November 15, 2019

Docket: PROJ0769

U.S. Nuclear Regulatory Commission
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**SUBJECT:** NuScale Power, LLC Submittal of Topical Report, "Rod Ejection Accident Methodology," TR-0716-50350, Revision 1

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

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

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

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

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

Sincerely,

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

**Enclosure 1:**

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

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**Enclosure 2:**

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

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Licensing Topical Report

# Rod Ejection Accident Methodology

November 2019

Revision 1

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## Department of Energy Acknowledgement and Disclaimer

This material is based upon work supported by the Department of Energy under Award Number DE-NE0008820.

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## Abstract

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

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

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

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

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

## Executive Summary

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

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

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

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

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

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

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

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

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

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

The REA methodology includes the following components:

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

With the rapid nature of the power increase in the REA VIPRE-01 calculations, several deviations from the subchannel methodology (described in Reference 8.2.11), were used to increase convergence and reliability of the final results. The deviations from the subchannel methodology are discussed and justified in this report using the following sensitivity studies:

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

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

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

# 1.0 Introduction

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

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

## 1.1 Purpose

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

## 1.2 Scope

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

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

## 1.3 Abbreviations and Definitions

Table 1-1 Abbreviations

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

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

Table 1-2 Definitions

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

# 2.0 Regulatory Considerations

## 2.1 Regulatory Requirements

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

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

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

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

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

## 2.2 Regulatory Guidance Background

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

### 2.2.1 Current Regulatory Guidance and Standard Review Plan Criteria

#### 2.2.1.1 Reactor Coolant System Pressure

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

#### 2.2.1.2 Fuel Cladding Failure

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

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

#### 2.2.1.3 Core Coolability

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

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

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

#### 2.2.1.4 Fission Product Inventory

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

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

where,

FGR = fission gas release, percent (must be $\geq 0$)

$\Delta H$ = fuel enthalpy increase ($\Delta$cal/g)

#### 2.2.1.5 Effects of Loss of Primary System Integrity

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

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

### 2.2.2 Nuclear Regulatory Commission Proposed Changes to Criteria

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

#### 2.2.2.1 Reactor Coolant System Pressure

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

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

### 2.2.2.2 Fuel Cladding Failure

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

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

### 2.2.2.3 Core Coolability

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

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

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

#### 2.2.2.4 Fission Product Inventory

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

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

where,

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

$\Delta H$ = fuel enthalpy increase ($\Delta$cal/g)

## 2.3 Regulatory Criteria for NuScale

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

Table 2-1 Method for addressing regulatory criteria

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

### 2.3.1 Reactor Coolant System Pressure

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

### 2.3.2 Fuel Cladding Failure

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

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

### 2.3.3 Core Coolability

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

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

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

### 2.3.4 Fission Product Inventory

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

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

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

### 2.3.5 Effects of Loss of Primary System Integrity

The effects addressed in the NuScale REA methodology include:

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

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

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

## 3.0 Overview and Evaluation Codes

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

## 3.1 Overview

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

### 3.1.1 Reactivity Considerations

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

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

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

#### 3.1.1.1 Prompt Critical

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

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

where,

$E_d$ = energy deposition,

ρ = static ejected CRA worth,

β = delayed neutron fraction,

$C_p$ = fuel heat capacity, and

$\alpha_D$ = Doppler temperature coefficient (DTC).

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

#### 3.1.1.2 Sub-Prompt Critical

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

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

where,

$P_j$ = prompt jump power, and

$P_o$ = initial power.

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

### 3.1.2 Reactor Coolant System Pressure Behavior

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

## 3.2 Analysis Computer Codes and Evaluation Flow

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

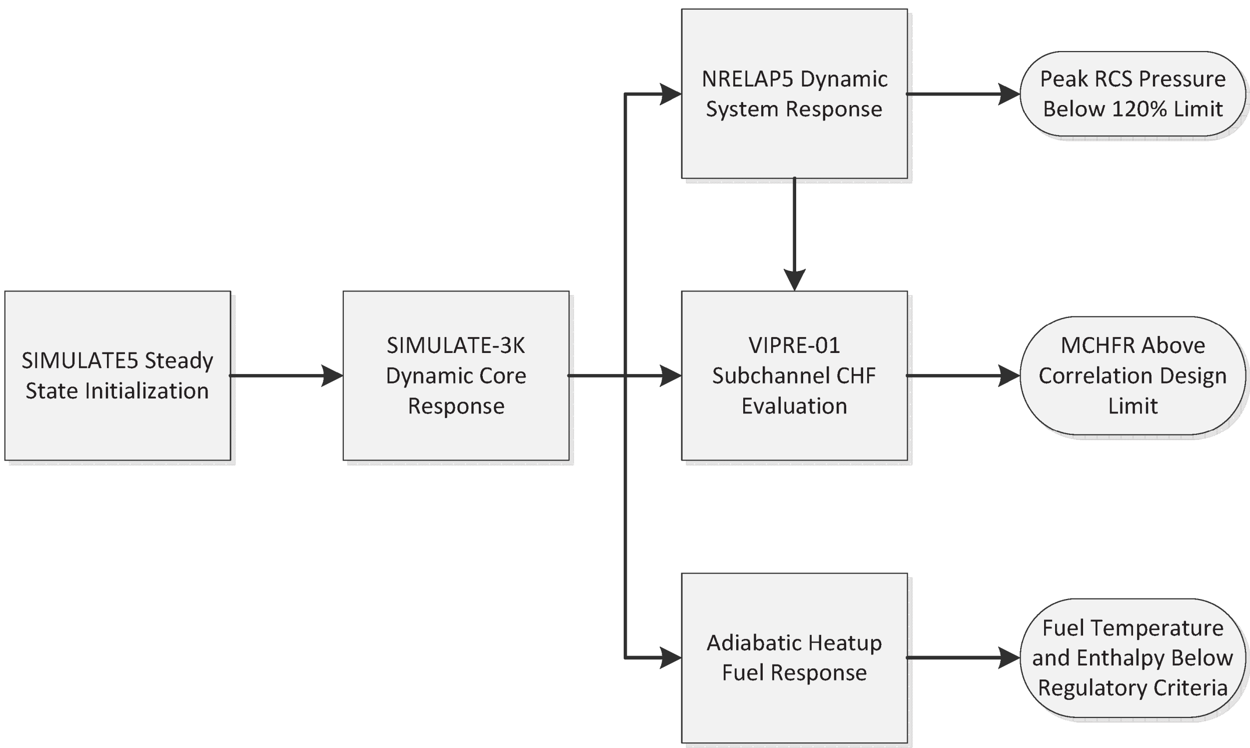


Figure 3-1 Calculation schematic for analyzing rod ejection accident

### 3.2.1 Core Response

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

#### 3.2.1.1 CASMO5

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

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

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

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

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

#### 3.2.1.2 SIMULATE5

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

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

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

#### 3.2.1.3 SIMULATE-3K

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

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

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

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

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

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

#### 3.2.1.4 Validation of SIMULATE-3K

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

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

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

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

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

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

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

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

### 3.2.2 System Response

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

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

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

### 3.2.3 Subchannel

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

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

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

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

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

### 3.2.4 Fuel Response

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

### 3.2.5 Accident Radiological Evaluation

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

## 4.0 Identification of Important Phenomena for Rod Ejection Accident

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

The overall goal of the evaluation of an REA is to

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

## 4.1 Industry Phenomena Identification and Ranking Table for Rod Ejection Accident

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

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

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

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

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

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

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

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

### 4.1.1 Plant Transient Analysis

#### 4.1.1.1 Calculation of Power History

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

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

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

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

#### 4.1.1.2 Calculation of Pin Fuel Enthalpy Increase

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

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

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

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

### 4.1.2 Fuel Response Analysis

#### 4.1.2.1 Initial Conditions

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

Cladding dimensions are used to calculate the maximum oxide to wall thickness ratio. This ratio is 0.0588 for the NuScale fuel; the fuel enthalpy rise limit is conservatively set at the inflection point of the 0.08 ratio in Figure 5-2. Using this ratio applies additional conservatism to the allowable fuel enthalpy rise.

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

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

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

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

#### 4.1.2.2 Fuel and Cladding Temperature Changes

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

For VIPRE-01 analyses, these parameters are addressed in the fuel rod conduction model. The fuel rod conduction model uses a calibration to COPERNIC (References 8.2.8 and 8.2.12) to develop conservative fuel property input that captures all of the effects of heat transfer from the fuel pellet to the fuel cladding, and ultimately to the coolant. Application of this model is discussed in Section 4.4 of the subchannel methodology topical report (Reference 8.2.11). As described in this report, calibration of VIPRE-01 fuel temperature predictions to the fuel performance analyses is performed for the fuel average, fuel surface, and cladding surface temperatures for each cycle. Fuel-design specific calibration results in temperature predictions that are conservative for MCHFR. The conservative bias for MCHFR is a high initial temperature of the fuel as well as a high gap conductance. This allows the amount of heat in the fuel to be conservatively high and transferred to the coolant the fastest. To ensure that the VIPRE-01 fuel conduction calculations are conservative, this methodology requires that the entire range of possible time-in-cycle parameters are evaluated using the COPERNIC fuel performance code, including exposure, uranium enrichment, gadolinium enrichment, gap conductance, and fuel density. The VIPRE-01 model is calibrated to ensure that it produces conservative temperatures for each fuel design.

## 4.2 Electric Power Research Institute Technical Report

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

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

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

## 4.3 Standard Review Plan Section 15.4.8 Initial Conditions

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

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

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

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

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

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

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

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

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

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

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

# 5.0 Rod Ejection Accident Analysis Methodology

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

## 5.1 Rod Ejection Accident Analysis General Assumptions

### 5.1.1 Cycle Design

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

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

### 5.1.2 Cycle Burnup

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

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

### 5.1.3 Core Power

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

### 5.1.4 Single Active Failure

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

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

### 5.1.5 Automatic System Response of Non-Safety Systems

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

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

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

### 5.1.6 Loss of Alternating Current Power

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

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

## 5.2 Core Response Methodology

### 5.2.1 Calculation Procedure

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

#### 5.2.1.1 Static Calculations

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

{{

}}2(a),(c)

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

#### 5.2.1.2 Transient Calculations with SIMULATE-3K

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

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

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

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

}}2(a),(c)

{{

}}2(a),(c)

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

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

}}2(a),(c)

### 5.2.2 Analysis Assumptions and Parameter Uncertainties for Core Response

#### 5.2.2.1 Control Rod Assembly Position

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

#### 5.2.2.2 Worst Rod Stuck Out

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

### 5.2.2.3 Input Parameters and Uncertainty Treatment

#### 5.2.2.3.1 Ejected Rod Time

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

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

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. Only the CRAs in the regulating bank are eligible for ejection and considered in the REA methodology.

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

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

#### 5.2.2.3.3 Reactor Trips

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

#### 5.2.2.3.4 Reactivity Feedback

The MTC and DTC are biased to be as least negative as possible. The effective delayed neutron fraction ($\beta_{eff}$) 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 $\beta_{eff}$ is conservative.

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

Table 5-1 Example uncertainties for rod ejection accident calculations

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

### 5.2.3 Results and Downstream Applicability

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

## 5.3 System Response

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

### 5.3.1 Calculation Procedure

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

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

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

- The SIMULATE-3K power response is used to maximize the impact on MCHFR. This tends to be a rapid, peaked power response due to using the maximum possible ejected CRA worth based on insertion to the PDIL.
- A reduced ejected CRA worth that raises the power quickly to just below both the high power and high power rate trip limits is used through the point kinetics model within NRELAP5, and reactivity feedback mechanisms are used to hold the power at this level. This delays the trip until the transient is terminated by high RCS pressure. These cases do not have an upstream SIMULATE-3K calculation.

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

#### 5.3.1.1 Minimum Critical Heat Flux Ratio

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

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

Consideration for conservative system conditions in MCHFR analysis includes

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

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

#### 5.3.1.2 Reactor Coolant System Pressurization

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

Considerations for conservative system conditions in peak pressure analysis include

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

### 5.3.2 Analysis Assumptions and Parameter Treatment for System Response

#### 5.3.2.1 Pressure Relief

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

#### 5.3.2.2 Core Power

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

#### 5.3.2.3 Direct Moderator and Cladding Heating

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

#### 5.3.2.4 Core Inlet Temperature

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

#### 5.3.2.5 Core Flow

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

#### 5.3.2.6 System Pressure and Pressurizer Level

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

### 5.3.3 Results and Downstream Applicability

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

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

## 5.4 Subchannel Response

### 5.4.1 Subchannel Calculation Procedure

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

#### 5.4.1.1 VIPRE-01 Deviations from Subchannel Methodology

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

- {{

}}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 and the pin peaking nuclear reliability factor are applied to the highest peaked $F_{\Delta H}$ rod. The uncertainties associated with $F_{\Delta H}$ are given in Table 5-1 and are combined using the root-sum-squared method similar to that discussed in Section 3.10.7 of Reference 8.2.11. The radial power distribution slope described in Section 3.10.6 of Reference 8.2.11 is used to determine the REA-specific normalized radial power distribution for use in VIPRE-01. In summary, the process for each case is to (i) determine the peak $F_{\Delta H}$ rod (ii) apply uncertainty to that rod only (iii) calculate a normalized power shape for both fully-detailed rods and lumped rods (iv) utilize artificial shape in VIPRE-01 simulation of the case.

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

#### 5.4.2.2 Axial Power Distribution

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

#### 5.4.2.3 Core Inlet Flow Distribution

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

#### 5.4.2.4 Fuel Conductivity and Gap Conductance

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

#### 5.4.2.5 Reactor Coolant System Pressure

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

### 5.4.3 Results and Downstream Applicability

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

## 5.5 Fuel Response

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

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

### 5.5.1 Fuel Temperature

The following equation defines the conservative temperature increase:

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

where,

$\Delta T$ = temperature increase,

$E_T$ = total energy,

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

$C_p$ = volumetric fuel heat capacity,

$V_{node}$ = nodal volume, and

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

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

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

### 5.5.2 Radial Average Fuel Enthalpy

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

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

where,

$h_i$ = maximum initial radial average fuel enthalpy,

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

$\rho_f$ = fuel density

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

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

where,

$\Delta h$ = radial average fuel enthalpy increase

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

### 5.5.3 Cladding Oxidation-Based Pellet-Cladding Mechanical Interaction Failure

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

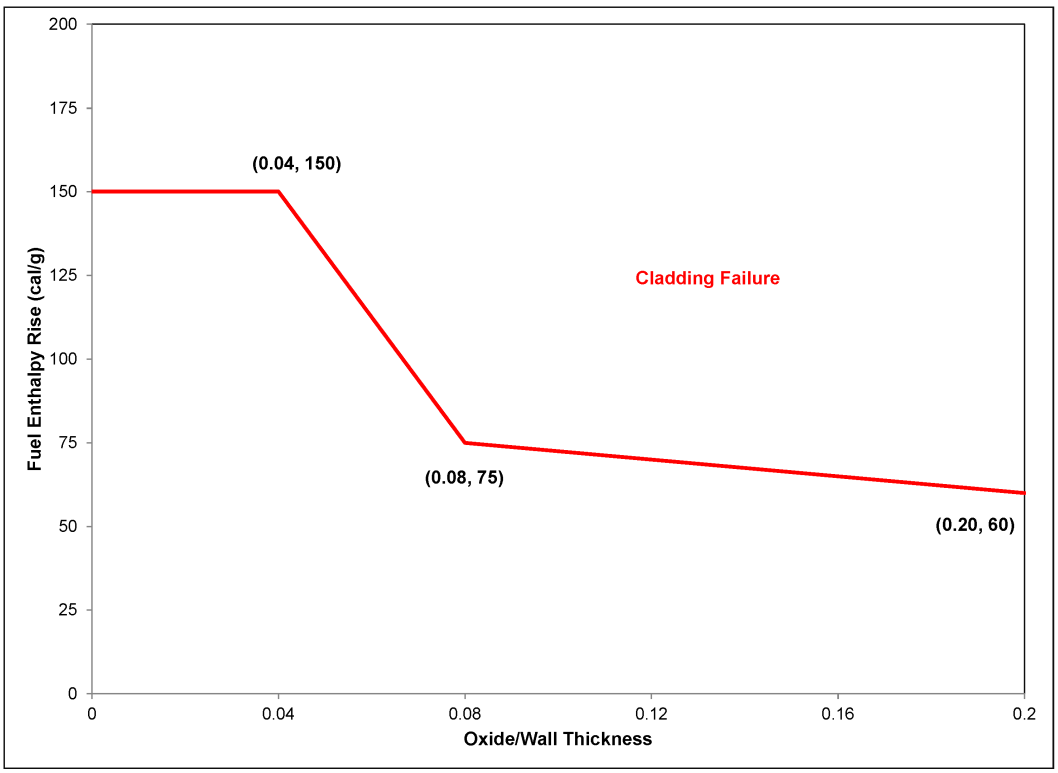


Figure 5-2 Pressurized water reactor pellet-cladding mechanical interaction fuel cladding failure criteria

## 5.6 Radiological Assessment

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

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

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

### 6.1 Rod Ejection Accident Sample Analysis System Pressure Response Results

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

Table 6-1 Conditions analyzed for sample calculations

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

### 6.2 Rod Ejection Accident Sample Analysis Fuel Response Results

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

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

The peak $F_Q$ before reactor trip is used to maximize the adiabatic heatup response for fuel enthalpy and temperature. $F_{\Delta H}$ at the peak reactor power is used in the VIPRE-01 for MCHFR analysis.

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

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

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

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

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

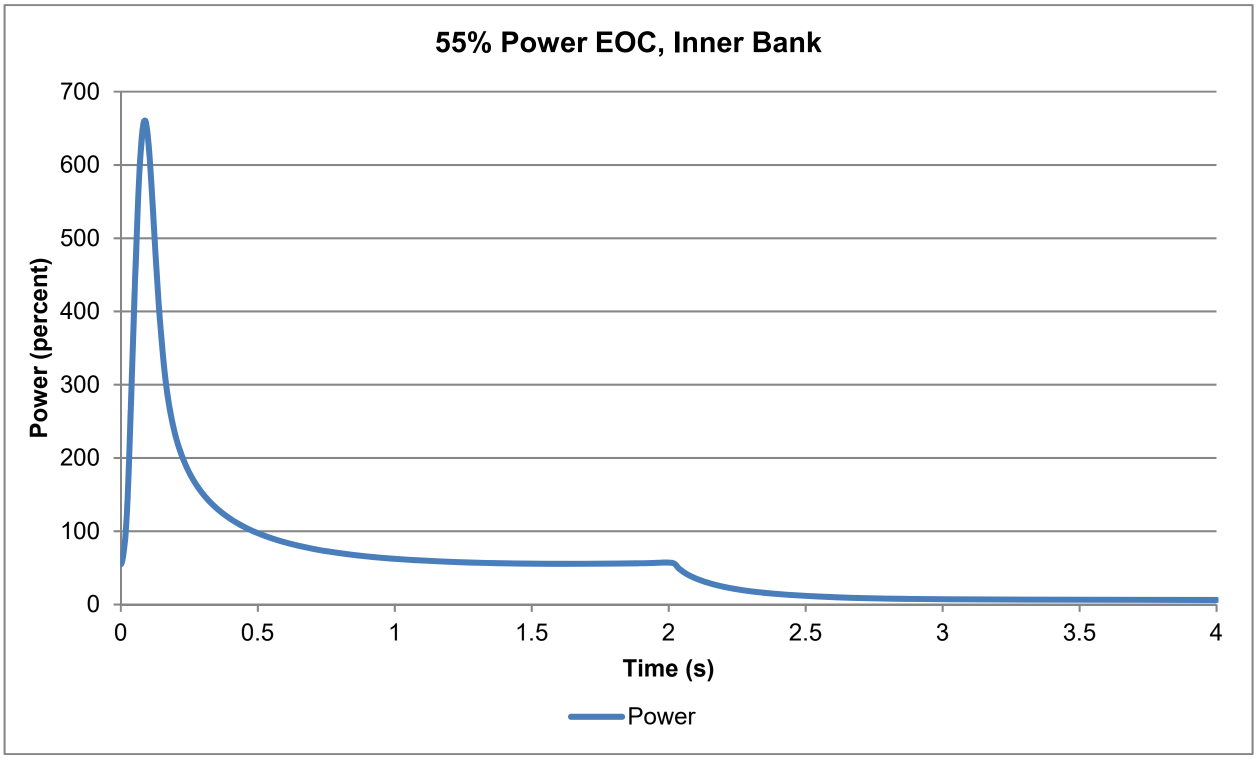


Figure 6-1 Power response at 55 percent power, end of cycle

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

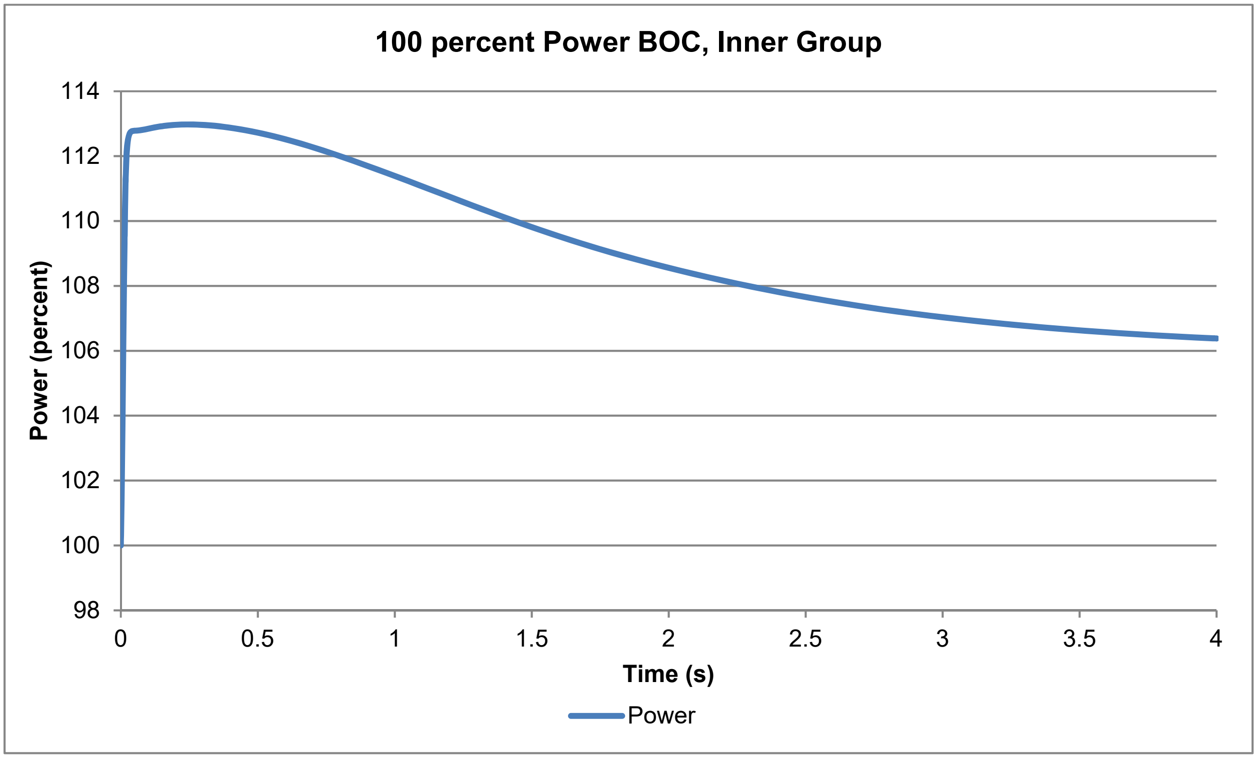


Figure 6-2 Power response at 100 percent power, beginning of cycle

## 6.3 Rod Ejection Accident Sample Analysis System Pressure Response Results

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

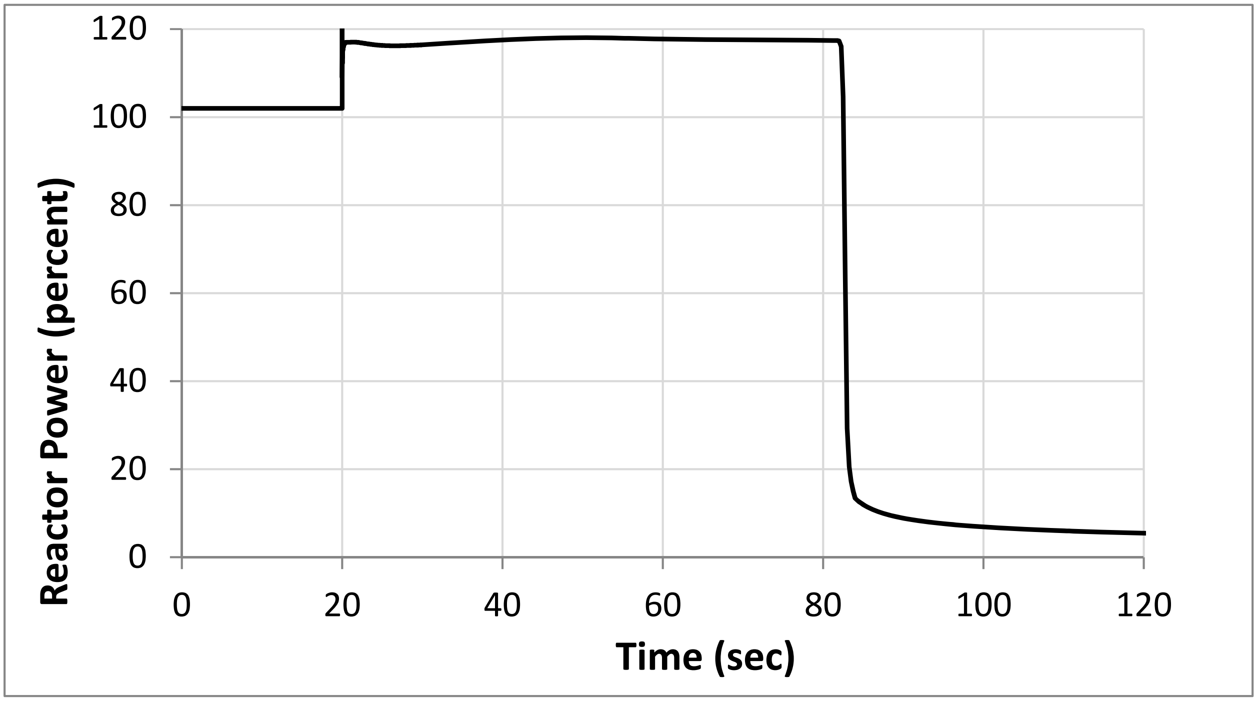


Figure 6-3 Power response for peak reactor coolant system pressure evaluation

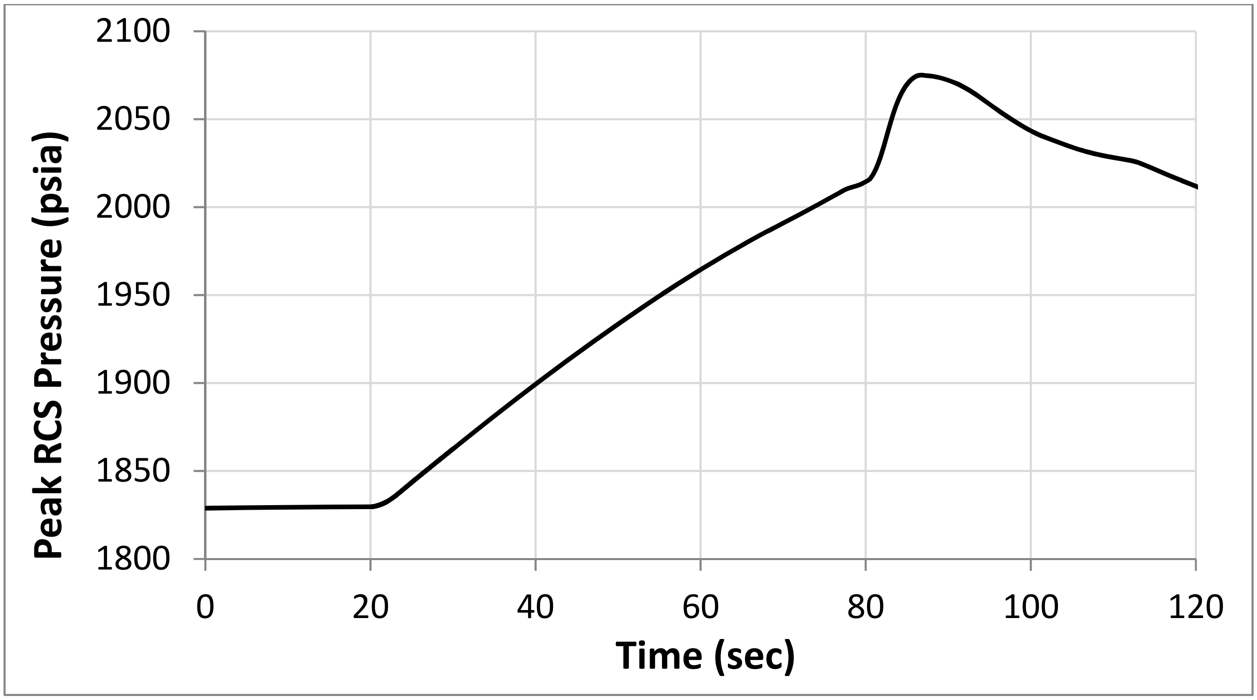


Figure 6-4 Pressure response for peak reactor coolant system pressure evaluation

### 6.4 Rod Ejection Accident Sensitivity Analysis Results

#### 6.4.1 NRELAP5 Minimum Critical Heat Flux Ratio Impacts

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

Table 6-4 NRELAP5 MCHFR impacts from sensitivity evaluation

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

### 6.4.2 VIPRE-01 Sensitivities

#### 6.4.2.1 Computational Time Steps

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

{{

}}2(a),(c)

Figure 6-5 Time step effect on power forcing function

#### 6.4.2.2 Code Axial Node Lengths

Figure 6-6 provides a comparison of various axial nodalizations used in VIPRE-01 compared to the resulting CHF value. The largest difference in the MCHFR from the nodalization used in the VIPRE-01 basemodel is {{ }}2(a),(c)

{{

}}2(a),(c)

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

### 6.4.2.3 Two-Phase Flow Correlation Options

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

{{

}}2(a),(c)

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

#### 6.4.2.4 Numerical Solution Damping Factors

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

}}2(a),(c)

{{

}}2(a),(c)

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

#### 6.4.2.5 Radial Power Distribution

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

{{

}}2(a),(c)

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

{{

}}2(a),(c)

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

{{

}}2(a),(c)

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

{{

}}2(a),(c)

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

{{

}}2(a),(c)

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

### 6.4.2.6 Fuel Rod Gap Conductance

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

{{

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

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

### 6.4.2.7 Reactor Coolant System Pressure

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

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

}}2(a),(c)

## 7.0 Summary and Conclusions

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

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

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

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

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

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

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

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

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

# 8.0 References

## 8.1 Source Documents

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

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

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

## 8.2 Referenced Documents

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

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

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

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

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

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

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

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

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

8.2.10 NuScale Topical Report, “Non-Loss-of-Coolant Accident Analysis Methodology,” TR-0516-49416 Revision 1, August 2017.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

**Enclosure 3:**

Affidavit of Zackary W. Rad, AF-1119-67635

**NuScale Power, LLC**

AFFIDAVIT of Zackary W. Rad

I, Zackary W. Rad, state as follows:

(1) I am the Director of Regulatory Affairs of NuScale Power, LLC (NuScale), and as such, I have been specifically delegated the function of reviewing the information described in this Affidavit that NuScale seeks to have withheld from public disclosure, and am authorized to apply for its withholding on behalf of NuScale.

(2) I am knowledgeable of the criteria and procedures used by NuScale in designating information as a trade secret, privileged, or as confidential commercial or financial information. This request to withhold information from public disclosure is driven by one or more of the following:

(a) The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by NuScale competitors, without a license from NuScale, would constitute a competitive economic disadvantage to NuScale.

(b) The information requested to be withheld consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage, as described more fully in paragraph 3 of this Affidavit.

(c) Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

(d) The information requested to be withheld reveals cost or price information, production capabilities, budget levels, or commercial strategies of NuScale.

(e) The information requested to be withheld consists of patentable ideas.

(3) Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale's competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying topical report reveals distinguishing aspects about the method by which NuScale develops its rod ejection accident methodology.

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

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

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

(4) The information sought to be withheld is in the enclosed topical report titled "Rod Ejection Accident Methodology," TR-0716-50350, Revision 1. The enclosure contains the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.

(5) The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies upon

the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC § 552(b)(4), as well as exemptions applicable to the NRC under 10 CFR § 2.390(a)(4) and 9.17(a)(4).

(6) Pursuant to the provisions set forth in 10 CFR § 2.390(b)(4), the following is provided for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld:

(a) The information sought to be withheld is owned and has been held in confidence by NuScale.

(b) The information is of a sort customarily held in confidence by NuScale and, to the best of my knowledge and belief, consistently has been held in confidence by NuScale. The procedure for approval of external release of such information typically requires review by the staff manager, project manager, chief technology officer or other equivalent authority, or the manager of the cognizant marketing function (or his delegate), for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside NuScale are limited to regulatory bodies, customers and potential customers and their agents, suppliers, licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or contractual agreements to maintain confidentiality.

(c) The information is being transmitted to and received by the NRC in confidence.

(d) No public disclosure of the information has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or contractual agreements that provide for maintenance of the information in confidence.

(e) Public disclosure of the information is likely to cause substantial harm to the competitive position of NuScale, taking into account the value of the information to NuScale, the amount of effort and money expended by NuScale in developing the information, and the difficulty others would have in acquiring or duplicating the information. The information sought to be withheld is part of NuScale's technology that provides NuScale with a competitive advantage over other firms in the industry. NuScale has invested significant human and financial capital in developing this technology and NuScale believes it would be difficult for others to duplicate the technology without access to the information sought to be withheld.

I declare under penalty of perjury that the foregoing is true and correct. Executed on November 15, 2019.

Zackary W. Rad

**Enclosure 4:**

Affidavit of Morris Byram, Framatome Inc.

**NuScale Power, LLC**
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360-0500 Fax 541.207.3928
www.nuscalepower.com

# AFFIDAVIT

1. My name is Morris Byram. I am Manager, Product Licensing, for FRAMATOME Inc. (FRAMATOME) and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by FRAMATOME to determine whether certain FRAMATOME information is proprietary. I am familiar with the policies established by FRAMATOME to ensure the proper application of these criteria.

3. I am familiar with the FRAMATOME information contained in the NuScale document TR-0916-50350, Revision 1, "Rod Ejection Accident Methodology," and referred to herein as "Document." Information contained in this Document has been classified by FRAMATOME as proprietary in accordance with the policies established by FRAMATOME Inc. for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by FRAMATOME and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by FRAMATOME to determine whether information should be classified as proprietary:

(a) The information reveals details of FRAMATOME's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for FRAMATOME.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for FRAMATOME in product optimization or marketability.

(e) The information is vital to a competitive advantage held by FRAMATOME, would be helpful to competitors to FRAMATOME, and would likely cause substantial harm to the competitive position of FRAMATOME.

The information in this Document is considered proprietary for the reasons set forth in paragraphs 6(b) and 6(e) above.

7. In accordance with FRAMATOME's policies governing the protection and control of information, proprietary information contained in this Document has been made available, on a limited basis, to others outside FRAMATOME only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. FRAMATOME policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

Moris E. Bergvall

SUBSCRIBED before me this 13th

day of November, 2019

OFFICIAL STAMP
TRACI MELINDA BALLER
NOTARY PUBLIC - OREGON
COMMISSION NO. 981969
MY COMMISSION EXPIRES JANUARY 06, 2023

Traci M Baller