February 11, 2019

Docket: PROJ0769

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

SUBJECT: NuScale Power, LLC Supplemental Response to NRC Request for Additional Information No. 9513 (eRAI No. 9513) on the NuScale Topical Report, "Non-Loss of Coolant Accident Analysis Methodology," TR-0516-49416, Revision 1

REFERENCES:

1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 9513 (eRAI No. 9513)," dated May 08, 2018
2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 9513 (eRAI No.9513)," dated July 09, 2018
3. NuScale Topical Report, "Non-Loss of Coolant Accident Analysis Methodology," TR-0516-49416, Revision 1, dated August 2017

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. 9513:

- 15.00.02-20

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 Paul Infanger at 541-452-7351 or at pinfanger@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. 9513

Enclosure 1:

NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9513

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

eRAI No.: 9513

Date of RAI Issue: 05/08/2018

NRC Question No.: 15.00.02-20

GDC 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs. GDC 13, “Instrumentation and control,” requires the provision of instrumentation to monitor variables and systems over their anticipated ranges of normal operation, including the effects of AOOs, and of appropriate controls to maintain listed variables and systems within prescribed operating ranges. TR-0516-49416-P supports the conclusions relative to GDC 10 and 13 in the NuScale FSAR.

TR Sections 7.2.13, “Uncontrolled Control Rod Assembly Bank Withdrawal from Subcritical or Low Power Startup Conditions,” and 7.2.14, “Uncontrolled Control Rod Assembly Bank Withdrawal at Power,” state that low power startup conditions exist until the reactor power reaches 15 percent rated thermal power. In addition, TR Section 8.3.1 of the same title assumes an initial core power of 15 percent rated thermal power. However, FSAR Tier 2, Section 15.4.1.3.2, “Input Parameters and Initial Conditions,” for the uncontrolled control rod assembly bank withdrawal from subcritical or low power startup conditions, states that the maximum initial power considered in the analysis is 25 percent, as the high power trip is set at 25 percent of full power for startup conditions. To ensure that future analyses encompass the appropriate operating ranges, address the apparent discrepancy in the scope of low power conditions, and update TR-0516-49416-P as appropriate.

NuScale Response:

The original NuScale response as submitted in NuScale correspondence RAIO-0718-60781 and dated June 9, 2018, is augmented with the following information.

The entry in Table 15.0-8 for initial power at the high level for Chapter 15.4.1 events was changed to 15% from 25% as a conforming change to the original response consistent with the range of initial power evaluated as indicated at the end of this response.

Impact on DCA:

Table 15.0-8 been revised as described in the response above and as shown in the markup provided in this response.

RAI 15.00.02-20S1

Table 15.0-8: Reactivity Coefficients

Section	Design Basis Event	Power Level % HFP (160 MWt)	Moderator Temperature Coefficient(1)	Doppler Coefficient(1)
15.1	**Increase in Heat Removal by the Secondary System**			
15.1.1	Decrease in Feedwater Temperature	102%	-43.0 pcm/°F	-1.40 pcm/°F
15.1.2	Increase in Feedwater Flow	102%	-43.0 pcm/°F	-1.40 pcm/°F
15.1.3	Increase in Steam Flow	102%	-43.0 pcm/°F	-1.40 pcm/°F
15.1.4	Inadvertent Opening of Steam Generator Relief or Safety Valve	102%	-43.0 pcm/°F	-1.40 pcm/°F
15.1.5	Steam Piping Failures Inside and Outside of Containment	102%	0.0 pcm/°F (RCS pressure case) -43.0 pcm/°F (MCHR case)	-1.40 pcm/°F -1.40 pcm/°F
15.1.6	Loss of Containment Vacuum/Containment Flooding	102%	-43.0 pcm/°F	-1.40 pcm/°F
15.2	**Decrease in Heat Removal by the Secondary System**			
15.2.1	Loss of External Load	102%	0.0 pcm/°F	-1.40 pcm/°F
15.2.2	Turbine Trip	102%	0.0 pcm/°F	-1.40 pcm/°F
15.2.3	Loss of Condenser Vacuum	102%	0.0 pcm/°F	-1.40 pcm/°F
15.2.4	Closure of Main Steam Isolation Valve	102%	0.0 pcm/°F	-1.40 pcm/°F
15.2.5	Steam Pressure Regulator Failure (Closed)	N/A	N/A	N/A
15.2.6	Loss of Non-Emergency AC to the Station Auxiliaries	102%	0.0 pcm/°F	-1.40 pcm/°F
15.2.7	Loss of Normal Feedwater Flow	102%	0.0 pcm/°F	-1.40 pcm/°F
15.2.8	Feedwater System Pipe Breaks Inside and Outside Containment	102%	0.0 pcm/°F	-1.40 pcm/°F
15.2.9	Inadvertent Operation of the Decay Heat Removal System	102%	-43.0 pcm/°F (EOC) 0.0 pcm/°F (BOC)	-1.40 pcm/°F
15.3	**Decrease in RCS Flow Rate (not applicable)**			
15.4	**Reactivity and Power Distribution Anomalies**			
15.4.1	Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power or Startup Condition	1W - ~~2~~15%	+6 pcm/°F	-1.40 pcm/°F