NuScale Power

RAIO-1117-57346

November 27, 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. 260 (eRAI No. 9108) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 260 (eRAI No. 9108)," dated October 13, 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. 9108:

- 19-34

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. 9108

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9108

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvalis, Oregon 97330, Office: 541.360.0500, Fax: 541.207.3928
www.nuscalepower.com

NuScale Power

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

eRAI No.: 9108
Date of RAI Issue: 10/13/2017

NRC Question No.: 19-34

Regulatory Basis

10 CFR 52.47(a)(27) states that a Design Certification (DC) application must contain a Final Safety Analysis Report (FSAR) that includes a description of the design-specific Probabilistic Risk Assessment (PRA) and its results. 10 CFR 52.47(a)(23) states that a DC application for light-water reactor (LWR) designs must contain an FSAR that includes a description and analysis of design features for the prevention and mitigation of severe accidents (e.g., challenges to containment integrity caused by core-concrete interaction, steam explosion, high-pressure melt ejection, hydrogen combustion, and containment bypass).

Regulatory 1.200, Revision 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" provides the NRC standards for performing PRA. Table 3, "Summary of Technical Characteristics and Attributes of a Level 2 PRA for Internal Events," of the regulatory guide states that the PRA should include an assessment of the credible severe accident phenomena via a structured process.

The Commission's 1995 PRA Policy Statement states that treatment of uncertainty is an important issue and notes that uncertainties are due to knowledge limitations. The Policy Statement notes that a probabilistic approach has exposed some of these limitations and has provided a framework to assess their significance and assist in developing a strategy to accommodate them in the regulatory process. NUREG-1855 provides guidance on how to treat uncertainties associated with PRAs.

Standard Review Plan 19.0, Revision 3, guidance is to compare the PRA results against a probabilistic goals that (1) the conditional containment failure probability be less than 0.1 for the composite of all core damage sequences assessed in the PRA and (2) the large release frequency be less than 1x10-6 per year.

Request for Additional Information

The staff reviewed the applicant's Level 2 analysis in FSAR Chapter 19, ER-P020-7024-Revision 0, Level 2 PRA Notebook, and ER-P020-5092, Revision 0, Assessment of Low Risk Severe Accident Phenomena for the NuScale Level 2 PRA. As described in these documents, the applicant screened out the following severe accident challenges to containment (i.e.,

---

assigned them a probability of zero in the Containment Event Tree (CET)): high pressure melt ejection, failure of the reactor pressure vessel and containment vessel (CNV) bottom heads by contact with corium, steam explosion, and hydrogen combustion. Assigning these severe accident phenomena a probability of zero can reduce the conditional containment failure probability as well as the large release frequency. The applicant is requested to provide a summary in FSAR Chapter 19 that describes the approach used to quantitatively screen these severe accident phenomena from the CET. This information is needed for the staff to find that the design meets the probabilistic goals.

The applicant also is requested to reconcile apparent inconsistencies with the Level 2 PRA and update the Level 2 notebook, the CET, and the FSAR as appropriate:

- The hydrogen combustion analysis assumes the containment has an initial air pressure of 0.1 psia. However, FSAR Table 7.1-4, "Engineering Safety Feature Actuation System Functions", states that the Containment System Isolation (CSI) High Narrow Range Containment Pressure is not actuated until containment pressure reaches 9.5 psia. FSAR Table 7.1-4 suggests that the containment could have an initial air pressure as high as 9.5 psia. Also, FSAR Section 9.3.6 states that the containment evacuation system (CES) is used to establish and maintain a vacuum in the containment vessel during operation by removing non-condensable gases from the containment vessel. However, the FSAR does not specify criteria for operating the CES or provide a technical specification that would limit containment vessel pressure. Please explain or clarify these apparent inconsistencies.
- For chemical and volume control system (CVCS) over-pressurization, the applicant has provided text in ER-P020-7024-R0 stating why this over-pressure condition cannot result in containment failure. The applicant assumed the likelihood that the operator continuously operates CVCS (which would result in over-pressurization of the reactor pressure vessel and the CNV) is small enough to be excluded from the CET. The staff is requesting an analysis of the sequence of events for a postulated CVCS over-pressurization to justify that the likelihood of operator error is low. This analysis should include as applicable, instrumentation, alarms, and required time that the operator needs to terminate CVCS injection.

NuScale Response:

<u>Severe accident screening in the containment event tree (CET)</u>

The approach used to quantitatively screen severe accident phenomena from the CET follows the method discussed in NUREG-1524 (A Reassessment of the Potential for an Alpha-Mode Containment Failure and a Review of the Current Understanding of Broader Fuel-Coolant Interaction Issues). In NUREG-1524, severe accident phenomena that are judged “physically unreasonable”, "vanishingly small", or “very unlikely” are not quantified because they are of “little or no significance to the overall risk” from a nuclear power plant; the quantitative value used for such phenomena was less than $1x10^{-3}$. A probability this low is negligible with respect to the conditional containment failure probability (CCFP) safety goal of less than 0.1 for the composite of all core damage sequences, as stated in FSAR Section 19.1.9.1. Based on the analyses summarized in FSAR Section 19.2, severe accident phenomena such as high pressure melt ejection, failure of the reactor pressure vessel (RPV) and containment vessel (CNV) bottom heads by contact with corium, steam explosion, and hydrogen combustion were found to be physically unrealistic in the NuScale design. As such, these severe accident phenomena were judged to be negligible contributors to the large release frequency and CCFP, and not included in the CET.

FSAR Section 19.1.4.2.1.2 has been revised to add a quantitative criterion for screening severe accident phenomena from the CET.

Containment pressures used for analysis

As discussed in FSAR Section 19.2.3.3.2, a near vacuum is maintained in the CNV during normal operation by the containment evacuation system (CES); the nominal pressure for normal operation is 0.1 psia, as listed in FSAR Table 7.1-2.

As described in TR-0716-50424 (FSAR References 3.8.2-4 and 6.2-3), an initial containment pressure of 1 psia is used for modeling the response to design basis accidents. A conservative initial containment pressure of 2 psia is used in FSAR Chapter 15 for design basis evaluations (FSAR Table 15.0-6) and FSAR Chapter 6 for hydrogen combustion analysis (TR-0516-49084, FSAR Reference 6.2-1). FSAR Sections 19.1.4.1 and 19.2.3.2 use an initial containment pressure of 1 psia for PRA analyses, with the exception of the analysis of hydrogen generation in a beyond design basis event (described in FSAR Section 19.2.3.3.2), which uses a conservative initial containment pressure of 9.5 psia. Similarly, TR-0716-50424 uses an initial containment pressure of 9.5 psia for modeling hydrogen generation in response to a beyond design basis accident.

As discussed in the response to RAI 8793 (Question 06.02.01-2), provided in NuScale letter RAIO-0917-56135 dated September 22, 2017, the containment pressure must be less than 3 psia for acceptable leak rate detection.

The following table summarizes the containment pressures that are relevant to NuScale analyses:

Containment Pressure (psia)	Description	Use
0.1	Normal operating pressure which is maintained by CES operation	FSAR Table 7.1-2
1	• Initial pressure for design basis analysis of hydrogen control • Initial pressure for PRA analysis (except hydrogen generation)	• TR-0716-50424 • FSAR Section 19.1.4.1, FSAR Section 19.2.3.2
2	Conservative initial pressure for Section 15 design basis evaluations	FSAR Table 15.0-6, TR-0516-49084
< 3	Technical Specification Limiting Condition of Operation (LCO)	Technical Specification LCO 3.4.7, FSAR Section 5.2.5.1
9.5	Initial pressure for beyond design basis analysis of hydrogen control	FSAR Section 19.2.3.3.2, TR-0716-50424

Containment overpressurization by the chemical and volume control system (CVCS)

The likelihood of overpressurizing the integrated reactor coolant system (RCS) and the CNV volumes due to continuous CVCS operation is so small that inclusion in the PRA is unnecessary. Operators would have hours to terminate CVCS injection before filling the module with water; this time frame is based on conservatively accounting for only the volume of the RCS and assumes the maximum flow rate of both CVCS pumps, as indicated in FSAR Tables 5.1-1 and 9.3.4-1, respectively. A high CNV water level alarm is provided to operators by "High CNV level", as indicated in FSAR Table 6.3-1. If operators do not isolate the CVCS, an automatic isolation of the CVCS containment isolation valves will occur on a "High Pressurizer Level", as indicated in FSAR Table 7.1-4. Because of the extended time for operator action and the automatic isolation function, CVCS overpressurization of the CNV has not been included in the CET.

Impact on DCA:

FSAR Section 19.1.4.2.1.2 has been revised as described in the response above and as shown in the markup provided in this response.

to perform an intermediate grouping or detailed "binning" of Level 1 core damage sequences; i.e., all core damage sequences identified in the Level 1 analysis are binned into the "CD" plant damage state.

The Level 2 event tree models the progression of a severe accident from core damage to the point of a potential release. The Level 2 event tree is also referred to as the containment event tree (CET). End states of the CET define the conditions that characterize the effect of the sequence on the environment, i.e., the potential radionuclide release. As such, end states reflect release characteristics such as timing and magnitude. Due to the simplicity of the design, only two CET end states are used to model radionuclide release. The end state "NR" is associated with a release that may be attributed to leakage from the boundary of an isolated containment; the end state "LR" is associated with a release from an unisolated containment. Each of these end states is assigned to a release category (RC) to represent the radionuclide source term.

19.1.4.2.1.2 Containment Event Tree

In the NuScale PRA, the CET is directly linked to the end state of the Level-1 event trees. Therefore, there is no development of plant damage states (PDS) to group sequences by similar characteristics. Instead, each core damage accident sequence that is not a success is directly linked to the CET by the transfer event LEVEL2-ET. As such, each core damage accident sequence is directly linked and propagated through the CET. As summarized below, most containment failure modes typically considered in Level 2 PRA analyses are physically unrealistic in the NuScale design. As such, all Level 1 sequences that are classified as core damage (i.e., whose end state is not "OK") transfer to a single CET initiating event, Level2-ET, as illustrated in Figure 19.1-15.

Severe Accident Processes and Phenomena

Potential severe accident phenomena are evaluated to determine their applicability to the NuScale design. The evaluation considers phenomena listed in the ASME/ANS PRA standard, Section 19.0 of the Standard Review Plan, the ASME/ANS PRA Standard, NUREG/CR-2300 (Reference 19.1-38) and NUREG/CR-6595 (Reference 19.1-39).

RAI 19-34

The characteristics of the NuScale design provide an inherent degree of safety. As a result, severe accident phenomena that may challenge containment in currently operating plants are shown in Section 19.2 to be physically unrealistic in the NuScale design. These phenomena are not included in the CET consistent with the approach taken in NUREG-1524 (Reference 19.1-64) in which phenomena that are judged to be "physically unreasonable", "vanishingly small", or "very unlikely" have a probability of less than 1 E-3. Because this probability is small with respect to the LRF and the conditional containment failure probability (CCFP) safety goal of less than 0.1, such events are not explicitly included in the CET.

19.1-54 Quanterion Automated Databook: Electronic Parts Reliability Data 2014 (EPRD-2014), Nonelectric Parts Reliability Data 2011 (NPRD-2011), Failure Mode/Mechanism Distribution 2013 (FMD-2013), Quanterion Solutions Incorporated, 100 Seymour Rd Kunsela Hall Suite C106 Utica, NY 13502.

19.1-55 EPRI TR- 1021167, "An Analysis of Loss of Decay Heat Removal and Loss of Inventory Event Trends (1990-2009)," Electric Power Research Institute, Palo Alto, CA, 2010.

19.1-56 DC/COL-ISG-020, "Implementation of a Seismic Margin Analysis for New Reactors Based on Probabilistic Risk Assessment," U.S. Nuclear Regulatory Commission, March 2010.

19.1-57 EPRI 103959, Electric Power Research Institute, "Methodology for Developing Seismic Fragilities," Electric Power Research Institute, Palo Alto, CA, June 1994.

19.1-58 EPRI 1019200, Electric Power Research Institute, "Seismic Fragility Applications Guide Update," Electric Power Research Institute, Palo Alto, CA, December 2009.

19.1-59 EPRI 3002000507, Electric Power Research Institute, "Utility Requirements Document," Approved Version 13, ALWR Passive Plant, Electric Power Research Institute, Palo Alto, CA, December 2014.

19.1-60 NUREG/CR-5485, "Guidelines on Modeling Common-Cause Failures in Probabilistic Risk Assessment," U.S. Nuclear Regulatory Commission, June 1998.

19.1-61 NUREG/CR-5497 - 2012 Update, "Common Cause Failure Parameter Estimations," U.S. Nuclear Regulatory Commission, January 2012.

19.1-62 Microsemi Reliability Report, No. 51000001-11/05.13, May 2013.

RAI 19-27

19.1-63 SAND2011-0128, "Accident Source Terms for Light Water Nuclear Power Plants Using High-Burnup or MOX Fuel," Sandia National Laboratories, January 2011.

RAI 19-34

19.1-64 NUREG-1524, "A Reassessment of the Potential for an Alpha-Mode Containment Failure and a Review of the Current Understanding of Broader Fuel-Coolant Interaction Issues," U.S. Nuclear Regulatory Commission, July 1996.