NuScaleTRRaisPEm Resource

From: Cranston, Gregory
Sent: Friday, June 15, 2018 3:59 PM
To: Request for Additional Information
Cc: Lee, Samuel; Karas, Rebecca; Burja, Alexandra; Franovich, Rani; NuScaleTRRaisPEm Resource; Chowdhury, Prosanta
Subject: Request for Additional Information Letter No. 9390 (eRAI No. 9393) Topical Report, LOCA , 15.06.01, SRSB
Attachments: Request for Additional Information No. 9390 (eRAI No. 9390)-Public.pdf

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

The NRC Staff recognizes that NuScale has preliminarily identified that the response to one or more questions in this RAI is likely to require greater than 60 days. NuScale is expected to provide a schedule for the RAI response by email within 14 days.

If you have any questions, please contact me.

Thank you.

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Request for Additional Information No. 9390 (eRAI No. 9390)

Issue Date: 06/15/2018
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: Section 8.3 Scaling & Distorion

QUESTIONS

15.06.05-18

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 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 evaluation model (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.

a. The NIST-1 facility was sized based on scaling the natural circulation phase at { }, which the staff audited in support of the loss of coolant accident [LOCA] Topical Report [TR]). As the reactor will be in steady state for natural circulation, the two dominant phenomena, buoyancy and friction should match. { } steady state operation, as these two forces will be balanced as was shown in Section 4.4.1 { }. Please explain this discrepancy.

b. Regarding depressurization equations for the system { }, the staff is concerned that { } may not be properly preserved. Therefore, the current approach could average the effect of heat addition over the whole primary system. As a result, the depressurization rate could be distorted as the impact of heat addition might be diluted in the reactor pressure vessel (RPV) and similarly the pressurization rate in the containment could be also be altered. Please provide justifications to show this { } and approach does not adversely impact the blow-down phenomenon simulation in NIST.

c. The scaling report { } showed that there is distortion in { }. At peak CNV pressure { }. Also, at the peak pressure, all terms in the energy equation should be balanced, i.e, the energy input from the RPV and the energy lost to the containment wall surface should be equal. This is not evident from Table 6-20. Please explain. { } for CNV pressure indicates { }. What are additional distortions that compensate for it?

d. Because the Section 8.3.2 only has a brief summary of the scaling analysis and distortion evaluation based on the actual scaling report and distortion analysis for NIST, however these evaluations form a large portion of the justification for the model, provide important findings and conclusions from the scaling analysis report and the distortion evaluation report with specific references (e.g. section/page/figure number) that support the conclusions in Section 8.3.2." These findings should also include distortions in power distribution, initial fluid and heat structure stored energy in the vessel, NIST CNV initial conditions; such as pressure, CNV wall temperature, HTP temperature, condensate liquid level, building pool temperature, and NIST initial vessel pressure, and the impact of these distortions on the test data for figures of merit. The accurate documentation of initial and boundary conditions are essential for NRELAP5 code validation.

15.06.05-19

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 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 evaluation model (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 evaluation model (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 purpose of the NIST-1 facility is to provide realistic data for evaluation model validation. As there are no other counterpart tests, the NIST-1 facility is critical for code validation. The validation with NIST-1 for a set of loss of coolant accident (LOCA) tests demonstrates the code's ability to predict NuScale Power Module (NPM) behaviors. However, for NIST-1, this requires an additional step of evaluating the scaling distortion. Significant distortion in initial conditions, boundary conditions and important scaling similarity groups (PI group) need to be assessed before the code is qualified to predict the figures of merit for the NPM, such as the containment peak pressure and the reactor vessel level that is a surrogate for clad temperature in a traditional pressurized water reactor (PWR).

In the scaling distortion report { }, which the staff audited in support of the LOCA Topical Report (TR), the applicant attempts to quantify the effect of the scaling distortion by performing sensitivity calculations for the NPM and NIST-1 configuration. In the course of review, the staff identified several discrepancies and concerns, as listed below. These discrepancies were not adequately addressed in audit discussions with the applicant. Assessment of scaling distortions is essential to predicting the figures of merit for a NPM LOCA transient. One of the elements of review is the consistency of sensitivity calculations and their impact on predicted figures of merit in response to scaling distortions, which is not clearly shown based on the following observations. The staff needs to understand how these discrepancies and distortions contribute to the LOCA figures of merit (e.g. reactor level, containment pressure) quantitatively. An integral estimate of quantified uncertainty of figures of merit due to scaling distortions as identified below is also needed.

1. { } show primary side collapsed liquid level for HP-06. The timing of NPM emergency core cooling systems (ECCS) actuation and the level after ECCS starts varies among plots. Distortion of the time of ECCS actuation reflects a scaling distortion of several parameters which contribute to LOCA figures of merit. Evaluation of the HP06b test is essential since { }. Provide a discussion regarding HP06b level behavior and the measured ECCS actuation timing.

2. { } shows pressure differentials at the break, and the NPM shows { } However, { }. There exists { }, which affects the vessel inventory prediction. The staff requests an explanation of this inconsistency and the implication to the vessel inventory prediction.

3. { } for test HP05 indicates that there is a better match for core flow data between the scaled NPM than NIST. It should be the other way around as NIST loss coefficients { } which the staff audited in support of the LOCA TR, describes { } This distortion is not explained sufficiently. The impact of this distortion is on stored energy in the vessel component, and its effects on the overall LOCA response. Provide quantitative justification to show that the distortion falls into the relevant PI number uncertainty range.