RAIO-0218-58876

February 27, 2018

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 346 (eRAI No. 9291) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 346 (eRAI No. 9291)," dated January 29, 2018

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. 9291:

- 12.02-24

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 Steven Mirsky at 240-833-3001 or at smirsky@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9291

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9291

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

eRAI No.: 9291
Date of RAI Issue: 01/29/2018

NRC Question No.: 12.02-24

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. 10 CFR Part 20 requires the use of engineering features to control and minimize the amount of radiation exposure to occupational workers, from both internal and external sources. 10 CFR 50.49(e)(4) requires applicants to identify the type of radiation and the total dose expected during normal operation over the installed life of the equipment. Appendix A to Part 50—General Design Criteria (GDC) 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. GDC 4 requires applicants to ensure that structures, systems, and components important to safety are designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation. NuScale DSRS 12.2 DSRS and DSRS 3.11 Acceptance Criteria states that the applicant should describe the radiation fields in sufficient detail for evaluating the inputs to shielding codes, and determination of radiation dose to electrical equipment important to safety as described in 10 CFR 50.49, and GDC 4.

Background

NuScale DCD, Tier 2 Revision 0, Table 3C-6: "Normal Operating Environmental Conditions," states that the 60 Years Integrated Gamma Dose (Rads), which the table indicates includes fission gammas, N-gammas, and coolant, for the area outside of the containment vessel and under the bioshield is 4.35E4 rads (0.087 rads/hour). DCD subsection 12.2.1.2 "Reactor Coolant System," states that the primary coolant gamma spectra are provided in DCD Table 12.2-3 and DCD Table 12.2-4. However, there is no quantitative discussion of how these values are derived. For instance, DCD subsection 9.3.4 "Chemical and Volume Control System," notes that reactor coolant system (RCS) gas removal operations are infrequent operations that only occur when non- condensable gas accumulation in the pressurizer impacts RCS pressure control. Based on this discussion, the staff expects fission product gases to accumulate in the pressurizer gas space. In addition, the Chemical and Volume Control System let down isolation valves and lines are located in the bioshield area. Based on operating experience at commercial Pressurized Water Reactors (PWR), dose rates from the valves alone could exceed the gamma values listed in Table 3C-6.

Also, NuScale DCD Table 3C-6 states that the 60 Years Integrated N Dose (Rads) for the area outside of the containment vessel and under the bioshield is 1.85E6 rads (3.7 rads/hour). There is no discussion in DCD subsection 3.11 nor DCD 12.2 about the gamma dose rate from activation of the containment vessel (CNV) steel, the stainless steel lining of the bioshield cover, the steel main steam and main feedwater lines etc. NuScale Technical Report TR-0116-20781-P Rev. 0 “Fluence Calculation Methodology and Results,” Table 5-1 “Best estimate of fluence expected to be experienced in various NuScale Power Module components and locations,” describes the neutron fluence to the reactor vessel and containment vessel, in the vicinity of the core, but does not provide any neutron fluence information above the reactor vessel flange area.

The gamma information evaluated during the staff review under NuScale DSRS 12.2, are used as inputs for the evaluation performed by the staff for NuScale DSRS 12.3-12.4 and DSRS 3.11, related to the acceptability of the shielding design, the establishment of radiation zones, the impact on systems, structures and components. This is consistent with NuScale DSRS 12.2 Acceptance Criteria, which states that the source descriptions should include all pertinent information required for input to shielding codes used in the design process, establishment of related facility design features, and determination of radiation dose to electrical equipment important to safety as described in 10 CFR 50.49, and GDC 4, as well as the controlling radiation exposure to workers and members of the public, consistent with 10 CFR 20 and GDC 61. DSRS 12.2 also states that unless described within other sections of the FSAR, source descriptions should include the methods, models, and assumptions used as the bases for all values provided in FSAR Section 12.2. These acceptance criteria are consistent with the relevant requirements of 10 CFR Part 50 and 10 CFR Part 52.

The DCD does not provide sufficient bases for fully determining the gamma dose rates under the bio shield, nor does it clearly articulate how they were derived. The staff needs to ascertain the gamma dose rates resulting from operation of the plant and evaluate appropriate supporting information to assess the impact on a variety of review areas, including equipment qualification, radiation streaming into adjacent areas, the amount of gamma radiation from neutron activation of materials, and operational radiation exposure for maintenance activities.

Key Issue: It is unclear what the gamma dose rates under the bio shield are, and how they were derived. The staff needs to know the gamma dose rates resulting from operation of the plant and sufficient information to justify the assumed values. The staff uses these values to assess the impact on a variety of topics considered in the review, including equipment qualification, radiation streaming into adjacent areas, the amount of gamma radiation from neutron activation of materials, and operational radiation exposure for maintenance activities.

Question

To facilitate staff understanding of the application information sufficient to make appropriate regulatory conclusions, with the respect to the kinds and quantities of radioactive materials and radiation fields within the facility, the staff requests that the applicant:

- Explain/Justify the methods, models and assumptions used to calculate the gamma dose rates during operation above the top of the pressurizer, inside the containment vessel above the reactor vessel, and under the bioshield wall (including gamma dose rates from neutron activation of materials, N-16, and all other sources).
- As necessary, revise and update the NuScale DCD, Tier 2, Revision 0, Section 12.2, to describe the gamma dose rate at the area identified above, and the assumptions and input parameters used.

OR

Provide the specific alternative approaches used and the associated justification.

NuScale Response:

For the doses in and around the NuScale power module (NPM), particle transport and shielding calculations are performed for various source terms using MCNP6.

NuScale modeled the core neutron source term (FSAR Table 12.2-1) using a U-235 Watt fission spectrum with a NuScale specific neutron source strength intensity. The maximum neutron release rate for the modeled core design configurations (U-235 enrichment and Gd_2O_3 loading) and fuel burnup was used to conservatively adjust the neutron source term, thus creating a bounding neutron source term. In addition, this conservative neutron source term was multiplied by an assembly peaking factor of 1.461. The MCNP6 code is then used to transport the core neutron source term through the modeled NPM to determine the exposure rates in the various parts of the module and above the module.

The neutron induced gamma source strength is generated internally by MCNP6 using this same neutron source strength. The dose from the gamma radiation induced by the fission neutrons is tallied.

The fission gamma source term is modeled as the SCALE6.1 PWR energy spectrum distribution with a normalized source strength of 160 MW_{th}, and NuScale specific cross section libraries. This gamma source term is then transported in and around the NPM using MCNP6.

Lastly, the gamma output from the reactor coolant system is based on the gamma energy spectrum from the design basis reactor coolant isotopics calculations (as described in FSAR Section 11.1.1 and FSAR Table 11.1-4). Reactor coolant isotopics includes N-16 and other water activation products. This gamma spectrum is transported from the primary coolant in and around the module, using MCNP6.

The gamma dose rates above the pressurizer inside the containment vessel are 3.0E+3 mrem/hr from neutron induced gammas, 1.4E+1 mrem/hr from fission gammas, and 1.1E+0 mrem/hr from reactor coolant gammas. The gamma dose rates above the containment vessel under the bioshield are 9.3E+1 mrem/hr from neutron induced gammas, and was calculated using MCNP. The dose rates above the containment vessel from fission gammas and the reactor coolant gammas were conservatively assumed to be equal to their respective dose rates above the pressurizer inside the containment vessel. This is conservative because it neglects the gamma attenuation due to the distances and intervening NPM material. The MCNP results from the sources described above showed that neutron induced gamma source is the dominant gamma source above the pressurizer inside the containment vessel, and above the containment vessel under the bioshield. The pressurizer was modeled as a void for the sources described above which results in conservative dose rates from these sources. A voided pressurizer resulted in conservatively high gamma dose rates from neutron induced gammas and fission gammas, the two largest contributors to the locations described above, due to reduced shielding material (i.e., water in the pressurizer).

Impact on DCA:

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