NuScale Power

RAIO-0817-55366

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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. 93 (eRAI No. 8897) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 93 (eRAI No. 8897)," dated July 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. 8897:

- 15.00.03-7

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
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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 8897

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NuScale Response to NRC Request for Additional Information eRAI No. 8897

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

eRAI No.: 8897
Date of RAI Issue: 07/13/2017

NRC Question No.: 15.00.03-7

Requirements for technical support center (TSC) occupancy and habitability during accidents are provided in 10 CFR 50.47(b)(8) and (b)(11), and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. NuScale design-specific review standard (DSRS) section 15.0.3 provides additional guidance on the evaluation of TSC radiological habitability, including dose acceptance criteria for dose to TSC occupants. The design basis accident (DBA) dose analyses in DCD Tier 2 Chapter 15 were performed, in part, to show compliance with the TSC habitability requirements.

During the audit of the applicant's DBA dose calculations, the staff noted that the technical support center (TSC) was modeled using different analysis inputs than were used for the main control room. Please provide the modeling assumptions for the TSC as used in the DBA dose analyses, including revisions to DCD text to clearly document the basis for the analyses, similar to the information provided in the response to RAI letter No. 13, RAI 8774, Question 15.00.03-2 (ADAMS Accession No. ML17144A451) related to the main control room model.

NuScale Response:

FSAR Section 15.0.3.7.2, Technical Support Center Design, has been added along with Table 15.0-18, Technical Support Center Parameters. These provide the modeling assumptions and parameters used in the DBA dose analyses.

Note, the previous Section 15.0.3.7.2, Reactor Building Pool Boiling Radiological Consequences, has been renumbered to Section 15.0.3.7.3.

Impact on DCA:

FSAR Section 15.0.3.7.2 and Table 15.0-18 have been revised as described in the response above and as shown in the markup provided in this response.

15.0.3.3.13 Dose Conversion Factors

Consistent with RG 1.183, dose conversion factors from Environmental Protection Agency Federal Guidance Report No. 11 (Reference 15.0-8) and Report No. 12 (Reference 15.0-9) are used for dose analysis.

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15.0.3.4 Containment Leakage

Containment leakage is described in Reference 15.0-4 ~~and~~, is consistent with the recommendations of RG 1.183, and is listed in Table 12.2-28.

15.0.3.5 Secondary-Side Decontamination

The helical coil steam generators of the NuScale Power Plant design are different than that of a large PWR because the primary coolant is on the outside of the tubes. As a result, there is no bulk water volume in which decontamination can easily occur. Reference 15.0-4 provides the details about the decontamination factor used in the helical coil steam generators as well as the treatment of iodine deposition in the main steam piping and the condenser.

15.0.3.6 Reactor Building Decontamination Factors

Reactor Building RXB decontamination factors are described in Reference 15.0-4.

15.0.3.7 Receptor Location Considerations

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Potential on-site radiological receptor locations considered in this evaluation are the control room and TSC; potential off-site locations are the EAB and LPZ. Figure 15.0-3 shows the schematic of the RADTRAD code nodalization used to model these locations for leakage paths from the containment or RXB. Figure 15.0-4 shows the RADTRAD code nodalization for the SGTF and MSLB events in which the principal release path is through the steam generator.

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A summary of control room and TSC characteristics are provided in ~~Reference 15.0-4~~Section 15.0.3.7.1 and Section 15.0.3.7.2, respectively. The variables associated with the derivation of these receptors are presented in Table 15.0-13.

RAI 15.00.03-2

15.0.3.7.1 Control Room Design

Accident analyses are performed for two control room emergency modes as follows:

- Uninterrupted power supply with continuous filtered airflow to the control room envelope for the event duration.

- Immediate loss of power with control room habitability system (CRHS) activation, and restored filtered airflow to the control room envelope at the time of CRHS depletion (72 hours).

RAI 15.00.03-2

Simplifying assumptions are made for the control room ventilation system design. Figure 15.0-3 and Figure 15.0-4 show the control room RADTRAD code nodalization used in the dose analyses. The key design features assumed for the control room are summarized as follows:

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- The nonsafety-related normal control room ventilation system inlet filters remove 99 percent of iodine. Although recirculation flow as shown in Figure 15.0-3 and Figure 15.0-4 indicate a filter, it is conservatively assumed that recirculation flow is unfiltered in the accident evaluations.
- The nonsafety-related normal control room ventilation is isolated by a safety-related control system once the radioactivity measured at the duct intake reaches the isolation signal setpoint. The setpoint for the radiation monitor to redirect air through the air filtration unit is 10-times background. The setpoint for CRHS initiation and CRE isolation is 10-times the expected radiation out of the filtration unit following a DBE, which indicates a failure of the filtration unit to remove sufficient radioactivity. The time between when the radiation concentration reaches the detector setpoint and radiation enters the control room or technical support center (TSC) envelopes is assumed to be zero seconds. The electric signal travels much faster than air, thus the mechanical dampers will be closed before the radiation enters the control room or TSC envelopes.
- An emergency source of pressurized air with the control room habitability system (CRHS) provides clean air for 72 hours.
- After 72 hours of CRHS operation, the normal control room ventilation system is available for use.
- The control room is habitable during a loss of normal AC power as the CRHS automatically activates after 10 minutes without normal AC power, as described in Section 6.4.3.
- Control room ventilation is designed to minimize in-leakage.

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- The control room is designed with a two-door air lock system. Therefore, in-leakage of 5 cfm is assumed for ingress and egress. An additional ~~10~~147~~-~~cfm of in-leakage is also assumed.

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The control room ventilation system design modeling assumptions are provided in ~~Reference 15.0-4~~Table 15.0-15. Details about system operation with CRHS are provided in Section 6.4 and Section 9.4.1.

No credit is taken for the use of personal protective equipment, such as beta radiation resistant protective clothing, eye protection, or self-contained breathing apparatus. No credit is taken for prophylactic drugs such as potassium iodide pills.

Potential shine radiological exposures to operators within the control room following a radiological release event are evaluated. Direct shine, sky-shine and shine from filters are evaluated using MCNP, as described in Section 15.0.2.4.7. Reference 15.0-4 provides additional details regarding the calculation of shine doses. The 30-day cumulative doses due to either recirculation filter or cloud-shine in the control room are added to the dose results from DBEs provided in Table 15.0-12.

Shine doses are well below the regulatory limit of 5 rem because of the heavy shielding provided by the wall and floors of the Control Building.

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15.0.3.7.2 Technical Support Center Design

Accident analyses are performed for one emergency mode: that of uninterrupted power supply with continuous filtered airflow to the Technical Support Center (TSC) envelope for the event duration. In the event of immediate loss of power with control room habitability system (CRHS) activation, TSC personnel are evacuated and the TSC function is transferred to an alternate site-specific location. With loss of power with CRHS activation, the TSC is evacuated since it is not serviced by the CRHS.

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The key design features assumed for the technical support center are summarized as follows:

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- The nonsafety-related normal TSC ventilation system filters remove 99 percent of iodine under accident conditions. Although the recirculation flow as shown in Figure 15.0-3 and Figure 15.0-4 indicate a filter, it is conservatively assumed that recirculation flow is unfiltered in the accident evaluations.

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- The nonsafety-related normal TSC ventilation is isolated by a safety-related control system once the radioactivity measured at the duct intake reaches the isolation signal setpoint. The setpoint for the radiation monitor to redirect air through the air filtration unit is 10-times background. The setpoint for CRHS initiation and CRE isolation is 10-times the expected radiation out of the filtration unit following a DBE, which indicates a failure of the filtration unit to remove sufficient radioactivity. The time between when the radiation concentration reaches the detector setpoint and radiation technical support center (TSC) envelopes is assumed to be zero seconds. The electric signal travels much faster than air, thus the mechanical dampers will be closed before the radiation enters the control room or TSC envelopes.

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- 10-cfm of in-leakage is assumed for ingress and egress. An additional 56 cfm of in-leakage is also assumed.

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The technical support center ventilation system design modeling assumptions are provided in Table 15.0-18.

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No credit is taken for the use of personal protective equipment, such as beta radiation resistant protective clothing, eye protection, or self-contained breathing apparatus. No credit is taken for prophylactic drugs such as potassium iodide pills.

RAI 15.00.03-7

Potential shine radiological exposures to operators within the TSC following a radiological release event are evaluated. Direct shine, sky-shine and shine from filters are evaluated using MCNP, as described in Section 15.0.2.4.7. Reference 15.0-4 provides additional details regarding the calculation of shine doses.

15.0.3.7.3 Reactor Building Pool Boiling Radiological Consequences

Without available power, decay heat from the reactors and spent fuel would heat the pool water and could eventually cause the reactor pool to boil. Table 9.2.5-2, in Section 9.2.5 shows that it takes longer than 72 hours for the pool to reach boiling after a loss of normal AC power event. However, if the pool were to boil, the dose would be less than 0.5 rem TEDE.

15.0.3.8 Consequence Analyses of Category 1 Events

15.0.3.8.1 Failure of Small Lines Carrying Primary Coolant Outside Containment

Failure of small lines carrying primary coolant outside containment is not an event addressed in RG 1.183. The methodology used for determining dose consequences, including the iodine spiking assumptions for this event, is similar to that used for the MSLB and SGTF. The event-specific transient analysis described in Section 15.6.2 defines the time-dependent release of activity into the RXB.

The small-line break outside containment can be a break in the chemical and volume control system (CVCS) letdown line or makeup line, or the pressurizer spray line. A non-mechanistic line break occurs in the RXB allowing primary coolant from the reactor to be released into the RXB. In addition, primary coolant in the CVCS equipment (heat exchangers, filters, etc.) and piping within the RXB flows out of the other side of the break contributing less than 15,000 lbm additional primary coolant to the release. The limiting radiological scenarios identified in Section 15.6.2 are:

- maximum mass release - double-ended break of the CVCS letdown line
- maximum time of iodine spiking - equivalent 100 percent cross-sectional area break of the CVCS makeup line

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Table 15.0-18: Technical Support Center Parameters

Parameter	Units	Value
TSC Normal Flow Rate	cfm	659
TSC Recirculation Flow Rate	cfm	11,891
TSC Unfiltered Ingress/Egress	cfm	10
TSC Unfiltered Inleakage	cfm	56
TSC Volume	ft^3	85,614