NuScaleTRRaisPEm Resource

From:	Chowdhury, Prosanta
Sent:	Tuesday, May 8, 2018 4:51 PM
To:	Request for Additional Information
Cc:	Lee, Samuel; Cranston, Gregory; Karas, Rebecca; Lu, Shanlai; Franovich, Rani; NuScaleTRRaisPEm Resource
Subject:	Request for Additional Information Letter No. 9476 (eRAI No. 9476) Topical Report, LOCA, 15.06.05, SRSB
Attachments:	Request for Additional Information No. 9476 (eRAI No. 9476).pdf

Attached please find NRC staff's request for additional information (RAI) concerning review of the NuScale Topical Report.

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.

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Office of New Reactors
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Request for Additional Information No. 9476 (eRAI No. 9476)

Issue Date: 05/08/2018
Application Title: NuScale Topical Report
Operating Company: NuScale
Docket No. PROJ0769
Review Section: 15.06.05 - Loss of Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary
Application Section: NuScale LOCA Topical Report

QUESTIONS

15.06.05-13

Title 10, Part 52, of the *Code of Federal Regulations* (10 CFR Part 52), "Licenses, Certifications, and Approvals for Nuclear Power Plants," Section 52.47, "Contents of Applications; Technical Information" (10 CFR 52.47), specifies that an application for certification of a nuclear power reactor design that uses simplified, inherent, passive, or other innovative means to accomplish its safety functions must meet the requirements of 10 CFR 50.43(e) (52 Part 52.47(c)(2)). 10 CFR 50.43(e) requires, in part, assessment of the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences. Regulatory Guide (RG) 1.203 describes a process that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable for use in developing and assessing evaluation models (EMs) that may be used to analyze transient and accident behavior that is within the design basis of a nuclear power plant.

As stated in RG 1.203, an EM is the calculational framework for evaluating the behavior of the reactor system during a postulated transient or design-basis accident. As such, the EM may include one or more computer programs, special models, and all other information needed to apply the calculational framework to a specific event, as illustrated by the following examples:

(1) Procedures for treating the input and output information (particularly the code input arising from the plant geometry and the assumed plant state at transient initiation)

(2) Specification of those portions of the analysis not included in the computer programs for which alternative approaches are used

(3) All other information needed to specify the calculational procedure

The entirety of an EM ultimately determines whether the results are in compliance with applicable regulations. Therefore, the development, assessment, and review processes must consider the entire EM.

The RELAP5 code geometry fixes the length of a junction fluid column joining two cells so that if a simple manometer is modeled, the inertia of the fluid column is determined by the geometry, not the length of the fluid column in the U-tube. Thus, the inertia of the fluid column in RELAP5 momentum formulation can be inconsistent with the junction geometry since this is fixed by the input. Comparisons of the NRELAP5 code with a simple oscillating manometer, with and without friction, shows that very quickly, RELAP5 predicts an incorrect period that is completely out of phase with the correct solution. Such behavior is attributable to the improper fluid inertia

specified by the RELAP5 geometric formulation. If this fundamental numerical deficiency exists, the code capability to accurately calculate the natural circulation flow during steady state and transient becomes questionable. Thus, the calculated two-phase water level above the core during a loss of coolant accident (LOCA), which is the figure of merit of the analysis, may not be conservative.

- Please discuss the impact of this modeling feature on the NRELAP5 prediction of fluid level and flow behavior in a manometer with and without friction and the impact on the predictive capabilities of heterogeneous fluid behavior in the vessel in the NuScale primary system under transient LOCA conditions.
- For the manometer input test problem (with and without friction) and solutions provided to NuScale by the staff, please provide the comparisons of NRELAP5 to the manometer solutions with and without friction, showing the flow rates and levels vs time.
- Please also discuss the impact of any conclusions from these comparisons to the ability of NRELAP5 to predict emergency core cooling system response in the small modular reactor following the limiting small break LOCA.