NuScale Power

RAIO-0318-59227

March 21, 2018

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. 342 (eRAI No. 9293) on the NuScale Design Certification Application

**REFERENCE:** U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 342 (eRAI No. 9293)," dated January 26, 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. 9293:

- 12.03-16

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,

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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9293

**Enclosure 1:**

NuScale Response to NRC Request for Additional Information eRAI No. 9293

NuScale Power™

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

**eRAI No.:** 9293
**Date of RAI Issue:** 01/26/2018

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**NRC Question No.:** 12.03-16

**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 set forth in 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 52.47(a) (22) requires applicants to demonstrate how the operating experience insights have been incorporated into the plant design.

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. The acceptance criteria of NuScale DSRS section 12.3-12.4 states that Regulatory Guide 8.19, "Occupational Dose Assessment in Light-Water Reactor Power Plants Design Stage Man-Rem Estimates," provides a method acceptable to the staff for performing an assessment of collective occupational radiation dose as part of the ongoing design review process so that such exposures will be ALARA. The DSRS Acceptance Criteria section of NuScale DSRS section 12.3-12.4, "Radiation Protection Design Features," states that the applications should describe how operating experience insights have been incorporated into the plant design, to reduce maintenance and improve reliability.

**Background**

DCD Tier 2 Section 12.4, "Dose Assessment," states that the dose assessment presented in DCD Section 12.4 includes the estimated radiation exposures to plant personnel performing work activities involving normal operations, maintenance and inspections, refueling activities, and waste handling, using the methodology presented in RG 8.19 to demonstrate that the facility design is compliant with 10 CFR Part 20. To estimate the occupational radiation exposures for the NuScale facility, various work activities and work durations are compiled along with the expected significant (>0.1 mrem/hr) radiation fields that would be encountered.

DCD section 12.4.1 states that the occupational radiation exposure (ORE) dose estimates for Special Maintenance and Refueling Activities were listed in DCD Table 12.4-5, “Occupational Dose Estimates from Special Maintenance,” and DCD Table 12.4-7, “Occupational Dose Estimates from Refueling Activities.”

Using information made available to the staff during the RPAC Chapter 12 Audit, the staff reviewed the bases for the estimated doses for the work activities described in these tables. The staff observed that the dose rates used for performing the dose estimates did not appear to be consistent with operating experience (e.g., Electric Power Research Institute (EPRI) Technical Report (TR) 1015119, “Application of the EPRI Standard Radiation Monitoring Program for PWR Radiation Field Reduction Final Report,” dated November, 2007.) For instance, the staff noted that the dose rates assumed for work in the vicinity of the steam generators did not appear to be within an order of magnitude to the median value depicted in TR-1015119. While it is not possible for the staff to quantitatively assess the change in dose rates associated with the smaller NuScale design, the staff did qualitatively consider the relative size of the plant on the assumed dose rates.

The guidance contained in RG 8.19 states that plant experience, available from industry groups like EPRI, provides useful information for performing this assessment. As noted in RG 8.19, an objective of the dose assessment process should be to develop a systematic process for considering and evaluating possible dose-reducing design changes and associated operating procedure changes as part of the comprehensive ongoing design review, and identification of principal ALARA-related changes resulting from the dose assessment. The occupational dose assessment should be based on anticipated radiation conditions after at least 5 years of plant operation. Analysis of the elements of the man-rem estimate (e.g., radiation levels, task duration, and frequency), treated qualitatively, can be of significant value in making engineering judgments regarding design changes for ALARA purposes. As noted in DSRS Section 12.3-12.4, the staff reviews the description of any additional dose-reducing measures taken because of the dose assessment process for specific functions or activities.

Key Issue 1

The ORE dose estimates provided in DCD Tier 2 Section 12.4, “Dose Assessment,” do not appear to have been based upon relevant plant data. Based on the cited industry experience, the component dose rates used to perform the ALARA analysis appear to be biased low. The potential result is that the method for identifying possible changes to the plant design for the purpose of minimizing ORE, as required by 10 CFR 20.1101(b), may improperly determine that the dose savings do not justify the proposed design change.

Question 1

To facilitate staff understanding of the application information sufficient to make appropriate regulatory conclusions with respect to potential ORE dose, the staff requests that the applicant:

- Justify/explain the basis of the dose rates assumed for performing the dose assessment, including the associated technical references,
- Justify/explain any design changes that were made as a result of the dose analysis

performed using the guidance of RG 8.19,
- As necessary, revised section DCD Section 12.4.1, to reflect any changes to the dose assessment,

OR

Provide the specific alternative approaches used and the associated justification.

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**NuScale Response:**

10 CFR 20.1101(b) and 10 CFR 20.1003 allow the use of both procedures and engineering controls to maintain exposures to radiation as far below the dose limits in 10 CFR Part 20 as is practical. As in the case of operating licensed nuclear power plants and previously NRC-approved design certification applications, the facility's design features work in concert with the radiation protection programs and procedures to comply with this regulation. Operational procedures are frequently relied upon to comply with regulations.

Also, the RAI includes a reference to 10 CFR 52.47(a)(22), which requires applicants to demonstrate how operating experience insights have been incorporated into the plant design, as another part of the regulatory basis. Regulatory Guide 1.206 and NUREG-0800 define operating experience insights specifically as NRC generic letters and bulletins issued after the most recent revision of the applicable standard review plan and six months prior to the docket date of the application. None of the cited references meet the definition of operating experience. The Electric Power Research Institute (EPRI) Technical Report (TR) 1015119, “Application of the EPRI Standard Radiation Monitoring Program for PWR Radiation Field Reduction Final Report,” dated November, 2007, is not included as part of the NuScale DSRS, Section 12.3-12.4 and does not meet the definition of operation experience, per 10 CFR 52.47(a)(22).

However, the EPRI Technical Report (TR) 1015119, does provide technical information related to typical pressurized water reactor doses from steam generators. NuScale has conducted a study that demonstrates that the estimated operator dose rates performing maintenance activities for the steam generator are appropriate for the NuScale design. The NuScale study assumes a dose rate inside the helical coil steam generator (HCSG) of 4 R/hr based on an approximate cold leg channel head center average from EPRI TR 1015119, Figure 5-7. It should be noted that the steam generator tube region inside the NuScale power module (NPM) is inaccessible by plant personnel. Steam generator maintenance activities during a refueling outage (e.g., in the dry dock) are performed on the secondary side (tube side) with personnel on the outside of the NPM. Therefore dose rates were evaluated at locations outside of the RPV and CNV.

Given that the radiation emanating from a HCSG inside an NPM must travel through the RPV steel wall, or through the RPV and CNV steel walls, the resultant attenuation significantly reduces the dose rate to plant personnel. The approximate thickness of the RPV steel walls can

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be found in FSAR Table 5.3-1, and the approximate thickness of the CNV steel walls can be found in FSAR Figure 6.2-2a. The conclusion of the study indicates that the radio-cobalt buildup in the steam generators is appropriately accounted for, and the results of the occupational radiation exposures in FSAR Section 12.4 are appropriate.

Therefore, the operator exposure estimates in FSAR Section 12.4 involving activities associated with the steam generator are appropriate for the NuScale design.

**Impact on DCA:**

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