LO-0620-70464

June 16, 2020

Docket No. 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 Submittal of the Approved Version of NuScale Topical Report, “Rod Ejection Accident Methodology,” TR-0716-50350, Revision 1

REFERENCES:

1. NRC Letter to NuScale Power, “Final Safety Evaluation for NuScale Power, LLC Topical Report TR-0716-50350, Revision 1, 'Rod Ejection Accident Methodology,'” dated June 3, 2020 (ML20157A223)
2. Letter from NuScale Power to NRC, “NuScale Power, LLC Submittal of Topical Report,’Rod Ejection Accident Methodology,’ TR-0716-50350, Revision 1,” dated November 15, 2019 (ML19319C685)

By referenced letter dated June 3, 2020 the NRC issued a final safety evaluation report documenting the NRC Staff conclusion that the NuScale topical report “Rod Ejection Accident Methodology,” TR-0716-50350, Revision 1, is acceptable for referencing in licensing applications for the NuScale small modular reactor design. The referenced NRC letter requested that NuScale publish the approved version of TR-0716-50350, within thirty days of receipt of the letter.

Accordingly, Enclosure 1 to this letter provides the approved version of TR-0716-50350-P-A, Revision 1. The enclosure includes the June 3, 2020 NRC letter and its final safety evaluation report.

Enclosure 1 contains proprietary information. NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 1 has also been determined to contain Export Controlled Information. This information must be protected from disclosure per the requirements of 10 CFR § 810. Enclosure 2 contains the nonproprietary version of the approved topical report package.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact John Fields at 541-452-7425 or at JFields@nuscalepower.com.

Sincerely,

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

Distribution: Gregory Cranston, NRC
Prosanta Chowdhury, NRC
Michael Dudek, NRC
Rani Franovich, NRC

Enclosure 1: "Rod Ejection Accident Methodology," TR-0716-50350-P-A, Revision 1, proprietary version
Enclosure 2: "Rod Ejection Accident Methodology," TR-0716-50350-NP-A, Revision 1, nonproprietary version
Enclosure 3: Affidavit of Zackary W. Rad, AF-0620-70465

Enclosure 1:

“Rod Ejection Accident Methodology,” TR-0716-50350-P-A, Revision 1, proprietary version

Enclosure 2:

“Rod Ejection Accident Methodology,” TR-0716-50350-NP-A, Revision 1, nonproprietary version

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Section A

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 3, 2020

Mr. Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC.
1100 Circle Boulevard, Suite 200
Corvallis, OR 97330

SUBJECT: FINAL SAFETY EVALUATION FOR NUSCALE POWER, LLC TOPICAL REPORT TR-0716-50350, REVISION 1, "ROD EJECTION ACCIDENT METHODOLOGY"

Dear Mr. Rad:

By letter dated December 30, 2016, NuScale Power, LLC (NuScale), submitted Topical Report (TR) TR-0716-50350, "Rod Ejection Accident Methodology," Revision 0, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16365A242) to the U.S. Nuclear Regulatory Commission (NRC) for review and approval.

By letter dated November 15, 2019, NuScale submitted, TR-0716-50350-P, Revision 1, (ADAMS Accession No. ML19319C685). The NRC staff has evaluated TR-0716-50350, Revision 1, and found that it is acceptable for referencing licensing applications for the NuScale small modular reactor design to the extent specified and under the conditions and limitations delineated in the enclosed safety evaluation report (SER).

The NRC staff requests that NuScale publish the applicable version(s) of the SER listed above within 30 days of receipt of this letter. The accepted version of the TR shall incorporate this letter and the enclosed SER and add "-A" (designated accepted) following the report identification number.

CONTACT: Bruce M. Bavol, NRR/DNRL
301-415-6715

If the NRC staff's criteria or regulations change, and its acceptability conclusion in the SER is invalidated, NuScale and/or the applicant referencing the SER will be expected to revise and resubmit its respective documentation; or submit justification for continued applicability of the SER without revision of the respective documentation.

After receiving the package with the "-A" version, the SER will be made available for public inspection through the publicly available records component of NRC's ADAMS.

If you have any questions or comments concerning this matter, please contact Bruce Bavol at 301-415-6715 or via e-mail address at Bruce.Bavol@nrc.gov.

Sincerely,

/RA/

Anna H. Bradford, Director
Division of New and Renewed Licenses
Office of Nuclear Reactor Regulation
Bob Caldwell for

Docket No. 52-048

Enclosures:
1. TR-0716-50350 SER (Public)
2. TR-0716-50350 SER (Proprietary)

cc: DC NuScale Power, LLC Listserv (w/o Enclosure 2)

SUBJECT: FINAL SAFETY EVALUATION FOR NUSCALE POWER, LLC TOPICAL REPORT TR-0716-50350, REVISION 1, "ROD EJECTION ACCIDENT METHODOLOGY"

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ADAMS Accession Nos.:
Pkg: ML20157A220
Letter: ML20157A223
Enclosure No. 1: ML20157A221 PUBLIC
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OFFICE	DNRL/NRLB: PM	DNRL/NRLB: LA	DNRL/NRLB: BC	DNRL: D
NAME	BBavol	CSmith*	MDudek*	RCaldwell for ABradford*
DATE	06/02/2020	06/02/2020	06/02/2020	06/03/2020

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TOPICAL REPORT TR-0716-50350-P, REVISION 1

"ROD EJECTION METHODOLOGY"

1 INTRODUCTION

1.1 Summary

By letter dated November 15, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19319C684), NuScale Power, LLC (NuScale), submitted, for U.S. Nuclear Regulatory Commission (NRC) staff review and approval, Topical Report (TR) TR-0716-50350-P, Revision 1, "Rod Ejection Accident Methodology" (Reference 1). A public version of this TR can be found at ADAMS Accession No. ML19319C685. This safety evaluation report is based on the submitted licensing TR and formal requests for additional information (RAIs).

In TR-0716-50350-P, Revision 1, the applicant described a method of analyzing the consequences of a control rod ejection accident (REA) for the NuScale reactor design. The methodology is based on a three-dimensional (3-D) nodal kinetics solution with both thermal-hydraulic and fuel temperature feedback.

The NRC carried out this review in conformance with the regulatory guidance as summarized in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP), Section 15.4.8, "Spectrum of Rod Ejection Accidents (PWR)" (Reference 2), and SRP Section 4.2, "Fuel System Design," Appendix B, "Interim Acceptance Criteria and Guidance for the Reactivity Initiated Accidents" (Reference 3).

The REA is analyzed using SIMULATE5 (Reference 4), SIMULATE-3K (Reference 5), NRELAP5 (Reference 6), VIPRE-01 (Reference 7), and an adiabatic heatup fuel response hand calculation. Rod failure is assumed if there is an addition of at least 100 calories per gram (cal/g) (180 British thermal units per pound mass (BTU/lb_m)) from zero power, if the local critical heat flux (CHF) thermal design limit is exceeded, or if the pellet clad mechanical interaction (PCMI) threshold listed in Figure B-1, "PWR PCMI Fuel Cladding Failure Criteria," of SRP Section 4.2, Appendix B, is exceeded.

1.2 Description of a Generic Rod Ejection Accident Transient Event

REAs are a class of accident transients that pressurized-water reactor (PWR) vendors are required to analyze to demonstrate compliance with the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," GDC 28, "Reactivity Limits" (as described in SRP Section 15.4.8), to obtain an NRC license for a particular reactor design. The staff based the following discussion on descriptions presented in the TR.

The postulated REA accident is initiated by the sudden ejection of a control rod assembly (CRA) from the core of a critical reactor. Initially, the reactor can be at hot full power (HFP) to hot zero power (HZP). In addition, the core could be at the beginning of cycle or the end of cycle. Thus, a total of at least four different combinations exist to be analyzed.

Enclosure 1

Partial power situations might be considered in particular cases to explore bounding conditions. In general, a large number of initial conditions can affect the transient response and its ultimate termination.

In a typical REA, a CRA is rapidly ejected and accelerated by the system pressure, resulting in a step change in reactivity. The sudden addition of reactivity results in a corresponding increase in power and fuel temperature. The only feedback mechanism that can counter this power increase is the Doppler Effect (Doppler) associated with the fertile component of the fuel (uranium-238). The increase in power causes the fuel temperature to increase, and the Doppler feedback becomes progressively more negative until it reverses the power increase, resulting in a typical power pulse. Finally, the ex-core power detectors trip the scram system and the transient is terminated. The typical duration of the transient is approximately 4 seconds, which is short enough to ignore all system-related changes to the coolant temperature and pressure.

However, if the time frame is long enough that system-level thermal-hydraulic changes are significant, a second transient type results. This can occur at HFP when the CRAs are mostly withdrawn. The power increase is comparatively small, causing a small amount of negative Doppler feedback and, thus, a small pulse followed by a slow increase in reactor power. In addition, the primary system boundary may be compromised because of the ejected rod creating a small-break loss of coolant. In this case, a system-level response is necessary, since the activation of reactor trips associated with system response will terminate the transient.

The two most important pieces of information resulting from an REA analysis involve (1) the number of fuel rods that failed as a result of the transient and (2) whether any of the regulatory requirements have been exceeded (see Section 2 of this report).

Two rod failure mechanisms are important in REA transients:

(1) those that occur during the initial power pulse, caused by PCMI

(2) those caused by fuel clad failure from a departure from nucleate boiling condition during the reactivity excursion

As long as the regulatory requirements are met, the transient analysis is considered complete. However, if the regulatory requirements are not met, the core design or reactor system needs to be reconfigured to ensure compliance. The applicable regulatory requirements are discussed below.

2 REGULATORY CRITERIA

2.1. Requirements

The applicant submitted TR-0716-50350 to support the rod ejection analysis summarized within the NuScale final safety analysis report (FSAR). As such, the staff used the regulatory requirements and guidance outlined by SRP Section 15.4.8 and SRP Section 4.2, Appendix B, in its review of this TR. These requirements concern cladding failure, coolability, and radiological release. The following summarizes the applicable criteria:

- GDC 28 assures that the effects of postulated reactivity accidents can neither damage the reactor coolant pressure boundary nor result in a disturbance sufficient to impair the core cooling capability.

2.2. Relevant Guidance

SRP Section 4.2, Appendix B, provides the interim acceptance criteria and guidance for reactivity-initiated accidents, of which the REA is a subset. By following the provided guidance, described as follows, an applicant can demonstrate compliance with GDC 28:

(1) Cladding Failure: The PCMI caused by the sudden rise in power during the pulse phase of an REA requires a limit on energy (cal/g) as a function of clad thickness (clad thickness change as a result of oxidation). The oxide thickness increases with burnup. Figure B-1 of SRP Section 4.2 provides guidance on a fuel enthalpy rise limit as a function of oxide/wall thickness. Additionally, clad failures can occur if the pin internal pressure is below system pressure when the total enthalpy exceeds 170 cal/g (306 BTU/lb), and if the pin internal pressure is above system pressure when the total energy exceeds 150 cal/g (270 BTU/lb). Both limits apply for core power levels below 5 percent. Finally, violating the thermal design limits for all power levels above 5 percent is assumed to lead to clad failure.

(2) Coolability: Pin cooling is assumed failed for all pins with a total enthalpy of 230 cal/g (414 BTU/lb). In addition, pin cooling is assumed to fail if there is incipient fuel melting. Furthermore, cooling failure will occur in all cases if there is a failure to preserve the reactor pressure boundary, reactor internals, and fuel assembly structural integrity. Finally, a loss of coolable geometry will result following clad and fuel fragmentation and clad ballooning.

(3) Radiological Impact: SRP Section 4.2, Appendix B, provides guidance related to the calculation of fission product inventory that would be available after an event. This inventory is to include both the steady-state gap inventory and fission gas released during the event[1]. SRP Section 4.2, Appendix B, provides a correlation between gas release and maximum fuel enthalpy increase that can be used to calculate the transient fission gas release.

The above guidance summarizes the limits that can be used to demonstrate compliance with GDC 28, which must be met in carrying out the analyses outlined in Section 3.2 of this report. These limits are used at the decision points for fuel temperature and cal/g determinations, as well as for the number of failed rods that imply unacceptable radiological release.

3 SUMMARY OF TECHNICAL INFORMATION

This chapter summarizes the applicant's methodology and briefly describes the codes used by the applicant, including their input, output, and analytic modeling.

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[1] Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors," (Reference 8) and Regulatory Guide 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," (Reference 9) provide further guidance.

3.1 Outline of Rod Ejection Accident PHYSICAL Phenomena, Modeling, and Overall Methodology

This section outlines the various physical phenomena that govern the progression of an REA transient. The software and methodology used in this analysis must be able to accurately (or conservatively) model these phenomena. The initial response of the core to an REA is generally a skewed increase in the power, which severely impacts the fuel temperature and cooling of the core in selected assemblies. In all these analyses, temperature-dependent cross-section data and temperature- and pressure-dependent thermo-physical properties are necessary to model the event accurately.

NuScale uses a 3-D space-time kinetic calculation to provide the nuclear analysis portion of the REA transient response. The calculated power versus time, F_Q versus time; radial power distribution; and axial power distribution information is passed to downstream calculations.

The applicant's fluid dynamics and heat transfer calculations cover the most highly challenged fuel assemblies, recognizing every fuel rod and allowing for both axial and transverse flow. This calculation takes input from the 3-D kinetics calculation and the variable thermo-physical data. The primary output from this analysis is the number of failed rods (if applicable), either from PCMI or from a violation of the SAFDLs.

Finally, those transients that result in a power spike of a lower magnitude but longer duration require a system-level code to determine the coolant temperature and pressure and to identify any phase change in cases where the pressure is dropping. This code takes input from the 3-D kinetics code SIMULATE-3K and the variable thermo-physical data. The system-level code uses a point kinetic model to simulate this event.

Rod failure caused by PCMI is determined by a threshold value of enthalpy deposited per gram of fuel material (cal/g) in conformance with SRP Section 4.2, Appendix B. All rods that exceed this limit are assumed failed and occur during the initial power pulse phase of the transient. In the longer term, additional rod failure may occur from a violation of the SAFDL (the minimum critical heat flux ratio (MCHFR) is greater than the SAFDL). According to the methodology, no fuel rod failures are considered acceptable. Therefore, no radiological consequences are to occur as the result of an REA.

Section 3.2 of TR-0716-50350 describes the computer codes used in the NuScale methodology and the evaluation flowpath. The starting point is the steady-state neutronics calculations performed with the FSAR Chapter 15 non-loss of coolant accident (non-LOCA) methods using CASMO5/SIMULATE5. CASMO5 is used to generate a cross-section data library for use by the 3-D transient nodal code, SIMULATE-3K, in another step. SIMULATE5 initializes the cycle-specific model and reactor conditions that are used as input into the SIMULATE-3K evaluation.

Then, SIMULATE-3K solves the transient 3-D, two-group neutron diffusion equations, starting with SIMULATE5 restart files. SIMULATE-3K analyzes the transient neutronic behavior of the REA at various times in the reactor life, power levels, control rod positions, and initial core conditions and provides the total core power, reactivity insertion, 3-D power distributions, and power peaking results.

The NRELAP5 code then calculates the system response based on input from SIMULATE-3K. Results from the RELAP5 dynamic system response analysis determine whether the reactor coolant system (RCS) pressure limit is exceeded. Additionally, the system thermal-hydraulic response results are passed to VIPRE-01.

VIPRE-01 is a subchannel analysis code, which calculates the CHF ratio and determines whether the acceptance criterion is met. VIPRE-01 uses radial and axial power distribution input from SIMULATE-3K and thermal-hydraulic response input from NRELAP5.

In addition to the codes mentioned above, the NuScale REA analysis methodology also includes an adiabatic heatup model to analyze the fuel response. Section 5.4 of TR-0716-50350 discusses this analysis.

In Section 4 of TR-0716-50350, NuScale presented an overview of the phenomena important for the REA, which is used to develop conservative assumptions for the analysis.

In Section 5 of TR-0716-50350, NuScale presented the REA methodology, based on the codes and methods from Section 3 and information developed in Section 4, which identifies the phenomena important for this accident. The methodology states that the REA analysis is to be performed for each core reload to ensure that any difference in power response is captured. The analysis methodology covers the beginning of cycle, the end of cycle, and the point of maximum energy rise hot channel factor ($F_{\Delta H}$), as well as a range of power levels from HZP to HFP. The analysis presents input assumptions for each stage, and Table 5-1 of TR-0716-50350 includes the uncertainties used in the REA analysis calculations.

NuScale presented a sample REA analysis in Section 6.0 of TR-0716-50350.

4 TECHNICAL EVALUATION

4.1 Software Applicability

Section 3.2 of TR-0716-50350-P presents the computer codes used in the NuScale REA methodology. It states that the CASMO5/SIMULATE5 code package is used for reactor core physics parameters, NRELAP5 is used for the transient system response, and VIPRE-01 is used for the subchannel analysis. TR-0616-48793-P-A, Revision 1, “Nuclear Analysis Codes and Methods Qualification,” dated December 14, 2018 (ADAMS Accession No. ML18348B035) (Reference 10), covers the applicability of these codes and methods to NuScale. A public version of this TR can be found at ADAMS Accession No. ML18348B036. The staff’s evaluation, as documented in the associated safety evaluation report, covers the applicability of these codes to the NuScale plant design.

Section 3.2 of TR-0716-50350-P also states that SIMULATE-3K is used to calculate the dynamic core response. Because TR-0616-48793-P-A, Revision 1 does not cover SIMULATE-3K, Section 3.2.1.3 includes the code description and Section 3.2.1.4 provides the validation. In Section 3.2.1.4, NuScale used data from the SPERT-III tests and a Nuclear Energy Agency Committee on Reactor Physics (NEACRP) control rod ejection benchmark problem to validate SIMULATE-3K for use in analyzing an REA for the NuScale plant design. In RAI 9306, Question 15.04.08-1 (Reference 11) NuScale provided support for the validation presented Section 3.2.1.4.

NuScale benchmarked SIMULATE-3K against a selection of SPERT-III cold startup tests for each statepoint, generally corresponding to the highest static worth for the statepoint. NuScale compared the SPERT-III conditions with the NuScale operating parameters and demonstrated that the SPERT-III test conditions were generally representative of the NuScale core design from a reactivity-initiated accident perspective. The staff reviewed the presented information and determined that NuScale's results demonstrate generally good agreement between the results predicted by SIMULATE-3K and the SPERT-III experimental results.

Additionally, NuScale provided a verification analysis of the NEACRP REA benchmark problem by Studsvik Scandpower with SIMULATE-3K (Reference 11). This analysis was performed under NuScale's approved 10 CFR Part 50, Appendix B, quality assurance program. The results presented in the RAI response demonstrate good agreement between NuScale's SIMULATE-3K results and the NEACRP benchmark reference solutions. Based on NuScale's analysis results, the staff finds that NuScale demonstrated that SIMULATE-3K can successfully model the NEACRP benchmarks for reactivity-initiated accidents.

The staff reviewed the NuScale validation of SIMULATE-3K against the SPERT-III experiments and the NEACRP benchmark suite, as discussed above, and concludes that NuScale demonstrated that SIMULATE-3K can be used in its methodology to accurately model a reactivity-initiated accident.

4.2 Methodology

Figure 3-1, "Calculation schematic for analyzing rod ejection accident," of TR-0716-50350-P is a flow diagram of the NuScale REA analysis methodology that describes the codes used for each part of the analysis. The staff evaluates the methodology below.

4.2.1 Steady-State Initialization

Section 3.2.1.2 of TR-0716-50350-P describes how SIMULATE5 initializes the cycle-specific model and reactor conditions, which SIMULATE-3K then uses to simulate the REA.
Section 5.2.1.1 describes the static calculations methodology. The static analysis consists of an assessment of the worst rod stuck out and the development of the restart file for initial conditions for SIMULATE-3K.

The approved referenced report, TR-0616-48793-P-A, Revision 1, describes the use of SIMULATE5 for non-LOCA analyses.

In Section 5.2.1.1 of the TR, NuScale states that the coolant mass flux is one of the initial conditions that it passes to SIMULATE-3K and VIPRE-01. In RAI 9306, Question 15.04.08-12, the staff asked NuScale to describe how it derives the coolant mass flux and how it varies with core power. In response, NuScale stated that the core flow, and thus the coolant mass flux, for a given initial power is held constant through a modeling option. NuScale determined the initial core flow as a function of initial core power based on the natural circulation flow curve. The staff finds that by setting the core flow as described in the RAI response, the mass flux is minimized, and the coolant temperature is maximized. This supports conservative downstream analyses such as MCHFR and maximum fuel centerline temperature. The staff finds that the RAI response is consistent with the non-LOCA accident methodology topical report, TR-0516-49416, Revision 3.

The staff finds that the method for developing steady-state conditions is consistent with the non-LOCA accident methodology as presented in TR-0516-49416 Revision 3, using the nuclear analysis codes and methods in TR-0616-48793-P-A and is therefore acceptable.

4.2.2 Dynamic Core Response

Section 5.2.1.2 of TR-0716-50350-P describes the transient system calculations performed with SIMULATE-3K for the NuScale REA analysis methodology. The methodology first determines conservative parameter uncertainties and then simulates the transient based on conservatively applying the uncertainties. The staff reviewed the spectrum of input values used in the dynamic core response analysis, the initial conditions considered, the ability to capture the most limiting case, and the analytical methods.

The staff also reviewed the conservatisms applied to the SIMULATE-3K calculation according to the methodology to ensure that the results would not underpredict fuel failures. In RAI 9306, Question 15.04.08-4, the staff requested an additional description of the way the methodology ensures that the parameters input to SIMULATE-3K are conservative. In response, NuScale provided additional information (ADAMS Accession No. ML18155A627) describing the methodology for conservatively modeling the dynamic core response. A public version of this response can be found at ADAMS Accession No. ML18155A628. The input core geometry and material compositions, core operating conditions, and core configuration come from a SIMULATE5 restart file according to the methodology described in TR-0616-48793-P-A. The VIPRE-01 thermal-hydraulic conditions are based on conservative NRELAP5 runs and include VIPRE-01-specific conservatisms consistent with the methodology presented in TR-0915-17564-P-A, "Subchannel Analysis Methodology," Revision 2, issued March 2019 (Reference 12). Additionally, the NuScale methodology includes removing the point kinetics while performing MCHFR analyses but continues to use them for the overpressure analyses. The staff reviewed this response and determined that NuScale's methodology ensures SIMULATE-3K conservatively calculates potential fuel failures by choosing conservative input values and following the approved methodology described in TR-0616-48793-P-A. This supports the statements provided in the topical report. The staff finds that NuScale has conservatively chosen input values to ensure that the consequences of a reactivity-initiated accident are not underpredicted and is therefore acceptable.

TR-0716-50350-P describes the process for performing the transient calculations once the uncertainties have been applied to the nuclear parameters. Each regulating group is set at the power-dependent insertion limit (PDIL) unless an unejected regulating CRA is identified as the worst rod stuck out. Additionally, in Section 5.1.3 of the TR, NuScale stated that a range of power levels are investigated (HZP to HFP) to ensure that the PDIL, axial offset limits, and moderator temperature are bounded. However, the staff could not make a finding based on the level of detail provided in the TR. In RAI 9306, Question 15.04.08-11, the staff asked NuScale to describe how the axial power shape is determined to bound the axial offset limits specified for all power levels. In its response (ADAMS Accession No. ML18155A627), NuScale described how moderator temperature, axial offset, and CRA insertion limits are conservatively chosen. A public version of this response can be found at ADAMS Accession No. ML18155A628. The staff reviewed the supporting information provided in the RAI response and confirmed that NuScale conservatively created top-peaked axial power shapes to conservatively maximize the ejected rod worth. Therefore, the staff finds that the additional information supports the statements in the TR on conservative parameters and is therefore acceptable.

Section 4.3(B) of TR-0716-50350-P states that the limiting rod worth for the REA occurs when the rods are at the PDIL and that is used as the starting point for the calculations. The staff notes that plant operation is allowed when the rods are at or above the PDIL. It was unclear to the staff whether a reactor trip when the rods are above the PDIL would result in higher deposited energy over a long-term transient. In RAI 9306, Question 15.04.08-8, the staff asked NuScale to demonstrate that the methodology bounds other allowed rod configurations (e.g., other than at PDIL) for scenarios in which a reactor trip is delayed or not reached. In response to the RAI (ADAMS Accession No. ML18155A627), NuScale stated that the case in which a rod ejection does not result in a reactor trip is bounded by a single rod event, as analyzed in NuScale FSAR Section 15.4.3. A public version of this response can be found at ADAMS Accession No. ML18155A628. The staff confirmed that the single rod event analysis bounds the scenario of a rod ejection from a rod insertion other than at PDIL. Therefore, the staff's review of TR-0716-50350-P does not cover this bounded scenario.

As discussed in Section 5.1.4 of TR-0716-50350-P, the conservative single active failure for an REA is a failure of the flux detector in the high-flux region. However, the staff was unable to ascertain how the ex-core detectors were implemented in the SIMULATE-3K analysis from the information provided in the TR. In RAI 9306, Question 15.04.08-5, the staff asked NuScale to describe how the ex-core detectors are implemented in the SIMULATE-3K analysis. In response (ADAMS Accession No. ML19031C977), NuScale provided additional information that demonstrated that the limiting cases are those that experience prompt (or near prompt) criticality as a result of the reactivity insertion. A public version of this response can be found at ADAMS Accession No. ML19031C978. The staff reviewed the cases screened by NRELAP5 and confirmed the information provided by NuScale and found that, for all cases, peak power and MCHFR occurred before control rods began to move. Therefore, the staff agrees that the ex-core detectors are not necessary to mitigate a reactivity-initiated accident.

4.2.3 Dynamic System Response

Section 5.3 of TR-0716-50350-P presents the system response for the REA analysis. These system response calculations determine the peak RCS pressure and provide thermal-hydraulic response inputs to the subchannel analysis for CHF determination. The NuScale methodology follows the non-LOCA evaluation methodology (Reference 13) but with modifications to ensure conservative results when modeling reactivity-initiated accidents.

4.2.3.1 Peak Pressure Calculations

The calculation procedure in Sections 5.3.1 and 5.3.1.2 of TR-0716-50350-P details the methods used to calculate the peak pressure resulting from an REA. To conservatively perform the peak pressure analysis, the methodology uses an ejected CRA worth, which results in a power increase just below the high power and high-power rate trip setpoints within NRELAP5. This maximizes the length of the transient, which is then terminated by high RCS pressure. These cases do not require an upstream SIMULATE-3K calculation. The staff reviewed the methodology and input assumptions in Section 5.3.1.2 of the TR and finds that the methodology as described would conservatively calculate the maximum RCS pressure and is therefore acceptable.

4.2.3.2 Minimum Critical Heat Flux Ratio

The calculation procedure detailed in Section 5.3.1 of TR-0716-50350-P states that NRELAP5 scoping cases determine the general trend for selecting the cases to be evaluated in the VIPRE-01 subchannel analysis for final confirmation that no MCHFR fuel failures occur. The MCHFR analyses use a SIMULATE-3K power response, which maximizes CRA worth by assuming insertion to the PDIL. The staff finds that the use of NRELAP5 in the method presented to determine the power level at which MCHFR occurs is within the code's capabilities and is therefore acceptable.

Section 5.3.1.1 of the TR provides the conservatisms included in the methodology for the MCHFR analyses. The staff agrees that the system condition assumptions used in the MCHFR analysis methodology are conservative, but, in RAI 9306, Question 12, the staff requested a description of how the coolant mass flux is determined and whether it varies with power. NuScale responded (Reference 11) that initial core flow is determined as a function of initial power based on the natural circulation flow curve and that core flow in the NRELAP5 analysis is allowed to increase but the increase is minimized. The staff reviewed the response and agrees that the methodology described would conservatively model coolant mass flux by not increasing coolant mass with increasing power and is therefore acceptable. The staff finds that the method for determining MCHFR is consistent with the methodology outlined in Reference 13 and is therefore acceptable.

4.2.4 Subchannel Critical Heat Flux Evaluation

Section 5.4 of TR-0716-50350-P presents the subchannel response methodology, which calculates the MCHFR and compares it against the MCHFR acceptance criteria to verify that CHF is not reached during the event for any rods.

As detailed in Section 5.4.1.1 of the TR, NuScale deviated from the referenced subchannel methodology described in Reference 12. Broadly speaking, the deviations are related to [ ].

The [ ] deviation was necessary to [ ]. In support, NuScale provided VIPRE-01 axial nodalization sensitivity results in Figure 6-6, "Effect of Axial Node Size (inches) on Critical Heat Flux," of the TR. Through this analysis, NuScale demonstrated that the MCHFR is relatively insensitive to axial node size in the range of interest and that, therefore, the deviation from the referenced subchannel methodology is acceptable. In Figure 6-7, "Effect of VIPRE-01 Two-phase Flow Model Options on Critical Heat Flux," of the TR, NuScale compared the profile-fit two-phase flow correlation to the nonprofile-fit subcooled void model. The results demonstrate that [ ] value.

The staff reviewed the description of the VIPRE-01 methodology deviations and the supporting sensitivity analyses as presented in the TR. The sensitivity analysis and conclusions agree with the similar sensitivity analysis presented in the approved subchannel methodology TR (Reference 12). The staff finds that the revised methodology continues to be consistent with the guidance provided in SRP Section 15.4.8(III)(2)(A) and is therefore acceptable.

4.2.5 Adiabatic Heatup Fuel Response

Section 5.5 of TR-0716-50350-P presents the adiabatic heatup fuel response methodology that calculates fuel temperature and radial average fuel enthalpy. The methodology uses two acceptance criteria: (1) the fuel is not allowed to melt, and (2) the peak fuel enthalpy and enthalpy rise must remain below the limits provided in SRP Section 4.2, Appendix B.

The staff reviewed the fuel temperature calculation as presented in Equation 5-2 of Section 5.5 of the TR. The staff confirmed that the equation resulted in a conservative bounding final fuel temperature by reviewing the input assumptions and calculation method. This included the assumption of no conduction from the pellet, the use of centerline temperature for a starting point, and the nodal peaking factors (plus uncertainty) as calculated by SIMULATE-3K.

Based on the staff's review of the adiabatic heatup fuel response calculation method and inputs, the staff finds the methodology outlined in Section 5.5 of TR-0716-50350-P to be acceptable.

5 CONCLUSIONS

The staff concludes that the analysis of the REAs is acceptable and meets GDC 28 requirements. This conclusion is based on the findings below.

The applicant met GDC 28 requirements for prevention of postulated reactivity accidents that could result in damage to the reactor coolant pressure boundary greater than limited local yielding or result in sufficient damage to impair the core cooling capability significantly. It met the requirements by demonstrating compliance with the regulatory guidance of SRP Section 15.4.8, "Spectrum of Rod Ejection Accidents (PWR)" (Reference 2). The staff has evaluated the applicant's analysis of the assumed control REA and finds the assumptions, calculation techniques, and consequences acceptable. As the calculations demonstrate peak fuel temperatures below melting conditions, prompt fuel rupture with consequent rapid heat transfer to the coolant from finely dispersed molten uranium dioxide presumably did not occur. The pressure surge results in a pressure increase below "Service Limit C" (as defined in Section III, "Nuclear Power Plant Components," of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code) (Reference 14) for the maximum control rod worths assumed. The staff believes that the calculations are sufficiently conservative, both in initial assumptions and analytical models, to maintain primary system integrity.

6 LIMITATIONS AND CONDITIONS

The staff's approval is limited to the application of this methodology to the NuScale reactor design.

7 REFERENCES

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12. NuScale Power, LLC, TR-0915-17564-P-A, Revision 2, “Subchannel Analysis Methodology,” March 2019, ADAMS Accession Nos. ML19067A255 (package) and ML19067A256 (public version).

13. NuScale Power, LLC, TR 0516 49416, “Non-Loss-of-Coolant Accident Analysis Methodology,” Revision 3, May 2020, ADAMS Accession Number ML20148M391.

14. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, “Rules for Construction of Nuclear Facility Components.”

Section B

Licensing Topical Report

Rod Ejection Accident Methodology

June 2020

Revision 1

Docket: PROJ0769

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Abstract

This report documents the NuScale Power, LLC, (NuScale) methodology for the evaluation of a control rod ejection accident (REA) in the NuScale Power Module. This methodology is used to demonstrate compliance with the requirements of 10 CFR 50, Appendix A, General Design Criterion (GDC) 13 and GDC 28, and the acceptance criteria and guidance in Regulatory Guide 1.77 as noted by NUREG-0800, Standard Review Plan, Sections 4.2 and 15.4.8. In addition, consideration is given to the acceptance criteria proposed in the March 16, 2015, memorandum "Technical and Regulatory Basis for the Reactivity-Initiated Accident Acceptance Criteria and Guidance, Revision 1," which contains the anticipated Nuclear Regulatory Commission criteria for the REA.

The methodology described herein uses a variety of codes and methods. The three-dimensional neutronic behavior is analyzed using SIMULATE5 and SIMULATE-3K; the reactor system response is analyzed using NRELAP5; and the subchannel thermal-hydraulic behavior and fuel response is analyzed using VIPRE-01. The software is validated for use to evaluate the REA. The fuel response is supplemented by the use of a bounding adiabatic heat-up calculation for the calculation of all transient fuel enthalpy and temperature increases during the REA.

This report includes the identification of important phenomena and input and specifies appropriate uncertainty treatment of the important input for a conservative evaluation. The methodology is discussed and demonstrated by the execution of sample problems and appropriate sensitivity analyses.

NuScale intends to use this methodology for REA analysis in support of the NuScale Design Certification Application and for future design work. This report is not intended to provide final design values or results; rather, example values for the various evaluations are provided for illustrative purposes in order to aid the reader's understanding of the context of the application of the methodology.

NuScale is requesting Nuclear Regulatory Commission review and approval to use the methodology described in this report for design-basis REA analyses in the NuScale Power Module.

Executive Summary

The purpose of this report is to describe the methodology that NuScale Power, LLC, intends to use for the analysis of rod ejection accidents (REAs) for the NuScale design certification application. NuScale is requesting Nuclear Regulatory Commission review and approval to use the methodology described in this report for analyses of design-basis REA events in the NuScale Power Module (NPM).

NUREG-0800, Standard Review Plan (SRP), Section 15.0 (Reference 8.2.4) categorizes the REA as a postulated accident due to frequency of occurrence and types it as a "Reactivity and Power Distribution Anomaly." The purpose of this report is to define and justify the methodology for analyzing the REA for the NPM design for the purpose of demonstrating that fuel failure does not occur. This is accomplished by conservatively applying regulatory acceptance criteria to bounding analyses. Specific regulatory acceptance criteria that are conservatively treated in this methodology include the following:

- hot zero power fuel cladding failure applies the worst case allowed peak rod differential pressure to the allowed radial average fuel enthalpy limit.
- pellet-cladding mechanical interaction (PCMI) failure threshold for cladding oxidation applies a bounding value of corrosion/wall thickness to assess fuel enthalpy rise limit.
- core coolability limit for fuel melt does not allow any fuel melt to occur.
- no fuel cladding failure due to minimum critical heat flux criteria (MCHFR) is allowed.

An REA is an assumed rupture of the control rod drive mechanism (CRDM) or of the CRDM nozzle. Upon this rupture, the pressure in the reactor coolant system (RCS) provides an upward force that rapidly ejects the control rod assembly (CRA) from the core. The ejection of the CRA results in a large positive reactivity addition, leading to a skewed and severely peaked core power distribution. As the power rapidly rises, fission energy accumulates in the fuel rods faster than it can be deposited into the coolant, raising the fuel temperature. The power rise is mitigated by fuel temperature feedback and delayed neutron effects.

The regulatory requirements for the REA are General Design Criterion (GDC) 13 and GDC 28 from 10 CFR 50, Appendix A (Reference 8.2.1.). In order to satisfy GDC 13 and GDC 28, this methodology utilizes the guidance provided in Regulatory Guide 1.77 (Reference 8.2.2), and SRP Sections 15.4.8 and 4.2, as amended in a Nuclear Regulatory Commission letter dated March 16, 2015 (hereafter called the "Clifford Letter," Reference 8.2.5). This guidance addresses: 1) maximum RCS pressure, 2) fuel cladding failure, 3) core coolability, and 4) fission product inventory. In general, the NuScale REA methodology has adopted the limiting criteria of the Clifford Letter with the exception of the Cladding H_2 uptake criteria for fuel cladding failure. This methodology instead utilizes the oxidation criteria from SRP Section 4.2, Appendix B, which is currently approved by the NRC.

This report describes the software codes used to evaluate the REA along with appropriate validation for its use in NuScale applications. The codes used for REA analysis are the following:

- CASMO5 – transport theory code that generates pin cell or assembly lattice physics parameters.
- SIMULATE5 – three-dimensional, steady-state, nodal diffusion theory reactor simulator code that calculates steady-state predictions (critical boron concentration, boron worth, reactivity coefficients, CRA worth, shutdown margin, power distributions, and peaking factors).
- SIMULATE-3K– three-dimensional nodal reactor kinetics code that couples core neutronics with detailed thermal-hydraulic models to supply power input to NRELAP5 and VIPRE-01.
- NRELAP5 – System thermal-hydraulic code produced by NuScale to produce boundary conditions to apply to the fuel sub-channel code.
- VIPRE-01 – Fuel thermal-hydraulic subchannel code predicts three-dimensional velocity, pressure, and thermal energy fields and radial fuel rod temperature profiles in reactor cores.

This report presents the findings documented in NUREG/CR-6742 (Reference 8.2.26), "Phenomena Identification and Ranking Table (PIRT) for Rod Ejection Accidents in Pressurized Water Reactors Containing High Burnup Fuel," identifying important phenomena. Associated with these phenomena, the Electric Power Research Institute (EPRI) topical report (Reference 8.2.14) for three-dimensional REA analysis identified the key parameters as the following:

- ejected CRA worth
- effective delayed neutron fraction
- moderator reactivity coefficient
- Doppler coefficient, and
- core power peaking

Appropriate biasing of these terms and other important parameters are addressed in this report. As the methodology is developed, each of the important parameters identified in the PIRT are evaluated and are biased appropriately for a conservative evaluation in addressing the NuScale REA regulatory criteria.

The REA methodology includes the following components:

- nuclear design and core response
- system response
- subchannel response
- fuel response

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 convergence and reliability of the final results. The deviations from the subchannel methodology are discussed and justified in this report using the following sensitivity studies:

- axial node size
- allowed minimum and maximum iterations
- damping factor for axial flow and cross-flow

This report describes representative sample problems employing the REA methodology and demonstrates how the REA behaves when modeling the NPM. However, NuScale is not seeking approval of the results provided in this report. Appropriately biased key inputs are used for the sample problems. The results are summarized for power levels ranging from hot zero power to 102 percent hot full power for fuel centerline temperature and enthalpy increase calculated by the adiabatic heat-up model, the MCHFR values from NRELAP5 and VIPRE-01, and the peak RCS pressure from NRELAP5. Sensitivity studies are documented for NRELAP5 for changes to RCS average temperature, loss of offsite power, and RCS flow. VIPRE-01 sensitivity calculation results are also provided. Results of the sample problems and sensitivity cases are discussed. Trends of the important parameters are also presented.

The REA methodology meets the regulatory requirements following the approved regulatory guidelines. The results of the sample problems using the REA methodology are provided in the report to demonstrate that the methodology meets the regulatory criteria from References 8.2.3, 8.2.4, and 8.2.5 by meeting the NuScale criteria defined in this report.

1.0 Introduction

A rod ejection accident (REA) is applicable to pressurized water reactor (PWR) designs with control rod assembly (CRA) insertions at the top of the reactor pressure vessel. An REA is an assumed rupture of the control rod drive mechanism (CRDM), or of the CRDM nozzle. Upon this rupture, the pressure in the reactor coolant system (RCS) provides an upward force that rapidly ejects the CRA from the core. The ejection of the CRA results in a large positive reactivity addition, leading to a highly skewed and severely peaked core power distribution. As the power rapidly rises, fission energy accumulates in the fuel rods faster than it can be deposited into the coolant, raising the fuel temperature. The power rise is mitigated by fuel temperature feedback and delayed neutron effects.

The CRDM design in the NuScale Power Module (NPM) is consistent with existing PWR designs (top entry), therefore, REA is the appropriate reactivity insertion accident to analyze for the NPM.

1.1 Purpose

The purpose of this report is to describe the methodology that NuScale intends to use for the analysis of REA for the NuScale design certification application. This methodology is used in the analysis that supports results reported in Section 15.4.8 of the NuScale Final Safety Analysis Report.

1.2 Scope

This report describes the assumptions, codes, and methodologies used to perform REA analysis. This report is intended to provide the methodology for performing this analysis; the input values and analysis results presented in the report are for demonstration of the analytical methodology and are not meant to represent final analysis results or design values. Analysis results and comparisons to applicable specified regulatory criteria from regulatory guidance are provided for illustration to aid the understanding of the context of the application of these methodologies.

The intention of the methodology herein is to demonstrate that no fuel failure occurs, therefore there is no dose consequence associated with the REA.

1.3 Abbreviations and Definitions

Table 1-1 Abbreviations

Term	Definition
BOC	beginning of cycle
CHF	critical heat flux
CRA	control rod assembly
CRDM	control rod drive mechanism
DTC	Doppler temperature coefficient
EOC	end of cycle
EPRI	Electric Power Research Institute
FGR	fission gas release
FTC	fuel temperature coefficient
GDC	general design criterion
HFP	hot full power
HZP	hot zero power
IR	importance ratio
KR	knowledge ratio
LOCA	loss-of-coolant accident
MCHFR	minimum critical heat flux ratio
MOC	middle of cycle
MTC	moderator temperature coefficient
NPM	NuScale Power Module
NRC	Nuclear Regulatory Commission
NRF	nuclear reliability factor
PCMI	pellet-cladding mechanical interaction
PDIL	power dependent insertion limit
PIRT	phenomena identification and ranking table
PWR	pressurized water reactor
RCS	reactor coolant system
REA	rod ejection accident
RPV	reactor pressure vessel

Term	Definition
SAF	single active failure
SRP	Standard Review Plan
TH	thermal-hydraulics
WRSO	worst rod stuck out

Table 1-2 Definitions

Term	Definition
β_{eff}	effective delayed neutron fraction
Courant number	A stability criterion for numerical analysis that is calculated by: u × Δt/Δx, where u is the axial velocity, Δt is the time step size, and Δx is the axial node size. It is a dimensionless number used as a necessary condition for convergence of numerical solutions of certain sets of partial differential equations.
$F_{\Delta H}$	enthalpy rise hot channel factor
F_Q	heat flux hot channel factor (total peaking factor)
IR	importance ratio: phenomena score on a scale between 0 and 100 with an increasing score representing increasing importance to the methodology
KR	knowledge ratio: phenomena score on a scale between 0 and 100 with an increasing score representing increasing knowledge of phenomena
MWd/MTU	megawatt days per metric ton of uranium

2.0 Regulatory Considerations

2.1 Regulatory Requirements

The REA is the PWR design basis accident under the scope of reactivity insertion accidents. The regulatory basis for the REA is fundamentally derived from the General Design Criteria (GDC) of 10 CFR 50 (Reference 8.2.1) Appendix A, specifically GDC 13 and GDC 28.

General Design Criterion 13 addresses the use of plant design features and instrumentation that are involved in the termination of a REA. General Design Criterion 28 addresses the design of the reactivity control system to limit the degree of power excursion possible during an REA.

Two sets of regulatory criteria are considered for this REA methodology. The first set is the current approved methodology as described in NUREG-0800, the Standard Review Plan (SRP), Sections 15.0 and 4.2 (Reference 8.2.3 and Reference 8.2.4). The second set is the proposed criteria as of March 2015, documented in the Clifford Letter (Reference 8.2.5).

Evaluation criteria specific to REAs, or more generally to reactivity insertion accidents, have been identified in this section to provide a basis for satisfying the above noted GDCs. These criteria can, in general, be grouped into four categories: RCS pressure, fuel cladding failure, core coolability, and fission product inventory. Section 2.3 identifies where in this report each of these specific criteria are addressed.

This report presents the NuScale REA methodology and demonstrates that the applicable regulatory acceptance criteria, described in this section, are met.

2.2 Regulatory Guidance Background

The regulatory criteria discussed in this report address the current approved regulatory guidance and interim criteria for the REA. The interim criteria was developed by the Nuclear Regulatory Commission (NRC) and published in March of 2015 (Reference 8.2.5). Sections 2.2.1 and 2.2.2 below delineate the current, approved REA regulatory criteria, and the interim criteria, respectively.

2.2.1 Current Regulatory Guidance and Standard Review Plan Criteria

2.2.1.1 Reactor Coolant System Pressure

The maximum RCS pressure acceptance criterion is defined in References 8.2.2 and 8.2.4 as "*The maximum reactor pressure during any portion of the assumed excursion should be less than the value that result in stresses that exceed the "Service Limit C" as defined in the American Society of Mechanical Engineers Boiler and Pressure Vessel Code.*" This acceptance criterion can be met by showing the maximum RCS pressure does not exceed 120 percent of the design pressure.

2.2.1.2 Fuel Cladding Failure

The regulatory criteria for evaluating fuel cladding failure are defined in Reference 8.2.3. These criteria are the following:

- For zero power conditions, the high temperature cladding failure threshold is expressed in the following relationship based on the internal rod pressure:
  - Internal rod pressure ≤ system pressure: Peak radial average fuel enthalpy = 170 cal/g, and
  - Internal rod pressure > system pressure: Peak radial average fuel enthalpy = 150 cal/g.
- For intermediate and full power conditions, fuel cladding failure is presumed if local heat flux exceeds the critical heat flux (CHF) thermal design limit.
- The pellet-cladding mechanical interaction (PCMI) failure threshold is a change in radial average fuel enthalpy greater than the corrosion-dependent limit depicted in Figure 5-2 (Figure B-1 of Reference 8.2.3).

2.2.1.3 Core Coolability

The regulatory criteria for evaluating core coolability are defined in Reference 8.2.3. These criteria are the following:

- Peak radial average fuel enthalpy must remain below 230 cal/g.
- Peak fuel temperature must remain below incipient fuel melting conditions.
- Mechanical energy generated as a result of (1) non-molten fuel-to-coolant interaction and (2) fuel rod burst must be addressed with respect to reactor pressure boundary, reactor internals, and fuel assembly structural integrity.
- No loss of coolable geometry due to (1) fuel pellet and cladding fragmentation and dispersal and (2) fuel rod ballooning.

Core coolability conditions due to fuel failure are avoided for the NuScale REA methodology in that CHF is not permitted to occur. Given that CHF does not occur, the fuel rods do not heat up enough to rupture, and core coolability issues due to post-CHF conditions are not possible. Also, PCMI failures are precluded by assuring that the criterion for limiting cladding oxidation delineated in Section 2.2.1.2 above is met. In addition, the NuScale criteria adopted and delineated in Section 2.3.3 establishes significant margin to the first two criteria. Therefore the last two criteria above are eliminated.

2.2.1.4 Fission Product Inventory

The regulatory criteria for evaluating the fission product inventory are defined in Reference 8.2.3. The transient fission gas release (FGR) correlation presented in Reference 8.2.3 is listed below. The total fission product inventory is equal to the steady state gap inventory plus the transient FGR derived with the following correlation:

Transient FGR (percent) = $[(0.2286 * \Delta H) - 7.1419]$

where,

FGR = fission gas release, percent (must be ≥ 0)

ΔH = fuel enthalpy increase (Δcal/g)

2.2.1.5 Effects of Loss of Primary System Integrity

The effects of the loss of primary system integrity are discussed in Regulatory Guide 1.77 (Reference 8.2.2). The two effects addressed are:

- the NPM depressurization effects of the prediction of CHF
- the resultant NPM mass and energy released to the containment.

2.2.2 Nuclear Regulatory Commission Proposed Changes to Criteria

Consideration is given to the acceptance criteria proposed in the March 16, 2015, memorandum “Technical and Regulatory Basis for the Reactivity-Initiated Accident Acceptance Criteria and Guidance, Revision 1,” (Reference 8.2.5), which contains the anticipated acceptance criteria for a future revision of RG 1.77.

2.2.2.1 Reactor Coolant System Pressure

This acceptance criterion can be met by showing the maximum RCS pressure does not exceed 120 percent of the design pressure.

The maximum RCS pressure acceptance criterion defined in Reference 8.2.5 is unchanged.

2.2.2.2 Fuel Cladding Failure

The criteria for evaluating fuel cladding failure are defined in Reference 8.2.5. These criteria are the following:

- For zero power conditions, the high temperature cladding failure threshold is expressed in the following relationship based on the cladding differential pressure:
  - $\Delta P \leq 1.0$ MPa: Peak radial average fuel enthalpy = 170 cal/g
  - 1.0 MPa < ΔP < 4.5 MPa: Peak radial average fuel enthalpy = 170 – ((ΔP – 1.0)*20) cal/g
  - $\Delta P \geq 4.5$ MPa: Peak radial average fuel enthalpy = 100 cal/g
- Predicted cladding differential pressure must consider the impact of transient FGR on internal gas pressure.
- Fuel cladding failure for intermediate and full power conditions based on CHF is unchanged.
- The PCMI failure threshold is a change in radial average fuel enthalpy greater than the excess hydrogen dependent limit depicted in Figures 3.2.2-21 and 3.2.2-22 of Reference 8.2.5.
- If fuel temperature anywhere in the pellet exceeds incipient fuel melting conditions, then fuel cladding failure is presumed. Fuel temperature predictions must be based upon design-specific information accounting for manufacturing tolerances and modeling uncertainties using NRC approved methods, including burnup-enhanced effects on pellet radial power distribution, fuel thermal conductivity, and fuel melting temperature. Increases in radiological source term because of predicted fuel melting must be accounted for in dose calculations.

2.2.2.3 Core Coolability

The regulatory criteria for evaluating core coolability are defined in Reference 8.2.5. These criteria are the following:

- Peak radial average fuel enthalpy is unchanged.
- A limited amount of fuel melting is acceptable provided it is restricted to (1) fuel centerline region and (2) less than 10 percent of any pellet volume. For the outer 90 percent of the pellet volume, peak fuel temperature must remain below incipient fuel melting conditions.
- The mechanical energy generation criterion is unchanged.
- Criterion for no loss of coolable geometry is unchanged.

Per Reference 8.2.5, until regulatory guidance exists to address the last two items above, applicants need only demonstrate compliance to the first two coolability criteria. Fuel cladding failure is addressed in Sections 2.2.1.2 and 2.2.2.2 of this report.

2.2.2.4 Fission Product Inventory

The regulatory criteria for evaluating the fission product inventory are defined in Reference 8.2.5. The revised transient FGR correlations are listed below. The total fission product inventory is equal to the steady state gap inventory plus the transient FGR derived with the following correlations:

- Peak Pellet Burnup < 50 GWd/MTU: Transient FGR (percent) = [(0.26 * ΔH) - 13]
- Peak Pellet Burnup ≥ 50 GWd/MTU: Transient FGR (percent) = [(0.26 * ΔH) - 5]

where,

FGR = fission gas release, percent (must be > 0)

ΔH = fuel enthalpy increase (Δcal/g)

2.3 Regulatory Criteria for NuScale

Table 2-1 summarizes how the regulatory acceptance criteria from References 8.2.3, 8.2.4, and 8.2.5 are addressed and applied to the NuScale REA methodology within this report.

Table 2-1 Method for addressing regulatory criteria

Criteria	Criteria Section	Method Section
Maximum RCS pressure	2.3.1	5.3
Hot zero power (HZP) fuel cladding failure	2.3.2	5.5.2
FGR effect on cladding differential pressure	2.3.2	N/A
CHF fuel cladding failure	2.3.2	5.4.1
Cladding oxidation-based PCMI failure	2.3.2	5.5.3
Cladding excess hydrogen-based PCMI failure	2.3.2	N/A
Incipient fuel melting cladding failure	2.3.2	5.5.1
Peak radial average fuel enthalpy for core cooling	2.3.3	5.5.2
Fuel melting for core cooling	2.3.3	5.5.1
Fission product inventory	2.3.4	5.6

2.3.1 Reactor Coolant System Pressure

The maximum RCS pressure acceptance criterion of 120 percent of design pressure as defined in Reference 8.2.5 is used in the methodology. The NPM design pressure is 2100 psia. Therefore, the peak pressure during the REA is limited to 2520 psia. Reactor Coolant System conditions are calculated with the NRELAP5 code.

2.3.2 Fuel Cladding Failure

The criteria for evaluating fuel cladding failure are defined in Reference 8.2.5. These criteria are the following:

- For zero-power conditions, the high-temperature cladding-failure threshold is expressed in cladding differential pressure. The peak radial average fuel enthalpy is below the 100 cal/g associated with the maximum peak rod differential pressure of $\Delta P \geq 4.5$ MPa. Thus, the predicted cladding differential pressure does not need to be calculated and the impact of transient FGR on internal gas pressure need not be included for the REA.
- For intermediate- and full-power conditions, fuel cladding failure is presumed if local heat flux exceeds the CHF thermal design limit. Detailed thermal-hydraulic (TH) conditions are calculated using the VIPRE-01 code.
- The PCMI failure threshold is a change in radial average fuel enthalpy greater than the corrosion-dependent limit depicted in Figure 5-2. This report does not include a methodology to address excess hydrogen in the cladding and the associated effect on PCMI-based cladding failure, because the hydrogen-based limits are not yet approved by the NRC.
- If fuel temperature anywhere in the pellet exceeds incipient fuel melting conditions, then fuel cladding failure is presumed. Fuel temperature predictions must be based upon design-specific information accounting for manufacturing tolerances and modeling uncertainties using NRC approved methods including burnup-enhanced effects on pellet radial power distribution, fuel thermal conductivity, and fuel melting temperature. Incipient fuel melt has been determined to be **[ ]** degrees F for the NuScale fuel (Reference 8.2.12) for a conservative pellet burnup of **[ ]** MWd/MTU.

2.3.3 Core Coolability

The regulatory criteria for evaluating core coolability are defined in Reference 8.2.3 and 8.2.5. Criteria 1 and 2 (as follows) are adopted for the NuScale REA methodology in a bounding fashion:

- Peak radial average fuel enthalpy will remain below 230 cal/g.
- No fuel melt will occur.

Core coolability conditions due to fuel failure are avoided for the NuScale REA methodology in that CHF is not permitted to occur. Given that CHF does not occur, the fuel rods do not heat up enough to rupture, and coolability issues due to post-CHF conditions are not possible. Also, PCMI failures are precluded by assuring that the criterion for limiting cladding oxidation delineated in Section 2.3.2 above is met. In addition, the first two core coolability NuScale criteria delineated above establishes significant margin to the first two criteria from Section 2.2.1. Therefore the last two criteria from Section 2.2.1 are eliminated.

2.3.4 Fission Product Inventory

The regulatory transient FGR criteria do not apply to the NuScale REA methodology for the following two reasons:

- This methodology requires that no fuel failure occurs, whether due to fuel melt, or transient enthalpy increase, or cladding failure due to minimum critical heat flux ratio (MCHFR), and therefore, the pellet-to-cladding gap will not be breached.
- The regulatory fuel cladding failure criteria in Section 2.3.2,based on cladding differential pressure, incorporates the most limiting criteria for $\Delta P \geq 4.5$ MPa, therefore any increase in pressure that could occur during the transient due to FGR will not change allowed peak radial average fuel enthalpy.

Based on the above two items, the acceptance criterion in SRP Section 15.4.8 (Reference 8.2.4) to perform a dose analysis is not required for the NuScale REA methodology.

2.3.5 Effects of Loss of Primary System Integrity

The effects addressed in the NuScale REA methodology include:

- the NPM depressurization effects of the prediction of CHF
- the resultant NPM mass and energy released to the containment

The CHF effect of the depressurization is addressed in the sensitivity study results presented in Section 6.4.2.7. The sensitivity study found that an increasing RCS pressure yields lower MCHFR results for the system pressure at which MCHFR is expected to occur for the REA. MCHFR is evaluated due to the reactor pressure vessel (RPV) pressurization when the power excursion occurs. Therefore, it is conservative for the NuScale REA methodology to not include system depressurization effects.

The mass and energy release effect is bounded by other RPV releases, which are evaluated for containment peak pressure. This evaluation included the additional energy generated during the REA.

3.0 Overview and Evaluation Codes

This section provides a general overview of REA and the applicable codes used to model the event for the NPM.

3.1 Overview

A general overview of the cause and progression of the REA is described in References 8.2.2 and 8.2.4. For the NPM, the REA is an assumed rupture of the CRDM or of the CRDM nozzle. An REA will lead to a rapid positive reactivity addition resulting in a power excursion and a skewed and peaked core power distribution. As power rises rapidly, the fission energy accumulates in the fuel rods faster than it can migrate to the coolant, resulting in raised fuel temperatures. The power rise is mitigated by fuel temperature feedback and delayed neutron effects. A reactor trip on high power rate is generated within a few hundredths of a second of the rod ejection and there is a delay before the CRAs are inserted. Some cases with low ejected CRA worth or large negative values of reactivity feedback may not hit the high power rate trip setpoint and will instead settle at a new steady state condition. The reactor core is protected against severe fuel failure by the reactor protection system and by restrictions of the power dependent insertion limit (PDIL) and axial offset window, which determine the depth of CRA insertion and initial power distribution allowed in the core.

3.1.1 Reactivity Considerations

The REA can behave differently based on the static worth of the ejected CRA. For example, REA can behave as follows:

- Reactivity insertion close to or greater than effective delayed neutron fraction; this scenario results in a prompt critical scenario.
- Reactivity insertion much less than the delayed neutron fraction; this scenario is considered sub-prompt critical.

In general, CRAs that are inserted deeper into the core will have a higher static worth. As the PDILs increase with reducing power (until they level off at an intermediate power), higher power cases will produce a lower ejected CRA worth, and thus will tend towards the sub-prompt critical scenario. A higher ejected CRA worth at reduced power can result in prompt critical power excursions. Similarly, a core with greater axial offset will produce a higher static worth.

3.1.1.1 Prompt Critical

In a prompt critical scenario, the energy deposition can be defined by the following equation:

$$E_d = \frac{2 * (\rho - \beta) * C_p}{\alpha_D} \qquad \text{Equation 3-1}$$

where,

E_d = energy deposition,

ρ = static ejected CRA worth,

β = delayed neutron fraction,

C_p = fuel heat capacity, and

α_D = Doppler temperature coefficient (DTC).

This equation (Equation 5-90 of Reference 8.2.13) implies that the key parameters affecting the energy deposition during a prompt critical REA are the ejected CRA worth, delayed neutron fraction, fuel heat capacity, and the DTC.

3.1.1.2 Sub-Prompt Critical

In a sub-prompt critical scenario, the delayed neutrons limit the power excursion, and instead a jump in power occurs. This prompt jump in power can be approximated by the following equation:

$$\frac{P_j}{P_o} = \frac{\beta}{(\beta - \rho)} \qquad \text{Equation 3-2}$$

where,

P_j = prompt jump power, and

P_o = initial power.

This equation (Equation 3-35 of Reference 8.2.13) implies that, for a given CRA worth, a higher initial power will result in a larger prompt jump power, and for these cases, the relationship between β and ρ has the most significant impact.

3.1.2 Reactor Coolant System Pressure Behavior

The trend of CHF with RCS pressure is described in Section 5.3. Differences between the bounding CHF and RCS overpressure calculations are described in Section 5.2.1.

3.2 Analysis Computer Codes and Evaluation Flow

The safety analyses of NuScale Final Safety Analysis Report Chapter 15 non-loss of coolant accident (non-LOCA) transients and accidents are performed using the CASMO5/SIMULATE5 code package for reactor core physics parameters, NRELAP5 for the transient system response, and VIPRE-01 for the subchannel analysis. The REA methodology follows a similar approach for use of code packages. The nuclear analysis portion of the REA transient response is performed using the three-dimensional space-time kinetics code SIMULATE-3K. NRELAP5 is used to simulate the RCS response to the core power excursion, and the VIPRE-01 code is used to model the core thermal response and to calculate the MCHFR. A conservative adiabatic heatup model is used for determination of the peak fuel temperature and enthalpy. Figure 3-1 depicts the computer codes used and the flow of information between codes and evaluations to address the regulatory acceptance criteria. Note that the adiabatic heatup evaluation is a manual calculation as opposed to software.

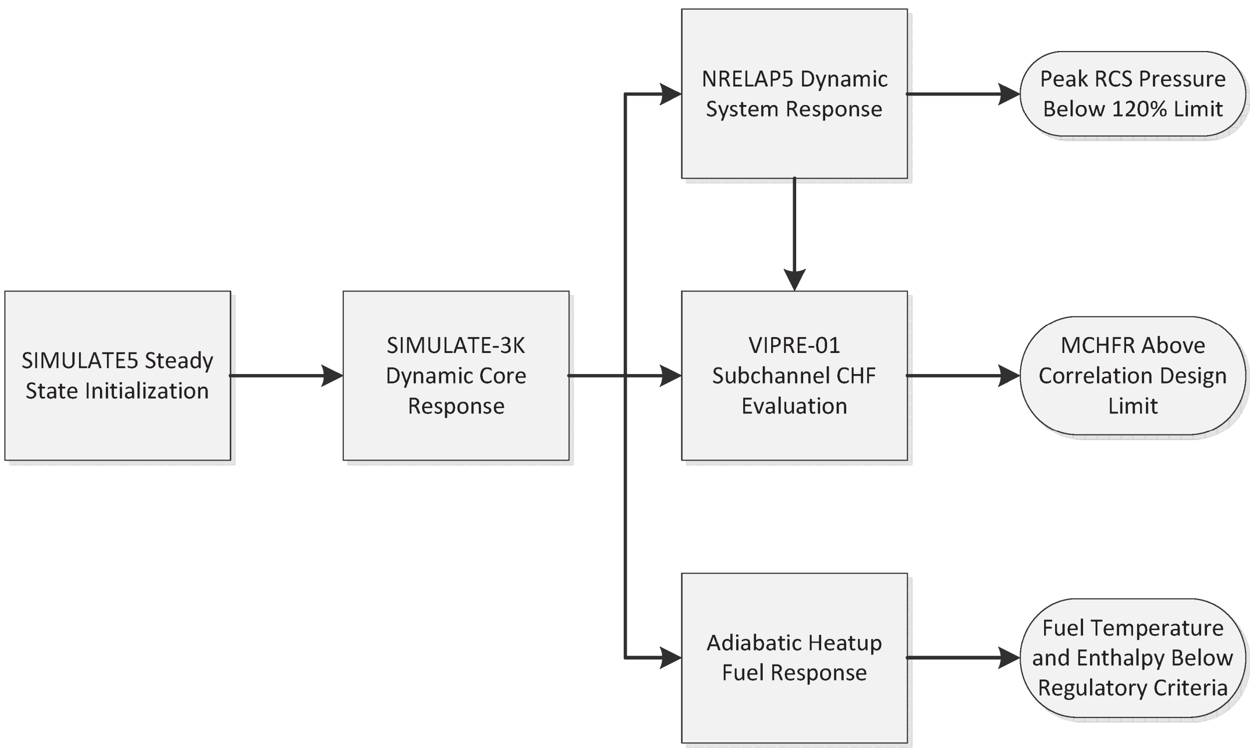

Figure 3-1 Calculation schematic for analyzing rod ejection accident

3.2.1 Core Response

Reference 8.2.7 provides the validation of CASMO5/SIMULATE5 to perform steady state neutronics calculations for the NuScale design. Validation of SIMULATE-3K for the NuScale design is described in this section.

3.2.1.1 CASMO5

CASMO5 (Reference 8.2.16) is a multi-group two-dimensional transport theory code used to generate pin cell or assembly lattice physics parameters, including cross-sections, nuclide concentrations, pin power distributions, and other nuclear data used for core performance analysis for light water reactors. The code is used to generate a neutron data library for use in the three-dimensional steady-state nodal diffusion code SIMULATE5, and the three-dimensional transient nodal code SIMULATE-3K.

CASMO5 solves the two-dimensional neutron transport equation by the Method of Characteristics. The code produces a two-dimensional transport solution based upon heterogeneous model geometry. The CASMO5 geometrical configuration consists of a square pitch array containing cylindrical fuel rods of varying composition. The code input may include burnable absorber rods, cluster control rods, in-core instrument channels, and water gaps, depending on the details of the assembly lattice design.

The CASMO5 nuclear data library consists of 586 energy groups covering a range from 0 to 20 mega electron volts (MeVs). Macroscopic cross sections are directly calculated from the geometries and material properties provided from the code input. Resonance integrals are used to calculate effective absorption and fission cross sections for each fuel rod in the assembly, and Dancoff factors are calculated to account for the shadowing effect in an assembly between different rods.

CASMO5 runs a series of depletions and branch cases to off-nominal conditions in order to generate a neutron data library for SIMULATE5 or SIMULATE-3K. These calculations form a case matrix, which functionalize boron concentration, moderator temperature, fuel temperature, shutdown cooling (isotopic decay between cycles or over long outage times), and CRA positioning with respect to exposure. The same neutron data library produced by the automated case matrix structure in CASMO5 and used for steady-state neutronic analysis in SIMULATE5 can be used for transient neutronic analysis in SIMULATE-3K.

For the REA analysis, CASMO5 is used to produce a neutron data library for steady-state neutronic calculations performed with SIMULATE5, and for transient neutronic calculations performed with SIMULATE-3K. The use of CASMO5 in this report is consistent with the methodology presented in Reference 8.2.7.

3.2.1.2 SIMULATE5

SIMULATE5 (Reference 8.2.17) is a three-dimensional, steady-state, nodal diffusion theory, reactor simulator code. It solves the multi-group nodal diffusion equation, employing a hybrid microscopic-macroscopic cross-section model that accounts for depletion history effects. SIMULATE5 output includes steady state nuclear analysis predictions, such as critical boron concentration, boron worth, reactivity coefficients, CRA worth, shutdown margin, power distributions, and peaking factors.

For the REA analysis, SIMULATE5 is used to initialize the cycle-specific model and reactor conditions for the REA simulation in SIMULATE-3K. SIMULATE5 writes an initial condition restart file containing the core model geometry, including CRA positioning, reactor operating conditions, and detailed depletion history, to establish the initial core conditions before the start of the REA transient. The restart file contains the explicit neutron library data produced in CASMO5 necessary for SIMULATE-3K calculations, and automatically accounts for differences between the SIMULATE5 calculation model and the data necessary for the SIMULATE-3K calculation model to properly execute.

The use of SIMULATE5 in this report is consistent with the methodology presented in Reference 8.2.7.

3.2.1.3 SIMULATE-3K

SIMULATE-3K (References 8.2.19, 8.2.20) is a three-dimensional nodal reactor kinetics code that couples core neutronics with detailed TH models. The neutronic model solves the transient three-dimensional, two-group neutron diffusion equations using the quadratic polynomial analytic nodal solution technique, or the semi-analytic nodal method. The code incorporates the effects of delayed neutrons, spontaneous fission in the fuel, alpha-neutron interactions from actinide decay, and gamma-neutron interactions from long term fission product decay.

The TH module consists of a conduction model and a hydraulics model. The conduction model calculates the fuel pin surface heat flux and within-pin fuel temperature distribution. Heat conduction in the fuel pin is governed by the one-dimensional radial heat conduction equation. The heat source is comprised of prompt fission and decay heat. Material properties are temperature and burnup dependent, and gap conductance is dependent on exposure and fuel temperature. The three-dimensional hydraulic model is nodalized with one characteristic TH channel per fuel bundle (no cross flow) and a variable axial mesh. The hydraulics model calculates the flow, density, and void distributions for the channel.

The TH module is coupled to the neutronics module through the fuel pin heat generation rate, which is based on reactor power. The TH module provides the neutronics module with data to determine cross-section feedback associated with the local thermal conditions. Cross-section feedback is based on coolant density, fuel temperature, CRA type, fuel exposure, void history, control rod history, and fission product inventory. The heat transferred from the fuel to the coolant provides the hydraulic feedback.

The SIMULATE-3K core model is established from SIMULATE5 restart files, which provide core model geometry and loading pattern, fuel assembly data, nodal information containing radial and axial mesh, and detailed depletion history. SIMULATE-3K uses the same cross-section library created from CASMO5 data that was used in SIMULATE5.

SIMULATE-3K is used for transient neutronic analysis of the REA at various times in core life, power levels, CRA positions, and initial core conditions. The transient REA analysis determines total core power, reactivity insertion, three-dimensional power distributions, and power peaking.

A combination of CASMO5, SIMULATE5, and SIMULATE-3K are used to calculate the core response and reactivity-related inputs for the downstream evaluations discussed in the following sections. The power response for the accident is determined by SIMULATE-3K for both NRELAP5 and VIPRE-01.

3.2.1.4 Validation of SIMULATE-3K

The validation of SIMULATE-3K to determine the transient neutronic response of the NuScale reactor during an REA includes comparisons to steady state neutronics calculations from SIMULATE5, and multiple transient benchmark problems performed by the code vendor, Studsvik Scandpower Inc. (Studsvik).

Steady-state neutronics calculation comparisons between SIMULATE-3K and SIMULATE5 demonstrate the ability of the SIMULATE-3K neutronics calculation methodology to accurately predict core physics parameters important to the REA event. These parameters include reactivity coefficients, including moderator temperature coefficient (MTC) and DTC, CRA and ejected worth, delayed neutron fraction, radial and axial power distributions, and power peaking factors. For all parameters except MTC, SIMULATE-3K results were in very good agreement with SIMULATE5 results. SIMULATE-3K MTC results were close to SIMULATE5 results, with SIMULATE-3K values generally more positive than the SIMULATE5 values. This is conservative for the REA analysis, because a more positive MTC limits the negative reactivity insertion from moderator feedback during the event.

SIMULATE-3K REA analysis for NuScale includes uncertainty factors on key core physics parameters important to reactivity. These parameters include delayed neutron fraction, ejected CRA worth, inserted CRA worth, MTC, and DTC. Uncertainties are applied to these parameters to either increase the positive reactivity insertion associated with an ejected CRA, or decrease the negative reactivity insertion associated with moderator and fuel temperature feedbacks and associated with the worth of the CRAs after a reactor trip. The agreement between SIMULATE-3K and SIMULATE5 calculations of these core physics parameters allow for the adoption of the nuclear reliability factors (NRFs) determined for SIMULATE5 (Reference 8.2.7) to be used by SIMULATE-3K for NuScale REA analysis.

In addition to steady-state comparisons, Studsvik has performed numerous benchmarks demonstrating the ability of SIMULATE-3K to model and accurately predict core physics parameters during reactor transients. Two of these benchmarks for REA analysis include experiments performed at the SPERT III E-core research reactor (Reference 8.2.21), and the NEACRP control rod ejection problem computational benchmark (Reference 8.2.22).

The Studsvik SPERT III benchmark provides measured REA transient data for comparison to SIMULATE-3K. SPERT III was a pressurized water nuclear research reactor that analyzed reactor kinetic behavior under conditions similar to commercial reactors. The SPERT III core resembled a commercial reactor, but of a reduced size more closely resembling the NuScale core size. The fuel type (uranium dioxide), moderator, system pressure, and certain initial operating conditions considered for SPERT III are also representative of NuScale. This benchmark demonstrates the ability of SIMULATE-3K to model fast reactivity transients in a PWR core (Reference 8.2.22). Similarities between the NuScale design and the SPERT III core, and notably the small core size, demonstrate applicability and suitability for SIMUALTE-3K REA transient analysis of the NuScale core.

In addition to the Studsvik benchmarks aforementioned, NuScale has performed a benchmark of the dynamic reactor response simulated by SIMULATE-3K of the SPERT III experiment. The original experiment included on the order of one hundred unique tests at five different sets of thermal-hydraulic conditions, with varying initial static worths at each statepoint. One test from each condition set that generally corresponds to the highest static worth for the statepoint has been benchmarked. A comparison of key parameters demonstrates that SIMULATE-3K compares to SPERT with generally excellent agreement; differences are within the experimental uncertainty (with few exceptions), and the major and minor phenomena are correctly predicted.

The NEACRP control rod ejection problem is a computational benchmark that includes a reference solution provided by the PANTHER code, and SIMULATE-3K REA transient results are compared against the reference solution. In this benchmark, a rod ejection accident in a typical commercial PWR at HZP conditions is analyzed. The fuel type (uranium dioxide), moderator, system pressure, and certain initial operating conditions considered for NEACRP are also representative of NuScale. The capability of SIMULATE-3K to model reactivity insertions in the NEACRP benchmark analysis (Reference 8.2.24 and 8.2.25) demonstrates suitability of the code for reactivity transient applications, and specifically REA analysis applications.

The SPERT III and NEACRP benchmarks demonstrate the combined transient neutronic, TH, and fuel pin modeling capabilities of SIMULATE-3K. SIMULATE-3K results for maximum power pulse, time to peak power, inserted reactivity, energy release, and fuel centerline temperature were in excellent agreement with the results from the two benchmark problems. The SIMULATE-3K results for each of these benchmark problems establish the ability of the code to accurately model an REA transient event and predict key reactivity and power-related parameters.

3.2.2 System Response

The NRELAP5 code was developed based on the Idaho National Laboratory RELAP5-3D© computer code. RELAP5-3D©, version 4.1.3 was procured by NuScale and used as the baseline development platform for the NRELAP5 code. Subsequently, features were added to address unique aspects of the NuScale design and licensing methodology.

The NRELAP5 code includes models for characterization of hydrodynamics, heat transfer between structures and fluids, modeling of fuel, reactor kinetics models, and control systems. NRELAP5 uses a two-fluid, non-equilibrium, non-homogenous fluid model to simulate system TH responses.

The validation and applicability of NRELAP5 to the NuScale design is described in Reference 8.2.9.

3.2.3 Subchannel

The analysis software VIPRE-01 was developed primarily based on the COBRA family of codes by Battelle Pacific Northwest Laboratories for the Electric Power Research Institute. The intention was to evaluate nuclear reactor parameters including minimum departure from nucleate boiling ratio, critical power ratio, fuel and cladding temperatures, and reactor coolant state in normal and off-normal conditions.

The three-dimensional velocity, pressure, and thermal energy fields and radial fuel rod temperature profiles for single- and two-phase flow in reactor cores are predicted by VIPRE-01. These predictions are made by solving the field equations for mass, energy and momentum using finite differences method for an interconnected array of channels assuming incompressible thermally expandable flow. The equations are solved with no channel size restrictions for stability and with consideration of lateral scaling for key parameters in lumped channels. Although the formulation is based on the fluid being homogeneous, non-mechanistic empirical models are included for subcooled boiling non-equilibrium and vapor/liquid phase slip in two-phase flow.

Like other core TH codes, the VIPRE-01 modeling structure is based on subchannel analysis. The core or section of symmetry is defined as an array of parallel flow channels with lateral connections between adjacent channels. These channels characterize the dominant, longitudinal flow (vertical) by nodalization with various models and correlations predicting TH phenomena that contribute to inter-channel exchange of mass, enthalpy, and momentum. These channels can represent all or fractions of the coolant channel bordered by adjacent fuel rods (hence "subchannel") in rod bundles. The axial variation in channel geometry may also be modeled with VIPRE-01. Channels may represent closed tubes as well as larger flow areas consisting of several combined (lumped) subchannels or rod bundles. These channels communicate laterally by diversion crossflow and turbulent mixing.

The original VIPRE-01 version (MOD-01) was submitted to the NRC in 1985 for use in PWR and boiling water reactor licensing applications. A safety evaluation report by the NRC was issued the following year (1986) (Reference 8.2.27). The NRC accepted MOD-01 with several specific restrictions and qualifications, limiting its use to PWR licensing applications for heat transfer regimes up to the point of CHF. This approval was contingent on: (a) the CHF correlation and its limit used in the application is approved by the NRC and (b) each organization using VIPRE for licensing calculations are to submit separate documentation justifying their input selection and modeling assumptions. In 1990, the MOD-02 version of VIPRE-01 was submitted to the NRC to review an improved and updated version, including changes and corrections from the MOD-01 version. This version was approved with an issued SER in 1993 with the same requirements and qualifications as in the MOD-01 SER (Reference 8.2.15). Unless otherwise stated, in the remainder of this report a reference to VIPRE-01 is referring to the MOD-02 version.

The validation and applicability of VIPRE-01 to the NuScale design is described in Reference 8.2.11.

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.

4.0 Identification of Important Phenomena for Rod Ejection Accident

Reference 8.2.26 presents the phenomena identification and ranking tables (PIRT) for REA. The PIRT addresses the parameters for consideration in modeling the REA to address the relevant regulatory guidance. Note that this PIRT is an industry PIRT based on large-scale reactors and is not an internally developed NuScale PIRT. This PIRT is applicable to the NuScale design because the PIRT is focused on PCMI-related cladding failures, and the fuel design used for NuScale is consistent with that used in larger PWRs (see Reference 8.2.8). Phenomena important to the REA are also identified in Section 15.4.8 of the SRP (Reference 8.2.4) and the EPRI technical report for three-dimensional analysis of REA (Reference 8.2.14).

The overall goal of the evaluation of an REA is to

- evaluate the integrity of the fuel pin during the power transient.
- confirm no fuel failures due to exceeding the CHF design limit.
- evaluate the integrity of the RCS during the pressure increase.

4.1 Industry Phenomena Identification and Ranking Table for Rod Ejection Accident

Use of the PIRT information allows the development of conservative assumptions in the REA methodology. These assumptions are addressed in more detail in Section 5.0. The PIRTs are split into four categories, two of which are applicable to the NuScale REA methodology: plant transient analysis and fuel rod transient analysis. The other categories relate to testing, which is not within the scope of this methodology.

Each phenomenon in the PIRT is assigned two scores, the importance ratio (IR) and knowledge ratio (KR). These are on scales of 0-100, with 100 IR being extremely important and 100 KR being very well-known and understood. IR scores above 75 signify highly important criteria. Therefore, this section will address those items with an IR of 75 or greater for evaluating REA against the regulatory acceptance criteria.

The rod ejection accident PIRT (Reference 8.2.26) provides the REA analysis parameters in Tables 3-1 and 3-3. Tables 4-1 and 4.2 list the important phenomena for the two applicable categories that apply to the NuScale REA methodology: Table 4-1 for the plant transient analysis and Table 4-2 for the fuel response. Note that for Table 4-2, only the initial conditions and fuel and cladding temperature change items are considered.

Table 4-1 Plant transient analysis phenomena identification and ranking table rankings

Phenomenon	IR Score	KR Score
Calculation of Power History During Pulse (Includes Pulse Width)		
Ejected CRA worth	100	100
Fuel temperature feedback	100	96
Delayed neutron fraction	95	96
Fuel cycle design	92	100
Calculation of Pin Fuel Enthalpy Increase During Pulse (Includes Cladding Temperature)		
Heat capacities of fuel and cladding	94	90
Pin peaking factors	97	100

Table 4-2 Fuel response phenomena identification and ranking table rankings

Phenomenon	IR Score	KR Score
Initial Conditions		
Gap size	96	82
Gas distribution	79	50
Pellet and cladding dimensions	91	96
Hydrogen distribution	100	50
Power distribution	100	89
Fuel-clad gap friction coefficient	75	30
Condition of oxidation (spalling)	100	46
Coolant conditions	93	96
Bubble size and bubble distribution	83	20
Transient power specification	100	94
Fuel and Cladding Temperature Changes		
Heat resistances in fuel, gap, and cladding	75	77
Heat capacities of fuel and cladding	88	93
Coolant conditions	85	88

It should be noted that additional parameters for the CHF and pressurization calculations not listed above were considered in the NuScale REA methodology. Discussion of other parameters considered for the methodology is provided in Section 5.3.

4.1.1 Plant Transient Analysis

4.1.1.1 Calculation of Power History

Ejected CRA worth is calculated by SIMULATE-3K. A larger worth is conservative, as it will maximize the power pulse. In order to maximize the worth, uncertainty factors are applied to the insertion depth of the CRAs and to the static CRA worth.

Fuel temperature feedback, in the form of DTC, is calculated by SIMULATE-3K. A less negative DTC is conservative, as DTC is the primary component that arrests the power pulse. In order to make DTC less negative, an uncertainty factor is applied.

Delayed neutron fraction, β_{eff}, is calculated by SIMULATE-3K. A smaller value of β_{eff} is conservative, as is shown in Equation 3-1 and Equation 3-2. In order to minimize β_{eff}, an uncertainty factor is applied.

Fuel cycle design is performed using CASMO5 and SIMULATE5. The sample calculations provided in this report were developed using an equilibrium cycle. In order to capture effects of the fuel cycle design, the REA is analyzed at beginning of cycle (BOC), end of cycle (EOC), and at a middle of cycle (MOC) point where $F_{\Delta H}$ is maximum, as well as at various reactor power values ranging from HZP to hot full power (HFP).

4.1.1.2 Calculation of Pin Fuel Enthalpy Increase

The pin fuel enthalpy increase is calculated using a conservative adiabatic heatup model. This model assumes all of the energy created during the event and before the movement of the CRAs is deposited into the fuel pellets.

Heat capacity of the fuel is used to calculate the enthalpy and temperature increases in the fuel pellets during the event. The heat capacity is assumed to be that at 600 degrees F, and does not credit the increase in heat capacity as the temperature increases. Initial fuel temperatures are greater than this temperature above HZP. Because fuel capacity increases with temperature, which is not limiting from a fuel temperature perspective, this assumption is conservative.

Heat capacity of the fuel cladding is not modeled as part of the adiabatic heatup model. The regulatory criteria for fuel response only address fuel enthalpy and temperature. Assuming no heat leaves the fuel pellets during the REA event is conservative, and therefore, modeling the heat capacity of the fuel cladding is not required.

Pin peaking factors are calculated by SIMULATE-3K. For the fuel enthalpy increase, the largest pin peaking factor, F_Q, during the event and before the movement of the CRAs is used to model the limiting node. An uncertainty factor is applied to F_Q that captures manufacturing tolerances and modeling uncertainties.

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; 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. 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 heat capacities of the fuel and fuel cladding, and coolant conditions are addressed in the VIPRE-01 CHF evaluation. The 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-design specific calibration results in temperature predictions that are conservative for MCHFR. The conservative bias for MCHFR is a high initial temperature of the fuel as well as a high gap conductance. This allows the amount of heat in the fuel to be conservatively high and transferred to the coolant the fastest. To ensure that the VIPRE-01 fuel conduction calculations are conservative, this methodology requires that the entire range of possible time-in-cycle parameters are evaluated using the COPERNIC fuel performance code, including exposure, uranium enrichment, gadolinium enrichment, gap conductance, and fuel density. The VIPRE-01 model is calibrated to ensure that it produces conservative temperatures for each fuel design.

4.2 Electric Power Research Institute Technical Report

The EPRI technical report (Reference 8.2.14) has identified several key parameters for the three-dimensional analysis methodology. These key parameters are the following:

- ejected CRA worth
- delayed neutron fraction
- MTC
- fuel temperature (Doppler) coefficient
- core peaking factor
- time-in-cycle

The EPRI topical report states that uncertainty is applied to the ejected CRA worth, and the MTC and DTC. The MTC and time-in-cycle are the only parameters not already addressed as part of the PIRT. The MTC value is calculated by SIMULATE-3K. A less negative MTC is limiting, as the moderator heating during the event will reduce the power excursion. In order to make this value conservative, an uncertainty factor is applied. The REA is evaluated at BOC, MOC, and EOC to determine the worst time-in-cycle. Uncertainty application for each of the key parameters except time-in-life is discussed in Section 5.0.

4.3 Standard Review Plan Section 15.4.8 Initial Conditions

In addition to the PIRT and the EPRI topical report, the SRP Section 16.4.8 (Reference 8.2.4) provides considerations for the initial conditions of the event. The items identified are as follows:

A. *A spectrum of initial conditions, which must include zero, intermediate, and full-power, is considered at the beginning and end of a reactor fuel cycle for examination of upper bounds on possible fuel damage. At-power conditions should include the uncertainties in the calorimetric measurement.*

   This spectrum is evaluated. The two percent power uncertainty is applied at HFP conditions.

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, primary core flow is conservatively treated to minimize any flow increase, as increased flow would cause an increase in MCHFR. 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.

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 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.

5.1.4 Single Active Failure

The conservative single active failure for radially asymmetric scenarios such as REA is a failure of the flux detector in the high flux region. This is implemented by requiring all four detectors to exceed the high power rate in order to cause a reactor trip.

This single active failure does not necessarily increase the severity of the accident. However, there are no known single active failures that would increase the severity. No safety-related systems besides analytical reactor trip limits in the module protection system such as those based on power or pressure are credited. The module protection system provides reactor trip limits that are sufficiently redundant and therefore, a CRA insertion delay is assumed.

5.1.5 Automatic System Response of Non-Safety Systems

In an REA scenario, the automatic systems would work to limit the power, pressure, and level excursions. The following balance-of-plant and control system responses are treated conservatively:

- Pressure control is disabled to ensure maximum pressure.
- Inventory control is disabled to maximize pressurizer level, and thus RPV pressure.
- Feedwater flow is assumed constant, keeping flow from increasing due to the increase in moderator average temperature.
- Steam pressure is not permitted to decrease as the power increases.
- CRA motion, besides the ejection and insertion of the CRAs, are not modeled.

The above conservatisms are appropriate for both the MCHFR and maximum pressure cases.

5.1.6 Loss of Alternating Current Power

The REA analysis, for the purpose of calculating MCHFR, assumes that loss of alternating current (AC) power occurs at the time of reactor trip. The timing of the loss of alternating current power has no effect on the rod ejection accident MCHFR results, as shown in Table 6-4.

For the purpose of determining the limiting RCS pressure, the REA is evaluated with loss of AC power at both the time of event initiation and at the time of reactor trip. The timing of the loss of AC power is an integral part of the biasing considerations listed in Section 5.3.1.2.

5.2 Core Response Methodology

5.2.1 Calculation Procedure

The core response REA methodology has two distinct stages. The first stage involves static calculations that use SIMULATE5. This stage establishes the initial conditions for the event. The second stage is the transient simulations with SIMULATE-3K. This stage establishes boundary conditions for the downstream plant response and subchannel calculations. These calculations are performed at various bounding combinations of power and burnup to determine the conditions where it is necessary to examine the plant response and perform subchannel analyses. The power levels that should be considered in the SIMULATE-3K analyses must cover the entire operating domain, and must take into consideration power levels where changes in behavior of safety systems or plant conditions occur (such as changes in allowed CRA positions).

5.2.1.1 Static Calculations

SIMULATE5 is used to run the static portion of the REA calculations for the core response analysis. This static assessment involves two calculations: assessment of the worst rod stuck out (WRSO) and development of the restart file to feed the initial conditions to SIMULATE-3K.

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}}2(a),(c)

The initial conditions of reactor power, inlet temperature, coolant mass flux, fission product material, identification of the CRA groups, positions of the CRAs, and information about the spacer grids are passed as input to SIMULATE-3K for use in the REA simulation.

5.2.1.2 Transient Calculations with SIMULATE-3K

The transient core response to the REA event is analyzed with SIMULATE-3K. The transient simulation involves two calculations: conservatively addressing parameter uncertainties, and final simulation of the transient.

Conservatism is applied to key nuclear parameters in SIMULATE-3K to produce a conservative transient response from the code. Conservative factors are applied to the delayed neutron fraction, fuel temperature coefficient (FTC), MTC, and the worth for the ejected CRA and the inserted CRAs after reactor trip. These parameters are adjusted to account for the uncertainty determined for their calculation in SIMULATE-3K. This uncertainty is characterized by the NRFs previously determined for SIMULATE5 (Reference 8.2.7) and demonstrated to be applicable to SIMULATE-3K.

The conservative factors are numerical multipliers which are used to adjust the nuclear parameters by a desired conservative factor, where the conservative value is a reference value determined from SIMULATE-3K for a particular parameter, plus or minus the applicable NRF. Conservative factors are applied to case-specific key nuclear parameters that vary with time in life and initial conditions before the event.

For the DTC, CRA worth, and delayed neutron fraction, a separate multiplier is applied which reflects the relative uncertainty from Table 5-1. To conservatively incorporate uncertainties for the MTC, {{

}}2(a),(c)

{{

}}2(a),(c)

Once the nuclear parameter uncertainties have been incorporated into the input file, the final transient calculation is performed. For each statepoint identified as part of the scope, a case is run for each regulating group. The process for creating the input is as follows:

- The regulating groups are set at the PDIL, unless an un-ejected regulating CRA is identified as the WRSO.
- The axial power shape is chosen such that the axial offset is at the highest allowable value.
- {{

}}2(a),(c)

5.2.2 Analysis Assumptions and Parameter Uncertainties for Core Response

5.2.2.1 Control Rod Assembly Position

The regulating groups of CRAs are placed at the appropriate PDIL. This assumption will maximize the worth of the ejected CRA. The shutdown bank is assumed to be at the all rods out position. Uncertainty for the CRA position is found in Table 5-1.

5.2.2.2 Worst Rod Stuck Out

REA is analyzed with the WRSO. This assumes that the highest worth CRA remains stuck out of the core after the trip. The WRSO is determined for each fuel burnup and power level that is analyzed, and is chosen to be in the same quadrant as the ejected CRA. The assumption of a WRSO covers the potential for an ejected CRA to damage a nearby CRDM.

5.2.2.3 Input Parameters and Uncertainty Treatment

5.2.2.3.1 Ejected Rod Time

The time to eject the CRA from the core is defined by Equation 5-1.

$$Rod\ Ejection\ Time = \sqrt{\frac{\left(2 \cdot distance(cm)\right)}{acceleration\left(\frac{cm}{s^2}\right)}} \qquad \text{Equation 5-1}$$

The acceleration is calculated based on the CRA cross-sectional area and weight of the CRA and control rod driveshaft. The distance is the depth in the core that the CRA is inserted.

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.

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

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 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
$F_{\Delta H}$ engineering uncertainty		VIPRE-01
$F_{\Delta H}$ pin peaking nuclear reliability factor	}}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.
- A reduced ejected CRA worth that raises the power quickly to just below both the high power and high power rate trip limits is used through the point kinetics model within NRELAP5, and reactivity feedback mechanisms are used to hold the power at this level. This delays the trip until the transient is terminated by high RCS pressure. These cases do not have an upstream SIMULATE-3K calculation.

For calculations using the SIMULATE-3K power response, the power forcing functions from the SIMULATE-3K analysis are converted from percent power into units of MW for input into the NRELAP5 calculations.

5.3.1.1 Minimum Critical Heat Flux Ratio

The cases that typically provide the most limiting MCHFR results are those where the static ejected CRA worth is close to or in excess of one dollar. These are the cases analyzed with SIMULATE-3K, generally at powers where the CRA is deeper in the core.

Parameters with uncertainties and/or biases such as total system flow, inlet temperature, and outlet pressure that are used by the downstream VIPRE-01 calculations are addressed within the NRELAP5 system calculations.

Consideration for conservative system conditions in MCHFR analysis includes

- maximized net RCS heat input; this is performed by maximizing the difference between reactor power and heat removal through the steam generator.
- high initial RCS temperature; this forces the liquid temperature closer to saturation, which increases the rate at which vapor, and thus pressure, is generated.
- Variable (high and low) core pressure: the flow will be subject to a sensitivity study of both increased and decreased pressure in the core. This sensitivity study is required for rod ejection due to the unique nature of the rapid power change and possible impacts on core flow.

- high reactor power before reactor trips; this requires starting at a high power or sustaining a large power run-up, and is related to a large ejected CRA worth and low Doppler and moderator feedback.
- high RCS pressurization rate; this is caused by high power and high pressurizer level.

5.3.1.2 Reactor Coolant System Pressurization

The cases that generate the highest pressures are those following the second scenario described above; operating at a power just below the high-power reactor trip limits until reactor trip on high pressure.

Considerations for conservative system conditions in peak pressure analysis include

- maximized net RCS heat input during the transient; this is performed by maximizing the difference between reactor power and heat removal through the steam generator.
- low initial pressure and high initial RCS temperature; this forces the liquid temperature closer to saturation, which increases the rate at which vapor, and thus pressure, is generated.
- low inlet flow; the flow is reduced by a pressure surge arising from within the core.
- high reactor power prior to reactor trip; this requires starting at a high power or sustaining a large power run-up, and is related to a large ejected CRA worth and low Doppler and moderator feedback.
- high RCS pressurization rate; this is caused by high power and high pressurizer level.
- delayed reactor trip and lower reactor trip worth.
- unavailability of automatic pressure-limiting systems, including pressurizer spray, pressurizer heater control, RPV volume control, and feedwater and steam pressure control.
- delay of the high-steam superheat reactor trip signal; reactor trip on high pressure is more conservative, and this can be done by increasing the steam pressure.

5.3.2 Analysis Assumptions and Parameter Treatment for System Response

5.3.2.1 Pressure Relief

No pressure reduction is assumed. Reference 8.2.2 states that no credit should be taken for any possible pressure reduction because of the failure of the CRDM or CRDM housing.

5.3.2.2 Core Power

Power is biased high to account for the calorimetric uncertainty (Table 5-1). This calorimetric uncertainty is applied for the HFP cases by increasing the SIMULATE-3K core power response by a factor of 1.02.

5.3.2.3 Direct Moderator and Cladding Heating

Direct moderator and cladding heating is modeled in NRELAP5 calculations. Reference 8.2.2 states that prompt heat generation in the coolant should be considered for pressure surge calculations.

5.3.2.4 Core Inlet Temperature

Core inlet temperature is assumed to be constant. High initial temperature is conservative for both overpressure and MCHFR (see Sections 5.2.1.1 and 5.2.1.2).

5.3.2.5 Core Flow

Low core flow is conservative for both overpressure and MCHFR calculations (see Sections 5.2.1.1 and 5.2.1.2).

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 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.

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5.4.2 Analysis Assumptions and Parameter Treatment for Subchannel Response

5.4.2.1 Radial Power Distribution

The radial power distribution to be used for the subchannel REA evaluations is a case-specific conservative artificial distribution based on the highest peaked $F_{\Delta H}$ rod at the time of peak neutron power as predicted in the SIMULATE-3K analysis. This condition will occur after the ejected CRA is fully out of the core. In addition, the $F_{\Delta H}$ engineering uncertainty and the pin peaking nuclear reliability factor are applied to the highest peaked $F_{\Delta H}$ rod. The uncertainties associated with $F_{\Delta H}$ are given in Table 5-1 and are combined using the root-sum-squared method similar to that discussed in Section 3.10.7 of Reference 8.2.11. The radial power distribution slope described in Section 3.10.6 of Reference 8.2.11 is used to determine the REA-specific normalized radial power distribution for use in VIPRE-01. In summary, the process for each case is to (i) determine the peak $F_{\Delta H}$ rod (ii) apply uncertainty to that rod only (iii) calculate a normalized power shape for both fully-detailed rods and lumped rods (iv) utilize artificial shape in VIPRE-01 simulation of the case.

The conservative nature of this modeling is described in Section 6.4.2.5. Additionally, as described in Section 6.4.2 of Reference 8.2.11, the radial power distribution more than a few rows removed from the hot subchannel has a negligible impact on the MCHFR results. Analysis of different power distributions of the NuScale core demonstrate that rod powers a few rod rows beyond the hot rod or channel have a negligible impact on the MCHFR.

5.4.2.2 Axial Power Distribution

The axial power distribution to be used will be a normalized representation of the SIMULATE-3K assembly-average axial power at time of maximum core neutron power for the assembly containing the highest peak $F_{\Delta H}$ rod.

5.4.2.3 Core Inlet Flow Distribution

The inlet flow distribution for subchannel analyses is described in Reference 8.2.11. For REA calculations, the limiting inlet flow fraction is applied to the assembly containing the rod with the highest $F_{\Delta H}$ as described above.

5.4.2.4 Fuel Conductivity and Gap Conductance

Large fuel conductivity and gap conductance values are assumed. Sensitivity studies show that high values are more conservative for REA CHF calculations. Section 6.4.2.6 discusses the effect of a wide range of gap conductance values on MCHFR.

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. 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.

The nodal volume is calculated from the cross-sectional area of the fuel pellet and nodal height. By considering a single node in the core, and skewing the power deposited by the pin peaking factor F_Q, including the uncertainty in Table 5-1, the calculation will maximize the energy deposited in the node, and therefore maximize the temperature increase. The number of nodes is calculated by multiplying the number of axial nodes in the SIMULATE-3K analysis by the number of fuel rods in the core.

5.5.2 Radial Average Fuel Enthalpy

For the peak radial average fuel enthalpy, a similar adiabatic calculation is used with an adiabatic heatup assumption. First, the initial maximum nodal fuel enthalpy is calculated using the following equation:

$$h_i = \frac{C_p * T_{f,max}}{\rho_f} \qquad \text{Equation 5-3}$$

where,

h_i = maximum initial radial average fuel enthalpy,

$T_{f,max}$ = maximum pre-transient fuel centerline temperature, and

ρ_f = fuel density

Next, the integrated energy is converted to an enthalpy increase, taking into consideration fuel geometry and power peaking factors. The following equation defines the conservative radial average fuel enthalpy increase:

$$\Delta h = \frac{E_T * F_{q,max}}{V_{node} * \rho_f * n_{nodes}}$$ Equation 5-4

where,

Δh = radial average fuel enthalpy increase

The acceptance criteria use a combination of both enthalpy rise and peak enthalpy during the transient. Using h_i as the bounding starting value, and adding the calculated Δh to this value would provide a bounding final enthalpy for the fuel. If the calculated enthalpy rise and peak enthalpy values, using the conservatism within this calculation, are below the acceptance criteria, then it is determined that all enthalpy-based acceptance criteria are met.

5.5.3 Cladding Oxidation-Based Pellet-Cladding Mechanical Interaction Failure

Using the method discussed in Section 5.5.2, the peak radial average fuel enthalpy is compared to the limitation for a given oxide thickness or more precisely, the ratio of the maximum oxide to wall thickness. The NuScale maximum oxide to wall thickness is calculated to be 0.0588. The allowed transient enthalpy rise in cal/gm is given in the below figure from SRP 4.2 Appendix B. Using a bounding value of 0.08 for the oxide to wall thickness, the transient enthalpy rise is limited to 75 cal/g. Using 0.08 for the oxide to wall thickness instead of the evaluated 0.0588 value, results in adding significant margin to the limit. In addition, the limiting enthalpy rise conditions are more likely to occur at the BOC and MOC time period, where the greatest oxide thickness occurs at EOC. Therefore, applying the maximum oxide thickness at all times in life is an added conservatism.

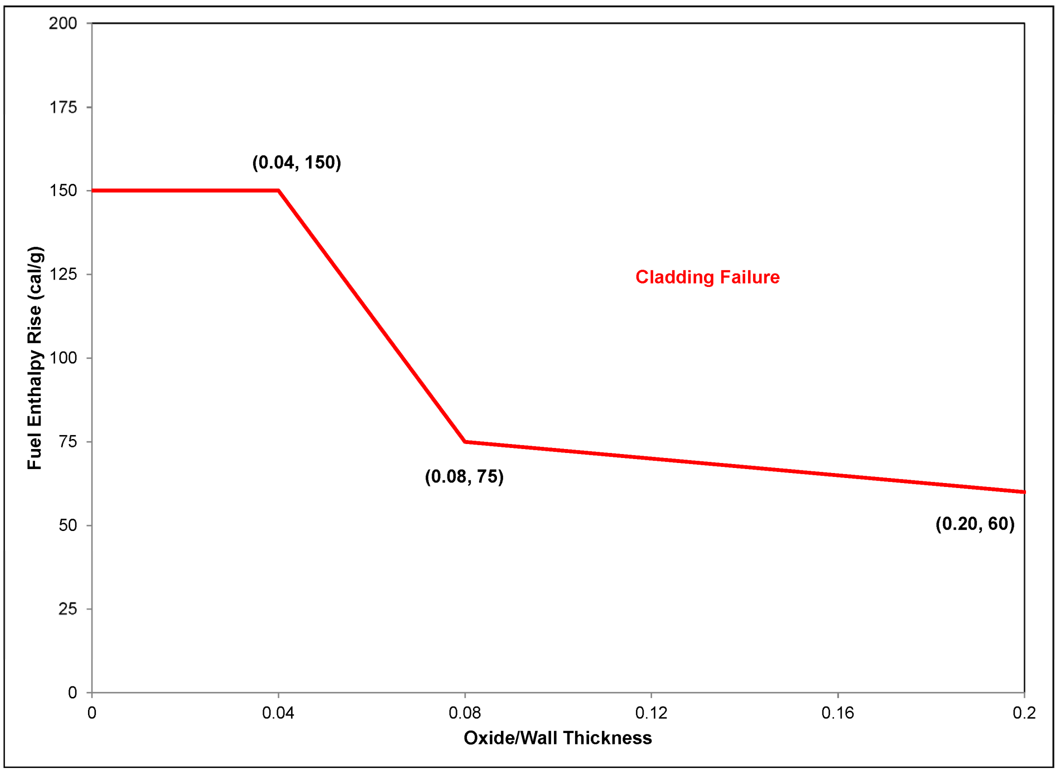

Figure 5-2 Pressurized water reactor pellet-cladding mechanical interaction fuel cladding failure criteria

5.6 Radiological Assessment

An accident radiological calculation is not performed because no fuel failures are predicted.

6.0 Sample Rod Ejection Accident Analysis and Sensitivity Results for the NuScale Design

For each power level and time in life, two sample REA calculations were performed. The first case analyzed an ejection of the CRA of the inner regulating group, CRA 1 in Figure 5-1. The second case analyzed an ejection of a CRA in the outer regulating group, CRA 5 in Figure 5-1. Because these two cases can vary significantly in terms of the ejected CRA worth and ensuing power response, Table 6-2 and Table 6-3 provide only the most limiting of the respective results from both calculation cases.

6.1 Rod Ejection Accident Sample Analysis System Pressure Response Results

The nominal conditions for each of the power levels evaluated for the REA is given in the Table 6-1 below.

Table 6-1 Conditions analyzed for sample calculations

Power Level (%)	0	10	25	45	50	55	60	70	80	100
Time in life	BOC, EOC	EOC	EOC	EOC	BOC, MOC, EOC	EOC	EOC	BOC, MOC, EOC	BOC, EOC	BOC, EOC
PDIL outer group (steps withdrawn)	140	140	140	140	140	140	140	140	140	170
PDIL inner group (steps withdrawn)	125	125	125	125	125	125	125	125	140	170
Core average temperature (°F)	425.0	500.0	543.3	543.3	543.3	543.3	543.3	543.3	543.3	543.3
System flow (kg/s)	29.3	237.1	339.8	426.1	443.7	460.7	477.2	507.1	535.3	587.0

6.2 Rod Ejection Accident Sample Analysis Fuel Response Results

The results of the REA sample evaluation are given below in Table 6-2 and Table 6-3. The SIMULATE-3K code produced the ejected rod worth, β_{eff}, MTC, and FTC values. Each of these values is biased to a conservative value based on the method discussion in Section 5. The peak power and transient F_Q and $F_{\Delta H}$ are outputs of the SIMULATE-3K calculation. The maximum enthalpy rise (Δcal/g) in the hot node, the maximum total enthalpy (cal/gm) in the hot node and the maximum fuel centerline temperature are calculated using the conservative adiabatic fuel heat-up model.

The MCHFR results were first screened using the NRELAP5 code, and those that were most likely to be the limiting conditions using the VIPRE-01 subchannel code were evaluated. Both the NRELAP5 MCHFR and VIPRE-01 MCHFR results are presented. The VIPRE-01 CHF analytical limit, using the NSP2 correlation, is 1.262 and the VIPRE-01 CHF analytical limit, using the NSP4 correlation is 1.284 (Reference 8.2.11). Criteria Limits for the Table 6-2 and 6-3 results are included in Table 7-1.

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.

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

Table 6-3 Sample results for rod ejection accident analysis, end of cycle, both regulating groups

Parameter	EOC, 0% Power	EOC, 10% Power	EOC, 25% Power	EOC, 45% Power	EOC, 50% Power	EOC, 55% Power	EOC, 60% Power	EOC, 70% Power	EOC, 80% Power	EOC, 102% Power
Ejected rod worth ($)	1.048	0.967	1.008	0.992	0.984	0.977	0.965	0.938	0.717	0.222
MTC (pcm/°F)	{{									}}$^{2(a),(c),ECI}$
FTC (pcm/°F)	-1.38	-1.38	-1.38	-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)	75	195	521	642	649	661	649	614	262	127
Maximum Δcal/g, hot node	18.1	18.1	19.4	23.5	23.7	24.6	25.0	25.9	25.4	18.5
Maximum cal/g, hot node	34.6	43.0	52.5	65.1	67.2	70.1	72.4	76.9	79.8	74.6
Maximum fuel centerline temperature (°F)	890	1106	1350	1676	1730	1802	1862	1978	2053	1920
NRELAP5 MCHFR (-)	{{									}}$^{2(a),(c),ECI}$
VIPRE-01 MCHFR (-)	{{						}}$^{2(a),(c),ECI}$	1.469	{{	}}$^{2(a),(c),ECI}$
Predicted rod failures (%)	0	0	0	0	0	0	0	0	0	0

Figure 6-1 shows an example of the power response at 55 percent and EOC, which is the highest power case of those analyzed. The large CRA worth, which is effectively a prompt critical reactivity insertion, results in a rapid power increase. This power increase is quickly turned around by the negative MTC and DTC feedback. The reactor trip signal is given early in the transient, as soon as the two operating detectors show a 15 percent power increase, and a delay of two seconds is assumed. After the large, narrow pulse, with a pulse width at half height of 0.12 seconds, a nearly steady state power of around 56 percent is reached until the CRAs start moving.

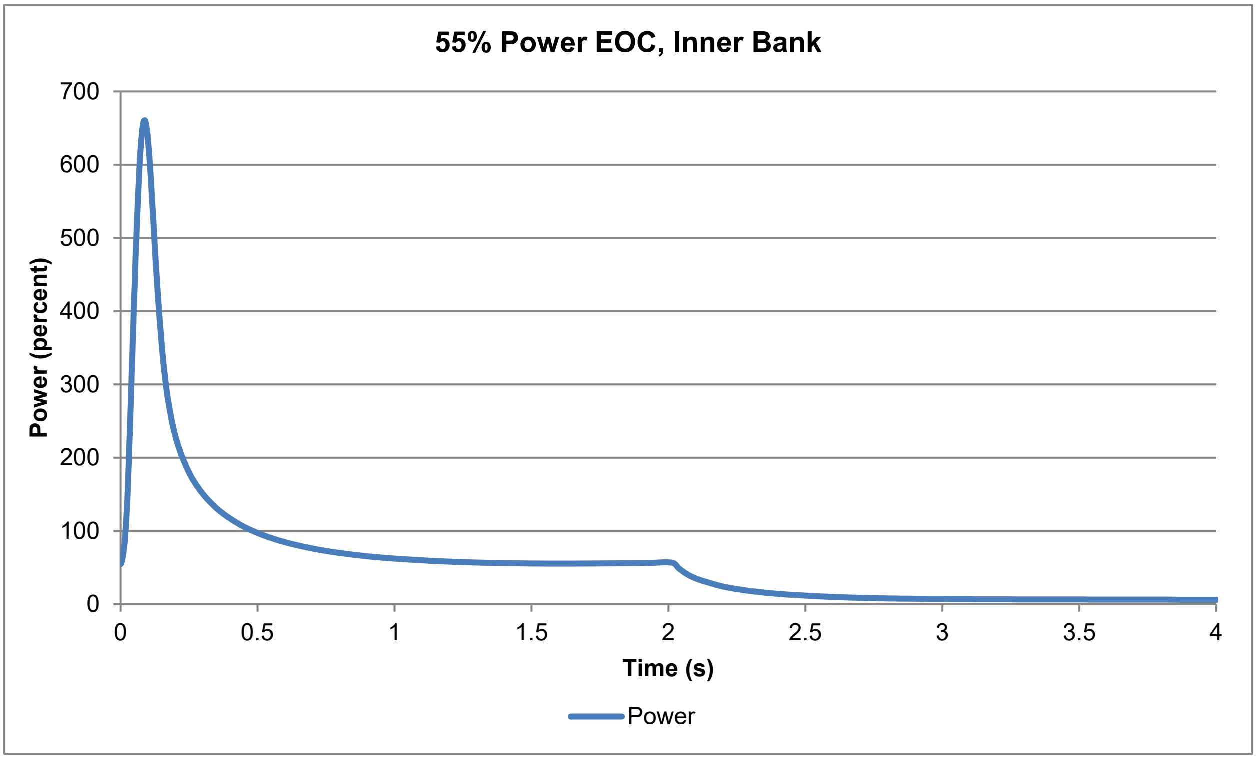

Figure 6-1 Power response at 55 percent power, end of cycle

In comparison, Figure 6-2 shows an example of the power response of an REA occurring at 100 percent and BOC. At these conditions, the low ejected worth results in a power response of smaller magnitude compared to the prompt response in Figure 6-1. The long term power comes to a new equilibrium steady state power around 106 percent. These conditions are not sufficient to violate CHF, fuel enthalpy, or fuel temperature, and thus are not analyzed against these failure criteria as they are bounded by HFP EOC cases that do reach the reactor trip limits.

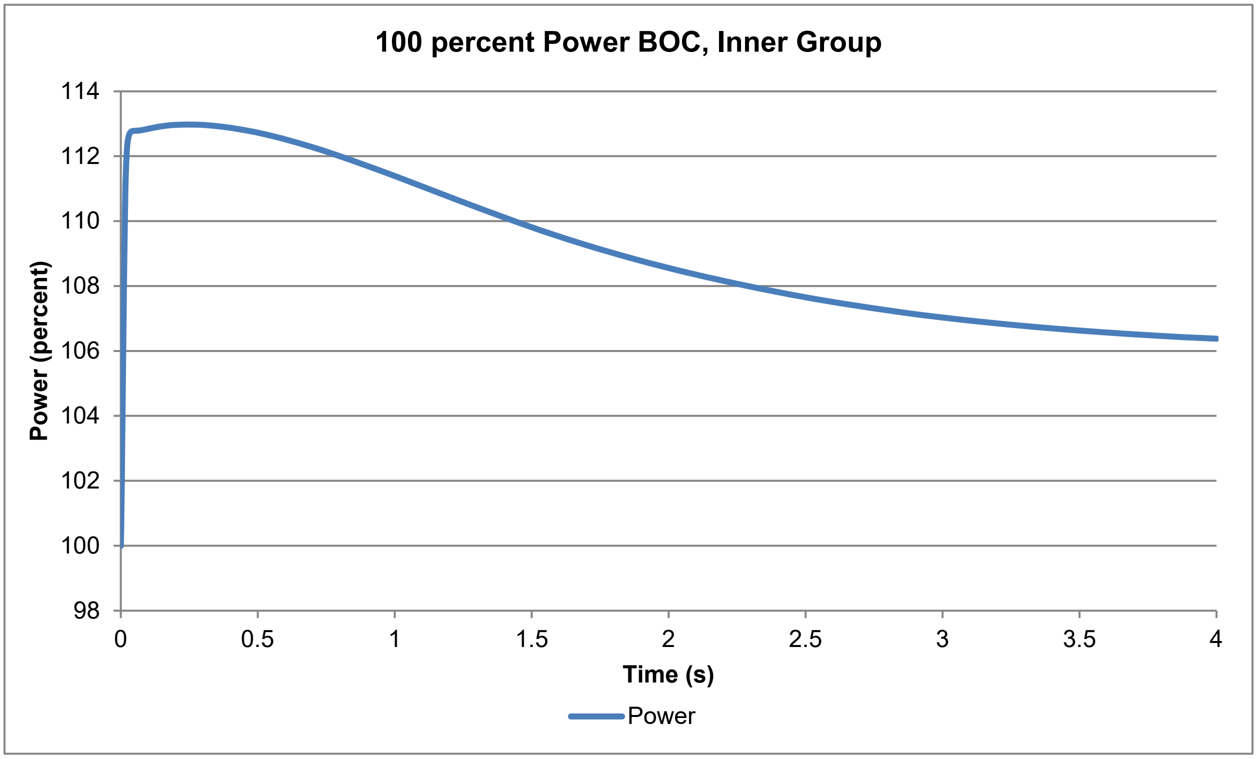

Figure 6-2 Power response at 100 percent power, beginning of cycle

6.3 Rod Ejection Accident Sample Analysis System Pressure Response Results

Figure 6-3 provides the power response for the peak RCS pressure evaluation. Figure 6-4 provides the peak RCS pressure response with this power forcing function. This calculation, as noted in the NRELAP5 methodology presented in Section 5.3, uses reactivity insertion and feedback inputs that allow the reactor power to jump to a level that is just below the trip setpoints for high reactor power and high power rate. The power is then held at this level until the reactor trip on reactor pressure is reached. The peak pressure reached during the REA is 2076 psia.

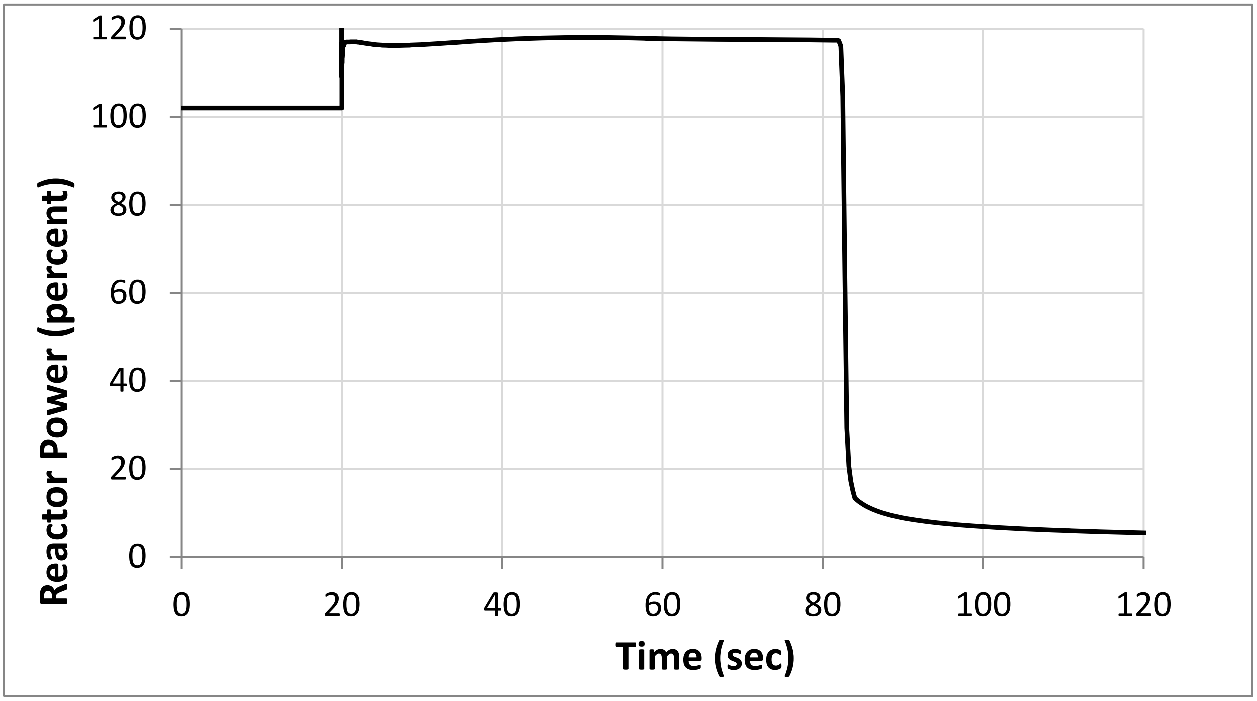

Figure 6-3 Power response for peak reactor coolant system pressure evaluation

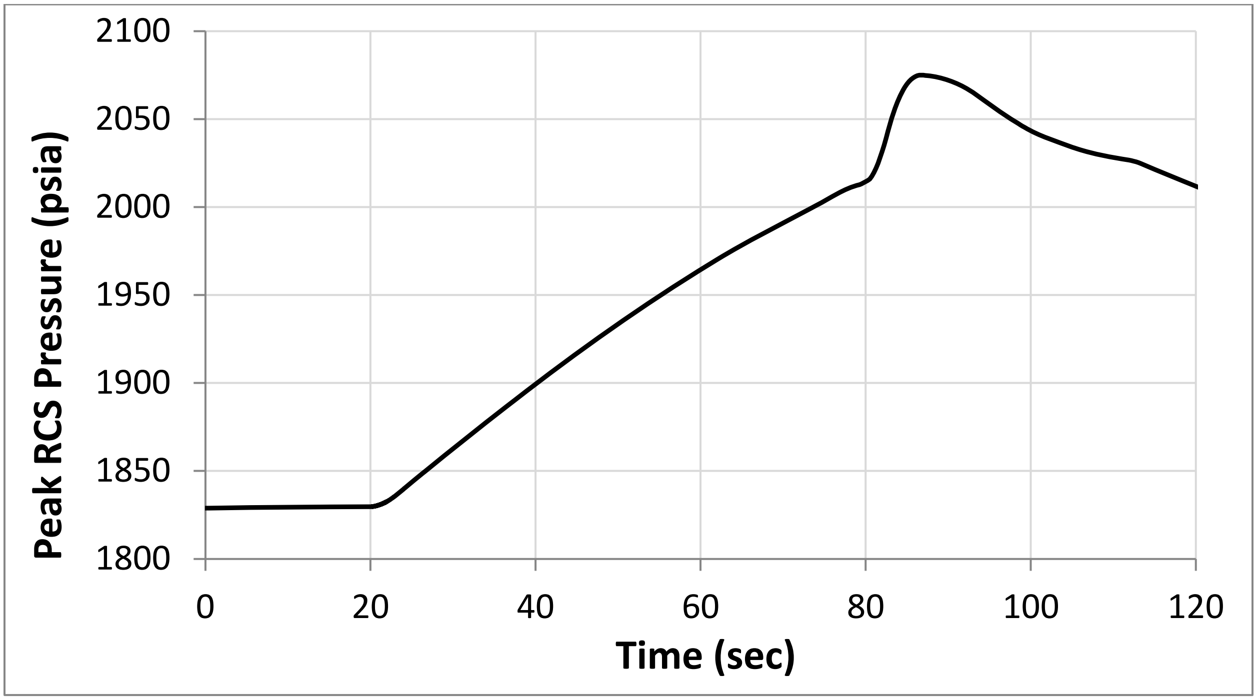

Figure 6-4 Pressure response for peak reactor coolant system pressure evaluation

6.4 Rod Ejection Accident Sensitivity Analysis Results

6.4.1 NRELAP5 Minimum Critical Heat Flux Ratio Impacts

Table 6-4 provides an evaluation of sensitivity calculations performed for the MCHFR in NRELAP5. The data shows the comparative effect on the MCHFR in terms of a percent difference from a nominal example case, based on the EOC 50 percent SIMULATE-3K core response.

Table 6-4 NRELAP5 MCHFR impacts from sensitivity evaluation

Parameter	Change	MCHFR Impact
RCS average temperature	T_{avg} +10°F	{{
Loss of offsite power	Loss of offsite power initiated concurrent with REA	
RCS Flow	Minimum design flow at 50% power	}}2(a),(c),ECI

6.4.2 VIPRE-01 Sensitivities

6.4.2.1 Computational Time Steps

Figure 6-5 provides a comparison between the time step size and power forcing functions used by VIPRE-01 and NRELAP5. VIPRE-01 assumes a time step of {{ }}2(a),(c) seconds, and the markers on the VIPRE-01 trendline are the actual VIPRE-01 time steps; VIPRE-01 linearly interpolates the power between these points.

{{

}}2(a),(c)

Figure 6-5 Time step effect on power forcing function

6.4.2.2 Code Axial Node Lengths

Figure 6-6 provides a comparison of various axial nodalizations used in VIPRE-01 compared to the resulting CHF value. The largest difference in the MCHFR from the nodalization used in the VIPRE-01 basemodel is {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 6-6 Effect of axial node size (inches) on critical heat flux

6.4.2.3 Two-Phase Flow Correlation Options

Figure 6-7 provides a comparison of the profile-fit model (EPRI) against the non-profile fit subcooled void model (HOMO). This provides additional evidence for robustness of the time step size used and any potential violations of the Courant limit. The MCHFR occurs at the same time step, and all time steps are within {{ }}$^{2(a),(c)}$ in CHFR.

{{

}}$^{2(a),(c)}$

Figure 6-7 Effect of VIPRE-01 two-phase flow model options on critical heat flux

6.4.2.4 Numerical Solution Damping Factors

Figure 6-8 shows a comparison of damping factors used in solving the VIPRE-01 numerical solution. {{

}}2(a),(c)

{{

}}2(a),(c)

Figure 6-8 Effect of VIPRE-01 damping factors on critical heat flux

6.4.2.5 Radial Power Distribution

Figure 6-9 provides the artificial radial power distribution used in the VIPRE-01 analysis, while Figure 6-10 provides the hot assembly radial power distribution from the limiting statepoint at time of peak power. Figures 6-11 and 6-12, cases 'Actual-1' and 'Acutal-2' respectively, are modified hot assembly radial power distributions that place the hot channel in potentially limiting locations. These modified power distributions are based on the power distribution shown in Figure 6-10, applying the $F_{\Delta H}$ uncertainty to the limiting rod. Figure 6-13 shows the comparison of the CHF behavior for these three power distributions when using the 51 channel model that uses fully detailed channels for the center assembly. This shows that the radial power distribution used in the VIPRE-01 analysis, Figure 6-9, is bounding.

{{

}}2(a),(c)

Figure 6-9 Radial power distribution for VIPRE-01 51 channel model, 70 percent power, end of cycle (Artificial)

{{

}}2(a),(c)

Figure 6-10 Radial power profile values for hot assembly at peak power

{{

}}2(a),(c)

Figure 6-11 Eighth-assembly radial power profile for VIPRE-01, peak rod on diagonal (Actual-1)

{{

}}2(a),(c)

Figure 6-12 Eighth-assembly radial power profile for VIPRE-01, peak rod near center (Acutal-2)

{{

}}2(a),(c)

Figure 6-13 Radial power profile effects on critical heat flux response

6.4.2.6 Fuel Rod Gap Conductance

Sensitivity calculations were performed to analyze the impact of applying various uncertainties or input options. Figure 6-14 below shows the comparison of high and low heat transfer inputs, specifically fuel rod gap conductance values of {{ }}2(a),(c) BTU/hr-ft^2-°F and the effect on CHF. This trend shows that the high heat transfer is limiting for the MCHFR.

{{

}}2(a),(c)

Figure 6-14 Effect of heat transfer inputs on critical heat flux

6.4.2.7 Reactor Coolant System Pressure

The effect of pressure on CHF involves the physical properties of the water coolant and the inlet subcooling effect. If subcooling is removed as a contributing factor (i.e. inlet subcooling is held constant with varying pressure) then changes in water properties with varying pressure lead to a negative trend of CHF versus pressure. The latent heat of vaporization of water has a negative trend with pressure, which is the primary driver of the negative trend in CHF versus pressure, because liquid-to-vapor phase conversion requires more enthalpy as pressure decreases. The specific vapor volume has an exponential relationship with pressure that is relatively flat above 3.0 to 4.0 MPa, but increases rapidly below this point. This increase in vapor volume at low pressures leads to increased vapor crowding on the surface of the heated rods and a subsequent decrease in heat transfer capability, resulting in lower CHF. These two competing effects are responsible for the change from a negative trend in CHF versus pressure to a positive one below 3.0 to 4.0 MPa. This trend is demonstrated by numerous CHF tests of various designs at multiple testing facilities.

When the subcooling effect is included, which is more appropriate for non-LOCA transient event calculations with VIPRE-01, the trends discussed above do not necessarily hold true. In traditional PWRs, pressure uncertainties are negatively applied (i.e. uncertainty is subtracted from best estimate value). This practice is based on the sensitivity of CHF to pressure seen historically in PWRs. The NPM operates in a different manner than traditional PWRs in that it does not rely on forced circulation via reactor coolant pumps to cool the core, but instead relies upon natural circulation. Relying on natural circulation results in a much lower mass flux (coolant flow) than is experienced in traditional PWR designs. The subcooling effect is influenced greatly by coolant flow in a reactor for a given amount of power. As mass flux increases the subcooling effect grows stronger due to decreasing enthalpy rise, leading to decreasing thermodynamic quality values and higher CHF. At high flows the subcooling effect is dominant and allows for a greater power capacity as pressure increases. {{

}}2(a),(c)

7.0 Summary and Conclusions

This report described the methodology for the evaluation of an REA in the NPM. This methodology was developed to demonstrate compliance with the requirements of GDC 13 and GDC 28, and the acceptance criteria and guidance in Regulatory Guide 1.77, SRP Sections 4.2 and 15.4.8, and the proposed guidance in the Clifford Letter. NuScale intends to use this methodology for REA analysis in support of the NuScale Design Certification Application and for future design work. The methodology presented is not generic for different core designs, therefore cycle-specific analysis must be performed for each core design.

The methodology described herein uses a variety of codes and methods. The three-dimensional neutronic behavior is analyzed using SIMULATE5 and SIMULATE-3K; the reactor system response is analyzed using NRELAP5; and the subchannel TH behavior and fuel response is analyzed using VIPRE-01. The software is validated for use to evaluate the REA. The fuel response is supplemented by the use of a bounding adiabatic heat-up calculation for the calculation of all transient fuel enthalpy and temperature increases during the REA.

This report includes the identification of important phenomena and input and specifies appropriate uncertainty treatment of the important input for a conservative evaluation. The methodology is discussed and demonstrated by the execution of sample problems and appropriate sensitivity analyses.

Section 6 of this report provides aggregate data from sample REA calculations. These results include a complete spectrum of initial conditions as well as relevant sensitivity evaluations. These results provide confirmation that the regulatory acceptance criteria outlined in Section 2.1 are achieved. The four main regulatory acceptance criteria that were demonstrated as being met are

- maximum RCS pressure. Results from the sample analysis using the NRELAP5 system code that evaluates the peak NPM pressure due to the power pulse from a worst-case rod ejection demonstrates that the maximum system pressure is well below the criteria of 120 percent of design or 2520 psia.
- fuel cladding failure. The adiabatic heat model demonstrates that transient enthalpy rise is well below the criteria for HZP, intermediate, and HFP conditions considering fuel rod differential pressure at HZP and oxidation due to corrosion with a wide margin. The adiabatic model also predicts that the peak fuel centerline temperature is well below the incipient melting point. For the limiting critical heat flux (CHF) case at 70 percent full power, VIPRE-01 predicts ample margin to CHF.
- core coolability. The results associated with core coolability of peak radial average fuel enthalpy are met with ample margin. Incipient fuel melt is precluded by a wide margin.

- fission product inventory. The fission product inventory effects are not applicable to the NuScale design, because no fuel rod failure is allowed and the highest rod differential pressure is assumed for the HZP requirement of transient fuel enthalpy rise.

The sample REA analysis quantitative results compared to the regulatory acceptance criteria are summarized below in Table 7-1.

Table 7-1 Summary of NuScale criteria and sample evaluation results

Parameter	Criteria	Sample Evaluation Results – Limiting Case
Maximum RCS pressure	2520 psia	2076 psia
HZP fuel cladding failure	100 cal/g	34.6 cal/g
FGR effect on cladding differential pressure	2.3.4 (item 2)	N/A
CHF fuel cladding failure	MCHFR > 1.262	1.47
Cladding oxidation-based PCMI failure	< 75 Δcal/g	28.7 Δcal/g
Cladding excess hydrogen-based PCMI failure	2.3.2 (item 3)	N/A
Incipient fuel melting cladding failure	< [ ]	2162 °F
Peak radial average fuel enthalpy for core coolability	230 cal/g	84.0 cal/g
Fuel melting for core cooling	< [ ]	2162°F
Fission product inventory	2.3.4	N/A

8.0 References

8.1 Source Documents

8.1.1 American Society of Mechanical Engineers, *Quality Assurance Program Requirements for Nuclear Facility Applications*, ASMENQA-1-2008, ASME NQA-1a-2009 Addenda, as endorsed by Regulatory Guide 1.28, Revision 4.

8.1.2 *U.S. Code of Federal Regulations,* "Domestic Licensing of Production and Utilization Facilities," Part 50, Title 10, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," (10 CFR 50 Appendix B).

8.1.3 NuScale Topical Report, "NuScale Topical Report: Quality Assurance Program Description for the NuScale Power Plant," NP-TR-1010-859-NP-A, Revision 3.

8.2 Referenced Documents

8.2.1 *U.S. Code of Federal Regulations,* Part 50, Title 10, "Domestic Licensing of Production and Utilization Facilities" (10 CFR 50).

8.2.2 U.S. Atomic Energy Commission, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," Regulatory Guide (RG) 1.77, May 1974.

8.2.3 U.S. Nuclear Regulatory Commission, Standard Review Plan, "Fuel System Design," NUREG-0800, Section 4.2, Rev. 3, March 2007.

8.2.4 U.S. Nuclear Regulatory Commission, Standard Review Plan, "Spectrum of Rod Ejection Accidents (PWR)," NUREG-0800, Section 15.4.8, Rev. 3, March 2007.

8.2.5 Letter from Paul M. Clifford to Timothy J. McGinty, "Technical and Regulatory Basis for the Reactivity-Initiated Accident Acceptance Criteria and Guidance, Revision 1," March 16, 2015.

8.2.6 NuScale Topical Report, "NuScale Power Critical Heat Flux Correlations," TR-0116-21012-P-A, Revision 1, dated December 2018.

8.2.7 NuScale Topical Report, "Nuclear Analysis Codes and Methods Qualification," TR-0616-48793-P-A, Revision 1, dated November 2018.

8.2.8 NuScale Topical Report, "Applicability of AREVA Fuel Methodology for the NuScale Design," TR-0116-20825-P-A, Revision 1, dated June 2016.

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,” TR-0516-49416 Revision 1, August 2017.

8.2.11 NuScale Topical Report, “Subchannel Analysis Methodology,” TR-0915-17564-P-A, Revision 2, February 2019.

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.

8.2.23 H. Finnemann, A. Galati. "NEACRP 3-D LWR Core Transient Benchmark Final Specifications," NEACRP-L-335 Revision 1. EOCD Nuclear Energy Agency, January 1992.

8.2.24 G. Grandi, "Effect of the Discretization and Neutronic Thermal Hydraulic Coupling on LWR Transients." Proceedings of NURETH-13, Kanazawa City, Japan, September 27-October 2, 2009.

8.2.25 LWR Core Reactivity Transients, SIMULATE-3K Models and Assessments, SSP-04/443 Revision 3. Studsvik Scandpower, July 2011.

8.2.26 U.S. Nuclear Regulatory Commission, "Phenomenon Identification and Ranking Tables (PIRTs) for Rod Ejection Accidents in Pressurized Water Reactors Containing High Burnup Fuel," NUREG/CR-6742 (LA-UR-99-6810), September 2001.

8.2.27 "Safety Evaluation Report on EPRI NP-2511-CCM VIPRE-01," May 1986.

Section C

RAI Number	eRAI Number	NuScale Letter Number
9306	9306	RAIO-0618-60285
9306	9306	RAIO-0119-64377 - Supplemental
9306	9306	RAIO-0219-64616 - Supplemental
9306	9306	RAIO-1019-67479 - Supplemental

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 2:

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

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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:

$$\Delta T = \frac{E_T * F_{Q,max}}{C_p * V_{node} * n_{nodes}}$$ 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

---

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

---

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.

NUSCALE POWER

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 <u>both</u> 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-

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

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

NRC Question No.: 15.04.08-8

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.

NuScale Power

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.

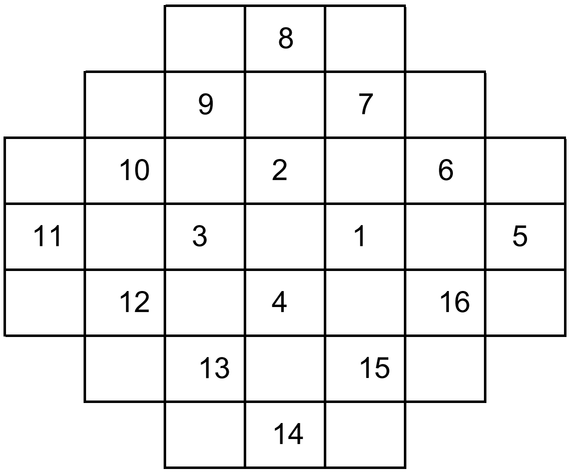

Figure 5-1 Control rod assembly layout for the NuScale Power Module

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.

RAIO-0119-64377

January 31, 2019

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 Supplemental 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 Power, LLC Response to NRC "Request for Additional Information No. 9306 (eRAI No.9306)," dated June 04, 2018
3. 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) supplemental response to the referenced NRC Request for Additional Information (RAI).

The Enclosures to this letter contain NuScale's supplemental response to the following RAI Questions from NRC eRAI No. 9306:

- 15.04.08-5
- 15.04.08-6
- 15.04.08-15
- 15.04.08-16

Enclosure 1 is the proprietary version of the NuScale Supplemental 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,

Carrie Fosaaen
Supervisor, Licensing
NuScale Power, LLC

Distribution: Gregory Cranston, NRC, OWFN-8H12
Samuel Lee, NRC, OWFN-8H12
Rani Franovich, NRC, OWFN-8H12

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

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

Enclosure 3: Affidavit of Thomas A. Bergman, AF-0119-64378

Enclosure 2:

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

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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:

<u>NuScale Supplement Response</u>

The original NuScale response as submitted in NuScale correspondence RAIO-0618-60285 and dated June 4, 2018, is augmented with the following information.

As described in Section 4.3, item D 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 are those that experience prompt, or near prompt, criticality due to the reactivity insertion. This example behavior is generic to all power transient initial conditions screened by NRELAP5 as being possibly limiting (see Section 6.2 of the topical report for more detail), as may be observed in the following Table 1, Figure 1, and Figure 2. As an example, the time of minimum critical heat flux ratio (MCHFR) for cases 'EOC 50' and 'EOC 70' are effectively the same. The peak power for 'EOC 50' is slightly higher, but occurs slightly slower. There is no case in which a reactor trip mitigates the consequences of the transient. Table 1 shows that for all cases, peak power and MCHFR occurred well before the control rods would have started to move, 2 seconds after a trip signal, should a trip signal have occurred.

Table 1. Summary of Example Cases Screened by NRELAP5

Case Name	Cycle Exposure (GWd/MT)	Initial Power (% Rated)	Peak Power (% Rated)	Time Peak Power (sec)	Time of MCHFR (sec)	MCHFR
4GW 50	4	50	185.5	0.0823	{{	
4GW 70	4	70	240.2	0.0780		
BOC 50	0	50	133.0	0.0930		
BOC 70	0	70	177.5	0.0701		
EOC 45	12.1	45	642.4	0.0928		
EOC 50	12.1	50	648.5	0.0917		
EOC 55	12.1	55	660.5	0.0890		
EOC 60	12.1	60	649.2	0.0856		
EOC 70	12.1	70	614.5	0.0837		
EOC 80	12.1	80	261.7	0.0762		}}2(a),(c),ECI

Figure 1 and Figure 2 illustrate the transient progression of power and MCHFR, respectively, for the cases listed in Table 1. The limiting values for both of these parameters occur very early in the transient.

{{

}}2(a),(c),ECI

Figure 1. Comparison of Input Core Power Forcing Functions

{{

}}2(a),(c),ECI

Figure 2. Comparison of Minimum CHF Ratio

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:

The original NuScale response as submitted in NuScale correspondence RAIO-0618-60285 and dated June 4, 2018, is augmented with the following information.

As described in Section 5.2.1.2, conservatism is applied to key nuclear parameters in SIMULATE-3K to produce a conservative transient response from the code. The conservatisms are also referred to as nuclear reliability factors (NRFs). Conservatism is applied to the effective delayed neutron fraction (β_{eff}), fuel temperature coefficient (FTC), moderator temperature coefficient (MTC), and control rod assembly (CRA) worth via the ‘KIN.MUL’ card in SIMULATE-3K.

For β_{eff}, the conservatism is applied as a {{

}}2(a),(c) The delayed neutron data is supplied by the cross-section (neutron data) library created by CASMO5 and input into the code.

{{

}}2(a),(c)

For the FTC, the Doppler feedback can be estimated as the product of the FTC and the change in fuel temperature with respect to the steady-state condition:

{{

}}2(a),(c)

{{

}}2(a),(c) and is applied in SIMULATE-3K to account for conservatism in the Doppler feedback. Since the NRF for FTC is a relative value, the multiplier is directly applied and no iterations are necessary.

For MTC, the SIMULATE-3K methodology is similar to FTC, but the NuScale NRF is an absolute value, so it is not directly applied as the multiplier. The multiplier must be iterated upon to determine a relative value corresponding to an adjusted MTC accounting for the application of the NRF in a conservative manner.

For CRA worth, the {{ }}2(a),(c) the multiplier must be iterated upon to determine a value corresponding to an adjusted rod worth accounting for the application of the NRF in a conservative manner.

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.: 15.04.08-15

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:

The original NuScale response as submitted in NuScale correspondence RAIO-0618-60285 and dated June 4, 2018, is augmented with the following information.

The rod ejection methodology is a cycle-specific approach to evaluate rod ejections for each core reload. As discussed in Section 5.4.2.1 of the topical report, the radial power distribution used in the minimum critical heat flux ratio (MCHFR) evaluation is a conservative artificial distribution contrived from the peaking results in the SIMULATE-3K analysis. In addition to the mentioned $F_{\Delta H}$ engineering uncertainty of {{ }}2(a),(c) applied to the peak rod, the uncertainty for the pin peaking nuclear reliability factor (NRF) of {{ }}2(a),(c) was incorporated. This additional pin peaking NRF is consistent with the steady-state uncertainty discussed in the Nuclear Analysis Codes and Methods Qualification Report (TR-0716-48793). Text was added to indicate the incorporation of the pin peaking NRF into the NuScale rod ejection accident methodology (Section 5.4.2.1 and Table 5-1 of TR-0716-50350) as indicated at the end of this response.

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.

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
$F_{\Delta H}$ engineering uncertainty		VIPRE-01
$F_{\Delta H}$ pin peaking nuclear reliability factor	}}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.

{{

}}2(a),(c)

5.4.2 Analysis Assumptions and Parameter Treatment for Subchannel Response

5.4.2.1 Radial Power Distribution

The radial power distribution to be used for the subchannel REA evaluations is a case-specific conservative artificial distribution based on the highest peaked $F_{\Delta H}$ rod at the time of peak neutron power as predicted in the SIMULATE-3K analysis. This condition will occur after the ejected CRA is fully out of the core. In addition, the $F_{\Delta H}$ engineering uncertainty ~~is~~and the pin peaking nuclear reliability factor are applied to the highest peaked $F_{\Delta H}$ rod. The uncertaintiesy associated with $F_{\Delta H}$ are~~is~~ given in Table 5-1 and are combined using the root-sum-squared method similar to that discussed in Section 3.10.7 of Reference 8.2.11. The radial power distribution slope described in Section 3.10.6 of Reference 8.2.11 is used to determine the REA-specific normalized radial power distribution for use in VIPRE-01. In summary, the process for each case is to (i) determine the peak $F_{\Delta H}$ rod (ii) apply uncertainty to that rod only (iii) calculate a normalized power shape for both fully-detailed rods and lumped rods (iv) utilize artificial shape in VIPRE-01 simulation of the case.

The conservative nature of this modeling is described in Section 6.4.2.5. Additionally, as described in Section 6.4.2 of Reference 8.2.11, the radial power distribution more than a few rows removed from the hot subchannel has a negligible impact on the MCHFR results. Analysis of different power distributions of the NuScale core demonstrate that rod powers a few rod rows beyond the hot rod or channel have a negligible impact on the MCHFR.

5.4.2.2 Axial Power Distribution

The axial power distribution to be used will be a normalized representation of the SIMULATE-3K assembly-average axial power at time of maximum core neutron power for the assembly containing the highest peak $F_{\Delta H}$ rod.

5.4.2.3 Core Inlet Flow Distribution

The inlet flow distribution for subchannel analyses is described in Reference 8.2.11. For REA calculations, the limiting inlet flow fraction is applied to the assembly containing the rod with the highest $F_{\Delta H}$ as described above.

5.4.2.4 Fuel Conductivity and Gap Conductance

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

eRAI No.: 9306

Date of RAI Issue: 04/04/2018

NRC Question No.: 15.04.08-16

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:

<u>NuScale Supplement Response</u>

The original NuScale response as submitted in NuScale correspondence RAIO-0618-60285 and dated June 4, 2018, is augmented with the following information.

For the rod ejection accident, the fuel is modeled in two different manners for the two different sets of fuel failure acceptance criteria, referred to as (1) critical heat flux (CHF) and (2) non-CHF related for the purposes of this response. The SIMULATE-3K (S3K) 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.

- **CHF:** The critical heat flux ratio is calculated in VIPRE-01 using the power pulse and power peaking as input. 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 code 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.

- **Non-CHF**: The adiabatic calculation described in Section 5.5 of the topical report, with conservative modeling and assumptions, utilizes the NRC-approved fuel performance code COPERNIC for the initial fuel temperature calculation. From this and other inputs, the various parameters are calculated and compared to the acceptance criteria. The initial fuel temperature is ensured to be bounding for a given fuel-design by conducting a fuel design-specific evaluation, similar to that performed for the subchannel analysis described in Section 4.4 of TR-0915-17564. Specifically, this methodology requires that the entire range of possible time-in-cycle parameters (i.e., exposure, uranium enrichment, gadolinium enrichment, gap conductance, and fuel density) are evaluated using the COPERNIC fuel performance code.

The S3K code is not directly relied upon to perform initial or maximum fuel pellet temperature calculations. S3K uses the fuel average temperature as the main feedback mechanism (92%) to calculate the Doppler feedback. S3K uses pre-calculated radial profiles that vary as a function of exposure and does not explicitly model the pellet rim. This use of S3K, in conjunction with the uncertainty treatment described in Section 5.2 of the topical report assures conservative fuel performance modeling, and is appropriate for calculating the power pulse and power peaking for use in the downstream analysis for rod ejection accidents.

Impact on DCA:

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

RAIO-0219-64616

February 21, 2019

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 Supplemental 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 Power, LLC Response to NRC "Request for Additional Information No. 9306 (eRAI No.9306)," dated June 04, 2018
3. 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) supplemental response to the referenced NRC Request for Additional Information (RAI).

The Enclosures to this letter contain NuScale's supplemental response to the following RAI Question from NRC eRAI No. 9306:

- 15.04.08-1

Enclosure 1 is the proprietary version of the NuScale Supplemental 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 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,

Carrie Fosaaen
Supervisor, Licensing
NuScale Power, LLC

Distribution: Gregory Cranston, NRC, OWFN-8H12
Samuel Lee, NRC, OWFN-8H12
Rani Franovich, NRC, OWFN-8H12

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

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

Enclosure 3: Affidavit of Thomas A. Bergman, AF-0219-64617

Enclosure 2:

NuScale Supplemental 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:

The original NuScale response as submitted in NuScale correspondence RAIO-0618-60285 and dated June 4, 2018, is augmented with the following information.

NuScale has performed a benchmark of the dynamic reactor response simulated by SIMULATE-3K (S3K) to the transient special power excursion reactor test III E-Core experiment (SPERT). This experiment performed by the Atomic Energy Commission (AEC) was a pressurized water nuclear research reactor that analyzed reactor kinetic behavior under conditions similar to commercial pressurized water reactors (Reference 1 and 2). The SPERT core resembled such reactor designs, but of a reduced size more closely resembling the NuScale core size. The fuel type, moderator, system pressure, and certain initial operating conditions considered for SPERT are also representative of NuScale as demonstrated in Table 1.

Table 1. Range of Applicability Comparison

Parameter	Units	SPERT	NuScale
Reactor Type	-	PWR	PWR
Fuel Material	-	Uranium dioxide	Uranium dioxide
UO2 Enrichment	w/o	4.8	≤4.95
Clad Material	-	Stainless Steel	Zircaloy Alloy (M5)
Active Fuel Length	in	38.3	78.74
Core Diameter	in	~26	~68
Rated Power	MWt	20	160
Rated Flow	kg/s	1,260	680
Design Core Exit Temp.	F	650	590
Design Pressure	psia	2,515	1,850

The original experiment included on the order of one hundred unique tests at five different sets of thermal-hydraulic statepoints, with varying initial static worths at each statepoint. For the purposes of this RAI supplement, one test from each statepoint that generally corresponds to the highest static worth for the statepoint is provided in tabulated and plotted format. Table 2 provides a summary and definition of the statepoint conditions of the selected cases.

Table 2. Summary of Selected Cases

Test #	Statepoint Condition	Initial Coolant Temp. (F)	Reactivity Insertion ($)
43	Cold Startup	78	1.210
70	Hot Startup	250	1.210
60	Hot Startup	500	1.230
81	Hot Standby	500	1.170
86	Full Power	500	1.170

Table 3 provides a tabulated comparison between SIMULATE-3K results and the experiment for the three key parameters of peak power, integrated energy, and reactivity compensation. Comparison plots for the selected cases are presented in Figure 1 through Figure 5. Due to the experimental values of the energy release to time of peak power and reactivity compensation at peak power being only approximate for hot standby and full power conditions (Tests #81 and #86), no comparison between SIMULATE-3K results and the experiment is performed for these parameters.

The following tables and figures for the selected comparisons of key parameters demonstrate that SIMULATE-3K compares to SPERT with generally excellent agreement; differences are within the experimental uncertainty (with few exceptions), and the major and minor phenomena are correctly predicted per the benchmark criteria defined in Reference 3. The most extreme difference in the benchmark is the peak power for test 81, which is {{ }}[2(a),(c)], as compared to the stated experimental uncertainty of ±15%. The magnitude in which {{ }}[2(a),(c)] of the stated uncertainty. Additionally, the experimental uncertainty of the initial reactivity insertion is between 0.03$ and 0.05$. Varying the initial reactivity insertion within the stated uncertainty is sufficient to be a possible explanation for the differences observed between the experiment and simulations. For these reasons, SIMULATE-3K exhibits no deficiencies in modeling the SPERT experiment and may be used with confidence in similar applications.

The SPERT peak power magnitudes are on the order of up to 3,000% rated power. For context, example NuScale peak power magnitudes presented in TR-0716-50350 are on the order of 600% of rated power and occur at the statepoints of medium power levels (~50% to ~80% of rated power). Thus, the example NuScale dynamic conditions are bounded by those of the experiment. Therefore, this benchmark provides justification that SIMULATE-3K can accurately model a rod ejection accident transient event and predict key reactivity and power-related parameters.

Table 3. Tabulated Results and Comparisons of Selected Cases

Test #	Peak Power (MW) [Exp. Uncertainty=±15%]			Integrated Energy (MW-sec) [Exp. Uncertainty=±17%]			Reactivity Compensation ($) [Exp. Uncertainty=±11%]		
	S3K	SPERT	% Diff	S3K	SPERT	% Diff	S3K	SPERT	% Diff
43	{{	280	{{	{{	6	{{	{{	0.22	{{
70		280			6.3			0.22	
60		410		}}2(a),(c)	8.5	}}2(a),(c)	}}2(a),(c)	0.24	}}2(a),(c)
81		330							
86	}}2(a),(c)	610	}}2(a),(c)						

{{

}}2(a),(c)

Figure 1. Test 43 SIMULATE-3K Comparison to SPERT

{{

}}[2(a),(c)]

Figure 2. Test 70 SIMULATE-3K Comparison to SPERT

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}}[2(a),(c)]

Figure 3. Test 60 SIMULATE-3K Comparison to SPERT

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}}2(a),(c)

Figure 4. Test 81 SIMULATE-3K Comparison to SPERT {{

}}2(a),(c)

Figure 5. Test 86 SIMULATE-3K Comparison to SPERT

References

1. U.S. Atomic Energy Commission, IDO-17281, “Reactivity Accident Test Results and Analyses for the SPERT III E-CORE - A Small, Oxide-Fueled, Pressurized Water Reactor”, March 1969, ADAMS Accession ML080320431
2. U.S. Atomic Energy Commission, IDO-17036, “SPERT III Reactor Facility” E-CORE Revision”, November 1965, ADAMS Accession ML080320408
3. U.S. Nuclear Regulatory Commission, “Transient and Accident Analysis Methods”, Regulatory Guide 1.203, December, 2005.

Impact on DCA:

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

RAIO-1019-67479

October 10, 2019

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 Supplemental 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 Power, LLC Response to NRC "Request for Additional Information No. 9306 (eRAI No.9306)," dated June 04, 2018
3. NuScale Topical Report, "Rod Ejection Accident Methodology," TR-0716-50350, Revision 0, dated December 2016
4. NuScale Power, LLC Supplemental Response to "NRC Request for Additional Information No. 9306 (eRAI No. 9306)" dated February 21, 2019

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

The Enclosure to this letter contains NuScale's supplemental response to the following RAI Question from NRC eRAI No. 9306:

- 15.04.08-1

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

If you have any questions on this response, please contact Matthew Presson at 541-452-7531 or at mpresson@nuscalepower.com.

Sincerely,

Michael Melton
Manager, Licensing
NuScale Power, LLC

Distribution: Gregory Cranston, NRC, OWFN-8H12
Samuel Lee, NRC, OWFN-8H12
Rani Franovich, NRC, OWFN-8H12
Michael Dudek, NRC, OWFN-8H12

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

Enclosure 1:

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

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvalis, Oregon 97330, Office: 541.360.0500, Fax: 541.207.3928
www.nuscalepower.com

Response to Request for Additional Information
Docket: PROJ0769

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:

The original NuScale response was submitted in NuScale correspondence RAIO-0618-60285 and was dated June 4, 2018. A supplement to this RAI response was submitted in NuScale correspondence RAIO-0219-64616, dated February 21, 2019, which detailed the results of a benchmark of the dynamic reactor response simulated by SIMULATE-3K (S3K) to the transient special power excursion reactor test III E-Core experiment (SPERT).

This supplement provides a mark-up to the Rod Ejection Accident Methodology Topical Report (TR-0716-50350), Section 3.2.1.4, which adds a summary of the NuScale SIMULATE-3K to the SPERT III benchmark results as indicated below.

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.

and the NEACRP control rod ejection problem computational benchmark (Reference 8.2.22).

The Studsvik SPERT III benchmark provides measured REA transient data for comparison to SIMULATE-3K. SPERT III was a pressurized water nuclear research reactor that analyzed reactor kinetic behavior under conditions similar to commercial reactors. The SPERT III core resembled a commercial reactor, but of a reduced size more closely resembling the NuScale core size. The fuel type (uranium dioxide), moderator, system pressure, and certain initial operating conditions considered for SPERT III are also representative of NuScale. This benchmark demonstrates the ability of SIMULATE-3K to model fast reactivity transients in a PWR core (Reference 8.2.22). Similarities between the NuScale design and the SPERT III core, and notably the small core size, demonstrate applicability and suitability for SIMUALTE-3K REA transient analysis of the NuScale core.

In addition to the Studsvik benchmarks aforementioned, NuScale has performed a benchmark of the dynamic reactor response simulated by SIMULATE-3K of the SPERT III experiment. The original experiment included on the order of one hundred unique tests at five different sets of thermal-hydraulic conditions, with varying initial static worths at each statepoint. One test from each condition set that generally corresponds to the highest static worth for the statepoint has been benchmarked. A comparison of key parameters demonstrates that SIMULATE-3K compares to SPERT with generally excellent agreement; differences are within the experimental uncertainty (with few exceptions), and the major and minor phenomena are correctly predicted.

The NEACRP control rod ejection problem is a computational benchmark that includes a reference solution provided by the PANTHER code, and SIMULATE-3K REA transient results are compared against the reference solution. In this benchmark, a rod ejection accident in a typical commercial PWR at HZP conditions is analyzed. The fuel type (uranium dioxide), moderator, system pressure, and certain initial operating conditions considered for NEACRP are also representative of NuScale. The capability of SIMULATE-3K to model reactivity insertions in the NEACRP benchmark analysis (Reference 8.2.24 and 8.2.25) demonstrates suitability of the code for reactivity transient applications, and specifically REA analysis applications.

The SPERT III and NEACRP benchmarks demonstrate the combined transient neutronic, TH, and fuel pin modeling capabilities of SIMULATE-3K. SIMULATE-3K results for maximum power pulse, time to peak power, inserted reactivity, energy release, and fuel centerline temperature were in excellent agreement with the results from the two benchmark problems. The SIMULATE-3K results for each of these benchmark problems establish the ability of the code to accurately model an REA transient event and predict key reactivity and power-related parameters.

3.2.2 System Response

The NRELAP5 code was developed based on the Idaho National Laboratory RELAP5-3D© computer code. RELAP5-3D©, version 4.1.3 was procured by NuScale and used as the baseline development platform for the NRELAP5 code. Subsequently, features

Section D

November 15, 2019 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 Submittal of Topical Report, "Rod Ejection Accident Methodology," TR-0716-50350, Revision 1

REFERENCE: Letter from NuScale Power, LLC to U.S. Nuclear Regulatory Commission, "Submittal of 'Rod Ejection Accident Methodology,' Revision 0," dated December 30, 2016 (ML16365A242)

NuScale Power, LLC (NuScale) hereby submits Revision 1 of "Rod Ejection Accident Methodology," TR-0716-50350.

Enclosure 1 is the proprietary version of the report titled "Rod Ejection Accident Methodology," TR-0716-50350, Revision 1. NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavits (Enclosure 3 and Enclosure 4) support this request. Enclosure 3 pertains to the NuScale proprietary information, denoted by double braces (i.e., "{{ }}"). Enclosure 4 pertains to the Framatome Inc. (formerly AREVA Inc.) proprietary information, denoted by brackets (i.e., "[ ]"). Enclosure 1 has also been determined to contain Export Controlled Information. This information must be protected from disclosure per the requirements of 10 CFR § 810. Enclosure 2 is the nonproprietary version of the report titled "Rod Ejection Accident Methodology," TR-0716-50350, Revision 1.

This letter makes no regulatory commitments or revisions to any existing regulatory commitments.

If you have any questions, please feel free to contact Matthew Presson at (541) 452-7531 or at mpresson@nuscalepower.com.

Sincerely,

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

Distribution: Samuel Lee, NRC, OWFN-8H12
Gregory Cranston, NRC, OWFN-8H12
Rani Franovich, NRC, OWFN-8H12
Michael Dudek, NRC, OWFN-8H12

Enclosure 1: “Rod Ejection Accident Methodology,” TR-0716-50350-P, Revision 1, proprietary version
Enclosure 2: “Rod Ejection Accident Methodology,” TR-0716-50350-NP, Revision 1, nonproprietary version
Enclosure 3: Affidavit of Zackary W. Rad, AF-1119-67635
Enclosure 4: Affdavit of Morris Byram, Framatome, Inc.

Enclosure 2:

"Rod Ejection Accident Methodology," TR-0716-50350-NP, Revision 1, nonproprietary version

Note: this enclosure to NuScsale's November 15, 2019 letter to the NRC is Identical to the topical report included in Section B of the current NuScale Letter with two exceptions: the Section B version includes "-A" in the document identification number and the date was updated on the cover page.

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360-0500 Fax 541.207.3928
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Enclosure 3:

Affidavit of Zackary W. Rad, AF-0620-70465

NuScale Power, LLC

AFFIDAVIT of Zackary W. Rad

I, Zackary W. Rad, state as follows:

(1) I am the Director of 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 topical report reveals distinguishing aspects about the method by which NuScale develops its rod ejection accident methodology.

NuScale has performed significant research and evaluation to develop a basis for this method 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 Enclosure 1 to the "NuScale Power, LLC Submittal of the Approved Version of the NuScale Topical Report, 'Rod Ejection Accident Methodology,' TR-0716-50350, Revision 1." 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 16, 2020.

Zackary W. Rad