NuScaleDCRaisPEm Resource

From:	Cranston, Gregory
Sent:	Tuesday, April 25, 2017 11:44 AM
To:	RAI@nuscalepower.com
Cc:	NuScaleDCRaisPEm Resource; Karas, Rebecca; Lee, Samuel; Burja, Alexandra; Franovich, Rani; Chowdhury, Prosanta
Subject:	Request for Additional Information No. 05 (eRAI No. 8766) Section 15.06.03 (SRSB)
Attachments:	Request for Additional Information No. 05 (eRAI No. 8766).pdf

Attached please find NRC staff's request for additional information concerning review of subject RAI.

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.

Gregory Cranston, Senior Project Manager
Licensing Branch 1 (NuScale)
Division of New Reactor Licensing
Office of New Reactors
U.S. Nuclear Regulatory Commission
301-415-0546

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Request for Additional Information No. 05 (eRAI No. 8766)

Issue Date: 04/25/2017
Application Title: NuScale Standard Design Certification - 52-048
Operating Company: NuScale Power, LLC
Docket No. 52-048
Review Section: 15.06.03 - Radiological Consequences of Steam Generator Tube Failure (PWR) 07/1981
Application Section: 15.6.3

QUESTIONS

15.06.03-1

Title 10 of the *Code of Federal Regulations* (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 performed, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses; 10 CFR 50, Appendix A, General Design Criterion (GDC) 19 for control room radiological habitability; and the requirements related to the technical support center in Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements, and Standard Review Plan (SRP) Section 15.0.3, "Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors," specifies the dose acceptance criteria. The fission product inventory released from all failed fuel rods is an input to the radiological evaluation under SRP Section 15.0.3. The NRC staff needs to ensure that the analysis showing no failure of fuel is suitably conservative.

FSAR Tier 2, Table 15.6-7, "Steam Generator Tube Failure - Sequence of Events - Limiting Reactor Pressure Vessel Pressure," shows that the time at which the maximum intact steam generator pressure is reached is 14 s into the transient; however, FSAR Tier 2, Figure 15.6-22, "Steam Generator Tube Failure - Limiting Reactor Pressure Vessel Pressure Scenario – Reactor Pressure Vessel and Steam Generator Pressures," shows the time of maximum intact steam generator pressure to be about 39 s. Because these timelines are not consistent, the staff lacks confidence that the peak pressure calculation has been reported accurately. RCS pressure is one acceptance criterion for postulated accidents. Therefore, please clarify the correct peak RCS pressure timing, and update the FSAR to correct any erroneous information.