June 21, 2017 Docket No. 52-048

U.S. Nuclear Regulatory Commission
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**SUBJECT:** NuScale Power, LLC Submittal of Changes to Final Safety Analysis Report, Sections 5.3.1 and 5.3.2.

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

During a May 23, 2017 public teleconferance with Bruce Bavol of the NRC staff, NuScale Power, LLC (NuScale) discussed potential updates to Sections 5.3.1 and 5.3.2 of the NuScale Final Safety Analysis Report (FSAR). The Enclosure to this letter provides a mark-up of the FSAR pages incorporating revisions to FSAR Sections 5.3.1 and 5.3.2, in redline/strikeout format. NuScale will include these changes as part of a future revision to the NuScale Design Certification.

NuScale and the NRC staff also discussed NuScale technical report TR-1015-18177, Pressure and Temperature Limits Methodology, on the same teleconference. NuScale agreed to provide a schedule for completing a revision to TR-1015-18177 to address a number of issues discussed. NuScale expects to provide a revision to TR-1015-18177 and a related revision to FSAR Section 5.3.1 by October 31, 2017.

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

Please feel free to contact Jennie Wike at 541-360-0539 or at jwike@nuscalepower.com if you have any questions.

Sincerely,

Zackary W. Rad
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Enclosure: "Changes to NuScale Final Safety Analysis Report Sections 5.3.1 and 5.3.2"

**Enclosure:**

"Changes to NuScale Final Safety Analysis Report Sections 5.3.1 and 5.3.2"

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

| Item No. | Description of COL Information Item | Section |
|---|---|---|
| COL Item 5.2-6: | A COL Applicant that references the NuScale Power Plant design certification will develop site-specific preservice examination, inservice inspection, and inservice testing program plans in accordance with Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code and will establish implementation milestones. If applicable, a COL Applicant that references the NuScale Power Plant design certification will identify the implementation milestone for the augmented inservice inspection program. The COL applicant will identify the applicable edition of the American Society of Mechanical Engineers Code utilized in the program plans consistent with the requirements of 10 CFR 50.55a | 5.2 |
| COL Item 5.2-7: | A COL Applicant that references the NuScale Power Plant design certification will establish plant-specific procedures that specify operator actions for identifying, monitoring, trending, and locating reactor coolant system leakage in response to prolonged low leakage conditions that exist above normal leakage rates and below the technical specification limits. The objective of the methods of detecting and locating the reactor coolant pressure boundary leak will be to provide the operator sufficient time to take actions before the plant technical specification limits are reached. | 5.2 |
| COL Item 5.3-1: | A COL Applicant that references the NuScale Power Plant design certification will establish measures to control the onsite cleaning of the RPV during construction in accordance with RG 1.28. | 5.3 |
| COL Item 5.3-2: | A COL Applicant that references the NuScale Power Plant design certification will develop operating procedures to ensure that transients will not be more severe than those for which the reactor design adequacy had been demonstrated. | 5.3 |
| COL Item 5.3-3 | A COL applicant that references the NuScale Power Plant design certification will describe their reactor vessel material surveillance program consistent with NUREG 0800, Section 5.3.1. | 5.3 |
| COL Item 5.4-1: | A COL Applicant that references the NuScale Power Plant design certification will develop and implement a Steam Generator Program for periodic monitoring of the degradation of steam generator components to ensure that steam generator tube integrity is maintained. The Steam Generator Program will be based on NEI 97-06, "Steam Generator Program Guidelines," Revision 3 and applicable EPRI steam generator guidelines. The elements of the program will include: assessment of degradation, tube inspection requirements, tube integrity assessment, tube plugging, primary-to-secondary leakage monitoring, primary and secondary side water chemistry control, foreign material exclusion, loose parts management, contractor oversight, self-assessment, and reporting. | 5.4 |
| COL Item 6.2-1: | A COL Applicant that references the NuScale Power Plant design certification will develop a "Containment Leakage Rate Testing Program" which will identify which Option is to be implemented under 10 CFR 50, Appendix J. Option A defines a prescriptive-based testing approach whereas Option B defines a performance-based testing program. | 6.2 |
| COL Item 6.3-1: | A COL Applicant that references the NuScale Power Plant design certification will describe a containment cleanliness program that limits debris within containment. The program should contain the following elements: | 6.3 |
| COL Item 6.4-1: | A COL Applicant that references the NuScale Power Plant design certification will comply with RG 1.78 Revision 1, "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release." | 6.4 |
| COL Item 6.4-2: | A COL Applicant that references the NuScale Power Plant design certification will specify operator training and qualification in the use of self-contained portable breathing apparatus. | 6.4 |
| COL Item 6.4-3: | A COL Applicant that references the NuScale Power Plant design certification will specify the technical resources to be stored within the CRE. | 6.4 |
| COL Item 6.4-4: | A COL Applicant that references the NuScale Power Plant design certification will specify food, water, and medical supplies to be stored within the CRE. | 6.4 |
| COL Item 6.4-5: | A COL Applicant that references the NuScale Power Plant design certification will specify testing and inspection requirements for the CRHS, including CRE integrity testing. | 6.4 |
| COL Item 6.6-1: | A COL Applicant that references the NuScale Power Plant design certification will implement an Inservice InspectionTesting Program in accordance with 10 CFR 50.55a(f). | 6.6 |

The chemical composition of unstabilized austenitic stainless steel materials that are welded or exposed to sensitizing temperatures in the range of 800 to 1500 degrees F subsequent to solution annealing, including weld filler materials, have a maximum carbon content of 0.03 wt%, consistent with Regulatory Guide (RG) 1.44, Revision 1.

COL Item 5.3-1: A COL applicant that references the NuScale Power Plant design certification will establish measures to control the onsite cleaning of the RPV during construction in accordance with RG 1.28.

### 5.3.1.2 Special Processes Used for Manufacture and Fabrication of Components

Forged low alloy steel is selected for the RPV assembly shells that surround the reactor core, pressurizer, and SGs. Forgings are used to form the various required geometries with minimum amount of welding.

~~The low alloy steel RPV is clad with austenitic stainless steel to avoid corrosion from borated primary coolant water. The outside surfaces of the RPV are clad with a minimum of one layer of weld deposited austenitic stainless steel. The inside surfaces and sealing surfaces of the RPV are clad with a minimum of two layers of weld deposited austenitic stainless steel. The first layer is type 309L and the subsequent layers are type 308L. Controls are exercised to limit the occurrence of underclad cracking in low alloy steel safety-related components clad with stainless steel. These controls conform to the guidance in RG 1.43, Revision 1. All the weld cladding processes are qualified in accordance with ASME BPVC, Section III, Subsection NB-4300.~~RPV cladding is addressed in Section 5.2.3.

Measures are taken to prevent sensitization of austenitic stainless steel materials during component fabrication. Heat treatment parameters comply with ASME BPVC, Section II. The austenitic stainless steel materials are either cooled by water quenching or cooled quickly enough through the sensitization temperature range to avoid carbide formation at the grain boundaries. When means other than water quenching are used, nonsensitization of the base material are verified by corrosion testing in accordance with Practice A or E of ASTM A262 (Reference 5.3-3).

Due to necessary component welding, it is unavoidable that the heat-affected zone within the austenitic stainless steel materials will be subjected to the sensitizing temperature range, 800 to 1500 degrees F, during fabrication. Welding practices and material composition are controlled to manage the sensitization while the material is in this temperature range and unstabilized Type 3XX austenitic stainless steels and corresponding austenitic stainless steel weld filler metals have a carbon content not exceeding 0.03 wt% to prevent undue sensitization. In addition, where unstabilized Type 3XX austenitic stainless steels are subjected to sensitizing temperatures for greater than 60 minutes during a post-weld heat treatment, non-sensitization of the materials are verified by testing in accordance with ASTM A262 Practice A or E, as required by RG 1.44.

### 5.3.1.3 Special Methods for Nondestructive Examination

The RPV pressure retaining and integrally attached materials examinations meet the requirements specified in ASME BPVC Section III. The examination methods are in accordance with ASME BPVC Section V, except as modified by Section III and any additional requirements listed below.

~~All surfaces to be clad are magnetic particle or liquid penetrant examined in accordance with subsection NB-2545 or subsection NB-2546 of Section III prior to cladding.~~Non-destructive examination of the reactor coolant pressure boundary is addressed in Section 5.2.3.

Preservice examinations are in accordance with subsection NB-5280 of Section III and subsection IWB-2200 of Section XI for ASME Code Class 1 pressure boundary and attachment welds using the examination methods in Section V, except as modified by subsection NB-5111 of Section III. These preservice examinations include 100 percent of the pressure boundary welds.

For ASME Code Class 2 pressure boundary items, preservice examinations are in accordance with subsection IWC-2200 of Section XI.

~~Preservice eddy current examinations for the SG tubing is in accordance with Section XI and Electric Power Research Institute, (EPRI) guideline 1013706, "Steam Generator Management Program: Pressurized Water Reactor Steam Generator Examination Guideline" (Reference 5.3-4).~~

#### 5.3.1.4 Special Controls and Special Processes Used for Ferritic Steels and Austenitic Stainless Steels

Welding of ferritic steels used for components in the reactor vessel is conducted utilizing procedures qualified in accordance with the applicable requirements of ASME Code, Section III, NB-4300 and Section XI (Reference 5.3-5). Further information is provided in Section 5.2.3.3.

Welding of austenitic stainless steel used for components in the reactor vessel is conducted utilizing procedures qualified in accordance with the applicable requirements of ASME Code, Sections III and XI. Further information is provided in Section 5.2.3.4.

In addition, electroslag welding processes are not utilized for joining materials. Electroslag welding processes are allowed for cladding low alloy steel and comply with RG 1.34 requirements.

Tools for abrasive work are addressed in Section 4.5.2.4.

Use of cold worked austenitic stainless steel is addressed in Section 4.5.1.1.

#### 5.3.1.5 Fracture Toughness

The fracture toughness properties of the RCPB pressure-retaining materials comply with the requirements of 10 CFR 50, Appendix G, "Fracture toughness requirements," and ASME BPVC, Section III, NB-2300.

The RPV is designed against non-ductile fracture in accordance with ASME BPVC, Section III, Appendix G, ASME BPVC, Section XI, Appendix G, and 10 CFR 50, Appendix G.

The pressure-temperature limits are developed in accordance with ASME BPVC, Section XI, Appendix G, 10 CFR 50, Appendix G, and RG 1.99, Revision 2.

Reactor vessel beltline materials are evaluated to ensure a minimum end-of-life Charpy V-notch upper shelf energy value of 50 ft-lb. The initial Charpy V-notch upper shelf energy value is 75 ft-lb minimum. Table 5.3-3 provides the 1/4-T adjusted reference temperature and upper shelf energy projections that were estimated using RG 1.99 for the end-of-life neutron fluence at the 1/4-T locations.

### 5.3.1.6 Material Surveillance

The Material Surveillance program monitors changes in the fracture toughness properties. Specimens are periodically removed and tested in order to monitor changes in fracture toughness in accordance with ASTM E185-82, as required by 10 CFR 50, Appendix H.

All material used for the specimens will be taken from the actual production forging, and from a weldment made of the same weld wire heat and flux lot combination used in the production weld. The limiting base metal is expected to be the lower RPV shell forging. However, unit-specific limiting base metal will be based on actual forging chemistry and initial $RT_{NDT}$, and can be either the lower RPV shell or the RPV bottom head forging. The predicted limiting weld metal is the lower RPV shell to bottom head circumferential weld. The predicted limiting heat affected zone is the region of the predicted limiting base metal next to the weld.

In accordance with ASTM E185-82, archive materials fill at least two capsules. The archived materials are the same limiting materials used to machine the baseline and capsule specimens, except the archive materials will be maintained as full-thickness sections during plant operating life. The purpose of archive materials is to fabricate additional capsules for unforeseeable contingencies, such as to monitor the effect of a major core change. Because the preferred type and size of specimens may change with time, the archive materials are maintained as full-thickness sections instead of being machined into specimens. Therefore, the archive materials for Reactor Pressure Vessel Surveillance Program are taken from the actual production forgings, and from weldments made from the same weld wire heat and flux lot combination used in the production weld.

Table 5.3-4 lists the specimen matrix for the Material Surveillance program requirements. As shown in the table, the number of specimens meets the ASTM E185-82 (Reference 5.3-6) minimum requirements.

The NuScale reactor vessel is designed for 60 years. Therefore, for the first 40 years of the 60-year design life, the capsule withdrawal schedule complies with Table 1 of ASTM E185-82, which is based on 32 effective full-power years (EFPY). Three capsules are sufficient to cover the initial 40-year operation per E185-82. ~~For the remaining 20-year~~

~~design lifetime, a fourth capsule is included. This fourth capsule is consistent with the license renewal requirements of NUREG-1801, Revision 2, for the 20-year license renewal period after the initial 40-year license.~~ The capsule withdrawal schedule is provided in Table 5.3-5.

The capsules are inside capsule holders that are attached to the outside of the core barrel at mid-height of the core. The capsules are positioned to achieve a lead factor between 1.5 and 4.5. The four capsules are positioned approximately 90 degrees apart around the circumference of the core support assembly. Figure 5.3-2 shows the core barrel horizontal cross-section and the location of the four capsule holders and capsule elevation on the core barrel.

The neutron flux and fluence calculations are contained in NuScale Technical Report TR-0116-20781, "Fluence Calculation Methodology and Results" (Reference 5.3-7).

The transition temperature upper shelf energy changes are calculated in accordance with RG 1.99, and are shown in Table 5.3-8, Table 5.3-9, and Table 5.3-10. Section 5.3.2 provides further information.

COL Item 5.3-3: A COL applicant that references the NuScale Power Plant design certification will describe their reactor vessel material surveillance program consistent with NUREG 0800, Section 5.3.1.

### 5.3.1.7 Reactor Vessel Fasteners

The RPV closure studs, nuts, and washers use SB-637 Alloy 718, instead of low alloy steels such a SA-540 Grade B23 or B24. The selection of Alloy 718 over traditional low alloy steels is to prevent general corrosion when the bolting is submerged during the plant startup and shutdown process. Because of its resistance to general corrosion, the concerns addressed by RG 1.65, Revision 1, position 2(b) do not apply to Alloy 718. Alloy 718 is an austenitic, precipitation-hardened, nickel-based alloy permitted for bolting materials by ASME BPVC Section III (Reference 5.3-1), Subsection NB-2128.

Furthermore, because Alloy 718 is not a ferritic material, the fracture toughness requirements of NB-2333 are not required. Further information is provided in Section 3.13.

## 5.3.2 Pressure-Temperature Limits, Pressurized Thermal Shock, and Charpy Upper-Shelf Energy Data and Analyses

### Analyses

The information provided in this section describes the bases for setting operational limits on pressure and temperature for the RCPB and ensures the requirements of 10 CFR 50, Appendices G and H, and 10 CFR 50.61 are complied with throughout the 60-year life of the plant.

### 5.3.2.1 Limit Curves

Using the methodology provided in ASME BPVC Section XI, Appendix G, and the requirements in 10 CFR 50 Appendix G, a generic set of pressure-temperature limits at 57 EFPY is calculated for various conditions. The methodology also accounts for vessel embrittlement due to neutron fluence in accordance with RG 1.99. The pressure-temperature limits for normal heatup and criticality conditions, normal cooldown, and inservice leak and hydrostatic tests are provided in Figure 5.3-3, Figure 5.3-4, and Figure 5.3-5, respectively. The corresponding numerical values are listed in Table 5.3-6 and Table 5.3-7. These pressure-temperature curves meet all the pressure and temperature requirements for the RPV listed in Table 1 of 10 CFR 50, Appendix G. The RCS pressure should be maintained below the limit of the pressure-temperature limit curves to ensure protection against non-ductile failure. Acceptable pressure and temperature combinations for reactor vessel operation are below and to the right of the applicable pressure-temperature curves. These pressure-temperature curves do not include any location correction or instrument uncertainty. For the purpose of location correction, the allowable pressure in the pressure-temperature curves can be taken as the pressure at the RPV bottom. The reactor is not permitted to be critical until the pressure-temperature combinations are to the right of the criticality curve shown in Figure 5.3-3.

The $\Delta RT_{NDT}$ at the 1/4 -T adjusted reference temperature at end of life is provided in Table 5.3-3, as described in Section 5.3.1.5.

Further information on the specific methodology is provided in NuScale Technical Report TR-1015-18177, "Pressure and Temperature Limits Methodology" (Reference 5.3-8).

### 5.3.2.2 Operating Procedures

Development of plant operating procedures to ensure that the pressure-temperature limits are not exceeded is addressed in Section 13.5. These procedures will ensure compliance with the technical specifications during normal power operating conditions and anticipated transients.

COL Item 5.3-2: A COL applicant that references the NuScale Power Plant design certification will develop operating procedures to ensure that transients will not be more severe than those for which the reactor design adequacy had been demonstrated. These procedures will be based on material properties of the as-built reactor vessels.

### 5.3.2.3 Pressurized Thermal Shock

The pressurized thermal shock (PTS) screening uses the methodology in 10 CFR 50.61. The attenuated fluence below the inside diameter cladding surface and the temperature factor for $RT_{NDT}$ shift (including the nominal irradiation temperature correction factor) are calculated using the RG 1.99 equations. As shown in Table 5.3-8, the predicted $RT_{PTS}$ remains below the 10 CFR 50.61 PTS screening criteria at 32-EFPY fluence and at 57-EFPY fluence, which bounds the NuScale 60-year design life.

### 5.3.3.6 Operating Conditions

Operating conditions as they relate to the integrity of the reactor vessel are presented in Section 5.2.2 and Section 5.3.2 and the plant technical specifications.

### 5.3.3.7 Inservice Surveillance

Inservice surveillance of the RPV is described in Section 5.2.4 and Section 5.3.1.

### 5.3.3.8 Threaded Fasteners

Threaded fasteners are discussed in Section 3.13 and Section 5.3.1.

## 5.3.4 References

5.3-1 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2013 Edition, Section III, Rules for Construction of Nuclear Facility Components, New York, NY.

5.3-2 American Society of Mechanical Engineers, NQA-1-2008/1a-2009 Addenda, "Quality Assurance Requirements for Nuclear Facility Applications."

5.3-3 ASTM A262-15, "Standard Practices for Detecting Susceptibility to Intergranular Attack in Austenitic Stainless Steels," West Conshohocken, PA.

5.3-4 ~~Electric Power Research Institute, EPRI-1013706, "Steam Generator Management Program: Pressurized Water Reactor Steam Generator Examination Guidelines: Revision 8," Palo Alto, CA.~~Not Used

5.3-5 American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2013 Edition, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, New York, NY.

5.3-6 ASTM International, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," ASTM E185-82, West Conshohocken, PA.

5.3-7 NuScale Power, LLC, "Fluence Calculational Methodology and Results," TR-0116-20781, Revision 0.

5.3-8 NuScale Power, LLC, "Pressure and Temperature Limits Methodology," TR-1015-18177, Revision 0.

**Table 5.3-5: Surveillance Capsule Withdrawal Schedule**

| Sequence | ASTM E185-82[a] or NUREG-1801 Rev. 2[b] | Estimated Withdrawal |
|---|---|---|
| 1st [a] | Whichever comes first<br>• 6-EFPY<br>• Capsule fluence > 5E+18 n/cm$^2$, E > 1 MeV<br>• Highest predicted $\Delta RT_{NDT}$ > ~50°F of all encapsulated materials | 3.5 EFPY for capsule fluence to reach 5E+18 n/cm$^2$, E > 1 MeV.<br>• From Table 2-2, 57 EFPY peak RPV inside surface fluence is 1.91E+19 n/cm$^2$, E > 1 MeV<br>• (5E+18/1.91E+19)*(57 EFPY/4.3) = 3.5 EFPY |
| 2nd [a] | Whichever comes first<br>• 15 EFPY<br>• Capsule fluence > peak 32-EFPY RPV inside surface fluence | 7.4 EFPY for capsule fluence to reach peak 32-EFPY RPV inside surface fluence<br>• 32 EFPY/4.3 = 7.4 EFPY |
| 3rd [a] | Capsule fluence is between 1 and 2 times of peak 32-EFPY inside surface fluence | Between 7.4 EFPY and 14.9 EFPY<br>• 32 EFPY/4.3 = 7.4 EFPY<br>• 64 EFPY/4.3 = 14.9 EFPY |
| 4th [b] | Capsule fluence is between 1 and 2 times of peak 57-EFPY inside surface fluence | Between 13.3 EFPY and 26.5 EFPY<br>• 57 EFPY/4.3 = 13.3 EFPY<br>• 114 EFPY/4.3 = 26.5 EFPY |

Notes:

a. The withdrawal schedule for the first three capsules is in accordance with 10 CFR 50 Appendix H and ASTM E185-82 for the initial 40-year operating license period.

b. The 4th capsule is used to cover the last 20 years of a 60-year design life.~~, and its withdrawal schedule is based on NUREG-1801 Rev. 2 for the license renewal period.~~

**Figure 5.3-2: Location of Surveillance Capsules**