RAIO-0418-59368

April 02, 2018

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 358 (eRAI No. 9335) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 358 (eRAI No. 9335)," dated February 02, 2018

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The Enclosure to this letter contains NuScale's response to the following RAI Question from NRC eRAI No. 9335:

- 05.02.01.01-7

This letter and the enclosed response make no new regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions on this response, please contact Carrie Fosaaen at 541-452-7126 or at cfosaaen@nuscalepower.com.

Sincerely,

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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9335

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9335

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9335
Date of RAI Issue: 02/02/2018

NRC Question No.: 05.02.01.01-7

In accordance with Title 10 of the Code of Federal Regulations (10 CFR) Section 50.55a, certain systems and components of the NuScale Small Modular Reactor (SMR) design are to meet the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) as well as additional conditions promulgated in 10 CFR 50.55a. These requirements help ensure that facilities will also meet the requirements of 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 1 such that structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

During a clarification call on September 15, 2017, NuScale and the NRC staff discussed the information requests in RAIs 8914 and 8917, which involved compliance with the Codes and Standards Rule and the use of ASME Code Cases. Part of this discussion pertained to the need for NuScale to provide sufficient information to make the necessary safety findings for all components within the scope of SRP 5.2.1.1.

At this time, the NRC staff has not received adequate information to address the full scope of components subject to the Codes and Standards Rule, 10 CFR 50.55a. Specifically, NuScale has discussed the components of the reactor coolant pressure boundary in their responses to RAI 8914, but has not adequately described the requirements for non-RCPB components, which is necessary to make the necessary safety findings for SRP 5.2.1.1. This follow-up RAI is issued to request a supplement to certain topics discussed in RAI 8914 that require additional information to fully address the scope of components subject to the Codes and Standards Rule, 10 CFR 50.55a.

The response to Question 30096 (05.02.01.01-5) requires additional information, as there is no statement in the DCD indicating that Quality Group B and C components meet the applicable conditions promulgated in 10 CFR 50.55a(b). Section 3.2.2.2 and 3.2.2.3 of the DCD indicate that Quality Group B and C SSCs meet the requirements for Class 2 and Class 3 components in Section III, Division 1 of the ASME B&PV Code, respectively, but is silent on meeting the applicable conditions promulgated in 10 CFR 50.55a(b), which is a regulatory requirement. The applicant is requested to confirm within the DCD that Quality Group B and C components are designed, fabricated, constructed, tested, and inspected as Class 2 and 3 (respectively) in accordance with Section III, of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) and meet the applicable conditions promulgated in 10 CFR 50.55a(b).

The response to Question 30098 (05.02.01.01-6) requires additional information, as there is no affirmative confirmation that there are no proposed alternatives to compliance with 10 CFR 50.55a with regards to Quality Group B and C components. NuScale's RAI response only discussed RCPB components. Staff must confirm that there are no proposed alternatives to 10 CFR 50.55a for Quality Group B and C components in order to make a finding regarding the acceptability of proposed alternatives to compliance with 10 CFR 50.55a. The applicant is requested to confirm within the DCD that there are no proposed alternatives to compliance with 10 CFR 50.55a for Quality Group B and C components in the NuScale design.

The RAI response for RAI 8914, Question 30093 (05.02.01.01-2) indicated that adding statements regarding compliance of ASME Code Section II, Section V, and Section IX to 10 CFR 50.55a is not appropriate because 10 CFR 50.55a does not address these ASME Code Sections. Staff agrees that 10 CFR 50.55a does not directly mention these sections. However, GDC 1 requires structures, systems, and components important to safety to be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Furthermore, the ASME Code is an integrated set of requirements, with references to other sections interwoven into each section. These references are not incorporated by reference in 10 CFR 50.55a. Therefore, to make a safety finding under 10 CFR 50 Appendix A, GDC 1, the standards used for material selection, examination, and welding need to be established within the DCD. The applicant is requested to identify within the DCD what Codes will be used for construction base materials and welding materials, inspection of structures, systems, and components constructed in accordance with ASME Code Section III, and qualification of welding procedures and welding operators (such as ASME Code Section II, V, and IX, respectively). Staff notes that the response to RAI 8914, Question 30093 (05.02.01.01-2) identified that 'qualification of welding procedures and welding operators to be in accordance with ASME BPVC Code Section IX, "Welding and Brazing Qualifications."' and "the materials selected for fabrication of the RCPB comply with the requirements of ASME BPVC, Section II," so identification of these Codes in the DCD maintains consistency with the original RAI response, but properly expands them to also include non- RCPB components still subject to ASME Code requirements.

NuScale Response:

NuScale reviewed the NRCs Request for Additional Information above and determined there were three separate requests included:

1 - ***Quality Group B and C Component Design***

The applicant is requested to confirm within the DCD that Quality Group B and C components *are designed, fabricated, constructed, tested, and inspected as Class 2 and 3 (respectively) in accordance with Section III, of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) and meet the applicable conditions promulgated in 10 CFR 50.55a(b).*

<u>2 - Proposed Alternatives to Compliance with 10 CFR 50.55a for Quality Group B and C Components</u>

The applicant is requested to confirm within the DCD that there are no proposed alternatives to compliance with 10 CFR 50.55a for Quality Group B and C components in the NuScale design.

<u>3 - Codes used for Construction Base Materials, Welding Materials, Inspection of SSCs</u>

The NRC requested the following information be provided: The applicant is requested to identify within the DCD what Codes will be used for construction base materials and welding materials, inspection of structures, systems, and components constructed in accordance with ASME Code Section III, and qualification of welding procedures and welding operators (such as ASME Code Section II, V, and IX, respectively)… include non-RCPB components still subject to ASME Code requirements.

A response to each of these three NRC question is provided below.

<u>Question 1 Response:</u>

NRC Standard Review Plan (SRP) SRP 5.2.1.1 (I)(1) specifies that the reactor coolant pressure boundary (RCPB) and safety-related fluid systems comply with 10 CFR 50.55a. The RCPB and safety related fluid systems for the NuScale Power Module (NPM) are addressed within this RAI.

<u>Scope of Systems</u>

The systems located within the NPM are the containment system (CNTS), reactor coolant system (RCS), steam generator system (SGS), decay heat removal system (DHRS) and emergency core cooling system (ECCS). The containment evacuation system (CES) and containment flooding and drain system (CFDS) connect to the CNTS at the NPM disconnect flanges. The SGS and DHRS are safety-related systems used for core heat removal. The portions of the CNTS that form part of the pressure boundary for the RCS, SGS and DHRS are addressed in this RAI. The CES and CFDS discussed in FSAR Section 9.3.6 are not addressed in this RAI, because they are not considered safety-related systems. Cooling for the control rod drive mechanisms is provided by the control rod drive system (CRDS) discussed in FSAR Section 4.6 and is not part of the RCPB or a safety-related system. The ECCS consists of the reactor vent valves (RVV), reactor recirculation valves (RRV), RVV trip-reset pilot valves and RRV trip-reset pilot valves. Since the ECCS is a Quality Group A, Class 1 system it is not addressed as part of this response, but the applicable American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) requirements for ECCS are provided in FSAR Section 5.2.

Containment System Interface

Fluid systems which pass through containment use system specific equipment labels beyond the NPM disconnect flanges. Beyond the containment isolation valves (CIVs) these systems are non-safety related and are not addressed in this RAI, but are identified to provide clarification. Beyond the NPM disconnect flanges the main steam and feedwater lines which flow to and from the SGS are labeled main steam system (MSS) and condensate and feedwater system (CFWS). These systems are discussed in FSAR Sections 10.3 and 10.4.7 respectively. The chemical and volume control system (CVCS) connections to the RCS are labeled as CVCS equipment beyond the NPM disconnect flange and are discussed in FSAR Section 9.3.4. The reactor component cooling water system (RCCWS) connections provided to the CRDS are labeled as RCCWS equipment beyond the NPM disconnect flange and discussed in FSAR Section 9.2.2.

A schematic of the CNTS, systems within containment and systems beyond the NPM disconnect flanges are shown in FSAR Figure 6.2-4 with the respective system names identified.

The DHRS functional flow path includes components of the CNTS and SGS to form the safety-related fluid system within the NPM. The RCPB is formed primarily by RCS components but also includes CNTS and SGS components. The Quality Group of the RCS, SGS, and DHRS components are identified below. Additionally, the FSAR sections identifying the system and where Quality Group B components are specified to be designed, fabricated, constructed, tested, and inspected as Class 2 in accordance with Section III, of the ASME BPVC are also provided. There are no Quality Group C, Class 3 RCPB or safety-related components in the NPM.

The NuScale CNTS is discussed in FSAR Section 6.2 and includes the containment vessel (CNV), CNV supports, passive containment isolation barriers, containment isolation valves (CIVs), and containment instruments. The CNV includes the piping nozzle penetrations and safe ends attached to the inside and outside of the penetrations.

The MS and FW nozzle penetrations through the CNV which connect to the outside safe ends and the SGS to the inside safe ends, plus the DHRS nozzle penetrations of the CNV and safe ends are CNTS components that form part of the safety-related fluid systems. The passive containment isolation barrier includes the MS piping from the outside safe end of the CNV nozzle to the main steam isolation valve (MSIV) as well as the DHRS penetration and safe ends. FSAR Section 6.2.4 provides additional discussion of the passive containment isolation barrier.

FSAR Section 6.2.4.2.2 identifies that the primary system CIVs (PSCIVs) that are open to the CNV (i.e. CES and CFDS) or connected to closed loop piping inside containment (i.e. CVCS) are designated as Quality Group B, but designed, fabricated, constructed, tested and inspected as Quality Group A, Class 1. FSAR Section 6.2.4.2.2 also identifies that the PSCIVs for the RCS injection, RCS discharge, PZR spray and RPV high point degasification penetrations are designed, fabricated, constructed, tested and inspected as Quality Group A, Class 1. The feedwater isolation valve (FWIV), main steam isolation valve (MSIV) and main steam bypass isolation valve (MSBIV) are secondary system CIVs (SSCIVs) and are designed, fabricated, constructed, tested and inspected as Quality Group B, Class 2.

FSAR Table 6.2-4 identifies the individual CNV penetration Quality Group designation. FSAR Section 6.2.4.2.2 identifies that the PSCIVs connected to RCPB piping are designed and constructed to Class 1 and the SSCIVs are designed as Class 2. FSAR Section 6.2.4.2.2.3 identifies that the main steam piping from the CNV safe end to the CIV is constructed to Class 2. FSAR Sections 6.2.4.2.2 and 6.2.4.2.2.3 have been modified to state that the SSCIVs and main steam piping are designed, fabricated, constructed, tested and inspected as Class 2 in accordance with Section III of the ASME BPVC. FSAR Section 6.2.4.2.2 has also been modified to state that the PSCIVs are designed, fabricated, constructed, tested and inspected as Class 1 in accordance with Section III of the ASME BPVC.

Reactor Coolant System

FSAR Section 5.2 defines the RCS components which consist of the RPV including the integrated pressurizer, reactor safety valves (RSVs), ECCS, pressure housing of the control rod drive mechanisms, pressurizer heater bundles and piping from the CNTS for RCS injection, RCS discharge, PZR spray and RPV high point degasification. FSAR Section 5.2.1.1 identifies these components are classified as Quality Group A components, which do not include any Quality Group B or C components, and that they are designed, fabricated, constructed, tested and inspected as Class 1 in accordance to Section III of the ASME BPVC.

Steam Generator System

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

The CNV nozzle penetrations and safe ends on the inside and outside of the penetration for the SGS are part of the CNTS. Likewise, the main steam piping from the safe end on the outside of the CNV penetration to the MSIV is part of the CNTS. FSAR Figure 5.4-9 shows a schematic diagram of the SGS.

The feedwater plena, SG tubes and main steam plena also form part of the RCPB and are identified as Quality Group A components in FSAR Table 3.2-1. The SGS is discussed in FSAR Section 5.4.1.2 and specifies the SGS RCPB components as Quality Group A designed to Class 1 requirements. FSAR Section 5.4.1.2 has been modified to state that SGS RCPB components are designed, fabricated, constructed, tested and inspected as Class 1 in accordance with Section III of the ASME BPVC.

FSAR Table 3.2-1 under the SGS system identifies the main steam and feedwater piping within the CNV, feedwater nozzle, main steam nozzles and thermal relief valves as Quality Group B components. Section 5.4.1.2 identifies that the main steam and feedwater lines between the feedwater isolation valve and main steam isolation valve are designed as Class 2 components. FSAR Section 5.4.1.2 has been modified to state that SGS components are designed, fabricated, constructed, tested and inspected as Class 2 in accordance with Section III of the ASME BPVC.

Decay Heat Removal System

The DHRS consists of the steam inlet piping from the tee in the CNTS main steam pipe, between the MSIVs and CNV up to the DHRS actuation valves (DHRV), the DHRV, DHRS piping from the DHRV to the DHRS condenser, piping from the condenser to the CNV, and piping from the CNV to the tee in the SGS feedwater piping. The CNV nozzle penetration and safe end on the inside and outside of the penetration are part of the CNTS. FSAR Figure 5.4-10 shows a schematic diagram of the DHRS.

FSAR Table 3.2-1 identifies that the DHRS actuation valve and condenser are Quality Group B components. FSAR Section 5.4.3 discusses this system and Section 5.4.3.1 identifies that the DHRS is Quality Group B and designed as Class 2 in accordance with Section III of the ASME BPVC. FSAR Section 5.4.3.1 has been modified to state that the DHRS is designed, fabricated, constructed, tested and inspected as Class 2 in accordance with Section III of the ASME BPVC.

Conclusion

In conclusion, the NuScale design for the applicable portions of the CNTS, SGS, and DHRS are designed, fabricated, constructed, tested, and inspected as Class 2 in accordance with Section III, of the ASME BPVC and meet the applicable conditions promulgated in 10 CFR 50.55a(b). There are no NPM components that form the RCPB or a safety-related system that are designated as Class 3. See Table 1 below, for a summary of the statements provided in response to Question 1.

Question 2 Response:

The “Augmented Design Requirements” column in FSAR Table 3.2-1 documents that there are no proposed alternatives to compliance with 10 CFR 50.55a for the Quality Group B and C components in the NPM design.

Question 3 Response:

The NPM design meets the following code requirements:

- Construction and welding of structures, systems, and components are performed in accordance with the ASME BPVC, Section III.
- Specifications for construction base materials and welding materials satisfy the requirements of ASME BPVC, Section II.
- Specifications for Qualification of Welding Procedures and Welding Operators satisfy the requirements of ASME BPVC Section IX
- The NPM is designed and provides access to enable the performance of the Inservice Inspection requirements specified in ASME BPVC, Section XI.
- Preservice examination requirements are specified in accordance with the requirements set forth in the editions and addenda of Section XI and the ASME BPVC incorporated by reference in paragraph (a)(1)(ii) of 10 CFR 50.55a as applied to the construction of the component.
- Inspections of SSCs constructed in accordance with ASME Section III are performed in accordance with ASME BPVC, Section V.
- The NPM is designed to provide access to enable the performance of inservice testing of valves in accordance with 10 CFR 50.55a(f)(3)(iii)(B).

Table 1 below, identifies the FSAR sections where information regarding the applicable sections of the ASME BPVC used for design, material, inspection and welding can be found. These same FSAR sections also provide the fabrication, construction, and testing information described in the Question 1 response.

Table 1 - FSAR Sections Providing Responses to Questions 1 and 3

System	Component	Class	Applicable FSAR Sections			
			Design	Material	Inspection	Welding
RCS	RCS Injection Piping RCS Discharge Piping RCS PZR Spray Piping RPV High-Point Degasification Piping Check Valves	1	5.4.2.5	5.4.2.5	5.4.2.5	5.4.2.5
CNTS	CNV Penetration and Safe Ends	1	6.2.1.1.3.2	3.8.2.2.2	6.2.1.1.6	3.8.2.6
	PSCIVs		6.2.4.2.2	6.2.4.2.2	6.2.4.4	6.2.4.2.2
	SSCIVs MS Piping	2	6.2.4.2.2 / 6.2.4.2.2.3	6.2.4.2.2	6.2.4.4	6.2.4.2.2
SGS	Feed Plenum Port and Access Cover SG Tubes Steam Cap	1	5.4.1.2	5.4.1.5	5.4.1.1	5.4.1.5
	FW Piping Steam Access Port and Access Cover MS Piping Thermal Relief Valve	2	5.4.1.2	5.4.1.5	5.4.1.1	5.4.1.5
DHRS	Piping DHRV Condenser	2	5.4.3.1	5.4.3.2	5.4.3.4	5.4.3.2

Impact on DCA:

FSAR Sections 5.4.1, 5.4.2, 5.4.3, and 6.2.4 have been revised as described in the response above and as shown in the markup provided in this response.

5.4 Reactor Coolant System Component and Subsystem Design

The reactor pressure vessel (RPV) of each integrated NuScale Power Module (NPM) contains the reactor and reactor vessel internals; a pressurizer; two steam generators (SGs); reactor safety valves (RSVs); emergency core cooling system (ECCS) valves; reactor coolant system (RCS) injection, discharge, pressurizer spray, and high-point degasification vent lines; and a decay heat removal system (DHRS).

The design basis and description of the reactor and reactor vessel internals are provided in Chapter 4. The design basis and description of the RSVs are provided in Section 5.2.2 and the design basis and description of the ECCS valves (i.e., reactor vent valves (RVVs) and reactor recirculation valves (RRVs)) are provided in Section 6.3 and Section 5.2.2.

5.4.1 Steam Generators

RAI 05.02.01.01-7

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

The SGs in the NPM are integral to the RPV. The RPV forms the SG shell and provides the outer pressure boundary of the SGs. The steam generator tube, tube-to-tubesheet welds, and tubesheets provide part of the reactor coolant pressure boundary (RCPB). Refer to Section 5.2 and Section 5.3 for description and design basis information regarding the RPV and the RCPB.

5.4.1.1 Design Basis

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

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

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

The secondary flow oscillation magnitude is limited by flow restriction devices at the secondary side inlet of each individual SG tube.

prevent adverse interaction with non-plugged tubes. Access to the internal (secondary) and external (reactor coolant) sides of tubesheets affords opportunity for inspection, and for removal of foreign objects. See Figure 5.4-4 and Figure 5.4-5 for an illustration of the steam and feed plena inspection ports.

RAI 05.02.01.01-7

The ~~SG~~FW plenum and MS plenum SGS subcomponents and SG tubes that form portions of the RCPB are classified Quality Group A and ~~designed~~are designed, fabricated, constructed, tested and inspected as Class 1 in accordance with Section III of the BPVC. Steam and feedwater ~~lines~~ piping between the ~~FWIVs and MSIVs~~CNTS and RPV, including those portions that interface with and support operation of the DHRS, are ~~designed~~designed, fabricated, constructed, tested and inspected as Class 2 in accordance with the BPVC, Section III. The SGS and connected components up to the ~~FWIVs and MSIVs~~CNTS are Seismic Category I components. Details of the SG classification designations and the scope of their applicability are provided in Table 3.2-1. Chapter 3 provides detailed information regarding the design basis and qualification of structures, systems, and components based on these classifications and designations. Figure 6.6-1 shows the BPVC Section III, Class 1 and 2 boundaries for the SGS.

RAI 05.04.02.01-6

Steam Generator Tube Supports and Steam Generator Supports

RAI 05.04.02.01-6

Based on the use of seamless helical tubing to comprise the tube bundle, typical SG tube support plates are not used. Instead, the NPM steam generator employs a system of austenitic stainless steel tube support assemblies. The design of the stainless steel supports includes full-circumferential support of the tubes. The circumferential support is not continuous and therefore limits the potential for crevices between the tube and support. By choosing materials that limit the potential for generation and buildup of corrosion products and a geometry that minimizes crevices and facilitates flow (further limiting potential for corrosion product buildup), two of the most significant historical contributors to tube degradation by the tube supports are precluded.

RAI 05.04.02.01-6

The tube support assemblies are located between each column of helical tubes. Stamped tabs in the tube supports envelope part of the circumference of tubes both above and below, and provide vertical tube support as shown in Figure 5.4-7. The overall design of the tube support assemblies and tabs minimizes the stagnation of flow at the tube-to-support interface precluding the buildup of deposits. Likewise, the tube support structure is located within the primary coolant environment; therefore, no ingress path exists for general corrosion products from the secondary system to deposit on the shell side of the SG as may occur in traditional SG designs. Outer and middle spacers welded into the pockets in the back of the tube supports (see Figure 5.4-6) allow for the tabs from each adjacent column to nest with each other to create a continuous support path through the columns. The circumferential spacing of the tube supports is optimized to provide the minimum possible tube free span lengths, while still accommodating the transition of the tubes to the steam and feedwater plena.

RAI 05.04.02.01-6

The SG tubes are supported for vibration and seismic loads by vertical bars that extend through the tube bundle from the feed to the steam plena. As shown on Figure 5.4-6, the tube support assemblies are attached to upper SG supports that are welded to the inner surface of the RPV and also interface with lower SG supports that are welded to the inner surface of the RPV. The SG tube support assemblies in the SG provides contact with each tube at eight separate circumferential locations. The use of 8 sets of tube support assemblies limits the unsupported tube lengths, which ensures SG tube modal frequencies are sufficiently high to preclude unacceptable flow-induced vibration.

RAI 05.04.02.01-6

As shown in Figure 5.4-6, the lower SG supports permit thermal growth and provide lateral support of the tube supports.

Inlet Flow Restrictors

RAI 05.04.02.01-8

A flow restriction device at the inlet to each tube ensures secondary-side flow stability and precludes density wave oscillations. The SG tube inlet flow restrictors provide the necessary secondary-side pressure drop for flow stability. The flow restrictors are mounted on a plate in each feed plenum that is attached to the secondary-side face of the tubesheets with stud bolts to avoid attaching the restrictors directly to the tube. The flow restrictor stud bolts are welded to the tubesheet at each mounting location. Mounting plate spacers hold the flow restrictor mounting plate off the surface of the tubesheet (see Figure 5.4-5). Spacers are located at each mounting plate attachment point. As shown in Figure 5.4-8, the individual flow restrictors extend into the tubes and are removable to support SG inspection, cleaning, tube plugging, or other maintenance and repair activities. The flow restrictor bolts are located at the center of the flow restrictor bolt assembly. The flow restrictor bolt runs the length of the assembly and holds the flow restrictor subcomponent. The flow restrictor bolts or nuts and the flow restrictor stud bolts or nuts include a locking feature to minimize the potential for loose parts generation.

Thermal Relief Valves

RAI 05.02.01.01-7

To establish desired SG and DHRS chemistry during startup and shutdown, the SG and DHRS are flushed to the condenser, creating a water solid condition. Unintended containment isolation during these flushing evolutions could result in overpressure conditions caused by changes in fluid temperature. A single thermal relief valve is located on each feedwater line upstream of the tee that supplies the feed plenums (see Figure 5.4-9) to provide overpressure protection during shutdown conditions for the secondary side of the SGs, feedwater and steam piping inside containment, and the DHRS when the secondary system is water solid and the containment is isolated. The thermal relief valves are spring-operated, balanced-bellows relief valves that vent directly into the containment. The thermal relief valves are classified Quality Group B and ~~designed~~are designed, fabricated, constructed, tested and inspected as Class 2 in accordance with Section III of the BPVC and are Seismic Category I components.

The thermal relief valves provide investment protection for the secondary system components during shutdown conditions and are not credited for safety-related overpressure protection for these systems during operation. Overpressure protection during operation is provided by system design pressure and the RSVs as described in Section 5.2.2.

Main Steam and Feedwater Plena Vent and Drain Valves

Manual valves allow draining the main steam and feedwater plena prior to cover removal to facilitate outage maintenance and testing. The valves are used for maintenance only and are normally closed and capped.

Compatibility of Steam Generator Tubing with Primary and Secondary Coolant

The chemistry of the primary and secondary water is controlled in accordance with industry guidelines suitably modified to address the unique NPM design and to ensure compatibility with the primary and secondary coolant. Section 5.2.3 describes the compatibility aspects of the reactor coolant chemistry that provide corrosion protection for stainless steels and nickel alloys, including SG components exposed to the reactor coolant. Section 6.1 describes the compatibility aspects of the secondary coolant chemistry that provide corrosion protection for stainless steels and nickel alloys, including the SG components exposed to the secondary system coolant and Section 10.3.5 describes the secondary water quality control program which is in accordance with Nuclear Energy Institute (NEI) 97-06 (Reference 5.4-1).

Copper deposits are a major source of SG corrosion products in nuclear plants with copper alloys in their secondary system. To minimize internal SG tube corrosion, low-melting point metals such as lead, antimony, cadmium, indium, mercury, zinc, bismuth, copper, tin, and their alloys and high sulfur materials; with the exception of strong acid cation resin; are excluded from use in reactor coolant primary system components and secondary system components.

Estimated radioactivity design limits for the secondary side of the SGs during normal operation and the basis are addressed in Section 11.1.2. The radiological effects associated with an SG tube failure are provided in Section 15.0.3.8.2.

5.4.1.3 Performance Evaluation

A single RCS natural circulation flow loop is entirely contained within the RPV, thereby eliminating distinct RCS piping loops and the associated potential for a large pipe break (i.e., large break loss-of-coolant accident [LOCA]) event. This design, combined with the intertwined SGs tube bundle configuration, eliminates the potential for asymmetric core cooling and temperatures as a result of a loss of a single SG function. Isolation or other loss-of-heat transfer capability by either of the two intertwined SGs does not introduce asymmetrical cooling in the reactor coolant vessel or system because the tube configuration of the remaining functional SG continues to provide repair is designed in an accessible position to minimize time and radiation exposure during refueling and maintenance outages. Workers can access SG components without being placed at risk for dose or situations where excessive plates, shields, covers, or piping must be moved or removed in order to access components.

The SG program is based on NEI 97-06 and documented in Section 5.5.4 of the technical specifications. Because the SG tube design does not contain U-bends, a volumetric examination is performed on the entire length of the SG tubing as specified in Item B16.10 of Table IWB-2500-1 (B-Q).

Preservice examinations are performed in accordance with the BPVC, Section III, Paragraph NB-5280 and Section XI, Subarticle IWB-2200 using examination methods of BPVC Section V except as modified by Section III, Paragraph NB-5111. These preservice examinations include 100 percent of the pressure boundary welds.

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

As discussed above, the operational inservice testing and inspection programs described in Section 5.2.4 and the SG program described in Section 5.4.1.6 provide testing and inspection requirements following initial plant startup.

5.4.1.5 Steam Generator Materials

Pressure boundary materials used in the SGs and associated components are selected and fabricated in accordance with the requirements of BPVC Section III and Section II as described in Section 5.2.3, and the materials used in the fabrication of the SGs are identified in Table 5.2-4.

RAI 05.02.01.01-7

The RCPB materials used in the ~~SGs and associated components~~SGS are classified as Quality Group A and are designed, fabricated, constructed, tested, and inspected as Class 1 in accordance with the BPVC and the applicable conditions promulgated in 10 CFR 50.55a(b). The ~~SG tubes~~SGS materials forming the RCPB, including weld materials, conform to fabrication, construction, and testing requirements of BPVC, Section III, Subsection NB. The materials selected for fabrication conform to the applicable material specifications provided in BPVC, Section II and meet the requirements of Section III, Article NB-2000. The SG tubes are fabricated with Alloy 690 (UNS N06690) and all SGS materials forming the RCPB are in accordance with BPVC, Section II, ~~Specification SB-163 and~~and meet the requirements of Section III, Article NB-2000. ~~s~~Surfaces of pressure retaining parts of the SGs, including weld filler materials and bolting material, are corrosion-resistant materials, such as stainless steel or nickel-based alloy. The SGs are constructed of materials with a proven history in light water reactor environments and the SG materials associated with the RCPB are listed in Table 5.2-4.

RAI 05.02.01.01-7

The ~~SG~~FW and MS piping from the ~~FWIVs and MSIVs~~CNTS to the SGs, thermal relief valve, steam plenum access ports, and plenum access covers are classified as Quality Group B and are designed, fabricated, constructed, tested, and inspected as Class 2 in accordance with the BPVC and the applicable conditions promulgated in 10 CFR 50.55a(d). The ~~SG~~FW and MS piping, thermal relief valve, steam plenum access ports, and plenum access covers, including weld materials, conform to fabrication, construction, and testing requirements of BPVC, Section III, Subsection NC. The materials selected for fabrication conform to the applicable material specifications provided in BPVC, Section II and meet the requirements of Section III, Article NC-2000. The materials and applicable specifications of the ~~SG~~MS and FW piping, associated reducers and elbows, steam plenum access ports, and plenum access covers and fasteners are provided in Table 5.4-3.

RAI 05.02.01.01-7

Welding of the RCPB portions of the SGS is conducted utilizing procedures qualified in accordance with the applicable requirements of ASME BPVC, Section III, Subarticle NB-4300 and Section IX. Welding of the secondary side portions of the SGS is conducted utilizing procedures qualified in accordance with the applicable requirements of ASME BPVC, Section III, Subarticle NC-4300 and Section IX.

The inside and outside surfaces of the integral steam and feed plenum access ports are clad with austenitic stainless steel. The cladding on the inside surfaces is deposited with at least two layers: the first layer is Type 309L and subsequent layers are Type 308L. The cladding on the outside surfaces is deposited with at least one layer of Type 309L.

The SG weld filler metals are listed in Table 5.2-4 and Table 5.4-3 and are in accordance with BPVC, Section II, Part C.

RAI 05.04.02.01-6

The SG supports and SG tube supports, including weld materials, conform to fabrication, construction, and testing requirements of BPVC, Section III, Subsection NG as a guide.

RAI 05.04.02.01-6

including weld materials, conform to fabrication, construction, and testing requirements of ASME BPVC, Section III, Subsection NB. The materials selected for fabrication conform to the applicable material specifications provided in ASME BPVC, Section II and meet the requirements of ASME BPVC, Section III, Article NB-2000. The check valves are constructed of materials with a proven history in light water reactor environments. Surfaces of pressure retaining parts of the valves, including weld filler materials and bolting material, are corrosion resistant materials such as stainless steel or nickel-based alloy. Materials used for the RCS check valves and associated weld filler metals are provided in Table 6.1-3.

RAI 05.02.01.01-7

Refer to Section 5.2.3 and Section 5.2.4 for additional description of material compatibility, fabrication and process controls, ~~and~~ welding controls and inspections related to the ASME Class 1 components.

5.4.3 Decay Heat Removal System

5.4.3.1 Design Basis

The DHRS provides cooling for non-LOCA design basis events when normal secondary-side cooling is unavailable or otherwise not utilized. The DHRS is designed to remove post-reactor trip residual and core decay heat from operating conditions and transition the NPM to safe shutdown conditions without reliance on external power.

The safety-related DHRS function is an engineered safety feature of the NPM design. Reliability of DHRS is evaluated using the reliability assurance program described in Section 17.4 and risk significance is determined using the guidance described in Chapter 19. The DHRS classification and risk categories are included in Table 3.2-1.

RAI 09.03.06-2S1

The DHRS design ensures the RCS average temperature is below 420 degrees F within 36 hours after an initiating event without challenging the RCPB or uncovering the core. An RCS average temperature of 420 degrees F was chosen based on the safe shutdown temperature proposed by EPRI for passive plant designs in the EPRI Advanced Light Water Reactor Utility Requirements Document (Reference 5.4-3) and determined to be acceptable by the Nuclear Regulatory Commission as documented in SECY-94-084.

The DHRS heat removal function does not rely on actuating ECCS. Any ECCS actuation after a DHRS actuation allows continued residual heat removal by both systems from the reactor core as described in Section 6.3.

Applicable 10 CFR 50 Appendix A, General Design Criteria and Other Design Requirements

RAI 05.02.01.01-7

GDC 1, 2, and 4 - The DHRS is classified Quality Group B and ~~designed~~is designed, fabricated, constructed, tested and inspected as Class 2 in accordance with Section III of the ASME BPVC and is designed, fabricated, and tested to the highest quality standards in accordance with Quality Assurance Program described in Chapter 17. The

DHRS is designed to withstand the effects of natural phenomena without loss of capability to perform its safety function. The DHRS is designed to accommodate the effects of, and be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. The design of the Reactor Building structure, NPM operating bays, and location of the NPM within the operating bays provides protection from possible sources of external or internal generated missiles. The DHRS is protected from pipe whip as described in Section 3.6.

GDC 5 - The DHRS does not share any active or passive components between individual NPMs necessary for performance of the DHRS safety functions. The NPMs share the reactor pool as the ultimate heat sink for removal of decay heat from the DHRS passive condensers. The shared Reactor Building and other structures are described in Chapters 1 and 3 and the reactor pool is described in Section 9.2.5. DHRS active components fail-safe on a loss of power. Therefore, shared power supplies between NPMs do not impact the capability of performing the DHRS safety functions.

GDC 14 - The DHRS is connected to the secondary system and does not directly interface with the RCPB. The SGs are described in Section 5.4.1 and the containment system piping coupling the DHRS to the SGs is described in Section 6.2.4. There are no other interfaces or shared components between the DHRS and the RCPB.

RAI 09.03.06-2S1

GDC 19 - The DHRS is operated from the control room, and is capable of prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown conditions. The DHRS is also actuated and monitored from an alternate shutdown location outside the control room. Once the reactor reaches safe shutdown conditions, non-safety systems are used to lower RCS temperature and pressure to the point the containment can be flooded with reactor pool water allowing the NPM to reach transition conditions using convection and conduction heat transfer.

RAI 09.03.06-2S1

PDC 34 (refer to Section 3.1 for the definition of PDC 34)- The DHRS is a passive design that utilizes natural circulation flow from the SGs to dissipate residual and decay core heat to the reactor pool. The DHRS consists of two independent trains each capable of performing the system safety function in the event of a single failure. The DHRS actuation valves fail open using nitrogen accumulation pressure when electrical power is interrupted to the valves. Therefore, electrical power is not required for system function. A nonsafety-related containment flood and drain system is used to flood the containment to allow cooldown to cold conditions by convection heat transfer from the RPV shell to the CNV shell to allow for disconnection and transfer of the NPM to the refueling area. During NPM transfer to the refueling area, residual and decay heat removal is provided by heat convection and conduction from the reactor to the reactor pool via the RCS, flooded containment, and the RPV and CNV walls. Refer to Section 9.2.5 for discussion related to the reactor pool for GDC 45 and 46, and PDC 44. Refer to Section 3.1 for the definition of PDC 44.

be opened to promote circulation between the RCS and flooded containment. Operation of the RVVs and RRVs is described in Section 6.3.

During NPM movement to and from the refueling area and during refueling, the DHRS does not provide a decay heat removal function. Residual and core decay heat removal during these shutdown conditions is provided by utilizing conduction through the RPV and containment shell with the RVVs and RRVs open or direct contact with the reactor pool water during refueling.

RAI 05.02.01.01-7

The DHRS actuation valves, piping, and passive condensers are classified Quality Group B and designed as Class 2 in accordance with Section III of the BPVC and remain operable following a design basis seismic event. The DHRS supports are designed and fabricated as Class 2 in accordance with BPVC, Section III, Subsection NF. Details of the classification designations and the scope of their applicability are provided in Chapter 3. It provides information regarding the design basis and qualification of structures, systems, and components based on these classifications and designations. Section 6.1 provides details regarding material selection and fabrication methods and discusses compatibility with fluids that the DHRS components may be exposed to during normal, accident, maintenance, and testing conditions. Table 6.1-1 provides a list of material and material specifications for engineered safety feature components, including the DHRS components and supports. Welding of the DHRS is conducted utilizing procedures qualified in accordance with the applicable requirements of ASME BPVC, Section III, Subarticle NC-4300 and Section IX.

The DHRS condenser, actuation valves, and DHRS piping are Seismic Category I components. The DHRS is protected from other natural phenomena by the reactor building structure. The DHRS instrumentation and control components are seismically qualified in accordance with Institute of Electrical and Electronic Engineers (IEEE) 344-2004, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," as modified by the NRC staff position stated in RG 1.100, Revision 3 (Reference 5.4-4).

A portion of the DHRS is submerged in the reactor pool and protected from internal generated missiles by the NPM operating bay walls. There are no credible sources of internally generated missiles in the area above the NPM as there is no rotating equipment in proximity to the NPM. Section 3.6.5 provides additional information on DHRS protection from pipe whip and internally generated missiles.

Potential LOCA loads do not affect the DHRS components because it is not connected to the RCS, and critical components are located outside the CNV. For discussion of stress analyses associated with the feedwater, DHRS, and steam piping inside the containment that connects the DHRS to the SGs, refer to Section 3.9.

5.4.3.2.1 Components

Actuation Valves

compared to the 100 degrees Fahrenheit initial condition. Based on these results, the DHRS design is capable of cooling the RCS to below a safe shutdown temperature of 420 degrees Fahrenheit in less than 36 hours with one DHRS train in operation assuming limiting off-normal conditions and a single active failure of the associated MSIV to close.

Figure 5.4-11 and Figure 5.4-13 show the hot and cold leg temperatures difference increase as the water level in the RPV drops to near the top of the riser. When the liquid level is near the top of the riser, the reduced flow area causes more losses and impedes RCS natural circulation that increases the temperature difference. Oscillations in natural circulation of the RCS could occur once the level drops to near the top of the riser due to vapor build up in the top of the core and lower riser. The vapor eventually is discharged into the upper riser and condenses as it rises. During these potential surges, liquid water is pushed over the top of the riser and into the downcomer. Results show that the potential oscillations do not affect the ability of the DHRS to remove heat. The DHRS has been shown to be capable of removing heat in excess of decay heat after 36 hours with the RCS at 420 degrees Fahrenheit and the pool at boiling conditions.

Refer to Chapter 15 for plant initial conditions, assumptions, and response to design basis events that result in DHRS actuation.

5.4.3.4 Tests and Inspections

RAI 05.02.01.01-7

Preservice and ~~I~~inservice inspection requirements of Section XI are met for Class 2 components of the BPVC are applicable to the DHRS components including the steam piping, actuation valves, condensers, and condensate piping.

RAI 09.03.06-2S1

The DHRS actuation valves are classified as Category B valves in accordance with OM Code Subparagraph ISTC-1300(b) because seat leakage in the closed position is inconsequential for fulfillment of the required function(s). Exercising the actuation valves while at power is not practicable. Therefore, the valves are full-stroke exercised during the equivalent of cold shutdown conditions as allowed by OM Code, Subparagraph ISTC-3521(c). As described in Section 3.9.6, NuScale Mode 3 safe shutdown with reactor coolant temperatures < 200 degrees Fahrenheit is considered to be the equivalent of cold shutdown as defined in the OM Code ISTA-2000. The DHRS actuation valves that are fully cycled as part of a plant shutdown satisfy the exercising requirements provided they meet the observation requirements for testing in accordance with ASME OM Code, Paragraph ISTC-3550. In addition, loss of valve actuator power and position verification testing is performed in accordance with OM Code, Paragraphs ISTC-3560 and ISTC-3700, respectively.

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

I&C for each valve within a PSCIV assembly are provided by independent divisions of the MPS.

RAI 05.02.01.01-7

The PSCIVs connected to lines that directly contact the reactor coolant (RCS injection, RCS discharge, PZR spray, and RPV high-point degasification) during normal operation are ~~designed and constructed~~designed, fabricated, constructed, tested and inspected 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~~CRDS, CES and CFDS 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~~designed, fabricated, constructed, tested and inspected 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.

RAI 05.02.01.01-7

The SSCIVs are ~~designed~~designed, fabricated, constructed, tested and inspected 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.

RAI 05.02.01.01-7

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

RAI 05.02.01.01-7

Attachment welding of the PSCIVs are conducted utilizing procedures qualified in accordance with the applicable requirements of ASME BPVC, Section III, Subarticle NB-4300 and Section IX. Welding of the SSCIVs and short section of MS piping are conducted utilizing procedures qualified in accordance with the applicable requirements of ASME BPVC, Section III, Subarticle NC-4300 and Section IX.

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).

RAI 06.02.04-2

Hydraulic actuators are used for both PSCIV and SSCIV designs. Each actuator is equipped with a hydraulic cylinder applying opening force to the valve and a pneumatic cylinder that applies closing force to the valve. To open the valve, hydraulic pressure is increased to exceed the force applied by the passive pneumatic cylinder. To close the valve, hydraulic fluid is vented, allowing the passive pneumatic cylinder pressure force to exceed the hydraulic pressure force. The pneumatic cylinders are precharged with nitrogen prior to plant operation. The hydraulic cylinders are pressurized by a non-safety related hydraulic skid and vented by redundant safety related pilot valves.

RAI 06.02.04-2, RAI 10.03-5

Two styles of actuators are used: standard capacity and high capacity. The standard capacity actuator consists of pilot solenoid valves arranged in two parallel vent paths (refer to Figure 6.2-7). The high capacity actuator consists of pilot solenoid valves and pilot dump valves arranged in two parallel vent paths (refer to Figure 6.2-7). For the high capacity actuator, the pilot solenoid valve actuates the pilot dump valve. Actuating the pilot dump valve vents the hydraulic line, thereby closing the valve. The pilot valves are remotely located on hydraulic skids. Figure 6.2-8 depicts the relationship between the CIVs, their pilot valve subcomponents and hydraulic skids. The two hydraulic skids are located in separate areas of the RXB.

RAI 06.02.04-2

The pilot solenoid valves are controlled by the module protection system (MPS) as described by Chapter 7. The CIVs fail to the safe (closed) position.

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.

RAI 05.02.01.01-7

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~~designed, fabricated, constructed, tested and inspected 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.

RAI 06.02.04-2

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

RAI 05.02.01.01-7

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~~designed, fabricated, constructed, tested, and inspected 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~~designed, fabricated, constructed, tested, and inspected 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~~designed, fabricated, constructed, tested, and inspected 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.

RAI 06.02.04-2

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. The valves are also designed to close under loss of power. 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

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.

RAI 05.02.01.01-7

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. Preservice and ISI requirements associated with the PSCIVs ASME BPV Class 1 components are summarized in Section 5.2. Preservice and ISI requirements associated with the SSCIVs and MS piping meet the applicable inspection requirements of Section XI of the ASME BPV. The SSCIVs and MS piping components are designed such that the ISI requirements of ASME BPVC, Section XI can be performed, including the preservice inspections of ASME Section III

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.