NuScaleTRRaisPEm Resource

From: Cranston, Gregory
Sent: Friday, June 23, 2017 6:07 PM
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Cc: NuScaleTRRaisPEm Resource; Lee, Samuel; Skarda, Raymond; Karas, Rebecca; Schmidt, Jeffrey; Chowdhury, Prosanta; Bavol, Bruce
Subject: Topical Report (TR-0516-49417) - Request for Additional Information Letter No. 8802 (eRAI No. 8802
Attachments: Request for Additional Information No. 8802 (eRAI No. 8802).pdf

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

Please submit your 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.

Hearing Identifier: NuScale_SMR_DC_TR_Public
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Subject: Topical Report (TR-0516-49417) - Request for Additional Information Letter No. 8802 (eRAI No. 8802
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Request for Additional Information No. 8802 (eRAI No. 8802)

Issue Date: 06/23/2017
Application Title: NuScale Topical Report
Operating Company: NuScale
Docket No. PROJ0769
Review Section: 01 - Introduction and Interfaces
Application Section: 01

QUESTIONS

01-11

Title 10 of the Code of Federal Regulations (CFR), Part 50, Appendix A, General Design Criterion (GDC), "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). Title 10 of CFR, Part 50, Appendix A, GDC 12, "Suppression of reactor power oscillations," requires that the reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding SAFDLs are not possible or can be reliably and readily detected and suppressed.

Standard Review Plan (SRP) Section 15.0.2, "Review of Accident and Transient Analysis Methods," states that the reviewers should confirm that sources of code uncertainty have been addressed, including uncertainties in plant model input parameters for plant operating conditions." SRP 15.0.2 refers to Regulatory Guide 1.203, "Transient and Accident Analysis Methods," which indicates that model and correlation information including original source, supporting database, accuracy and applicability should be documented as part of a models and document evaluation report. SRP 15.0.2 also states that the chosen mathematical models and numerical solution of those models must be able to predict important physical phenomena reasonably well from both qualitative and quantitative points of view.

Section 5.6.1 of TR-0516-49417-P, "Neutron Kinetics," describes the reactor kinetics models, but does not adequately describe the methods for determining the nuclear data used in the analysis. To demonstrate compliance with GDCs 10 and 12:

1. Provide a description of the methods used to generate the nuclear data used in PIM.

Generally, methods such as steady-state nuclear design codes such as CASMO/SIMULATE are used to generate kinetics data such as reactivity coefficients, delayed neutron fraction, etc. If this is the case:

2. Describe how off-rated conditions are treated. For example, PIM calculations are performed at various power levels; are the kinetics parameters calculated using another tool at these same off-rated conditions to be supplied to PIM?

Justify any assumptions or approximations in the methods for specifying the nuclear data. In the specific case of the moderator temperature coefficient, the TR is not clear as to whether the method captures the reactivity effect from the temperature and density changes of the moderator in one lumped parameter.

3. Justify assumptions regarding the method for calculating the moderator temperature coefficient. This justification should consider the reactivity effect of changes to both the moderator temperature and density.

4. Confirm that kinetics data (i.e., delayed neutron fraction, decay group constants, coefficients, etc.) used in licensing evaluations will be based on NRC-approved nuclear design methods and that these methods will account for the most limiting point in cycle and most limiting thermal-hydraulic conditions of allowable operation or that conservatively bounding parameters will be used.