NuScaleTRRaisPEm Resource

From:	Chowdhury, Prosanta
Sent:	Wednesday, April 4, 2018 11:17 AM
To:	Request for Additional Information
Cc:	Lee, Samuel; Cranston, Gregory; Karas, Rebecca; Van Wert, Christopher; Franovich, Rani; NuScaleTRRaisPEm Resource
Subject:	Request for Additional Information Letter No. 9306 (eRAI No. 9306) Topical Report, Rod Ejection Accident Methodology, 15.04.08, SRSB
Attachments:	Request for Additional Information No. 9306 (eRAI No. 9306).pdf

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

Please submit your technically correct and complete 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.

Prosanta Chowdhury, Project Manager
Licensing Branch 1 (NuScale)
Division of New Reactor Licensing
Office of New Reactors
U.S. Nuclear Regulatory Commission
301-415-164

Hearing Identifier: NuScale_SMR_DC_TR_Public
Email Number: 81

Mail Envelope Properties (BN7PR09MB2609197AF0C2B0F088406D489EA40)

Subject: Request for Additional Information Letter No. 9306 (eRAI No. 9306) Topical Report, Rod Ejection Accident Methodology, 15.04.08, SRSB
Sent Date: 4/4/2018 11:16:33 AM
Received Date: 4/4/2018 11:16:38 AM
From: Chowdhury, Prosanta

Created By: Prosanta.Chowdhury@nrc.gov

Recipients:
"Lee, Samuel" <Samuel.Lee@nrc.gov>
Tracking Status: None
"Cranston, Gregory" <Gregory.Cranston@nrc.gov>
Tracking Status: None
"Karas, Rebecca" <Rebecca.Karas@nrc.gov>
Tracking Status: None
"Van Wert, Christopher" <Christopher.VanWert@nrc.gov>
Tracking Status: None
"Franovich, Rani" <Rani.Franovich@nrc.gov>
Tracking Status: None
"NuScaleTRRaisPEm Resource" <NuScaleTRRaisPEm.Resource@nrc.gov>
Tracking Status: None
"Request for Additional Information" <RAI@nuscalepower.com>
Tracking Status: None

Post Office: BN7PR09MB2609.namprd09.prod.outlook.com

Files	**Size**	**Date & Time**	
MESSAGE	531	4/4/2018 11:16:38 AM	
Request for Additional Information No. 9306 (eRAI No. 9306).pdf			110687

Options
Priority: Standard
Return Notification: No
Reply Requested: No
Sensitivity: Normal
Expiration Date:
Recipients Received:

Request for Additional Information No. 9306 (eRAI No. 9306)

Issue Date: 04/04/2018
Application Title: NuScale Topical Report
Operating Company: NuScale
Docket No. PROJ0769
Review Section: 15.04.08 - Spectrum of Rod Ejection Accidents (PWR)
Application Section:

QUESTIONS

15.04.08-1

In accordance with 10 CFR 50 Appendix A GDC 28, “Reactivity Limits,” the reactivity control systems must be designed with appropriate limits on potential reactivity increases so the effects of a rod ejection accident (REA) can result in neither damage to the reactor coolant pressure boundary nor result in sufficient disturbance to significantly impair the core cooling capability. SRP Section 15.4.8 provides review guidance related to the spectrum of REAs. For an applicant to accurately analyze its plant design for an REA, the underlying software used as part of the applicant’s methodology must be properly verified and validated.

Section 3.2.1.4 of Topical Report TR-0716-50350-P, “Rod Ejection Accident Methodology,” Revision 0, provides the validation of SIMULATE-3K, which is used to provide a three-dimensional nodal reactor kinetics solution. This section indicates that the SPERT-III benchmark and the NEACRP REA problem were used to validate SIMULATE-3K for the purpose of REA analyses. The references for the validation of SIMULATE-3K against SPERT-III and NEACRP appear to be based on conference proceedings. Neither a summary of results nor an analysis of bias or uncertainty is provided. The referenced conference proceedings are not part of the applicant’s Appendix B quality assurance program and, therefore, the robustness of the validation is not demonstrated. As such, the staff makes the following requests:

a. Provide a plot of the comparison between the SIMULATE-3K model and the SPERT-III benchmark results.
b. Provide a summary of the SIMULATE-3K comparison against the NEACRP REA benchmark problem.
c. Provide a reference for a complete verification/validation analysis of SIMULATE-3K under an Appendix B quality assurance program.

15.04.08-2

In accordance with 10 CFR 50 Appendix A GDC 28, “Reactivity Limits,” the reactivity control systems must be designed with appropriate limits on potential reactivity increases so the effects of a REA can result in neither damage to the reactor coolant pressure boundary nor result in sufficient disturbance to significantly impair the core cooling capability. For an applicant to correctly predict fuel failures resulting from overheating of the fuel cladding in support of demonstrating compliance with GDC 28, the fuel melting analysis methodology must be shown to conservatively calculate the fuel centerline temperature.

In Section 5.5.1, NuScale provides Equation 5-2, which is used to calculate the temperature increase. The staff notes that the equation uses the maximum nodal peaking factor input before the control rod assembly (CRA) moves. It is unclear to the staff if using the maximum F_Q calculated before any CRA moves would bound the use of F_Q calculated after the rod is ejected.

Provide justification for using the maximum F_Q as determined before the beginning of the transient to calculate the maximum fuel temperature.

15.04.08-3

In accordance with 10 CFR 50 Appendix A GDC 28, “Reactivity Limits,” the reactivity control systems must be designed with appropriate limits on potential reactivity increases so the effects of a REA can result in neither damage to the reactor coolant pressure boundary nor result in sufficient disturbance to significantly impair the core cooling capability. SRP Section 15.4.8 provides review guidance related to the spectrum of REAs. For an applicant to correctly predict fuel failures resulting from overheating of the fuel cladding in support of demonstrating compliance with GDC 28, any analysis must demonstrate that the limiting condition is analyzed.

In Section 5.3.3 of TR-0716-50350-P, NuScale states, “[s]coping of the [maximum critical heat flux ration (MCHFR)] can be performed to determine the generally limiting scenarios; final MCHFR calculations will defer to the sub-channel analyses.” It is unclear to the staff how the scoping analysis ensures that the limiting case(s) are performed in the VIPRE-01 sub-channel analysis.

Provide additional description of the scoping study used to provide assurance that the limiting RELAP5 MCHFR cases correctly determine which VIPRE-01cases are analyzed

15.04.08-4

In accordance with 10 CFR 50 Appendix A GDC 28, “Reactivity Limits,” the reactivity control systems must be designed with appropriate limits on potential reactivity increases so the effects of a REA can result in neither damage to the reactor coolant pressure boundary nor result in sufficient disturbance to significantly impair the core cooling capability. SRP Section 15.4.8 and SRP Section 4.2, Appendix B, provide review guidance related to the spectrum of REAs. The applicant must use computer codes to demonstrate its compliance with appropriate limits and utilize models that represent the phenomena associated with the event being analyzed. In addition, the applicant must use conservative inputs to ensure that the analysis bounds allowed plant operation accounting for uncertainties.

Section 3.2 of TR-0716-50350-P describes the computer codes and analysis flow that make up the methodology for analysis of the REA. In addition, reference is made to a manual calculation that is used for the adiabatic heat-up for the fuel response. The staff requires additional information concerning the models and inputs used in the REA analysis methodology to determine compliance with the above regulation and guidance.

a. Please describe the models used for the REA analysis for each code. The staff specifically requests a description of how the core is represented with SIMULTATE 5 and SIMULATE-3K and the thermal hydraulic parameters passed from SIMULATE5 to SIMULATE-3K to establish initial conditions for the SIMULATE-3K analysis.
b. Similarly, describe the parameters passed from SIMULATE-3K to both NRELAP and VIPRE-01.
c. State whether or not the models used in the REA for NRELAP5 and VIPRE-01 differ from those described in the referenced topical reports for each code. If the models differ, provide further description and justification for the changes.
d. Describe how the thermal hydraulic initial conditions (including uncertainties) are determined to conservatively calculate MCHFR.

15.04.08-5

In accordance with 10 CFR 50 Appendix A GDC 28, “Reactivity Limits,” the reactivity control systems must be designed with appropriate limits on potential reactivity increases so the effects of a REA can result in neither damage to the reactor coolant pressure boundary nor result in sufficient disturbance to significantly impair the core cooling capability. SRP Section 15.4.8 and SRP Section 4.2, Appendix B, provide review guidance related to the spectrum of REAs. The applicant must use computer codes to demonstrate the compliance with appropriate limits and utilize models that capture the phenomena associated with the event being analyzed.

Section 3.2.1.3 of TR-0716-50350 states that SIMULATE-3K is used to determine the power response for the accident, which is subsequently used in NRELAP5 and VIPRE-01. The power response is dependent on the timing of the reactor trip and is critical in the analysis of the REA in limiting clad damage. For the most limiting cases a reactor trip is expected from high flux rate or high neutron flux signal. TR-0716-50350-P does not describe how SIMULATE-3K modeled the excore detectors.

Describe how the excore detectors are modeled in the SIMULATE-3K analysis

15.04.08-6

In accordance with 10 CFR 50 Appendix A GDC 28, “Reactivity Limits,” the reactivity control systems must be designed with appropriate limits on potential reactivity increases so the effects of a REA can result in neither damage to the reactor coolant pressure boundary nor result in sufficient disturbance to significantly impair the core cooling capability. SRP Section 15.4.8 and SRP Section 4.2, Appendix B, provide review guidance related to the spectrum of REAs. In performing the analysis of the REA the applicant must select inputs to assure a bounding calculation that would envelop plant operation and possible future cycle designs and reflect limits in Technical Specifications or COLR.

Sections 4.1.1.1, 4.1.1.2, and 5.2.1.1 of TR-0716-50350-P discuss the application of uncertainty factors applied to SIMULATE-3K for the rod ejection analysis. For intrinsically (code determined) parameters in Table 5-1 (DTC, B_{eff}, ejected CRA worth, MTC) it is unclear to the staff how the multipliers are applied to SIMULATE-3K.

Describe in detail how these uncertainty multipliers for intrinsically determined parameters are applied to SIMULATE-3K.

15.04.08-7

In accordance with 10 CFR 50 Appendix A GDC 28, “Reactivity Limits,” the reactivity control systems must be designed with appropriate limits on potential reactivity increases so the effects of a REA can result in neither damage to the reactor coolant pressure boundary nor result in sufficient disturbance to significantly impair the core cooling capability. SRP Section 15.4.8 and SRP Section 4.2, Appendix B, provide review guidance related to the spectrum of REAs. To demonstrate compliance with the above the applicant must model the fuel to calculate the amount of energy deposited throughout the REA and whether or not clad damage occurs.

Section 4.1.2 of TR-0716-50350-P states that several effects are not modeled because of the assumption that all of the energy is deposited in the fuel pellet with no loses from conduction. Section 4.1.2.2 of TR-0716-50350-P further states that fuel cladding is considered in <u>both</u> the VIPRE01 CHF evaluation and the adiabatic heat-up calculation.

Please clarify this apparent discrepancy.

15.04.08-8

In accordance with 10 CFR 50 Appendix A GDC 28, “Reactivity Limits,” the reactivity control systems must be designed with appropriate limits on potential reactivity increases so the effects of a REA can result in neither damage to the reactor coolant pressure boundary nor result in sufficient disturbance to significantly impair the core cooling capability. SRP Section 15.4.8 and SRP Section 4.2, Appendix B, provide review guidance related to the spectrum of REAs. In demonstrating that the above criteria are met the applicant must consider all possible control rod configurations allowed.

Section 4.3 B of TR-0716-50350-P identifies the limiting rod worth for the REA and states this will occur when the rods are at the power-dependent insertion limits (PDIL) and all calculations will begin from this point consistent with Appendix A of Regulatory Guide 1.77 "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors". However, the staff notes that plant operation per Technical Specification 3.1.6, Regulating Group Insertion Limits, allows operation with rod positions above the PDILs (FSAR Figure 4.3-2). As noted in Regulatory Guide 1.77, "a sufficient number of initial reactor states to completely bracket all possible operational conditions of interest should be analyzed...". If a rod above the PDILs is ejected a reactor trip may be delayed or may not occur at all which could be limiting from a deposited energy or MCHFR perspective.

Provide justification for the assertion that other allowed rod configurations (other than at PDIL) would not result in a more limiting case (more closely approach acceptance limits) for scenarios in which a reactor trip is delayed or not achieved.

15.04.08-9

In accordance with 10 CFR 50 Appendix A GDC 28, “Reactivity Limits,” the reactivity control systems must be designed with appropriate limits on potential reactivity increases so the effects of a REA can result in neither damage to the reactor coolant pressure boundary nor result in sufficient disturbance to significantly impair the core cooling capability. SRP Section 15.4.8 and SRP Section 4.2, Appendix B, provide review guidance related to the spectrum of REAs. In demonstrating that the above is met the applicant must select conservative inputs to bound allowable plant operation.

Section 4.3.E of TR-0716-50350-P states that the primary core flow for the REA is not allowed to increase. The method for determining the core flow is unclear to the staff.

Please describe the process for determining the initial core flow to ensure a conservative calculation for each initial core power and operating condition.

15.04.08-10

In accordance with 10 CFR 50 Appendix A GDC 28, “Reactivity Limits,” the reactivity control systems must be designed with appropriate limits on potential reactivity increases so the effects of a REA can result in neither damage to the reactor coolant pressure boundary nor result in sufficient disturbance to significantly impair the core cooling capability. SRP Section 15.4.8 and SRP Section 4.2, Appendix B, provide review guidance related to the spectrum of REAs. In demonstrating that the above is met the applicant must select conservative inputs to bound allowable plant operation.

Section 5.1.2 of TR-0716-50350-P indicates that the REA is analyzed at three burnup points during the cycle: beginning of cycle (BOC), end of cycle (EOC), and at the point of maximum $F_{\Delta H}$. It is unclear to the staff if this methodology assures a conservative set of parameters for the critical heat flux (CHF) and adiabatic fuel rod heat-up calculations.

a. Please provide justification that the point of maximum $F_{\Delta H}$ results in a conservative set of parameters in the REA analysis of both CHF and adiabatic fuel rod heat-up.
b. Does the maximum $F_{\Delta H}$ occur at the same burnup as the maximum F_Q?

15.04.08-11

In accordance with 10 CFR 50 Appendix A GDC 28, “Reactivity Limits,” the reactivity control systems must be designed with appropriate limits on potential reactivity increases so the effects of a REA can result in neither damage to the reactor coolant pressure boundary nor result in sufficient disturbance to significantly impair the core cooling capability. SRP Section 15.4.8 and SRP Section 4.2, Appendix B, provide review guidance related to the spectrum of REAs. In demonstrating that the above is met the applicant must select conservative inputs to bound allowable plant operation.

Section 5.1.3 states that analysis of the REA will be performed at power levels from hot zero power (HZP) to hot full power (HFP) to bound the PDIL, axial offset limits, and moderator

temperature. It is unclear to the staff, from the methodology described, how these values will be applied.

Describe the process for selecting and biasing these parameters to ensure a conservative analysis for the REA. For example, at low power levels the limits on axial offset are unbounded. Describe how the axial shape is determined to bound the axial offset limits specified for all power levels.

15.04.08-12

In accordance with 10 CFR 50 Appendix A GDC 28, “Reactivity Limits,” the reactivity control systems must be designed with appropriate limits on potential reactivity increases so the effects of a REA can result in neither damage to the reactor coolant pressure boundary nor result in sufficient disturbance to significantly impair the core cooling capability. SRP Section 15.4.8 and SRP Section 4.2, Appendix B, provide review guidance related to the spectrum of REAs. In demonstrating that the above is met the applicant must select conservative inputs to bound allowable plant operation.

Section 5.2.1.1, “Static Calculations,” and Section 5.3.1.1, “Minimum Critical Heat Flux Ratio,” of TR-0716-50350-P state that the coolant mass flux is one of the initial conditions that are passed to SIMULATE-3K and VIPRE-01. However, the method for deriving the coolant mass flux is not described.

How is this coolant mass flux derived and how does it vary with core power?

15.04.08-13

In accordance with 10 CFR 50 Appendix A GDC 28, “Reactivity Limits,” the reactivity control systems must be designed with appropriate limits on potential reactivity increases so the effects of a REA can result in neither damage to the reactor coolant pressure boundary nor result in sufficient disturbance to significantly impair the core cooling capability. SRP Section 15.4.8 and SRP Section 4.2, Appendix B, provide review guidance related to the spectrum of REAs. In demonstrating that the above is met the applicant must select conservative inputs to bound allowable plant operation.

Section 6.0 of TR-0716-50350-P describes a series of sample calculations illustrating the REA methodology. The staff requires additional information on how the initial thermal hydraulic conditions selected (including uncertainties applied) are derived in the REA analysis.

a. How is the initial T_{avg} selected as a function of power in the power dependent initial conditions selected for the REA analysis?
b. What is the flow rate assumed for the HZP cases, what is the basis for this value and how is it controlled as part of the rod ejection analysis?

15.04.08-14

In accordance with 10 CFR 50 Appendix A GDC 28, “Reactivity Limits,” the reactivity control systems must be designed with appropriate limits on potential reactivity increases so the effects of a REA can result in neither damage to the reactor coolant pressure boundary nor result in sufficient disturbance to significantly impair the core cooling capability. SRP Section 15.4.8 and SRP Section 4.2, Appendix B, provide review guidance related to the spectrum of REAs. In demonstrating that the above is met the applicant must select conservative inputs to bound allowable plant operation.

Section 6.2 of TR-0716-50350-P states that “…hot zero power MCHFR calculations are not a part of the REA analysis scope…” However, the staff notes that no justification is provided for this assumption. In addition, the staff notes on sample calculation results provided in Table 6-2, “Sample results for rod ejection accident analysis, beginning of cycle and middle of cycle, both regulating groups” that the BOC, 80% power and BOC, 100% power, NRELAP5 screening cases where not performed. It is unclear to the staff why NRELAP5 screening is not performed for these conditions.

a. Provide justification that MCHFR calculations at HZP are not part of the REA analysis scope.
b. Provide information or justification as to why these cases are not part of the rod ejection MCHFR screening methodology.

15.04.08-15

In accordance with 10 CFR 50 Appendix A GDC 28, “Reactivity Limits,” the reactivity control systems must be designed with appropriate limits on potential reactivity increases so the effects of a REA can result in neither damage to the reactor coolant pressure boundary nor result in sufficient disturbance to significantly impair the core cooling capability. For an applicant to correctly predict fuel failures resulting from overheating of the fuel cladding in support of demonstrating compliance with GDC 28, the fuel melting analysis methodology must be shown to conservatively calculate the fuel centerline temperature.

Table 5-1, “Uncertainties for REA calculations,” of TR-0716-50350 provides the uncertainties applied to the rod ejection analysis. It is unclear to the staff if the uncertainties in Table 5-1 will be updated as described in Section 7.0 of the “Nuclear Analysis Codes and Methods Qualification” topical report (TR-0616-48793, Rev. 0). The staff also notes that the $F_{\Delta H}$ provided in Table 5-1 is less conservative than the $F_{\Delta H}$ given in Section 7.7.1, “Base Nuclear Reliability Factors,” of TR-0616-48793.

a. Please indicate if the uncertainties in Table 5-1 will be updated consistent with TR-0616-48793. If the uncertainties will not be updated as discussed in TR-0616-48793, either describe the method for updating them or provide a justification as to why an update is not necessary. If the uncertainties in Table 5-1 will be updated, modify TR-0716-50350 to indicate the method by which updates will be made.
b. Justify the use of a lower $F_{\Delta H}$ uncertainty for the rod ejection analysis relative to the steady-state $F_{\Delta H}$ uncertainty.

15.04.08-16

In accordance with 10 CFR 50 Appendix A GDC 28, “Reactivity Limits,” the reactivity control systems must be designed with appropriate limits on potential reactivity increases so the effects of a REA can result in neither damage to the reactor coolant pressure boundary nor result in sufficient disturbance to significantly impair the core cooling capability. SRP Section 15.4.8 provides review guidance related to the spectrum of REAs.

Section 5.5.1 of TR-0716-50350-P states that the change in fuel centerline temperature determined by Equation 5-2 is added to the initial fuel centerline temperature as the bounding starting temperature. Likewise, the change in enthalpy, as calculated by Equation 5-4, is dependent on the maximum pre-transient fuel centerline temperature as described by Equation 5-3. Section 3.2.1.3 of TR-0716-50350-P, “SIMULATE-3K,” states that within-pin fuel temperature distribution is governed by the one-dimensional radial heat conduction equation. Section 3.2.1.3 of TR-0716-50350-P goes on to state that material properties are temperature and burnup dependent, and gap conductance is dependent on exposure and fuel temperature. This method assumes the transient, within pellet radial temperature distribution remains constant (i.e., initial steady-state, within pellet radial shape is preserved). In a rod ejection transient, within pellet radial power distributions may not remain constant (e.g., radial power profile may become more edge peaked).

Demonstrate that the proposed method produces a conservative, maximum fuel pellet temperature. As part of this demonstration describe how SIMULATE-3K is used to determine the initial within pellet radial temperature distribution and provide comparisons, including the effects of burnup-dependent thermal conductivity degradation, to either experimental data or an NRC approved fuel performance code to show a reasonably conservative initial (steady-state) temperature distribution.