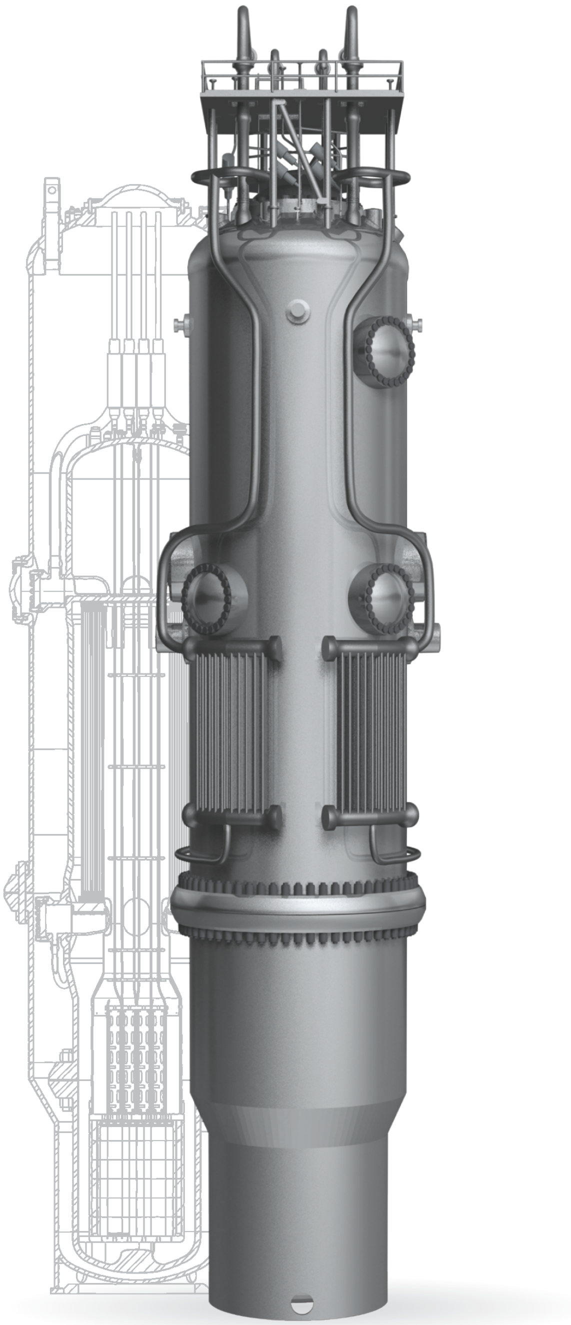

NuScale Standard Plant
Design Certification Application

Chapter Six
Engineered Safety Features

PART 2 - TIER 2

Revision 0
December 2016

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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.

ESF system components are designed, fabricated, erected, and tested to quality standards commensurate with the importance of their intended safety functions (GDC 1). NuScale ESF systems include the containment system (CNTS), emergency core cooling system (ECCS), and decay heat removal system (DHRS). Details of the ECCS and CNTS are described in Sections 6.3 and 6.2, respectively. The DHRS is described in Subsection 5.4.3.

Material selection and fabrication methods ensure the ESF components are compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. In addition, ESF pressure-retaining components are fabricated of materials that 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). Further details of code and regulatory applicability are provided in Chapter 3.

This section also provides information on materials within the CNV that are associated with non-ESF systems. As depicted by Figure 6.2-4, the reactor coolant system (RCS), containment system (CNTS), steam generator system (SGS) and control rod drive system (CRDS) have lines inside the CNV. The materials for these systems are selected and fabricated to be compatible 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

ESF components have a design life of 60 years and are constructed using materials permitted by ASME B&PV Code Section III, Division 1 with proven experience in light water reactor environments. 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 BPV Code, Sections II and III and meet the requirements of NB-2000, NC-2000 or NF-2000. These materials are compatible with the coolant system fluids and their selection is consistent with Parts A, B, and C of ASME BPV Code Section II and Appendix I to ASME BPV Code Section III.

The design, fabrication, and materials of construction of the CNV includes sufficient margin to provide reasonable assurance that the CNV pressure boundary will not undergo brittle fracture, and the probability of a rapidly propagating fracture is minimized, under operating, maintenance, testing, and postulated accident conditions for the 60 year design life.

Materials used in regions of the CNV subjected to neutron irradiation are selected for 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.

The core region of the CNV, which has a 60-year design life peak fluence of less than 1E19 n/cm^2 (>1MeV), is fabricated from austenitic stainless steel. Additional CNV design detail is provided in Section 6.2.1.

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

The use of austenitic stainless steel in the systems that may be exposed to reactor coolant is a common practice in currently operating nuclear power plants to address material corrosion. Operating plant experience shows that there is minimal degradation of these components.

The upper shell, top head and associated supports of the CNV are fabricated from low alloy steel. The lower shell, bottom head and associated supports are fabricated from solution annealed austenitic stainless steel forgings. All inner and outer surfaces of the CNV low alloy steel, including the low alloy steel welds are clad with low carbon grade stainless steel. Unstabilized Type 3XX series austenitic stainless steel materials meet the requirements of RG 1.44.

The passive skirt support, support skirt ring and support skirt are made of SA-965 [or SA-182], Grade F304/304L. The lower shell and bottom head components are made of SA-965, Grade FXM-19.

The cladding applied to the interior surface of the low alloy steel consists of at least one layer of Type 309L austenitic stainless steel.

The cladding applied to the CNV exterior surface of the low alloy steel consists of at least two layers of austenitic stainless steel. The first layer consists of Type 309L stainless steel and the subsequent layers are Type 308L stainless steel. The CNV through-bolt holes are clad with one layer of Type 309L stainless steel.

To avoid cracking of the base material, the stainless steel weld overlay cladding process conforms to the guidelines of RG 1.43 and the underlying low alloy steel satisfies fine grain requirements. The weld cladding processes are qualified in accordance with ASME Code Section III, Subsection NB-4300 and the low-alloy steel forgings that have cladding applied are manufactured to an ASTM grain size of 5 or finer.

Implementation of RG 1.44 guidelines minimizes the potential for stainless steel intergranular stress corrosion cracking. Prior to fabrication, unstabilized austenitic stainless steel of the AISI Type 3XX series is solution treated per the guidance of RG 1.44, which describes acceptable criteria for preventing intergranular corrosion of stainless steel components. Where austenitic stainless steel materials are subjected to sensitizing temperatures for greater than 60 minutes during post weld heat treatment, non-sensitization of the materials is verified by testing in accordance with ASTM A262, Practice A or E. Furnace-sensitized austenitic stainless steel is not used in NuScale ESF components.

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

The guidelines of RG 1.31 or ASME BPV Code, Section III, Subsection NB-2433 for determining the delta ferrite of stainless steel welds do not apply to the filler materials used for depositing cladding because the cladding is deposited as a corrosion resistant layer and does not have a structural function. The delta ferrite content in the austenitic stainless steel weld cladding is controlled between ferrite number 5 and 20.

The chemical composition of low-alloy steel filler metals used for the ESF components meets the requirements of the ASME BPV Code, Section II material specifications. The weld metal filler metals are listed in Table 6.1-1.

Pressure retaining bolting and stud materials (studs, nuts and flat washers) used in ESF systems are fabricated of corrosion resistant alloys; SB-637, Alloy 718 (UNS N07718) for connection of the upper and lower CNV shell flanges, top head and reactor pressure vessel (RPV) to CNV support ledge shell; and, SA-564, Grade 630, heat treated at 1100 °F for CNV appurtenance flanges, and manways and inspection or access ports. These bolting and stud materials are consistent with ASME BPV Code Section III, Subsection NB-2128 and are inspected to the requirements of NB-2580.

Threaded inserts for CNV bolting are fabricated of corrosion resistant alloy SA-479, Type 304/304L.

To minimize the potential for stress corrosion cracking, cold-working of austenitic stainless steel surfaces from abrasive work, such as grinding or wire brushing, is minimized. When abrasive work is required, the tools are controlled to ensure that ferritic carbon steel contaminants are not introduced. The use of cold worked austenitic stainless steel is avoided. If cold work is used, the yield strength as determined by the 0.2% offset method is limited to 90 ksi maximum.

Thermal insulation (metallic or non-metallic) is not used inside the CNV. Insulation used for the exterior upper CNV head above the reactor pool water level is reflective metallic insulation. The use of fiber or mineral insulation is not permitted.

The qualification of welders for making welds in areas where access is limited, and the methods for monitoring and certifying such welds, is in accordance with RG 1.71. Procedure qualification records and welding procedure specifications, including preheat temperatures, for welding low-alloy steel comply with the guidance in RG 1.50. Preheat temperatures also comply with the recommendations of ASME Section III, Appendix D, Article D-1000. Moisture controls for low-hydrogen welding materials comply with the guidelines in Section III of the ASME BPV Code.

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 meets the applicable requirements of 10 CFR 50, Appendix B Criterion XIII.

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 in supporting the ESF function and also the portions of the additional non-ESF functional systems (CFDS, CVCS, MS, FW and RCCWS) that penetrate the CNV boundary and become part of the RCS, CNTS, SGS, and CRDS inside the CNV.

The ECCS function includes three 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 RVVs, RRVs, and their hydraulic lines are intermittently covered by RCS/reactor pool water during periodic shutdown/cooldown/refueling operations and during ECCS operation. The actuator assemblies for the RVVs and RRVs are normally immersed in the reactor pool. The ECCS RVVs, RRVs, their actuators, and the connecting hydraulic lines are designed to be compatible with the RCS chemistry that would be present under LOCA conditions or intermittent exposure to reactor pool water. A more detailed discussion and description of the ECCS is provided in Section 6.3.

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 immersed within the reactor pool. The DHRS piping including the portion that penetrates the CNV boundary is designed to ASME Class 2 criteria. The DHRS piping material internal and external to the CNV is selected to be compatible with the secondary fluid in contact with the DHRS components and with borated water present in the reactor coolant system and the reactor pool. A more detailed discussion and description of the DHRS is provided in Section 5.4.

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

During normal power operations the interior environment of the CNV is maintained dry, at a partial vacuum. The CNV is partially flooded with reactor pool water during cooldown prior to the movement of an NPM for refueling operations.

Emergency core cooling for the NuScale plant is facilitated by the reactor coolant discharged into the CNV. Reactor coolant chemistry is maintained consistent with the guidance found in the EPRI PWR 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 are exposed to primary reactor coolant (internally or externally) are selected to be compatible with reactor coolant chemistry and the NuScale design prohibits the use of materials within the CNV that could significantly alter post-accident coolant chemistry. Additional information on reactor coolant water chemistry is located in Section 5.2.3.

The materials for ESF components that are partially immersed within the reactor pool are selected to be compatible with the reactor pool chemistry conditions maintained in the pool. No significant corrosion is expected based on the purity of the reactor pool water outside of the CNV. 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 9.1.3-2. A corrosion allowance is not included for ESF materials exposed to process fluids or reactor pool chemistry.

Piping, supports and components associated with CFDS and located in the CNV interior but defined as part of the CNTS are designed to be compatible with the reactor coolant chemistry that would be present under operation of ECCS conditions. The CNTS piping, fittings, pipe supports and components are constructed of austenitic stainless steel Type 304 or 304L and are ASME Class 2 components. Non-ESF components in the CNV are listed in Table 6.1-2.

Piping, supports, and components associated with the functional systems that communicate through the CNV boundary and defined RCS or SGS are designed to be compatible with the reactor coolant chemistry that would be present under ECCS operation conditions. The RCS and SGS piping, fittings, pipe supports and components interior to the CNV are constructed of austenitic stainless steel Type 304 or 304L and are designed to ASME Class 1 or Class 2 criteria as appropriate for the associated functional system. Non-ESF components in the CNV are listed in Table 6.1-2. Vendor supplied material supports conform to the fabrication, construction, and testing requirements of ASME BPV Code, Sections II and III and are compatible with the coolant system fluids.

No materials, paint or coatings are used within the NuScale CNV that contribute to corrosion related hydrogen production or alters post-LOCA coolant chemistry to 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 designed free of organic material in the insulation and sheath.

Table 6.1-1: Material Specifications for ESF Components

Component	Material/Grade/Type
CNV Lower Shell Assembly	
CNV Lower (Core Region) Shell CNV Lower Transition Shell CNV Lower Flange CNV Bottom Head	SA-965, FXM-19
Upper/Lower Shell Flange Studs, nuts, and bolts	SB-637, UNS N07718
CNV Upper Shell Assembly	
CNV Upper Flange CNV RPV Support Ledge Shell CNV Steam Plenum Access Shell CNV Upper Shell CNV Support Lugs	SA-508, Grade 3, Class 2
Set Screws Hex Nuts	SA-193, Grade B8,Class 1,
Threaded Inserts	SA-479 Type 304/304L
PZR Heater Access Cover CNV Manway/SG Inspection Port Covers	SA-240, Type 304/304L
Leak Port Tube	TP316L SMLS
Nozzles (CNV22, CNV23, CNV26, CNV27, CNV28, CNV29, CNV30, CNV31, CNV32, CNV33, CNV34, CNV35, CNV36 and CNV40)	SA-508, Grade 3, Class 2
Hex Head Bolts Hex Head, Flat Washers, Nuts for the RPV Support Lug Shell	SB-637 UNS N07718
CNV Head Assembly	
Inspection Port and Manway Cover Nuts Studs, Nuts and Flat Washers for CNV15 thru 20, CNV24, CNV25, CNV38, CNV39	SA-564, Grade 630, Condition H1100
Inspection Ports and Manway Cover Threaded Inserts Vessel Alignment Pin Threaded Insert	SA-479, Type 304/304L
Set Screws	SA-193, Grade B8, Class 1
CNV Top Head CNV Top Head Nozzles (FW, MS, RCCW, CES, CVCS, High Point Vent and CFDS)	SA-508, Grade 3, Class 2
CNV Top Head Cover	SA-182, Grade F304/F304L
CNV Head Manway Blind Flange Top Head Cover Lifting Lugs CRDM Blind Flange (CNV 37)	SA-240, Type 304/304L
RPI Grp 1 and Grp 2 Electrical Penetration Assembly Flanges (CNV38 and CNV39)	SA-240, Type 304/304L
CNV Nozzle Safe Ends	
FW (internal and external), MS (internal and external), RCCW (external), CES, CFDS (external), CVC PZR Spray (external), CVC letdown (external), CVC makeup (external) RPV high point vent (external)	SB-166 [or SB-167] UNS N06690
RCCW supply and return (internal), CVC makeup (Internal), CVC PZR Spray (internal), CVC letdown (internal), CFDS (internal), RPV high point vent (internal), DHR (external), ECCS Trip/Reset (external)	SB-166, UNS N06690
CNV Supports	
Passive Skirt Support, CNV Support Skirt and CNV Support Skirt Ring	SA-182 or SA-965, Grade F304/304L
RPV Ledge Bolts, Nuts, and Flat Washers	SB-637, UNS N07718
RPV Support Ledge and Gussets	SA-533, Type B, Class 2

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

Component	Material/Grade/Type
RXM Platform Mount Assemblies	SA-508, Grade 3, Class 2
Weld Filler	
Low alloy steel weld filler	SFA-5.5, SFA-5.23, SFA-5.28 or SFA-5.29
Austenitic SS weld filler	SFA-5.4: E209, E308, E308L, E309, E309L, E316, E316L SFA-5.9: ER209, ER308, ER308L, ER309, ER309L, ER316, ER316L, EQ308L, EQ309L SFA-5.22: E308, E308L, E309, E309L, E316, E316L
Nickel Based (Ni-Cr-Fe) alloy weld filler	SFA-5.11: ENiCrFe-7 SFA-5.14: ERNiCrFe-7, ERNiCrFe-7A
Stainless Steel Weld Filler for Cladding on low alloy Steel Base Metal	SFA-5.4: E308L, E309L SFA-5.9: ER308L, ER309L, EQ308L, EQ309L SFA-5.22: E308L, E309L
Containment Isolation Valves	Austenitic stainless steel or Ni-Cr-Fe alloy
DHRS	
Steam and Condensate Piping Condenser Tubing and Header Pipes	SA-312, TP304/304L
Actuator Valves	Austenitic stainless steel or Ni-Cr-Fe alloy
Steam Supply Piping Elbows and Tees Condenser Threadolets and Weldolets	SA-182, Grade F304/F304L
Steam Supply Side Pipe Clamp Condenser Manifold Brackets and Retaining Bar	SA-240, Type 304/304L
Condenser Fittings (Elbows, Tees, Caps, Reducers)	SA-403, WP304/304L
Condenser Bolts	SA-193, Grade B8M
Condenser Washers	304 SST
Weld Filler Material	SFA-5.4, E308, E308L, E316, E316L SFA 5.9, ER308, ER308L, ER316, ER316L
ECCS	
Reactor Vent (RVV) and Recirculation (RRV) Valves (Valve Body)	Austenitic stainless steel or Ni-Cr-Fe alloy
RVV and RRV Bolts and Nuts	Austenitic stainless steel or Ni-Cr-Fe alloy
Valve Actuator Manifold Assembly (Trip and Reset)	Austenitic stainless steel or Ni-Cr-Fe alloy
Hydraulic Actuator Tubing	SA-213, Grade TP304/TP304L
Actuator Tubing Reducers and Tees	SA-182, Grade TP304/304L
Weld Filler Material	SFA-5.4: E308, E308L, E309, E309L, E316, E316L SFA-5.9: ER308, ER308L, E309, E309L, ER316, ER316L

Table 6.1-2: Material Specifications for CNV Related non-ESF Components

Component	Material/Grade/Type
Containment Flood and Drain System	
Piping	SA-312, Grade TP304/304L
Piping Supports and Guides (flat plates, supports and rotational restraint)	SA-240, Type 304/304L SA-479, Type 304/304L
Coupler	SA-182, Grade F304/F304L
Reactor Coolant System (CVCS, RSVs)	
Piping	SA-312, Grade TP304/304L
Piping Supports (short, long, tube)	SA-479, Type 304/304L
Pressurizer Support Anchor and Support Plate 180 Degree Piping Supports 90 Degree Piping Supports	SA-240, Type 304/304L
Tee connection to ECCS Reset Valves	SA-182, Grade F304/F304L
Check valves and Excess Flow Check Valves	Stainless Steel
Reactor Safety Valves	Stainless Steel
Control Rod Drive System	
Piping	SA-312, Grade TP304/304L
Supports (brackets and bolts)	SA-479, Type 304/304L
Nozzle fittings, pipe caps, reducers, flange weld neck	SA-182, Grade F304/F304L
Flex hose	TP316L stainless steel
Steam Generator System	
Piping	SA-312, Grade TP304/304L
Piping Supports	SA-479, Type 304/304L
Piping reducers and elbows	SA-182, Grade F304/F304L
Weld Filler Materials (stainless steel to be compatible with base material)	SFA-5.4: E308, E308L, E309, E309L, E316, E316L SFA-5.9: ER308, ER308L, E309, E309L, ER316, ER316L
Instrumentation	
Sensor Line Tubing	SA-312, Grade Type 316L SMLS
Instrument Enclosure (Base and Cover)	SA-182, Grade F304/F304L
Swageloks	SS304
Nuts and Studs	A479-74, Grade XM-19
Flat Washers	SA-240, Type 304
Sensor Cabling (Sheath)	TP304L Stainless Steel

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 NuScale containment system (CNTS) includes the containment vessel (CNV), CNV supports, containment isolation valves (CIVs), passive containment isolation barriers, and containment instruments. (See 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 from a combination of low alloy steel 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 maintained 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 that is designed, analyzed, fabricated, inspected, tested and stamped as an ASME Code Class 1 pressure vessel.

The CNV is designed to provide a barrier against the release of fission products while accommodating the calculated pressures and temperatures resulting from postulated mass and energy release inside containment with margin such that design leakage rates are not exceeded (General Design Criterion (GDC) 50). In concert with the containment isolation valves (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 CNV design specifications also take into consideration the pressures and temperatures associated with combustible gas deflagration. The CNV design includes no internal sub-compartments which eliminates the potential for collection of combustible gases and differential pressures resulting from postulated high-energy pipe breaks within containment.

The CNV is designed to withstand the full spectrum of primary and secondary system mass and energy releases (loss-of-coolant accident (LOCA), valve opening events and non-LOCA) while considering the worst case single active failure and loss of power conditions. Calculated peak containment pressures and temperatures are shown by analysis to remain less than the CNV internal design pressure and temperature for analyzed events.

The limiting primary system pipe break (LOCA) event peak pressure is 921 psia, resulting from a reactor coolant system injection line break. The LOCA peak pressure provides approximately 8 percent margin to the CNV design pressure of 1000 psia. The peak CNV wall temperature for this event is 523 degrees F.

The overall CNV peak pressure is 951 psia resulting from inadvertent opening of an emergency core cooling system valve. The overall peak CNV temperature is 523 degrees F as discussed above. The peak pressure and CNV wall temperature results for secondary system line break events are bounded by the LOCA results. These results demonstrate that the CNV design provides margin to the CNV design pressure of 1000 psia and CNV design temperature of 550 degrees F.

The supporting analyses results are presented in Chapter 5 of the containment response analysis methodology report (Reference 6.2-1). The supporting analyses are discussed by Reference 6.2-1, as well as Section 6.2.1.3 and Section 6.2.1.4.

The CNV is evaluated to demonstrate it can withstand deflagration, incident detonation and deflagration-to-detonation events for 72 hours after event initiation. Structural analysis demonstrates that the CNV is capable of withstanding the resultant combustion loads with margin to stress and strain limits as required by 10 CFR 50.44. Further details are provided in Section 6.2.5.

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 and the reactor module 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.

Instrumentation is provided to monitor 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 containment parameters monitored.

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

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 is designed so that CNV pressure and temperature are rapidly reduced and maintained at acceptably low levels following postulated mass and energy releases, including LOCA, into containment (principal design criterion 38 (Section 3.1.4)). The containment heat removal function is described in Section 6.2.2.

The containment and associated systems are designed to establish an essentially leak tight barrier against an uncontrolled release of radioactivity to the environment, and assures 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 is protected against the dynamic effects that may result from external events (GDC 4). A discussion of these design considerations is provided in Chapter 3.

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 is provided in Chapter 3.

Information discussing offsite and control room dose consequences associated with the containment functional design is provided in Chapter 15 and a shutdown risk assessment, including containment analysis, is provided in Chapter 19.

The evaluation methodology used to determine CNV peak pressure and temperature utilizes the NRELAP5 code with conservative initial and boundary conditions applied to predict the bounding containment pressures and temperatures. The NRELAP5 model used in the containment response analyses for primary system releases (LOCA and valve opening events) CNV response analyses is consistent with the model used in the LOCA evaluation model (Reference 6.2-2). The NRELAP5 model used in the secondary system pipe break CNV response analysis is similar to the NRELAP5 evaluation model described in the Non-LOCA Transient Analysis Methodology (Reference 6.2-4). The significant differences between the primary system release and secondary system break CNV response analysis models and the evaluation models are related to the objective of determining the maximum containment peak pressure and peak temperature scenarios. This is accomplished by conservatively maximizing the mass and energy release, and minimizing the performance of the containment heat removal system. The primary system release and secondary system break CNV response analysis models are detailed in Reference 6.2-1.

The NRELAP5 base plant model used for CNV analysis considers relevant design features of the NPM including fluid volumes (RCS, steam generator system (SGS), and CNV), heat transfer surfaces, valve locations and closure rates and module protection system logic with analytical limits.

The CNV response analysis methodology considers all of the relevant sources of energy for primary system releases including:

- core power equal to rated thermal power plus an allowance for uncertainties
- conservative decay heat model
- maximized RCS and fuel stored energy
- secondary system stored energy sources

These sources of energy are also included in secondary break release analyses, with the addition of energy resulting from feedwater pump runout.

Reference 6.2-1, Section 5.1 presents the results of the NRELAP5 base case analyses of the spectrum of primary mass and energy release scenarios for the NPM, that are determined using the containment response analysis methodology. Section 5.1 also describes sensitivity analyses used to determine the limiting primary release case assumptions for CNV pressure and wall temperature and presents their results. Sections 5.2 and 5.3 of Reference 6.2-1 present the results of NRELAP5 limiting analyses of main steam line and feedwater line break (FWLB) scenarios, respectively. 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), as well as, limiting secondary system break scenarios.

The sources and amounts of energy released to the containment and the post-accident time-dependence of mass and energy releases of postulated primary system events are described in Section 6.2.1.3.

The sources and amounts of energy released to the containment and the post-accident time-dependence of mass and energy releases of postulated secondary system pipe ruptures inside containment (main steam and feedwater line breaks) are described in Section 6.2.1.4.

The capability to remove energy from the CNV (depressurization rate) is determined by the heat transfer rate from the CNV to the reactor pool. In all postulated events, containment pressure is shown to be reduced to less than 50 percent of the peak calculated pressure in less than 24 hours after the postulated accident (principal design criterion 38 (Section 3.1.4)). Specifically, for the limiting peak pressure case, the CNV pressure is reduced to less than 50 percent of its peak value in less than two hours.

6.2.1.1.2 Design Features

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

The CNTS design features passively support:

- enclosure of the RPV, RCS and associated components
- containment of fission product releases from the 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. The CNV is partially immersed in a below-grade reactor pool that provides a passive heat sink and is absent of internal sumps or subcompartments that could entrap water or gases. The CNV and the reactor pool are housed within a Seismic Category 1 Reactor Building (RXB).

Vertical and lateral support for the CNV is provided by the RXB via a support skirt at the bottom of the vessel. Lateral support for the CNV is provided by the reactor compartment walls 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 containment design parameters, operating parameters and information relevant to the CNV. Containment general arrangement drawings are provided in Figure 6.2-2a and Figure 6.2-2b.

During normal operation, the CNV is maintained 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 convective heat transfer from the reactor vessel is minimized during normal reactor operation. Combustible gas control is further described in Reference 6.2-3.

The NPM is designed to be transported 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. Core decay heat removal during this process is discussed in Section 9.2.5, Ultimate Heat Sink.

The postulated primary system release events evaluated for CNV response include RCS injection and discharge line breaks, RPV high point vent degasification line breaks, pressurizer spray line breaks, control rod drive housing failure, reactor safety valve (RSV) opening and inadvertent ECCS valve (reactor vent valve (RVV) and reactor recirculation valve (RRV)) opening. In primary system release events, the CNV design assures that sufficient inventory is retained within the CNV in a configuration that, when given a return path to the RPV (e.g., the ECCS RRVs), maintains the reactor core covered.

Any primary system release event or secondary system release event transfers mass and energy into the CNV and has the potential to increase containment temperature and pressure. Peak calculated pressures and temperatures associated with these postulated events are bounded by the containment design pressure and temperature to ensure that the containment's functional capabilities are maintained under worse case conditions. Analyses of containment response to RCS mass and energy releases are provided in Section 6.2.1.3, and secondary system mass and energy releases are provided in Section 6.2.1.4.

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 or main steam or feedwater 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, the reactor coolant is condensed and collected until coolant level within the RPV has lowered to the ECCS actuation setpoint. Actuation of the safety system opens the RVVs and RRVs, further depressurizing the RPV and increasing the discharge of RPV inventory to the CNV. 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 is returned to the RPV through the ECCS recirculation valves for core cooling. Opening of the RVVs and RRVs establishes the CNV shell as the outer boundary of the coolant circulation flow path. This method of passive coolant circulation and heat removal is further described in Section 6.2.2.

For a secondary system mass and energy release into containment, the released steam or feedwater is captured within the CNV by closure of the CIVs. The collected inventory is condensed and retained with the heat energy transferred to the reactor pool.

The design of the CNV is consistent with the functional requirements of the ECCS and its associated acceptance criteria. Acceptable models for evaluating emergency core cooling during the postulated mass and energy releases are defined in 10 CFR 50 Appendix K.

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 gasses through the containment boundary while preserving the ability of the boundary to prevent or limit the escape of fission products in the event of postulated events. The containment isolation valves are described in Section 6.2.4.

The CNV components and appurtenances are designed to ensure pressure boundary integrity for the life of the plant when considering fatigue, corrosion and wear. The CNV components and penetrations (piping, electrical and instrumentation and controls (I&C)) are designed for and tested to harsh environment conditions (temperature, pressure, radiation, and submergence). Refer to Chapter 3 for additional component design detail.

To ensure leak tightness and functional capability, the CNV is rigorously inspected and its appurtenances are tested to ensure that functional capability is maintained under calculated design basis conditions. The CNV is designed to support the leakage testing requirements of 10 CFR 50 Appendix J, with the exception of Type A testing as discussed in Section 3.1.5 and Section 6.2.6. Access is provided to allow for all inspections, testing, maintenance and removal of components contained within the CNTS as required by the inservice inspection (ISI) and Inservice Testing 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, RSVs, pressurizer heaters, instruments, electrical connections, check valves, welds, supports, and piping inside containment).

The CNTS components and supports are designed such that the ASME BPVC, Section XI inspection requirements for Class 1, Class 2 and Class MC are met including the preservice inspection requirements.

The CNV design information addressing structural loads and loading combinations, including consideration of loading generated as a result of fuel clad metal-water reaction, hydrogen deflagration and hydrodynamic loads, is provided in Section 3.8.2.

6.2.1.1.3 Design Evaluation

Design specific analyses demonstrate the functional capabilities of the NuScale containment design to provide:

- a leak tight barrier that can withstand worse 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 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 withstand hydrogen combustion during an accident while maintaining containment structural integrity and preclude the release of radioactivity.
- 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.

The applicable codes, standards and guides that apply to the containment design are addressed in Section 3.8.2. The CNV design and operating parameters are listed in Table 6.2-1, and general arrangement drawings of the CNV are also provided in Figure 6.2-2a and Figure 6.2-2b. The dynamic effects of mass and energy releases into containment resulting from postulated piping ruptures are considered in the structural design as discussed in Section 3.6.

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 valve opening
- control rod drive housing failure
- bounding anticipated operational occurrence event with ECCS actuation
- secondary system pipe breaks inside containment

The peak containment pressure and temperature resulting from a mass and energy release in containment is dependent on the nature, size, and location of the postulated breach. The NuScale CNV is designed to contain 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 NuScale design, is the code used to predict the bounding CNV pressures and temperatures. The evaluation methodology developed by NuScale addresses applicable regulatory guidance contained in Design Specific Review Standard Section 6.2.1 as discussed by Reference 6.2-1, Section 2.1. The results are shown to be within the design pressure and design temperature of the CNV.

A spectrum of possible locations, sizes and types of mass and energy releases are analyzed in accordance with the NuScale mass and energy release and containment pressure and temperature response methodology. The spectrum of mass and energy release events analyzed for the NuScale design includes the following:

- RCS discharge line break (Case 1)
- RCS injection line break (Case 2)
- RPV high point vent degasification supply line break (Case 3)
- inadvertent opening of a RVV (Case 4)
- inadvertent opening of a RRV (Case 5)
- steam line break
- FWLB

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

The selection process used to determine initial conditions and boundary condition assumptions, reflecting the unique NuScale design, that are used for evaluation of containment response to postulated primary system mass and energy releases into containment are described in Reference 6.2-1, Section 3.5. Secondary system pipe break analysis initial and boundary condition assumptions and their selection process are described in Reference 6.2-1, Section 3.5. 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.

The results of NRELAP5 primary system release event analyses are presented by Reference 6.2-1, Section 5.1. Additionally, Reference 6.2-1, Section 5.1 discusses the insights obtained from the sensitivity studies, used to determine limiting assumptions and single failures, that create a bounding set of assumptions. These assumptions result in the limiting CNV peak temperature and pressure for primary release event Cases 1 though 5. Similarly, Reference 6.2-1, Sections 5.2 and 5.3 present the limiting CNV pressure and temperature results for main steam line and feedwater events, respectively, along with the analysis assumptions that provide these limiting results.

Each mass and energy release event analyzed also includes the consideration of the worst case single active failure as identified by sensitivity cases and a determination of how the availability of normal AC and DC power affects the results, as described in detail by Reference 6.2-1.

The limiting LOCA peak calculated containment pressure and temperature, based on the mass and energy release spectrum analyses, is postulated to occur as the result of a double-ended break of the RCS injection line. (Case 2). Considering the results of sensitivity analyses, the analysis assumes a combined simultaneous loss of normal AC power that occurs at event initiation, conservatively biased ECCS actuation setpoints, and the single failure of one RRV to open. The peak calculated pressure is 921 psia, providing a margin of 79 psia to the CNV design pressure of 1000 psia. The peak calculated temperature is 523 degrees F, providing a margin of 27 degrees F to the CNV design temperature of 550 degrees F.

The overall limiting peak calculated containment pressure, based on the mass and energy release spectrum analyses, is postulated to occur as the result of the spurious opening of a RRV anticipated operational occurrence (Case 5). 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. Additional assumptions accounting for the results of sensitivity analyses, include the loss of normal AC power and highly reliable DC power system (EDSS) postulated to occur at event initiation and an inadvertent actuation block (IAB) release pressure of 1000 psid and minimum primary system flow. The results of single failure sensitivity studies demonstrated no adverse CNV pressure impact for postulated single failures. The peak calculated pressure is 951 psia, providing a 49 psi margin to the CNV design pressure of 1000 psia. Reference 6.2-1, Section 5.4 discusses the analytical and design margin incorporated into the CNV design.

The peak calculated containment pressure resulting from a secondary side mass and energy release is postulated as the result of a double-ended FWLB inside containment. The analysis assumes a loss of normal AC power and DC power that occurs simultaneously with a turbine trip, an IAB release pressure of 1200 psid, with DHRS available, and a failure of the associated FWIV to close. The peak calculated pressure is 442 psia.

The peak calculated containment temperature resulting from a secondary side mass and energy release is postulated as the result of a double-ended steam line break inside containment. The analysis assumes normal AC and DC power available, with decay heat removal system (DHRS) available, and a failure of the associated feedwater isolation valve (FWIV) to close. The peak calculated temperature is 427 degrees F.

The secondary system mass and energy release event results are bounded by the primary system mass and energy release events.

The CNV external design pressure is 60 psia which is based on an internal pressure of 0 psia and an external pressure resulting from 100 feet of pool water static pressure.

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 is discussed in Section 3.11.

The CNV instrumentation provided to monitor and record the required containment parameters and the capability to operate in post-accident environments is discussed in Chapter 7 and Section 3.11.

6.2.1.2 Containment Subcompartments

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

6.2.1.3 Mass and Energy Release Analyses for Primary System Release Events

The NuScale containment receives the primary system mass and energy released following a postulated rupture of piping containing reactor coolant, inadvertent opening of an ECCS or RSV. The CNV response analysis methodology, described in technical report TR-0516-49084 - Containment Response Analysis Methodology Technical Report, Rev 0 (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 identifies and justifies differences in the containment response methodology, in comparison to the LOCA evaluation model.

The containment response analysis methodology is based on the NRELAP5 system thermal-hydraulic code with appropriate initial and boundary conditions. The NRELAP5 code is a NuScale modified version of the RELAP5-3D Version 4.1.3 code. The NRELAP5 code is used for LOCA transient analyses. The NRELAP5 code has been qualified or assessed to separate and integral effects tests as described by Reference 6.2-2 to demonstrate the capability of the code to model LOCAs in the NPM. The LOCA evaluation model qualification activities in Reference 6.2-2 are adequate to demonstrate the capability of the NRELAP code to model containment response to LOCA events. Modifications to the inputs to the code are incorporated in order to predict maximum containment peak pressures and temperatures for the various event scenarios by conservatively maximizing the mass and energy release while minimizing the performance of containment heat removal.

The LOCA analysis break spectum modeled by the NuScale containment response analysis methodology considers the same break spectrum as considered by the LOCA evaluation model (Reference 6.2-2). The maximum break opening area is modeled. The inadvertent ECCS valve opening event analysis uses the largest valve area. Break locations are chosen such that mass and energy releases to containment are maximized. The dominant consideration is the timing of the second mass and energy release that occurs when the ECCS valves open. Sensitivity studies are performed on the timing of the ECCS valve opening, and this is effectively a break size sensitivity combined with the maximum initial mass and energy release for each break location. This approach assures that the limiting cases have been identified for the peak CNV pressure and peak CNV temperature.

Critical flow is evaluated using the Henry-Fauske and Moody models for subcooled and two-phase flow conditions as discussed by Reference 6.2-1. The maximum valve area and Cv values (RVV and RRV) are used to determine ECCS valve flows. Smaller values would result in less mass and energy release to containment; therefore, the maximum values are utilized.

The containment response analysis methodology uses the heat transfer correlation package in the NRELAP5 computer code. The LOCA evaluation model report demonstrates these correlations are applicable to the NPM design (Reference 6.2-2). The local fluid conditions and the local heat structure surface temperatures determine the heat transfer mode. Forced convection, natural convection, condensation, conduction, and nucleate boiling are included in the code and are selected if the local conditions are appropriate. Initial and boundary conditions are selected to maximize containment pressure and temperature response. Further details are provided in Reference 6.2-1.

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:

- radiative heating of CNV to maximize the initial inside diameter temperature and thereby minimize the initial condensation rate
- high initial CNV pressure to maximize the non-condensable gas concentration
- 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 low reactor pool level
- conservative high reactor pool temperature

Reference 6.2-1, Table 3-2 shows the heat transfer correlations and models for all of the processes that could impact the CNV peak pressure and temperature response.

The CNV modeling in the NRELAP5 LOCA containment response analysis model is similar to the LOCA Evaluation Model (Reference 6.2-2), with certain changes made to maximize the mass and energy release and consequential containment pressure and temperature, as detailed by Reference 6.2-1. The CNV and reactor pool models for secondary system pipe break containment response analysis methodology are the same as the modeling for primary release events.

Simplified diagrams of the nodalization used in the containment response analysis methodology are provided by Reference 6.2-1.

The primary system mass and energy release events analyzed include the following:

- pipe breaks
  - RCS discharge line
  - RCS injection line
  - RPV high point vent degasification line
- inadvertent RVV opening
- inadvertent RRV opening

The NuScale LOCA evaluation model divided the NPM LOCA scenarios into two phases for phenomena identification:

- LOCA blowdown phase (Phase 1a)
- ECCS recirculation (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 any of the following conditions:

- low RPV level
- high CNV level
- loss of normal AC power and the EDSS

The RVVs and RRVs open under the following conditions:

- If the pressure differential across the valves is greater than the IAB threshold when the ECCS signal actuates, then the valves stay closed until the pressure differential decreases to below the IAB release pressure.
- If the pressure differential across the valves has decreased to below the IAB threshold pressure when the ECCS signal actuates, then the valves open at that time.

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. This terminates the reactor vessel level decrease prior to 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, the period of peak containment pressure and temperature have been completed, and a gradual depressurization and cooling phase begins.

Sensitivity cases are performed to determine the effect of loss of power (AC and/or DC) scenarios, as well as postulated single failures, on the primary system mass and energy release scenarios considered by the NuScale containment response analysis methodology. The insights obtained from the results of the sensitivity cases, discussed in Reference 6.2-1, are used to determine the limiting cases for CNV pressure and temperature.

6.2.1.3.1 Mass and Energy Release Data - Primary System Release Events

The maximum containment peak pressure and peak temperature scenarios are determined by conservatively modeling the mass and energy release and minimizing the performance of the containment heat removal function of containment.

Reference 6.2-1, Section 5.1 provides the results of the NRELAP5 limiting analyses of the spectrum of the five primary system mass and energy release scenarios for the NPM. The limiting primary system release event (Case 5) CNV pressure results are depicted by figures contained in Reference 6.2-1, Section 5.1. 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 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, since 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 (163.2 MW).
- 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, main steam and feedwater piping inside containment) based on conservative initial conditions of steam pressure and feedwater 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

During normal power operation (normal AC and DC power available), the primary system release scenarios start with the blowdown of the primary inventory through the pipe break or valve opening into the CNV. The reactor trips on high CNV pressure, and that causes a turbine trip along with main steam isolation and feedwater isolation. The primary system depressurizes as the CNV pressurizes, and the coolant inventory accumulates in the CNV. Steam released into the CNV condenses on the CNV inner surface that is cooled by conduction and convection to the reactor pool. When the primary system inventory reaches the low level setpoint, or the CNV level reaches the high level setpoint, the ECCS actuates. The ECCS valves subsequently open as described in Section 6.2.1.3.

The NRELAP5 primary release event model is developed from engineering information, drawings and associated reference documents to develop a thermal-hydraulic simulation model that calculates the mass and energy released from the RCS during blowdown.

The containment response analysis methodology 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 based on the NRELAP5 computer code, and the modeling approach is very similar to the NuScale LOCA Evaluation Model that complies with the applicable portions of 10 CFR 50 Appendix K. Specific changes to the LOCA Evaluation Model required to model primary system mass release events are described by Reference 6.2-1. A discharge coefficient of 1.0 is applied to the applicable critical flow correlation. Reference 6.2-2 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 containment response analysis methodology uses the heat transfer correlation package in the NRELAP5 computer code. The LOCA Evaluation Model report demonstrates these correlations are applicable to the NPM design (Reference 6.2-2).

6.2.1.3.4 Description of the Emergency Core Cooling System Recirculation Model

The containment response analysis methodology 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 recirculation phase is the same as previously described for the blowdown phase.

Operation of the ECCS is further discussed in Section 6.3.

6.2.1.3.5 Description of the Long-Term Cooling Model

The containment response analyses demonstrate that the CNV pressure and temperature are rapidly reduced and maintained at acceptably low levels following postulated mass and energy releases, including LOCA, into containment (principal design criterion 38 (Section 3.1.4)). The previously described methodology and model are utilized for this purpose. This demonstrates adequate long term containment heat removal.

6.2.1.3.6 Single-Failure Analysis

Potential single failures are considered in the containment response analysis methodology and the results show in each case that the necessary safety function can be met when the worst case single failure is included. Reference 6.2-1, Section 5.1 discusses the insights obtained from the sensitivity studies used to 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

Due to the absence of significant post-LOCA cladding heat-up, the NuScale methodology does not model additional energy resulting from cladding metal-water reaction.

6.2.1.3.8 Energy Inventories - Loss-of-Coolant Accident

The integrated mass and energy release rates for the primary system release pressure and temperature limiting events are provided in Reference 6.2-1.

6.2.1.3.9 Additional Information Required for Confirmatory Analysis

Information supporting confirmatory analysis is contained in Table 6.2-1 and in the containment response analysis methodology report (Reference 6.2-1).

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

The NuScale containment receives the secondary system mass and energy released following a postulated main steam line or feedwater line break. The containment response analysis methodology, described in technical report TR-0516-49084 -

Containment Response Methodology Technical Report, Rev 0 (Reference 6.2-1) is an extension of NuScale non-LOCA methodology (Reference 6.2-4) developed in accordance with the guidance of Regulatory Guide (RG) 1.203, "Transient and Accident Analysis Methods" December 2005. Reference 6.2-1 identifies and justifies differences in the containment response methodology, in comparison to the non-LOCA methodology.

The containment response analysis methodology for secondary system events is based on the NRELAP5 system thermal-hydraulic code with appropriate initial and bounding conditions. The NRELAP5 code is a NuScale modified version of the RELAP5-3D Version 4.1.3 code. The NRELAP5 code is used for non-LOCA transient analyses. The NRELAP5 non-LOCA model described in the non-LOCA transient analysis methodology report (Reference 6.2-4) is used to develop the MSLB and FWLB analyses in the containment response analysis methodology. Modifications to the inputs to the code are incorporated in order to predict maximum containment peak pressures and temperatures for various event scenarios accomplished by conservatively maximizing the mass and energy release while minimizing the performance of containment heat removal.

The limiting MSLB event and FWLB event are double-ended ruptures of the largest main steam line and feedwater line pipes.

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 feedwater supply and main steam lines are isolated.

Conservative modeling of secondary system mass and energy release scenarios ensures a bounding analysis. All breaks are considered with a maximum break size at each location. Critical flow is evaluated using the Moody and Henry-Fauske models for subcooled and two-phase flow conditions as discussed in Reference 6.2-1.

The containment response analysis methodology uses the heat transfer correlation package in the NRELAP5 computer code for secondary system pipe break analysis. The Non-LOCA transient analysis methodology report demonstrates these correlations are applicable to the NPM design (Reference 6.2-4). The local fluid conditions and the local heat structure surface temperatures determine the heat transfer mode. Nucleate boiling heat transfer is included in the code and is selected 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 are selected to maximize containment pressure and temperature response.

A description of each postulated secondary system mass and energy release event is provided in Reference 6.2-1. Results of the limiting analyses are provided in Reference 6.2-1 and for FWLBs. The secondary system mass and energy analyses are fully bounded by the primary system limiting events.

6.2.1.4.1 Mass and Energy Release Data - Secondary System

Similar to primary system mass and energy release scenarios, the maximum containment peak pressure and peak temperature scenarios for secondary system releases into containment are determined by conservatively modeling the mass and energy release and minimizing the performance of the heat removal function of containment.

6.2.1.4.2 Single-Failure Analysis - Secondary System

Potential single failures are considered in the containment response analysis methodology. Due to the simplicity of the NPM design, there are few candidate single failures for the secondary system mass and energy release scenarios. Failure of ECCS valves to open would obviously reduce the mass and energy release and are not analyzed. Failures of main steam isolation valves (MSIVs) or FWIVs to close are analyzed as sensitivity studies.

6.2.1.4.3 Initial and Boundary Conditions - Secondary System

Initial conditions for secondary system line break containment response analyses are selected to ensure a conservative CNV peak pressure and peak temperature result. The process of selecting the initial conditions is consistent with applicable DSRS guidance. The selection process ensures that energy sources are maximized and energy sinks are minimized. Initial conditions associated with primary side parameters for MSLB analyses are similar to those described for the primary mass and energy release events, with exceptions noted by Reference 6.2-1. In addition to the primary system initial conditions, secondary system initial conditions for MSLB analyses are listed in Reference 6.2-1. Initial conditions associated with primary side parameters for FWLB analyses are similar to those described for the primary mass and energy release events, with one exception noted by Reference 6.2-1. The FWLB analyses use the same secondary system initial conditions as the steam line break (SLB) analyses. Boundary conditions for secondary system line break containment response analyses are selected to ensure a conservative CNV peak pressure and peak temperature result. The process of selecting the boundary conditions is consistent with applicable DSRS guidance. The selection process ensures that energy sources are maximized, and energy sinks are minimized. Boundary conditions assumed by MSLB analyses are the same as those used in primary release event analyses except for those listed by Reference 6.2-1. Boundary condition assumptions for FWLB analyses are the same as those used by MSLB analyses, with one exception discussed by Reference 6.2-1.

6.2.1.4.4 Description of Blowdown Model - Secondary System

The MSLB is modeled as a double-ended break of a main steam line inside the CNV that depressurizes the secondary system and pressurizes the CNV. Cross connected main steam lines downstream of the main steam isolation results in both SGs discharging to containment until the steam lines isolate. A high containment pressure signal results in closure of the main steam and feedwater isolation valves, reactor trip and actuation of DHRS. Actuation of DHRS establishes long-term decay heat removal using the unaffected SG and the DHRS. After the initiation of the break, there are two potential limiting events depending on the evolution of the scenario with continued AC power, or following a loss of normal AC and DC power. Analysis of the two above scenarios has determined that the case with continued

AC power results in the peak CNV pressure and peak CNV temperature results, as discussed by Reference 6.2-1.

The FWLB is modeled as a double-ended break of the largest feedwater pipe inside the 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 main steam and feedwater isolation valves, reactor trip and actuation of the DHRS. Actuation of DHRS establishes long-term decay heat removal using the unaffected SG and the DHRS. A single failure of the FWIV to close on the affected SG is mitigated by closure of the feedwater regulating valve. The limiting case, described by Reference 6.2-1, also assumes a loss of normal AC and DC power at time of turbine trip, and that results in ECCS actuation and a loss of AC power to the pressurizer heaters. With the DHRS actuation, the primary system begins a gradual cooldown and depressurization. The maximum FWLB 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 are the same as 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 previously discussed.

6.2.1.4.6 Additional Information Required for Confirmatory Analyses - Secondary System

Information supporting confirmatory analysis is contained in Table 6.2-1 and in the containment response analysis methodology report (Reference 6.2-1).

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

For conventional pressurized water reactor designs, the ECCS system supplies water to the reactor vessel to reflood and cool the reactor core. The core reflooding rate for these plants depends directly on containment pressure (i.e., the core flooding rate increases with increasing containment pressure). Accordingly, a minimum containment pressure analysis for ECCS performance capability is applicable for these plants.

For the NuScale facility design, 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 coolant that has condensed and collected within the CNV volume being returned to the RPV. The driving force for the flow back to the RPV is provided by the static head of coolant in the CNV that collects above the return to the vessel (the ECCS RRVs). In the event of large coolant leaks (e.g., LOCAs or valve opening events) the CNV provides for the retention of adequate reactor coolant inventory to prevent core uncovery or loss of core cooling. In this passive coolant system arrangement, CNV internal pressure is irrelevant to ECCS performance.

6.2.1.6 Testing and Inspection

The Inservice Inspection and Inservice Testing Program identifies 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 provided 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 NuScale CNV meets ASME Code Section III and Section XI requirements for a steel CNV. The preservice testing and inspection of the NuScale CNV pressure retaining and integrally attached materials is performed in accordance with written non-destructive examination procedures as required by the ASME BPVC. The examinations, using the methods of ASME BPVC Section V except as modified by Section NB or NF, meets the applicable requirements of ASME NB-5000 and NF-5000.

During fabrication, magnetic particle or liquid penetrant examination of surfaces that are clad is performed in accordance with NB-2545 or NB-2546 prior to application of the cladding.

Preservice examinations of ASME Code Class 1 containment pressure boundary items are conducted in accordance with NB-5280 and IWB-2200 using the examination methods of ASME BPVC Section V except as modified by NB-5111. Preservice inspections include 100 percent of the pressure boundary welds.

The CNV is hydrostatically tested in accordance with ASME BPVC Section III, Subsection NB-6000 at a minimum test pressure (highest point) of 1250 psig and maximum test pressure (lowest point) of 1325 psig.

Based on the high pressure and the safety function of a NuScale CNV, enhanced inspection requirements are provided for the CNV in excess of the Class MC requirements of ASME BPVC, Section XI, Subsection IWE. The CNV augmented inspections are based on Class 1 requirements of ASME BPVC, Section XI. Specifically, rather than just a visual examination as required for an ASME Class MC containment, many of the NuScale CNV pressure boundary welds are required to have a volumetric or surface examination performed per ASME BPVC, Section XI, Subsection IWB.

The CNV inspection elements are provided in Table 6.2-3. An inspection element is a combination of a component and the inspection requirements.

A description of the ISI requirements for Class 2 and 3 components is provided in Section 6.6.

6.2.1.7 Instrumentation Requirements

Instrumentation is provided to monitor the conditions inside the containment and to actuate the appropriate engineered safety features, should those conditions exceed predetermined levels. Instruments are provided to measure containment pressure, temperature, and water level. Instrumentation to monitor RCS leakage into containment and compliance with RG 1.45 is described in subsection 5.2.5.

Containment pressure instrumentation is provided for continuous control room indication to monitor containment pressure boundary integrity, RCS pressure boundary integrity and ECCS performance and to support the actuation of critical safety functions (reactor trip, decay heat removal actuation, CVCS isolation and containment isolation functions).

Containment pressure is measured and monitored by four narrow range, Class 1E, instruments and two wide range non-Class 1E instruments. The narrow range sensors (transducer/transmitter type) are located 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 located 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 is provided for continuous control room indication to monitor containment pressure boundary integrity, RCS pressure boundary integrity and ECCS performance and to support the actuation of critical safety functions.

Containment water level is measured and monitored by four Class 1E instruments. The sensors (digital type) are located at the reactor pressure boundary interface. There are four independent channels of CNV water level instrumentation.

Containment air temperature instrumentation is provided for continuous control room indication to monitor the environment in containment.

Containment air temperature is measured and monitored by two non-Class 1E, instruments. The sensors (resistance temperature detector type) are located inside the CNV near the top of the CNV head. There are two independent channels of CNV air temperature instrumentation.

Additional containment instrumentation design detail addressing power supplies, actuation logic and initiation signals for the engineering safety feature functions is provided in Section 7.

6.2.2 Containment Heat Removal

Containment heat removal for accident conditions is based on CNV material selection and the physical configuration of the NuScale Power Plant design. The steel CNV is partially immersed in the reactor pool and heat is transferred 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 maintained 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 by radiation. The radiated heat energy is conducted through the CNV wall to the exterior surface where it is released via convection into the reactor pool water. For normal operations, the water level in the pool submerges the CNV in 69 ft of water with the pool water level just below the upper head of the CNV. Most of the CNV is in contact with the pool water and used for containment heat removal during operations. Section 9.1.3 describes the removal of the heat from the pool water by the active reactor pool cooling system.

Containment vacuum is maintained during normal operation by the containment evacuation system (CES) described in Section 9.3.6. Maintaining the containment under a vacuum during normal operation serves to minimize heat transfer from the RPV to the CNV and the associated loss of efficiency.

During postulated primary and secondary release events into containment, the released inventory is collected and accumulates within the CNV. 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 main steam and feedwater lines and single failures. The inventory released into containment flashes, condenses, and accumulates within the CNV.

The steel wall of the NuScale 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. The UHS is a system common to the NPMs. Conformance with GDC 5 is addressed in Sections 3.1.1 and 9.2.5.

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

6.2.2.1 Design Basis

The NuScale CNTS design has the CNV partially immersed and is therefore capable of removing the thermal energy from the containment for accident conditions. Following a DBE that results in containment pressurization, containment pressure is rapidly reduced and maintained below the design value without operator action. For the postulated DBEs described in Chapter 15, containment pressure is reduced to less than 50 percent of the peak calculated pressure in less than 24 hours after the postulated accident and meets the requirement in principal design criterion 38 (Section 3.1.4) for reducing pressure rapidly after an accident. Specifically, for the limiting peak pressure case, the CNV pressure is reduced to less than 50 percent of its peak value in less than 2 hours.

The NuScale CNTS and UHS are also designed to provide long-term cooling that removes sensible and decay heat from the reactor core and containment atmosphere following DBEs. With the exception of the backup secondary islation 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 1 percent to 2 percent of CNV design pressure during long-term cooling, which limits the cumulative leakage of coolant inventory from the pressure boundary of the CNV.

The NPMs and the UHS are protected by the RXB, which is designed to withstand the effects of natural phenomena such as earthquakes, winds, tornadoes, or floods. The CNTS structures, systems, and components are designed and constructed to Seismic Category I requirements.

The components of the CNTS that perform containment heat removal are designed to allow inspection in accordance with the intervals specified in the ASME Code, Section XI (Reference 6.2-7). 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. Testing of the passive containment heat removal function for LOCA conditions was performed as described in Reference 6.2-2 and Reference 6.2-5. The NuScale design supports an exemption from GDC 40. Section 3.1.4 addresses the NuScale exemption from GDC 40.

6.2.2.2 System Design

Passive containment cooling is accomplished via direct communication between two safety-related systems, the CNTS and the UHS. The passive containment cooling function makes use of 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 is heated 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 is sufficient to prevent the containment from exceeding its design pressure and temperature following the postulated design basis accidents (DBAs) identified in Chapter 15.

The passive containment cooling function makes use of the steel CNV. Section 3.8.2 describes the design features of the CNV including the dimensions, materials, penetrations, and attachments.

The generation of post-accident debris from coatings, insulation, latent debris, and chemical effects is considered along with debris transport and downstream effects in the CNTS design. Components located within the CNV do not contribute to the post-accident debris load and are not permitted to use fibrous or organic insulation materials except when encapsulated in a manner that prevents debris generation (i.e. conduit or sheathed). The materials selected for the components within containment take into consideration the anticipated water chemistry conditions. Protective coatings are not allowed within the CNV and susceptible materials (cables, etc.) are designed to withstand anticipated accident conditions within containment. Mineral insulated, metallic sheathed (MI) cable is used and does not contain fibrous insulation so it does not contribute to debris loading. Cable that is not MI cable is installed inside conduit. Thermal insulation is not used inside containment. Section 6.3.2.5 describes conformance with RG 1.82 and the approach used to address Generic Safety Issue 191 (GSI-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 located below grade 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 are also designed to 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 large inventory of water in the UHS pools during power operations means that sufficient containment heat removal capability is present at the start of an event. Table 9.2.5-1 provides the normal operating level range for the water in the UHS pools and also shows that lower pool levels would be sufficient to provide an adequate supply of cooling after an accident. As shown in the table, the pool water level can be several feet lower than the normal level and provide 30 days of water coverage over the DHRS condensers and support ECCS operation. These results indicate the conservatism in the initial inventory of water available in the UHS pools to support containment heat removal.

The LTC methodology report (Reference 6.2-5) 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 analytical model includes modifications made to the base NRELAP5 model described by the LOCA evaluation model (Reference 6.2-2) to support LTC calculations. Assessments comparing the NRELAP5 model predictions to experimental results for significant LTC parameters demonstrate excellent fidelity. 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 and that the cladding temperature remains acceptably low. These results demonstrate sufficient LTC capability for 72 hours, without operator action, following a design basis event with ECCS operation with or without power available. For further detail, see Reference 6.2-5.

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.

The CNTS design fully complies with the regulatory positions of RG 1.82 and addresses GSI-191. The generation of post-accident debris, debris transport, and downstream effects are considered in the design as described in Section 6.3.2.5.

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

6.2.2.4 Testing and Inspection

Information is provided in Section 6.2.1.6 on the test programs for the CNV for preservice and initial plant testing (preoperational and startup). After startup, operability and performance of the passive containment heat removal function is ensured through inspection in accordance with GDC 39. Periodic inservice inspection of the containment heat removal surfaces is performed to assess for 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.

Testing and inspection for the UHS are addressed in Section 9.2.5.5.

6.2.2.5 Instrumentation Requirements

Instrumentation for the CNV is addressed in Section 6.2.1.7 and for the UHS in Section 9.2.5.6.

6.2.3 Secondary Containment Functional Design

The NuScale Power Plant is not a dual containment design (i.e., the design does not require a secondary containment function) facility. The design of the NuScale CNV is described in Section 3.8.2, Section 6.1 and Section 6.2.1.

6.2.4 Containment Isolation System

The RXB has up to 12 NPMs, and each NPM has a containment system (CNTS) with a containment boundary designed to prevent or limit the release of radioactive materials under postulated accident conditions. The containment boundary is formed by the CNV and by the containment isolation valves (CIVs) and the passive containment isolation barriers that are used to prevent releases through the penetrations in the CNV. Although a "containment isolation system" is not defined for a NuScale Power Plant, the CIVs for an NPM are similar to such a system for existing light water reactor plants.

Figure 3.8.2-1 provides an overview of the CNTS showing the CNV components surrounding an RPV. Figure 6.2-1 has a cutaway view of an NPM. Figure 6.2-2a and Figure 6.2-2b show the CNV top head and side penetrations. Figure 6.2-3a and Figure 6.2-3b show and list the penetrations by nozzle number. Figure 3.8.2-4 identifies the process flow for each of the CNV top head mechanical penetrations. Figure 3.8.2-5 identifies electrical, I&C, and access penetrations in the CNV top head.

Table 6.2-4 lists the 40 penetration openings in the CNV. As shown in the table, there are the following types of penetrations:

- mechanical penetrations for process fluids or gases - 20 total with 12 on the top head for process flows and 8 on the side with 6 penetrations for ECCS valve actuator assemblies and 2 penetrations for DHRS process flows
- electrical penetrations for power supply - 3 total on the top head
- I&C penetrations for signals - 8 total on the top head
- access and inspection port penetrations - 9 total with 2 on the top head and 7 on the side

Note that in Table 6.2-4 the CNV penetration number 21 is not used and that the opening of the CNV at the main flange is not considered a penetration opening.

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

- Eight mechanical penetrations through the CNV top head have two hydraulically-operated 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 connected to the RCPB or open to the atmosphere inside of a CNV.
- Four mechanical penetrations through the CNV top head have a single hydraulically-operated 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 not connected 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, a main steam isolation bypass valve, and the two parallel DHRS branch lines.
- Six 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 not connected 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 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 are designed to 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.

The CIVs are in the CNTS and are not part of the various process systems that penetrate the CNV upper head. The valves are however labeled with the interfacing system acronym for easier identification, i.e., CES, containment flooding and drain system (CFDS), CVCS, reactor component cooling water system, 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 6.2-4 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 6.2-4 also shows the systems to which the process lines connect inside of containment, i.e., CFDS, RCS, control rod drive system (CRDS), 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 a CIVs on the line outside of the CNV. The containment isolation design features for the penetrations used for the DHRS piping are described further below.

The passive containment isolation barriers are: 1) the closed piping in the MSS, FWS, and DHRS inside of the CNV, 2) the piping between a CNV safe-end and an MSIV, an MSIBV, and the branch lines for the DHRS, 3) the ECCS valve actuator assemblies, 4) the DHRS closed piping outside of containment, and 5) the flange connections. 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 primary objective of protecting against the release of radioactive material to the environment as a result of an accident is accomplished based in part on leakage rates 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. Both of these capabilities are maintained by periodic inservice inspection and testing of these containment isolation components. Table 6.2-4 provides a list the containment penetrations and shows which penetrations are used for which process system fluids or gases.

Table 6.2-5 provides a list of the CIVs. The CIVs are grouped into PSCIVs and SSCIVs. Table 6.2-5 shows the valves in each group. The PSCIVs have two hydraulically-operated valves in series on each line through containment and the pair of valves meets the intent of GDC 55 or 56. The SSCIVs have a single hydraulically-operated valve on each line (and bypass line) and meet GDC 57. The bases for meeting the GDCs, or the intent of the GDCs, are explained below, as are the exemptions needed from the GDCs and their justifications. Table 7.1-4 provides the conditions under which containment isolation becomes mandatory.

Consistent with GDC 1, the CIVs and the passive containment isolation barriers are designed, fabricated, erected, and tested to the quality standards commensurate with the importance of the safety functions to be performed. Consistent with GDC 2, these containment isolation components are located in the RXB, which is designed to withstand the effects of natural phenomena hazards, such as, earthquakes, winds, tornadoes, and floods. These components are protected from natural phenomena hazards by the RXB and meet Seismic Category I requirements. A summary discussion of compliance with GDC 1 and GDC 2 is provided in Section 3.1.1.

Consistent with GDC 4, CIVs and barriers are designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents (including LOCAs). These containment isolation components are also protected against 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 provided in Section 3.6.2.1.2. A summary discussion of compliance with GDC 4 is provided in Section 3.1.1.

Consistent with GDC 5, the CIVs and barriers are designed so that the containment isolation components on one NPM are not shared among the other NPMs at a NuScale Power Plant. A summary discussion of compliance with GDC 5 is provided in Section 3.1.1.

Consistent with GDC 16, the containment isolation components (valves and barriers) are designed such that isolation of the penetrations and the associated fluid systems, together with the CNV, establish 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 provided in Section 3.1.2.

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

Consistent with GDC 56 except for the location of isolation valves, each line that penetrates the containment boundary and is connected directly to the containment atmosphere is provided with 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 located outside of the CNV for the NuScale design. A summary discussion of compliance with GDC 56 is provided 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 are provided with 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 provided in Section 3.1.5.

With the exception of 10 CFR 50.34(f)(2)(xiv)(E), the CIVs, including associated controls, are designed in accordance with 10 CFR 50.34(f)(2)(xiv) and conform to the requirements of RG 1.141 through adherence to ANSI/ANS-56.2. The NuScale 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 provided in Section 1.9.

The CIVs are hydraulically operated and are designed to fail to the closed position, which is the safe position, on loss of power or loss of hydraulic pressure. The PSCIVs are operated by hydraulic actuators powered by nonsafety-related hydraulics and controlled by two divisions of the module protection system (MPS). The actuators are powered from two different hydraulic control skids located on different levels of the RXB to satisfy the single-failure separation requirement. Safety-related, redundant, normally energized Division I and II vent solenoids mounted on the hydraulic control skids de-energize to vent hydraulic pressure from the CIV actuator on an MPS close signal or loss of power. Stored energy in a nitrogen accumulator closes a PSCIV when the hydraulic pressure is vented. The SSCIVs are similar except that these valves have one hydraulically-operated valve and each division is powered from one hydraulic skid.

6.2.4.2 System Design

6.2.4.2.1 General Description

The containment pressure boundary includes the steel CNV (described in Sections 3.8.2 and 6.1) and the CIVs and barriers that close the penetrations in the CNV. Table 6.2-4, "Containment Penetrations," provides a listing of the containment penetrations and associated isolation data. A schematic of CNV penetrations is provided in Figure 6.2-3a and Figure 6.2-3b.

The passive containment isolation barriers use the following design features.

The piping between a CNV safe-end and a CIV consists of the main steam lines from the containment nozzle safe-end to an MSIV and a MSIBV, including the tees to the DHRS piping. Unlike the other CIVs, the MSIVs and bypass valves are not welded to a safe-end on the CNV top head. The short length of piping is leak tight based on the welded design (as described in Section 3.6.2.5), and this piping is provided to allow for a tee for each of the two DHRS piping branch lines (as shown in Figure 6.2-6a).

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

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 NRC 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). The closure flanges are provided with 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 design basis events 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 described in Section 7.1.

Containment isolation of process piping lines in the NuScale design is provided by two barriers. Lines that communicate directly with the RCPB or with the containment atmosphere are equipped with 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 are equipped with 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 CIVs and barriers are classified by safety, quality group, and seismic category. Table 3.2-1 provides the classifications of the CNTS components. The CIVs, their valve actuators, and their penetration safe-ends are designated safety-related and Seismic Category I. The CIVs connected to lines that directly contact the reactor coolant are classified as 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 classified as Quality Group B, as shown in Table 6.2-4. With the exception of the DHRS, the process systems that penetrate the containment pressure boundary are not required for accident mitigation or for safe shutdown.

The CIVs are designed to fail to the position that provides for greater safety on a loss of power to the actuators. Electrical power is not required for actuation of the isolation function. The valves are designed to close upon de-energization to perform the containment isolation function. In addition, as shown in Table 7.1-4, on detection of low AC voltage to the battery chargers, the MPS actuates the CIVs.

The CIVs are mounted 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 GDC. 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 NuScale design provides two isolation valves in a single valve body outside of containment welded to a CNV nozzle safe-end. The benefits of this approach include eliminating 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 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 NuScale design provides two isolation valves in a single valve body outside of containment welded to a CNV nozzle safe-end. The benefits of this approach include eliminating 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 used on the CRDS lines because these lines are not credited as an containment boundary even though the lines form a closed loop inside containment and are not connected to the RCPB and are 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 SSCIVs consist of the MSIVs, the main steam isolation bypass valves, and the feedwater isolation valves (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 6.2-6a, each line with an MSIV has a main steam isolation bypass valve (MSIBV) in a parallel flow arrangement to bypass a MSIV. The MSIBV is a normally closed valve that is in a parallel flow path with the larger diameter MSIV. The MSIBV are used to introduce steam for secondary system startup prior to opening the MSIVs. General Design Criterion 57 requires closed system isolation valves to be located as close to containment as practical. The MSIVS are located approximately 4 ft from the CNV which meets the intent of this requirement.

As shown on Figure 6.2-6b, a FWIV and feedwater isolation check valves are housed in the same valve body. These valves are welded directly to the CNV nozzle safe-end. The purpose of the FWIV is for containment isolation and to provide a DHRS pressure boundary. The feedwater isolation check valve closes more rapidly (<1 second) than the FWIV during a FWLB outside containment. The safety function of the check valve is to close fast and preserve DHRS inventory.

The CIVs are located such that there is sufficient access to allow for the following:

- inservice inspection
- 10 CFR 50 Appendix J, Type C testing
- maintenance and repair

The general and functional arrangements of the CIVs are provided on the piping and instrumentation diagram shown in Figure 6.2-4.

The NuScale CNTS design does not include instrumentation process sensing lines that penetrate the containment pressure boundary. Pressure, temperature, level, and flow sensors that monitor processes within an NPM are located inside the containment pressure boundary with digital or analog signals coupled to equipment outside of the CNV for processing. The NuScale design conforms to the intent of 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 6.2-5, 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 CNV nozzle safe-end. Each valve is a ball type design arranged in a cartridge configuration where the ball, seat, and seals are removed as an assembly from the valve body for maintenance, repair, or replacement. Valve operation is based on a hemispherical design that rotates such that the ball presses into the valve seat to close the valve.

In accordance with the regulatory requirements for redundancy, each PSCIV assembly contains two, independent balls and actuators. The single body design welded to a nozzle safe-end 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. The I&C for each valve within a PSCIV assembly are provided by independent divisions of the MPS.

The PSCIVs connected to lines that directly contact the reactor coolant during normal operation are designed and constructed in accordance with the ASME Code, Section III, Class 1, Subsection NB, Quality Group A, and Seismic Category I criteria. The PSCIVs on the other lines are designed and constructed as Class 1; however, these valves are classified the same as the lines, which are not part of the RCPB inside of containment. These PSCIVs are Quality Group B components and are on lines designed and constructed in accordance with ASME Code, Section III, Class 2, Subsection NC, Quality Group B, and Seismic Category I criteria.

As shown in Figure 6.2-6a and Figure 6.2-6b, the single valve design consists of one ball-type valve in a cartridge configuration like the PSCIVs. The SSCIVs have a ball, seat, and seals that allow for removal as an assembly from the valve body for maintenance, repair, or replacement. Valve operation is based on a design with the ball positioned off-center to provide for a tight seal on both the upstream and downstream metal seats.

The SSCIVs are designed to the ASME Code, Section III, Class 2, Subsection NC, Quality Group B, and Seismic Category I criteria. The FWIVs, MSIVs, and main steam isolation bypass valves are designed to these criteria. The MSIVs and main steam isolation bypass valves are welded to a short section of piping that is welded directly to a CNV top head nozzle safe-end. The FWIVs are welded to a CNV top head nozzle safe-end.

When the PSCIVs are actuated to close, the fluid between the two valves could then heat up. Design overpressure for this condition is taken into consideration in the dual valve design. As the fluid heats up and expands, the pressure increase that is applied in the direction against the inboard valve ball changes the force balance on the hemispherical ball and unseats the ball to open a gap between the ball and the seat to relieve the excess pressure to the CNV.

The PSCIV design provides a bonnet closure with a double seal with a test connection in the space between the seals to allow for detection of leakage past the valve bonnet seals. The valve design also provides for Appendix J, Type C testing via the use of "testing only" and "inservice" test inserts that allow the pressurization of each of the two valves (Figure 6.2-5).

Hydraulic actuators are used for both PSCIV and SSCIV designs. The hydraulic system consists of two remote skids, which supply pressurized hydraulic fluid to the valve actuators. Each valve uses a quarter turn rack and pinion type hydraulic actuator. Maintaining the hydraulic system pressure supplies the force to keep the valve in the open position. Pre-charged, nitrogen-filled cylinders are mounted on the "closed" side of each actuator. The gases in the cylinders are compressed as part of the valve opening action. Compression of the gas in each cylinder provides the passive stored energy used for valve closure.

Dual solenoid valves are positioned on the supply side of each of the hydraulic lines. When energized, the solenoids allow supply of pressurized hydraulic fluid to an actuator to open the valve and maintain it in the open position. When de-energized, the solenoids vent the hydraulic fluid in a supply line to a reservoir which depressurizes the valve hydraulic supply and allows the stored energy in the nitrogen cylinders to close the valve.

6.2.4.2.2.1 Piping Systems Connected to the Reactor Coolant Pressure Boundary

Isolation of each containment penetration that is part of the RCPB (RCS injection, RCS discharge, RCS high point degasification and pressurizer spray) is provided by a dual valve, single body PSCIV.

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 incorporated into each single valve body.

The NuScale design of the PSCIVs on these lines locates both valves outside of the harsh CNV environment which is a justified approach for meeting GDC 55 for penetrations with lines containing reactor coolant. Compliance with GDC 55 is discussed in Section 3.1.5.

Each PSCIV is remotely operated from the main control room with valve position indicated and each valve automatically closes under accident conditions that require containment isolation. These valves also are designed to close under loss of power to the hydraulic solenoids or loss of hydraulics to the actuator assemblies. Table 6.2-4 lists the PSCIV open or closed position for normal and accident conditions.

6.2.4.2.2.2 Piping Systems Open to the Containment

The NuScale design includes two systems that penetrate the containment boundary and are open to the containment atmosphere. These systems are the CES and the 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 are not connected to the RCPB and like the piping, the PSCIVs are classified as ASME Class 2 valves.

When closed, the PSCIVs on these lines 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 incorporated into a single body.

The NuScale design of the PSCIVs on these lines locates both valves outside of the harsh CNV environment which is a justified approach for meeting GDC 56 for lines that are open to the containment atmosphere. Compliance with GDC 56 is discussed in Section 3.1.5.

The I&C for each valve within a dual valve, single body PSCIV are provided by independent divisions of the MPS. Each PSCIV can be remotely operated from the main control room and automatically closes under accident conditions that require containment isolation.

Table 6.2-4 lists the PSCIV open or closed position for normal and accident conditions.

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 is provided with an SSCIV that is a single CIV outside of containment, or is provided with 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:

- main steam system lines (single SSCIV)
- feedwater system lines (single SSCIV)
- decay Heat Removal System piping (closed piping loop outside of containment)

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

Each MSIV and main steam isolation bypass valve is welded to a short section of piping that is welded directly to a CNV top head nozzle safe-end. The short sections of piping are constructed to ASME Code, Section III, Class 2, Subsection NC criteria. The closed loop section of the piping inside containment is constructed to the same criteria. The main steam line is also fitted with a tee for each of the two branch lines that connect to the DHRS.

The SSCIVs on the main steam and feedwater lines are Seismic Category I, Quality Group B components capable of remote operation from the control room. The SSCIVs and the actuators are constructed in accordance with ASME Code, Section III, Class 2, Subsection NC criteria.

Each SSCIV is remotely operated from the main control room and automatically closes under accident conditions that require containment isolation. Each valve has remote Class 1E position indication in the main control room. The valves are also designed to close under loss of power to the solenoid or loss of hydraulics to the actuator assembly. Table 6.2-4 lists the SSCIV open or closed position for normal and accident conditions.

The CRDS cooling water lines have CIVs that are of the dual valve, single body design constructed to ASME BPVC, Section III, Class 1, Subsection NB, criteria. These valves are classified as Quality Group B, ASME BPVC, Section III, Class 2, Subsection NC, but are designed and constructed as Class 1 components. The classification is based on the following. The CRDS piping and pressure retaining components inside containment form a closed loop and are constructed to ASME BPVC, Section III, Class 2, Subsection NC. The CIVs for these CRDS lines are classified as ASME Class 2 components. The isolation valves are welded directly to the CNV top head nozzle safe-end and the actuators are designed and constructed to ASME BPVC, Section III, Class 1, Subsection NB.

When closed, the PSCIVs on the CRDS lines isolate reactor component cooling water system piping to and from the CRDMs. The design configuration for CRDS piping inside containment satisfies GDC 57 criteria for a closed system inside containment, however, because this piping is not credited as a containment boundary, the dual valve, single body PSCIVs are used. Compliance with GDC 57 criteria is discussed in Section 3.1.5.

Each PSCIV on a CRDS line is remotely operated from the main control room and automatically closes under accident conditions that require containment isolation. Valve position is provided in the main control room and classified Class 1E. The valves are also designed to close under loss of power or hydraulics to the actuator assemblies. Table 6.2-4 lists the PSCIV open or closed position for normal and accident conditions.

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 is provided by closed loop Seismic Category I, Quality Group B, ASME Class 2 piping inside and outside of the containment boundary. The system is described here to the extent that the SG walls and DHRS piping address the GDC 57 dual barrier criteria. The containment pressure vessel penetrations for the DHRS are listed in Table 6.2-4.

The DHRS steam side piping branches off each main steam line external to the CNV and upstream of a 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 than the FWIV in the event of a FWLB outside containment to preserve DHRS inventory.

The DHRS actuation valves are Seismic Category I, Quality Group B components capable of remote operation from the control room and automatic actuation based on the isolation of main steam and feedwater to the RPV. The actuation valves and actuators are constructed in accordance with ASME BPVC Section III, Class 2, Subsection NC.

The NuScale design configuration for the DHRS satisfies the intent of GDC 57 with the exception that closed loop piping provides an isolation barrier outside of containment rather than an isolation valve. The NuScale 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 justified 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 DHRS piping outside of containment performs the isolation function and establishes the containment boundary outside of containment without the need for valve actuation. Compliance with GDC 57 is discussed in Section 3.1.5. Section 5.4.3 provides a description of the DHRS.

6.2.4.2.3 System Operation

Piping systems penetrating the containment are provided with leak detection, isolation, and containment capabilities. 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 NuScale designs for the PSCIVs and SSCIVs allow for normal operation of the systems shown in Figure 6.2-4 and automatic isolation of the valves for accident conditions that require containment isolation. Table 6.2-4 provides the time for each valve to receive the signal to start closing and the time to complete closing. The designs of the CIVs permit periodic operability and leak testing. Testing details are discussed further in Section 6.2.4.4 and Section 6.2.6.

6.2.4.3 Design Evaluation

The primary purpose of the containment isolation function is to prevent the release of radioactive materials with an essentially leak tight barrier. This purpose is accomplished by the integrity of the CNV and with isolation of the penetrations for the CNV. Containment isolation actuation and control logic, valve actuation and control features, and valve closure times are taken into consideration.

The components in the CNTS are designed to Seismic Category I criteria. The CIVs are designed in accordance with ASME Code, 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. Conformance with RG 1.26 is discussed in Section 1.9.1.

The quality groups for CNTS components are provided in Table 3.2-1. Quality standards applicable to the respective quality groups are described in Section 3.2.

The CIVs are protected from external hazard effects by their placement within the RXB structure. Protection from internal hazards is addressed in Section 3.4 (flooding), Section 3.5 (missiles), Section 3.6 (pipe rupture), Section 3.7 (earthquake), and Section 9.5 (fire). Environmental qualification of CNTS components is addressed in Section 3.11.

Redundancy is provided for containment isolation of the lines penetrating containment. Isolation of each piping penetration is provided by two barriers in series consisting 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 with a closed piping system inside of containment. With this arrangement, no single active failure can prevent containment isolation of a piping penetration. Containment boundary valve and barrier components are redundant, in that each isolation valve and barrier arrangement provides for backup in the event of accident conditions. The isolation valve and barrier arrangements satisfy the intent of GDCs 54, 55, 56, and 57 and conform to RG 1.141. Conformance with GDCs 54, 55, 56, and 57 is discussed in Section 3.1.5. Conformance with RG 1.141 is addressed in Section 1.9.1.

Where the isolation boundary is provided by a CIV, the valve is designed to fail to the safe (isolate) position on loss of hydraulics, loss of signal, or loss of power to the actuator.

Redundant CIV actuator vent solenoid valves are powered from redundant safety-related divisions of MPS. Redundancy in the instrumentation and control system for the CIS is discussed further in Section 7.1.

As described in Section 8.4, the fail-safe feature of the CIVs and passive containment cooling design ensures containment integrity is maintained in the event of a station blackout. The NuScale containment isolation function is initiated upon loss of power and the isolation valves remain closed for the duration of design basis events. The passive cooling design of the NuScale reactor plant provides a reliable coping capability that is able to achieve safe shutdown and maintain core cooling and containment integrity for an indefinite duration, independent of AC power sources.

Conformance with General Design Criterion 5 is discussed in Section 3.1.1.

The closed piping systems within the containment are designed to withstand CNV design conditions for external design pressure and temperature, and to withstand the environmental effects resulting from a LOCA. The closed piping systems are protected against LOCA effect missiles, pipe whip, and jet forces. Qualification of the closed piping inside containment also takes into consideration 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 are designed to withstand reactor vessel design pressure and temperature. The piping forms a passive containment isolation barrier located outside of containment and is designed to function under the most adverse anticipated environmental conditions to which the piping may be exposed.

The piping systems that penetrate containment are provided with leak detection, isolation, and containment capability having redundancy, reliability, and performance capabilities commensurate with the containment isolation function. Leakage detection capability and the leakage detection test program are addressed in Section 6.2.6. The CIV operability tests are discussed in Section 3.9.6. Redundancy and reliability of the CIV actuation system is discussed in Section 7.1. Taken together, these programs establish the overall reliability of the CIVs.

The CIVs described in Section 6.2.4.2 are located as close as practical to the CNV wall by welding directly to a CNV nozzle safe-end, or for the main steam lines, to a short length of piping welded to a safe-end.

The CIVs and barriers are capable of performing their intended function under the maximum integrated radiation levels to which they may be exposed during the component's service life. The accumulative effect includes lifetime operational exposure plus the maximum anticipated exposure following a design basis accident.

Each line penetrating containment that is either part of the RCPB, connects directly to the containment atmosphere, or is not credited as meeting the requirements of a closed piping system inside of containment is provided with a dual isolation valve assembly (single body design) located outside of containment with automatic isolation capability that is welded directly to a CNV nozzle safe-end.

Each line that penetrates containment that is neither part of the RCPB nor connected directly to the containment atmosphere and satisfies the requirements of a closed piping system inside of containment is provided with piping and components that are constructed to ASME Code Section III, Class 2, Subsection NC criteria with a design pressure and temperature equivalent to the RCS.

The containment boundary design for the DHRS justifies an exemption from the GDC 57 criteria. Instead of a single isolation valve located outside of containment, the DHRS relies on the passive isolation barrier of a closed piping system outside of containment. The DHRS piping penetrations have a closed piping system inside of containment and closed piping loops outside of containment. The DHRS closed piping loops and actuation valves are designed in accordance with ASME Code Section III, Subsection NC, Quality Group B, Seismic Category I criteria and provide a "passive" alternative to an "active" CIV located outside of containment.

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 engineered safety features systems, including the initiation of the containment isolation function. Refer to Section 6.2.2.5 for additional detail. The design basis of the MPS is discussed in Section 7.1.

The NuScale control systems are designed for resetting an isolation signal without automatically reopening the valves. An override control function is performed by deliberate operator action and only after actuation signal reset. Reset does not automatically cause any isolation valve to change position. The reopening of more than one CIV at a time is precluded by design. Reopening of isolation valves is performed valve by valve or line by line.

Actuation of the containment isolation function consists of the closure of the CIVs on the MSS, FWS, reactor component cooling water system, CVCS, CES, and CFDS lines. As shown in Table 7.1-4, an ESFAS containment system isolation actuation signal is initiated on high containment pressure, 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 provided in Section 7.1.

Closure times for CIVs are established to 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. Closure times are specified in Table 6.2-4. 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 are designed such that closure times assume the valve begins to stroke within two seconds of de-energization or loss of power to the pilot valve solenoid and fully stroke closed within five seconds of beginning the stroke for a total of seven seconds.

The PSCIVs are designed with the capability of stopping line break flows of 200 percent within a five second valve stroke time.

The MSIVs are capable of stopping fully developed steam line break flows of 100 percent and four percent steam conditions within a five second valve stroke time. The main steam isolation bypass valve is capable of closure within 10 seconds of receipt of a closure signal or loss of power. The FWIVs are capable of stopping fully developed FWLB flows of 200 percent in the forward direction and together with the internal safety-related check valve are capable of closure within 1 second on fully-developed reverse flow.

The open or closed status of CIVs is provided in the main control room.

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

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

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

The NuScale design provides the capability to periodically test the CIVs for leakage and functionality, the details of which are provided in Section 3.9.6.3 and Section 6.2.6. During normal operation, any bolted connection or valve stem packing that forms part of the pressure boundary of the PSCIVs includes a double seal and means to detect, measure, and terminate leakage past the seals. The SSCIVs are designed with a means to preclude or monitor and capture leakage past the bonnet seals and valve stem packing.

6.2.4.4 Tests and Inspections

The CIVs and barrier components are designed to permit rigorous inspections and performance of tests to ensure that functional capability of the components is maintained under design basis accident conditions.

The PSCIVs connected to lines that directly contact reactor coolant are inspected as Class 1 components and those that directly contact the containment atmosphere or form a closed loop inside containment are inspected as Class 2 components. The SSCIVs are classified and inspected as Class 2 components.

The periodic inspections and testing programs meet ASME BPV and 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 provided in Chapter 14.

CIVs are testable through the entire sequence initiated by a containment isolation signal. The CIVs are verified to close within the time specified in Table 6.2-4.

6.2.4.4.2 Periodic Operability Testing

The leak-tight integrity of the CNV and the isolation valves and barriers is established and verified periodically as described in Section 6.2.6. Additional information addressing the inservice testing of containment isolation components is provided Section 3.9.6.3.

Each CIV body is designed with test connections that facilitate Appendix J, Type C, leakage testing. Refer to Section 6.2.6 and Section 6.6 for 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 are provided to ensure adequate safety. The protection system is designed to sense operating conditions and automatically initiate the actuation of the CIVs and other components or systems needed for accident mitigation. Manual control capability to mitigate the consequences of faulted conditions is also provided, at the division level, via the instrumentation and control system.

The MPS provides instrumentation and controls to sense accident conditions and initiate the operation of necessary engineered safety features. These instruments and control devices are served by instruments that support achieving their requirements for redundancy, independence, and testability to ensure the systems safety functions are accomplished. The occurrence of a limiting fault, such as a LOCA, FWLB or a steamline break, requires a reactor trip and actuation of one or more of the engineered safety features in order to ensure containment integrity.

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

The ESFAS, which is part of the MPS, consists of four redundant separation groups that monitor the plant parameters that activate the operation of the ESFAS containment isolation function. MPS uses a two-out-of-four coincidence with two independent, diverse, and redundant divisions of ESFAS such that a single failure will not prevent the containment isolation function from being accomplished when it is required.

Each ESFAS signal measurement channel trips when the associated parameter exceeds a predetermined set point. The ESFAS signal coincident logic is designed so that no single failure can prevent a safeguards actuation when required, and no single failure in a single measurement channel can generate an unnecessary safeguards actuation.

To satisfy single failure requirements, containment isolation signals are actuated 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-low pressurizer level, high narrow range containment pressure, low AC voltage, and high under-the-bioshield temperature.

The MPS is designed such that once initiated, automatic or manual, the protective action sequence continues to completion with deliberate operator action required to return MPS to a non-tripped (reset) state. A seal-in of the ESFAS logic is provided at the division level. When plant conditions allow, the operator can manually reset the division level actuation signal and reposition the isolation valves. Operator action is allowed only after the equipment has fully actuated to the required position by the engineered safety feature safety function.

The MPS allows for periodic on-line surveillance testing during power operations without adversely affecting safety or operability of the generating station.

Refer to Section 7.1 for further details on the fundamental design principles of instrumentation and control, and Section 7.2.12 for details on automatic and manual control.

6.2.5 Combustible Gas Control in the Containment Vessel

This section addresses control of combustible gases that have the potential to accumulate in containment following postulated accidents including beyond-design basis accidents involving up to 100 percent fuel clad-coolant reaction in conformance with GDC 5, principal design criterion 41 (Section 3.1.4) and 10 CFR 50.44.

Regulatory requirements dictate the control of generated combustible gases by establishing the capability to ensure a mixed containment atmosphere, controlling combustible gas concentrations in containment during and following an accident that releases hydrogen (i.e., LOCA) and the capability to continuously measure containment hydrogen concentrations following a significant beyond design-basis accident.

The combustible gas control requirements for future water cooled reactor containments of reactor designs that have a comparable potential for the production of combustible gases to the current light water reactor designs are prescribed in 10 CFR 50.44(c).

The potential for generation and accumulation of combustible gases within containment exists following an accident as a result of fuel cladding-coolant reaction and radiolysis of reactor coolant.

Sufficient hydrogen and oxygen could be generated from these sources and accumulate into a flammable or detonable mixture. However, hydrogen combustion scenarios occurring within the first 72 hours following design basis or beyond design basis event initiation, have no adverse effect on containment integrity or plant safety functions.

6.2.5.1 Design Bases

The NuScale CNV design, does not require an inerted atmosphere or means to limit hydrogen concentrations in order to prevent a hydrogen combustion event that could result in a loss of containment structural integrity or accident mitigation features. The CNV can withstand the environmental conditions created due to combustible gas deflagrations, reflected detonations, and deflagration-to-detonation transitions (DDT) for design basis or severe accident scenarios. For the NuScale design, CNV structural integrity and availability of equipment necessary for safe shutdown are assured without reliance on an inert atmosphere or low hydrogen concentrations. Hydrogen combustion scenarios, occurring 72 hours following an event initiation, have no adverse effect on containment integrity or plant safety functions. The CNV design does not use combustible gas control systems. The NuScale design supports an exemption from the hydrogen control requirements of 10 CFR 50.44(c)(2).

NuScale examined the potential for combustible gas (hydrogen and oxygen) accumulation in the containment during and following postulated severe accidents. The type of severe accidents considered were those that assume an intact containment boundary and result in varying degrees of core damage. One example of this type of severe accident is a LOCA inside containment with an ECCS failure that prevents the recirculation of coolant from the CNV back into the RPV. The 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 accumulate within the CNV.

The CNV is capable of maintaining a mixed containment atmosphere during design-basis and significant beyond design-basis accidents that is sufficient to prevent the concentration of combustible gases in any part of the containment from exceeding levels that support combustion or detonation events that could result in a loss of containment integrity.

Adequate mixing of the CNV is ensured by virtue of 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 NuScale design does not rely upon active components within the first 72 hours of the event to limit hydrogen concentration in containment. The containment is designed to withstand the bounding loads from deflagration, incident detonation, reflected detonation, and DDT events for 72 hours. The calculated combustion loads are evaluated against ASME Level C and D limits for all components that could be exposed to a combustion event. Beyond 72 hours, the accumulation of combustible gases can be managed by licensee implementation of severe accident management guidelines or other ad hoc actions because of the long period of time available to take such action. A structural evaluation of the impact of limiting combustion pressures demonstrates the ability of the CNV to withstand the design loads with no impact on structural integrity or safety functions. This evaluation is described in more detail by Reference 6.2-3, Section 3.3. Although, the combustion loads were evaluated against ASME Level C and D limits for design basis events, the NuScale design is not required to comply with the additional TMI requirements given by 10 CFR 50.34, paragraph (f)(3)(v)(A)(1). 10 CFR 50.34, paragraph (f) specifically states that applicants for design certification under Part 52 do not need to demonstrate compliance with paragraph (f)(3)(v).

The systems and components within CNV that are relied upon to establish and maintain safe shutdown or support containment structural integrity remain capable of performing their required functions after containment hydrogen burn or detonation events that assume a hydrogen accumulation based on fuel cladding-coolant reactions involving 100 percent of the cladding surrounding the active fuel region. Systems and components within containment that are relied upon to establish and maintain safe shutdown or support containment structural integrity are evaluated for the impact of combustion loading.

Equipment is provided for monitoring in-containment hydrogen that is reliable and remains functional and capable of continuously measuring the concentration of hydrogen following a significant beyond design-basis accident for accident management and emergency planning.

Hydrogen generation criteria associated with the ECCS performance criteria requirements of 10 CFR 50.46 are addressed in Section 6.3.

Consistent with GDC 5, the NuScale design relies on a passive containment design for control of combustible gases that does not involve sharing between reactor modules (i.e., nuclear power units).

Compliance with principal design criterion 41 (Section 3.1.4) is met with the NuScale passive design with respect to hydrogen and oxygen control and cleanup. The NuScale design does not rely on active systems to control the concentration of hydrogen, oxygen or other substances released into containment following postulated events in order to assure that containment integrity is maintained. Structural analysis demonstrates that the CNV can withstand the environmental conditions created by the burning of hydrogen during the first 72 hours of design-basis and beyond design-basis accidents, while maintaining structural integrity and safe shutdown capability. Consideration of combustion loading is included in the ASME design specifications for CNV components to ensure that structural integrity and safe shutdown capability is ensured.

Compliance with guidance provided in RG 1.7 is addressed in Section 1.9.

6.2.5.2 System Design

The NuScale CNV is a metal containment, Class MC pressure vessel, that is designed, analyzed, fabricated, inspected, tested and stamped as an ASME Code Class 1 pressure vessel that is maintained partially immersed in a reactor pool common to eleven other reactor modules. The CNV is a passive design that does not rely on an actively inerted or vented containment during normal power operation or for the first 72 hours following any design basis or significant beyond design basis event for combustible gas control.

The principal functions of the CNTS include the following:

1) contain fission product releases from the RCPB

2) contain the mass and energy released in a postulated LOCA

3) support ECCS operation by containing the reactor coolant and transferring the heat through the containment wall to the UHS

The NuScale CNV is designed to meet the 10 CFR 50.44 requirements to safely accommodate the hydrogen generated by the equivalent of up to a 100 percent fuel-cladding metal water reaction as described in SECY 93-087. This type accident is considered to be a significant beyond-design-basis (severe) accident in which hydrogen generation could exceed the flammability limits.

A partial vacuum is initially established in the CNV by the CES prior to reactor module startup that is maintained during reactor operation. The initial combustible gas composition in the CNV is calculated based in the initial CNV pressure. The CES is addressed in Section 9.3.6.

Following a LOCA resulting from a pipe break or component failure, energy is released into CNV that increases temperature and pressure. The CNV retains the RCS water within the CNV allowing heat transfer to the UHS for an extended period of time. Simultaneous actuation of the CIS as described in Section 6.2.4 provides an essentially leak tight barrier against the uncontrolled release of radioactivity for all design events.

The CNV is designed to passively provide for a mixed post-accident atmosphere, mitigate combustible gas generated during design-basis and beyond-design basis accidents and withstand the effects of postulated combustible gas deflagration or detonation without the loss of containment integrity.

When RCS discharge to the containment occurs, the dynamic nature of the event creates a mixed atmosphere due to the high turbulent condition that is induced. As turbulence subsides later in the event, continued mixing is ensured through convection and molecular diffusion. There are no partitions or subcompartments to impede these natural mixing forces. Convective mixing is ensured for relevant events because it is driven by decay heat. Turbulence in the CNV is evaluated as discussed in Reference 6.2-3. The analysis shows that turbulent convective mixing exists in the CNV throughout the first 72 hours of a DBE or beyond design basis event.

The NuScale CNV design, does not require a means to limit hydrogen concentrations, for the first 72 hours, in order to prevent a hydrogen combustion event that could result in a loss of containment structural integrity or accident mitigation features. The configuration of the NuScale containment coupled with the dynamics of the LOCA and mitigating components assures adequate mixing within the containment volume during and following events that generate and release combustible gases to containment. Combustible gas control is relevant in any event that results in a successful actuation of ECCS to maintain RPV liquid level above the top of the core. Because of effective heat removal from the containment, temperature and pressure decrease rapidly. Small amounts of non-condensables are discharged to the CNV that includes residual hydrogen used for RCS water chemistry control and any that may have accumulated in the upper pressurizer during extended periods of reactor operation.

Hydrogen and oxygen gases are also radiolytically generated and accumulate stoichiometrically following ECCS initiation. The structural loading that could occur as a result of combustion is assessed at 72 hours. The structural analysis demonstrates that the CNV is capable of withstanding the combustion loads with significant margin to stress and strain limits. Combustion loads on components required to maintain CNV integrity or ensure safe shutdown capability are shown to be bounded by the CNV analysis loads or are considered in the ASME design specifications.

The NuScale CNV is designed such that the pressure capacity is not exceeded by design basis events. The pressure capacity of the containment is higher than the pressure resulting from the assumed 100 percent fuel clad-coolant reaction. There is margin to the containment pressure capacity limit such that there is no need for containment overpressure protection.

6.2.5.3 Design Evaluation

This section provides the results of evaluations that demonstrate: (1) the effective control of combustible gases in the NuScale CNV; and (2) assurance of CNV integrity and safe shutdown following any postulated design-basis accident or significant beyond design-basis event (pursuant to 10 CFR 50.44). Further details concerning these evaluations is provided in the Combustible Gas Control Technical Report (Reference 6.2-3).

NuScale 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 rate of post-accident oxygen and hydrogen generation by radiolysis in the NuScale containment following a severe accident determines the time required for concentrations in containment to increase to a value that would constitute a combustible gas condition. The rate of radiolytic production is dependent on the amount of fission products released to the reactor coolant and the related power decay profile and the associated energy deposition. The energy deposition results in a proportional radiolytic production of hydrogen and oxygen. The integrated energy deposition into water for design basis events and the radiolytic production of combustible gases for 72 hours is depicted by Reference 6.2-3, Figures 3-1 and 3-2, respectively.

In the design-basis LOCA, the reactor vessel depressurizes through the break or opening as reactor coolant inventory is discharged into the CVN where the coolant flashes to steam and is subsequently condensed. The released coolant inventory is collected, retained and eventually returned to the reactor vessel via natural circulation after the emergency core cooling valves open.

The oxygen and hydrogen generated from radiolytic decomposition and any non-condensables released from the RPV or stripped from the reactor coolant during the steam flashing and condensation process gradually accumulate within the containment.

The sources of hydrogen in containment following a design basis or beyond-design-basis (severe) accident are limited to

- oxidation of zirconium in the fuel cladding.
- radiolysis of water (reactor coolant).
- initial amount of dissolved hydrogen in the RCS.
- amount of hydrogen accumulated in the upper region of the RPV (i.e., the pressurizer).

Materials within the CNV that have the potential to yield hydrogen gas due to contact with liquid contents in the CNV (upon ECCS actuation or other condition involving liquid in containment) are restricted. Any such materials are identified in Section 6.1.

Analyses have been performed to demonstrate CNV structural integrity is maintained in the event of an accident that releases hydrogen generated from a 100 percent fuel clad-coolant reaction accompanied by hydrogen burning.

The results of the analysis for containment response show that the containment can withstand a global combustion without loss of integrity. The structural analyses show a 60 percent margin to the design stress limits for the reflected detonation load, and 15 percent margin to the design stress limits for the DDT load. An elastic-plastic analysis quantifies the integrity of the containment for a severe accident core melt event which demonstrates an 85 percent margin for the highest membrane hoop strain.

The structural integrity of the NuScale CNV is demonstrated by compliance with the guidance described in RGs 1.7, 1.57 and 1.216. NuScale compliance with the RGs is addressed and the analyses used to demonstrate structural integrity are described in Section 1.9.

6.2.5.4 Inspection and Testing

Inspection and testing of the CNV and its components is described in 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.

6.2.5.5 Instrumentation

Hydrogen and oxygen analyzers are provided within the containment sampling system portion of the process sampling system. During normal operation the containment gas

discharge from the CES vacuum pumps is continuously routed to the containment sampling system sample panel for online analysis of hydrogen and oxygen concentrations, with continuous indication in the main control room. The monitors meet the criteria specified in Regulatory Guide (RG) 1.7 for non-safety-related commercial-grade monitors. The hydrogen and oxygen monitoring equipment is designed to be functional, reliable and operable in design basis or beyond design basis environmental conditions.

The CES is isolated during design basis and beyond design basis events. However, the monitors can be used for combustible gas control and accident management, including emergency planning. This is described in more detail by Reference 6.2-3, Section 2.7.

6.2.6 Containment Leakage Testing

Containment leakage rate testing is designed to verify the leak tight integrity of the reactor containment. The CIVs on CNV piping penetrations, and the passive containment isolation barriers are designed to permit the periodic leakage testing described in GDCs 53 and 54 to ensure leakage through the CNTS and components does not exceed the allowable leakage rate specified in Technical Specifications. Compliance with GDCs 52, 53 and 54 is further described in Section 3.1. The NuScale design supports an exemption from the integrated leak rate testing specified in the GDC 52 criterion. Further details are provided by Reference 6.2-6.

The preoperational and periodic containment leakage testing requirements and acceptance criteria that demonstrate leak-tight integrity of the CNTS and associated components are prescribed in 10 CFR 50, Appendix J and implemented through the reactor containment leakage rate testing program described in Section 5 of the Technical Specifications.

The design of containment penetrations support performance of local leak rate tests (Type B and Type C tests) in accordance with the guidance provided in ANSI/ANS 56.8, Regulatory Guide 1.163, and NEI 94-01. The NuScale system design, in conformance with 10 CFR 50.54(o), accommodates the 10 CFR 50, Appendix J, test method frequencies of Option A or Option B.

COL Item 6.2-1: A COL applicant that references the NuScale Power Plant design certification will develop a "Containment Leakage Rate Testing Program" which will identify which Option is to be implemented under 10 CFR 50, Appendix J. Option A defines a prescriptive-based testing approach whereas Option B defines a performance-based testing program.

To verify the leak tightness of the reactor containment, Type B, and C tests are performed prior to initial operations and periodically thereafter to assure that leakage rates through the containment and the systems or components that penetrate containment do not exceed the maximum allowable leak rate.

The specified maximum allowable containment leak rate, La, is 0.20 weight percent of the containment air mass per day at the calculated peak accident pressure, Pa, identified in Section 6.2.1. Containment leak rate testing is designed to verify that leakage from containment remains within the prescribed Technical Specification limits.

The reactor containment, containment penetrations, and isolation barriers are designed to permit periodic leakage rate testing in accordance with GDC 53 and GDC 54 independent of other reactor modules.

6.2.6.1 Containment Integrated Leakage Rate Test

The NuScale CNV design is different from traditional containments and exempt from GDC 52 criterion because integrated leakage rate testing as described of 10 CFR 50 Appendix J, Type A tests, are not required to meet the purpose of the rule. Specifically, the CNV is

- a high pressure vessel.
- an ASME Class MC component constructed to ASME Class 1 vessel rules.
- constructed of all stainless steel clad or stainless materials.
- designed with penetrations that are either ASME Class 1 flanged joints capable of Type B testing or ASME Class 1 welded nozzles with isolation valves capable of Type C testing.
- inaccessible (interior) to personnel during startup, shutdown and normal operation.
- under a vacuum and partially immersed in borated water during normal operation.
- constantly monitored during normal operation for containment vacuum and leakage into containment.

- disassembled by separating the upper and lower CNV shells during outages for refueling, maintenance and inspection.

GDC 52 requires that containments are designed so that periodic integrated leakage rate testing can be conducted at containment design pressure. The purpose of GDC 52 is to provide design capability for testing to verify leakage tightness to ensure continued leakage integrity of the CNTS. The CNTS meets the purpose of the rule due to the unique features of the NuScale Power containment design.

- Manufacturing and preservice test and inspections are similar to RPV requirements.
- All known leakage pathways will be Type B or Type C tested.
- Comprehensive ISI will meet ASME Class 1 criteria to ensure no new leakage pathways develop.

The CNV is an ASME Subsection NE, Class MC containment, and is designed, fabricated, and stamped as an ASME Subsection NB, Class 1 pressure vessel, except that overpressure protection is in accordance with NE 7000, see Subsection 3.8.2. The CNV is made of corrosion resistant materials, has a low number of penetrations, and no penetrations have resilient seals. The use of all welded nozzles and testable flange seals at every containment penetration ensure that Type B and C testing provides an adequate assessment of containment leak rate.

The NuScale design has fewer and smaller potential leak pathways than traditional, large pressurized water reactors, which provides a meaningful safety advantage. The small size of the containment allows for factory fabrication, which facilitates increased quality and testing control than field construction.

Pressure retaining and integrally attached materials meet the requirements of ASME Subsections NB-5000 and NF-5000 using the examination methods of ASME Section V. All surfaces to be clad will be magnetic particle or liquid penetrant examined in accordance with ASME Subsections NB-2545 or NB-2546, respectively.

Preservice examinations for ASME Class 1 pressure boundary items will be performed in accordance with ASME Subsection NB-5280 and ASME Section XI, Subsection IWB-2200 using the examination methods of ASME Section V, except as modified by ASME Subsection NB-5111. These preservice examinations include 100 percent of the pressure boundary welds.

Final preservice examinations will be performed after hydrostatic testing but prior to code stamping.

The CNV is hydrostatically tested in accordance with ASME Subsection NB-6000. The test is conducted with the RPV installed and vented. The water-filled CNV is pressurized to a minimum of 25 percent over design pressure for at least ten minutes. Pressure is then reduced to design pressure and held for at least four hours prior to examining for leaks. The acceptance criterion is no leakage indications at the examination pressure (design pressure).

ASME Class MC, Section IWE, only requires visual examination for SSCs subject to normal degradation and aging. Surface areas that are subject to accelerated degradation and aging require an ultrasonic thickness exam. However, based on the high pressure and safety functions of the CNV, the Inservice Inspection Program requires the CNV to meet ASME Class 1 requirements, similar to the RPV. The CNV design allows for visual inspection of the entire inner and outer surfaces; therefore, developing a leak through the metal pressure boundary is implausible.

All CIVs are located outside of the CNV. The reduced ISI requirements for small primary system pipe welds between the CNV and the CIVs are not applied to these welds. Welds between the CNV and the CIVs are ASME Class 1 and will be inspected with a volumetric and surface exam each test interval. The CNV is designed to accommodate comprehensive inspections of welds, including volumetric and surface inspections. All welds are accessible and there are no areas that cannot be inspected.

The CNV is maintained at a vacuum during normal operation. Leakage into the CNV is continuously monitored via liquid level in the containment evacuation sample vessel and possible changes in CNV pressure. Leakage into the CNV can be either from the UHS or pressurized systems inside the CNV. The source of the leakage can be identified by sampling the containment evacuation sample vessel. Maintaining the CNV at a vacuum facilitates a constant on-line, low pressure leakage monitoring of the containment boundary during normal operations.

Known pathways will be tested periodically by Type B and Type C LLRTs, as discussed in Section 6.2.6.2 and Section 6.2.6.3. All penetrations are either ASME Class 1 flanged joints capable of Type B testing or ASME Class 1 welded nozzles with isolation valves capable of Type C testing. There are 40 total penetrations in the CNV. There are 26 Type B penetrations, see Table 6.2-4. An additional Type B test is performed at the main CNV flange. There are 8 total pipe penetrations with CIVs that will be Type C tested, (see Table 6.2-4). These 8 pipe penetrations are each protected by two, 2-inch PSCIVs. The other 6 pipe penetrations are protected by either a single SSCIV or a closed loop (DHRS). The SSCIVs are leak tested as a DHRS boundary, but are not Type C tested.

No instrument lines penetrate containment; therefore, there are no small diameter fluid lines without isolation capability that are not subject to Type B or C local leak rate test. There are no air locks, flexible sleeves, or non-metallic boundaries. There are no penetrations in the NuScale design that would only be tested in an integrated leak rate testing. Entry into the CNV is precluded during normal operation by personnel safety constraints and most openings will be underwater. The integrity of the Type B pathways is not expected to be disturbed except when the NPM is in a refueling outage or disassembled for emergent repairs. All Type B and Type C pathways will be tested to peak CNTS accident pressure (Pa). All Type C pathways are designed such that an individual valve can be tested in the same direction as that when the valve would be required to perform its safety function.

Continued leakage integrity is assured by the advanced design of the CNTS. The CNTS is a small, high pressure, ASME Class 1 vessel with a reduced number of penetrations. Manufacturing and preservice tests and inspections are similar to the RPV in a factory environment. Comprehensive ISI, applying ASME Class 1 criteria, ensures continued system integrity. All surface areas and welds are accessible for inspection. All penetration pathways will be tested to Type B or C criteria at Pa. These features ensure that continued leakage integrity of the CNTS is maintained.

6.2.6.2 Containment Penetration Leakage Rate Test

The CNV is designed for Type B pneumatic tests (local penetration leak tests) to detect and measure leakage across the pressure retaining, leakage limiting boundaries that include flange openings (bolted connections), flanges, I&C penetration assemblies, and electrical penetration assemblies. The leakage limiting boundary is pressurized with air or nitrogen and the pressure decay or the leak flow rate is measured.

Preoperational and periodic Type B leakage rate testing is performed in accordance with 10 CFR 50, Appendix J, NEI 94-01, and ANSI-56.8 within the defined test intervals. The containment penetrations subject to Type B tests are identified in Table 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
- electrical penetration assemblies
- ECCS trip/reset valve body-to-bonnet connection

All CNV bolted closures have dual o-ring seals a testing port between the seals.

All CNV flange openings have bolted connections are designed and constructed to ASME Class 1. All of these openings have an identical double o-ring seals with a test port to facilitate Type B testing by pressurizing between the seals. The main CNV flange has a similar double o-ring and test port arrangement.

Electrical penetration assemblies (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 installed in a CNV penetration which includes a testable flange connection. Installed EPAs are limited to local leak rate test acceptance criteria. The NuScale 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.

There are five ECCS main valves supported by eleven 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. This seal is both a containment and RCS boundary and will be Type B tested at RCS design pressure.

Type B tests are performed by local pressurization at containment peak accident pressure, Pa, using either the pressure-decay or flowmeter method of detection. For the pressure-decay method, a test volume is pressurized with air or nitrogen to at least Pa. The rate of decay of pressure in the known test volume is monitored to calculate a leakage rate using the pressure-decay method. For the flowmeter method, the required test pressure is maintained in the test volume by making up air or nitrogen, through a calibrated flowmeter. The flowmeter fluid flow rate is the leakage rate from the test volume.

The combined leakage rate for all penetrations and valves subject to Type B and C tests is limited to less than 0.60 La.

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

6.2.6.3 Containment Isolation Valve Leakage Rate Test

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

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

Isolation valves are tested using either the pressure decay or flowmeter method. For the pressure decay method the test volume is pressurized with air or nitrogen. The rate of decay of pressure in the known volume is monitored to calculate the leak rate. For the flowmeter method pressure is maintained in the test volume by supplying air or nitrogen through a calibrated flowmeter. The measured makeup flow rate is the isolation valve leak rate.

Pressure to the CIV is applied in the same direction as it the pressure applied when the valve is required to perform its safety function.

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

Each CIV to be tested is closed by normal means without any preliminary exercising or adjustment. Piping is drained and vented as needed and a test volume is established that, when pressurized, will produce a differential pressure across the valve. The valve is prepared for testing by removing the normal insert and replacing it with a test blank insert in the valve body (Figure 6.2-5, Figure 6.2-6a, and Figure 6.2-6b). The test blank is closed to the CNV to establish the pressure boundary for the test in the same direction as that when the valve would be required to perform its safety function. Test equipment is then installed on the test port that is located between the test blank and the inboard ball. The valves are aligned so that a vent path is established downstream of the tested valve. The valve is then tested on air or nitrogen using either the pressure decay or flowmeter method as specified in the owner's Appendix J program.

When valve testing is completed, the test equipment can be vented and the valves can be realigned. The tested valve is opened and the second CIV closed. The test alignment for the second CIV is now established. The test equipment can be re-pressurized and the second valve tested.

Isolation valves whose seats may be exposed to the containment atmosphere during a LOCA are tested with air or nitrogen at a pressure not less than Pa.

The leak rates of penetrations and valves subject to Type B and C testing are combined in accordance with 10 CFR 50, Appendix J. The combined leakage rate for all penetrations and valves subject to Type B and C tests shall be less than 0.60 La. If repairs are required to meet this limit, the results shall be reported in a separate summary to the NRC, to include the structural conditions of the components which contributed to the failure. As each Type B or C test, or group of tests, is completed the combined total leak rate is revised to reflect the latest results. Thus, a reliable current summary of containment leak tightness is maintained. Leak rate limits and the criteria for the combined leakage results are described in the plant Technical Specifications.

6.2.6.4 Scheduling and Reporting of Periodic Tests

Schedules for performance of the periodic Type B and C leak rate tests are specified in the plant Technical Specifications [Section 5.5.9]. Provisions for reporting test results are described in the containment leakage rate testing 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 does not exceed the maximum allowable interval specified in the containment leakage rate testing program.

Each time a Type B or C test is completed, the overall total leakage rate for all required Type B and C tests is updated to reflect the most recent test results. In addition to the periodic tests, any major modification or replacement of a component that is part of the reactor containment boundary performed after the preoperational leakage rate test is followed by either a Type B or C test (as applicable) for the area affected by the modification.

The leakage test summary report includes descriptions of the containment inspection method, any repairs necessary to meet the acceptance criteria, and the test results. The summary report includes periodic leakage test results from the Type B and C tests. Leakage test results from Type B and C tests that failed to meet the acceptance criteria are included in a separate accompanying summary report that includes an analysis and interpretation of the test data, the least squares fit analysis of the test data, the instrumentation error analysis, and the structural conditions of the containment or components, if any, that contributed to the failure in meeting the acceptance criteria. Results and analyses of the supplemental verification test employed to demonstrate the validity of the leakage rate test measurements are also included.

6.2.6.5 Special Testing Requirements

Major modifications or replacement of components that are part of the containment boundary performed after preoperational leakage rate testing is 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.

6.2.7 Fracture Prevention of Containment Vessel

The NuScale CNTS encloses the RPV providing a CNV pressure boundary and provides an essentially leak-tight final barrier against the release of radioactive fission products resulting from postulated accidents. Design of the steel containment is addressed in Section 3.8.2.

The CNV pressure boundary components include the CNV, all 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 any valves required to isolate the system and provide a pressure boundary for the containment function.

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 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, and the probability of rapidly propagating fracture will be minimized, under operating, maintenance, testing, and postulated accident conditions 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 ASME BPVC Section XI Appendix G apply.

The NuScale 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 for containment pressure boundary components fabricated of ferritic materials. Fracture prevention of the containment pressure boundary is assured.

Information addressing the integrity and fracture toughness of CNV bolting and threaded fasteners is addressed in Section 3.13.

Portions of the lower CNV have 60-year design fluence in excess of 1E+17 neutrons/cm2, E>1 MeV, with the peak fluence in the lower CNV not exceeding 5.5E+18 neutrons/cm2, E > 1 MeV. To remove neutron embrittlement concerns in the CNV, the lower CNV is made of austenitic stainless steels which are more resistant to neutron embrittlement than ferritic materials.

6.2.8 References

6.2-1 TR-0516-49084, Rev 0, Containment Response Analysis Methodology Technical Report

6.2-2 TR-0516-49422, Rev 0, "LOCA Evaluation Model," July 2016

6.2-3 TR-0716-50424, Rev 0, "Combustible Gas Control Technical Report," 6.2-4 TR-0516-49416, Rev. 0, "Non-LOCA Transient Analysis Methodology Report," January 2017

6.2-5 TR-0916-51299, Rev. 0, "Long Term Core Cooling Methodology". January 2017.

6.2-6 TR-1116-51962, Rev. 0, "NuScale Containment Leakage Integrity Assurance", December 2016.

6.2-7 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2013 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	1000 psia*
• External Design Pressure	
• Design Temperature	550 °F
• Design Maximum Containment Leakage	
• UHS Pool Water (Avg) Temperature	
• Reactor Building Air Temperature (°F)**	65 °F - 85 °F
Normal Operating Conditions (nominal)	
• Internal CNV Pressure	less than 0.1
• External CNV Pressure	60
• CNV Temperature (Atmosphere)	
• UHS Pool Water Level (See Figure 9.1.3-5)	68 - 69 ft (pool level) 93 - 94 ft (building elevation)
• UHS Pool Water Volume - from normal operating level (69 ft) to top of weir (45 ft) as shown in Figure 9.1.3-5	4 million gallons
• UHS Pool Water (Avg) Temperature	
• Reactor Building Air Temperature (°F)	75 ±10 °F
• Lowest Service Temperature	40°F

* Hydrostatically tested at 1250 psig (at highest point of test boundary)/70°F (minimum) to 140°F (maximum)

** Additional Reactor Building 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 Break	Base Case	884	499
RCS Discharge Break	Limiting Sensitivity Case Results	906	502
RCS Injection Line Break	Base Case	904	518
RCS Injection Line Break	Limiting Sensitivity Case Results	921	523[2]
Pressurizer Spray Supply Line Break	Base Case	586	476
Pressurizer Spray Supply Line Break	Limiting Sensitivity Case Results	864	484
Inadvertent RVV Actuation	Base Case	802	481
Inadvertent RVV Actuation	Limiting Sensitivity Case Results	880	482
Inadvertent RRV Actuation	Base Case	933	495
Inadvertent RRV Actuation	Limiting Sensitivity Case Results	951[1]	506
Main Steam Line Break	Limiting Results	419	427
Feedwater Line Break	Limiting Results	442	414

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

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

Table 6.2-3: Containment Vessel Inspection Elements

Component Description	Examination Category	Examination Method	Notes
CNV Shell and Head Welds			
Bottom head to core region shell	B-A	Volumetric	
Core region shell to lower transition shell	B-A	Volumetric	
Lower transition shell to lower flange	B-A	Volumetric	
Upper flange to RPV support lug shell	B-A	Volumetric	
RPV support lug shell to steam plenum access shell	B-A	Volumetric	
Steam plenum access shell to upper shell	B-A	Volumetric	
Upper shell to top head	B-A	Volumetric	
Support Welds			
Support skirt ring to support skirt	F-A	Visual	
Support skirt to bottom head	B-K	Surface or volumetric	
CNV shipping/storage support lug	B-K	Surface	
CNV support lug	B-A	Volumetric	
Alignment pin to threaded insert	B-K	Surface	
Instrument enclosure base to CNV	None	None	Exempted by IWF-1230
CNV Nozzle to Shell Welds			
Feedwater lines	B-D	Volumetric	
Main steam lines	B-D	Volumetric	
CRDS return line	B-D	Volumetric	
CVCS makeup line	B-D	Volumetric	
CVCS pressurizer spray line	B-D	Volumetric	
I&C Divisions	B-D	Not required	See Table IWB-2500-1 (B-D) Note 1
Containment evacuation system line	B-D	Volumetric	
Containment flood and drain system line	B-D	Volumetric	
CRDS supply line	B-D	Volumetric	
CVCS letdown line	B-D	Volumetric	
RPV high point degasification line	B-D	Volumetric	
Pressurizer heater power (Elect-1 and 2)	B-D	Not required	See Table IWB-2500-1 (B-D) Note 1
I&C channels A-D	B-D	Not required	See Table IWB-2500-1 (B-D) Note 1
Decay heat removal system lines	B-D	Volumetric	Single sided, shell side

Table 6.2-3: Containment Vessel Inspection Elements (Continued)

Component Description	Examination Category	Examination Method	Notes
Head manway	B-D	Not required	See Table IWB-2500-1 (B-D) Note 1
CRDM access opening	B-D	Not required	See Table IWB-2500-1 (B-D) Note 1
CNV manway	B-D	Not required	See Table IWB-2500-1 (B-D) Note 1
SG inspection ports	B-D	Not required	See Table IWB-2500-1 (B-D) Note 1
Pressurizer heater access ports	B-D	Not required	See Table IWB-2500-1 (B-D) Note 1
RRV and RVV trip/reset	B-D	Not required	See Table IWB-2500-1 (B-D) Note 1
CRDM power	B-D	Not required	See Table IWB-2500-1 (B-D) Note 1
RPI groups	B-D	Not required	See Table IWB-2500-1 (B-D) Note 1
Nozzle-to-Safe-end Dissimilar Metal Welds (SE)			
Feedwater lines SE (inner and outer)	B-F	Surface and Volumetric	
Main steam lines SE (inner and outer)	B-F	Surface and Volumetric	
CRDS return line SE (outer)	B-F	Surface and Volumetric	
CRDS return lines SE (inner)	B-F	Surface	
CVCS makeup line SE (outer)	B-F	Surface and Volumetric	
CVCS makeup line SE (inner)	B-F	Surface	
CVCS pressurizer spray line SE (outer)	B-F	Surface and Volumetric	
CVCS pressurizer spray line SE (inner)	B-F	Surface	
Containment evacuation system line SE	B-F	Surface	
Containment flood and drain system line SE (inner and outer)	B-F	Surface	
CRDS supply line SE (inner and outer)	B-F	Surface	
CVCS letdown line SE (inner and outer)	B-F	Surface	
RPV high point degasification line SE (inner and outer)	B-F	Surface	
Decay heat removal system lines SE (inner and outer)	B-F	Surface	
RRV and RVV trip/reset SE	B-F	Surface	
Bolting			
CNV main flange bolts	B-G-1	See Notes	Per Note 7 of B-G-1, surface examination is permitted when bolts are removed.
CNV bolting two inches or less in diameter	B-G-2	VT-1	Examine if removed.

Table 6.2-4: Containment Penetrations

Containment Penetration	System/Component	Nominal Size (Opening)	CIV Connection (Nozzle)	Location (elevation) Note 1	Valve Type/ Operator or Closure	Appendix J Type B/C Test	Process Fluid/ Gas	Closure Time	Isolation Signal	Valve Position Norm/ Shutdown	Valve Position Post-Accident/ Power Failure	Piping Penetration Regulatory Reference/ Compliance
CNV1	Feedwater Line 1	NPS 5	NPS 5	CNV Head	Ball/Hydraulic	TS, DHRS operability	Water	5 sec	2 sec	Open/Closed	Closed/Closed	Meets GDC 57
CNV2	Feedwater Line 2	NPS 5	NPS 5	CNV Head	Ball/Hydraulic	TS, DHRS operability	Water	5 sec	2 sec	Open/Closed	Closed/Closed	Meets GDC 57
CNV3	Main Steam Line 1	NPS 12	NPS 12	CNV Head	Ball/Hydraulic	TS, DHRS operability	Steam	5 sec	2 sec	Open/Closed	Closed/Closed	Meets intent of GDC57 Exemption justified - Note 2
CNV4	Main Steam Line 2	NPS 12	NPS 12	CNV Head	Ball/Hydraulic	TS, DHRS operability	Steam	5 sec	2 sec	Open/Closed	Closed/Closed	Meets intent of GDC57 Exemption justified - Note 2
CNV5	CRDS Return	NPS 2	NPS 4	CNV Head	Ball/Hydraulic	C	Water	5 sec	2 sec	Open/Closed	Closed/Closed	Meets the intent of GDC 56 Exemption justified - Note 3
CNV6	CVCS Makeup	NPS 2	NPS 4	CNV Head	Ball/Hydraulic	C	Borated Water	5 sec	2 sec	Open/Closed	Closed/Closed	Meets the intent of GDC 55 Exemption justified - Note 3
CNV7	CVCS Pressurizer Spray	NPS 2	NPS 4	CNV Head	Ball/Hydraulic	C	Borated Water	5 sec	2 sec	Open/Closed	Closed/Closed	Meets the intent of GDC 55 Exemption justified - Note 3
CNV8	I&C Division 1	NPS 3	N/A	CNV Head	Flange	B	N/A	N/A	N/A	N/A	N/A	N/A
CNV9	I&C Division 2	NPS 3	N/A	CNV Head	Flange	B	N/A	N/A	N/A	N/A	N/A	N/A
CNV10	Containment Evacuation System	NPS 2	NPS 4	CNV Head	Ball/Hydraulic	C	Non-condensables, Gas, Vapor	5 sec	2 sec	Open/Closed	Closed/Closed	Meets the intent of GDC 56 Exemption justified - Note 3
CNV11	Containment Flood and Drain System	NPS 2	NPS 4	CNV Head	Ball/Hydraulic	C	Borated Water	5 sec	2 sec	Closed/Closed	Closed/Closed	Meets the intent of GDC 56 Exemption justified - Note 3
CNV12	CRDS Supply	NPS 2	NPS 4	CNV Head	Ball/Hydraulic	C	Water	5 sec	2 sec	Open/Closed	Closed/Closed	Meets the intent of GDC 56 Exemption justified - Note 3
CNV13	CVCS Letdown	NPS 2	NPS 4	CNV Head	Ball/Hydraulic	C	Borated Water	5 sec	2 sec	Open/Closed	Closed/Closed	Meets the intent of GDC 55 Exemption justified - Note 3
CNV14	RPV High Point Degasification	NPS 2	NPS 4	CNV Head	Ball/Hydraulic	C	Non-condensables, Gas, Vapor	5 sec	2 sec	Closed/Closed	Closed/Closed	Meets the intent of GDC 55 Exemption justified - Note 3
CNV15	Pressurizer Heater Power (Elect-1)	NPS 12	N/A	CNV Head	Flange	B	N/A	N/A	N/A	N/A	N/A	N/A
CNV16	Pressurizer Heater Power (Elect-2)	NPS 12	N/A	CNV Head	Flange	B	N/A	N/A	N/A	N/A	N/A	N/A
CNV17	I&C Channel A	NPS 8	N/A	CNV Head	Flange	B	N/A	N/A	N/A	N/A	N/A	N/A
CNV18	I&C Channel B	NPS 8	N/A	CNV Head	Flange	B	N/A	N/A	N/A	N/A	N/A	N/A
CNV19	I&C Channel C	NPS 8	N/A	CNV Head	Flange	B	N/A	N/A	N/A	N/A	N/A	N/A
CNV20	I&C Channel D	NPS 8	N/A	CNV Head	Flange	B	N/A	N/A	N/A	N/A	N/A	N/A
CNV21	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
CNV22	Decay Heat Removal 1	NPS 2		377″	Closed piping - Note 2	N/A	Borated Water	N/A	N/A	N/A	N/A	Meets the intent of GDC 57 Exemption justified - Note 2
CNV23	Decay Heat Removal 2	NPS 2		377″	Closed piping - Note 2	N/A	Borated Water	N/A	N/A	N/A	N/A	Meets the intent of GDC 57 Exemption justified - Note 2

Notes:
1 Containment isolation valves are located outside of the containment vessel. Elevations are approximate as measured from global zero to top of safe-end or nozzle cover.
2 Exemption allows the use of a closed piping system (DHRS) outside of the containment vessel rather than providing an isolation valve.
3 Exemption allows placement of both containment isolation valves outside of the containment boundary.
4 Borated water flows to and from each pilot valve remain within the containment boundary.

Table 6.2-5: Containment Isolation Valve Design Information

Containment Isolation Valve	Process System Inside of Containment	Process System Outside of Containment	Penetration	Valve Type	Seismic Category	Quality Group (Valve/ Actuator)	ASME Design Code	Pressure Class	OM Category	Design Pressure	Design Temperature
CVC-ISV-0331	RCS Injection	CVCS Makeup	CNV6	Ball/PSCIV	I	A/B	Section III, NB-2000	ANSI B16.34	A	2100 psia	650°F
CVC-ISV-0329	RCS Injection	CVCS Makeup	CNV6	Ball/PSCIV	I	A/B	Section III, NB-2000	ANSI B16.34	A	2100 psia	650°F
CVC-ISV-0325	RCS Pressurizer Spray	CVCS Pressurizer Spray	CNV7	Ball/PSCIV	I	A/B	Section III, NB-2000	ANSI B16.34	A	2100 psia	650°F
CVC-ISV-0323	RCS Pressurizer Spray	CVCS Pressurizer Spray	CNV7	Ball/PSCIV	I	A/B	Section III, NB-2000	ANSI B16.34	A	2100 psia	650°F
CVC-ISV-0336	RCS Discharge	CVCS Letdown	CNV13	Ball/PSCIV	I	A/B	Section III, NB-2000	ANSI B16.34	A	2100 psia	650°F
CVC-ISV-0334	RCS Discharge	CVCS Letdown	CNV13	Ball/PSCIV	I	A/B	Section III, NB-2000	ANSI B16.34	A	2100 psia	650°F
CVCS-ISV-0403	RCS High Point Degasification	Pressurizer High Point Degasification	CNV14	Ball/PSCIV	I	A/B	Section III, NB-2000	ANSI B16.34	A	2100 psia	650°F
CVCS-ISV-0401	RCS High Point Degasification	Pressurizer High Point Degasification	CNV14	Ball/PSCIV	I	A/B	Section III, NB-2000	ANSI B16.34	A	2100 psia	650°F
CE-ISV-0101	Containment Evacuation System	Containment Evacuation System	CNV10	Ball/PSCIV	I	B/B	Section III, NB-2000	ANSI B16.34	A	2100 psia	650°F
CE-ISV-0102	Containment Evacuation System	Containment Evacuation System	CNV10	Ball/PSCIV	I	B/B	Section III, NB-2000	ANSI B16.34	A	2100 psia	650°F
CFD-ISV-0130	Containment Flood and Drain System	Containment Flood and Drain System	CNV11	Ball/PSCIV	I	B/B	Section III, NB-2000	ANSI B16.34	A	2100 psia	650°F
CFD-ISV-0129	Containment Flood and Drain System	Containment Flood and Drain System	CNV11	Ball/PSCIV	I	B/B	Section III, NB-2000	ANSI B16.34	A	2100 psia	650°F
RCCW-ISV-0185	CRDS Supply	CRDS Supply	CNV12	Ball/PSCIV	I	B/B	Section III, NB-2000	ANSI B16.34	A	2100 psia	650°F
RCCW-ISV-0184	CRDS Supply	CRDS Supply	CNV12	Ball/PSCIV	I	B/B	Section III, NB-2000	ANSI B16.34	A	2100 psia	650°F
RCCW-ISV-0190	CRDS Return	CRDS Return	CNV5	Ball/PSCIV	I	B/B	Section III, NB-2000	ANSI B16.34	A	2100 psia	650°F
RCCW-ISV-0191	CRDS Return	CRDS Return	CNV5	Ball/PSCIV	I	B/B	Section III, NB-2000	ANSI B16.34	A	2100 psia	650°F
FW-ISV-1003	SGS Feed	Feedwater 1	CNV1	Ball/SSCIV	I	B/B	Section III, NC-2000	ANSI B16.34	A	2100 psia	650°F
FW-ISV-2003	SGS Feed	Feedwater 2	CNV2	Ball/SSCIV	I	B/B	Section III, NC-2000	ANSI B16.34	A	2100 psia	650°F
ISV-1005	SGS Steam 1	Main Steam 1	CNV3	Ball/SSCIV	I	B/B	Section III, NC-2000	ANSI B16.34	A	2100 psia	650°F
ISV-1006	SGS Steam 1	Main Steam 1 Bypass	CNV3	Ball/SSCIV	I	B/B	Section III, NC-2000	ANSI B16.34	A	2100 psia	650°F
ISV-2005	SGS Steam 2	Main Steam 2	CNV4	Ball/SSCIV	I	B/B	Section III, NC-2000	ANSI B16.34	A	2100 psia	650°F
ISV-2006	SGS Steam 2	Main Steam 2 Bypass	CNV4	Ball/SSCIV	I	B/B	Section III, NC-2000	ANSI B16.34	A	2100 psia	650°F

PSCIV - primary system containment isolation valve - see Figure 6.2-5

SSCIV - secondary system containment isolation valve - see Figure 6.2-6a and Figure 6.2-6b

Figure 6.2-1: Containment System

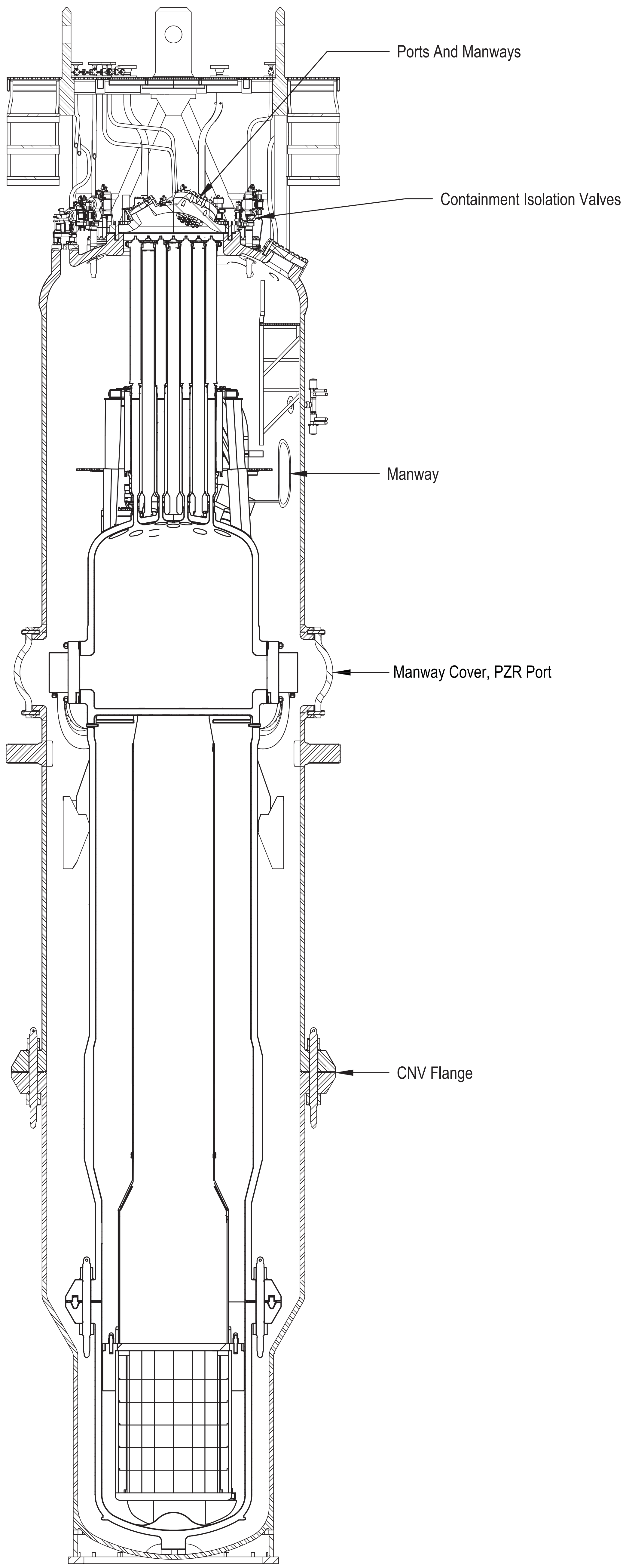
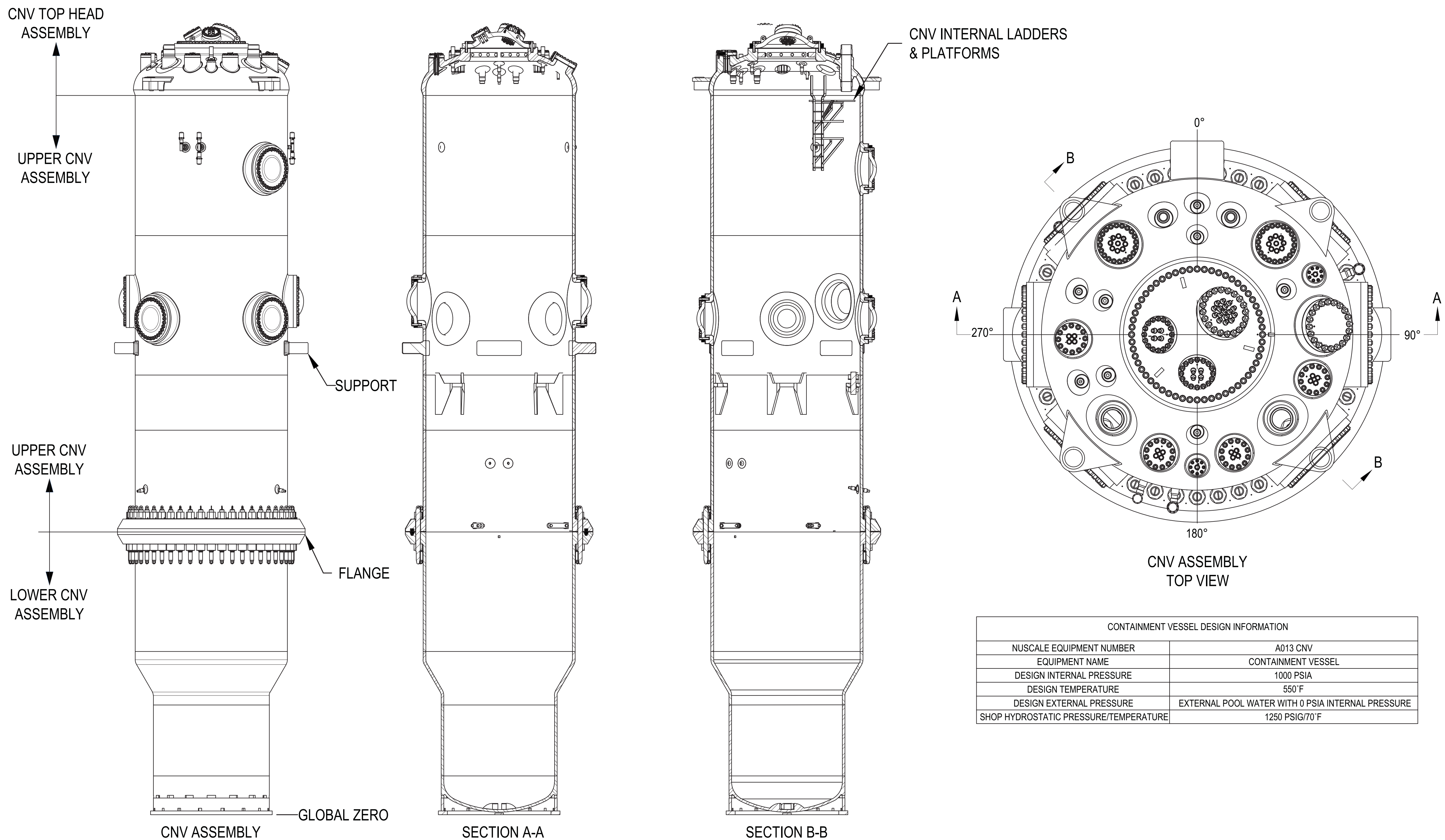


Figure 6.2-2a: Containment Vessel Assembly



CONTAINMENT VESSEL DESIGN INFORMATION	
NUSCALE EQUIPMENT NUMBER	A013 CNV
EQUIPMENT NAME	CONTAINMENT VESSEL
DESIGN INTERNAL PRESSURE	1000 PSIA
DESIGN TEMPERATURE	550°F
DESIGN EXTERNAL PRESSURE	EXTERNAL POOL WATER WITH 0 PSIA INTERNAL PRESSURE
SHOP HYDROSTATIC PRESSURE/TEMPERATURE	1250 PSIG/70°F

Figure 6.2-2b: Containment Vessel Assembly

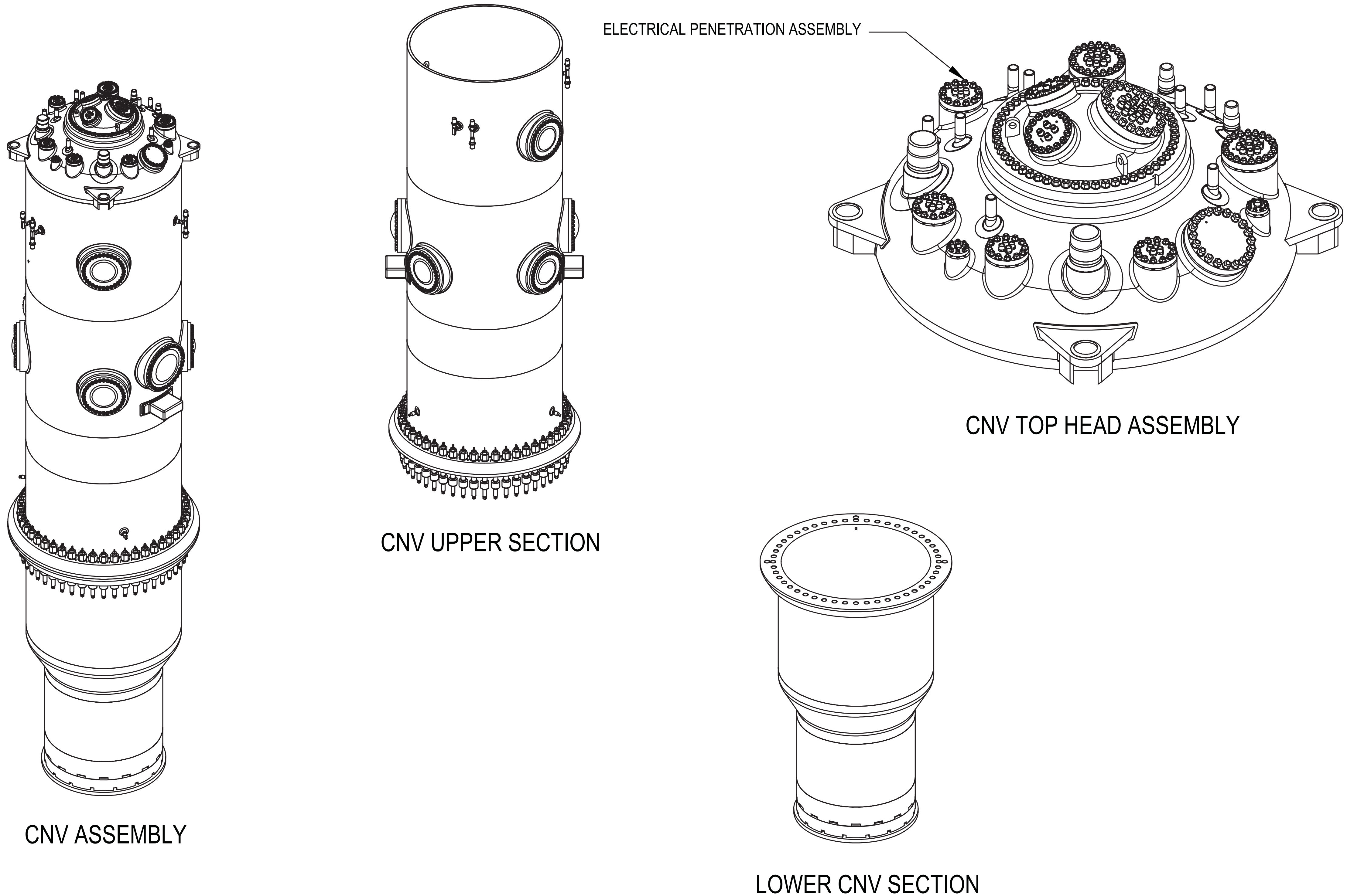
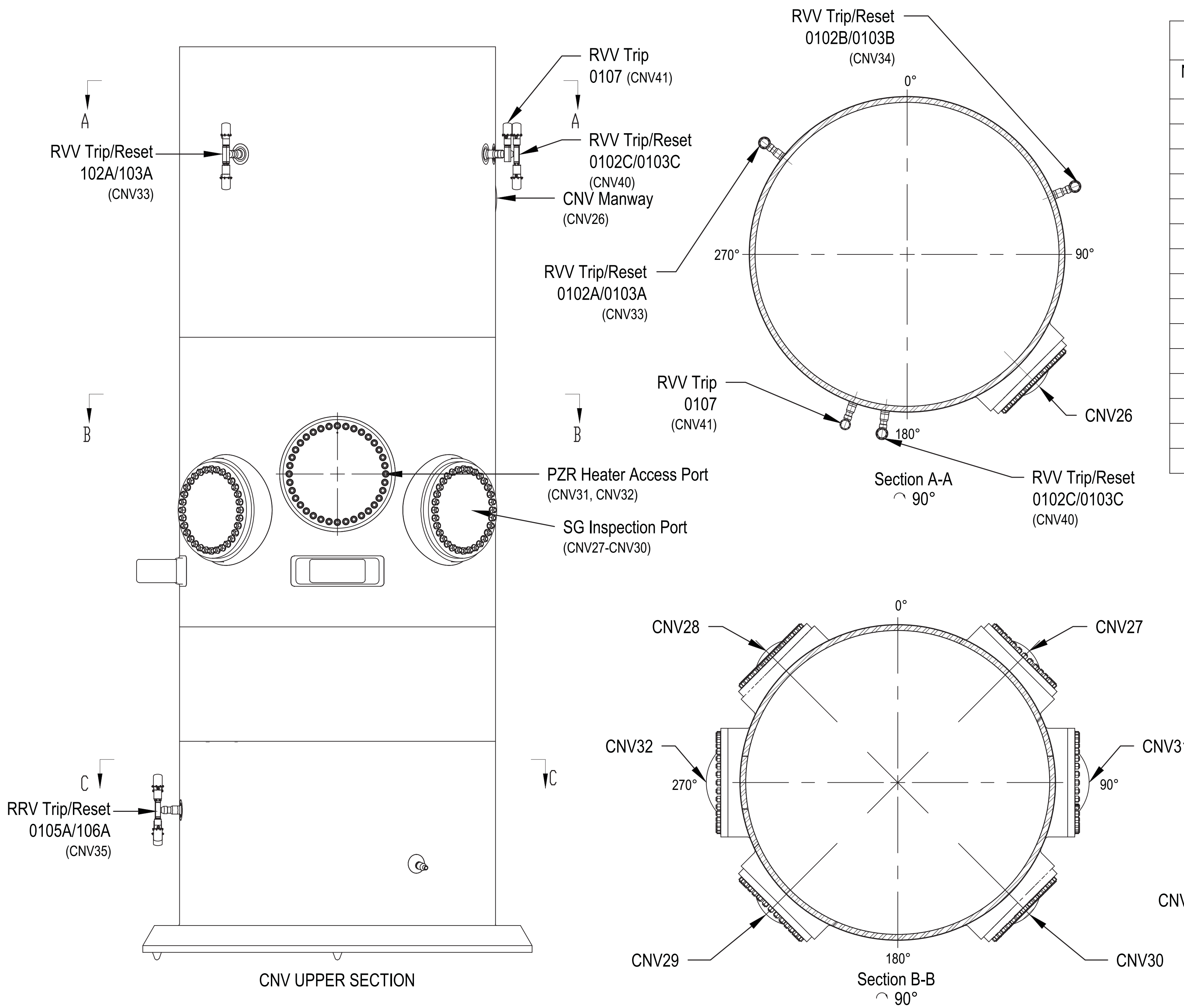


Figure 6.2-3a: Containment Vessel Penetrations



CNV NOZZLE SCHEDULE		
NOZZLE No.	NOZZLE OPENING OR NOMINAL SIZE	DESCRIPTION
CNV22	NPS 2	DECAY HEAT REMOVAL 1
CNV23	NPS 2	DECAY HEAT REMOVAL 2
CNV26	38	CNV MANWAY
CNV27	38	SG INSPECTION PORT 1
CNV28	38	SG INSPECTION PORT 2
CNV29	38	SG INSPECTION PORT 3
CNV30	38	SG INSPECTION PORT 4
CNV31	44	PZR HEATER ACCESS PORT 1
CNV32	44	PZR HEATER ACCESS PORT 2
CNV33	NPS 3	RVV TRIP/ RESET 0102A/0103A
CNV34	NPS 3	RVV TRIP/ RESET 0102B/0103B
CNV35	NPS 3	RRV TRIP/ RESET 0105A/0106A
CNV36	NPS 3	RRV TRIP/ RESET 0105B/0106B
CNV40	NPS 3	RVV TRIP/ RESET 0102C/0103C
CNV41	NPS 3	RVV TRIP 0107

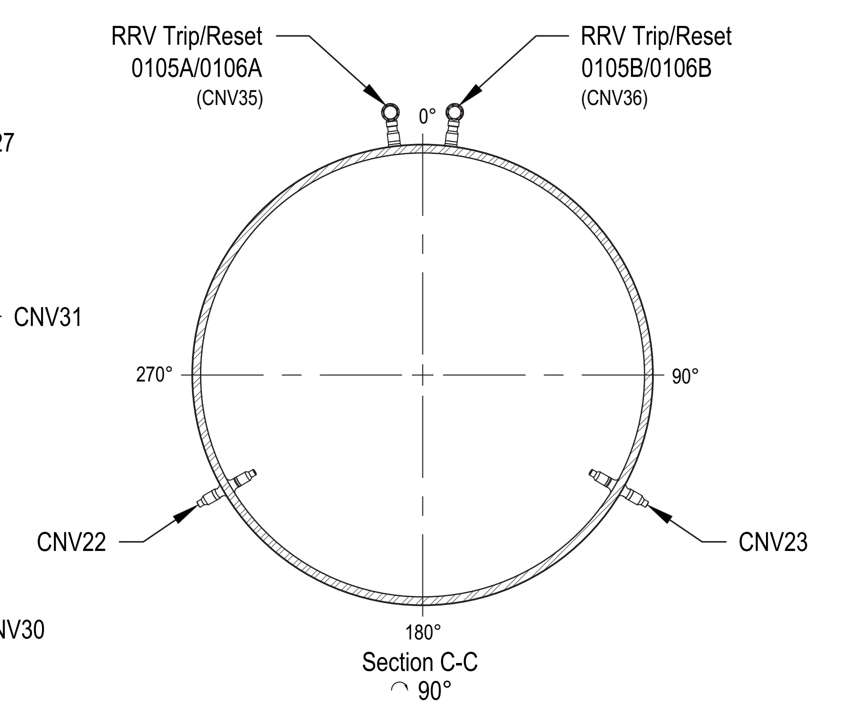
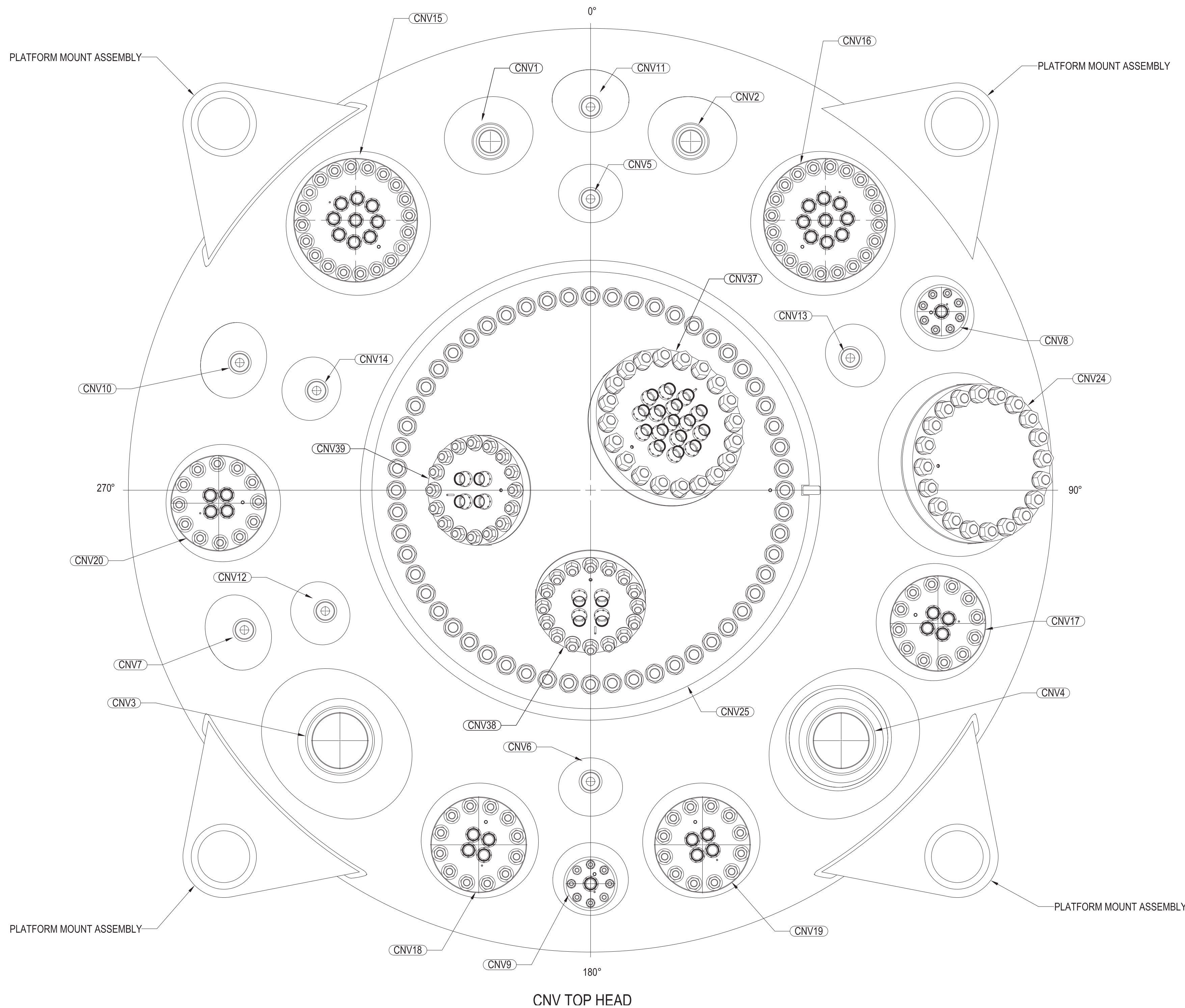


Figure 6.2-3b: Containment Vessel Penetrations



CNV NOZZLE SCHEDULE

NOZZLE NO.	CONTAINMENT ISOLATION VALVE CONNECTION SIZE	FLUID LINE OR OPENING SIZE	DESCRIPTION
CNV1	NPS 5	NPS 5	FEEDWATER LINE 1
CNV2	NPS 5	NPS 5	FEEDWATER LINE 2
CNV3	NPS 12	NPS 12	MAIN STEAM LINE 1
CNV4	NPS 12	NPS 12	MAIN STEAM LINE 2
CNV5	NPS 4	NPS 2	RCCW RETURN
CNV6	NPS 4	NPS 2	CVC INJECTION
CNV7	NPS 4	NPS 2	CVC PRESSURIZER
CNV8	N/A	NPS 3	I&C - DIVISION 1
CNV9	N/A	NPS 3	I&C - DIVISION 2
CNV10	NPS 4	NPS 2	CONTAINMENT EVACUATION
CNV11	NPS 4	NPS 2	CONTAINMENT FLOODING & DRAINING
CNV12	NPS 4	NPS 2	RCCW SUPPLY
CNV13	NPS 4	NPS 2	CVC DISCHARGE
CNV14	NPS 4	NPS 2	RPV HIGH POINT DEGASIFICATON
CNV15	N/A	NPS 12	ELECTRICAL 1 - PZR HEATER POWER
CNV16	N/A	NPS 12	ELECTRICAL 2 - PZR HEATER POWER
CNV17	N/A	NPS 8	I&C - CHANNEL A
CNV18	N/A	NPS 8	I&C - CHANNEL B
CNV19	N/A	NPS 8	I&C - CHANNEL C
CNV20	N/A	NPS 8	I&C - CHANNEL D
CNV24	N/A	NPS 18	CNV HEAD MANWAY
CNV25	N/A	67	CRDM ACCESS OPENING
CNV37	N/A	NPS 18	CRDM POWER
CNV38	N/A	NPS10	RPI GROUP 1
CNV39	N/A	NPS 10	RPI GROUP 2

Figure 6.2-4: Containment System (Isolation Valves) Piping and Instrumentation Diagram

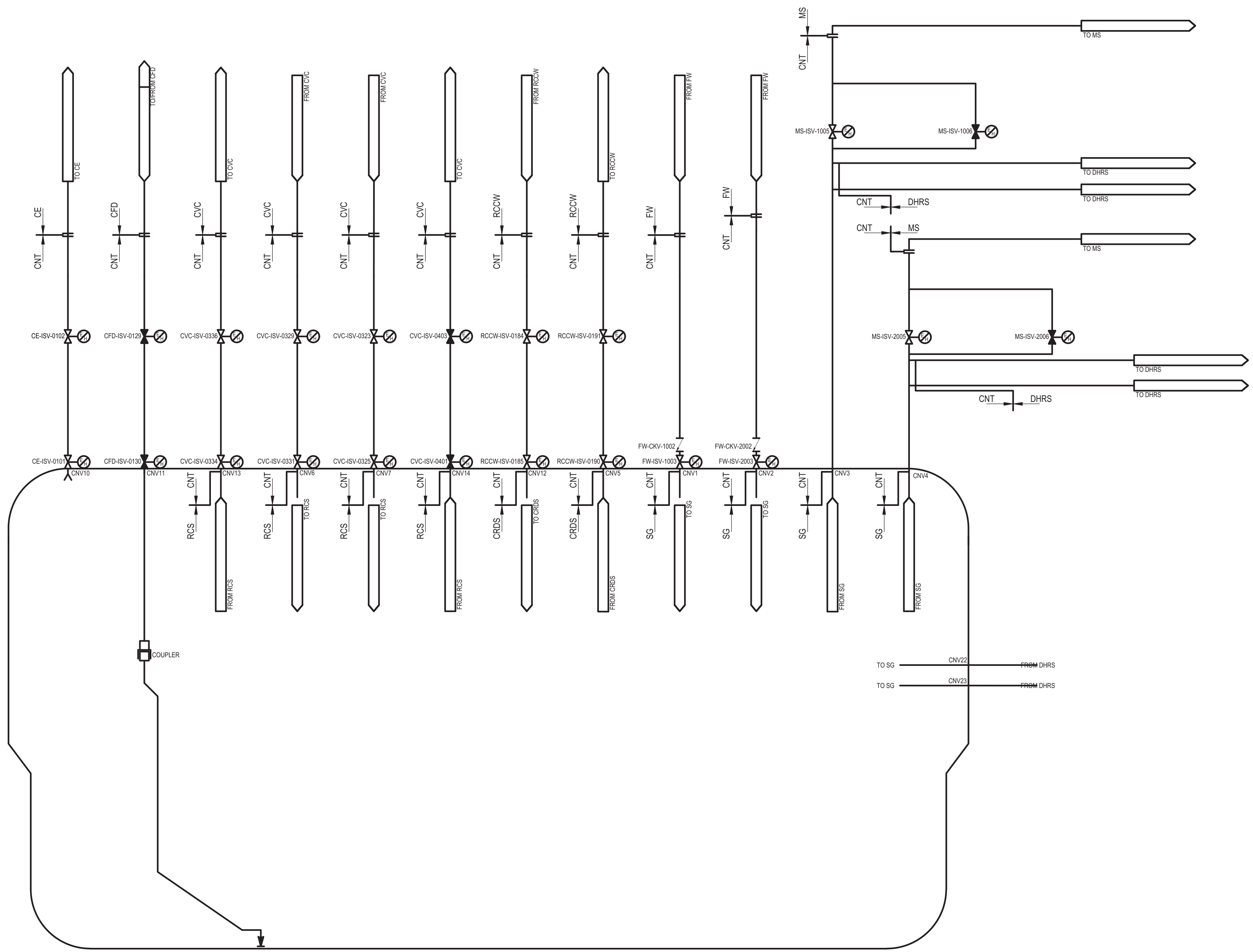
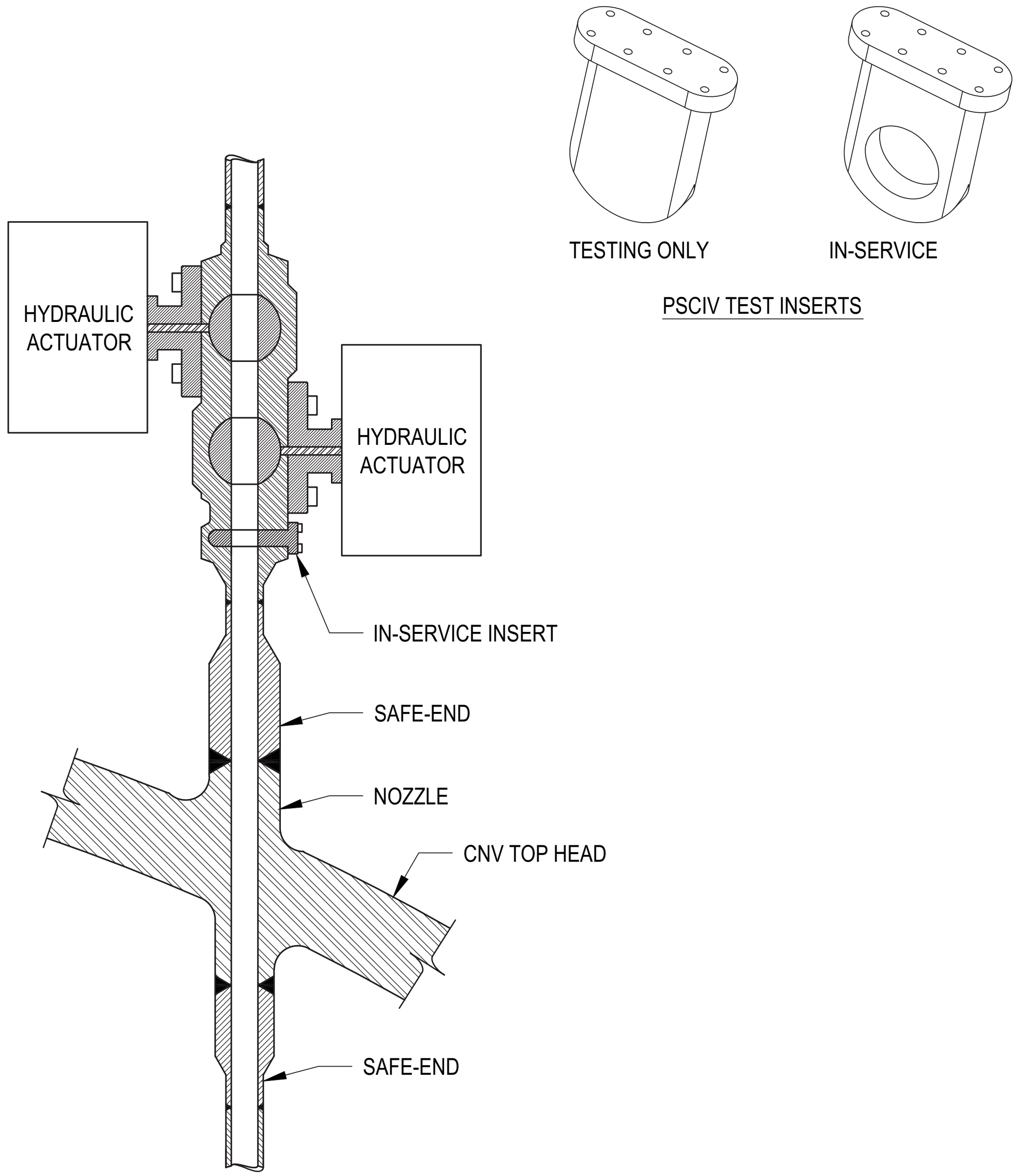
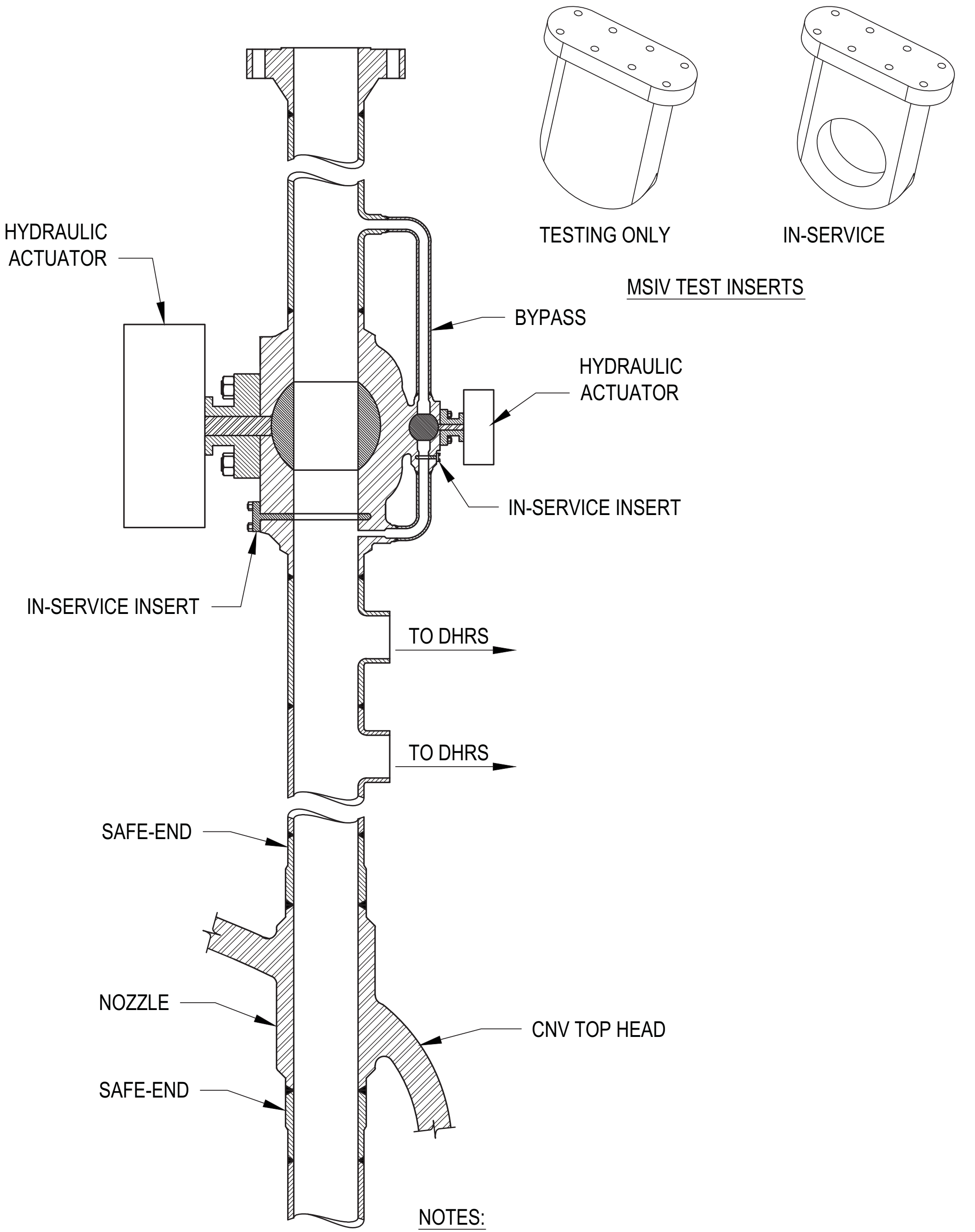


Figure 6.2-5: Primary System Containment Isolation Valves Dual Valve, Single Body Design



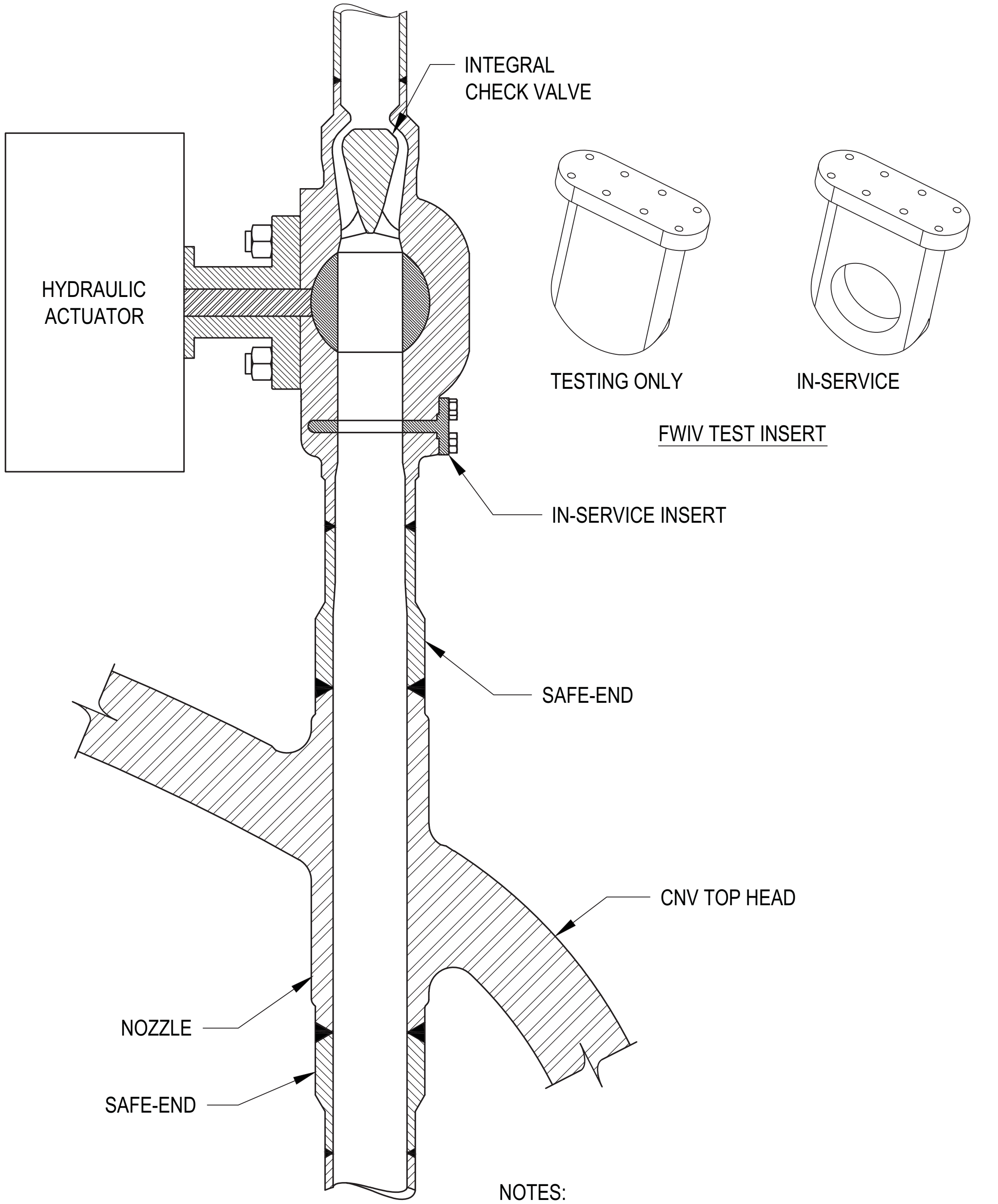
NOTES:
- VALVES SHOWN IN NORMAL OPERATING POSITIONS
- NOT TO SCALE
- COMPONENTS ROTATED FOR CLARITY

Figure 6.2-6a: Main steam Isolation Valve with Bypass Valve and Actuator Assembly



NOTES:
- VALVES SHOWN IN NORMAL OPERATING POSITIONS
- NOT TO SCALE
- COMPONENTS ROTATED AND REPOSITIONED FOR CLARITY

Figure 6.2-6b: Feedwater Isolation Valve with Nozzle Check Valve and Actuator Assembly



NOTES:
- VALVES SHOWN IN NORMAL OPERATING POSITIONS
- NOT TO SCALE
- COMPONENTS ROTATED FOR CLARITY

6.3 Emergency Core Cooling System

The emergency core cooling system (ECCS) provides core cooling during and after anticipated operational occurrences (AOOs) 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 loss of coolant that exceeds makeup capability.

The ECCS consists of three 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. All five valves are closed during normal plant operation and open to actuate the system during applicable accident conditions. 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 is actuated, the coolant level in the RPV is maintained 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.

After actuation, 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 are maintained in the closed position during normal plant operation and receive an actuation signal upon predetermined event conditions (initiated by low RPV level or high containment level) to depressurize the RPV and allow flow of reactor coolant between the CNV and the RPV.

Reactor coolant inventory released during a LOCA event is collected and retained within the CNV which precludes the requirement to provide the makeup capacity necessary to replace coolant inventory lost to the core cooling function. The ECCS does not provide replacement or addition of inventory from an external source and does not provide a reactivity control function.

Facility design relies on passive design provisions that ensure sufficient coolant inventory is retained in the module to maintain the core covered and cooled. Makeup (addition) of reactor coolant inventory is not necessary or relied upon to protect against breaks. Reactor coolant inventory released from the reactor vessel during an in-containment unisolatable LOCA is collected and maintained within the CNV. After the ECCS valves open, the collected RCS inventory is returned to the reactor vessel by natural circulation. This return path to the vessel ensures that the core remains covered. The coolant inventory maintained in the reactor coolant system (RCS), assuming minimum allowed pressurizer (PZR) level, is adequate to provide sufficient coolant level in the CNV during a postulated design basis LOCA and maintain the reactor core covered. This is further discussed in Section 6.2.2.

The reactor core is covered at all times during ECCS operation. The analyzed loss-of-coolant events do not result in periods of refill or reflood and the design does not include or require pumps for forced circulation. Actuation and operation of the ECCS establishes a natural circulation path whereby coolant heated in the core leaves as steam or vapor through the reactor vents and condenses in the CNV, where it is collected and retained for return to the RPV.

The RRV penetrations are oriented horizontally on opposite sides of the RPV shell about six feet above the top of the reactor core. The locations of the ECCS valves are shown in Figure 6.3-2, a schematic depicting ECCS conditions (accident conditions). During ECCS operation, all five ECCS valves are open and the water levels in the concentric vessels (RPV and CNV) stabilize above the reactor core.

The ECCS valves are designed to 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 five valves 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, and fuel and cladding damage that could interfere with long-term effective core cooling is prevented. The ECCS is capable of providing adequate core cooling with two RVVs and one RRV in the open position. The cooling and associated depressurization functions are accomplished using safety-related equipment.

The functional requirements and system performance requirements of 10 CFR 50.46 are satisfied for the postulated LOCA conditions. The 10 CFR 50.46 requirements are described in more detail in Section 6.3.3.

The ECCS is designed such that no single failure event prevents the ECCS from performing its safety function including electrical power (normal AC or DC power availability, busses, electrical and mechanical parts, cabinets and wiring), initiation logic, and single active or passive component failure.

To ensure that RPV pressure-temperature limits are not exceeded, the ECCS provides LTOP when RPV temperatures are less than 325 degrees F. The analytical limit for this LTOP function is a variable pressure value determined by the calculated saturation pressure for the RPV cold leg temperature plus 250 degrees F. When the wide range RCS cold temperature is below 162 degrees F, the limit is fixed at a minimum pressure of 350 psia. The module protection system (MPS) logic provides the actuation signal that opens the RVVs. Once the wide range RCS cold temperature exceeds 325 degrees F during startup, the LTOP function is disabled. A more detailed discussion of the LTOP function is provided in Chapter 5.

Long-term cooling requirements that call for the removal of decay heat via the passive containment heat removal function are discussed in Section 6.2.

The ECCS meets General Design Criterion (GDC) 2 requirements as 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 provided in Section 3.7 and conformance with Regulatory Guide (RG) 1.29, Revision 5, is addressed in Section 1.9. The ECCS is designed to withstand the effects of natural phenomenon and designed to be structurally robust to ensure that its safety function is preserved in the event of a safe shutdown earthquake. Components of the ECCS are also protected from physical damage due to their installation inside the containment pressure boundary and are not subject to pipe whip and internal missiles. Missile protection and the effects of pipe whip are addressed in Section 3.5 and Section 3.6, respectively.

The ECCS main valves are protected from fire by their physical location within the CNV. The valve actuators are attached to the exterior surface of the CNV shell (underwater) and contained within the defined containment pressure boundary.

The ECCS meets the intent of GDC 4 with respect to the environmental and dynamic effects associated with the normal operation, maintenance, testing, and postulated accidents. The components are not subject to flow instabilities and loads (e.g., water hammer) because the natural circulation on which system operation is based is not subject to these instabilities. Protection of ECCS components against the dynamic effects resulting from valve discharge or equipment failures inside containment is further discussed in Section 3.6 and Section 4.6.2. Protection from the dynamic effects of events and conditions outside the NuScale Power Module (NPM) are provided by their physical location within the containment pressure boundary. Additional information addressing the environmental qualifications of the ECCS components is provided 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 (UHS), is common to all NPMs. Compliance with GDC 5 is described in further detail in Section 9.2.

The ECCS meets the regulatory requirements of GDC 14 in that those portions of the ECCS that are part of the RCPB are designed, fabricated, and tested to have an extremely low probability of abnormal leakage, rapidly propagating failure, or gross failure. Component descriptions with applicable design codes and classifications are further discussed in Section 6.3.2.

The ECCS design does not require alternating current (AC) or direct current (DC) power to effectively cool the core. The ECCS provides a safety-related passive system designed to maintain core cooling and containment integrityindependent of AC or DC power sources, by requiring SSC to transition to their safety state upon loss of control power or motive power. Because the electric power systems have no safety function and are not important to safety in the facility design, the intent of GDC 17 is met.

The NPM does not rely on poison addition by the ECCS for reliably controlling reactivity changes. Compliance with principal design criterion 27, “Combined Reactivity Systems Capability” is further discussed in Section 3.1.3.

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 the components are designed, fabricated, and tested to the highest quality standards practical, and designed with sufficient margin to ensure that the RCPB behaves in a non-brittle manner and the probability of a rapidly propagating failure is minimized. Component descriptions with the applicable design codes and classifications are further discussed in Section 6.3.2.

Facility design meets the regulatory requirements of principal design criterion 35, and GDCs 36 and 37 as they relate to the ECCS being designed to provide 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 is provided to ensure the system safety function can be accomplished assuming the single failure criteria.

The MPS provides the capability to perform periodic pressure and functional testing of the ECCS that ensures operability and performance of system components and the operability and performance of the system as a whole. This is further discussed in Section 7.2.15.

The ECCS meets the intent of the regulatory requirements of GDC 50, 51, 52 and 53 for those portions of the ECCS that are part of the containment pressure boundary in that the components are designed such that they 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 property, stresses (residual, steady state and transient), and flaw size. The components are designed to accommodate the required containment integrated leakage rate and periodic inspection, surveillance, and penetration testing requirements. ECCS components that are part of the containment boundary are capable of withstanding the effects of hydrogen combustion without loss of containment integrity. Compliance with 10CFR 50.44 is further discussed in Section 6.2.5.

The ECCS meets the intent of 10 CFR 50.34(f)(1)(vii) (TMI Action Plan II.K.3.18 of NUREG 0737) in that ECCS is automatically initiated and manual operator action is not necessary 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.

Environmental qualification information for the ECCS is addressed in Section 3.11.

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, the generation of radioactive waste and facilitation of facility decommissioning are minimized by limiting operation of ECCS to the RPV and CNV components.

Facility design meets the intent of 10 CFR 50.34(f)(2)(xi) (TMI Action Plan II.D.3 of NUREG 0737) in that valve position indication is provided in the control room for the ECCS valves, trip and reset actuator valves, and the reactor safety valves. In addition, solenoid power indication for the ECCS trip and reset valves is provided in the control room.

Facility design meets the intent of 10 CFR 50.34(f)(2)(xviii) (TMI Action Plan II.D.3 of NUREG 0737). 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 is provided for a diverse selection of monitored parameters in order to ensure that conditions indicative of inadequate core cooling are easily identified and unambiguous. Parameters monitored include core inlet and exit temperatures, RPV water level and pressure, and degree of subcooling. Control room indication and instrumentation is further described in Section 7.2.

The LOCA Evaluation Model (EM) (Reference 6.3-1) has been developed based upon satisfying 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 6.3-4). The NuScale Power Plant design complies with 10 CFR 50.46 with respect to the ECCS being designed such that cooling performance is in accordance with an acceptable evaluation model. The evaluation model is based on 10 CFR 50 Appendix K requirements.

6.3.2 System Design

6.3.2.1 Schematic Piping and Instrumentation Diagrams

A schematic of the ECCS configuration is shown in Figure 6.3-1.

6.3.2.2 Equipment and Component Descriptions

The ECCS consists of five main valves: three independent RVVs and two independent RRVs with associated actuator assemblies, instrumentation and controls (I&C). A typical ECCS main valve is shown in Figure 6.3-3. Each main valve is a power-actuated relief valve that is hydraulically closed, spring-assist to open, normally closed, and fails open. The valve is welded to the RPV with a nozzle and safe end configuration. The ECCS valve design and normal operating information is provided in Table 6.3-2.

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

During power operations, the ECCS is maintained in a standby condition with the main valve disks held closed against spring force by a pressurized control chamber. The main valve is initially closed (reset) by operator action that supplies chemical and volume control system (CVCS) water to pressurize the control chamber and close the valve against spring pressure. Once closed, the valve is maintained closed by a pressurized control chamber. Control chamber pressure is maintained by reactor pressure ported through an internal orifice located in the body of the main valve disk or by maintaining CVCS supply to the valve control chamber until the RPV is sufficiently pressurized.

The main valve is opened 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 two pilot valves: a trip valve and a reset valve, each with an associated solenoid as shown in Figure 6.3-3. To allow for actuation by either ESFAS division, one RVV is provided with an additional (redundant) trip valve solenoid assembly. Each ECCS actuator assembly is welded 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 entire RVV actuator assembly is 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 DC solenoid-operated valve. Energizing the reset pilot valve solenoid ports CVCS (RCS/CVCS coolant) to the main valve to pressurize the control chamber and 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 is de-energized to close the reset valve.

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

Each ECCS 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 is located in the path from the main valve 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 setpoint, 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 main ECCS valve is set to [1100 +/- 100]psid. The IAB valves block the opening and the ECCS main valves remain closed until RPV pressure has lowered sufficiently to establish a differential pressure of less than the IAB valve release pressure when actuated on a valid ECCS actuation signal.

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

The RVVs are six-inch power-actuated relief valves attached to the reactor vessel head and connected directly to the PZR steam space of the RPV.

The RRVs are four-inch, power-actuated relief valves attached to the upper shell section of the RPV, above the flange and about six feet above the top of the reactor core. The valves are connected directly to the downcomer space of the RPV and are designed with a minimum flow coefficient of 55.

Stainless steel bolt-on flow diffusers are provided on the discharge of the RVVs to diffuse the high pressure steam and water flow discharged to the CNV. RRVs do not require diffusers since they are smaller and more distant from equipment requiring protection. The RVV and diffuser, as a combined unit, are designed with a minimum flow coefficient of 375.

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 heat transfer to the reactor pool. The CNV is described in Section 6.1 with additional information on the heat removal function provided in Section 6.2.2.

The capability for containment heat removal (long-term ECCS operation) is maintained without operator action for at least 72 hours. The reactor pool (UHS) is described in detail in Section 9.2.5.

The containment heat removal function includes heat transfer through the CNV wall to the cooling pool. The maximum calculated containment pressure for design basis events is less than design pressure.

Upon a sensed loss of AC power to the highly reliable DC power system battery chargers, reactor trip, decay heat removal actuation, demineralized water system isolation, and containment isolation are initiated by the MPS to reduce battery load. In addition, three 24-hour digital timers in each division of the MPS are started. 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. If an ECCS actuation signal (low RPV riser or high CNV water level) is received during the 24-hour timing period, the ECCS immediately initiates. This ECCS hold mode maintains ECCS trip valve solenoids energized when an actuation signal is not present, but sheds the load at 24 hours to ensure sufficient battery power for post-accident monitoring for at least 72 hours.

6.3.2.3 Applicable Codes and Classifications

The NuScale Power Plant design complies with applicable industry codes and standards, and regulatory requirements commensurate with the appropriate safety functions. The components of the ECCS (valves, hydraulic lines, and actuator assemblies) are Quality Group A, Seismic Category I components designed to the requirements of ASME Boiler and Pressure Vessel Code (BPVC), Section III, Subsection NB, 2013 Edition. Additional information addressing compliance with the applicable codes and classification of ECCS components is provided in Section 1.9 and Section 3.2. Valve position indication for the ECCS is qualified for seismic loads in accordance with IEEE 344. The IEEE codes applicable to the controls and power supplies are further addressed in Section 7.1.

6.3.2.4 Material Specifications and Compatibility

The external operating environment for ECCS components varies based on the operating conditions. During normal operation, the main valves operate in a high temperature, evacuated environment. During shutdown, the RRVs and hydraulic actuator lines are submerged in borated water. Under accident conditions, the ECCS valve exterior and hydraulic lines may be subjected to pressures up to 1000 psia.

The operating environment for the pilot valve actuator assemblies remains fairly constant with the assemblies submerged in the reactor pool. The pool temperatures range from 90 to 120 degrees F during normal operation, but can be as high as the post-accident saturation temperature for accident pressure.

The ECCS valves are of a robust physical design constructed from corrosion-resistant materials that have a proven history in light water reactor environments. All wetted portions of the ECCS valves, submerged lines, and actuators are constructed of stainless steel material resistant to boric acid corrosion. All surfaces of the ECCS valves in contact with reactor coolant or reactor pool water during refueling are constructed of corrosion-resistant materials and have been shown to not exhibit unacceptable degradation in service based on exposure to operating chemistry.

The valves are qualified to retain the ability to perform their safety function under all postulated events for the CNV and RPV. Additional information describing the ECCS main valve, piping, and actuator valve materials of construction is provided in Table 6.1-1.

The ECCS valves and hydraulic lines are designed for an internal pressure and temperature of 2100 psia and 650 degrees F, respectively, and an external pressure of 1000 psia. Overpressurization of the components is protected by the reactor safety valves.

The ECCS actuator assemblies are designed for submergence and external pressure and temperature of 50 psia and 32 to 250 degrees F, respectively. The actuator assemblies are designed for an internal pressure and temperature of 2100 psia and 650 degrees F, respectively.

The RRVs and actuators are designed for a minimum neutron exposure of 2.5E17 neutrons/cm^2. Neutron exposure of the RVVs is negligible due to their distance from the core beltline region.

During reactor shutdown and post-LOCA events, the surfaces of ECCS components in the CNV are exposed to borated water. Eliminating or minimizing chloride levels and maintaining low levels of oxygen in the water reduces the potential for stress corrosion-cracking. The post-LOCA coolant is reactor coolant that satisfies RCS chemistry criteria. Water chemistry during shutdown conditions is controlled to preclude stress corrosion-cracking initiation using water treatment methods that are consistent with the spent fuel pool chemistry recommended in the Electric Power Research Institute Primary Water Chemistry Guidelines.

6.3.2.5 System Reliability

The ECCS is designed to function by transferring heat from the reactor coolant to the UHS through the CNV. The ECCS does not require operator action or support from nonsafety-related systems for continued operation and is capable of post-accident extended long-term cooling of the core for at least 30 days.

Reliability of the ECCS is provided by redundant valves remotely and separately actuated by two divisions of the ESFAS function of the MPS. The ECCS valves and actuators are assigned to separate divisions for instrumentation and electric power. One RVV is designed for actuation by either division of ESFAS by redundant trip valves controlled by separate divisions of instrumentation and electric power.

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 on 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.

No single active or passive failure prevents ECCS initiation or the capability of the system from performing its core cooling or LTOP safety function. The effect and consequences of single failures are discussed in Section 6.3.3 and are provided in Table 6.3-3.

The ECCS main valves are not susceptible to water hammer. The system consists of five main valves attached to the RPV. The system design includes no pumps or piping which precludes the susceptibility to water hammer mechanisms. The ECCS actuator lines and trip reset valves have some susceptibility to water hammer due to the possibility for flow in the actuator lines to transition to two-phase choked flow as the hot RCS coolant in the lines is discharged to CNV during actuation. Accordingly, the actuators are designed to accommodate the effects of water hammer.

The NuScale design fully complies with the regulatory positions of RG 1.82.

The generation of post-LOCA or high-energy line break debris and the presence of latent debris was evaluated with respect to its impact on ECCS and long-term core cooling operation. Based upon the NuScale design and cleanliness requirements, minimal debris generation and accumulation is expected and does not adversely impact the ability of ECCS to perform its required functions. Debris limits were developed to address generic safety issue GSI-191. An evaluation of the effects of fibrous, particulate, and chemical debris in the reactor coolant on the long-term cooling capability was performed with the results demonstrating that long-term core cooling is not adversely impacted. Evaluations were conducted to assess the debris impact on ECCS components, the fuel, and the core.

Latent debris is used as the basis for evaluating debris because insulation, paint, and coatings used in typical PWRs are not used or allowed in the NuScale 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 (cables, etc.) are designed to 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. Latent debris has no impact on core cooling during ECCS operation. The amount of latent debris, which is limited by operational cleanliness requirements in concert with the core design, 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: A COL applicant that references the NuScale Power Plant design certification 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 limit the introduction of coating materials into containment.
- An inspection program to confirm containment vessel cleanliness prior to closing for normal power operation.

6.3.2.6 Protection Provisions

Protection is provided in the design against missiles, pipe whip, flooding, thermal stresses, LOCA loading, and seismic effects. The ECCS main valves are protected from external missiles by the Reactor Building, NPM bio-shield, and CNV wall. There is no rotating equipment beneath the bio-shields or in the CNV that could malfunction and generate a missile.

The ECCS is protected against the effects of missiles, pipe whip (that might result from piping failures), jet impingement, and environment changes (wetting and temperature change). Due to the limited space inside the CNV, the usual protection methods of separation by distance, installation of pipe whip restraints, or shielding are not used. Integral shield and pipe whip restraint (ISR) devices (described in Section 3.6.5) are mounted directly to the high energy lines. Protection from the direct dynamic effects of pipe breaks and from loads applied by the impinging fluid released are further described in Section 3.6.1 for safety-related SSC, including the ECCS. Protection against environment changes (wetting and temperature) are addressed in the design requirements of the individual components. Section 3.6.2 provides additional discussion of pipe break location selection and dynamic effects.

In general, refer to Section 3.10 for seismic qualification of equipment, Section 3.3 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 provided on the outlets of the RVVs to protect plant cables and other components from the blowdown force when the RVVs open.

6.3.2.7 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 are designed to accommodate the preservice and inservice testing and inspection requirements of IWC-2500 and ISTC-3100 of the ASME Code with the valves in place. Additional discussion is provided in Section 3.9.6.

No maintenance, inspection, or testing of ECCS components is conducted during normal operations. Inspection and maintenance of the ECCS main valves are conducted only during NPM refueling outages. Because the CNV interior is inaccessible during normal operation, the required maintenance and inspections are performed in the NPM inspection bay during reactor outages.

The ECCS valves are power-actuated relief valves (O&M 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 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 are therefore conducted 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 are tested in accordance with ISTC-3630 and Appendix J, Type B criteria. The requirements of Section XI, Subsection IWB of the ASME BPVC are applicable to all pressure-retaining (RCPB and containment pressure boundary) ECCS components.

Additional information addressing inservice testing of ECCS valves is provided in Section 3.9.6.

6.3.2.8 Manual Actions

Manual actuation of the ECCS from the control room is possible, but operator action is not credited or required for design basis events, including those that involve active failures of the ECCS. The operator is alerted if a valve fails to change to its safety position on command. An alert is generated when the position instrumentation on an ECCS valve disagrees with the MPS demand position.

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

Manual override or interruption of an ECCS actuation while a valid safety system armed or active signal exists is prohibited by the ESFAS design. The ECCS actuation signal is sealed-in by the MPS to ensure completion of ECCS actuation (see Section 7.2).

6.3.3 Performance Evaluations

The conditions that result in actuation of the passive ECCS discussed in Chapters 15 and 19 include:

- LOCA
- Inadvertent ECCS actuation

Performance of the ECCS is determined by evaluating 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 evaluation model is based on the guidelines in RG 1.203.

The LOCA evaluation model is described in TR-0516-49422 (Reference 6.3-1).

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 presented in Section 15.6.

The LOCA EM (Reference 6.3-1) is used to evaluate a break spectrum as specified by Appendix K to 10 CFR 50 to provide assurance that the most severe postulated LOCA has been identified. The LOCA analyses are presented in Section 15.6.

The long-term cooling capability following ECCS operation is discussed in Technical Report TR-0916-51299 (Reference 6.3-2).

The applicable ECCS performance acceptance criteria taken from 10 CFR 50.46 is shown in Table 6.3-4.

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

Conformance to the 10 CFR 50.46 acceptance criteria for postulated accidents is demonstrated for the NuScale fuel parameters. 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.

Calculations were performed to determine that maximum boron concentrations in the core region following actuation of the ECCS do not result in boron precipitation to the extent that natural circulation flow in the core and long-term cooling capability are interrupted. The calculation established a boron concentration maximum in the core region and an associated boric acid solubility temperature.

Analyses show that under bounding conditions of core mixing volume for long-term core cooling conditions (quasi-static state), boron precipitation does not occur at temperatures greater than 80 degrees F.

Containment cooling is provided 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.

The long-term cooling analysis shows that the UHS 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. In consideration of potential water losses, the UHS analysis includes the heat input from all NPMs as well as the heat input from the spent fuel racks at maximum capacity when normal pool cooling is not available.

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 to the spent fuel in the spent fuel pool and to each of the shutdown NPMs for more than 72 hours of post-accident cooling.

When available, the reactor pool cooling water system transfers heat from the reactor pool to the site cooling water system and the pool surge control system can be used to add water from an offsite source to the reactor pool.

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, the effects of fibrous, particulate, and chemical species or precipitates on long-term core cooling were evaluated, including the potential for debris accumulation at the RRVs, core inlet, and in the heated core region.

Acceptance criteria are applied separately to 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 acceptance criteria applied:

- fuel rod scale thickness
- peak cladding temperature
- pressure drop due to debris

Debris deposition on any fuel rod is limited to 50 mils to preclude deposition on two adjacent fuel rods from touching and reducing convective heat removal from the rods. The 50-mils limit includes clad oxide, crud layer, and debris deposition. The minimum clearance between adjacent fuel rods is calculated from fuel assembly dimensions within the NuScale fuel design. A calculated pin-to-pin distance of 122 mils provides a conservative margin when comparing half the distance between rods to the 50-mil limit for each fuel rod.

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.

The core inlet pressure drop due to debris accumulation at the core inlet is limited to less than 0.1 psid.

Analysis assumptions include:

- flow approaching the core inlet is uniform
- debris is assumed to accumulate in a uniform manner
- fiber and particulate are well mixed in a homogeneous solution
- debris transport to the core inlet is proportional to the ECCS flow rate
- fiber is in its constituent form and neutrally buoyant
- individual fiber diameter is 7 microns
- particulate is in its constituent form with an assumed diameter of 10 microns
- chemical species are assumed to take the form of aluminum oxyhydroxide and precipitate out of solution
- localized buildup of debris within an intermediate spacer grid is assumed to be completely insulated on the outside surface so that cladding heat removal is only by axial conduction in the cladding
- the decay heat level during the initial portion of the transient is applied for the entire event to maximize energy removal requirements
- the core is 50 percent voided in consideration of the liquid volume in the core
- power at 160 MWt with a two percent measurement uncertainty
- 120 percent ANS 1971 decay heat
- fibrous debris density of 62.4 lbm/ft^3
- particulate debris density of 100 lbm/ft^3

The NuScale design minimizes debris generation by restricting the use of insulation, paint, and coatings within containment. In addition, the zone of influence for debris generated by fluid jet forces is limited by ISR devices located on potential line break areas and diffusers installed on the discharge of RVVs. The combination of devices to limit the zone of influence, restrictions on the use of debris-generating materials (e.g., insulation, paint, coatings, etc.) and components within containment that are designed to withstand accident conditions, limits debris generation to less than the amount needed to block or restrict long-term cooling flowpaths.

An estimate of the amount of debris present in a NuScale Power Plant was evaluated by examining the latent debris data in current operating plants. The use of operating plant information to establish a latent-debris source term established a conservative estimate because the NuScale design does not have the large containment with machinery, insulation, and personnel traffic during outages typical of a traditional operating plant. The total estimated mass of debris was conservatively determined to be 2.86 lbm (1297g).

The fiber content of the latent debris constitutes between 5 and 16 percent of the mixture with the particulate component accounting for 84 to 95 percent. The maximum fraction of 16 percent is used to estimate the amount of fiber in the RCS with the calculated amount determined to be 0.46 lbm (207.3 g or 5.6 g/FA). The maximum fraction of 95 percent is used to estimate the amount of particulate with the calculated amount determined to be 2.72 lbm (1234 g or 33.4 g/FA). 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. Chemical species are typically generated when containment liquid is at high temperature and appear in solution as a result of the corrosion or erosion of materials in containment. These species combine with other chemical species or buffering agents and precipitate out of solution to form a solid as the liquid cools. The evaluation assumes aluminum as the primary chemical species in solution; that, when combined with a buffering agent precipitates out as aluminum oxyhydroxide.

Chemical species are not expected to form in the NuScale design. Boron is used to control reactivity and buffering agents are not included in containment. The containment and components within containment are either fabricated from or clad with stainless steel to preclude 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.

The evaluated debris accumulation is based on conditions associated with an inadvertent opening of a RRV. 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. At about 145 seconds, conditions 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 that is 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 is dependent on the nature of the debris and the flow field in the containment and RPV. The earliest that debris arrives in the RPV follows ECCS actuation and occurs about 460 seconds after event initiation. The fiber and particulate collected in the containment coolant are assumed to be well mixed within a homogeneous solution and debris transport is assumed to be proportional to the ECCS flow rate into the RPV. Fiber is assumed to be neutrally buoyant and all fiber approaches the core inlet in proportion to the ECCS flow. In consideration of the flow velocities below the RRVs and within the RPV, particulate of the assumed 10 micron size also approaches the core inlet in proportion to ECCS flow.

The flow rate through the RPV and the CNV liquid volume determine 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 an initial 35 lbm debris mass, of which half is delivered to the core in about 45 minutes and 99 percent is delivered in less than 5 hours.

Debris that approaches the core inlet and is captured on the lower end fittings or structural grid is conservatively treated as if it were captured in 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 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. Particulate and precipitates are typically too small to be captured without fibrous debris accumulating first, so a fiber bed must first accumulate in order to capture the particulate.

Fuel assembly testing demonstrates that little head loss is expected for fiber loads of up to 7.5g/FA. The testing addresses a range of flow rates from 3 to 10.4 gpm/FA, which is higher than the 2.4 gpm/FA (or less) expected. The results indicate a pressure drop through the debris bed of less than the 0.1-psid acceptance criteria and 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 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.

Due to 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 still met. The debris plug is assumed to be completely insulated with heat removal only by axial conduction through the cladding. The temperature at the center of the blockage is calculated to reach 592 degrees F which remains 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.

At concentrations above 30 percent (debris in suspension), the insulating effects of the debris outweigh the beneficial effects of particulate assisted turbulence. The calculated values of fiber based on conservative estimates from operating plants demonstrate a 0.085 percent fiber concentration. The particulate and chemical precipitate calculation demonstrates that up to 944 lbm of particulate and chemical species can be tolerated in the core and not affect core heat transfer. The 2.86 lbm (1297 g) 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 was used to examine 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 prior to the event with deposition occurring as impurities transport into the crud deposit through large pores. Small particulate and formed precipitates are also assumed to be drawn into and merge with the growing scale. The analysis shows that up to 7.5 g/FA of fiber and 30 lbm of particulate in addition to 271 lbm of aluminum deposited on the fuel rods meets the acceptance criteria with additional margin.

The results of the fuel testing, engineering analysis, and debris estimates determined from operating plant data provide the basis for the following limits:

• total debris mass	2.86 lbm (1297 grams)
– fiber	0.46 lbm (207.3 grams) 5.6 g/FA
– particulate	2.72 lbm (1234 grams) 33.4 g/FA
• chemical	27.1 lbm

Analyses demonstrate adequate design margin with respect to the defined acceptance criteria. Adequate core cooling is ensured at debris levels of up to 7.5 gm/FA (fiber), 30 lbm (particulate), and 271 lbm (chemical). Additional analysis demonstrated acceptable cooling performance with up to 944 lbm of particulate and chemical species. A restriction of 0.5 for density increase is the basis for using the 30 lbm particulate value.

6.3.4 Tests and Inspections

Preoperational testing of the ECCS function is conducted to ensure 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 incorporated into the preoperational testing described in Section 14.2.

Preservice and inservice testing and inspection programs are described in Sections 3.9.6 and 6.6. The ECCS operational surveillance requirements are addressed in Chapter 16.

The ECCS-related Inspections, Tests, Analyses and Acceptance Criteria are addressed in Section 14.3.

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 provided in the control room through the safety display, indication system and module control system that includes ECCS valve position, containment isolation valve position, RPV riser level, RCS hot and cold temperature, pressurizer pressure, wide range RCS pressure, CNV water level, containment temperature, and containment pressure. The ECCS-related instrumentation is addressed in Section 7.2.

The ECCS is supplied with 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 provided from independent and redundant sensors. The ECCS is automatically actuated and requires no operator action during the first 72 hours following event initiation. Applicable parameters with analytical limits are listed in Table 6.3-1.

The ESFAS uses four redundant sensors (channels) to monitor ECCS-associated actuation parameters (low RPV riser level and high CNV water level) 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 actuators for the ECCS solenoid valves and ECCS valve position indications are supplied with power by the highly reliable DC power system. This power may not necessarily be available during an accident, and valve closure is not required during an accident. Position indication cabling is qualified in accordance with IEEE 323-1974 for the design conditions (temperature, humidity, submergence, pressure, radiation) of containment.

The ECCS performance monitoring is accomplished with instrumentation provided by the MPS 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).

6.3.6 References

6.3-1 NuScale, "LOCA Evaluation Model Topical Report," TR-0516-49422-P, September 2016.

6.3-2 NuScale, "Long Term Cooling Methodology," TR-0916-51299, 2016.

Table 6.3-1: Emergency Core Cooling System Alarms and Actuation

Parameter	Analytical Limit
Low RPV water level alarm	N/A
Low RPV water level actuation	371 inches above reactor pool floor
High CNV level alarm	N/A
High CNV level actuation	221 inches above reactor pool floor
RPV low temperature & high pressure warning (LTOP) alarm	N/A
RPV low temperature & high pressure (LTOP) actuation	RPV temperature < 320°F and RCS pressure at RPV flange > 260 psia or the calculated pressure value of RCS cold leg T_{sat} + 265°F (max value applies)

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	2100 psia	2100 psia	2100 psia
	External design pressure	1000 psia	1000 psia	50 psia
	Design temperature	650°F	650°F	650°F
Normal operating conditions	Internal pressure	1850 psia	1850 psia	1850 psia
	External pressure	0.037 psia	0.037 psia	45 psia
	Fluid temperature	500°F	590°F	40 to 160°F
	Valve external temperature	470°F	525°F	100°F
Hydrostatic testing	External pressure	1250 psig	1250 psig	
	Internal pressure	2782 psig	2625 psig	
	CNV test temperature	70 to 140°F	70 to 140°F	
Accident conditions	External pressure	1000 psia	1000 psia	
	Design temperature	F		

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

Component	Function	Failure Mode	Failure Mechanism	Effect	Failure Detection
Reactor vent valve ECC-HOV-0101A ECC-HOV-0101B ECC-HOV-0101C	Maintain closed position	Spurious opening	Mechanical	The function of maintining a closed position has failed and the ECCS function actuates when water level reaches the actuation setpoint. No impact on the core cooling safety function of ECCS.	• Reactor trip • Valve position indication • PZR pressure indication • PZR level indication • Containment pressure indication
			Electrical/I&C	Requires an additional failure of the inadvertent block valve to allow the main valve to open.	
		Leakage exceeding technical specifications limits	Mechanical	No impact on the core cooling function of ECCS.	• Containment pressure indication • Leakage detection system* • Valve inspection/testing
	Vent path for ECCS operation	Failure to open	Mechanical electrical/I&C	Venting limited to redundant RVVs to ensure the core cooling function of ECCS is maintained.	• Valve position indication • Containment water level indication • RPV/CNV ΔP • Valve inspection/testing
		Slow opening/ extended stroke time/ delayed actuation	Mechanical electrical/I&C	Greater portion of vent flow is through redundant RVVs until the affected valve opens completely. Redundant RVVs ensure the core cooling safety function.	• Valve position indication • Containment water level indication • RPV/CNV ΔP • Valve inspection/testing
		Spurious closure	Electrical/I&C operator error	Redundant RVVs ensure adequate venting and the core cooling safety function.	• Valve position indication • Containment water level indication • RPV/CNV ΔP • Valve inspection/testing
		Flow blockage	Mechanical	Reduced or nonexistent flow through an affected valve. Flow through redundant RVVs ensures the ECCS core cooling safety function.	• Containment water level indication • RPV/CNV ΔP

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

Component	Function	Failure Mode	Failure Mechanism	Effect	Failure Detection
Reactor vent valve ECC-HOV-0101A ECC-HOV-0101B ECC-HOV-0101C (cont)	LTOP depressurization	Failure to open	Mechanical electrical/I&C	Slower rate of depressurization through redundant RVVs. Core cooling safety function is accomplished.	• Valve position indication • Depressurization rate • Valve inspection/testing
		Slow opening	Mechanical electrical/I&C		• Valve position indication • Depressurization rate • Valve inspection/testing
		Spurious closure	Electrical/I&C operator error		• Valve position indication • Depressurization rate change
		Flow blockage	Mechanical	Reduced or no flow through the affected valve. Depressurization through redundant RVVs ensures the core cooling safety function.	• None
Reactor recirculation valve ECC-HOV-0104A ECC-HOV-0104B	Maintain closed position	Spurious opening	Mechanical	The function of maintaining a closed position has failed. The ECCS function actuates when RPV water level reaches the actuation setpoint. Core cooling safety function of ECCS is not impacted.	• Reactor trip • Valve position indication • PZR pressure indication • PZR level indication • Containment pressure indication
			Electrical/I&C	Requires an additional failure of the inadvertent block valve to allow the main valve to open.	
		Leakage exceeding technical specifications limit	Mechanical	No impact on the core cooling function of ECCS. Potential for boric acid buildup along the leakage path.	• Containment pressure indication • Leakage detection system* • Valve inspection/testing

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

Component	Function	Failure Mode	Failure Mechanism	Effect	Failure Detection
Reactor recirculation valve ECC-HOV-0101A ECC-HOV-0101B ECC-HOV-0101C (cont)	Coolant return from CNV to RPV	Failure 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.	• Valve position indication • Valve inspection/testing
		Slow opening/ extended stroke time/ delayed actuation	Mechanical electrical/I&C	Greater portion of the recirculation flow is through the redundant RRV until the affected valve is fully open. The redundant RRV ensures the core cooling function of ECCS is not compromised.	• Valve position indication • Valve inspection/testing
		Spurious closure	Electrical/I&C operator error	ECCS recirculation flow is exclusively through the redundant RRV. The core cooling safety function is maintained.	• Valve position indication • Containment water level indication • RPV/CNV ΔP
		Flow blockage	Mechanical	Reduced or loss of flow through the affected valve. Adequate recirculation flow is maintained through the redundant RRV to ensure the core cooling safety function is maintained.	• Containment water level indication • RPV/CNV ΔP
ECCS valves controlled by one ESFAS division (ECC-HOV-0101A, -0101C, -01004A) or (ECC-HOV-0101B, -0101C, -01004B)	Maintain closed position	Spurious opening	Electrical/I&C operation error	The function of maintaining a closed postion has failed. The ECCS function actuates when RPV water level reaches the actuation setpoint. The core cooling safety function is maintained. Requires an additional failure of an associate inadvertent block valve to allow the main valve to open is reactor pressure is above the threshold pressure.	• Reactor trip • Valve position indication

Notes:
*The leakage detection system cannot identify leakage from a specific location, it can be used to distinguish between RCS and non-RCS leakage into the CNV, but cannot provide the location or quantify the leakage from a specific leak path

Table 6.3-4: Acceptance Criteria for Emergency Core Cooling System Performance

Criterion	Value	Conformance
Peak Cladding Temperature (PCT)	Calculated maximum fuel element cladding temperature shall not exceed 2200°F.	Maximum PCT is and the maximum fuel centerline temperature is .
Maximum cladding oxidation	Calculated total local oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.	The core remains substantially submerged and does not encounter high temperature steam. Measurable cladding oxidation is not predicted.
Maximum hydrogen generation	Calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all the metal in the cladding cylinder surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.	Measurable cladding oxidation is not predicted. Hydrogen generation from metal-water chemical reaction is similar to normal operation.
Coolable geometry	Calculated changes in core geometry shall be such that the core remains amenable to cooling.	Based on calculated core temperatures, core geometry does not change measurably.
Long-term cooling	After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.	Calculation demonstrates long term success of ECCS.

Figure 6.3-1: Emergency Core Cooling System

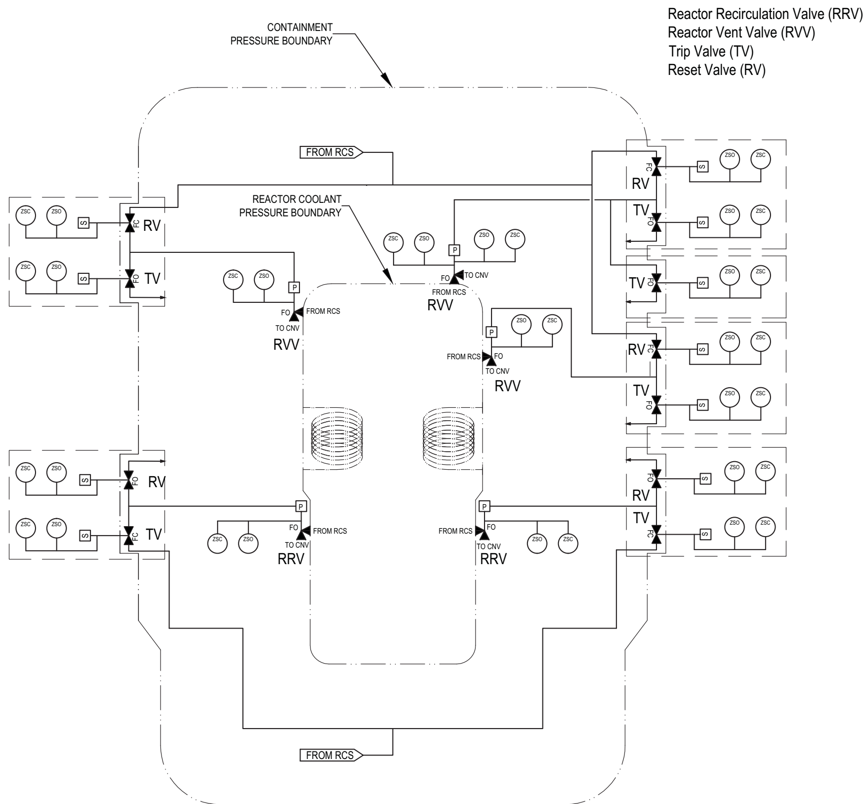


Figure 6.3-2: Emergency Core Cooling System Operation

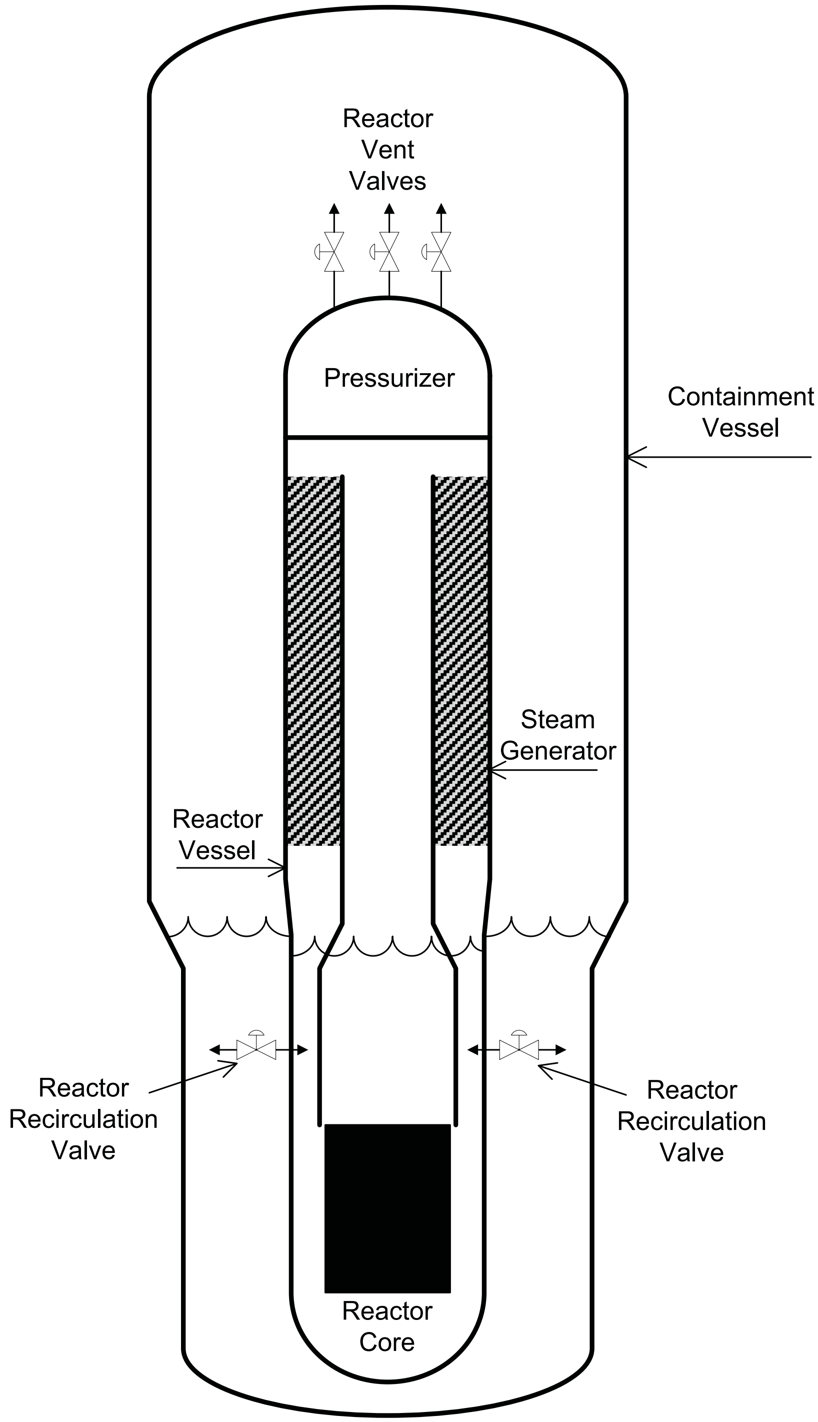
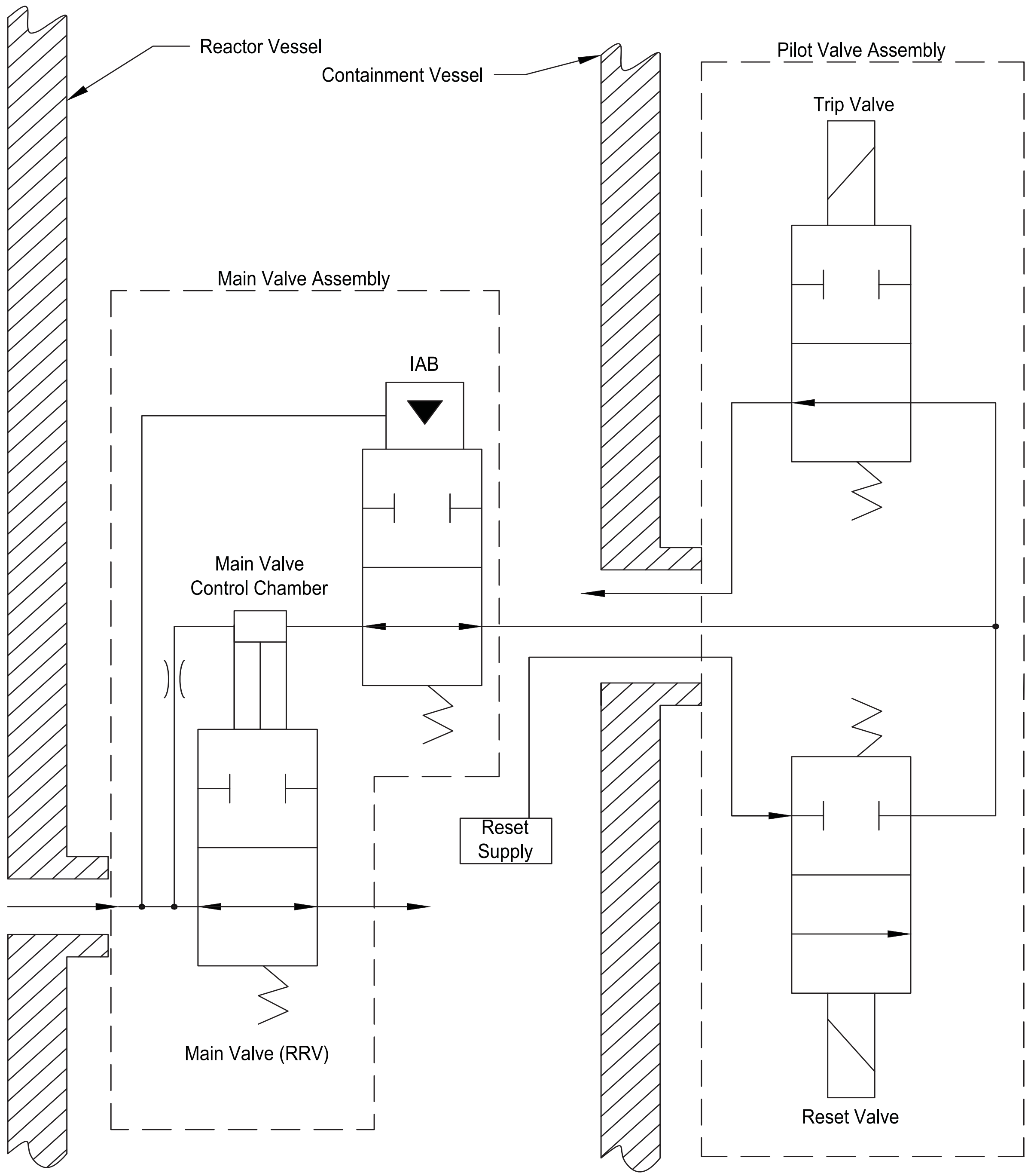


Figure 6.3-3: Emergency Core Cooling System Valve and Actuator Hydraulic Schematic



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. Habitability functions are provided by systems and equipment to protect control room operators against postulated releases such as radioactive materials, toxic gases and smoke. Control room habitability functions include:

- missile protection (see Section 3.5.1)
- radiation shielding (see Chapter 12)
- normal pressurization, air filtration, and air-conditioning (see Section 9.4.1)
- fire protection (see Section 9.5.1)
- radiation monitoring (see Section 9.4.1 and Section 11.5)
- smoke detection (see Section 9.4.1)
- lighting (see Section 9.5.3)

This section describes the control room habitability system (CRHS) which provides breathable air to the control room during the first 72 hours following an accident. After 72 hours, the normal control room HVAC system (CRVS), described in Section 9.4.1, is expected to be available to provide heating ventilation and air conditioning service to the control building (CRB) for the remainder of the accident recovery period.

6.4.1 Design Basis

This section identifies the CRHS required or credited functions, the regulatory requirements that govern the performance of those functions, and the controlling parameters and associated values that ensure that the functions are fulfilled. Together, this information represents the design bases, defined in 10 CFR 50.2, as required by 10 CFR 52.47(a) and (a)(3)(ii).

The CRHS is a nonsafety-related system designed to provide clean breathing air to the control room envelope and maintain a positive control room pressure for habitability and control of radioactivity when conditions prohibit the CRVS from fulfilling these functions.

GDC 2 was considered in the design of the CRHS. Natural phenomena, including earthquakes, do not prevent required components of the CRHS from performing their intended function.

GDC 4 was considered in the design of the CRHS. Failure of the CRHS or malfunction of the CRHS does not adversely affect other SSC that are located in the vicinity of CRHS components. Also, the CRHS components are located in the CRB, which provides protection from potential adverse environmental conditions.

GDC 5 was considered in the design of the CRHS. The CRHS services the control room that contains the controls of up to 12 NuScale Power Modules and is designed such that a failure of one portion of the system does not significantly impair the ability to perform its regulatory required functions including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining unit(s).

The CRHS complies with GDC 19, as it relates to maintaining the control room in a safe condition under accident conditions and providing adequate radiation protection.

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

6.4.2 System Design

6.4.2.1 Definition of Control Room Envelope

The control room envelope (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. All of these areas are either frequently or continuously occupied. The CRE includes air locks for ingress and egress.

6.4.2.2 Ventilation System Design

Normal heating ventilation and air conditioning service to the CRE is provided by the CRVS as described in Section 9.4.1. The CRVS includes redundant isolation dampers that close to isolate the CRE. The CRHS provides emergency air to the CRE from a bottled air supply. The CRHS is designed to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. Design capacity of the air bottles is defined in Section 6.4.2.3. The supply of breathing air limits the concentration of carbon dioxide in the CRE to less than 5000 ppm for 72 hours after the CRE is isolated.

The CRHS is designed to Seismic Category I standards, except for the compressor and associated equipment and piping up to the isolation valve between the compressor and the air bottles which are Seismic Category III. The compressor is not located in the same area as the bottles, so it does not need be designed to Seismic Category II.

A simplified diagram of the CRHS is shown in Figure 6.4-1. System design parameters are presented in Table 6.4-1. 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.

6.4.2.3 Component Descriptions

High Pressure Air Compressor

An air compressor, located in the CRB, is provided to charge the emergency air storage bottles. The compressor uses intake air filters to provide breathing quality air that meets the standards of Compressed Gas Association G7.1 Level D (Reference 6.4-1).

Air Bottles

The CRHS air storage bottles are located on an elevation below the control room and can provide breathable air for 20 personnel in the CRE and positive CRE pressurization for 72 hours with up to 25 percent of the bottles out of service. The assumption of 20 personnel includes six operators, two NRC staff, two security officers, and two unspecified individuals. In addition the 20 includes an allowance of seven TSC staff and one emergency communicator in the event that the TSC should become uninhabitable and a decision is made to transfer the TSC plant management function to the main control room.

Bottles are arranged in groups to allow isolation by group. The air storage bottles are integrally forged vessels and conform to Section VIII of the ASME Code (Reference 6.4-2). Design information related to the air storage bottles is noted in Table 6.4-2.

Air Bottle Racks

The CRHS air bottle racks are welded structures that support the CRHS air bottles and are designed to meet Seismic Category I requirements. The design also includes capturing features to ensure the air bottles do not become internal missiles within the CRB.

Eductor

The air supply into the control room uses an eductor to help mix the CRHS breathing air with the CRE air volume and increase air movement in CRE spaces. This passive component has a bypass line provided to allow airflow through an alternate line if the eductor fails to provide air flow.

Silencers

The CRHS includes silencers on the outlets of the piping that supplies air into the CRE. A silencer is also provided on the inlet line to the eductor from the CRE.

Valves

The CRHS air storage bottles supply air to the system through isolation valves that are normally open. These valves are located downstream of the air bottles and upstream of the pressure regulating valves.

Compressed air supply lines contain pressure regulating valves located downstream of the system common header. The pressure regulating valves reduce pressure to provide stable outlet pressure despite supply pressure changes. The pressure at the outlet of the regulating valves is controlled by self-contained pressure control operators. The outlet pressure of the regulating valves is set to function with a downstream orifice to maintain pressure within the CRE at a minimum of one-eighth inch water gauge relative to surroundings.

The pressure boundary of the compressed air storage bottles is maintained by two normally closed air supply isolation valves in parallel in the main supply line. These solenoid operated valves are remotely operated, and are located within the CRE pressure boundary downstream of the pressure regulating valves and orifice. These valves automatically 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 is provided as a means of manually activating the alternate air delivery flowpath in the event the main air delivery flowpath is inoperable. This alternative flowpath is provided with separate pressure regulating equipment.

The CRHS has a permanent connection feature to allow hook-up of portable compressed air for off-site assistance. This connection, accessible to a temporary source of air, is located between the compressor and the air storage bottles.

There are two openings in the wall separating the MCR from the vestibule area. Each of these openings contains a manual balancing valve followed by a pressure relief isolation valve (refer to Figure 6.4-1). The CRE pressure relief valves allow air to exit the CRE during CRHS operation. The balancing valves are set manually and locked into position to maintain CRE pressure at one-eighth inch water column. These valves can also be adjusted manually as required during CRHS operation.

CRHS Piping

The CRHS piping is stainless steel built to ASME B31.1 standards (Reference 6.4-3).

6.4.2.4 Leak Tightness

Control room doors, walls, cable penetrations, and other penetrations of the CRE boundary are designed to minimize leakage in to or out of the main control room. Isolation dampers are part of the CRVS as described in Section 9.4.1.

6.4.2.5 Interaction with Other Zones and Pressure-Containing Equipment

Other than the CRHS supply lines, the only pressurized lines that penetrate the CRE are the CRVS, fire protection, and potable water supply lines. Fire protection and potable water are not gas sources. CRVS isolation dampers, which form part of the CRE boundary, are closed while CRHS is in service, so no air infiltration is expected from CRVS duct work. There are no pressurized tanks in the CRE other than any self-contained breathing apparatus that may be required for protection against toxic gases.

6.4.2.6 Shielding Design

The CRE is surrounded by thick concrete walls that provide shielding from radiation that may be present in the atmosphere following an accident. The reactor building walls provide additional protection from radioactive sources.

The CRVS charcoal filters are located two floors above the control room so that any radiation emanating from that source will be attenuated by two thick concrete floors.

6.4.3 System Operation

6.4.3.1 Normal Operation

The CRHS is in standby mode during normal plant operation. Air bottles are maintained pressurized to provide breathing air and CRE pressurization when actuated.

6.4.3.2 Off-Normal Operation

In the event of a loss of alternating current power to both CRVS air handler units for ten minutes, or if high radiation is detected downstream of the CRVS air filtration unit, or if power is unavailable to all four of the EDSS-C battery chargers, the plant protection system generates a signal which automatically initiates the following actions:

- 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

These actions provide CRE occupants with clean breathing air under conditions in which the normal air supply may be contaminated.

Operation of the CRHS can also be initiated by manual actuation, for example in response to a hazardous chemical spill.

After the CRHS isolation valves are opened, the air supply pressure is regulated by the self-contained regulating valves. These valves maintain a constant downstream pressure despite upstream pressure changes as the bottled air inventory is supplied to the CRE. A constant air flow rate is maintained by the orifice downstream of the pressure regulating valves.

In the event of insufficient or excessive flow in the main delivery line, the main delivery line is isolated and the alternate delivery line is manually actuated. The alternate delivery line contains the same components as the main delivery line with the exception of the remotely operated isolation valves, and thus is capable of supplying compressed air to the CRE pressure boundary at the required air flow rate.

The regulated breathing air flow rate from the CRHS air bottles is sufficient to maintain the CRE pressure boundary at one-eighth inch water column positive differential pressure with respect to the surroundings.

Differential pressure between the CRE and the surrounding area is monitored to ensure that a positive pressure is maintained in the control room with respect to its surroundings. The wall separating the MCR area from the vestibule contains two pressure relief lines with balancing valves which discharge air from the CRE general area into the CRE vestibule. This air movement maintains the vestibule at a higher pressure than the CRB corridor pressure, reducing the potential for radioactive material being transported into the CRE when operators enter. Two vestibule discharge openings provide a purge flow path from the vestibule to the corridor.

6.4.4 Design Evaluation

As noted in Section 15.0.0, no operator actions are required or credited to mitigate the consequences of design basis events. As such, the operators perform no safety-related functions, consistent with the definition in 10CFR50.2. Therefore, although a habitable control room is provided for the operators, consistent with GDC 19, to perform other important non-safety related functions, the control room envelope and supporting habitability systems and components, including the CRHS, are not safety-related.

GDC 2 was considered in the design of the CRHS. Natural phenomena, including earthquakes, do not prevent regulatory required components of the CRHS from performing their intended function. The CRHS is designed and constructed to Seismic Category I specifications except for the compressor and the piping up to the first isolation valve between the compressor and the air bottles. The compressor is not located in the same space as the air bottles. The CRHS is located within the CRB, a Seismic Category I concrete building protecting its contents from the effects of severe weather.

GDC 4 was considered in the design of the CRHS. CRHS components are not subject to pipe whipping or fluids discharging from nearby systems that could degrade their performance. CRHS materials are compatible with the expected environmental conditions encountered during all phases of plant operation.

Although the CRHS is a shared system for 12 NPMs, its use during an accident on one unit does not affect the ability to safely shutdown and cooldown the remaining units, as it provides air to the control room common to all units. Thus, in compliance with GDC 5, the CRHS design does not create conditions that would cause an accident in one unit to propagate to other units.

The CRHS, in conjunction with the CRVS, provides compliance with GDC 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, toxic gas monitors, and smoke detectors located in the outside air intake as described in Section 9.4.1. Upon detection of smoke or toxic gas in the outside air duct, the outside air isolation dampers are closed to isolate the CRB from the environment. The CRB will not be pressurized under these conditions.

Upon a detection of a high radiation level in the outside air intake, the normal outside air flow path is isolated with isolation dampers and 100 percent of the outside air is routed through the CRVS air filtration unit, which includes charcoal and HEPA filters. If high levels of radiation are detected downstream of the air filtration unit or if normal power is not available to the CRVS air handlers or the EDSS-C battery chargers, the CRVS provides isolation of the CRE from the surrounding areas and outside environment via the CRE isolation dampers. The CRHS is then relied upon to maintain a habitable environment in the CRE. The CRHS air bottles have sufficient capacity to pressurize the CRE for 72 hours. The thermal mass of the CRB provides passive cooling, maintaining CRE temperatures suitable for equipment and personnel for 72 hours.

After 72 hours, backup power is expected to be available and the CRVS will then be utilized to provide air conditioning and building pressurization. The CRHS also includes an external air supply connection so that the air bottles can be replenished from an offsite source if needed.

The TSC is not served by the CRHS and therefore does not receive pressurization air 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 determined to be uninhabitable and is evacuated. The TSC function is then transferred to another location in accordance with the emergency plan.

The design of the CRHS satisfies CFR 50.34(f)(2)(xxviii) in that it provides assurance that, in the event of an accident, radiation doses to operators will not exceed acceptable limits and consequently will not prevent operators from performing required functions. The CRHS does not interface with any other systems that would provide a potential pathway for radioactive materials. The CRHS consists of pressurized air bottles that are charged with breathing quality air. There is an external air connection point that will allow the connection of a post 72 hour air supply from off-site air bottles to supply air and pressurization to the CRE for extended accident conditions if needed.

Radiological Protection

In the presence of significant airborne radiation downstream of the CRVS air filtration unit, 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 GDC 19 limit.

The CRHS provides bottled air to the CRE for 72 hours following CRE isolation and maintains the CRE at a higher pressure than its surroundings. After 72 hours, the CRHS bottled air supply is depleted and the CRVS (Section 9.4.1) is returned to operation, providing the CRE and the rest of the CRB with filtered air.

With the CRHS in operation, the CRE is maintained at a positive pressure of one-eighth inch water column with respect to its surroundings. The only source of unfiltered leakage into the CRE is due to ingress and egress; up to 5 cfm is considered for this type of inleakage. The dose analysis conservatively includes an additional 10 cfm of inleakage. Before the isolation dampers have closed in response to the high radiation signal, a certain amount of radioactive material will have entered the CRE. This material is gradually diluted over the duration of the accident as air enters and an equal amount of air leaves the CRE. Finally, control room operators will receive a small amount of radiation dose due to airborne radiation outside the CRE (sky shine and direct shine) and from the filters in the CRVS (filter shine).

For the purposes of radiation dose, accident duration is considered to be 30 days in accordance with RG 1.183. Analysis shows that the sum of radiation doses to control room personnel from all sources is less than 5 rem for the duration of any postulated accident. These design features provide compliance with 10 CFR 50.34(f)(2)(xxviii). Radiological dose to control room operators is further addressed in Section 15.0.3.

Toxic Gas Protection

In accordance with RG 1.78, the station maintains a supply of self-contained portable breathing apparatus with air bottles stored onsite. These bottles are used if hazardous chemicals are suspected to be present. The amount of stored air is sufficient to provide a 6-hour supply of breathable air for the number of main control room occupants identified in Technical Specifications Section 5.2.

COL Item 6.4-1: A COL applicant that references the NuScale Power Plant design certification will comply with RG 1.78 Revision 1, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release."

COL Item 6.4-2: A COL applicant that references the NuScale Power Plant design certification will specify operator training and qualification in the use of self-contained portable breathing apparatus.

Other Habitability Considerations

When normal air conditioning from the CRVS is not available, the thermal mass of the CRB and its contents limit the temperature increase as shown in Table 6.4-3 for the first 72 hours following an accident. The peak temperature at three hours is the result of a conservative assumption in the analysis, that control room equipment powered by the normal DC power system remains powered for three hours. After 72 hours, the CRVS (Section 9.4.1) provides cooling to the CRE.

The MCR is provided with adequate lighting for safe operation of the plant during accidents and all modes of operation. Refer to Section 9.5.3 for more information.

The CRE includes access to procedures, drawings, and other technical resources useful for mitigating accident conditions. The CRE includes a lavatory, kitchen facilities, and a dining area. Food, water, and first-aid medical supplies are stored in an accessible location within the CRE. The food and water supply is adequate for the minimum needs of the number of main control room occupants identified in Technical Specifications Section 5.2 for 72 hours.

COL Item 6.4-3: A COL applicant that references the NuScale Power Plant design certification will specify the technical resources to be stored within the CRE.

COL Item 6.4-4: A COL applicant that references the NuScale Power Plant design certification will specify food, water, and medical supplies to be stored within the CRE.

6.4.5 Testing and Inspection

Refer to Section 14.2 for information regarding preoperational testing.

Inservice testing of the CRHS is conducted in accordance with the surveillance requirements specified in the technical specifications. Leak tightness testing of the CRE pressure boundary is conducted in accordance with the frequency specified in the technical specifications. Inservice testing includes demonstration of the integrity of the CRE in accordance with RG 1.197.

In addition to periodic tests, CRE testing is performed when changes are made to structures, systems, and components that could impact CRE integrity, including systems internal and external to the CRE. These tests are commensurate with the types and degrees of modifications and repairs and the potential impact upon integrity. Additional CRE testing is also performed if a new limiting condition or alignment arises for which no in-leakage data is available (e.g., a hazardous chemical source appears where previously there was none). In-leakage in excess of the licensing basis value for the particular challenge to CRE integrity constitutes test failure.

COL Item 6.4-5: A COL applicant that references the NuScale Power Plant design certification will specify testing and inspection requirements for the CRHS, including CRE integrity testing.

6.4.6 Instrumentation Requirements

The CRHS design includes instrumentation and controls to ensure safe, efficient, and reliable operation. Instrumentation and controls for the CRHS are part of the plant protection system, which is described in Chapter 7.

6.4.7 References

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

6.4-2 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2013 Edition, Section VIII, Division 1, “Rules for Construction of Pressure Vessels,” New York, NY.

6.4-3 American Society of Mechanical Engineers, ASME Code for Pressure Piping, B31, ASME B31.1, “Power Piping,” New York, NY.

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 one-eighth-inch water gauge with respect to adjacent areas.
Breathing air supply flow rate	At least 80 cfm in order to maintain acceptable levels of CO_2, but not greater than 100 cfm in order to ensure a 72 hours 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 one-eighth-inch water gauge with respect to surrounding areas.	10* cfm plus 5 cfm assumed for air lock operation.

Note:

*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.

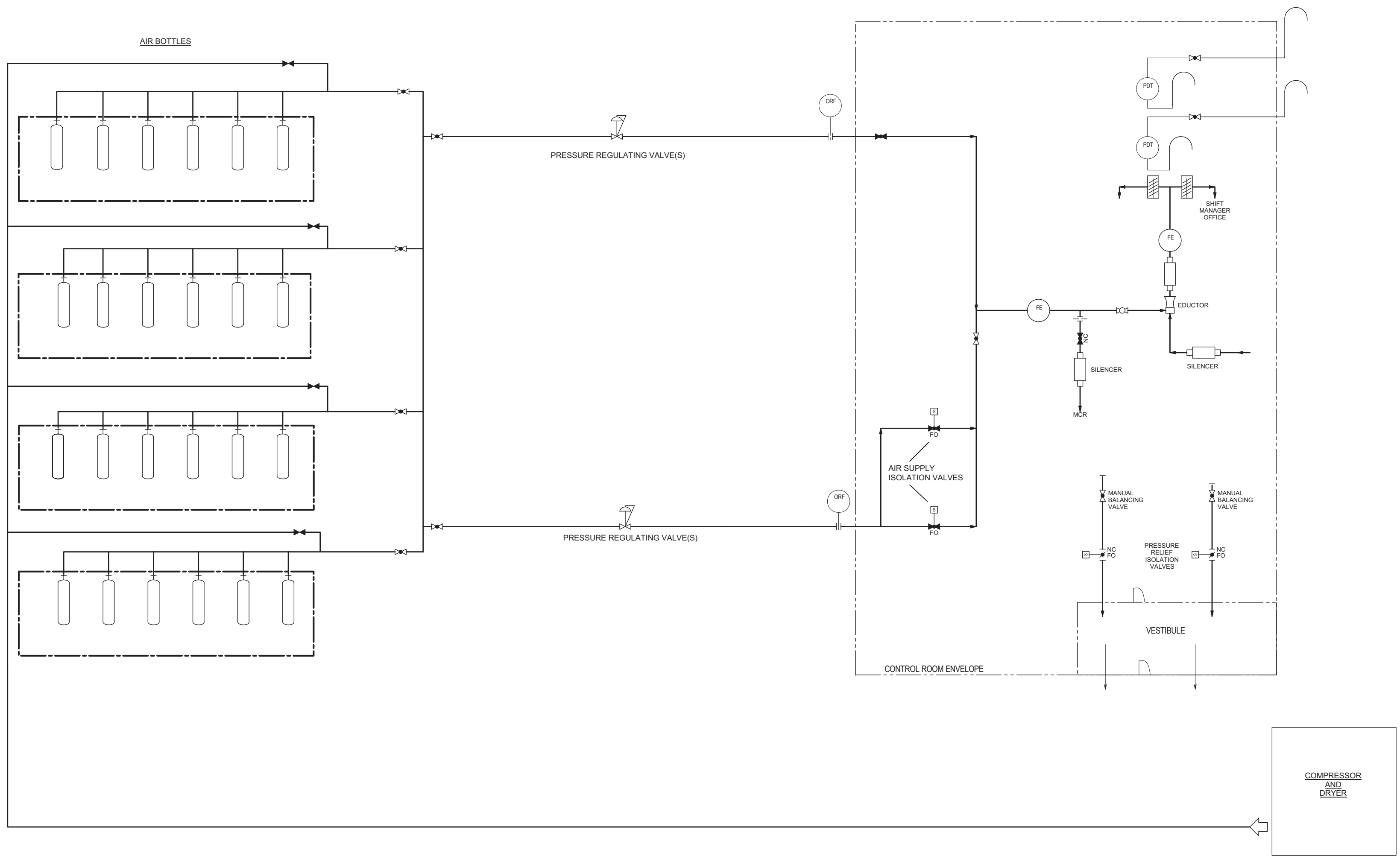
Table 6.4-2: Control Room Habitability System Air Bottle Design Parameters

Design Data/Parameter	Value
Design Pressure	4000 psig
Design Temperature	120°F
Size	24 in. diameter, 84 in. length
Capacity	15.4 ft^3 per Air Bottle
Total number of Bottles	192

Table 6.4-3: Main Control Room Temperature under Passive Cooling Conditions

Time after loss of normal cooling (hours)	Temperature (°F)	Relative Humidity (%)	Wet Bulb Globe Temperature (°F)
0	78.0	60.0	70.6
1	92.7	37.3	78.0
2	97.2	32.0	80.1
3	99.7	29.3	81.1
4	93.9	34.5	78.2
8	92.5	34.8	77.1
24	94.4	30.9	77.5
48	96.1	28.9	78.2
72	97.2	28.0	78.8

Figure 6.4-1: Control Room Habitability System Diagram



6.5 Fission Product Removal and Control Systems

6.5.1 Engineered Safety Features Filter Systems

The NuScale Power Plant design does not utilize engineered safety feature (ESF) filter systems or ESF ventilation systems to mitigate the consequences of a design-basis accident (DBA). In the NuScale Power Plant design there is a nonsafety-related reactor building heating ventilating and air conditioning (HVAC) system that includes filtering; however, it is not credited in the dose analysis.

6.5.2 Containment Spray Systems

The NuScale Power Plant design does not utilize 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 NuScale Power Plant design. Numerous failures must be postulated in order for significant core damage to occur. The only ESF fission product control system credited to mitigate the consequences of a design basis accident, or the Maximum Hypothetical Accident described in Section 15.0.3, in the NuScale Power Plant design is the containment vessel (CNV) in conjunction with the containment isolation system (CIS). Active fission product removal systems are not required to meet regulatory requirements in the NuScale Power Plant design. Natural aerosol removal mechanisms inherent in the containment design include thermophoresis, diffusiophoresis, hygroscopicity, and sedimentation. These passive removal processes, which deplete elemental iodine and particulates in the containment atmosphere, are discussed in Section 15.0.3. 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. Because the NuScale Power Plant design does not include the containment atmosphere cleanup systems that would be subject to GDC 42 and GDC 43, GDC 42 and GDC 43 do not apply to the NuScale Power Plant design.

6.5.3.1 Primary Containment

An ESF function of the containment is to provide a passive barrier to fission product migration to the environment in the event of a release of core fission products to the containment atmosphere. The CNV is an ASME B&PV Code Class MC (steel) containment that is constructed and stamped as an ASME B&PV Code Section III, Class 1 vessel. The CNV consists of an upright cylinder with top and bottom head closures. The CNV is partially immersed in the reactor pool. The CNV design is described in greater detail in Section 6.2.1. The CNV along with the CIS described in Section 6.2.4 is designed to preclude leakage from exceeding the value shown in Table 6.5-1, as discussed in Section 6.2.6 and the containment leakage rate testing program described in Section 5.5 of the Technical Specifications. Containment isolation times are provided in Table 6.2-4.

6.5.3.2 Secondary Containments

The NuScale Power Plant design does not utilize a secondary containment to mitigate the consequences of a DBA. The reactor building may delay releases to the environment and allow 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 certification of the NuScale Power Plant design.

6.5.4 Ice Condenser as a Fission Product Cleanup System

The NuScale Power Plant design does not utilize an ice condenser to mitigate the consequences of a DBA.

6.5.5 Pressure Suppression Pool as a Fission Product Cleanup System

The NuScale Power Plant 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	0.2 wt% per day
Containment free volume	1.1×10^7 in^3

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 are inspected and tested to quality standards commensurate with their safety functions. Where generally recognized Codes and Standards are used, they are identified and evaluated to determine their applicability and adequacy and are supplemented and modified as necessary to assure quality in keeping 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 provided in Chapter 17.

The ISI Program requirements for Class 2 and Class 3 systems and components, as defined in Section III of the American Society of Mechanical Engineers (ASME) Code, are described in this section.

The ISI Program for Class 2 and 3 components is based on 10 CFR 50.55a(g)(3), which requires that ASME Code Class components be designed to allow for the inspections detailed in the ASME Boiler and Pressure Vessel Code (hereinafter "the ASME Code"), Section XI, "Rules for In-service Inspection of Nuclear Power Plant Components" (Reference 6.6-1).

The initial ISI Program incorporates the latest edition and addenda of the ASME Code approved in 10 CFR 50.55a(b) on the date 128 months prior to fuel load. ISI of components and system pressure tests conducted during successive 120-month inspection intervals comply with the requirements of the latest edition and addenda of the Code incorporated by reference in 10 CFR 50.55a(b) 18 months prior to fuel load, subject to the limitations and modifications listed in 10 CFR 50.55a(b). In addition, the optional ASME Code cases listed in Regulatory Guide (RG) 1.147 (Reference 6.6-4) may be used. The ASME Code includes requirements for system leakage tests for active components. The requirements for system leakage tests are defined in ASME Section XI, Article IWC-5220 for Class 2 pressure retaining components and ASME Section XI, Article IWD-5220 for Class 3 pressure retaining components. 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 inspection. The ISI Program complies with applicable ISI and testing provisions of 10 CFR 50.55a(g) and (f).

There are no Exemptions regarding the ISI of Class 2 and 3 SSC. There are no relief requests necessary for preservice inspection and first interval ISI examinations for Class 2 and 3 components.

An In-service Testing (IST) Program that meets the requirements set forth in the ASME Code for Operation and Maintenance of Nuclear Power Plants (Reference 6.6-2), as stipulated in 10 CFR 50.55a(f) is implemented. The IST of valves is addressed in Section 3.9.6.

COL Item 6.6-1: A COL applicant that references the NuScale Power Plant design certification will implement an Inservice Testing Program in accordance with 10 CFR 50.55a(f).

The ISI requirements for Class 1, 2, and 3 threaded fasteners are described in Section 3.13.2.

The administrative controls and surveillance requirements specified by 10 CFR 50.63 are provided in Section 16.

COL Item 6.6-2: A COL applicant that references the NuScale Power Plant design certification will develop a preservice inspection and Inservice Inspection Program plans in accordance with Section XI of the ASME Code, and will establish the implementation milestones for the program. The COL applicant will identify the applicable edition of the ASME Code utilized in the program plan consistent with the requirements of 10 CFR 50.55a. The COL applicant will, if needed, address the use of a single ISI Program for multiple NPMs, including any Alternative to the Code that may be necessary to implement such an ISI Program.

6.6.1 Components Subject to Examination

Preservice inspections and ISIs are performed on ASME Code Class 2 and 3 components in accordance with ASME Section XI (Reference 6.6-1). These components are considered RG 1.26, Quality group B and C components.

The ASME Class 2 boundaries, based on RG 1.26 (Reference 6.6-3), for Quality Group B, are

1) portions of the reactor coolant pressure boundary (RCPB), but which are excluded from the Class 1 boundary pursuant to Section 5.2.4.

2) systems or portions of systems that are designed for reactor shutdown or residual heat removal.

3) portions of the steam systems extending from the outermost CIV up to but not including the turbine stop and bypass valves and connected piping up to and including the first valve that is either normally closed or capable of automatic closure during normal reactor operation.

4) systems or portions of systems that are connected to the RCPB and are not capable of being isolated from the boundary during normal reactor operation by two valves, each of which is normally closed or capable of automatic closure.

5) systems or portions of systems that are designed for:

   (1) emergency core cooling, (2) post-accident containment heat removal, or (3) post-accident fission product removal.

The ASME Class 3 boundaries, based on RG 1.26 (Reference 6.6-3), for Quality Group C are not part of the RCPB but are

1) cooling water systems or portions of cooling water systems that are designed for emergency core cooling, post-accident containment heat removal, post-accident containment atmosphere cleanup, or residual heat removal from the reactor and from the spent fuel storage pool (including primary and secondary cooling systems). Portions of these systems that are required for their safety functions and that do not operate during normal operation and cannot be tested adequately, however, are included in Class 2.

2) cooling water and seal water systems or portions of these systems that are designed for the functioning of other Class 2 or 3 components and systems.

3) systems or portions of systems that are connected to the RCPB and are capable of being isolated from that boundary during normal reactor operation by two valves, each of which is normally closed or capable of automatic closure.

4) systems other than radioactive waste management systems, not covered by items (1), (2) and (3) above, that contain or may contain radioactive material and whose postulated failure would result in conservatively calculated potential offsite doses (RGs 1.3 and 1.4), that exceed 0.5 rem to the whole body or its equivalent to any part of the body.

The ASME Code Class 2 and 3 pressure-retaining components and supports subject to examination include pressure vessels, piping, valves, and their bolting. Preservice and in-service examinations, tests and inspections are performed in accordance with ASME Code Section XI (Reference 6.6-1), including associated Mandatory Appendices. Impingement shield and restraints are not considered pressure retaining components or attachments and are not subject to examination within the ISI Program.

The control rod drive mechanism cooling coils are ASME Class 2 components. The testing and inspection programs for these components are described in Section 3.9.4.

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

Table 6.6-1 specifies the defined Code ANSI Safety Class for components that have a safety function in accordance with ASME Section III (Reference 6.6-2), Article NCA-2000 and defines the relationship between these Code Classes and NRC RG 1.26 (Reference 6.6-3).

The ISI Program periodically monitors systems or components requiring ISI in order to identify and repair the indications thus ensuring their capability to perform their required functions.

The ASME Code Class 2 pressure retaining components and integral attachments are examined in accordance with the requirements of ASME Section XI Article IWC-2500.

The ASME Code Class 3 pressure retaining components and integral attachments are examined in accordance with the requirements of ASME 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) are designed and provided with access to enable the performance of in-service examinations.

The design and layout of the Code Class 2 and 3 systems allows for the performance of the ISI requirements contained in ASME Section XI, Articles IWC-2000 and IWD-2000, and as defined in the ISI Program. An inspection method that complies with the performance demonstration requirements of ASME Section XI Appendix VIII and 10 CFR 50.55a(b)(2)(xvi)(B) and 10 CFR 50.55a(b)(2)(xv)(A)(2) is provided as part of the inspection procedures. Examinations are PDI qualified.

The ASME Code Class 2 and 3 components requiring inspection are designed for and are provided with access to enable the performance of ASME Section XI inspections. Systems and components are designed so that design, materials, and geometry do not restrict inspections required by ASME Section XI. The piping arrangement allows for adequate separation of piping welds so that space is available to perform ISI. Adjacent welds are separated by sections of straight pipe of sufficient length to conduct inspections. Welds in piping that passes through walls are located away from the wall as required by ASME Code Section XI. Component nozzles, tees, elbows, valves, branch connections, and other fittings are not connected to each other unless they are specifically designed with an extended tangent length adjacent to the weld to permit weld examination.

Space is provided in accordance with IWA-1500(d) for the performance of examinations alternative to those specified in the event that structural defects or modifications are revealed that require alternative examinations. Space is also provided per IWA-1500(e) for necessary operations associated with repair or replacement activities.

Piping and pipe support locations, integral jet impingement shield and pipe whip restraints (ISRs), insulation, hangers, and stops are designed so as not to interfere with the inspection equipment and personnel. Where this cannot be done, the components are designed to be removable. ISRs are designed to be removable to allow piping and weld examinations.

Welds and other areas requiring periodic inspection are accessible. Reinforcing pads, supports, piping, and equipment are located so as not to obstruct welds. Where insulating materials are used above the containment vessel (CNV), the insulation is designed to be removable to provide accessibility for ISI.

Working platforms are provided in areas requiring inspection and servicing of valves. Temporary or permanent working platforms, walkways, scaffolding, and ladders are provided to 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 Class 2 and 3 components are included in modules fabricated offsite and shipped to the site. The modules are designed and engineered to provide access for preservice inspections, ISI, and maintenance activities.

6.6.3 Examination Techniques and Procedures

The examination techniques used for ISI 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 Section XI, Articles IWA-2000, IWC-2000 and IWD-2000. Preservice inspection and subsequent ISI are conducted with equivalent equipment and techniques.

Preservice inspections are performed once, in accordance with ASME Section XI, IWC-2000 and IWD-2000.

Ultrasonic examination, equipment, and procedures are qualified in accordance with ASME Section XI, Appendices VII and VIII.

No ASME Code Cases, as listed in NRC RG 1.147 (Reference 6.6-4), are used.

The following alternative examination methods, combination of methods, and newly developed techniques are used to implement the Class 2 and Class 3 ISI Program. These methods and techniques meet the acceptance standards of Section XI, Articles IWC-3000 and IWD-3000.

Alternate examination methods are not used.

6.6.4 Inspection Intervals

The inspection intervals for Class 2 examinations in the ISI program are in accordance with Section XI, Articles IWA-2400 and IWC-2400. An example of the selection of Code Class 2 items and examinations to be conducted within a 10-year interval are described in Table 6.6-1.

The inspection intervals for Class 3 examinations in the ISI program are in accordance with Section XI, Articles IWD-2400 and IWF-2400. An example of the selection of Code Class 3 items and examinations to be conducted within a 10-year interval are described in Table 6.6-1.

Examinations may be performed on-line or during normal plant outages, such as refueling or maintenance outages, that occur within the specified inspection interval.

The examination program for the 120-month inspection interval is described in the Reactor Module Test Inspection Elements report and are fully developed in the Owners ISI Program to be developed as part of the COL (see COL Item 6.6-2). The initial in-service inspection program incorporates the latest edition and addenda of the ASME Code approved in 10 CFR 50.55a(b) on the date 18 months prior to fuel load. ISI of components and system pressure tests are conducted during successive 120-month inspection intervals conform with the requirements of the latest edition and addenda of the Code incorporated by reference in 10 CFR 50.55a(b) 18 months before the start of the 120-month interval, 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 Section XI, Article IWC-2400 and Table IWC-2500-1.

For Class 3 systems and components, the examination categories and requirements are in conformance with 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 Section XI, Article IWF-1200.

Table 6.6-1 provides a summary of the Class 2 and Class 3 SSC, their Examination Categories and Methods.

6.6.6 Evaluation of Examination Results

Evaluations of Class 2 examination results are performed in accordance with Section XI, Articles IWA-3000 and IWC-3000.

Evaluations of Class 3 examination results are performed in accordance with Section XI, Articles IWA-3000 and IWD-3000.

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

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

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

6.6.7 System Pressure Tests

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

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

6.6.8 Augmented In-service 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 (refer to Section 6.1.1).

High-energy piping systems are defined in Section 3.6.2 as fluid systems that, during normal plant conditions, are either in operation or maintained pressurized under conditions where either the maximum operating temperature exceeds 200 deg. F or the maximum operating pressure exceeds 275 psig.

The areas subject to examination, method of examination and frequency are summarized in Table 6.6-1. The areas to be examined include the high-energy fluid piping systems described in Section 3.6.1 and 3.6.2.

These areas include

1) the high-energy fluid system piping between CIVs and, where no isolation valve is used inside containment, between the first rigid pipe connection to the containment penetration or the first pipe whip restraint inside containment and the outside isolation valve.

2) the break exclusion portion of the high-energy fluid systems described in Section 3.6.1.

The protective measures, structures, and ISRs installed to prevent postulated high energy line failures are designed to allow personnel access to perform the ISIs.

6.6.9 References

6.6-1 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2013 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, 2013 Edition, Section III Division 1, "Rules for Construction of Nuclear Facility Components," New York, NY.

6.6-3 U.S. Nuclear Regulatory Commission, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Regulatory Guide 1.26, Rev. 4, March 2007.

6.6-4 U.S. Nuclear Regulatory Commission, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," Regulatory Guide 1.147, Rev. 17, August 2014.

6.6-5 U.S. Nuclear Regulatory Commission, "Combined License Applications for Nuclear Power Plants (LWR Edition)," June 2007.

6.6-6 U.S. Nuclear Regulatory Commission, "Design-Specific Review Standard for NuScale SMR Design (DSRS) 6.6 In-service Inspection and Testing of Class 2 and 3 Components," Chapter 6, Section 6.6, June 2016.

6.6-7 U.S. Nuclear Regulatory Commission, "Developing Inservice Testing and Inservice Inspection Programs Under 10 CFR Part 52," Regulatory Issue Summary 2012-08, Rev 1, July 2013.

Table 6.6-1: Examination Categories and Methods

Class 2 Piping and Support Welds		
Description	**Examination Category**	**Methods**
Decay Heat Removal Piping		
DHR INNER CNV SE TO PIPING	C-F-1	Surface and Volumetric
DHR CONDENSER FW OUTLET TO PIPING	C-F-1	Surface and Volumetric
DHR CONDENSER STEAM INLET TO PIPING	C-F-1	Surface and Volumetric
DHR HEADER TO FIT-UP PIPE	C-F-1	Surface and Volumetric
DHR HEADER TO TEE	C-F-1	Surface and Volumetric
DHR HEADER TO FIT-UP PIPE	C-F-1	Surface and Volumetric
PIPE TO SWAGELOK	None	Exempted by IWC- 1221(a)(1) NPS4 and smaller
DHR TEE TO PIPE	C-F-1	Surface and Volumetric
DHR PIPE TO ELBOW	C-F-1	Surface and Volumetric
DHR ELBOW TO VALVE	C-F-1	Surface and Volumetric
PIPE TO VALVE	C-F-1	Surface and Volumetric
PIPE TO FITTING LINE HEADER	None	Exempted by IWC-1221(a)(1) NPS4 and smaller
PIPE TO WELDOLET LINE	None	Exempted by IWC-1221(a)(1) NPS4 and smaller
FITTING TO FITTING	None	Exempted by IWC-1221(a)(1) NPS4 and smaller
PIPE TO CLAMP	F-A	VT-3
Containment Evacuation Piping		
DUAL BODY ISOLATION VALVE TO SAFE-END (CNV10)	C-F-1	Surface
Containment Flooding and Drain Piping		
DUAL BODY ISOLATION VALVE TO SAFE-END (CNV11)	C-F-1	Surface and Volumetric
PIPE TO SAFE-END (CNV11)	None	Exempted by IWC 1221(a)(1) NPS4 and smaller
PIPE TO PIPE	None	Exempted by IWC 1221(a)(1) NPS4 and smaller
PIPE TO LUG	None	Exempted by IWF-1230
PIPE TO SUPPORT	F-A	VT-3
RCCWS Piping		
DUAL BODY ISOLATION VALVE TO SAFE-END (CNV5)	C-F-1	Surface
DUAL BODY ISOLATION VALVE TO SAFE-END (CNV12)	C-F-1	Surface
PIPE TO SAFE-END	C-F-1	Surface
PIPE TO CAP	C-F-1	Surface
PIPE TO NOZZLE TO BASE	None	Exempted by IWC 1221(a)(1) NPS4 and smaller
FITTING TO PIPE	None	Exempted by IWC 1221(a)(1) NPS4 and smaller
PIPE TO REDUCER	None	Exempted by IWC 1221(a)(1) NPS4 and smaller
REDUCER TO REDUCER	None	Exempted by IWC 1221(a)(1) NPS4 and smaller

Table 6.6-1: Examination Categories and Methods (Continued)

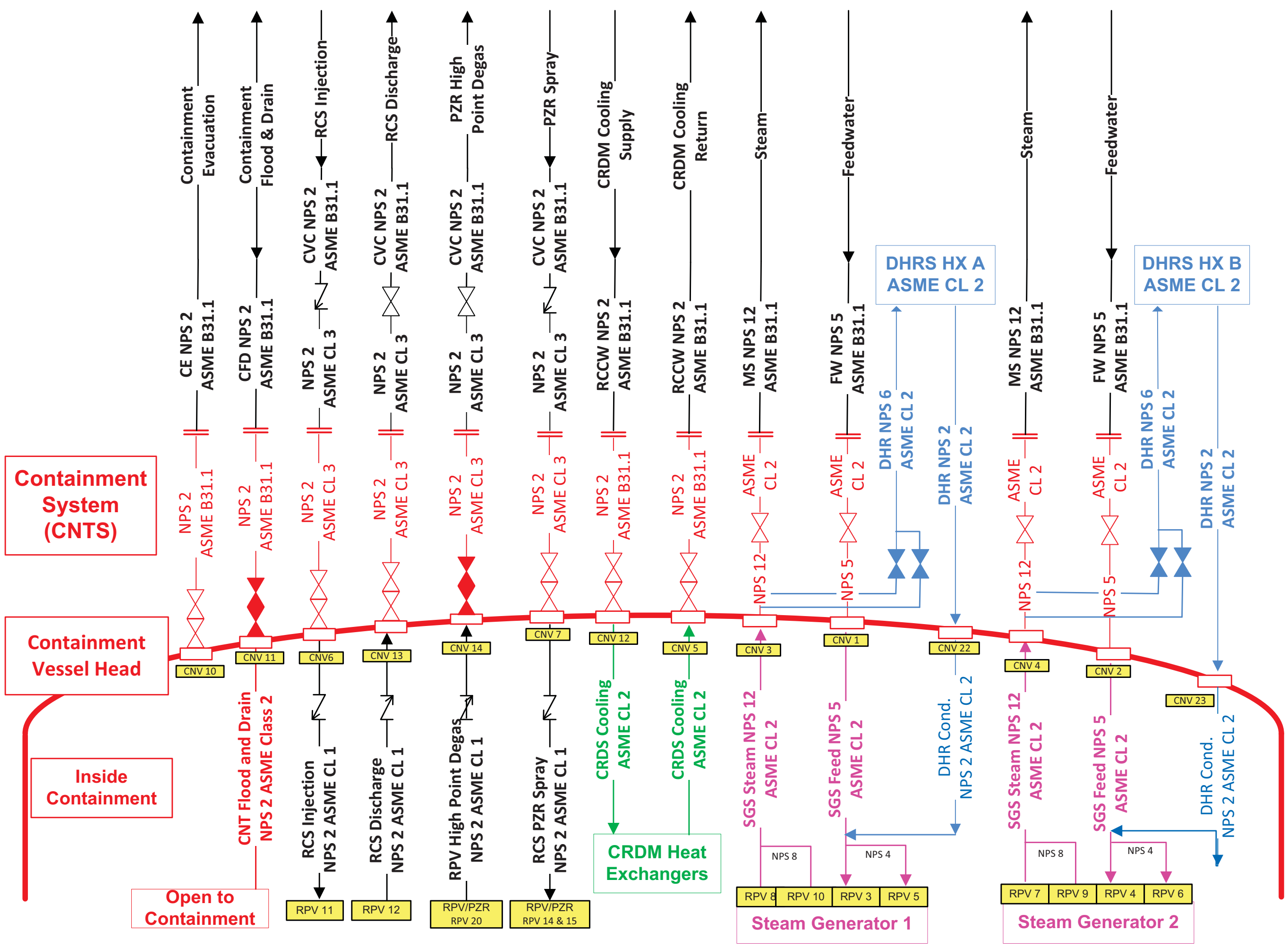
Class 2 Piping and Support Welds		
Description	**Examination Category**	**Methods**
REDUCER TO FLANGE	None	Exempted by IWC 1221(a)(1) NPS4 and smaller
U-BOLT TO SUPPORT BRACKET	None	Exempted by IWF-1230
Main Steam Piping		
CUSTOM TEE TO ELBOW	C-F-1	Surface and Volumetric
CUSTOM TEE TO SAFE-END	C-F-1	Surface and Volumetric
PIPE TO ELBOW	C-F-1	Surface and Volumetric
PIPE TO SAFE END	C-F-1	Surface and Volumetric
PIPE TO PIPE	C-F-1	Surface and Volumetric
PIPE TO CUSTOM TEE	C-F-1	Surface and Volumetric
PIPE TO VALVE	C-F-1	Surface and Volumetric
PIPE TO FLANGE	C-F-1	Surface and Volumetric
PIPE TO SOCKOLET	C-F-1	Exempted by IWC-1221(a)(1) NPS4 and smaller
PIPE TO THERMOWELL	C-F-1	Exempted by IWC-1221(a)(1) NPS4 and smaller
PIPE TO WELDOLET	None	Exempted by IWC-1221(a)(1) NPS4 and smaller
CUSTOM TEE TO VALVE	C-F-1	No description.
CUSTOM TEE TO DHR PIPE	C-F-1	Surface and Volumetric
CUSTOM TEE TO SAFE END	C-F-1	Surface and Volumetric
CUSTOM TEE TO CUSTOM TEE	C-F-1	Surface and Volumetric
Feedwater Piping		
PIPE TO PIPE	C-F-1	Surface and Volumetric
PIPE TO SAFE-END	C-F-1	Surface and Volumetric
PIPE TO CUSTOM TEE	C-F-1	Surface and Volumetric
VALVE TO PIPE	C-F-1	Surface and Volumetric
PIPE TO FLANGE	C-F-1	Surface and Volumetric
PIPE TO ELBOW	C-F-1	Surface and Volumetric
THERMOWELL TO PIPE	C-F-1	Exempted by IWC-1221(a)(1) NPS4 and smaller
VALVE TO SAFE END	C-F-1	Surface and Volumetric
FEEDWATER DHR TEE TO DHR INLET	None	Exempted by IWC-1221(a)(1) NPS4 and smaller
HSS TO HSS	F-A	VT-3
CVCS Piping		
CVCS DISCHARGE - PIPE TO DUAL BODY ISOLATION VALVE	D-A	None - Exempted by IWD-1220 (a) NPS4 and smaller
CVCS DISCHARGE - PIPE TO TEE	D-A	None - Exempted by IWD-1220 (a) NPS4 and smaller
CVCS DISCHARGE - FLANGE TO TEE	D-A	None - Exempted by IWD-1220 (a) NPS4 and smaller
CVCS DISCHARGE - TEE TO REDUCER	D-A	None - Exempted by IWD-1220 (a) NPS4 and smaller
CVCS DISCHARGE - REDUCER TO FLANGE	D-A	None - Exempted by IWD-1220 (a) NPS4 and smaller
CVCS INJECTION - PIPE TO DUAL BODY ISOLATION VALVE	D-A	None - Exempted by IWD-1220 (a) NPS4 and smaller

Table 6.6-1: Examination Categories and Methods (Continued)

Class 2 Piping and Support Welds		
Description	**Examination Category**	**Methods**
CVCS INJECTION - PIPE TO TEE	D-A	None - Exempted by IWD-1220 (a) NPS4 and smaller
CVCS INJECTION - FLANGE TO TEE	D-A	None - Exempted by IWD-1220 (a) NPS4 and smaller
CVCS INJECTION - TEE TO REDUCER	D-A	None - Exempted by IWD-1220 (a) NPS4 and smaller
CVCS INJECTION - REDUCER TO FLANGE	D-A	None - Exempted by IWD-1220 (a) NPS4 and smaller
CVCS PRESSURIZER - PIPE TO DUAL BODY ISOLATION VALVE	D-A	None - Exempted by IWD-1220 (a) NPS4 and smaller
CVCS PRESSURIZER - PIPE TO TEE	D-A	None - Exempted by IWD-1220 (a) NPS4 and smaller
CVCS PRESSURIZER - FLANGE TO TEE	D-A	None - Exempted by IWD-1220 (a) NPS4 and smaller
CVCS PRESSURIZER - TEE TO REDUCER	D-A	None - Exempted by IWD-1220 (a) NPS4 and smaller
CVCS PRESSURIZER - REDUCER TO FLANGE	D-A	None - Exempted by IWD-1220 (a) NPS4 and smaller
CVCS HIGH POINT VENT -PIPE TO DUAL BODY ISOLATION VALVE	D-A	None - Exempted by IWD-1220 (a) NPS4 and smaller
CVCS HIGH POINT VENT - PIPE TO TEE	D-A	None - Exempted by IWD-1220 (a) NPS4 and smaller
CVCS HIGH POINT VENT - FLANGE TO TEE	D-A	None - Exempted by IWD-1220 (a) NPS4 and smaller
CVCS HIGH POINT VENT - TEE TO REDUCER	D-A	None - Exempted by IWD-1220 (a) NPS4 and smaller
CVCS HIGH POINT VENT - REDUCER TO FLANGE	D-A	None - Exempted by IWD-1220 (a) NPS4 and smaller
Support Welds		
RXM TOP SUPPORT WELDS	F-A	VT-3
CVCS DISCHARGE - SUPPORT TO PLATFORM	None	Exempted by IWF-1230
CVCS INJECTION - SUPPORT TO PLATFORM	None	Exempted by IWF-1230
CVCS PRESSURIZER - SUPPORT TO PLATFORM	None	Exempted by IWF-1230
CVCS HIGH POINT VENT - SUPPORT TO PLATFORM	None	Exempted by IWF-1230

General: The preservice examination includes the items selected for in-service examination in each examination category with the exception of Category C-H, in accordance with IWC-2200, and the VT-2 examination of pressure retaining surfaces in Categories D-A, D-B and D-C, in accordance with IWD-2200. Preservice examination of supports are performed following the initiation of hot functional or power ascension tests.

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.