NuScaleDCRaisPEm Resource

From: Cranston, Gregory
Sent: Saturday, August 05, 2017 11:48 AM
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Cc: NuScaleDCRaisPEm Resource; Lee, Samuel; Chowdhury, Prosanta; Karas, Rebecca; Thurston, Carl; Franovich, Rani
Subject: RE: Request for Additional Information No. 138, RAI 8794 (15.6.3)
Attachments: Request for Additional Information No. 138 (eRAI No. 8794).pdf

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

Please submit your technically correct and complete 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. 138 (eRAI No. 8794)

Issue Date: 08/05/2017
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:

QUESTIONS

15.06.03-2

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 (FSAR) 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 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, and 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 19 for control room radiological habitability. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in Standard Review Plan (SRP) Section 15.0.3. NRC staff needs to ensure that a suitably conservative estimate is determined for the radiological release associated with the steam generator tube rupture (SGTR) event. In addition, 10 CFR Part 50, Appendix A, GDC 54, “Piping systems penetrating containment,” requires piping systems penetrating primary reactor containment to be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems.

As indicated by the applicant in FSAR Tier 2, Section 15.6.3.1, “[t]he design of the helical coil steam generators, described in Section 5.4, is different from the design of SGs in conventional pressurized water reactors [PWRs] because primary coolant is located on the outside, or shell side, of the tubes.” In addition, the staff notes that the inventory of the SGs is also very small, so the radiological consequences of a SGTR could be more severe than for conventional PWRs. The mitigation of the SGTR event is totally dependent upon closure of the main steam isolation valve (MSIV) or the secondary MSIV, depending on the single active failure assumed. FSAR Figure 15.6-19 indicates the affected SG level is rapidly increasing as the isolation valve is closing, and Figures 15.6-20 and 15.6-21 confirm that liquid fraction is quickly increasing as the valve is closing. The staff is concerned about the ability of the isolation valve to close under potentially water-solid conditions since, as noted, the volume of NuScale SG secondary is quite small compared to that of conventional PWRs and may be prone to filling during the SGTR event.

For these reasons, the NRC staff requests that the applicant confirm that the case for “Limiting Mass Release” per FSAR Section 15.6.3.2 is the limiting case for highest potential to refill the SGs or provide an evaluation of the limiting case of SG filling. In addition, the NRC staff requests that the applicant confirm that the isolation valves are qualified to close under the worst predicted steam quality conditions.