From:	Getachew Tesfaye
Sent:	Thursday, October 19, 2023 4:04 PM
To:	Request for Additional Information
Cc:	Alina Schiller; Mahmoud Jardaneh; Griffith, Thomas; Osborn, Jim; NuScale-SDA-720RAIsPEm Resource
Subject:	NuScale SDAA Section 12.3 - Request for Additional Information No. 006 (RAI-10101-R1)
Attachments:	SECTION 12.3 - RAI-10101-R1-FINAL.pdf

Attached please find NRC staff's request for additional information (RAI) concerning the review of NuScale Standard Design Approval Application for its US460 standard plant design (Agencywide Documents Access and Management System (ADAMS) Accession No. ML222339A066).

Please submit your technically correct and complete response by the agreed upon date to the NRC Document Control Desk.

If you have any questions, please do not hesitate to contact me.

Thank you,

Getachew Tesfaye (He/Him)
Senior Project Manager
NRC/NRR/DNRL/NRLB
301-415-8013

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Options
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Reply Requested: No
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REQUEST FOR ADDITIONAL INFORMATION No. 006 (RAI-10101-R1)
BY THE OFFICE OF NUCLEAR REACTOR REGULATION
NUSCALE STANDARD DESIGN APPROVAL APPLICATION
DOCKET NO. 05200050
CHAPTER 12, "RADIATION PROTECTION"
SECTION 12.3, "RADIATION PROTECTION DESIGN FEATURES"
ISSUE DATE: 10/19/2023

Background

By letter dated December 28, 2022, NuScale Power, LLC (NuScale or the applicant), submitted Part 2 – Final Safety Analysis Report (FSAR), Chapter 12, "Radiation Protection," Revision 0, (Agencywide Documents Access and Management System Accession No. ML22362A116) of the NuScale Standard Design Approval Application (SDAA) for its US460 standard plant design. The applicant submitted the US460 plant SDAA in accordance with the requirements of Title 10 *Code of Federal Regulations* (10 CFR) Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Subpart E, "Standard Design Approvals." The NRC staff has reviewed the information in Chapter 12 of the SDAA and determined that additional information is required to complete its review.

Question 12.3-1

Regulatory Basis

10 CFR Part 50, Appendix A, General Design Criteria (GDC) 61 requires in part that fuel storage and handling, radioactive waste, and other systems shall be designed with suitable shielding for radiation protection.

10 CFR 52.137(a)(5) requires that an application for an SDA must include information on the kinds and quantities of radioactive materials expected to be produced in operation and the means for controlling and limiting radioactive effluents and radiation exposures within the limits set forth in part 20.

Issue

FSAR Section 12.3.2.3 indicates that the credited radiation shielding barriers for the reactor building and radioactive waste building are provided in terms of nominal concrete equivalent thicknesses and that the design provides "equivalent density thicknesses" for the barrier described using a variety of structural design solutions. During the audit, NuScale described the approach of using equivalent density thickness as using the ratio of densities of the materials to determine the thickness of the replacement material (for example, if a material has a density 2.25 times that of the concrete shield thickness identified in Tables 12.3-5 or 12.3-6 in FSAR Chapter 12 of the NuScale SDAA, the replacement material thickness will be 2.25 times less thick than the provided concrete value). However, the amount of radiation attenuated by a shield is not only dependent on the ratios of density and thickness. Even for gamma radiation, mass attenuation coefficients are different for different materials. Determining appropriate radiation

shielding is even more complex for neutron radiation. Therefore, if a replacement radiation shielding material is used, the replacement shielding material should provide at least equivalent radiation attenuation as the specified material.

Furthermore, NuScale SDAA, Part 8, Table 3.11-1, item 4 and Table 3.12-1, item 1 provide the Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for the reactor building and radioactive waste building radiation shielding barriers. The acceptance criteria for these ITAAC include that a report exists and concludes that the radiation attenuation capability of the radiation shielding barriers is greater than or equal to the required attenuation capability of the approved design (note that Table 3.11-1, item 5 also includes a similar ITAAC for reactor building radiation shield doors). As a result, the equivalent density thickness approach described in FSAR Section 12.3.2.3 appears inconsistent with the ITAAC acceptance criteria for these ITAAC.

Information Requested

Provide additional information describing how the alternative radiation shielding approach of using “equivalent density thicknesses,” as mentioned in FSAR Section 12.3.2.3 and as described during the audit, is acceptable. Include information on how radiation protection of workers and equipment will be ensured using the specified approach and discuss potential impacts on designated plant radiation zones and equipment qualification radiation zones and total integrated dose specifications in the FSAR.

Also, discuss the apparent discrepancy between the “equivalent density thicknesses” approach and the ITAAC acceptance criteria for ITAAC, item 4 in Table 3.11-1 and item 1 in Table 3.12-1 of SDAA, Part 8 and how the ITAAC will be addressed if alternative radiation shielding is used.

As appropriate, update the SDAA to incorporate any updated information or to address any inconsistencies.

Question 12.3.4.2-1

Regulatory Basis

10 CFR 50.34(f)(2)(xix) requires that instrumentation be provided for adequate monitoring of plant conditions following an accident that includes core damage.

10 CFR 50.34(f)(2)(xvii), item D requires, in part, that instrumentation is provided to measure, record and readout containment radiation intensity (high level) in the control room.

10 CFR Part 50, Appendix A, General Design Criterion 64 requires that means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

Issue

NuScale SDA Section 12.3.4.2 states that the fixed area radiation monitors used for post-accident monitoring (PAM) have ranges that consider the maximum calculated accident levels

and are designed to operate effectively under the environmental conditions caused by an accident. It also states that the PAM monitors conform to Regulatory Guide (RG) 1.97, Revision 5, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants.”

The staff notes that the under-the-bioshield radiation monitors in the NuScale design serve as the alternative to the containment high range monitors in large light water reactors (LWRs). NUREG-0737, “Clarification of TMI Action Plan Requirements,” Table II.F.1-3 specifies that the containment high range radiation monitors shall have the capability to detect and measure the radiation levels within the reactor containment during and following the accident. NUREG-0737 also references RG 1.97, Revision 2, “Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident.” Specifically, RG 1.97, Revision 2, states that one of the purposes of the containment high range radiation monitors is long-term surveillance. In large LWRs, the containment high range radiation monitors are typically qualified to function long term following a core damage accident and are used to aid in accident diagnosis and control and are used in emergency classifications.

The NuScale under-the-bioshield radiation monitors are designated as Type B, C, and F post-accident monitoring variables in the NuScale SDAA design. RG 1.97, Revision 5 references Institute of Electrical and Electronics Engineers (IEEE) Standard (Std.) 497-2016, “IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations,” which specifies that for Type C variables, the required operating time shall be the duration for which the measured variable is required by the plant’s licensing basis document (LBD) or at least 100 days following the start of an accident. However, SDA Table 19.2-8 specifies an equipment survivability duration in Table 19.2-8 of only 48 hours after core damage.

The staff notes that the radiological core damage conditions under the bioshield are generally not more severe than the core damage conditions inside containment in large LWRs and it does not appear that other environmental conditions such as temperatures or pressures are more severe either.

In addition, in exemption 16, NuScale requests an exemption from taking post-accident sampling. Post-accident sampling requirements are in place as a means to provide data in the post-core accident environment. With the under-the-bioshield radiation monitors only surviving for 48 hours and no post-accident sampling, it appears that there may be no means to determine the radiological conditions inside containment or under the bioshield beyond 48 hours. Furthermore, there doesn’t appear to be any radiation monitoring equipment anywhere in the facility designated to operate more than 48 hours following a core damage accident.

SDA Section 12.3.4.2 provides information about the under-the-bioshield monitors; however, the SDA doesn’t provide much detail for its specific uses in accident conditions, beyond that the monitors are intended to detect fuel damage.

Information Requested

Provide additional information on the intended uses of the NuScale under-the-bioshield radiation monitors, following a core damage accident. It is the staff’s understanding that the radiation levels under the bioshield could be thousands or more R/hr after the first 48 hours following the start of the core damage accident. Please provide justification as to why it is acceptable for the

information regarding the radiological conditions under the bioshield to potentially be unavailable after the first 48 hours for the plant staff to take appropriate actions, provided the information in IEEE 497-2016, especially considering accidents that may not progress as anticipated.