NuScaleDCRaisPEm Resource

From:	Cranston, Gregory
Sent:	Tuesday, June 27, 2017 12:56 PM
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Cc:	NuScaleDCRaisPEm Resource; Lee, Samuel; Chowdhury, Prosanta; Hayes, Michelle; Franovich, Rani; Schaperow, Jason
Subject:	RE: Request for Additional Information No. 71, RAI 8889
Attachments:	Request for Additional Information No. 71 (eRAI No. 8889).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.

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. 71 (eRAI No. 8889)

Issue Date: 06/27/2017
Application Title: NuScale Standard Design Certification - 52-048
Operating Company: NuScale Power, LLC
Docket No. 52-048
Review Section: 19 - Probabilistic Risk Assessment and Severe Accident Evaluation
Application Section: 19

QUESTIONS

19-13

Regulatory Basis

10 CFR 52.47(a)(27) states that a design certification application (DCA) must contain a Final Safety Analysis Report (FSAR) that includes a description of the design-specific probabilistic risk analysis (PRA) and its results. 10 CFR 52.47(a)(2) states that the standard plant should reflect through its design, construction, and operation an extremely low probability for accidents that could result in the release of radioactive fission products. 10 CFR 52.47(a)(4) states that each DCA must contain an FSAR that includes and analysis and evaluation of the design and performance of systems, structure and components (SSCs) with the objective of assessing the risk to public health and safety resulting from operation of the facility and including a determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the SSCs provided for the prevention of accidents and the mitigation of the consequences of accidents.

Request for additional information

NuScale FSAR Chapter 19, page 19.2-27 states "The probability of [a steam generator tube failure (SGTF)] during high-temperature severe accident conditions was developed conservatively assuming the primary side was depressurized and the secondary side was pressurized. The probability of such a failure is incorporated into the Level 2 PRA as described in Section 19.1.4.2."

The applicant's analysis of the probability of SGTF is described in FSAR Section 6.4.1 "Thermal-Induced Steam Generator Tube Failure" of ER_P020_7024_R0, "Level 2 Probabilistic Risk Assessment Notebook." The applicant's analysis uses the methodology described in Section 2.5 "SGTF Probability under Severe Accident Conditions" of ER_P010_3782_R0 "Steam Generator Tube Failure Probabilistic Risk Assessment Report."

For severe accident scenarios in pressurized water reactors (PWRs) that feature U-tube steam generators with the primary system at high pressure and a dry secondary system at low pressure (known as the "high-dry-low" scenario), a counter-current flow of hydrogen and superheated steam occurs in the hot legs and in the steam generator tubes. This phenomenon was demonstrated in the Westinghouse $1/7^{th}$ scale experiments. In the high-dry-low scenario, ex-vessel piping is exposed to high temperatures and high internal pressures. For the high-dry-low scenario, studies have been performed to estimate the probability of a steam generator tube rupture before another ex-vessel piping rupture (which could result in higher offsite radiological consequences if a tube ruptures).

The applicant extrapolated the results of one of these studies (Liao, Y., and Guentay, S., "Potential steam generator tube rupture in the presence of severe accident thermal challenge and tube flaws due to foreign object wear," Nuclear Engineering and Design, Vol. 239, Issue 6, pp: 1128-1135, 2009) to estimate the probability of thermally induced steam generator tube rupture for NuScale. However, the applicability of the Liao and Guentay study to NuScale is unclear because of design differences. For example, the probability distribution developed by Liao and Guentay represents the probability of a steam generator tube rupturing before other ex-vessel piping (hot leg and surge line) ruptures. NuScale does not have ex-vessel piping. Also, the design analyzed by Liao and Guentay has primary coolant on the inside of the steam generator tubes. NuScale has primary coolant on the outside of the steam generator tubes. The applicant is requested to justify the applicability of these studies to NuScale's design.