April 17, 2019

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 Supplemental Response to NRC Request for Additional Information No. 350 (eRAI No. 9278) on the NuScale Design Certification Application

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 350 (eRAI No. 9278)," dated January 29, 2018
2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 350 (eRAI No.9278)," dated May 16, 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 Questions from NRC eRAI No. 9278:

- 12.03-31
- 12.03-32
- 12.03-33
- 12.03-34
- 12.03-35
- 12.03-36
- 12.03-37
- 12.03-39
- 12.03-40

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
NuScale Power, LLC

Distribution: Gregory Cranston, NRC, OWFN-8H12
Samuel Lee, NRC, OWFN-8H12
Getachew Tesfaye, NRC, OWFN-8H12

Enclosure 1: NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9278

Enclosure 1:

NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9278

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

eRAI No.: 9278

Date of RAI Issue: 01/29/2018

NRC Question No.: 12.03-31

Regulatory Basis

10 CFR 52.47(a)(8) requires that the final safety analysis report provide the information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), except paragraphs (f)(1)(xii), (f)(2)(ix), and (f)(3)(v).

10 CFR 50.34(f)(2)(vii) requires that applicants perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain accident source term radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment.

10 CFR 50.34(f)(2)(viii) requires that applicants provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain accident source term radioactive materials without radiation exposures to any individual exceeding 5 rems to the whole body or 50 rems to the extremities.

10 CFR 50.49 and 10 CFR Part 50, Appendix A, Criterion 4 require that certain components important to safety be designed to withstand environmental conditions, including the effects of radiation, associated with design basis events, including normal operation, anticipated operational occurrences, and design basis accidents.

NUREG-0737 and DSRS section 12.3-12.4 provide additional guidance on acceptable methods of meeting these requirements. These documents indicate that post accident radiation zones should consider access to, stay time in, and egress from these vital areas. NUREG-0737 specifies that any area which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident is to be designated as a vital area. RAI-8775 Question 12.03-1 focused on the sample station, sample analysis, and other areas requiring infrequent access. NUREG-0737 provides a list of other areas that should be considered in determining the vital areas and stipulates that if these areas are not considered vital areas, justification should be provided for not including them. Areas specified include the containment isolation reset control area, motor control centers, instrument panels, emergency power supplies, and radwaste control panels. In addition, any other areas that may need to be accessed during an accident are to be identified. As specified, the plant should be designed so that the dose to an individual should not exceed the occupational dose criteria to perform the vital missions, including accessing and egressing from the areas.

Background

The applicant's response to RAI-8775 Question 12.03-1, dated June 26h 2017, provided a new proposed DCD Table 12.4-8: "Post- Accident Sampling Operator Dose," which provides a time line and estimated dose for steps of the sampling process. The response provided proposed Figures 12.3-4a-12.3-4c depicting reactor building post-accident radiation zone maps for the 50', 75' and 100' elevations. The response also proposed changes to DCD Section 12.4.1.8 "Post-Accident Actions."

Key Issue 1:

DCD Tier 2 Section 9.3.2.2.3 "System Operation," states that the sample line purge fluid may be collected in a temporary disposal tank if radiation levels are expected to be too high for disposal to the LRWS. Proposed Table 12.4-8: "Post-Accident Sampling Operator Dose," states that a purge collection tank may be used to collect water from the sample line. The table also states that ¼" of temporary lead equivalent shielding material may be staged to reduce dose rates during sampling. Based on need to purge multiple line volumes in order to ensure a representative sample is obtained, and operating experience from Three Mile Island (TMI) regarding the specific activity of reactor coolant system (RCS) fluids following an accident, the temporary surge tank is a significant source of radiation exposure that is not reflected in the post-accident dose calculations.

Question 1

To facilitate staff understanding of the application information sufficient to make appropriate regulatory conclusions, with respect to estimated radiation exposure for accessing areas following an accident, the staff requests that the applicant:

- As necessary, revise proposed Table 12.4-8: “Post-Accident Sampling Operator Dose,” to include the dose from temporary purge collection tank, or provide the specific alternative approaches used and the associated justification.

OR

Provide the specific alternative approaches used and the associated justification

---

NuScale Response:

On January 31, 2019, NuScale submitted a request to be exempted from the dose analysis aspect of 10 CFR 50.34(f)(2)(viii), as described in DCA Part 7, Section 16. Therefore, based on this exemption request, this supplemental RAI response replaces the previous RAI response.

This RAI requested that additional details be provided regarding the NuScale analyses related to operator dose during post-accident sampling activities, and other areas that may need to be accessed following an accident. Given the NuScale exemption request in DCA Part 7, Section 16, such requested information is not required. Also, as described in the original response to this RAI, there are no areas outside of the main control room that require operator access post-accident in the NuScale design.

For the associated DCA changes, please see NuScale letter LO-0319-65027, dated March 29, 2019.

Impact on DCA:

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

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

eRAI No.: 9278
Date of RAI Issue: 01/29/2018

NRC Question No.: 12.03-32

The Regulatory Basis and Background are in RAI-9278 Question 31017

Question 2
To facilitate staff understanding of the application information sufficient to make appropriate regulatory conclusions, with respect to radiation levels in areas requiring access following an accident, the staff requests that the applicant:

- As necessary, revise DCD Section 12.3 to include figures depicting post-accident radiation zone maps of areas containing pipes or components that may contain highly radioactive fluid resulting from sampling activities;

And,

- Explain/Justify any changes needed to support the qualification of equipment (e.g., DCD Table 3C-1),
- As necessary, revise the DCD to include the description of the changes needed to support the qualification of equipment (e.g., DCD Table 3C-1),

Or,

Provide the specific alternative approaches used and the associated justification.

NuScale Response:

See the NuScale supplemental response to RAI 12.03-31.

Impact on DCA:

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

NUSCALE
Power for all humankind

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

eRAI No.: 9278
Date of RAI Issue: 01/29/2018

NRC Question No.: 12.03-33

The Regulatory Basis and Background are in RAI-9278 Question 31017

Key Issue 3:

Proposed changes to DCD Section 12.4.1.8 “Post-Accident Actions,” states the operator's exposure to airborne activity was considered as part of the dose evaluation. Post-accident airborne activity is generated from containment leakage into the Reactor Building atmosphere. The air space above the reactor pool is isolated from the other Reactor Building air spaces, such that the areas accessed by operators to perform sampling activities are not subjected to post-accident airborne contamination. However, the staff was unable to identify any design features on Figure 12.3-4c: “Reactor Building Post-Accident Radiation Zone Map - 100' Elevation,” that would serve to isolate the air space in this area of the facility from the air spaces in other portions of the facility.

Question 3

To facilitate staff understanding of the application information sufficient to make appropriate regulatory conclusions, with respect to airborne radioactivity levels in areas requiring access following an accident, the staff requests that the applicant:

- Explain/Justify the design features relied upon to provide isolation of the air space above the ultimate heat sink pool from other areas of the Reactor Building (RXB)
- As necessary, revise DCD Section 12.3 to include a description of those design features relied upon to provide isolation of the air space above the ultimate heat sink pool from other areas of the Reactor Building (RXB),

OR

Provide the specific alternative approaches used and the associated justification.

NuScale Response:

See the NuScale supplemental response to RAI 12.03-31.

Impact on DCA:

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

NUSCALE™
Power for all humankind

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

eRAI No.: 9278
Date of RAI Issue: 01/29/2018

NRC Question No.: 12.03-34

The Regulatory Basis and Background are in RAI-9278 Question 31017

Key Issue 4:
Proposed changes to DCD Section 12.4.1.8 “Post-Accident Actions,” states the operator's exposure to airborne activity was considered as part of the dose evaluation. Post-accident airborne activity is generated from containment leakage into the Reactor Building atmosphere. However, the response does not address potential airborne activity and the resultant personnel exposure, resulting from the purging and sampling activities.

Question 4
To facilitate staff understanding of the application information sufficient to make appropriate regulatory conclusions, with respect to airborne radioactivity levels resulting from taking samples following an accident, the staff requests that the applicant:

- Explain/Justify the methods, models and assumptions used to determine the exposure to the operator from airborne radioactive material during resulting from sample purging and acquisition, following an accident,
- As necessary, revise DCD Section 12.3 to include the assessed dose resulting from airborne activity radioactive material evolving from purging or sampling fluids,

OR

Provide the specific alternative approaches used and the associated justification.

NuScale Response:

See the NuScale supplemental response to RAI 12.03-31.

Impact on DCA:

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

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

eRAI No.: 9278

Date of RAI Issue: 01/29/2018

NRC Question No.: 12.03-35

The Regulatory Basis and Background are in RAI-9278 Question 31017

Key Issue 5:

DCD Tier 2 Revision 0 Section 9.3.2.2.3 "System Operation," states that the post-accident water level in the reactor vessel for some events may be below the level of the normal CVCS reactor vessel nozzle (for reactor coolant discharge to CVCS). The reactor coolant post-accident sample is collected via the CVCS sample line flow path to the primary sampling system sample panel. DCD Section 9.3.2.2.3 also states that RCS pressure is used, as in normal operation, to force the RCS sample to the sample panel. At temperatures below 200 degrees Fahrenheit with insufficient RCS pressure, a nitrogen overpressure can be established. However, the proposed changes to DCD Table 12.4-8: "Post-Accident Sampling Operator Dose," do not include dose estimates for the activities of adding water to the reactor, or raising the pressure of the containment vessel.

Question 5

To facilitate staff understanding of the application information sufficient to make appropriate regulatory conclusions, with respect to personnel exposure to radiation resulting from taking samples following an accident, the staff requests that the applicant:

- Explain/Justify the methods, models and assumptions used to assess the dose resulting from adding water to the reactor and raising containment pressure, as needed, to support sample acquisition following an accident,
- As necessary, revise DCD Table 12.4-8: "Post-Accident Sampling Operator Dose," to include the assessed dose resulting from adding water to the reactor, and raising containment pressure,

OR

Provide the specific alternative approaches used and the associated justification.

NuScale Response:

See the NuScale supplemental response to RAI 12.03-31.

Impact on DCA:

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

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

eRAI No.: 9278
Date of RAI Issue: 01/29/2018

NRC Question No.: 12.03-36

The Regulatory Basis and Background are in RAI-9278 Question 31017

Key Issue 6:
Based on information made available to the staff during the RPAC NuScale Chapter 12 Audit, the staff became aware that the methodology used to develop the photon source strength from the post-accident fluid did not account for some principle radiation emitting isotopes in the fluid stream. For instance, Ba-137m is in secular equilibrium with the parent Cs-137 radionuclide, the specific activity of Ba-137m should be within 94 percent of the Cs-137 specific activity, within 20 minutes. However, the information reviewed by the staff indicated that the source term used by the applicant to perform the analysis of dose resulting from the sample fluid, did not properly account for Ba-137m, thus resulting in an underestimation of the dose rate from the sample fluid.

Question 6

To facilitate staff understanding of the application information sufficient to make appropriate regulatory conclusions, with respect to personnel exposure to radiation resulting from taking samples following an accident, the staff requests that the applicant:

- Explain/Justify the methods, models and assumptions used to determine the dose rates from systems, structures and components containing reactor coolant fluids following an accident,
- As necessary, revise DCD Table 12.4-8: “Post-Accident Sampling Operator Dose,” to describe the necessary assumptions used to reflect the correct source term specific activity, and the resultant dose rates and the doses to operator(s),

OR

Provide the specific alternative approaches used and the associated justification.

NuScale Response:

See the NuScale supplemental response to RAI 12.03-31.

Impact on DCA:

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

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

eRAI No.: 9278
Date of RAI Issue: 01/29/2018

NRC Question No.: 12.03-37

The Regulatory Basis and Background are in RAI-9278 Question 31017

Key Issue 7:
The dose calculations performed to support the specific task dose information provided in proposed DCD Table 12.4-8: "Post- Accident Sampling Operator Dose," assumed that fluid contained within the CVCS heat exchangers was at the normal RCS fluid specific activity. However, because DCD Tier 2 Revision 0 Figure 9.3.4-1: "Chemical and Volume Control System Diagram," does not show any design features to prevent flow through the CVCS heat exchanger once the containment isolation valves are opened, and proposed DCD Table 12.4-8: "Post-Accident Sampling Operator Dose," does not include an activity to prevent flow of post-accident radioactive fluid through the CVCS heat exchangers, the staff is unable to determine if the dose estimate should include dose from the CVCS heat exchangers containing post-accident fluids.

Question 7

To facilitate staff understanding of the application information sufficient to make appropriate regulatory conclusions, with respect to personnel exposure to radiation resulting from taking samples following an accident, the staff requests that the applicant:

- Explain/Justify the methods, models and assumptions used for determining the radiation exposure to operators from SSCs containing reactor coolant system fluids following an accident,
- As necessary, revise the proposed DCD Table 12.4-8: "Post-Accident Sampling Operator Dose," dose rates and the resultant operator doses, to reflect the activities to prevent flow of post-accident fluids through the CVCS heat exchanger,

OR

- As necessary, revise proposed DCD Table 12.4-8 to include dose from the CVCS heat exchanger containing post- accident fluid,

OR

Provide the specific alternative approaches used and the associated justification.

NuScale Response:

See the NuScale supplemental response to RAI 12.03-31.

Impact on DCA:

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

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

eRAI No.: 9278
Date of RAI Issue: 01/29/2018

NRC Question No.: 12.03-39

The Regulatory Basis and Background are in RAI-9278 Question 31017

Key Issue 9:
DCD Tier 2 Revision 0 Section 9.3.2.2.3 “System Operation,” and proposed DCD Section 12.4.1.8 “Post-Accident Actions,” indicate that the post-accident primary coolant sample is collected via the normal CVCS sample line flow path to the primary system sample at a panel located in the CVCS gallery, but appears to indicate that the sample does not flow through the normal Plant Sampling System (PSS) sample panel. Because DCD Section 9.3.2.2.2 “Component Description,” discusses first and second stage cooling for samples streams over approximately 100 degrees Fahrenheit but also notes that second stage cooling is provided when the sample stream is directed to an analysis panel. Proposed DCD Table 12.4-8: “Post-Accident Sampling Operator Dose,” does not include expected dose from post-accident fluid contained in sample coolers.

Question 9

To facilitate staff understanding of the application information sufficient to make appropriate regulatory conclusions, with respect to design features protect personnel, and reduce personnel exposure to radiation to personnel, resulting from taking samples following an accident, the staff requests that the applicant:

- Explain/Justify the methods and assumptions used to determine the conditions (e.g., temperature, pressure, volatile gas content,) of reactor coolant system fluids present at the sampling point, following an accident,
- As necessary, revise the DCD to include a description of how post-accident sample streams that could potentially be over 100 degrees Fahrenheit will be cooled to a low

enough temperature to allow for safe purge volume collection, and sample collection while minimizing airborne radioactive material release from vaporization;

And,

- As necessary, revise the proposed DCD Table 12.4-8: “Post-Accident Sampling Operator Dose,” dose rates, and the resultant doses, to reflect the dose from the post-accident fluids in the sample cooler(s),

OR

Provide the specific alternative approaches used and the associated justification.

NuScale Response:

See the NuScale supplemental response to RAI 12.03-31.

Impact on DCA:

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

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

eRAI No.: 9278
Date of RAI Issue: 01/29/2018

NRC Question No.: 12.03-40

The Regulatory Basis and Background are in RAI-9278 Question 31017

Key Issue 10:
DCD Tier 2 Revision 0 Section 9.3.2.2.3 “System Operation,” and proposed DCD Section 12.4.1.8 “Post-Accident Actions,” states that the sample line purge volume may be directed to the liquid radioactive waste system (LWRS). DCD Figure 9.3.3-1:
“Radioactive Waste Drain System Diagram,” shows that the drains in the Reactor Building (RXB) go to sumps 41 A/B – 46 A/B. As noted in DCD Section 9.3.2.2.3, additional operator actions, such as raising containment pressure, may be required to allow sampling. Also, as noted in DCD Section 9.3.2.2.3, sample fluids greater than 100 degrees Fahrenheit may require cooling. There is no discussion in DCD Section 9.3.2.2.3 or the response to RAI-8775 Question 12.03-1 that addresses any actions needed to assure proper operation of the these systems during post-accident conditions. As noted above, NUREG-0737 provides a list of other areas that should be considered in determining the vital areas and stipulates that if these areas are not considered vital areas, justification should be provided for not including them.

Question 10

To facilitate staff understanding of the application information sufficient to make appropriate regulatory conclusions, with respect to the levels of radiation that may be present in the facility resulting from taking samples of reactor coolant system fluids following an accident, the staff requests that the applicant:

- Explain/Justify the methods, models and assumptions used to assess the radiological impact on area dose rates, in other areas of the plant (e.g., radioactive waste collection systems,) resulting from acquiring samples of reactor coolant system fluids following an accident,

- As necessary, revise proposed DCD Table 12.4-8: “Post-Accident Sampling Operator Dose,” dose rates, and the resultant doses, to reflect the dose from accessing those areas,

Or,

Provide the specific alternative approaches used and the associated justification.

NuScale Response:

See the NuScale supplemental response to RAI 12.03-31.

Impact on DCA:

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