NuScale Power

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Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Submittal of Changes to Final Safety Analysis Report, Section 5.4.

REFERENCES:
1. Letter from NuScale Power LLC, to Nuclear Regulatory Commission, "NuScale Power, LLC Submittal of the NuScale Standard Plant Design Certification Application," dated December 31, 2016 (ML17013A229)

2. NRC Meeting Summary of June 28th, 2017, Public Meeting with NuScale Power, LLC, dated July 12, 2017 (ML17192A591)

During a June 28th, 2017 public teleconference with Bruce Bavol and other members of the NRC staff, NuScale Power, LLC (NuScale) discussed potential updates to Final Safety Analysis Report (FSAR), Section 5.4.1, "Steam Generators". As a result of this discussion, NuScale changed Tier 2, Section 5.4.1 and Tier 1, Section 2.1.1. The Enclosure to this letter provides a mark-up of the FSAR pages incorporating revisions to Sections 5.4.1, in redline/strikeout format. NuScale will include this change as part of a future revision to the NuScale Design Certification Application.

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

Please feel free to contact Marty Bryan at 541-452-7172 or at mbryan@nuscalepower.com if you have any questions.

Sincerely,

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Enclosure: Changes to NuScale Final Safety Analysis Report Tier 2, Sections 5.4.1 and Tier 1, Section 2.1.1

Enclosure:

Changes to NuScale Final Safety Analysis Report Tier 2, Sections 5.4.1 and Tier 1, Section 2.1.1

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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

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 ~~reactor coolant pressure boundary (~~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.

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

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

The SGs perform an integral part of the reactor residual and decay heat removal process when the DHRS is in operation. They transfer heat from the primary coolant to the naturally circulating closed loops that transfer decay heat to the reactor pool. 10 CFR 50.55a(g) requires the inservice inspection (ISI) program to meet the applicable inspection requirements of Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC). The steam generator system (SGS) components are designed such that the ISI requirements of ASME BPVC, Section XI can be performed, including the preservice inspections of ASME Section III. A SG program, based on NEI 97-06 and described in Section 5.5.4 of the technical specifications, is used for implementing ASME Code Section III and XI for the SG tubes. The primary and secondary sides of the SGs are designed to permit implementation of a SG program that provides reasonable assurance the structural and leakage integrity of the SG tubes is maintained. Integrity of SGs, integral steam plenum, and feedwater plena that make up portions of the RCPB is discussed in Section 5.2.

5.4.1.2 System Design

Each SG, located inside the RPV, is comprised of interlacing helical tube columns connecting to two feed and two steam plena. The feed and steam plena comprising a single SG are configured 180 degrees apart. As shown in Figure 5.4-1 and Figure 5.4-2, the configuration of the helical tube columns of the two SGs form an intertwined bundle of tubes around the upper riser assembly with a total of four feed and four steam plena located 90 degrees apart around the RPV. Figure 5.4-2 shows the cross-sectional arrangement of the integral steam plena and feed plena, while Figure 5.4-3 and Figure 5.4-4 show individual cross-sectional views of an individual steam and feed plenum. The primary reactor coolant circulates outside the SG tubes with steam formation occurring inside the SG tubes.

Each SG tube is comprised of a helix with bends at each end that transition from the helix to a straight configuration at the entry to the tubes sheets as shown in Figure 5.4-1. The helical tubes are seamless with no intermediate welds. The helical tubes terminate at the feed and steam plenum tubesheets, where the tubes are secured to the tubesheet by expansion fit and are welded to the tubesheet on the secondary side. Crevices between the SG tubes and the tube supports and tubesheets are minimized to limit the buildup of corrosion products. Minimal quantities of corrosion products are present because the SG tube-to-tubesheet contact is within the primary coolant environment. Crevices at the tube-to-tubesheet face are prevented by full-length expansion of the tube within the tubesheet bore. The tubes are expanded into both the steam and feed plenum tubesheets.

The SG has no secondary side crevices or low-flow regions that could concentrate corrosion products or impurities accumulated during the steam generation process. The once-through SG design does not contain a bulk reservoir of water at the inlet plena where the accumulation or concentration of material could occur. The concept of SG blowdown to remove these deposits is not applicable to the once-through NuScale Power Module SG design based on the geometry of the design and flow characteristics that do not allow accumulation of corrosion products within a fluid reservoir. Therefore, a blowdown system that could be implemented would only serve to divert feedwater flow from the SG and would not be capable of removing corrosion products or impurities. Based on these factors, no SG blowdown system is included in the NPM design.

Secondary coolant impurities and corrosion products may deposit directly on the interior tube surfaces as a scale or film, or be removed from the SG by carryover. The

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 5.4-8) 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 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.

5.4.1.4 Tests and Inspections

The SGs are tested and inspected to ensure conformance with the design requirements as described in Section 5.2.4. The configuration of equipment requiring inspection or 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 tube test and inspection requirements are provided in NEI 97-06 and include the examination requirements for SG tubing specified in the BPVC, Section XI, Table IWB-2500-1 (B-Q).~~ 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). ~~The SG program follows the guidance of NEI 97-06 and is described in the plant technical specifications.~~

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.

~~Preservice eddy current examinations for the SG tubing are in accordance with the applicable requirements of the Electric Power Research Institute (EPRI) Steam Generator Management Program guidelines (Reference 5.4-2) and BPVC Section XI.~~A volumetric, full-length preservice inspection of 100% 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. Any tubes with flaws that exceed 40% of the nominal tube wall thickness shall be plugged. Any tubes with flaws that could potentially compromise tube integrity prior to the performance of the first inservice inspection and any tubes with indications that could affect future inspectability of the tube shall also be plugged. The volumetric technique used for the PSI 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, ~~T~~the operational inservice testing and inspection programs described in Section 5.2.4 and the SG program described in ~~Section 5.4.2.2~~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

Refer to Section 5.2.3 for additional description of material compatibility, fabrication and process controls, and welding controls related to the ASME Class 1 components. Refer to Section 6.6 for additional description of material compatibility, fabrication and process controls, and welding controls related to the ASME Class 2 components.

Threaded fasteners are described in Section 3.13.

5.4.1.6 Steam Generator Program

The SG program monitors the performance and condition of the SGs to ensure they are capable of performing their intended functions. The program provides monitoring and management of tube degradation and degradation precursors that permit preventative and corrective actions to be taken in a timely manner, if needed. The SG program is described in the plant technical specifications and is a part of the overall ISI program. The program implements applicable portions of Section XI of the BPVC and specifically addresses 10 CFR 50.55a(b)(2)(iii). Appendix B to 10 CFR 50 applies to implementation of the SG program.

The NuScale SG Program follows NEI 97-06 and EPRI guidance. Application of established commercial SG Program requirements to the NuScale design are appropriate based on the historical causes of SG tube degradation and the features of the NuScale SG design. The NuScale design incorporates design improvements necessary to restrict SG tube degradation and has additional design features that reduce the risk of SG tube degradation compared to existing PWR designs.

Historically, significant SG tube degradation has occurred in the operating PWR SG fleet due to various corrosion mechanisms, including wastage and both primary and secondary side stress corrosion cracking. These corrosion mechanisms were related to materials selection, plant chemistry control, and control of the ingress of impurities and corrosion products to the SGs. In the operating fleet, detrimental SG corrosion has been effectively mitigated based on use of A690TT SG tubing, application of EPRI primary and secondary plant chemistry control, and design of condensate systems (including extensive use of polishing resin beds and improved materials). These improvements have been implemented in the NuScale design~~; therefore, these factors do not provide a basis for NuScale to deviate from the established NEI 97-06 SG Program requirements that have led to high levels of SG reliability and integrity in the operating commercial fleet~~.

In addition to chemistry and materials considerations, where the NuScale design is equivalent to the existing PWR fleet, there are two areas where the NuScale design has reduced SG tube degradation risk. ~~The NuScale SG tube wall thickness is thicker than existing designs based on incorporation of a substantial degradation allowance (additional tube wall thickness above minimum required for design) as shown in Table 5.4-3 and discussed in Section 5.4.1.2.~~The NuScale SG tube wall thickness is thicker than existing designs (see Table 5.4-3) based on incorporation of a substantial degradation allowance (additional tube wall thickness above minimum required for design) as discussed in Section 5.4.1.2. The NPM reactor coolant flowrates are lower than the flowrates across the SG tubes in PWR recirculating steam generators as discussed in Section 5.1. This low flow rate reduces the flow energy available to cause flow induced vibration (FIV) wear degradation of SG tubes. Based on the additional tube wall margin and the additional margin against FIV turbulent buffeting wear (the most likely SG tube degradation mechanism), application of the existing PWR SG Program requirements to the NuScale design is appropriate.

For SGs in the operating PWR fleet with A690TT SG tubing, the only observed degradation has been wear as a result of flow induced vibration (tube-to-tube or tube-to-support plate) or wear due to foreign objects. With respect to the risk of introduction of foreign objects, the NPM is at no greater risk than existing designs, therefore no deviations from existing SG program guidelines are warranted. From the standpoint of SG tube design, the two significant differences between the NuScale SG design and existing designs is the helical shape of the SG tubing and the SG tube supports structures. The helical shape of the SG tubing itself does not represent risk of degradation based on the minimum bend radius of the helical tubing being within the experience base of operating PWR SG designs. The SG tube support design is novel. However, as discussed in Section 5.4.1.2, it preserves attributes of the existing tube support (plate) designs. Prototypic testing of the SG tube supports is performed to validate acceptable performance (including wear) of the SG tube support design. Implementation of a typical SG program is appropriate based on evaluation of the design of the SG tube supports.

5.4.1.6.1 Degradation Assessment

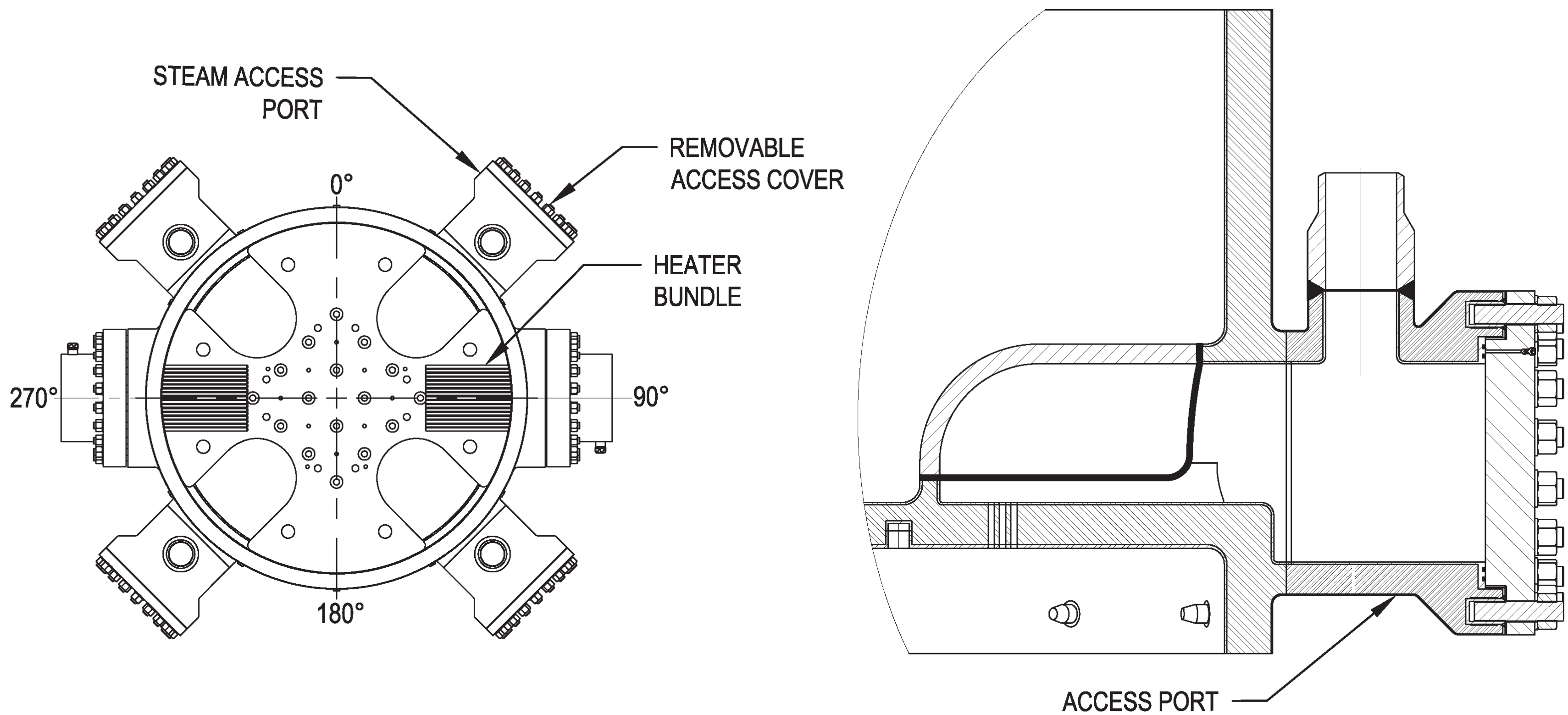
A SG degradation assessment of the NPM SG identified several potential degradation mechanisms. As observed in the operating PWR fleet, wear is the most likely degradation mechanism. The preliminary SG degradation assessment also identified the potential for several secondary side corrosion mechanisms, including under deposit pitting and intergranular attack based on the once through design with secondary boiling occurring inside the tubes. The estimated growth rates for these potential defects is sufficiently low that the SG tube plugging criteria for the NPM SG is a 40% through wall defect, consistent with the existing PWR SG fleet. Based on the ability to implement tube plugging criteria consistent with the operating PWR SG fleet, consistent implementation of other elements of the SG Program, including SG inspection frequency, is appropriate.

COL Item 5.4-1: A COL applicant that references the NuScale Power Plant design certification will develop and implement a Steam Generator Program for periodic monitoring of the degradation of steam generator components to ensure that steam generator tube integrity is maintained. The Steam Generator Program will be based on NEI 97-06, "Steam Generator Program Guidelines," Revision 3 and applicable EPRI steam generator guidelines. The elements of the program will include: assessment of degradation, tube inspection requirements, tube integrity assessment, tube plugging, primary-to-secondary leakage monitoring, primary and secondary side water chemistry control, foreign material exclusion, loose parts management, contractor oversight, self-assessment, and reporting.

Table 5.2-4: Reactor Coolant Pressure Boundary Component Materials Including Reactor Vessel, Attachments, and Appurtenances

Component	Specification	Alloy Designation (Grade, Class, or Type)
Reactor Vessel		
Lower RPV section flange shell RPV bottom head Core support blocks	SA-508	Grade 3, Class 1
RPV top head PZR Shell Integral steam plenum Upper RPV flanged transition shell Integral steam plenum Steam plenum access ports Upper RPV SG shell Lower RPV SG shell Feed plenum access ports ~~Upper and lower RPV steam generator shells~~	SA-508	Grade 3, Class 2
RPV support gussets RPV support plates	SA-533	Type B, Class 2
Core barrel guides	SA-193	Type 304/304L; Grade B8, Class 1
Vessel alignment pins RPV flange stud threaded inserts Pressure instrument tap swagelok reducers	SA-479	Type 304/304L
Instrumentation and Controls (I&C) access port covers	SA-240	Type 304/304L
I&C access port cover threaded fasteners	SB-637	Alloy 718 (UNS N07718)
RPV flange leak detection tube	SA-312	Type 316L; Seamless
RPV flange closure stud bolts, nuts, and washers RSV flange threaded fasteners, nuts, and washers	SB-637	Alloy 718 (UNS N07718)
I&C swagelok male connectors	SA-479	Type 316/316L
PZR pressure taps Thermowell nozzles	SB-166	Alloy 690 (UNS N06690)
Safe ends for: • RRV • CVCS charging and letdown nozzles • CRDM nozzles • RVV • High point degasification nozzle • Spray nozzle	SB-166 or SB-167	Alloy 690 (UNS N06690)
PZR heater closure flange	SB-168	Alloy 690 (UNS N06690)
Ultrasonic testing sensor nozzles	SA-182	Grade F304/F304L
Low alloy steel weld filler material	SFA 5.5 SFA 5.23 SFA-5.28 SFA-5.29	Weld filler metal classifications compatible with low alloy steel base metal

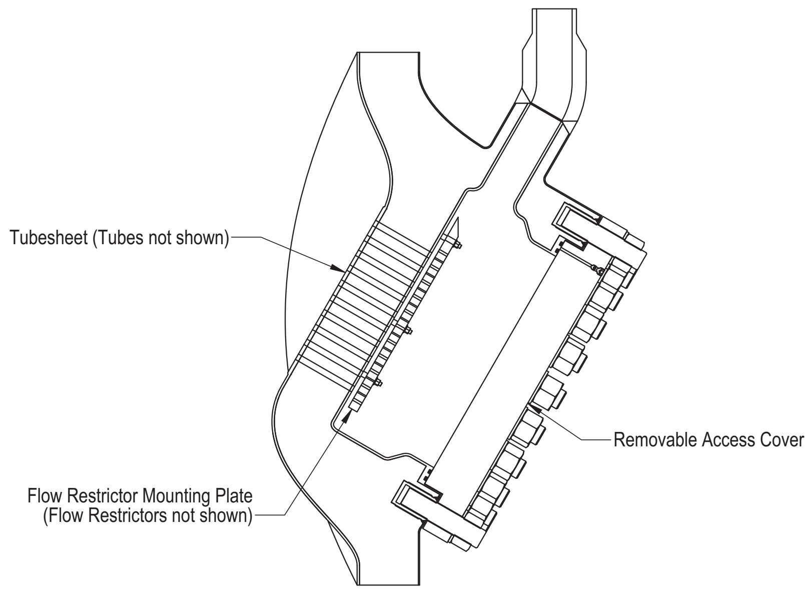
Figure 5.4-4: Integral Steam Plenum



INTEGRAL STEAM PLENUM ASSEMBLY TOP VIEW
(OVERHEAD VIEW)

INTEGRAL STEAM PLENUM
(SIDE VIEW - CUTAWAY)

Figure 5.4-5: Feedwater Plenum Access Port



FEED PLENUM ACCESS PORT CUTAWAY

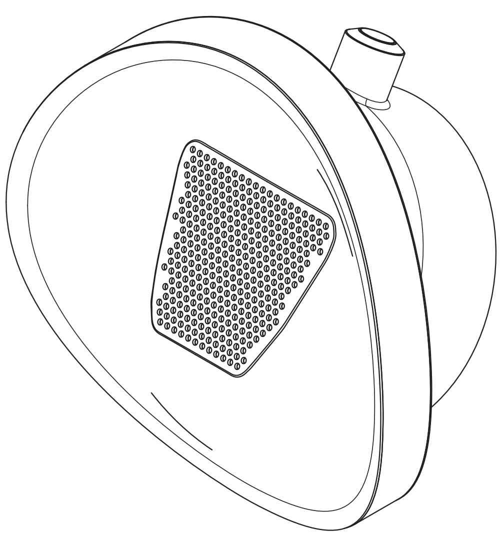
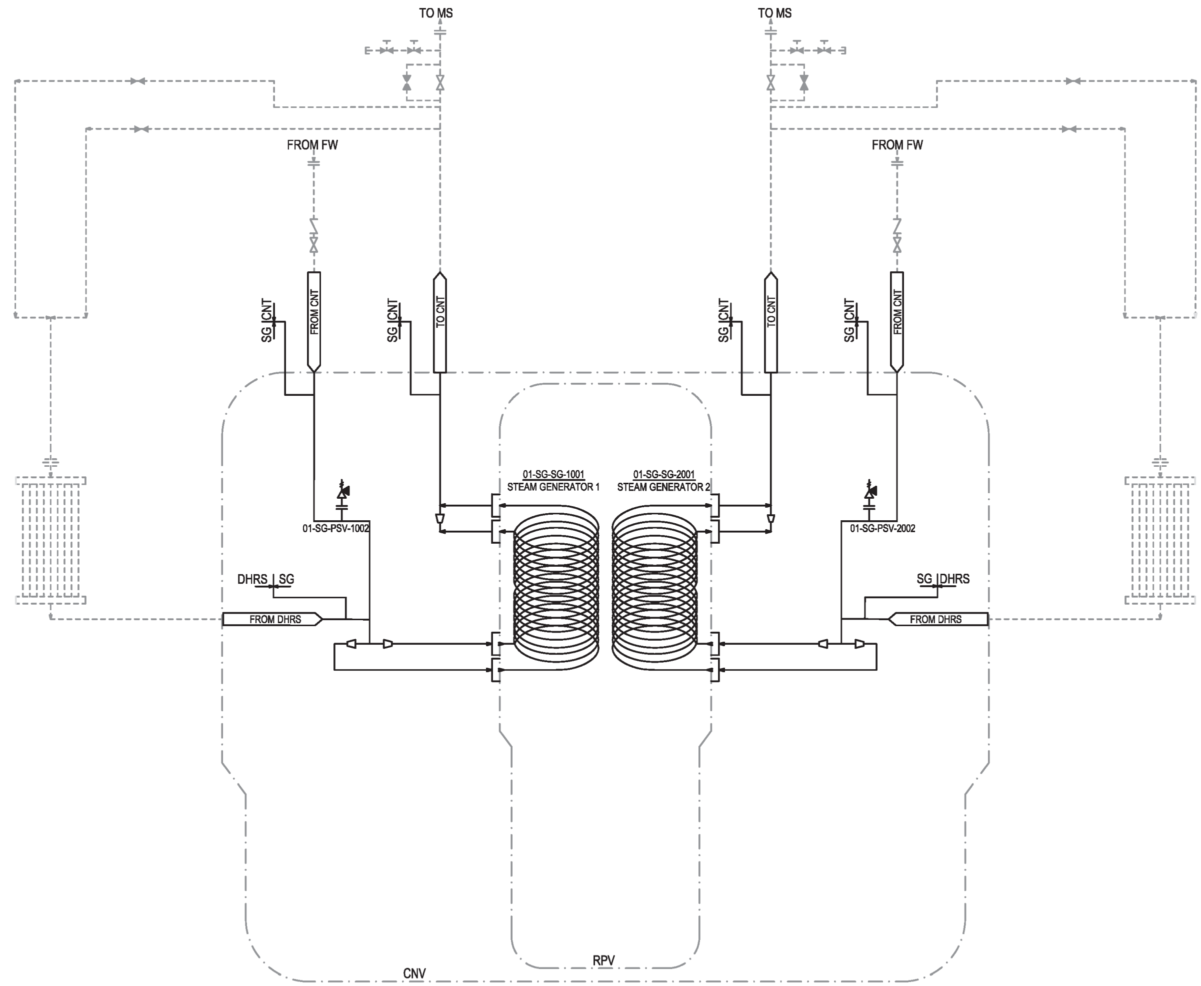


Figure 5.4-9: Steam Generator Simplified Diagram



2.1 NuScale Power Module

2.1.1 Design Description

<u>System Description</u>

The scope of this section is the NuScale Power Module (NPM) and its associated systems. The NPM is installed in the reactor pool in the Reactor Building (RXB). Up to 12 NPMs may be installed in the Reactor Building. The systems contained within the boundary of the NPM are the

- reactor coolant system (RCS), including the reactor pressure vessel (RPV), pressurizer, steam generator (SG), reactor vessel internals (RVI), and associated piping and valves.
- control rod drive system (CRDS), including the control rod drive mechanisms (CRDM) with embedded cooling water tubes, cables, and associated cooling water piping. The CRDS also includes instrumentation to provide control rod position indication information.
- containment system (CNTS), including the containment vessel (CNV) and containment isolation valves (CIVs) and associated piping.
- emergency core cooling system (ECCS) valves.
- decay heat removal system (DHRS), including associated piping and valves.

The NPM includes the pressure retaining structures of these systems because they are part of either the reactor coolant pressure boundary (RCPB) or the CNV pressure boundary. Therefore, the mechanical design and arrangement of the piping, CRDS, and NPM valves (emergency core cooling, reactor safety, and containment isolation) are included in this section.

The CRDM pressure housings form the pressure boundary between the environments inside the RPV and the CNV. The CRDM pressure housings consist of the latch housing~~ and the~~<u>,</u> rod travel housing<u>, and rod travel housing plug</u>.

The NPM performs the following ~~non~~safety-related~~, risk-significant~~ functions that ~~is~~<u>are</u> verified by Inspections, Tests, Analyses, and Acceptance Criteria:

- The RCS supports the CNTS by supplying the RCPB and a fission product boundary via the RPV and other appurtenances.
- The CRDS supports the RCS by maintaining the pressure boundary of the RPV.
- The SG supports the RCS by suppl<u>y</u>ing part of the RCPB.
- The ECCS supports the RCS by providing a portion of the RCPB for maintaining the RCPB integrity.
- The CNTS supports the RXB by providing a barrier to contain mass, energy, and fission product release from a degradation of the RCPB.
- The ECCS supports the CNTS by providing a portion of the containment boundary for maintaining containment integrity.