RAIO-0619-65974

June 17, 2019

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Supplemental Response to NRC Request for Additional Information No. 325 (eRAI No. 9268) on the NuScale Design Certification Application

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 325 (eRAI No. 9268)," dated January 08, 2018
2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 325 (eRAI No.9268)," dated June 14, 2019

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. 9268:

- 12.02-11

This supplemental RAI response replaces the previous response, except for the Design Certification Application markups originally provided.

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
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Enclosure 1: NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9268

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Enclosure 1:

NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9268

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

eRAI No.: 9268
Date of RAI Issue: 01/08/2018

NRC Question No.: 12.02-11

Regulatory Basis

10 CFR 52.47(a)(5) requires applicants to identify the kinds and quantities of radioactive materials expected to be produced in the operation and the means for controlling and limiting radiation exposures within the limits of 10 CFR Part 20.

Appendix A to Part 50—General Design Criteria for Nuclear Power Plants, Criterion 61—"Fuel storage and handling and radioactivity control," requires systems which may contain radioactivity to be designed with suitable shielding for radiation protection and with appropriate containment, confinement, and filtering systems.

10 CFR 20.1101(b) and 10 CFR 20.1003, require the use of engineering controls to maintain exposures to radiation as far below the dose limits in 10 CFR Part 20 as is practical. NuScale DSRS section 12.2 "Radiation Source," regarding the identification of isotopes and the methods, models and assumptions used to determine dose rates. NuScale DSRS section 12.3 "Radiation Protection Design Feature," states in the specific acceptance criteria that areas inside the plant structures should be subdivided into radiation zones, with maximum design dose rate zones and the criteria used in selecting maximum dose rates identified.

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.

Background

DCD Tier 2 Revision 0 Section 12.2.1.13 “Post-Accident Sources,” discusses post-accident sources and points to several tables in DCD Section 12.2 for additional information. Table 12.2-28: “Post-Accident Source Term Input Assumptions,” identifies the assumptions used to derive the post-accident sources of radiation. DCD Table 12.2-29: “Post-Accident Core Inventory Release Fractions,” provides a listing of the radionuclide groups and the associated release fractions. DCD Table 12.2-30: “Post-Accident Containment Aerosol Removal Rates,” describes how some radionuclides are removed from the contain atmosphere following an accident. DCD Table 12.2-31: “Post-Accident Integrated Energy Deposition and Integrated Dose,” provides the Integrated MeV energy deposition, and the integrated dose at specific time intervals during post-accident conditions.

Key Issue:

Because DCD Tier 2 Revision 0 Section 12.2 does not contain a listing of the isotopic inventory (i.e., isotope identification and concentration) during post-accident conditions, the staff is unable to determine how the assumptions listed in Tables 12.2-28 and 12.2-29 have been applied. The post-accident isotopic concentrations in the containment (CNV) air volume liquid are used to determine the post-accident radiation levels in a variety of areas, including but not limited to; areas above the reactor building (RXB) pool due to shine from the air volume in the CNV, the area above the CNV but below the bioshield, areas adjacent to the bioshield subject to radiation penetrating shielding or streaming through opening. The post-accident isotopic concentrations in the reactor coolant system (RCS) liquid are used to determine the post-accident radiation levels in a variety of areas, including but not limited to; in areas with pipes containing RCS

liquids (e.g., chemical and volume control system (CVCS), Plant Sample System (PSS) and liquid radioactive waste system (LWRS).

The staff uses the calculated radiation levels to compare design features described in the DCD to the acceptance criteria in DSRS 12.2, 12.3 and 3.11.

Question

Please provide, as a revision in the DCD (Section 12.2) a listing of radionuclide concentrations in the CNV air volume for post- accident conditions, and provide in the DCD a listing of radionuclide concentrations in the RCS liquid volume for post-accident conditions.

Or,

Provide the specific alternative approaches used and the associated justification.

NuScale Response:

A revised accident source term topical report (TR-0915-17565, Rev. 3) was submitted to the NRC on April 22, 2019. This topical report describes the methodology used to develop the revised NuScale accident source term. The reduced magnitude of the revised NuScale accident source term reduces the overall magnitude of the resulting post-accident doses.

For the purposes of determining the post-accident integrated doses to equipment within the volume of interest (i.e., containment vapor space or primary coolant liquid space), the entire post-accident radionuclide inventory is assumed to be located within that volume (except that nobles gases are not assumed to be in the liquid volume), as a bounding and simplifying assumption. For equipment located within the bioshield envelope, the post-accident integrated dose includes contributions from photon shine from the CNV vapor space, immersion in the leakage from the containment vessel into the bioshield envelope, plus the largest design basis accident radionuclide release into the bioshield envelope space.

The maximum primary coolant radionuclide concentrations have been added to the NuScale FSAR in Table 12.2-34. The total radionuclide inventory in either the RCS liquid or CNV vapor can be determined using the volumes and densities provided in FSAR Table 12.2-28.

It should be noted that the equipment qualification (EQ) integrated dose is based on an activity concentration of 0.2 µCi/g DE I-131, plus a coincident iodine spike, and a crud burst. The NuScale primary coolant activity technical specification has since been revised to a lower value, as described in the supplemental response to RAI 8759 (12.02-1S1), however the radionuclide concentrations presented in Table 12.2-34 are conservatively based on the previous primary coolant activity technical specification of 0.2 µCi/g DE I-131, including a coincident iodine spike and a crud burst. The containment vapor space radionuclide concentration conservatively assumes this same liquid radionuclide inventory is entirely vaporized within the containment vapor space. No credit for aerosol deposition within the containment volume is taken in this analysis.

The radionuclide concentration in the bioshield envelope space is determined by the largest design basis accident radionuclide release into the bioshield envelope, plus subsequent containment vessel leakage into the envelope at the technical specification leak rate for the first 24 hours, and half of that for the remaining duration. This activity is assumed to remains within the envelope for the duration of the event, except for radioactive decay (i.e., no leakage, plateout or deposition, from the bioshield envelope space was assumed). This assumption was also used for earlier designs of the bioshield, which did not include open ventilation pathways. The current bioshield design does include open ventilation pathways to the Reactor Building pool room, therefore this assumption results in conservatively high dose rates, as compared to modeling the dispersion of the activity from under the bioshield. The limiting design basis accident for the radionuclide concentration within the bioshield envelope is the small line break outside containment, but under the bioshield (FSAR Section 15.6.2).

The revised FSAR markups were provided with the original RAI response. FSAR Section 12.2.1.13 and Table 12.2-28 (title only) were revised and markups were provided with the NuScale Topical Report TR-0915-17565, Revision 3 (NuScale letter # LO-0419-65280; ML19112A220).

Impact on DCA:

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