RAIO-0218-58628

February 12, 2018

Docket No. 52-048

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 296 (eRAI No. 9231) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 296 (eRAI No. 9231)," dated December 13, 2017

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 Questions from NRC eRAI No. 9231:

- 05.04.02.01-2
- 05.04.02.01-3
- 05.04.02.01-4
- 05.04.02.01-5
- 05.04.02.01-6
- 05.04.02.01-7
- 05.04.02.01-8
- 05.04.02.01-9
- 05.04.02.01-10
- 05.04.02.01-11
- 05.04.02.01-12
- 05.04.02.01-13
- 05.04.02.01-14

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,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Gregory Cranston, NRC, OWFN-8G9A
Samuel Lee, NRC, OWFN-8G9A
Bruce Bavol, NRC, OWFN-8G9A

Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9231

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9231

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

eRAI No.: 9231
Date of RAI Issue: 12/13/2017

NRC Question No.: 05.04.02.01-2

Section 52.6(a) of Title 10 of the *Code of Federal Regulations* (10 CFR) requires that information provided to the U.S. Nuclear Regulatory Commission (NRC) by an applicant for a standard design certification be complete and accurate in all material respects. Therefore, please revise Tier 2 of the Final Safety Analysis Report (FSAR) to address the following changes made in your August 3, 2017, supplements (Accession Nos. ML17215A977 and ML17215A978 in the NRC's Agencywide Documents Access and Management System (ADAMS)).

a. It appears that "integral steam plenum" was inadvertently added twice to Table 5.2-4, "Reactor Coolant Pressure Boundary Component Materials Including Reactor Vessel, Attachments, and Appurtenances." The NRC staff believes that the intent was to also add "integral feed plenum."
b. The figure referenced in the subsection entitled, "Thermal Relief Valves," of Section 5.4.1.2, "System Design," for illustrating the thermal relief valve on each feedwater line appears to be incorrect. The referenced figure, 5.4-8, is the steam generator (SG) flow restrictor assembly.
c. The table referenced in Section 5.4.1.6, "Steam Generator Program," for SG tube wall thickness in the fourth paragraph appears to be incorrect. The referenced table, 5.4-3, "Steam Generator Piping, Tube and Piping Supports, and Flow Restrictor Materials," identifies materials specifications.

NuScale Response:

a. "Integral steam plenum" was added to FSAR Table 5.2-4 twice in error and this error was corrected in a NuScale letter dated December 12, 2017. Please note that "Feed plenum access ports" and "Lower RPV SG shell" were added to FSAR Table 5.2-4 as shown in the NuScale letter dated December 12, 2017 to capture the feed plenum access ports and the associated RPV shell (see FSAR Figures 5.3-4 and 5.4-3 for the feed plenum access port and lower RPV SG shell configuration).
b. The correct figure reference for the thermal relief valve location (FSAR Figure 5.4-9) has been added to the "Thermal Relief Valves" subsection in FSAR Section 5.4.1.2 . This

addition is shown in the FSAR changes provided in response to RAI 9231 - 05.04.02.01-6.

c. The table referenced in FSAR Section 5.4.1.6 for SG tube wall thickness has been corrected. FSAR Section 5.4.1.6 now references FSAR Table 5.4-2 for SG tube wall thickness. This addition is shown in the FSAR changes provided in response to RAI 9231 - 05.04.02.01-6.

Impact on DCA:

Changes to FSAR Section 5.4.1.6 have been revised as described in the response above and as shown in the markup provided with the response to question 05.04.02.01-6.

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eRAI No.: 9231
Date of RAI Issue: 12/13/2017

NRC Question No.: 05.04.02.01-3

Statements that the SG program is “based” on Nuclear Energy Institute (NEI) 97-06, “Steam Generator Program Guidelines,” were added to Sections 5.4.1.1, “Design Basis,” and 5.4.1.4, “Tests and Inspections,” in your August 3, 2017, supplements. For consistency with other statements in the FSAR, this statement should be revised to make it clear that the SG program follows or conforms to NEI 97-06 because the term “based on” implies that there may be exceptions. If there are exceptions, then they should be identified and justified. This request is made to ensure that your SG program meets the requirements of General Design Criteria (GDC) 32, 10 CFR 50.55a, 10 CFR 50.36, Appendix B in 10 CFR Part 50, and 10 CFR 50.65 as they relate to implementation of a SG program to maintain the structural and leakage integrity of the SG tubes.

NuScale Response:

Use of the term "based on" in FSAR Sections 5.4.1.1 and 5.4.1.4 is consistent with the wording used in FSAR Section 5.2.3.2.1, "Reactor Coolant Chemistry." The wording "based on" was used in the same way that the NRC used it in Section 5.2.3.4 of NUREG-2124, Volume 1, "Final Safety Evaluation Report Related to the Combined Licenses for Vogtle Electric Generating Plant, Units 3 and 4."

NEI 97-06, “Steam Generator Program Guidelines,” Rev. 3, Section 1.5 endorses use of a set of six EPRI Steam Generator Guidelines as an industry consensus standard for implementing Steam Generator Programs. NEI 97-06, Rev. 3, Section 1.5 also provides methods that a licensee may follow to deviate from the the Steam Generator Program through following the requirements of NEI 03-08. Finally, NEI 97-06, Rev. 3, Section 1.5 recognizes that the guidelines contain "mandatory" or "needed" elements. The EPRI Steam Generator Guidelines also have determined that some elements are considered "recommended" - related to good practices - and are subject to change (see EPRI, "PWR Secondary Water Chemistry Guidelines," section 8).

See the response to NuScale RAI 9117 - 10.04.06-3 for a similar discussion related to primary

water chemistry and implementation of the EPRI Guidelines.

Therefore, based on the considerations in NEI 97-06, the likelihood of changes in the industry steam generator programs, and the use of the term by the NRC in another DCA safety evaluation, use of "based on" is an acceptable term to demonstrate the requirements of GDC 32, 10 CFR 50.55a, 10 CFR 50.36, Appendix B in 10 CFR Part 50, and 10 CFR 50.65 are met as they relate to implementation of a SG program to maintain the structural and leakage integrity of the SG tubes.

Based on the discussion above, no changes are required to the NuScale FSAR.

Impact on DCA:

There are no impacts to the DCA as a result of this response.

NuScale Power

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eRAI No.: 9231
Date of RAI Issue: 12/13/2017

NRC Question No.: 05.04.02.01-4

The NRC staff notes that Table 5.4-2 in Tier 2 of the FSAR, "Steam Generator Design Data," includes the minimum steam and feed tubesheet thickness (i.e., the thickness of the ferritic base material of the tubesheets). Revise this table to include the steam and feed tubesheet cladding thickness and the total thickness (tubesheet thickness plus cladding thickness). The NRC staff requests this information be added to the table because this material will form part of the reactor coolant pressure boundary (RCPB) since it is joined to the SG tube ends. Therefore, the selection, processing, fabrication, and testing of this cladding must meet the requirements of Appendix A in 10 CFR Part 50, GDC 1, GDC 30, and 10 CFR 50.55a. Adding this information will also meet the requirement in 10 CFR 52.6(a) to submit complete and accurate information in a standard design certification.

NuScale Response:

The cladding thickness on the primary and secondary sides of the steam and feed tubesheets as well as the total thickness of the tubesheets (i.e., tubesheet thickness plus cladding thickness) has been added to FSAR Table 5.4-2.

The NuScale reactor pressure vessel - including the steam generator, both steam and feed plenum tubesheets, and associated cladding - is designed, analyzed, and fabricated in accordance with ASME Boiler and Pressure Vessel Code Section III, Subsection NB. In accordance with paragraph NB-3122.1, no structural strength shall be attributed to the cladding for the purposes of satisfying primary stresses. Therefore, the cladding is not considered to directly contribute to the structural integrity of the reactor coolant pressure boundary (RCPB), but rather to indirectly support structural integrity of the RCPB by preventing corrosion of the base metal.

Impact on DCA:

Table 5.4-2 has been revised as described in the response above and as shown in the markup provided in this response.

RAI 05.04.02.01-4

Table 5.4-2: Steam Generator Design Data

Parameter	Value
Type	Helical, once-through
Total number of helical tubes per NPM	1380
Number of helical tube columns per NPM	21
Internal pressure - secondary (psia)	2100
External pressure - primary (psia)	2100
External pressure - SG piping in containment (psia)	1000
Internal temperature - secondary (°F)	650
External temperature - primary (°F)	650
External temperature - SG piping in containment (°F)	550
Tube wall outer diameter (inches)	0.625
Tube wall thickness (inches)	0.050
~~Minimum s~~Steam tubesheet thickness, without clad (inches)	4.0
~~Minimum f~~Feed tubesheet thickness, without clad (inches)	6.0
Steam and feed tubesheet clad thickness - secondary (inches)	0.250
Steam and feed tubesheet clad thickness - primary (inches)	0.375
Steam tubesheet thickness, with clad (inches)	4.625
Feed tubesheet thickness, with clad (inches)	6.625
Total heat transfer area (ft^2)	17928
Fouling factor (hr-ft^2-°F/BTU)	0.0001
Minimum SG tube transition bend radius (inches)	≥ 6.250

NuScale Power

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eRAI No.: 9231
Date of RAI Issue: 12/13/2017

NRC Question No.: 05.04.02.01-5

Section 5.1.3.7 in Tier 2 of the FSAR, “Steam Generators,” states that the tubes, tube supports, steam and feedwater piping inside containment, tube inlet flow restrictors, and steam and feed plenums makeup the NuScale SG system. Table 3.2-1 in Tier 2 of the FSAR, “Classification of Structures, Systems, and Components,” notes that the SG system also includes the main steam and feedwater supply nozzles. Section 3.12.5.8.3 of Tier 2 of the FSAR, “Feedwater Line Stratification,” states that the SG feedwater nozzles are located on the feedwater inlet plenums. Section 52.6(a) of 10 CFR requires that information provided to the NRC by an applicant for a standard design certification be complete and accurate in all material respects. Therefore, please revise Chapter 5 in Tier 2 of the FSAR to include:

a. A statement that the SG system also includes the main steam and feedwater supply nozzles.
b. Identification of the main steam and feedwater supply nozzles on Figures 5.4-4, “Integral Steam Plenum,” and 5.4-5, “Feedwater Plenum.”
c. A statement of the location of the main steam and feedwater supply nozzles (e.g., the SG main steam supply nozzles are located on the steam plenums and the SG feedwater supply nozzles are located on the feedwater inlet plenums as shown in Figures 5.4-4 and 5.4-5).
d. A description of how the nozzles are attached to the steam and feed plenums. The main steam and feedwater supply nozzle materials in Table 5.4-3, including the nozzle weld materials.

NuScale Response:

a. A statement that the steam generator system (SGS) also includes the main steam and feedwater supply nozzles has been added to FSAR Section 5.4.1.2. All SGS components and the associated classification designations are also provided in FSAR Table 3.2-1.
b. The main steam and feedwater nozzles have been identified on FSAR Figures 5.4-4 and 5.4-5, respectively.
c. A statement of the location of the main steam and feedwater supply nozzles has been added to FSAR Section 5.4.1.2.

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d. The main steam and feedwater nozzles are integral to the steam plenum access ports and feed plenum access ports, respectively (i.e., there are no welds between the nozzles and the associated access ports). This information has been added to FSAR Section 5.4.1.2. Steam and feed plenum access port materials are listed in FSAR Table 5.4-3.

Impact on DCA:

FSAR Section 5.4.1.2, Figure 5.4-4 and Figure 5.4-5 have been revised as described in the response above and as shown in the markup provided in this response.

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

RAI 05.04.02.01-5

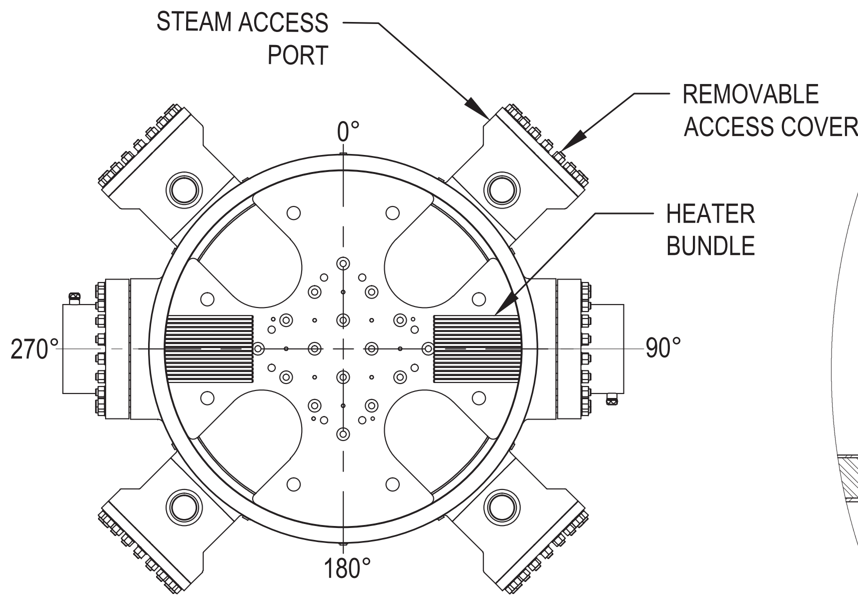
Each SG, located inside the RPV, is comprised of interlacing helical tube columns connecting to two feed and two steam plena. The feed and steam plena comprising a single SG are configured 180 degrees apart. As shown in Figure 5.4-1 and Figure 5.4-2, the configuration of the helical tube columns of the 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 5.4-3 shows the cross-sectional arrangement of the integral steam plena and feed plena, while Figure 5.4-4 and Figure 5.4-5 show individual cross-sectional views of an individual steam and feed plenum. The main steam supply nozzles and the feedwater supply nozzles are also part of the SGS. Each SG has a pair of feedwater and main steam supply nozzles. The main steam supply nozzles are integral to the steam plenum access ports and the feedwater supply nozzles are integral to the feed plenum access ports as shown in Figure 5.4-4 and Figure 5.4-5, respectively. 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.

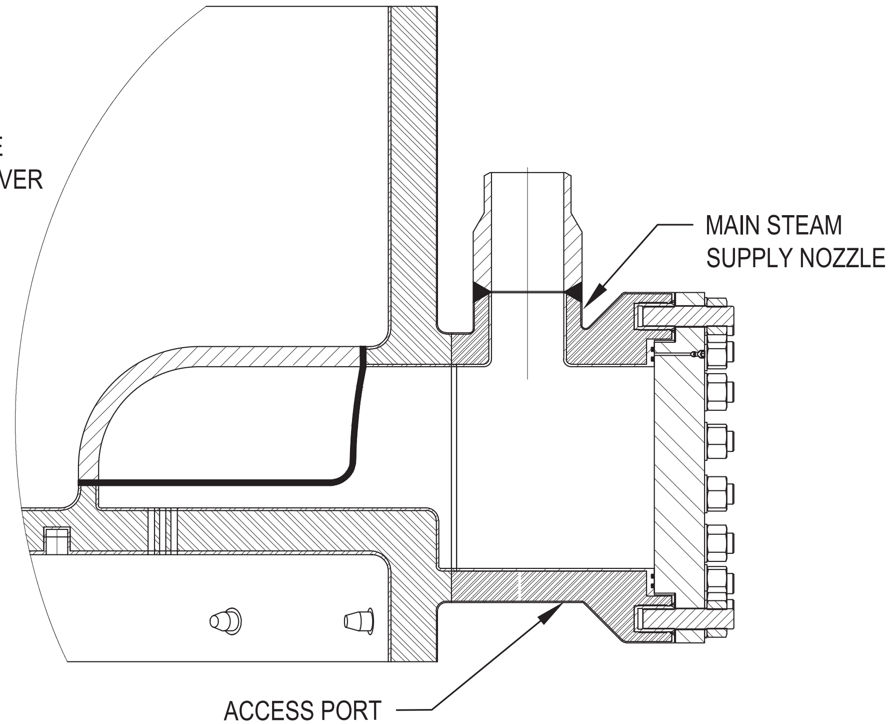
The SG has no secondary side crevices or low-flow regions that could concentrate corrosion products or impurities accumulated during the steam generation process. The once-through SG design does not contain a bulk reservoir of water at the inlet plena where the accumulation or concentration of material 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

RAI 05.04.02.01-5

Figure 5.4-4: Integral Steam Plenum



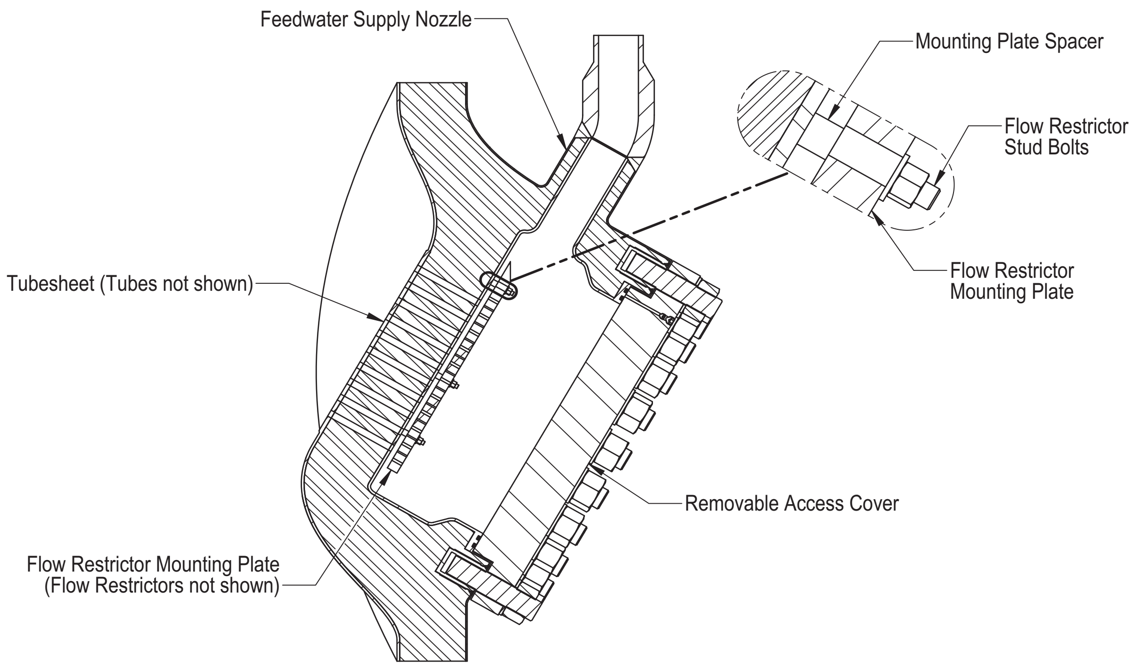
INTEGRAL STEAM PLENUM ASSEMBLY TOP VIEW
(OVERHEAD VIEW)



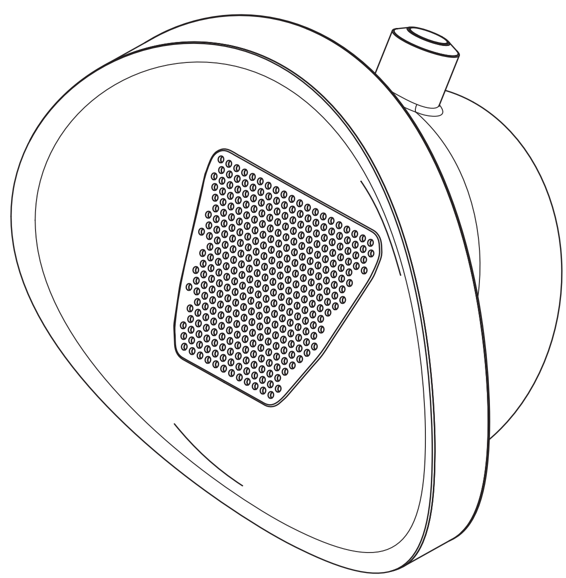
INTEGRAL STEAM PLENUM
(SIDE VIEW - CUTAWAY)

RAI 05.04.02.01-5

Figure 5.4-5: Feedwater Plenum Access Port



FEED PLENUM ACCESS PORT CUTAWAY



Response to Request for Additional Information
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eRAI No.: 9231
Date of RAI Issue: 12/13/2017

NRC Question No.: 05.04.02.01-6

Steam generator tube supports must be designed to meet the requirements of GDC 4, 14, 15, and 31 in Appendix A of 10 CFR Part 50, as they relate to maintaining the integrity of the RCPB. Given that this is a first-of-a-kind design, please provide the following information to address how the support structures meet these requirements:

a. A description of the testing and analyses performed to evaluate the design of the tube support structures with respect to providing adequate support, limiting accumulation of corrosive or particulate materials, and resisting degradation. Please also include a discussion of the likelihood and extent of loose parts generated from the tube support structures.
b. Clarification of the design requirements for the tube support structures and how they are addressed in the Tier 1 and Tier 2 material in the FSAR. Section 5.4.1.5 in Tier 2 of the FSAR, “Steam Generator Materials,” states that the tube supports conform to the requirements of Subsection NG of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. This appears to contradict Table 3.2-1 in Tier 2 of the FSAR, “Classification of Structures, Systems, and Components,” which identifies the supports as Quality Group A, corresponding to ASME Code Class 1. In addition, it is not clear to the NRC staff whether or not Inspection, Test, Analysis, and Acceptance Criteria No. 2 in Table 2.1-4 in Tier 1 of the FSAR, “NuScale Power Module Inspections, Tests, Analyses, and Acceptance Criteria,” regarding ASME Code Class 1 and 2 components applies to the SG tube support structures (Class NG).
c. Table 5.2-7, “Reactor Vessel Internals Inspection Elements,” and Figure 5.4-7, “Steam Generator Tube Support Tabs,” in Tier 2 of the FSAR refer to plates. Given that the FSAR states that tube support plates used in the currently operating fleet are not used in the NuScale design, please confirm that the “plates” referred to in this table and figure are the vertical support bars and revise this table and figure to that effect.
d. Table 5.2-7 refers to five tube support plate-to-tube support plate welds. Given that the FSAR states that tube support plates used in the currently operating fleet are not used in the NuScale design, please confirm that there are no tube support plate-to- tube support plate welds in the NuScale design and revise this table to that effect.

NuScale Response:

a. The Steam Generator (SG) tube supports are designed and fabricated in accordance with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section III, Subsection NG as a guide. Specifically, NuScale is applying the requirements in Article NG-5000, as described below, and Subarticles NG-2400, NG-3200, NG-4300, and NG-4400 to the SG tube support design and fabrication. During NuScale Power Module fabrication, NG-5000 Surface Examinations (PT examinations) will be performed for major joint welds in the SG tube supports. Visual inspections during fabrication will be performed for other welds (e.g. the spacers welded into the SG tube supports). These requirements provide adequate basis that the structural design and fabrication are sufficient to mitigate the generation of loose parts. Also, the SG tube supports are analyzed to demonstrate that the design satisfies the requirements of Subarticle NG-3200 for design, test, and service level conditions.

   In addition to ASME Code analysis requirements, the SG tube supports are also subject to analysis and testing in accordance to Regulatory Guide 1.20. Flow and modal testing of a helical SG tube bundle, using prototypic SG tube supports, to evaluate flow induced vibration (FIV) is performed as part of the comprehensive vibration assessment program (CVAP) described in TR-0716-50439-P. The following tests, included in the CVAP, utilize a full scale five column model of the helical steam generator and its supports:

   - In-air natural frequencies and mode shapes of the steam generator tubes.
   - In-water natural frequencies, mode shapes, vibration amplitudes and damping of the steam generator tubes.
   - Characterization of FIV amplitudes under primary coolant flow rates at and above prototypic maximum design flow conditions.

   The data from the CVAP testing will be used to validate the design and analysis of the steam generator and its supports and will demonstrate expected functionality of the SG tube supports.

   The tube supports are constructed of corrosion resistant materials (i.e. stainless steel) and therefore corrosion in a primary chemistry environment is expected to be minimal. There is no plausible source of "corrosive materials" in the reactor coolant system (RCS) based on the materials of construction (FSAR Table 5.2-4), cleanliness requirements (FSAR section 5.2.3.4.2) and chemistry controls (FSAR section 5.2.3.2.1). Therefore no testing or evaluation of accumulation of corrosive materials is performed. For the same reasons, the quantity of particulate material in the RCS is expected to be extremely low, limiting the potential for particulate material deposition. The tube supports do not provide continuous circumferential contact with the SG tubes (see FSAR Figure 5.4-6). This configuration limits the potential capture points (small clearance areas between the SG tube and the support interface) for any deposition to occur. Based on the limited

potential particulate capture points in the NuScale SG tube support design, and SGs located in a primary coolant environment (more restrictive chemistry controls, less expected corrosion) as compared to a secondary chemistry environment for typical commercial SG designs, the potential for corrosion or accumulation of particles in the NuScale design is reduced.

b. The SG tube supports are classified as “Internal Structures” and are constructed in accordance with ASME BPVC, Section III, Subsection NG as a guide. The Seismic Category of the SG tube supports is Seismic Category I. Inspection, Test, Analysis, and Acceptance Criteria No. 2 in FSAR Table 2.1-4 is not applicable to the SG tube supports.

   The information above is consistent with FSAR Sections 5.4.1.2 and 5.4.1.5 and FSAR Table 3.2-1. As revised in the response to RAI 8948, Question 03.02.02-3; FSAR Table 3.2-1 no longer classifies the SG tube supports as Quality Group A. The SG tube supports do not form part of the RCPB, therefore, previous classification of the SG tube supports as Quality Group A components was incorrect. FSAR Section 3.9.3.1.2 has also been revised in response to RAI 8901 Question 03.09.05-16 to clarify the SG tube support classification.

c. The steam generator tube support design has been modified, including changes in terminology, to align with the CVAP. As part of the design change, naming of some tube support components has changed, including use of the term “tube support plates,” which has been removed. There are no structures in the NuScale design that are comparable to "tube support plates," (i.e. solid metal plates with drilled holes that the tubes pass through) as they exist in currently operating commercial SGs. The design changes implemented are limited to features related to manufacturing details (e.g. specific placement of welds, method of forming specific geometric features, etc.) of the SG tube supports. The interface between the tubes and the supports, the general structural design and the overall function of the supports is unchanged.

   FSAR Sections 3.2.2, 3.9.3.1.2, and 5.4.1, FSAR Figures 5.4-6 and 5.4-7, and FSAR Tables 3.2-1, 5.2-4, 5.2-6, 5.2-7, and 5.4-3 have been revised to reflect the updated design of the steam generator tube supports including revised language used to describe the steam generator tube supports and the upper and lower steam generator supports.

d. As discussed above, the term "tube support plates" has been changed, thus there are no longer any welds described as "tube support plate-to-tube support plate welds" in the NuScale design. However, welds between sections of the steam generator tube supports are still included in the design. Table 5.2-7 has been updated to reflect the current weld design of the steam generator tube supports.

Impact on DCA:

FSAR Sections 3.2.2, 3.9.3.1.2, and 5.4.1, FSAR Figures 5.4-6 and 5.4-7, and FSAR Tables

3.2-1, 5.2-4, 5.2-6, 5.2-7, and 5.4-3 have been revised as described in the response above and as shown in the markup provided in this response.

Identifiers A - C correspond to ASME Class 1 through 3 and align with quality groups A - C. Code identifier D corresponds to Quality Group D as described in RG 1.26.

Safety-related instrument sensing lines are designed and constructed in accordance with ANSI/ISA-67.02.01-1999 (Reference 3.2-2) as described in RG 1.151. The standard ANSI/ISA-67.02.01-1999 establishes the applicable code requirements and code boundaries for the design and installation of instrument sensing lines interconnecting safety-related piping and vessels with both safety-related and nonsafety-related instrumentation. This is further discussed in Section 7.2.2.

RAI 03.02.01-2, RAI 03.02.02-3, RAI 05.04.02.01-6

The following subsections also describe the codes and standards applicable to supports for Quality Group A, B, C, and D components. The reactor vessel internals (see Section 3.9.5) and steam generator supports and tube supports (see Section 5.4.1.5) comply with the design and construction requirements of Subsection NG of Section III, Division 1 of the ASME B&PV Code (Reference 3.2-1).

3.2.2.1 Quality Group A

RAI 03.02.02-3

Quality Group A applies to pressure-retaining components ~~and their supports~~ that form part of the reactor coolant pressure boundary, except those that can be isolated from the reactor coolant system by two automatically-closed or normally-closed valves in series.

RAI 03.02.02-3, RAI 05.04.02.01-6

Quality Group A SSC meet the requirements for Class 1 components in Section III, Division 1 of the ASME B&PV Code (Reference 3.2-1). Supports for Quality Group A SSC meet the requirements for Class 1 supports in Section III, Division 1, Subsection NF of the ASME B&PV Code and are not separately listed in Table 3.2-1. Exceptions exist for supports within the pressure retaining boundary of the RPV. See Section 3.2.2 and Section 5.4.1.5 for additional information.

The remaining portions of the reactor coolant pressure boundary are in Quality Group B.

3.2.2.2 Quality Group B

Quality Group B applies to water- and steam-containing pressure vessels, heat exchangers (other than turbines and condensers), storage tanks, piping, pumps, and valves that are:

- part of the reactor coolant pressure boundary but are excluded from Quality Group A.
- safety-related or risk-significant systems or portions of systems that are designed for (i) emergency core cooling, (ii) post-accident containment heat removal, or (iii) post-accident fission product removal.

RAI 03.02.01-2, RAI 03.02.01-3, RAI 03.02.02-2, RAI 03.02.02-6, RAI 05.04.02.01-6, RAI 06.02.04-2, RAI 09.02.02-1, RAI 09.02.04-1, RAI 09.02.05-1, RAI 09.02.06-1, RAI 09.02.07-4, RAI 09.02.07-5, RAI 09.02.09-2, RAI 09.03.04-5, RAI 09.04.02-1, RAI 10.04.07-2, RAI 11.02-1, RAI 19-14

Table 3.2-1: Classification of Structures, Systems, and Components

SSC (Note 1)	Location	SSC Classification (A1, A2, B1, B2)	RTNSS Category (A,B,C,D,E)	QA Program Applicability (Note 2)	Augmented Design Requirements (Note 3)	Quality Group / Safety Classification (Ref RG 1.26 or RG 1.143) (Note 4)	Seismic Classification (Ref. ~~RG 1.189 or~~ RG 1.29 or RG 1.143) (Note 5)
CNTS, Containment System							
All components (except as listed below)	RXB	A1	N/A	Q	None	~~A~~B	I
• CVC Injection & Discharge Nozzles • CVC PZR Spray Nozzle • CVC PZR Spray CIV • CVC RPV High Point Degasification Nozzle • CVC RPV High Point Degasification CIV • RVV & RRV Trip/Reset # 1 & 2 Nozzles • RVV Trip 1 & 2/Reset #3 Nozzles • CVC Injection & Discharge CIVs	RXB	A1	N/A	Q	None	A	I
• RXM Lifting Lugs • Top ~~Auxiliary Mechanical Access~~Support Structure • Top ~~Auxiliary Mechanical Access~~Support Structure Diagonal Lifting Braces	RXB	B1	None	AQ-S	• ANSI/ANS 57.1-1992 • ASME NOG-1 • NUREG-0554	N/A	I
• CNV Fasteners • Hydraulic skid • CNV Seismic Shear Lug • CNV CRDM Support Frame • Containment Pressure Transducer (Narrow Range) • Containment Water Level Sensors (Radar Transceiver) • SG 1 & 2 Steam Temperature Sensors (RTD)	RXB	A1	N/A	Q	None	N/A	I
CNTS CFDS Piping in containment	RXB	B2	None	AQ-S	None	B	II
Piping from (CES, CFDS, ~~CVCS,~~ FWS, MSS, and RCCWS) CIVs to disconnect flange (outside containment)	RXB	B2	None	AQ-S	None	D	I
CVCS Piping from CIVs to disconnect flange (outside containment)	RXB	B2	None	AQ-S	None	C	I
~~Hydraulic Skid for valve reset~~	~~RXB~~	~~B2~~	~~None~~	~~None~~	~~None~~	~~D~~	~~III~~
CIV Close and Open Position Sensors: • CES, Inboard and Outboard • CFDS, Inboard and Outboard • CVCS, Inboard and Outboard PZR Spray Line • CVCS, Inboard and Outboard RCS Discharge • CVCS, Inboard and Outboard RCS Injection • CVCS, Inboard and Outboard RPV High-Point Degasification • FWS, Supply to SGs and DHR HXs FWIV • RCCWS, Inboard and Outboard Return and Supply • SGS, Steam Supply CIV/MSIVs and CIV/MSIV Bypasses	RXB	B2	None	AQ-S	IEEE 497-2002 with CORR 1	N/A	I
Containment Pressure Transducer (Wide Range)	RXB	B2	None	AQ-S	IEEE 497-2002 with CORR 1	N/A	~~II~~I
• Containment Air Temperature (RTDs) • FW Temperature Transducers	RXB	B2	None	AQ-S	None	N/A	II
SGS, Steam Generator System							
• SG tubes • Feedwater plenums • Steam plenums ~~SG tube supports~~	RXB	A1	N/A	Q	None	A	I
• SG tube supports • Upper and lower SG supports	RXB	A1	N/A	Q	None	N/A	I
• Steam piping inside containment • Feedwater piping inside containment • Feedwater supply nozzles • Main steam supply nozzles • Thermal relief valves	RXB	A2	N/A	Q	None	B	I
Flow restrictors	RXB	A2	N/A	Q	None	N/A	I

Table 3.2-1: Classification of Structures, Systems, and Components (Continued)

SSC (Note 1)	Location	SSC Classification (A1, A2, B1, B2)	RTNSS Category (A,B,C,D,E)	QA Program Applicability (Note 2)	Augmented Design Requirements (Note 3)	Quality Group / Safety Classification (Ref RG 1.26 or RG 1.143) (Note 4)	Seismic Classification (Ref. ~~RG 1.189 or~~ RG 1.29 or RG 1.143) (Note 5)
RXC, Reactor Core System							
Fuel assembly (RXF)	RXB	A1	N/A	Q	None	N/A	I
Fuel Assembly Guide Tube	RXB	A2	N/A	Q	None	N/A	I
Incore Instrument Tube	RXB	B2	None	AQ-S	None	N/A	I
CRDS, Control Rod Drive System							
• Control Rod Drive Shafts • Control Rod Drive Latch Mechanism	RXB	A1	N/A	Q	None	N/A	I
CRDM Pressure Boundary (Latch Housing, Rod Travel Housing, Rod Travel Housing Plug)	RXB	A2	N/A	Q	None	A	I
CRDS Cooling Water Piping and Pressure Relief Valve	RXB	B2	None	AQ-S	None	B	II
Rod Position Indication (RPI) Coils	RXB	B2	None	AQ-S	None	N/A	I
• Control Rod Drive Coils • CRDM power cables from EDN breaker to MPS breaker • CRDM power cables from MPS breaker to CRDM Cabinets	RXB	B2	None	AQ-S	None	N/A	II
• CRDM Control Cabinet • CRDM Power & Rod Position Indication Cables • Rod Position Indication Cabinets (Train A/B)	RXB	B2	None	AQ	None	N/A	III
CRA, Control Rod Assembly							
All components	RXB	A2	N/A	Q	None	N/A	I
NSA, Neutron Source Assembly							
All components	RXB	B2	None	AQ-S	None	N/A	I
RCS, Reactor Coolant System							
All components (except as listed below)	RXB	A1	N/A	Q	None	A	I
• Reactor vessel internals (upper riser assembly, lower riser assembly, core support assembly, flow diverter, and pressurizer spray nozzles) • Narrow Range Pressurizer Pressure Elements • PZR/RPV Level Elements • Narrow Range RCS Hot Leg Temperature Elements • Wide Range RCS Hot Leg Temperature Elements • RCS Flow Transmitters (Ultrasonic)	RXB	A1	None	Q	None	N/A	I
• Wide Range RCS Pressure Elements • Wide Range RCS Cold Leg Temperature Elements	RXB	A2	N/A	Q	None	N/A	I
Reactor Safety Valve Position Indicator	RXB	B2	None	AQ-S	Environmental Qualification Power from EDS	N/A	I
• PZR Control Cabinet • PZR Vapor Temperature Element • PZR heater power cabling from MPS breaker to PZR heaters • Pressurizer Liquid Temperature Element • Narrow Range RCS Cold Leg Temperature Element	RXB	B2	None	AQ-S	None	N/A	II
PZR heater power cabling from ELV breaker to MPS breaker	RXB	B2	None	None	None	N/A	III
CVCS, Chemical and Volume Control System							
DWS Supply Isolation Valves	RXB	A1	N/A	Q	None	C	I
Position Indication for DWS Supply Isolation Valves	RXB	B2	None	AQ-S	IEEE 497-2002 with CORR 1	N/A	I

The lifting, handling, and transportation load contains a 15 percent dynamic load factor, for a total load of 115 percent times the DW load applied at the lifting and transportation support points.

RAI 03.09.03-1

Lifting, handling, and transportation loads are not required to meet ASME stress limits. However, the Service Level B primary limits are used as the allowable limits for the lifting, handling, and transportation loads. The platform mounting assemblies are analyzed to ensure minimum safety factors of five for material ultimate strength and three for material yield strength, and are maintained for dual-load-path loading conditions considering the dynamic load factor specified above.

RAI 03.09.03-1

Hydrogen Detonation

Short duration pressure pulse due to hydrogen detonation and hydrogen detonation with deflagration-to-detonation transition resulting from a combustible gas that results from a fuel-clad metal-water reaction followed by an uncontrolled hydrogen burn during a post-accident condition is evaluated per the rules defined in 10 CFR 50.44, 10 CFR 50.34 and RG 1.7, "Control of Combustible Gas Concentrations in Containment." Revision 3.

3.9.3.1.2 Load Combinations and Stress Limits

The RPV is a Seismic Category 1, ASME Section III, Class 1 component. The load combinations and stress limits for the RPV and its supports are presented in Table 3.9-3.

The CNV is a Seismic Category 1 component. The ASME classification of the CNV and its supports is described in Section 3.8.2.2. The load combinations and stress limit for CNV and its supports are presented in Table ~~3.9-4~~3.8.2-2.

The RVI are Seismic Category 1 components. Portions of the RVI, which perform a core support function, are classified as Class CS components in accordance with ASME Section III, Subsection NG. The remaining portions of the RVI are designated as internal structures; however, they are designed using NG-3000 as a guide and constructed to ASME Subsection NG. The load combinations and stress limit are presented in Table 3.9-5.

RAI 03.09.05-16, RAI 05.04.02.01-6

The SG supports and SG tube supports are Seismic Category 1 components. The SG supports and SG tube supports are designated as internal structures and are designed using ASME Section III, Subsection NG as a guide. The load combinations and stress limit are consistent with those presented in Table 3.9-5.

The portions of the CRDM providing a RCPB function are ASME Code Class 1, Seismic Category I components. The CRDM coil heat exchangers, tubes, and connections, which provide cooling water and are external to the RCPB, are ASME

RAI 05.02.03-1, RAI 05.02.03-9, RAI 05.02.03-12, RAI 05.03.01-3, RAI 05.04.02.01-6, RAI 06.01.01-3

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

Component	Specification	Alloy Designation (Grade, Class, or Type)[1]
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 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~~SA-479 or SA-240	Type 304~~/304L; Grade B8, Class 1~~ with 0.03% max carbon
~~Vessel alignment pins~~ ~~RPV flange stud threaded inserts~~Pressure instrument tap swagelok reducers Threaded inserts for: RSV flanges Instrumentation and controls (I&C) access ports PZR heater access ports Steam plenum access ports Feed plenum access ports ~~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~~Threaded fasteners, nuts, and washers for: Main RPV flange RSV flanges I&C access ports PZR heater access ports Steam plenum access ports Feed plenum access ports	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)

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

Component	Specification	Alloy Designation (Grade, Class, or Type)[1]
Safe ends for: • RRV • CVCS charging and letdown nozzles • CRDM nozzles • RVV • High point degasification nozzle • Pressurizer 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
Stainless steel weld filler material (includes filler material for cladding)	SFA 5.4 SFA 5.9 SFA-5.22	E308, E308L, E309, E309L, E316, E316L ER308, ER308L, ER309, ER309L ER316, ER316L, EQ308L, EQ309L E308, E308L, E309, E309L, E316, E316L
Nickel-based alloy weld filler material	SFA-5.11 SFA-5.14	ENiCrFe-7 ERNiCrFe-7, ERNiCrFe-7A, ~~ERNiCrFe-13,~~ EQNiCrFe-7, EQNiCrFe-7A
Steam Generators		
SG tubes	~~SB-163Alloy 690 (UNS N06690)~~See Section 5.4.1.5	
SG tube supports	SA-240	Type 304/304L
Upper and Lower SG supports	SA-240	Type 304/304L
Integral steam plenum cap	SB-564	Alloy 690 (UNS N06690)
Nickel-based alloy weld filler material	SFA-5.11 SFA-5.14	ENiCrFe-7 ERNiCrFe-7, ERNiCrFe-7A, ~~ERNiCrFe-13,~~ EQNiCrFe-7, EQNiCrFe-7A
Piping	See Table 5.4-3	
Piping supports		
Piping reducers and elbows		
RVVs and RRVs		
Refer to Table 6.1-~~1~~3		
RCS ~~Injection and Discharge and High Point Vent Class I~~ Piping		
~~Containment to check valve piping~~ ~~Check valve to RPV piping~~RCS injection line, CNV to RPV RCS discharge line, RPV to CNV RPV high point degasification line, RPV to CNV PZR spray supply line, CNV to RPV	SA-312	~~Type~~ Grade TP304/304L
Stainless steel weld filler materials[2]	SFA 5.4 SFA 5.9	E308, E308L, E316, E316L ER308, ER308L, ER316, ER316L
RCS piping reducers and elbows	SA-479	Type 304/304L
Tee connection to ECCS reset valves	SA-182	Grade F304/F304L
RCS Check Valves		

RAI 05.02.04-3, RAI 05.03.01-3, RAI 05.04.02.01-6, RAI 06.06-3

Table 5.2-6: Reactor Pressure Vessel Inspection Elements

Description	Examination Category	Examination Method	Notes
RPV Shell and Head Welds			
Lower RPV flange shell to RPV bottom head Upper RPV flanged transition shell to lower SG shell Lower SG shell to upper SG shell Upper SG shell to integral steam plenum Integral steam plenum to PZR shell PZR shell to RPV top head Steam plenum cap to integral steam plenum	B-A	Volumetric	
RPV Internal Welds			
Core support block to RPV bottom head Core support block to latch Core barrel guide to lower RPV flange shell Upper ~~tube support bar~~SG support to lower RPV integral steam plenum Lower ~~tube support cantilever~~SG support to upper RPV	B-N-2	VT-3	
Instrumentation and Controls Sleeve Welds	None	None	These welds are part of the cladding.
Flow diverter to RPV lower head RPV interior surfaces and attachment welds	B-N-1	VT-3	B-N-1 is for the space above and below the core made accessible by removal of components during a normal refueling outage
RPV External Welds			
RPV support plate to RPV support gussets RPV support plate to upper RPV SG shell 1-4	F-A	VT-3	
RPV support plate to upper RPV SG shell RPV support gussets to upper RPV SG shell RPV lateral support lug	B-K	Surface or Volumetric	
RPV Nozzle to Shell and Head Welds			
Reactor recirc valve~~s~~ flange Feedwater nozzles RCS discharge	B-D	Volumetric	Inside corner. All welds examination requirement IWB-2500-7(d).
Main steam nozzles	B-D	Volumetric	Examination requirement IWB-2500-7(d)
RCS injection PZR spray supply lines	B-D	N/A	No inside corner

RAI 04.05.02-2, RAI 05.04.02.01-6

Table 5.2-7: Reactor Vessel Internals Inspection Elements

Description	Location	Examination Cat~~a~~egory	Examination Method	Notes
Core Support Components				
Reflector Block - Bottom	Core Support Assembly	B-N-3	VT-3	
Reflector Block Intermediate	Core Support Assembly	B-N-3	VT-~~3~~1	Required VT-3 augmented to VT-1. Exam will be of the interior surface, checking for a gap developing between reflector blocks.
Reflector Block Top	Core Support Assembly	B-N-3	VT-3	
Reflector Block Alignment Pins	Core Support Assembly	B-N-3	VT-~~3~~1	Inspection only required when reflector blocks are removed for another reason
Core Barrel	Core Support Assembly	B-N-3	VT-~~3~~1	Required VT-3 augmented to VT-1 of accessible surfaces
Lower core Plate	Core Support Assembly	B-N-3	VT-~~3~~1	Required VT-3 augmented to VT-1 of accessible surfaces
Upper Core Plate	Lower Riser Assembly	B-N-3	VT-3	
Lower Core Plate Alignment Pins	Core Support Assembly	B-N-3	VT-3	
Upper Support Block	Core Support Assembly	B-N-2	VT-1	
Core Barrel to Lower Core Plate	Core Support Assembly	B-N-2	VT-1	
Fuel Pins	Lower Riser Assembly	B-N-3	VT-~~3~~1	Required VT-3 augmented to VT-1
Fuel Pins Caps	Lower ~~r~~Riser Assembly	B-N-3	VT-~~3~~1	Required VT-3 augmented to VT-1
Fuel Pin Capture Weld	Lower Riser Assembly	B-N-2	VT-1	
Shared Fuel Pins and Nuts	Core Support Assembly	B-N-3	VT-~~3~~1	Required VT-3 augmented to VT-1
Lower Riser to Upper Core Plate	Lower Riser Assembly	B-N-3	VT-3	
ICIGT Bottom Flag ICIGT 1 to Upper Core Plate	Lower Riser Assembly	B-N-3	VT-1	Required VT-3 augmented to VT-1
Upper Seismic Belleville Washers	Core Support Assembly	B-N-3	VT-3	
Lower Seismic Belleville Washers	Core Support Assembly	B-N-3	VT-1	VT-1 measurement of the lower core plate height relative to the core support block
Upper Seismic Belleville Retaining Nut	Core Support Assembly	B-N-3	VT-3	

Table 5.2-7: Reactor Vessel Internals Inspection Elements (Continued)

Description	Location	Examination ~~Cataegory~~ Category	Examination Method	Notes
Riser Backing Strip A	Upper Riser Assembly	B-N-1	VT-3	
Riser Backing Strip B	Upper Riser Assembly	B-N-1	VT-3	
Upper CRDS Support	Upper Riser Assembly	B-N-1	VT-3	
Brace to Upper Riser Hanger Ring	Upper Riser Assembly	B-N-1	VT-3	
Brace to Upper Riser Section	Upper Riser Assembly	B-N-1	VT-3	
Upper Riser to Bellows	Upper Riser Assembly	B-N-1	VT-3	
Bellows to Upper Riser Transition	Upper Riser Assembly	B-N-1	VT-3	
Pipe to Bellows	Upper Riser Assembly	B-N-1	VT-3	
Elbow to Flexible Pipe	Upper Riser Assembly	B-N-1	VT-3	
Flexible Pipe to Rigid Pipe	Upper Riser Assembly	B-N-1	VT-3	
Brace to Pipe	Upper Riser Assembly	B-N-1	VT-3	
Pipe to Cap	Upper Riser Assembly	B-N-1	VT-3	
ICI Centering Plate 1-12	Upper Riser Assembly	B-N-1	VT-3	
ICIGT Link to Upper Riser Hanger Ring	Upper Riser Assembly	B-N-1	VT-3	
Injection to RPV11	Upper Riser	B-N-1	VT-3	
ICIGT to Integrated Steam Plenum	PZR	B-N-1	VT-3	
Pressurizer Spray Nozzle	PZR Spray Nozzle to Safe End (RPV14-RPV15)	B-N-1	VT-3	
Surveillance Capsule	Core Support Assembly	B-N-1	VT-3	
Steam Generator				
Col 01-21 ~~Tube Support Plate to Tube Support Plate Weld 1~~Tube Support Top/ Bottom Sections to Tube Support middle Section	SG	~~F-A~~N/A	~~VT-3~~General Visual	Augmented exam. Examine where accessible.
Col 01-21 ~~Tube Support Plate to Tube Support Plate Weld 2~~Outer Spacer to Tube Support	SG	~~F-A~~N/A	~~VT-3~~General Visual	Augmented exam. Examine where accessible.
Col 01-21 ~~Tube Support Plate to Tube Support Plate Weld 3~~Middle Spacer to Tube Support	SG	~~F-A~~N/A	~~VT-3~~General Visual	Augmented exam. Examine where accessible.
~~Col 01-21 Tube Support Plate to Tube Support Plate Weld 4~~	~~SG~~	~~F-A~~	~~VT-3~~	
~~Col 01-21 Tube Support Plate to Tube Support Plate Weld 5~~	~~SG~~	~~F-A~~	~~VT-3~~	

Table 5.2-7: Reactor Vessel Internals Inspection Elements (Continued)

Description	Location	Examination ~~Cataegory~~ Category	Examination Method	Notes
~~Col 01-21 Tube Support Plate to Top Support Bracket A~~	~~SG~~	~~F-A~~	~~VT-3~~	
~~Col 01-21 Tube Support Plate to Bottom Support Bracket A~~	~~SG~~	~~F-A~~	~~VT-3~~	
~~Col 01-21 Backing Strip A-1 Inner~~	~~SG~~	~~F-A~~	~~VT-3~~	
~~Col 01-21 Backing Strip A-1 Outer~~	~~SG~~	~~F-A~~	~~VT-3~~	
~~Col 01-21 Backing Strip A-2 Inner~~	~~SG~~	~~F-A~~	~~VT-3~~	
~~Col 01-21 Backing Strip A-2 Outer~~	~~SG~~	~~F-A~~	~~VT-3~~	
~~Col 01-21 Tube Support Plate to Top Support Bracket B~~	~~SG~~	~~F-A~~	~~VT-3~~	
~~Col 01-21 Tube Support Plate to Bottom Support Bracket B~~	~~SG~~	~~F-A~~	~~VT-3~~	
~~Col 01-21 Backing Strip B-1 Inner~~	~~SG~~	~~F-A~~	~~VT-3~~	
~~Col 01-21 Backing Strip B-1 Outer~~	~~SG~~	~~F-A~~	~~VT-3~~	
~~Col 01-21 Backing Strip B-2 Inner~~	~~SG~~	~~F-A~~	~~VT-3~~	
~~Col 01-21 Backing Strip B-2 Outer~~	~~SG~~	~~F-A~~	~~VT-3~~	

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 ~~Cantilevers~~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 ~~be~~ used. Instead, the NPM steam generator employs a system of austenitic stainless steel tube support ~~bar~~ 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 ~~bar~~ assemblies are located between each column of helical tubes. Stamped tabs in the ~~support bars~~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 ~~bar~~ 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. ~~The~~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 ~~support bars~~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 ~~vertical~~ tube support ~~bar~~ assemblies are attached to upper ~~tube support bars~~SG supports that are welded to the inner surface of the RPV and also interface with lower ~~tube support cantilevers~~SG supports that are welded to the inner surface of the RPV. The SG tube support ~~bar~~ assemblies in the SG provides contact with each tube at eight separate circumferential locations. The use of 8 sets of tube support ~~bar~~ 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 of Figure 5.4-6, the lower tube support cantilevers support the vertical tube support bars and permit thermal growth of the tube support bars while providing~~

~~lateral restraint for the support bars.~~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

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 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.

To ~~prevent~~control instabilities in individual SG tubes due to ~~two-phase conditions in the SG tubes as the~~ fluid ~~is~~ brought to boiling conditions as it travels up the tubes, inlet flow ~~resistance is~~restrictors are added at the feedwater inlet ~~plena~~plenum interface. Analysis shows that SG secondary side flow oscillations are decoupled from primary side flow oscillations and thus secondary side flow instabilities do not cause reactor power oscillations. In-phase oscillation of secondary flow does not occur, and the out-of-phase oscillatory flow in individual tubes cancels out, so that the net secondary flow is not oscillating. NRELAP5 analysis shows that an inlet loss coefficient (K) of at least 900 ensures that the tube mass flow rate is stable with fluctuations of less than ±10 percent for all power levels above 5 percent. Additional pressure loss is added to the inlet restriction to provide margin based on a comparison of the NRELAP5 results to test data.

RAI 05.04.02.01-6

The comprehensive vibration assessment program conforms to the guidance of Regulatory Guide (RG) 1.20, Revision 3. Based on the integral design of the NPM, the SG pressure retaining components are located within the fluid volume of the RPV, along with the reactor internal components. Therefore, ~~the SGs and main steam piping up to and including the MSIVs and the SG tube supports~~ the SGs, main steam piping up to and including the MSIV, SG supports, and the SG tube supports are included in the comprehensive vibration assessment plan.

A set of flow-induced vibration screening criteria were developed for the comprehensive vibration assessment plan as described in Section 3.9. Under normal operating conditions, the flow energy available to excite tube vibration is low due to the low primary coolant flow rates in the NPM design.

Based on an evaluation of the screening criteria, the following lists the flow-induced vibration mechanisms and susceptible SG components that require flow-induced vibration analysis:

- fluid elastic instability: SG tubes are susceptible.

RAI 05.04.02.01-6

- vortex shedding: SG tubes and lower SG ~~tube support cantilevers~~supports are susceptible.

RAI 05.04.02.01-6

- turbulent buffeting: SG tubes, SG tube ~~support bars~~supports, ~~SG, lower tube support cantilevers,~~lower SG supports and SG inlet flow restrictors are susceptible.
- acoustic resonance: Steam nozzles, steam piping, integral steam plenum, and MSIVs are susceptible.
- leakage flow: SG inlet flow restrictors are susceptible.

RAI 05.04.02.01-6

- galloping and flutter: ~~SG lower tube support cantilevers~~Lower SG supports are susceptible.

A fluid elastic instability analysis was performed for the tube bundle over a range of 50 vibration modes based on comparison of the effective cross-flow velocity to a critical

velocity for fluid elastic instability determined using the Connor's criteria (Reference 5.4-7). The vibration modes considered in the analysis included both beam modes and lower frequency helical "breathing" modes. Fluid elastic instability testing of helical tube arrays suggests that the onset of fluid elastic instability in helical tube arrays is governed by the same parameters as for arrays of straight tubes. Connor's constants selected for the fluid elastic instability evaluation considered the results of this testing. Comparison of effective cross-flow velocities to calculated critical velocities for the NPM steam generator showed a minimum margin of 33 percent to the onset of fluid elastic instability.

Vortex shedding is precluded for downstream tubes based upon the overall turbulence created due to flow through the tube bundle disrupting the formation of coherent vortices. However, the tubes at the inlet to the tube bundle are evaluated to ensure that vortex shedding is precluded based on tube structural frequencies, damping, overall flow velocity, and a combination of these factors. The tubes at the bottom of the bundle (cold leg) are the only tubes susceptible to vortex shedding because they have no adjacent downstream obstruction to disrupt vortices.

Turbulent flow was evaluated to determine the mechanical wear of the SG tubes by turbulence in the RCS flow and tube internal axial flow. A 20.8 percent fatigue usage factor is calculated based on tube impact stress. Maximum fretting wear thickness ratio is calculated to be 34.2 percent during the 60-year design life, which is less than 40 percent allowable tube wall wear depth.

The SG tube flow restrictors are rigid components and have a high fundamental frequency, which provides for low turbulent buffeting vibration amplitudes. Due to the low vibration amplitudes, fatigue stresses are negligible and impact between the flow restrictor and the SG tube inner diameter does not occur.

RAI 05.04.02.01-6

The SG tube support~~ bar~~ assemblies span the full height of the helical tube bundle and are anchored at the top by the ~~tube support bars~~upper SG supports and at the bottom by connection to the ~~tube support cantilevers~~lower SG supports. The confinement of the ~~bars~~tube supports within the tube bundle, where the tortuous flow path creates continuous turbulence, ensures formation of stable vortices does not occur. The axial alignment of the tube ~~support bars~~supports to the prevailing flow through the bundle ensures that the angle of attack is close to 0.0 and the aspect ratio is > 2.0, thus precluding galloping and flutter.

RAI 05.04.02.01-6

The upper ~~tube support bars~~SG supports are welded to the RPV and the integral steam plenum, and excluded from analysis due to their rigidity. The lower ~~tube support cantilevers~~SG supports are welded to the RPV shell below the SG tube bundle. Individual tube ~~support bars~~supports are joined to the ~~tube support cantilevers~~lower SG supports. Below the tube bundle, vortex shedding from the ~~cantilevers~~lower SG supports is possible. Based on their form as solid bars and an aspect ratio of four, they are not subject galloping flow-induced vibration mechanisms.

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.

The SG piping from the FWIVs and MSIVs to the SGs, steam plenum access ports, and plenum 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 piping, steam plenum access ports, and plenum 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 piping, associated reducers and elbows, steam plenum access ports, and plenum covers and fasteners are provided in Table 5.4-3.

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

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

The SG inlet flow restrictors ~~are classified as Quality Group B and~~ are designed, fabricated, constructed, tested, and inspected as ~~non-ASME Code components~~non-structural attachments.

RAI 05.04.02.01-11

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 5.2.3.4.2 for cleaning and cleanliness controls for the SGs. 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 (Reference 5.4-2). 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~~.

RAI 05.04.02.01-2

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 5.4-2) based on incorporation of a substantial degradation allowance (additional tube wall thickness above minimum required for design) as discussed in Section 5.4.1.2. 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.

RAI 05.04.02.01-6

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 support~~s~~ structure~~s~~. 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.

RAI 05.04.02.01-14

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 the latest revision of Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines," ~~Revision 3~~ and applicable ~~EPRI~~Electric Power Research Institute steam generator guidelines at the time of the COL application. The elements of the program will include: assessment of degradation, tube inspection requirements, tube integrity assessment, tube plugging, primary-to-secondary leakage monitoring, shell side integrity and accessibility assessment, steam plant corrosion product deposition assessment, primary and secondary side water chemistry control, foreign material exclusion, loose parts management, contractor oversight, self-assessment, and reporting.

RAI 05.04.02.01-6, RAI 05.04.02.01-7

Table 5.4-3: Steam Generator Piping, ~~Tube and~~ Piping Supports, and Flow Restrictor Materials

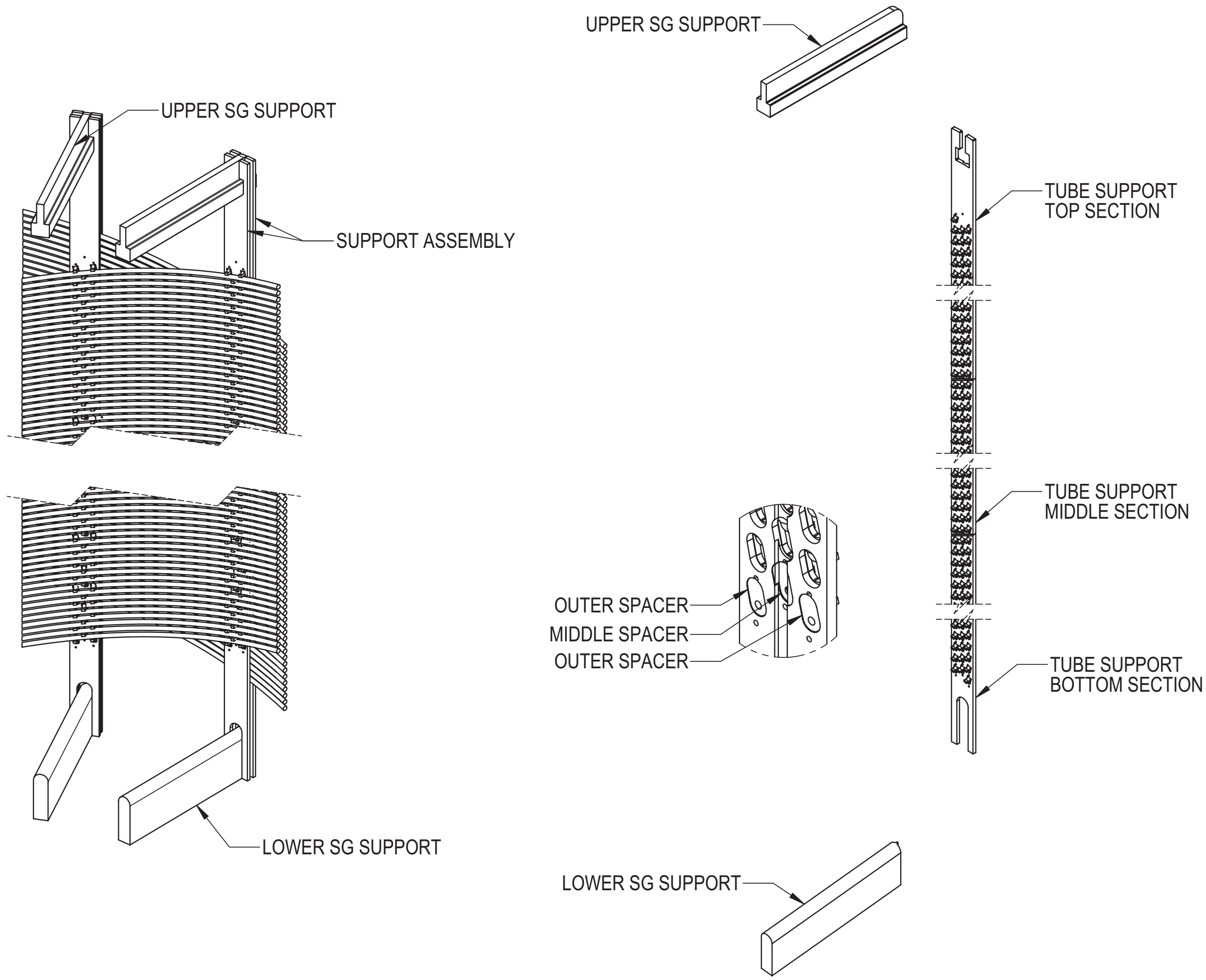
Component	Specification	Alloy Designation[1] (Grade, Class, or Type)
SG Class 2 piping	SA-312	Type ~~340~~304/304L
SG piping reducers and elbows	SA-182	Grade F304/F304L
SG piping supports	SA-479	Type 304/304L
SG tubes	See Section 5.4.1.5	
Flow restrictors and flow restrictor bolts	SA-479	Type 304
Flow restrictor mounting plates	SA-240	Type 304
Flow restrictor mounting plate spacer	SB-166	Alloy 690 (UNS N06690)
Flow restrictor stud bolts, nuts, and washers Integral steam and feed plenum port cover threaded fasteners	SB-637	Alloy 718 (UNS N07718)
Integral steam and feed plenum access ports	SA-508	Grade 3, Class 2
Integral steam and feed plenum access port covers	SA-240	Type 304/304L
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
Stainless steel weld filler material (includes filler material for cladding)	SFA 5.4 SFA 5.9 SFA-5.22	E308, E308L, E309, E309L, E316, E316L ER308, ER308L, ER309, ER309L ER316, ER316L, EQ308L, EQ309L E308, E308L, E309, E309L, E316, E316L
Nickel-based alloy weld filler material	SFA-5.11 SFA-5.14	ENiCrFe-7 ERNiCrFe-7, ERNiCrFe-7A, ~~ERNiCrFe-13,~~ EQNiCrFe-7, EQNiCrFe-7A

Notes:

1 - When the material is designated as Type or Grade 304/304L, this refers to dual certified stainless steel material.

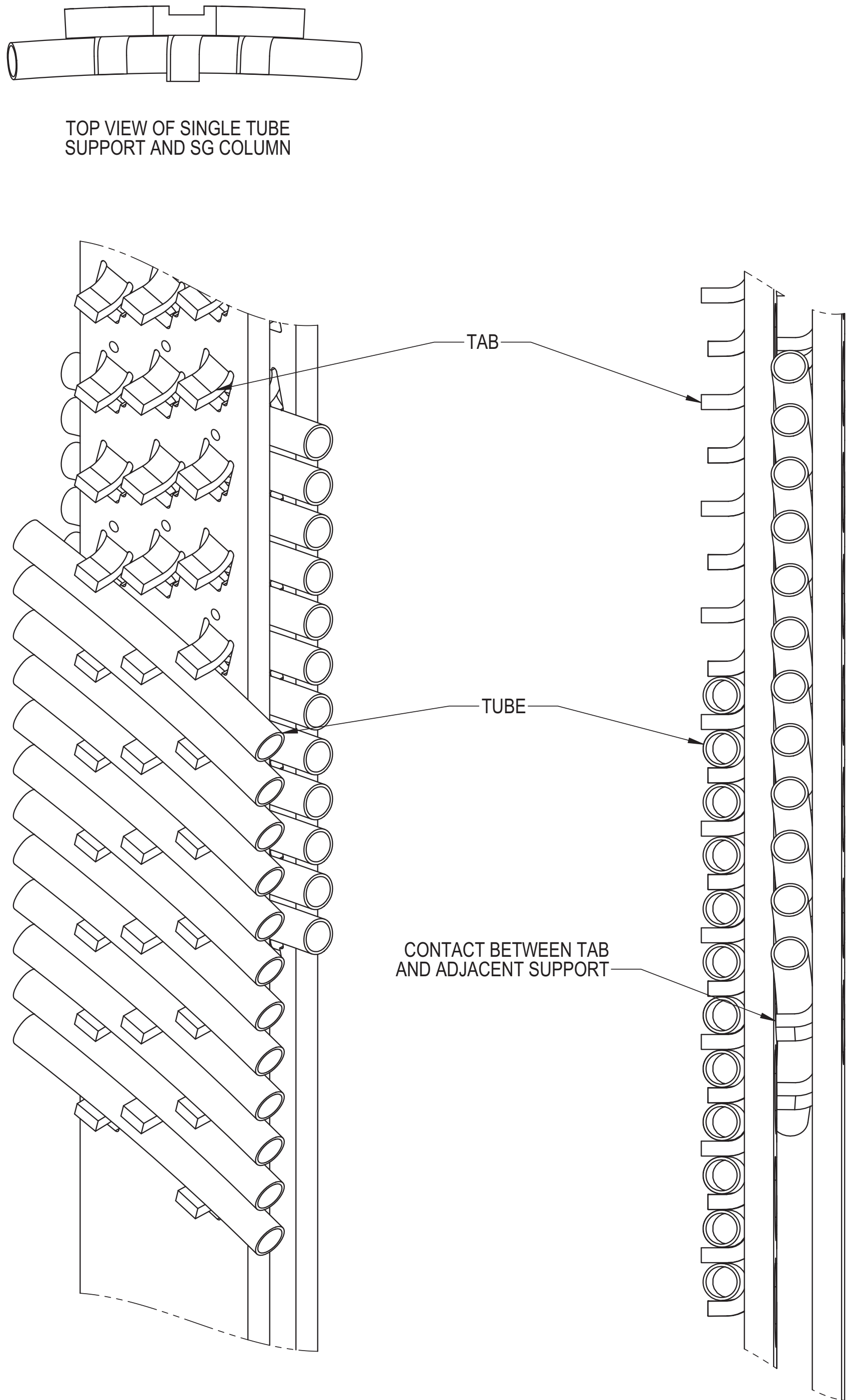
RAI 03.09.02-17, RAI 05.04.02.01-6

Figure 5.4-6: ~~Steam Generator Tube Support Bars and Cantilevers~~Steam Generator Tube Supports and Steam Generator Supports



RAI 05.04.02.01-6

Figure 5.4-7: Steam Generator Tube Support Tabs



NuScale Power

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

eRAI No.: 9231
Date of RAI Issue: 12/13/2017

NRC Question No.: 05.04.02.01-7

The NRC staff notes that Table 5.4-3 in Tier 2 of the FSAR indicates “Type 340/304L” for the SG Class 2 piping. Section 52.6(a) of 10 CFR requires that information provided to the NRC by an applicant for a standard design certification be complete and accurate in all material respects. Therefore, please revise this table to read “Type 304/304L.”

NuScale Response:

The requested change has been made. See markup to Table 5.4-3 provided in the response to question 05.04.02.01-6.

Impact on DCA:

FSAR Table 5.4-3 has been revised as described in the response above and as shown in the markup provided with the response to question 05.04.02.01-6.

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

eRAI No.: 9231
Date of RAI Issue: 12/13/2017

NRC Question No.: 05.04.02.01-8

The flow restrictors in the SG tube ends must enable the full range of design flow rates and resist degradation that could lead to damage of the SG tubes and other downstream components in the steam and feedwater system. This is necessary for the flow restrictors to satisfy the requirements of 10 CFR Part 50, Appendix A, GDC 1, 4, 15, 30, and 31, as they relate to ensuring the integrity of the RCPB. Please provide the following information about the flow restrictors:

a. A description of the inspection requirements for the flow restrictors. Table 3.2-1 in Tier 2 of the FSAR indicates that while the flow restrictors are safety-related, Quality Group and Safety Classifications are not applicable. This appears to be inconsistent with the statement in Section 5.4.1.5 in Tier 2 of the FSAR that the flow restrictors are Quality Group B, but designed, fabricated, constructed, tested, and inspected as non-ASME Code components.
b. A description of the evaluations performed for potential degradation related to the flow restrictors and associated hardware, including a discussion of the likelihood and extent of loose parts generated from the flow restrictors, and the provisions for managing that degradation. In addition, a description of the evaluations performed to determine the impact of removing flow restrictors in order to plug tubes.
c. Table 5.4-3 in Tier 2 of the FSAR includes components called “flow restrictor bolts,” “flow restrictor stud bolts,” and “flow restrictor mounting plate spacer.” Revise Figure 5.4-8 of Tier 2 of the FSAR to clarify their location and revise Section 5.4.1.2 in Tier 2 of the FSAR to clarify their functions.

NuScale Response:

a. As stated in FSAR Section 5.4.1.5, the Steam Generator (SG) inlet flow restrictors are designed, fabricated, constructed, tested, and inspected as non-structural attachments. The inconsistency between FSAR Table 3.2-1 and FSAR Section 5.4.1.5 regarding Quality Group Classification has been corrected.
b. The NuScale SG inlet flow restrictors are constructed of corrosion resistant materials to prevent degradation of the restrictors. A separate effects test is planned for the SG inlet

flow restrictors as part of the comprehensive vibration assessment program (CVAP) to evaluate vibration performance (see TR-0716-50439-P). The flow restrictors are removable components by design that can be inspected for degradation and replaced, if necessary, during outages. Since the flow restrictors are designed to be removed, there is no impact to the flow restrictor assembly when removing the restrictors to plug a tube. The potential for loose part generation is minimized for the SG inlet flow restrictors based on use of a locking mechanism on the flow restrictor bolts or nuts and flow restrictor stud bolts or nuts.

c. FSAR Figures 5.4-5 and 5.4-8 and FSAR Section 5.4.1.2 have been revised to clarify the location and function of the “flow restrictor bolts,” “flow restrictor stud bolts,” and “flow restrictor mounting plate spacer.”

Impact on DCA:

FSAR Section 5.4.1.2, Section 5.4.1.5, Figure 5.4-5 and Figure 5.4-8 have been revised as described in the response above and as shown in the markup provided in this response.

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

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 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 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.

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

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

The SG inlet flow restrictors ~~are classified as Quality Group B and~~ are designed, fabricated, constructed, tested, and inspected as ~~non-ASME Code components~~non-structural attachments.

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

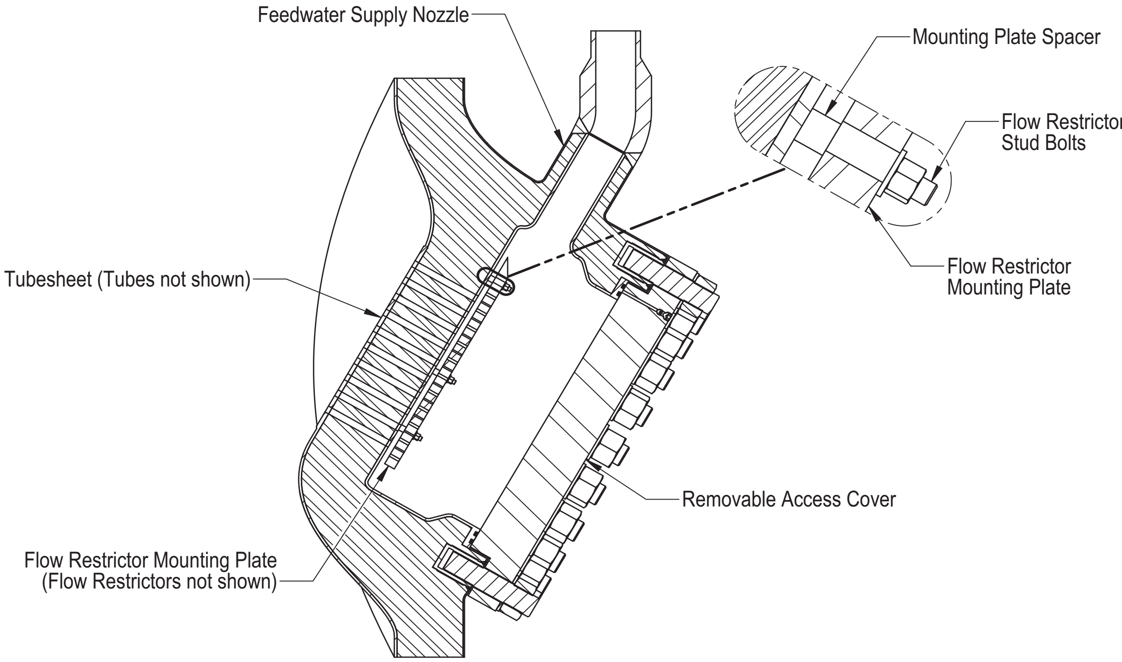
The SG program monitors the performance and condition of the SGs to ensure they are capable of performing their intended functions. The program provides monitoring and management of tube degradation and degradation precursors that permit preventative and corrective actions to be taken in a timely manner, 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.

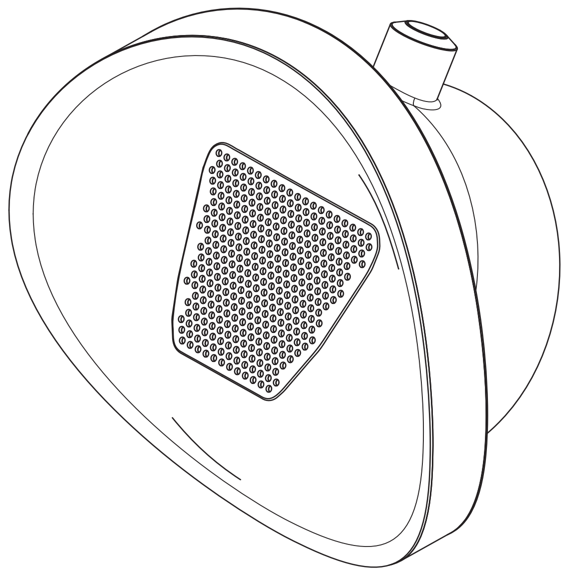
Historically, significant SG tube degradation has occurred in the operating PWR SG fleet due to various corrosion mechanisms, including wastage and both primary and secondary side stress corrosion cracking. These corrosion mechanisms were related to materials selection, plant chemistry control, and control of the ingress of impurities and

RAI 05.04.02.01-5

Figure 5.4-5: Feedwater Plenum Access Port

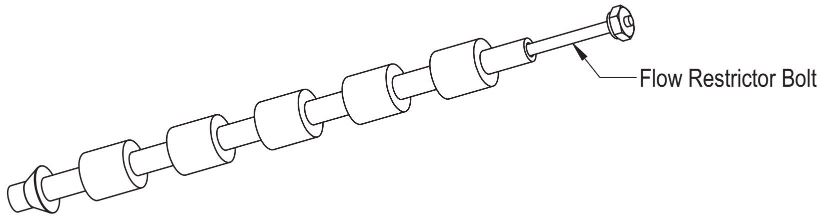


FEED PLENUM ACCESS PORT CUTAWAY

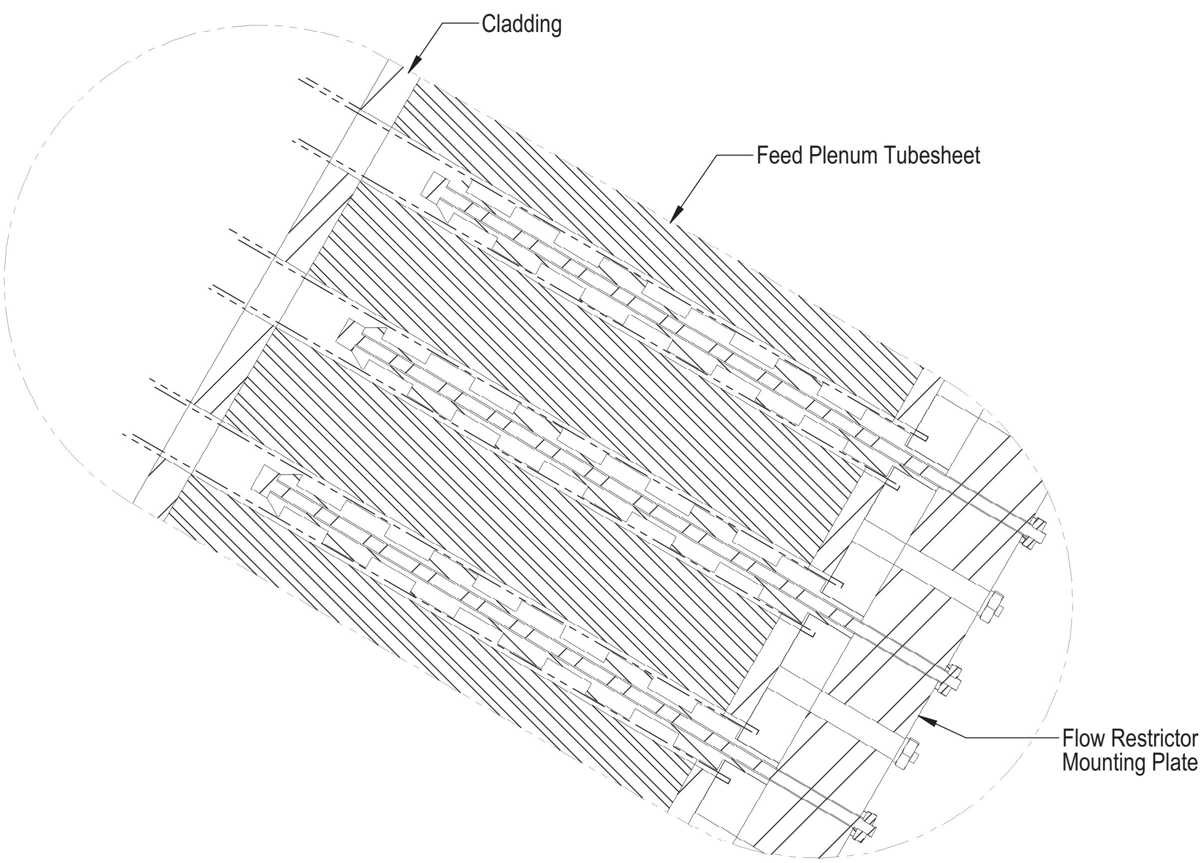


RAI 05.04.02.01-8

Figure 5.4-8: Steam Generator Flow Restrictor Assembly



Flow Restrictor



Flow Restrictor Mounting

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

eRAI No.: 9231
Date of RAI Issue: 12/13/2017

NRC Question No.: 05.04.02.01-9

Section 5.4.1.2 in Tier 2 of the FSAR states that the "chemistry of the primary and secondary water is controlled in accordance with industry guidelines suitably modified to address the unique NPM (NuScale Power Module) design and to ensure compatibility with the primary and secondary coolant." In this case, the industry guidelines are the Electric Power Research Institute (EPRI) Pressurized Water Reactor Primary Water Chemistry Guidelines and Pressurized Water Reactor Secondary Water Chemistry Guidelines (EPRI Guidelines). Explain what is meant by "suitably modified" or, if the EPRI Guidelines will be followed without modification, revise Section 5.4.1.2 to state that water chemistry will be controlled "in accordance with" the EPRI Guidelines. The NRC staff requires this information because the primary and secondary coolant water chemistry should be controlled consistent with the requirements of GDC 4.

So that you may ensure consistency in your responses and changes to the FSAR, the NRC staff notes that related requests for information (RAI) have been issued under Sections 5.2.3 (RAI 9233), 9.3.4 (RAI 9194), and 10.4.6 (RAI 9117, Question 10.04.06-1).

NuScale Response:

The industry guidelines in Chapter 1, Introduction and Management Responsibilities, of the EPRI PWR Primary Water Chemistry Guidelines and EPRI PWR Secondary Water Chemistry Guidelines provide the guidance for the development of an optimized primary water chemistry program and secondary water chemistry program. The Guidelines define three elements of the water chemistry programs. The elements are categorized as "mandatory", "shall" and "recommended" specifications. The EPRI PWR Primary Water Chemistry Guidelines defines these categories as follows:

- Guideline elements designated as "mandatory" are important to primary system pressure boundary integrity or fuel-cladding integrity and should not be deviated from by any utility.
- Guideline elements designated as "shall" are important to long-term primary system pressure boundary reliability or fuel-cladding reliability but could be subject to legitimate

deviations due to plant differences and special situations.

- Guideline elements designated as “recommendations” are good or best practice that utilities should try to implement when practical.

The EPRI PWR Secondary Water Chemistry Guidelines defines these parameters as follows:

1. Guideline elements designated as “mandatory” are important to steam generator tube integrity and should not be deviated from by any utility. Steam generator tube integrity is defined as meeting the performance criteria as specified in NEI 97-06. Each utility is ultimately responsible for the operation of their plant(s) and actions taken at those plants, but must realize that it is highly unlikely that any deviations from mandatory elements would be supported by the industry.
2. Guideline elements designated as “shall” are important to long-term steam generator reliability but could be subject to legitimate deviations due to plant differences and special situations.
3. Guideline elements designated as “recommendations” are good or best practices that utilities should try to implement when practical.

The “mandatory” and “shall” elements are those elements that affect pressure boundary integrity and fuel cladding integrity for primary chemistry control. Those chemistry elements are listed as control parameters and are contained in Tables 3-3, 3-7, and 3-8 of the EPRI PWR Primary Water Chemistry Guidelines. (Lithium is also defined as a control parameter; compliance with the principles in Table 3-1, also in Chapter 3 of the EPRI Guidelines is defined as a “shall” requirement.) The “mandatory” and “shall” elements are those elements that affect steam generator tube integrity and steam generator reliability for secondary chemistry control. The EPRI PWR Secondary Water Chemistry Guidelines list control parameters for once-through steam generators in Tables 6-1, 6-2, 6-4, 6-5, 6-7, and 6-8.

All diagnostic parameters are designated as “recommendations” in the EPRI PWR Primary Water Chemistry Guidelines and EPRI PWR Secondary Water Chemistry Guidelines. Recommendations are considered good or best practices that utilities should implement when practical. Their purpose is to assist in determining the potential cause of an off-normal chemistry condition. They are listed in Tables 3-5, 3-6, and 3-9 in Chapter 3 of the EPRI PWR Primary Water Chemistry Guidelines and Table 6-1, Table 6-2, Table 6-4, Table 6-5, Table 6-7, Table 6-8, and Table 6-9.

Additional “recommendations” listed in the EPRI PWR Primary Water Chemistry Guidelines include consideration of operation in the upper portion of the 25 to 50 cc/kg hydrogen band, consideration of zinc injection into the RCS, and footnotes 2, 3, and 5 to Table 3-3 that recommend increased sampling to address potential resin ingress, changes in RCS lithium, and changes in RCS hydrogen respectively.

Additional “recommendations” listed in the EPRI PWR Secondary Water Chemistry Guidelines address the use of nitrogen blankets, covering the steam generator tubes with layup solution,

and the use of elevated hydrazine levels if a nitrogen blanket is not used or the steam generator tubes are not covered.

The EPRI PWR Primary Water Chemistry Guidelines and EPRI PWR Secondary Water Chemistry Guidelines address deviations to the elements as follows.

> "*Deviations to mandatory and shall requirements shall be handled in accordance with the guidance in the current revision of the Steam Generator Management Program (SGMP) Administrative Procedures. Additionally, these Guidelines recommend that any exception to monitoring of a diagnostic (recommended) parameter be documented in the Strategic Water Chemistry Plan.*"

Chapter 4, Methodology for Plant Specific Optimization, of both the EPRI PWR Primary Water Chemistry Guidelines and Secondary Water Chemistry Guidelines describe the methodology to develop a Strategic Water Chemistry Plan. Included in that methodology are checklists to ensure that the requirements of the Guidelines for NEI 97-06 and NEI 03-08 are met. Completion of the checklists requires that the "mandatory", and "shall" elements in the Guidelines are implemented. Changes to the plant-specific action levels, hold values, or sample frequencies that are outside the bounds of the limits specified in the Guideline tables require that the technical justification be included in the strategic plan to document the deviation. The deviation must also be handled in accordance with the guidance in the current revision of the SGMP Administrative Procedures. Implementation of the recommended parameters is also required. Deviations are required to be identified and justified in the strategic water chemistry plan.

Chapter 6, Water Chemistry Guidelines Once-Through Steam Generators, of the EPRI PWR Secondary Water Chemistry Guidelines states the following:

> "*As discussed in Chapters 3 and 4, it is intended that plant-specific optimized strategic water chemistry plans be developed for each plant. It is recognized that steam generator designs vary significantly as do company management philosophies and economic conditions. Therefore, implementation of these guidelines requires "customization" to ensure they are specific to the needs of a given power station. However, as discussed in Chapter 1, this "customization" needs to be accomplished within the framework of meeting mandatory and "shall" requirements, which are identified in Chapter 8.*"

COL item 5.2-4 requires an applicant that references the NuScale Power Plant design certification to develop and implement a Strategic Water Chemistry Plan (mandatory requirement). The plant specific primary and secondary water chemistry program will be controlled by plant procedures that implement the "shall" and "recommended" requirements of the EPRI Guidelines. As noted in RAI 9194, sample frequencies will vary due to plant operating conditions. They may also vary due to the required plant or utility chemistry program optimization (customization) process directed by the EPRI Guidelines. In that case, any deviations to the "shall" or "recommended" elements cited in the EPRI Guidelines, including

sample frequencies, will be identified, evaluated, and documented as directed by the EPRI Guidelines.

The EPRI Guidelines state the following in Chapter 1:

> *"With respect to the advanced nuclear technology designs, the current guidelines are being evaluated for applicability to new plants. Any changes identified based on this evaluation will be issued as Interim Guidance and incorporated in the next revision to these guidelines."*

This review for the NuScale design has not been performed yet. NuScale expects that SMR design differences (i.e. lack of a SG blowdown system, lack of reactor coolant pumps) will require changes to the current EPRI guidelines to address these differences. In the event that this evaluation has not been performed at the time of a NuScale plant licensing, the SGMP administrative procedures for processing a deviation will drive the change to the EPRI Guidelines.

Based on the discussion above, no changes are proposed to the FSAR.

Impact on DCA:

There are no impacts to the DCA as a result of this response.

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

eRAI No.: 9231
Date of RAI Issue: 12/13/2017

NRC Question No.: 05.04.02.01-10

Compliance with ASME Code Section III (NB-2160, NB-3121, NC-2160, and NC-3121) requires an appropriate allowance for corrosion and other forms of degradation. Section 5.4.1.2 in Tier 2 of the FSAR states that "a lifetime degradation allowance of 0.010 inch is included in the design nominal wall thickness." Please revise the FSAR to include a discussion of the basis for the degradation allowance specified.

NuScale Response:

The lifetime degradation allowance of 0.010 inch is determined from the difference between the design nominal wall thickness (0.050 inch) and the minimum steam generator tube wall thickness (0.040 inch) as calculated in accordance with ASME Code Case N-759-2. This allowance is implemented to ensure tube integrity is maintained in light of the identified potential degradation mechanisms discussed in FSAR Section 5.4.1.6.

Impact on DCA:

There are no impacts to the DCA as a result of this response.

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eRAI No.: 9231
Date of RAI Issue: 12/13/2017

NRC Question No.: 05.04.02.01-11

General Design Criteria 30 in Appendix A of 10 CFR Part 50 requires that components in the RCPB be designed, fabricated, erected, and tested to the highest quality standards practical. Appendix B of 10 CFR Part 50, which is applicable to reactor coolant system components, provides quality assurance requirements that ensure the highest quality standards practicable. Please address the following related to SG cleaning and cleanliness controls and quality assurance requirements:

a. If the onsite cleaning and cleanliness controls of the SGs are included in the overall cleaning and cleanliness controls for the RCPB, then revise Section 5.4.1 in Tier 2 of the FSAR to include a statement to that effect and a reference to where a description of the applicable controls are located in the FSAR (e.g., Section 5.2.3.4.2 in Tier 2 of the FSAR, “Cleaning and Contamination Protection Procedures”).
b. Revise Table 1.9-2 in Tier 2 of the FSAR, “Conformance with Regulatory Guides,” to state that Regulatory Guide (RG) 1.28, “Quality Assurance Program Criteria (Design and Construction),” applies to Section 5.4.1 in Tier 2 of the FSAR. Compliance with RG 1.28 is described in the NuScale Design Specific Review Standard (DSRS) acceptance criteria for Section 5.4.2.1.

NuScale Response:

a. FSAR Section 5.4.1.5 has been revised to state that the cleaning and cleanliness controls provided in FSAR Section 5.2.3.4.2 for the reactor coolant pressure boundary are applicable to the steam generators.
b. FSAR Table 1.9-2 has been revised to state that Regulatory Guide 1.28 applies to FSAR Section 5.4.1.

Impact on DCA:

FSAR Table 1.9-2 and FSAR Section 5.4.1.5 have been revised as described in the response above and as shown in the markup provided in this response.

RAI 02.03.01-5, RAI 05.02.03-13, RAI 05.03.01-3, RAI 05.04.02.01-11, RAI 06.01.01-8, RAI 06.01.01-9, RAI 08.01-1, RAI 08.01-1S1, RAI 08.02-4, RAI 08.02-6, RAI 08.03.02-1, RAI 09.02.06-1

Table 1.9-2: Conformance with Regulatory Guides

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.3	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors	2	Not Applicable	Not Applicable	This guidance is only applicable to BWRs.	Not Applicable
1.4	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors	2	Not Applicable	Not Applicable	This RG pertains to existing reactors; RG 1.183 is specified in SRP Section 15.0.3 to be used for new reactors.	Not Applicable
1.5	Safety Guide 5 - Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors	-	Not Applicable	Not Applicable	This guidance is only applicable to BWRs.	Not Applicable
1.6	Safety Guide 6 - Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems	-	Partially Conforms	Applicable	The onsite electrical AC power systems do not contain any Class 1E distribution systems. The EDSS design conforms to the guidance for independence of standby power sources and their distribution systems.	8.3
1.7	Control of Combustible Gas Concentrations in Containment	3	Not Applicable	Not Applicable	The containment vessel design is such that its integrity does not rely on combustible gas control systems.	6.2
1.8	Qualification and Training of Personnel for Nuclear Power Plants	3	Not Applicable	Applicable	Site-specific programmatic and operational activities are the responsibility of the COL applicant.	Not Applicable
1.9	Application and Testing of Safety-Related Diesel Generators in Nuclear Power Plants	4	Not Applicable	Not Applicable	Based on reduced reliance on AC power, the design does not require or include safety-related emergency diesel generators.	8.3
1.11	Instrument Lines Penetrating the Primary Reactor Containment	1	Not Applicable	Not Applicable	No lines penetrate the NPM containment.	6.2

Table 1.9-2: Conformance with Regulatory Guides (Continued)

RG	Division Title	Rev.	Conformance Status	COL Applicability	Comments	Section
1.26	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants	4	Conforms	Applicable	The quality group classification from RG 1.26 applicable to a specific component is described throughout the FSAR.	3.2 5.2 5.4 6.2 9.1 9.3 10.3 10.4
1.27	Ultimate Heat Sink for Nuclear Power Plants (for Comment)	3	Not Applicable	Not Applicable	RG does not apply to plants that utilize a passive containment cooling system as their ultimate heat sink.	9.2.5
1.28	Quality Assurance Program Criteria (Design and Construction)	4	Conforms	Applicable	The NuScale design is based on NQA12008 and the NQA1a2009 addenda (rather than NQA11994), as endorsed in RG 1.28, Rev. 4. The design for threaded fasteners meet the cleaning criteria in RG 1.28.	3.13 4.5 5.2.3 5.4.1 6.1 7.2 ~~10.3~~ 14.2 17.1 17.5

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.

The SG piping from the FWIVs and MSIVs to the SGs, steam plenum access ports, and plenum 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 piping, steam plenum access ports, and plenum 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 piping, associated reducers and elbows, steam plenum access ports, and plenum covers and fasteners are provided in Table 5.4-3.

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.

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

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

The SG inlet flow restrictors ~~are classified as Quality Group B and~~ are designed, fabricated, constructed, tested, and inspected as ~~non-ASME Code components~~non-structural attachments.

RAI 05.04.02.01-11

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 5.2.3.4.2 for cleaning and cleanliness controls for the SGs. 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.

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NRC Question No.: 05.04.02.01-12

Provide a description of the analyses or tests that have been performed to evaluate the potential for and extent of sludge accumulation in the integral feed plena and possible adverse effects, such as blockage of flow through the tubes or concentration of corrosion products. This information is necessary to ensure the integral feed plena satisfy the requirements of 10 CFR Part 50, Appendix A, GDC 1, 4, 15, 30, and 31, as they relate to ensuring the integrity of the RCPB.

NuScale Response:

By design, the feedwater plena contains only subcooled liquid. No boiling occurs in the plena; therefore no physical mechanism exists to concentrate corrosion products in this location. No testing is necessary as the types of corrosion products that could be deposited in the NuScale SG feedwater plena are bounded by existing commercial experience.

Use of EPRI *Secondary Water Chemistry Guidelines* and selection of materials for the secondary plant (see FSAR Section 10.3.6) will maintain levels of suspended corrosion products (the only possible source of deposits in the plenum location) at low levels.

Given the flow rate of the feedwater at the inlet to each plenum, the flow will be highly turbulent. Based on the high levels of turbulence and the relatively open volume of the plenum it is not plausible that the suspended corrosion products will deposit in one highly localized area (e.g. lower part of the plenum at the entrance to the lowest column of tubes) that could impact SG performance. Dispersed deposits are expected. It is also expected that a significant fraction (at least half) of any dissolved solids will pass into the SG tubes and therefore will not deposit in the plena. Based on these considerations, the quantity that could be deposited in each feedwater plenum is too small to cause a functional issue. Removable access covers on the feedwater plena allow for detection and cleaning of any unexpected sludge deposits that may accumulate.

Impact on DCA:

There are no impacts to the DCA as a result of this response.

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

eRAI No.: 9231
Date of RAI Issue: 12/13/2017

NRC Question No.: 05.04.02.01-13

Steam Generator preservice inspection (PSI) and inservice inspection (ISI) are required by 10 CFR 50.55a and are identified in SECY-05-0197, “Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria,” dated October 28, 2005 (ADAMS Accession No. ML052770225), as operational programs to be implemented by licensees. Since SGs have distinct PSI and ISI requirements, please address the following:

a. Reference the SG PSI and ISI programs separately by adding reference to Section 5.4 in Section 13.4 in Tier 2 of the FSAR, “Operational Programs.” For example, “Inservice inspection programs (refer to Section 5.2, Section 6.6, and Section 5.4)” and “Pre-service inspection programs (refer to Section 5.2 and Section 5.4).”

b. Revise Table 1.8-2, “Combined License Information Items,” and Section 5.4.1, “Steam Generators,” in Tier 2 of the FSAR to add a COL information item that requires COL applicants referencing the NuScale design certification to prepare PSI and ISI programs for the SGs.

NuScale Response:

a. Reference to Section 5.4 has been added to Section 13.4 “Operational Programs” and Table 1.8-2, "Combined License Information Items."

b. COL Item 5.4-1 requires COL applicants referencing the NuScale Power Plant design certification to develop and implement a SG program. As stated in COL Item 5.4-1, inspections are a SG program element and so this COL Item includes the requirement for development of both preservice inspection (PSI) and inservice inspection (ISI) programs for the SGs. In addition, Technical Specification 5.5.4 provides a full discussion of the SG inspection program. Therefore, no change to the COL item is necessary.

Impact on DCA:

FSAR Table 1.8-2 and Section 13.4 have been revised as described in the response above and as shown in the markup provided in this response.

RAI 02.04.13-1, RAI 03.04.02-1, RAI 03.04.02-2, RAI 03.04.02-3, RAI 03.05.01.04-1, RAI 03.05.02-2, RAI-03.06.02-15, RAI 03.07.01-2, RAI 03.07.01-3, RAI 03.07.02-8, RAI 03.07.02-12, RAI 03.09.02-15, RAI 03.09.02-48, RAI 03.09.03-12, RAI 03.09.06-5, RAI 03.09.06-6, RAI 03.09.06-16, RAI 03.09.06-27, RAI 03.11-8, RAI 03.11-14, RAI 03.13-3, RAI 05.04.02.01-13, RAI 05.04.02.01-14, RAI 06.04-1, RAI 09.01.02-4, RAI 09.01.05-3, RAI 09.01.05-6, RAI 09.03.02-3, RAI 09.03.02-4, RAI 09.03.02-5, RAI 09.03.02-6, RAI 09.03.02-8, RAI 10.02-1, RAI 10.02-2, RAI 10.03.06-1, RAI 10.04.06-1, RAI 10.04.06-2, RAI 10.04.06-3, RAI 10.04.10-2, RAI 13.01.01-1, RAI 13.01.01-1S1, RAI 13.02.02-1, RAI 13.03-4, RAI 13.05.02.01-2, RAI 13.05.02.01-2S1, RAI 13.05.02.01-3, RAI 13.05.02.01-3S1, RAI 13.05.02.01-4, RAI 13.05.02.01-4S1, RAI 19-31

Table 1.8-2: Combined License Information Items

Item No.	Description of COL Information Item	Section
COL Item 1.1-1:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will identify the site-specific plant location.	1.1
COL Item 1.1-2:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will provide the schedules for completion of construction and commercial operation of each power module.	1.1
COL Item 1.4-1:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will identify the prime agents or contractors for the construction and operation of the nuclear power plant.	1.4
COL Item 1.7-1:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will provide site-specific diagrams and legends, as applicable.	1.7
COL Item 1.7-2:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will list additional site-specific P&IDs and legends as applicable.	1.7
COL Item 1.8-1:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will provide a list of departures from the certified design.	1.8
COL Item 1.9-1:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will review and address the conformance with regulatory criteria in effect six months before the docket date of the COL application for the site-specific portions and operational aspects of the facility design.	1.9
COL Item 1.10-1:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will evaluate the potential hazards resulting from construction activities of the new NuScale facility to the safety-related and risk significant structures, systems, and components of existing operating unit(s) and newly constructed operating unit(s) at the co-located site per 10 CFR 52.79(a)(31). The evaluation will include identification of any management and administrative controls necessary to eliminate or mitigate the consequences of potential hazards and demonstration that the limiting conditions for operation of an operating unit would not be exceeded. This COL item is not applicable for construction activities (build-out of the facility) at an individual NuScale Power Plant with operating NuScale Power Modules.	1.10
COL Item 2.0-1:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will demonstrate that site-specific characteristics are bounded by the design parameters specified in Table 2.0-1. If site-specific values are not bounded by the values in Table 2.0-1, the COL applicant will demonstrate the acceptability of the site-specific values in the appropriate sections of its combined license application.	2.0
COL Item 2.1-1:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will describe the site geographic and demographic characteristics.	2.1
COL Item 2.2-1:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will describe nearby industrial, transportation, and military facilities. The COL applicant will demonstrate that the design is acceptable for each potential accident, or provide site-specific design alternatives.	2.2
COL Item 2.3-1:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will describe the site-specific meteorological characteristics for Section 2.3.1 through Section 2.3.5, as applicable.	2.3
COL Item 2.4-1:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will investigate and describe the site-specific hydrologic characteristics for Section 2.4.1 through Section 2.4.14, as applicable.	2.4

Table 1.8-2: Combined License Information Items (Continued)

Item No.	Description of COL Information Item	Section
COL Item 13.4-1:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will provide site-specific information, including implementation schedule, for operational programs: • Inservice inspection programs (refer to Section 5.2, Section 5.4, and Section 6.6) • Inservice testing programs (refer to Section 3.9 and Section 5.2) • Environmental qualification program (refer to Section 3.11) • Pre-service inspection program (refer to Section 5.2 and Section 5.4) • Reactor vessel material surveillance program (refer to Section 5.3) • Pre-service testing program (refer to Section 3.9.6, Section 5.2, and Section 6.6) • Containment leakage rate testing program (refer to Section 6.2) • Fire protection program (refer to Section 9.5) • Process and effluent monitoring and sampling program (refer to Section 11.5) • Radiation protection program (refer to Section 12.5) • Non-licensed plant staff training program (refer to Section 13.2) • Reactor operator training program (refer to Section 13.2) • Reactor operator requalification program (refer to Section 13.2) • Emergency planning (refer to Section 13.3) • Process control program (PCP) (refer to Section 11.4) • Security (refer to Section 13.6) • Quality assurance program (refer to Section 17.5) • Maintenance rule (refer to Section 17.6) • Motor-operated valve testing (refer to Section 3.9) • Initial test program (refer to Section 14.2)	13.4
COL Item 13.5-1:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will describe the site-specific procedures that provide administrative control for activities that are important for the safe operation of the facility consistent with the guidance provided in RG 1.33, Revision 3.	13.5
COL Item 13.5-2:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will describe the site-specific procedures ~~that licensed operators perform in the control room including normal operating procedures, abnormal operating procedures, and emergency operating procedures (EOPs), and describe the classification system for these types of procedures and general format and content for each classification.~~that operators use in the main control room and locally in the plant, including normal operating procedures, abnormal operating procedures, and emergency operating procedures (EOPs). The COL applicant will describe the classification system for these procedures, and the general format and content of the different classifications.	13.5
COL Item 13.5-3:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will describe the site-specific ~~program for developing~~ maintenance and other operating procedures, ~~including the classification system, objectives, and organizational responsibility for the following types of procedures~~including how these procedures are classified, and the general format and content of the different classifications. The categories of procedures listed below should be included: • plant radiation protection procedures • emergency preparedness procedures • ~~instrument~~ calibration and test procedures • chemical-radiochemical control procedures • radioactive waste management procedures • maintenance and modification procedures • material control procedures • plant security procedures	13.5

13.4 Operational Programs

COL Item 13.4-1: A COL applicant that references the NuScale Power Plant design certification will provide site-specific information, including implementation schedule, for operational programs:

RAI 05.04.02.01-13

- Inservice inspection programs (refer to Section 5.2, Section 5.4, and Section 6.6)
- Inservice testing programs (refer to Section 3.9 and Section 5.2)
- Environmental qualification program (refer to Section 3.11)

RAI 05.04.02.01-13

- Pre-service inspection program (refer to Section 5.2 and Section 5.4)
- Reactor vessel material surveillance program (refer to Section 5.3)
- Pre-service testing program (refer to Section 3.9.6, Section 5.2, and Section 6.6)
- Containment leakage rate testing program (refer to Section 6.2)
- Fire protection program (refer to Section 9.5)
- Process and effluent monitoring and sampling program (refer to Section 11.5)
- Radiation protection program (refer to Section 12.5)
- Non-licensed plant staff training program (refer to Section 13.2)
- Reactor operator training program (refer to Section 13.2)
- Reactor operator requalification program (refer to Section 13.2)
- Emergency planning (refer to Section 13.3)
- Process control program (PCP) (refer to Section 11.4)
- Security (refer to Section 13.6)
- Quality assurance program (refer to Section 17.5)
- Maintenance rule (refer to Section 17.6)
- Motor-operated valve testing (refer to Section 3.9)
- Initial test program (refer to Section 14.2)

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Docket No. 52-048

eRAI No.: 9231
Date of RAI Issue: 12/13/2017

NRC Question No.: 05.04.02.01-14

General Design Criteria 32, 10 CFR 50.55a, 10 CFR 50.36, Appendix B in 10 CFR Part 50, and 10 CFR 50.65 include requirements to ensure that implementation of a SG program maintains the structural and leakage integrity of the SG tubes. Combined License Information Item 5.4-1 in Tier 2 of the FSAR requires that a COL applicant referencing the NuScale design certification develop and implement a SG program based on NEI 97-06, Revision 3, and applicable EPRI SG guidelines. Please revise the FSAR to address the following regarding the SG program COL item:

a. For consistency with other statements in the FSAR, COL Information Item 5.4-1 should be revised to make it clear that the SG program follows or conforms to NEI 97-06. The term "based on" implies that there may be exceptions, in which case they should be identified and justified.
b. The required program elements in COL Information Item 5.4-1 do not include maintenance of the SG secondary-side integrity (primary side for NuScale design) which is a program element in NEI 97-06. Please add an appropriate program element that meets the intent of the SG secondary-side integrity program element in NEI 97-06 to this COL item to be consistent with NEI 97-06.
c. Given that a newer revision of NEI 97-06 may be available prior to a COL applicant referencing the NuScale design certification, please remove reference to a specific revision of NEI 97-06 from COL Information Item 5.4-1. The NRC staff notes that NEI 97-06, Revision 3, and EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines," Revision 4, are listed under References in Bases B 3.4.5.

NuScale Response:

a. Based on the discussion provided in the response to eRAI 9231 - 05.04.02.01-3, no changes to the term "based on" are needed for COL Item 5.4-1.
b. Shell side integrity and accessibility assessment and steam plant corrosion product deposition assessment have been added to COL Item 5.4-1 to meet the intent of the maintenance of steam generator secondary-side integrity program element for the NuScale SGs.

c. Reference to Revision 3 of NEI 97-06 has been removed from COL Item 5.4-1 and FSAR Table 1.8-2. COL Item 5.4-1 and FSAR Table 1.8-2 now state that the SG program will be based on the latest revision of NEI 97-06, “Steam Generator Program Guidelines,” and applicable EPRI guidelines at the time of the COL application. Brackets have been added to specific revisions of NEI 97-06, “Steam Generator Program Guidelines,” and EPRI, “Pressurized Water Reactor Primary-to-Secondary Leak Guidelines,” in the References section in Bases B 3.4.5 and B 3.4.9.

Impact on DCA:

FSAR Table 1.8-2, COL Item 5.4-1 and TS Bases 3.4.5 and 3.4.9 have been revised as described in the response above and as shown in the markup provided in this response.

RAI 02.04.13-1, RAI 03.04.02-1, RAI 03.04.02-2, RAI 03.04.02-3, RAI 03.05.01.04-1, RAI 03.05.02-2, RAI-03.06.02-15, RAI 03.07.01-2, RAI 03.07.01-3, RAI 03.07.02-8, RAI 03.07.02-12, RAI 03.09.02-15, RAI 03.09.02-48, RAI 03.09.03-12, RAI 03.09.06-5, RAI 03.09.06-6, RAI 03.09.06-16, RAI 03.09.06-27, RAI 03.11-8, RAI 03.11-14, RAI 03.13-3, RAI 05.04.02.01-13, RAI 05.04.02.01-14, RAI 06.04-1, RAI 09.01.02-4, RAI 09.01.05-3, RAI 09.01.05-6, RAI 09.03.02-3, RAI 09.03.02-4, RAI 09.03.02-5, RAI 09.03.02-6, RAI 09.03.02-8, RAI 10.02-1, RAI 10.02-2, RAI 10.03.06-1, RAI 10.04.06-1, RAI 10.04.06-2, RAI 10.04.06-3, RAI 10.04.10-2, RAI 13.01.01-1, RAI 13.01.01-1S1, RAI 13.02.02-1, RAI 13.03-4, RAI 13.05.02.01-2, RAI 13.05.02.01-2S1, RAI 13.05.02.01-3, RAI 13.05.02.01-3S1, RAI 13.05.02.01-4, RAI 13.05.02.01-4S1, RAI 19-31

Table 1.8-2: Combined License Information Items

Item No.	Description of COL Information Item	Section
COL Item 1.1-1:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will identify the site-specific plant location.	1.1
COL Item 1.1-2:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will provide the schedules for completion of construction and commercial operation of each power module.	1.1
COL Item 1.4-1:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will identify the prime agents or contractors for the construction and operation of the nuclear power plant.	1.4
COL Item 1.7-1:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will provide site-specific diagrams and legends, as applicable.	1.7
COL Item 1.7-2:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will list additional site-specific P&IDs and legends as applicable.	1.7
COL Item 1.8-1:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will provide a list of departures from the certified design.	1.8
COL Item 1.9-1:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will review and address the conformance with regulatory criteria in effect six months before the docket date of the COL application for the site-specific portions and operational aspects of the facility design.	1.9
COL Item 1.10-1:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will evaluate the potential hazards resulting from construction activities of the new NuScale facility to the safety-related and risk significant structures, systems, and components of existing operating unit(s) and newly constructed operating unit(s) at the co-located site per 10 CFR 52.79(a)(31). The evaluation will include identification of any management and administrative controls necessary to eliminate or mitigate the consequences of potential hazards and demonstration that the limiting conditions for operation of an operating unit would not be exceeded. This COL item is not applicable for construction activities (build-out of the facility) at an individual NuScale Power Plant with operating NuScale Power Modules.	1.10
COL Item 2.0-1:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will demonstrate that site-specific characteristics are bounded by the design parameters specified in Table 2.0-1. If site-specific values are not bounded by the values in Table 2.0-1, the COL applicant will demonstrate the acceptability of the site-specific values in the appropriate sections of its combined license application.	2.0
COL Item 2.1-1:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will describe the site geographic and demographic characteristics.	2.1
COL Item 2.2-1:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will describe nearby industrial, transportation, and military facilities. The COL applicant will demonstrate that the design is acceptable for each potential accident, or provide site-specific design alternatives.	2.2
COL Item 2.3-1:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will describe the site-specific meteorological characteristics for Section 2.3.1 through Section 2.3.5, as applicable.	2.3
COL Item 2.4-1:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will investigate and describe the site-specific hydrologic characteristics for Section 2.4.1 through Section 2.4.14, as applicable.	2.4

Table 1.8-2: Combined License Information Items (Continued)

Item No.	Description of COL Information Item	Section
COL Item 5.2-6:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will develop site-specific preservice examination, inservice inspection, and inservice testing program plans in accordance with Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code and will establish implementation milestones. If applicable, a COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will identify the implementation milestone for the augmented inservice inspection program. The COL applicant will identify the applicable edition of the American Society of Mechanical Engineers Code utilized in the program plans consistent with the requirements of 10 CFR 50.55a.	5.2
COL Item 5.2-7:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will establish plant-specific procedures that specify operator actions for identifying, monitoring, trending, and locating reactor coolant system leakage in response to prolonged low leakage conditions that exist above normal leakage rates and below the technical specification limits. The objective of the methods of detecting and locating the reactor coolant pressure boundary leak will be to provide the operator sufficient time to take actions before the plant technical specification limits are reached.	5.2
COL Item 5.3-1:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will establish measures to control the onsite cleaning of the RPV during construction in accordance with RG 1.28.	5.3
COL Item 5.3-2:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will develop operating procedures to ensure that transients will not be more severe than those for which the reactor design adequacy had been demonstrated.	5.3
COL Item 5.3-3	A COL applicant that references the NuScale Power Plant design certification will describe their reactor vessel material surveillance program consistent with NUREG 0800, Section 5.3.1.	5.3
COL Item 5.4-1:	A COL ~~Applicant~~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 the latest revision of NEI 97-06, "Steam Generator Program Guidelines," ~~Revision 3~~ and applicable EPRI steam generator guidelines at the time of the COL application. The elements of the program will include: assessment of degradation, tube inspection requirements, tube integrity assessment, tube plugging, primary-to-secondary leakage monitoring, shell side integrity and accessibility assessment, steam plant corrosion product deposition assessment, primary and secondary side water chemistry control, foreign material exclusion, loose parts management, contractor oversight, self-assessment, and reporting.	5.4
COL Item 6.2-1:	A COL ~~Applicant~~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.	6.2
COL Item 6.3-1:	A COL ~~Applicant~~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:	6.3
COL Item 6.4-1:	A COL ~~Applicant~~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."	6.4
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.~~Not used.	6.4
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.~~Not used.	6.4
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.~~Not used.	6.4
COL Item 6.4-5:	A COL ~~Applicant~~applicant that references the NuScale Power Plant design certification will specify testing and inspection requirements for the CRHS, including CRE integrity testing.	6.4

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.

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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 the latest revision of NEI 97-06, "Steam Generator Program Guidelines," ~~Revision 3~~ and applicable EPRI steam generator guidelines at the time of the COL application. The elements of the program will include: assessment of degradation, tube inspection requirements, tube integrity assessment, tube plugging, primary-to-secondary leakage monitoring, shell side integrity and accessibility assessment, steam plant corrosion product deposition assessment, primary and secondary side water chemistry control, foreign material exclusion, loose parts management, contractor oversight, self-assessment, and reporting.

BASES

REFERENCES

1. 10 CFR 50, Appendix A GDC 30.
2. Regulatory Guide 1.45, May 2008.
3. FSAR Chapter 15, “Accident Analysis.”
4. NEI-97-06, Rev. [3], “Steam Generator Program Guidelines.”
5. EPRI, “Pressurized Water Reactor Primary-to-Secondary Leak Guidelines,” Rev. [4].

BASES

REFERENCES

1. NEI 97-06, Rev. [3], “Steam Generator Program Guidelines.”

2. 10 CFR 50 Appendix A, GDC 19.

3. 10 CFR 100.

4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.

5. Draft Regulatory Guide 1.121, “Basis for Plugging Degraded Steam Generator Tubes,” August 1976.

6. EPRI, “Pressurized Water Reactor Steam Generator Examination Guidelines,” Rev. [4].