RAIO-0818-61441

August 16, 2018

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Supplemental Response to NRC Request for Additional Information No. 345 (eRAI No. 9294) on the NuScale Design Certification Application

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 345 (eRAI No. 9294)," dated January 26, 2018
2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 345 (eRAI No.9294)," dated March 21, 2018

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 Questions from NRC eRAI No. 9294:

- 12.03-23
- 12.03-25
- 12.03-27

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. 9294

Enclosure 1:

NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9294

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

eRAI No.: 9294
Date of RAI Issue: 01/26/2018

NRC Question No.: 12.03-23

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 10 CFR 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.
The Acceptance Criteria of DSRS Section 12.3-12.4, "Radiation Protection Design Features," contains a number of criteria related to the design of the shielding, including:

- That the areas inside the plant structures, as well as in the general plant yard, should be subdivided into radiation zones, with maximum design dose rate zones and the criteria used in selecting maximum dose rates identified.
- That the composition of the shielding material should be selected to minimize, to the extent practicable, the potential for the shield itself to become a radiation source (either from activation of the shield material or production of secondary radiation resulting from interactions with the primary radiation).
- Where the applicant's shielding design incorporates material subject to degradation, such as through the effects of radiation (e.g., depletion of boron neutron absorbers,) temperature extremes (e.g., degradation of polymer based materials because of high temperature,) density changes (e.g., sagging or settling of shielding material with age,) methods are in place to ensure that ORE remains ALARA, and the equipment exposures are maintained in accordance with the provisions of 10 CFR 50.49 should be specified
- The application should identify the allowable constraints (e.g., minimum cooling air flow, maximum shielding material temperature, and maximum allowable neutron flux,) and how those parameters are measured and assessed over the design life of the facility.

- That accessible portions of the facility that are capable of having radiation levels greater than 1 gray (Gy) per hour (100 rads per hour) are shielded, and are clearly marked with a sign stating that potentially lethal radiation fields are possible. If removable shielding is used to reduce dose rates to less than 1 Gy per hour, it must also be explicitly marked as above.

Background

DCD Tier 2 Revision 0 Section 12.3.2, “Shielding,” describes some of the design considerations for radiation shielding, such as stating that material used for a significant portion of plant shielding is concrete.

DCD Section 12.3.2.2, “Design Considerations,” states that the selection of shielding materials considers the ambient environment and potential degradation mechanisms. The material used for a significant portion of plant shielding is concrete. In addition to concrete, other types of materials such as steel, water, tungsten, and polymer composites are considered for both permanent and temporary shielding. DCD Section 12.3.2.4.3, “Reactor Building,” states that cubicle walls are concrete supported by carbon steel plates, called structural steel partition walls.

DCD Table 12.3-6, “Reactor Building Shield Wall Geometry,” provides the nominal thickness of concrete for some of the walls in the RXB. DCD Table 12.3-8, “Reactor Building Radiation Shield Doors,” lists the shielded doors located in the RXB. DCD Table 12.3-7, “Radioactive Waste Building Shield Wall Geometry,” provides the nominal thickness of concrete for some of the walls in the RWB. DCD Table 12.3-9, “Radioactive Waste Building Radiation Shield Doors,” list the shielded doors located in the RWB.

Using information made available to the staff during the RPAC Chapter 12 Audit, the staff identified that some shielding design calculations referenced the use of additional steel (i.e., in addition to the structural steel partition walls already noted,) shielding to limit dose rates in adjacent areas.

Key Issue 1

DCD Tier 2 Revision 0 Section 12.3.2, “Shielding,” does not identify the specific areas where additional shielding is required. DCD Table 12.3-6, “Reactor Building Shield Wall Geometry,” and DCD Table 12.3-7, “Radioactive Waste Building Shield Wall Geometry,” provide the nominal thickness of concrete for some of the walls in the RXB and RWB. However, neither table identifies the location of nor the minimum thickness of any additional steel shielding material.

Question 1

To facilitate staff understanding of the application information sufficient to make appropriate regulatory conclusions regarding the adequacy of the radiation shielding, the staff requests that the applicant:

- Describe the locations in the RXB and RWB where additional steel shielding is credited for the radiation shielding design,
- Justify/explain the assumptions used to perform the shielding analysis in the RXB and the RWB, supporting the amount of steel shielding identified, including the associated methods, models and assumptions used to establish the identified values,
- As necessary, revised section DCD Section 12.3.2, to describe the these steel thicknesses, and the associated assumptions,

OR

Provide the specific alternative approaches used and the associated justification.

NuScale Response:

During a public NRC RAI clarification call on July 26, 2018, NuScale agreed to provide a supplemental response to RAI 9294 (Q12.03-23) by revising the end notes of FSAR Table 12.3-7 to say "provided," instead of "credited," and to add the words, "equivalent attenuation."

Impact on DCA:

FSAR Table 12.3-7 has been revised as described in the response above and as shown in the markup provided in this response.

RAI 12.03-23, RAI 12.03-23S1, RAI 12.03-25

Table 12.3-7: Radioactive Waste Building Shield Wall Geometry

Elev.	Room #	Room Name	North Wall	East Wall	South Wall	West Wall	Floor	Ceiling	Source Term
71′	030-004	Tank Room	20″ Concrete	20″ Concrete	20″ Concrete	48″ Concrete Wall (Facility External Wall)	60″ Concrete (Facility Basemat)	24″ Concrete	GRWS Charcoal Beds
71′	030-005	Tank Room	20″ Concrete	20″ Concrete	20″ Concrete	48″ Concrete Wall (Facility External Wall)	60″ Concrete (Facility Basemat)	24″ Concrete	GRWS Charcoal Beds
71′	030-012	Tank Room	36″ Concrete	48″ Concrete Wall (Facility External Wall)	15″ Concrete	15″ Concrete	60″ Concrete (Facility Basemat)	24″ Concrete	Phase Separator Tank
71′	030-013	Tank Room	36″ Concrete	15″ Concrete	15″ Concrete	24″ Concrete	60″ Concrete (Facility Basemat)	24″ Concrete	Phase Separator Tank
71′	030-015	Tank Room	36″ Concrete	24″ Concrete	24″ Concrete	24″ Concrete	60″ Concrete (Facility Basemat)	24″ Concrete	HCW Collection Tank
71′	030-016	Tank Room	36″ Concrete	24″ Concrete	24″ Concrete	24″ Concrete	60″ Concrete (Facility Basemat)	24″ Concrete	HCW Collection Tank
71′	030-018	Tank Room	24″ Concrete	24″ Concrete	36″ Concrete	24″ Concrete	60″ Concrete (Facility Basemat)	24″ Concrete	LCW Collection Tank
71′	030-019	Tank Room	24″ Concrete	24″ Concrete	36″ Concrete	24″ Concrete	60″ Concrete (Facility Basemat)	24″ Concrete	LCW Collection Tank
71′	030-020	Tank Room	24″ Concrete	24″ Concrete	36″ Concrete	24″ Concrete	60″ Concrete (Facility Basemat)	24″ Concrete	LCW Sample Tank
71′	030-021	Tank Room	24″ Concrete	24″ Concrete	24″ Concrete	24″ Concrete	60″ Concrete (Facility Basemat)	24″ Concrete	LCW Sample Tank
71′	030-024	Tank Room	24″ Concrete	24″ Concrete	24″ Concrete	24″ Concrete	60″ Concrete (Facility Basemat)	24″ Concrete	HCW Sample Tank
71′	030-025	Tank Room	24″ Concrete	34″ Concrete	24″ Concrete	24″ Concrete	60″ Concrete (Facility Basemat)	24″ Concrete	HCW Sample Tank
71′	030-026	Tank Room	36″ Concrete	34″ Concrete	36″ Concrete	34″ Concrete	60″ Concrete (Facility Basemat)	24″ Concrete	Spent Resin Storage Tank
71′	030-027	Tank Room	36″ Concrete	48″ Concrete Wall (Facility External Wall)	36″ Concrete	34″ Concrete	60″ Concrete (Facility Basemat)	24″ Concrete	Spent Resin Storage Tank
71′	030-033	HIC Filling Room	36″ Concrete (Note 1)	36″ Concrete (Note 1)	36″ Concrete (Note 1)	36″ Concrete (Note 1)	60″ Concrete (Facility Basemat)	24″ Concrete	HIC

Table 12.3-7: Radioactive Waste Building Shield Wall Geometry (Continued)

Elev.	Room #	Room Name	North Wall	East Wall	South Wall	West Wall	Floor	Ceiling	Source Term
71′	030-034	Class A/B/C HIC Room	36″ Concrete	36″ Concrete	36″ Concrete	48″ Concrete Wall (Facility External Wall)	60″ Concrete (Facility Basemat)	24″ Concrete	HIC
82′	---	Pipe Chase	24″ Concrete	24″ Concrete	24″ Concrete	24″ Concrete	20″ Concrete	24″ Concrete	Resin transfer pipe
100′	030-105	LRW Mobile Processing Area	24″ Concrete	36″ Concrete	24″ Concrete	36″ Concrete	24″ Concrete	12″ Concrete - Facility Ceiling (Note 2)	LCW GAC; LCW TUF; LCW RO; HCW GAC; HCW TUF; HCW RO; LCW Cation Demineralizer; LCW Anion Demineralizer; LCW Mixed Bed Demineralizer; LCW Cesium Demineralizer.
100′	030-106	Drum Dryer Room A	24″ Concrete	36″ Concrete	24″ Concrete	12″ Concrete	24″ Concrete	12″ Concrete - Facility Ceiling (Note 3)	Drum Dryer

Note 1: The equivalent attenuation to an additional 4.5 inches of lead is provided for a HIC process shield.

Note 2: The equivalent attenuation to an additional one inch of steel on top of the LCW demineralizers and GAC processing skids is provided.

Note 3: The equivalent attenuation to an additional two inches of steel on top of the drum dryer skid is provided.

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

eRAI No.: 9294
Date of RAI Issue: 01/26/2018

NRC Question No.: 12.03-25

The Regulatory Basis and Background are in RAI-9294 Question 31054

Key Issue 3

The acceptance criteria of NuScale DSRS section 12.3-12.4, states that accessible portions of the facility that are capable of having radiation levels greater than 1 gray (Gy) per hour (100 rads per hour rads/hr) should be shielded. All accessible portions of the facility capable of having radiation levels greater than 1 Gy per hour (100 rads per hour) are to be clearly marked with a sign stating that potentially lethal radiation fields are possible. If removable shielding is used to reduce dose rates to less than 1 Gy per hour, it must also be explicitly marked as above. DCD Tier 2 Revision 0 Section 12.3.2.4.3, "Reactor Building," and DCD Section 12.3.2.4.4 , "Radioactive Waste Building," identify a number of areas, such as resin demineralizers, filters, spent resin storage tanks etc., which may contain quantities of radioactive material resulting in radiation dose rates exceeding 100 rads/hr.

However, DCD Tier 2 Revision 0 Section 12.3.2, "Shielding," does not identify the specific areas where removable shielding is used. DCD Table 12.3-6, "Reactor Building Shield Wall Geometry," and DCD Table 12.3-8, "Reactor Building Radiation Shield Doors and DCD Table 12.3-7, "Radioactive Waste Building Shield Wall Geometry," provide the nominal thickness of concrete for some of the walls in the RXB and RWB. However, neither table identifies the location of removable shielding material. DCD Tier 2 Revision 0 Section 12.3.2, "Shielding," does not specify that those portions of the facility capable of having radiation levels greater than 1 Gy per hour (100 rads per hour) where removable shielding is used, are clearly marked with a sign stating that potentially lethal radiation fields are possible.

Question 3

To facilitate staff understanding of the application information sufficient to make appropriate regulatory conclusions regarding the adequacy of the radiation shielding, the staff requests that the applicant:

- Describe the locations in the RXB and RWB where removable shielding is credited for the radiation shielding design,
- For those portions of the facility exceeding 100 rads/hr where removable shielding is used describe how the areas are marked,
- As necessary, revised section DCD Section 12.3.2, to include these descriptions of removable shielding and markings,

OR

Provide the specific alternative approaches used and the associated justification.

NuScale Response:

During a public NRC RAI clarification call on July 26, 2018, NuScale agreed to provide a supplemental response to RAI 9294 (Q12.03-25) by revising the FSAR to state that radiation shield plugs have equivalent attenuation as the floor in which they are located.

Impact on DCA:

FSAR Section 12.3 has been revised as described in the response above and as shown in the markup provided in this response.

Consistent with RG 8.8, shielding analysis employs accurate modeling techniques and conservative approaches in the determination of shielding thickness. Source terms, geometries, and field intensities are analyzed conservatively. In addition to normal and shutdown conditions, source terms include transient conditions such as resin transfers.

RAI 12.03-58

The material used for a significant portion of plant shielding is concrete. For most applications, concrete shielding is designed in accordance with ANSI/ANS 6.4-2006 (Reference 12.3-1). Table 12.3-6 and Table 12.3-7 shows~~s~~ the ~~nominal~~ shielding thicknesses assumed in the shielding analyses~~for rooms~~ in plant buildings. In addition to concrete, other types of materials such as steel, water, tungsten, and polymer composites are considered for both permanent and temporary shielding. The use of lead is minimized.

For shield walls that contain a door, the door provides an equivalent radiation attenuation as the shield wall that contains the door. A listing of radiation shield doors is provided in Table 12.3-8 for the RXB and Table 12.3-9 for the RWB.

RAI 12.03-25S1

Shield floor plugs provide an equivalent radiation attenuation as the shield floor that contains the plug.

12.3.2.3 Calculation Methods

The primary computer program used to evaluate shielding is Monte Carlo N-Particle Transport Code (MCNP6) (Reference 12.3-2) which was developed by Los Alamos National Laboratory. The MCNP6 code is a Monte Carlo radiation transport code designed to track a variety of particles over a broad spectrum of energies. The MCNP6 code is used for shielding calculations and for dose rate determinations. ANSI/ANS 6.1.1-1977, "Gamma Flux to Dose Conversion Factors," (Reference 12.3-3) is used to convert gamma flux at each detector location to a corresponding dose rate.

RAI 12.02-14

Radioactive components in the RXB and RWB are modeled using MCNP6. The codes used to prepare source strength input data are described in Section 12.2. A three-dimensional shielding model is constructed for radioactive components using structure, location, and equipment data. Source geometries and source term distributions and intensities are conservatively determined. ~~Source terms associated with resin transfers and crud bursts are included.~~ In general, the component source geometries are modeled as cylindrical volumes which incorporate the full volume of the component.

Shielding credit and material selections for MCNP6 cells are conservatively applied. The material compositions for air, concrete, water, and stainless steel are taken from PNNL-25870 (Reference 12.3-4). Structural steel composition is in accordance with plant drawings and ASTM standards. Credit is not taken for reinforcing steel bars in the concrete.

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

eRAI No.: 9294
Date of RAI Issue: 01/26/2018

NRC Question No.: 12.03-27

The Regulatory Basis and Background are in RAI-9294 Question 31054

Key Issue 5

DCD Tier 1 Revision 0 Section 3.11, ”Reactor Building,” and DCD Tier 1 Section 3.12, ”Radioactive Waste Building,” contain the Inspections, Tests, Analyses, & Acceptance Criteria (ITAAC,) related to the radiation shielding.

DCD Tier 1 Section 3.11, ”Reactor Building,” states that the RXB includes radiation shielding barriers for normal operation and post- accident radiation shielding. It further states that DCD Tier 1 Table 3.11-2, “Reactor Building Inspections, Tests, Analyses, and Acceptance Criteria,” contains the inspections, tests, and analyses for the RXB. DCD Tier 1 Table 3.11-1 item 4 Acceptance Criteria states that the thickness of RXB radiation shielding barriers is greater than or equal to the required thickness specified in DCD Tier 1 Table 3.11-1. DCD Tier 1 Section 3.11 further states that the RXB includes radiation attenuating doors for normal operation and post-accident radiation shielding. These doors have a radiation attenuation capability that meets or exceeds that of the wall within which they are installed.

DCD Tier 1 Section 3.12, ”Radioactive Waste Building,” states that the RWB includes radiation shielding barriers for normal operation and post-accident radiation shielding. Also, the RWB includes radiation attenuating doors for normal operation and for post-accident radiation shielding. These doors have a radiation attenuation capability that meets or exceeds that of the wall within which they are installed. DCD Tier 1 Section 3.12 further states that DCD Tier 1 Table 3.12-2: “Radioactive Waste Building ITAAC” contains the inspections, tests, and analyses for the RWB. DCD Tier 1 Table 3.12-2 item 1 Acceptance Criteria states that the thickness of RWB radiation shielding barriers is greater than or equal to the required thickness specified in DCD Tier 1 Table 3.12-1, “Radioactive Waste Building Shield Wall Geometry.”

DCD Tier 2 Section 12.3.2.2, “Design Considerations,” states that DCD Tier 2 Table 12.3-6 and DCD Tier 2 Table 12.3-7, show the nominal shielding thicknesses for rooms in the RXB and the RWB, respectively. DCD Tier 2 Table 12.3-6, “Reactor Building Shield Wall Geometry,” provides

the nominal thickness of concrete for some of the walls in the RXB. DCD Tier 2 Table 12.3-7, “Radioactive Waste Building Shield Wall Geometry,” provides the nominal thickness of concrete for some of the walls in the RWB.

Question 5

To facilitate staff understanding of the application information sufficient to make appropriate regulatory conclusions regarding the adequacy of the radiation shielding and the associated ITAAC, the staff requests that the applicant:

- As necessary, revise DCD Tier 1 Section 3.11 and DCD Tier 1 Section 3.12, and the associated tables, to reflect type and the minimum thicknesses of radiation shielding material, in addition to concrete, used in the RXB and RWB,

OR

Provide the specific alternative approaches used and the associated justification.

NuScale Response:

During a public NRC RAI clarification call on July 26, 2018, NuScale agreed to provide a supplemental response to RAI 9294 (Q12.03-27) that revises the end notes in DCD Tier 1 Table 3.12-1 to replace the word "provided" with the word "credited," and to add the term "equivalent attenuation."

Impact on DCA:

DCA Tier 1 Table 3.12-1 has been revised as described in the response above and as shown in the markup provided in this response.

RAI 12.03-27, RAI 12.03-27S1

Table 3.12-1: Radioactive Waste Building Shield Wall Geometry

Elev.	Room Name	North wall	East wall	South wall	West wall	Floor	Ceiling
71′	Tank room	20″ concrete	20″ concrete	20″ concrete	48″ concrete wall (Facility external wall)	60″ concrete (Facility basemat)	24″ concrete
71′	Tank room	20″ concrete	20″ concrete	20″ concrete	48″ concrete wall (Facility external wall)	60″ concrete (Facility basemat)	24″ concrete
71′	Tank room	36″ concrete	48″ concrete wall (Facility external wall)	15″ concrete	15″ concrete	60″ concrete (Facility basemat)	24″ concrete
71′	Tank room	36″ concrete	15″ concrete	15″ concrete	24″ concrete	60″ concrete (facility basemat)	24″ concrete
71′	Tank room	36″ concrete	24″ concrete	24″ concrete	24″ concrete	60″ concrete (facility basemat)	24″ concrete
71′	Tank room	36″ concrete	24″ concrete	24″ concrete	24″ concrete	60″ concrete (facility basemat)	24″ concrete
71′	Tank room	24″ concrete	24″ concrete	36″ concrete	24″ concrete	60″ concrete (facility basemat)	24″ concrete
71′	Tank room	24″ concrete	24″ concrete	36″ concrete	24″ concrete	60″ concrete (facility basemat)	24″ concrete
71′	Tank room	24″ concrete	24″ concrete	36″ concrete	24″ concrete	60″ concrete (facility basemat)	24″ concrete
71′	Tank room	24″ concrete	24″ concrete	24″ concrete	24″ concrete	60″ concrete (facility basemat)	24″ concrete
71′	Tank room	24″ concrete	24″ concrete	24″ concrete	24″ concrete	60″ concrete (facility basemat)	24″ concrete
71′	Tank room	24″ concrete	34″ concrete	24″ concrete	24″ concrete	60″ concrete (facility basemat)	24″ concrete
71′	Tank room	36″ concrete	34″ concrete	36″ concrete	34″ concrete	60″ concrete (facility basemat)	24″ concrete
71′	Tank room	36″ concrete	48″ concrete wall (Facility external wall)	36″ concrete	34″ concrete	60″ concrete (facility basemat)	24″ concrete
71′	High integrity container filling room	36″ concrete (Note 1)	36″ concrete (Note 1)	36″ concrete (Note 1)	36″ concrete (Note 1)	60″ concrete (facility basemat)	24″ concrete
71′	Class A/B/C high integrity container room	36″ concrete	36″ concrete	36″ concrete	48″ concrete wall (Facility external wall)	60″ concrete (facility basemat)	24″ concrete
82′	Pipe chase	24″ concrete	24″ concrete	24″ concrete	24″ concrete	20″ concrete	24″ concrete
100′	Liquid radioactive waste mobile processing area	24″ concrete	36″ concrete	24″ concrete	36″ concrete	24″ concrete	12″ concrete - Facility ceiling (Note 2)
100′	Drum dryer room A	24″ concrete	36″ concrete	24″ concrete	12″ concrete	24″ concrete	12″ concrete - Facility ceiling (Note 3)

Note 1: The equivalent attenuation to an additional 4.5″ of lead is provided for a high integrity container process shield.
Note 2: The equivalent attenuation to an additional one inch of steel on top of the low-conductivity waste demineralizers and granulated activated charcoal processing skid inside the liquid radioactive waste mobile processing area is provided.
Note 3: The equivalent attenuation to an additional two inches of steel on top of the drum dryer is provided.