



January 25, 2019

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Supplemental Response to NRC Request for Additional Information No. 175 (eRAI No. 9069) on the NuScale Design Certification Application

REFERENCES: 1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 175 (eRAI No. 9069)," dated August 12, 2017
2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 175 (eRAI No.9069)," dated December 03, 2018

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

The Enclosure to this letter contains NuScale's supplemental response to the following RAI Question from NRC eRAI No. 9069:

- 03.12-5

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 Marty Bryan at 541-452-7172 or at mbryan@nuscalepower.com.

Sincerely,

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

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Enclosure 1: NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9069



Enclosure 1:

NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9069

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

eRAI No.: 9069

Date of RAI Issue: 08/12/2017

NRC Question No.: 03.12-5

For applications for light-water-cooled nuclear power plants, 10 CFR 52.47(a)(2) requires an application contain a final safety analysis report (FSAR), that includes a description and analysis of the structures, systems, and components (SSCs) of the facility. The description shall be sufficient to permit understanding of the system designs and their relationship to the safety evaluations.

For applications for light-water-cooled nuclear power plants, 10 CFR 52.47(a)(9) requires an evaluation of the standard plant design against the Standard Review Plan (SRP) revision in effect 6 months before the docket date of the application. The evaluation required shall include an identification and description of all differences in design features, analytical techniques, and procedural measures proposed for the design and those corresponding features, techniques, and measures given in the SRP acceptance criteria. Where a difference exists, the evaluation shall discuss how the proposed alternative provides an acceptable method of complying with the Commission's regulations, or portions thereof, that underlie the corresponding SRP acceptance criteria. The SRP is not a substitute for the regulations, and compliance is not a requirement.

ASME BPV Section III, mandated by 10 CFR 50.55a, requires that piping analysis considers combinations of various loadings, including deadweight, pressure, seismic, thermal expansion and transient loads. NuScale FSAR Section 3.12 states that NuScale has adapted the graded approach for piping design, which the NRC staff had proposed in the March 4, 2014 NRC white paper - Piping Level of Detail for Design Certification (ML14065A067). Accordingly, the FSAR identifies piping that has been selected for preliminary and for final as-designed pipe stress analysis for the design certification. Provide the following information.

1. Clarify in FSAR 3.12 what piping sections are included in the chemical and volume control system (CVCS) RCS discharge piping and feedwater math models for detailed final piping

design analysis. If not all the piping within the reactor building is included in the analysis model, justify the termination points.

2. Clarify in FSAR 3.12 whether preliminary pipe stress evaluation is performed for all piping within the NuScale Power Module (NPM). Provide a justification for piping that is left out of the preliminary piping evaluations.
3. FSAR Section 3.12 identifies that preliminary pipe stress evaluation is performed for decay heat removal system (DHRS) lines up to the first 6-way rigid restraint beyond the containment isolation valves. The staff does not understand the reference of the 6-way rigid restraint beyond the containment isolation valves regarding the DHRS. Discuss whether the DHRS piping system is included in its entirety in the preliminary piping analysis and provide a justification if portions of the piping system are excluded.
4. The FSAR identifies that all ASME Class 1 piping is NPS 2.
According to operating experience (see EPRI TR-111188), Failures of socket welded piping connections continue to occur frequently in U.S. nuclear power plants, resulting in degraded plant systems. Small bore piping fatigue failures in primary loop has mostly occurred at the socket weld joints (IAEA-CN-155-055).

Please clarify whether the class 1 piping is socket welded or butt welded. If socket welded connections are utilized for NuScale piping, define the socket-weld detail and discuss how failure in these socket welded connections will be prevented.

5. It is stated in FSAR 3.12 that the RCS discharge (RCS/CVCS letdown) line and a FW line have been chosen for detailed completed stress analysis. The MS line has been identified in the NRC white paper as a system in the category of the most significant piping systems for which detailed information is required for providing the staff with sufficient basis to make a safety determination. Include the detailed final as-designed piping analysis for main steam or provide a justification for not including the main steam in the completed detailed stress analysis pipe line selection.
- 6a. FSAR Section 3.12.1 indicates that the detailed piping stress analyses considered loads due to deadweight, seismic and thermal expansion, and that fatigue analysis, including environmentally assisted fatigue, has been performed for class 1 piping. These are not the

only loads that a piping system could potentially experience. Please clarify whether all applicable loads listed in Section 3.12.5.3 have been considered in pipe stress analysis or provide a justification for loads that are left out?

- 6b. NRC white paper recommendation for preliminary piping evaluations is that these evaluations need to consider deadweight, seismic, thermal, dynamic and fatigue piping analyses, as applicable. FSAR Section 3.12.1 shows that in the preliminary pipe stress evaluations only loads due to deadweight, seismic and thermal expansion have been considered. This is a departure from the NRC white paper's recommendation. Please provide a technical justification for considering only these loads for the preliminary analyses of piping that may be needed for pipe routing, pipe support selections, postulating pipe break locations, leak-before-break (LBB) analyses and pipe whip and jet impingement protection.
- 6c. Please provide a technical justification for the rationale of using ASME Class 2 rules for ASME Class 1 piping in the preliminary piping evaluations that is indicated in Section 3.12.1.
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NuScale Response:

The initial response to RAI 9069 Question 03.12-5, as submitted by NuScale letter RAIO-1218-63674, December 3, 2018, is supplemented by the following.

The response to Sub-question 1 is replaced in its entirety by:

1. Final detailed piping analyses are performed for the high energy pipe routings as described in the NuScale September 8, 2015 presentation on use of the Graded Approach, 'Level of Detail for Piping Analysis' [ML15260A521]. Specifically:
 - The CVCS/RCS discharge line from the reactor pressure vessel nozzle connection to the first anchor (i.e., restraint in six degrees of freedom) on the outboard side of the reactor bay wall.
 - The feedwater lines from the reactor pressure vessel nozzle connection to the anchor support(s) on the outboard side of the reactor bay wall.



NRC agreement with this analysis approach was documented in the NRC Summary of September 8, 2015, Open and Closed Meeting with NuScale to Discuss Piping and Pipe Rupture Hazards Analyses for Design Certification, October 29, 2015 [ML15295A360].

Additionally, final detailed piping analyses are performed for the high energy pipe routings that fall within the area designated as containment penetration area, as described in FSAR Section 3.6.2.1.2.2. These analyses evaluate the MSS piping from the reactor pressure vessel nozzle connection to the anchor support(s) on the outboard side of the reactor bay wall.

The response to Sub-question 5 is replaced in its entirety by:

5. The detailed final as-designed analysis of the portions of the main steam lines inside the containment vessel (i.e., the portions for which LBB methodology is applied) is complete. (See the response to RAI 8938 Question 03.12-1, submitted separately.)

The detailed final as-designed analysis of the portions of the main steam lines within the containment penetration area is complete. (See the responses to RAI 8836 Question 03.06.02-2 and RAI 8938 Question 03.12-1, submitted separately.)

The responses to Sub-questions 2 through 4 and 6 are unchanged.

Impact on DCA:

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