April 11, 2025 Docket No. 52-050

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 Approved "-A" Version of Topical Report "Rod Ejection Accident Methodology," TR-0716-50350-A, Revision 3

REFERENCES:
1. NRC email to NuScale, "Final Safety Evaluation for NuScale Rod Ejection Accident Methodology TR," dated April 7, 2025
2. NuScale letter to NRC, "NuScale Power, LLC Submittal of Topical Report "Rod Ejection Accident Methodology," TR-0716-50350, Revision 3," dated October 20, 2023 (ML23293A292)

By referenced email dated April 7, 2025 (Reference 1), 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-A, Revision 3, is acceptable for referencing in licensing applications for the NuScale small modular reactor design. Reference 1 requested that NuScale publish the approved version of TR-0716-50350-A as soon as possible.

Enclosure 1 contains the proprietary version of the report entitled, "Rod Ejection Accident Methodology," TR-0716-50350-A, Revision 3. 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 report.

The proprietary and nonproprietary versions of the report in Enclosures 1 and 2, respectively, are identical to the versions previously transmitted in Reference 2, with the following exceptions:

- The report number is revised to add the "-A" at the end.
- The report date and copyright is updated to match the date of this letter.
- Several NuScale documents in the list of references in Section 8 of the report are updated to the latest versions reviewed by the NRC.

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

If you have any questions, please contact Amanda Bode at 541-452-7971 or at abode@nuscalepower.com.

I declare under penalty of perjury that the foregoing is true and correct. Executed on April 11, 2025

Sincerely,

Mark W. Shaver

Mark W. Shaver
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Mahmoud Jardaneh, Chief New Reactor Licensing Branch, NRC
Getachew Tesfaye, Senior Project Manager, NRC
Stacy Joseph, Senior Project Manager, NRC

Enclosure 1: "Rod Ejection Accident Methodology," TR-0716-50350-P-A, Revision 3, Proprietary Version
Enclosure 2: "Rod Ejection Accident Methodology," TR-0716-50350-NP-A, Revision 3, Nonproprietary Version
Enclosure 3: Affidavit of Mark W. Shaver, AF-181136

Enclosure 1:

"Rod Ejection Accident Methodology," TR-0716-50350-P-A, Revision 3, Proprietary Version

Enclosure 2:

“Rod Ejection Accident Methodology,” TR-0716-50350-NP-A, Revision 3, Nonproprietary Version

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Contents

Section A

Cohen, Tamela

From: Stacy Joseph <Stacy.Joseph@nrc.gov>
Sent: Monday, April 7, 2025 11:14 AM
To: Regulatory Affairs
Cc: Griffith, Thomas; Getachew Tesfaye; Mahmoud -MJ- Jardaneh; Bode, Amanda; Lynn, Kevin
Subject: Final Safety Evaluation for NuScale Rod Ejection Accident Methodology TR (Proprietary)
Attachments: Final Safety Evaluation for NuScale Rod Ejection Accident Methodology Topical Report Final_NON-Public.pdf; Final Safety Evaluation for NuScale Rod Ejection Accident Methodology Topical Report Final_PUBLIC.pdf

The NRC staff has prepared a final safety evaluation for TR-0716-50350-P, Revision 3 (ML23293A292/ ML23293A293). The non-proprietary and proprietary final safety evaluations are enclosed. The NRC staff has found TR-0716-50350-P, Revision 3, to be acceptable for referencing in licensing applications for the NuScale small modular reactor design to the extent specified and under the conditions and limitations delineated in the attached final safety evaluation.

The NRC staff requests that NuScale publish the accepted version of this TR as soon as possible following receipt of this electronic mail. The accepted version shall incorporate this electronic mail and the enclosed final safety evaluation after the title page. It must be well indexed such that information is readily located. Also, it must contain historical review information, including NRC requests for additional information and accepted responses. The accepted version of the TR shall include an "-A" (designated accepted) following the report identification number.

If the NRC's criteria or regulations change such that the NRC staff's conclusion in this electronic mail (that the TR is acceptable) is invalidated, NuScale and/or the applicant referencing the TR will be expected either to revise and resubmit its respective documentation or to submit justification for continued applicability of the TR without revision of the respective documentation.

If you have any questions or comments concerning this matter, I can be reached via e-mail at Stacy.Joseph@nrc.gov. The attached documents are both password protected. Password to follow in a separate email.

Sincerely,

Stacy K. Joseph
Senior Project Manager
USNRC/NRR/DNRL/NRLB

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT TR-0716-50350-P, REVISION 3

"ROD EJECTION ACCIDENT METHODOLOGY"

NUSCALE POWER, LLC

Proprietary information pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) section 2.390, "Public inspections, exemptions, requests for withholding," has been redacted from this document. Redacted information is identified by blank space enclosed within bolded double brackets, as shown here: **{{ }}**.

1 INTRODUCTION

1.1 Summary

By letter dated December 17, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21351A399), NuScale Power, LLC (NuScale, the applicant), submitted, for U.S. Nuclear Regulatory Commission (NRC) staff review and approval, Topical Report (TR) TR-0716-50350-P, Revision 2, "Rod Ejection Accident Methodology" (Reference 1). NuScale supplemented its submittal by letter dated September 14, 2022 (Reference 2) in response to requests for additional information (RAI), RAI No. 9936, from the NRC staff. The NRC staff conducted a limited scope audit for TR-0716-50350, Revision 2, starting on April 19, 2023 (Reference 3). On October 20, 2023, NuScale submitted Revision 3 of TR-0716-50350-P (Reference 4), hereafter referred to as "the TR."

In the TR, the applicant described a method for analyzing the consequences of a control rod ejection accident (REA) for a NuScale Power Module (NPM) design. The staff performed a review of the methodology presented in the TR, information made available as part of the audit (Reference 3), as well as the applicant's responses to RAI No. 9936, questions NTR-01 and NTR-02 (Reference 2).

The NRC staff review utilized the guidance in Regulatory Guide (RG) 1.236, "Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drop Accidents," (Reference 5). Based on its review, as provided below, the staff determined that the TR provides a methodology for analyses of REAs with the limitations and conditions as listed in Section 6 of this SER.

1.2 Description of a Generic Rod Ejection Accident Transient Event

REAs are a class of postulated reactivity accidents 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," General

Design Criteria (GDC) 28, “Reactivity Limits” (as described in Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants SRP Section 15.4.8 (Reference 6)), to obtain an NRC license for a particular reactor design. Additionally, REAs must be considered (among other postulated accidents) in dose consequence analysis required by 10 CFR 100.11, “Determination of exclusion area, low population zone, and population center distance,” or equivalent, such as Subsection (a)(2)(iv) of 10 CFR 52.47, “Contents of applications; technical information.”

The postulated REA is initiated by the sudden ejection of a control rod assembly (CRA) from the core of a reactor that is critical. The reactor can be at any state from hot zero power to hot full power, and the core could be at any stage of the reactor operation, from the beginning of cycle to the end of cycle. Partial power situations should be considered to explore bounding conditions. In general, a large number of initial conditions can affect the transient response and its ultimate termination and therefore must be examined to assess the safety of the reactor with fuel damage in focus.

In a typical rod ejection event, a CRA is rapidly ejected and accelerated by the system pressure, resulting in a near step insertion of positive reactivity to the core. The sudden addition of positive reactivity results in a corresponding rapid increase in local power and local fuel temperature. The only feedback mechanism that can counter this power increase is the Doppler effect (Doppler) of the fuel, which adds negative reactivity as the fuel temperature increases. The Doppler feedback accumulates 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 after non-ejected control rods are inserted and stable cooling is established. The duration of the rod ejection is approximately the scram delay time, which is short enough to ignore all system-related changes to the coolant temperature and pressure as the control system will trip the reactor with either a high-rate power increase or high-power level.

A second type of transient may occur when the worth of ejected CRA is relatively small. In this scenario, the rate of power increase is relatively small and slow. Consequently, the ex-core detectors do not reach the setpoint that trip the reactor because both the integrated flux increase and the rate of flux increase are small. In this case, a system-level response occurs, since the activation of reactor trips associated with the system response will terminate the transient.

1.3 Scope of the NRC Staff’s Review and Approval

The purpose of the TR is to describe the methodology that NuScale intends to use for REA analysis, as stated Section 1.1 of the TR. Accordingly, the NRC staff reviewed the REA methodology presented in the TR.

Because the supporting calculations provided in Sections 5 and 6 of the TR use NPM-20 as the target reactor design that is not finalized and the applicant requested approval of the TR as a generic methodology for REA analysis, the NRC staff placed Limitation and Condition 1 (as listed in Section 6 of this SER) on the TR identifying the need for an applicability review when the TR is referenced by an applicant or licensee. Limitation and Condition 1 applies to the NPM-20 design, once finalized, as well as to other NPM designs. Applicability should be addressed in specific licensing applications referencing this TR.

2 REGULATORY REQUIREMENTS, RELEVANT REGULATORY GUIDANCE, AND ACCEPTANCE CRITERIA

2.1 Regulatory Requirements

The REA methodology presented in the TR was developed to support compliance with the regulatory requirements in GDC 28 of 10 CFR Part 50, Appendix A, which states:

> *Criterion 28—Reactivity limits. The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.*

In addition, the REA methodology is predicated on compliance of the instrumentation and control system for a specific application with GDC 13:

> *Criterion 13—Instrumentation and control. Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.*

2.2 Relevant Guidance

The TR references the acceptance criteria and guidance outlined in SRP Section 4.2 (Reference 7), SRP Section 15.4.8 (Reference 6), and RG 1.236 (Reference 5) for reactivity-initiated accidents. By following the provided guidance, described as follows, an applicant can demonstrate compliance with GDC 28. The following is a summary of this guidance and the associated acceptance criteria:

(1) Cladding Failure: RG 1.236 describes cladding failure phenomena and fuel rod cladding failure thresholds that are acceptable to the NRC staff. The pellet cladding mechanical interaction (PCMI) caused by the sudden rise in power during the pulse phase of a REA requires a limit on the total energy (calories per gram (cal/g)) deposited as a function of cladding hydrogen content. Cladding failure can occur also because of high-temperature post-DNB (departure from nucleate boiling) oxygen-induced embrittlement and fragmentation, or high-temperature cladding creep. RG 1.236 combines these two cladding failure mechanisms into a composite high-temperature failure threshold. Specifically, fuel cladding failure is presumed if predicted fuel temperature anywhere in the pellet exceeds incipient fuel melting conditions. Finally, fuel cladding failure is presumed if heat flux exceeds the critical heat flux criterion.

(2) Coolability: Per RG 1.236, pin cooling is assumed failed for all pins with a total enthalpy of 230 cal/g or greater. In addition, pin cooling is assumed to fail if there is incipient fuel melting in the outer 90 percent of the fuel volume.

(3) Radiological Impact: RG 1.236 provides guidance related to the calculation of fission product inventory that would be available after an event. These limits are used at the decision points for fuel temperature and enthalpy determinations, as well as for the number of failed rods that may lead to unacceptable radiological release.

2.3 Rod Ejection Accident Analysis Method Acceptance Criteria

The methodology defined in the TR identifies a set of acceptance criteria for the performance of a reactor under a REA. Section 3.2.4 of the TR states that the methodology requires that no fuel failure occurs, and therefore no radiological release will occur as the result of a REA. NuScale defines the acceptance criteria in Section 2.2.2 of the TR to assure that no fuel failure will occur, as summarized below.

- Fuel cladding failure is presumed if local heat flux exceeds the critical heat flux (CHF) thermal design limit.
- The increase in radial average fuel enthalpy is limited to less than 33 cal/g.
- The peak radial average fuel enthalpy is below 100 cal/g.
- Fuel cladding failure is presumed if fuel temperature anywhere in the pellet exceeds incipient melting conditions.

Section 2.2.3 of the TR provides additional criteria to ensure core coolability in order to meet the requirements of GDC 28. Integrity of the reactor coolant pressure boundary is assured by ensuring that the maximum reactor coolant system (RCS) pressure remains below 120 percent of design pressure, as described in Section 2.2.1 of the TR. The staff's evaluation of these acceptance criteria is described in Section 4.2 of this SER.

3 SUMMARY OF TECHNICAL INFORMATION

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

3.1 General Information

The methodology presented in the TR consists of the following three major parts:

(1) General Rod Ejection Progression, Phenomena Identification and Ranking Analysis.
(2) Computer codes and cross-section library used for the REA methodology.
(3) REA Methodology.

The REA methodology uses CASMO5 (Reference 8), SIMULATE5 (Reference 9), SIMULATE-3K (Reference 10), NRELAP5 (Reference 11), and VIPRE-01 (References 12 and 13) computer codes to perform analyses of system response to a REA. ENDF/B-VII cross-section library is used in the nuclear analyses that are performed with the computer codes CASMO5, SIMULATE5, and SIMULATE-3K. The applicant provides a flow diagram in Figure 3-1 of the TR to show the interfaces between these different computer codes. Table 3-1 of the TR shows the parameters that are passed along between these computer codes.

The applicant provides a sample analysis to illustrate how the methodology can be used for REA analysis in Section 6 of the TR.

3.2 Outline of Rod Ejection Accident Physical Phenomena, Modeling, and Overall Methodology

A REA is postulated to be caused by a failure of a control rod drive mechanism housing, which allows a control rod to be rapidly ejected from the reactor by the RCS pressure. The rate of reactivity insertion is dependent on the speed of the ejected rod and the differential rod worth (which is dependent on the location of the CRA).

Section 3.2 of the TR describes the computer codes used in the NuScale methodology and the evaluation flow-path is shown in Figure 3-1. The analysis starts with the steady-state neutronics calculations performed with the methods using CASMO5/SIMULATE5 computer codes that the NRC staff approved in TR-0616-48793-P-A, Revision 1 (Reference 14). CASMO5 is used to generate a cross-section data library for use by the 3-D steady-state nodal code SIMULATE5 and 3-D transient analysis code SIMULATE-3K. SIMULATE5 initializes the cycle-specific model and reactor conditions that are used as input into the SIMULATE-3K evaluation.

SIMULATE5 is used to determine the steady-state initial condition portion of the REA calculations for the core response analysis. This steady-state assessment involves two calculations: determining the worst rod stuck out and development of the initial conditions to SIMULATE-3K model in a format of SIMULATE5 restart file.

Then, SIMULATE-3K solves the transient 3-D, two-group neutron diffusion equations, using the SIMULATE5 restart files. The transient simulation involves two calculations: simulation of the transient and determination of the parameter uncertainties used to bias inputs to the transient simulation. SIMULATE-3K analyzes the transient neutronic behavior under a REA at various times in the reactor cycle, power levels, control rod positions, and initial core conditions and provides the total core power, 3-D power distributions, and peaking factors during the REA transient.

The transients that result in a power spike with a lower magnitude, but longer duration necessitate a system-level code to determine the coolant temperature and pressure and to identify any phase change in cases where the pressure is dropping. The TR employs the NRELAP5 code to calculate the system response based on input from SIMULATE-3K. Results from the system response analysis determine whether the reactor coolant system pressure limit is exceeded. **{{**

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}}. Section 4.4.3.2 of this SER provides further discussion on the screening method and criteria.

The applicant used the VIPRE-01 subchannel analysis computer code to examine the fluid dynamics and heat transfer behaviors of the hot channel identified by SIMULATE-3K. 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. VIPRE-01 uses radial and axial power distribution input from SIMULATE-3K and core exit pressure, system flow, and core inlet temperature forcing functions from NRELAP5. The primary output from this analysis is detection of rod failure based on the fuel failure criteria, as identified in Section 2.3 of this SER.

The subchannel methodology also implicitly relies on the fuel performance code COPERNIC (Reference 15) to supply application-specific fuel thermal properties, as described in Section 4.4 of TR-0915-17564-P-A, Revision 2 (Reference 12). In addition, the NuScale REA methodology uses the statistical subchannel analysis methodology as defined in TR-108601-P, Revision 4 (Reference 13), to evaluate fuel failure criteria and uncertainties associated with the input parameters for subchannel analyses.

In Section 4 of the TR, NuScale presented an overview of the phenomena important for the REA, which is used to develop conservative assumptions for the analysis. There is no change in this revision of the TR in phenomenon identification and ranking table (PIRT) results compared to that of the previous NRC approved Revision 1 of this TR (Reference 16).

Section 5 of the TR describes the REA methodology. The methodology requires a REA analysis for each core reload design. Section 5.1 discusses the general assumptions and considerations used in the REA analysis. These assumptions and considerations include reload cycle-specific core design features, the time in the cycle when a REA occurs, core power level, single active failure, automatic system response of non-safety systems, and loss of alternating current power.

Section 5.2.1 of the TR describes calculation procedures for the rod ejection analysis methodology. These procedures include initial condition calculations and transient calculations, determination of the worst rod stuck out and treatments of key parameters such as Doppler temperature coefficient (DTC) uncertainty, moderator temperature coefficient uncertainty, travel time of the ejected rod, and ex-core detector modeling.

Section 5.2.2 of the TR provides the assumptions and treatment of uncertainties associated with the key system parameters that must be used in rod ejection analyses. These assumptions include the worst rod is stuck out and the regulating groups of control rod assemblies are inserted at the power dependent insertion limit (PDIL).

Section 5.2.2 of the TR discusses key input parameters. These parameters include rod ejection time, rod ejection location, and modeling of the reactor trip signals. The values for the reactor trip setpoints are input parameters that are specified based on the NPM-20 design. The TR provides high power trip and high power rate trip setpoints used in the methodology.

In addition, Section 5.2.2 of the TR discusses treatments of reactivity coefficients DTC and moderator temperature coefficient (MTC) and the effective delayed neutron fraction variation as a function of core burnup and time in the cycle at which a REA takes place.

Table 5-1 of the TR provides example uncertainties for rod ejection calculations. The TR requires that the uncertainties associated with the key parameters listed in Table 5-1 must be reexamined for each rod ejection analysis for each reload design.

TR Section 5.3 provides calculation procedures for the system response of the rod ejection analysis accident of the NuScale NPM-20 reactor. The objective is to determine the system responses to a sudden injection of a large reactivity resulting from a REA. The two most paramount system parameters are the peak reactor coolant system pressure and MCHFR.

The purposes of the system response calculations are to determine the peak RCS pressure analysis and to provide inputs to the subchannel analysis for CHF determination. The 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.

The TR considers competing scenario evaluations between the peak pressure and the MCHFR calculations. To examine the MCHFR, the system responses to maximum ejected CRA worth scenarios for different initial power levels are calculated. To examine the system pressurization, a reduced reactivity insertion rate assumption is used to allow the reactor power to rise less rapidly without triggering the high power rate setpoint that will trip the reactor.

Section 5.3.2 of the TR discusses assumptions and parameter treatment used in the system response calculation. No pressure reduction associated with the postulated failure of the control rod drive mechanism is assumed. Parameter treatments include initial power, inlet temperature, coolant flow, and model options governing direct moderator and cladding heating.

In addition, the axial power distribution is used as the assembly-average axial power at time of maximum core neutron power for the assembly containing the highest peak $F_{\Delta H}$ rod that is calculated using the SIMULATE-3K computer code.

Section 5.4 of the TR provides detailed procedures for performing thermal-hydraulic and fuel response analyses. The VIPRE-01 (References 12, 13, and 17) code is used to perform subchannel analyses. However, several deviations **{{**

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The TR states that **{{**

}}. For these reasons, cycle-specific sensitivity studies must be performed for determining the adequacy of these values.

Power distributions within the fuel assembly in which the ejected rod is inserted and across the core are of a concern for rod ejection analyses because power shape could potentially affect the worth of the control rod. Figure 6-9 provides an example core radial power distribution, while Figure 6-10 provides an example hot assembly radial power distribution from the limiting statepoint at time of peak power. In the default radial nodalization, SIMULATE-3K power distribution inputs would be used to represent all fuel rods in the core. **{{**

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As discussed in Section 3.2.4 of the TR, the REA methodology does not include assessment of inventory or dispersion of radiological materials because the acceptance criterion of the methodology requires that no fuel failure occurs and, therefore, there is no need to include such analyses.

The staff reviewed Sections 1, 2, 3, 4, and 5 of the TR. The staff finds that Section 1 of the TR provides an overview of the methodology for REA analysis; Section 2 provides regulatory considerations for a REA; and Sections 3, 4, and 5 provide a detailed description of the methodology for performing analysis of a REA associated with the NuScale NPM-20 design. The TR also specifies the assumptions and limitations of the REA methodology.

Section 6 of the TR includes results of selected sample calculations and sensitivity analyses. Table 6-1 provides a list of these sample calculations and sensitivity analyses. Notably, however, some of the calculations are used as justifications of the adequacy of the REA methodology and referenced in Section 5 of the TR; the results are for illustration purpose only. Therefore, unless being referenced in Section 5 of the TR, these sample calculations are not part of the methodology for which NuScale is seeking NRC staff's approval.

4 TECHNICAL EVALUATION

4.1 General Information

The NRC staff reviewed Section 1 of the TR and finds it to be consistent with the purpose and scope of the TR, subject to an applicability determination, reviewed and approved by the NRC, when applied to a specific design.

Section 2 of the TR includes discussion of applicable regulatory requirements and acceptance criteria, including limits to evaluate RCS integrity, fuel cladding failure, and core coolability. Section 4.2 of this SER provides detailed evaluations of the acceptance criteria with respect to the regulatory requirements.

Section 4 of the TR includes a PIRT. The staff compared this PIRT to the previously approved PIRT in TR-0716-50350-P-A, Revision 1 (Reference 16) and noted there were no changes. However, the staff noted that Revision 3 of the TR is intended for higher power levels and different thermal-hydraulic conditions than Revision 1, and includes changes that may alter the magnitudes of parameters such as power peaking, enthalpy increase, or RCS pressure; however, the NRC staff determined that these changes do not introduce new phenomena or alter the relative importance of phenomena for the rod ejection accident. Based on the above discussion, the NRC staff determined that use of the PIRT in this TR is acceptable.

Revision 4 of the SIMULATE5 and Revision 7 of the CASMO5 code and the ENDF/B-VII cross-section library are used to generate core neutronic characteristic parameters for the SIMULATE5 code to calculate the reactor power distribution and control rod reactivity worth. These codes and library have been previously reviewed and approved by the NRC staff (Reference 14) and the staff finds them to be acceptable for REA analysis.

The applicant updated important core neutronic characteristics, such as the reactivity, effective delayed neutron fraction (β_{eff}), Doppler coefficient, moderator reactivity coefficient, and other parameters, as functions of time in the cycle, with the increased power rate using the same methodology as approved by the staff (Reference 16). As discussed in Section 4.3 of this SER, the staff reviewed these revised parameters and the supporting calculations and finds the new values to be acceptable for use in demonstrating the REA analyses methodology. However, because the values of these parameters are design specific, they will be determined when this REA methodology is applied.

The staff reviewed the general descriptions of the rod ejection methodology as presented in the TR and finds it to be acceptable because the TR provides an overall description of the methodology with sufficient details to evaluate consistency with the guidance provided in RG 1.236 (Reference 5).

4.2 Acceptance Criteria

The NRC staff compared fuel failure criteria identified in TR Section 2.2 to those in RG 1.236. The rod ejection analysis methodology ensures that PCMI fuel failure does not occur by setting a maximum radial average energy deposition of 33 cal/g. The applicant determined this value by examining the RG 1.236 PCMI failure threshold for the fuel with unlined recrystallization annealed (RXA) cladding that is used in NuScale designs, as stated in Section 5.4.3 of the TR.

The methodology also requires that peak radial average fuel enthalpy remain below 100 cal/g. This limit corresponds to the cladding-pressure-dependent fuel failure criterion for brittle and ductile fuel failure modes at high temperature. The NRC staff compared the peak radial average fuel enthalpy criterion in the TR to that in Figure 1 of RG 1.236 and finds that the TR limit is below the RG 1.236 threshold at all cladding pressures. On this basis, the NRC staff finds this criterion is conservative and acceptable.

The NRC staff finds that NuScale’s approach of applying steady-state threshold that is below the RG 1.236 hydrogen-concentration-dependent PCMI failure threshold and cladding-pressure-dependent high-temperature failure threshold is consistent with the regulatory position in 2.2.1.5 of RG 1.236.

NuScale evaluates MCHFR fuel failure for all analyzed power levels and exposure points, as discussed in Sections 5.1.2 and 5.1.3 of the TR. The NRC staff finds this to be consistent with the RG 1.236 guidance that fuel cladding failure should be presumed if local heat flux exceeds thermal design limits in prompt critical scenarios which experience a prolonged power level following the prompt pulse and non-prompt critical excursions, regardless of initial power level.

In addition, the methodology assumes cladding failure if fuel temperature anywhere in the pellet exceeds incipient melting conditions, which is consistent with the guidance in RG 1.236.

The staff notes that the TR fuel failure acceptance criterion is established based on the PCMI fuel failure criterion of RXA cladding. Therefore, cladding type must be evaluated when this TR is referenced in licensing applications, consistent with Limitation and Condition 1.

The TR provides criteria to ensure core coolability in Section 2.2.3. The NRC staff reviewed these criteria and finds them to be consistent with guidance provided in RG 1.236 and therefore, acceptable.

4.3 Software Applicability

Section 3.2 of the TR presents the computer codes used in the NuScale REA methodology and states that as part of the applicability review these computer codes have been previously reviewed and approved by the staff for the applicants and licensees to use in TR-0616-48793-P-A, Revision 1 (Reference 14), with the exception of SIMULATE-3K. Applicability needs to be demonstrated by the applicant or licensee and reviewed by NRC staff when the TR is referenced. However, NRC staff reviewed the rod ejection methodology with the codes and methods approved in TR-0616-48793-P-A, Revision 1 (Reference 14) as part of the methodology. It is important to note that use of different nuclear analysis method may represent a significant change to an element of the methodology and would likely require NRC staff review, as identified in Limitation and Condition 3 in Section 6 of this SER.

Section 3.2.1.3 of the TR includes the code description and Section 3.2.1.4 describes the validation of SIMULATE-3K. In TR Section 3.2.1.4, NuScale used data from the SPERT-III tests and a Nuclear Energy Agency Committee on Reactor Physics control rod ejection benchmark problem to validate SIMULATE-3K for use in analyzing a REA. Details for the validation are provided in Appendix A of the TR.

The applicant has demonstrated that SIMULATE5 and SIMULATE-3K computer codes used in the REA methodology can capture the impacts of power tilt in calculating the reactivity worth of the ejected control rod, total reactivity insertion, and the power excursion response inside the fuel assembly in which the ejected rods were inserted. Based on the staff’s review of the information provided by the applicant and the staff’s confirmatory analyses, the staff determined that the power tilt inside the fuel assemblies and its impacts on rod ejection accident is adequately captured by the codes and calculation procedures. The staff also performed confirmatory analyses to confirm the SIMULATE5 and SIMULATE-3K computer codes are capable of reliably predicting the power tilt within a fuel assembly and its impact on the reactivity worth of the control rod. The NRC staff performed confirmatory analysis using the POLARIS lattice physics module of SCALE, and the Purdue Advanced Reactor Core Simulator (PARCS). The results of the confirmatory analyses provide evidence that SIMULATE5 and SIMULATE-3K are capable of reliably predicting assembly-wise power distribution and control rod worth, and that ejected control rod worth is not sensitive to the detailed flux distribution within individual assemblies.

4.4 Methodology

Section 3 of the TR provides a general description of the computer codes that are used in the rod ejection analysis methodology and a flowchart to show the interfaces between different codes.

4.4.1 Steady-State Initialization

Section 3.2.1.2 of the TR 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 steady-state calculations methodology. The steady-state 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 NRC staff-approved TR-0616-48793-P-A, Revision 1 (Reference 14) describes the use of SIMULATE5 for non-LOCA analyses. 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. In Section 5.2.1.1, NuScale stated that the core flow, and thus the coolant mass flux, for a given initial power is held constant through a modeling option. The staff finds that this is consistent with previously approved TR-0716-50350-P-A, Revision 1 (Reference 16). The NRC staff considered whether other changes (that is, changes in the methodology, changes in the expected application, or changes in NRC staff guidance) would require detailed review of these initial conditions. The NRC staff did not identify any such changes and determined that there is no need for detailed review for this TR.

The staff reviewed Section 5.2 of the TR and finds that the method for developing steady-state conditions is consistent with the non-LOCA accident methodology, as presented in TR-0516-49416-P, Revision 5 (Reference 11), using the nuclear analysis codes and methods in TR-0616-48793-P-A, Revision 1 (Reference 14) and is, therefore, acceptable.

The TR does not describe steady-state initialization methodologies for non-baseload operation. Because some non-baseload operation schemes may involve operation with regulating rods being inserted, the axial and radial power may be significantly skewed compared with baseload operation. The insertion of the regulating rods will suppress power density (and therefore suppress fissile material depletion) in the upper region of the core, which exacerbates the power excursion under a rod ejection. Further, the staff considered the fact that the worth of an ejected rod is dependent on the magnitude of the axial offset, and that the axial offset assumptions drive the level of conservatism in the analysis. To estimate the significance of the impact, the staff conducted a confirmatory analysis. The codes used in the analysis are described in Section 4.3 of this safety evaluation. The staff examined sensitivity of the control rod worth to axial offset by varying the axial xenon distribution present during a control rod ejection. The analysis confirmed that there is a significant sensitivity between the ejected rod worth and the magnitude of the axial offset. Since changes to axial offset from non-baseload operation are not accounted for in NuScale's REA methodology, the methodology cannot be applied to analysis of REA events where core operation includes control rod insertion resulting in significant skewed axial fuel depletion. Additionally, depending on the non-baseload operation scheme and attendant frequency, magnitude, and rate of power changes, assumptions regarding axial power shape and xenon distribution may require modification or additional justification or clarification. This limitation is reflected in Limitation and Condition 2.

4.4.2 Core Response

Section 5.2.1.2 of this TR describes the transient core response calculations performed using SIMULATE-3K for the NuScale REA 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.

NuScale applies numerical multipliers to conservatively bias the Doppler temperature coefficient, moderator temperature coefficient, effective delayed neutron fraction, and CRA worth according to the assessed uncertainty. The magnitude of the uncertainty is based on Nuclear Reliability Factors assessed in TR-0616-48793-P-A, Revision 1 (Reference 14) for SIMULATE-5, and NuScale stated in TR Section 3.2.1.4 that the benchmark comparison between SIMULATE-5 and SIMULATE-3K establishes applicability of these uncertainties to SIMULATE-3K. **{{**

}}. As discussed in Section 5.2.2.3.4 of the TR, MTC and DTC are biased to be as least negative as possible, and the effective delayed neutron fraction is biased to be as small as possible. The worth of the ejected CRA is increased, and the worth of the CRAs inserted with reactor trip is decreased. The NRC staff finds these biases to be appropriately conservative as they will increase the severity of the analyzed accident. Additionally, because the methodology for determining the maximum rod worth accounts for calculation uncertainties in neutronic parameters, the NRC staff finds that this treatment is consistent with position 2.2.1.4 of RG 1.236.

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 (Reference 14). 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-108601-P, “Statistical Subchannel Analysis Methodology,” Revision 4 (Reference 13). Additionally, the NuScale methodology includes turning off the point kinetics in NRELAP5 while performing MCHFR analyses (as power pulses from SIMULATE-3K are used directly) but continues to use them for the overpressure analyses. The staff reviewed the information provided in the TR and TR-108601-P (Reference 13) and determined that NuScale’s methodology ensures SIMULATE-3K conservatively calculates potential fuel failures by choosing conservative input values and following the NRC staff-approved methodology described in TR-0616-48793-P-A (Reference 14). This supports the statements provided in the

TR. 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.

The TR describes the process for performing the transient calculations once the uncertainties have been applied to the nuclear parameters. The staff finds that the information is the same as presented in the previously approved TR-0716-50350-P-A, Revision 1 (Reference 16). The NRC staff considered whether other changes (that is, changes in the methodology, changes in the expected application, or changes in NRC staff guidance) would require detailed review. The NRC staff did not identify any such changes and therefore, did not perform a detailed review.

Section 5.2.2.3.1 of the TR describes the calculation for time to eject the CRA from the core. The ejection time is calculated using the appropriate PDIL depth at the initial power and the CRA acceleration equation in Equation 5-1. Section 5.2.2.3.1 of the TR states that the acceleration is based on the CRA cross-sectional area and weight of the CRA and control rod driveshaft. The NRC staff audited the example calculations and confirmed that no pressure barrier restriction is assumed (Reference 3). Because the method of calculating the rate of ejection, including the assumption of no pressure barrier restriction, is consistent with position 2.2.1.7 of RG 1.236, the NRC staff finds it acceptable.

Section 4.3(B) of the TR 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 only allows the rods at or above the Power Dependent Insertion Limit (PDIL) (see response to NRC Question No. 15.04.08-8 in RAI No. 9306 in Reference 16). The staff finds this conclusion acceptable because, with all else equal, a rod ejection event starting with CRAs above the PDIL will result in a smaller reactivity insertion and hence a smaller power pulse. The NRC staff considers the assumption that control rods are initially at PDIL, factoring in uncertainty for the CRA position, to be consistent with position 2.2.1.4 of RG 1.236 and, therefore, acceptable.

Because the reactivity worth of an ejected CRA is dependent on its location in the core and the three-dimensional core power distribution, and rod ejections from different locations in the core may produce different sizes (widths and heights) of pulses and levels of power peaking, the TR requires examining multiple CRA ejections. Explicit evaluation is performed for each regulating rod unless the core design is quarter-core or eighth-core symmetric. In this case, regulating rod ejections in each unique quadrant-symmetric location are evaluated. The NRC staff considers this an acceptable method of identifying the location of the limiting control rod consistent with position 2.2.1.4 of RG 1.236 because it evaluates ejection from each control rod location that will produce a different result.

Examples of trip setpoints are provided in Section 5.2.2.3.3 of the TR. NRC staff expects that fuel temperature, enthalpy, and heat flux will be insensitive to these trip setpoints in prompt critical scenarios, as limiting transients for the expected application will involve a rapid increase in power to well above the initial condition. The outcome of rod ejections with smaller initial power increases will be more sensitive to trip setpoints. Reactor trip delays are assumed, as discussed in Section 5.1.4 and Section 5.3.1.2 of the TR.

As discussed in Section 5.1.4 of the TR, the conservative single active failure for a REA is a failure of the neutron flux detector in the high-flux region. The staff verified that this is identical to the previously approved TR-0716-50350-P-A, Revision 1 (Reference 16). The NRC staff considered whether other changes (that is, changes in the methodology, changes in the expected application, or changes in NRC staff guidance) would require additional review of this assumption. The NRC staff did not identify any such changes, and, therefore, did not perform additional review.

4.4.3 Dynamic System Response

Section 5.3 of the TR 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 REA methodology follows the non-LOCA evaluation methodology in TR-0516-49416-P, Revision 5 "Non-Loss-of-Coolant Accident Analysis Methodology" (Reference 11) but with modifications to ensure conservative results when modeling reactivity-initiated accidents. The following sections discuss NRC staff evaluation of the peak pressure and MCHFR portions of the system response calculations.

4.4.3.1 Peak Pressure Calculations

The calculation procedure in Sections 5.3.1 and 5.3.1.2 of the TR details the methods used to calculate the peak pressure resulting from a 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 for the reactor. 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 because the calculation uses bounding assumptions in the transient analysis and is, therefore, acceptable.

The peak pressure calculation methodology is unchanged from the previously approved TR-0716-50350-P-A, Revision 1 (Reference 16). The NRC staff considered whether other changes (that is, changes in the methodology, changes in the expected application, or changes in NRC staff guidance) would require additional review of this assumption. The NRC staff did not identify any such changes and therefore, did not perform a detailed review of the peak pressure calculation methodology.

4.4.3.2 Minimum Critical Heat Flux Ratio

The calculation procedure detailed in Section 5.3.1 of the TR 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. Section 5.3.3 of the TR states that the scoping methodology described in Section 4.3.5 of TR-0516-49416-P, Revision 5 (Reference 11), is used to determine generally limiting scenarios for final MCHFR calculation with the subchannel analysis methodology. The screening criteria are defined in Section 4.3.5 of TR-0516-49416-P, Revision 5 (Reference 11). The staff reviewed the MCHFR calculation with the subchannel analysis, the initial conditions considered, the ability to capture the most limiting case, and the analytical methods. Because of the interdependence between the scoping methodology for the rod ejection methodology and the screening criteria of TR-

0516-49416-P, Revision 5 (Reference 11), NRC staff assessed this TR and TR-0516-49416-P, Revision 5 (Reference 11) together for consistency.

Section 5.3.1.1 of the TR provides the conservatisms included in the methodology for the MCHFR analyses. The staff concludes that the system condition assumptions used in the MCHFR analysis methodology are conservative, in that they result in increased fuel and coolant temperatures, which is conservative for MCHFR. Additionally, Section 5.3.1.1 states that high and low pressure conditions are investigated due to the unique nature of REA and the potential impact on core flow. The staff finds that the method for determining MCHFR is consistent with the methodology outlined in TR-108601-P, Revision 4 (Reference 13) and is, therefore, acceptable.

4.4.4 Detailed Thermal-Hydraulic and Fuel Response

Section 5.4 of the TR presents the subchannel response methodology, which calculates the MCHFR, fuel enthalpy and fuel temperature and compares them against the relevant fuel failure criteria to verify that no fuel failure occurs.

As described in Section 3.2.3 of the TR, VIPRE-01 contains a fuel rod model with radial profile, theoretical density, and gap conductance supplied from a fuel design-specific calibration to COPERNIC. Additional detail of this process is described in Section 4.4 of TR-0915-17564-P-A, Revision 2 (Reference 12), which clarifies that the entire range of possible time-in-cycle parameters are evaluated, and the VIPRE-01 model is calibrated to ensure that it produces conservative fuel temperatures for each fuel design. The NRC staff previously approved the use of this fuel rod model in evaluation of rod ejection accidents in the previous revision (Revision 1) of this TR (Reference 16). The current revision (Revision 3) of the rod ejection methodology requires cycle-specific sensitivity studies on fuel heat transfer parameters to ensure conservative evaluation of all fuel failure quantities of interest, such as critical heat flux and radial average fuel enthalpy increase. Because these studies cover the full range of the fuel rod's lifetime, and cycle-specific sensitivity studies are used to identify limiting biases for each fuel failure criterion, NRC staff finds that this approach is consistent with position 2.2.1.9 of RG 1.236 and is, therefore, acceptable.

The radial power distribution used in each case-specific VIPRE-01 case comes from the SIMULATE-3K pin power reconstruction model. In order to account for uncertainty in radial peaking, **{{ }}**. Section 5.4.2.1 of the TR describes two options for evaluating fuel failure criteria since the MCHFR limit implicitly includes $F_{\Delta H}$ engineering uncertainty and non-CHF fuel failure limits do not. The first option **{{**

}}. The NRC staff finds both options acceptable, as they both account for radial power distribution uncertainties identified in TR-0915-17564-P-A (Reference 12), which the NRC staff previously found to be acceptable, when evaluating relevant fuel failure criteria.

Section 5.4.5 of the TR describes the sensitivity studies necessary for applying the REA analysis method. Table 5-3 lists the sensitivity studies. Optional sensitivity studies are performed when any non-default VIPRE-01 calculation control parameters or options are used. When a non-default parameter is selected, the user will perform additional calculations with greater and lesser values than the selected parameter value. To ensure that the calculation is converged, NuScale will ensure that "excellent agreement" is achieved as defined in RG 1.203, "Transient and Accident Analysis Methods" (Reference 18), as stated in Section 5.4.5 of the TR. These sensitivity studies will demonstrate that convergence is achieved by ensuring that changes in the VIPRE-01 parameters do not change the results or the progression of the transient. TR-108601-P, Revision 4 (Reference 13) Section 7.2.1, bullet 3 describes procedures for ensuring convergence is appropriately achieved, as stated in Section 5.4.1.1 of the TR (i.e., TR-0716-50350, Revision 3).

Section 5.4.1.1 of the TR describes several considerations taken in VIPRE-01 calculations in order to capture the rapid power increase associated with the REA transient. **{{**

}} and is, therefore, acceptable.

Cycle-specific sensitivity studies on axial and radial nodalization are only performed when a deviation from the default nodalizations described in the statistical subchannel report are taken. Based on NRC staff review and approval of these nodalizations in the statistical subchannel report TR-108601-P, Revision 4 (Reference 13), as well as the sensitivity results presented in Section 6 of the TR, and the REA methodology requirement to perform sensitivity studies when non-default nodalizations are used, the NRC staff finds these default nodalizations acceptable.

The REA methodology requires that applicants and licensees perform other "mandatory" sensitivity studies to address the VIPRE-01 timestep and fuel heat transfer inputs. Timestep selection affects the Courant number, which was discussed above. The VIPRE-01 timestep must also be sufficiently small to resolve the time-dependent power as calculated by SIMULATE-3K. The REA methodology requires the user to plot time-dependent power with SIMULATE-3K and VIPRE-01 timesteps and evaluate whether "excellent agreement" is achieved. Based on this, and on the preceding Courant limit discussion, the NRC staff finds treatment of VIPRE-01 timestep within the rod ejection methodology acceptable.

Cycle-specific sensitivity studies on fuel heat transfer inputs (such as gap conductance) are performed to ensure that different fuel failure criteria are evaluated conservatively. NuScale provided sensitivity study results in Section 6.3.6 of the TR which demonstrate that high fuel heat transfer cases can be bounding for MCHFR. However, maximizing heat transfer from the fuel may not be bounding for other fuel failure criteria, such as radial average fuel enthalpy rise. Because fuel heat transfer inputs are varied to ensure that limiting biases are identified for each fuel failure criterion, the NRC staff finds this sensitivity study acceptable.

Table 5-1 of the TR provides a list of the parameters that need to account for uncertainties for each time a rod ejection analysis is performed. As described in Section 5.2.2.3.4 of the TR, the uncertainties defined in Table 5-1 must be verified to ensure they are current and consistent with References 8.2.6, "Nuclear Analysis Codes and Methods Qualification," TR-0616-48793-P-A, Revision 1," and 8.2.10, "Statistical Subchannel Analysis Methodology," TR-108601-P, Revision 4." This is consistent with Limitation and Condition 3 identified in Section 6 of this SER.

5 CONCLUSIONS

Based upon the NRC staff's review, as discussed above, the NRC staff concludes that TR-0716-50350-P, Revision 3 provides a systematic methodology for performing REA analysis. This conclusion is based on the following, as summarized below:

The applicant uses 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. The REA methodology is developed with the acceptance criteria provided in the regulatory guidance of RG 1.236. The staff has evaluated the applicant's methodology for analyzing of the assumed control REA and finds the assumptions, calculation techniques, and consequences acceptable. The staff finds that acceptance criteria for fuel enthalpy and fuel enthalpy rise under a REA to be conservative for preventing fuel damage. The staff determined that the calculational procedures are clear and the acceptance criteria are sufficiently conservative, both in initial assumptions and analytical models, to maintain primary system integrity subject to the limitations and conditions listed in Section 6 of this SER. On these bases, the staff determined that the methodology presented in the TR is acceptable for REA analyses.

6 LIMITATIONS AND CONDITIONS

The staff's approval is limited to the application of this methodology to the NuScale reactor design with the following limitations and conditions:

1. An applicant or licensee referencing this report is required to demonstrate the applicability of the REA methodology to the specific NPM design. The use of this methodology for a specific NPM design requires the NRC staff review and approval of the applicant or licensee determination of applicability.

2. The REA methodology is limited to evaluation of REAs for fuel that has not experienced significant depletion with control rods inserted, such as from non-baseload operation.

3. The staff's approval is limited to the use of the REA methodology with TR-0616-48793-P-A, Revision 1 (Reference 14), "Nuclear Analysis Codes and Methods Qualification," and TR-108601-P, Revision 4 (Reference 13), "Statistical Subchannel Analysis Methodology, Supplement 1 to TR-0915-17564-P-A, Revision 2, Subchannel Analysis Methodology."

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Section B

Licensing Topical Report

Rod Ejection Accident Methodology

April 2025

Revision 3

Docket: 99902078

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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 (NPM). 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 (RG) 1.236, NUREG-0800 Standard Review Plan (SRP) Section 4.2, and SRP Section 15.4.8.

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, including transient fuel enthalpy and temperature increases, is analyzed using VIPRE-01. The software is validated for use to evaluate 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 calculations and appropriate sensitivity analyses.

NuScale intends to use this methodology for REA analysis of NPM designs. 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 (NRC) review and approval to use the methodology described in this report for design-basis REA analyses of NPM designs.

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Executive Summary

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

NUREG-0800, SRP, Section 15.4.8 (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 designs 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 applies a bounding value of cladding excess hydrogen content 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 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 RG 1.236 (Reference 8.2.2), and SRP Sections 15.4.8 and 4.2. This guidance addresses: 1) maximum RCS pressure, 2) fuel cladding failure, 3) core coolability, and 4) fission product inventory.

This report describes the software codes used to evaluate the REA along with appropriate validation for its use in NPM applications. The software is controlled under the NuScale quality assurance program (Reference 8.1.3). 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).

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- 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, thermal energy fields, radial fuel rod temperature and enthalpy profiles in reactor cores.

This report presents the findings documented in NUREG/CR-6742 (Reference 8.2.25), "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 8.2.13) 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
- detailed thermal-hydraulic and 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 8.2.10), were used to increase convergence and reliability of the final results. The deviations from the subchannel methodology are discussed and justified in this report.

This report describes representative sample calculations employing the REA methodology and demonstrates how the REA behaves when modeling NPM designs. However, NuScale is not seeking approval of the results provided in this report. Appropriately biased key inputs are used for the sample calculations. The NRELAP5 sensitivity studies evaluate changes to RCS average temperature, loss of offsite power, and RCS flow. VIPRE-01 sensitivity calculation results are also provided. Results of the sensitivity cases are discussed. Trends of the important parameters are also presented.

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The REA methodology meets the regulatory requirements following the approved regulatory guidelines. The results of the sample calculations using the REA methodology are provided in the report to demonstrate that the methodology meets the regulatory criteria from References 8.2.2, 8.2.3, and 8.2.4 by meeting the NuScale criteria defined in this report.

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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 NPM designs.

1.1 Purpose

The purpose of this report is to describe the methodology that NuScale intends to use for REA analysis of NPM designs. This methodology is used in the analysis that supports results reported in Section 15.4.8 of a 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

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Term	Definition
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
RG	regulatory guide
RPV	reactor pressure vessel
SAF	single active failure
SRP	Standard Review Plan
TH	thermal-hydraulics
WRSO	worst rod stuck out

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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
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

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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 derived from the General Design Criteria (GDC) of 10 CFR 50 (Reference 8.2.1) Appendix A, specifically GDC 13 and GDC 28.

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

This methodology considers the criteria provided in NUREG-0800, the Standard Review Plan (SRP), Sections 4.2 and 15.4.8 (Reference 8.2.3 and Reference 8.2.4) and the guidance described in Regulatory Guide (RG) 1.236 (Reference 8.2.2).

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 be grouped into the following categories: RCS pressure, fuel cladding failure, core coolability, and fission product inventory. Section 2.2 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.

2.1.1 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.

2.1.2 Fuel Cladding Failure

The regulatory criteria for evaluating fuel cladding failure are defined in References 8.2.2 and 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.

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- 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 B-1 of Reference 8.2.3. This criterion is bounded by the conservative application of the change in enthalpy limit as a function of cladding excess hydrogen given in Reference 8.2.2.

2.1.3 Core Coolability

The regulatory criteria for evaluating core coolability are defined in References 8.2.2 and 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 excess hydrogen content delineated in Section 2.1.2 is met. In addition, the NuScale criteria adopted and delineated in Section 2.2.3 establishes significant margin to the first two criteria. Therefore the last two criteria above are eliminated.

2.1.4 Fission Product Inventory

The regulatory criteria for evaluating the fission product inventory are defined in Appendix B of Reference 8.2.2 and in Reference 8.2.3. This criteria is not applicable because fuel failures are not permitted in the methodology described in this topical report.

The revised transient fission gas release (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) = $[(0.26 * \Delta H) - 13]$
- Peak Pellet Burnup ≥ 50 GWd/MTU: Transient FGR (percent) = $[(0.26 * \Delta H) - 5]$

where,

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

ΔH = fuel enthalpy increase (Δcal/g)

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2.2 Regulatory Criteria for NuScale

Table 2-1 summarizes how the regulatory acceptance criteria from References 8.2.2, 8.2.3, and 8.2.4 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	2.2.1	5.3
Hot zero power (HZP) fuel cladding failure	2.2.2	2.2.2
FGR effect on cladding differential pressure	2.2.2	N/A
CHF fuel cladding failure	2.2.2	2.2.3
Cladding excess hydrogen-based PCMI failure	2.2.2	5.4.3
Incipient fuel melting cladding failure	2.2.2	2.2.2
Peak radial average fuel enthalpy for core cooling	2.2.3	2.2.4
Fuel melting for core cooling	2.2.3	2.2.3
Fission product inventory	2.2.4	5.5

2.2.1 Reactor Coolant System Pressure

The maximum RCS pressure acceptance criterion of 120 percent of design pressure is used in the methodology. For an NPM design pressure of 2200 psia, for example, the peak pressure during the REA is limited to 2640 psia. RCS conditions are calculated with the NRELAP5 code.

2.2.2 Fuel Cladding Failure

The criteria for evaluating fuel cladding failure are listed below.

- 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.
- 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 cladding excess hydrogen dependent limit depicted in Figure 5-2. A conservative treatment of Figure 5-2 is applied through a single acceptance criteria of 33 cal/g.
- 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

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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 is determined using Equation 12-3 from Reference 8.2.11 while applying a conservative pellet burnup value. Equation 12-3 is applicable for peak rod average burnup to 62 GWd/MTU as identified in Reference 8.2.11.

2.2.3 Core Coolability

The regulatory criteria for evaluating core coolability are defined in Reference 8.2.2 and 8.2.3. The following criteria 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 concerns 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. PCMI failures are precluded by assuring that the criterion for limiting cladding excess hydrogen content delineated in Section 2.2.2 above is met. In addition, the core coolability NuScale criteria delineated above establishes significant margin to the first two criteria from Section 2.1.3. Therefore the last two criteria from Section 2.1.3 are eliminated.

2.2.4 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 cladding fission product barrier will not be breached.
- The regulatory fuel cladding failure criteria in Section 2.2.2, 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 Reference 8.2.4 to perform a dose analysis is not required for the NuScale REA methodology.

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3.0 Overview and Evaluation Codes

This section describes the REA and the applicable codes used to model the event for NPM designs.

3.1 Overview

The cause and progression of the REA is described in References 8.2.2 and 8.2.4. For NPM designs, 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 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. PDIL insertion depth increases as power decreases. Therefore, higher power cases produce lower ejected CRA worth, and 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 a greater positive 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:

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

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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 8.2.12) 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 8.2.12) 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 5.3.1.

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 and fuel response. 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

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the core thermal response and to calculate the MCHFR, peak fuel temperature, and enthalpy. The software is controlled under the NuScale quality assurance program (Reference 8.1.3). Figure 3-1 depicts the computer codes used and the flow of information between codes and evaluations to address the regulatory acceptance criteria.

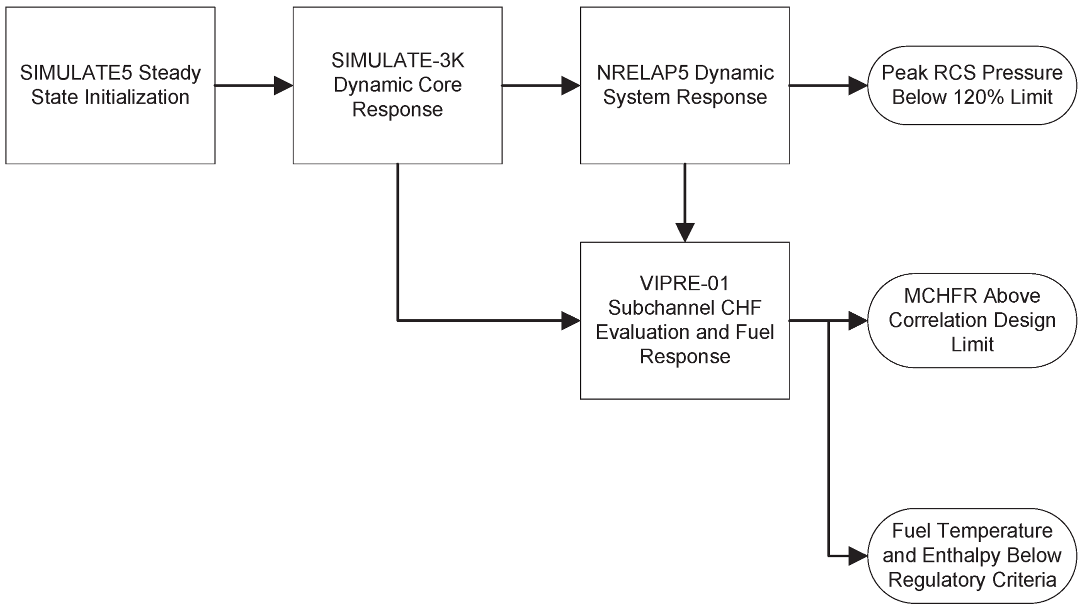

Figure 3-1 Calculation schematic for analyzing rod ejection accident

Section 5.2 through Section 5.5 further describe how the power as a function of time and elements of the power distributions calculated by SIMULATE-3K are used as input to NRELAP5 and VIPRE-01. 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 3-1, below, arranged such that the discipline in the row receives input from the discipline defined in the column.

Table 3-1 High-level discipline and code interface

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
Subchannel (VIPRE-01)	N/A	Radial power distribution (includes $F_{\Delta H}$), axial power distribution	Event thermal-hydraulic response (power, flow, temperature, pressure)

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3.2.1 Core Response

Reference 8.2.6 provides the validation of CASMO5/SIMULATE5 to perform steady state neutronics calculations for NPM designs. Validation of SIMULATE-3K for NPM designs is described in this section.

3.2.1.1 CASMO5

CASMO5 (Reference 8.2.15) 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 8.2.6.

3.2.1.2 SIMULATE5

SIMULATE5 (Reference 8.2.16) 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

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worth, shutdown margin, power distributions, delayed neutron fraction, and peaking factors.

In general, the SIMULATE5 core model is based on input of the core geometry and material compositions, core operating conditions, and core configuration. Core geometry is fully represented with radial nodes corresponding to a quarter of an assembly at numerous axial levels. Material properties and cross-sections are assigned to each node.

Section 4.1 identifies that ejected rod worth is the highest ranked phenomena for this event, as well as describing the two sub-components of worth: control rod and flux redistribution. Static worth is the difference between two static criticality calculations. Reference 8.2.6 provides a robust justification for the ability of SIMULATE5 to accurately predict critical conditions, power distributions, and depletion for of a broad range of reactor designs and operating conditions. Therefore, the SIMULATE5 code provides an excellent tool for predicting ejected rod worths. Additionally, code uncertainty, in the form of nuclear reliability factors (NRFs), conservatively bound differences in code prediction to measurement results and corresponding measurement uncertainty.

Control rod worth is primarily a function of cross-sections and number densities, resulting in a straightforward validation assessment.

The other component of ejected rod worth, flux redistribution, is more complex, with dependence on current power distribution (and the corresponding depletion history that is the integration of power distribution throughout irradiation history). As a result, the intra-assembly power gradient is important for the determination of ejected control rod worth, and thus for the dynamic consequences of the event. Like existing PWRs, an intra-assembly power gradient exists in all assemblies in the NPM core, especially for assemblies on the periphery. This gradient is due to geometric buckling (i.e., radial leakage). Such gradients occur regardless of core design, though lower leakage cores will tend to exhibit lower intra-assembly power gradients than higher leakage cores. Therefore, bounding evaluations of all intra-assembly power gradients, which inherently includes the corresponding depletion history, are evaluated in each application of this method. Because SIMULATE5 has been shown to reliability predict critical conditions, power distributions, and depletion, it also reliably predicts the ejected rod worth, including consideration of phenomena such as intra-assembly power and depletion gradients.

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 8.2.6.

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3.2.1.3 SIMULATE-3K

SIMULATE-3K (References 8.2.17, 8.2.18, and 8.2.19) 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.

The SIMULATE-3K input file is modified for differences between the codes, {{

}}2(a),(c) 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 5.2.1.1), and providing transient-specific inputs (Section 5.2.1.2).

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.

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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 during an REA includes comparisons to steady state neutronics calculations from SIMULATE5, multiple transient benchmark studies performed by the code vendor, Studsvik Scandpower Inc. (Studsvik), and benchmarks performed by NuScale described in this section.

Steady-state neutronics calculation comparisons between SIMULATE-3K and SIMULATE5 demonstrate the ability of the SIMULATE-3K neutronics calculation methodology to accurately predict the effects of 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.

Section 3.2.1.2 provides further discussion on the robust demonstration of the SIMULATE5 predictions of static ejected rod worths and power distributions, including the consideration of important phenomena such as intra-assembly power and depletion gradients. SIMULATE5 to SIMULATE-3K benchmarks over a broad range of operating conditions show excellent agreement for rod worths and power distributions. Therefore, SIMULATE-3K appropriately models these phenomena. Furthermore, code uncertainty is applied in the analysis to conservatively ensure an under-estimation of rod worth does not occur.

SIMULATE-3K REA analysis for NuScale includes SIMULATE5 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 calculated with SIMULATE5 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 steady-state SIMULATE-3K and SIMULATE5 calculations of the effects of these core physics parameters allow for the adoption of the NRFs determined for SIMULATE5 (Reference 8.2.6) to be used by SIMULATE-3K for NuScale REA analysis. Section 7 of Reference 8.2.6 provides additional detail on the determination and application of NRFs to account for code uncertainty in the transient analyses described in Section 5.2.1.

In addition to steady-state comparisons, Studsvik has performed numerous benchmarks demonstrating the ability of SIMULATE-3K to model and accurately predict core physics

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parameters during reactor transients. Two of these benchmarks for REA analysis include experiments performed at the SPERT III E-core research reactor (Reference 8.2.20), and the NEACRP control rod ejection study 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 core size of NPMs. The fuel type (uranium dioxide), moderator, system pressure, and certain initial operating conditions considered for SPERT III are also representative of NPMs. This benchmark demonstrates the ability of SIMULATE-3K to model fast reactivity transients in a PWR core (Reference 8.2.21). Similarities between the NPM designs and the SPERT III core, and notably the small core size, demonstrate applicability and suitability for SIMULATE-3K REA transient analysis of the NPMs.

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 III with generally excellent agreement; differences are within the experimental uncertainty (with few exceptions), and the major and minor phenomena are correctly predicted. In general, the NPM designs are much closer in size and transient conditions to SPERT III than existing PWRs (i.e., there is less distortion). The modeling node size of the experiment and NPM applications is equivalent at one node per 20 to 25 fuel rods (i.e., 4x4 to 5x5 rod matrix); corresponding to a node size of a quarter of an assembly for NPM applications.

The NEACRP control rod ejection study 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 NPMs. The capability of SIMULATE-3K to model reactivity insertions in the NEACRP benchmark analysis (Reference 8.2.23 and 8.2.24) 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 studies. Because peak power is proportional to the square of ejected rod worth above prompt critical, the excellent agreement of peak power for the same ejected rod worth in the benchmarks also demonstrates excellent agreement of ejected rod worth. The SIMULATE-3K results for each of these benchmark studies establish the ability of the code to accurately model an REA transient event and predict key reactivity and power-related

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parameters. See Appendix A for further details on the NRC acceptance of the validation of SIMULATE-3K.

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 NPM designs is described in References 8.2.8 and 8.2.9.

3.2.3 Detailed Thermal-Hydraulic and Fuel Response

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.

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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 (Reference 8.2.26). 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 (Reference 8.2.14) with the same requirements and qualifications as in the MOD-01 SER. Unless otherwise stated, in the remainder of this report a reference to VIPRE-01 is referring to the MOD-02 version.

The fuel rod model utilized in VIPRE-01 is important to the fuel failure modes of critical heat flux, fuel temperature, and fuel enthalpy as described in Section 2.1. These parameters are addressed in the fuel rod conduction model, where a fuel design-specific calibration to COPERNIC is performed as described in Reference 8.2.11. This calibration calculation develops a conservative radial profile, theoretical density, and gap conductance that captures the effects of heat transfer from the fuel pellet to the clad, and ultimately to the coolant. In the application of the method, sensitivity studies on bounding fuel heat transfer inputs must be performed to determine the limiting condition. This calibration is applicable to rod ejection because extreme rod ejection example cases are utilized in the calibration. Additionally, performing time step sensitivities in application calculations demonstrates the simulation adequately addresses the unique heat generation and conduction characteristics of this event, which impacts heat flux and timing. These sensitivity studies confirm the appropriate resolution of the numerical solution.

The validation and applicability of VIPRE-01 to the NPM designs is described in Reference 8.2.10.

3.2.4 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.

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4.0 Identification of Important Phenomena for Rod Ejection Accident

Reference 8.2.25 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 8.2.7). 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 8.2.13).

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 8.2.25) provides the REA analysis parameters in Tables 3-1 and 3-3. Table 4-1 and Table 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.

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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 identified in Section 5.3.

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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.

The positive reactivity insertion of ejected CRA worth is traditionally separated into two components and reflected accordingly in codes such as SIMULATE-3K, that of control rod and flux shape. Control rod reactivity is designated as the specific absorption of neutrons in the control rod absorbers, resulting in reduced neutron multiplication and thus less reactivity. Flux shape reactivity is the change due to changes in neutron production, absorption and leakage as a result of the power distribution. For example, shifting power from a location of low enrichment to high enrichment will result in higher production and less absorption, resulting in an increase in reactivity. The opposite is also true, shifting power from a location of high enrichment to low enrichment will result in a decrease in reactivity.

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), middle of cycle (MOC), and end of cycle (EOC), as well as at various reactor power values ranging from HZP to hot full power (HFP).

Heat capacity of the fuel is used to calculate the enthalpy and temperature increases in the fuel pellets during the event.

Pin peaking factors are calculated by SIMULATE-3K. The largest pin peaking during the event is used to model the limiting node. An uncertainty factor is applied that captures manufacturing tolerances and modeling uncertainties.

4.2 Electric Power Research Institute Technical Report

The EPRI technical report (Reference 8.2.13) 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

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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 15.4.8 (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. At insertion depths less than the PDIL (i.e., with the rod less inserted), the ejected rod has a smaller static worth. For both prompt critical and sub-prompt critical responses described in Sections 3.1.1.1 and 3.1.1.2, respectively, a smaller static worth ejection for the same rod at the same conditions (i.e., same kinetics parameters and reactivity coefficients) is less limiting.

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 5.0.

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

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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 occur well after the power peak, and consequently well after MCHFR.

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 Sections 3.1.1 and 3.1.2. 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 8.2.25, will still be modeled in a conservative manner as noted in the above item D.

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5.0 Rod Ejection Accident Analysis Methodology

As discussed in Section 3.2, 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 an NPM. 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 calculation results provided in this report are from evaluations performed using an equilibrium cycle.

5.1.2 Cycle Burnup

The REA is analyzed at BOC, MOC, and EOC burnups to bound core reactivity conditions. For prompt critical CRA ejections, it is expected that the limiting MCHFR case will occur at EOC because the delayed neutron fraction is minimized at this time, and a smaller delayed neutron fraction typically maximizes the peak power of the event for a given initial power level. For sub-prompt critical jumps, the limiting MCHFR may not be associated with the maximum peak power.

When analyzing MOC, the time in cycle of maximum peaking will be considered if it does not occur at BOC. This time in cycle may not necessarily correspond to a burnup halfway between BOC and EOC. In the event that MOC is more limiting than BOC or EOC, additional analyses at other MOC points should be performed to ensure the limiting case is identified.

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.

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. For low powers, top peaked axial power shapes are produced which bound possible axial power shapes while operating the core with rods inserted. Therefore, the rod ejection occurs through a bounding top peaked shape to maximize the rod worth. These top peaking conditions provide a simple method to bound operations. The conditions are realistic axial shapes

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because allowed operations could in principle arrive at such an axial offset through a unique operation history. While changing axial power does affect the radial power distribution, this is both expected and necessary given that the two are inherently coupled.

The moderator temperature is a function of core power and is set by the operating strategy for the plant. The NRELAP5 analysis accounts for the flow and temperature response based on power, 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. As a result, this treatment of moderator temperature ensures conservatism of the analysis conditions.

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 control systems could respond to limit the power, pressure, and level excursions. The following balance-of-plant and control system automatic responses are therefore not credited:

- 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, typically assumes that loss of alternating current (AC) power occurs at the time of reactor trip. However, the timing of the loss of AC power has no effect on the rod ejection accident MCHFR results, as shown in Table 6-2.

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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. The core response calculations are performed at various bounding combinations of power and burnup as described in Section 5.1.3 and Section 5.1.2, respectively, 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.

{{

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

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. Treatment of system flow in the NRELAP5 and VIPRE-01 models is described in Section 5.3 and Section 5.4, respectively. Coolant mass flux is a unit conversion from system flow, core bypass fraction, and core flow cross-sectional area. The coolant mass flux passed to SIMULATE-3K for a given initial power is then held constant through a modeling option. Core depletion calculations for entire cycles are performed in SIMULATE5 as described in Section 3.2.1.2 at nominal, high, and low bounds of the system flow analytical limit. The depletion calculations are used to bound variations in cross-section feedback from flow and temperature variation and distributions, both instantaneous and long-term.

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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 8.2.6) and demonstrated to be applicable to SIMULATE-3K. Section 7 of Reference 8.2.6 provides additional detail on the determination and application of the NRFs used to account for code uncertainty.

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)

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. The WRSO is identified for each ejected CRA. If a non-ejected CRA is the WRSO, then it is left at the PDIL position after reactor trip.
- The axial power shape is chosen such that the axial offset is at the highest allowable value.
- {{

}}2(a),(c)

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{{

}}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 applied.

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 a postulated ejected CRA to damage a nearby CRDM.

The power pulse, minimum critical heat flux ratio, peak enthalpy, and peak temperature occur prior to control rod insertion from reactor trip for prompt critical cases. Therefore, the WRSO assumption has no impact on the limiting results for those cases. Regardless of the impact for a given case, WRSO is assumed in the REA analyses.

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. If the core design does not exhibit a one-eighth core or quarter-core symmetric pattern then all regulating control rod locations must be explicitly evaluated. Only the CRAs in the regulating bank are eligible for ejection and considered in the REA methodology.

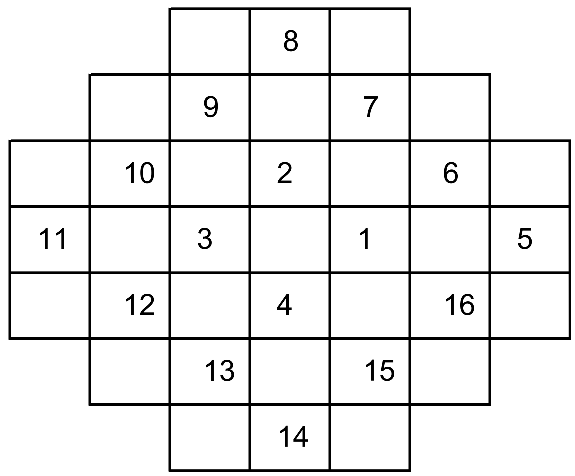

Figure 5-1 Control rod assembly layout for the NuScale Power Module

5.2.2.3.3 Reactor Trips

The NPM includes reactor trip signals for high power and high power rate. These reactor trip signals are modeled in SIMULATE-3K using the output of the excore detectors as described in Section 5.2.1.2. The values for the reactor trip setpoints are input parameters that are specified based on the NPM design.

The example high power rate reactor trip signal used in the sample calculations in Section 6.0 of this report is produced when the core power increases more than 7.5 percent from the initial power level within 30 seconds. The example high power reactor trip signal is produced when the core power exceeds 115 percent of rated power if the initial condition is above 15 percent power; the example low power 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 are used to minimize the power decrease that occurs after the initial power jump. Specific uncertainties applied are listed in Table 5-1.

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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 8.2.6 and 8.2.10.

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
$F_{\Delta H}$ 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 8.2.9); for the system analysis using NRELAP5, REA utilitizes this methodology. However, in order to assess the NuScale-specific criteria outlined in Section 2.2, 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.

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

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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.

The initialization and treatment of uncertainties of the system thermal-hydraulic parameters of moderator temperature and system flow are described as follows. 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. In the NRELAP5 analysis, 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. For hot zero power, the flow rate in NRELAP5 is modeled based on the natural circulation curve of a very low power (e.g., 0.001 percent) and the flow rate in SIMULATE-3K is modeled assuming a conservatively low value (e.g., 5 percent of rated flow).

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.

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- Variable (high and low) core pressure: the flow is 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.

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5.3.2.2 Core Power

Initial 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 for an example core power uncertainty of 2%.

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 MCHFR and overpressure (see Sections 5.3.1.1 and 5.3.1.2).

5.3.2.5 Core Flow

Low core flow is conservative for both MCHFR and overpressure (see Sections 5.3.1.1 and 5.3.1.2).

5.3.2.6 System Pressure and Pressurizer Level

System pressure and pressurizer level are addressed for MCHFR and system pressurization (see 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 8.2.9); 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 Detailed Thermal-Hydraulic and Fuel 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.

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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 8.2.10 are used to increase the convergence and reliability of the final results. These changes are described below.

- {{

  }}2(a),(c)

- The radial nodalization of the subchannel basemodel is a {{

  }}2(a),(c) The phenomenological characteristics of the rod ejection event is unique compared to other events. For a rod that does not experience critical heat flux, the thermal-hydraulics change negligibly while the nuclear physics change dramatically. Sensitivity studies are used to confirm the radial nodalization of the model accurately maintains the hot channel flow field and results in a conservative MCHFR. Reference 8.2.10 demonstrates {{

  }}2(a),(c)

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{{

}}2(a),(c)

The default convergence parameters and options for use in VIPRE-01 input is provided in Table 5-2.

Table 5-2 Default VIPRE-01 convergence parameters and options

Variable	Value	Description
{{		
		}}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 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. The peak neutron power will occur after the rod is fully ejected and therefore will represent a skewed power distribution.

With the statistical subchannel methodology defined in Reference 8.2.10, radial peaking uncertainties other than the $F_{\Delta H}$ pin peaking NRF are treated within the CHF analysis limit. However, the applicable $F_{\Delta H}$ engineering uncertainty for the enthalpy and fuel temperature acceptance criteria (contained in the CHF analysis limit for the MCHFR acceptance criteria) must be applied for these acceptance criteria. Therefore, two options exist for appropriately treating the radial peaking uncertainties for all acceptance criteria.

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- {{

}}2(a),(c)

Because the analysis is performed for each unique operating cycle core design, the three-dimensional power distribution for each of the screened cases from SIMULATE-3K are analyzed in VIPRE-01. Therefore, design variations, such as enrichment, burnable poison, and loading patterns are captured in these detailed power shapes. The process for transferring the SIMULATE-3K power distribution output at the time of peak power inputs to VIPRE-01 for each case is described. SIMULATE-3K offers choices for editing this information, from fully-detailed powers at each time step, to only a specific assembly at a specific time step. Depending on the preference of the analyst, simple lookup of the required output or an iterative approach may be needed to determine the correct time-step and hot assembly. Once the availability and location of this SIMULATE-3K information is determined, it is transferred to VIPRE-01 input. {{

}}2(a),(c)

The conservative nature of this modeling is described in Section 5.4.1.1. Additionally, as described in Section 6.4.2 of Reference 8.2.10, 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 NPM cores 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.

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5.4.2.3 Core Inlet Flow Distribution

The inlet flow distribution for subchannel analyses is described in Reference 8.2.10. 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 Heat Transfer

Bounding fuel heat transfer inputs are used. Sensitivity studies show that high values are more conservative for REA CHF calculations. Section 6.3.6 discusses the effect of a wide range of heat transfer values on MCHFR.

5.4.3 Fuel Response Calculation Procedure

VIPRE-01 is used to calculate the peak radial average fuel enthalpy and maximum rise in order to evaluate acceptance criteria established in Reference 8.2.3. For cladding excess hydrogen the NuScale fuel design uses cladding which is an unlined recrystallization annealed (RXA) fuel cladding. Empirically-based PCMI cladding failure threshold curves for RXA at or above 500°F and below 500°F (from Reference 8.2.3) are applicable to the NuScale fuel design and are shown in Figure 5-2. The most conservative application of these criteria are applied; a limit of 33 cal/g is established so the initial cladding temperature and exposure is not tracked and the excess cladding hydrogen content is not calculated.

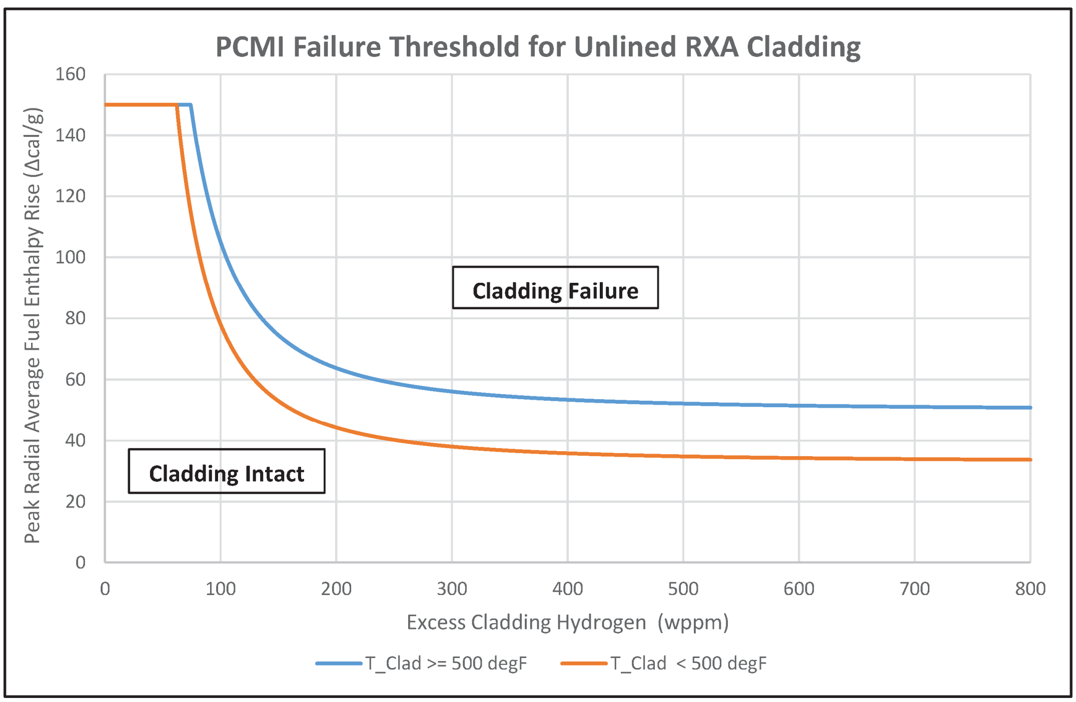

Figure 5-2 PCMI failure threshold curves for unlined RXA fuel cladding temperatures equal to or above 500 °F, and below 500 °F

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5.4.4 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.

5.4.5 Sensitivity Studies

Table 5-3 provides a summary and cross-reference of the sensitivity studies associated with the subchannel REA evaluations. Some of the sensitivity studies are mandatory while others are optional. Mandatory sensitivity studies are those that are required in each implementation of the method. Optional sensitivity studies are those that are only required to be performed if a non-default nodalization, parameter, or option is utilized.

In general, the purpose of these parametric sensitivity studies is to demonstrate a reliable and converged solution. In addition to the VIPRE-01 assessment of convergence (yes or no), Courant number, and mass and energy errors, several other output parameters are evaluated. The sensitivity studies focus on the MCHFR results because MCHFR is typically the most limiting acceptance criteria. In most cases, there is no difference in sensitivity among the different acceptance criteria. The sensitivity studies are performed to look for trends or differences that would indicate that the solutions are not properly converged. The procedure for each sensitivity study is to change an input parameter or modeling feature to alternate values that cover a range of possible values above and below the reference value. If the input parameter or modeling feature is a binary variable rather than a continuous variable, then the sensitivity study uses the single other value.

Regulatory Guide 1.203 (Reference 8.2.30) provides acceptance criteria for the ability of a computer code to model phenomena of interest when compared to experimental data. In the context of code-to-data comparisons, RG 1.203 provides a definition for the phrase “Excellent Agreement”. This definition is used as a model for developing acceptance criteria for the rod ejection subchannel sensitivity studies. Each sensitivity study should exhibit “Excellent Agreement” between the reference case and the other cases with different values. Specifically, “Excellent Agreement” is defined in this context to be when “sensitivity cases (as compared to a reference case) exhibit no deficiencies in modeling a given behavior. Major and minor phenomena and trends are correctly predicted and agree closely with the reference case.” If “Excellent Agreement” is not achieved, then the simulation is unreliable and therefore cannot be used. If this occurs, the analyst shall investigate the cause of the issue and make necessary adjustments to analyses or the cycle-specific core design and operating limits.

There are three exceptions from the general formulation described above. The first exception is with respect to the fuel heat transfer inputs described in Section 5.4.2.4. Consistent with Table 5-3 (row 1), the purpose of the sensitivity study for fuel heat transfer inputs that is performed in each implementation of the methodology is to determine appropriate inputs to calculate limiting values for the acceptance criteria. The direction of fuel heat transfer inputs that lead to limiting results may change depending on the specific event trajectory. The second exception is the time-step size plot that compares the power as a function of time calculated by SIMULATE-3K versus the discretization of that boundary condition input used in the VIPRE-01 simulation. The acceptance criterion for this plot is that the linear interpolation between two discrete points of the VIPRE-01

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boundary condition must be in "Excellent Agreement" with the reference. The third exception is for the radial nodalization sensitivity study. It is expected that the single channel model will be more conservative than the other models as shown in Figure 6-11 and discussed in Section 6.3.5. The acceptance criterion for this sensitivity is to demonstrate conservatism of the selected model since the trends for the models may not meet the threshold for excellent agreement. This acceptance criterion includes both an explicit quantitative comparison of the values and a reasonable explanation for observed differences.

Table 5-3 Sensitivity studies for rod ejection subchannel evaluations

Description	Required?	Purpose	Acceptance Criteria	Comments	Example
Fuel heat transfer inputs	Mandatory	Determine limiting value for acceptance criteria	N/A	None	Figure 6-12
Time-step size plot	Mandatory	Confirm valid solution	Excellent Agreement	Resolve phenomena	Figure 6-5
Two-phase flow correlations	Mandatory	Confirm valid solution	Excellent Agreement	Courant number	Figure 6-7
Axial nodalization	Optional – only required if subchannel default nodalization is not used	Confirm valid solution	Excellent Agreement	Courant number, resolve phenomena	Figure 6-6
Radial nodalization	Optional – only required if subchannel default nodalization is not used	Confirm valid solution	Demonstrate Conservatism	Appropriate and conservative local conditions	Figure 6-11
Convergence parameters	Optional – only required if subchannel default parameters are not used	Confirm valid solution	Excellent Agreement	Prevent false convergence	Figure 6-8
Convergence option deviations	Optional – only required if subchannel default options are not used	Confirm valid solution	Excellent Agreement	Prevent false convergence	N/A

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5.5 Radiological Assessment

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

5.6 Method Summary

As introduced in Section 3.2, four separate codes are required to model each rod ejection transient with a high-level flow depicted in Figure 3-1. At each code interface, Table 3-1 provides the information transferred between codes. However, to appropriately bound allowed operations, apply uncertainties, and confirm validity of simulation results, more than one simulation per code is often required. Table 5-4 provides a summary of the simulation types required to fully implement the method.

Table 5-4 Summary of simulation types needed to implement method

Code	Simulation Type Description	Downstream Code	Purpose
SIMULATE5	Worst rod stuck out	-	Valid input
SIMULATE5	Pre-ejection conditions	SIMULATE-3K	-
SIMULATE-3K	Iterative determination of MTC	-	Valid input
SIMULATE-3K	Confirm rod worth multiplier	-	Valid input
SIMULATE-3K	Nuclear transient simulation	NRELAP5	-
SIMULATE-3K	Re-run transient with power distribution edits	VIPRE-01	-
NRELAP5	Core boundary conditions	VIPRE-01	-
NRELAP5	Peak pressurization case	-	Acceptance criteria
VIPRE-01	Core thermal-hydraulics simulation	-	Acceptance criteria
VIPRE-01	Sensitivity - fuel heat transfer	-	Limiting value
VIPRE-01	Sensitivity - two-phase flow correlations	-	Valid solution
VIPRE-01	Sensitivity - axial nodalization if non-default	-	Valid solution
VIPRE-01	Sensitivity - radial nodalization if non-default	-	Valid solution
VIPRE-01	Sensitivity - convergence parameters if non-default	-	Valid solution
VIPRE-01	Sensitivity - convergence options if non-default	-	Valid solution

These simulation types are applied for each combination of inputs in the case matrix, which is unique to each discipline as described in this section. The core response

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(SIMULATE5 and SIMULATE-3K) develop a case matrix that parameterizes initial power level, cycle exposure, and ejected rod location. The cases that correspond to the limiting peak powers for each initial power level are provided for system response (NRELAP5). In the system response, each case is run with initial reactor coolant system temperature bias. Transient core boundary conditions for each case are provided for use in detailed core thermal-hydraulics and fuel response (VIPRE-01). Additionally, simulations to check that the peak reactor coolant system pressure is found are performed. VIPRE-01 simulations of transients are performed to determine the limiting case. Upon determination of the limiting case, the required sensitivity studies as dictated by Section 5.4.5 are performed. As a result of completion of the simulation types for the cases in each case matrix, limiting values for each acceptance criteria may be compared to the criteria defined in Table 7-1.

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6.0 Sample Rod Ejection Calculations

Examples of implementing calculations, as well as sensitivity studies, are presented to provide context. The assumptions and inputs used in the examples in Section 6.0 are not required to be used in the applications of the methodology. Instead, application-specific sensitivity studies or other evaluations are required to determine the appropriate assumptions and input for a given core design, consistent with the methodology in Section 5.0.

Table 6-1 lists each of the example figures provided in this section and identifies the purpose of the figure. Specifically, Table 6-1 identifies whether the figure is a sub-step from an example calculation or an example of a sensitivity study as described in Section 5.4.5.

Table 6-1 Sensitivity studies for rod ejection subchannel evaluations

Figure Number	Figure Purpose	Figure Title
Figure 6-1	Example result	Power response at 55 percent power, end of cycle
Figure 6-2	Example result	Power response at 100 percent power, beginning of cycle
Figure 6-3	Example result	Power response for peak reactor coolant system pressure evaluation
Figure 6-4	Example result	Pressure response for peak reactor coolant system pressure evaluation
Figure 6-5	Example sensitivity study (Table 5-3)	Time step effect on power forcing function
Figure 6-6	Example sensitivity study (Table 5-3)	Effect of axial node size (inches) on critical heat flux
Figure 6-7	Example sensitivity study (Table 5-3)	Effect of VIPRE-01 two-phase flow model options on critical heat flux
Figure 6-8	Example sensitivity study (Table 5-3)	Effect of VIPRE-01 damping factors on critical heat flux
Figure 6-9	Example input	Example case-specific core radial power distribution at time of peak power
Figure 6-10	Example input	Example case-specific hot assembly radial power distribution at time of peak power
Figure 6-11	Example sensitivity study (Table 5-3)	Radial nodalization sensitivity MCHFR comparison
Figure 6-12	Example sensitivity study (Table 5-3)	Effect of heat transfer inputs on critical heat flux

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Figure 6-1 shows an example of the power response at 55 percent and EOC, which is the highest power case of an example core design and operational limits. 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 due to the uncertainty treatment until the CRAs start moving.

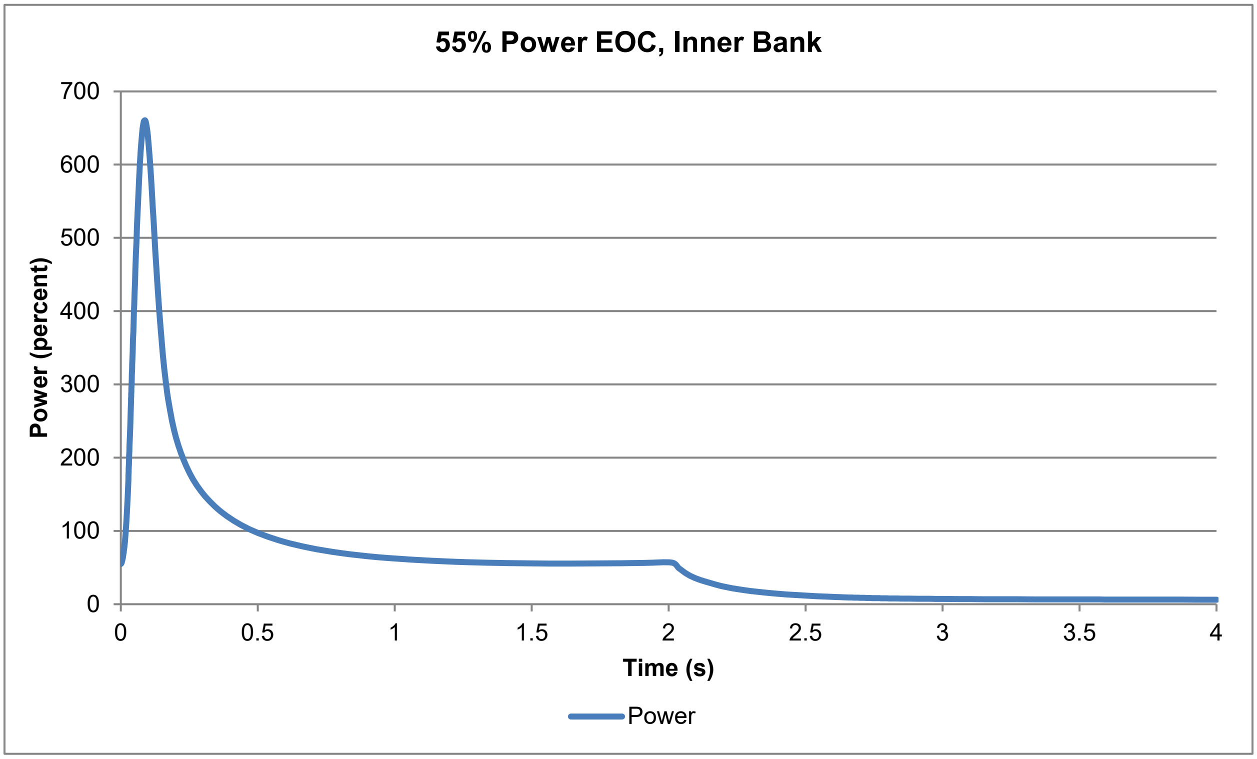

Figure 6-1 Power response at 55 percent power, end of cycle

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 module protection system limits are not reached and the long term power comes to a new equilibrium steady state power around 106 percent.

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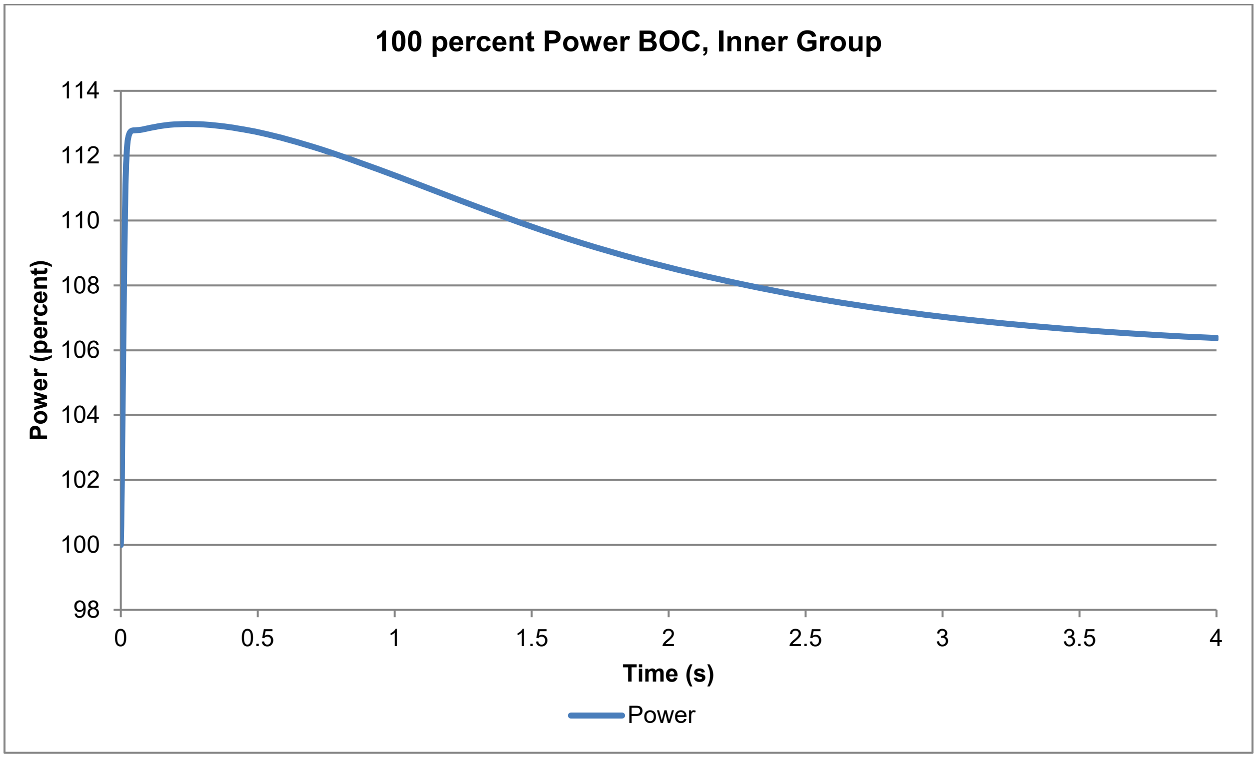

Figure 6-2 Power response at 100 percent power, beginning of cycle

6.1 Rod Ejection Accident Sample Analysis System Pressure Response Results

Figure 6-3 provides the power response for an example peak RCS pressure evaluation. Figure 6-4 provides the peak RCS pressure response with this example 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 high reactor pressure is reached. Pressure increases because of the increased power and also because of the loss of AC power assumed at the time of reactor trip. The peak pressure reached during this example REA is 2076 psia.

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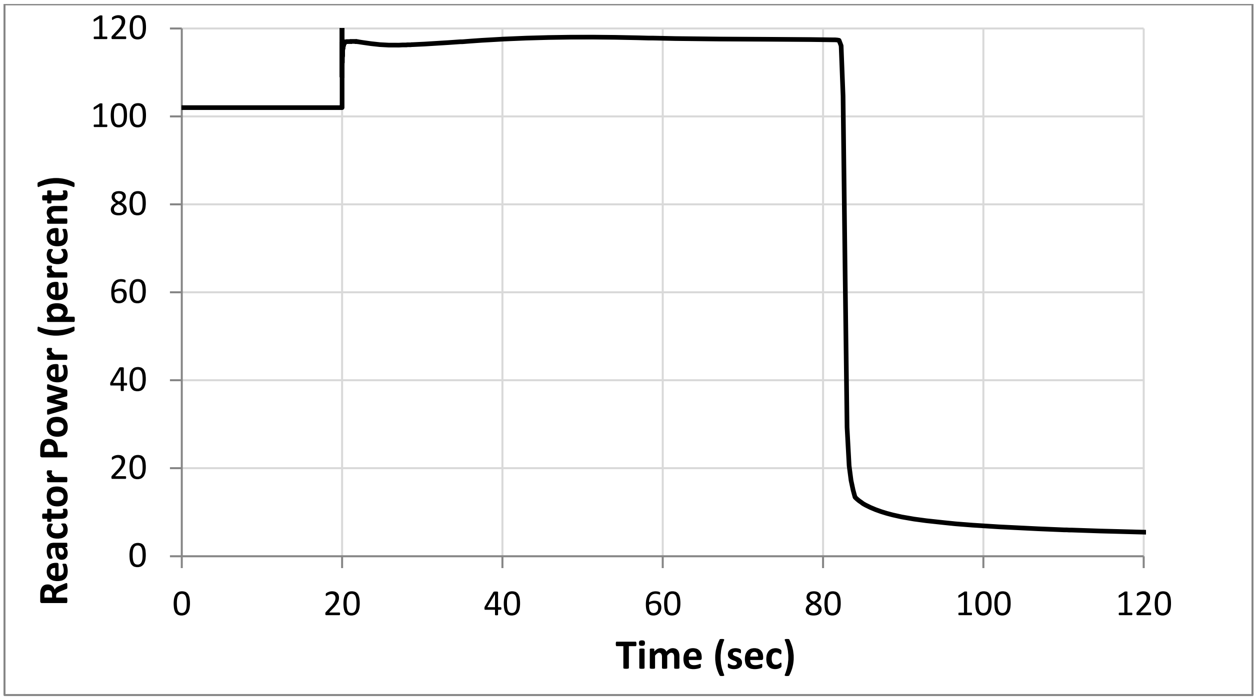

Figure 6-3 Power response for peak reactor coolant system pressure evaluation

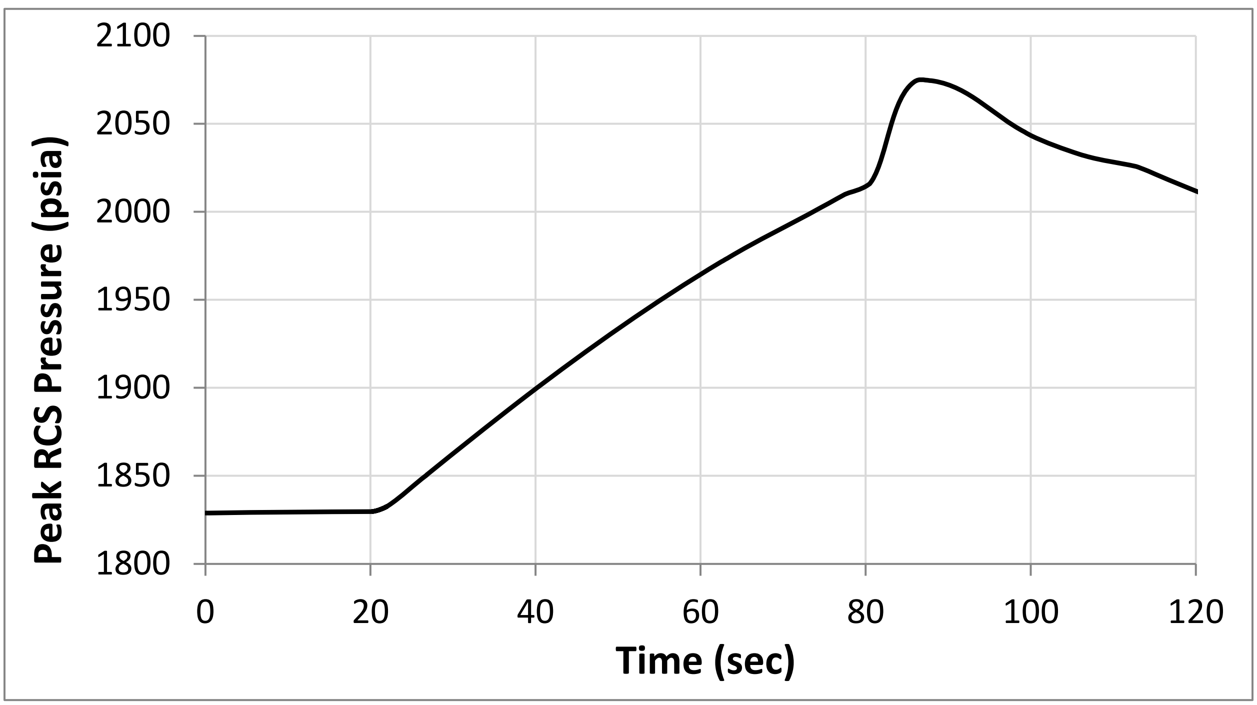

Figure 6-4 Pressure response for peak reactor coolant system pressure evaluation

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6.2 NRELAP5 Minimum Critical Heat Flux Ratio Impacts

Table 6-2 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 6-2 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

6.3 VIPRE-01 Sensitivities

Section 5.4.5 defines the mandatory and optional subchannel sensitivity analyses. The following sections provide example results for the subchannel sensitivity analyses.

6.3.1 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.

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{{

}}2(a),(c)

Figure 6-5 Time step effect on power forcing function

6.3.2 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) As described in Section 5.4.1.1, the axial nodalization used must be shown to result in a reliable and converged solution on an application-specific basis.

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}}2(a),(c)

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

6.3.3 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

6.3.4 Numerical Solution Damping Factors

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

}}2(a),(c)

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}}2(a),(c)

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

6.3.5 Radial Power Distribution and Nodalization

Figure 6-9 provides an example core radial power distribution, while Figure 6-10 provides the hot assembly radial power distribution from the limiting statepoint at time of peak power. In the default radial nodalization, these power distribution inputs would be used to represent all fuel rods in the core. {{

}}2(a),(c)

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{{

}}2(a),(c)

Figure 6-9 Example case-specific core radial power distribution at time of peak power

{{

}}2(a),(c)

Figure 6-10 Example case-specific hot assembly radial power distribution at time of peak power

If the default radial nodalization is not used, a sensitivity study is required as described in Section 5.4.5. An example of the single channel radial nodalization for a different case with a peak power of roughly 300% rated power is provided. For this sensitivity study, three different nodalization schemes are examined of {{

}}2(a),(c) The results from this sensitivity study are plotted in Figure 6-11.

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{{

}}2(a),(c)

Figure 6-11 Radial nodalization sensitivity MCHFR comparison

As expected from the reasoning provided in Section 5.4, the timing and magnitude of the decrease in MCHFR as the power increases and then is turned around by the Doppler feedback is close for the three cases, with the {{

}}2(a),(c) This sensitivity provides an example justification that the single channel radial nodalization is appropriate for this particular case. As noted above, each implementation of the single channel model for a limiting case requires a similar sensitivity to confirm applicability.

6.3.6 Fuel Rod Heat Transfer

Figure 6-12 provides an example sensitivity study as described in Section 5.4.5 for the purpose of determining limiting values for acceptance criteria. Figure 6-12 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. {{

}}2(a),(c)

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{{ }}2(a),(c) Therefore, the fuel heat transfer input sensitivity analyses required by Section 5.4.5 consider the difference in impact for different acceptance criteria.

{{

}}2(a),(c)

Figure 6-12 Effect of heat transfer inputs on critical heat flux

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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 1.236 and SRP Sections 4.2 and 15.4.8. NuScale intends to use this methodology for REA analysis of NPM designs. 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, including transient fuel enthalpy and temperature increases, is analyzed using VIPRE-01. The software is validated for use to evaluate 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 calculations and sensitivity analyses.

Section 6.0 of this report provides sample REA sensitivity calculations. These data provide confirmation that the method for satisfying the regulatory acceptance criteria outlined in Section 2.1 are appropriate. The regulatory acceptance criteria 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 pressure.
- fuel cladding failure. Transient enthalpy rise is well below the criteria for HZP, intermediate, and HFP conditions considering fuel rod differential pressure at HZP and cladding excess hydrogen with a wide margin. The subchannel model also predicts that the peak fuel centerline temperature is well below the incipient melting point. For the limiting critical heat flux (CHF) cases 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, because no fuel rod failure is allowed and the highest rod differential pressure is assumed for the HZP requirement of transient fuel enthalpy rise.

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

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

Parameter	Criteria	Sample Evaluation Results – Limiting Case
Maximum RCS pressure	≤ 120% design	2076 psia (94.4% design)
HZP fuel cladding failure (average enthalpy)	< 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 > CHF analysis limit	1.47
Cladding excess hydrogen-based PCMI failure	< 33 Δcal/g	11.9 Δcal/g
Incipient fuel melting cladding failure	< incipient fuel melt limit	2162 °F
Peak radial average fuel enthalpy for core coolability	< 230 cal/g	84.0 cal/g
Fuel melting for core cooling	< incipient fuel melt limit	2162°F
Fission product inventory	2.3.4	N/A

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8.0 References

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Appendix A. NRC Acceptance of NuScale Validation of SIMULATE-3K

The NRC reviewed NuScale's benchmark of SIMULATE-3K against a selection of SPERT-III cold startup tests for each statepoint, generally corresponding to the highest static worth for the statepoint (Reference 8.2.21). NuScale compared the SPERT-III conditions with the NPM operating parameters and demonstrated that the SPERT-III test conditions were generally representative of the NPM core designs from a reactivity-initiated accident perspective (References 8.2.27 and 8.2.28). Table A-1 provides the comparison of conditions to SPERT-III, Table A-2 summarizes the SPERT-III cases selected, and Table A-3 provides the results. Additional comparison results are shown in Figure A-1 through Figure A-5. The NRC determined that the NuScale results demonstrated generally good agreement between the results predicted by SIMULATE-3K and the SPERT-III experimental results.

Table A-1 Range of comparison for SPERT-III

Parameter	Units	SPERT-III	NuScale Power Module
Reactor Type	-	PWR	PWR
Fuel Material	-	Uranium dioxide	Uranium dioxide
UO_2 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	160-250
Rated Flow	kg/s	1,260	680-820
Design Core Exit Temperature	F	650	590-610
Design Pressure	psia	2,515	1,850-2,000

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Table A-2 Summary of selected SPERT-III cases

Test #	Statepoint Condition	Initial Coolant Temperature (°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

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Table A-3 Tabulated results and comparisons of selected SPERT-III cases

Test #	Peak Power (MW) [Exp. Uncertainty = ±15%]			Integrated Energy (MW-sec) [Exp. Uncertainty = ±17%]			Reactivity Compensation ($) [Exp. Uncertainty = ±11%]		
	S3K	SPERT-III	% Difference	S3K	SPERT-III	% Difference	S3K	SPERT-III	% Difference
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)						

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{{

}}2(a),(c)

Figure A-1 Test 43 SIMULATE-3K comparison to SPERT-III

{{

}}2(a),(c)

Figure A-2 Test 70 SIMULATE-3K comparison to SPERT-III

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{{

}}2(a),(c)

Figure A-3 Test 60 SIMULATE-3K comparison to SPERT-III

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}}2(a),(c)

Figure A-4 Test 81 SIMULATE-3K comparison to SPERT-III

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}}2(a),(c)

Figure A-5 Test 86 SIMULATE-3K comparison to SPERT-III

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Additionally, the NRC reviewed NuScale's verification analysis of the NEACRP REA benchmark performed by Studsvik Scandpower with SIMULATE-3K (Reference 8.2.27). This analysis was performed under NuScale's approved 10 CFR Part 50, Appendix B, quality assurance program. The results of this analysis are presented below.

Table A-4 provides a comparison of the SIMULATE-3K results obtained by NuScale against the NEACRP benchmark reference solutions.

Table A-4 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			
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			}}2(a),(c)

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Parameter	Case	NEACRP	S3K	Δ	%Δ
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)

After review the NRC determined that the results demonstrated good agreement between NuScale's SIMULATE-3K results and the NEACRP benchmark reference solutions. Based on NuScale's analysis results, the NRC found that NuScale demonstrated that SIMULATE-3K can successfully model the NEACRP benchmarks for reactivity-initiated accidents.

The NRC concluded that the NuScale validation of SIMULATE-3K against the SPERT-III experiments and the NEACRP benchmark suite, as discussed above, were acceptable and demonstrated that SIMULATE-3K can be used in its methodology to accurately model a reactivity-initiated accident (Reference 8.2.29).

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Section C

RAI Number	Question Number	NuScale Letter Number
9936	NTR-01 NTR-02	RAIO-125601
N/A	A-RE-02	LO-152560
9306	15.04.08-1 15.04.08-3 15.04.08-4 15.04.08-5 15.04.08-6 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	RAIO-0618-60285
9306	15.04.08-5 15.04.08-6 15.04.08-15 15.04.08-16	RAIO-0119-64377
9306	15.04.08-1	RAIO-0219-64616
9306	15.04.08-1	RAIO-1019-67479

RAIO-125601

September 14, 2022

Docket: 99902078

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 (RAI No. 9936) on the NuScale Topical Report, "Rod Ejection Accident Methodology," TR-0716-50350, Revision 2

REFERENCES:
1. NRC Letter Final Request for Additional Information (RAI) 9936, dated July 13, 2022, RAI# 9936
2. NuScale Topical Report Rod Ejection Accident Methodology, dated December 2021, TR-0716-50350, Revision 2

This letter provides NuScale's response to Reference 1.

NuScale's response to the following RAI Questions from NRC RAI# 9936 are provided in the attached enclosures:

- NTR-01
- NTR-02

Enclosures are grouped with all proprietary version responses first, followed by all nonproprietary version responses. NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit supports this request. 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.

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

Please contact Thomas Griffith at 541-452-7813 or at tgriffith@nuscalepower.com if you have any questions.

Sincerely,

Mark W. Shaver

Mark Shaver
Manager, Licensing
NuScale Power, LLC

Distribution: Bruce Bavol, NRC
Getachew Tesfaye, NRC
Michael Dudek, NRC

Enclosure 1: NuScale Response to NRC Request for Additional Information RAI No. 9936, proprietary

Enclosure 2: NuScale Response to NRC Request for Additional Information RAI No. 9936, nonproprietary

Enclosure 3: Affidavit of Mark Shaver, AF-125602

Enclosure 2:

NuScale Response to NRC Request for Additional Information eRAI No. 9936, nonproprietary

NuScale Power, LLC
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Response to Request for Additional Information
Docket: 99902078

RAI No.: 9936
Date of RAI Issue: 07/13/2022

NRC Question No.: NTR-01

Regulatory Basis:

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. A spectrum of initial conditions for an REA should be considered to ensure the analysis of the event is appropriately bounded.

Regulatory Guide 1.236, Section 2.2.1, "PWR CRE Initial Conditions" states accident analyses should consider the full range of cycle operation from beginning of cycle (BOC) to end of cycle (EOC) and full range of power operation.

Issue:

Section 5.1.2 of TR-0716-50350, Rev 2, states the REA is analyzed at BOC and EOC burnups to bound core reactivity conditions and the expected limiting MCHFR case will occur at EOC. Middle of cycle (MOC) conditions are not explicitly considered, and no justification is provided for the statement that EOC is limiting. Consistent with RG 1.236, the REA method should address the full spectrum of cycle burnup and range of power operation, which includes consideration of all MOC conditions.

Request:

To support a finding that the REA analysis is appropriately bounded, the staff requests that the TR be updated to include consideration of the full spectrum of cycle burnup and range of power operation in the set of initial conditions for the REA analysis, or provide justification for excluding MOC conditions from these initial conditions.

NuScale Response:

As discussed in the background of Regulatory Guide 1.236, the uncontrolled movement of a single control rod out of the core that results in a positive reactivity insertion and prompt local core power increase is considered the limiting reactivity insertion accident. For prompt critical rod ejections, the Fuchs-Nordheim point-kinetics model can be used to predict the maximum power increase as shown below:

$$P_{max} = \frac{1}{2\Lambda\alpha K} \times \left(\frac{\Delta k}{\beta} - 1\right)^2$$

where:

P_{max} = maximum instantaneous power

Δk = static worth of ejected rod

Λ = prompt neutron lifetime

α = fuel temperature reactivity coefficient

β = effective delayed neutron fraction

K = inverse fuel heat capacity

Larger maximum powers occur for larger static worths and smaller prompt neutron lifetimes, fuel temperature reactivity coefficients, and effective delayed neutron fractions. These parameters may vary during an operating cycle. The effective delayed neutron fraction is smallest at end of cycle (EOC) and tends to dominate the other parameters. With a dominant effective delayed neutron fraction minimized at EOC, the Fuchs-Nordheim model would predict the maximum peak power for prompt critical rod ejections at EOC. This predicted behavior has been confirmed by numerous SIMULATE-3K calculations that have demonstrated the maximum powers for EOC are larger than the maximum powers for either beginning of cycle (BOC) or middle of cycle (MOC). For these reasons, TR-0716-50350-P, Rev. 2, referred to EOC as the expected limiting case for minimum critical heat flux (MCHFR).

The power dependent insertion limits (PDILs) restrict the amount by which regulating bank groups can be inserted at power. For higher PDILs, rods have lower static worths and a postulated rod ejection may result in a sub-prompt critical scenario. Equation 3-2 in TR-0716-50350-P, Rev. 2, provides an approximation for determining the prompt jump power. Depending on the static worth, the prompt jump power may not exceed module protection system limits. Instead, the power would reach a new steady state power as shown in the sample result in Figure 6-2 of TR-0716-50350-P, Rev. 2. For these sub-prompt critical scenarios, the MCHFR may be driven by the integrated energy deposited in addition to the prompt jump power. As a result, the limiting MCHFR may not be associated with only the EOC cases that minimize the effective delayed neutron fraction.

Based on the above discussion, BOC and MOC should be considered in addition to EOC. Evaluating a rod ejection accident at BOC, MOC, and EOC is also consistent with Regulatory Guide 1.236, Section 2.2.1.1.

TR-0716-50350-P, Rev. 2, is revised to state that MOC will be considered and to clarify the discussion regarding EOC as limiting.

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.

Additional Information:

The markup also includes correction of unrelated minor typographical errors.

Licensing Topical Report

Rod Ejection Accident Methodology

~~December 2021~~

Draft Revision 32

Docket: 99902078

NuScale Power, LLC

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Licensing Topical Report

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TABLES

FIGURES

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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 (NPM). 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 (RG) 1.236, NUREG-0800 Standard Review Plan (SRP) Section 4.2, and SRP Section 15.4.8.

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, including transient fuel enthalpy and temperature increases, is analyzed using VIPRE-01. The software is validated for use to evaluate 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 calculations and appropriate sensitivity analyses.

NuScale intends to use this methodology for REA analysis in support of the NuScale standard design approval application and for future applications that are appropriately justified and approved. 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 (NRC) review and approval to use the methodology described in this report for design-basis REA analyses in the NPM.

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Executive Summary

The purpose of this report is to describe the methodology that NuScale Power, LLC, intends to use for the analysis of REAs. 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 NPM.

NUREG-0800, SRP, Section 15.4.8 (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 applies a bounding value of cladding excess hydrogen content 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 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 RG 1.236 (Reference 8.2.2), and SRP Sections 15.4.8 and 4.2. This guidance addresses: 1) maximum RCS pressure, 2) fuel cladding failure, 3) core coolability, and 4) fission product inventory.

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).

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- 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, thermal energy fields, radial fuel rod temperature and enthalpy profiles in reactor cores.

This report presents the findings documented in NUREG/CR-6742 (Reference 8.2.25), “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 8.2.13) 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
- detailed thermal-hydraulic and 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 8.2.10), were used to increase convergence and reliability of the final results. The deviations from the subchannel methodology are discussed and justified in this report.

This report describes representative sample calculations 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 calculations. The NRELAP5 sensitivity studies evaluate changes to RCS average temperature, loss of offsite power, and RCS flow. VIPRE-01 sensitivity calculation results are also provided. Results of the sensitivity cases are discussed. Trends of the important parameters are also presented.

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The REA methodology meets the regulatory requirements following the approved regulatory guidelines. The results of the sample calculations using the REA methodology are provided in the report to demonstrate that the methodology meets the regulatory criteria from References 8.2.2, 8.2.3, and 8.2.4 by meeting the NuScale criteria defined in this report.

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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 standard design approval application (SDAA) and other future applications that are appropriately justified and approved. 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.

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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
RG	regulatory guide

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Term	Definition
RPV	reactor pressure vessel
SAF	single active failure
SDAA	standard design approval application
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
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

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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 derived from the General Design Criteria (GDC) of 10 CFR 50 (Reference 8.2.1) Appendix A, specifically GDC 13 and GDC 28.

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

This methodology considers the criteria provided in NUREG-0800, the Standard Review Plan (SRP), Sections 4.2 and 15.4.8 (Reference 8.2.3 and Reference 8.2.4) and the guidance described in Regulatory Guide (RG) 1.236 (Reference 8.2.2).

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 be grouped into the following categories: RCS pressure, fuel cladding failure, core coolability, and fission product inventory. Section 2.2 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.

2.1.1 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.

2.1.2 Fuel Cladding Failure

The regulatory criteria for evaluating fuel cladding failure are defined in References 8.2.2 and 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.

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- 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 B-1 of Reference 8.2.3. This criterion is bounded by the conservative application of the change in enthalpy limit as a function of cladding excess hydrogen given in Reference 8.2.2.

2.1.3 Core Coolability

The regulatory criteria for evaluating core coolability are defined in References 8.2.2 and 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 excess hydrogen content delineated in Section 2.1.2 is met. In addition, the NuScale criteria adopted and delineated in Section 2.2.3 establishes significant margin to the first two criteria. Therefore the last two criteria above are eliminated.

2.1.4 Fission Product Inventory

The regulatory criteria for evaluating the fission product inventory are defined in Appendix B of Reference 8.2.2 and in Reference 8.2.3. This criteria is not applicable because fuel failures are not permitted in the methodology described in this topical report.

The revised transient fission gas release (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) = $[(0.26 * \Delta H) - 13]$
- Peak Pellet Burnup ≥ 50 GWd/MTU: Transient FGR (percent) = $[(0.26 * \Delta H) - 5]$

where,

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

ΔH = fuel enthalpy increase (Δcal/g)

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2.2 Regulatory Criteria for NuScale

Table 2-1 summarizes how the regulatory acceptance criteria from References 8.2.2, 8.2.3, and 8.2.4 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	2.2.1	5.3
Hot zero power (HZP) fuel cladding failure	2.2.2	2.2.2
FGR effect on cladding differential pressure	2.2.2	N/A
CHF fuel cladding failure	2.2.2	2.2.3
Cladding excess hydrogen-based PCMI failure	2.2.2	5.4.3
Incipient fuel melting cladding failure	2.2.2	2.2.2
Peak radial average fuel enthalpy for core cooling	2.2.3	2.2.4
Fuel melting for core cooling	2.2.3	2.2.3
Fission product inventory	2.2.4	5.5

2.2.1 Reactor Coolant System Pressure

The maximum RCS pressure acceptance criterion of 120 percent of design pressure is used in the methodology. For an NPM design pressure of 2200 psia, for example, the peak pressure during the REA is limited to 2640 psia. RCS conditions are calculated with the NRELAP5 code.

2.2.2 Fuel Cladding Failure

The criteria for evaluating fuel cladding failure are listed below.

- 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 cladding excess hydrogen dependent limit depicted in Figure 5-3.
- 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

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effects on pellet radial power distribution, fuel thermal conductivity, and fuel melting temperature. Incipient fuel melt is determined using Equation 12-3 from Reference 8.2.11 while applying a conservative pellet burnup value. Equation 12-3 is applicable for peak rod average burnup to 62 GWd/MTU as identified in Reference 8.2.11.

2.2.3 Core Coolability

The regulatory criteria for evaluating core coolability are defined in Reference 8.2.2 and 8.2.3. The following criteria 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 concerns 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. PCMI failures are precluded by assuring that the criterion for limiting cladding excess hydrogen content delineated in Section 2.2.2 above is met. In addition, the core coolability NuScale criteria delineated above establishes significant margin to the first two criteria from Section 2.1.3. Therefore the last two criteria from Section 2.1.3 are eliminated.

2.2.4 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 cladding fission product barrier will not be breached.
- The regulatory fuel cladding failure criteria in Section 2.2.2, 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 Reference 8.2.4 to perform a dose analysis is not required for the NuScale REA methodology.

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3.0 Overview and Evaluation Codes

This section describes the REA and the applicable codes used to model the event for the NPM.

3.1 Overview

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 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. PDIL insertion depth increases as power decreases. Therefore, higher power cases produce lower ejected CRA worth, and 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 a greater positive 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:

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

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 8.2.12) 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 8.2.12) 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 5.3.1.

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 and fuel response. 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

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core thermal response and to calculate the MCHFR, 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.

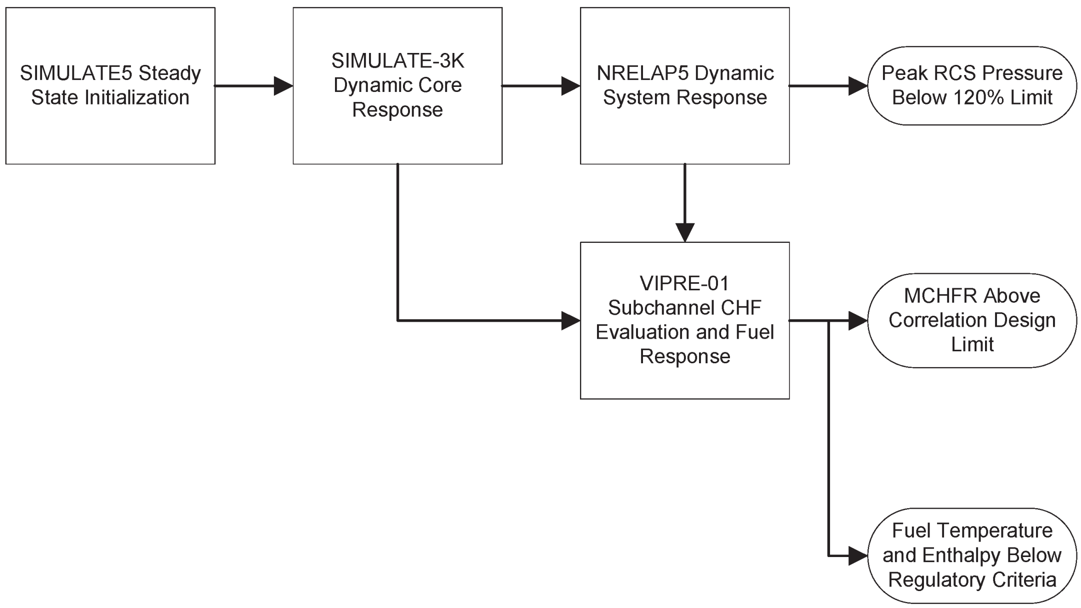

Figure 3-1 Calculation schematic for analyzing rod ejection accident

3.2.1 Core Response

Reference 8.2.6 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 8.2.15) 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

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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 8.2.6.

3.2.1.2 SIMULATE5

SIMULATE5 (Reference 8.2.16) 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 8.2.6.

3.2.1.3 SIMULATE-3K

SIMULATE-3K (References 8.2.17, 8.2.18, and 8.2.19) 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.

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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 studies 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.

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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 8.2.6) 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 8.2.20), and the NEACRP control rod ejection study 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 8.2.21). Similarities between the NuScale design and the SPERT III core, and notably the small core size, demonstrate applicability and suitability for ~~SIMUALTE~~SIMULATE-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 study 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 8.2.23 and 8.2.24) demonstrates suitability of the code for reactivity transient applications, and specifically REA analysis applications.

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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 studies. The SIMULATE-3K results for each of these benchmark studies establish the ability of the code to accurately model an REA transient event and predict key reactivity and power-related parameters. See Appendix A for further details on the NRC acceptance of the validation of SIMULATE-3K.

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 References 8.2.8 and 8.2.9.

3.2.3 Detailed Thermal-Hydraulic and Fuel Response

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

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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 (Reference 8.2.26). 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 (Reference 8.2.14) with the same requirements and qualifications as in the MOD-01 SER. Unless otherwise stated, in the remainder of this report a reference to VIPRE-01 is referring to the MOD-02 version.

The fuel rod model utilized in VIPRE-01 is important to the fuel failure modes of critical heat flux, fuel temperature, and fuel enthalpy as described in Section 2.1. These parameters are addressed in the fuel rod conduction model, where a fuel design-specific calibration to COPERNIC is performed as described in Reference 8.2.11. This calibration calculation develops a conservative radial profile, theoretical density, and gap conductance that captures the effects of heat transfer from the fuel pellet to the clad, and ultimately to the coolant. In the application of the method, sensitivity studies on bounding fuel heat transfer inputs must be performed to determine the limiting condition. This calibration is applicable to rod ejection because extreme rod ejection example cases are utilized in the calibration. Additionally, performing time step sensitivities in application calculations demonstrates the simulation adequately addresses the unique heat generation and conduction characteristics of this event, which impacts heat flux and timing. These sensitivity studies confirm the appropriate resolution of the numerical solution.

The validation and applicability of VIPRE-01 to the NuScale design is described in Reference 8.2.10.

3.2.4 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.

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4.0 Identification of Important Phenomena for Rod Ejection Accident

Reference 8.2.25 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 8.2.7). 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 8.2.13).

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 8.2.25) provides the REA analysis parameters in Tables 3-1 and 3-3. Table 4-1 and Table 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.

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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 identified in Section 5.3.

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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), middle of cycle (MOC), and end of cycle (EOC), as well as at various reactor power values ranging from HZP to hot full power (HFP).

Heat capacity of the fuel is used to calculate the enthalpy and temperature increases in the fuel pellets during the event.

Pin peaking factors are calculated by SIMULATE-3K. The largest pin peaking during the event is used to model the limiting node. An uncertainty factor is applied that captures manufacturing tolerances and modeling uncertainties.

4.2 Electric Power Research Institute Technical Report

The EPRI technical report (Reference 8.2.13) 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.

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4.3 Standard Review Plan Section 15.4.8 Initial Conditions

In addition to the PIRT and the EPRI topical report, the SRP Section 15.4.8 (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 5.0.

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 occur well after the power peak, and consequently well after MCHFR.

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 Sections 3.1.1 and 3.1.2. 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,

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though not explicitly important per Reference 8.2.25, will still be modeled in a conservative manner as noted in the above item D.

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5.0 Rod Ejection Accident Analysis Methodology

As discussed in Section 3.2, 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. 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 calculation results provided in this report are from evaluations performed using an equilibrium cycle.

5.1.2 Cycle Burnup

The REA is analyzed at BOC, MOC, and EOC burnups to bound core reactivity conditions. ~~It~~For prompt critical CRA ejections, it is expected that the limiting MCHFR case will occur at EOC because the delayed neutron fraction is minimized at this time, and a smaller delayed neutron fraction ~~increases the reactivity insertion for CRA ejection and~~typically maximizes the ~~dynamic response~~peak power of the event for a given initial power level. For sub-prompt critical jumps, the limiting MCHFR may not be associated with the maximum peak power.

When analyzing MOC, the time in cycle of maximum peaking will be considered if it does not occur at BOC. This time in cycle may not necessarily correspond to a burnup halfway between BOC and EOC. In the event that MOC is more limiting than BOC or EOC, additional analyses at other MOC points should be performed to ensure the limiting case is identified.

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.

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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 control 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 AC power has no effect on the rod ejection accident MCHFR results, as shown in Table 6-1.

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. The core response 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).

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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 8.2.6) 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)

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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. The WRSO is identified for each ejected CRA. If a non-ejected CRA is the WRSO, then it is left at the PDIL position after SCRAM.
- 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 applied.

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 a postulated ejected CRA to damage a nearby CRDM.

The power pulse, minimum critical heat flux ratio, peak enthalpy, and peak temperature occur prior to SCRAM insertion for limiting cases. The power pulse width is on the order of 10 milliseconds and analytical limits for the control rod insertion initial movement and drop times are approximately 2 seconds each. Thus, for limiting cases the worst consequences of this event do not depend on reactor scram.

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.

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

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. If the core design does not exhibit a one-eighth core or quarter-core symmetric pattern then all regulating control rod locations must be explicitly evaluated. Only the CRAs in the regulating bank are eligible for ejection and considered in the REA methodology.

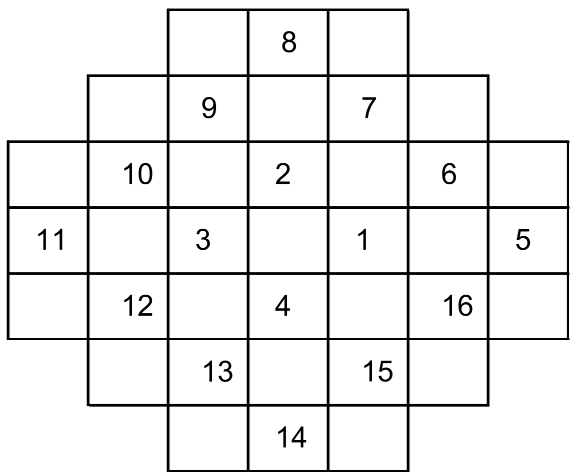

Figure 5-1 Control rod assembly layout for the NuScale Power Module

5.2.2.3.3 Reactor Trips

The example high power rate reactor trip signal used in this report is produced when the core power increases more than 7.5 percent from the initial power level within 30 seconds. The example high power reactor trip signal is produced when the core power exceeds 115 percent of rated power if the initial condition is above 15 percent power; the example low power 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.

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For the low CRA worth calculations to determine peak pressure, BOC reactivity feedback parameters are 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 8.2.6 and 8.2.10.

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
$F_{\Delta H}$ 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 8.2.9); for the system analysis using NRELAP5, REA utilitizes this methodology. However, in order to assess the NuScale-specific criteria outlined in Section 2.2, 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.

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

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 is 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.

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- 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

Initial 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 for an example core power uncertainty of 2%.

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.

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5.3.2.4 Core Inlet Temperature

Core inlet temperature is assumed to be constant. High initial temperature is conservative for both MCHFR and overpressure (see Sections 5.3.1.1 and 5.3.1.2).

5.3.2.5 Core Flow

Low core flow is conservative for both MCHFR and overpressure (see Sections 5.3.1.1 and 5.3.1.2).

5.3.2.6 System Pressure and Pressurizer Level

System pressure and pressurizer level are addressed for MCHFR and system pressurization (see Sections 5.3.1.1 and 5.3.1.2).

5.3.2.7 Generic Assessment

If the peak power is {{ }}[2(a),(c)], a generic assessment has demonstrated that the pressure acceptance criteria is generically satisfied for the module protection system analytical limits of peak power, power rate change, and peak pressure. Significant changes to these analytical limits that would change the event trajectory require a corresponding generic analysis. A generic calculation is appropriate due to the fact that the deposited energy is too small to pressurize the reactor coolant for the prompt critical peak powers considered as compared to other event trajectories. Rather, the worst pressurization possible is from a sub-prompt critical jump in power to just under the high power trip analytical limit (not a prompt critical) and the reactor eventually trips on high pressure. An example comparison of this is shown in Figure 5-2. The blue line (rea-37) depicts an example bounding prompt-critical case with a lower pressure than the red line (rea-44) of a sub-prompt-critical case.

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}}2(a),(c)

Figure 5-2 Lower plenum pressure response for prompt and sub-prompt critical event trajectories

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 8.2.9); 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 Detailed Thermal-Hydraulic and Fuel 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 8.2.10 were used to increase the convergence and reliability of the final results. These changes are described below.

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  }}2(a),(c)

- The radial nodalization of the subchannel basemodel is a {{

  }}2(a),(c) The phenomenological characteristics of the rod ejection event is unique compared to other events. For a rod that does not experience critical heat flux, the thermal-hydraulics change negligibly while the nuclear physics change dramatically. Sensitivity results presented in Figure 6-7 and Figure 6-8 for two lumped models of different sizes and a fully detailed model are compared for a variety of operating conditions. The radial nodalization of the basemodel is confirmed to accurately maintain the hot channel flow field and results in a conservative MCHFR with the largest deviations in MCHFR of 0.1 CHF points or less, an insignificant difference. Since cross-flow impacts are minimal on the calculated MCHFR, a {{

  }}2(a),(c)

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}}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 $F_{\Delta H}$ pin peaking nuclear reliability factor is applied to the highest peaked $F_{\Delta H}$ rod. The peak neutron power will occur after the rod is fully ejected and therefore will represent a skewed power distribution. With the statistical subchannel methodology defined in Reference 8.2.10, all radial peaking uncertainties are treated within the CHF analysis limit. Therefore, no additional modifications are made to the best-estimate radial power distribution as calculated by SIMULATE-3K.

The conservative nature of this modeling is described in Section 5.4.1.1. Additionally, as described in Section 6.4.2 of Reference 8.2.10, 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 8.2.10. 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 Heat Transfer

Bounding fuel heat transfer inputs are used. Sensitivity studies show that high values are more conservative for REA CHF calculations. Section 6.3.7 discusses the effect of a wide range of heat transfer values on MCHFR.

5.4.3 Fuel Response Calculation Procedure

VIPRE-01 is used to calculate the peak radial average fuel enthalpy and maximum rise in order to evaluate acceptance criteria established in Reference 8.2.3. For cladding excess hydrogen the NuScale fuel design uses cladding which is an unlined recrystallization annealed (RXA) fuel cladding. Empirically-based PCMI cladding failure threshold curves

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for RXA at or above 500°F and below 500°F (from Reference 8.2.3) are applicable to the NuScale fuel design and are shown in Figure 5-3. The most conservative application of these criteria are applied; a limit of 33 cal/g is established so the initial cladding temperature and exposure is not tracked and the excess cladding hydrogen content is not calculated.

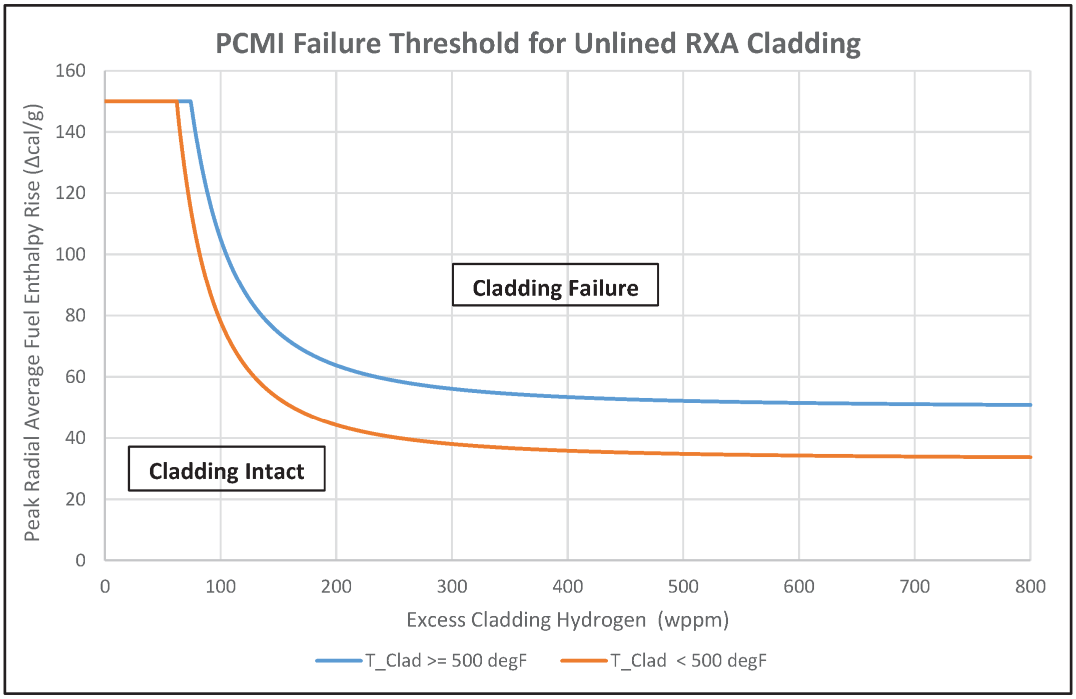

Figure 5-3 PCMI failure threshold curves for unlined RXA fuel cladding temperatures equal to or above 500 °F, and below 500 °F

5.4.4 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.

The following are sensitivity cases used to demonstrate applicability for each rod ejection subchannel calculation as described in this report.

- fuel heat transfer inputs (e.g., fuel conductivity and gap conductance)
- axial nodalization and Courant number
- time-step size
- two-phase flow correlations and Courant number
- convergence parameters

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- convergence option deviations
- radial nodalization (if default not used)

5.5 Radiological Assessment

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

6.0 Sample Rod Ejection Sensitivity Results for the NuScale Design

Examples of key sensitivity results are presented to provide context and augment the theoretical assessments made in the previous sections.

Figure 6-1 shows an example of the power response at 55 percent and EOC, which is the highest power case of an example core design and operational limits. 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 due to the uncertainty treatment until the CRAs start moving.

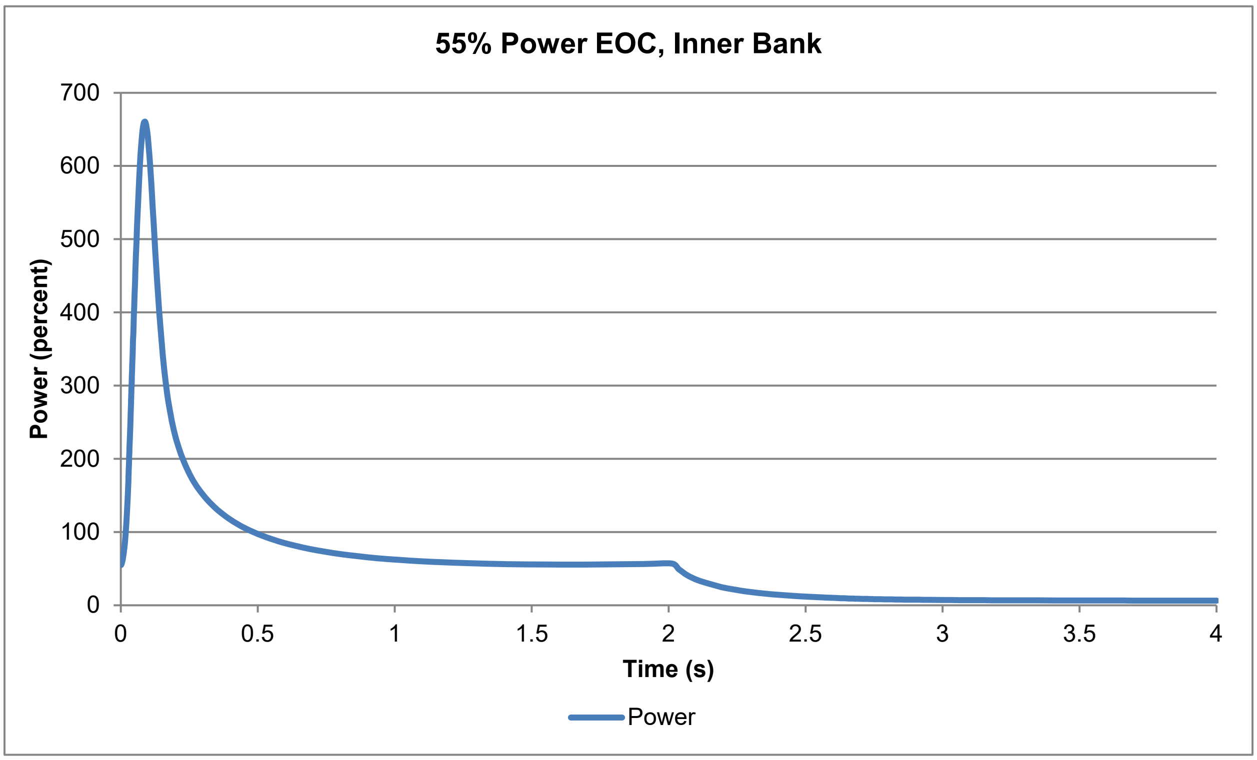

Figure 6-1 Power response at 55 percent power, end of cycle

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 module protection system limits are not reached and 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.~~

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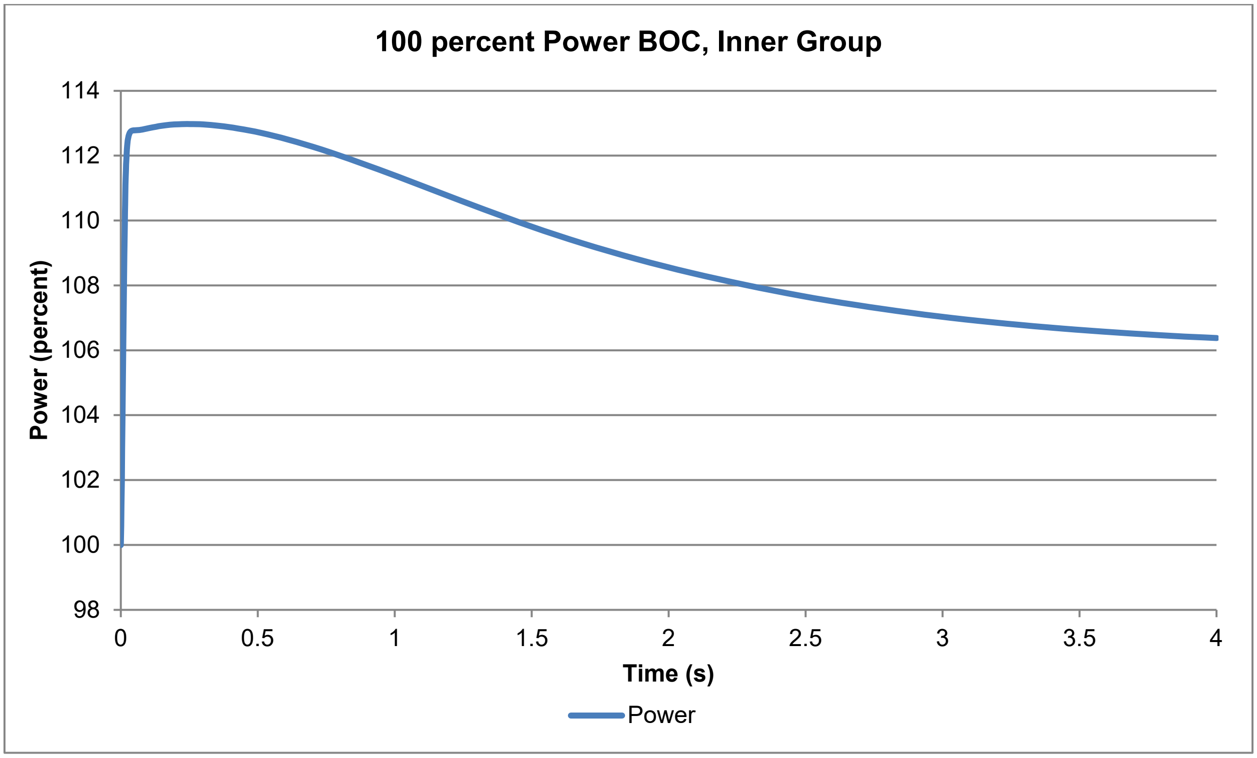

Figure 6-2 Power response at 100 percent power, beginning of cycle

6.1 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.

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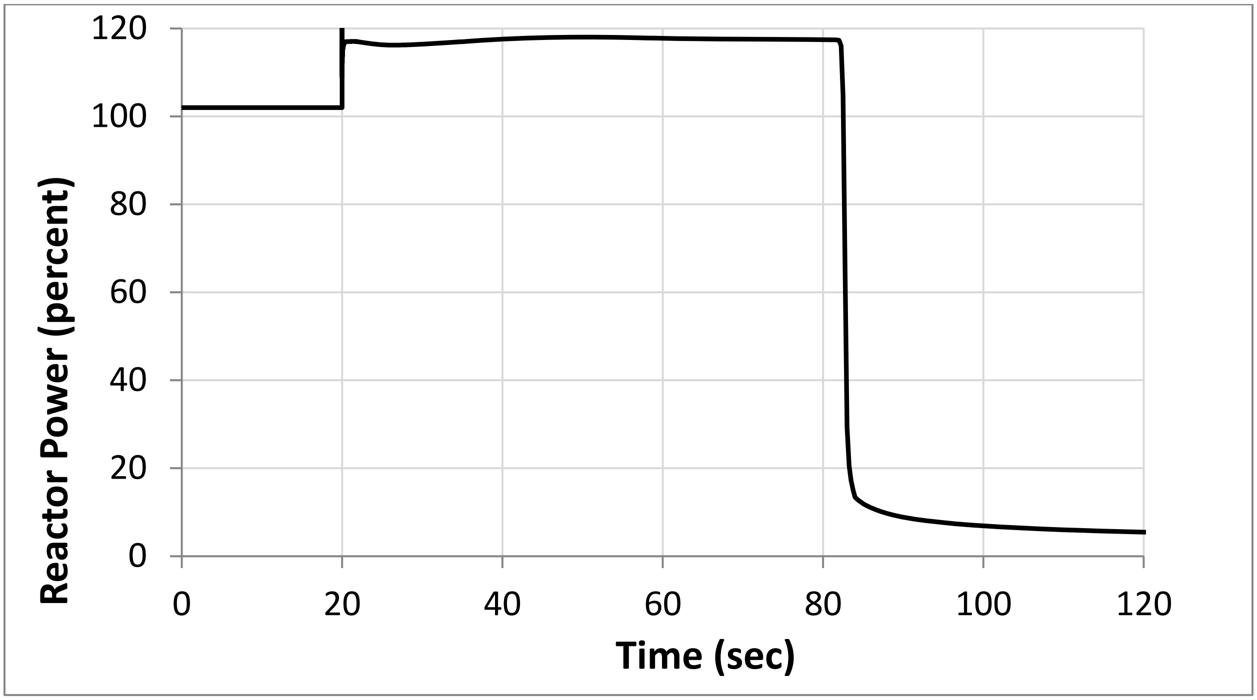

Figure 6-3 Power response for peak reactor coolant system pressure evaluation

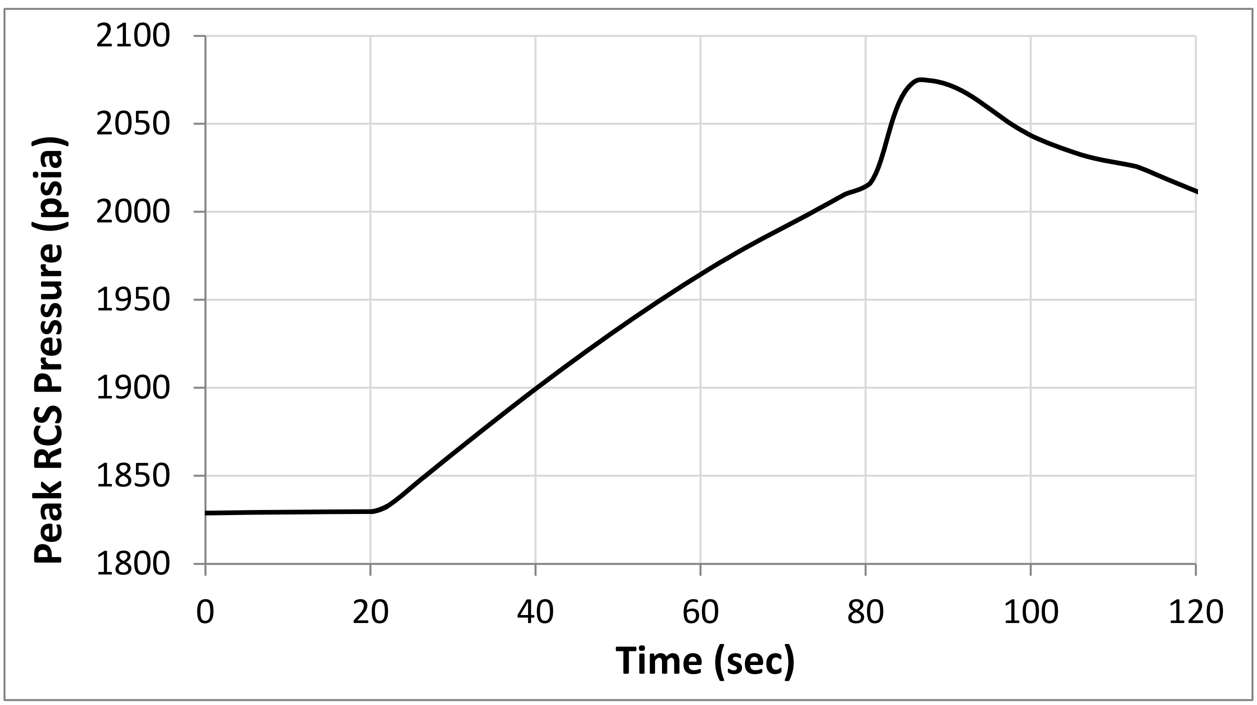

Figure 6-4 Pressure response for peak reactor coolant system pressure evaluation

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6.2 NRELAP5 Minimum Critical Heat Flux Ratio Impacts

Table 6-1 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 6-1 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]

6.3 VIPRE-01 Sensitivities

6.3.1 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

6.3.2 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

6.3.3 Code Radial Nodalization

Figure 6-7 presents a comparison of two lumped models of different sizes and a fully detailed model that are compared for MCHFR as a function of axial elevation. Figure 6-8 presents mass flux for the same models. The radial nodalization of the basemodel is confirmed to accurately maintain the hot channel flow field and results in a conservative MCHFR with the largest deviations in MCHFR of {{

}}2(a),(c)

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{{

}}2(a),(c)

Figure 6-7 Radial geometry nodalization hot channel CHFR ~~verses~~versus axial elevation

{{

}}2(a),(c)

Figure 6-8 Radial geometry nodalization mass flux versus axial elevation

6.3.4 Two-Phase Flow Correlation Options

Figure 6-9 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 6-9 Effect of VIPRE-01 two-phase flow model options on critical heat flux

6.3.5 Numerical Solution Damping Factors

Figure 6-10 shows a comparison of damping factors used in solving the VIPRE-01 numerical solution. {{

}}2(a),(c)

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{{

}}2(a),(c)

Figure 6-10 Effect of VIPRE-01 damping factors on critical heat flux

6.3.6 Radial Power Distribution

Figure 6-11 provides an example artificial radial power distribution, while Figure 6-12 provides the hot assembly radial power distribution from the limiting statepoint at time of peak power. Figure 6-13 and Figure 6-14, cases 'Actual-1' and 'Actual-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 6-12, applying the $F_{\Delta H}$ uncertainty to the limiting rod. Figure 6-15 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 validates the statement made in Section 5.4.1.1 that accurately maintaining the hot channel flow field is the only significant requirement for the conservative calculation of MCHFR. Simplification of the radial nodalization a few rows away from the hot rod results in insignificant deviations in MCHFR.

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{{

}}2(a),(c)

Figure 6-11 Radial power distribution for VIPRE-01 51 channel model, 70 percent power, end of cycle (Artificial)

{{

}}2(a),(c)

Figure 6-12 Radial power profile values for hot assembly at peak power

{{

}}2(a),(c)

Figure 6-13 Eighth-assembly radial power profile for VIPRE-01, peak rod on diagonal (Actual-1)

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{{

}}2(a),(c)

Figure 6-14 Eighth-assembly radial power profile for VIPRE-01, peak rod near center (Actual-2)

{{

}}2(a),(c)

Figure 6-15 Radial power profile effects on critical heat flux response

An example of the single channel radial nodalization for a different case with a peak power of roughly 300% rated power is provided. For this sensitivity study, three different nodalization schemes are examined of {{

}}2(a),(c)

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{{

}}2(a),(c) The results from this sensitivity study and plotted in Figure 6-16.

{{

}}2(a),(c)

Figure 6-16 Radial nodalization sensitivity MCHFR comparison

As expected from the reasoning provided in Section 5.4, the timing and magnitude of the decrease in MCHFR as the power increases and then is turned around by the Doppler feedback is close for the three cases, with the {{

}}2(a),(c) This sensitivity provides an example justification that the single channel radial nodalization is appropriate for this particular case. As noted above, each implementation of the single channel model for a limiting case requires a similar sensitivity to confirm applicability.

6.3.7 Fuel Rod Heat Transfer

Sensitivity calculations were performed to analyze the impact of applying various uncertainties or input options. Figure 6-17 below shows the comparison of high and low

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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 6-17 Effect of heat transfer inputs on critical heat flux

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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 1.236 and SRP Sections 4.2 and 15.4.8. NuScale intends to use this methodology for REA analysis in support of the NuScale standard design approval application and for future applications that are appropriately justified and approved. 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, including transient fuel enthalpy and temperature increases, is analyzed using VIPRE-01. The software is validated for use to evaluate 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 calculations and sensitivity analyses.

Section 6.0 of this report provides sample REA sensitivity calculations. These data provide confirmation that the method for satisfying the regulatory acceptance criteria outlined in Section 2.1 are appropriate. The regulatory acceptance criteria 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 pressure.
- fuel cladding failure. Transient enthalpy rise is well below the criteria for HZP, intermediate, and HFP conditions considering fuel rod differential pressure at HZP and cladding excess hydrogen with a wide margin. The subchannel model also predicts that the peak fuel centerline temperature is well below the incipient melting point. For the limiting critical heat flux (CHF) cases 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.

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

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

Parameter	Criteria	Sample Evaluation Results – Limiting Case
Maximum RCS pressure	≤ 120% design	2076 psia (94.4% design)
HZP fuel cladding failure (average enthalpy)	< 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 > CHF analysis limit	1.47
Cladding excess hydrogen-based PCMI failure	< 33 Δcal/g	11.9 Δcal/g
Incipient fuel melting cladding failure	< incipient fuel melt limit	2162 °F
Peak radial average fuel enthalpy for core coolability	< 230 cal/g	84.0 cal/g
Fuel melting for core cooling	< incipient fuel melt limit	2162°F
Fission product inventory	2.3.4	N/A

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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. Nuclear Regulatory Commission, "Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drop Accidents," Regulatory Guide 1.236, June 2020.

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 NuScale Power, LLC, "NuScale Power Critical Heat Flux Correlations," TR-0116-21012-P-A, Revision 1.

8.2.6 NuScale Power, LLC, "Nuclear Analysis Codes and Methods Qualification," TR-0616-48793-P-A, Revision 1.

8.2.7 NuScale Power, LLC, "Applicability of AREVA Fuel Methodology for the NuScale Design," TR-0116-20825-P-A, Revision 1.

8.2.8 NuScale Power, LLC, "Loss-of-Coolant Accident Evaluation Model," TR-0516-49422-P-A, Revision 2.

8.2.9 NuScale Power, LLC, "Non-Loss-of-Coolant Accident Analysis Methodology," TR-0516-49416-P-A Revision 3.

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8.2.10 NuScale Power, LLC, “Statistical Subchannel Analysis Methodology,” TR-108553-P, Revision 0.

8.2.11 BAW-10231P-A, “COPERNIC Fuel Rod Design Computer Code,” January 2004.

8.2.12 Hetrick, D. L., “Dynamics of Nuclear Reactors,” ANS, Illinois, pp. 64 and 166, 1993.

8.2.13 EPRI Technical Report 1003385, “Three-Dimensional Rod Ejection Accident Peak Fuel Enthalpy Analysis Methodology,” November 2002.

8.2.14 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.

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

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

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

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

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

8.2.20 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.

8.2.21 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.

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

8.2.23 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.

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

8.2.25 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.

8.2.26 “Safety Evaluation Report on EPRI NP-2511-CCM VIPRE-01,” May 1986.

8.2.27 NuScale Power, LLC, “Response to RAI 9306, Question 15.04.08-1,” June 4, 2018, ADAMS Accession Nos. ML18155A627 (package) and ML18155A628 (public version).

8.2.28 U.S. Nuclear Regulatory Commission, Final Safety Evaluation for NuScale Power, LLC Topical Report TR-0716-50350, Revision 1, "Rod Ejection Accident Methodology," dated June 3, 2020 (ML20157A223).

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Appendix A. NRC Acceptance of NuScale Validation of SIMULATE-3K

The NRC reviewed NuScale's benchmark of SIMULATE-3K against a selection of SPERT-III cold startup tests for each statepoint, generally corresponding to the highest static worth for the statepoint (Reference 8.2.21). 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 (Reference 8.2.27). The NRC determined that the NuScale results demonstrated generally good agreement between the results predicted by SIMULATE-3K and the SPERT-III experimental results.

Additionally, the NRC reviewed NuScale's verification analysis of the NEACRP REA benchmark performed by Studsvik Scandpower with SIMULATE-3K (Reference 8.2.27). This analysis was performed under NuScale's approved 10 CFR Part 50, Appendix B, quality assurance program. The results of this analysis are presented below.

Table A-1 provides a comparison of the SIMULATE-3K results obtained by NuScale against the NEACRP benchmark reference solutions.

Table A-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)

Parameter	Case	NEACRP	S3K	Δ	%Δ
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)

After review the NRC determined that the results demonstrated good agreement between NuScale's SIMULATE-3K results and the NEACRP benchmark reference solutions. Based on NuScale's analysis results, the NRC found that NuScale demonstrated that SIMULATE-3K can successfully model the NEACRP benchmarks for reactivity-initiated accidents.

The NRC concluded that the NuScale validation of SIMULATE-3K against the SPERT-III experiments and the NEACRP benchmark suite, as discussed above, were acceptable and demonstrated that SIMULATE-3K can be used in its methodology to accurately model a reactivity-initiated accident (Reference 8.2.28).

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Response to Request for Additional Information
Docket: 99902078

RAI No.: 9936
Date of RAI Issue: 07/13/2022

NRC Question No.: NTR-02

Regulatory Basis:

Regulatory Guide 1.236, section 3.3, "Molten Fuel Cladding Failure Threshold" states that "Fuel cladding failure is presumed if predicted fuel temperature anywhere in the pellet exceeds incipient fuel melting conditions." Section 6, "Allowable Limits on Damaged Core Coolability," states that in medium- to high-burnup rods, fuel melting outside the centerline region should be precluded.

Issue:

Section 2.2.2 "Fuel Cladding Failure" of TR-0716-50350, Rev 2, states that burnup-enhanced incipient fuel melt temperature is determined using equation 12-3 from BAW-10231P-A, "COPERNIC Fuel Rod Design Computer Code," and using a conservative burnup value. The NRC staff has previously approved the methodology in BAW-10231P-A for uranium dioxide fuels with M5 cladding to a peak rod average burnup of 62 GWd/MTU; however, NuScale has not stated the range over which this methodology will be applied.

Request:

The staff requests that NuScale update the LTR to: a) state the range of burnup over which the REA methodology may be applied, and b) if that range extends beyond 62 GWd/MTU, provide justification for applicability of equation 12-3 up to the maximum burnup allowed.

NuScale Response:

The use of Equation 12-3 from BAW-10231P-A, "COPERNIC Fuel Rod Design Computer Code," in TR-0716-50350-P, Rev. 2, is intended to be consistent with the range of peak rod average burnup of up to 62 GWd/MTU identified in BAW-10231P-A. TR-0716-50350-P, Rev. 2, is revised to explicitly identify the range of applicability.

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 the RAI No. 9936 Question No. NTR-01 response.

October 31, 2023 Docket No. 052-050

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 NRCs Request for Docketing of Resolved Audit Responses

REFERENCES: 1. Letter from the NRC to NuScale Power entitled, “Audit Status – The Staff Review of the NuScale Power, LLC Standard Design Approval Application – NuScale US460,” Dated September 29, 2023 (ML23269A062)

The purpose of this letter is to provide NuScale's response to the NRC’s request to docket certain audit responses, noted in Reference 1 above. The responses to the individual audit items are provided in the attachments.

This letter contains NuScale's responses to the following audit questions from the NRC.

Chapter 3

1.	3.4.2-2	Information related to FSAR Section 3.4.2, “Flood Protection from External Sources” nonprop
2.	3.4.2-2	Information related to FSAR Section 3.4.2, “Flood Protection from External Sources” prop
3.	3.7.2-1	Information related to FSAR Section 3.7.2, “Seismic System Analysis”
4.	3.7.2-6	Information related to FSAR Section 3.7.2, “Seismic System Analysis” nonprop
5.	3.7.2-6	Information related to FSAR Section 3.7.2, “Seismic System Analysis” prop
6.	3.8.2-1	Information related to FSAR Section 3.8.2, “Steel Containment”
7.	3.8.2-2	Information related to FSAR Section 3.8.2, “Steel Containment”
8.	3.9.1-2	Information related to FSAR Section 3.9.1, “Special Topics for Mechanical Components”
9.	3.9.2-2	Information related to FSAR Section 3.9.2, “Dynamic Testing and Analysis of Systems, Components, and Equipment” nonprop
10.	3.9.2-2	Information related to FSAR Section 3.9.2, “Dynamic Testing and Analysis of Systems, Components, and Equipment” prop
11.	3.9.2-3	Information related to FSAR Section 3.9.2, “Dynamic Testing and Analysis of Systems, Components, and Equipment” nonprop
12.	3.9.2-3	Information related to FSAR Section 3.9.2, “Dynamic Testing and Analysis of Systems, Components, and Equipment” prop
13.	3.9.2-8	Information related to FSAR Section 3.9.2, “Dynamic Testing and Analysis of Systems, Components, and Equipment” nonprop

14.	3.9.2-8	Information related to FSAR Section 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment" prop
15.	3.9.2-9	Information related to FSAR Section 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment" nonprop
16.	3.9.2-9	Information related to FSAR Section 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment" prop
17.	3.9.2-10	Information related to FSAR Section 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment" nonprop
18.	3.9.2-10	Information related to FSAR Section 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment" prop
19.	3.9.2-11	Information related to FSAR Section 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment" nonprop
20.	3.9.2-11	Information related to FSAR Section 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment" prop
21.	3.9.2-12	Information related to FSAR Section 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment" nonprop
22.	3.9.2-12	Information related to FSAR Section 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment" prop
23.	3.9.4-1	Information related to FSAR Section 3.9.4, "Control Rod Drive System" nonprop
24.	3.9.4-1	Information related to FSAR Section 3.9.4, "Control Rod Drive System" prop
25.	3.9.4-4	Information related to FSAR Section 3.9.4, "Control Rod Drive System" nonprop
26.	3.9.4-4	Information related to FSAR Section 3.9.4, "Control Rod Drive System" prop
27.	3.9.4-5	Information related to FSAR Section 3.9.4, "Control Rod Drive System" nonprop
28.	3.9.4-5	Information related to FSAR Section 3.9.4, "Control Rod Drive System" prop
29.	3.9.6-1	Information related to FSAR Section 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints" nonprop
30.	3.9.6-1	Information related to FSAR Section 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints" prop
31.	3.9.6-5	Information related to FSAR Section 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints"
32.	3.9.6-9	Information related to FSAR Section 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints"
33	3.9.6-12	Information related to FSAR Section 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints" nonprop
34.	3.9.6-12	Information related to FSAR Section 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints" prop

35.	3.9.6-13	Information related to FSAR Section 3.9.6, “Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints”
36	3.9.6-14	Information related to FSAR Section 3.9.6, “Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints”
37.	3.9.6-16	Information related to FSAR Section 3.9.6, “Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints”
38.	3.12-3	Information related to FSAR Section 3.12, “ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and Associated Supports”
39.	3.12-4	Information related to FSAR Section 3.12, “ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and Associated Supports” nonprop
40.	3.12-4	Information related to FSAR Section 3.12, “ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and Associated Supports” prop
41	3.12-6	Information related to FSAR Section 3.12, “ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and Associated Supports”
42	3.12-9	Information related to FSAR Section 3.12, “ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and Associated Supports” nonprop
43.	3.12-9	Information related to FSAR Section 3.12, “ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and Associated Supports” prop
Chapter 4		
44.	4.2-2	Information related to FSAR Section 4.2, “Fuel System Design”
45.	4.3-8	Information related to FSAR Section 4.3, “Nuclear Design”
46.	4.6-1	Information related to FSAR Section 4.6, “Functional Design of Control Rod Drive System”
Chapter 5		
47.	5.2.1.1-4	Information related to FSAR Section 5.2.1.1, “Compliance with 10 CFR 50.55a”
48.	5.2.1.1-5	Information related to FSAR Section 5.2.1.1, “Compliance with 10 CFR 50.55a”
49.	5.2.1.2-3	Information related to FSAR Section 5.2.1.2, “Compliance with Applicable Code Cases”
Chapter 7		
50.	7.1-1	Information related to FSAR Section 7.1, “Fundamental Design Principles”
Chapter 9		
51.	9.2.9-2	Information related to FSAR Section 9.2.9, “Utility Water Systems”
52.	9.3.4-1	Information related to FSAR Section 9.3.4, “Chemical and Volume Control System”
Chapter 10		
53.	10.2-1	Information related to FSAR Section 10.2, “Turbine Generator”
54.	10.3-2	Information related to FSAR Section 10.3, “Main Steam System”

55.	10.3-3	Information related to FSAR Section 10.3, “Main Steam System” nonprop
56.	10.3-3	Information related to FSAR Section 10.3, “Main Steam System” prop
57.	10.3.6-1	Information related to FSAR Section 10.3.6, “Steam and Feedwater System Materials”
58.	10.4-1	Information related to FSAR Section 10.4, “Other Features of Steam and Power Conversion System” nonprop
59.	10.4-1	Information related to FSAR Section 10.4, “Other Features of Steam and Power Conversion System” prop
60.	10.4.5-6	Information related to FSAR Section 10.4.5, “Condensate Polisher Skid and Resin Regeneration System”
Chapter 11		
61.	11.1-2	Information related to FSAR Section 11.1, “Source Terms”
62.	11.1-3	Information related to FSAR Section 11.1, “Source Terms”
63.	11.3-2	Information related to FSAR Section 11.3, “Gaseous Waste Management System”
64.	11.4-1 (follow-up)	Information related to FSAR Section 11.4, “Solid Waste Management System” - FOLLOW-UP
65.	11.4-1	Information related to FSAR Section 11.4, “Solid Waste Management System”
66.	11.4-2 (follow-up)	Information related to FSAR Section 11.4, “Solid Waste Management System” – FOLLOW-UP
67.	11.4-2	Information related to FSAR Section 11.4, “Solid Waste Management System”
68.	11.4-3	Information related to FSAR Section 11.4, “Solid Waste Management System”
69.	11.4-4	Information related to FSAR Section 11.4, “Solid Waste Management System”
70.	11.4-5	Information related to FSAR Section 11.4, “Solid Waste Management System”
71.	11.5-1	Information related to FSAR Section 11.5, “Process and Effluent Radiation Monitoring Instrumentation and Sampling System”
72.	11.5-2	Information related to FSAR Section 11.5, “Process and Effluent Radiation Monitoring Instrumentation and Sampling System”
73.	11.5-3	Information related to FSAR Section 11.5, “Process and Effluent Radiation Monitoring Instrumentation and Sampling System”
74.	11.5-4 (follow-up)	Information related to FSAR Section 11.5, “Process and Effluent Radiation Monitoring Instrumentation and Sampling System” – FOLLOW-UP
75.	11.5-4	Information related to FSAR Section 11.5, “Process and Effluent Radiation Monitoring Instrumentation and Sampling System”
Chapter 12		
76.	12.2.1.8-1	Information related to FSAR Section 12.2.1.8, “Reactor Pool Water,”
77.	12.2-1	Information related to FSAR Section 12.2, “Radiation Sources” nonprop
78.	12.2-1	Information related to FSAR Section 12.2, “Radiation Sources” prop
79.	12.2-3	Information related to FSAR Section 12.2, “Radiation Sources”

80.	12.2-5	Information related to FSAR Section 12.2, “Radiation Sources”
81.	12.3.1.1-2	Information related to FSAR Section 12.3.1.1, “Equipment Design”
82.	12.3.2.2-1	Information related to FSAR Section 12.3.2.2, “Design Considerations”
83.	12.4.1.9-1 (follow-up)	Information related to FSAR Section 12.4.1.9, “Construction Activities” - FOLLOW-UP
84.	12.4.1.9-1	Information related to FSAR Section 12.4.1.9, “Construction Activities”
Chapter 15		
85.	15.5.1-1	Information related to FSAR Section 15.5.1, “Chemical and Volume Control System Malfunction”
86.	15.6.3-1	Information related to FSAR Section 15.6.3, “Steam Generator Tube Failure (Thermal Hydraulic)” nonprop
87.	15.6.3-1	Information related to FSAR Section 15.6.3, “Steam Generator Tube Failure (Thermal Hydraulic)” prop
Chapter 16		
88.	16-1	Information related to FSAR Chapter 16, “Technical Specifications”
89.	16.1.1-1	Information related to FSAR Section 16.1.1, “Introduction to Technical Specifications”
90.	16.5.5.09-1	Information related to Generic Technical Specification 5.5.9, “Containment Leakage Rate Testing Program”
91.	16.5.5.09-2	Information related to Generic Technical Specification 5.5.9, “Containment Leakage Rate Testing Program”
92.	16.5.5.09-3	Information related to Generic Technical Specification 5.5.9, “Containment Leakage Rate Testing Program”
Chapter 17		
93.	17.4-1	Information related to FSAR Section 17.4, “Reliability Assurance Program”
Chapter 18		
94.	18-1	Information related to FSAR Chapter 18, “Human Factors Engineering”
95.	18.2-1	Information related to FSAR Section 18.2, “Operating Experience Review”
96.	18.3-2	Information related to FSAR Section 18.3, “Functional Requirements Analysis and Function Allocation”
97.	18.3-3	Information related to FSAR Section 18.3, “Functional Requirements Analysis and Function Allocation” nonprop
98.	18.3-3	Information related to FSAR Section 18.3, “Functional Requirements Analysis and Function Allocation” prop
99.	18.4-2	Information related to FSAR Section 18.4, “Task Analysis” nonprop
100.	18.4-2	Information related to FSAR Section 18.4, “Task Analysis” prop
101.	18.5-1	Information related to FSAR Section 18.5, “Staffing and Qualifications”
102.	18.5-1S (follow-up)	Information related to FSAR Section 18.5, “Staffing and Qualifications” – FOLLOW-UP
103.	18.6-2	Information related to FSAR Section 18.6, “Treatment of Important Human Actions” nonprop

104.	18.6-2	Information related to FSAR Section 18.6, “Treatment of Important Human Actions” prop
105.	18.7-4	Information related to FSAR Section 18.7, “Human-System Interface Design” nonprop
106.	18.7-4	Information related to FSAR Section 18.7, “Human-System Interface Design” prop
107.	18.7-5	Information related to FSAR Section 18.7, “Human-System Interface Design”
108.	18.7-6	Information related to FSAR Section 18.7, “Human-System Interface Design”
109.	18.10-1	Information related to FSAR Section 18.10, “Human Factors Verification and Validation”
110.	18.10-2	Information related to FSAR Section 18.10, “Human Factors Verification and Validation”
Chapter 19		
111.	19.1-20	Information related to FSAR Section 19.1, “Probabilistic Risk Assessment”
112.	19.1-25	Information related to FSAR Section 19.1, “Probabilistic Risk Assessment” nonprop
113.	19.1-25	Information related to FSAR Section 19.1, “Probabilistic Risk Assessment” prop
114.	19.1-27	Information related to FSAR Section 19.1, “Probabilistic Risk Assessment” nonprop
115.	19.1-27	Information related to FSAR Section 19.1, “Probabilistic Risk Assessment” prop
116.	19.1-40	Information related to FSAR Section 19.1, “Probabilistic Risk Assessment”
117.	19.3-1	Information related to FSAR Section 19.3, “Regulatory Treatment of Nonsafety Systems” nonprop
118.	19.3-1	Information related to FSAR Section 19.3, “Regulatory Treatment of Nonsafety Systems” prop
119.	19.5-12	Information related to FSAR Section 19.5, “Adequacy of Design Features and Functional Capabilities Identified and Described for Withstanding Aircraft Impacts”
Rod Ejection Accident Methodology LTR		
120.	RE-02	Information related to Rod Ejection Accident Methodology LTR, TR-0716-50350, Revision 2 (ML21351A399). This information is discussed in a separate limited scope audit that began on April 19, 2023 (ML23107A227) nonprop
121.	RE-02	Information related to Rod Ejection Accident Methodology LTR, TR-0716-50350, Revision 2 (ML21351A399). This information is discussed in a separate limited scope audit that began on April 19, 2023 (ML23107A227) prop

The audit items specified above have been attached to the NRC EIE submittal. The responses that are labeled as proprietary include both proprietary and nonproprietary versions. NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit supports this request. Some responses have also been determined to contain Export Controlled

Information. This information must be protected from disclosure per the requirements of 10 CFR § 810.

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

If you have any questions, please contact Thomas Griffith at 541-452-7813 or tgriffith@nuscalepower.com.

Sincerely,

Mark W. Shaver

Mark W. Shaver
Director, Regulatory Affairs
NuScale Power, LLC

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Enclosure 1: Affidavit of Carrie Fosaaen, AF-152561

Response to SDAA Audit Question

Question Number: A-RE-02

Receipt Date: 07/24/2023

Question:

Not all previous responses to RAIs were added to the topical report in response to audit questions. It is not clear why some of the responses are not included. These RAI responses contain information that is part of the bases of the methodology for the rod ejection analysis.

Applicable portions of RAI responses provided during review of revision 1 of the topical report should be included in the -A version of revision 2. List RAI responses that will be included in the -A version of revision 2. Provide justifications for RAI responses that are no longer applicable. If part of a previous RAI response is applicable, provide markup to the response.

Response:

In addition to the original audit question, NuScale has also reviewed the supplemental information provided by the NRC. The response below considers the supplemental information provided by the NRC as well as discussions of this audit question that took place during the in-person audit meeting on September 6, 2023.

Revision 1 of the rod ejection accident methodology topical report, TR-0716-50350-P was approved by the NRC on June 3, 2020. NuScale submitted TR-0716-50350-P-A, Revision 1, on June 16, 2020 and included the NRC safety evaluation report (SER). The submittal of TR-0716-50350-P-A, Revision 1, also included the responses to requests for additional information (RAIs) from the NRC's review that led to approval of Revision 1. Some of those RAI responses resulted in changes to the topical report that were incorporated into Revision 1, but others did not. The acceptability of those RAI responses, including the associated topical report changes or lack thereof, was addressed by the NRC approval of Revision 1. For the purpose of this response, those RAIs and their responses will be called the "Rev. 1 RAIs" throughout the remainder of this response.

NuScale submitted Revision 2 of the topical report on December 17, 2021. During the NRC review of Revision 2, the NRC issued RAI 9936 and NuScale responded on September 14,

2022. The NRC also performed an audit of Revision 2 of the topical report in 2023 and NuScale provided responses to multiple audit questions. Some of the audit questions received in 2023 were the same as or similar to the Rev. 1 RAIs. For the purpose of this response, the RAIs and audit questions associated with the NRC review of Revision 2 and their responses will be called the “Rev. 2 RAIs and audit questions” throughout the remainder of this response.

This current audit question states that the Rev. 1 RAIs should continue to be included with the next -A version of the topical report. NuScale understands the request, but believes it is not necessary to include the Rev. 1 RAIs in the next -A version of the topical report for the following reasons:

- The relevant details of the Rev. 1 RAI were incorporated into Revision 1 at the time of its approval and remain in Revision 2 (or its markups). In this case, the relevant information is contained within the topical report itself and it is not necessary to continue to include the Rev. 1 RAI in the next -A version.
- The relevant details of the Rev. 1 RAI were not incorporated into Revision 1 at the time of its approval, but are incorporated into Revision 2 (or its markups). In this case, the relevant information is contained within the topical report itself and it is not necessary to continue to include the Rev. 1 RAI in the next -A version.
- The Rev. 1 RAI concerns a topic which is no longer applicable to Revision 2 (or its markups). In this case, there is no relevant information from the Rev. 1 RAI that needs to be included in the next -A version.

Table 1 below lists each of the Rev. 1 RAIs and provides a disposition of whether it needs to be included in the next -A version, consistent with the justifications above. As indicated in Table 1, there are no Rev. 1 RAIs that need to be included in the next -A version of the topical report.

Table 1: Review of Rev. 1 RAIs

Rev. 1 RAI Question	NuScale Response	Sections Revised in Rev. 1-A	Related Rev. 2 RAI or Audit Question	Sections Revised in Rev. 2 or Draft[1] Rev. 3	Assessment of Need to Append Rev. 1 RAI to Future[1] Rev. 3-A Upon NRC Approval
9306 15.04.08-1	RAIO-0618-60285 (dated 6/4/2018) RAIO-0219-6416 (dated 2/21/2019) RAIO-1019-67479 (dated 10/10/2019)	3.2.1.4	Draft Condition and Limitation #3	Ex.Sum., 3.2, App. A	No. The Rev. 1 RAI concerned code validation. The response included code validation information that was not incorporated in Rev. 1-A, but was incorporated in Rev. 2 as Appendix A. The first supplemental response included further code validation information that was not incorporated in Rev. 1-A, but is being added to the topical report Appendix A as part of NuScale's response to NRC proposed condition and limitations discussed in an audit meeting on September 6, 2023 (markup to be provided separately in the eRR). The topic of software control under the quality assurance program is also being added to the topical report in the Executive Summary and Section 3.2 as part of that same markup. In addition, the second supplemental response included markups to incorporate relevant information into Rev. 1-A. Because the relevant information from the Rev. 1 RAI is present in the current revision of the topical

Rev. 1 RAI Question	NuScale Response	Sections Revised in Rev. 1-A	Related Rev. 2 RAI or Audit Question	Sections Revised in Rev. 2 or Draft[1] Rev. 3	Assessment of Need to Append Rev. 1 RAI to Future[1] Rev. 3-A Upon NRC Approval
					report, it is not necessary to include the Rev. 1 RAI in the next -A version of the topical report.
9306 15.04.08-2	RAIO-0618-60285 (dated 6/4/2018)	4.1.2.1, 4.1.2.2, 5.5.1			No. The relevant information from the response to the Rev. 1 RAI regarding the fuel response calculation was incorporated into Rev. 1-A. In Rev. 2, those sections were revised because the fuel response methodology was revised to use VIPRE-01. As a result of the fuel response methodology change, the information in the Rev. 1 RAI is no longer applicable. Because the relevant information from the Rev. 1 RAI is not applicable to the current revision of the topical report, it is not necessary to include the Rev. 1 RAI in the next -A version of the topical report.
9306 15.04.08-3	RAIO-0618-60285 (dated 6/4/2018)	5.3.3			No. The response to the Rev. 1 RAI provided a markup and the markups were incorporated into Rev. 1-A. The changes made in Rev. 1-A are still present in the the current revision. Because the relevant information from the Rev. 1 RAI is present in the current revision of the topical

Rev. 1 RAI Question	NuScale Response	Sections Revised in Rev. 1-A	Related Rev. 2 RAI or Audit Question	Sections Revised in Rev. 2 or Draft[1] Rev. 3	Assessment of Need to Append Rev. 1 RAI to Future[1] Rev. 3-A Upon NRC Approval
					report, it is not necessary to include the Rev. 1 RAI in the next -A version of the topical report.
9306 15.04.08-4	RAIO-0618-60285 (dated 6/4/2018)		A-RE-02	3.1, 3.2.1.2, 3.2.1.3	No. The information in the response to the Rev. 1 RAI was not incorporated into Rev. 1-A. In the response to a Rev. 2 audit question (i.e., this question), markups are provided to include the relevant information from the Rev. 1 RAI in the topical report. Because the relevant information from the Rev. 1 RAI is present in the current revision of the topical report, it is not necessary to include the Rev. 1 RAI in the next -A version of the topical report.
9306 15.04.08-5	RAIO-0618-60285 (dated 6/4/2018) RAIO-0119-64399 (dated 1/31/2019)		A-RE-5.2.1.2-02	5.2.1.2	No. The response and supplemental response to the Rev. 1 RAI provided an explanation, but the information was not incorporated into Rev. 1-A. In the response to a Rev. 2 audit question, markups were provided to include the relevant information from the Rev. 1 RAI in the topical report. Because the relevant information from the Rev. 1 RAI is present in the current revision of the topical report, it is not necessary to

Rev. 1 RAI Question	NuScale Response	Sections Revised in Rev. 1-A	Related Rev. 2 RAI or Audit Question	Sections Revised in Rev. 2 or Draft[1] Rev. 3	Assessment of Need to Append Rev. 1 RAI to Future[1] Rev. 3-A Upon NRC Approval
					include the Rev. 1 RAI in the next -A version of the topical report.
9306 15.04.08-6	RAIO-0618-60285 (dated 6/4/2018) RAIO-0119-64399 (dated 1/31/2019)		A-RE-3.2.1.4-01, A-RE-5.2.1.2-01	3.2.1.2, 3.2.1.4, 5.2.1.2	No. The response and supplemental response to the Rev. 1 RAI provided an explanation, but the information was not incorporated into Rev. 1-A. In the response to Rev. 2 audit questions, markups were provided to include the relevant information from the Rev. 1 RAI in the topical report. Because the relevant information from the Rev. 1 RAI is present in the current revision of the topical report, it is not necessary to include the Rev. 1 RAI in the next -A version of the topical report.
9306 15.04.08-7	RAIO-0618-60285 (dated 6/4/2018)	4.1.2.1, 4.1.2.2			No. The relevant information from the response to the Rev. 1 RAI regarding the fuel response calculation was incorporated into Rev. 1-A. In Rev. 2, those sections were revised because the fuel response methodology was revised to use VIPRE-01. As a result of the fuel response methodology change, the information in the Rev. 1 RAI is no longer applicable. Because the relevant information from the

Rev. 1 RAI Question	NuScale Response	Sections Revised in Rev. 1-A	Related Rev. 2 RAI or Audit Question	Sections Revised in Rev. 2 or Draft[1] Rev. 3	Assessment of Need to Append Rev. 1 RAI to Future[1] Rev. 3-A Upon NRC Approval
					Rev. 1 RAI is not applicable to the current revision of the topical report, it is not necessary to include the Rev. 1 RAI in the next -A version of the topical report.
9306 15.04.08-8	RAIO-0618-60285 (dated 6/4/2018)		A-RE-4.3-01	4.3	No. The response to the Rev. 1 RAI provided an explanation, but the information was not incorporated into Rev. 1-A. In the response to a Rev. 2 audit question, markups were provided to include the relevant information from the Rev. 1 RAI in the topical report. Because the relevant information from the Rev. 1 RAI is present in the current revision of the topical report, it is not necessary to include the Rev. 1 RAI in the next -A version of the topical report.
9306 15.04.08-9	RAIO-0618-60285 (dated 6/4/2018)	4.3			No. The response to the Rev. 1 RAI provided a markup and the markups were incorporated into Rev. 1-A. The changes made in Rev. 1-A are still present in the the current revision. Because the relevant information from the Rev. 1 RAI is present in the current revision of the topical report, it is not necessary to include the Rev. 1 RAI in the next

Rev. 1 RAI Question	NuScale Response	Sections Revised in Rev. 1-A	Related Rev. 2 RAI or Audit Question	Sections Revised in Rev. 2 or Draft[1] Rev. 3	Assessment of Need to Append Rev. 1 RAI to Future[1] Rev. 3-A Upon NRC Approval
					-A version of the topical report.
9306 15.04.08-10	RAIO-0618-60285 (dated 6/4/2018)	5.1.2	9936 NTR-01	5.1.2	No. The response to the Rev. 1 RAI provided a markup and the markups were incorporated into Rev. 1-A. The changes made in Rev. 1-A were removed in Rev. 2. In the response to a Rev. 2 RAI, markups were provided to restore the relevant information from the Rev. 1 RAI in the topical report. The Rev. 2 RAI will be included with the next -A version of the topical report per the usual process. Because the relevant information from the Rev. 1 RAI is present in the current revision of the topical report and the Rev. 2 RAI will be included in the next -A version of the topical report, it is not necessary to include the Rev. 1 RAI in the next -A version of the topical report.
9306 15.04.08-11	RAIO-0618-60285 (dated 6/4/2018)		A-RE-5.1.3-01, A-RE-5.2.1-01	5.1.3, 5.2.1	No. The response to the Rev. 1 RAI provided an explanation, but the information was not incorporated into Rev. 1-A. In the response to Rev. 2 audit questions, markups were provided to include the relevant information from the Rev. 1 RAI in the topical report.

Rev. 1 RAI Question	NuScale Response	Sections Revised in Rev. 1-A	Related Rev. 2 RAI or Audit Question	Sections Revised in Rev. 2 or Draft[1] Rev. 3	Assessment of Need to Append Rev. 1 RAI to Future[1] Rev. 3-A Upon NRC Approval
					Because the relevant information from the Rev. 1 RAI is present in the current revision of the topical report, it is not necessary to include the Rev. 1 RAI in the next -A version of the topical report.
9306 15.04.08-12	RAIO-0618-60285 (dated 6/4/2018)		A-RE-5.2.1.1-01	5.2.1.1	No. The response to the Rev. 1 RAI provided an explanation, but the information was not incorporated into Rev. 1-A. In the response to a Rev. 2 audit question, markups were provided to include the relevant information from the Rev. 1 RAI in the topical report. Because the relevant information from the Rev. 1 RAI is present in the current revision of the topical report, it is not necessary to include the Rev. 1 RAI in the next -A version of the topical report.
9306 15.04.08-13	RAIO-0618-60285 (dated 6/4/2018)		A-RE-02	5.3.1	No. The information in the response to the Rev. 1 RAI was not incorporated into Rev. 1-A. In the response to a Rev. 2 audit question (i.e., this question), markups were provided to include the relevant information from the Rev. 1 RAI in the topical report. Because the relevant information from the Rev. 1 RAI is present in

Rev. 1 RAI Question	NuScale Response	Sections Revised in Rev. 1-A	Related Rev. 2 RAI or Audit Question	Sections Revised in Rev. 2 or Draft[1] Rev. 3	Assessment of Need to Append Rev. 1 RAI to Future[1] Rev. 3-A Upon NRC Approval
					the current revision of the topical report, it is not necessary to include the Rev. 1 RAI in the next -A version of the topical report.
9306 15.04.08-14	RAIO-0618-60285 (dated 6/4/2018)	6.2	A-5.1.3-01	5.1.3	No. The response to the Rev. 1 RAI provided a markup and the markups were incorporated into Rev. 1-A. The changes made in Rev. 1-A were regarding sample results and were replaced in Rev. 2 with different sample results. The old sample results are no longer applicable. In the response to a Rev. 2 audit question, markups were provided to add the remaining relevant information from the Rev. 1 RAI in the topical report. Because the relevant information from the Rev. 1 RAI is present in the current revision of the topical report and the rest of the information was determined to be no longer applicable, it is not necessary to include the Rev. 1 RAI in the next -A version of the topical report.
9306 15.04.08-15	RAIO-0618-60285 (dated 6/4/2018) RAIO-0119-	5.2.2.3.4	A-RE-5.4.2.1-01	5.4.2.1	No. The response and supplemental response to the Rev. 1 RAI provided a markup and the

Rev. 1 RAI Question	NuScale Response	Sections Revised in Rev. 1-A	Related Rev. 2 RAI or Audit Question	Sections Revised in Rev. 2 or Draft[1] Rev. 3	Assessment of Need to Append Rev. 1 RAI to Future[1] Rev. 3-A Upon NRC Approval
	64399 (dated 1/31/2019)				markups were incorporated into Rev. 1-A. Some of the changes made in Rev. 1-A were regarding the fuel response calculation. In Rev. 2, those sections were revised because the fuel response methodology was revised to use VIPRE-01. As a result of the fuel response methodology change, that portion of the information in the Rev. 1 RAI is no longer applicable. The remainder of the changes made in Rev. 1-A are still present in the the current revision. In the response to a Rev. 2 audit question, markups were provided to add other relevant information from the Rev. 1 RAI in the topical report. Because the relevant information from the Rev. 1 RAI is present in the current revision of the topical report and the rest of the information was determined to be no longer applicable, it is not necessary to include the Rev. 1 RAI in the next -A version of the topical report.
9306 15.04.08-16	RAIO-0618-60285 (dated 6/4/2018) RAIO-0119-	3.2.4			No. The relevant information from the response and supplemental response to the Rev. 1 RAI regarding the fuel response

Rev. 1 RAI Question	NuScale Response	Sections Revised in Rev. 1-A	Related Rev. 2 RAI or Audit Question	Sections Revised in Rev. 2 or Draft[1] Rev. 3	Assessment of Need to Append Rev. 1 RAI to Future[1] Rev. 3-A Upon NRC Approval
	64399 (dated 1/31/2019)				calculation was incorporated into Rev. 1-A. In Rev. 2, those sections were revised because the fuel response methodology was revised to use VIPRE-01. As a result of the fuel response methodology change, the information in the Rev. 1 RAI is no longer applicable. Because the relevant information from the Rev. 1 RAI is not applicable to the current revision of the topical report, it is not necessary to include the Rev. 1 RAI in the next -A version of the topical report.

1: The changes to Rev. 2 have been provided to the NRC as a markup view in draft Rev. 3. Submittal of Rev. 3 will occur at a later date, upon completion of the NRC review. For the purpose of this response, this table assumes that Rev. 3 will be the version approved by the NRC and then will be resubmitted for the -A version.

Although Table 1 and the response above support NuScale's position that there are no Rev. 1 RAIs that need to be included in the next -A version of the topical report, this audit question was discussed further in an audit meeting on September 6, 2023. At that meeting, it was agreed that appending the applicable Rev. 1 RAIs to the next -A version of the topical report would be the most efficient resolution of the issue raised by this audit question. Therefore, NuScale will append the Rev. 1 RAIs identified in Table 2 to the next -A version of the topical report. As indicated by comparison of Table 2 to Table 1, questions 15.04.08-2 and 15.04.08-7 will not be appended as no part of the RAI or response remains applicable to the topical report.

Table 2: Rev. 1 RAIs to be Appended to Next -A Version of Topical Report

Rev. 1 RAI Question	NuScale Response
9306 15.04.08-1	RAIO-0618-60285 (dated 6/4/2018) RAIO-0219-6416 (dated 2/21/2019) RAIO-1019-67479 (dated 10/10/2019)
9306 15.04.08-3	RAIO-0618-60285 (dated 6/4/2018)
9306 15.04.08-4	RAIO-0618-60285 (dated 6/4/2018)
9306 15.04.08-5	RAIO-0618-60285 (dated 6/4/2018) RAIO-0119-64399 (dated 1/31/2019)
9306 15.04.08-6	RAIO-0618-60285 (dated 6/4/2018) RAIO-0119-64399 (dated 1/31/2019)
9306 15.04.08-8	RAIO-0618-60285 (dated 6/4/2018)
9306 15.04.08-9	RAIO-0618-60285 (dated 6/4/2018)
9306 15.04.08-10	RAIO-0618-60285 (dated 6/4/2018)
9306 15.04.08-11	RAIO-0618-60285 (dated 6/4/2018)
9306 15.04.08-12	RAIO-0618-60285 (dated 6/4/2018)
9306 15.04.08-13	RAIO-0618-60285 (dated 6/4/2018)
9306 15.04.08-14*	RAIO-0618-60285 (dated 6/4/2018)*
9306 15.04.08-15*	RAIO-0618-60285 (dated 6/4/2018)* RAIO-0119-64399 (dated 1/31/2019)*
9306 15.04.08-16*	RAIO-0618-60285 (dated 6/4/2018)* RAIO-0119-64399 (dated 1/31/2019)*

* Although the response contains some information related to the fuel response methodology or example results that are no longer applicable, other portions of the RAI and response remain applicable and therefore the entire RAI and response will be appended.

Markups of the affected changes, as described in the response, are provided below:

the core thermal response and to calculate the MCHFR, peak fuel temperature, and enthalpy. The software is controlled under the NuScale quality assurance program (Reference 8.1.3). Figure 3-1 depicts the computer codes used and the flow of information between codes and evaluations to address the regulatory acceptance criteria.

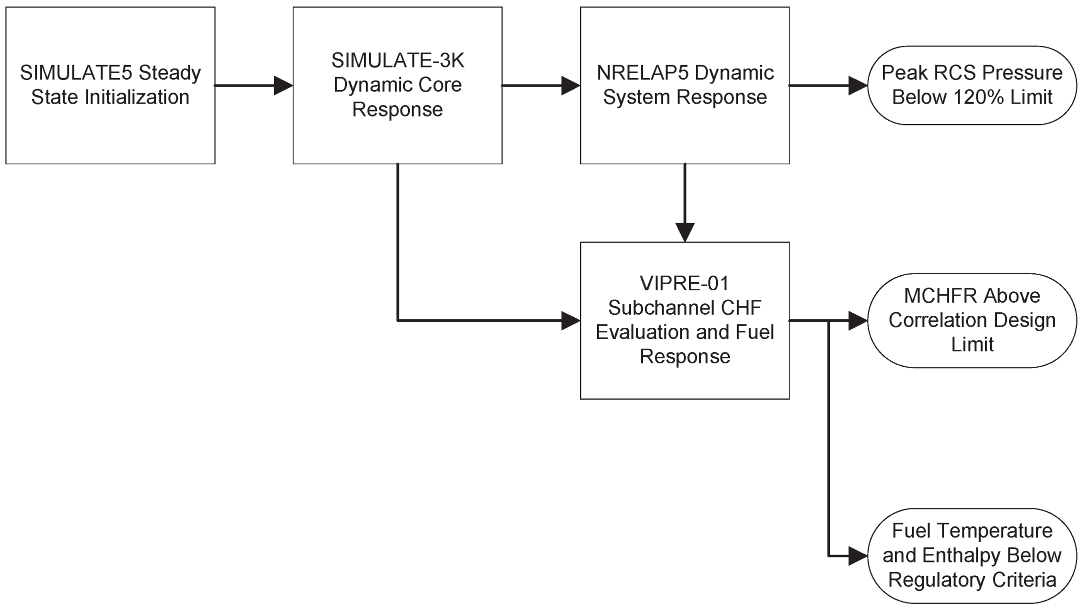

Figure 3-1 Calculation schematic for analyzing rod ejection accident

Section 5.2 through Section 5.5 further describe how the power as a function of time and elements of the power distributions calculated by SIMULATE-3K are used as input to NRELAP5 and VIPRE-01. 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 3-1, below, arranged such that the discipline in the row receives input from the discipline defined in the column.

Table 3-1 High-level discipline and code interface

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
Subchannel (VIPRE-01)	N/A	Radial power distribution (includes $F_{\Delta H}$), axial power distribution	Event thermal-hydraulic response (power, flow, temperature, pressure)

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3.2.1 Core Response

Reference 8.2.6 provides the validation of CASMO5/SIMULATE5 to perform steady state neutronics calculations for NPM~~the NuScale~~ designs. Validation of SIMULATE-3K for NPM~~the NuScale~~ designs is described in this section.

3.2.1.1 CASMO5

CASMO5 (Reference 8.2.15) 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 8.2.6.

3.2.1.2 SIMULATE5

SIMULATE5 (Reference 8.2.16) 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

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worth, shutdown margin, power distributions, delayed neutron fraction, and peaking factors.

In general, the SIMULATE5 core model is based on input of the core geometry and material compositions, core operating conditions, and core configuration. Core geometry is fully represented with radial nodes corresponding to a quarter of an assembly at numerous axial levels. Material properties and cross-sections are assigned to each node.

Section 4.1 identifies that ejected rod worth is the highest ranked phenomena for this event, as well as describing the two sub-components of worth: control rod and flux redistribution. Static worth is the difference between two static criticality calculations. Reference 8.2.6 provides a robust justification for the ability of SIMULATE5 to accurately predict critical conditions, power distributions, and depletion for of a broad range of reactor designs and operating conditions. Therefore, the SIMULATE5 code provides an excellent tool for predicting ejected rod worths. Additionally, code uncertainty, in the form of nuclear reliability factors (NRFs), conservatively bound differences in code prediction to measurement results and corresponding measurement uncertainty.

Control rod worth is primarily a function of cross-sections and number densities, resulting in a straightforward validation assessment.

The other component of ejected rod worth, flux redistribution, is more complex, with dependence on current power distribution (and the corresponding depletion history that is the integration of power distribution throughout irradiation history). As a result, the intra-assembly power gradient is important for the determination of ejected control rod worth, and thus for the dynamic consequences of the event. Like existing PWRs, an intra-assembly power gradient exists in all assemblies in the NPM core, especially for assemblies on the periphery. This gradient is due to geometric buckling (i.e., radial leakage). Such gradients occur regardless of core design, though lower leakage cores will tend to exhibit lower intra-assembly power gradients than higher leakage cores. Therefore, bounding evaluations of all intra-assembly power gradients, which inherently includes the corresponding depletion history, are evaluated in each application of this method. Because SIMULATE5 has been shown to reliability predict critical conditions, power distributions, and depletion, it also reliably predicts the ejected rod worth, including consideration of phenomena such as intra-assembly power and depletion gradients.

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 8.2.6.

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3.2.1.3 SIMULATE-3K

SIMULATE-3K (References 8.2.17, 8.2.18, and 8.2.19) 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.

The SIMULATE-3K input file is modified for differences between the codes, {{

}}2(a),(c) 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 5.2.1.1), and providing transient-specific inputs (Section 5.2.1.2).

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.

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

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.

The initialization and treatment of uncertainties of the system thermal-hydraulic parameters of moderator temperature and system flow are described as follows. 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. In the NRELAP5 analysis, 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. For hot zero power, the flow rate in NRELAP5 is modeled based on the natural circulation curve of a very low power (e.g., 0.001 percent) and the flow rate in SIMULATE-3K is modeled assuming a conservatively low value (e.g., 5 percent of rated flow).

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

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NuScale Power

RAIO-0618-60285

June 04, 2018

Docket: PROJ0769

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

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 9306 (eRAI No. 9306) on the NuScale Topical Report, "Rod Ejection Accident Methodology," TR-0716-50350, Revision 0

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 9306 (eRAI No. 9306)," dated April 04, 2018
2. NuScale Topical Report, "Rod Ejection Accident Methodology," TR-0716-50350, Revision 0, dated December 2016

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

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

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

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

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

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

Sincerely,

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

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

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

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

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

Enclosure 2:

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

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

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

NRC Question No.: 15.04.08-1

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

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

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

NuScale Response:

a. Studsvik Scandpower performed the SPERT-III benchmark demonstrating the ability of SIMULATE-3K to model the transient response of the reactor (Reference 8.2.22 of TR-0716-50350). Although not performed under NuScale's approved Appendix B quality assurance (QA) program, Studsvik performed this benchmark as part of their V&V program to demonstrate the ability of SIMULATE-3K to perform reactivity insertion events for super-prompt conditions. The SIMULATE-3K and SPERT-III benchmark comparisons show good agreement between SIMULATE-3K and the experimental results providing confidence that SIMULATE-3K can predict core power excursions for reactivity insertion events as seen in Figure 3 (S3K vs SPERT-III Cold Start-up Test 43. Inserted Reactivity 1.21$), Figure 5 (S3K vs SPERT-III Hot Start-up Test 70. Inserted Reactivity 1.21$), Figure 7 (S3K vs SPERT-III Hot Start-up Test 60. Inserted Reactivity 1.23$), Figure 9 (S3K vs SPERT-III Hot Stand-by Test 81. Inserted Reactivity 1.17$), and Figure 11 (S3K vs SPERT-III Full PowerTest 86. Inserted Reactivity 1.17$) of Reference 8.2.22. Differences in peak power shown in Figure 3 and Figure 9 of Reference 8.2.22 are attributed to experimental uncertainty in the initial position of the transient control rod, leading to uncertainty in the initial reactivity insertion.

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

Table 1: NEACRP Benchmark Results Comparison

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

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

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

   The software package delivery to NuScale was accompanied by installation test cases and user manual, methodology, and version change documentation. Upon delivery, configuration control is initiated and the software was subjected to appropriate controls within the NuScale QA program (Reference 8.1.3). In addition to the V&V activities performed by Studsvik Scandpower, software validation is performed for applications and use specific to NuScale. The QA program is compliant with Reference 8.1.1 of TR-0716-50350. The QA program governs activities associated with acquisition of commercial grade software, configuration control, validation, and dedication of the SIMULATE-3K code.

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

Impact on DCA:

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

Response to Request for Additional Information
Docket: PROJ0769

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

NRC Question No.: 15.04.08-3

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.

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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.: 15.04.08-4

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

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

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

NuScale Response:

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

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

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

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

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

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

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

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

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

Impact on DCA:

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

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

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

NRC Question No.: 15.04.08-5

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

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

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

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NuScale Response:

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

Figure 6-6 of the topical report is an example MCHFR plot as a function of time based on the power pulse provided in Figure 6-5. The peak power occurs at approximately 80 milliseconds, with a half-width-half-max pulse width lasting 60 milliseconds. An additional 60 milliseconds past the time of the peak pulse, the minimum CHFR occurs at approximately 140 milliseconds after the start of the event. The analytical limit delay for the reactor trip to begin (approximately 2 seconds of rod movement for full insertion) once detected is less than 2.5 seconds. Therefore, the key elements of the event are completely over before the excore detectors could be credited to intiate the reactor trip.

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

Impact on DCA:

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

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

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

NRC Question No.: 15.04.08-6

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

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

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

NuScale Response:

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

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

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

Impact on DCA:

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

Response to Request for Additional Information
Docket 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.

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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.

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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.: 15.04.08-11

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

Section 5.1.3 states that analysis of the REA will be performed at power levels from hot zero power (HZP) to hot full power (HFP) to bound the PDIL, axial offset limits, and moderator temperature. It is unclear to the staff, from the methodology described, how these values will be applied.

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

NuScale Response:

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

- Moderator Temperature : The moderator temperature is a function of core power and is set by the operating strategy for the plant. In the NRELAP5 analysis, the core flow is allowed to increase, however, the analysis is performed to minimize the flow increase with temperature, calculated to satisfy mass and energy conservation. The VIPRE-01 analysis uses the calculated core flow and core inlet temperature directly from NRELAP5 as an input forcing function.

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

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

Impact on DCA:

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

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

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

NRC Question No.: 15.04.08-12

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

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

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

NuScale Response:

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

Impact on DCA:

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

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

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

NRC Question No.: 15.04.08-13

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

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

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

NuScale Response:

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

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

Impact on DCA:

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

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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

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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.

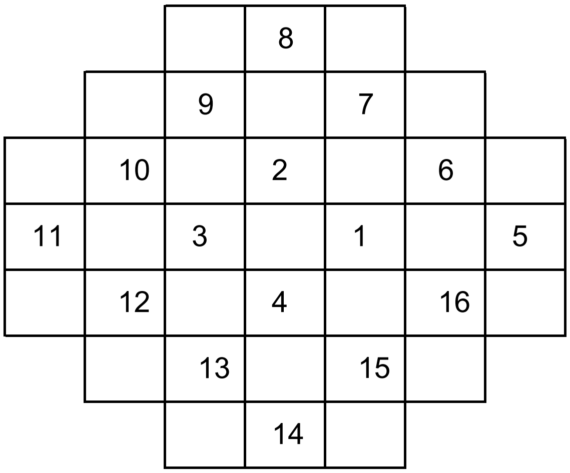

Figure 5-1 Control rod assembly layout for the NuScale Power Module

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.

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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.

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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.

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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

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:

<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 $F\Delta H$ given in Section 7.7.1, “Base Nuclear Reliability Factors,” of TR-0616-48793.

a. Please indicate if the uncertainties in Table 5-1 will be updated consistent with TR-0616-48793. If the uncertainties will not be updated as discussed in TR-0616-48793, either describe the method for updating them or provide a justification as to why an update is not necessary. If the uncertainties in Table 5-1 will be updated, modify TR-0716-50350 to indicate the method by which updates will be made.

b. Justify the use of a lower $F_{\Delta H}$ uncertainty for the rod ejection analysis relative to the steady-state $F_{\Delta H}$ uncertainty.

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.

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{{

}}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

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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.: 15.04.08-16

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.

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

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

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

{{

}}2(a),(c)

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

{{

}}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

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Enclosure 1:

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

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

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Enclosure 3:

Affidavit of Mark W. Shaver, AF-181136

NuScale Power, LLC

AFFIDAVIT of Mark W. Shaver

I, Mark W. Shaver, state as follows:

(1) I am the Director of Regulatory Affairsof 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 report reveals distinguishing aspects about the process 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 enclosed report entitled, "Rod Ejection Accident Methodology," TR-0716-50350-A, Revision 3. 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 April 11, 2025.

Mark W. Shaver

Mark W. Shaver