NuScaleDCRaisPEm Resource

From: Cranston, Gregory
Sent: Saturday, June 9, 2018 8:45 AM
To: Request for Additional Information
Cc: Lee, Samuel; Chowdhury, Prosanta; Jackson, Diane; Haider, Syed; Tabatabai, Omid; NuScaleDCRaisPEm Resource
Subject: Request for Additional Information No. 487 eRAI No. 9549 (6.2.1.1A)
Attachments: Request for Additional Information No. 487 (eRAI No. 9549)public.pdf

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

If you have any questions, please contact me.

Thank you.

Hearing Identifier: NuScale_SMR_DC_RAI_Public
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Request for Additional Information No. 487 (eRAI No. 9549)

Issue Date: 06/09/2018
Application Title: NuScale Standard Design Certification - 52-048
Operating Company: NuScale Power, LLC
Docket No. 52-048
Review Section: 06.02.01.01.A - PWR Dry Containments, Including Subatmospheric Containments
Application Section: FSAR Section 6.2.1.1 Containment Structure

QUESTIONS

06.02.01.01.A-21

NIST-1 RRV Opening Test and Validation for the Containment Response Analysis Methodology

10 CFR 50, Appendix A, Criterion 16, Containment design, states,

"Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require."

10 CFR 50, Appendix A, Criterion 50, Containment design basis, states,

"The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by § 50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters."

10 CFR 50.43(e) states, in part, the use of simplified, inherent, passive, other innovative means to accomplish their safety functions will be approved only if:

(1)(i) The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof; and

(1)(iii) Sufficient data exist on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions.

The design evaluation for the containment system provided in FSAR, Tier 2, Section 6.2.1.1.3 references the Containment Response Analysis Methodology (CRAM) technical report, TR-0516-49084. NuScale used the CRAM to determine the peak containment pressure as the key figure of merit for containment integrity, for all design basis events (DBEs) analyzed for the NuScale power module (NPM). The NPM containment response analysis results presented in FSAR, Tier 2, Table 6.2-2 show that (1) the limiting peak containment pressure DBE is an inadvertent opening of a reactor recirculation valve (RRV), which is the largest liquid space discharge, and (2) this event results in a peak calculated containment pressure within 5 percent of the containment design pressure. The CRAM technical report also references the NRELAP5 code assessment described in the Loss-of-Coolant Accident (LOCA) evaluation model (EM) topical report, TR-0516-49422. The NRELAP5 code assessment includes a single NIST-1 test to characterize the liquid-space discharge from a smaller chemical and volume control system (CVCS) line than the RRV opening. The assessment results for this event, presented in Sections 7.5.6.5 and 7.5.6.6 of TR-0516-49422, show that NRELAP5 {{…………………………………}}. Even though an inadvertent RRV actuation as the initiating event could lead to the most severe containment pressurization, no integral test was conducted to demonstrate that the NRELAP5 code can conservatively model the largest liquid space discharge from the RRV at the lowest discharge point in the NIST-1 containment, with bounding uncertainties.

As the NRELAP5 code assessment does not include the limiting peak containment pressure event, and the non-limiting liquid space discharge included in the NRELAP5 code assessment appears to show that NRELAP5 {{…………………………………}}, the NRC staff is unable to reach a reasonable assurance finding that sufficient test data exist to assess NREALP5 over a sufficient range of transient and accident conditions. Therefore, the NRC staff requests that NuScale perform an assessment of the NRELAP5 code using an integral effects test where the initiating event is a discharge from an RRV at a representative altitude. The applicant is also requested to provide or make available for audit the additional test data and code assessment results for the limiting peak containment pressure event, and update the licensing documentation (e.g., CRAM technical report, LOCA EM topical report), accordingly.