NuScale Power

RAIO-0418-59690

April 24, 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. 325 (eRAI No. 9268) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 325 (eRAI No. 9268)," dated January 08, 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. 9268:

- 12.02-11

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. 9268

Enclosure 1:

NuScale 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) is scheduled to be issued in late summer 2018. This topical report will describe 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, 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/gram DEI-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 here are conservatively based on the previous primary coolant activity technical specification of 0.2 µCi/g DEI-131. 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 activity concentrations provided in FSAR Table 12.2-34 are based on this lower technical specification value (DEI-131 = 0.037 µCi/gram), and does not contain a crud burst.

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 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). 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).

FSAR Tables 12.2-28, 12.2-29, 12.2-30 and 12.2-31 have been revised to reflect a new accident source term methodology. This new methodology will be documented in the NuScale

Topical Report TR-0915-17565, Revision 3. The FSAR Section 12.2.1.13 description of post-accident sources has also been revised to reflect the new methodology.

Impact on DCA:

FSAR Section 12.2.1.13, Table 12.2-28, Table 12.2-29, Table 12.2-30, and Table 12.2-31 have been revised as described in the response above and as shown in the markup provided in this response.

12.2.1.11 Control Rods and Secondary Source Rods

<u>Control Rod Assemblies</u>

The control rod assemblies are irradiated during reactor operations. Because the reactor core operates in an all-rods-out configuration, it is assumed that only the tip of the control rod is irradiated. This portion of the control rod assembly (CRA) consists of Ag-In-Cd neutron absorber. The major input assumptions are listed in Table 12.2-25. The CRA gamma spectra are listed in Table 12.2-26.

<u>Secondary Source Rod</u>

The secondary source is antimony and beryllium (Sb-Be) and is irradiated for nine cycles. Flux is the same as for the in-core instruments (Section 12.2.1.10).

The gamma ray source strengths associated with the secondary source rods are listed in Table 12.2-27 for various times after shutdown.

12.2.1.12 Secondary Coolant System

The secondary coolant system is expected to contain minimal radioactivity during normal operations. Primary-to-secondary leaks through the steam generator can introduce primary coolant activity into the secondary system with the resultant contamination level being dependent upon the activity level in the primary coolant and the magnitude of the steam generator leak. Because the condensate polishing system is a full flow system, the condensate polishers were evaluated for the radioactive material that could accumulate on the resins during the period between resin regenerations. Assuming the secondary coolant is at the design basis concentrations (Table 11.1-5), resin decontamination factors consistent with NUREG-0017, and a ten day resin regeneration period, the accumulation of radioactive material is less than 100 mCi.

12.2.1.13 Post-Accident Sources

RAI 12.02-3, RAI 12.02-11

Consistent with 10 CFR 50.34(f)(2)(vii), areas that could contain post-accident sources were evaluated for equipment protection and access. ~~The accident source term used to evaluate equipment qualification is based on the core isotopic inventory presented in Table 11.1-1, using the methodology presented in Regulatory Guide 1.183. Because the design does not include post-accident recirculation piping outside containment, the post-accident source remains within the containment vessel except for assumed leakage into the bioshield envelope, which is above the surface of the pool water and below the bioshield. The radionuclide activity released from assumed fuel damage is assumed to be instantaneously and homogeneously released into containment.~~ <u>The maximum primary coolant activity released from design basis accidents, that could be released into the bioshield envelope, is assumed to be released instantaneously and homogeneously throughout the bioshield envelope volume. The remaining primary coolant inventory is assumed to be released instantaneously and homogeneously throughout the containment atmosphere, where it shines and leaks into the bioshield</u>

envelope at the technical specification leak rate and remains in the envelope's volume for the duration of the event. Table 12.2-34 provides the maximum post-accident activity concentrations in the NPM on a mass basis. These concentrations apply to both the liquid and vapor spaces. Other major assumptions for the post-accident source term are listed in Table 12.2-28. ~~The release fractions and containment aerosol removal rates are shown in Table 12.2-29 and Table 12.2-30, respectively.~~ Plateout of activity onto containment surfaces is neglected due to the small containment volume and the lack of surface coatings inside containment. There is also no aerosol removal assumed. The containment air and water volumes are determined based on the reactor vessel being initially full of water and the reactor vessel and containment vessel water levels being in equilibrium. There are ~~four~~three volumes that are evaluated ~~for~~, which includes the post-accident source term. Table 12.2-31 lists the integrated post-accident source energy deposition versus time for both photons and electrons for these ~~four~~three volumes. Table 12.2-31 also tabulates the integrated doses for various times post-accident. For additional details on equipment qualification, see Section 3.11 and Appendix 3.C.

12.2.1.14 Other Contained Sources

There are no other identified contained sources that exceed 100 mCi, including HVAC filters. To evaluate the accumulation of radioactive material on the Reactor Building HVAC system HEPA filters, the airborne radioactivity in the Reactor Building due to pool evaporation and primary coolant leaks was deposited on filters assuming a 99 percent particulate efficiency and two years of operation. For the pool evaporation portion, the Reactor Building HVAC system provides a ventilation flow rate equivalent to one air volume change per hour. For the primary coolant leakage portion, the activity that becomes airborne is captured and filtered by the ventilation system. The resultant accumulation of radioactive material is less than 100 mCi.

COL Item 12.2-1: A COL applicant that references the NuScale Power Plant design certification will describe additional site-specific contained radiation sources that exceed 100 millicuries (including sources for instrumentation and radiography) not identified in Section 12.2.1.

12.2.2 Airborne Radioactive Material Sources

This section describes the airborne radioactive material sources that form part of the basis for design of ventilation systems and personnel protective measures, and also are considered in personnel dose assessment.

12.2.2.1 Reactor Building Atmosphere

Airborne radioactivity may be present in the RXB atmosphere due to reactor pool evaporation or primary coolant leakage. The airborne concentration is modeled as a buildup to an equilibrium concentration based on Bevelacqua (Reference 12.2-1) given the production rate and removal rate. The airborne concentration in the air space above the reactor pool is determined by using the peak reactor pool water source term. The input parameters are listed in Table 12.2-32.

RAI 12.02-3, RAI 12.02-11

Table 12.2-28: Post-Accident Source Term Input Assumptions

Parameter	Value
Containment release delay	~~4.9 hours~~0 hours
Containment release duration	~~1.5 hours~~1.0E-05 hours
Containment leak rate	0.2%/day
Containment leak rate after 24 hours	0.1%/day
Aerosol fraction of non-noble gases released	~~95~~100%
Bioshield envelope volume	~~6711~~6475 ft^3
Primary coolant water density	~~46.9~~43.6 lb/ft^3
Air density	0.07 lb/ft^3
Containment air volume	~~4549~~3635 ft^3
~~Reactor vessel air volume~~	~~1595 ft^3~~
Combined water volume	2500 ft^3

RAI 12.02-3, RAI 12.02-11

Table 12.2-29: ~~Post-Accident Core Inventory Release Fractions~~Not Used

~~Radionuclide Group~~	~~Release Fraction~~
~~Noble gases~~	~~0.68~~
~~Halogens~~	~~0.22~~
~~Alkali metals~~	~~0.24~~
~~Tellurium group~~	~~0.0057~~
~~Alkaline group~~	~~0.21~~
~~Molybdenum group~~	~~0.045~~
~~Noble metals~~	~~0.0011~~
~~Lanthanides~~	~~4.5E-08~~
~~Cerium group~~	~~4.5E-08~~

RAI 12.02-3, RAI 12.02-11

Table 12.2-30: ~~Post-Accident Containment Aerosol Removal Rates~~Not Used

~~Time after Core Damage (hours)~~	~~Removal Rate (hour^{-1})~~
~~4.9~~	~~48.6~~
~~5.94~~	~~6.9~~
~~6.94~~	~~4.03~~
~~7.94~~	~~5.92~~
~~8.94~~	~~7.84~~
~~22.82~~	~~34.3~~
~~36.71~~	~~68.3~~
~~50.59~~	~~100~~
~~64.48~~	~~100~~
~~78.36~~	~~0~~

RAI 12.02-3, RAI 12.02-11

Table 12.2-31: Post-Accident Integrated Energy Deposition and Integrated Dose

Volume	Medium	Time	~~Integrated Deposition (MeV)~~	Integrated Dose (Rad)
Reactor and containment	Water	1 hour	~~0.000E+00~~	~~0.000E+00~~2.73E+03
		~~End release~~	~~1.705E+21~~	~~5.136E+05~~
		~~1 day~~	~~1.872E+22~~	~~5.639E+06~~
		36 hour	~~2.709E+22~~	~~8.159E+06~~2.99E+04
		3 day	~~4.713E+22~~	~~1.420E+07~~4.01E+04
		30 day	~~2.492E+23~~	~~7.507E+07~~9.15E+04
		100 day	~~6.105E+23~~	~~1.839E+08~~1.17E+05
~~Reactor vessel~~	~~Air~~	~~1 hour~~	~~0.000E+00~~	~~0.000E+00~~
		~~End release~~	~~2.350E+21~~	~~7.435E+08~~
		~~1 day~~	~~2.471E+22~~	~~7.816E+09~~
		~~36 hour~~	~~3.549E+22~~	~~1.123E+10~~
		~~3 day~~	~~6.126E+22~~	~~1.938E+10~~
		~~30 day~~	~~2.896E+23~~	~~9.162E+10~~
		~~100 day~~	~~6.582E+23~~	~~2.082E+11~~
Containment vessel	Air	1 hour	~~0.000E+00~~	~~0.000E+00~~1.19E+06
		~~End release~~	~~2.350E+21~~	~~2.607E+08~~
		~~1 day~~	~~2.471E+22~~	~~2.741E+09~~
		36 hour	~~3.549E+22~~	~~3.936E+09~~1.35E+07
		3 day	~~6.126E+22~~	~~6.795E+09~~1.84E+07
		30 day	~~2.896E+23~~	~~3.213E+10~~4.85E+07
		100 day	~~6.582E+23~~	~~7.301E+10~~7.78E+07
Bioshield envelope	Air and water	1 hour	~~0.000E+00~~	~~0.000E+00~~7.35E+03
		~~End release~~	~~7.127E+16~~	~~5.359E+03~~
		~~1 day~~	~~5.078E+18~~	~~3.818E+05~~
		36 hour	~~1.033E+19~~	~~7.764E+05~~8.02E+04
		3 day	~~2.935E+19~~	~~2.207E+06~~1.11E+05
		30 day	~~4.792E+20~~	~~3.603E+07~~4.30E+05
		100 day	~~2.105E+21~~	~~1.583E+08~~1.30E+06

RAI 12.02-3, RAI 12.02-11

Table 12.2-34: Maximum Post-Accident Radionuclide Concentrations

Isotope	RCS Peak Concentration (µCi/g)
Kr-83m	2.69E+00
Kr-85m	2.14E-01
Kr-85	3.73E+01
Kr-87	6.52E-02
Kr-88	1.90E-01
Xe-131m	1.87E+00
Xe-133m	5.47E+00
Xe-133	2.07E+02
Xe-135m	1.73E+01
Xe-135	1.05E+02
Xe-137	1.39E-02
Xe-138	4.77E-02
Br-82	5.67E-01
Br-83	2.67E+00
Br-84	9.06E-01
Br-85	9.52E-02
I-130	4.45E+00
I-131	1.20E+02
I-132	4.60E+01
I-133	1.75E+02
I-134	2.09E+01
I-135	1.04E+02
Rb-88	3.31E-01
Rb-89	1.08E-02
Cs-134	1.84E-01
Cs-136	3.86E-02
Cs-137	1.27E-01
Cs-138	9.98E-02
Ni-63	4.41E-02
Sr-89	1.10E-02
Zr-95	6.51E-02
Nb-95	6.44E-02
Mo-99	3.88E-02
Tc-99m	7.01E-02
Ag-110m	2.17E-01
Te-132	1.56E-02
Ba-137m	2.26E-01
Na-24	9.11E+00
Cr-51	5.19E-01
Mn-54	2.67E-01
Fe-55	2.00E-01
Fe-59	5.01E-02
Co-58	7.68E+00
Co-60	8.83E-02
W-187	4.65E-01

Table 12.2-34: Maximum Post-Accident Radionuclide Concentrations (Continued)

Isotope	RCS Peak Concentration (μCi/g)
Zn-65	8.50E-02
H-3	9.90E-01
Ar-41	2.07E-01