NuScale Power

RAIO-1017-56865

October 26, 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. 210 (eRAI No. 8999) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 210 (eRAI No. 8999)," dated September 01, 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. 8999:

- 19-30

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 Darrell Gardner at 980-349-4829 or at dgardner@nuscalepower.com.

Sincerely,

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

Distribution: Gregory Cranston, NRC, OWFN-8G9A
Samuel Lee, NRC, OWFN-8G9A
Rani Franovich, NRC, OWFN-8G9A

Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 8999

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 8999

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

eRAI No.: 8999
Date of RAI Issue: 09/01/2017

NRC Question No.: 19-30

Section 15.2.1 of the Part 7 of the design certification application, “Exemptions”, provides the technical basis for a proposed exemption from 10 CFR 50, Appendix A, Criterion 27, “Combined Reactivity Control Systems Capability.” As part of the technical basis, the applicant states that a bounding probability for a return to power following reactor trip is calculated to be less than 1E-6 per reactor year, which is considered by the applicant to be insignificant. The applicant makes the following three assumptions in performing this calculation:

1. The probability per demand of each individual control rod failing to insert is 1.35 E-05.
2. The probability of a chemical and volume control system (CVCS) failure to insert soluble boron is 8E-3 per demand.
3. The probability that the reactor is in a state which could result in a return to power with the highest worth control rod stuck out (based on time in cycle and time at power) is 4E-2 to 1E-1 per year.

The bases for these probabilities are not provided in the design certification application. The staff requests that the applicant describe the method by which these probabilities were calculated, including any applicable operating experience or reliability analyses, to allow the staff to confirm the applicant’s assertion that a return to power following reactor trip is highly unlikely.

NuScale Response:

Item 1: The probability per demand of an individual control rod to insert is listed in FSAR Table 19.1-9; as indicated in the table, the source of this probability is NUREG/CR-6928, FSAR Reference 19.1-23 (specifically, Table 5-1 of initial issue). FSAR Reference 19.1-23 has been clarified to include the initial issue of NUREG/CR-6928 as well as the 2010 update. Use of data from currently operating plants was judged appropriate for the NuScale design because (i) the NuScale core is shorter than a typical currently operating plant and thus, warping and binding that could inhibit control rod insertion is less likely and (ii) as illustrated in FSAR Figure 4.6-1, the control rod drive shaft provides additional weight and corresponding additional force to facilitate control rod assembly insertion.

Item 2: The probability that the chemical and volume control system (CVCS) fails to insert soluble boron when demanded was calculated by solving the fault tree associated with the event tree top event “CVCS-T01”. As described in FSAR Table 19.1-4, CVCS is the system modeled in the PRA that provides primary coolant makeup. Key information used to support this calculation is summarized in the following paragraph.

FSAR Table 19.1-76 identifies the boron addition system as a means to provide reactivity control when aligned as a suction source to the CVCS. FSAR Table 19.1-7 describes the success criteria for top event CVCS-T01, which is used to calculate the probability that the CVCS fails to insert soluble boron. As described in Table 19.1-7, success of CVCS requires operator action and activating a makeup pump. FSAR Section 19.1.4.1.1.5 identifies the sources of data used in the PRA for component failure rates, equipment unavailabilities, human error probabilities, and common cause failure parameters. FSAR Section 19.1.4.1.1.3 describes that the PRA mission time is 72 hours (i.e., the time systems and components are required to operate).

Item 3: There are a number of factors, which are subject to change during the operating life cycle, that influence the potential for a return to power. These include moderator temperature coefficient, decay heat, and the concentration of soluble boron in the reactor coolant. As described in FSAR Section 4.3.1, control rods are designed to control power during normal operation and anticipated operational occurrences; the control rod assemblies, with all control rods inserted, are capable of holding the reactor subcritical. However, as described in FSAR Section 15.0.6, under the conditions that (1) the highest worth control rod assembly is assumed not to insert, (2) the CVCS is unavailable, and (3) during a small window of operational conditions when boron concentration is low and decay heat is low, a return to power could occur following subcriticality after a shutdown.

Operational conditions during the fuel cycle were evaluated to estimate the time window when the reactor could return to power, with an assumed worst rod stuck out (WRSO), following subcriticality after a reactor trip. As described in Design Certification Application, Part 7 Section 15.2.1, a return to power is precluded at the beginning of a fuel cycle due to a favorable moderator temperature coefficient and the higher boron concentration associated with beginning-of-cycle operation. A return to power is also precluded during most of the fuel cycle because higher decay heat levels prevent a return to power due to the negative reactivity from voiding. The probability that the reactor could return to power following subcriticality after a reactor trip, with a WRSO, is based on the time window after a restart when decay heat levels are still relatively low. The estimated range for this time window is 2 to 6 weeks (depending on the operational life cycle of the module); this time span bounds the operating window at the start of a fuel cycle when decay heat levels could be sufficiently low to allow a return to power as a result of cooling down the reactor coolant system. The corresponding probabilities for this operating window, assuming one shutdown per year, are 4E-2 (i.e., 2 weeks / 52 weeks) to 1E-1 (i.e., 6 weeks / 52 weeks).

Impact on DCA:

FSAR Section 19.1.10 has been revised as described in the response above and as shown in the markup provided in this response.

19.1-12 EPRI 1016747, “Program on Technology Innovation: Comprehensive Risk Assessment Requirements for Passive Safety Systems, “ Electric Power Research Institute, Palo Alto, CA, December 2008.

19.1-13 IAEATECHDOC-1752, “Progress in Methodologies for the Assessment of Passive Safety System Reliability in Advanced Reactors,” International Atomic Energy Institute, Vienna, Austria, 2014.

19.1-14 EPRI 1025291, “Pilot Application of Risk-Informed Safety Margins of Power Uprates for Loss of Main Feedwater Events,” Electric Power Research Institute, Palo Alto, CA, December 2012.

19.1-15 EPRI TR-100741, “MAAP Thermal-Hydraulic Qualification Studies,” Electric Power Research Institute, Palo Alto, CA, June 1992.

19.1-16 NUREG/CR-6890 "Reevaluation of Station Blackout Risk at Nuclear Power Plants, Analysis of Loss of Offsite Power Events: 1986-2004" with subsequent updates including: "Analysis of Loss of Offsite Power Events 2013 Update."

19.1-17 EPRI NP-2230 "ATWS: A Reappraisal. Part 3. Frequency of Anticipated Transients."

19.1-18 NUREG/CR-2680 "Seismic Safety Margins Research Program: Equipment Fragility Data Base," Lawrence Livermore National Laboratory, January 1983, U.S. Nuclear Regulatory Commission.

19.1-19 NUREG/CR-3558 "Seismic Safety Margins Research Program: Handbook of Nuclear Power Plant Seismic Fragilities," Lawrence Livermore National Laboratory, June 1985. U.S. Nuclear Regulatory Commission.

19.1-20 NUREG/CR-4659 "Seismic Fragility of Nuclear Power Plant Components," Brookhaven National Laboratory, August 1991, U.S. Nuclear Regulatory Commission.

19.1-21 EPRI 1002989, "Seismic Probabilistic Risk Assessment Implementation Guide," Electric Power Research Institute, Palo Alto, CA, December 2003.

19.1-22 NUREG/CR-6883 "The SPAR-H Human Reliability Analysis Method," INL/EXT-05-00509, August 2005. U.S. Nuclear Regulatory Commission.

RAI 19-30

19.1-23 NUREG/CR-6928~~-~~, Initial Issue 2007 and 2010 Update, "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants," U.S. Nuclear Regulatory Commission, September 2012.

19.1-24 INL/EXT-16-37873 "Analysis of Loss-of-Offsite-Power Events 1997-2014", Idaho National Laboratory, February 2016.