RAIO-1217-57696

December 18, 2017

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. 267 (eRAI No. 9193) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 267 (eRAI No. 9193)," dated October 20, 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. 9193:

- 05.02.03-1
- 05.02.03-2
- 05.02.03-3
- 05.02.03-4
- 05.02.03-5
- 05.02.03-6
- 05.02.03-7
- 05.02.03-8
- 05.02.03-9
- 05.02.03-10
- 05.02.03-11
- 05.02.03-12
- 05.02.03-13
- 05.02.03-14
- 05.02.03-15
- 05.02.03-16

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
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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9193

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9193

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

eRAI No.: 9193
Date of RAI Issue: 10/19/2017

NRC Question No.: 05.02.03-1

Regulatory basis: Title 10 Code of Federal Regulations (CFR) Part 50, Appendix A, General Design Criteria (GDC) 1 and GDC 30 require that components in the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practicable. 10 CFR Part 50, Appendix A, GDC 4 requires Structures, Systems, and Components (SSCs) to be designed and fabricated to accommodate the effects of environmental conditions during normal, off normal, and accident conditions.

In DCD Tier 2, FSAR Table 5.2-4, "Reactor Coolant Pressure Boundary Component Materials Including Reactor Vessel, Attachments, and Appurtenances," the "RCS Check Valves" entry references DCD Tier 2, FSAR Subsection 5.4.2.5.

Subsection 5.4.2.5 provides a generic statement that the RCS check valve materials, including weld materials, conform to ASME Boiler and Pressure Vessel Code, Section III, Subsection NB and that the surfaces of pressure retaining parts of the valves, including weld filler materials and bolting material are corrosion resistant such as stainless steel or a nickel-based alloy.

This information is insufficient for the staff to make a finding on GDC 1 and GDC 4 since not all types and grades of stainless steel and/or nickel-based alloys would be acceptable for such applications.

Revise the DCD Tier 2 information to provide acceptable material specifications, types, and grades which may be used for all RCS components including the RCS check valves.

NuScale Response:

The title of Tier 2, FSAR Table 6.1-3 has been modified to indicate that this table of pressure-retaining materials is also applicable to reactor coolant pressure boundary (RCPB) valves.

Tier 2, FSAR Table 5.2-4, and applicable portions of FSAR Section 5.4.2.5 have been updated

to add a note to direct the user to see Table 6.1-3 for a list of materials that may be used for RCPB valves.

Impact on DCA:

Table 5.2-4, Section 5.4.2.5 and Table 6.1-3 have been revised as described in the response above and as shown in the markup provided in this response.

pressure continues to increase followed by opening of the RVV. The analysis results indicate the peak pressure remains below the brittle fracture stress limit.

RAI 03.09.06-6

COL Item 5.2-2: A COL applicant that references the NuScale Power Plant design certification will provide a certified Overpressure Protection Report in compliance with American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Subarticles NB-7200 and NC-7200 to demonstrate the reactor coolant pressure boundary and secondary system are designed with adequate overpressure protection features, including low temperature overpressure protection features.

5.2.2.3 Piping and Instrumentation Diagram

Figure 5.1-2 provides the RCS piping and instrument diagram and illustrates the design configuration of the RSVs and RVVs showing the number and location with respect to the RPV.

5.2.2.4 Equipment and Component Description

5.2.2.4.1 Reactor Safety Valves

The RSVs are redundant, safety-related, Seismic Category I, Quality Group A components designed to maintain pressure below 110 percent of design pressure, 2310 psia. Each valve is sized to provide 100 percent of the required relief capacity.

RAI 05.02.03-1

The RSV design information is provided in Table 5.2-2, and materials of the RSV components are provided in Table 6.1-3~~5.2-3~~. The RSV pressure boundary design life is for a service life of 60 years. Each RSV is a pilot operated relief valve designed in accordance with the requirements of ASME BPVC, Section III, Subsubarticle NB-7520. The valve is designed to allow RCS pressure routed to a chamber located above the pilot disc where it expands the pilot valve bellows and seats the pilot disc. Relief pressure is determined by the spring pre-load of the pilot valve and the main valve closing spring pressure. A simplified diagram of the RSV and associated pilot valve is provided in Figure 5.2-1 and Figure 5.2-2. RSVs are designed for 300 cycles over the design life. Environmental qualification information associated with the RSVs is provided in Section 3.11.

8914 RAI 05.02.01.01-6

~~The RSV blowdown is set > 5 percent below set pressure and is a deviation from the ASME BPVC, Section III, Subparagraph NB-7522.6. The basis for this deviation is to minimize successive number of lifts.~~

5.2.2.4.2 Reactor Vent Valves

Three RVVs are safety-related, Seismic Category I, Quality Group A components constructed in accordance with ASME BPVC, Section III, Subsection NB, each designed with sufficient relief capacity to prevent brittle fracture stress limits being

RAI 05.02.03-1

Table 5.2-3: Not Used~~Reactor Safety Valve Materials~~

~~Component~~	~~Specification~~	~~Alloy Designation (Grade, Class, or Type)~~
~~Valve main body~~	~~SA-182~~	~~Grade F316~~
~~Valve end caps~~	~~SA-479~~ ~~OR~~ ~~SA-182~~	~~Type 316 or Grade F316~~
~~Pilot body, bonnet, disc and seat~~	~~SA-479~~	~~Type 304~~ ~~Type 304L~~
~~RSV cap bolts~~	~~SA-453~~	~~A286~~
~~Main disk, piston, and rings~~	~~SA-564~~	~~Type 17-4; Grade 630; Condition H1100~~
~~Main and pilot springs~~	~~AMS 5699~~	~~Alloy X-750~~
~~Welding material~~	~~SFA 5.4~~ ~~SFA 5.9~~	~~E308, E308L, E309, E309L, E316, E316L~~ ~~ER308, ER308L, ER309, ER309L, ER316, ER316L~~

RAI 05.02.03-1, RAI 05.02.03-9, RAI 05.03.01-3, 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)
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)
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-163	Alloy 690 (UNS N06690)
SG tube 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	SA-312	Grade TP304/304L
Piping Supports	SA-479	Type 304/304L
Piping Reducers and Elbows	SA-182	Grade F304/F304L
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[1]	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		
Refer to ~~Section 5.4.2.5~~Table 6.1-3		

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]
RCS Excess Flow Check Valves		
Refer to Table 6.1-3		
RCS Injection and Discharge Isolation Valves		
Refer to Table 6.1-~~2~~3		
Reactor Safety Valves		
Refer to Table 6.1-3~~5.2-3~~		
RCS Piping Supports		
RCS Piping Supports (short, long, tube)	SA-479	Type 304/304L
Pressurizer Support Anchor and Support Plate 180 Degree Piping Supports 90 Degree Piping Supports	SA-240	Type 304/304L
CRDM Pressure Retaining Components		
Latch housing Rod travel housing Rod travel housing Plug	SA-965	Grade F304LN

Note:
(1) Carbon Content of unstabilized Type 3XX weld filler materials is restricted to 0.03% maximum.

maximum expected RCS discharge flow during a plant startup. The RPV high-point degasification line excess flow check valve is designed to be capable of stopping fully developed flow of 150 percent in the forward direction within one second during accident conditions.

5.4.2.3 Performance Evaluation

Section 3.9, Section 3.12, and Section 5.2 provide information regarding the RCS piping criteria, methods, and materials, and include the design, fabrication, and operational provisions to control those factors that contribute to stress-corrosion cracking. The RCS piping supports the functional aspects of the chemical volume and control system (CVCS) as summarized in Section 9.3.4.

The RCS piping reverse flow and excess flow check valves provide a backup to the containment isolation valves in the event of a line break outside containment and both containment isolation valves fail to isolate the line. The RCS piping reverse flow and excess flow check valves do not provide an operational safety function but they do form a portion of the RCPB along with the RCS piping. Analysis demonstrates that flow induced vibration is either not predicted to occur or the effects are shown to be acceptable for the design life of the RCS piping and check valves.

5.4.2.4 Tests and Inspections

Preservice and ISI requirements associated with ASME Class 1 components, which include the RCS piping and associated check valves, are summarized in Section 5.2. No socket welds are used for RCS piping including piping of NPS 2 or less.

The reverse flow check valves and excess flow check valves are included in the augmented inservice testing and stroke tested in accordance with ASME Operational and Maintenance Code (OM Code) OM-2012, Division 1, Paragraph ISTC-3522. The testing is performed every 96 months on a staggered test basis between the reverse flow and excess flow check valves during refueling conditions.

5.4.2.5 Reactor Coolant System Piping and Check Valve Materials

Descriptions of the RCPB and materials associated with the RCS piping are provided in Section 5.2.

RAI 05.02.03-1

The RCS check valves are classified as Quality Group A and are designed, fabricated, constructed, tested, and inspected as Class 1 in accordance with ASME BPVC and the applicable conditions promulgated in 10 CFR 50.55a(b). The check valve materials, including weld materials, conform to fabrication, construction, and testing requirements of ASME BPVC, Section III, Subsection NB. The materials selected for fabrication conform to the applicable material specifications provided in ASME BPVC, Section II and meet the requirements of ASME BPVC, Section III, Article NB-2000. The check valves are constructed of materials with a proven history in light water reactor environments. Surfaces of pressure retaining parts of the valves, including weld filler materials and bolting material, are corrosion resistant materials such as stainless steel

or nickel-based alloy. Materials used for the RCS check valves and associated weld filler metals are provided in Table 6.1-3.~~The RCS check valve weld filler metals are in accordance, as applicable, with SFA-5.4 and SFA-5.9, of BPVC, Section II, Part C.~~

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.

5.4.3 Decay Heat Removal System

5.4.3.1 Design Basis

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

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

The DHRS design ensures the RCS average temperature is below 420 degrees F within 36 hours after an initiating event without challenging the RCPB or uncovering the core. An RCS average temperature of 420 degrees F was chosen based on the safe shutdown temperature proposed by EPRI for passive plant designs in the EPRI Advanced Light Water Reactor Utility Requirements Document (Reference 5.4-3) and determined to be acceptable by the Nuclear Regulatory Commission as documented in SECY-94-084. Per SECY-94-084 and NUREG-1242, Volume 3, Part 2, transition of a passive plant from safe shutdown conditions to cold shutdown conditions may be reached using nonsafety-related systems. The nonsafety-related containment flood and drain system is used to flood the containment to allow passive long term decay heat removal via convection and conduction to the reactor pool via the RCS, RPV shell, flooded containment, and CNV shell.

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

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

GDC 1, 2, and 4 - The DHRS is classified Quality Group B and designed as Class 2 in accordance with Section III of the ASME BPVC and is designed, fabricated, and tested to the highest quality standards in accordance with Quality Assurance Program described in Chapter 17. The DHRS is designed to withstand the effects of natural phenomena without loss of capability to perform its safety function. The DHRS is designed to accommodate the effects of, and be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. The design of the Reactor Building structure, NPM operating bays, and location of the NPM within the operating bays provides protection from possible sources of external or

RAI 05.02.03-1, RAI 06.01.01-1, RAI 06.01.01-2, RAI 06.01.01-3, RAI 06.01.01-4

Table 6.1-3: Pressure Retaining Materials for RCPB and ESF Valves

Bodies	SA-182 (Note 1)	Grade F304, F304L, F304LN, F316, F316L, F316LN
	SA-351 (Note 2)	Grade CF3, CF3A, CF3M, CF8, CF8A, CF8M
	SA-479 (Note 1)	Type 304, 304L, 304LN, 316, 316L, 316LN
Bonnets	SA-182 (Note 1)	Grade F304, F304L, F304LN, F316, F316L, F316LN
	SA-240 (Note 1)	Type 304, 304L, 304LN, 316, 316L, 316LN
	SA-351 (Note 2)	Grade CF3, CF3A, CF3M, CF8, CF8A, CF8M
	SA-479 (Note 1)	Type 304, 304L, 304LN, 316, 316L, 316LN
Discs	SA-182 (Note 1)	Grade F304, F304L, F304LN, F316, F316L, F316LN
	SA-351 (Note 2)	Grade CF3, CF3A, CF3M, CF8, CF8A, CF8M
	SA-479 (Note 1)	Type 304, 304L, 304LN, 316, 316L, 316LN, XM-19
	SA-564	Type 630 Condition H1100 or H1150
	SB-637	UNS N07718
Stems	SA-479 (Note 1)	Type 304, 304L, 304LN, 316, 316L, 316LN, XM-19
	SA-564	Type 630 Condition H1100 or H1150
	SB-637	UNS N07718
Pressure Retaining Studs, Bolts, and Screws	SA-193 (Note 3)	Grade B8, B8A, B8M, B8MA, B8R, B8RA, B8S, B8SA
	SA-453	Grade 660 Class A or B
	SA-564	Type 630 Condition H1100
	SB-637 (Note 4)	UNS N07718
Pressure Retaining Nuts	SA-193 (Note 3)	Grade B8, B8A, B8M, B8MA, B8R, B8RA, B8S, B8SA
	SA-194	Grade 8, 8A, 8M, 8MA, 8R, 8RA, 8S, 8SA
	SA-453 (Note 4)	Grade 660 Class A or B
	SA-564	Type 630 Condition H1100
	SB-637 (Note 4)	UNS N07718
Filler Metals for Pressure Retaining Welds	SFA-5.4 (Note 5)	E308, E308L, E309, E309L, E316, E316L
	SFA-5.9 (Note 5)	ER308, ER308L, E309, E309L, ER316, ER316L

1. Carbon is limited to 0.03% maximum for unstablized Type 3XX that are welded or exposed to temperature range between 800°F and 1500°F subsequent to final solution anneal.
2. Carbon is limited to 0.03% maximum. Delta ferrite is limited to 20% maximum, except to 14% maximum for CF3M and CF8M.
3. B8A, B8MA, B8R, and B8RA can only be used for Class 1 valves.
4. Solution treatment temperature range prior to precipitation hardening treatment is restricted to 1800°F to 1850°F.
5. Carbon is limited to 0.03% maximum. The ferrite number is in the range of 5FN to 20FN except Type 316 and Type 316L are in the range of 5FN to 16FN.

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

eRAI No.: 9193
Date of RAI Issue: 10/19/2017

NRC Question No.: 05.02.03-2

10 CFR Part 50, Appendix A, GDC 4 requires SSCs to be designed and fabricated to accommodate the effects of environmental conditions during normal, off normal, and accident conditions. 10 CFR Part 52.6 requires that a standard design certification submitted for approval under 10 CFR Part 52, "shall be complete and accurate in all material respects." As described below the staff finds that portions of the NuScale application have conflicting or unclear statements. The conflicting or unclear statements prevent the staff from reaching a conclusion if NuScale design meets GDC 4.

In DCD Tier 2, FSAR Table 5.2-4, "Reactor Coolant Pressure Boundary Component Materials Including Reactor Vessel, Attachments, and Appurtenances," weld filler materials E/ER308, E/ER309, and E/ER316 are permitted.

In DCD Tier 2, FSAR Section 5.2.3.4.1, "Prevention of Sensitization and Intergranular Corrosion of Austenitic Stainless Steel," the applicant states that avoidance of intergranular attack in austenitic stainless steels is accomplished, in part, by use of austenitic stainless steels with a carbon content not exceeding 0.03 wt%.

In ASME/ASTM SFA-5.4 the maximum carbon content for the weld filler materials are: 0.08 percent for E/ER308, 0.15 percent for E/ER309, and 0.08 percent for E/ER316.

Therefore, there appears to be a conflict between the material specifications contained in FSAR Section 5.2.3.4.1 and the materials specified in Table 5.2-4. The staff also notes that welding L-grade base metal with normal grade weld filler metal may result in carbon migration during welding. This practice would not be consistent with the staff guidance in Regulatory Guide (RG) 1.44 "Control of the Processing and Use of Stainless Steel."

Revise the DCD Tier 2 information to address this apparent inconsistency in weld filler material specifications.

NuScale Response:

NuScale previously modified FSAR Section 5.2.3.4.1 to remove the statement that avoidance of intergranular attack in austenitic stainless steels is accomplished, in part, by use of austenitic stainless steels with a carbon content not exceeding 0.03 wt%. NuScale replaced that statement to indicate that compliance with Regulatory Guide 1.44, Rev. 1 would be performed for unstabilized Type 3XX austenitic stainless steels to avoid sensitization and intergranular attacks. This change was previously provided to the NRC in a letter dated August 3, 2017 (ML17215A977).

NuScale previously modified FSAR Table 5.2-4 to insert a footnote to Table 5.2-4 to limit SFA-5.4 and 5.9 unstablized Type 3XX filler metals to 0.03% max carbon. This change was previously provided to the NRC in a letter dated November 20, 2017, entitled "Subject: NuScale Power, LLC Response to the NRC Request for Additional Information No. 233 (eRAI No. 9111) on the Nuscale Design Certification Application." The response to RAI 06.01.01-3 in this letter provided the requested change to Table 5.2-4.

These previously performed changes eliminate any apparent inconsistency. No further changes are required.

Impact on DCA:

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

NuScale Power™

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

eRAI No.: 9193
Date of RAI Issue: 10/19/2017

NRC Question No.: 05.02.03-3

10 CFR Part 52.6 requires that a standard design certification submitted for approval under 10 CFR Part 52 “shall be complete and accurate in all material respects.” As described below the staff finds that portions of the NuScale application have conflicting or unclear statements.

The feed plenum is not mentioned in DCD Tier 2, FSAR Table 5.2-4, “Reactor Coolant Pressure Boundary Component Materials Including Reactor Vessel, Attachments, and Appurtenances.” DCD Tier 2 FSAR Figure 5.4-5 indicates that the feed plenum tube sheet and the feed plenum forging should retain Reactor Coolant System (RCS) pressure. The feedwater plenum forging and the feedwater plenum tub sheet are not described in DCD Tier 2, FSAR Table 5.2-4.

Revise DCD Tier 2 information, as appropriate, to ensure that the feed plenum, as well as other RCS pressure boundary components are included in Table 5.2-4.

NuScale Response:

NuScale FSAR Table 5.2-4 identifies the Feed Plenum Access Ports, as SA-508 Grade 3 Class 2 components. The feedwater plenum tubesheets are integral to the Feed Plenum Access Ports and thus is part of the SA-508 Grade 3 Class 2 forging. No changes to the FSAR are required.

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.: 9193
Date of RAI Issue: 10/19/2017

NRC Question No.: 05.02.03-4

Regulatory basis: 10 CFR Part 52.6 requires that a standard design certification submitted for approval under 10 CFR Part 52 "shall be complete and accurate in all material respects." As described below the staff finds that portions of the NuScale application have conflicting or unclear statements. 10 CFR Part 50, Appendix A, GDC 4 requires SSCs to be designed and fabricated to accommodate the effects of environmental conditions during normal, off normal, and accident conditions. NuScale utilizes cladding to ensure that the ferritic steel base metal does not degrade during operations.

In DCD Tier 2, FSAR Section 5.2.3, the applicant provides a discussion on the cladding for ferritic steel components. The applicant specifies the materials used for cladding, the preheat requirements for cladding, non-destructive evaluation (NDE) requirements for base metal prior to cladding application, preheat controls and qualification requirements to prevent under-bead cracking, and post weld heat treatment requirements. The discussion on preheat, NDE of base metal, and post-weld heat treatment for cladding is focused on ensuring that the process does not impact the pressure retaining and structural integrity functions of the base metal. The applicant also states that welding of ferritic steels will be qualified in accordance with American Society of Mechanical Engineers (ASME) Code Section IX but the discussion on welding ferritic materials is focused on the joining of components. The FSAR does not provide clear statements on which requirements apply to the cladding.

Revise the DCD Tier 2 information to provide:

1. A statement that the qualification of cladding welding and welders will meet the requirements of ASME Code Section IX. If ASME Code Section IX will not be used, the applicant should describe how the qualification of cladding welding and welders will be conducted.
2. The code of construction for the cladding welding process or process controls that ensure that production welding is consistent with the qualifications process.

NuScale Response:

In response to the two requests, NuScale has revised FSAR Section 5.2.3.3.2 to insert the following statement:

> *Weld overlay cladding is conducted utilizing procedures qualified in accordance with the applicable requirements of ASME BPVC, Section III, Subarticle NB-4300 and Section IX.*

This change provides the qualification of cladding welding and welders and also provides the code of construction for the cladding welding process.

Impact on DCA:

FSAR Section 5.2.3.3.2 has been revised as described in the response above and as shown in the markup provided in this response.

5.2.3.2.2 Compatibility of Construction Materials with Reactor Coolant

The RCPB ferritic low alloy and carbon steels used in pressure retaining applications have austenitic stainless steel cladding on surfaces that are exposed to the reactor coolant. Low alloy steel forgings have an average grain size of five or finer in accordance with American Society for Testing and Materials standards. The cladding of ferritic type base material receives a post-weld heat treatment as required by ASME BPVC, Section III, Subsubarticle NB-4622.

RAI 05.02.03-4

The inside and outside surfaces of the RPV low alloy steels 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. Weld overlay cladding is conducted utilizing procedures qualified in accordance with the applicable requirements of ASME BPVC, Section III, Subarticle NB-4300 and Section IX.

The through-holes in the baffle plate of the low alloy steel integral steam plenum for the twelve incore instrumentation guide tubes are sleeved with Alloy 690 inserts. Larger through-holes in the baffle plate, for the CRD shafts and holes for pressurizer insurge and outsurge flow, approximately 2 in diameter or greater, are cladded with austenitic stainless steel.

The use of cobalt based alloys is minimized and limits are established to minimize cobalt intrusion into the reactor coolant. Cobalt based alloys are used for hard surfacing and wear resistant parts in the CRDMs and the core support locking assembly. Refer to Section 4.5 for additional details regarding the materials of the CRDMs and reactor vessel internals. Low cobalt or cobalt-free alloys may be used for hardfacing and wear resistant parts in contact with the reactor coolant if their wear and corrosion resistance are qualified by testing.

5.2.3.3 Fabrication and Processing of Ferritic Materials

5.2.3.3.1 Fracture Toughness

The fracture toughness properties of the RCPB components comply with the requirements of 10 CFR 50, Appendix G, "Fracture toughness requirements," and ASME BPVC, Section III, Subarticle NB-2300. Discussion of the fracture toughness requirements of the RPV materials are provided in Section 5.3.

5.2.3.3.2 Welding Control - Ferritic Materials

Welding of ferritic materials used for components of the RCPB is conducted utilizing procedures qualified in accordance with the applicable requirements of ASME BPVC, Section III, Subarticle NB-4300 and Section IX.

Stainless steel corrosion resistant weld overlay cladding of low alloy steel components conforms to the requirements of RG 1.43, Revision 1. Controls to limit

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

eRAI No.: 9193
Date of RAI Issue: 10/19/2017

NRC Question No.: 05.02.03-5

10 CFR Part 52.6 requires that a standard design certification submitted for approval under 10 CFR Part 52 "shall be complete and accurate in all material respects." As described below the staff finds that portions of the NuScale application have conflicting or unclear statements.

In DCD Tier 2, FSAR Section 5.2.3.2.2, "Compatibility of Construction Materials with Reactor Coolant" the applicant states the following:

> The RCPB ferritic low alloy and carbon steels used in pressure retaining applications have austenitic stainless steel cladding on surfaces that are exposed to the reactor coolant.

The staff is not clear regarding the scope of this statement. The cladding appears to be limited to pressure-retaining materials but some components may be part of the ASME Code Section III, Class 1 system and not perform a pressure-retaining function.
Additionally, based upon the information in DCD Tier 2 FSAR Table 5.2-4 "Reactor Coolant Pressure Boundary Component Materials Including Reactor Vessel, Attachments, and Appurtenances" there are no carbon steel components in the RCPB.

If the applicant intended the statement in DCD Tier 2 FSAR Section 5.2.3.2.2 to encompass all ferritic materials part of the reactor coolant pressure boundary, revise the aforementioned state this more clearly. An example of this would be:

> The RCPB ferritic low alloy ~~and carbon steels~~ used in **<u>the RCPB</u>** have austenitic…

Additionally, DCD Tier 2 SAR Section 5.2.3.1 states that Alloy 52/152 will be used as cladding. During the June 27th public meeting the applicant clarified that the pressurizer baffle plate in the area of the integral steam plenum is clad with Alloy 52/152. The statement in DCD Tier 2 FSAR Section 5.2.3.2.2 appears to only allow austenitic stainless steel cladding.

Update the FSAR as necessary to ensure that in FSAR Section 5.2.3.2.2 clearly states which components and surfaces will be clad.

NuScale Response:

FSAR Section 5.2.3.2.2 has been revised to provide clarifying information. "Carbon steel" has been removed as suggested by the NRC. In addition, other changes have been made to clarify the use of Ni-CR-Fe cladding deposited with Alloy 52/152.

Impact on DCA:

FSAR Section 5.2.3.2.2 has been revised as described in the response above and as shown in the markup provided in this response.

5.2.3.2.2 Compatibility of Construction Materials with Reactor Coolant

RAI 05.02.03-5, RAI 05.02.03-6

The RCPB ferritic low alloy ~~and carbon~~ steels used in pressure retaining applications have austenitic stainless steel cladding or Ni-Cr-Fe cladding on surfaces that are exposed to the reactor coolant. Low alloy steel forgings have an average grain size of five or finer in accordance with American Society for Testing and Materials standards. The cladding of ferritic type base material receives a post-weld heat treatment as required by ASME BPVC, Section III, Subsubarticle NB-4622.

RAI 05.02.03-4, RAI 05.02.03-5, RAI 05.02.03-6

The inside and outside surfaces of the RPV low alloy steels including RPV attachments and appurtenances in contact with reactor coolant, secondary water or pool water are clad with austenitic stainless steel or Ni-Cr-Fe. The austenitic stainless steel 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 austenitic stainless steel cladding on the outside surfaces is deposited with at least one layer of Type 309L. The Ni-Cr-Fe cladding is deposited with Alloy 52/152. Weld overlay cladding is conducted utilizing procedures qualified in accordance with the applicable requirements of ASME BPVC, Section III, Subarticle NB-4300 and Section IX.

The through-holes in the baffle plate of the low alloy steel integral steam plenum for the twelve incore instrumentation guide tubes are sleeved with Alloy 690 inserts. Larger through-holes in the baffle plate, for the CRD shafts and holes for pressurizer insurge and outsurge flow, approximately 2 in diameter or greater, are cladded with austenitic stainless steel.

The use of cobalt based alloys is minimized and limits are established to minimize cobalt intrusion into the reactor coolant. Cobalt based alloys are used for hard surfacing and wear resistant parts in the CRDMs and the core support locking assembly. Refer to Section 4.5 for additional details regarding the materials of the CRDMs and reactor vessel internals. Low cobalt or cobalt-free alloys may be used for hardfacing and wear resistant parts in contact with the reactor coolant if their wear and corrosion resistance are qualified by testing.

5.2.3.3 Fabrication and Processing of Ferritic Materials

5.2.3.3.1 Fracture Toughness

The fracture toughness properties of the RCPB components comply with the requirements of 10 CFR 50, Appendix G, "Fracture toughness requirements," and ASME BPVC, Section III, Subarticle NB-2300. Discussion of the fracture toughness requirements of the RPV materials are provided in Section 5.3.

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

eRAI No.: 9193
Date of RAI Issue: 10/19/2017

NRC Question No.: 05.02.03-6

10 CFR Part 52.6 requires that a standard design certification submitted for approval under 10 CFR Part 52 "shall be complete and accurate in all material respects." As described below the staff finds that portions of the NuScale application have conflicting or unclear statements.

In DCD Tier 2, FSAR Section 5.2.3.2.2, "Compatibility of Construction Materials with Reactor Coolant," the applicant states the following:

> The inside and outside surfaces of the RPV low alloy steels 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 staff seeks clarification on the scope of this statement.

Confirm that the aforementioned statement refers to all of the low-alloy steel components in DCD Tier 2, FSAR Table 5.2-4 "Reactor Coolant Pressure Boundary Component Materials Including Reactor Vessel, Attachments, and Appurtenances" or identify which components would not be clad.

The staff believes that the aforementioned the statement is intended to state that all low-alloy steel components in the RCS which make contact with the reactor coolant and portions of the RCS that are in contact with the containment environment are clad with austenitic stainless steel. Confirm that the staff interpretation is correct or provide additional clarification. Revise the DCD Tier 2 information as appropriate to clarify the intent of the statement in Section 5.2.3.2.2.

NuScale Response:

NuScale response to RAI 9193, 05.02.03-5 clarified the cladding scope to include low alloy steel

reactor pressure vessel (RPV) attachments and appurtenances that are in contact with reactor coolant, secondary water, or pool water.

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.: 9193
Date of RAI Issue: 10/19/2017

NRC Question No.: 05.02.03-7

Regulatory basis: 10 CFR Part 50, Appendix A, GDC 1 and GDC 30 require that components in the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practicable. As discussed in SRP 5.2.3, the imposition of appropriate welding controls are needed to satisfy the quality requirements of GDC 1 and 30. Limited accessibility conditions present a challenge in meeting the high quality components requirements of GDC1 and GDC 30 because ASME Code Section IX welding procedure qualification and welder qualification does not require an exact mockup of the operation to be performed. Experience has shown that a welder may not be able to produce acceptable welds if the accessibility to the weld area is restricted.

The applicant provides welding controls for ferritic steels (DCD Tier 2, FSAR Section 5.2.3.3.2) and austenitic stainless steels (DCD Tier 2, FSAR Section 5.2.3.4.4).

DCD Tier 2, FSAR Table 5.2-4 states that Alloy 690 is used to fabricate components in the reactor coolant pressure boundary. Alloy 690 (a nickel alloy) does not fall into either of the categories discussed in FSAR Section 5.2.3.3.2 or 5.2.3.4.4. The staff was unable to locate similar requirements for the fabrication of nickel alloys.

1) Nickel alloy components that are part of the RCPB should meet the welding requirements of ASME Code, Section III, Article NB- 4000

2) The qualification of welding of Nickel alloy components should meet ASME Code Section IX.

3) The applicant should impose controls for Nickel alloy components consistent with the ferritic and austenitic stainless steel controls described in RG 1.71 for welding in areas of limited accessibility.

Revise the DCD Tier 2 information to include appropriate requirements for the welding and welding qualification of nickel alloys.

NuScale Response:

In accordance with the NRC request, NuScale has updated FSAR Section 5.2.3.5 to provide the requirements for welding, and control of welding of nickel-based alloys in the RCPB. See the FSAR change provided below for details.

Impact on DCA:

FSAR Section 5.2.3.5 has been revised as described in the response above and as shown in the markup provided in this response.

- limiting the sulfur content of nickel-based alloy base metal in contact with RCS primary fluid to maximum 0.02 wt%.

RAI 05.02.03-7

The nickel-based alloy materials that are used in the RCPB, including weld materials, conform to the fabrication, construction, and testing requirements of ASME BPVC Section III. Material specifications comply with ASME BPVC Section II Parts B and C. Welding of nickel-base alloys in the RCPB is performed in accordance with procedures qualified to the requirements of ASME BPVC, Section III, Subarticle NB-4300 and Section IX. Control of welding variables, as well as examination and testing during procedure qualification and production welding, are performed in accordance with ASME Code requirements. Welders and welding operators are qualified in accordance with ASME Section IX and RG 1.71, Revision 1.

Chemistry, mechanical properties, and thermo-mechanical processing requirements are controlled in nickel-based alloy base metal by solution annealing and by thermal treating to produce an optimum microstructure for resistance to intergranular corrosion.

EPRI Materials Reliability Program Reports MRP-111 (Reference 5.2-6) and MRP-258 (Reference 5.2-7) detail the Alloy 690, 52/52M, and 152 resistance to PWSCC. These documents conclude that Alloy 690 and its weld metals are highly corrosion resistant materials deemed acceptable for PWR applications. No signs of PWSCC of Alloy 690 materials have been observed in operating PWRs and Alloy 690 has proven resistant to PWSCC initiation in a wide variety of laboratory tests.

These EPRI reports provide a comprehensive summary of Alloy 690 stress corrosion cracking laboratory test data from simulated primary water environments which provides reasonable assurance of the high resistance to PWSCC for Alloy 690 and its weld metals.

5.2.3.6 Threaded Fasteners

Threaded fasteners used in the RPV main closure flange, PZR heater bundle closures, RCS piping flanges, and RSV flanges are nickel-based Alloy 718. Threaded fastener materials conform to the applicable requirements of ASME BPVC, Sections II and III and are selected for their compatibility with the borated water environment in the RCS and reactor pool water.

Section 3.13 provides further description of the design of threaded fasteners for the RPV and pressure retaining components including design requirements for the use of Alloy 718 for the mitigation of SCC.

5.2.4 Reactor Coolant Pressure Boundary Inservice Inspection and Testing

Preservice and inservice inspection and testing of ASME BPVC Class 1 pressure-retaining components (including vessels, pumps, valves, bolting, and supports) within the RCPB are performed in accordance with ASME BPVC, Section XI pursuant to 10 CFR 50.55a(g), including ASME BPVC Section XI mandatory appendices.

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Docket No. 52-048

eRAI No.: 9193
Date of RAI Issue: 10/19/2017

NRC Question No.: 05.02.03-8

Regulatory basis: 10 CFR Part 50, Appendix A, GDC 1 and GDC 30 require that components in the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practicable. 10 CFR Part 50, Appendix A, GDC 4 requires SSCs to be designed and fabricated to accommodate the effects of environmental conditions during normal, off normal, and accident conditions. The ASME Boiler and Pressure Vessel Code, as incorporated by reference in 10 CFR 50.55a, ensures high quality components by requiring qualified Non-Destructive Examination.

In DCD Tier 2, FSAR Section 5.2.3.4.2, "Nondestructive Examination for Austenitic Stainless Steel Tubular Products" the applicant states the following:

For Class 1 piping welds requiring an ultrasonic preservice examination, the welds meet the surface finish and marking requirements of ASME BPV Code, Section III, Subparagraph NB-4424.2 except that the surface finish has an average roughness (Ra) of 125 in or better and the surface flatness is less than 0.03125 inches for a minimum distance of 2 times the thickness of the part from the weld centerline.

Based on information contained in DCD Tier 2 FSAR Figure 3.6-1 "Piping Systems Associated with the NuScale Power Module" and DCD Tier 2 FSAR Table 5.2-6, "Reactor Pressure Vessel Inspection Elements" the staff understands that this provision may be applicable to the RCS discharge piping and other components.

DCD Tier 2, FSAR Figure 3.6-1, "Piping Systems Associated with the NuScale Power Module" shows that several ASME Code Class 1 components have a size of nominal pipe size (NPS) 2.

If the staff assumes that the piping utilizes standard thickness, the statement in FSAR Section 5.2.3.4.2 would mean that surface preparation for the preservice ultrasonic examination would only be required for 0.31 inches on either side of the weld centerline.

The staff is concerned that a surface preparation of 0.31 inches from the weld centerline would be insufficient. A small area of surface preparation may result in rocking or lift off of the ultrasonic transducer thereby challenging the ability to conduct a proper examination..

Additionally, considering that most of the NuScale components are NPS 2, the ratio of the size of the heat affected zone to the thickness of the component may be greater than would be expected on larger pipes (NPS 12 and greater). The ASME Code requires a minimum surface preparation of 6 inches or 4 inches plus twice the thickness of the component to ensure that the entirety of weld and the heat affected area is covered. In contrast, the applicant does not provide a similar "buffer zone" to ensure that the entirety of the heat affected zone is covered.

The applicant should provide a justification that the twice the component thickness from the weld centerline (generally expected to be approximately 0.31 inch) surface preparation distance is sufficient. The staff understands that the ASME BPV Code, Section III, Subparagraph NB-4424.2 minimum surface preparation of 6 inches may be excessive considering the size of some piping systems.

NuScale Response:

The additional surface preparation requirements have been removed and are now only required to meet NB-4424.2. The surface preparation length of 6 inches or 2t+4 inches will be enough to ensure a full volumetric exam can be completed without challenging the ability to conduct a proper examination.

Impact on DCA:

FSAR Section 5.2.3.4.5 has been revised as described in the response above and as shown in the markup provided in this response.

- complete removal of temporary markings prior to heating, welding, heat treating, assembly, or shipment

Controls are established to minimize the introduction of potentially harmful contaminants including chlorides, fluorides, and low melting point alloys on the surface of austenitic stainless steel components. In accordance with RG 1.44, cleaning solutions, processing equipment, degreasing agents, and other foreign materials are removed at any stage of processing prior to elevated temperature treatments. Acid pickling is avoided on stainless steel.

Use of abrasive work is minimized to avoid surface coldwork and contamination. Tools for abrasive work such as grinding, polishing, or wire brushing are not permitted to be contaminated by previous usage on carbon or low alloy steels or other non-corrosive resistant materials that could contribute to intergranular cracking or stress-corrosion cracking.

5.2.3.4.3 Compatibility of Construction Materials with External Reactor Coolant

The external surfaces of the RPV are clad with austenitic stainless steel. External surfaces of the RCPB do not contain exposed ferritic materials and are compatible with a borated water environment and resistant to general corrosion.

5.2.3.4.4 Control of Welding - Austenitic Stainless Steel

Welding is conducted utilizing procedures qualified according to the rules of ASME BPVC, Sections III, Subarticle NB-4300 and IX. Control of welding variables, as well as examination and testing during procedure qualification and production welding, is performed in accordance with ASME Code requirements.

Welders and welding operators are qualified in accordance with ASME Section IX and RG 1.71, Revision 1.

5.2.3.4.5 Nondestructive Examination for Austenitic Stainless Steel Tubular Products

RAI 05.02.03-8

Nondestructive examinations performed on austenitic stainless steel tubular products to detect unacceptable defects comply with ASME BPVC, Section III, Subsubarticles NB-2550 through NB-2570, and Section XI examination requirements. For Class 1 piping welds requiring an ultrasonic preservice examination, the welds meet the surface finish and marking requirements of ASME BPV Code, Section III, Subparagraph NB-4424.2~~except that the surface finish has an average roughness (Ra) of 125 in or better and the surface flatness is less than 0.03125 inches for a minimum distance of 2 times the thickness of the part from the weld centerline~~.

5.2.3.5 Prevention of Primary Water Stress-Corrosion Cracking for Nickel-Base Alloys

Nickel-base alloy components in the RCS are protected from PWSCC by:

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Docket No. 52-048

eRAI No.: 9193
Date of RAI Issue: 10/19/2017

NRC Question No.: 05.02.03-9

10 CFR Part 52.6 requires that a standard design certification submitted for approval under 10 CFR Part 52 "shall be complete and accurate in all material respects." As described below the staff finds that portions of the NuScale application does not conform to the design drawings.

In FSAR Table 5.2-4, "Reactor Coolant Pressure Boundary Component Materials Including Reactor Vessel, Attachments, and Appurtenances," the applicant states that the core barrel guides will meet the SA-193 specification.

In NuScale drawing NP12-01-A011-M-SA-2639-501, Rev. 2, "Lower RPV Section," the bill of material and detail B-B show that core barrel guide is made from a different product form and uses a different material specification.

The applicant should confirm the material specification for the core barrel guides and revise the FSAR accordingly.

NuScale Response:

The Core Barrel Guide materials identified in FSAR Table 5.2-4 were incorrect and have been corrected as shown in the markup provided below.

Impact on DCA:

FSAR Table 5.2-4 has been revised as described in the response above and as shown in the markup provided in this response.

RAI 05.02.03-9, RAI 05.03.01-3, 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)
Reactor Vessel		
Lower RPV section flange shell RPV bottom head Core support blocks	SA-508	Grade 3, Class 1
RPV top head PZR Shell 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~~ 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)

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Docket No. 52-048

eRAI No.: 9193
Date of RAI Issue: 10/19/2017

NRC Question No.: 05.02.03-10

GDC 1 states: Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function.

In the public meeting on June 20, 2017, the staff informed the applicant that verbatim compliance with the ASME Code alone may not be adequate and sufficient to meet the quality requirements in GDC 1. Specifically, parts of the ASME Code are written with an underlying assumption that small diameter piping and components may be less safety significant than larger piping and components. The staff informed the applicant that additional controls and requirements may be necessary for small diameter piping and components.

Following the public meeting the staff reviewed the Design Specifications for the NuScale Power Module. The staff found the Design Specifications informative but ultimately lacking a description on how the adequacy and sufficiency of the ASME Code requirements for small diameter piping and components were considered and documented for the NuScale design.

The staff provides the following examples from ASME Code Section III, NB-2000 which illustrate the issue:

1. In NuScale drawing NP12-01-A013-M-PY-4035-S10, Rev. 0, “Containment System Piping Layout Drawing,” the RPV High Point Degasification Line is detailed. Additional details on the High Point Degasification connection to the RPV are shown in drawing NP12-01-A011-M-SA-2645-S03, Rev. 2, “RPV Head Section.”

   These drawings provide information on the inlet and outlet piping connections (both NPS 2) for the “RPV High Point Degasification Line – Dual Body Isolation Valve.” Considering this information, the staff believes that verbatim compliance with ASME Code, Section III subsubparagraph NB-2510(a) would allow for a containment isolation valve with a safety-related function to be exempt from a non-destructive volumetric examination

In a similar situation, NuScale drawing NP12-01-A013-M-PY-4035-S03, Rev. 0, “Containment System Piping Layout Drawing,” depicts the inlet and outlet piping connections (both NPS 4) for the “Feedwater Isolation Valve W/ Actuator W/ Check Valve.”

Considering this information, the staff believes that verbatim compliance with ASME Code, Section III subsubparagraph NB-2510(b) would allow for a containment isolation valve with a safety-related function to have a surface examination (PT or MT) and a volumetric examination limited to the area around the safe end welds (i.e., the majority of the valve body would be exempt from volumetric non-destructive examination).

2. In DCD Tier 2, FSAR Table 6.1-1, “Materials Specifications for ESF Components,” the applicant states that the containment isolation valves will be manufactured from “austenitic stainless steel or a Ni-Cr-Fe alloy.”

   Without a material specification the staff cannot determine which requirements in ASME Code, Section III NB-2500 would apply. If the RPV High Point Degasification line containment isolation valve was made from cast austenitic stainless steel, verbatim compliance with ASME Code Section III subsubparagraph NB-2573.5(b) would allow the containment isolation valve to be repaired by a material organization without a subsequent volumetric examination (though, subsubparagraph NB2573.7(c) would still apply).

In order to determine if the NuScale design meets GDC 1, the applicant must provide a justification that verbatim compliance with the ASME Code as conditioned in 10 CFR 50.55a meets GDC 1. This justification should identify all relevant ASME Code requirements based upon small diameter piping/component size and identify whether these requirements support finding that the NuScale design meets GDC 1. The applicant should justify that strict application of ASME Code requirements, including reduced requirements for small diameter piping/component sizes, provides adequate assurance that SSCs important to safety have been designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. This justification should address the impact of reduced quality activities on safety functions to be performed for small diameter components, including consideration of increased uncertainty due to the effects of: less robust non-destructive examinations to determine the material condition, fewer periodic inspections, and lower quality assurance requirements.

NuScale Response:

NuScale will follow the ASME BPVC verbatim as required by 10 CFR 50 to ensure compliance with the GDC requirements specified in 10 CFR 50 Appendix A. Compliance with the ASME BPVC, 10 CFR 50.55a and 10 CFR 50 Appendix A assures a high degree of quality and integrity for the structures, systems and components designed to these standards. A more detailed response to this RAI was included in eRAI 9109, 06.06-1. See the response to RAI 06.06-1 for additional information, including the basis for the NuScale position.

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.: 9193
Date of RAI Issue: 10/19/2017

NRC Question No.: 05.02.03-11

Regulatory basis: 10 CFR Part 50, Appendix A, GDC 1 and GDC 30 require that components in the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practicable. 10 CFR Part 50, Appendix A, GDC 4 requires SSCs to be designed and fabricated to accommodate the effects of environmental conditions during normal, off normal, and accident conditions.

In DCD Tier 2, FSAR Table 5.2-4, "Reactor Coolant Pressure Boundary Component Materials Including Reactor Vessel, Attachments, and Appurtenances," the "Low alloy steel weld filler material" entry states that the materials will be "weld filler classifications compatible with low alloy steel base metal."

This information is insufficient for the staff to make a conclusion on GDC 1. Revise the DCD Tier 2 information to include the specific, acceptable types and grades of filler metal which may be used for the RCPB low-alloy steel components.

NuScale Response:

FSAR Table 5.2-4 identifies the SFA specifications permitted for low alloy steel weld filler metals, and states that weld filler metal classifications are compatible with the low alloy steel base metal. These weld metals are compatible with the base metal mechanical requirements and meet applicable ASME BPVC Section II Part C, Section III NB, and Section IX requirements. The specific filler metal grades within the permitted SFA specifications will be selected by Reactor Pressure Vessel (RPV) fabricators based on their equipment and fabrication experience.

The level of detail currently provided in Table 5.2-4 is consistent with the information provided by other Design Certification Applicants. For example, no specific low alloy steel weld filler metal grades were identified in the FSARs for the USEPR, Revision 6, Table 5.3-2 for the RPV or for the AP1000, Revision 19, Table 5.2-1 for the RCPB.

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.: 9193
Date of RAI Issue: 10/19/2017

NRC Question No.: 05.02.03-12

10 CFR Part 50, Appendix A, GDC 4 requires SSCs to be designed and fabricated to accommodate the effects of environmental conditions during normal, off normal, and accident conditions. 10 CFR Part 52.6 requires that a standard design certification submitted for approval under 10 CFR Part 52, "shall be complete and accurate in all material respects."

In DCD Tier 2, FSAR Table 5.2-4, "Reactor Coolant Pressure Boundary Component Materials Including Reactor Vessel, Attachments, and Appurtenances," the "Vessel alignment pins" entry states that the materials will be "Type 304/304L." Confirm that it is the intent of this designation to indicate that the material will be dual certified (meeting the minimum and maximum chemistry requirements and mechanical properties of both specifications).

Revise the DCD Tier 2 information to clarify that the materials noted above will be dual certified to meet both Type 304 and Type 304L requirements.

NuScale Response:

A note has been added to Tier 2 FSAR Table 5.2-4 to indicate that the term Type or Grade 304/304L refers to dual certified stainless steel. Type or Grade 304/304L dual certified material has a maximum carbon content that is the same as Type or Grade 304L. Type or Grade 304/304L dual certified material has the same mechanical properties as Type or Grade 304. See markup of Table 5.2-4 provided below.

Impact on DCA:

FSAR Table 5.2-4 has been revised as described in the response above and as shown in the markup provided in this response.

RAI 05.02.03-1, RAI 05.02.03-9, RAI 05.02.03-12, RAI 05.03.01-3, 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-163	Alloy 690 (UNS N06690)
SG tube 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	SA-312	Grade TP304/304L
Piping Supports	SA-479	Type 304/304L
Piping Reducers and Elbows	SA-182	Grade F304/F304L
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		
Refer to ~~Section 5.4.2.5~~Table 6.1-3		

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]
RCS Excess Flow Check Valves		
Refer to Table 6.1-3		
RCS Injection and Discharge Isolation Valves		
Refer to Table 6.1-~~2~~3		
Reactor Safety Valves		
Refer to Table 6.1-3~~5.2-3~~		
RCS Piping Supports		
RCS Piping Supports (short, long, tube)	SA-479	Type 304/304L
Pressurizer Support Anchor and Support Plate 180 Degree Piping Supports 90 Degree Piping Supports	SA-240	Type 304/304L
CRDM Pressure Retaining Components		
Latch housing Rod travel housing Rod travel housing Plug	SA-965	Grade F304LN

Note:

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

(2) Carbon Content of unstabilized Type 3XX weld filler materials is restricted to 0.03% maximum.

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

eRAI No.: 9193
Date of RAI Issue: 10/19/2017

NRC Question No.: 05.02.03-13

Regulatory basis: 10 CFR Part 50, Appendix A, GDC 30 requires that components in the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. 10 CFR Part 50, Appendix B provides quality assurance requirements that ensure the highest quality standards practicable which is applicable to reactor coolant system components.

In DCD Tier 2, FSAR Section 5.2.3.4.2, "Cleaning and Contamination Protection Procedures," the applicant states the following:

> Cleaning of austenitic stainless steel components complies with ASME NQA-1 requirements (Reference 5.2-5). The fina; cleanness of the RCPB internal surfaces meets the requirements for "Class B" of Subpart 2.1.
>
> Handling, storage, and shipping of austenitic stainless steel components comply with ASME NQA-1-2008, Part I, Requirement 13. Packaging, shipment, handling, and storage of RCPB components meet the applicable requirements of ASME NQA-1a-2009, Part II, Subpart 2.2 (Reference 5.2-5).

1) The applicant does not provide a statement whether procedures will be in compliance the staff endorsement of ASME NQA-1 (RG 1.28, "Quality Assurance Program Criteria Design and Construction").

   DCD Tier 2, FSAR Table 1.9-2, "Conformance with Regulatory Guides" shows that RG 1.28 is omitted from being applicable to Section 5.2.3 but is applicable to other components with similar material concerns (threaded fasteners, control rod drive mechanisms, engineered safety features, the steam and feedwater system, etc). Compliance with RG 1.28 or RG 1.37 (the withdrawn standard that was subsumed by RG 1.28) is described in the SRP acceptance criteria for Section 5.2.3.

   Revise the DCD to state that RG 1.28 will apply to FSAR Section 5.2.3.

2) The aforementioned statements are subtopics in DCD Tier 2, Section 5.2.3.4 "Fabrication and Processing of Austenitic Stainless Steels." As shown in ASME Code Section III-NCA Table NCA-7100-2, NQA-1 applies to the design, construction, installation, and testing of all ASME Code Section III, Division 1 systems. The requirements in NQA-1 are not limited to austenitic stainless steel components as is evident in NQA-1 Subpart 2.1 which contains requirements for "corrosion-resistant alloys" and "carbon and low-alloy steels." The discussion of NQA-1 Subparts 2.1 and 2.2 in FSAR Section 5.2.3 is too narrowly focused on austenitic stainless steel components and does not include other materials for which cleaning, handling, and shipping requirements exist (e.g., nickel alloys and low-alloy steels).

   Revise the DCD Tier 2 information to state that ASME NQA-1 Subpart 2.1 and Subpart 2.2 requirements will be applied to all materials in the RCS.

NuScale Response:

Part 1) Response:

In accordance with the NRCs request, NuScale has modified FSAR Table 1.9-2 to indicate that RG 1.28 applies to FSAR Section 5.2.3.

Part 2) Response:

To clarify the scope of Subpart 2.1 and Subpart 2.2, NuScale has revised FSAR section 5.2.3.4.2 to indicate the applicability of ASME NQA-1 cleaning, handling, storage and shipment to reactor coolant pressure boundary (RCPB) components rather than austenitic stainless steel components. This change modifies these paragraphs to be consistent with RCPB requirements.

Impact on DCA:

FSAR Table 1.9-2 and Section 5.2.3.4.2 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 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 6.1 7.2 ~~10.3~~ 14.2 17.1 17.5

5.2.3.4.2 Cleaning and Contamination Protection Procedures

RAI 05.02.03-13

Cleaning of ~~austenitic stainless steel~~RCPB components complies with ASME NQA-1 requirements (Reference 5.2-5). The final cleanness of the RCPB internal surfaces meets the requirements for "Class B" of Subpart 2.1. The final cleanness of the RCPB external surfaces meets the requirements for "Class C" of Subpart 2.1.

RAI 05.02.03-13

Handling, storage, and shipping of ~~austenitic stainless steel~~ RCPB components comply with ASME NQA-1-2008, Part I, Requirement 13. Packaging, shipment, handling, and storage of RCPB components meet the applicable requirements of ASME NQA-1a-2009, Part II, Subpart 2.2 (Reference 5.2-5).

Austenitic stainless steel materials used in the fabrication, installation, and testing of nuclear steam supply components and systems are handled, protected, stored, and cleaned according to recognized and accepted methods that are designed to minimize contamination which could lead to stress corrosion cracking.

Procedures provide cleanliness controls during the various phases of manufacture and installation including final flushing. The suppliers implement a written cleanliness control plan prior to and during manufacturing and assembly of components and continues until components are sealed for shipment. The cleanliness control plan includes specific provisions for:

- maintenance of cleanliness
- controls to prevent foreign material from being introduced into the hardware
- water purity control
- controls to prevent detrimental material from contacting hardware
- support system cleanliness and inspection
- use of temporary plugs or seals to prevent entry of foreign material and objects and, as practical, prevent mechanical damage
- use of stickers or other devices identifying cleanliness control requirements, affixed to temporary plugs and seals in such a manner that removal of the plug or seal cannot be accomplished without breaking the sticker
- detection and removal of foreign objects
- maintenance of cleanliness immediately prior to and during welding, brazing, and heat treating
- tools and loose parts accountability
- complete removal of temporary markings prior to heating, welding, heat treating, assembly, or shipment

Controls are established to minimize the introduction of potentially harmful contaminants including chlorides, fluorides, and low melting point alloys on the

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

eRAI No.: 9193
Date of RAI Issue: 10/19/2017

NRC Question No.: 05.02.03-14

Regulatory basis: 10 CFR Part 50, Appendix A, GDC 30 requires that components in the reactor coolant pressure boundary shall be designed to allow for the detection and identifying the location of leakage. 10 CFR Part 50, Appendix A, GDC 4 requires SSCs to be designed and fabricated to accommodate the effects of environmental conditions during normal, off normal, and accident conditions.

In DCD Tier 2, FSAR Table 1.8-2, the applicant provides the following COL Information Item:

Item No.	Description
5.2-5	A COL applicant that references the NuScale Power Plant design certification will develop and implement a Boric Acid Control Program that includes: inspection elements to ensure the integrity of the reactor coolant pressure boundary components for subsequent service, the type of visual or other nondestructive inspections to be performed, and the required inspection frequency.

Considering that the annulus between the RPV and the containment vessel is filled with water during refueling operations and that some of the RCS piping is submerged at all times, the staff is concerned that any boric acid deposits would go into solution prior to detection.

Explain how the design of the NuScale reactor will allow for the implementation of an effective Boric Acid Control Program.

NuScale Response:

The Boric Acid Control Program (BACP) manages the effects of aging caused by loss of material, loss of mechanical closure integrity, and corrosion of connector surfaces. This program applies to structures, systems, and components containing, or exposed to, borated water, including the external surfaces of piping, valves, tanks, and bolting made of carbon steel or other locations where boric acid may drip. The program relies on implementing the

recommendations of Generic Letter 88-05, “Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR plants,” dated March 17, 1988.

In Generic Letter 88-05, the NRC identified that the most effective way to prevent boric acid corrosion is to minimize reactor coolant leakages. This can be achieved by frequent monitoring of the locations where potential leakages could occur and repairing the leaky components as soon as possible. Review of the locations where leakages have occurred in the past indicates that the most likely locations are (1) valves; (2) flanged connections in steam generator manways, reactor head closure, etc.; (3) primary coolant pumps where leakages occur at cover to-casing connections as a result of defective gaskets; and (4) defective welds.

In Bulletin 2001-01 the NRC identified that vessel head penetration (VHP) nozzles are one of the primary locations for weld leakage. NRC regulations in 10 CFR 50.55a state that ASME Class 1 components (which include VHP nozzles) must meet the requirements of Section XI of the ASME Boiler and Pressure Vessel Code. Table IWB-2500-1 of Section XI of the ASME Code provides examination requirements for VHP nozzles and references IWB-3522 for acceptance standards. IWB-3522.1(c) and (d) specify that conditions requiring correction include the detection of leakage from insulated components and discoloration or accumulated residues on the surfaces of components, insulation, or floor areas which may reveal evidence of borated water leakage, with leakage defined as through-wall leakage that penetrates the pressure retaining membrane. Therefore, 10 CFR 50.55a, through its reference to the ASME Code, does not permit through-wall cracking of VHP nozzles.

For through-wall leakage identified by visual examinations in accordance with the ASME Code, acceptance standards for the identified degradation are provided in IWB-3142. Specifically, supplemental examination (by surface or volumetric examination), corrective measures or repairs, analytical evaluation, and replacement provide methods for determining the acceptability of degraded components.

In RIS 2003-13 the NRC described typical industry methods for detection of reactor coolant system (RCS) leakage. The NRC observed that detection of reactor coolant leakage during power operation depends upon inventory balance calculations, generally performed once per day, as supplemented by information from monitoring of humidity, radiation, and sump level. NUREG/CR-6582, “Assessment of Pressurized Water Reactor Primary System Leaks,” indicates that due to large RCS volumes and instrument inaccuracies, this approach is not sufficiently sensitive to detect small leaks. Based on the staff reviews and audits, licensees generally do not take action to locate leaks until inventory balance calculations indicate an increase of between 0.1 to 0.2 gallons per minute (gpm) over the previous level of identified plus unidentified RCS leakage. At this leakage rate, licensees typically would promptly inspect charging and letdown systems outside containment but, if leakage is not found outside containment, may not take action for several days to inspect inside containment. Further, there is no uniform industry guidance on this issue. Industry procedures to address this issue are weak, with actions typically taken on a case-by-case basis.

In the NuScale Power module (NPM), emergency core cooling system valves, reactor safety valves, as well as, flanged covers for steam generator and feedwater plenums are located inside the containment vessel (CNV). The NPM CNV is held at vacuum conditions during normal operations and leakage from the steam generator and feedwater plenum covers would leak into the CNV and be monitored by the containment evacuation system (CES). There are no primary coolant pumps in the NPM design, so there is no credible leakage mechanism for this location. Finally, defective welds in the reactor pressure vessel (RPV), RCS piping or the CNV would all be detectable by the CES. Excessive leakage would require a shutdown of the reactor in accordance with Technical Specification (TS) 3.4.5. TS 3.4.5 does not permit pressure boundary leakage and limits unidentified leakage to less than or equal to 0.5 gpm.

The NuScale CES provides the following functions applicable to this discussion:

- Establishes and maintains a vacuum in the CNV during NPM operation by removing non-condensable gases from the CNV.
- Measures CNV pressure during NPM operation via pressure sensors on the CES vacuum pump suction line to monitor leakage into the CNV from all sources.
- Removes water vapor from the CNV during NPM operation and provides a method to condense, collect, and sample the water removed from the CNV prior to the water being discharged.
- Quantifies the amount of water vapor removed from the CNV during NPM operation to monitor leakage into the CNV from all sources

The CES does not depend on a once per day RCS inventory calculation to determine if leakage is occurring. The CES provides a constant monitor of containment leakage. The CES continuously monitors for leakage by observing changes in containment pressure and by directly measuring removed liquid condensate. This leak detection design is more accurate than infrequent walkdowns of a large space with significantly more potential leak locations that are typically covered in insulation. Due to the nature of operating the containment at a vacuum, any observed leakage is captured from water vapor and a precise leak rate can be determined. See Tier 2 FSAR Section 5.2.5 for further discussion concerning the CES and methods for detecting RCS leakage.

The NPM containment surfaces are all stainless steel, and there is no insulation or coatings internal to the containment vessel that would disguise where leakage is occurring. RCS piping inside containment is never fully submerged. The likely CNV locations that would be submerged during refueling are the RPV and CNV flanges, the reactor recirculation valves, feedwater plenums, and the steam generator thermal relief valves. Other components and welds are above the CNV water line during refueling and would, if a leak occurs, show boric acid deposits.

NuScale is following the ASME BPVC for the design and planned inspection and repair of the NPM. Any RCS leak location will be inspected and repaired per BPVC code requirements.

Based on the above the NPM CES leakage detection capability is very sensitive, therefore, any leakage which does occur is critically examined to determine its source and leak rate. NuScale has concluded that the NPM design can effectively implement a BACP.

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.: 9193
Date of RAI Issue: 10/19/2017

NRC Question No.: 05.02.03-15

Regulatory basis: 10 CFR Part 50, Appendix A, GDC 4 requires SSCs to be designed and fabricated to accommodate the effects of environmental conditions during normal, off normal, and accident conditions. 10 CFR Part 50, Appendix A, GDC 14 requires reactor coolant pressure boundary SSCs to be designed, fabricated, erected, and tested as to have a low probability of rapidly propagating failure and gross rupture.

DCD Tier 2, FSAR Table 5.2-4, "Reactor Coolant Pressure Boundary Component Materials Including Reactor Vessel, Attachments, and Appurtenances" contains references to other sections in the FSAR which describe materials in the reactor coolant pressure boundary.

The RCS Check Valves (DCD Tier 2, FSAR Section 5.4.2.5), the Reactor Vent Valves (DCD Tier 2, FSAR Table 6.1-1 "Materials Specifications for ESF Components), the Reactor Recirculation Valves (DCD Tier 2, FSAR Table 6.1-1), and the Containment Isolation Valves (DCD Tier 2, FSAR Table 6.1-2) are specified as being made from an "austenitic stainless steel or Ni-Cr-Fe alloy."

This language permits the use of cast austenitic stainless steel for components part of the RCPB. Operating experience has shown that cast austenitic stainless steel material can become embrittled when exposed to high irradiation environments or elevated temperatures

Supplement the DCD with controls that ensure cast austenitic stainless steel materials retain sufficient toughness.

1. Describe any cast austenitic materials expected to experience irradiation embrittlement.
2. Describe controls to ensure that cast stainless steels will not experience thermal embrittlement which could cause sudden failure of the RCPB components. Staff guidance can be found in the Grimes letter (ADAMS Accession ML003717179).

NuScale Response:

NuScale does not use cast austenitic stainless steel for reactor coolant pressure boundary (RCPB) locations where irradiation embrittlement can be a concern. As indicated in Table 6.1-3, SA-351 Grades CF3, CF3A, CF3M, CF8, CF8A, and CF8M are among the permitted RCPB materials for valves. In addition, as indicated in Table 6.1-3, the delta ferrite is limited to 20% maximum for Grades CF3, CF3A, CF8, and CF8A, and to 14% maximum for Grades CF3M and CF8M. The limits are consistent with the screening criteria in the Grimes letter.

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.: 9193
Date of RAI Issue: 10/19/2017

NRC Question No.: 05.02.03-16

Regulatory Basis: Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," General Design Criteria (GDC) 1 requires that structures, systems, and components (SSCs) important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. GDC 4 requires that SSCs important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs). GDC 14 requires that the reactor coolant pressure boundary (RCPB) shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. GDC 30 requires that components which are part of the RCPB shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

GDC 31 requires that the RCPB shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. 10 CFR Part 50, Appendix G specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor pressure boundary to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime.

DCD Tier 2, FSAR, Section 3.9.4.2, "Applicable Control Rod Drive System Design Specifications," states that the control rod drive mechanism (CRDM) is part of the RCPB, and that the CRDM bolting is designed in accordance with American Society of Mechanical Engineers Code (ASME Code), Section III, as addressed in Section 3.13.

Since CRDM bolting is part of the RCPB, the staff expects that would be discussed in DCD Tier

2, FSAR, Section 5.2.3 and Table 5.2-4, “Reactor Coolant Pressure Boundary Component Materials Including Reactor Vessel, Attachments, and Appurtenances.” However, DCD Tier 2, FSAR, Table 5.2-4 under the “CRDM Pressure Retaining Components” header does not list any threaded fastener components that are part of the RCPB portion of the CRDMs. DCD Tier 2, FSAR, Section 5.2.3.6 also does not discuss any threaded fastener components that are part of the RCPB portion of the CRDMs.

Confirm that the CRDM bolting is part of the RCPB and revise DCD Tier 2, FSAR, Table 5.2-4 and Section 5.2.3.6 to include the pressure-retaining threaded fasteners that are part of the RCPB portion of the CRDM, or revise DCD Tier 2, FSAR, Section 3.9.4.2 to clarify that there are no pressure-retaining threaded fasteners that are part of the RCPB portion of the CRDMs.

NuScale Response:

No bolting or fasteners are used in portions of the NuScale Control Rod Drive Mechanism (CRDM) design that comprise the reactor coolant pressure boundary (RCPB). The rod travel housing is threaded into the latch housing and the rod travel housing plug is threaded into the rod travel housing. The threaded structures of the CRDM are shown in Figure 4.6-4.

The CRDM latch housing, rod travel housing, and rod housing plug provide the RCPB components of the CRDM. The top end of the latch housing is joined to the rod travel housing by means of a 5.55-4 ACME threaded joint and a toroidal seal weld. The rod housing plug is connected to the top of the rod travel housing by a 2.50-4 ACME threaded joint and seal weld. The threaded joints provide for removal of CRDM internal components with straight lifts and no rigging transfers. The toroidal seal welds provide the primary pressure boundary.

The CRDM components that interface with the RCPB are specified in Table 5.2-4. NuScale has revised FSAR section 3.9.4.2 to clarify the CRDM design and to remove reference to CRDM bolting in the RCPB.

Impact on DCA:

FSAR Section 3.9.4.2 has been revised as described in the response above and as shown in the markup provided in this response.

RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-6, RAI 03.09.04-7, RAI 03.09.04-9

The main control of the stepping cycle is the voltage profile that is imposed on the three drive coils (SG, MG, and lift). The maximum allowed duration for each one way step (either up or down) is 1.5 seconds. This is derived by dividing the 0.375 inch step by the maximum required velocity of 15 in/min.

3.9.4.2 Applicable Control Rod Drive System Design Specifications

RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-6, RAI 03.09.04-7, RAI 03.09.04-9

The design, fabrication, construction, examination, testing, inspection, and documentation of the RCPB pressure boundary parts of the CRDS are in accordance with the requirements of ASME BPVC, ~~Code~~2013 Edition, Section III (Reference 3.9-1), Division I, Subsection NB. Classification of the pressure retaining portions of the CRDS is addressed in Section 3.2.2.

RAI 03.09.04-1, RAI 03.09.04-2, RAI 03.09.04-4, RAI 03.09.04-5, RAI 03.09.04-6, RAI 03.09.04-7, RAI 03.09.04-9

The design, fabrication, examination, testing, inspection and documentation for the CRDM coil heat exchangers, cooling tubes and cooling water connectors are in accordance with the requirements of ASME BPVC, 2013 Edition, Section III (Reference 3.9-1), Division 1, Subsection NC. These components are conservatively classified Quality Group B to minimize the potential for fluid leakage inside containment, as discussed by Section 4.5.1. The pressure retaining components of the CRDS are designed, fabricated, constructed, and tested in accordance with ASME BPVC, 2013 Edition, Section III Division 1 and are consistent with the requirements of 10 CFR 50.55a.

The pressure boundary materials are in accordance with the requirements of ASME BPVC, Section II. These pressure boundary materials are described in Section 5.2.3. The non-pressure boundary materials of the CRDS are described in Section 4.5.1.

RAI 05.02.03-16

The CRDM, which is considered part of the reactor coolant pressure boundary (RCPB), is designed in accordance with 10 CFR 50.55a. The pressure boundary components are designed to meet the stress limits and design and transient conditions specified in Table 3.9-6. The preservice and inservice inspection requirements of ASME Code, Section XI (Reference 3.9-2) are applicable to the CRDM. Welding is performed in accordance with the ASME BPVC Code, Section III, Division I, Subsection NB. The requirements to prevent brittle fracture presented in ASME BPVC Code, Section III, Division I, Subsection NB are also applicable to the CRDM. The CRDM threaded connections are ~~bolting is~~ designed in accordance with the ASME BPVC Code, Section III~~, as addressed in Section 3.13~~. The threaded connections in the CRDM pressure housing sections use acme threads, and canopy welds as the pressure seals. The CRDM threaded joint configurations are provided in Figure 4.6-4. Additional information on compliance with codes and code cases for the RCPB is provided in Section 5.2.1.