rule and regulatory commitments
RCS leakage detection</td><td>Design and construction requirements, inspections/ examinations performed in accordance with ASME BPVC Section XI, the maintenance rule and regulatory commitments</td></tr>
<tr><td>Detection of leakage from age-based failure mechanisms (age-related degradation)</td><td>ILRT</td></tr>
<tr><td>Post-repair/ modification verification of leakage integrity</td><td>Hydrostatic testing per ASME BPVC Section XI and LLRT</td><td>Hydrostatic testing per ASME BPVC Section XI</td><td>ILRT/LLRT</td></tr>
<tr><td>Post-repair/ modification verification of structural integrity</td><td>Hydrostatic testing per ASME BPVC Section XI</td><td>Hydrostatic testing per ASME BPVC Section XI</td><td>ILRT</td></tr>
</table>

2.2 Type B Testing

Type B pneumatic tests detect and measure leakage across the pressure-retaining, leakage-limiting boundaries in the CNV. Preoperational and periodic Type B leakage rate testing is performed in accordance with 10 CFR 50, Appendix J, NEI 94-01 Rev. 3-A, and ANSI/ANS 56.8 within the test intervals defined by the licensee. The containment penetrations subject to Type B tests are identified in Appendix B. As described further in Section 3.2, the design of CNV penetrations allows accurate LLRT results to be obtained to quantify overall containment penetration leak rates.

The design of CNV Type B penetrations is described in Section 3.2.

2.3 Type C Testing

The CIVs are designed to support Type C pneumatic tests. Preoperational and periodic Type C leakage rate testing of CIVs is performed in accordance with 10 CFR 50, Appendix J, NEI 94-01 Rev. 3-A, ANSI/ANS 56.8, and the licensee TS. The CIVs subject to Type C tests are identified in Appendix C. As described further in Section 3.3, the design of CIVs allows accurate LLRT results to be obtained to quantify overall CIV leak rates.

The design of CIVs is described in Section 3.3.

2.4 Containment Overall Leakage Limits

Per 10 CFR 50, Appendix J, L_a is defined as the maximum allowable containment leakage rate in weight percent per day at peak containment accident pressure P_a. For the design, L_a is selected to be 0.20 weight percent of the containment air mass per day (over 24 hours) at the peak containment accident pressure (P_a) provided in FSAR Section 6.2.1. L_a is established as a safety analysis operational limit for the design. The values are used in calculations to confirm accident radiological containment leakage to the environment is within acceptable limits.

An evaluation of containment penetrations and access flanges (i.e., leakage pathways) was performed to determine whether the design can reliably meet the 10 CFR 50, Appendix J, leakage criteria using the maximum allowable leak rate of L_a of 0.2 weight percent of the containment air mass per day at design pressure. The evaluation concluded the combined maximum expected leakage from local penetrations, with conservative margin for degradation, is less than 0.60 L_a, at the peak accident pressure, P_a, which is the acceptance criterion for LLRT per 10 CFR 50, Appendix J.

Table 2-2 documents containment design basis leakage rate criteria. The CLRT leakage rate limits for LLRT are developed from these design basis limits to meet 10 CFR 50, Appendix J, leakage criteria.

Testing to meet less than $0.6L_a$ at P_a ensures the operational limit of 0.20 weight percent of containment air mass per day can be met. Design Specific Review Standard for NuScale Small Modular Reactor Section 6.2.1.1.A, Acceptance Criteria 4 states that to satisfy GDC 38 to rapidly reduce the containment pressure, the pressure should be reduced to less than 50 percent of the peak calculated pressure for the design basis loss-of-coolant accident within 24 hours after the postulated accident. Following the peak containment pressure design basis accident, containment pressure drops from P_a to approximately 100 psia in less than five hours.

Table 2-2 Maximum Allowable Containment Leakage Rate Limits

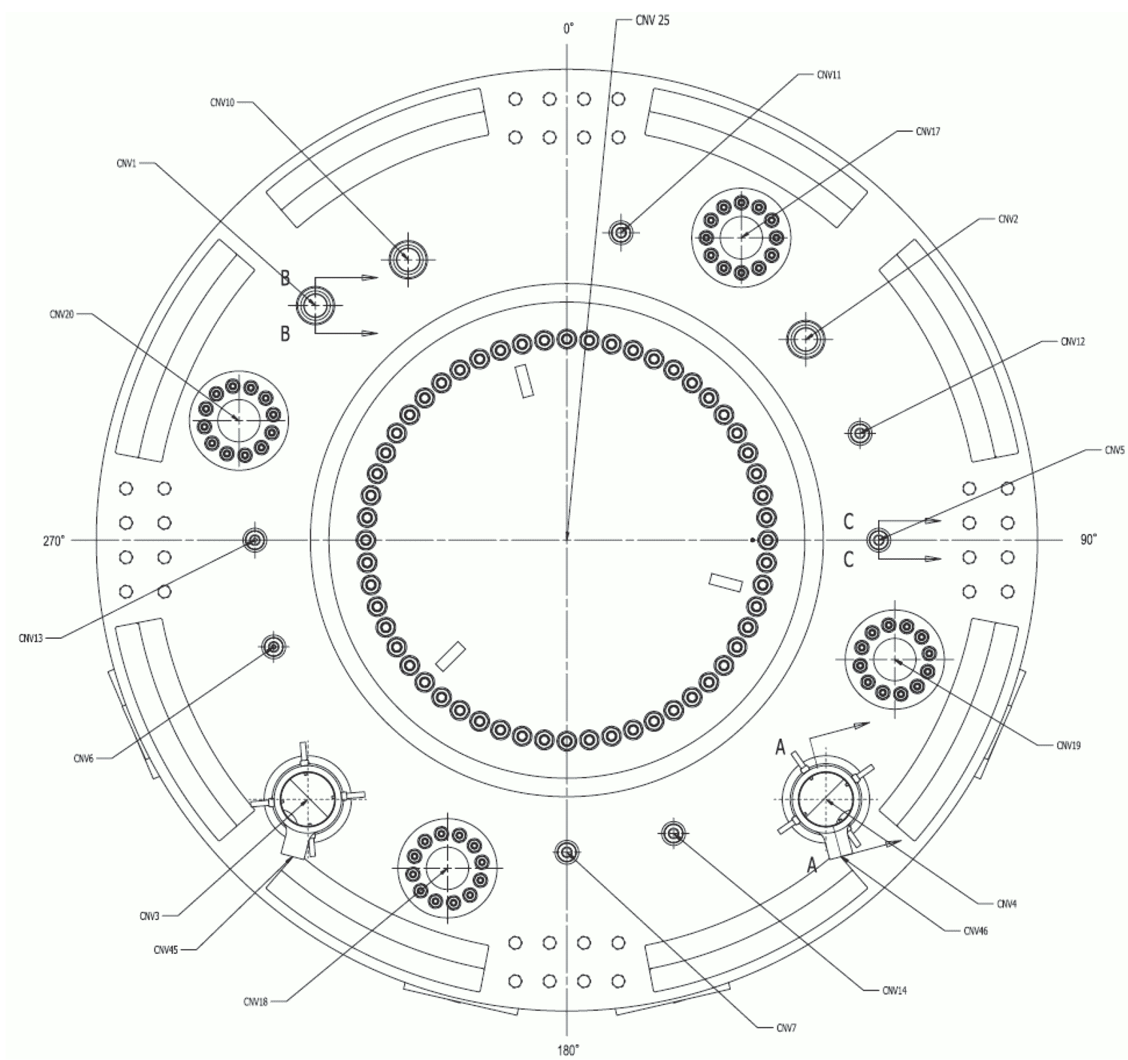
	Leak Rate	**Pressure**
Containment leakage rate limit	0.2 weight percent of containment air mass per day (L_a)	P_a
Containment leakage rate evaluation parameters	0.2 weight percent of containment air mass per day	1,200 psia (CNV internal design pressure)

The LLRT limits are developed based on the values of Table 2-2, and are based on a L_a at P_a and to meet less than $0.6L_a$. The peak containment accident pressure (P_a) is identified in FSAR Section 6.2.1.The CLRT is described further in Section 7.0.

3.0 NuScale Containment System Design

The CNTS is designed around an ASME Code Class MC pressure vessel. The simplicity of the NPM design minimizes the number of containment penetrations required. There are a limited number of access ports (quantity nine), electrical penetration assemblies (EPAs, quantity 12), and instrument seal assemblies (ISAs, quantity four flanges with four modules per flange) that use the same flange seal design. The CNV closure flange, which separates the upper and lower CNV assemblies, uses the same seal design as the access ports. There are a limited number of fluid lines (quantity 14) penetrating containment (Figure 3-1). Of these, 12 are protected by CIVs and two are protected by a closed loop.

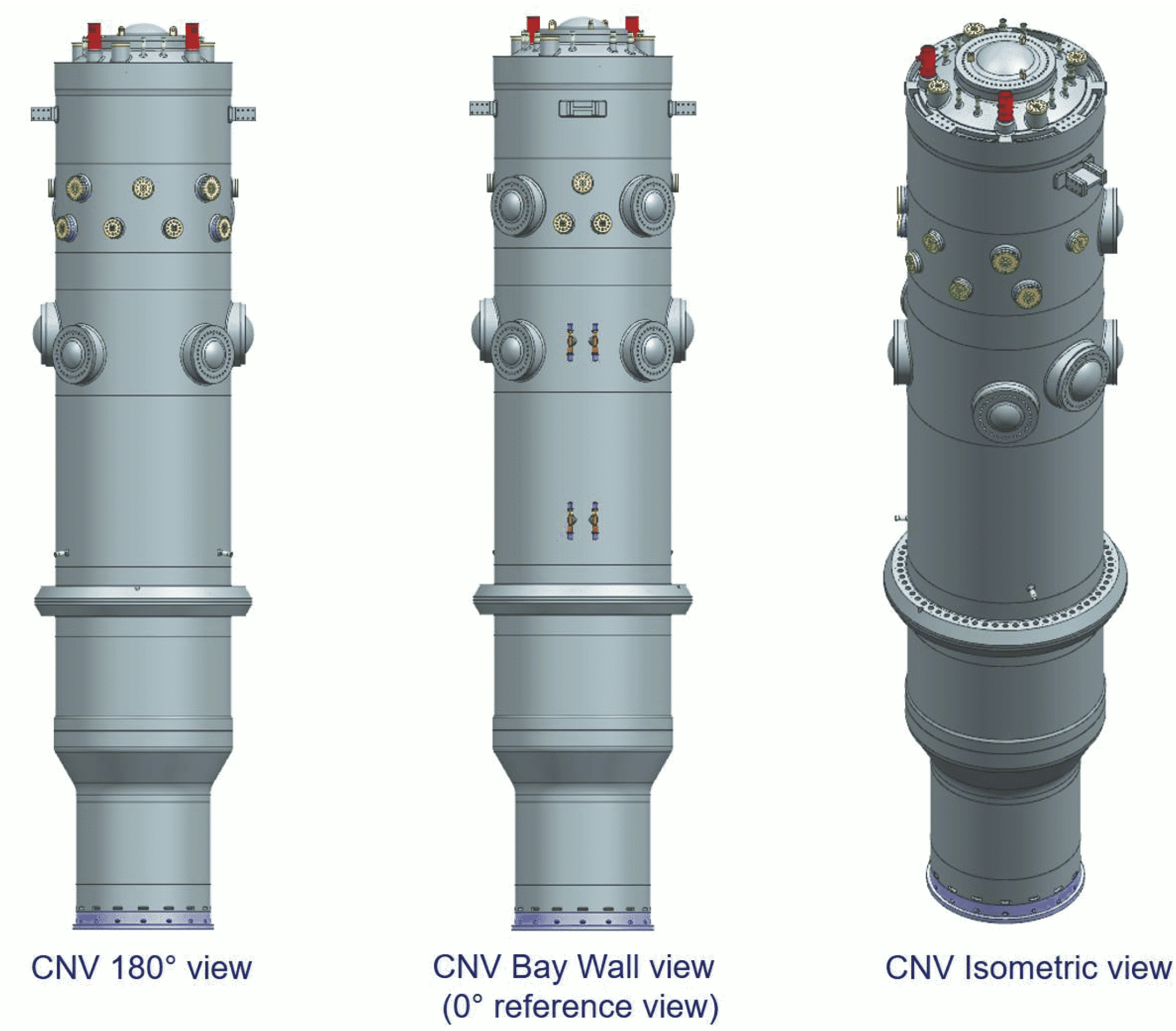
Figure 3-1 Containment Vessel Head



3.1 Containment Vessel Design

The CNV is partially immersed in the ultimate heat sink (UHS) that removes residual core heat during normal and accident conditions. The design of the CNV features a relatively low and simple volume compared to other PWR containments: an approximately 6000 ft^3 free volume steel vessel with no internal sub-compartments. The design prevents isolated pockets of concentrated gases. The upper and lower portions of the CNV are constructed of stainless steel. The CNV are shop-fabricated, which facilitates enhanced fabrication quality and testing control.

Figure 3-2 Containment Vessel



3.1.1 Preservice Tests and Inspections (Containment Vessel)

Nondestructive examination of the CNV after fabrication includes the following preservice examinations that are performed after hydrostatic testing but before code stamping:

- general visual examinations for pressure retaining surfaces above the reactor pool level in accordance with Paragraph IWE-2200
- VT-3 visual examinations for pressure retaining surfaces below the reactor pool level in accordance with Paragraph IWE-2200
- VT-1 visual examinations for pressure retaining bolting in accordance with Paragraph IWE-2200
- volumetric examinations for select welds in support of the break exclusion zone requirement in accordance with augmented requirements
- volumetric examinations for the: (a) CNV upper head to CNV upper seismic support shell, (b) CNV lower shell to CNV lower transition shell, and (c) CNV lower core shell to CNV lower head circumferential vessel welds in accordance with augmented requirements (note this inspection supports the Type A test exemption discussed in Section 1.3)

The CNV is hydrostatically tested after construction in accordance with ASME BPVC Section III, Paragraph NB-6000. The water-filled CNV is pressurized to a minimum of 25 percent over design pressure, and the pressure is held for at least ten minutes. Pressure is then reduced to design pressure and then held for leakage examination. The acceptance criterion for the test is that there are no leakage indications at the examination pressure (i.e., design pressure).

For the preservice design pressure leakage test, the CNV is tested with water at a minimum pressure of 1,200 psia to a maximum of 1,275 psia with the pressure held for 30 minutes prior to examining for leaks. The CNV bolted flange covers are attached to the vessel in their design condition during the preservice design pressure test. Covers with electrical and instrumentation penetration may be substituted with blank covers having the same sealing design as the design covers. The design seals are installed and the flanges are bolted using design bolting materials installed to normal operational preload values. Verification of bolting preload values is performed before vessel water-filled pressurization to design pressure. The testing is performed at a temperature between 70 degrees F and 140 degrees F which minimizes the possibility of brittle fracture. Once the vessel is at design pressure, the vessel bolted connections are visually examined for water leakage. The test is considered satisfactory if there is no indication of water leakage at the flange connections.

3.2 Preservice Tests and Inspections (Type B and Type C Components)

With the exception of the emergency core cooling system (ECCS) pilot valve body-to-bonnet seals (Type B penetrations), penetrations are either ASME Code Class 1 bolted flanged connections capable of Type B testing or ASME Code Class 1 welded nozzles with isolation valves capable of Type C testing.

The CNV upper shell includes 12 EPAs and the CNV head includes four ISA flanges, as well as nine ports at various locations (Appendix B). The ISA flanges include four modules that are individually tested. The seal design for Type B penetrations, including the CNV closure flange is similar, with the exception being the size and model of the seals. These penetrations are tested periodically by Type B LLRTs.

Type C testing is required for eight CNV penetrations, seven of which are 2-inch nominal pipe size (NPS) pipe penetrations and one of which is 4-inch NPS (Appendix C). These CIVs are of identical design and construction. These penetrations are tested periodically by Type C LLRTs. The other six penetrations are main steam, feedwater, and decay heat removal system (DHRS) condensate penetrations that are connected to the steam generator (SG), which are closed loops inside containment. These penetrations are not required to be Type C tested in accordance with ANSI/ANS 56.8-2002, Section 3.3.1 and NEI 94-01 Rev. 3-A.

No instrument tubing penetrates containment; therefore, there are no small diameter fluid lines without isolation capability that are not subject to Type B or Type C LLRT. There are no air locks, flexible sleeves, or non-metallic boundaries. There are no penetrations in the NuScale design that would only be tested during an ILRT. Because of personnel safety and operational constraints, entry into the CNV does not occur during normal operation.

Type B pathway integrity is not expected to be disturbed except when the NPM is in a refueling outage or disassembled for emergent maintenance activities. Type C pathways are designed such that an individual valve can be tested in the same direction as the pressure applied when the valve would be required to perform its safety function. Type B and Type C pathways are tested to P_a.

NEI 94-01 Rev. 3-A provides guidance for implementing the performance-based option of 10 CFR 50, Appendix J (commonly referred to as Option B), for traditional large containment structures:

> *The purpose of Type A testing is to verify the leakage integrity of the containment structure. The primary performance objective of the Type A test is not to quantify an overall containment system leakage rate. The Type A testing methodology as described in ANSI/ANS-56.8-2002, and the modified testing frequencies recommended by this guideline, serves to ensure continued leakage integrity of the containment structure. Type B and Type C testing assures that individual leakage rates support the leakage tightness of primary containment by minimizing potential leakage paths. A review of performance history has concluded that almost all containment leakage is identified by local leakage rate testing.*

Existing PWRs incorporate large containment building structures, however, leakage integrity is ensured in the NPM design by the welded-metal vessel design of the CNTS. The CNTS is a small, high-pressure, ASME BPVC Class MC vessel with a significantly reduced number of penetrations and no internal sub-compartments. Preservice tests and inspections are similar to those performed on the RPV. Comprehensive ISI against ASME BPVC acceptance criteria also ensures continued system integrity. The surface areas and welds are accessible for inspection. Penetration pathways are tested to Type B or

Type C criteria at accident or design pressures. This ensures leakage integrity of the CNTS is maintained without the need for Type A testing.

3.2.1 Type B Penetrations

Type B components in the scope of the CLRT program are listed in Attachment A.1.

The CNV is designed to support Type B local leak rate tests to detect and measure leakage across pressure-retaining, leakage-limiting boundaries. The following containment penetrations are subject to preoperational and periodic Type B leakage rate tests.

- flanged openings with bolted connections (i.e., access ports), nine total per NPM
- electrical penetration assemblies for various instrumentation and power cables, 12 total per NPM
- instrument seal assemblies for incore instrumentation, four total per NPM (each ISA includes four modules, resulting in a total of 20 Type B components)
- ECCS trip/reset valve body-to-bonnet seals, four total per NPM
- containment vessel closure flange, one total per NPM

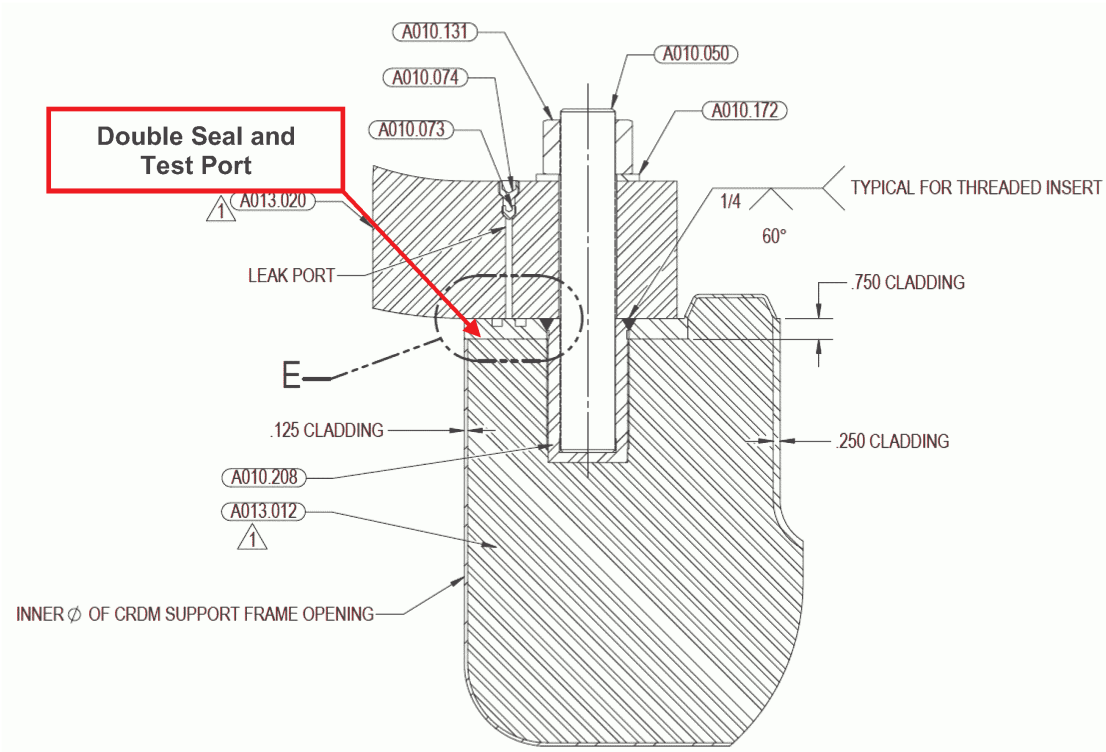
Type B penetrations are bolted closures that have dual metal seals with testing ports between the seals. Type B penetration assemblies are designed and constructed to ASME Code Class 1 requirements. The CNV closure flange has a similar double seal and test port arrangement.

In addition to the penetrations listed above, most CIVs incorporate a test fixture with a removable cover that is subject to Type B testing. Section 3.3 provides details.

3.2.2 Access Ports

The nine access ports are bolted closures that have dual metal seals with testing ports between the seals. Figure 3-3 shows the CNV head and control rod drive mechanism (CRDM) access flange. This double seal and test port design is used for every flange seal subject to Type B testing.

Figure 3-3 Containment Vessel Head / Control Rod Drive Mechanism Access Flange

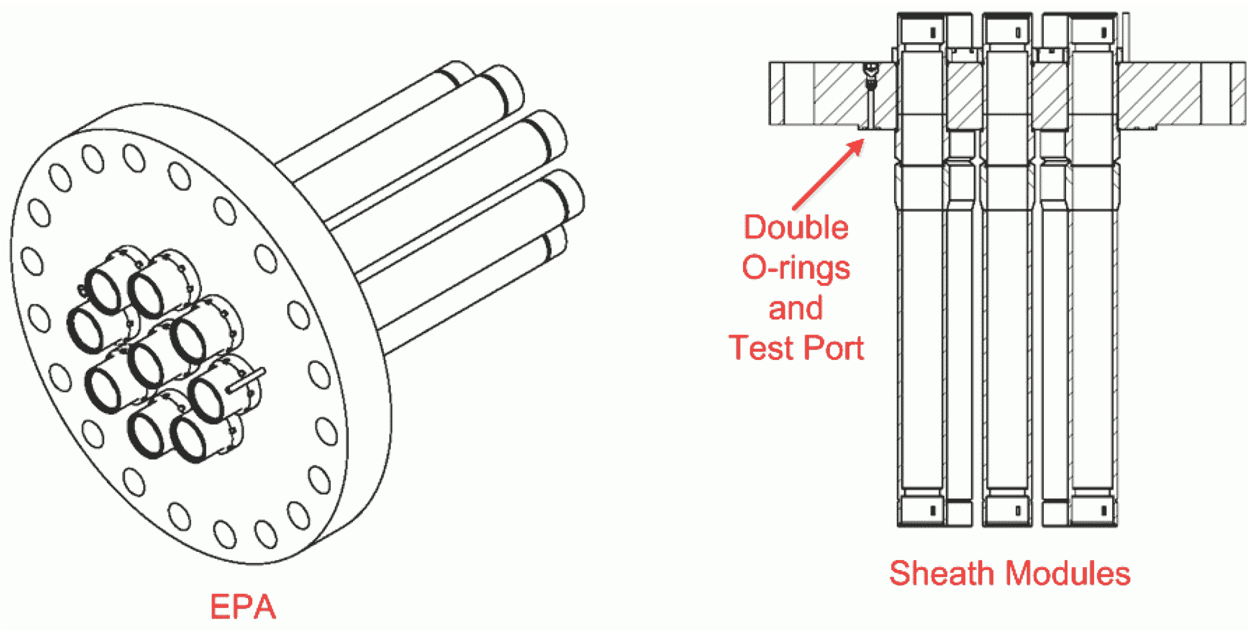


3.2.3 Electrical Penetration Assemblies

A total of 12 EPA penetrations are located at various locations on the CNV upper shell (Appendix B). The EPAs use sheath modules that use a glass-to-metal sealing technology that is not vulnerable to thermal or radiation aging, does not require periodic maintenance, and can achieve a less than minimum detectable leak rate. The performance of the glass-to-metal EPA seal has been proven in currently operating nuclear plants.

The EPA, with installed modules, is bolted to CNV flange penetrations similarly to the flanged access ports. Figure 3-4 depicts the pressurizer (PZR) heater power supply EPA. This configuration is typical for EPAs in the design. The design includes the ability to test the double seals by pressurizing between the seals of the EPA similar to the flanged access ports. Sheath modules are only disassembled from an EPA for modification or if leakage is indicated. If disassembly is performed, then retest of the module or EPA seal is required before installing the EPA in the CNV.

Figure 3-4 Electrical Penetration Assembly Modules (Typical)



3.2.4 Instrument Seal Assemblies

There are four ISA flanges located on the CNV head located at the incore instrumentation (ICI) penetrations. Each ISA flange contains four modules to accommodate ICI stringer assemblies and PZR level sensors. The ISA flanges and modules are leak tested separately.

The ISA contain ICI containment vessel compression seal fittings on its outer diameter for the 12 ICI stringer assemblies installed within the NPM, permitting the ICI stringer assemblies to penetrate the top head of the CNV while maintaining the containment pressure boundary during normal operations and design-basis events.

The ICI containment vessel compression seal fitting features a test port to allow local pneumatic pressurization of the region between the primary seal and test seals for Type B testing. The test port and test seals perform no function during normal plant operation.

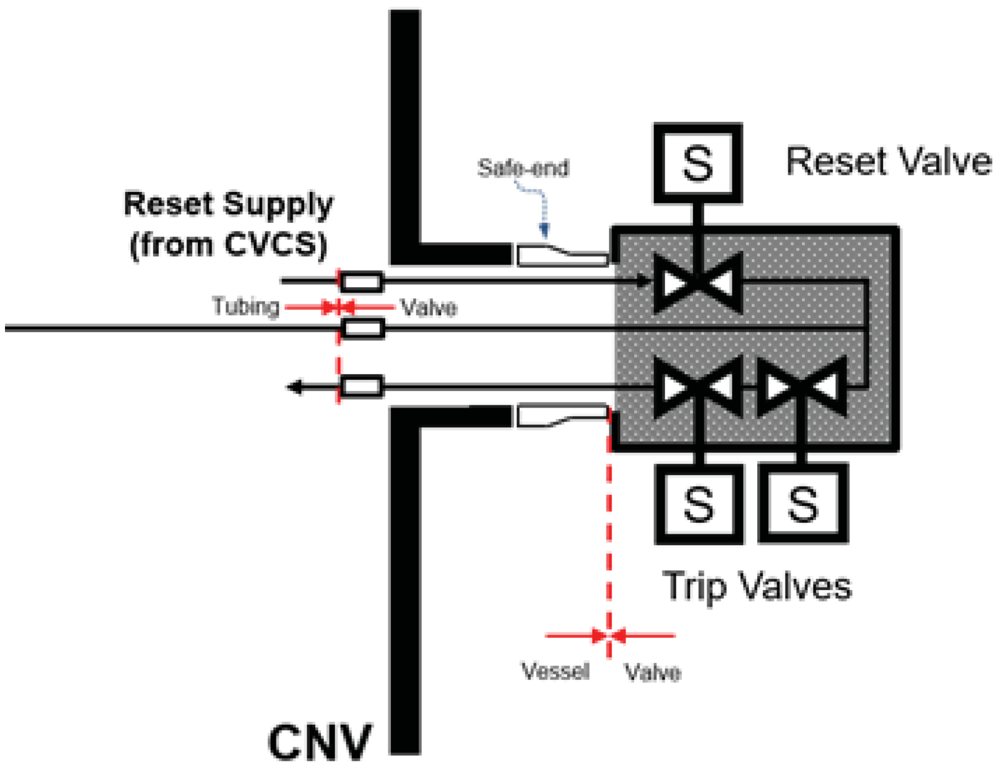
3.2.5 Emergency Core Cooling System Trip and Reset Body-to-Bonnet Seals

Four penetrations on the upper CNV shell accommodate ECCS valve trip/reset pilot assemblies. The trip/reset pilot valve body is located outside the CNV and is an RCPB. The ECCS valve trip/reset pilot assembly safe-end penetrations are welded to the external side of the penetration nozzle. Figure 3-5 shows the boundaries.

The safe-ends and penetration nozzle-to-safe-end welds are part of the CNV. The valve assembly is welded to the penetration nozzle safe-end. The boundary is in the valve assembly-to-safe-end welds and the welds are part of the CNV. Pilot valve body-to-bonnet interfaces have a double seal with monitoring capability to allow Type B leakage testing. Trip and reset valve pressure boundaries are designed and constructed to ASME Code Class 1 requirements.

The ECCS trip valve and reset valve body-to-bonnet joint seals are not considered to be a flanged connection and are not included in the preservice design pressure leakage test or containment flange bolting calculation.

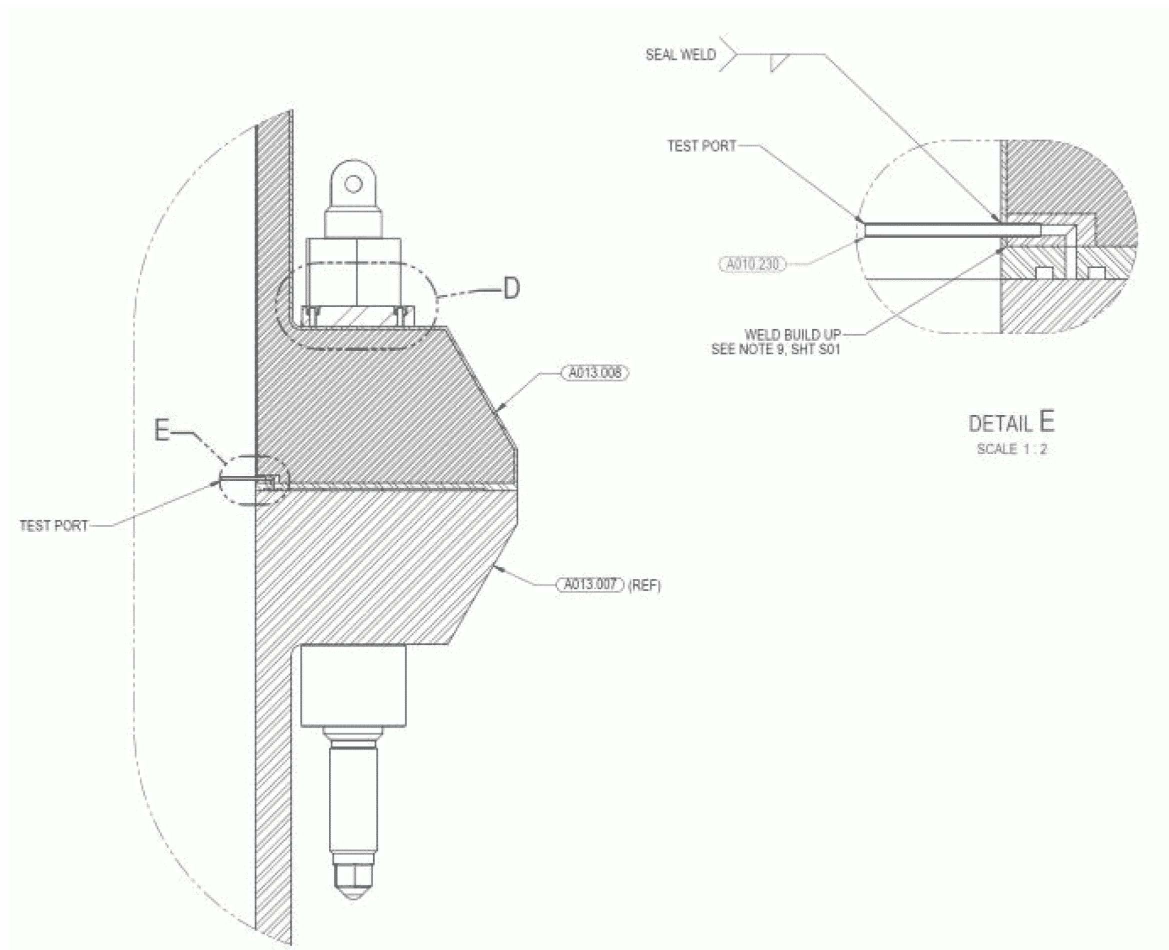
Figure 3-5 Emergency Core Cooling System Valve Trip/Reset Pilot Assembly (Simplified Diagram)



3.2.6 Containment Vessel Closure Flange

The CNV closure flange allows disassembly of the CNV every outage for refueling, maintenance, testing, and inspection of the NPM. Figure 3-6 shows the CNV closure flange has a double seal with a test port. The design is similar to the flanged access ports. The test port attaches to tubing that runs from the CNV closure flange to an accessible point near the CNV manway access ports.

Figure 3-6 Containment Vessel Closure Flange Seal and Test Port



3.3 Type C Penetrations

Type C components in the scope of the CLRT Program are listed in Appendix C. Figure 3-7 shows a general depiction of these components, which are summarized below:

- Primary system piping penetrations:
  - CNV5: Reactor component cooling water system (RCCWS) supply to CRDMs
  - CNV6: Chemical and volume control system (CVCS) injection
  - CNV7: CVCS PZR spray
  - CNV10: Containment evacuation system (CES)
  - CNV11: Containment flooding and drain system
  - CNV12: RCCWS return from CRDMs
  - CNV13: CVCS discharge

  - CNV14: CVCS reactor coolant system high point degasification
- Secondary system piping penetrations:
  - CNV1 and CNV2: Feedwater
  - CNV3 and CNV4: Main steam
  - CNV22 and CNV23: DHRS condensate return

There are eight primary system piping penetrations into the CNV. Four piping penetrations (CNV6, CNV7, CNV13, and CNV14) are part of the RCPB and are subject to the requirements of GDC 55. These penetrations are protected by dual ASME BPVC Class 1 primary system containment isolation valves (PSCIVs) of identical design. The other four piping penetrations (CNV5, CNV10, CNV11, and CNV12) are open to containment atmosphere and are subject to the requirements of GDC 56. The RCCWS is conservatively considered to be open to containment. These ASME BPVC Class 2 penetrations are protected by dual ASME BPVC Class 1 PSCIVs of identical design, with the exception of the CES penetration, which has ASME BPVC Class 2 valves.

There are six secondary system piping penetrations into the CNV, none of which penetrations require Type C testing. These six penetrations are open to a closed loop inside containment (the SGS) and are subject to the requirements of GDC 57. Four of these penetrations (CNV1, CNV2, CNV3, and CNV4) are protected by single ASME Code Class 2 secondary system containment isolation valves (SSCIVs) and nonsafety-related backup valves. The other two penetrations (CNV22 and CNV23) are protected by an ASME Code Class 1 and Class 2 closed loop inside containment and an ASME Code Class 2 closed loop outside containment. The DHRS penetrations do not feature CIVs. The design supports an exemption from GDC 57 to clarify the use of a closed loop system inside and outside containment.

Figure 3-7 Primary System Containment Isolation Valve and Secondary System Containment Isolation Valve Arrangement on Containment Vessel Head

{{

}}2(a),(c)

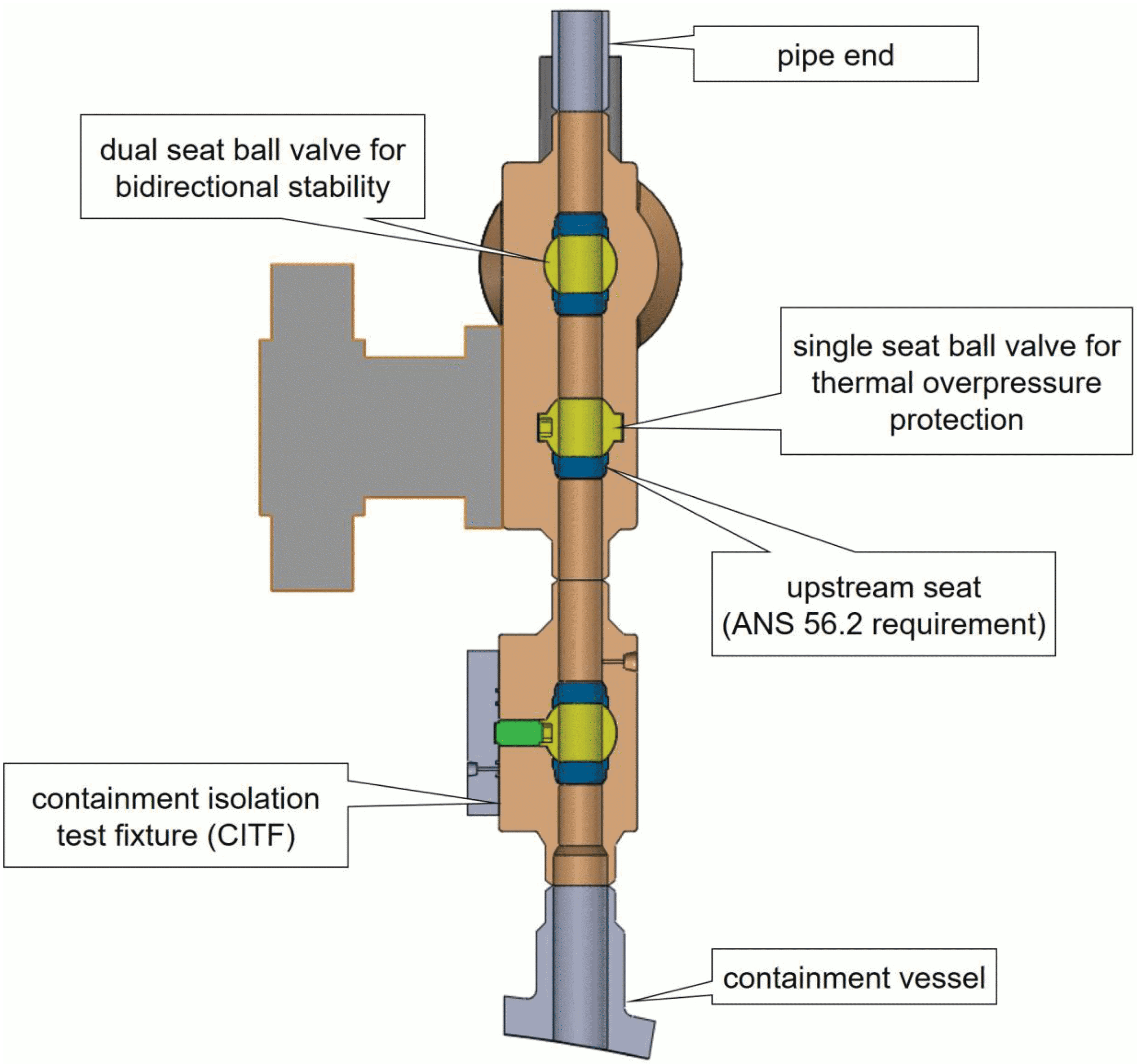
The CIVs on both primary and secondary systems are quarter-turn ball valves with the same actuator design. The size and ball design varies between primary and secondary valves.More differences are found in the main steam isolation valves (MSIVs) and feedwater isolation valves (FWIVs), but the majority of design features are identical.

The CIV pairs feature a containment isolation test fixture (CITF) valve located between the shared valve body and the CNV head (Figure 3-8.), with the exception of the MSIVs and associated bypass valves. The CITF valve is a single, top-entry ball valve that features a test port to accommodate Type C LLRTs of the downstream CIVs. The CITF valve features a cover that is keyed to the ball such that the CITF valve remains locked open during operation and the cover must be removed and rotated to close the CITF valve for testing. This cover features a double o-ring cap-to-body seal with a test port for Type B pneumatic leakage testing of the seal. The leak rate measured on each CITF valve cover following maintenance and testing is included in the combined Type B and Type C leak rate summation.

3.3.1 Primary System Containment Isolation Valves

The PSCIVs are the only piping penetration isolation valves required to meet 10 CFR 50, Appendix J, Type C test criteria. Both GDC 55 and GDC 56 penetrations are protected by PSCIVs of identical design and construction. These PSCIVs are 2-inch NPS (except for two valves on the containment evacuation system, which are 4-inch NPS) and have a dual-actuator, single-body arrangement. Four PSCIVs protect GDC 55 penetrations, and four PSCIVs protect GDC 56 penetrations. The PSCIVs are designed and constructed to ASME Code Class 1 except the two CES PSCIVs, which are ASME Code Class 2 valves.

Figure 3-8 Primary System Containment Isolation Valve Dual Valve, Single Body Design



Each dual valve assembly is welded directly to its respective CITF, which in turn is welded to the CNV head via integral vessel nozzle safe end and butt weld arrangements. The containment flooding and drain system and RCCWS valves are classified as Quality Group B, but are specified to be designed and constructed to ASME BPVC Class 1.

3.3.2 Secondary System Containment Isolation Valves

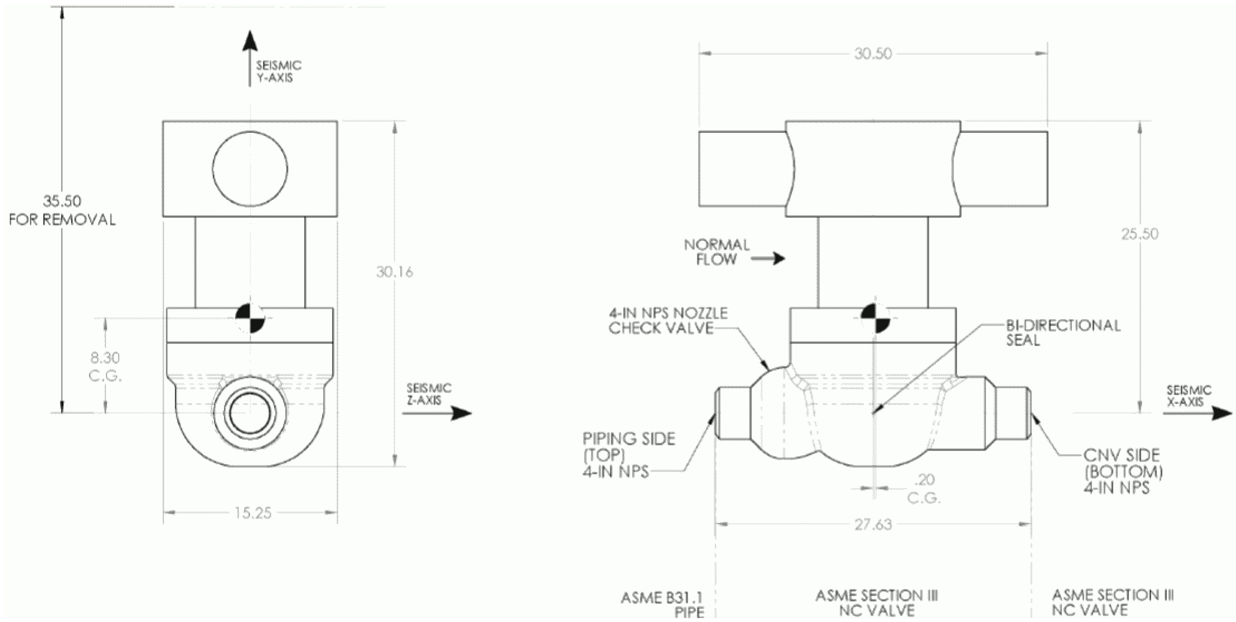
3.3.2.1 Feedwater Isolation Valves

There are two feedwater containment penetrations. Both of these piping penetrations are 5-inch NPS. Each penetration is protected by a dual ASME BPVC Class 2 SSCIV. The FWIV has a dual valve, single body arrangement. Each assembly consists of an actuated isolation valve and an integral self-actuating check valve. The dual valve assembly is welded to a CITF that is welded directly to the CNV head via integral vessel nozzle safe end and butt weld arrangements.

The isolation valve is a hydraulic-to-open, ball valve that uses a stored energy device to close. It is the inboard valve in the dual valve arrangement. The outboard valve is a safety-related nozzle check valve. The function of the check valve is to close more rapidly than the inboard isolation valve to preserve DHRS inventory in the event of a feedwater line break outside containment. The FWIV has no containment isolation function and no other specific leakage criteria.

Neither the FWIV nor the feedwater isolation check valve has a Type C test requirement as defined by the IST Plan, which is typical for FWIV of a PWR design; however, the FWIV is classified as IST Category A because of the specific leakage criteria needed for DHRS operability. As such, the CITF on each FWIV is used for IST purposes and no Type B test is performed on the cover.

Figure 3-9 Feedwater Isolation Valve Dual Valve, Single Body Design



3.3.3 Steam Generator Thermal Relief Valves

Two thermal relief valves are provided inside containment with one on each of the SG closed loops. These relief valves provide secondary side overpressure protection during chemistry control evolutions in support of startup and shutdown. Feedwater lines, SG, and main steam lines could experience water solid conditions if the SSCIVs inadvertently close during these evolutions with decay heat in the reactor core.

The thermal relief valves are designed to relieve thermal overpressure during water solid conditions to maintain the integrity of the SG closed loops. The valves are installed on flanges to facilitate removal for periodic replacement as established by the IST Program. These valves form part of the SG closed loop boundary as part of the containment boundary. However, they are not leak tested per 10 CFR 50, Appendix J, because these valves

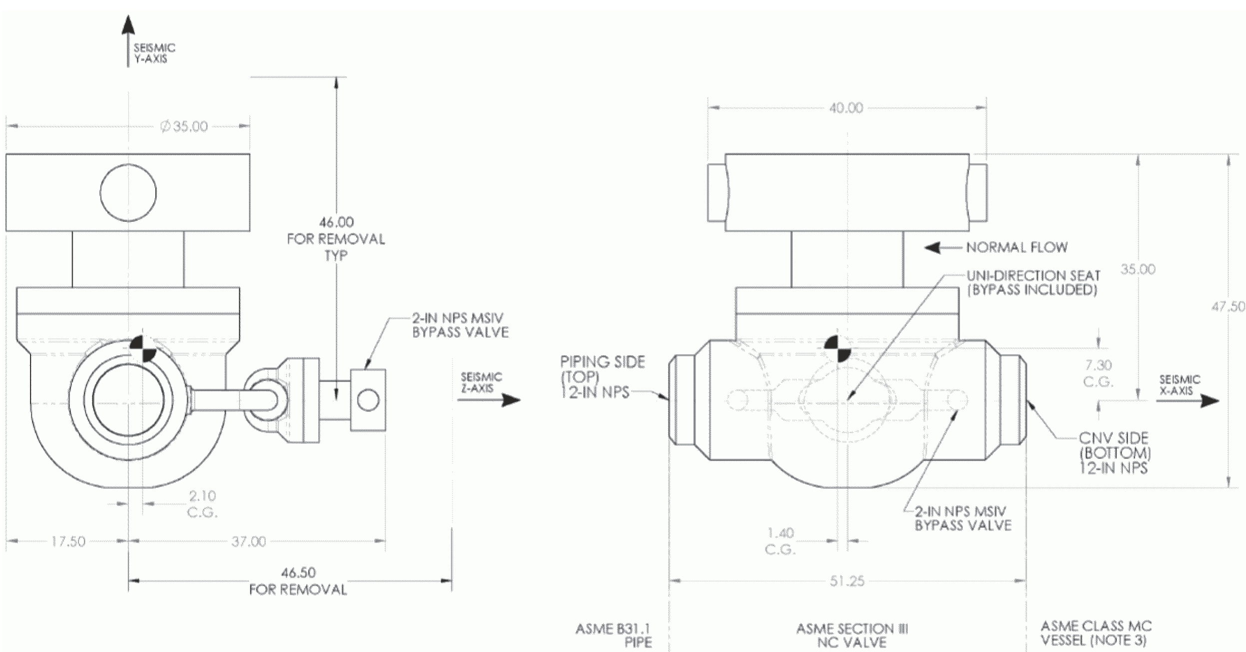
- are ASME BPVC Class 2, Quality Group B.
- relieve into the CNV.
- have a set pressure approximately 1,000 psia above peak operating pressure (2,200 psia vs. 1,200 psia).
- are leak tested pursuant to ASME Operation and Maintenance Code Mandatory Appendix I with the licensee leakage criteria for replacement valves specified so that reliable operation can be reasonably established throughout the ten-year test interval.

3.3.4 Main Steam Isolation Valves and Bypass Valves

There are two main steam containment penetrations, both of which are 12-inch NPS. Each penetration is protected by an ASME BPVC Class 2 SSCIV with an integral bypass valve on a 2-inch NPS bypass line. Both the MSIVs and the main steam isolation bypass valve (MSIBV) have a single actuator, single body arrangement. The MSIBV is integral to the MSIV in a parallel arrangement but operated independently. The MSIV and MSIBV are hydraulic-to-open,ball valves that use a stored energy device to close. Each valve assembly is welded directly to the CNV head via integral vessel nozzle safe end and butt weld arrangements.

The functions of the MSIV and the MSIBV are containment isolation, main steam isolation, and DHRS boundary during DHRS actuation. Neither the MSIV nor the MSIBV has a 10 CFR 50, Appendix J, Type C test requirement as defined by the IST Plan. However, both the MSIV and the MSIBV are classified as IST Category A because of the specific leakage criteria needed for DHRS operability. Thus, the leak testing features on these valves is only used for IST purposes.

Figure 3-10 CombinedMain Steam Isolation Valve and Main Steam Isolation Bypass Valve, Single Unit Design



3.3.5 Feedwater Plenum Cover Drain Valves

Each SG feedwater plenum cover is equipped with a drain valve to facilitate draining the SG during outage maintenance evolutions. As they are used only for system maintenance, they are exempt from the IST Plan, as well as 10 CFR 50, Appendix J testing.

4.0 Inservice Inspection and Testing of Containment

Inservice inspection and IST are required by 10 CFR 50.55a(g) and (f), respectively, to ensure periodic requisite inspection and testing is performed on the CNTS that ensures leak-tight integrity is maintained. Type B testing is specified in the ISI Plan and Type C testing in the IST Plan. Both the ISI and IST Programs are an integral part of the CLRT Program.

4.1 Inservice Inspection

Inservice inspection provides an essential function for the CLRT Program by confirming CNTS integrity and ensuring no new leakage paths are present as discussed in the following sections.

4.1.1 Weld Inspections

The specified surface, volumetric (ultrasonic), and visual examinations within the ISI Program provide reasonable assurance that no new leakage paths will develop over the service life of the NPM.

The CNV design allows comprehensive inspections of welds, including volumetric and surface inspections. The CIVs are located outside the CNV and pressure boundary welds are accessible with no areas that cannot be inspected. The reduced ISI requirements from ASME BPVC Section XI for small primary system pipe welds between the CNV and the CIV are not applied to these welds. Welds between the CNV and the CIV are inspected each test interval.

4.1.2 Visual Inspections

ASME BPVC, Class MC, Section IWE requires visual examination for structures, systems, and components subject to normal degradation and aging. Surface areas subject to accelerated degradation and aging require an ultrasonic thickness exam. Additionally, based on the high pressure and safety functions of the CNV, the ISI Program requires augmented examinations of the CNV in accordance with ASME Code, Class 1 requirements. The CNV design allows visual inspection of the entire inner and outer surfaces; therefore, developing an undetected leak through the metal pressure boundary is unlikely.

4.1.3 Bolting Inspections

Inspection of CNV bolting is required per ASME BPVC, Section XI. For bolting provided for the CNV closure flange (i.e., CNV main closure studs), these inspections are limited to visual examination (VT-1) per ASME BPVC, Section XI, Subsection IWE-2500, Category E-G for metal containments.

Bolting of other flanges installed on the CNV are subject to visual examination (VT-1) per ASME BPVC, Section XI, Subsection IWE-2500, Category E-G. These examinations may be performed without disassembling the joint; however, if the joint is disassembled during the inspection interval, the examination is required to be performed with the joint disassembled.

4.1.4 Steam Generator Inspections and Controls

Each SG forms part of a GDC 57 closed loop containment barrier for PWRs; therefore, the reliability of its integrity and its failure mechanisms contribute to the reliability of the containment boundary. The NuScale SG design is different from traditional SGs. Major differences include:

- The SG is located inside the RPV and is not a separate component attached by RCS piping.
- The tubes are helically coiled in the annular space between the walls of the RPV and a concentric upper riser internal to the RPV.
- Steam is generated on the inside of the tubes with lower steam pressure inside the tube and higher RCS pressure on the outside.

As a GDC 57 closed loop system, each SG is isolated by single SSCIV (FWIV on the inlet, MSIV and MSIBV on the outlet). The SG is an ASME BPVC Class 1 RCPB. Detailed inspection requirements for the SG tubing are part of the ISI program. Technical Specification Section 5.5.4 establishes a Steam Generator Program to ensure SG tube integrity is maintained.

4.1.5 Type B Testing

Type B testing is local pneumatic pressure leak rate testing of containment penetrations, specifically the EPAs, IPAs, access ports, ECCS pilot valve bodies, and the CNV closure flange. These tests are inservice inspections specified in the ISI Plan. The ISI Program specifies Type B LLRTs.

4.2 Inservice Testing

The IST Plan identifies valves in the scope of the IST Program with specific leakage criteria. Valves with specific leakage criteria as a containment boundary are identified as “LTJ,” which denotes a valve with a 10 CFR 50, Appendix J, Type C leakage test requirement. The IST Plan also specifies test frequencies pursuant to the requirements of NEI 94-01 Rev. 3-A.

5.0 Type B Local Leak Rate Testing

Type B tests of the double seals on the containment bolted closures are performed by local pressurization at containment peak accident pressure, P_a. Pressurized gas such as air or nitrogen is applied to the test ports, which are provided between the two seals in each bolted closure and the pressure decay over time or the leak flow rate is measured.

Type B tests use either the pressure decay or flow makeup method of detection. For the pressure-decay method, a test volume is pressurized with air or nitrogen to at least P_a. The rate of decay of pressure in the known test volume is monitored over time to calculate a leakage rate using the pressure-decay method. For the flow makeup method, the required test pressure is maintained constant in the test volume by making up test fluid, such as air or nitrogen, through a calibrated flowmeter. With pressure held constant, the makeup flow rate is equivalent to the leakage rate from the test volume.

The design combined leakage rate for penetrations and valves subject to Type B and Type C tests is limited to less than $0.6L_a$. An overall leak rate of less than $0.6L_a$ (Section 7.2) is confirmed by LLRT before the startup of each NPM. In accordance with 10 CFR 50, Appendix J, Type B tests are performed during each reactor shutdown for refueling or at other convenient intervals in accordance with the CLRT program.

5.1 Type B Test Method

Type B penetrations are tested each refueling outage. An as-found test is required to be performed before any Type B penetration is opened or manipulated in a way that would affect the leak-tightness of the penetration (Section 7.5 has a discussion of test considerations, including preconditioning). Test equipment is installed on the test port located between the double seals. The seal is then tested with compressed air or nitrogen using either the pressure decay or flow makeup method to measure the leakage as specified in the CLRT program. Once as-found testing is performed and documented, the penetration can be opened. Just inside the CNV head manway is a small tubing connection to the CNV closure flange test port. The Type B test rig is connected at this point and an as-found test of the CNV closure flange is performed (Figure 3-6).

Once the refueling outage is completed and penetrations are closed for the final time, an as-left Type B test is performed on penetrations. If a penetration was not opened and no bolts were manipulated, and the as-found test was within CLRT acceptance criteria, then the as-found test may be credited as the as-left test with no further testing needed.

The CNV closure flange is tested twice after it is reassembled. The first as-left test occurs in the refueling area to ensure the new CNV closure flange seals are installed properly and are sealed. The second (final) as-left test occurs after the NPM is moved to the operating bay. The second as-left test ensures CNV movement had no adverse effect on the leak-tightness of the CNV closure flange seal. After the CNV closure flange seal is tested in the operating bay, then the CNV head manway cover is reinstalled and tested.

Section 7.5 contains additional discussions on as-found and as-left testing.

5.2 Electrical Penetration Assemblies

The EPA sheath modules are installed and tested at the shop. Glass-to-metal seals (penetrations), exclusive of the flange-to-nozzle seals, are designed for leakage rates not to exceed 1.0 x 10^{-3} standard cm^3/s [1.27 x 10^{-4} standard cubic foot per hour (scfh)] of dry nitrogen at design pressure and ambient temperature. Glass-to-metal seals typically achieve leak rates in the undetectable range, or 1.0 x 10^{-7} standard cm^3/s of dry nitrogen at design pressure and ambient temperature.

The glass-to-metal module seal is an established sealing technology that is not vulnerable to thermal or radiation aging and does not require periodic maintenance or testing. This module-to-EPA seal does not require periodic testing. It would only be tested after completing maintenance activities that affect the seal. The EPA flange seal is the same double o-ring seal design of all Type B penetration seals. The required installation acceptance criterion for leakage rate of each EPA penetration is 1.0 x 10^{-2} standard cm^3/s (1.27 x 10^{-3} scfh) per Institute of Electrical and Electronics Engineers Standard 317-1983 (Reference 9.5). The leakage margin allotment for Type B testing is preliminarily selected to be {{ }}$^{2(a),(c)}$ the installation acceptance criterion. With this allotment, the EPA contribution to the overall containment leakage rate does not challenge the acceptance criterion of $0.6L_a$.

5.3 Instrument Seal Assemblies

The ICI containment vessel compression seal fitting is a mechanical seal device that contains a primary seal, which is solely responsible for maintaining the CNV pressure boundary at the outside diameter of the ICI stringer assembly. The primary seal uses metal ferrules to form a leak tight compression seal with the OD of the ICI stringer assembly. Mechanical requirements are specified to aid with proper swaging of the ferrules, ensuring a leak tight seal is formed when properly installed. A leakage test port in the ICI containment vessel compression seal fitting allows local pneumatic pressurization of the region between the primary seal and test seals to allow Type B testing. The test port and the test seals perform no function during normal plant operation.

5.4 Access Ports

The CNV access port flange seals feature an identical double seal design. The leakage performance of these seals is similar to the EPA flanges based on an evaluation of leakage performance for off-the-shelf metal seals.

5.5 Emergency Core Cooling System Pilot Valve Bodies

There are four 3-inch NPS containment penetrations for the ECCS trip and reset valve assemblies (two of each). A Type B test is required at the double seal between the valve bonnet and body (Figure 3-5). The rest of these valve bodies are self-contained metal barriers that form part of the containment pressure boundary. Leakage criteria for these seals are lower than other Type B boundaries because of the smaller size of the seals.

5.6 Containment Vessel Closure Flange

The CNV closure flange is a large, double seal design with an approximately 45-foot circumference. This seal maintains the containment boundary between the upper and lower CNV assemblies. The CNV closure flange leakage limit for the CLRT program is estimated to be {{ }}[2(a),(c)] based on the linear seal length and performance of off-the-shelf metal seals.

5.7 Bolting

The CNV bolting design for the closure flange and EPAs, instrument seal assemblies, and access ports is in accordance with ASME BPVC, Section III, Division 1, Subsection NB. The preload needed to maintain a tight joint maintains seal integrity with design pressure in containment. Preload requirements are:

- The bolt preload for the design pressure is sufficient to resist the hydrostatic end force and maintain a compression load on the gasket contact surface to ensure a tight joint when the design pressure is applied to the internal surfaces.
- Preload is applied to the joint at atmospheric conditions without the presence of internal pressure.

The CNV bolted closure design and preload design requirements ensure Type B flange seals, including EPAs and ISAs, remain in contact at accident temperature concurrent with peak accident pressure.

Flanges are as-found tested in accordance with 10 CFR, 50 Appendix J, before removal for refueling outage activities. The licensee's administrative controls are used during reassembly, including preload verification and quality control hold points, to ensure EPAs, ISAs, access ports, and flange seals are reassembled with fasteners at the correct preload. An as-left Type B test on the penetration seal verifies leakage is within the CLRT program limit.

6.0 Type C Local Leak Rate Testing

The PSCIVs are tested using either the pressure decay or flow makeup method at containment peak accident pressure, P_a. For the pressure decay method, the test volume is pressurized with air or nitrogen. These test methods are described in Section 5.0. Pressure to the PSCIV is applied in the same direction as the pressure applied when the valve is required to perform its safety function. The CITF valve is closed to provide a test boundary upstream of the PSCIV.

There are no Type C leak test requirements for the SSCIVs (FWIVs, MSIVs, and MSIBVs), although these valves do have specific leakage criteria for DHRS operability. Leak testing of the SSCIVs is in accordance with the TS and the IST Plan to maintain DHRS operability. The leak testing features on these valves is used only for these purposes.

6.1 Type C Test Method

Each CIV to be tested is closed by normal means without preliminary exercising or adjustments (Section 7.5 has a discussion of test considerations, including preconditioning). This closure can be achieved via the periodic closed-stroke test required by the IST Program. Piping is drained and vented as needed and a test volume is established to produce a differential pressure across the valve when pressurized.

The CIV is then prepared for testing by removing the cover of the upstream CITF valve and closing the internal ball valve, then reinstalling the cover to lock the ball valve in place. The test port on the CITF assembly is then used to establish test pressure in the same direction as the pressure applied when the CIV would be required to perform its safety function (i.e., upstream from the CNV). Test equipment is installed on the test port and system valves are aligned so a vent path is established downstream of the tested valve. The CIV is then tested via air or nitrogen using either the pressure decay or flow makeup method as specified in the CLRT program.

When testing the first CIV in the penetration is completed, the test equipment is vented and the valves are realigned. The first CIV is opened and the second CIV is closed to establish the test alignment for the latter.The test equipment is re-pressurized and the second CIV is then tested.

Once the LLRT is completed on the inboard and outboard CIVs, the system is vented and the test equipment is disconnected from the CITF test port. Then, the desired post-test system lineup is established, including opening the CITF ball valve and re-orienting the cover to lock it in place.

7.0 Containment Leakage Rate Test Program

The CLRT program contains the following attributes:

- Limits are applied that are established in the plant's design basis and the TS to establish LLRT criteria to ensure penetrations meet the preservice and periodic limit of $0.6L_a$ at P_a for the combined leakage rate of penetrations and valves subject to Type B and C tests.
- Type B LLRT is performed in accordance with the ISI Plan frequency.
- Type C LLRT is performed in accordance with the IST Plan frequency.
- The results of containment system ISI are documented.
- The results of found and as-left Type B and Type C LLRTs are documented.
- Post-maintenance testing results on a Type B and Type C pressure boundary are documented.
- An adverse condition is analyzed for generic considerations. Type B seals are the same double seal design, and Type C valves are the identical 2-inch design, except the two containment evacuation system CIVs, which are 4-inch. Additionally, each site employing the design has six identical NPMs and CNTS.
- Records are maintained to produce periodic leakage test summary reports that are available onsite for NRC review in accordance with NEI 94-01 Rev. 3-A.

7.1 Challenges Associated with Type A Testing

Besides the GDC 52 exemption basis discussed above, the actual performance of Type A integrated leak rate testing on an NPM poses significant challenges that render the test either invalid or infeasible. These challenges include but are not limited to

- temperature variations due to continuous heat transfer between the CNV (which is normally filled with water when fully assembled in the operating bay during refueling outages) and the UHS in which it is mostly immersed, as well as from core heat from the fuel at the bottom of the RPV.
- the procurement, arrangement, and installation of a multitude of high-precision temperature sensors beyond normally installed instrumentation to ensure temperature variations are properly detected and compensated in both liquid and air spaces.
- greater accuracy requirements for test instrumentation given the larger proportional impact of temperature changes and instrument errors on pressure at the magnitude of P_a, which is much higher compared to traditional PWR designs.
- the lack of available, off-the-shelf sensors capable of measuring dew point temperature and relative humidity at high-pressure, no-flow conditions that can also be arranged and calibrated per the requirements of Reference 9.5.
- the application of standard ILRT acceptance criteria (75 percent of L_a) to a much smaller CNV volume, resulting in an exceedingly low allowable leak rate that may not

be achievable during the ILRT, especially considering the increased effects of instrument errors.

Appendix D provides additional discussion on these challenges.

7.2 Containment Leakage Limits

The leak rates of penetrations and valves subject to Type B and Type C testing are combined in accordance with 10 CFR 50, Appendix J. The combined leakage rate for penetrations and valves subject to Type B and Type C tests shall be less than $0.6L_a$ at P_a. The CLRT limits are derived from the design basis limits to meet $0.6L_a$ for LLRT. If repairs are required to meet CLRT limits, the results are reported in a separate summary to the NRC, in accordance with 10 CFR 50, Appendix J, to include the structural conditions of the components that contributed to the failure. As each Type B or Type C test (or group of tests) is completed, the combined total leak rate is revised to reflect the latest results. Thus, a reliable summary of containment leak-tightness is maintained current. Leak rate limits and the criteria for the combined leakage results are described in the plant TS.

7.3 Test Frequency

Schedules for performing periodic Type B tests are specified in the owner's ISI Plan and periodic Type C tests are specified in the owner's IST Plan. The CLRT Program is endorsed in the plant TS Section 5.5.9. Provisions for reporting test results are described in the CLRT Program.

Conditional testing is in accordance with the owner's procedures, but includes Type B or Type C testing when repair, replacement, or modification to a containment pressure boundary takes place.

Upon initial startup of each NPM, Type B tests are performed during reactor shutdown or refueling, or at other convenient intervals, but in no case at intervals greater than two years (as specified in the owner's ISI Plan) per 10 CFR 50, Appendix J, Option B. Type C tests are performed during reactor shutdown or refueling, but in no case at intervals greater than 30 months (as specified in the IST Plan) per 10 CFR 50, Appendix J, Option B and NEI 94-01 Rev 3-A.

Performance-based (i.e., extended) test frequencies under 10 CFR 50, Appendix J, Option B can be adopted for Type B and Type C tests once satisfactory performance is established. Section 7.5 provides further details.

7.4 Test Results and Reporting Requirements

The CLRT Program reporting requirements are pursuant to 10 CFR 50, Appendix J, Option B. Preoperational and periodic tests are documented in a summary report that is made available for inspection, upon request, at the plant site. The summary report includes, at a minimum, a schematic arrangement of the leakage rate measurement system, the instrumentation used, the supplemental test method, the test program selected as applicable to the preoperational test, and subsequent periodic tests. The report contains an analysis and interpretation of the leakage rate test data for the Type B and Type C test results and the applicable ISI results to the extent necessary to demonstrate the acceptability of the containment leakage rate in meeting acceptance criteria.

For each periodic test, leakage test results from Type B and Type C tests are included in the summary report. The summary report contains an analysis and interpretation of the Type B and Type C test results and the applicable ISI results that were performed since the last inspection interval (usually the last refueling outage). Leakage test results from Type B and Type C tests that failed to meet the CLRT Program acceptance criteria are included in a separate accompanying summary report that includes interpretation of the test data, and the structural conditions of the containment or components, if any, that contributed to the failure in meeting the acceptance criteria.

If performance-based (i.e., extended) test frequencies under 10 CFR 50, Appendix J, Option B are adopted for Type B and Type C tests, additional reporting requirements are imposed by NEI 94-01, Rev. 3-A. To facilitate the transition from baseline frequency testing (30 months) to extended intervals, these requirements can be included as part of standard CLRT Program reporting from initial startup. Section 7.5 provides additional details.

7.5 Special Testing Considerations

7.5.1 As-Found Testing

As-found testing is performed to determinehow a component would perform if called upon in an accident scenario. It is the first actuation of the component that has been in standby mode since the last time it was tested or for normal operation. Technical specification limiting conditions for operation criteria are generally as-found values. The as-found results are used to determine whether the leak-tightness of a component and the overall containment degraded over time.

7.5.2 As-Left Testing

As-left testing is the final performance of a surveillance test or calibration of a component to determine its functional performance before placing it back into service. Technical specification surveillance test criteria are generally as-left values. 10 CFR 50, Appendix J, LLRT, requires as-found testing of Type B and Type C penetrations when entering a refueling outage, and as-left testing when reassembling Type B penetrations or performing post-maintenance testing on a PSCIV (if maintenance was performed that affected the leak tightness of the valve). The as-left results establish operational readiness until the next scheduled LLRT and ensure the leak-tightness of a component (and the overall containment) does not degrade to an unacceptable level over time.

7.5.3 Preconditioning

Preconditioning occurs when a component is exercised, adjusted, or otherwise manipulated before as-found testing performance. The ASME and NRC requirements do not allow preconditioning for the performance of any as-found testing. For Type C penetrations, the owner balances the requirements of the IST Program and the CLRT Program. As-found stroke time testing is required for IST and as-found LLRT is required for 10 CFR 50, Appendix J. These tests are coordinated to preclude preconditioning.

7.5.4 Reverse Direction Testing

Special considerations from References 9.4and 9.6 apply when testing a component in the reverse direction (i.e., applying test pressure in the opposite direction of post-accident pressure). For the design, no reverse direction testing is currently performed, but these requirements are reviewed if any reverse testing becomes necessary as the design matures.

7.5.5 Modifications After Preoperational Testing

Any major modification or replacement of components that are part of the containment pressure boundary performed after preoperational leakage rate testing are followed by a Type B or Type C test as applicable for the area affected by the modification. The measured leakage from the test is included in the summary report.

7.5.6 Option B Performance-Based Testing

Applicants that reference the NuScale power plant US460 standard design approval may not be able to initially adopt the test method frequencies specified in 10 CFR 50, Appendix J, Option B, Performance-Based Requirements. However, the licensee is expected to adopt Option B once sufficient operating history is obtained under Option A to use this performance-based approach.

Multi-module testing does not impact test frequencies of the owner’s CLRT Program. Multi-module testing does also not affect the test frequencies of either the ISI or IST Programs. Risk-informed methods are not available to initial ISI or IST Programs, yet multi-module testing is a factor in CLRT, ISI, and IST. Generic considerations of adverse conditions not only potentially affect similar components in the affected NPM. Consideration must also be given to similar components across the NPMs. A plant with six NPMs nominally plans for three refueling outages annually. This outage frequency provides a rapid accumulation of performance history for the CLRT, ISI, and IST Programs. With NRC approval, risk-informed methods could be applied sooner compared to a traditional one- or two-reactor plant.

Under Option B, the performance-based testing requirements established by NEI 94-01 Rev 3-A applies for Type B and Type C local leak rate testing. To summarize these requirements:

- The LLRT intervals for Type B components are increased from the baseline frequency of 30 months up to a maximum of once per 120 months (10 years) following the completion of two consecutive periodic as-found tests that satisfy administrative limits.
- The LLRT intervals for Type C components are increased from the baseline frequency of 30 months up to a maximum of once per 75 months following the completion of two consecutive periodic as-found tests that satisfy administrative limits.
- Though not required, the expected industry best practice is to obtain three satisfactory as-found tests before adopting a 75-month extended LLRT interval.
- If a 75-month interval is adopted for Type C LLRTs, the Limitations and Conditions provided in Reference 9.6 apply.
- If a valve is replaced or engineering judgment determines that modification of a valve has invalidated the valve performance history (for example, replacement of a part that affects seat tightness), the valve is tested at the baseline frequency of 30 months.
- For both Type B and Type C components, as-found testing is performed before a maintenance, repair, modification, or adjustment activity if the activity could affect the penetration's leak tightness. An as-left test is performed following maintenance, repair, modification, or adjustment activities.
- If results are not acceptable, then the testing interval is set at the baseline frequency, and a cause determination is performed and corrective actions identified that focus on activities that can eliminate the identified cause of failure with appropriate steps to prevent recurrence.
- A post-outage report is prepared presenting results of the previous cycle Type B and Type C tests, as well as Type B and Type C tests performed during that outage.
- The technical contents of the report are generally described in Reference 9.4 and are available on-site for NRC review. The report documents that the applicable performance criteria are met, and serves as a record that continued performance is acceptable.
- The report includes the combined Type B and Type C leakage summation, and the margin between the Type B and Type C leakage rate summation and its regulatory limit (Topical Condition 1 discusses further). Adverse trends in the Type B and Type C leakage rate summation are identified in the report and a corrective action plan is developed to restore the margin to an acceptable level.

Certain limitations apply as described in Section 11.0 of Reference 9.6. Beyond this standard, the following considerations also inform the owner's Option B testing plan:

- An as-found test is performed before work is done that can affect the leak rate of a component whose leakage integrity is suspect (Section 3.3.4.1 of Reference 9.4).

- In the event mid-cycle maintenance is required between refueling outages for a component on an extended LLRT interval, the as-left test from the last refueling outage may be counted as the as-found test.
- Extended LLRT intervals need not be short-cycled to accommodate Type A tests (i.e., ILRT) because an exception is being pursued for these tests as discussed in Section 2.0.

Additionally, the NRC Safety Evaluation Report incorporated into NEI 94-01 Rev. 3-A imposes two additional conditions that must be considered for Option B testing. Topical Condition 1 requires if a 75-month extended interval is adopted for Type C tests, then (1) a licensee's post-outage report shall include the margin between the Type B and Type C leakage rate summation and its regulatory limit and (2) a corrective action plan shall be developed to restore the margin to an acceptable level. Topical Condition 2 requires if a Type C test interval is extended beyond 60 months (i.e., up to 75 months), then CLRT program trending and monitoring must include an estimate of the amount of understatement in the Type B and Type C leakage rate summation that must be included in a licensee's post-outage report. The report must include the reasoning and determination of the acceptability of the extension, demonstrating that the LLRT totals calculated represent the actual leakage potential of the penetrations. The common industry practice is to include an additional 25 percent of the measured leak rate on any Type C component on a 75-month interval in the Type B and Type C leakage rate summation (e.g., if a particular valve has a measured leak rate of 0.1 scfh, a value of 0.125 scfh would be included in the total).

Upfront consideration of Option B requirements can facilitate the eventual transition from baseline frequency testing (30 months) to extended intervals if CLRT Program documentation, procedures, and tools used for scheduling, trending, and monitoring are initially developed with these requirements already (or readily able to be) incorporated.

8.0 Material Selection and Aging Degradation Leakage Rate Test Program

The containment design includes material selection that supports leakage integrity assurance. Potential degradation can be identified by inspection and examination before the formation of containment leakage pathways.

8.1 Material Selection and Operating Conditions

Table 8-1 lists the CNV pressure-retaining materials. Nonpressure-retaining materials are not included or discussed because they are not involved in maintaining the leak tightness for the CNV. Most of the CNV is immersed in reactor pool water during plant operation. The CNV operating temperature is 100 degrees F. Although the minimum specified pool water temperature is 65 degrees F, the typical pool water temperature is approximately 100 degrees F under operating conditions.

During the plant shutdown process, the CNV is flooded with reactor pool water when the operating condition is in the safe shutdown mode and the RCS coolant temperature drops below 300 degrees F.

The portion of the CNV in contact with RCS coolant during plant operation is the only portion that is part of the RCPB. The CNV components that are in contact with RCS coolant include the following nozzles and their safe ends:

- CVCS injection nozzle (CNV6)
- CVCS discharge nozzle (CNV13)
- PZR spray nozzle (CNV7)
- RPV high point degasification nozzle (CNV14)

The CNV (shells, flanges, top head, nozzles, and covers) is made of Grade F6NM martensitic stainless steel that is included as an allowable material by Regulatory Guide (RG) 1.84 by approval of ASME BPVC Case N-774 or SA-182 except for a portion of the lower CNV. The lower CNV (lower head, core region shell, and transitional shell) with peak 60-year design fluence exceeding 1E+17 n/cm^2, > 1 million electron volts (MeV) is made of SA-965 Grade FXM-19 austenitic stainless steel.

Table 8-1 Containment Vessel Pressure-Retaining Materials

Item	Material
CNV Vessel	
CNV top head cover	SA-182 Grade F6NM
CNV top head and upper CNV shells and flange	ASME BPVC Case N-774 (SA-336) Grade F6NM
Lower CNV (flange and lower shell)	ASME BPVC Case N-774 (SA-336) Grade F6NM
Lower CNV (lower head, core region shell, and transition shell)	SA-965 Grade FXM-19 (Note 2)
Nozzles and Access Ports	
Nozzles and access ports	SA-182 Grade F6NM
Safe-ends for nozzles	SA-182 Grade F304 (Note 1)
Covers for access ports	SA-182 Grade F6NM
Pressure-Retaining Bolting	
Bolting for CNV main closure flange	SB-637 UNS N07718 (Note 3)
Bolting for other than CNV main closure flange	SB-637 UNS N07718 (Note 3) SA-564 Type 630 Condition H1100 SA-193 Grade B8 Class1 SA-194 Grade 8
Weld Filler Metals	
2XX austenitic stainless steel weld filler metals	SFA-5.4: E209, E240 (Note 2) SFA-5.9: ER209, ER240 (Note 2)
3XX austenitic stainless steel weld filler metals	SFA-5.4: E308, E308L, E309, E309L, E316, E316L (Note 4) SFA-5.9: ER308, ER308L, ER309, ER309L, ER316, ER316L (Note 4) SFA-5.22: E308, E308L, #309, E309L, ER316, ER316 (Note 4)
4XX martensitic stainless steel weld filler metals	SFA-5.4: E410NiMo SFA-5.9: ER410NiMo
Nickel-base alloy weld filler metals	SFA-5.11: ENiCrFe-7 SFA-5.14: ERNiCrFe-7, ERNiCrFe-7A, EQNiCrFe-7, EQNiCrFe-7A

Notes:
1. 0.03 percent maximum carbon if unstabilized Type 3XX base metals are welded or exposed to temperature range of 800 degrees F to 1500 degrees F subsequent to final solution anneal.
2. 0.04 percent maximum carbon for FXM-19 and Type 2XX weld filler metals.
3. SB-637 UNS N07718 solution treatment temperature of range before precipitation hardening treatment restricted to “1800 degrees F to 1850 degrees F.” In addition, nuts for the CNV main flange closure bolting are case-hardened to reduce galling.
4. 0.03 percent 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.

8.1.1 Pool Water Chemistry

The reactor pool chemistry is maintained consistent with the spent fuel pool chemistry requirements in the Electric Power Research Institute (EPRI) Primary Water Chemistry Guidelines (Reference 9.14). The reactor pool water chemistry parameters are listed in Table 8-2. No target limit and monitoring frequency are set for lithium, hydrogen peroxide, magnesium, calcium, or aluminum. Lithium is not expected to be added in the pool and hydrogen peroxide addition is only required as needed. The control limit and monitoring requirement for calcium, magnesium, and aluminum are specified by the fuel vendor.

Control limits are not included for pH and conductivity. The monitoring frequency is the only limit specified. The control limit is established for boron, rather than pH and conductivity, because the pool pH is primarily determined by boron concentration. The remainder of the NuScale reactor pool water chemistry parameters are listed in Table 8-2 with their target limits.

Table 8-2 Target Limits for Reactor Pool Water

Chemistry Parameter	Units	Value
Boric acid (as boron)	ppm	≥ 2000
Chloride	ppm	≤ 0.15
Fluoride	ppm	≤ 0.15
Sulfate	ppm	≤ 0.15
Silica	ppm	≤ 1.5
Aluminum	ppb	≤ 80
Magnesium	ppb	≤ 40
Calcium	ppb	≤ 40
Turbidity	NTU[1]	≤ 3.0
Gamma isotopic activity	mCi/gram	0.001
Tritium	mCi/gram	Trend
Note 1: NTU refers to a measure of the amount of suspended solids in a liquid.		

8.1.2 Reactor Coolant System Coolant Chemistry

The plant limits follow the EPRI Primary Water Chemistry Guidelines. Limits for chemical species are provided in Table 8-3. These reactor coolant chemistry specifications conform to the recommendations of RG 1.44 (Reference 9.17).

The RCS water chemistry is controlled to minimize corrosion of RCS surfaces and to minimize corrosion product transport during normal operation. The CVCS provides the means for adding chemicals through charging flow and for removing chemicals through dilution or purification. For reactivity control, boric acid is added as a soluble neutron poison. The concentration of boric acid is varied throughout reactor operation as needed for reactivity control.

To maintain the alkalinity of the coolant, lithium hydroxide enriched with the lithium-7 isotope is added to the coolant. Slight alkalinity is maintained in accordance with the recommendations of the EPRI Primary Water Chemistry Guidelines. This chemical is chosen for its compatibility with boric acid, stainless steel, zirconium alloys, and nickel-base alloys. Lithium hydroxide is added to the coolant through the charging flow of the CVCS. It is removed from the coolant by the purification systems of the CVCS or reduced in concentration by dilution.

The coolant pH is determined based on the recommendations in the EPRI PWR Primary Water Chemistry Guidelines and fuel vendor limits. The coolant maintains its reducing environment by adding dissolved hydrogen to the coolant. Hydrogen is used because of its compatibility with the aqueous environment and its ability to suppress radiolytic oxygen generation during normal operation. Dissolved hydrogen is added to the coolant by direct injection of high-pressure gaseous hydrogen into the CVCS charging flow. During startup, oxygen is removed by a combination of mechanical degasification by the CES and by chemical degasification using hydrazine. Hydrazine is an effective oxygen scavenger at low temperatures and is added to the coolant by the charging flow of the CVCS.

Table 8-3 Reactor Coolant System Coolant Chemistry

Parameter)	Normal Operating Range	RG 1.44 Limit
Chloride (ppm)	≤ 0.15	0.15
Fluoride (ppm)	≤ 0.15	0.15
Dissolved oxygen (ppm), operating	≤ 0.10	0.10
Sulfate (ppm)	≤ 0.05	N/A
Hydrogen (cc/kg), operating	15 - 50	N/A
Boron (ppm)	0 - 2,000	N/A

8.2 Aging Degradation Assessment

This section assesses the following aging degradations for the CNV pressure boundary materials:

- fatigue
- boric acid corrosion
- primary water stress corrosion-cracking (PWSCC)
- stress corrosion-cracking (SCC) of austenitic stainless steels
- SCC of pressure-retaining bolting materials
- irradiation embrittlement of lower CNV
- SCC of F6NM martensitic stainless steel

8.2.1 Fatigue

Pressure-retaining components of the CNV were analyzed for fatigue in accordance with applicable subsections of ASME Code Section III. The CNV components that are part of the RCPB are described in Section 8.1. For the CNV nozzles and their safe ends that are in contact with RCS coolant during normal plant operation, the fatigue analysis considers the environmental effects in accordance with RG 1.207 (Reference 9.18) and NUREG/CR-6909 (Reference 9.19). Therefore, cracking of CNV pressure-retaining components due to fatigue loading is unlikely during the design lifetime.

8.2.2 Boric Acid Corrosion

The pressure-retaining components for the CNV do not use low-alloy steel cladded or non-cladded with austenitic stainless steel. Pressure-retaining materials are either stainless steels or nickel-base alloys. Because CNV surfaces in contact with borated pool water or reactor coolant are corrosion-resistant stainless steels or nickel-base alloys, boric acid corrosion is not an applicable aging degradation mechanism for the CNV.

8.2.3 Primary Water Stress Corrosion-Cracking

Alloy 600 and its weld Alloy 82/182 are susceptible to PWSCC when exposed to high-purity deaerated, hydrogenated primary water at elevated temperatures.

The CNV components that are part of the RCPB are described in Section 8.1. The nickel-base alloy in contact with primary water is limited to Alloy 52/152 for the dissimilar metal welds between F304 safe-ends and the F6NM CNV top head. Extensive laboratory testing and PWR operating experience have confirmed Alloy 690/52/152 are highly resistant to PWSCC (Reference 9.7). The NuScale primary chemistry follows EPRI Primary Water Chemistry Guidelines that also minimize PWSCC. Therefore, PWSCC of Alloy 52/152 welds is unlikely.

8.2.4 Stress Corrosion-Cracking of Austenitic Stainless Steels

The austenitic stainless steels for the CNV pressure-retaining components other than pressure-retaining bolting are the following:

- SA-965 Grade FXM-19 (UNS S20910) for lower CNV
- Type 2XX weld filler metals for welding SA-965 Grade FXM-19
- SA-182 Grade F304 safe-ends
- Type 3XX weld filler metals for welding SA-182 Grade F304 safe-ends

Type 3XX weld filler metals are used for welds between Type 304 safe-ends and Type 304 piping. The circumferential welds between SA-965 Grade FXM-19 components in the lower CNV are joined with Type 2XX weld filler metals.

The carbon content of SA-182 Grade F304 safe-ends and Type 3XX weld filler metals is limited to 0.03 percent maximum. The carbon content of SA-965 Grade FXM-19 and Type 2XX weld filler metals is limited to 0.04 percent maximum. The limit on carbon content minimizes intergranular precipitation Cr-carbides due to exposure to elevated temperatures during welding and post-weld heat treatment (PWHT). If water quenching is not used following final solution anneal of SA-182 Grade F304, non-sensitization is verified by the American Society for Testing and Materials (ASTM) A262 Practice A or E.

The pressure-retaining austenitic stainless steels or welds are in contact with reactor pool water or RCS coolant during plant operation or NPM movement. Section 8.1 describes that the reactor pool chemistry is maintained consistent with the spent fuel pool chemistry requirements in EPRI Primary Water Chemistry Guidelines.

Type 304 austenitic stainless steel has been used for spent fuel pool liners and spent fuel pool racks with excellent operating experience. Occasionally, transgranular SCC has been observed in Type 304 piping when exposed to borated water near ambient temperatures in PWRs. This observation has been attributed to sensitization, elevated chloride concentration, and high residual stresses from welding (Reference 9.8). However, transgranular SCC is unlikely to occur in CNV SA-182 Grade F304 safe-ends based on the following considerations:

- Sensitization is prevented by limiting carbon content to 0.03 percent maximum. Nonsensitization is verified by ASTM A262 Practice A or E in accordance with RG 1.44 as described above.
- The pool chemistry is maintained consistent with the spent fuel pool chemistry requirements in EPRI Primary Water Chemistry Guidelines. Deleterious species in the water are monitored to remain below acceptable limits. Halogens such as chloride and fluoride are kept to 0.15 ppm maximum.

The SA-965 Grade FXM-19 (UNS S20910) is used for the lower CNV (Figure 8-1). This material is commonly referred to as XM-19 or Nitronic 50. Because of its higher yield strength and better SCC resistance than Type 304, XM-19 has been used extensively in boiling water reactor internals. According to Reference 9.9 there has been no failure or cracking of XM-19 after more than 25 years of service.

8.2.5 Stress Corrosion-Cracking of Pressure-Retaining Bolting Materials

The pressure-retaining bolting materials in the CNV are:

- SB-637 UNS N07718 (also known as Alloy 718)
- SA-564 Grade 630 (also known as Type 17-4PH), H1100
- SA-193 Grade B8 Class 1 and SA-194 Grade 8 (also known as Type 304)

Alloy 718 is used for studs and nuts for the CNV main closure flange between the upper CNV and lower CNV, and for various closure covers in the upper CNV shell. Type 17-4PH is used for studs and nuts for various closure covers in the CNV top head.

The pressure-retaining bolting materials are in contact with the pool water only during plant operation or NPM movement. As described in Section 8.1, the pool chemistry is maintained consistent with the spent fuel pool chemistry requirements in EPRI Primary Water Chemistry Guidelines.

Alloy 718 is an austenitic, precipitation-hardenable alloy whose composition is adjusted to enable strengthening by heat treatment. Alloy 718 has been used inside PWRs because of its excellent SCC resistance in primary water, although intergranular stress corrosion-cracking (IGSCC) of Alloy 718 has been reported (Reference 9.10). The Alloy 718 bolting for the CNV is submerged in the pool water during plant operation. Intergranular SCC of nickel-base alloys is unlikely at the pool water temperature of 100 degrees F. Alloy 718 contains at least 50 percent nickel. Alloys containing more than 30 percent nickel are extremely resistant to chloride-induced transgranular SCC (Reference 9.11). Therefore, SCC of Alloy 718 bolting in the CNV is unlikely.

Type 17-4PH is a martensitic precipitation hardenable stainless steel. Type 17-4PH in the H900 condition is relatively susceptible to SCC. However, laboratory SCC testing showed Type 17-4PH in the overaged H1100 condition is much more resistant to SCC (Reference 9.12). Type 17-4PH for CNV bolting is used in the overaged H1100 condition. There have been no reports of SCC of Type 17-4PH in the H1100 condition in PWR applications. Failure of Type 17-4PH in the H1100 condition due to thermal embrittlement has been reported after exposure to temperatures above 500 degrees F (Reference 9.13). However, thermal aging embrittlement is not a concern because Type 17-4PH is used in CNV locations where normal operation temperature is below 500 degrees F.

The pool chemistry is maintained consistent with the spent fuel pool chemistry requirements in EPRI Primary Water Chemistry Guidelines. Deleterious species in the water are monitored to remain below acceptable limits. Halogens such as chloride and fluoride are kept to 0.15 ppm maximum. Therefore, SCC of Type 17-4PH bolting in the CNV is unlikely.

Type 304 austenitic stainless bolting materials (i.e., SA-193 Grade B8 and SA-194 Grade 8) are used in the solution-annealed condition for CNV pressure-retaining applications. If water quenching is not used following final solution anneal, non-sensitization is verified by ASTM A262 Practice A or E. Bolting materials are not subject to sensitization temperature range of 800 to 1500 degrees F after final solution anneal. Therefore, SCC of Type 304 bolting in the CNV is unlikely.

8.2.6 Irradiation Embrittlement of Lower Containment Vessel

Figure 8-1 shows the pressure-retaining materials in the lower CNV. The lower CNV beltline region with a peak 57 effective full-power years (EFPY) fluence exceeding 1E+17 n/cm^2, > 1 MeV is bounded by the lower head, lower core region shell, and lower transitional shell. To avoid irradiation embrittlement in the lower CNV beltline region, SA-965 Grade FXM-19 austenitic stainless steel and compatible Type 2XX austenitic stainless steel weld filler metal are selected for the base metal and associated welds, respectively.

The lower flange and lower shell are farther away from the core and their peak 57 EFPY fluence is below 1E+17 n/cm^2, > 1 MeV. For the lower CNV non-beltline region, ASME BPVC Case N-774 SA-336 Grade F6NM martensitic stainless steel and compatible E/ER-410NiMo martensitic weld filler metal are selected for the lower flange and F6NM lower shell, and the weld between the two F6NM forgings, respectively. Because their fluence is below 1E+17 n/cm^2, > 1 MeV, the materials for the lower CNV non-beltline region materials do not have an irradiation embrittlement concern.

The 57 EFPY peak fluence in the lower CNV beltline region are the following: (1) CNV beltline base metal (SA-965 Grade FXM-19): 5.0E+18 n/cm^2, E > 1 MeV; and (2) CNV beltline weld metal (Type 2XX): 2.5E+18 n/cm^2, E > 1 MeV.

Fluence is usually converted to average number of displacements per atom (dpa) to enable comparison of irradiation embrittlement data originated from different reactor types, fluence is usually converted to average number of displacements per atom (dpa). A typical conversion factor for light water reactors is 1dpa = 6.7E+20 n/cm^2 per MRP-175 (Reference 9.20). Using this conversion factor, the dpa equivalent of the lower CNV peak fluence is the following: (1) CNV beltline base metal: 5.0E+18 n/cm^2, E > 1 MeV = 0.0075 dpa; and (2) CNV beltline weld metal: 2.5E+18 n/cm^2, E > 1 MeV = 0.0037 dpa.

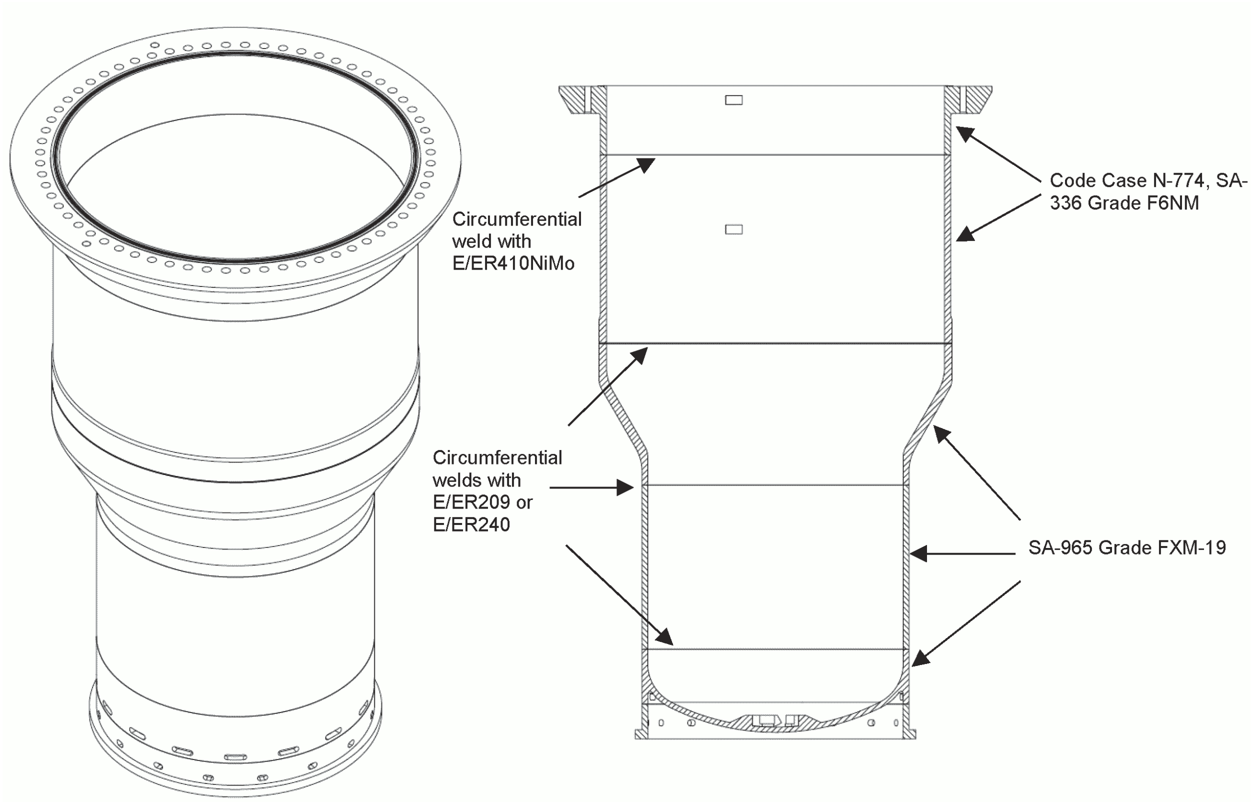
Based on extensive irradiated Type 3XX fracture toughness data, MRP-175 proposed the following screening fluence for irradiation embrittlement in PWR reactor internals: (1) wrought austenitic stainless steels > 1.5 dpa; and (2) austenitic stainless steel welds or cast austenitic stainless steel > 1 dpa.

However, NUREG/CR-7027 (Reference 9.15) proposed the following more conservative threshold fluence for irradiation embrittlement in Type 3XX austenitic stainless steels than the MRP-175 screening fluence: (1) wrought austenitic stainless steels = 0.5 dpa; and (2) austenitic stainless steel welds or cast austenitic stainless steel = 0.3 dpa.

Although XM-19 and E209/ER209 or E240/ER240 welds contain higher manganese and nitrogen content than Type 3XX, there is no data to indicate such differences have a pronounced effect on irradiation embrittlement. Fracture toughness testing of irradiated solution annealed XM-19 was performed under an EPRI- Department of Energy program (Reference 9.16). After 0.28 dpa at 340 degrees C (644 degrees F), fracture toughness was found to be J_{Ic} = 198 kJ/m^2 (K_{Jc} = 212 MPavm) when tested at 289 degrees C (552 degrees F). This XM-19 test data point is well within the scatter band of Type 3XX austenitic stainless steels reviewed by MRP-175 or NUREG/CR-7027. Therefore, XM-19 irradiation embrittlement behavior is similar to Type 3XX, at least up to 0.28 dpa.

The peak 57 EFPY fluence is 0.0075 dpa for the lower CNV beltline base metal and 0.0035 dpa for the lower CNV beltline welds. These peak fluence values are tiny fractions of either MRP-175 screening fluence or NUREG/CR-7027 threshold fluence for irradiation embrittlement. Therefore, loss of fracture toughness in the lower CNV beltline SA-965 Grade FXM-19 base metal or associated weld metal from neutron irradiation during the design lifetime is negligible.

Figure 8-1 Lower Containment Vessel Pressure-Retaining Materials



8.2.7 Stress Corrosion-Cracking of F6NM Martensitic Stainless Steel

Except for the beltline of the lower CNV, which is made of austenitic stainless steel, the CNV (top head, top head cover, nozzles, access ports, covers for access ports, and flanges) is made of Grade F6NM martensitic stainless steel per ASME BPVC Case N-774 or per SA-182 (Table 8-1).

The F6NM base metal is tempered at 1095 ± 25 degrees F for 8 hours in two steps of 4 hours each. After each tempering step, the base metal is cooled to below 175 degrees F. F6NM-to-F6NM welds are made with 4XX weld filler metals in Table 8-1. For dissimilar metal welds between F6NM and FXM-19 and between F6NM and Type 304, the F6NM base metal is buttered with 2XX weld filler metals or with nickel-base alloy weld filler metals in Table 8-1, respectively. The PWHT after welding of F6NM-to-F6NM or after buttering of F6NM base metal is in accordance with ASME

BPVC, Section III, NB-4622 for P-No.6 Gr.4. The PWHT temperature is 1075 ± 25 degrees F for minimum holding time per ASME BPVC, Section III, Table NB-4622.1-1.

The CRDM pressure housings are known examples of welded martensitic stainless steel pressure-retaining components in LLWRs. Type 403 martensitic stainless steel has been used in CRDM pressure housing of United States PWRs since the 1970s. In such CRDM designs, the Type 403 section was welded to austenitic stainless steel by dissimilar metal welds to form a complete pressure housing. There have been no reports of degradation of CRDM pressure housings containing Type 403, except for one leaking CRDM pressure housing at Prairie Island Unit 2 in 1998. Subsequent failure analysis (Reference 9.21) identified a pre-existing crack from fabrication and concluded the leak was due to a fabrication defect.

Alloy F6NM martensitic stainless steel has been used in CRDM pressure housings of German Konvoi PWRs since 1988. The Konvoi CRDM design was later adopted by US EPR. According AREVA's responses to NRC requests for additional information (RAIs) (Reference 9.22 and Reference 9.23), the Konvoi CRDM pressure housing had been in service for 19 years (at the time of AREVA response to RAIs in 2009) without crack indications or leakages. The upper dissimilar metal weld was inspected by eddy-current examination from the inside diameter, and by ultrasonic examination and liquid penetrant examination from the outside diameter. No intergranular corrosion attack, cracking due to IGSCC, transgranular SCC, and thermal embrittlement, or leakages had been detected in any Konvoi CRDM pressure housings.

The Konvoi PWRs in Germany are still operating as of October 2022, but are scheduled for permanent shutdown by the end of 2022. They have operated for about 34 years since 1988 without reports of cracking or degradation issues related to the CRDM pressure housings and the dissimilar metal welds.

In the response to NRC RAIs (Reference 9.22 and Reference 9.23), AREVA stated that F6NM is not susceptible to SCC based on laboratory SCC testing at 599 degrees F. The specimens were loaded to 90 percent of room temperature yield strength. No cracking was observed in any of the 177 specimens tested under simulated PWR primary water after 4200 hours. Additional testing was performed under more aggressive water chemistry:

a. 1750 ppm H_3BO_3 + 10 ppm chloride/N_2H_4 for 880 hours

b. 1750 ppm H_3BO_3 + 100 ppm chloride/N_2H_4 for 984 hours

c. Oxygen saturated water containing 100 ppm chloride

No cracking was observed under conditions (a) and (b), but cracking was observed under condition (c). However, AREVA noted condition (c) was far outside normal PWR primary water chemistry.

The design ensures the CNV inside surface is exposed to vacuum during operation. The top portion of the upper CNV is not submerged during plant operation. The rest of the CNV outside surface is in contact with reactor pool water, whose water chemistry is similar to PWR primary water during shutdown period (Section 8.1).

Type 410 martensitic stainless steel is susceptible to SCC in LLWRs if it is tempered below 1050 degrees F, resulting in hardness in excess of Rockwell C hardness number (HRC) 33. Some examples of SCC of Type 410 are listed below.

- The NRC Information Notice No. 85-59 (Reference 9.24) described several incidents of cracked Type 410 valve stems and shafts due to IGSCC. In all cases, the cracking was attributed to excessive hardness levels corresponding to tempering temperature between 700 to 1050 degrees F.
- The NRC Information Notice No. 86-39 (Reference 9.25) described cracked Type 410 wear rings in residual heat removal pumps due to IGSCC. The cracking was attributed to excessive hardness HRC 33 to 39.
- The NRC Information Notice No. 88-85 (Reference 9.26) described cracked Type 410 studs due to IGSCC. The cracking was attributed to excessive hardness HRC 36.
- The NRC Bulletin 89-02 (Reference 9.27) described cracked Type 410 bolting for a swing check valve in the residual heat removal system. Cracking was due to SCC and was attributed to excessive hardness HRC 36. It was not reported if the SCC was IGSCC or transgranular SCC.
- The NRC Information Notice No. 95-26 (Reference 9.28) described cracked Type 416 lock nuts for safety-injection pumps due to IGSCC. The cracking was attributed to excessive hardness HRC 47. Type 416 is the free-machining grade of Type 410 (intentionally adding 0.15 percent minimum sulfur).

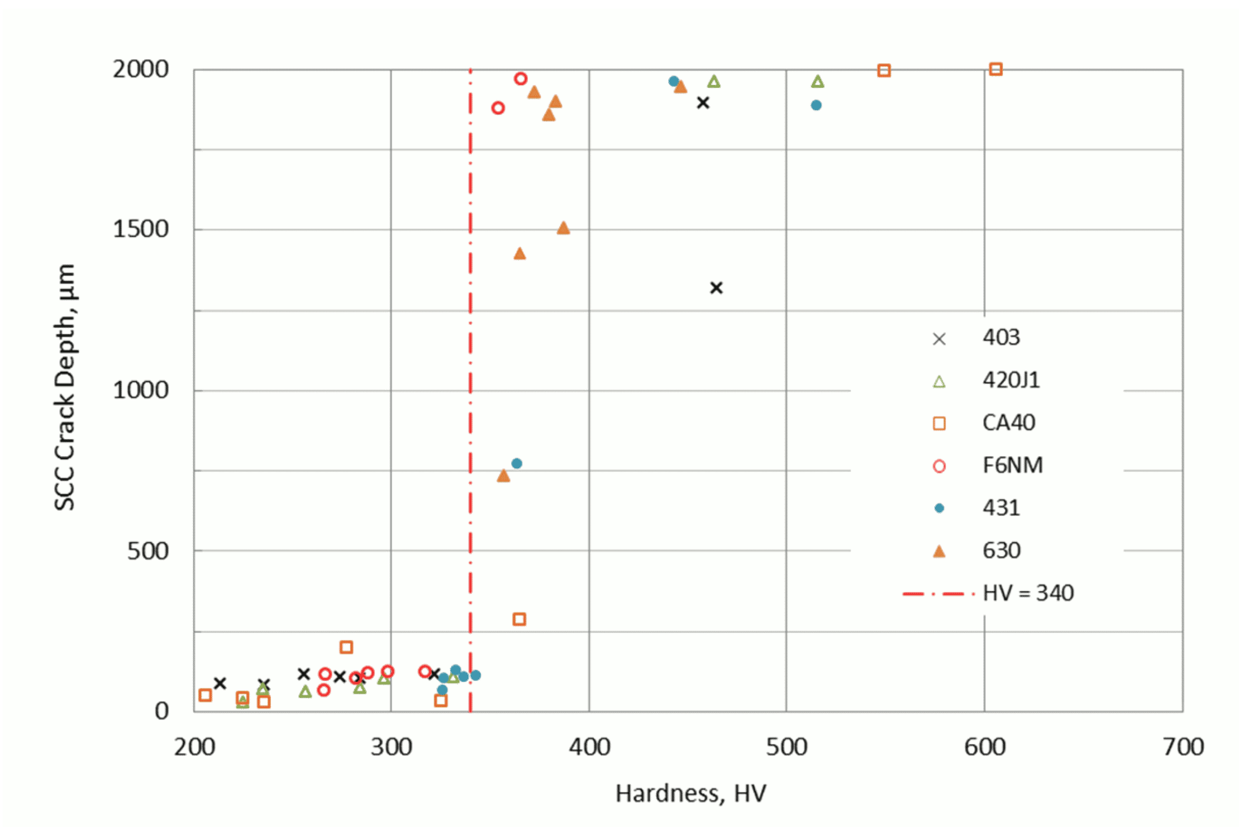
Tsubota (Reference 9.29) performed laboratory SCC tests of different martensitic stainless steels including F6NM using creviced bent beam type specimens. To examine SCC sensitivity to tempering temperature, each material was tempered at several temperatures. Alloy F6NM tempered at 580 and 600 degrees C (1076 and 1112 degrees F) is consistent with the F6NM used for the CNV.

The SCC testing was performed in 550-degrees F water saturated with oxygen at room temperature. After 500 hours, the creviced bent beam specimens were examined for crack depth, which was plotted as a function of hardness. Figure 8-2 shows an abrupt increase in SCC susceptibility when hardness exceeded 340 Vickers Hardness number (HV) for the materials tested. The SCC susceptibility is low if F6NM hardness level is kept below 340 HV, which is equivalent to HRC 34. The laboratory test results were also consistent with inservice SCC failures of insufficiently tempered Type 410 in LLWRs.

The hardness control is confirmed to be effective by Konvoi CRDM latch housings made of F6NM. During the Konvoi CRDM housing production, the F6NM latch housing hardness was limited to 350 HV maximum (Reference 9.22 and Reference 9.23). In the case of NuScale, the maximum hardness for F6NM base metal for the CNV is limited to Brinell hardness number 295 maximum by ASME BPVC Case N-774 and SA-182. Brinell hardness number 295 is equivalent to 310 HV or HRC 31. In addition, the minimum PWHT temperature for F6NM-to-F6NM welds and buttering of F6NM in the CNV is 1050 degrees F in accordance with ASME Section III, NB-4622.

Therefore, based on the operating experience of welded CRDM pressure housings containing martensitic stainless steels and laboratory SCC test results of martensitic stainless steels in simulated PWR and BWR primary water, SCC of Grade F6NM used in the CNV is unlikely.

Figure 8-2 Stress Corrosion Cracking Depth as a Function of Hardness, Martensitic Stainless Steels
(Reference 9.29)



9.0 References

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9.24 NRC Information Notice No. 85-59: Valve Stem Corrosion Failures.

9.25 NRC Information Notice No. 86-39: Failures of RHR Pump Motors and Pump Internals.

9.26 NRC Information Notice No. 88-85: Broken Retaining Block Studs on Anchor Darling Check Valves.

9.27 NRC Bulletin 89-02: Stress Corrosion Cracking of High-Hardness Type 410 Stainless Steel Internal Preloaded Bolting in Anchor Darling Model S350w Swing Check Valves or Valves of Similar Design.

9.28 NRC Information Notice No. 95-26: Defect In Safety-Related Pump Parts Due To Inadequate Heat Treatment.

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Appendix A Containment Isolation Summary Figures

The following figures are provided to show the isolation valves and closed loop piping systems that form part of the containment pressure boundary for the GDC 55, GDC 56 and GDC 57 piping penetrations of the CNV. Collectively, these figures identify the fluid service penetrations of the NuScale containment.

Table A-1 Simplified Figures Illustrating the Containment Pressure Boundary for the General Design Criteria 55, General Design Criteria 56 and General Design Criteria 57 Piping Systems of the Containment Vessel.

A-1	CNTS
A-2	DHRS
A-3	SGS

Figure A-1 Containment System

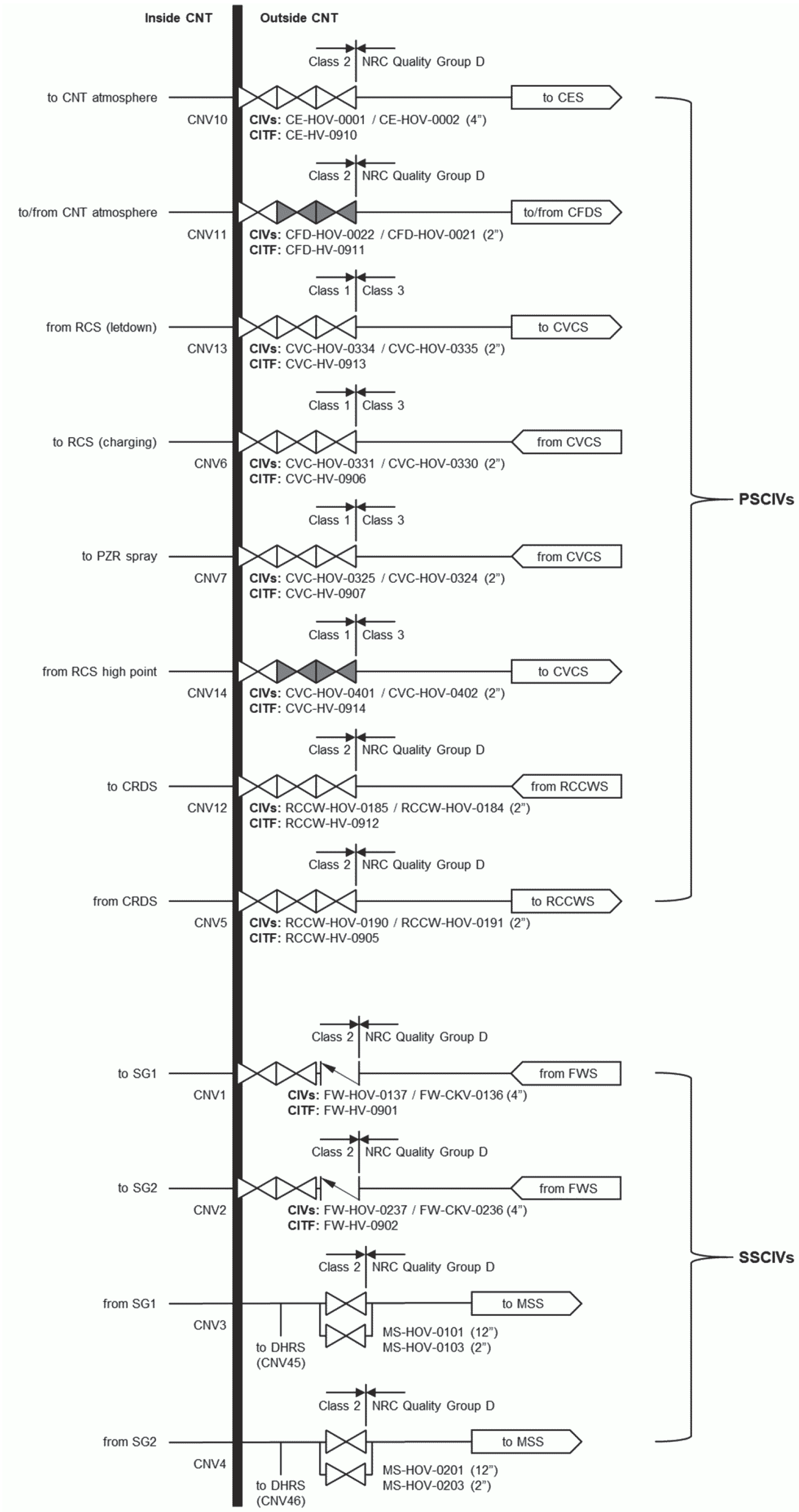


Figure A-2 Decay Heat Removal System

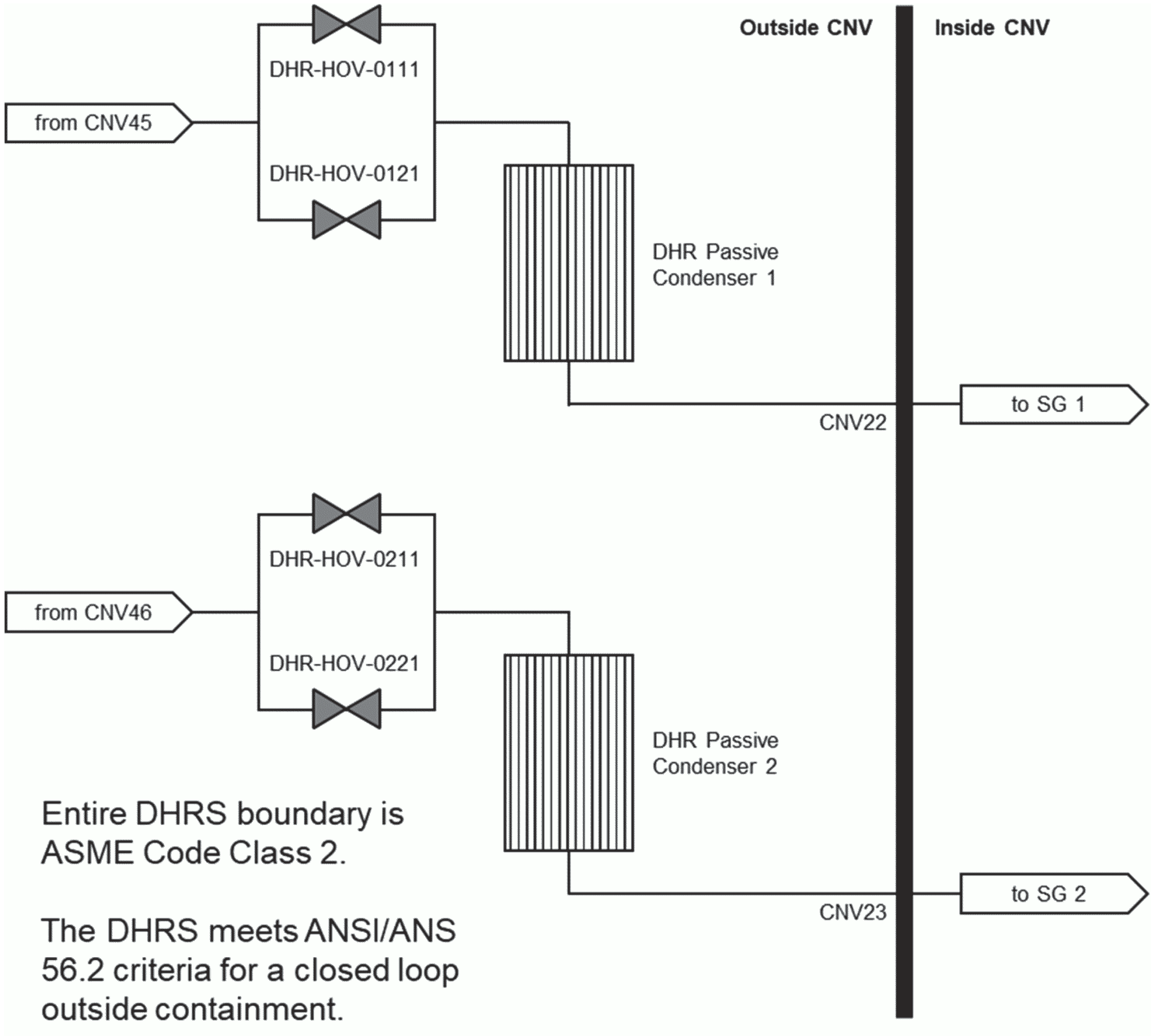
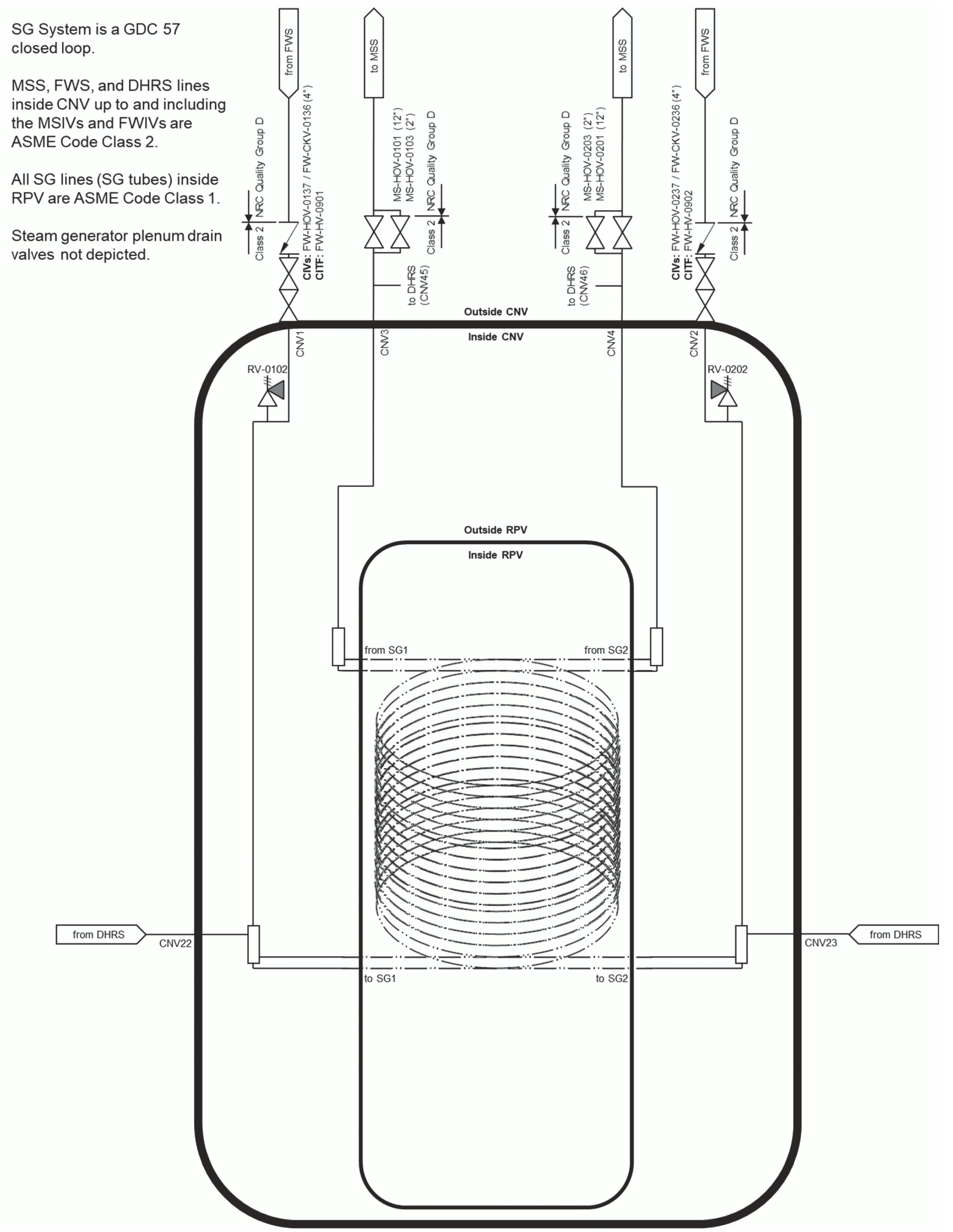


Figure A-3 Steam Generator System



Appendix B Type B Containment Penetrations

Table B-1 Type B Containment Penetrations

Pen.	[Type] Component	Nominal Size (Opening)	Location (Azimuth, Elevation)	Quantity of EPA Sheath Modules	Type B Leakage Test?	Notes
CNV8	[EPA] Instrumentation and controls (I&C) Division 1	NPS 3	348°, 702”	4	Yes	3 blanks
CNV9	[EPA] I&C Division 2	NPS 3	12°, 702”	4	Yes	3 blanks
CNV15	[EPA] PZR Heater Power 1	NPS 12	235°, 702”	9	Yes	3 blanks
CNV16	[EPA] PZR Heater Power 2	NPS 12	125°, 702”	9	Yes	3 blanks
CNV17	[ISA] ICI Channel A	8 inch	CNV Head	N/A	Yes	3 ICI points, 1 blank
CNV18	[ISA] ICI Channel C	8 inch	CNV Head	N/A	Yes	3 ICI points, 1 blank
CNV19	[ISA] ICI Channel B	8 inch	CNV Head	N/A	Yes	3 ICI points, 1 blank
CNV20	[ISA] ICI Channel D	8 inch	CNV Head	N/A	Yes	3 ICI points, 1 blank
CNV24	[Port] CNV Manway Access 1	NPS 18	315°, 719”	N/A	Yes	
CNV25	[Port] CRDM Access	67 inch	CNV Head	N/A	Yes	
CNV26	[Port] CNV Manway Access 2	38 inch	315°, 719”	N/A	Yes	
CNV27	[Port] SG Access 1	38 inch	45°, 576”	N/A	Yes	
CNV28	[Port] SG Access 2	38 inch	315°, 576”	N/A	Yes	
CNV29	[Port] SG Access 3	38 inch	225°, 576”	N/A	Yes	
CNV30	[Port] SG Access 4	38 inch	135°, 576”	N/A	Yes	
CNV31	[Port] PZR Heater Access 1	44 inch	90°, 596”	N/A	Yes	
CNV32	[Port] PZR Heater Access 2	44 inch	270°, 596”	N/A	Yes	
CNV33	[Valve] RVV Trip/Reset A(3)	NPS 3(4)	353°, 585”	N/A	Yes	Notes 3 and 4
CNV34	[Valve] RVV Trip/Reset B(3)	NPS 3(4)	7°, 585”	N/A	Yes	Notes 3 and 4
CNV35	[Valve] RRV Trip/Reset A(3)	NPS 3(4)	353°, 408”	N/A	Yes	Notes 3 and 4
CNV36	[Valve] RRV Trip/Reset B(3)	NPS 3(4)	7°, 408”	N/A	Yes	Notes 3 and 4
CNV37	[EPA] CRDM Power 1	NPS 18	135°, 742”	9	Yes	1 blank
CNV38	[EPA] RPI Group 1	NPS 10	105°, 742”	4	Yes	No blanks
CNV39	[EPA] RPI Group 2	NPS 10	270°, 742”	4	Yes	No blanks
CNV40	[EPA] I&C Separation Group A	NPS 8	348°, 727”	4	Yes	1 blank
CNV41	[EPA] I&C Separation Group B	NPS 8	12°, 727”	4	Yes	No blanks
CNV42	[EPA] I&C Separation Group C	NPS 8	73°, 742”	4	Yes	No blanks
CNV43	[EPA] I&C Separation Group D	NPS 8	287°, 742”	4	Yes	1 blank
CNV44	[EPA] CRDM Power 2	NPS 18	225°, 742”	9	Yes	1 blank
N/A	[N/A] CNV Closure Flange	170 inch	328.3”	N/A	Yes	

Table B-1 Type B Containment Penetrations (Continued)

Pen.	[Type] Component	Nominal Size (Opening)	Location (Azimuth, Elevation)	Quantity of EPA Sheath Modules	Type B Leakage Test?	Notes
CNV5	[CITF] RCCW-HV-0905[5]	NPS 2	CNV Head	N/A	Yes	CITF cover only
CNV6	[CITF] CVC-HV-0906[5]	NPS 2	CNV Head	N/A	Yes	CITF cover only
CNV7	[CITF] CVC-HV-0907[5]	NPS 2	CNV Head	N/A	Yes	CITF cover only
CNV10	[CITF] CE-HV-0910[5]	NPS 4	CNV Head	N/A	Yes	CITF cover only
CNV11	[CITF] CFD-HV-0911[5]	NPS 2	CNV Head	N/A	Yes	CITF cover only
CNV12	[CITF] RCCW-HV-0912[5]	NPS 2	CNV Head	N/A	Yes	CITF cover only
CNV13	[CITF] CVC-HV-0913[5]	NPS 2	CNV Head	N/A	Yes	CITF cover only
CNV14	[CITF] CVC-HV-0914[5]	NPS 2	CNV Head	N/A	Yes	CITF cover only

Notes:

1. Penetration ID number CNV21 is not used.
2. Elevations are measured from global zero (bottom of the CNV support skirt) measured to the top of the respective safe-end or nozzle cover.
3. RVV and RRV trip/reset valves are part of the reactor coolant pressure boundary.
4. CNV33-36 are 3-inch penetrations for ECCS trip and reset valves. Each penetration has two bolted connections (trip and reset valve) that each require a Type B test at the body-to-bonnet joint.
5. Each CITF is common to both CIVs in a given penetration.

Appendix C List of Type C Containment Penetrations

Table C-1 Penetration CNV1

Category	Parameter	Penetration CNV1	
		Valve FW-HOV-0137	Valve SG-RV-0102
Overview	Name	Feedwater Isolation Valve (FWIV) #1	SGS thermal relief valve
	Location	CNV head	FW line inside CNV
	Valve Type	Ball valve	Spring-actuated relief valve
	Operator Type	Hydraulic to open, stored energy device to close	Spring close, line pressure to open
	Fluid	Water	Water
	Nominal Size (Opening)	NPS 4 (inlet), NPS 5 (outlet)	NPS ¾ x 1
	CIV Configuration	GDC 57	Connected to GDC 57 closed loop
	Type C Leakage Test?	No, TS leak test per IST program	No, TS leak test per IST program
Positions	Normal	Open	Closed
	Shutdown	Closed	Closed
	Safety Function	Closed	Open
	Failure	Closed	N/A
Actuation	Primary	Automatic	Self-actuating
	Secondary	Remote manual	N/A
	Power Source	Hydraulic actuator with failsafe stored energy device	Spring
Design	Design Pressure	2200 psia	2200 psia
	Design Temperature	650 degrees F	650 degrees F
	Seismic Category	I	I
	Design Code Valve	Section III, NC	Section III, NC

Table C-2 Penetration CNV2

Category	Parameter	Penetration CNV2	
		Valve FW-HOV-0237	Valve SG-RV-0202
Overview	Name	Feedwater Isolation Valve (FWIV) #2	SGS thermal relief valve
	Location	CNV head	FW line inside CNV
	Valve Type	Ball Valve	Spring-actuated relief valve
	Operator Type	Hydraulic to open, stored energy device to close	Spring close, line pressure to open
	Fluid	Water	Water
	Nominal Size (Opening)	NPS 4 (inlet), NPS 5 (outlet)	NPS ¾ x 1
	CIV Configuration	GDC 57	Connected to GDC 57 closed loop
	Type C Leakage Test?	No, TS leak test per IST program	No, TS leak test per IST program

Table C-2 Penetration CNV2 (Continued)

Category	Parameter	Penetration CNV2	
		Valve FW-HOV-0237	**Valve SG-RV-0202**
Positions	**Normal**	Open	Closed
	Shutdown	Closed	Closed
	Safety Function	Closed	Open
	Failure	Closed	N/A
Actuation	**Primary**	Automatic	Self-actuating
	Secondary	Remote manual	N/A
	Power Source	Hydraulic actuator with failsafe stored energy device	Spring
Design	**Design Pressure**	2200 psia	2200 psia
	Design Temperature	650 degrees F	650 degrees F
	Seismic Category	I	I
	Design Code Valve	Section III, NC	Section III, NC

Table C-3 Penetration CNV3

Category	Parameter	Penetration CNV3	
		Valve MS-HOV-101	Valve MS-HOV-103
Overview	**Name**	Main Steam Isolation Valve (MSIV) #1	Main Steam Isolation Bypass Valve (MSIBV) #1
	Location	CNV head	CNV head, parallel to MSIV
	Valve Type	Ball valve	Ball valve
	Operator Type	Hydraulic to open, stored energy device to close	Hydraulic to open, stored energy device to close
	Fluid	Steam	Steam
	Nominal Size (Opening)	NPS 12	NPS 2
	CIV Configuration	GDC 57	GDC 57
	Type C Leakage Test?	No, TS leak test per IST program	No, TS leak test per IST program
Positions	**Normal**	Open	Open
	Shutdown	Closed	Closed
	Safety Function	Closed	Closed
	Failure	Closed	Closed
Actuation	**Primary**	Automatic	Automatic
	Secondary	Remote manual	Remote manual
	Power Source	Hydraulic actuator with failsafe stored energy device	Hydraulic actuator with failsafe stored energy device
Design	**Design Pressure**	2200 psia	2200 psia
	Design Temperature	650 degrees F	650 degrees F
	Seismic Category	I	I
	Design Code Valve	Section III, NC	Section III, NC

Notes:
(1) DHRS penetration CNV45 is integral to CNV3.

Table C-4 Penetration CNV4

Category	Parameter	Penetration CNV4	
		Valve MS-HOV-0201	Valve MS-HOV-0203
Overview	**Name**	Main Steam Isolation Valve (MSIV) #2	Main Steam Isolation Bypass Valve (MSIBV) #2
	Location	CNV head	CNV head, parallel to MSIV
	Valve Type	Ball valve	Ball valve
	Operator Type	Hydraulic to open, stored energy device to close	Hydraulic to open, stored energy device to close
	Fluid	Steam	Steam
	Nominal Size (Opening)	NPS 12	NPS 2
	CIV Configuration	GDC 57	GDC 57
	Type C Leakage Test?	No, TS leak test per IST program	No, TS leak test per IST program
Positions	**Normal**	Open	Open
	Shutdown	Closed	Closed
	Safety Function	Closed	Closed
	Failure	Closed	Closed
Actuation	**Primary**	Automatic	Automatic
	Secondary	Remote manual	Remote manual
	Power Source	Hydraulic actuator with failsafe stored energy device	Hydraulic actuator with failsafe stored energy device
Design	**Design Pressure**	2200 psia	2200 psia
	Design Temperature	650 degrees F	650 degrees F
	Seismic Category	I	I
	Design Code Valve	Section III, NC	Section III, NC
Notes: (1)DHRS penetration CNV46 is integral to CNV4.			

Table C-5 Penetration CNV5

Category	Parameter	Penetration CNV5	
		Valve RCCW-HOV-0190	**Valve RCCW-HOV-0191**
Overview	**Name**	RCCWS Return Containment Isolation Valve, Inboard	RCCWS Return Containment Isolation Valve, Outboard
	Location	CNV head, CNV side	CNV head, piping side
	Valve Type	Ball valve	Ball valve
	Operator Type	Hydraulic to open, stored energy device to close	Hydraulic to open, stored energy device to close
	Fluid	Water	Water
	Nominal Size (Opening)	NPS 2	NPS 2
	CIV Configuration	GDC 56	GDC 56
	Type C Leakage Test?	Yes	Yes
Positions	**Normal**	Open	Open
	Shutdown	Closed	Closed
	Safety Function	Closed	Closed
	Failure	Closed	Closed
Actuation	**Primary**	Automatic	Automatic
	Secondary	Remote manual	Remote manual
	Power Source	Hydraulic actuator with failsafe stored energy device	Hydraulic actuator with failsafe stored energy device
Design	**Design Pressure**	2850 psia(1)	2850 psia(1)
	Design Temperature	650 degrees F(1)	650 degrees F(1)
	Seismic Category	I	I
	Design Code Valve	Section III, NB(1)	Section III, NB(1)

Notes:

(1) Valve provides a containment boundary and is classified as ASME BPVC Class 2 with a minimum design pressure and temperature requirement equivalent to the CNTS. However, primary system CIVs are designed to ASME BPVC Section III NB with a design pressure and temperature requirement equivalent to the CVCS injection piping (2850 psia, 650 degrees F).

Table C-6 Penetration CNV6

Category	Parameter	Penetration CNV6	
		Valve CVC-HOV-0331	**Valve CVC-HOV-0330**
Overview	**Name**	CVC Injection Containment Isolation Valve, Inboard	CVC Injection Containment Isolation Valve, Outboard
	Location	CNV head, CNV side	CNV head, piping side
	Valve Type	Ball valve	Ball valve
	Operator Type	Hydraulic to open, stored energy device to close	Hydraulic to open, stored energy device to close
	Fluid	Water	Water
	Nominal Size (Opening)	NPS 2	NPS 2
	CIV Configuration	GDC 55	GDC 55
	Type C Leakage Test?	Yes	Yes
Positions	**Normal**	Open	Open
	Shutdown	Closed	Closed
	Safety Function	Closed	Closed
	Failure	Closed	Closed
Actuation	**Primary**	Automatic	Automatic
	Secondary	Remote manual	Remote manual
	Power Source	Hydraulic actuator with failsafe stored energy device	Hydraulic actuator with failsafe stored energy device
Design	**Design Pressure**	2850 psia	2850 psia
	Design Temperature	650 degrees F	650 degrees F
	Seismic Category	I	I
	Design Code Valve	Section III, NB	Section III, NB

Table C-7 Penetration CNV7

Category	Parameter	Penetration CNV7	
		Valve CVC-HOV-0325	**Valve CVC-HOV-0324**
Overview	**Name**	PZR Spray Containment Isolation Valve, Inboard	PZR Spray Containment Isolation Valve, Outboard
	Location	CNV head, CNV side	CNV head, piping side
	Valve Type	Ball Valve	Ball Valve
	Operator Type	Hydraulic to open, stored energy device to close	Hydraulic to open, stored energy device to close
	Fluid	Water	Water
	Nominal Size (Opening)	NPS 2	NPS 2
	CIV Configuration	GDC 55	GDC 55
	Type C Leakage Test?	Yes	Yes
Positions	**Normal**	Open	Open
	Shutdown	Closed	Closed
	Safety Function	Closed	Closed
	Failure	Closed	Closed
Actuation	**Primary**	Automatic	Automatic
	Secondary	Remote manual	Remote manual
	Power Source	Hydraulic actuator with failsafe stored energy device	Hydraulic actuator with failsafe stored energy device
Design	**Design Pressure**	2850 psia	2850 psia
	Design Temperature	650 degrees F	650 degrees F
	Seismic Category	I	I
	Design Code Valve	Section III, NB	Section III, NB

Table C-8 Penetration CNV10

Category	**Parameter**	**Penetration CNV10**	
		Valve CE-HOV-0001	**Valve CE-HOV-0002**
Overview	**Name**	Containment Evacuation Containment Isolation Valve, Inboard	Containment Evacuation Containment Isolation Valve, Outboard
	Location	CNV head, CNV side	CNV head, piping side
	Valve Type	Ball Valve	Ball Valve
	Operator Type	Hydraulic to open, stored energy device to close	Hydraulic to open, stored energy device to close
	Fluid	Steam and air	Steam and air
	Nominal Size (Opening)	NPS 4	NPS 4
	CIV Configuration	GDC 56	GDC
	Type C Leakage Test?	Yes	Yes
Positions	**Normal**	Open	Open
	Shutdown	Closed	Closed
	Safety Function	Closed	Closed
	Failure	Closed	Closed
Actuation	**Primary**	Automatic	Automatic
	Secondary	Remote manual	Remote manual
	Power Source	Hydraulic actuator with failsafe stored energy device	Hydraulic actuator with failsafe stored energy device
Design	**Design Pressure**	1200 psia	1200 psia
	Design Temperature	600 degrees F	600 degrees F
	Seismic Category	I	I
	Design Code Valve	Section III, NC	Section III, NC

Table C-9 Penetration CNV11

Category	Parameter	Penetration CNV11	
		Valve CFD-HOV-0022	**Valve CFD-HOV-0021**
Overview	**Name**	Containment Flooding Containment Isolation Valve, Inboard	Containment Flooding Containment Isolation Valve, Outboard
	Location	CNV head, CNV side	CNV head, piping side
	Valve Type	Ball valve	Ball valve
	Operator Type	Hydraulic to open, stored energy device to close	Hydraulic to open, stored energy device to close
	Fluid	Water	Water
	Nominal Size (Opening)	NPS 2	NPS 2
	CIV Configuration	GDC 56	GDC 56
	Type C Leakage Test?	Yes	Yes
Positions	**Normal**	Closed	Closed
	Shutdown	Closed	Closed
	Safety Function	Closed	Closed
	Failure	Closed	Closed
Actuation	**Primary**	Automatic	Automatic
	Secondary	Remote manual	Remote manual
	Power Source	Hydraulic actuator with failsafe stored energy device	Hydraulic actuator with failsafe stored energy device
Design	**Design Pressure**	2850 psia(1)	2850 psia(1)
	Design Temperature	650 degrees F(1)	650 degrees F(1)
	Seismic Category	I	I
	Design Code Valve	Section III, NB(1)	Section III, NB(1)

Notes:

(1) Valve provides a containment boundary and is classified as ASME BPVC Class 2 with a minimum design pressure and temperature requirement equivalent to the CNTS. However, primary system CIVs are designed to ASME BPVC Section III NB with a design pressure and temperature requirement equivalent to the CVCS injection piping (2850 psia, 650 degrees F).

Table C-10 Penetration CNV12

Category	Parameter	Penetration CNV12	
		Valve RCCW-HOV-0185	**Valve RCCW-HOV-0184**
Overview	**Name**	RCCWS Supply Containment Isolation Valve, Inboard	RCCWS Supply Containment Isolation Valve, Outboard
	Location	CNV head, CNV side	CNV head, piping side
	Valve Type	Ball valve	Ball valve
	Operator Type	Hydraulic to open, stored energy device to close	Hydraulic to open, stored energy device to close
	Fluid	Water	Water
	Nominal Size (Opening)	NPS 2	NPS 2
	CIV Configuration	GDC 56	GDC 56
	Type C Leakage Test?	Yes	Yes
Positions	**Normal**	Open	Open
	Shutdown	Closed	Closed
	Safety Function	Closed	Closed
	Failure	Closed	Closed
Actuation	**Primary**	Automatic	Automatic
	Secondary	Remote manual	Remote manual
	Power Source	Hydraulic actuator with failsafe stored energy device	Hydraulic actuator with failsafe stored energy device
Design	**Design Pressure**	2850 psia(1)	2850 psia(1)
	Design Temperature	650 degrees F(1)	650 degrees F(1)
	Seismic Category	I	I
	Design Code Valve	Section III, NB(1)	Section III, NB(1)

Notes:

(1) Valve provides a containment boundary and is classified as ASME BPVC Class 2 with a minimum design pressure and temperature requirement equivalent to the CNTS. However, primary system CIVs are designed to ASME BPVC Section III NB with a design pressure and temperature requirement equivalent to the CVCS injection piping (2850 psia, 650 degrees F).

Table C-11 Penetration CNV13

Category	Parameter	Penetration CNV13	
		Valve CVC-HOV-0334	Valve CVC-HOV-0335
Overview	**Name**	CVC Discharge Containment Isolation Valve, Inboard	CVC Discharge Containment Isolation Valve, Outboard
	Location	CNV head, CNV side	CNV head, piping side
	Valve Type	Ball valve	Ball valve
	Operator Type	Hydraulic to open, stored energy device to close	Hydraulic to open, stored energy device to close
	Fluid	Water	Water
	Nominal Size (Opening)	NPS 2	NPS 2
	CIV Configuration	GDC 55	GDC 55
	Type C Leakage Test?	Yes	Yes
Positions	**Normal**	Open	Open
	Shutdown	Closed	Closed
	Safety Function	Closed	Closed
	Failure	Closed	Closed
Actuation	**Primary**	Automatic	Automatic
	Secondary	Remote manual	Remote manual
	Power Source	Hydraulic actuator with failsafe stored energy device	Hydraulic actuator with failsafe stored energy device
Design	**Design Pressure**	2850 psia	2850 psia
	Design Temperature	650 degrees F	650 degrees F
	Seismic Category	I	I
	Design Code Valve	Section III, NB	Section III, NB

Table C-12 Penetration CNV14

Category	Parameter	Penetration CNV14	
		Valve CVC-HOV-0401	Valve CVC-HOV-0402
Overview	**Name**	RPV High Point Degasification Containment Isolation Valve, Inboard	RPV High Point Degasification Containment Isolation Valve, Outboard
	Location	CNV head, CNV side	CNV head, piping side
	Valve Type	Ball valve	Ball valve
	Operator Type	Hydraulic to open, stored energy device to close	Hydraulic to open, stored energy device to close
	Fluid	Steam and air	Steam and air
	Nominal Size (Opening)	NPS 2	NPS 2
	CIV Configuration	GDC 55	GDC 55
	Type C Leakage Test?	Yes	Yes
Positions	**Normal**	Closed	Closed
	Shutdown	Closed	Closed
	Safety Function	Closed	Closed
	Failure	Closed	Closed
Actuation	**Primary**	Automatic	Automatic
	Secondary	Remote manual	Remote manual
	Power Source	Hydraulic actuator with failsafe stored energy device	Hydraulic actuator with failsafe stored energy device
Design	**Design Pressure**	2850 psia	2850 psia
	Design Temperature	650 degrees F	650 degrees F
	Seismic Category	I	I
	Design Code Valve	Section III, NB	Section III, NB

Table C-13 Penetrations CNV22 and CNV23

Category	Parameter	Penetration CNV22 (no valves)	Penetration CNV23 (no valves)
Overview	**Name**	Decay Heat Removal System Train 1 Condensate Penetration	Decay Heat Removal System Train 2 Condensate Penetration
	Location	N/A	N/A
	Valve Type	N/A	N/A
	Operator Type	N/A	N/A
	Fluid	Water	Water
	Nominal Size (Opening)	NPS 2	NPS 2
	CIV Configuration	GDC 57(1)	GDC 57(1)
	Type C Leakage Test?	N/A	N/A
Positions	**Normal**	N/A	N/A
	Shutdown	N/A	N/A
	Safety Function	N/A	N/A
	Failure	N/A	N/A
Actuation	**Primary**	N/A	N/A
	Secondary	N/A	N/A
	Power Source	N/A	N/A
Design	**Design Pressure**	2200 psia	2200 psia
	Design Temperature	650 degrees F	650 degrees F
	Seismic Category	I	I
	Design Code Valve	Section III, NC	Section III, NC

Notes:

(1) The DHRS lines have the attributes of both a closed loop inside and outside of containment. An exemption is provided to clarify the system design within GDC 57 criteria. CNV45 and CNV46 (which are integral to CNV3 and CNV4, respectively) are the DHRS lines that penetrate containment. The DHRS becomes a closed system outside containment when the FWIVs and MSIVs shut, creating the DHRS boundary. The test for this system is the leakage test of the FWIVs, MSIVs, and main steam bypass valves in accordance with the IST Program.

Appendix D Type A Testing Challenges

D.1 Overview

The design supports an exemption from the requirements of GDC 52 and 10 CFR 50, Appendix J, Type A tests, which specify the design for and performance of preoperational and periodic ILRT at containment design pressure. However, NuScale reviewed the requirements of GDC 52 and 10 CFR 50, Appendix J, Type A testing to assess the potential of performing ILRT with the NuScale design. The inherent safety feature of a small metal containment in direct contact with the UHS presents unique challenges to performing ILRT with the NuScale design.

At the conclusion of a normal shutdown for refueling, the CNV is filled with water to provide heat transfer during reactor refueling by filling the containment with water up to a level near the reactor PZR baffle plate. The heat transfer across the reactor vessel wall into the containment filled with water and through the containment wall into the UHS water provides cooling for the fuel in the RPV (Figure D-1). The high heat transfer ability of the system coupled with the changing decay heat from the core, as well as the UHS heat transfer coupled to the rest of the NPMs in the UHS pool, creates a highly variable temperature system.

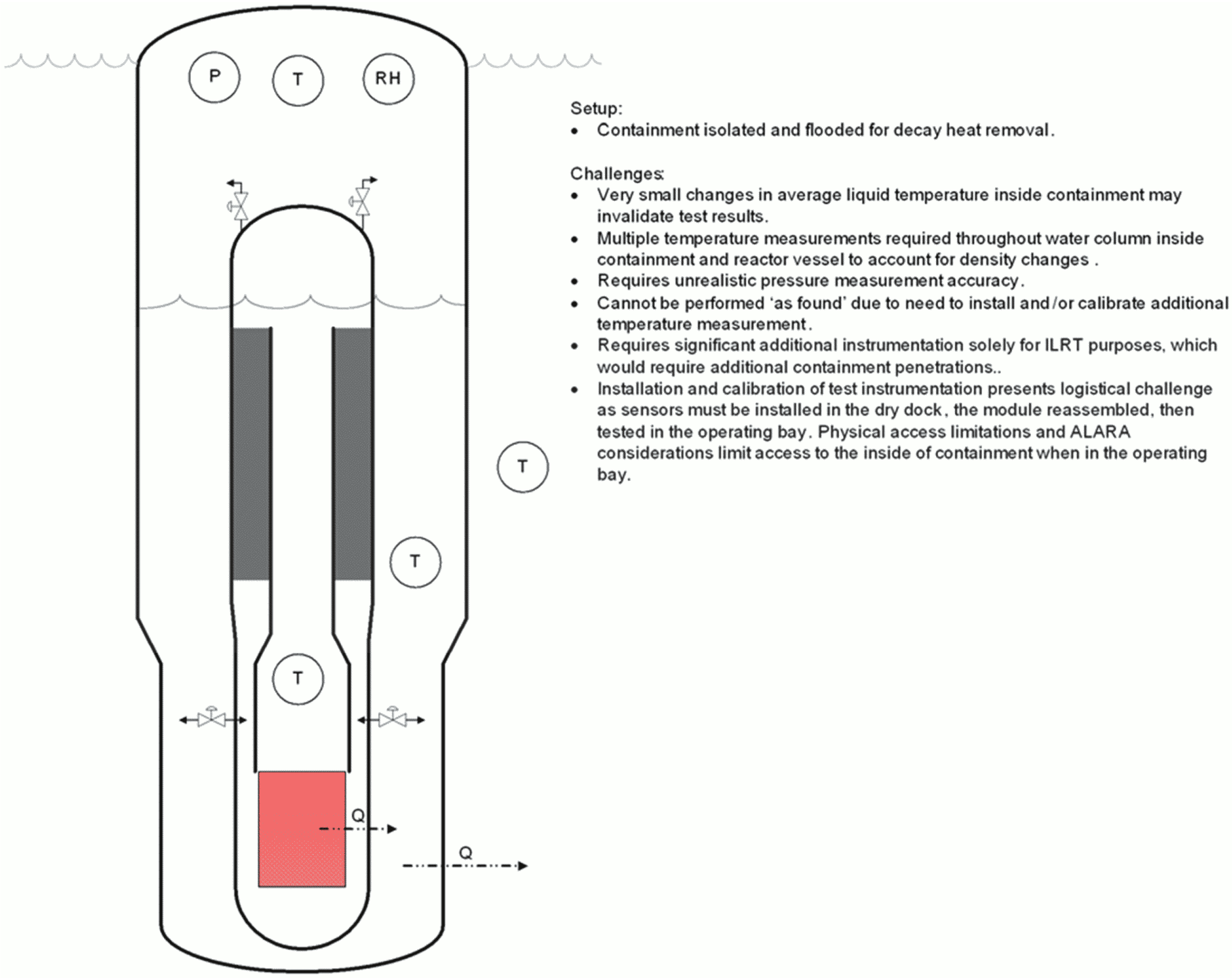
D.2 Temperature

To ensure temperature variations are detected and offset, high-precision sensors both in the top of the PZR and in the containment gas space are provided (Figure D-1). If the RPV water level were lower than the baffle plate, then the additional area under the baffle would need to be individually instrumented.

Sensors needed to monitor temperature changes of the coolant in the RPV and the CNV would need to be more accurate and placed in different locations than the normal plant temperature sensors inside the RPV and the CNV.

While the exact number of additional sensors required for ILRT is unknown, including these sensors permanently in the design would complicate CNV instrumentation and add more signal leads to those already required. Additionally, permanently installed sensors may not be in the optimal locations for a given test. Differing conditions (e.g., water level, air and water temperature) than those for the "design" test requires review and possible reconfiguration of instrument quantities and locations to provide meaningful ILRT test results.

Figure D-1 Reactor Pressure Vessel, Containment Vessel, and Ultimate Heat Sink Temperature Gradients



D.3 Temperature Changes

During ILRT, a coincident undetected temperature change of the gas volume would result in an uncompensated change in pressure of magnitude similar to the allowable pressure change associated with the test leak rate limit.

From the Combined Gas Law, a 0.1 degree F temperature rise increases CNV pressure proportionately:

$$\frac{P_1}{P_2} = \frac{T_1}{T_2}$$

The equation may be rearranged to determine P1:

$$P_1 = P_2\left(\frac{T_1}{T_2}\right)$$

For the NuScale design, P_2 at 1,200 psia (design pressure), T_1 at 100 degree F (559.7 degree R), and T_2 representing a 0.1 °F degree change at 99.9 degree F (559.6 degree R) yields a pressure of:

$$P_1 = (1200\ \text{psia})\left(\frac{559.7\ °\text{R}}{559.6\ °\text{R}}\right) = 1200.214\ \text{psia}$$

The equivalent average temperature change in a large standard plant design with a 60 psia test pressure would have a resulting pressure of:

$$P_1 = (60\ \text{psia})\left(\frac{559.7\ °\text{R}}{559.6\ °\text{R}}\right) = 60.011\ \text{psia}$$

Thus, for the NuScale design, a 0.1 degree F undetected temperature increase in the gas volume would cause a corresponding pressure rise of 0.214 psi. The allowable pressure change to meet the leakage criteria for the NuScale design is approximately 0.06 psia. Therefore, a 0.1 degree F change in average temperature of the gas in the CNV results in more than three times the allowable pressure change associated with the maximum allowable leak rate at 1,200 psia. Changes in average temperature of the fluid inside containment have a similar although less pronounced impact.

The impact of temperature emphasizes the need for highly accurate temperature measurements to obtain a representative average temperature of the CNV atmosphere during ILRT. It also highlights a challenge in obtaining accurate pressure measurement as high-precision gauges available for field installation are typically accurate within 0.01 percent of full scale, or 0.1 psia for a nominal 1,200 psia measurement.

The pressure change of 0.011 psi for a large standard plant is more than 19 times smaller and would not be expected to cause failure of ILRT.

D.4 Instrumentation

Sensors to measure dew point temperature or relative humidity are not currently included in the CNV or RPV instrumentation. Multiple dew point sensors to perform the ILRT are needed in various regions and elevations, such as:

- near the top of the CNV where the insulated head is above the reactor pool surface
- mid-height of the gas volume
- just above the top of the internal CNV water level to ensure these different environments are monitored
- inside the PZR and possibly under the baffle plate to monitor the RPV gas space

In general, ensuring proper placement, accuracy, and calibration is difficult. Furthermore, meeting ANSI/ANS 56.8 criteria is impractical and even doing so appears insufficient for providing accurate test results.

Dew point sensors capable of withstanding 500 to 1,200 psia are uncommon but available. However, the most accurate sensors (e.g., gravimetric, chilled mirror hygrometer) are not suitable for field installation and usually are designed for applications involving flow past the sensor. Without dew point or relative humidity measurements, the effect of evaporative increases in vapor pressure would have to be approximated for the ILRT.

Allowance for uncertainties due to these effects are included in establishing acceptance criteria. These allowances apply even if the plant conditions requiring them do not occur during the test (i.e., even if no temperature or dew point variations actually exist, the data would have to be adjusted on the assumption that the lack of ability to sense such variations was the result of insufficient monitoring capability).

D.5 Leak Rate Criteria

The allowable leak rates of large PWRs are typically above 1 standard cubic foot per minute (scfm) with many being around 5 scfm. The test acceptance criterion for NuScale is approximately 0.226 scfm at 1000 psia. This leak rate value is specified for illustrative purposes; it is based on the latest available references and is updated as the NPM-20 design matures. As a result, ILRT for an NPM must include monitoring accuracy that is 27 times better than commonly used. Because large PWRs sometimes have difficulty meeting their acceptance criteria for stability or accuracy, the challenge for NuScale is even greater.

The acceptance criterion for passing ILRT is $0.75L_a$. Typically, actual leakage is 30 to 50 percent of this acceptance criterion with the remainder reserved for operational margin to allow for some allowable degradation over future operating cycles without requiring immediate repair.

Because the combination of uncertainty values may result in reducing the acceptable leak rate result to less than half of $0.75L_a$, it is more likely that ILRT on NPMs will fail repeatedly on assumed and actual data uncertainty. Subsequent ILRT would need to be re-performed at a considerably higher rate than existing plants.

Normal operational instrumentation provides insufficient accuracy and coverage for ILRT. More accurate sensors are needed for NPM integrated leak rate testing because the leak rate to be detected is approximately one-thirtieth of that for a large PWR. The sensors must also function at 1,000 psia. Such instrumentation must be assumed to be either permanently or temporarily installed inside the module rather than being located outside with sensors inserted through external test points, the inclusion of which would complicate the CNV design.

D.6 Alternate Testing Arrangements

Testing under dry containment conditions was also considered. Such conditions could be achieved through a complete core offload, or by completing the test with the ECCS valves closed and decay heat removal through the normal operating pathways (i.e., the SGs and, to a lesser extent, the CVCS).

A full-core offload would eliminate impacts from core heat of the NPM being tested, but would not eliminate heat transfer from the reactor pool to the containment gas space. Numerous temperature and humidity measurements throughout the containment would still be required, which presents the same challenges already discussed. This approach also defeats the purpose of an as-found or as-left containment leakage test as neither cannot reasonably be truly performed if a full core offload is first required.

Testing with the ECCS valves closed would require reactor pressure to be safely above the containment test pressure because of the passive nature of the ECCS valves, which begin to open when CNV pressure is close to or above RPV pressure. Reactor pressure, therefore, has to be greater than approximately 1,100 psia with a corresponding PZR temperature of approximately 556 degree F. This scenario presents an unacceptable negative safety impact to ECCS operation and introduces additional significant impacts to containment gas space temperature. Testing in these conditions also does not eliminate the challenges already presented.

D.7 Conclusions

NuScale reviewed the requirements of GDC 52 and 10 CFR 50, Appendix J, Type A testing to assess the potential of performing ILRT with the design. The inherent safety features of the NPM limit the ability of the design to conform with 10 CFR 50, Appendix J, Type A testing acceptance criteria and limit the effectiveness of Type A tests for the design. The heat transfer mechanisms and high heat transfer ability of the NPM creates a variable temperature and pressure atmosphere within containment. The prescriptive 10 CFR 50, Appendix J, Type A testing requirements and acceptance criteria are impractical for the design. The temperature and pressure impacts on Type A testing and associated acceptance criteria for the design increases the likelihood of inaccurate results, false test failures, and multiple iterations of testing. Application of Type A testing requirements to the CNV would likely yield inaccurate leakage results because of the limited effectiveness of Type A acceptance criteria when applied to the design.

The evaluation of bolted flange connections provides reasonable assurance that the Type B measured leakage is representative of CNV leakage at design basis conditions. Additionally, the sealing studies performed on the CNV flange as part of technology maturation further justifies that the CNV design does not require Type A testing.

Accessibility constraints within containment and the installation of a large quantity of additional CNV instrumentation (permanent or temporary) for Type A testing would expose occupational radiation workers to unnecessary radiation doses to support testing without a commensurate safety benefit. This unnecessary exposure would be required to support installation, maintenance, and calibration of the equipment necessary to perform

Type A tests. These dose impacts would multiply if additional ILRT is required following any failed tests.

In summary, conformance with GDC 52 and 10 CFR 50, Appendix J, Type A testing requirements is impractical for the design. The Containment Leak Rate Testing Program, supported by the design, provides sufficient leakage integrity assurance for the containment.

Enclosure 3:

Affidavit of Carrie Fosaaen, AF-132201

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 method by which NuScale develops its Engineered Safety.

NuScale has performed significant research and evaluation to develop a basis for this method 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 Engineered Safety. 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