RAIO-1117-57263

November 20, 2017

Docket: PROJ0769

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 9104 (eRAI No. 9104) 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. 9104 (eRAI No. 9104)," dated September 21, 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 Questions from NRC eRAI No. 9104:

- 01-45
- 01-46
- 01-47

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. 9104

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9104

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Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 9104
Date of RAI Issue: 09/21/2017

NRC Question No.: 01-45

Title 10 of the Code of Federal Regulations (CFR), Part 50, Appendix A, General. Design Criterion (GDC) 10, "Reactor design," states 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 are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. GDC12, "Suppression of reactor power oscillations," requires that oscillations be either not possible or reliably detected and suppressed. The Standard Review Plan (SRP) 15.0.2 acceptance criteria with respect to evaluation models specifies that 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.

Section 5.7, "Material Properties," of the topical report (TR), TR-0516-49417-P, indicates that steam generator (SG) material properties are input by the user and are selected from standard references.

In order to make an affirmative finding NRC staff requests NuScale to:
1) Provide the source of the material properties assumed for the SG tubes.
2) Clarify what is meant by the term "standard references" in the Stability TR.

NuScale Response:

Item 1:

The source of the material for the SG tubes (SB-163 Alloy 690) is American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, 2013 Edition, Section II, "Materials," New York, NY.

Item 2:

By "standard" the TR suggests the use of commonly cited open sources for the widely used stainless steel properties. The NuScale standard reference for the material properties for use in engineering calculations of the NPM include the ASME BPVC material properties.

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.

Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 9104
Date of RAI Issue: 09/21/2017

NRC Question No.: 01-46

Title 10 of the Code of Federal Regulations (CFR), Part 50, Appendix A, General. Design Criterion (GDC) 10, “Reactor design,” states 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 are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. GDC12, "Suppression of reactor power oscillations," requires that oscillations be either not possible or reliably detected and suppressed. The Standard Review Plan (SRP) 15.0.2 acceptance criteria with respect to evaluation models specifies that 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.

Section 5.6.4.3, "Numerical Solution Procedure," of the topical report (TR), TR-0516-49417-P of the topical report describes the method for calculating the pellet temperature based on the heat flux and gap conductance. The method is predicated on the pellet heat flux being equivalent to the heat flux on the cladding inside surface, however the fuel thermal time constant does not to take into account gap thermal resistance.

In order to make an affirmative finding NRC staff requests NuScale to:
1) Explain why the time constant does not include a term to account for the gap.
2) Justify any assumptions regarding the thermal resistance of the gap on the analysis.

NuScale Response:

Item 1:

Eqn. 5.88 in the TR calls for a fuel pellet time constant, which is not the same as a fuel rod time constant. Only the latter would need contribution from the pellet-clad gap and the clad wall. The time constant of the pellet represents a pellet only lumped parameter model which is coupled to the gap and clad models to produce the overall pin thermal response.

Item 2:

The gap conductance (reciprocal of resistance) is not computed internally in PIM, but is provided as a constant input parameter. The response to RAI 9098 includes figures representing the effect of wide variation of the pellet-clad gap conductance. The figures demonstrate that the effect of the gap on the stability of the module is not significant.

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.

Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 9104
Date of RAI Issue: 09/21/2017

NRC Question No.: 01-47

Title 10 of the Code of Federal Regulations (CFR), Part 50, Appendix A, General. Design Criterion (GDC) 10, “Reactor design,” states 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 are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. GDC12, "Suppression of reactor power oscillations," requires that oscillations be either not possible or reliably detected and suppressed. The Standard Review Plan (SRP) 15.0.2 acceptance criteria with respect to evaluation models specifies that 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.

In section 5.6.4.4, "Pellet Centerline and Average Temperature," of the stability topical report, TR-0516-49417-P, describes the pellet conductivity dependence on burnup. However rationale for using the local exposure dependent fuel thermal conductivity model noted in this section (section 5.6.4.4) is requested below.

In order to make an affirmative finding NRC staff requests NuScale to provide justification of the exposure dependent fuel thermal conductivity model. It is acceptable to respond to this RAI by providing validation of the model against data.

NuScale Response:

Eqn. 5.108 for irradiated UO_2 thermal conductivity given in stability topical report TR-0516-49417-P is a simplification of Eqn. of 2.52 of FRAPCON-3.5 code report (Ref. 1). The same simple form of the thermal conductivity as function of temperature and burnup is used in Eqn. 3.3 of FRAPCON-3 report (Ref. 2) in Section 3.2 Experimental Evidence for Burnup Degradation.

In Section 3.2 of Ref. 2, experimental evidence of thermal conductivity degradation cited the Halden Ultra-High-Burnup Experiment. The experiments indicated degradation in urania fuel thermal conductivity (averaged over the temperature range from ~750 to 1200K) of ~5 to 7%

relative per 10 GWd/MTU, for burnups up to 88 GWd/MTU. This magnitude of thermal conductivity degradation is in good agreement with the formula used in PIM as shown in Figure 1.

Figure 1 is a plot of the percent relative decrease of thermal conductivity per 10 MWD/kg burnup at a temperature of 900 K, versus burnup. The relative degradation is obtained directly from Eqn. 5.108 of the TR as:

$$\frac{\Delta k}{k} = \frac{0.00187\ \Delta B}{0.0452 + 2.46\times10^{-4}T + 0.00187B}$$

For burnup change of Δ*B* = 10 MWD/kg, and at a temperature of *T* = 900 K, the percent relative degradation is obtained as:

$$\frac{\Delta k}{k}(\text{percent per 10 MWD/kg}) = \frac{1.87}{0.2666 + 0.00187B}$$

Figure 1 shows the percent relative degradation in the correct experimental range of ~5 to 7%.

References

1. NUREG-CR-7022-V1, “FRAPCON-3.5: A Computer Code for the Calculation of Steady-State, Thermal-Mechanical Behavior of Oxide Fuel Rods for High Burnup”(NRC ADAMS: ML14295A539). (Reference 12.1.34 in the TR)

2. NUREG-CR-6534 Vol 1, “FRAPCON-3: Modifications to Fuel Rod Properties and Performance Models for High-Burnup Application”

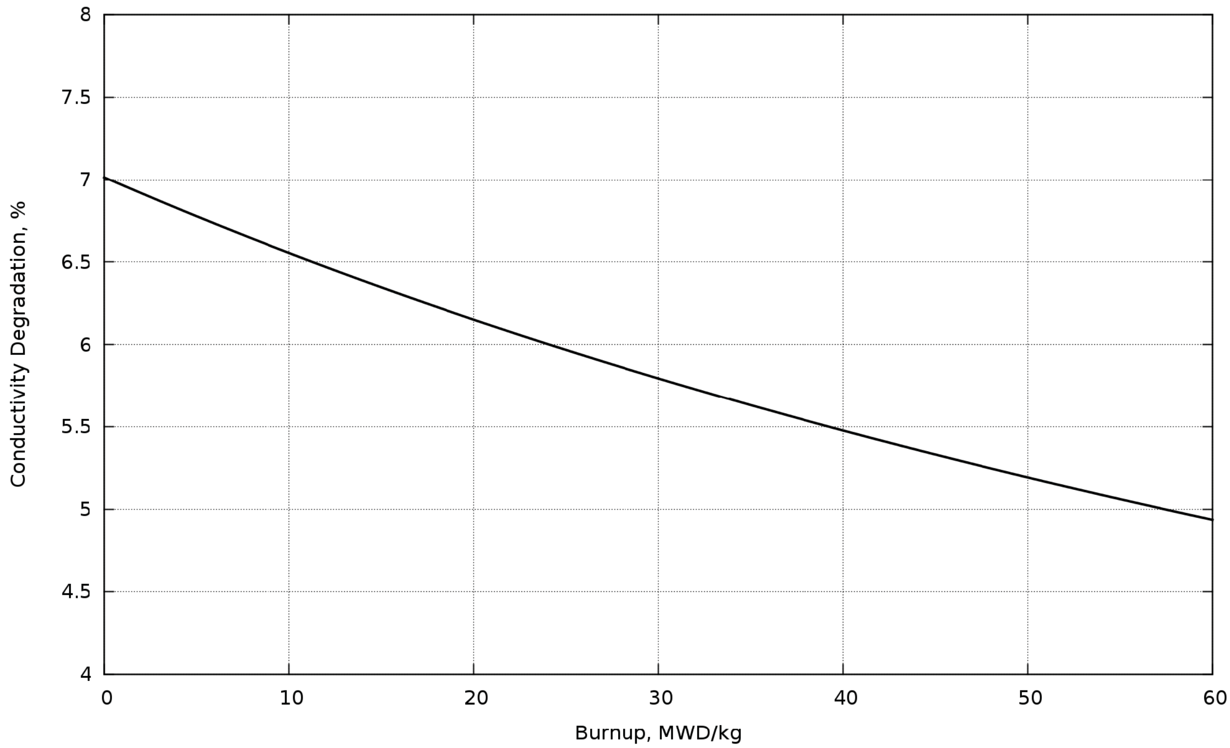

Figure 1 Percent relative thermal conductivity degradation per 10 MWD/kg burnup

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.