RAIO-0517-54215

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Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information (eRAI No. 8774) on the NuScale Design Certification Application

REFERENCE: 1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 13 (eRAI No. 8774)," dated April 25, 2017

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The Enclosures to this letter contain NuScale's response to the following RAI Questions from NRC eRAI No. 8774:

- 15.00.03-1
- 15.00.03-2

This letter and the enclosed response makes no new regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions on this response, please contact Darrell Gardner at 980-349-4829 or at dgardner@nuscalepower.com.

Sincerely,

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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 8774
RAI 15.00.03-1

Enclosure 2: NuScale Response to NRC Request for Additional Information eRAI No. 8774
RAI 15.00.03-2

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Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 8774 - RAI 15.00.03-1

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

eRAI No.: 8774
Date of RAI Issue: 04/25/2017

NRC Question No.: 15.00.03-1

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 10 CFR 52.47(a)(2)(iv)(A) and 10 CFR 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR Part 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 50.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents (DBAs) are evaluated against these regulatory requirements and the dose acceptance criteria given in NuScale DSRS 15.0.3.

Note that the referenced topical report cited in DCD Section 15.0.3.3. does not give the values for the (MHA) DBST source term release fractions, as implied in the last sentence of the text of this DCD Section. It only gives the method and example calculation.

Consequently, the staff requires the following information to conduct its review of the maximum hypothetical accident (MHA):

1. What accident scenario(s) are used to develop the design basis source term (DBST) for the MHA?
    a. Referenced topical report, TR-0915-17565-P, Rev.1, “Accident Source Term Methodology,” gives the methodology to choose scenarios – what was result of the use of the methodology?
    b. Are the severe accident scenarios used to develop the DBST the same ones evaluated in the PRA?
    c. Describe the accident scenario(s) that are used to develop the DBST, including initiating event and progression.

2. On DCD page 15.0-33 it states that core release fractions to containment are given in the referenced topical report. Where exactly in the topical report? This statement appears to be in conflict with various scope statements made in the topical report such as the following:
   - The topical report gives example release fractions related to example scenarios. In the second paragraph of Section 1.2 “Scope” of the topical report, it states that “This topical report is not intended to provide final DBST isotopic inventory values, final dose values, final atmospheric dispersion factors, or final values of any other associated accident evaluation; rather, example values for the various evaluations are provided for illustrative purposes.”
   - Topical report Section 4.2.3, “Release Timing and Magnitude,” describes the method for how the release timing and magnitude (fraction of core inventory) will be determined from the MELCOR calculations of candidate “source term DBAs,” but does not provide values for the core release fractions and their timing.
   - Section 5.0, “Example Calculation Results,” of the topical report states that the example calculation analyses and results are presented to demonstrate the application of the methodology, and that “NuScale plans to provide the final design values in the design certification application.”
3. What are the aerosol removal rates assumed in the DBA dose analyses?
4. DCD Tier 2 Tables 12.2-28, 12.2-29, and 12.2-30 look like they give MHA analysis parameters in discussing the post-accident source used in shielding analyses. If so, these tables should at least be referenced from DCD Tier 2 Section 15.0.3.9 or it may be more appropriate to move the information to Chapter 15.

NuScale Response:

Question 1:

The accident scenarios used to develop the design basis source term (DBST) for the maximum hypothetical accident (MHA) are selected using the methodology summarized in Section 4.2.1 of the Accident Source Term Methodology Topical Report (TR-0915-17565, Revision 1). All intact-containment internal events were considered when developing the DBST. The probabilistic risk assessment (PRA) is used to select a sample of sequences that span the range of module responses to a beyond-design basis accident, including both loss of coolant accidents (LOCA) and transient initiators, and different modes of emergency core cooling system (ECCS) failure. These sequences are logically displayed in the Final Safety Analysis Report (FSAR) Chapter 19, Figure 19.1-4 and Figure 19.1-10.

Although TR-0915-17565 and Revision 0 of the FSAR utilize the results of simulations using an older version of the MELCOR code and a preliminary version of the MELCOR plant input model, the results reported in the FSAR have been found to be bounding based on recent simulations. The more recent simulation results were produced using MELCOR 2.1 and an updated plant input model. The updated results are described below.

Five beyond-design basis intact-containment events were selected in which significant core damage was predicted to occur. Each of the five sequences involves failure of ECCS with either no valves opening, the reactor vent valves (RVVs) failing to open, or the reactor recirculation valves (RRVs) failing to open. In each case, the decay heat removal system (DHRS) is assumed available to remove heat, though DHRS was disabled in the model upon reaching core damage. The cases are:

1. A LOCA of the reactor coolant system (RCS) injection line with all ECCS valves failing to open.
2. A LOCA similar to case 1 with RVVs failing to open.
3. A LOCA similar to case 1 with RRVs failing to open.
4. Loss of DC power with the RVVs failing to open.
5. Loss of DC power with the RRVs failing to open.

Cases 1, 2, and 3 are represented in the event tree located in FSAR Figure 19.1-4, end-state 3, and are described in Section 19.1.4 (CVCS--ALOCA-CIC). The chemical and volume control charging line becomes the RCS injection line at the containment vessel wall. Case 1 includes the complete failure of the ECCS and results in the greatest release of fission products to the containment.

Case 2, which includes a failure of the RVVs to open, results in the most rapid onset of core damage. However, the open RRVs result in reflooding of the core following the transport of non-condensable gas to the containment. The result is a limited release of fission products from the fuel and a short release duration.

Case 3, which includes the failure of the RRVs to open, results in the second greatest fission product release to the containment. The lower release to the containment is due to the open RVVs resulting in a reduction in the driving force for flow through the failed injection line, which is the most direct pathway for fission products to the containment. Although the release to containment is reduced in comparison to Case 1, the onset of the release occurs sooner since the actuation of the RVVs increases the total coolant loss rate and accelerates uncovering of the fuel.

Cases 4 and 5 are represented in the event tree located in FSAR Figure 19.1-10, end-state 3, and are described in Section 19.1.4 (EDSS--LODC-----). Case 4, which is initiated with a loss of DC power and includes RVV failure, results in the lowest magnitude of release from the fuel and to the containment. The case is similar to Case 2 except for a more gradual accident progression and the fission product pathway to the containment is not as direct. The fission product pathway is through the RRVs in the downcomer, while the pathway for Case 2 is through the RCS injection line which lies above the core.

Case 5, which is initiated with a loss of DC and considers RRV failure, results in the greatest quantity of relocated core material and magnitude of release from the fuel. However, the release to the containment is less than Cases 1 and 3 since the release pathway to the containment is not as direct as the RCS injection line.

The range of release timing and fractions presented in FSAR Table 12.2-19 and Table 12.2-30 conservatively bound similar values generated from the PRA severe accident subset described above. Therefore, the FSAR range of release timing and fraction values used for the DBST are conservative. The severe accident subset, described above, which represent the source term design basis accidents, has been included in Section 15.0.3.

Questions 2, 3, and 4:

The release fractions and aerosol removal rates used for the DBST were determined using the methodology provided in TR-0915-17565, Accident Source Term Methodology Topical Report. The resultant release fractions used in the radiological consequences of the DBST are presented in FSAR Table 12.2-29. The aerosol removal rates are presented in FSAR Table 12.2-30. Additional DBST parameters are provided in FSAR Table 12.2-28. References to these tables have been added to FSAR Section 15.0.3.

Impact on DCA:

FSAR Sections 15.0.0.1, 15.0.3, and FSAR Table 15.0-12 have been revised as described in the response above and as shown in the markups provided in this response.

5) increase in reactor coolant inventory

6) decrease in reactor coolant inventory

7) radioactive release from a subsystem or component

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Table 15.0-1 lists the events selected for evaluation in Sections 15.1 through 15.7 and a list of computer codes used for analyzing each event. An additional event, the radiological consequences of the ~~maximum hypothetical accident (MHA)~~design basis source term (DBST) described in Section 15.0.3.9, is also included in Table 15.0-1.

15.0.0.2 Design Basis Event Classification

NuScale DBEs are classified by frequency of occurrence, including those events that are expected to occur within the NPM lifetime as well as those that are postulated but not expected to occur during the NPM lifetime. The NuScale DBE spectrum is developed by considering DBEs associated with current generation plants and unique events resulting from NuScale Power Plant design features, including review of PRA initiators. This approach ensures the design considers a broad spectrum of potential events. Classification by frequency of occurrence is used to assign the analysis acceptance criteria for the event.

The set of DBEs establishes the design adequacy of the NPM and NuScale Power Plant to limit radiological releases below regulatory guidelines.

15.0.0.2.1 Classification by Event Frequency and Type

Design basis event classification by frequency is based on four distinct categories:

- anticipated operational occurrences (AOOs)
- infrequent events (IEs)
- postulated accidents
- special events

Events that are expected to occur one or more times during an NPM lifetime are classified as AOOs. Events that are not expected to occur during an NPM lifetime are classified as IEs or postulated accidents or may be conservatively classified as AOOs. In general, events that are not considered to be within the design basis are evaluated in Chapter 19; however, those beyond design basis events (BDBEs) that are explicitly defined by regulation are addressed in this chapter. These events are termed special events.

Special events also encompass defense-in-depth and common cause failures (CCFs) of digital control systems, as described in Branch Technical Position 7-19. The IE category accommodates the anticipated lower frequency of NuScale event occurrence that results from the unique design features. These features include digital control systems that are redundant and fault tolerant. Infrequent events are

- integrated photon dose to the containment and total dose to the coolant
- initial mass of coolant
- mass of coolant
- temperature of coolant

The program then calculates the coolant temperature-dependent pH as a function of time.

15.0.2.4.7 MCNP6

The MCNP6 is used for evaluating potential shine radiological exposures or doses to operators in the control room following a radiological release event. Both sky-shine and shine from filters are evaluated. MCNP is a general-purpose tool used for neutron, photon, electron, or coupled neutron, photon, and electron transport. MCNP treats an arbitrary three-dimensional configuration of materials in geometric cells bounded by first- and second-degree surfaces and fourth-degree elliptical tori. The code is well-suited to performing fixed source calculations.

MCNP uses continuous energy cross-section data. For photons, the code accounts for incoherent and coherent scattering, the possibility of fluorescent emission after photoelectric absorption, and absorption in electron-positron pair production. Electron and positron transport processes account for angular deflection through multiple Coulomb scattering, collisional energy loss with optional straggling, and the production of secondary particles including x-rays, knock-on and Auger electrons, bremsstrahlung, and annihilation gamma rays from positron annihilation at rest. The MCNP code is commercially-grade dedicated under the NuScale NQA-1 program described in Reference 15.0-4.

15.0.3 Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors

RAI 15.00.03-1

This section presents the methodology used to perform the calculations associated with the radiological consequences of the ~~maximum hypothetical accident (MHA)~~DBST and other limiting event types. Table 15.0-11 identifies the list of events analyzed for radiological consequences. Results from the application of this methodology are provided in Table 15.0-12.

15.0.3.1 Introduction

RAI 15.00.03-1

This section describes the NuScale conservative methodology for developing accident source terms and performing the corresponding radiological consequence analyses. Key unique features of the NuScale methodology are the:

- use of ARCON96 to calculate off-site atmospheric dispersion factors

- development of a ~~design basis source term (~~DBST~~)~~ to meet the intent of 10 CFR 52.47(a)(2)(iv)
- use of the STARNAUA containment aerosol transport code in the range of NuScale containment conditions.

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10 CFR 52.47(a)(2)(iv) requires nuclear power reactor design certification applicants to evaluate the consequences of a fission product release into the containment assuming the facility is being operated at the maximum licensed power level and to describe those design features intended to mitigate the radiological consequences of an accident. NuScale follows the approach of the 2012 Nuclear Energy Institute (NEI) position paper on small modular reactor source terms (Reference 15.0-6) by referring to the scenario described in footnote 3 of 10 CFR 52.47(a)(2)(iv) as the maximum hypothetical accident (MHA). Source terms are divided into two principal categories titled "Category 1" and "Category 2."

The Category 1 source terms include deterministic accidents and are analyzed using the guidance of RG 1.183. Exceptions to RG 1.183 are due to differences between the NuScale Power Plant design and large LWRs, as outlined in Reference 15.0-4.

The Category 2 source term consists of the MHA scenario in which significant core damage occurs.

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The MHA has historically been linked to a large-break LOCA in large LWRs. The NPM has no large diameter primary coolant system piping; therefore, a large-break LOCA cannot be postulated as the basis for the MHA radiological consequence analysis for NuScale. ~~A surrogate methodology~~Surrogate accident scenarios, denoted as source term ~~DBA, is~~design basis accidents, are used to identify appropriate severe accident scenarios to address the MHA for the off-site and control room dose evaluations.

As stated in RG 1.183, "the design basis accidents were not intended to be actual event sequences, but rather, were intended to be surrogates to enable deterministic evaluation of the response of a facility's engineered safety features."

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The ~~term~~ DBST is ~~used for the surrogate core damage events discussed in RG 1.183~~composed of a set of key parameters, such as fuel release fractions and timing, derived from a spectrum of source term design basis accidents. The radiological consequence analysis of the MHA using the DBST is discussed in Section 15.0.3.9. However, there are no DBEs that result in significant core damage for the NuScale Power Plant.

Table 15.0-11 identifies design basis events evaluated for radiological consequences, cross references them to RG 1.183, and identifies the primary source of radiation for the event.

15.0.3.2 Methodology Overview

The methodology used to perform the Category 1 DBA and the Category 2 MHA events considers:

- atmospheric dispersion
- design basis source term
- containment aerosol generation and removal
- post-accident temperature dependent pH

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The Category 1 analyses follow the guidance of RG 1.183 methodology modified to reflect the difference in the NuScale Power Plant design from large PWRs. This methodology addresses the submersion and inhalation doses and the direct shine doses from contained or external sources. The key elements of this methodology are:

- Severe accident sequences are modeled using MELCOR.
- Thermal-hydraulic conditions are modeled using NRELAP.
- Source term and dose evaluations are calculated using RADTRAD.
- Meteorological dispersion is calculated using ARCON96.
- Fission product removal rates are calculated using approved methodology.
- ~~The gap and early in-vessel release values are described in Reference 15.0-4.~~
- The evaluation of post-accident pH on the chemical form of iodine is considered.

Section 15.0.2.4 summarizes the computer codes used for calculating MHA and DBE doses.

15.0.3.3 General Dose Analysis Inputs

The following sections summarize the key aspects for calculating DBA doses.

15.0.3.3.1 Core Radionuclide Inventory

The isotopic inventories of fuel assemblies are calculated using SCALE 6.1 which is described in Section 15.0.2.4. Isotopic concentrations are based on the detailed geometry of a fuel assembly, rated power plus uncertainty, maximum assembly average exposure, and a range of U-235 enrichments. The isotopic inventory is calculated at a number of time steps in the fuel cycle. Table 11.1-1 provides the maximum core isotopic inventory.

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15.0.3.3.2 Primary Coolant Radionuclide Inventory

For the radiological consequence analysis, the radioiodine concentrations in the primary coolant system are set at the maximum dose equivalent values permitted by design basis limits. Table 15.0-14 provides the primary coolant radionuclides and nominal inventory assumed in the dose analyses presented in Section 15.0.3.8. The iodine appearance rates, including the pre-incident appearance rates, are described in ~~Reference 15.0-4~~Section 15.0.3.8, where used.

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15.0.3.3.3 Secondary Coolant Activity

Large PWR designs contain a large volume of secondary system water on the "shell" side of the SG. Through primary-to-secondary leakage limits and monitoring by sampling, this water volume contains levels of iodine that are limited operationally. A sensitivity study was performed in Reference 15.0-4 for the steam generator tube failure (SGTF) and main steam line break (MSLB) events assuming the liquid secondary coolant in the SG was at the primary coolant design basis limit concentration. The sensitivity study demonstrated dose results are not sensitive to the initial secondary side activity. This conclusion is supported by comparing the secondary coolant source terms shown in Table 11.1-5 with the primary coolant source terms shown in Table 15.0-14.

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15.0.3.3.4 Source Term Release Fraction and Timing for Dose Analysis

The DBST is derived using the methodology described in Reference 15.0-4. The intact-containment internal events are considered when developing the DBST. A subset of these events are modeled using MELCOR to provide a representative range of release timing and fractions. Five beyond-design basis Level 1 PRA events were selected in which significant core damage was predicted to occur. Each of the five sequences involves failure of the ECCS with either no valves opening, the RVVs failing to open, or the RRVs failing to open. In each case, the DHRS is assumed available to remove heat. The cases are:

1) A LOCA of RCS injection line with ECCS valves failing to open.

2) A LOCA of RCS injection line with RVVs failing to open.

3) A LOCA of RCS injection line with RRVs failing to open.

4) Loss of DC power with the RVVs failing to open.

5) Loss of DC power with the RRVs failing to open.

The DBST on-site and off-site radiological consequences use event-specific radionuclide groups, release timing, release fractions, and aerosol removal, as described in Section 15.0.3.9. The aerosol removal rate as a function of time, and other key inputs such as atmospheric dispersion factors and isotopic inventories, are input into RADTRAD for calculating the radiological consequences.

~~Additional details about source term release fractions are provided in Reference 15.0-4.~~

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15.0.3.3.5 Release Timing and Magnitude

As with radionuclide groups, design-specific representative results for release timing and magnitude from severe accident evaluations are used for the methodology to reflect current practices and appropriately model the specific event. The approach used to identify the release magnitudes and timing are provided in Reference 15.0-4. Release timing and magnitude for the DBST are listed in Tables 12.2-28 and 12.2-29.

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15.0.3.3.6 Aerosol and Elemental Iodine Removal

Natural deposition phenomena including sedimentation, diffusiophoresis, thermophoresis, and hygroscopicity result in aerosol removal. The NuScale aerosol removal methodology uses the aerosol removal code STARNAUA to track these various deposition phenomena in calculating time-dependent airborne aerosol mass and removal rates. ~~A summary of the aerosol transport and removal calculation process is described in Reference 15.0-4.~~ Aerosol removal rates used for the DBST are located in Table 12.2-30.

~~15.0.3.3.7 Elemental Iodine Removal~~

~~Deposition of elemental iodine onto inner-containment surfaces results in removal.~~ A key assumption of the NuScale aerosol transport methodology is that there is no maximum limit on ~~elemental~~ iodine decontamination factor because removal is facilitated by natural processes as opposed to an active spray system. The NuScale removal rate calculation methodology is based on the calculated time-dependent airborne aerosol mass in accordance with Appendix A, Section 3.3 of RG 1.183. NuScale conservatively does not take credit for elemental iodine removal. Rather, only aerosol removal is credited. A summary of the aerosol transport and removal calculation process is described in Reference 15.0-4.

15.0.3.3.8 Aerosol Resuspension and Revaporization

Treatment of aerosol resuspension and revaporization is discussed in Reference 15.0-4.

15.0.3.3.9 RADTRAD Modeling

Consistent with RG 1.183:

- The RADTRAD decay and daughter product modeling option is used to include progeny from the decay of parent radionuclides that are significant with regard to radiological consequences and the released radioactivity. The calculated total effective dose equivalent (TEDE) is the sum of the committed effective dose equivalent from inhalation and the deep dose equivalent from external exposure from tracked isotopes.
- RADTRAD does not include corrections for depletion of the effluent plume by deposition on the ground.

- RADTRAD determines the maximum two-hour TEDE by calculating the postulated dose for a series of small time increments and performing a "sliding" sum over the increments of successive two-hour periods.

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15.0.3.3.10 Post-Accident pH_T Code

The DBST methodology calculates the post-accident temperature-dependent pH. The pH_T code is used to calculate the extent of iodine re-evolution inside containment. During the postulated DBST event, additional acids and bases may enter the coolant and cause a change in pH. The expected overall pH of the coolant is modeled over a time period of 30 days. Section 15.0.2.4 provides a discussion of the NuScale pH_T program used to calculate post- accident temperature-dependent pH.

Details about the methodologies used for evaluating post-accident temperature-dependent pH in coolant water following a significant core damage event are presented in Reference 15.0-4. The results of implementing the methodology show that the post-accident temperature-dependent pH inside containment is between 6.0 and 7.0.

15.0.3.3.11 Iodine Re-evolution

The DBST methodology assumes a negligible amount of iodine re-evolution occurs between temperature-dependent pH values of 6.0 and 7.0 and does not need to be explicitly included in the dose analysis calculation. This position simplifies the analysis without an impact to the conservatism of the calculated dose results. The treatment of iodine re-evolution is described in Reference 15.0-4.

15.0.3.3.12 Atmospheric Dispersion Factors (χ/Q), Breathing Rates, and Occupancy Factors

Atmospheric dispersion factor (χ/Q) inputs to RADTRAD are derived as described in Reference 15.0-4. Table 2.0-1 provides the accident release χ/Q values.

Control room and offsite breathing rate and control room occupancy factor inputs to RADTRAD, consistent with RG 1.183, are listed in Table 15.0-13.

15.0.3.3.13 Dose Conversion Factors

Consistent with RG 1.183, dose conversion factors from Environmental Protection Agency Federal Guidance Report No. 11 (Reference 15.0-8) and Report No. 12 (Reference 15.0-9) are used for dose analysis.

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15.0.3.4 Containment Leakage

Containment leakage is described in Reference 15.0-4 ~~and~~, is consistent with the recommendations of RG 1.183, and is listed in Table 12.2-28.

15.0.3.5 Secondary-Side Decontamination

The helical coil steam generators of the NuScale Power Plant design are different than that of a large PWR because the primary coolant is on the outside of the tubes. As a result, there is no bulk water volume in which decontamination can easily occur. Reference 15.0-4 provides the details about the decontamination factor used in the helical coil steam generators as well as the treatment of iodine deposition in the main steam piping and the condenser.

15.0.3.6 Reactor Building Decontamination Factors

Reactor Building RXB decontamination factors are described in Reference 15.0-4.

15.0.3.7 Receptor Location Considerations

RAI 15.00.03-2

Potential on-site radiological receptor locations considered in this evaluation are the control room and TSC; potential off-site locations are the EAB and LPZ. Figure 15.0-3 shows the schematic of the RADTRAD code nodalization used to model these locations for leakage paths from the containment or RXB. Figure 15.0-4 shows the RADTRAD code nodalization for the SGTF and MSLB events in which the principal release path is through the steam generator.

A summary of control room characteristics are provided in ~~Reference 15.0-4~~Section 15.0.3.7.1. The variables associated with the derivation of these receptors are presented in Table 15.0-13.

RAI 15.00.03-2

15.0.3.7.1 Control Room Design

Accident analyses are performed for two control room emergency modes as follows:

- Uninterrupted power supply with continuous filtered airflow to the control room envelope for the event duration.
- Immediate loss of power with control room habitability system (CRHS) activation, and restored filtered airflow to the control room envelope at the time of CRHS depletion (72 hours).

RAI 15.00.03-2

Simplifying assumptions are made for the control room ventilation system design. Figure 15.0-3 and Figure 15.0-4 show the control room RADTRAD code

15.0.3.8.5 Fuel Handling Accident

A fuel handling accident is postulated to occur during the movement of the fuel resulting in a dropped assembly onto the spent fuel racks, in the reactor vessel during refueling, in a spent fuel cask during loading, or on the weir wall between the reactor pool and SFP. The methodology for determining fuel handling accident radiological consequences is based on the guidance provided in Appendix B of RG 1.183. The analysis follows the guidance in Appendix B of RG 1.183, with the exception that the iodine decontamination factor is calculated with a generalized methodology instead of utilizing the values in RG 1.183 because the height of water above the damaged fuel is greater than 23 feet in the NuScale reactor and SFP. Reference 15.0-4 describes the methodology for determining the SFP iodine decontamination factor. The calculated iodine decontamination factor used in the fuel handling accident dose analysis is 250.

The inventory of fission products available for release at the time of the accident is dependent on a number of factors, such as the power history of the fuel assembly, the time delay between reactor shutdown and the beginning of fuel handling operations, the volatility of the nuclides, and the number of fuel rods damaged in a fuel assembly handling accident. The activity available for release is based on 102 percent power, bounding core inventory provided in Table 11.1-1, and a 1.4 radial peaking factor with 48 hours decay from time of reactor shutdown to the beginning of fuel handling operation. Activity is instantaneously released into the pool water from all fuel rods in the dropped assembly.

The following is a summary of the assumptions used from Appendix B of RG 1.183:

- radionuclides considered include xenon, krypton, halogen, cesium, and rubidium
- release fractions are from RG 1.183, Table 3
- ~~iodine chemical form released from the pool is 57 percent elemental iodine and 43 percent organic iodide~~
- no reduction or mitigation of noble gas radionuclides released from the fuel is assumed
- radionuclides are released to the environment over a two-hour period

There are no single failures assumed for this event. Noble gases and iodines are released from the pool, while the cesiums and rubidiums are particulates and remain in the pool. The activity released from the pool to the RXB is assumed to be instantaneously released to the environment without holdup or mitigation. Doses are determined at the EAB, LPZ, and for personnel in the control room and TSC. The control room model is described in Section 15.0.3.7.1. The potential radiological consequences of a fuel handling accident are summarized in Table 15.0-12.

RAI 15.00.03-1

15.0.3.9 Radiological Analysis of the Category 2 Maximum Hypothetical Accident

Section 15.0.3.1 discusses how a MHA has historically been linked to a large-break LOCA in large LWRs and that, for the NPM, a large-break LOCA cannot physically be postulated as the basis for the MHA radiological consequence analysis. Section 15.6.5 presents the LOCA analysis, which shows that no fuel failures occur. However, 10 CFR 52.47 (a)(2)(iv) requires nuclear power reactor design certification applicants to evaluate the consequences of a fission product release into the containment assuming substantial meltdown of the core. Therefore, this section ~~presents the MHA, with fuel damage, even though the substantial fuel damage cannot physically be postulated for the NuScale design~~presents source term design basis accidents. A source term design basis accident is a postulated accident scenario, meant as a surrogate to the large break LOCA typically evaluated by LWRs to meet the regulatory intent of addressing the MHA. Five source term design basis accidents derived from the Level 1 PRA were used to establish the DBST described in Section 15.0.3.3.4 in accordance with the methodology of Reference 15.0-4. Parameters associated with the DBST are presented in Table 12.2-28, Table 12.2.29, and Table 12.2-30.

RAI 15.00.03-1

To address 10 CFR 52.47 (a)(2)(iv), ~~a MHA~~the DBST is assumed to occur, resulting in significant core damage. Activity is assumed to be released from the fuel over a specified time period, as described in Reference 15.0-4 and presented in Table 12.2-28, and assumed to homogeneously mix in the containment atmosphere. Removal of aerosol in the containment occurs through natural deposition phenomena. The aerosol removal methodology utilizes the code STARNAUA to determine the time-dependent airborne aerosol mass and removal rates, as described in Reference 15.0-4. Activity is released to the atmosphere from the containment at the design basis leakage rate for 24 hours, and at 50% of the limit after 24 hours.

Reference 15.0-4 provides the methodology for the radiological consequences of the MHA, based on the guidance provided in Appendix A of RG 1.183. Assumptions used from Appendix A of RG 1.183 are:

- The chemical form of radioiodine released to the containment atmosphere is 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. Note that the methodology considers cesium iodide as an aerosol.
- The radioactivity released from the fuel is assumed to mix instantaneously and homogeneously throughout the free air volume of the containment.

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The radioactive source term is calculated from the maximum core inventory provided in Table 11.1-2, multiplied by the release fractions provided in ~~Reference 15.0-4~~Table 12.2-29. The timing of the release and the radionuclide groups assumed, and the iodine removal mechanisms in the containment are ~~also~~ provided in ~~Reference 15.0-4~~Table 12.2-28 and Table 12.2-30.

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RADTRAD is used to determine the dose, as outlined in Section 15.0.3.3.9. The control room model is described in Section 15.0.3.7.1. The potential radiological consequences of the ~~MHA~~DBST are presented in Table 15.0-12.

15.0.4 Safe, Stabilized Condition

Safety analyses of design basis events are performed from event initiation until a safe, stabilized condition is reached. A safe, stabilized condition is reached when the initiating event is mitigated, the acceptance criteria are met and system parameters (for example inventory levels, temperatures and pressures) are trending in the favorable direction. For events that involve a reactor trip, system parameters continue changing slowly as decay and residual heat are removed and the RCS continues to cool down. No operator action is required to reach or maintain a safe, stabilized condition.

Two additional considerations are discussed to show that Chapter 15 acceptance criteria are not challenged beyond the safe, stabilized condition. Long term decay and residual heat removal is discussed in Section 15.0.5 and a potential return to power is discussed in Section 15.0.6.

15.0.5 Long Term Decay and Residual Heat Removal

There are two systems that perform the safety-related function of decay and residual heat removal from the NPM following a DBE. The DHRS, described in Section 5.4.3, provides decay and residual heat removal while RCS inventory is retained inside the RPV, the containment is maintained in partially evacuated dry conditions, and power is available. The ECCS, described in Section 6.3, provides decay and residual heat removal when RCS inventory has been redistributed between the RPV and the CNV after the RVVs and RRVs are opened.

The DBEs listed in Table 15.0-1 progress from initiation of the event to effective DHRS or ECCS operation demonstrating that the NPM has reached a safe, stabilized condition, as described in Section 15.0.4. The decay heat removal process continues into the long-term phase, either with DHRS, natural circulation between the CNV and RPV through the RRVs and RVVs, or a combination of the two.

There are four decay and heat removal scenarios:

1) DHRS,

2) DHRS with the RVVs and RRVs opening 24 hours after a loss of normal AC power,

3) DHRS with the RVVs and RRVs opening after a loss of normal AC and normal DC power when the IAB pressure threshold is reached, and

4) ECCS actuation following an inadvertent opening of a reactor coolant pressure boundary (RCPB) valve or a LOCA.

Scenario 1 - Decay and Residual Heat Removal using DHRS

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Table 15.0-12: Radiological Dose Consequences for Design Basis Analyses

Event	Location	Acceptance Criteria (rem TEDE)	Dose (rem TEDE)
Failure of Small Lines Carrying Primary Coolant Outside Containment	EAB	6.3	0.10
	LPZ	6.3	0.17
	CR	5.0	0.32
Steam Generator Tube Failure (pre-incident iodine spike)	EAB	25.0	0.36
	LPZ	25.0	0.36
	CR	5.0	0.65
Steam Generator Tube Failure (coincident iodine spike)	EAB	2.5	0.05
	LPZ	2.5	0.05
	CR	5.0	0.02
Main Steam Line Break (pre-incident iodine spike)	EAB	25.0	<0.01
	LPZ	25.0	0.02
	CR	5.0	0.05
Main Steam Line Break (coincident iodine spike)	EAB	2.5	<0.01
	LPZ	2.5	<0.01
	CR	5.0	<0.01
Fuel Handling Accident	EAB	6.3	0.42
	LPZ	6.3	0.42
	CR	5.0	0.71
~~Maximum Hypothetical Accidents~~Design Basis Source Term (significant core damage)	EAB	25.0	0.50
	LPZ	25.0	1.44
	CR	5.0	2.29

Enclosure 2:

NuScale Response to NRC Request for Additional Information eRAI No. 8774 - RAI 15.00.03-2

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

eRAI No.: 8774
Date of RAI Issue: 04/25/2017

NRC Question No.: 15.00.03-2

Criteria on control room habitability, including dose to operators during accidents, are provided in 10 CFR Part 50, Appendix A, GDC 19. The design basis accident (DBA) dose analyses in DCD Tier 2 Chapter 15 were performed, in part, to show compliance with GDC 19.

In DCD Tier 2 Section 15.0.3.7.1 it states that the control room ventilation system design modeling assumptions are given in the referenced topical report, TR-0915-17565-P, Rev.1, "Accident Source Term Methodology." Although not specifically referenced, Section 3.3.4.1 of the topical report provides some information on control room ventilation design. Table 3-4 of the topical report is titled "Example control room characteristics," and, as noted in the text in Section 3.3.4.1, is only intended to provide the values used in the example calculations provided in the topical report. Information on the control room characteristics relevant to radiological protection and used in the estimation of dose to control room operators is also not evident in DCD Tier 2, Sections 6.4 and 9.4.1. Furthermore, the example values in the referenced topical report give much different values for component flow rates than those given in DCD 6.4 and 9.4.1.

In order to complete its review of the applicant's evaluation of the DBA control room radiological habitability, the staff requires additional information. Provide the following information used as assumptions and inputs to the applicant dose analyses that support the evaluations in Sections 6.4 and

of the DCD and make revision to the DCD to document:

- Control room envelope (CRE) volume
- Normal control room HVAC (heating, ventilation and air conditioning) system (CRVS) normal ventilation intake, unfiltered flow rate

- CRVS post-accident supplemental filtration mode intake flow rate (if supplemental filtration credited after 72 hours)
- CRVS recirculation flow rate and whether recirculation flow is filtered
- Intake radiation monitor setpoints for control room isolation and initiation of the control room habitability system (CRHS) and for initiation of the CRVS supplemental filtration mode
- CRHS initiation time based on intake radiation high signal, per DBA

NuScale Response:

Two control room Emergency Modes are evaluated for analyzing control room dose. The bounding results are reported in FSAR Table 15.0-12. The modes evaluated are:

1. Uninterrupted power supply with continuous filtered airflow to the control room envelope for the event duration.
2. Immediate loss of power with control room habitability system (CRHS) activation and restored filtered airflow to the control room envelope at the time of CRHS depletion (72 hours).

The control room and control room ventilation system parameters assumed in control room dose analyses have been added to FSAR Table 15.0-15.

The setpoint for the radiation monitor to redirect air through the air filtration unit is 10 times background radiation. The setpoint for CRHS initiation and control room envelope isolation is 10 times the expected radiation out of the filtration unit following a design basis event, which indicates a failure of the filtration unit to remove sufficient radioactivity. The time between when the radiation concentration reaches the detector setpoint and radiation enters the control room or technical support center (TSC) envelopes is assumed to be zero. The electric signal travels much faster than air, thus as a result of the location and distance between the sensors in relation to the dampers, the mechanical dampers will be closed before the radioactivity enters the control room or TSC envelopes.

Although FSAR Figure 15.0-3 and FSAR Figure 15.0-4 indicate a filter for recirculation flow, the filter is only for model sensitivity studies and no filtration of recirculation flow is assumed or credited in the dose analyses.

FSAR Section 15.0.3.7 has been revised to add clarifying text describing control room dose analyses assumptions.

Impact on DCA:

FSAR Section 15.0.3.7 and FSAR Table 15.0-15 have been revised as described in the response above and as shown in the markups provided in this response.

15.0.3.4 Containment Leakage

Containment leakage is described in Reference 15.0-4 ~~and~~, is consistent with the recommendations of RG 1.183, and is listed in Table 12.2-28.

15.0.3.5 Secondary-Side Decontamination

The helical coil steam generators of the NuScale Power Plant design are different than that of a large PWR because the primary coolant is on the outside of the tubes. As a result, there is no bulk water volume in which decontamination can easily occur. Reference 15.0-4 provides the details about the decontamination factor used in the helical coil steam generators as well as the treatment of iodine deposition in the main steam piping and the condenser.

15.0.3.6 Reactor Building Decontamination Factors

Reactor Building RXB decontamination factors are described in Reference 15.0-4.

15.0.3.7 Receptor Location Considerations

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Potential on-site radiological receptor locations considered in this evaluation are the control room and TSC; potential off-site locations are the EAB and LPZ. Figure 15.0-3 shows the schematic of the RADTRAD code nodalization used to model these locations for leakage paths from the containment or RXB. Figure 15.0-4 shows the RADTRAD code nodalization for the SGTF and MSLB events in which the principal release path is through the steam generator.

A summary of control room characteristics are provided in ~~Reference 15.0-4~~Section 15.0.3.7.1. The variables associated with the derivation of these receptors are presented in Table 15.0-13.

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15.0.3.7.1 Control Room Design

Accident analyses are performed for two control room emergency modes as follows:

- Uninterrupted power supply with continuous filtered airflow to the control room envelope for the event duration.
- Immediate loss of power with control room habitability system (CRHS) activation, and restored filtered airflow to the control room envelope at the time of CRHS depletion (72 hours).

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Simplifying assumptions are made for the control room ventilation system design. Figure 15.0-3 and Figure 15.0-4 show the control room RADTRAD code nodalization used in the dose analyses. The key design features assumed for the control room are summarized as follows:

- The nonsafety-related normal control room ventilation system inlet filters remove 99 percent of iodine. Figure 15.0-3 and Figure 15.0-4 indicate a filter for recirculation flow. The filter is only for model sensitivity studies. No filtration of recirculation flow is assumed.
- The nonsafety-related normal control room ventilation is isolated by a safety-related control system once the radioactivity measured at the duct intake reaches the isolation signal setpoint. The setpoint for the radiation monitor to redirect air through the air filtration unit is 10-times background. The setpoint for CRHS initiation and CRE isolation is 10-times the expected radiation out of the filtration unit following a DBE, which indicates a failure of the filtration unit to remove sufficient radioactivity. The time between when the radiation concentration reaches the detector setpoint and radiation enters the control room or technical support center (TSC) envelopes is assumed to be zero seconds. The electric signal travels much faster than air, thus the mechanical dampers will be closed before the radiation enters the control room or TSC envelopes.
- An emergency source of pressurized air with the control room habitability system (CRHS) provides clean air for 72 hours.
- After 72 hours of CRHS operation, the normal control room ventilation system is available for use.
- The control room is habitable during a loss of normal AC power as the CRHS automatically activates after 10 minutes without normal AC power, as described in Section 6.4.3.
- Control room ventilation is designed to minimize in-leakage.
- The control room is designed with a two-door air lock system. Therefore, in-leakage of 5 cfm is assumed for ingress and egress. An additional 10 cfm of in-leakage is also assumed.

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The control room ventilation system design modeling assumptions are provided in ~~Reference 15.0-4~~Table 15.0-15. Details about system operation with CRHS are provided in Section 6.4 and Section 9.4.1.

No credit is taken for the use of personal protective equipment, such as beta radiation resistant protective clothing, eye protection, or self-contained breathing apparatus. No credit is taken for prophylactic drugs such as potassium iodide pills.

Potential shine radiological exposures to operators within the control room following a radiological release event are evaluated. Direct shine, sky-shine and shine from filters are evaluated using MCNP, as described in Section 15.0.2.4.7. Reference 15.0-4 provides additional details regarding the calculation of shine doses. The 30-day cumulative doses due to either recirculation filter or cloud-shine in the control room are added to the dose results from DBEs provided in Table 15.0-12.

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Table 15.0-15: Control Room Parameters

Parameter	Units	Value
Control Room Envelope Volume	ft^3	74,680
Control Room Emergency Duration	hr	72
Control Room Emergency Flow Rate	cfm	100
Control Room Normal Flow Rate	cfm	742
Control Room Recirculation Flow Rate	cfm	7011
Control Room Unfiltered Ingress/Egress	cfm	5
Control Room Unfiltered Inleakage	cfm	147