NuScale Power

RAIO-0817-55461

August 16, 2017

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 71 (eRAI No. 8889) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 71 (eRAI No. 8889)," dated June 27, 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 Question from NRC eRAI No. 8889:

- 19-13

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
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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 8889

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 8889

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NuScale Power

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

eRAI No.: 8889
Date of RAI Issue: 06/27/2017

NRC Question No.: 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 1/7th 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.

NuScale Response:

As discussed in FSAR Section 19.1.4.1, the operating characteristics of the NuScale steam generators are opposite of those in conventional PWRs; notably, secondary coolant flows through the inside of the steam generator tubes and the higher-pressure primary coolant is on the outside such that the tubes are maintained in a constant state of compression. The NuScale steam generator is also helical by comparison to a U-tube or once-through design. Along with the use of Alloy-690, these design differences are expected to result in lower occurrences of tube failures. As such, NuScale uses the term steam generator tube “failure” (SGTF) to distinguish potential tube faults from the steam generator tube “rupture” (SGTR) terminology that is used in conventional PWRs.

In the NuScale design, reactor coolant flow is driven by natural circulation; reactor coolant flows upward through the core due to natural circulation where it absorbs heat and continues to flow up the central riser. At the top of the central riser, it is turned by the pressurizer baffle plate and flows downward across the integrated steam generator tube bundles. Heat is removed by helical steam generator tubes coiled inside the RPV at a distance above the reactor core. Feedwater enters the steam generators through the feedwater supply nozzles and feed plenums and flows up the helical tubes where it is heated, boiled, and superheated before it exits through the steam plenums and main steam supply nozzles to the steam lines. If the steam generator plenums or access covers were to fail before the tubes due to high pressure, integrity of the containment boundary would be retained which would eliminate the potential for offsite radiological consequences.

As discussed in FSAR Section 19.2.3.3.6, the high temperature and pressure differentials in the steam lines during a severe accident can induce an SGTF. Under severe accident conditions, tubes have a higher failure probability than during normal operation and failure is most likely due to high temperature creep failure. To estimate the probability of an SGTF during a severe accident in the NuScale design, an analysis was performed that relied on a scaling approach of a creep rupture model developed to predict tube failure in conventional PWR steam generators. The creep rupture model, validated by tests, predicts tube failure under constant as well as ramped temperature and pressure conditions.

Based on the creep rupture model, Liao and Guentay provide a distribution for the tube failure probability. Since the Liao and Guentay tube failure probability calculations are for Alloy-690 under tensile stresses, their creep and severe accident analysis is conservatively applied to the NuScale design with the difference being the severe accident temperature and pressure profiles; the scaling approach accounts for the NuScale temperature and pressure profiles during a severe accident. Based on the basic strength of materials, data for tensile failures bound compressive failures under the same conditions. As such, the modeling of the tubes with internal pressure during a severe-accident is a simplifying but conservative assumption that is used because of the lack of data for tubes with external compression.

Thermally-induced SGTFs that result in radiological consequences have a very low probability in the NuScale design because failure of at least two isolation valves is also required for a release. As shown by Sensitivity Study 3b, summarized in FSAR Tables 19.1-22 and 19.1-31, increasing the probability of a severe accident thermally-induced SGTF (to 2.2E-01) results in negligible changes in CDF and LRF. The probability used in the sensitivity study is an order of magnitude higher than the 0.025 mean calculated in the Liao and Guentay paper that developed a method for estimating the probability of a SGTR before an ex-vessel piping rupture. Thus, the NuScale CDF and LRF results are not sensitive to the use of Liao and Guentay probability distribution.

Impact on DCA:

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