RAIO-0618-60285

June 04, 2018

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. 9306 (eRAI No. 9306) on the NuScale Topical Report, "Rod Ejection Accident Methodology," TR-0716-50350, Revision 0

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 9306 (eRAI No. 9306)," dated April 04, 2018
2. NuScale Topical Report, "Rod Ejection Accident Methodology," TR-0716-50350, Revision 0, dated December 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 Enclosures to this letter contain NuScale's response to the following RAI Questions from NRC eRAI No. 9306:

- 15.04.08-1
- 15.04.08-2
- 15.04.08-3
- 15.04.08-4
- 15.04.08-5
- 15.04.08-6
- 15.04.08-7
- 15.04.08-8
- 15.04.08-9
- 15.04.08-10
- 15.04.08-11
- 15.04.08-12
- 15.04.08-13
- 15.04.08-14
- 15.04.08-15
- 15.04.08-16

Enclosure 1 is the proprietary version of the NuScale Response to NRC RAI No. 9306 (eRAI No. 9306). NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The proprietary enclosures have been deemed to contain Export Controlled Information. This information must be protected from disclosure per the requirements of 10 CFR § 810. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 2 is the nonproprietary version of the NuScale response.

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

If you have any questions on this response, please contact Paul Infanger at 541-452-7351 or at pinfanger@nuscalepower.com.

Sincerely,

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

Distribution: Gregory Cranston, NRC, OWFN-8G9A
Samuel Lee, NRC, OWFN-8G9A
Rani Franovich, NRC, OWFN-8G9A

Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9306, proprietary

Enclosure 2: NuScale Response to NRC Request for Additional Information eRAI No. 9306, nonproprietary

Enclosure 3: Affidavit of Zackary W. Rad, AF-0618-60286

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9306, proprietary

Enclosure 2:

NuScale Response to NRC Request for Additional Information eRAI No. 9306, nonproprietary

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9306
Date of RAI Issue: 04/04/2018

NRC Question No.: 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.

NuScale Response:

a. Studsvik Scandpower performed the SPERT-III benchmark demonstrating the ability of SIMULATE-3K to model the transient response of the reactor (Reference 8.2.22 of TR-0716-50350). Although not performed under NuScale's approved Appendix B quality assurance (QA) program, Studsvik performed this benchmark as part of their V&V program to demonstrate the ability of SIMULATE-3K to perform reactivity insertion events

for super-prompt conditions. The SIMULATE-3K and SPERT-III benchmark comparisons show good agreement between SIMULATE-3K and the experimental results providing confidence that SIMULATE-3K can predict core power excursions for reactivity insertion events as seen in Figure 3 (S3K vs SPERT-III Cold Start-up Test 43. Inserted Reactivity 1.21$), Figure 5 (S3K vs SPERT-III Hot Start-up Test 70. Inserted Reactivity 1.21$), Figure 7 (S3K vs SPERT-III Hot Start-up Test 60. Inserted Reactivity 1.23$), Figure 9 (S3K vs SPERT-III Hot Stand-by Test 81. Inserted Reactivity 1.17$), and Figure 11 (S3K vs SPERT-III Full PowerTest 86. Inserted Reactivity 1.17$) of Reference 8.2.22. Differences in peak power shown in Figure 3 and Figure 9 of Reference 8.2.22 are attributed to experimental uncertainty in the initial position of the transient control rod, leading to uncertainty in the initial reactivity insertion.

b. The validation of SIMULATE-3K includes the performance of the NEACRP REA benchmark problem by Studsvik (discussed in Section 3.2.1.4 of TR-0716-50350) as part of their V&V of the code. NuScale performed the NEACRP REA benchmark problem as part of the code validation under NuScale's approved Appendix B QA program (Reference 8.1.3 of TR-0716-50350). Table 1 provides a comparison of the SIMULATE-3K results obtained by NuScale against the NEACRP benchmark reference solutions. The results show good agreement between SIMULATE-3K and the benchmark reference solutions providing confidence that SIMULATE-3K can model and adequately predict results for the rod ejection event.

Table 1: NEACRP Benchmark Results Comparison

Parameter	Case	NEACRP	S3K	Δ	%Δ
Critical Boron Concentration (ppm)	A1	567.7	{{		
	A2	1160.6			
	B1	1254.6			
	B2	1189.4			
	C1	1135.3			
	C2	1160.6			
Reactivity Release (pcm)	A1	822			
	A2	90			
	B1	831			
	B2	99			
	C1	958			
	C2	78			
Maximum Power (%)	A1	117.9			
	A2	108.0			
	B1	244.1			
	B2	106.3			
	C1	477.3			
	C2	107.1			}}2(a),(c)

Time of Maximum Power (s)	A1	0.56	{{		
	A2	0.10			
	B1	0.52			
	B2	0.12			
	C1	0.27			
	C2	0.10			
Final Power (%)	A1	19.6			
	A2	103.5			
	B1	32.0			
	B2	103.8			
	C1	14.6			
	C2	103.0			
Final Average Doppler Temperature (°C)	A1	324.3			
	A2	554.6			
	B1	349.9			
	B2	552.0			
	C1	315.9			
	C2	553.5			
Final Maximum Centerline Temperature (°C)	A1	673.3			
	A2	1691.8			
	B1	559.8			
	B2	1588.1			
	C1	676.1			
	C2	1733.5			
Final Coolant Outlet Temperature (°C)	A1	293.1			
	A2	324.6			
	B1	297.6			
	B2	324.7			
	C1	291.5			
	C2	324.5			}}2(a),(c)

c. The software development of the SIMULATE-3K code was performed by Studsvik Scandpower and was delivered to NuScale as a compiled, commercial software package. V&V activities were performed by the code developer prior to delivery of the software package to demonstrate that the code can correctly perform the functions intended and accurately predict results. Multiple transient benchmark problems were performed by Studsvik as part of their V&V process, including SPERT-III and the NEACRP REA benchmark problem.

The software package delivery to NuScale was accompanied by installation test cases and user manual, methodology, and version change documentation. Upon delivery, configuration control is initiated and the software was subjected to appropriate controls within the NuScale QA program (Reference 8.1.3). In addition to the V&V activities performed by Studsvik Scandpower, software validation is performed for applications and use specific to NuScale. The QA program is compliant with Reference 8.1.1 of TR-0716-50350. The QA program governs activities associated with acquisition of

commercial grade software, configuration control, validation, and dedication of the SIMULATE-3K code.

The commercial software was placed under configuration control and installation testing was performed using the test case inputs and reference solutions included with the software delivery. This installation testing ensures that the software has been installed properly by comparing solutions of the test case inputs to the reference solutions and ensuring there are no unexpected differences in the results. After successful installation, validation and benchmarking demonstrates the code performs the functions intended for NuScale applications. Section 3.2.1.4 describes the SIMULATE-3K validation performed by NuScale. In addition to the code validation detailed in Section 3.2.1.4, The NEACRP benchmark results are provided in response to RAI question 15.04.08-1(b). All software used to support this topical report is appropriately controlled under the NRC approved NuScale QA program.

Impact on DCA:

There are no impacts to the DCA as a result of this response.

Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 9306
Date of RAI Issue: 04/04/2018

NRC Question No.: 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.

NuScale Response:

The wording in Section 4.1.2.1 and 5.5.1 of the Rod Ejection Accident Methodology topical report (TR-0716-50350) was clarificatied as indicated in the markup provided with this response. The definition for $F_{Q,max}$, "maximum nodal peaking factor before CRA moves" in Equation 5-2 in the topical report is referring to maximum nodal peaking during the reactor transient which occurs before the reactor scrams. The analysis uses the peak F_Q that occurs during the power pulse which occurs after the rod ejection, but before the remainder of the CRAs move as a result of reactor scram. Thus, the correct power peaking is utilized in the analysis.

Impact on Topical Report:

Topical Report TR-0716-50350, Rod Ejection Accident Methodology, has been revised as described in the response above and as shown in the markup provided in this response.

4.1.2 Fuel Response Analysis

4.1.2.1 Initial Conditions

The initial conditions from the industry PIRT noted in Table 4-2 are input to the adiabatic heatup analysis. However, several of these effects are not modeled because of the assumption that all of the energy is deposited into the fuel pellet with no losses due to conduction. Therefore, no consideration is given to gap size, gas distribution, hydrogen distribution, fuel-clad gap friction coefficient, coolant conditions, or bubble size and distribution.

Cladding dimensions are used to calculate the maximum oxide to wall thickness ratio. ~~{{~~This ratio is 0.0588 for the NuScale fuel~~}}~~[2(a),(c),ECI]; the fuel enthalpy rise limit is conservatively set at the inflection point of the 0.08 ratio in Figure 5-2. Using this ratio applies additional conservatism to the allowable fuel enthalpy rise.

Pellet dimensions are used when calculating the nodal volume for the adiabatic heatup calculations. A smaller pellet is conservative, as the enthalpy and temperature rise are inversely proportional to the volume as shown in Equation 5-3 and Equation 5-4. Manufacturing tolerances are thus applied to the pellet dimensions to conservatively calculate the fuel enthalpy and temperature.

Power distribution, in the form of pin peaking factors, is discussed in Section 4.1.1.2.

The condition of oxidation is accounted for in the maximum oxide to wall thickness ratio. As noted above in the cladding dimension discussion, using the inflection point, which corresponds to a higher allowed fuel enthalpy rise than that for the calculated ratio, is effectively applying an uncertainty factor to the oxidation condition.

The transient power is accounted for when integrating the thermal energy created by the power pulse~~ before CRA movement~~. This is conservatively accounted for by assuming all of the energy is deposited into the fuel pellet, including the area under the initial power level.

4.1.2.2 Fuel and Cladding Temperature Changes

Heat resistances and~~in the fuel and fuel cladding,~~ heat capacities of the fuel and fuel cladding, and coolant conditions are addressed ~~both~~ in the VIPRE-01 CHF evaluation~~and adiabatic heatup calculation~~. The~~se~~ parameters for the adiabatic heatup application are discussed in Section 4.1.1.2.

For VIPRE-01 analyses, these parameters are addressed in the fuel rod conduction model. The fuel rod conduction model uses a calibration to COPERNIC (References 8.2.8 and 8.2.12) to develop conservative fuel property input that captures all of the effects of heat transfer from the fuel pellet to the fuel cladding, and ultimately to the coolant. Application of this model is discussed in Section 4.4 of the subchannel methodology topical report (Reference 8.2.11). As described in this report, calibration of VIPRE-01 fuel temperature predictions to the fuel performance analyses is performed for the fuel average, fuel surface, and cladding surface temperatures for each cycle. Fuel-

5.4.2.5 Reactor Coolant System Pressure

It is appropriate to bias pressure in the positive direction (increase pressure) for pressures above {{ }}2(a),(c),ECI psia to achieve a conservative MCHFR. The MCHFR sensitivity to RCS initial pressure is provided in Section 6.4.2.7.

5.4.3 Results and Downstream Applicability

The VIPRE-01 analysis is used to demonstrate that no fuel failures are present, using the regulatory criteria outlined in Section 2.1.

5.5 Fuel Response

For the fuel response, namely the fuel temperature and radial average fuel enthalpy, simplified calculations assuming adiabatic heatup within the fuel is performed. For this calculation, the total energy during the transient is integrated. This energy is then converted into either a temperature or enthalpy increase. This calculation takes into consideration the fuel geometry, fuel heat capacity, and power peaking factors.

This approach is conservative as no energy is allowed to leave the fuel. The total reactor power is integrated from event initiation until the point at which CRAs begin entering the core during reactor trip, including the power below the initial power.

5.5.1 Fuel Temperature

The following equation defines the conservative temperature increase:

$$\Delta T = \frac{E_T * F_{Q,max}}{C_p * V_{node} * n_{nodes}} \qquad \text{Equation 5-2}$$

where,

ΔT = temperature increase,

E_T = total energy,

$F_{Q,max}$ = maximum nodal peaking factor before reactor trip~~CRA moves~~. Uncertainty is applied to this parameter (Table 5-1),

C_p = volumetric fuel heat capacity,

V_{node} = nodal volume, and

n_{nodes} = total number of nodes in the core.

Using the initial fuel centerline temperature as the bounding starting temperature, adding the calculated ΔT to this value provides a bounding final temperature for the fuel. If this final temperature, using the conservatism within this calculation is below the incipient fuel melting temperature of **[ ]** degrees F (Reference 8.2.12), core coolability is achieved.

Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 9306
Date of RAI Issue: 04/04/2018

NRC Question No.: 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

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NuScale Response:

Section 4.3.5 of the Non-LOCA Methodology topical report (TR-0516-49416) describes that NRELAP5 CHF calculations for the dummy hot rod are used as a screening tool to assist in determining limiting transient cases to be evaluated in downstream subchannel analyses. For this purpose it has been demonstrated that minimum CHFR calculated by NRELAP5 trends consistently with the VIPRE-01 minimum CHFR for given changes in power, flow, pressure, and inlet temperature. Thus, the use of CHF values calculated by NRELAP5 as part of the system transient pre-screening process are used to identify cases for downstream subchannel analysis. The NRELAP5 calculation is not used to demonstrate that margin to the minimum CHFR is maintained; the dummy hot rod results are used only to assist analysts in identifying potentially limiting transient cases to be evaluated in downstream subchannel analyses. This process is also applied in the Rod Ejection Accident Methodology topical report. Therefore, scoping of the

MCHFR is performed to determine the likely limiting scenarios and the final MCHFR calculations are performed in the subchannel analyses using VIPRE-01 and the approved CHF correlations to calculate the limiting MCFHR.

Information is added to the Rod Ejection Accident Methodology topical report in the markup provided with this response referencing the MCHFR scoping method in the Non-LOCA Methodology topical report.

Impact on Topical Report:

Topical Report TR-0716-50350, Rod Ejection Accident Methodology, has been revised as described in the response above and as shown in the markup provided in this response.

5.3.2.6 System Pressure and Pressurizer Level

System pressure and pressurizer level are addressed for MCHFR and system pressurization in Sections 5.3.1.1 and 5.3.1.2.

5.3.3 Results and Downstream Applicability

The primary result of the system response is the peak RPV pressure. Scoping of the MCHFR can be performed to determine the generally limiting scenarios as described in Section 4.3.5 of the Non-LOCA Methodology topical report (Reference 8.2.10); final MCHFR calculations for the limiting scenarios are performed by~~will defer to~~ the subchannel analyses.

The overall plant response determined by the NRELAP5 calculations is transferred to the subchannel and fuel response analysis for calculation of MCHFR and radial average fuel enthalpy to establish that fuel cladding failure has not occurred.

5.4 Subchannel Response

5.4.1 Subchannel Calculation Procedure

The subchannel scope of calculations considers the MCHFR acceptance criteria. A hot channel that applies all the limiting conditions bounding all other channels in the core is modeled. The boundary conditions from NRELAP5 of core exit pressure, system flow, and core inlet temperature and the power forcing function from SIMULATE-3K are applied to the VIPRE-01 model. The MCHFR calculations are performed to verify that CHF is not reached during the event for any rods.

5.4.1.1 VIPRE-01 Deviations from Subchannel Methodology

With the rapid nature of the power increase in the REA VIPRE-01 calculations, several deviations from the subchannel methodology described in Reference 8.2.11 were used to increase the convergence and reliability of the final results. These changes are described below.

- {{

}}2(a),(c)

8.2.9 NuScale Topical Report, "Loss-of-Coolant Accident Evaluation Model," TR-0516-49422, Revision 0, dated December 2016.

8.2.10 NuScale Topical Report, "Non-Loss-of-Coolant Accident Analysis Methodology~~Non-LOCA Methodologies~~," TR-0516-49416 Revision ~~0,~~1, August 2017.

8.2.11 NuScale Topical Report, "Subchannel Analysis Methodology," TR-0915-17564, Revision 0, October 2016.

8.2.12 BAW-10231P-A, "COPERNIC Fuel Rod Design Computer Code," January 2004.

8.2.13 Hetrick, D. L., "Dynamics of Nuclear Reactors," ANS, Illinois, pp. 64 and 166, 1993.

8.2.14 EPRI Technical Report 1003385, "Three-Dimensional Rod Ejection Accident Peak Fuel Enthalpy Analysis Methodology," November 2002.

8.2.15 U.S. Nuclear Regulatory Commission, "Safety Evaluation by the Office of Nuclear Reactor Regulation Relating to VIPRE-01 Mod 02 for PWR and BWR Applications, EPRI-NP-2511-CCMA, Revision 3," October 30, 1993.

8.2.16 CASMO5: A Fuel Assembly Burnup Program User's Manual, SSP-07/431 Revision 7. Studsvik Scandpower, December 2013.

8.2.17 SIMULATE5 Advanced Three-Dimensional Multigroup Reactor Analysis Code, SSP-10/438 Revision 4. Studsvik Scandpower, December 2013.

8.2.18 SIMULATE-3K Extended Fuel Pin Model, SSP-05/458 Revision 1. Studsvik Scandpower, March 2008.

8.2.19 SIMULATE-3K Input Specification, SSP-98/12 Revision 17. Studsvik Scandpower, September 2013.

8.2.20 SIMULATE-3K Models and Methodology, SSP-98/13 Revision 9. Studsvik Scandpower, September 2013.

8.2.21 R. McCardell, et.al., "Reactivity Accident Test Results and Analyses for the SPERT III E-Core – A Small, Oxide-Fueled, Pressurized Water Reactor," IDO-17281. March 1969.

8.2.22 G. Grandi, "Validation of CASMO5 / SIMULATE-3K Using the Special Power Excursion Test Reactor III E-Core: Cold Start-Up, Hot Start-Up, Hot Standby and Full Power Conditions." Proceedings of PHYSOR 2014, Kyoto, Japan, September 28-October 3, 2014.

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9306
Date of RAI Issue: 04/04/2018

NRC Question No.: 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.

NuScale Response:

Response to parts a) and b): For detailed specification of how the core is represented in

SIMULATE5 and SIMULATE-3K, please see Section 3.0 of the Nuclear Analysis Codes and Methods topical report, TR-0616-48793 (Reference 8.2.7). In general, the SIMULATE5 core model is based on input of the core geometry and material compositions, core operating conditions, and core configuration. At a high level, the core geometry is fully represented with radial nodes corresponding to a quarter of an assembly at numerous axial levels with material properties and cross-sections assigned to each node.

As described in Section 5.2 of the Rod Ejection Accident Methodology (REAM) topical report (TR) (TR-0716-50350), for the nuclear analysis component of the calculation, the core model defined in SIMULATE5 is passed to SIMULATE-3K via a detailed restart file establishing the initial conditions of the core before the start of the transient. The SIMULATE-3K input file may be modified for differences between the codes, including modifications for inlet temperature, spacer grid information, and CRA composition. SIMULATE-3K uses inlet temperature as input, and SIMULATE5 uses average temperature, thus that parameter is adjusted in SIMULATE-3K. SIMULATE5 treats the spacer grids explicitly, but SIMULATE-3K input must homogenize the spacer grids over the active fuel length, so spacer grid data must be adjusted for SIMULATE-3K. Also, CRA input limitations require simplifications of the SIMULATE5 CRA inputs (made conservatively) to model the NuScale CRAs in SIMULATE-3K. As described in Reference 8.2.25 of the REAM TR, the SIMULATE-3K has a different thermal-hydraulics model than SIMULATE5. In summary, the core model and initial conditions for the SIMULATE-3K analysis are set by reading the appropriate SIMULATE5 restart file, making required adjustments to account for differences between the codes, biasing reactivity coefficients (Section 5.2.1), and providing transient-specific inputs (Section 5.2.2).

Sections 5.3, 5.4, and 5.5 of the topical report, respectively, describe that the power as a function of time calculated by SIMULATE-3K is used as input into NRELAP5, VIPRE-01, and the adiabatic fuel response calculation. Additionally, elements of the power distributions are used as input to VIPRE-01 and the adiabatic fuel response calculations. The NRELAP5 calculation then provides the core power (same as the power provided by SIMULATE-3K), core inlet flow, core inlet temperature, and core exit pressure forcing functions to VIPRE-01.

A simplified definition of the discipline and code interfaces is presented in Table 1, below, arranged such that the discipline in the row receives input from the discipline defined in the column:

Table 1: High-Level Discipline/Code Interface Cross-Reference

Discipline	**Steady-State Nuclear (SIMULATE5)**	**Transient Nuclear (SIMULATE-3K)**	**Transients (NRELAP5)**
Transient Nuclear (SIMULATE-3K)	Steady-state boundary conditions	N/A	N/A
Transients (NRELAP5)	Reactivity coefficients Kinetics parameters	Power vs. Time	N/A
Adiabatic Fuel Response	N/A	Power vs. Time F_Q vs. Time	N/A
Subchannel (VIPRE-01)	N/A	Radial power distribution (includes $F_{\Delta H}$) Axial power distribution	Event thermal-hydraulic response (power, flow, temperature, pressure)

c) In general, the models for NRELAP5 and VIPRE-01 described in the REAM TR do not differ from those as described in their respective topical reports. Section 5.4.1.1 of the REAM TR describes the deviations of the VIPRE-01 model used, which are related to adjusting the model for convergence to accommodate smaller time steps than typically used for other events. For NRELAP5 cases, the only change is to the point kinetics model, which is removed and replaced by a case-specific power versus time forcing function input from the upstream SIMULATE-3K calculation.

d) The methodology presented in the REAM TR is a cycle-specific detailed analysis, generally using best-estimate tools and input conditions. A full spectrum of initial conditions are evaluated to ensure that a conservative value for each acceptance criterion is calculated. For some inputs as identified in the topical report, bounding assumptions through the biasing of input parameters are utilized to simplify the methodology (reduce the number of initial condition permutations explicitly analyzed), while maintaining conservatism in the calculation of each acceptance criterion. To ensure conservatism of the thermal-hydraulic conditions, the NRELAP5 analysis determines conservative treatment of system conditions. For specific conservative treatment of system conditions, please refer to Section 5.3.1.1 of the REAM TR. For the screened NRELAP5 cases, the subchannel analysis uses the calculated case-specific power, flow, temperature, and pressure forcing functions to conservatively calculate MCHFR.

Impact on DCA:

There are no impacts to the DCA as a result of this response.

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9306
Date of RAI Issue: 04/04/2018

NRC Question No.: 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

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NuScale Response:

As described in Section 4.3 of the Rod Ejection Accident Methodology topical report (TR-0716-50350), the reactor trip has a negligible effect on the limiting cases, because the limiting cases experience prompt or near prompt criticality due to the reactivity insertion. For the limiting cases, the Doppler feedback effectively mitigates the event before reactor trip occurs, thus reducing the importance of the excore detectors and reactor trip for mitigating the event.

Figure 6-6 of the topical report is an example MCHFR plot as a function of time based on the power pulse provided in Figure 6-5. The peak power occurs at approximately 80 milliseconds, with a half-width-half-max pulse width lasting 60 milliseconds. An additional 60 milliseconds past the time of the peak pulse, the minimum CHFR occurs at approximately 140 milliseconds after the start of the event. The analytical limit delay for the reactor trip to begin (approximately 2

seconds of rod movement for full insertion) once detected is less than 2.5 seconds. Therefore, the key elements of the event are completely over before the excore detectors could be credited to intiate the reactor trip.

The excore model in SIMULATE-3K requires a description of detector geometry relative to the center of the core and the outer radius of the pressure vessel, placed on-axis at 0°, 90°, 180°, and 270°. Trip signals are generated based on the change in flux calculated at the detector location relative to the initial condition flux estimate. Standard modeling techniques as recommended by the SIMULATE5 and SIMULATE-3K user guidance are utilized.

Impact on DCA:

There are no impacts to the DCA as a result of this response.

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9306
Date of RAI Issue: 04/04/2018

NRC Question No.: 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.

NuScale Response:

Conservative allowances are made for uncertainties in nuclear parameters that most significantly impact the modeling of the event. The ‘KIN.MUL’ card is used in SIMULATE-3K, which applies conservatism to a stated parameter equal to the uncertainty in that parameter. Conservatisms are applied to beta (β, delayed neutron fraction), FTC (fuel temperature coefficient, also known as Doppler coefficient), MTC (moderator temperature coefficient), and CRA (control rod assembly) worth. As a result, the cross-sections (reactivity feedback) are effectively adjusted based on the conservative factors applied to each parameter. Cases are run in steady-state to determine the correct multipliers to apply to the stated parameters to produce conservative results which bound the uncertainty in the stated parameters. These multipliers are then input to the SIMULATE-3K transient cases to account for the uncertainties in the nuclear parameters. Section 7.0 of TR-0616-48793 (Reference 8.2.7) provides more detail on the

background and derivation of the nuclear reliability factors utilized to account for code uncertainty.

The discussion below explains how uncertainty is incorporated for the intrinsically determined parameters of FTC, beta, MTC, and CRA worth in SIMULATE-3K:

- CRA worth uncertainty is applied to the ejected CRA worth, and to the worth of the CRAs inserted after the reactor trip. The 'KIN.MUL' card is used to apply conservatism to each based on the rod worth nuclear reliability factor. The 'KIN.MUL' input is iterated on until the result is equal to the assumed conservatism in the stated parameters. The uncertainty multiplier for inserted rod worth is set to a constant value that bounds the nuclear reliability factors applied to the rods after SCRAM. The ejected rod worth undergoes iteration to determine the correct multiplier so that the ejected rod worth is equal to the best-estimate rod worth for that location adjusted to include the nuclear reliability factor.
- The nuclear reliability factor for MTC is applied through the 'KIN.MUL' multiplier in SIMULATE-3K, which is iterated on until the correct MTC is achieved.
- For β and FTC, no iteration is necessary, because the uncertainty applies directly as a multiplier on the base value.

Impact on DCA:

There are no impacts to the DCA as a result of this response.

Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 9306
Date of RAI Issue: 04/04/2018

NRC Question No.: 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 both the VIPRE01 CHF evaluation and the adiabatic heat-up calculation.

Please clarify this apparent discrepancy.

NuScale Response:

The first paragraph in Section 4.1.2.2 was clarified as indicated in the markup provided with this response. The assumption that “all of the energy is deposited in the fuel pellet with no losses from conduction” only applies to the adiabatic calculation. The VIPRE-01 fuel rod conduction model does not make this assumption. Hence, the adiabatic model is not the appropriate method for calculating clad heat transfer for use in MCHFR assessments.

Impact on Topical Report:

Topical Report TR-0716-50350, Rod Ejection Accident Methodology, has been revised as described in the response above and as shown in the markup provided in this response.

4.1.2 Fuel Response Analysis

4.1.2.1 Initial Conditions

The initial conditions from the industry PIRT noted in Table 4-2 are input to the adiabatic heatup analysis. However, several of these effects are not modeled because of the assumption that all of the energy is deposited into the fuel pellet with no losses due to conduction. Therefore, no consideration is given to gap size, gas distribution, hydrogen distribution, fuel-clad gap friction coefficient, coolant conditions, or bubble size and distribution.

Cladding dimensions are used to calculate the maximum oxide to wall thickness ratio. ~~{{~~This ratio is 0.0588 for the NuScale fuel~~}}~~[2(a),(c),ECI]; the fuel enthalpy rise limit is conservatively set at the inflection point of the 0.08 ratio in Figure 5-2. Using this ratio applies additional conservatism to the allowable fuel enthalpy rise.

Pellet dimensions are used when calculating the nodal volume for the adiabatic heatup calculations. A smaller pellet is conservative, as the enthalpy and temperature rise are inversely proportional to the volume as shown in Equation 5-3 and Equation 5-4. Manufacturing tolerances are thus applied to the pellet dimensions to conservatively calculate the fuel enthalpy and temperature.

Power distribution, in the form of pin peaking factors, is discussed in Section 4.1.1.2.

The condition of oxidation is accounted for in the maximum oxide to wall thickness ratio. As noted above in the cladding dimension discussion, using the inflection point, which corresponds to a higher allowed fuel enthalpy rise than that for the calculated ratio, is effectively applying an uncertainty factor to the oxidation condition.

The transient power is accounted for when integrating the thermal energy created by the power pulse~~ before CRA movement~~. This is conservatively accounted for by assuming all of the energy is deposited into the fuel pellet, including the area under the initial power level.

4.1.2.2 Fuel and Cladding Temperature Changes

Heat resistances and~~in the fuel and fuel cladding,~~ heat capacities of the fuel and fuel cladding, and coolant conditions are addressed ~~both~~ in the VIPRE-01 CHF evaluation~~and adiabatic heatup calculation~~. The~~se~~ parameters for the adiabatic heatup application are discussed in Section 4.1.1.2.

For VIPRE-01 analyses, these parameters are addressed in the fuel rod conduction model. The fuel rod conduction model uses a calibration to COPERNIC (References 8.2.8 and 8.2.12) to develop conservative fuel property input that captures all of the effects of heat transfer from the fuel pellet to the fuel cladding, and ultimately to the coolant. Application of this model is discussed in Section 4.4 of the subchannel methodology topical report (Reference 8.2.11). As described in this report, calibration of VIPRE-01 fuel temperature predictions to the fuel performance analyses is performed for the fuel average, fuel surface, and cladding surface temperatures for each cycle. Fuel-

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Docket No. 52-048

eRAI No.: 9306
Date of RAI Issue: 04/04/2018

NRC Question No.: 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.

NuScale Response:

As described in Section 4.3 of the Rod Ejection Accident Methodology topical report (TR-0716-50350), the rod ejection event is driven by a rapid increase in local reactivity, resulting in a dynamic power excursion. There is a general correlation between the static reactivity worth of the ejected rod and the resulting height, width, and integrated energy of the power pulse when power is plotted as a function of time. This correlation is slightly noisy due to feedback

effects related from a variety of other key variables such as power distribution and reactivity feedback. The only other allowed rod configurations not analyzed by the methodology are those at insertion depths less than PDIL, in other words with rods less inserted, and thus, having a lower static worth. These configurations are non-limiting as the lower dynamic worth power excursions result in more benign transient conditions. This characteristic applies to the MCHFR and pressure acceptance criteria, as well as the fuel enthalpy criteria. For fuel enthalpy criteria (described in Section 5.5.3 of the topical report), the adiabatic heat up calculation does not take credit for a variable acceptance criteria, that is, a single value for the oxide wall thickness acceptance criteria is utilized. This is in contrast to alternate methodologies, in which individual best-estimate fuel rod enthalpy changes are compared to a variable acceptance criteria based on it's predicted oxide wall thickness. In this alternate methodology, if the event progression changes slightly, the location of the peak enthalpy change could occur in a different location in which the oxide wall thickness is greater, resulting in an acceptance criteria failure. As the NuScale methodology utilizes a conservative deterministic approach (a bounding calculation is compared to a single acceptance criteria), there is no risk of failure when the event progression changes. Therefore, the NuScale methodology results in a conservative evaluation of the event for all acceptance criteria.

A rod ejection that doesn't result in a reactor trip is bounded by a single rod withdrawal event, which is shown to result in acceptable MCHFR in Section 15.4.3 of the NuScale FSAR.

Impact on DCA:

There are no impacts to the DCA as a result of this response.

Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 9306
Date of RAI Issue: 04/04/2018

NRC Question No.: 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.

NuScale Response:

Paragraph 4.3.E in the Rod Ejection Accident Methodology topical report (TR-0716-50350) was modified to improve clarity as seen in the markup provided with this response. In the SIMULATE-3K calculation, the core flow for a given initial power is held constant through a modeling option input. The initial core flow is determined as a function of initial power based on the natural circulation flow curve. The reason for this modeling simplification is that the transient is effectively over much faster (<1 second) than the time it takes the primary coolant to transverse the coolant loop (~60 seconds at full power). In the NRELAP5 analysis, the core flow is allowed to increase, but the analysis is performed so that any flow increase is minimized through the use of the minimum design flow as described in Section 4.4.4.5.1 of the FSAR. The VIPRE-01 analysis uses the calculated core flow directly from NRELAP5 as an input forcing function.

Impact on Topical Report:

Topical Report TR-0716-50350, Rod Ejection Accident Methodology, has been revised as described in the response above and as shown in the markup provided in this response.

B. *From the initial conditions, considering all possible control rod patterns allowed by technical specification/core operating limit report power-dependent insertion limits, the limiting rod worths are determined.*

The limiting rod worths will occur when the rods are at the PDIL. All calculations will begin from this point.

C. *Reactivity coefficient values of the limiting initial conditions must be used at the beginning of the transient. The Doppler and moderator coefficients are the two of most interest. If there is no three-dimensional space-time calculation, the reactivity feedback must be weighted conservatively to account for the variation in the missing dimension(s).*

The application of the reactivity coefficients is discussed in Section 5.

D. *[…] control rod insertion assumptions, which include trip parameters, trip delay time, rod velocity curve, and differential rod worth.*

Reactor trip is conservatively applied in the methodology. However, for the REA evaluation, the reactor trip has a negligible effect on the limiting cases, because the limiting cases are those that experience prompt, or near prompt, criticality due to the reactivity insertion. These cases will turn around based on reactivity feedback, primarily due to DTC. Application of a reactor trip delay, reducing the reactor trip worth, or slowing the speed of CRA insertion capture effects that will occur well after the power peak, and consequently well after MCHFR. The reactor trip delay is used to determine the cutoff point for the energy integration for the adiabatic heatup evaluation of the fuel response, and for these cases a longer delay is conservative.

E. *[…] feedback mechanisms, number of delayed neutron groups, two-dimensional representation of fuel element distribution, primary flow treatment, and scram input.*

Feedback mechanisms are discussed in the section 3.1.1 and 3.2.1. The number of delayed neutron groups and two-dimensional representation of the fuel element are addressed in the code discussion in Section 3.2.1. For a given set of initial conditions, ~~P~~primary core flow is ~~not allowed to~~conservatively treated to minimize any flow increase, as increased flow would cause an increase in MCHFR~~, which is not conservative~~. Reactor trip input, though not explicitly important per Reference 8.2.26, will still be modeled in a conservative manner as noted in the above item D.

Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 9306
Date of RAI Issue: 04/04/2018

NRC Question No.: 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∆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∆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∆H occur at the same burnup as the maximum F_Q?

NuScale Response:

a) In general, end of cycle conditions maximize the dynamic response of the event. However, as part of a robust methodology a full spectrum of initial conditions are evaluated to ensure that a conservative value for each acceptance criterion is calculated. Beginning and end of cycle points bound the possible core reactivity conditions, with middle of cycle conditions between the two extremes. Evaluations of a middle of cycle point where $F_{\Delta H}$ is maximum are performed to ensure that the true limiting condition is found. It is expected that the limiting case will occur at the end of cycle because the delayed neutron fraction is minimized at this time, and a smaller delayed neutron fraction increases the reactivity insertion for a control rod assembly ejection. In the event that any middle of cycle points become limiting, additional analyses at a variety of middle of cycle points should be performed to ensure that the true limiting case is found .

This discussion has been added to Section 5.1.2 of the Rod Ejection Accident Methodology topical report (TR-0716-50350) as seen in the markup provided with this response.

b) The exposure at which the maximum $F_{\Delta H}$ occurs may not always be at the same exposure point as maximum F_Q. Both of these points typically do not occur at the end of cycle in which the limiting dynamic response occurs. With respect to the MCHFR calculation, the F_Q component is dependent on the treatment of the $F_{\Delta H}$ and the peak of the axial power shape (F_Z) as $F_Q = F_{\Delta H} * F_Z$ (F_Z must be defined on a rod basis for this equation to be true). In summary, the methodology utilizes a conservative determination of the limiting initial conditions (including exposure, power, and flow) that maximizes the dynamic response. For each unique dynamic response, the corresponding best-estimate power distribution is modeled in a conservative manner. Therefore, the limiting event is determined and modeled in a conservative manner.

Impact on Topical Report:

Topical Report TR-0716-50350, Rod Ejection Accident Methodology, has been revised as described in the response above and as shown in the markup provided in this response.

5.0 Rod Ejection Accident Analysis Methodology

As discussed in Section 3.0, the software used and the flow of information between specific codes in the REA analysis is depicted in Figure 3-1. This section describes the method for the use of these computer codes in the modeling of the REA in the unlikely event it should occur in the NuScale NPM. In addition, the methodology for the adiabatic heatup model is described. Major assumptions for each phase of the REA analysis are discussed within the text for that phase, while the general assumptions are presented at the beginning of this section.

5.1 Rod Ejection Accident Analysis General Assumptions

5.1.1 Cycle Design

The REA analysis will be performed for each core reload. Each reload may result in a different power response, both in magnitude as well as radial and axial distributions. As the underlying assumption for the NuScale REA methodology is that no fuel failures will occur, this assumption will need to be confirmed for any design changes that affect the input to the REA analysis.

The sample problem results provided in this report are from calculations performed using an equilibrium cycle.

5.1.2 Cycle Burnup

The REA is analyzed at three points during the cycle, BOC, EOC, and the point of maximum $F_{\Delta H}$. These three points ~~will~~should bound all core reactivity and power peaking considerations.

In general, end of cycle conditions maximize the dynamic response of the event. Beginning and end of cycle points bound the possible core reactivity conditions, with middle of cycle conditions between the two extremes. Evaluations of a middle of cycle point where $F_{\Delta H}$ is maximum are performed to ensure that the true limiting condition is found. It is expected that the limiting case will occur at the end of cycle because the delayed neutron fraction is minimized at this time, and a smaller delayed neutron fraction increases the reactivity insertion for CRA ejection. In the event that any middle of cycle points become limiting, additional analyses at a variety of middle of cycle points should be performed to ensure that the true limiting case is found.

5.1.3 Core Power

The REA is analyzed at power levels ranging from HZP to HFP. The power levels analyzed will bound the PDIL, axial offset limits, and moderator temperature over the NPM power range; these parameters feed into the reactivity insertion from a REA.

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9306
Date of RAI Issue: 04/04/2018

NRC Question No.: 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.

NuScale Response:

A full spectrum of initial conditions are evaluated to ensure that a conservative value for each acceptance criterion is calculated. For some inputs, as identified in the Rod Ejection Accident Methodology topical report (TR-0716-50350), bounding assumptions through the biasing of input parameters are utilized to simplify the methodology, while maintaining conservatism in the calculation of each acceptance criterion. To ensure conservatism of the analysis conditions, the following approach is utilized:

- Moderator Temperature : The moderator temperature is a function of core power and is set by the operating strategy for the plant. In the NRELAP5 analysis, the core flow is allowed to increase, however, the analysis is performed to minimize the flow increase

with temperature, calculated to satisfy mass and energy conservation. The VIPRE-01 analysis uses the calculated core flow and core inlet temperature directly from NRELAP5 as an input forcing function.

- Axial Offset : The xenon distribution is adjusted to provide a top peaked axial power shape at the axial offset window boundary, which maximizes the worth of the ejected rod. At low powers, no axial offset window boundary has been defined. For low powers, top peaked axial power shapes are produced which bound any axial power shapes possible while operating the core with rods inserted. Therefore, the rod ejection always occurs through a bounding top peaked shape to maximize the rod worth.

- Control Rod Assembly Insertion : Control rod assembly position is bounded by applying uncertainty to the PDIL at each given power level to maximize the initial insertion.

Impact on DCA:

There are no impacts to the DCA as a result of this response.

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9306
Date of RAI Issue: 04/04/2018

NRC Question No.: 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?

NuScale Response:

The core flow, and thus the coolant mass flux, in the SIMULATE-3K calculation for a given initial power is held constant through a modeling option. The initial core flow is determined as a function of initial power based on the natural circulation flow curve. The core flow in the NRELAP5 analysis is allowed to increase, but the analysis is performed to minimize the flow increase during the event. The VIPRE-01 analysis uses the calculated core flow directly from NRELAP5 as an input forcing function.

Impact on DCA:

There are no impacts to the DCA as a result of this response.

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9306
Date of RAI Issue: 04/04/2018

NRC Question No.: 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?

NuScale Response:

a. The moderator temperature is a function of core power and set by the operating strategy for the plant. In addition to the various safety analysis considerations such as thermal margins, the selection of the moderator temperature operating band is affected by thermodynamic efficiencies and the strategy for normal plant startup and shutdown. Section 4.4.4.5.1 of the FSAR provides more details on the primary coolant thermal-hydraulic characteristics. In the NRELAP5 analysis the temperature is initialized with a bounding high value. The VIPRE-01 analysis uses the calculated core flow and inlet temperature directly from NRELAP5 as an input forcing function.
b. In the plant flow will be established through a module heatup system as discussed in Section 5.1.4 of the FSAR at low flow (approximately 10% rated flow). In the NRELAP5 analysis the hot zero power flow rate is modeled based on the natural circulation curve of a very low power (for example 0.001%), which corresponds to the low flow of the module

heatup system. In the SIMULATE-3K analysis, the flow is modeled assuming a conservatively low value of 5% rated flow.

Impact on DCA:

There are no impacts to the DCA as a result of this response.

Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 9306
Date of RAI Issue: 04/04/2018

NRC Question No.: 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.

NuScale Response:

a) This statement was based on the interpretation of the wording in SRP 4.2 Appendix B, Item B.1, which states:

> The high cladding temperature failure criteria for zero power conditions is a peak radial average fuel enthalpy greater than 170 cal/g for fuel rods with an internal rod pressure at or below system pressure and 150 cal/g for fuel rods with an internal rod pressure exceeding system pressure. For intermediate (greater than 5% rated thermal power) and full power conditions, fuel cladding failure is

presumed if local heat flux exceeds thermal design limits (e.g. DNBR and CPR).

The wording of the guidance implies that below 5% power (i.e., hot zero power (HZP)) cladding temperature failures are based on fuel enthalpy, not thermal design limits (i.e., MCHFR). Due to the robust methodology established by NuScale, the possibility of MCHFR failures at HZP is inherently included in the methodology and analysis performed to support the FSAR. Generally, cases from HZP have very mild power excursion (reach <100% rated peak power) as opposed to the limiting cases which reach a peak power of greater than 500% rated for cases with an initial power of ~70% rated thermal power (RTP). Therefore, the HZP cases are typically screened by the NRELAP5 analysis and no VIPRE-01 MCHFR analysis is explicitly performed. However, in the event that a HZP case does not screen out, explicit MCHFR analysis would be performed and additional lower power cases would be run to ensure the true limiting configuration is found (as was done in the FSAR analysis for initial powers between 50% and 100% RTP). The last two sentences in the third paragraph of Section 6.2 of the Rod Ejection Accident Methodology topical report (TR-0716-50350) were deleted as seen in the markup provided with this response for clarification. The information deleted was not salient to the intent of the paragraph. While it is true that the difference in MCHFR is negligible when either peak FΔH or FΔH at peak power is used, the intent of the paragraph was to delineate <u>where</u> the peak FQ and the limiting FΔH at peak power are used in the analysis.

b) The SIMULATE-3K calculation of the event calculates roughly 40 different combinations of initial conditions and corresponding transient responses. The 80% and 100% power BOC cases were seen to produce non-limiting peak power and peak transient FQ and FΔH compared to the lower power BOC cases as seen in Table 6-2. Also, the BOC cases were seen to produce non-limiting peak power and peak transient FQ and FΔH compared to EOC cases. Peak power for the BOC cases ranged from 7% RTP at 0% RTP to 178% RTP at 70% RTP as compared to a range of 75% RTP at 0% RTP to 661% RTP at 55% RTP for EOC conditions. Thus, the BOC 80% and 100% initial RTP cases were manually screened as non-limiting when considering the cases for which NRELAP5 and VIPRE-01 calculations were performed. The results of both the NRELAP5 and VIPRE-01 calculations are analyzed as part of each calculation to ensure the logic of the judgment of the manual screening remains sound.

Impact on Topical Report:

Topical Report TR-0716-50350, Rod Ejection Accident Methodology, has been revised as described in the response above and as shown in the markup provided in this response.

The peak F_Q before reactor trip is used to maximize the adiabatic heatup response for fuel enthalpy and temperature. $F_{\Delta H}$ at the peak reactor power is used in the VIPRE-01 for MCHFR analysis. ~~These two values may not occur at the same time step; however, the peak $F_{\Delta H}$ before the trip and at the peak reactor power are within 0.005 above HZP. Because hot zero power MCHFR calculations are not a part of the REA analysis scope, this difference is negligible and the MCHFR calculations are not impacted.~~

Table 6-2 Sample results for rod ejection accident analysis, beginning of cycle and middle of cycle, both regulating groups

Parameter	BOC, 0% Power	BOC, 50% Power	BOC, 70% Power	BOC, 80% Power	BOC, 100% Power	MOC, 50% Power	MOC, 70% Power
Ejected rod worth ($)	~~{{~~ 0.570	0.629	0.614	0.427	0.119	0.739	0.721
MTC (pcm/°F)	{{						}}2(a),(c),ECI
FTC (pcm/°F)	~~{{~~ -1.38	-1.38	-1.38	-1.38	-1.38	-1.38	-1.38
β_{eff} (-)	{{						
Peak transient F_Q (-)							
Peak transient $F_{\Delta H}$ (-)							}}2(a),(c),ECI
Peak power (% rated)	7	133	178	137	113	186	240
Maximum Δcal/g, hot node	N/A	24.6	28.7	26.0	N/A	24.3	27.5
Maximum cal/g, hot node	N/A	70.5	83.2	84.0	N/A	69.9	81.5
Maximum fuel centerline temperature (°F)	N/A	1813	2141	2162	N/A	1798	2097
NRELAP5 MCHFR (-)	{{						
VIPRE-01 MCHFR (-)							}}2(a),(c),ECI
Predicted rod failures (%)	0	0	0	0	0	0	0

Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 9306
Date of RAI Issue: 04/04/2018

NRC Question No.: 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 FΔH 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.

NuScale Response:

a. The uncertainties in Table 5-1 are updated as described in TR-0616-48793 (Reference 8.2.7) for all except the $F_{\Delta H}$ engineering uncertainty, which is updated consistent with the value in the Subchannel Analysis Methodology topical report (TR-0915-17564, Reference 8.2.11). The Rod Ejection Accident Methodology topical report (TR-0716-50350) was revised as indicated in the markup provided with this response to define the method in which the update is made. The title of Table 5-1 was also revised to

stipulate that the values listed are examples.

b. The methodology presented in the topical is a cycle-specific detailed analysis, generally using best-estimate tools and input conditions. This is opposed to standard steady-state $F_{\Delta H}$ uncertainty in which a set of bounding assumptions through the biasing of this input parameter are utilized to simplify the methodology. The rod ejection event does utilize the $F_{\Delta H}$ engineering uncertainty, which includes variations in pellet diameter, pellet density, enrichment, fuel rod diameter, fuel rod pitch, inlet flow distribution, flow redistribution, and flow mixing. The items in the standard steady-state $F_{\Delta H}$ uncertainty that are not included for rod ejection due to inapplicability are the $F_{\Delta H}$ measurement uncertainty and variations in peaking due to rod insertion (would be redundant with the use of best-estimate power peaking). Thus, it is appropriate for the event specific methodology to utilize the event-specific $F_{\Delta H}$ engineering uncertainty.

Impact on Topical Report:

Topical Report TR-0716-50350, Rod Ejection Accident Methodology, has been revised as described in the response above and as shown in the markup provided in this response.

5.2.2.3.2 Ejected Rod Location

The core is designed with quadrant symmetry, where CRAs 1, 5, 15, and 16 in Figure 5-1 represent all unique CRA positions in the core. Only the CRAs in the regulating bank are eligible for ejection and considered in the REA methodology.

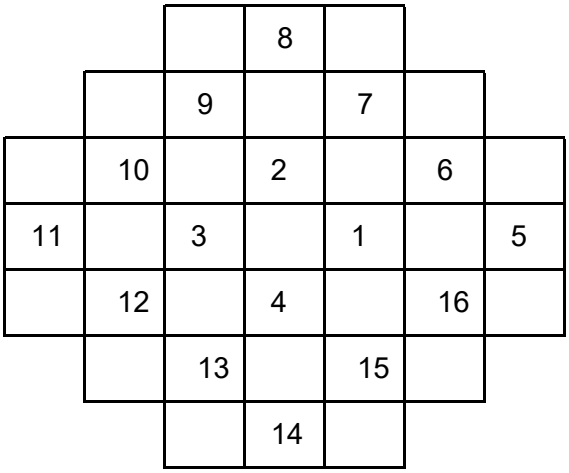

Figure 5-1 Control rod assembly layout for the NuScale Power Module

5.2.2.3.3 Reactor Trips

The high power rate reactor trip signal is produced when the core power increases more than 15 percent from the initial power level within one minute. The high power reactor trip signal is produced when the core power exceeds 120 percent of rated power if the initial condition is above 15 percent power; the setpoint is 25 percent of rated power if the initial power level is below 15 percent.

5.2.2.3.4 Reactivity Feedback

The MTC and DTC are biased to be as least negative as possible. The effective delayed neutron fraction (β_{eff}) is biased to be as small as possible.

For the low CRA worth calculations to determine peak pressure, BOC reactivity feedback parameters is used to minimize the power decrease that occurs after the initial power jump. Specific uncertainties applied are listed in Table 5-1.

For events that increase RCS and fuel temperatures, the least negative MTC and DTC are conservative. For events based on reactivity insertion, a smaller β_{eff} is conservative.

Each time a rod ejection analysis is performed, the example uncertainties defined in Table 5-1 will be verified to ensure they are current and updated, if applicable, consistent with References 8.2.7 and 8.2.11.

Table 5-1 Example ~~U~~uncertainties for rod ejection accident calculations

Parameter	Uncertainty	Analysis
Delayed neutron fraction	6 percent	SIMULATE-3K
Ejected CRA worth	12 percent	SIMULATE-3K
Doppler temperature coefficient	15 percent	SIMULATE-3K
MTC	2.5 pcm/°F	SIMULATE-3K
CRA position	6 steps	SIMULATE-3K
Initial power	2 percent	NRELAP5
F_Q	{{	Adiabatic Heatup
$F_{\Delta H}$	}}2(a),(c)	VIPRE-01

5.2.3 Results and Downstream Applicability

No explicit acceptance criteria are evaluated in the core response calculations. Instead, the boundary conditions are generated to be used by the system response, subchannel, and fuel response analyses. Applicable acceptance criteria are applied to these downstream analyses.

5.3 System Response

The generic non-LOCA methodology is discussed in more detail in the non-LOCA evaluation methodology topical report (Reference 8.2.10); for the system analysis using NRELAP5, REA utilitizes this methodology. However, in order to assess the NuScale criteria outlined in Section 2.3, some deviations or additions to the non-LOCA methodology are implemented. The event-specific analysis is discussed in this section.

5.3.1 Calculation Procedure

For the system response, calculations are performed for the purpose of determining the peak RCS pressure analysis and to provide inputs to the subchannel analysis for CHF determination. Because it is determined that pressurization, and not depressurization, is limiting for CHF, all NRELAP5 system calculations are performed assuming no depressurization effects.

Critical heat flux scoping cases are performed to determine the general trend and to select the cases to be evaluated in the VIPRE-01 subchannel analysis for final confirmation that no MCHFR fuel failures occur.

Competing scenario evaluations exist between the peak pressure and the MCHFR calculations. The two scenarios to consider within the system response are as follows:

- The SIMULATE-3K power response is used to maximize the impact on MCHFR. This tends to be a rapid, peaked power response due to using the maximum possible ejected CRA worth based on insertion to the PDIL.

Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 9306
Date of RAI Issue: 04/04/2018

NRC Question No.: 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.

NuScale Response:

The SIMULATE-3K calculation is not directly relied upon to perform initial or maximum fuel pellet temperature calculations, rather it calculates the power pulse and power peaking for use

in the downstream analysis. The adiabatic calculation, with conservative modeling and assumptions, utilizes the SIMULATE-3K input for the calculation of the maximum fuel pellet temperature. The initial fuel temperature is obtained from a bounding fuel performance calculation utilizing the NRC-approved fuel performance code COPERNIC and a combination of conservative conditions such as exposure and power peaking. The Rod Ejection Accident Methodology topical report (TR-0716-50350) was revised as seen in the markup provided with this response to reflect the appropriate source of the initial fuel temperature for the conservative calculation of the maximum fuel temperature and enthalpy.

As described in Section 4.4 of the Subchannel Analysis Methodology topical report (TR-0915-17564), for each fuel design the VIPRE-01 fuel conduction model is updated based on a fuel performance benchmark in order to ensure the MCHFR calculation conservatively accounts for the entire range of possible time-in-life parameters, including exposure, uranium enrichment, gadolinium enrichment, gap conductance, and fuel density.

Impact on Topical Report:

Topical Report TR-0716-50350, Rod Ejection Accident Methodology, has been revised as described in the response above and as shown in the markup provided in this response.

3.2.4 Fuel Response

The fuel response calculations are performed using a conservative adiabatic heatup model. Initial fuel temperatures are calculated by an NRC-approved fuel performance code. These evaluations are performed outside of a code package and are discussed in Section 5.4.

3.2.5 Accident Radiological Evaluation

This methodology requires that no fuel failure occurs, whether due to fuel melt, transient enthalpy increase, or cladding failure due to MCHFR, and therefore, the pellet/cladding gap shall not be breached. In addition, because the fuel enthalpy increase limit already incorporates the worst cladding differential pressure because of FGR, cladding failure as a result of cladding differential pressure will not occur. Therefore no accident radiological consequences will occur for the REA.

Enclosure 3:

Affidavit of Zackary W. Rad, AF-0618-60286

NuScale Power, LLC

AFFIDAVIT of Zackary W. Rad

I, Zackary W. Rad, state as follows:

1. I am the Director, Regulatory Affairs of NuScale Power, LLC (NuScale), and as such, I have been specifically delegated the function of reviewing the information described in this Affidavit that NuScale seeks to have withheld from public disclosure, and am authorized to apply for its withholding on behalf of NuScale.
2. I am knowledgeable of the criteria and procedures used by NuScale in designating information as a trade secret, privileged, or as confidential commercial or financial information. This request to withhold information from public disclosure is driven by one or more of the following:
   a. The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by NuScale competitors, without a license from NuScale, would constitute a competitive economic disadvantage to NuScale.
   b. The information requested to be withheld consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage, as described more fully in paragraph 3 of this Affidavit.
   c. Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.
   d. The information requested to be withheld reveals cost or price information, production capabilities, budget levels, or commercial strategies of NuScale.
   e. The information requested to be withheld consists of patentable ideas.
3. Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale's competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying Request for Additional Information response reveals distinguishing aspects about the methods by which NuScale develops its rod ejection analysis for the NuScale power module.

   NuScale has performed significant research and evaluation to develop a basis for this methods and has invested significant resources, including the expenditure of a considerable sum of money.

   The precise financial value of the information is difficult to quantify, but it is a key element of the design basis for a NuScale plant and, therefore, has substantial value to NuScale.

   If the information were disclosed to the public, NuScale's competitors would have access to the information without purchasing the right to use it or having been required to undertake a similar expenditure of resources. Such disclosure would constitute a misappropriation of NuScale's intellectual property, and would deprive NuScale of the opportunity to exercise its competitive advantage to seek an adequate return on its investment.

4. The information sought to be withheld is in the enclosed response to NRC Request for Additional Information No. 9306, eRAI 9306. The enclosure contains the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.
5. The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC § 552(b)(4), as well as exemptions applicable to the NRC under 10 CFR §§ 2.390(a)(4) and 9.17(a)(4).
6. Pursuant to the provisions set forth in 10 CFR § 2.390(b)(4), the following is provided for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld:
   a. The information sought to be withheld is owned and has been held in confidence by NuScale.
   b. The information is of a sort customarily held in confidence by NuScale and, to the best of my knowledge and belief, consistently has been held in confidence by NuScale. The procedure for approval of external release of such information typically requires review by the staff manager, project manager, chief technology officer or other equivalent authority, or the manager of the cognizant marketing function (or his delegate), for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside NuScale are limited to regulatory bodies, customers and potential customers and their agents, suppliers, licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or contractual agreements to maintain confidentiality.
   c. The information is being transmitted to and received by the NRC in confidence.
   d. No public disclosure of the information has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or contractual agreements that provide for maintenance of the information in confidence.
   e. Public disclosure of the information is likely to cause substantial harm to the competitive position of NuScale, taking into account the value of the information to NuScale, the amount of effort and money expended by NuScale in developing the information, and the difficulty others would have in acquiring or duplicating the information. The information sought to be withheld is part of NuScale's technology that provides NuScale with a competitive advantage over other firms in the industry. NuScale has invested significant human and financial capital in developing this technology and NuScale believes it would be difficult for others to duplicate the technology without access to the information sought to be withheld.

I declare under penalty of perjury that the foregoing is true and correct. Executed on June 4, 2018.

Zackary W. Rad