RAIO-1017-56550

October 11, 2017

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 168 (eRAI No. 8977) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 168 (eRAI No. 8977)," dated August 12, 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. 8977:

- 19-27

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
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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 8977

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NuScale Response to NRC Request for Additional Information eRAI No. 8977

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

eRAI No.: 8977
Date of RAI Issue: 08/12/2017

NRC Question No.: 19-27

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). For staff to make a finding that the applicant has performed an adequate evaluation of the risk from severe accidents in accordance with Standard Review Plan (SRP) 19.0, the applicant is requested to respond to the questions below.

Request for additional information

a. The applicant used a large release frequency metric of less than 10-6 large releases per year and defined a large release as an acute exposure of greater than 200 rem to an individual located at a distance of 0.167 miles from the reactor for 96 hours. SRP 19.0 directs the staff to determine whether the applicant has adequately demonstrated that the risk associated with the design compares favorably against the Commission's goals. In order to make this finding, the applicant is requested to add information to Chapter 19 of the NuScale FSAR to demonstrate its large release frequency metric, including its large release definition, is equivalent to or less than the Commission's Safety Goal Policy's quantitative health objective for prompt fatality risk.
b. The applicant is requested to clarify the text in Chapter 19 of the FSAR by adding the following:
    a. Identify the scenarios in which the applicant compared the predicted dose directly against the large release definition of 200 rem at 0.167 miles to classify whether the scenario results in a large release.
    b. Identify the scenarios in which the applicant compared the predicted radionuclide release against the MACCS back-calculated radionuclide release equivalent to the large release definition of 200 rem at 0.167 miles to classify whether the scenario

results in a large release.

c. For at-power accidents, “Probabilistic Risk Assessment Large Release Frequency Definition,” ER- P000-7004-R0, and “Release Fraction Determination for PRA Large Release,” ER-P000-7005, describe (1) the use of MACCS to translate the large release definition of 200 rem over 96 hours into an equivalent environmental radionuclide release and (2) a hand calculation showing that releases from a leaking containment (as opposed to a failed containment) are smaller than this (i.e., less than a large release). In the FSAR, the applicant used the iodine release fraction as the metric for this comparison. The applicant is requested to add clarifying information to Chapter 19 of the FSAR describing how the following were addressed in this comparison: (1) other aspects of the environmental release such as release timing, release rate, other radionuclides (e.g., cesium) and (2) other potentially important phenomena such as changes in wind direction during the 96- hour exposure period.

d. “Code Manual for MACCS2,” NUREG/CR-6613, Vol. 1, states “The dispersion of a plume of material released in the wake of a building is subject to a large degree of uncertainty. For that reason, MACCS should not be used for estimating doses at distances of less than 0.5 km [0.31 miles] from laboratory or industrial-facilities.” The applicant’s discussion on page 10 of “Probabilistic Risk Assessment Large Release Frequency Definition,” ER-P000-7004-R0, indicates that the applicant recognized this uncertainty and attempted to address it by applying a building wake model to both the short and long faces of the reactor building and identifying the largest dose. The applicant is requested to add information to Chapter 19 of the FSAR describing its validation of the assumptions and input used in its MACCS predictions of plume concentration at 0.167 miles from the reactor for its large release assessment. The information should include a discussion of its parameterization of the spatially dependent dispersion parameters (sigma-y and sigma-z) in MACCS’s Gaussian plume model and its treatment of meteorological phenomena such as building wake, plume lift, and meander, as applicable.

NuScale Response:

Item a): As stated in FSAR Section 19.1.4.2.1.4, NUREG-0396 (FSAR Reference 19.1-9) defines an acute 200 rem whole body dose as the dose at which significant early injuries start to occur. Therefore, this dose was used in the NuScale PRA to define a “large” release; the large release is evaluated at a "best-estimate" distance of 0.167 miles from a postulated release point to the site boundary. This best-estimate site boundary distance is defined as one-half of the shortest site dimension. FSAR Section 19.1.4.2.1.4 has been modified to clarify the distance used in the large release definition.

This definition of a large release is consistent with the NRC’s performance guideline and surrogate metric for prompt fatality risk as provided in FSAR Reference 19.1-36.

> *Guideline: Consistent with the traditional defense-in-depth approach and the accident mitigation philosophy requiring reliable performance of containment systems, the overall*

mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation (i.e., large release frequency [LRF] < 1 x 10^{-6} per reactor year).

The Commission proposed this performance guideline for the staff to use as a basis for determining whether a level of safety is consistent with the safety goal and quantitative health objective (QHO) for prompt fatality risk:

QHO: The risk to an average individual in the vicinity of an NPP or prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.

The NuScale calculated risk metrics, as shown in FSAR Table 19.1-80, are well below the NRC safety goals for LRF of less than 1 x 10^{-6} per reactor year, and for CDF of less than 1 x 10^{-4} per reactor year. These metrics are surrogates for the early fatality QHO and the latent cancer QHO, respectively, as discussed in TR-0515-13952-NP-A (FSAR Reference 19.1-8); thus NuScale demonstrates conformance with the QHOs.

FSAR Section 19.1.9.1 has been modified to include a discussion of the relationship between the large release definition and the QHOs.

Further, as noted in FSAR Section 19.1.9.2, the safety goals are independent of design, thus the size of the potential radionuclide source term is not considered in the core damage or large release frequency safety goals. The risk of operating a NuScale plant is further reduced because of the relatively small potential radionuclide source term.

Item b), Part a.): The predicted dose for one sequence, a module drop event, was evaluated directly against the large release definition of an acute 200 rem whole body dose. The module drop event was assumed to result in containment bypass. Dose consequences from the module drop event were calculated to be well below the large release criterion. Other containment bypass scenarios were not compared directly to the large release criterion, but were conservatively categorized as large release sequences. FSAR Section 19.1.6.2 has been clarified to refer to offsite dose, which was explicitly calculated for a postulated module drop.

Item b), Part b.): As described in FSAR Section 19.1.4.2.1.4, the predicted dose from individual sequences is not explicitly calculated or compared to the large release definition, with the exception of the module drop event. Instead, a bounding approach has been employed in which a maximum core inventory release from the containment was calculated to envelope severe accident sequences that are associated with an intact containment; as reported in FSAR Section 19.1.4.2.1.4, this value was calculated to be 0.8 percent of the iodine core inventory.

The bounding approach compared the largest possible iodine release fraction from an intact

containment scenario to the MACCS back-calculated radionuclide release equivalent to the large release definition. Considering that all the iodine in the core is assumed to be released to containment, the results confirmed that the iodine release fraction associated with containment leakage at Technical Specification limits (i.e., 0.8 percent of the iodine core inventory) is significantly lower than the MACCS back-calculated radionuclide release equivalent; as reported in FSAR Section 19.1.4.2.1.4, the MACCS back-calculated value is 2.9 percent of the iodine core inventory.

FSAR Section 19.1.4.2.1.4 has been modified to include discussion of these points.

Item c): The following provides additional clarifications to support the MACCS analysis of Release Category 2 (RC2):

- Release occurs immediately following core damage to maximize radionuclide inventory by minimizing radioactive decay
- Release rate is constant throughout the 2-hour release (i.e., each hour of the release contains the same release fraction)
- Radionuclide classes are scaled relative to iodine based on the guidelines in Table 11 of SAND2011-0128 for the gap release and in-vessel release phase of the proposed source term for PWRs using low burnup uranium dioxide
- Release occurs in hourly segments and travels in the direction of the wind at the time of release
- Wind direction, wind speed, and rain rate change hourly based on meteorological data
- No credit for reduced radionuclide concentration due to plume meander at low wind speeds
- Building wake effects are modeled from the short face of the reactor building

The following provides additional clarifications to support the MACCS analysis of Release Category 1 (i.e., RC1) as a hand calculation was used to determine the bounding environmental release from an intact containment due to Technical Specification leakage:

- Release occurs immediately following core damage to maximize radionuclide inventory by minimizing radioactive decay
- Release rate is constant throughout the 96-hour release consistent with maximum containment technical specification leakage
- Radionuclide groups are scaled relative to iodine consistent with Table 11 of SAND2011-0128
- Retention of radionuclides in the reactor vessel is credited for all radionuclide groups with the exception of xenon and iodine

FSAR Section 19.1.4.2.1.4 has been modified to include this additional information.

Item d) Validation of plume modeling within 0.31 miles of the release was accomplished through

a comparison of MACCS plume dispersion results with an independent atmospheric transport and dispersion code previously accepted by the NRC for use in the presence of building wakes at distances near the site boundary (i.e., ARCON96). The results were compared for an identical ground level release from the short face of the reactor building using meteorological data from five different sites across the U.S. The results show good agreement between relative plume concentrations calculated at and beyond the site boundary distance.

In addition, MACCS plume modeling of building wake effects follows best practices, using the height and width of the short face of the reactor building to calculate the initial sigma-y and sigma-z according to the equations in Section 5.10 of NUREG/CR-6613 (FSAR Reference 19.1-10). The parametrization of the spatial dependence of sigma-y and sigma-z during downwind transport is consistent with the State-of-the-Art Reactor Consequence Analyses (SOARCA) Project in NUREG/CR-7110. Plume buoyancy and plume meander are not modeled.

Therefore, MACCS produces reasonable results at the site boundary using the inputs and assumptions described in FSAR Section 19.1.4.2.1.4.

FSAR Section 19.1.4.2.1.4 has been modified to include discussion about the applicability of the MACCS plume dispersion modeling at the site boundary.

Impact on DCA:

FSAR Sections 19.1.4.2, 19.1.6.2, and 19.1.9.1 have been revised as described in the response above and as shown in the markup provided in this response.

19.1.4.2.1.4 Release Categories

The Level 2 event tree, provided as Figure 19.1-15, is completed by defining the end state of each sequence. The figure provides three end states, "CD", "NR" and "LR." The end state "CD" allows quantification of the CDF as it summarizes the sequences transferred from the Level 1 event trees. The end state "NR" represents a core damage sequence with intact containment; for this end state, the potential radionuclide release is due to allowable leakage as defined by the Technical Specifications. The "LR" end state represents a large release. Due to the small core used in the design, additional release categories to reflect a range of release possibilities were judged to be unnecessary. The release categories are:

- RC1 is core damage with successful containment isolation.
- RC2 is core damage with containment bypass or failure of containment isolation.

RC1: core damage with successful containment isolation.

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To ensure RC1 sequences are below the threshold of a large release, a bounding analysis is employed to envelope all intact containment sequences. A calculation of the maximum possible iodine release fraction to the environment from a single module accident with intact containment, assuming the Technical Specification leak rate limit, is calculated to be 0.8 percent of the iodine core inventory; this is well below the threshold of a large release, which is defined in RC2. The leakage is calculated for a 96 hour time period, with the following conservatisms:

- Core inventory is completely and instantaneously vaporized inside containment and remains airborne.
- No credit for iodine or xenon deposition in the reactor vessel.
- No credit for radionuclide deposition on the walls of the CNV.
- Containment leakage rate is the maximum allowable at 0.20% of containment air weight per day over the entire release.
- No credit for the biological shield above the reactor pool in deflecting or containing a release of radionuclides from a reactor module.
- No credit for reactor pool scrubbing.
- No credit for RXB filtration systems.
- No credit for RXB spray system.

RC2: core damage with containment bypass or failure of containment isolation

This release category represents the release associated with a core damage sequence that does not have successful isolation of the CNV. These sequences have a Level 2 end state of RC2 and are associated with a "large" release.

The large release frequency (LRF) is the quantified result of the Level 2 PRA, and is used to demonstrate conformance with the safety goal promulgated in NRC policy statement (Reference 19.1-36). While various definitions of "large release" have been considered, there is no established consensus definition. The definition used in the PRA is based on a threshold radionuclide dose that could result in early injuries.

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Specifically, NUREG-0396 (Reference 19.1-9) specifies 200 rem whole body dose as the dose at which significant early injuries start to occur. This dose was used as the basis for defining a "large release" in terms of a hypothetical individual located at the ~~reactor module site boundary~~site boundary; in the PRA, the "site boundary" is a best-estimate distance and is defined as one-half of the shortest site dimension, which is approximately 884 feet (0.167 miles). Based on simulation results using the MACCS code (Reference 19.1-10) a release fraction of 2.9 percent of the Iodine core inventory would cause an acute 200 rem whole body (red marrow) dose at the site boundary using 2006 Peach Bottom meteorology under the following bounding assumptions:

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- Target person is standing in the middle of the release plume, at the site boundary, ~~the minimum distance from a reactor module,~~

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- Building wake effects are modeled from the short face of the reactor building to maximize initial radionuclide concentration,
- Release occurs over a 2-hour period to minimize reduction in radionuclide concentration due to wind shift during release,

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- Release occurs immediately following core damage to maximize radionuclide inventory by minimizing radioactive decay,
- No credit for fission product pipe deposition, building retention (i.e., building filtration system or biological shield),
- No credit for reactor pool scrubbing,
- No credit for elevated release or plume buoyancy to decrease ground level air concentrations,

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- No credit for reduced radionuclide concentration due to plume meander at low wind speeds,
- 96-hour absorption window at the site boundary, which corresponds to an upper bound for the range of time factors discussed in Reference 19.1-40.

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The following best-estimate assumptions were also used:

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- Radionuclide classes are scaled relative to iodine based on the guidelines in Table 11 of SAND2011-0128 (Reference 19.1-63) for the gap release and in-

vessel release phase of the proposed source term for PWRs using low burnup uranium dioxide,

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- Release rate is constant throughout the 2-hour release (i.e., each hour of the release contains the same release fraction),

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- Release occurs in hourly segments and travels in the direction of the wind at the time of release,

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- Wind direction, wind speed, and rain rate change hourly based on meteorological data.

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MACCS plume dispersion results were compared to and found consistent with those provided by the ARCON96 code (described in FSAR Section 15.0.2) at and beyond the site boundary.

19.1.4.2.1.5 Data Sources and Analysis

This section provides the sources of numerical data used in the Level 2 PRA. Initiating event frequencies, component failure rates, equipment unavailabilities, human error probabilities, and common-cause failure parameters are discussed.

Containment Event Tree Initiating Event Frequency

The frequency of the CET initiating event, "LEVEL2-ET", is the summation of all core damage sequences.

Component Failure Rates and Equipment Unavailability

Because the NuScale reactor modules and plant have no operating history, failure rates are derived from generic data (i.e., based on industry information or other accepted practices and standards). The generic data sources to support quantification of top event CNTS-T01 are summarized in Section 19.1.4.1.1.5.

Because the CNV is maintained subatmospheric during power operation, to minimize heat loss, testing and maintenance on containment penetrations is expected to be performed during outages. As such, unavailability of the CIVs because of testing or maintenance is not included in the model. Unavailability because of testing or maintenance on the equipment providing the signals to close the valves is included in the model.

Human Error Probabilities

There is one post-initiator operator action modeled for containment isolation, CNTS--HFE-0001C-FTC-N. It is a recovery action following failure of the MPS

The mean value of the CDF due to internal events for a module during LPSD conditions is calculated to be 4.5E-13 per calendar year; the 5th and 95th percentile values are 1.4E-14 per calendar year and 1.7E-12 per calendar year, respectively.

The mean CDF due to module drop, MD-CD, is calculated to be 8.8E-08 per calendar year; the 5th and 95th percentile values are 1.5E-08 per calendar year and 2.5E-07 per calendar year, respectively. This frequency is equally divided between POS3 and POS5.

Large Release Frequency

The mean value of the large release frequency due to internal events for a module during LPSD conditions is calculated to be 2.3E-14 per calendar year; the 5th and 95th percentile values are 2.1E-16 per calendar year and 7.4E-14 per calendar year, respectively.

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Analysis shows that the scrubbing effect of the water in the reactor pool reduces the offsite ~~consequences~~dose to only a small fraction of the large release definition, and therefore precludes a large release following a module drop. Thus, a dropped module event does not contribute to the large release frequency.

Significant Core Damage and Large Release Sequences

Four initiating events are significant with regard to internal events: spuriously open ECCS valve, loss of DC power, reactor coolant system LOCA inside containment, and loss of support system transient. The loss of support system transient is associated with a sequence in which the RPV fails due to overpressure. The other three initiating events are significant because of sequences with an incomplete ECCS actuation. The increased significance of the spurious ECCS valve opening initiator in the LPSD probabilistic risk assessment is because the IAB is not credited for reducing the frequency of a spurious ECCS valve opening in POS1 or POS6, which increases the initiating event frequency for those POSs by several orders of magnitude.

The significant LRF sequences involve an unisolated CVCS pipe break outside containment in POS1, POS6, or POS7, followed by a series of failures that prevent the CVCS or CFDS from injecting coolant into the CNV. These sequences are not significant in POS2 and POS5 due to the short duration of applicability for these initiating events.

Module drop sequences are significant only in POS3 and POS5.

Significant Cutsets

Table 19.1-71 provides the top ten CDF and LRF cutsets for internal initiating events during LPSD operating modes. The cutsets for internal event initiators indicate that risk is distributed over a range of initiators and random failures. The cutsets associated with module drop contain very few events, reflecting the fact that mitigating actions are not credited.

Risk Significance

- the mean value of the CDF of a NuScale module is 3.0E-10/mcyr as compared to the CDF safety goal of 1.0 E-4 per reactor year.
  - The ATWS contribution to CDF is 2.2E-11/mcyr, significantly less than the target of 1.0E-5 per reactor year provided in SECY 83-293.
  - With regard to a multi-module configuration, the MM-CDF is about 10 percent of the CDF.
- the mean value large release frequency of a NuScale module, LRF, is 2.1E-11/mcyr as compared to the LRF safety goal of 1.0 E-6 per reactor year.
  - With regard to a multi-module configuration, the MM-LRF is less than 10 percent of the LRF.
- the composite CCFP of a module is less than the safety goal of 0.1.
- the evaluated external events (seismic, internal fire, internal flood, external flood, and high winds) do not pose a significant risk to the plant.

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The CDF and LRF risk metrics illustrate conformance with the quantitative health objectives (QHOs) defined in Reference 19.1-36. Conformance with the prompt fatality QHO is illustrated by an LRF that is well below the surrogate risk metric of less than 1×10^{-6} per reactor year. Similarly, risk results show that NuScale demonstrates conformance with the latent cancer QHO as illustrated by a CDF that is well below the surrogate metric of less than 1×10^{-4} per reactor year.

COL Item 19.1-8: A COL applicant that references the NuScale Power Plant design certification will confirm the applicability of assumptions and data and modify as necessary for the to the as-built/as-operated probabilistic risk assessment.

19.1.9.2 Perspective of the NuScale Small Core with Respect to Safety Goals

The safety goals are independent of design, thus the size of the potential radionuclide source term is not considered in the core damage or large release frequency safety goals. These goals are surrogates for potential public health consequences. With regard to potential consequences, an additional insight into the significance of a core damage event can be gained by considering the small NuScale radionuclide source term.

As a small reactor, the potential radionuclide source term associated with a severe accident is much smaller than that associated with typical currently operating and large advanced plant designs, e.g., the source term is five percent of that associated with a 1000 MWe design. Even the postulate of severe accidents occurring in all modules would produce a source term that is only a fraction of that associated with a larger design. Thus, while the risk to public health and safety is small as evidenced by the very low calculated CDF, LRF and CCFP risk metrics, the risk of operating a NuScale plant is further reduced because of the small potential radionuclide source term.

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.

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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.