NuScaleDCRaisPEm Resource

From:	Chowdhury, Prosanta
Sent:	Friday, April 13, 2018 4:31 PM
To:	Request for Additional Information
Cc:	Lee, Samuel; Cranston, Gregory; Franovich, Rani; Karas, Rebecca; Burja, Alexandra; NuScaleDCRaisPEm Resource
Subject:	Request for Additional Information No. 422 eRAI No. 9512 (15.04.03)
Attachments:	Request for Additional Information No. 422 (eRAI No. 9512).pdf

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

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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Division of New Reactor Licensing
Office of New Reactors
U.S. Nuclear Regulatory Commission
301-415-1647

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Request for Additional Information No. 422 (eRAI No. 9512)

Issue Date: 04/13/2018
Application Title: NuScale Standard Design Certification - 52-048
Operating Company: NuScale Power, LLC
Docket No. 52-048
Review Section: 15.04.03 - Control Rod Misoperation (System Malfunction or Operator Error)
Application Section: FSAR Section 15.4.3

QUESTIONS

15.04.03-2

General Design Criterion 10, "Reactor design," in Title 10 of the Code of Federal Regulations (10 CFR) Part 50, Appendix A, 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). GDC 13 requires the provision of instrumentation to monitor variables and systems over their anticipated ranges of normal operation, including the effects of AOOs, and of appropriate controls to maintain listed variables and systems within prescribed operating ranges.

According to TR-0915-17564-P, “Subchannel Analysis Methodology,” the operating boundary conditions that are input into the subchannel analysis must account for measurement uncertainty. The staff understands that if biases are applied to parameters in the transient code input, biases for those parameters need not be applied in the subchannel analysis. However, when considering a steady-state analysis such as a static control rod misalignment, the proper biases should be applied in the subchannel analysis. The staff audited engineering calculation (EC)-0000-4309, “Subchannel Analysis of a Control Rod Misalignment,” which supports FSAR Section 15.4.3, and notes that the applied system pressure bias of **[[ ]]** psia is not consistent with the 70 psia bias specified in FSAR Tier 2, Table 15.0-6, “Module Initial Conditions Ranges for Design Basis Event Evaluation.” Using a bias of the incorrect magnitude could produce non-limiting results for the minimum critical heat flux ratio (MCHFR) or linear heat generation rate evaluation. Therefore, please confirm whether the correct reactor coolant system pressure bias was applied in the subchannel analysis for this event. If it was not, either provide a revised analysis, or justify why the current analysis results remain valid. Update the FSAR as necessary.

In addition, the staff requests clarification of whether the **[[ ]]** bias in core inlet temperature listed in EC-0000-4309 is consistent with a 10°F bias in RCS average temperature, as specified in FSAR Tier 2, Table 15.0-6. If it is not, provide a revised analysis that uses a core inlet temperature bias consistent with a 10°F bias in RCS average temperature, or justify why the current analysis results remain valid. Update the FSAR as necessary.

Finally, the staff notes that the system pressure and core inlet temperature values and biases, along with other key inputs to the subchannel analysis, influence MCHFR and are therefore necessary for the staff's safety finding with respect to GDC 10 and 13. Therefore, update the FSAR to include the key inputs and assumptions for the control rod misalignment subchannel analysis, including, but not limited to, the primary side parameters identified in FSAR Table 15.0-6.

15.04.03-3

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. NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” (SRP) Section 15.4.3, “Control Rod Misoperation (System Malfunction or Operator Error),” provides guidance to the staff in determining compliance with GDC 10, among several other GDC, and states that for each failure event analyzed, the cases which result in a limiting fuel rod condition should be presented. Initial conditions and parameter values selected for these cases should be justified with a sensitivity analysis or discussion. Conditions of first-order importance for any time in cycle are initial power level and distribution, initial rod configuration, reactivity addition rate, moderator temperature, fuel temperature, and void reactivity coefficients.

While auditing EC-A021-2405, “Control Rod Misalignment Analysis,” which supports FSAR Section 15.4.3, the staff noted that open design item (ODI)-16-1030 requires verification that the low power hold point is at 25 percent power because the subchannel analysis does not postulate a misalignment event that occurs below this hold point. To ensure the case that results in a limiting fuel rod condition is presented, do one of the following: confirm the low power hold point is at 25 percent power, provide an analysis for a case with all control rod assemblies (CRAs) fully withdrawn except for one CRA fully inserted, or provide justification that such a case is less limiting than the misalignment case presented in the FSAR. Update the FSAR as appropriate.

Furthermore, while auditing EC-0000-4309, the staff noted that the axial power shape and system boundary conditions correspond to **[[ ]]** power, while the limiting radial augmentation factor is for **[[ ]]**. EC-0000-4309 further states that using the **[[ ]]**, but the staff notes that the axial power peaking increases with decreasing power. Given the reduced axial power peaking at **[[ ]]** power, please provide justification that assuming an axial power shape and/or boundary conditions associated with a lower power level would not be limiting for this event. Update the FSAR as necessary.

15.04.03-4

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. SRP Section 15.4.3 provides guidance to the staff in determining compliance with GDC 10, among several other GDC, and states that for each failure event analyzed, the cases which result in a limiting fuel rod condition should be presented. Initial conditions and parameter values selected for these cases should be justified with a sensitivity analysis or discussion. Conditions of first-order importance for any time in cycle are initial power level and distribution, initial rod configuration, reactivity addition rate, moderator temperature, fuel temperature, and void reactivity coefficients.

The staff notes that a maximum (least negative) MTC is typically limiting for a single CRA withdrawal event because it minimizes reactivity feedback that mitigates the power increase.

FSAR Tier 2, Table 15.4-6, “Key Inputs for Single CRA Withdrawal with Limiting MCHFR,” indicates that the moderator temperature coefficient (MTC) value used in the single CRA withdrawal analysis is a power-dependent value (-6 pcm/°F) based on the initial power level of 75%. However, based on FSAR Tier 2, Figure 4.3-13, “Moderator Temperature Coefficient of Reactivity at Full Power,” and Figure 4.3-14, “Moderator Temperature Coefficient of Reactivity at Zero Power,” the maximum MTC at an RCS temperature of 535°F (the temperature assumed in the analysis) is about -5 pcm/°F. Furthermore, the analysis in FSAR Section 15.4.2, “Uncontrolled Control Rod Assembly Withdrawal at Power,” assumes an MTC value of 0 pcm/°F for the analysis at 75 percent power.

Therefore, provide further justification that the -6 pcm/°F MTC is bounding and conservative for the single CRA withdrawal analysis, or alternatively, provide a new analysis using a bounding value for MTC, such as 0 pcm/°F. Update the FSAR as necessary.

In addition, Table 15.4-6 does not list the initial RCS flow assumed for the analysis. Because initial RCS flow affects MCHFR, update the table to include the value for initial RCS flow.

15.04.03-5

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. SRP Section 15.4.3 provides guidance to the staff in determining compliance with GDC 10, among several other GDC, and states that for each failure event analyzed, the cases which result in a limiting fuel rod condition should be presented. Initial conditions and parameter values selected for these cases should be justified with a sensitivity analysis or discussion. Conditions of first-order importance for any time in cycle are initial power level and distribution, initial rod configuration, reactivity addition rate, moderator temperature, fuel temperature, and void reactivity coefficients.

In auditing EC-0000-2139, Revision 1, “Control Rod Misoperation Transient Analysis,” one of the calculations supporting FSAR Section 15.4.3, the staff noted that the sensitivity cases to determine the limiting single CRA withdrawal case do not appear conclusive. In particular, only two cases **[[ ]]** were examined. Both **[[**

]]. Therefore, it appears that more limiting cases initiating from **[[ ]]** may be possible by biasing parameters to further delay the **[[ ]]** and by considering other reactivity insertion rates.

In light of this observation, provide additional justification that the limiting single CRA withdrawal case has been identified. If a more limiting case is identified, update the FSAR as appropriate.

15.04.03-6

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. SRP Section 15.4.3 provides guidance to the staff in determining compliance with GDC 10, among several other GDC, and states that for each failure event analyzed, the cases which result in a limiting fuel rod condition

should be presented. Initial conditions and parameter values selected for these cases should be justified with a sensitivity analysis or discussion. In addition, the specific acceptance criteria to ensure the SAFDLs are met include the departure from nucleate boiling ratio (for NuScale, MCHFR) being met and fuel centerline temperatures not exceeding the melting point.

FSAR Tier 2, Section 15.4.3, and the related Tables 15.4-6 and 15.4-8, “Key Inputs for CRA Drop with Limiting MCHFR,” discuss and provide input parameters and initial conditions for the limiting MCHFR case. However, there is no discussion of the initial conditions or results of the maximum linear heat generation rate (LHGR) case, which provides conclusions regarding the fuel centerline melting acceptance criterion, aside from the LHGR value itself. To enable the staff to ensure that the limiting results for fuel centerline temperature have been identified, update the FSAR to include the key initial conditions and a high-level discussion of the single CRA withdrawal and rod drop event results for their respective limiting LHGR cases.