



July 12, 2019

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Supplemental Response to NRC Request for Additional Information No. 450 (eRAI No. 9498) on the NuScale Design Certification Application

REFERENCES: 1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 450 (eRAI No. 9498)," dated May 01, 2018
2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 450 (eRAI No.9498)," dated June 28, 2018

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

The Enclosure to this letter contains NuScale's supplemental response to the following RAI Question from NRC eRAI No. 9498:

- 15-9

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

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

Sincerely,

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Enclosure 1: NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9498



Enclosure 1:

NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9498

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

eRAI No.: 9498

Date of RAI Issue: 05/01/2018

NRC Question No.: 15-9

Appendix A to Part 50 - General Design Criteria (GDC) for Nuclear Power Plants states, "...The principal Design Criteria establish the necessary design, fabrication, construction, testing and performance requirements for structures, systems and components important to safety..." The categorization of the Design Basis Events (DBEs) specified for the NuScale design in Final Safety Analysis Report (FSAR) Section 15.0 determines, in part, which of the GDCs apply to which events. NuScale DSRS Section 15.0 notes that the staff must ensure that the applicant's selection and assembly of the plant transient and accident analyses represent a sufficiently broad spectrum of transients and accidents, or initiating events. In particular, initiating events are categorized according to expected frequency of occurrence and by type to provide a basis for selection of the applicable analysis acceptance criteria and to provide a basis for comparison between events, which makes it possible to identify and evaluate the limiting cases.

The staff finds the reference to not applicable (N/A) in Table 15.0-1, Design Basis Events, confusing. Some events, such as startup of an inactive loop or boiling water reactor (BWR) specific events, are not possible based on the lack of design features. In these instances, N/A is appropriate. However, N/A also appears when specific design features exist such that the event falls into a beyond design basis category. For example, the NuScale Power Module (NPM) drop, described in FSAR section 15.7.6, appears to state that specific design features of the NPM movement system are single failure proof, and hence NPM drop is categorized as a beyond design basis event.

1) The staff is requesting the applicant modify FSAR Table 15.0-1 to clarify and distinguish events that are not applicable based on a lack of design features from events that are considered beyond design basis based on component or system design features. Further, the staff seeks to understand why station blackout is not included in the special events section of

Table 15.0-1, since FSAR Section 15.0.0.2 defines "special events" as beyond design bases events that are explicitly defined by regulation.

2) The staff also requests the long-term, return to power scenario described in FSAR Section 15.0.6, and the computer codes used to evaluate the event, be added to denote its design basis event classification since the scenario can occur within 72 hours following an abnormal operating occurrence or postulated accident using design basis assumptions.

NuScale Response:

In a conference call on December 11, 2018, the NRC requested that Final Safety Analysis Report (FSAR) Table 15.0-1 be clarified to indicate that the Return to Power Event is classified as an Anticipated Operational Occurrence (AOO). In a conference call on April 15, 2019, NuScale described to the NRC that the Return to Power Event is an event progression unique to the NuScale Power Module. NuScale agreed that the Return to Power Event is evaluated against AOO acceptance criteria. Table 15.0-1 was modified by two footnotes, one to clarify that 15.0.6 is an event progression, not an initiating event, and a second footnote to indicate which classes of events could result in a return to power.

Further, in the April 15, 2019 conference call, the NRC and NuScale discussed revising the FSAR to more clearly specify the NuScale Acceptance Criteria used for the Chapter 15 accidents and transients provided in FSAR Tables 15.0-2, 15.0-3 and 15.0-4. The acceptance criteria establish that specified acceptable fuel design limits (SAFDLs) are met for all events. Meeting SAFDLs is established by ensuring the reactor core has adequate heat transfer to provide sufficient margin to CHF with a 95-percent probability at the 95-percent confidence level (95/95 DNBR limit). Some additional acceptance criteria, that are more limiting for parameters other than fuel cladding, are also listed. For example, the 230 calorie per gram limit protects fuel pellet integrity for core coolability. The 95/95 DNBR limit also applies to core coolability to protect the cladding. NuScale has provided markups to the requested tables as well as markups to other sections of the FSAR to make the verbiage consistent throughout.

Also, in the April 15, 2019 conference call, NuScale agreed to provide a clarification to the Principal Design Criterion (PDC) 27 language provided in FSAR Section 3.1.3.8. Additional information was added to FSAR Section 3.1.3.8 to provide the rationale for the revised PDC 27 wording.

See the attached FSAR markup for details of these changes.



Impact on DCA:

FSAR Sections 3.1.3.8, 15.0.2, 15.0.5, 15.4-8 and Tables 15.0-1, 15.0-2, 15.0-3, 15.0-4 and 15.4-24 have been revised as described in the response above and as shown in the markup provided in this response.

Relevant FSAR Chapters and Sections

- Section 3.9.4 Control Rod Drive System
- Section 4.3 Nuclear Design
- Section 4.6 Functional Design of Control Rod Drive System
- Section 9.3 Process Auxiliaries
- Chapter 15 Transient and Accident Analyses

3.1.3.8 Criterion 27-Combined Reactivity Control Systems Capability

The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

Implementation in the NuScale Power Plant Design

GDC 27 is not applicable to the NuScale design. The following PDC has been adopted:

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The reactivity control systems shall be designed to have a combined capability of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained. Following a postulated accident, the control rods shall be capable of holding the reactor core subcritical under cold conditions with all rods fully inserted.

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~~Following a postulated accident, the control rods shall be capable of holding the reactor core subcritical under cold conditions, without margin for stuck rods, provided the specified acceptable fuel design limits for critical heat flux would not be exceeded by the return to power.~~

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~~The CVCS, with boron addition, and CRDS are designed for a combined capability of controlling reactivity changes that assures the capability to cool the core under postulated accident conditions with margin for stuck rods as explained in Section 4.3.1.5. Conservative analysis indicates that a return to power could occur following a reactor trip under the condition that the highest worth CRA does not insert, coincident with the CVCS being unavailable. Consequently, the GDC is modified for the NuScale design to address the shutdown capability for postulated accidents. Consistent with GDC 27, this PDC requires that the reactivity control systems function, together with heat removal systems, to protect the core from unacceptable damage under accident conditions. This protection function is met by providing sufficient reactivity control such that core cooling is maintained under accident conditions, analyzed using~~

conservative methodology and assumptions including margin equivalent to the highest worth rod stuck out. Under the NuScale design basis, during normal operation sufficient negative reactivity is maintained (instantaneous shutdown margin) to ensure that the capability to cool the core is maintained under accident conditions by rapid control rod insertion with the highest worth rod stuck out.

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The PDC also includes a post-accident holddown criterion specific to the NuScale design. This provision requires the control rods to be capable of maintaining the core subcritical under cold conditions following a postulated accident, without margin for the highest worth rod stuck out. Conservative analysis indicates that a post-accident return to power could occur following initial shutdown, under the condition that the highest worth CRA does not insert. The CVCS system is capable of providing negative reactivity but is not credited in this analysis since it is not a safety-related system. Section 15.0.6 demonstrates that the passive heat removal safety systems provide sufficient thermal margin such that a return to power does not result in the failure of the fuel cladding fission product barrier, as demonstrated by not exceeding SAFDLs for the analyzed events.

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The reactivity control capability required by either GDC 27 or PDC 27 provides assurance that even if a postulated accident damages fuel, continued core cooling will not be precluded and thus accident consequences can be maintained within acceptable limits. The NuScale design assures that fuel cladding integrity is maintained for all design basis events, including postulated accidents, such that the effect of a postulated return to power with failed fuel has not been evaluated in the analysis of accident consequences. Therefore to preclude unanalyzed accident consequences, NuScale's design basis implements PDC 27 in Chapter 15 to prohibit fuel failures under postulated accident conditions.

Conformance or Exception

The NuScale Power Plant design departs from GDC 27 and supports an exemption from the criterion. The NuScale Power Plant design conforms to PDC 27.

Relevant FSAR Chapters and Sections

- Section 3.9.4 Control Rod Drive System
- Section 4.2 Fuel System Design
- Section 4.3 Nuclear Design
- Section 4.6 Functional Design of Control Rod Drive System
- Section 6.3 Emergency Core Cooling System
- Section 9.3.4 Chemical and Volume Control System

not go into post-critical heat flux (CHF) heat transfer, and ensuring that the containment pressure and temperature for the limiting collapsed liquid level case remains below design limits ensures that the Appendix K limits for PCT, oxidation, and hydrogen production are not violated.

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There is no oxidation of the cladding as a result of a LOCA in the NPM. There are no changes in core geometry resulting from a LOCA that would prevent the core from being amenable to cooling. Therefore, the first four [10 CFR 50.46](#) acceptance criteria are met when the collapsed liquid level remains above the top of the core, the critical heat flux ratio is greater than 1.29, ~~and containment pressure and temperature remain below design limits for the entire LOCA.~~ [The NuScale specific LOCA acceptance criteria are listed in Table 15.0-4.](#)

The calculated core temperature is maintained at an acceptably low value and decay heat is removed in both the short-term and long-term of a LOCA in the NPM. The long-term evaluation of core temperature and decay heat removal is assessed in Reference 15.0-7.

15.0.2.2.2 Non-Loss-Of-Coolant Accident Methodology

The main steps of the non-LOCA system transient analysis process are:

- 1) Perform steady state and transient system analysis calculation with NRELAP5.
- 2) Evaluate results to confirm margin to RCS and steam generator pressure acceptance criteria.
- 3) Identify if a subchannel analysis is necessary based on system response.
- 4) Perform a subchannel analysis for those events identified in step 3.

For step 1, NRELAP5 is the thermal-hydraulics code used to calculate the NPM system response short-term transient event progression. The NuScale LOCA evaluation model was developed following the EMDAP guidelines of RG 1.203, as outlined in Reference 15.0-3. The NuScale non-LOCA EM starts with the LOCA EM and modifies it for use for non-LOCA events, as described in Reference 15.0-5. The requirements of the non-LOCA evaluation model capability are established based on the analysis purpose and plant design.

The EMDAP defined in RG 1.203 provides a four-element structured process to establish the adequacy of a methodology for evaluating complex events that are postulated to occur in nuclear power plant systems using the guidance of RG 1.203. The evaluation model has been developed using the guidance of RG 1.203 for simulating the NPM system transient response to non-LOCA events. Reference 15.0-5 describes the modifications made to the LOCA evaluation model to develop the non-LOCA evaluation model.

The short-term non-LOCA transient calculations presented in Reference 15.0-5 cover transient initiation and reactor trip, and demonstrate stable natural

circulation is achieved and effective DHRS operation has been established. The transient progression from this point is similar regardless of the specific initiating event, and the subsequent transient progression is treated as part of long-term decay and residual heat removal analysis discussed in Section 15.0.5.

The NPM parameters used in DBE evaluations are identified in Table 15.0-6. Table 15.0-7 lists the analytical limits and the associated time delays used in the Chapter 15 DBEs. Results of the DBE analyses are compared to the acceptance criteria identified in Table 15.0-2 through Table 15.0-5. System response is evaluated with respect to depressurization rates, to determine if the event is bounded by another DBE. Events with a slow depressurization rate that tend toward increasing CHF, which are bounded by events with a rapid depressurization rate, are not specifically analyzed for CHF with a subchannel analysis. VIPRE-01 is used to perform the subchannel analysis, and Table 15.0-1 identifies the DBEs for which a subchannel analysis is performed.

For the rod ejection calculations, a combination of CASMO5, SIMULATE5, and SIMULATE-3K (S3K) are used to calculate the core response and reactivity-related inputs. S3K is used to calculate fuel energy deposition and temperatures. The power response for the accident is determined by S3K for both NRELAP5 and VIPRE-01.

NRELAP5 is used to calculate system response including data such as flow rates, pressures and temperatures. NRELAP5 results are used as boundary conditions for the subchannel analysis.

VIPRE-01 is used to perform the detailed subchannel calculations to determine the minimum critical heat flux ratio (MCHFR) for each event, and to assess potential fuel damage based on CHF. The calculated fuel damage results are used to perform the accident radiological consequences using the methodology described in Section 15.0.3. Peak fuel temperatures, clad temperatures, radially averaged fuel enthalpy, and CHF calculations are performed with VIPRE-01 to address the event acceptance criteria. The description of the codes and the methodology for rod ejection accident analysis are described in more detail in Reference 15.0-11.

15.0.2.2.3 Flow Stability

The NPM system response is obtained by the NuScale proprietary computer code, PIM, which is used to demonstrate system stability at steady-state operation. The PIM code is described in Section 4.4.7. The PIM code relies on the published description of the theory and numerical methods of RAMONA, but is not a direct derivative of the coding. The PIM code has been developed independently to suit the geometry and specific needs of the NPM. The main advantage of the RAMONA-type algorithm is the absence or insignificance of numerical damping that affects other time-domain codes and requires extensive studies and adjustments before they can be successfully benchmarked and reliably used. Reference 15.0-10 provides details about the process used to select and qualify the PIM code.

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Table 15.0-1: Design Basis Events

Section	Type	Classification	Computer Code Used
15.0	Transient and Accident Analysis		
15.0.3	Radiological Consequences of Category 2 Maximum Hypothetical Accident <u>Iodine Spike Design-Basis Source Term (10 CFR 52.47(a)(2)(iv))</u>	Postulated Accident N/A ⁽⁷⁾	RADTRAD ORIGEN STARNAUA- pH_T ARCON96
15.0.6	Return to Power Event - NuScale specific <u>event progression</u> phenomenon	N/A ^(6,4)	NRELAP5
15.1	Increase in Heat Removal by Secondary System⁽⁸⁾		
15.1.1	Decrease in Feedwater Temperature	AOO	NRELAP5 VIPRE-01
15.1.2	Increase in Feedwater Flow	AOO	NRELAP5 VIPRE-01
15.1.3	Increase in Steam Flow	AOO	NRELAP5 VIPRE-01
15.1.4	Inadvertent Opening of Steam Generator Relief or Safety Valve	AOO	NRELAP5 VIPRE-01
15.1.5	Steam Piping Failures Inside and Outside of Containment	Postulated Accident	NRELAP5 VIPRE-01 RADTRAD ORIGEN ARCON96
15.1.6	Loss of Containment Vacuum/Containment Flooding	AOO	NRELAP5 VIPRE-01
15.2	Decrease in Heat Removal by the Secondary System⁽⁸⁾		
15.2.1	Loss of External Load	AOO	NRELAP5 VIPRE-01
15.2.2	Turbine Trip	AOO	NRELAP5 VIPRE-01
15.2.3	Loss of Condenser Vacuum	AOO	NRELAP5 VIPRE-01
15.2.4	Closure of Main Steam Isolation Valve	AOO	NRELAP5 VIPRE-01
15.2.5	Steam Pressure Regulator Failure (Closed)	N/A ⁽¹⁾	N/A

Table 15.0-1: Design Basis Events (Continued)

Section	Type	Classification	Computer Code Used
15.2.6	Loss of Non-Emergency AC to the Station Auxiliaries	AOO	NRELAP5 VIPRE-01
15.2.7	Loss of Normal Feedwater Flow	AOO	NRELAP5 VIPRE-01
15.2.8	Feedwater System Pipe Breaks Inside and Outside Containment	Postulated Accident	NRELAP5 VIPRE-01 RADTRAD ORIGEN ARCON96
15.2.9	Inadvertent Operation of the Decay Heat Removal System	AOO	NRELAP5 VIPRE-01
15.3	Decrease in RCS Flow Rate (not applicable)		
15.4	Reactivity and Power Distribution Anomalies⁽⁸⁾		
15.4.1	Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power or Startup	AOO	NRELAP5 VIPRE-01
15.4.2	Uncontrolled Control Rod Assembly Withdrawal at Power	AOO	NRELAP5 VIPRE-01
15.4.3	Control Rod Misoperation (System Malfunction or Operator Error)	AOO	VIPRE-01 SIMULATE5
15.4.4	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature	N/A ⁽¹⁾	N/A
15.4.5	Flow Controller Malfunction Causing an Increase in Core Flow Rate (Boiling Water Reactor)	N/A ⁽¹⁾	N/A
15.4.6	Inadvertent Decrease in Boron Concentration in Reactor Coolant System	AOO	N/A
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	IE	SIMULATE5, VIPRE-01
15.4.8	Spectrum of Rod Ejection Accidents	Postulated Accident	SIMULATE-3K NRELAP5 VIPRE-01 RADTRAD ORIGEN ARCON96
15.5	Increase in Reactor Coolant Inventory⁽⁸⁾		
15.5.1	Chemical and Volume Control System Malfunction	AOO	NRELAP5 VIPRE-01

Table 15.0-1: Design Basis Events (Continued)

Section	Type	Classification	Computer Code Used
15.6	Decrease in Reactor Coolant Inventory⁽⁸⁾		
15.6.1	Inadvertent Opening of Reactor Safety Valve	AOO	See 15.6.6
15.6.2	Failure of Small Lines Carrying Primary Coolant Outside Containment	IE	NRELAP5 RADTRAD ORIGEN ARCON96
15.6.3	Steam Generator Tube Failure	Postulated Accident	RADTRAD NRELAP5 ORIGEN ARCON96
15.6.4	Main Steam Line Failure Outside Containment (BWR)	N/A ⁽¹⁾	N/A
15.6.5	Loss-of-Coolant Accidents Resulting From a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	Postulated Accident	NRELAP5
15.6.6	Inadvertent Operation of Emergency Core Cooling System	AOO	NRELAP5
15.7	Radioactive Release from a Subsystem or Component		
15.7.1	Gaseous Waste Management System Leak or Failure	N/A ⁽²⁾	N/A
15.7.2	Liquid Waste Management System Leak or Failure	N/A ⁽²⁾	N/A
15.7.3	Postulated Radioactive Releases Due to Liquid Containing Tank Failures	N/A ⁽²⁾	RADTRAD, ORIGEN, ARCON96
15.7.4	Fuel Handling Accidents	Postulated Accident	RADTRAD, ORIGEN, ARCON96
15.7.5	Spent Fuel Cask Drop Accident	Postulated Accident	Not analyzed
15.7.6	NuScale Power Module Drop Accident	N/A ⁽³⁾	Not analyzed
Special Events			
15.8	Anticipated Transient Without Scram-(10 CFR 50.62)	Special Event	No analysis required.
15.9	Stability - note that stability is not an event. The NPM is protected from this phenomenon by MPS trips and technical specification initial conditions.	N/A ⁽⁴⁾	PIM

Table 15.0-1: Design Basis Events (Continued)

Section	Type	Classification	Computer Code Used
15.10	Core Damage Source Term (10 CFR 52.47(a)(2)(iv))	Special Event	RADTRAD ORIGEN STARNAUA pH _T ARCON96 MELCOR
8.4	Station Blackout (10 CFR 50.63)	N/A ⁽⁵⁾	NRELAP5

Notes:

- (1) Design feature is not part of NuScale design.
- (2) Events are described in Chapter 11.
- (3) Module drop is considered a Beyond Design Basis Event.
- (4) Event is analyzed to AOO Acceptance Criteria.
- (5) Event is included in the loss of non-emergency AC power analysis described in Section 15.2.6.
- (6) This is not an initiating event, however, AOO acceptance criteria are met. See Section 15.0.6 for details.
- (7) The iodine spike DBST is not an event, rather it serves as a bounding surrogate for design-basis loss of primary coolant into containment events described in Section 15.6.
- (8) A return to power can occur during the progression of events that involve a cooldown using DHRS or ECCS cooling when decay heat levels and boron concentration are low and control rods are not fully inserted.

RAI 15-9S1, RAI 15-15

Table 15.0-2: Acceptance Criteria-Thermal Hydraulic and Fuel

Classification ⁽⁵⁾	Fuel Clad ⁽¹⁾	RCS Pressure	Main Steam System Pressure	Containment	Event Progression
AOO	Fuel cladding integrity shall be maintained by ensuring that SAFDLs are met. ⁽⁶⁾ minimum DNBR remains above the 95/95 DNBR limit.	≤ 110% of system design pressure	≤ 110% of system design pressure	Peak pressure ≤ design pressure ⁽⁴⁾	An AOO should not develop into a more serious plant condition without other faults occurring independently. Satisfaction of this criterion precludes the possibility of a more serious event during the lifetime of the plant.
IE	Fuel cladding integrity will be maintained if the minimum DNBR remains above the 95/95 DNBR limit. If the minimum DNBR does not meet these limits, then the fuel is assumed to have failed. Fuel cladding integrity shall be maintained by ensuring that SAFDLs are met. ⁽⁶⁾	≤ 120% of system design pressure	≤ 120% of system design pressure	Peak pressure ≤ design pressure ⁽⁴⁾	Shall not, by itself, cause a consequential loss of required functions of systems needed to cope with the fault, including those of the RCS and the reactor containment system.
Postulated Accidents ^{(2),(3)}	Fuel cladding integrity will be maintained if the minimum DNBR remains above the 95/95 DNBR limit. If the minimum DNBR does not meet these limits, then the fuel is assumed to have failed. Fuel cladding integrity shall be maintained by ensuring that SAFDLs are met. ^{(6),(7)}	≤ 120% of system design pressure	≤ 120% of system design pressure	Peak pressure ≤ design pressure ⁽⁴⁾	Shall not, by itself, cause a consequential loss of required functions of systems needed to cope with the fault, including those of the RCS and the reactor containment system.
Special Event (SBO)	Core cooling	refer to Section 8.4	N/A	N/A	N/A

Notes:

(1) Minimum critical heat flux ratio (MCHFR) is used instead of minimum DNBR, as described in Section 4.4.2.

(2) See Table 15.0-3 for acceptance criteria for the Rod Ejection Accident.

(3) See Table 15.0-4 for acceptance criteria for Loss of Coolant Accidents.

(4) See Section 6.2.1.1 for containment pressure design limits.

(5) The iodine spike DBST and core damage event associated CDST do not have thermal hydraulic or fuel acceptance criteria.

(6) Specified Acceptable Fuel Design Limits (SAFDLs) are met by assuring that MCHFR is maintained above the 95/95 limit.

(7) SAFDLs are met during postulated accidents to ensure fuel cladding integrity is maintained should a return to power occur during the progression of the event.

Table 15.0-3: Acceptance Criteria Specific to Rod Ejection Accidents ⁽¹⁾

Purpose	Conditions	Parameter	Acceptance Criteria
For identification of failed fuel for radiological assessment <u>To assure no fuel failure occurs</u>	Zero power, fuel rods with internal pressure at or below system pressure	Maximum peak radial average fuel enthalpy (if exceeded, fuel is assumed to have failed)	≤ 100-170 cal/g
	Zero power, fuel rods with internal pressure greater than system pressure	Maximum peak radial average fuel enthalpy (if exceeded, fuel is assumed to have failed)	≤ 150 cal/g
	5% to 100% power	Minimum DNBR (if violated, fuel is assumed to have failed) ^(1,2)	≥ 95/95 DNBR limit ^(1,2)
	Function of cladding oxide/wall thickness ratio	<u>Change in radial average fuel enthalpy</u> Per Standard Review Plan 4.2, Figure B-1	Per Standard Review Plan 4.2, Figure B-1 <u>See Reference 15.0-11, Figure 5-2</u>
	<u>N/A</u>	<u>Maximum peak fuel temperature</u>	<u>≤ melting temperature</u>
For assessment of core coolability	N/A	Maximum peak radial average fuel enthalpy	≤ 230 cal/g <u>and MCHFR ≥ 95/95 DNBR limit⁽¹⁾</u>
	N/A	Maximum peak fuel temperature	≤ melting temperature
	N/A	Fuel pellet cladding fragmentation and dispersal	Coolable geometry maintained
	N/A	Fuel rod ballooning	Coolable geometry maintained

Notes:

~~(1) From Standard Review Plan 4.2, Appendix B.~~~~(2)~~ ⁽¹⁾ Minimum critical heat flux ratio (MCHFR) is used instead of minimum DNBR, as described in Section 4.4.2.

Table 15.0-4: Acceptance Criteria Specific to Loss of Coolant Accidents⁽¹⁾.

Parameter ⁽¹⁾	Acceptance Criteria
Maximum fuel element cladding temperature	$\leq 2200^{\circ}\text{F}$ MCHFR > 1.29 and collapsed water level above the top of active fuel.
Maximum total oxidation of cladding	$\leq 17\%$ of the total cladding thickness before oxidation MCHFR > 1.29 and collapsed water level above the top of active fuel.
Maximum total hydrogen generated from chemical reaction of cladding with water or steam	$\leq 1\%$ of the hypothetical amount if all of the metal in the cladding cylinders surrounding the fuel, excluding surrounding the plenum volume, were to react. MCHFR > 1.29 and collapsed water level above the top of active fuel.
Core geometry	Any calculated changes to core geometry are such that the core remains amenable to cooling. MCHFR > 1.29 and collapsed water level above the top of active fuel. ⁽²⁾
Long term cooling	Following initial successful operation of the ECCS, the calculated core temperature is maintained at an acceptably low value and decay heat is removed for the extended time required by the long-lived radioactivity remaining in the core.

Notes:

(1) From 10 CFR 50.46.

(2) Note that this is typically met by demonstrating that the ~~first three~~ acceptance criteria are met, and in addition, by assuring that the fuel deformation due to combined LOCA and seismic loads is specifically addressed.

a full or partial turbine bypass valve opening covers and bounds the steam increases due to an inadvertent opening of a MSSV. The analysis of a limiting increase in steam flow is presented in Section 15.1.3.

15.1.5 Steam Piping Failures Inside and Outside of Containment

15.1.5.1 Identification of Causes and Accident Description

A steam line break (SLB) event for the NuScale Power Plant design could range from a small break to a double ended rupture of the main steam line. This event could occur inside or outside of the containment vessel (CNV). A spectrum of SLB locations with varied core and plant conditions are analyzed to determine the scenarios with the most severe results.

A SLB inside the CNV would increase the pressure inside containment, reaching the high containment pressure analytical limit. The high containment pressure signal trips the core, isolates the CNV, and actuates DHRS on the intact SG train. The break flow would decrease due to SG depressurization until dryout due to feedwater isolation. The containment pressure is sensitive to any SLB size, so the protection system detects the break sooner than a comparable break outside of containment. A spectrum of breaks inside containment is evaluated to ensure that containment pressure is acceptable. The peak containment pressure remains below the design limit for all postulated events, as shown in Section 6.2. Aside from containment pressure, the plant conditions for a SLB inside containment are bounded by the analysis presented in this section for a SLB outside of containment.

A SLB outside the CNV would cause an increase in steam flow event that could either cause a low SG pressure signal or a high core power trip due to the reactor power response from the decreased RCS temperature. The break flow would be stopped by the closure of the MSIV and depressurization of the steam system piping. The largest steam line break outside containment could occur from a double-ended rupture of the portion of the main steam line located outside of the CNV. However, a double-ended rupture results in a low steam pressure signal that occurs earlier in the transient than for a small break outside of containment. A small SLB outside of containment is the most limiting type of SLB because it provides the longest event progression before detection by the protection system. A smaller break can result in a significant delay in detection time relative to a larger break, producing more limiting primary and secondary pressures, MCHFR, and integral mass and energy release.

RAI 15-9S1

A significant steam piping failure is not expected to occur during the life of the plant, so the event is classified as an accident. A smaller or secondary piping failure has a higher probability of occurrence, so the SLB accident will be evaluated against the ~~more~~ conservative 95/95 MCHFR AOO acceptance criteria.

15.1.5.2 Sequence of Events and Systems Operation

There are separate small SLB cases outside of containment that are limiting with respect to primary pressure, MCHFR, and radiological consequences. The SLB event is a

- Average RCS temperature biased high - The higher temperature corresponds to a higher coefficient of expansion. This exacerbates the REA-induced core pressure pulse and inlet flow slow-down, minimizing MCHFR.
- RCS flow biased low - The lower core flow minimizes MCHFR.
- Fuel and gap conductivities are maximized - Maximizing the conductivities increases the energy flow into the coolant, which maximizes the inlet flow slow-down.

The results from the thermal hydraulic evaluation are used as input to the subchannel analysis to determine if the MCHFR design limit is met for this event. Other key inputs and assumptions used in the subchannel analysis are provided in Reference 15.4-1. The results of the subchannel analysis and adiabatic fuel energy calculation determine if there is any potential fuel damage resulting from an REA. The REA event-specific methodology is provided in Reference 15.4-2.

15.4.8.3.3 Regulatory Criteria for NuScale

Reference 15.4-2 discusses the various REA regulatory acceptance criteria and how they apply to the NuScale design. A summary of these acceptance criteria are provided in this section.

Fuel Cladding Failure

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- For zero power conditions, the high temperature cladding failure threshold is expressed in cladding differential pressure. The peak radial average fuel enthalpy must be below ~~the 100 cal/g associated with the maximum peak rod differential pressure of $\Delta P \geq 4.5$ MPa.~~
- For intermediate and full power conditions, fuel cladding failure is presumed if local heat flux exceeds the critical heat flux (CHF) thermal design limit.

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- The PCMI failure ~~threshold limit~~ is a change in radial average fuel enthalpy ~~greater than 75 cal/g, based on~~ the corrosion-dependent limit depicted in ~~Figure 5-2 of Reference 15.0-11~~ ~~Figure B-1 of NUREG-0800 SRP 4.2.~~
- If fuel temperature anywhere in the pellet exceeds incipient fuel melting conditions, then fuel cladding failure is presumed.

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Core Coolability

- Peak radial average fuel enthalpy will remain below ~~230~~100 cal/g.
- No fuel melt will occur.
- MCHFR \geq 95/95 DNBR limit.

RCS Pressure

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Table 15.4-24: Spectrum of Rod Ejection Accidents (15.4.8) - Limiting Analysis Results

Acceptance Criteria	Limit	Analysis Value
Peak radial average fuel enthalpy at zero power	100 cal/g ⌘	34.6 cal/g ⌘
Change in radial average fuel enthalpy	75 cal/g ⌘	28.7 cal/g ⌘
Peak radial average fuel enthalpy	230 100 cal/g ⌘	84 cal/g ⌘
Maximum RCS pressure	2310 psia	2076 psia
Peak fuel temperature	4791 °F	2162 °F
MCHFR	1.284	2.477