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

"ROD EJECTION METHODOLOGY"

1 INTRODUCTION

1.1 Summary

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

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

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

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

1.2 Description of a Generic Rod Ejection Accident Transient Event

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

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

Enclosure 1

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

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

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

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

Two rod failure mechanisms are important in REA transients:

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

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

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

2 REGULATORY CRITERIA

2.1. Requirements

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

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

2.2. Relevant Guidance

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

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

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

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

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

3 SUMMARY OF TECHNICAL INFORMATION

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

4 TECHNICAL EVALUATION

4.1 Software Applicability

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

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

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

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

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

4.2 Methodology

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

4.2.1 Steady-State Initialization

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

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

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

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

4.2.2 Dynamic Core Response

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

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

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

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

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

4.2.3 Dynamic System Response

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

4.2.3.1 Peak Pressure Calculations

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

4.2.3.2 Minimum Critical Heat Flux Ratio

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

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

4.2.4 Subchannel Critical Heat Flux Evaluation

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

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

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

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

4.2.5 Adiabatic Heatup Fuel Response

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

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

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

5 CONCLUSIONS

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

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

6 LIMITATIONS AND CONDITIONS

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

7 REFERENCES

1. NuScale Power, LLC, TR-0716-50350-P, "Rod Ejection Accident Methodology," Revision 1, December 2016, ADAMS Accession No. ML19319C684.

2. U.S. Nuclear Regulatory Commission, NUREG-0800, "NUREG-0800 - Chapter 15, Section 15.4.8, Revision 3, Spectrum of Rod Ejection Accidents (PWR)," March 2007, ADAMS Accession No. ML070660036.

3. U.S. Nuclear Regulatory Commission, NUREG-0800, “NUREG-0800 - Chapter 4, Section 4.2, Revision 3, Fuel System Design,” March 2007, ADAMS Accession No. ML070740002.

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

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

6. NuScale Power, LLC, TR-0516-49422-P-A, “Loss-of-Coolant Accident Evaluation Model,” Revision 1, November 2019, ADAMS Accession Number ML19331B585.

7. 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. U.S. Nuclear Regulatory Commission, “Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” July 2000, ADAMS Accession No. ML003716792.

9. U.S. Nuclear Regulatory Commission, “Regulatory Guide 1.195, Methods and Assumptions for Evaluating Radiological Consequences of Design-Basis Accidents at Light-Water Nuclear Power Reactors,” May 2003, ADAMS Accession No. ML031490640.

10. NuScale Power, LLC, TR-0616-48793-P-A, “Nuclear Analysis Codes and Methods Qualification,” Revision 1, December 14, 2018, ADAMS Accession Nos. ML18348B035 (package) and ML18348B036 (public version).

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

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

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

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