NuScaleTRRaisPEm Resource

From: Chowdhury, Prosanta
Sent: Tuesday, May 8, 2018 1:21 PM
To: Request for Additional Information
Cc: Lee, Samuel; Cranston, Gregory; Karas, Rebecca; Burja, Alexandra; Franovich, Rani; NuScaleTRRaisPEm Resource
Subject: Request for Additional Information Letter No. 9466 (eRAI No. 9466) Topical Report, Non-LOCA Analysis Methodology, 15.00.02, SRSB
Attachments: Request for Additional Information No. 9466 (eRAI No. 9466).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. 9466 (eRAI No. 9466)

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QUESTIONS

15.00.02-6

General Design Criterion (GDC) 10, "Reactor design," requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs). In addition, GDC 15, "Reactor coolant system design," requires that the reactor coolant system (RCS) and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary (RCPB) are not exceeded during any condition of normal operation, including AOOs.

Topical report (TR) TR-0516-49416-P, "Non-Loss-of-Coolant Accident [Non-LOCA] Analysis Methodology," supports the conclusions relative to GDC 10 and 15 in the NuScale Final Safety Analysis Report (FSAR), which under 10 CFR 52.47 must describe the facility, present the design bases and the limits on its operation, and present a safety analysis of the structures, systems, and components and of the facility as a whole.

Chapter 15 of NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants," describes a subset of the transients and accidents that should be considered in the safety analyses. SRP Section 15.0.2, "Review of Transient and Accident Analysis Methods," directs the staff in reviewing methodologies used to conduct the safety analyses required by 10 CFR 52.47 and states:

> *When the code is used in a licensing calculation, the combined code and application uncertainty must be less than the design margin for the safety parameter of interest. The analysis must include a sample uncertainty evaluation for a typical plant application. In some cases, bounding values are used for input parameters as described in SRP sections or Regulatory Guides and are used for plant operating conditions such as accident initial conditions, set points, and boundary conditions.*

While the representative calculations for the non-LOCA evaluation model (EM) in TR-0516-49416-P, Section 8, include initial condition and parameter biasing, there is no assessment of uncertainty to ensure that the combined uncertainty in the code, plant inputs, and plant model is less than the design margin, such as that required by GDC 10 and 15. The representative analyses appear to meet non-LOCA figure of merit (FOM) acceptance criteria, but it is not clear how the code uncertainty associated with these best estimate NRELAP5 code results factor into the assessment.

In addition, as a result of the concerns in the third question of RAI 9351, the staff is seeking specific information regarding uncertainty related to the SIET assessments.

Information Requested:

1. Based upon the code assessment against the separate effects tests (SETs) and integral effects tests (IETs) (and any revised analyses related to the presented SETs and IETs), state the determined code uncertainties and biases.
2. Justify that the combined effect of conservative input parameters, parameters that are conservatively biased, the plant model, and the code uncertainty is less than the design margin for the FOMs.
3. Address parts (a) and (b) above specifically for the assessments against the SIET tests, i.e.:
    - Based upon the code assessment against the SIET tests and the response to the third question in RAI 9351, provide the NRELAP5 code steam generator model uncertainties and biases.
    - Justify that the combined effect of conservative input parameters and conservatively biased parameters bounds the steam generator model uncertainty.

15.00.02-7

GDC 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs. In addition, GDC 15 requires that the RCS and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs.

TR-0516-49416-P supports the conclusions relative to GDC 10 and 15 in the NuScale FSAR, which under 10 CFR 52.47 must describe the facility, present the design bases and the limits on its operation, and present a safety analysis of the structures, systems, and components and of the facility as a whole. SRP Section 15.0.2 provides the staff guidance on reviewing analytical models and computer codes used to analyze transient and accident behavior. SRP Section 15.0.2 states:

> *Models must be present for all phenomena and components that have been determined to be important or necessary to simulate the accident under consideration. The chosen mathematical models and the numerical solution of those models must be able to predict the important physical phenomena reasonably well from both qualitative and quantitative points of view.*

TR Section 6.1.2 discusses the core kinetics in the NRELAP5 plant model of the NuScale power module (NPM) and states:

> *The fission product decay type is specified as 'gamma-ac' with the 'ans73' model, which calculates decay heat in accordance with the 1973 ANS standard while adding the contribution from actinides. A fission product yield factor of 1.0 is specified in the base model, which can be changed to suit the scenario being analyzed.*

According to TR Section 7.1.5.3, decay heat is biased either low or high, depending on the transient being analyzed, by use of decay heat multipliers and specifying whether or not to include the actinide contribution. The staff requires justification that, as altered by multipliers and actinide contribution in the non-LOCA analyses, the 1973 model leads to conservative results. In response to an audit discussion (Round 2, Issue 5), the applicant stated in the quality assurance process for NRELAP5, some of the decay heat models were subjected to greater scrutiny, with the 1973 decay heat model being one of greater pedigree. Therefore, it was selected for use. The applicant stated that a calculation that examined the various decay heat models was performed, and the results helped determine that a multiplier for a maximum or minimum decay heat level was appropriate.

Furthermore, Page 112 of the LOCA submittal (TR-0516-49422-P) states that "λ and η values can be user-specified, or default values equal to those stated in the 1979 ANS standard (Table 6-4), the 1994 Standard, or the 2005 Standard can be used."

Because decay heat affects energy production in the core and therefore transient progression, the staff requires additional information about the use of the 1973 decay heat model and the methodology for selecting the values used for λ and η.

Information Requested:

Provide the actual values of λ and η used and references for these parameters, and justify that the use of these values, the specified multipliers, and inclusion (or lack thereof) of actinides in combination with the 1973 decay heat model leads to a conservative result for decay heat contribution for all event types. Update TR-0516-49416-P and any other affected documentation as appropriate.

15.00.02-8

GDC 15 requires that the RCS and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs. TR-0516-49416-P supports the conclusions relative to GDC 15 in the NuScale FSAR, which under 10 CFR 52.47 must describe the facility, present the design bases and the limits on its operation, and present a safety analysis of the structures, systems, and components and of the facility as a whole.

RG 1.203 describes the evaluation model development and assessment process (EMDAP), which the NRC staff considers acceptable for use in developing and assessing EMs used to analyze transient and accident behavior. Step 18 of the EMDAP (Section 1.4.6) discusses preparation of input and performance of calculations to assess system interactions:

> *"The ability of the EM to model system interactions should also be evaluated in this step, and plant input decks should be prepared for the target applications. Sufficient analyses should be performed to determine parameter ranges expected in the nuclear power plant."*

TR-0516-49416-P, Table 7-45 provides the results of representative sensitivity studies for the loss of normal AC power event. All peak primary pressures in Table 7-45 are **[[ ]]** .

In addition, Table 7-45 shows that the parameter with the largest effect on the peak SG pressure is **[[ ]]** .

The staff observes that reducing the **[[ ]]** .

The staff needs additional information to understand these sensitivity studies since similar sensitivity studies are conducted for the FSAR analyses to identify the initial conditions and biases that result in the most limiting figures of merit.

Information Requested:

1. Clarify the timing and delay times for the reactor trip, decay heat removal system (DHRS) activation time, DHRS valve time to complete stroke, the time to when AC power to the normal DC power system (EDNS) battery chargers is lost, and the time at which the containment is fully isolated.
2. Primary side peak pressure is **[[ ]]** would result in a higher primary side peak pressure.
3. Explain the reductions in the peak SG pressure for **[[ ]]** .
4. Explain the relatively large variation in the peak SG pressure for **[[ ]]** .

15.00.02-9

10 CFR 50.43(e) states that applications that reference simplified, inherent, passive, or other innovative means will be approved only if 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.

TR-0516-49416-P supports the conclusions in the NuScale FSAR, which under 10 CFR 52.47 must describe the facility, present the design bases and the limits on its operation, and present a safety analysis of the structures, systems, and components and of the facility as a whole. The NuScale Design-Specific Review Standard (DSRS) Section 15.0, "Introduction – Transient and Accident Analyses," directs the staff to verify that the implementation of models or codes is within the applicable ranges and conditions.

RG 1.203 describes the EMDAP, which the NRC staff considers acceptable for use in developing and assessing EMs used to analyze transient and accident behavior. Step 7 of the EMDAP (Section 1.2.3) discusses the identification and performance of SETs and IETs to complete the database against which the EM is assessed. Steps 14 and 18 of the EMDAP (Sections 1.4.2 and 1.4.6) discuss the preparation of input and performing of calculations to assess model fidelity or accuracy and system interactions and global capability, respectively.

Several aspects of the assessment of NRELAP5 predictions against test data in TR-0516-49416-P require further clarification, as discussed below.

Some of the phenomenological assumptions and changes to the NRELAP5 modeling parameters that are tuned to match the NLT-02a and NLT-02b test results do not appear to be fully supported by statements in the audited document **[[ ]]**.

[[ ]] does not contain specific recommendations based on the results of the benchmarking studies for application to NPM non-LOCA analyses. A number of changes were implemented in the model **[[ ]]** , which have the effect of "tuning" the model to achieve improved results. TR-0516-49416-P, Section 5.4 discusses generally how the NRELAP5 computer code has the ability to represent specific phenomena and processes. However, the TR does not discuss the important steps regarding how specific test simulations provided information to develop the NPM evaluation model.

In addition, in response to audit discussions in which the staff sought to understand the reasons for discrepancies between predictions and test data, the applicant described multiple sensitivity studies not included in TR-0516-49416-P that may demonstrate better agreement than what is currently reflected in TR-0516-49416-P. For example, the applicant noted (Round 3 of audit discussion) that a preliminary sensitivity study **[[ ]]** led to better agreement for the NIST low pressure test HP-03-01, and preliminary sensitivity studies **[[ ]]** led to better agreement for the NIST NLT-02a and NLT-02b tests. An understanding of the results of the sensitivity studies is necessary for the staff to conclude whether or not the noted discrepancies between prediction and data is due to model or code deficiencies.

Information Requested:

a. Describe how the code models are assessed in the data benchmarking studies when external adjustments are made to improve the simulation results. Update TR-0516-49416-P with an explanation, as appropriate.

b. Explain how the lessons learned from the NIST tests were translated into the NPM model development or provide a reference or document that explains this process.

c. For the medium pressure test (HP-03-02c) documented in document **[[ ]]** , several of the figures are labeled as HP-03-01 tests. Please confirm that the intended figures are shown, and provide corrections for any incorrect information.

d. When modifications are made to assess test data comparisons due to re-examinations or evaluations (such as for the cited HP-03-01 and NLT test sensitivity studies) provide the results of the revised analyses, and update TR-0516-49416-P as appropriate.

15.00.02-10

GDC 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs. In addition, GDC 15 requires that the RCS and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs.

TR-0516-49416-P supports the conclusions relative to GDC 10 and 15 in the NuScale FSAR, which under 10 CFR 52.47 must describe the facility, present the design bases and the limits on its operation, and present a safety analysis of the structures, systems, and components and of the facility as a whole. DSRS Section 15.0 directs the staff to confirm that the implementation of models or codes are within the applicable ranges and conditions. Furthermore, RG 1.203 describes the EMDAP. Step 4 of the EMDAP (Section 1.1.4) discusses the identification and ranking of key phenomena and processes and states:

> *"A key feature of the adequacy assessment is the ability of the EM or its component devices to predict appropriate experimental behavior. Once again, the focus should be on the ability to predict key phenomena, as described in the first principle. To a large degree, the calculational devices use collections of models and correlations that are empirical in nature. Therefore, it is important to ensure that they are used within the range of their assessment."*

TR-0516-49416-P, Section 5.3.1.3 attributes the reasonable to excellent agreement between NRELAP5 calculations and KAIST measured experimental data to **[[ ]]** . However, the NPM DHRS is designed to operate under natural circulation conditions.

Therefore, it is not clear that the use of the **[[ ]]** , within the range of the assessment, is appropriate for the natural circulation conditions within the NPM. Reasonable to excellent agreement for application of an inappropriate correlation may result from compensating errors, so additional justification is required to demonstrate the adequacy of the present **[[ ]]** within the NRELAP5 methodology framework.

Information Requested:

Demonstrate the fundamental adequacy of the present approach within the NRELAP5 methodology for application to the NPM safety analysis, for example by providing an assessment of various approaches to representation of condensation heat transfer under conditions of buoyancy-drive flows for the geometries of interest and their potential applicability to the NPM, discussing the range of the applicability of the correlations, the test conditions to which the correlations have been subjected, and the similarity to the conditions expected within the NPM. If a more applicable approach exists, compare it to the current approach, and justify the current approach. Update TR-0516-49416-P and any other affected documentation (e.g., the LOCA EM TR [TR-0516-49422-P]) as appropriate.

15.00.02-11

GDC 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs. In addition, GDC 15 requires that the RCS and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs.

TR-0516-49416-P supports the conclusions relative to GDC 10 and 15 in the NuScale FSAR, which under 10 CFR 52.47 must describe the facility, present the design bases and the limits on its operation, and present a safety analysis of the structures, systems, and components and of the facility as a whole. DSRS Section 15.0 guides the staff to confirm that the implementation of models or codes are within the applicable ranges and conditions. Furthermore, RG 1.203 describes the EMDAP. Step 4 of the EMDAP (Section 1.1.4) discusses the identification and ranking of key phenomena and processes and states:

> *"A key feature of the adequacy assessment is the ability of the EM or its component devices to predict appropriate experimental behavior. Once again, the focus should be on the ability to predict key phenomena, as described in the first principle. To a large*

degree, the calculational devices use collections of models and correlations that are empirical in nature. Therefore, it is important to ensure that they are used within the range of their assessment."

Page 256 of TR-0516-49416-P states that **[[ ]]** is used to calculate the heat transfer from the DHRS to the cooling pool. However, the staff notes that several audited documents mention use of **[[ ]]** to calculate the DHRS heat transfer to the cooling pool. These documents include **[[ ]]**, which provides references for the design inputs in the NRELAP5 model of an NPM; **[[ ]]**, a change to the calculation supporting NuScale FSAR Sections 15.2.1-15.2.3; and **[[ ]]**, the calculation supporting NuScale FSAR Section 15.1.1. Therefore, it is unclear what correlation is actually used in the non-LOCA methodology.

If the **[[ ]]** is used, it is unclear whether within the range of the assessment, it is appropriate for the pool boiling conditions within the NPM. To justify the adequacy of the use of **[[ ]]** under pool boiling conditions, the applicant performed a sensitivity study in which it changed **[[ ]]** , among other various sensitivities, and documented the sensitivity study in Section 4.2 of the document **[[ ]]**. The results of the sensitivity study are shown in Figures 4-7 and 4-8 of **[[ ]]** . It appears that the impact on the change in enthalpy across the DHRS resulting from the change **[[ ]]** is larger than the impact of other sensitivity results shown on Figure 4-7, but the co-plotting of the various sensitivity cases makes this unclear. Figure 4-8 appears to show an improved response in the predicted DHRS tube water level **[[ ]]** for the first two-thirds of the transient, before being roughly the same for the final one-third of the calculation. Therefore, if the **[[ ]]** is used, the staff requires additional information to justify whether it is appropriate for the NPM.

In general, if a correlation used in the FSAR analyses differs from those used in the code assessment, the correlation used in the FSAR analyses must be justified and documented. If the **[[ ]]** is used in the FSAR analyses but is not used in the non-LOCA assessments, a basis and justification for the **[[ ]]** must be provided.

In addition, the applicant may re-perform some test simulations in the future to resolve discrepancies or other RAIs. The correlation modeling the heat transfer from the DHRS to the cooling pool should be one that has been validated.

Information Requested:

a. Clarify what correlation is used to model heat transfer from the DHRS to the cooling pool in TR-0516-49416-P and in the FSAR non-LOCA analyses.

b. If **[[ ]]** is used to analyze non-LOCA events,

- Provide a figure equivalent to **[[ ]]** , Figure 4-7, in which only the data, the NRELAP5 base calculation, and the NRELAP5 **[[ ]]** calculation information are shown.
- Provide a figure showing the integrated delta enthalpy (kW) values for the data, the NRELAP5 base calculation, and the NRELAP5 **[[ ]]** calculation as a function of time.
- Provide a more in-depth discussion of the **[[ ]]** , Figure 4-8 results for the data, the base calculation, and the **[[ ]]** calculation, comparing the results, the bases for the differences, and the reason for the approximate agreement between the base calculation and the **[[ ]]** calculation after approximately 510 seconds.

- Based on the above information and the physical dynamics within the NPM, justify the applicability of **[[ ]]** to the pool boiling conditions in the NPM.

c. If **[[ ]]** is used to analyze non-LOCA events,

- Provide a basis and justification for use of the revised model, as the non-LOCA TR does not appear to have included this correlation in the assessment studies. Consider all impacted assessment studies. This may include some of the same comparisons as suggested in part (b).
- Assess the impact of the revised model on uncertainties and margin in the non-LOCA analyses.

d. Update TR-0516-49416-P and any other affected documentation as appropriate.

15.00.02-12

GDC 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs. In addition, GDC 15 requires that the RCS and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs.

TR-0516-49416-P supports the conclusions regarding GDC 10 and 15 in the NuScale FSAR, which under 10 CFR 52.47 must describe the facility, present the design bases and the limits on its operation, and present a safety analysis of the structures, systems, and components and of the facility as a whole. RG 1.203 describes the EMDAP, which the NRC staff considers acceptable for use in developing and assessing EMs used to analyze transient and accident behavior. RG 1.203 discusses the preparation of input and performance of calculations to assess model fidelity in Step 14 (Section 1.4.2) and states:

> *In particular, nodalization and option selection should be consistent between the experimental facility and similar components in the nuclear power plant. Nodalization convergence studies should be performed to the extent practicable in both the test facility and plant models.*

It is important to understand how the test facility nodalization differs from the nodalization proposed for use in the non-LOCA licensing transient analyses since the adequacy of the NPM non-LOCA model is based, in large part, on assessments against test data. Different nodalization could impact the margin to SAFDLs and RCS pressure limits specified in GDC 10 and 15.

The discussions related to the nodalization of the test facility and plant models in TR Sections 5 and 6 are not clear. The nodalization diagrams that have been provided for both the various experimental assessment cases and the plant model are incomplete. In addition, for each of the representative calculations documented in Section 8 of the TR, any changes that may have been made to the nodalization diagrams in Section 6 are not indicated.

Information Requested:

1. Provide a detailed, complete, and legible nodalization diagram of the non-LOCA model used as the base for each of the assessment cases documented in TR Section 5, and update the TR accordingly.
2. Identify and provide the bases for the differences between the NRELAP5 NIST-1 test facility nodalization and the NRELAP5 NPM nodalization. Explain how the NIST-1 assessment results were utilized for NPM model nodalization.
3. For each representative calculation documented in TR Section 8, provide either a detailed and complete nodalization diagram, or explain the specific changes to the base model nodalization as used for each transient calculation. Update the TR accordingly.

15.00.02-13

GDC 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs. In addition, GDC 15 requires that the RCS and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs.

TR-0516-49416-P supports the conclusions relative to GDC 10 and 15 in the NuScale FSAR, which under 10 CFR 52.47 must describe the facility, present the design bases and the limits on its operation, and present a safety analysis of the structures, systems, and components and of the facility as a whole. SRP Section 15.0.2 directs the staff to review analytical models and computer codes used to analyze transient and accident behavior and states that a theory manual that describes such items as field equations, closure relationships, numerical solution techniques, etc. should be included as part of the EM documentation.

The staff notes that TR-0516-49416-P refers to the NRELAP5 theory manual but does not include it as a reference in TR Section 11.0. The adequacy of NRELAP5 and its ability to calculate pertinent physical phenomena is dependent on the physical and numerical modeling described in the theory manual. Therefore, please add the NRELAP5 theory manual as a reference in TR-0516-49416-P.