RAIO-0119-64116

January 11, 2019

Docket No. 52-048

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

SUBJECT: NuScale Power, LLC Supplemental Response to NRC Request for Additional Information No. 489 (eRAI No. 9534) on the NuScale Design Certification Application

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 489 (eRAI No. 9534)," dated June 15, 2018
2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 489 (eRAI No.9534)," dated August 30, 2018
3. NuScale Power, LLC Supplemental Response to "NRC Request for Additional Information No.489 (eRAI No. 9534)" dated October 2, 2018

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. 9534:

- 06.04-4

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 Carrie Fosaaen at 541-452-7126 or at cfosaaen@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
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Enclosure 1: NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9534

Enclosure 1:

NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9534

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

eRAI No.: 9534
Date of RAI Issue: 06/15/2018

NRC Question No.: 06.04-4

Regulatory Basis:

10 CFR 52.47(a)(2) requires that a standard design certification application include a final safety analysis report (FSAR) that describes the design of the facility including the principal design criteria for the facility, for which NuScale used the 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

General Design Criterion (GDC) 19 requires that a control room be provided with adequate radiation protection to permit access and occupancy of the control room under accident conditions without the personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

Question:

The applicant's response to request for information (RAI) 9079 discusses the basis for the dose analysis modeling assumption that after 72 hours (after the control room habitability system (CRHS) is exhausted) the normal control room heating ventilation and air conditioning system (CRVS) is assumed to operate in supplemental filtration mode. The discussion of the CRVS reliability and operability is mainly focused on isolation and filtration component capabilities and backup power. The RAI response did not discuss augmented quality with respect to the capability to recover the CRVS for reasons other than loss of power. NuScale does not consider failure of the CRVS to operate post-72 hours concurrent with a design basis accident (DBA) to be within the design basis for evaluation of the radiological consequences of DBAs. However, if recovery of the CRVS supplemental filtration mode capability within 72 hours is not sufficiently and reliably shown to be ensured for accident conditions, then the FSAR Chapter 15 dose analysis assumptions on filtration and removal of radioactive material in the control room

ventilation intake are not justified and the dose results may exceed the dose criterion of GDC 19.

Considering that the NuScale FSAR does not include technical specifications for the CRVS, specific testing and inspection requirements for the CRVS are left to the combined license (COL) applicant (COL Item 9.4-1), and the CRVS is not classified as Seismic Category I except for the components that isolate the control room, the staff requires additional information regarding the CRVS supplemental filtration capability to limit dose to control room operators under accident conditions. Specifically, the staff requests the following information in order to complete its review by fully evaluating the importance of the post-72 hours operation of the CRVS supplemental filtration mode on the NuScale design ability to meet the requirements of GDC 19:

Provide a sensitivity analysis, including both a qualitative and quantitative assessment, evaluating the effect on the control room operator dose for DBAs for the case where after the CRHS is exhausted, the CRVS supplemental filtration mode is not recovered within 72 hours as assumed in the DBA control room dose analyses described in FSAR Chapter 15.0.3. Describe the analysis assumptions and inputs, as well as the dose results. For this sensitivity case, would the GDC 19 dose criterion of 5 rem TEDE be met for all DBAs without credit for CRVS filtration after the CRHS is exhausted?

NuScale Response:

Background:

NuScale agreed to submit a supplemental response during a meeting on September 9, 2018 to address the following NRC requests for clarification regarding sensitivity analysis provided in the original response.

- Clarify the assumptions used for the source term sensitivity analysis.
- Clarify the following sentence in the response, “It is further noted that the calculation revisions necessitated by this sensitivity study incorporated updated primary coolant source term input to the applicable steam generator tube failure, main steam line break, and small line break DBA evaluations.”
- Clarify the assumptions used for the 3 MHA cases. For example, what was used for the aerosol deposition coefficients, and also, what coolant activity concentration were used, and what were the iodine spiking values?

The original response was supplemented to reflect these clarifications (reference RAIO-1018-62021, October 2, 2018).

In a subsequent meeting on November 13, 2018, NuScale agreed to provide follow-on information and evaluate the need for a second supplemental response to address the NRC request for clarification regarding the inleakage assumption of 5 cfm applied in the sensitivity analyses provided in the original response.

Specifically, NuScale was requested to provide justification for the assumed total non-CRHS air intake of 5 cfm ingress/egress inleakage for the sensitivity analyses in comparison with the minimum breathing air supply flow rate value of 80 cfm prescribed in FSAR Table 6.4-1 in order to maintain acceptable levels of CO_2 in the control room.

NuScale investigated specifics of the airflow assumption including use of the minimum breathing air supply flow rate value of 80 cfm for breathability listed in FSAR Table 6.4-1 and compared this value with other control room habitability CO_2 concentration estimates. It is noted that 80 cfm is the bounding required air supply value, which is based on Equation 3-1 of ASHRAE 62.1, and assumes 20 personnel in the control room envelope (CRE).

In comparison, information on control room habitability from NUREG-0800 SRP 6.4, page 6.4-10, 2, "Control Room Personnel Capacity", states that "…The air inside a 2,830 m^3 (100,000 ft^3) control room would support five persons for at least 6 days. This CO_2 buildup in an isolated emergency zone is not normally considered a limiting problem."

Therefore the required ventilation rate for 20 personnel in the NuScale CRE could alternatively be determined as follows:

$$V_0 = \frac{100{,}000\ ft^3}{5\ people \cdot 6\ days \cdot 24 \frac{hr}{day}} = 138.9 \frac{ft^3}{person - hr}$$

$$V_{total} = 20\ people \cdot \frac{138.9\ ft^3}{person - hr \cdot 60 \frac{min}{hr}} = 46.3 \frac{ft^3}{min}$$

This SRP-guidance-based calculation is a better estimate of minimum air supply value than the 80 cfm value which is developed for the purpose of a bounding CRHS performance specification.

A maximum occupancy time until unsafe CO_2 buildup for the NuScale control room can be determined using the NuScale design-specific control room volume of 74,680 ft^3 and the SRP-based minimum breathable air requirement as follows:

$$maximum\ occupancy\ time\ (days) = \frac{74{,}680\ ft^3}{20\ people \cdot 138.9 \frac{ft^3}{person - hr} \cdot 24 \frac{hr}{day}} = 1.1\ days$$

Given these derived values, an alternative inleakage scenario can be assumed wherein after 1.1 days of 5 cfm inleakage/exhaust due to ingress/egress, and otherwise stagnant airflow, the trip of a CO_2 high-level alarm prompts a nonspecific action, such as propping open a door, to restore 46.3 cfm of unfiltered airflow to maintain acceptable levels of CO_2 in the control room.

However, it is noted that an alternative inleakage flow rate and activation time such as the one discussed here does not have a specific attributable technical basis. It is also noted that in considering alternative airflow sources to support habitability for beyond-design-basis conditions, one might in kind assume nonspecific operator action is taken to restore an alternative source of clean, uncontaminated air to maintain acceptable levels of CO_2 in the control room for an accident duration, e.g., filtered airflow from the door or bringing in outside clean bottled air.

Therefore, NuScale asserts the 5 cfm inleakage to the CRE constitutes a credible scenario for operator dose evaluation and is a reasonable and appropriate assumption as applied in the provided sensitivity analyses for the following reasons:

- The sensitivity analyses provided are beyond-design-basis scenarios, i.e., considered to already transcend credibility notwithstanding the assumed CRE air inflow.
- Consideration of an alternative inflow of unspecified origin would represent an additional layer of beyond-design-basis perturbation crediting operator action to a beyond-design-basis sensitivity scenario which is a level of modeling complexity unnecessary to demonstrate safety.
- Similar operator physical limitations to that of CO_2 toxicity are not typically addressed in precedence, e.g., evaluating for sufficient drinking water.
- Operator action could be taken in the event of uninhabitable temperatures or atmospheric conditions to preclude the need for supplemental airflow in a beyond-design-basis sensitivity scenario similar to actions described in the response to eRAI 9079/06.04-1 (reference RAIO-1017-56676, October 18, 2017) *"As added protection against overexposure to radiation, the control room is equipped with area radiation*

monitors. If the radiation level should exceed preset limits, which will be determined by the licensee in accordance with their radiation protection program, the operators would trip any operating reactors, initiate decay heat removal and containment isolation, and vacate the control room. Each reactor module would then reach safe shutdown conditions without operator action."

Impact on DCA:

There are no impacts to the DCA as a result of this response.