## NuScaleDCRaisPEm Resource

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**Sent:** Wednesday, June 28, 2017 4:02 PM
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**Subject:** Request for Additional Information No. 76, RAI 8792
**Attachments:** Request for Additional Information No. 76 (eRAI No. 8792).pdf

Attached please find NRC staff's request for additional information concerning review of the NuScale Design Certification Application.

Please submit your response within 60 days of the date of this RAI to the NRC Document Control Desk.

If you have any questions, please contact me.

Thank you.

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301-415-0546

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**Request for Additional Information No. 76 (eRAI No. 8792)**

Issue Date: 06/28/2017
Application Title: NuScale Standard Design Certification - 52-048
Operating Company: NuScale Power, LLC
Docket No. 52-048
Review Section: 15.00.03 - Design Basis Accidents Radiological Consequence Analyses for Advanced Light Water Reactors
Application Section: DCD 15.0.3

QUESTIONS

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.

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.

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.

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.