RAIO-0817-55443

August 15, 2017

Docket: PROJ0769

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 8802 (eRAI No. 8802) on the NuScale Topical Report, "Evaluation Methodology for Stability Analysis of the NuScale Power Module," TR-0516-49417, Revision 0

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 8802 (eRAI No. 8802)," dated June 23, 2017
2. NuScale Topical Report, "Evaluation Methodology for Stability Analysis of the NuScale Power Module," TR-0516-49417, Revision 0, dated July 2016

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The Enclosure to this letter contains NuScale's response to the following RAI Question from NRC eRAI No. 8802:

- 01-11

This letter and the enclosed response make no new regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions on this response, please contact Darrell Gardner at 980-349-4829 or at dgardner@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 8802

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NuScale Response to NRC Request for Additional Information eRAI No. 8802

Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 8802
Date of RAI Issue: 06/23/2017

NRC Question No.: 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 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.

NuScale Response:

The neutron kinetics parameters in PIM were prepared by processing SIMULATE5 analyses of a representative NuScale reactor core (RXC) on a generic basis; there is no intention of representing cycle-specific data. User-controlled multipliers are programmed in PIM to test for sensitivities, however, analyses with maximum nuclear data variations due to beginning versus end of cycle depletion states demonstrate no such sensitivities and the generic nature of the analysis (not cycle-specific) is supported. The same nuclear data apply to rated and off-rated conditions.

The moderator reactivity is fitted from SIMULATE5 analyses of a representative RXC with exposure dependence. The moderator reactivity is dependent on moderator density, where the effect of moderator temperature at constant density is negligible. When the moderator is in single-phase, the moderator density itself is temperature-dependent. In this case, representing the reactivity feedback via a moderator density coefficient (MDC) or a moderator temperature coefficient (MTC) is equivalent. Traditionally, PWR moderator reactivity is represented as a temperature coefficient, while for BWR a density (or void) coefficient is used. For NuScale, the code PIM can optionally apply MTC or MDC, where the latter is needed in order to be able to capture reactivity changes even after the moderator temperature reaches saturation and the temperature cannot increase further and the reactivity change would be erroneously truncated.

It should be noted that the independent variables in fitting the moderator coefficient in PIM are density and burnup. Reactivity effects due to moderator temperature as an independent variable are insignificant, but they are implied in the SIMULATE5 analysis and there is no need to make an explicit representation for both density and temperature.

Impact on Topical Report:
There are no impacts to the Topical Report TR-0516-49417, Evaluation Methodology for Stability Analysis of the NuScale Power Module, as a result of this response.