NuScale Power

RAIO-0517-54211

May 24, 2017

Docket No. 52-048

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

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information (eRAI No. 8766) on the NuScale Design Certification Application

REFERENCE: 1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 05 (eRAI No. 8766)," dated April 25, 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 Question from NRC eRAI No. 8766:

- 15.06.03-1

This letter and the enclosed response makes no new regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions on this response, please contact Darrell Gardner at 980-349-4829 or at dgardner@nuscalepower.com.

Sincerely,

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

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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 8766
RAI 15.06.03-1

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Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 8766 - RAI 15.06.03-1

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

eRAI No.: 8766
Date of RAI Issue: 04/25/2017

NRC Question No.: 15.06.03-1

Title 10 of the *Code of Federal Regulations* (10 CFR) 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are performed, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses; 10 CFR 50, Appendix A, General Design Criterion (GDC) 19 for control room radiological habitability; and the requirements related to the technical support center in Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements, and Standard Review Plan (SRP) Section 15.0.3, “Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors,” specifies the dose acceptance criteria. The fission product inventory released from all failed fuel rods is an input to the radiological evaluation under SRP Section 15.0.3. The NRC staff needs to ensure that the analysis showing no failure of fuel is suitably conservative.

FSAR Tier 2, Table 15.6-7, “Steam Generator Tube Failure - Sequence of Events - Limiting Reactor Pressure Vessel Pressure,” shows that the time at which the maximum intact steam generator pressure is reached is 14 s into the transient; however, FSAR Tier 2, Figure 15.6-22, “Steam Generator Tube Failure - Limiting Reactor Pressure Vessel Pressure Scenario – Reactor Pressure Vessel and Steam Generator Pressures,” shows the time of maximum intact steam generator pressure to be about 39 s. Because these timelines are not consistent, the staff lacks confidence that the peak pressure calculation has been reported accurately. RCS pressure is one acceptance criterion for postulated accidents. Therefore, please clarify the correct peak RCS pressure timing, and update the FSAR to correct any erroneous information.

NuScale Response:

FSAR Table 15.6-7: Steam Generator Tube Failure - Sequence of Events - Limiting Reactor Pressure Vessel Pressure incorrectly lists the time when the maximum pressure occurs for the intact steam generator. Consistent with FSAR Figure 15.6-22, the maximum pressure for the intact steam generator occurs at approximately 40 seconds after the initiation of the event. FSAR Table 15.6-7 has been corrected to reflect the correct time for the intact steam generator maximum pressure.

Impact on DCA:

FSAR Table 15.6-7 have been revised as described in the response above and as shown in the markups provided in this response.

RAI 15.06.03-1

Table 15.6-7: Steam Generator Tube Failure - Sequence of Events - Limiting Reactor Pressure Vessel Pressure

Event	Time (s)(1)
SGTF at Top of Steam Generator	0
Loss of Normal AC Power	0
Turbine Stop Valves Closure	1
High Pressurizer Pressure Signal (2000 psia)	8
Reactor trip	10
DHRS Actuation	10
MSIV Closure	10
High Steam Line Pressure (800 psia) for faulted SG	10
High Steam Line Pressure (800 psia) for intact SG	12
Maximum RPV Pressure reached	12
MSIVs fully closed	15
Maximum Intact SG pressure reached	~~14~~40
~~MSIVs fully closed~~	~~15~~
Maximum Faulted SG pressure reached	150
Low Pressurizer Pressure (1600 psia)	2963
Low Pressurizer Level (35%)	3649
Pressurizer Heater Trip	3650
Minimum Pressurizer Level	6000

Notes: (1) Time rounded up to second.