NuScale Power

RAIO-0817-55190

August 01, 2017

Docket No. 52-048

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

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 76 (eRAI No. 8792) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 76 (eRAI No. 8792)," dated June 28, 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 Enclosure to this letter contains NuScale's response to the following RAI Questions from NRC eRAI No. 8792:

- 15.00.03-3
- 15.00.03-4
- 15.00.03-5
- 15.00.03-6

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 Darrell Gardner at 980-349-4829 or at dgardner@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 8792

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 8792

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

eRAI No.: 8792
Date of RAI Issue: 06/28/2017

NRC Question No.: 15.00.03-3

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.

Design certification document (DCD) Tier 2 Sections 15.0.3 and 15.6.2 present information on the radiological consequence analysis for the failure of small lines carrying primary coolant outside containment. The staff requests the following additional information to understand the applicant's analyses and complete its review:

a. DCD Section 15.0.3.8.1 states that flow from the chemical and volume control system (CVCS) through the side of the break away from the reactor contributes less than 15,000 pounds (mass) (lbm) of additional primary coolant to the release. Is this release contribution included in the mass release values listed in DCD Table 15.6-5, or is it an additional mass release? Please clarify how this release from the CVCS is modeled in the dose analysis, including necessary revisions to the DCD text.

b. What is the assumed initial mass of primary coolant in the CVCS?

c. DCD Section 15.0.3.8.1 states that the post-isolation leakage through one containment isolation valve for either break location (CVCS letdown line or CVCS makeup line) is assumed to occur at the maximum design basis leak rate. Is the referenced design basis leak rate the technical specification (TS) 3.4.5 LCO limit value for RCS identified leakage (0.5 gallons per minute (gpm)), unidentified leakage (2 gpm) the total of both identified and unidentified leakage (2.5 gpm), or some other value? Please clarify what parameter value was used in the dose analysis and provide its basis, including necessary revisions to the DCD text.

NuScale Response:

Response to item a:

FSAR Section 15.0.3.8.1 states "In addition, primary coolant in the CVCS equipment (heat exchangers, filter, etc.) and piping within the RXB flows out of the other side of the break contributing less than 15,000 lbm additional primary coolant to the release." FSAR Table 15.6-5 provides the mass release for the maximum mass release and maximum iodine spiking time thermal hydraulic cases. FSAR Figure 15.6-2 provides the integrated break mass flow for the maximum mass release scenario, which shows that the mass release from the RCS side of the break is equivalent to the results reported in FSAR Table 15.6-5. The conservatively assumed release from the CVCS equipment of 14,150 lbm is in addition to the mass release from the RCS side of the break reported in FSAR Table 15.6-5. FSAR Section 15.0.3.8.1 has been revised to clarify that the mass released from CVCS equipment and piping is added to the mass released from the reactor.

Response to item b:

The initial mass of primary coolant in the CVCS is calculated by summing the total volume of the CVCS and module heatup system (MHS) piping and equipment (recirculation, purification, makeup, letdown, and MHS lines). Note that the MHS is only used during the startup process to bring the system up to operating temperature. However, MHS is conservatively included in the volume assumed to drain into the reactor building. The calculated volume is increased by 30% to obtain the total volume of water external to containment that is assumed to drain into the reactor building. As an additional conservatism, when converting from volume to mass, the water is assumed to be at 70 degrees Fahrenheit. Thus, the initial mass of primary coolant from the CVCS is approximately 14,043 lbm. This value is then conservatively increased to 14,150 lbm for the radiological consequence analysis. This value is added to the RCS mass release given in Table 15.6-5 for the dose calculations discussed in Item a.

Response to item c:

The leakage rate from the reactor pressure vessel to the reactor building is based on Technical Specification 3.6.1.1, Containment Systems leakage rate. The containment is designed with an allowable leakage rate of 0.20% of containment air weight per day after a design basis accident. The analysis assumes that one containment isolation valve has failed open and that the entire allowable leakage rate is through the one closed containment isolation valve, which equates to 17.5 scfh, which equates to 8.2E-3 cfm.

Impact on DCA:

FSAR Section 15.0.3.8.1 has been revised as described in the response above and as shown in the markup provided in this response.

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Table 15.6-5 provides the assumed mass released from the reactor and the break isolation times for the two scenarios. The total mass released from the event is the sum of the mass released from the reactor provided in Table 15.6-5 and the primary coolant from CVCS equipment and piping discussed above.

Before containment isolation occurs, primary coolant flows out of the reactor vessel through the break at a rate and duration as described in Section 15.6.2. The coolant flow results in a time-dependent release of activity in the RXB that is conservatively modeled as a direct release to the environment. After containment isolation, primary coolant leaks through one containment isolation valve (the redundant in-series valve is assumed to fail open) at the maximum leak rate allowed by design basis limits. The activity from this leak path is also assumed to flow directly to the environment with no mitigation or reduction by intervening structures. After 30 hours, the reactor is assumed to be shut down and depressurized, and releases through the containment isolation valve stop.

The following is a summary of the assumptions used from Appendix E (main steam line break) of RG 1.183:

- coincident iodine spiking factor- 500
- duration of coincident iodine spike- 8 hours
- iodine chemical form- 97 percent elemental iodine and 3 percent organic iodide
- activity released from the fuel due to the iodine spike is assumed to mix instantaneously and homogeneously within the primary coolant in the reactor vessel
- no reduction or mitigation of noble gas radionuclides released from the primary system

The primary coolant in the reactor vessel and CVCS equipment and piping in the RXB initially contains the allowable concentration of dose equivalent (DE) I-131 of 0.2 μCi/gm and DE Xe-133 of 60 μCi/gm.

There are no single failures for this event that affect the thermal-hydraulic response of the NPM. However, the failure of one of the two containment isolation valves on the faulted line is assumed in the dose consequence analysis.

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 small lines carrying primary coolant break outside containment event are presented in Table 15.0-12.

15.0.3.8.2 Steam Generator Tube Failure

Radiological consequences of the SGTF are calculated based on the guidance provided in Appendix F of RG 1.183.

NuScale Power

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NRC Question No.: 15.00.03-4

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. The staff requests the following additional information to understand the applicant's analyses and complete its review:

Design certification document (DCD) Tier 2 Sections 15.0.3 and 15.1.5.4 present information on the radiological consequence analysis for the main steam line break (MSLB) outside containment. DCD Section 15.0.3.8.3, Item 3 states that the analysis assumption for primary-to-secondary flow through the intact steam generator (SG) is the maximum leak rate of 150 gallons per minute allowed by design basis limits. TS LCO 3.4.5 limits primary-to-secondary flow through the SGs to 150 **gallons per day**. Please clarify which value is used in the MSLB dose analysis.

NuScale Response:

The primary-to-secondary leak rate assumed in the main steam line break analysis is 150 gallons per day. FSAR Section 15.0.3.8.3, item 3, has been corrected.

Impact on DCA:

FSAR Section15.0.3.8.3 has been revised as described in the response above and as shown in the markup provided in this response.

As described in Section 15.6.3, a single failure of the main steam isolation valve (MSIV) for the faulted SG delays isolating the steamline, resulting in a larger release. A loss of normal AC power causes the steamline to isolate earlier, limiting the release. Therefore, a loss of normal AC power is not assumed to occur for the portion of the SGTF analysis used to determine the radiological releases. However, a loss of normal AC power is assumed to occur for the thermal-hydraulic portion of the SGTF analysis used to determine peak pressures presented in Section 15.6.3.

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 dose results for the SGTF event are presented in Table 15.0-12.

15.0.3.8.3 Main Steam Line Break Outside Containment Accident

Radiological consequences of the MSLB outside containment accident are calculated based on the guidance provided in Appendix E of RG 1.183. Section 15.1.5 describes the sequence of events and thermal-hydraulic response to a MSLB outside containment.

The radiological dose consequence analysis considers the MSLB event with two different initial iodine concentrations, one based on a pre-incident iodine spike and the other based on a coincident iodine spike. A description of the scenario evaluated is summarized as follows

1) An MSLB occurs in one of the two main steam lines.

2) The iodine and noble gas coolant activity is calculated based on the maximum concentrations allowed by design basis limits for each of the iodine spiking scenarios. The primary coolant contains a concentration of 0.2 µCi/gm DE I-131 for the coincident Iodine spike scenario and 12 µCi/gm DE I-131 for the pre-incident Iodine spike scenario. For both iodine spiking scenarios, the primary coolant contains 60 µCi/gm DE Xe-133.

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3) Primary coolant leaks into the secondary side of the intact SGs at the maximum leak rate of 150 gallons per day~~minute~~ allowed by design basis limits. The leakage continues until the primary system pressure is less than the secondary system pressure.

4) A time-dependent release is modeled that effectively releases the activity directly to the environment through the break.

5) The non-faulted steam line continues to release a small quantity of radiation through valve leakage.

The assumptions used from Appendix E of RG 1.183 are:

- coincident iodine spiking factor- 500
- duration of coincident iodine spike- 8 hr
- density for leak rate conversion- 62.4 lbm/ft^3

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NRC Question No.: 15.00.03-5

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.

Design certification document (DCD) Tier 2 Sections 15.0.3 and 15.7.4 present information on the radiological consequence analysis for the fuel handling accident (FHA). Although the DCD refers to the methodology in topical report TR-0915-17565-P, Rev.1, “Accident Source Term Methodology,” the staff notes that the assumed iodine effective decontamination factor (DF) for the reactor pool is different than the example given in the topical report. The staff requests the following additional information to understand the applicant's analyses and complete its review of the DCD:

a. DCD Section 15.0.3.8.5 states that the height of water above the damaged fuel is greater than 23 feet. What water depth was assumed in the calculation of the iodine effective decontamination factor for the reactor pool? Explain how the pool depth assumed in the FHA dose analysis is assured by the TS 3.5.3 Ultimate Heat Sink minimum level of 68 ft.

b. What is the depth of water above the damaged fuel when the dropped assembly lands on the following locations, including consideration of whether the fuel settles to a horizontal position or may be upright or at an angle to the surface:
    i. spent fuel storage racks or the fuel within the racks
    ii. the fuel in the reactor vessel when in the refueling stand
    iii. fuel in a spent fuel cask during loading
    iv. other structures in the reactor pool or spent fuel pool, including the weir wall between the reactor pool and the spent fuel pool

c. DCD Section 15.0.3.8.5 gives values for the relative percentage of iodine chemical forms released from pool that are the same as given in RG 1.183, which assumes an iodine effective DF which is less than that assumed in the NuScale FHA dose analysis. However, given the same values for fractional release from the fuel by iodine chemical form as assumed in RG 1.183, the ratio of inorganic to organic iodine released from the pool would be different than the values given in RG 1.183 when the iodine effective DF is different. The estimated DF for inorganic iodine forms and the iodine effective DF are calculated using the methodology from the referenced topical report, with the resulting iodine effective DF a function of the inorganic iodine DF. No retention of organic iodine in the pool water is assumed. The difference in the DF between the inorganic and organic forms of iodine causes a shift in the inorganic/organic ratio released from the pool as compared to the release from the fuel. The iodine chemical form is important when evaluating the effect of fission product mitigation systems (such as control room ventilation filtration systems) where the mitigation capabilities may be different for the different chemical forms of iodine. Therefore, please provide corrected information on re-normalized iodine chemical forms for the release from the pool based on the calculated DF for inorganic forms of iodine that is the basis for the iodine effective DF.

NuScale Response:

Response to item a:

The fuel handling accident (FHA) is being revised to use the decontamination factor based on 23 feet of water above the damaged fuel rods, consistent with Regulatory Guide 1.183. FSAR Section 15.0.3.8.5 has been revised to present the FHA consistent with Regulatory Guide 1.183.

The FHA assumes that a fuel assembly detaches from the assembly hook during movement of the assembly from one point to another point in the reactor pool. The dropped assembly comes to rest horizontally on top of the weir wall separating the refueling pool from the spent fuel pool. The top of the weir wall is the highest point in the pool area, as seen in FSAR Figure 9.1.3-5. Any other rest location would provide additional water above the dropped assembly. The top of the weir wall is 48 feet below the minimum pool level of 68 feet. Thus, the pool depth assumed in the FHA of 23 feet is assured by TS 3.5.3.

Response to item b.

As stated in the response to item a, it is assumed in the FHA that the fuel assembly comes to rest in the horizontal position on top of the weir wall. Any other rest location for the fuel assembly will provide additional water above the dropped assembly. From FSAR Figure 9.1.3-5, the top of the spent fuel storage racks is at 34 feet and 4 inches building elevation, which is 10 feet and 8 inches lower (or greater height of water) than the weir wall at 45 feet elevation.

The top of the reactor flange is 39.6 feet elevation, which is 5.4 feet lower than the weir wall. The top of fuel in the reactor vessel is lower than that.

The weir wall is the highest structure in the pool, thus if the dropped fuel assembly came to rest on other structures in the pool, including a spent fuel shipping cask, the height of water above the assembly would be greater.

Response to item c.

NuScale will now be following the guidance of Regulatory Guide 1.183 for the FHA. Therefore, the relative percentages of iodine chemical forms released from the pool are the same as provided in Regulatory Guide 1.183.

Impact on DCA:

FSAR Sections 15.0.3.8.5 and Table 15.0-12 have been revised as described in the response above and as shown in the markup provided in this response.

- iodine chemical form of 97 percent elemental iodine and 3 percent organic iodide
- no reduction or mitigation of noble gas radionuclides released from the primary system

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 steam system piping failure outside the primary containment are summarized in Table 15.0-12.

15.0.3.8.4 Rod Ejection Accident

Radiological consequences of a rod ejection accident (REA) are calculated based on the guidance provided in Appendix H of RG 1.183. Section 15.4.8 describes the sequence of events and thermal-hydraulic response to an REA which shows that the REA does not result in fuel failure. Therefore, per Appendix H of RG 1.183, a radiological analysis is not required as the consequences of this event are bounded by the consequences of other analyzed events.

15.0.3.8.5 Fuel Handling Accident

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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 weir wall provides the highest point in the reactor pool on which a fuel assembly could come to rest. Therefore, it is assumed that the dropped fuel assembly lands horizontally on the top of the weir wall providing the minimum water depth above the dropped assembly. The methodology for determining fuel handling accident radiological consequences is ~~based on~~consistent with 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

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- depth of water above the damaged fuel of 23 feet is assumed

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- overall effective decontamination factor of 200 is assumed
- 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.

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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.

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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

RAI 15.00.03-1, RAI 15.00.03-5

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	0.50~~0.42~~
	LPZ	6.3	0.50~~0.42~~
	CR	5.0	0.88~~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

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eRAI No.: 8792
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NRC Question No.: 15.00.03-6

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. The staff requests the following additional information to understand the applicant's analyses and complete its review.

DCD Tier 2 Section 15.0.3.7.2 presents information on the radiological consequence analysis for the reactor building pool boiling. This discussion indicates that if the pool were to boil as a result of a loss of normal AC power event, the dose would be less than 0.5 roentgen equivalent man (rem) total effective dose equivalent (TEDE). Is the dose from reactor pool boiling added to the total dose results at each of the dose receptor locations (exclusion area boundary (EAB), low population zone (LPZ) and main control room and technical support center) for each of the DBAs evaluated in DCD Chapter 15?

a. If so, please provide information on the analysis assumptions and inputs for the pool boiling dose calculation.
b. If not, please clarify why not and the purpose of the discussion of the reactor pool boiling dose.

NuScale Response:

The dose from the postulated pool boiling event is not added to the results of each of the design basis accidents (DBAs). The pool boiling event is not a DBA, nor an extension of a DBA. The purpose of the discussion of the pool boiling event in FSAR Chapter 15 is to provide additional information concerning the potential event.

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