U.S. NUCLEAR REGULATORY COMMISSION

AUDIT SUMMARY FOR THE REGULATORY AUDIT

OF NUSCALE POWER, LLC, FINAL SAFETY ANALYSIS REPORT CHAPTER 15,

"TRANSIENT AND ACCIDENT ANALYSES"

1 BACKGROUND

NuScale Power, LLC (NuScale) submitted by a letter dated December 31, 2016, to the U.S. Nuclear Regulatory Commission (NRC) a Final Safety Analysis Report (FSAR) for its Design Certification Application (DCA) of the NuScale design (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17013A229). The NRC staff initiated this DCA review on March 27, 2017. Revision 0 of the FSAR was superseded by the March 15, 2018, submittal of Revision 1 (ADAMS Accession No. ML18086A090), followed by the October 30, 2018, submittal of Revision 2 (ADAMS Accession No. ML18311A006).

The NRC staff initiated a regulatory audit related to the review of Chapter 15, "Transient and Accident Analyses," on June 15, 2017 (ADAMS Accession No. ML17157B592). The audit was conducted according to Office of New Reactors (NRO) Office Instruction NRO-REG-108, "Regulatory Audits" (ADAMS Accession No. ML081910260). The audit was performed primarily via the NuScale electronic reading room (ERR), with limited in-person audits at NuScale's Office in Rockville, Maryland, and through several telephone audit discussions with the applicant.

The purposes of this audit were to: (1) gain a better understanding or confirm the NRC staff's understanding of the detailed calculations, analyses, and bases underlying the NuScale FSAR Chapter 15, (2) confirm information in the DCA and evaluate its conformance with the NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP) or technical guidance, and (3) identify any information needed on the docket to support the basis of a reasonable assurance finding. Additional background is available in the audit plan associated with this audit summary (ADAMS Accession No. ML17157B592).

2 REGULATORY AUDIT BASIS

Title 10 of the *Code of Federal Regulations* (CFR), Section 52.47(a)(4), states that a DC application must contain an FSAR that includes:

> *An analysis and evaluation of structures, systems, and components with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents.*

The staff determined the need for an audit to confirm the basis for the safety conclusions made in the applicant's Chapter 15 transient and accident analyses.

3 AUDIT LOCATION AND DATES

The audit was conducted from NRC headquarters via NuScale's ERR, by telephone, and at NuScale's Rockville office.

Dates:

Phase 1: June 15, 2017, through September 15, 2017

Phase 2: September 18, 2017, through March 20, 2018

Locations:

U.S. Nuclear Regulatory Commission Headquarters
(via NuScale's ERR)
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5 APPLICANT PARTICIPANTS

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6 AUDIT DOCUMENTS

The staff audited the following documents:

FSAR Section 15.0.6

- EC-0000-4820, Revision 1, "Overcooling Return to Power Analysis"
- EC-0000-4848, Revision 0, "ECCS Overcooling Reactivity Coping Analysis"

FSAR Section 15.1.1

- EC-0000-2017, Revision 0, "Decrease in Feedwater Temperature Analysis," and attached ECN-0000-4863, "Decrease in Feedwater Temperature Impact Analysis"

FSAR Section 15.1.2

- EC-0000-2016, Revision 0, "Increase in Feedwater Flow Analysis," and attached ECN-0000-4864, "Increase in Feedwater Flow Impact Analysis"
- EC-0000-3077, Revision 1, "Subchannel Analysis of an Increase in Feedwater Flow"

FSAR Section 15.1.5

- EC-0000-2714, Revision 0, "Steam System Piping Failure Analysis," and attached ECN-0000-4957, Revision 0, "Steam Piping Failure Analysis – Impact Analysis"

FSAR Section 15.1.6

- EC-0000-2878, Revision 0, "Loss of Containment Vacuum/Containment Flooding," and attached ECN-0000-4910, Revision 0, "Loss of Containment Vacuum/Containment Flooding – Impact Analysis"; and ECN-0000-5020, Revision 0, "Additional Plots and Event Detail for CNV Flooding and Loss of Vacuum Events"

FSAR Sections 15.2.1-15.2.3

- EC-0000-1997, Revision 0, "Loss of External Load, Turbine Trip, Loss of Condenser Vacuum," and attached ECN-0000-4523, "Additional Summary Tables for Loss of External Load, Turbine Trip, Loss of Condenser Vacuum Analysis"; ECN-0000-4862, "Loss of External Load, Turbine Trip, Loss of Condenser Vacuum – Impact Analysis"; and ECN-0000-4950, "Loss of External Load, Turbine Trip, Loss of Condenser Vacuum – Impact Analysis Timetable Results for Subchannel Analysis"

FSAR Sections 15.2.4

- EC-0000-2995, Revision 1, “Closure of Main Steam Isolation Valve Transient Analysis”

FSAR Section 15.2.6

- EC-0000-2908, Revision 1, “Loss of Non-Emergency AC Power to the Station Auxiliaries Analysis”

FSAR Section 15.2.7

- EC-0000-1998, Revision 0, “Loss of Normal Feedwater Flow Analysis”

FSAR Section 15.2.8

- EC-0000-2250, Revision 0, “Feedwater Piping Failure Analysis”
- EC-0000-2357, Revision 1, “Subchannel Analysis of Feedwater System Pipe Breaks”

FSAR Section 15.4.1

- EC-0000-2910, Revision 1, “Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition,” and attached ECN-0000-5268, “Additional UCRWS Plots”
- EC-0000-3080, Revision 1, “Subchannel Analysis of an Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power,” and attached ECN-0000-5726, “NSP4 Update – Subchannel Analysis of UCRW from Subcritical or Low Power”

FSAR Section 15.4.2

- EC-0000-1999, Revision 2, “Uncontrolled Control Rod Assembly Withdrawal at Power Transient Analysis”
- EC-0000-2899, Revision 2, “Subchannel Analysis of Uncontrolled Rod Assembly Withdrawal at Power,” and attached ECN-0000-5262, “Change of time scales in Figure 5-1”; ECN-0000-5635, “Additional Subchannel Cases for Uncontrolled Control Bank Withdrawal at Power”; and ECN-0000-5714, “NSP4 Update – Subchannel Analysis of Uncontrolled Control Rod Assembly Withdrawal at Power”

FSAR Section 15.4.3

- EC-A021-1977, Revision 1, “Control Rod Assembly Drop Analysis,” and attached ECN-A021-5136, “Supplementary Single Rod Drop Summary Results”
- EC-A021-2405, Revision 1, “Control Rod Misalignment Analysis”
- EC-0000-2139, Revision 1, “Control Rod Misoperation Transient Analysis,” and attached ECN-0000-5067, “Linear Power Range Negative Rate Trip Definition”

- EC-0000-2897, Revision 1, “Subchannel Analysis of Control Rod Misoperation,” and attached ECN-0000-5263, “Change of time scales in Figure 5-1”; and ECN-0000-5715, “NSP4 Update – Subchannel Analysis of Control Rod Misoperation”

- EC-0000-4309, Revision 1, “Subchannel Analysis of a Control Rod Misalignment,” and attached ECN-0000-5716, “NSP4 Update – Subchannel Analysis of a Control Rod Misalignment”

FSAR Section 15.4.7

- EC-0000-2646, “Subchannel Analysis of Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position,” and attached ECN-0000-5727, “NSP4 Update – Subchannel Analysis of Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position”

FSAR Section 15.6.2

- EC-0000-2786, Revision 2, “Failure of Small Lines Carrying Primary Coolant Outside Containment”

FSAR Section 15.6.3

- EC-0000-1735, Revision 0, “NuScale Steam Generator Tube Failure”

FSAR Section 15.6.5

- EC-0000-4888, Revision 0, “NuScale LOCA Evaluation Model Supporting Calculations”

- EC-0000-2749, Revision 0, “NuScale LOCA Spectrum Calculations”

- EC-A010-1507, Revision 3, “System Transient Model Input Parameters Calculation”

- ER-0000-2486, Revision 4, “Safety Analysis Analytical Limits Report”

- ER-0000-3921, Revision 0, “Long Term Core Cooling Methodology Report”

- EC-A010-4270, Revision 1, “Long Term Cooling Analysis”

- EC-0000-4576, Revision 0, “Reduction of Collapsed Primary Water Level in LTC due to Loss of Inventory”

- EC-T080-4505, Revision 0, “NRELAP5 Assessment of Spurious RVV Opening Test NIST-1 HP-19a”

- EC-T080-4506, Revision 0, “NRELAP5 Assessment of Spurious RVV Opening Test NIST-1 HP-19b”

FSAR Section 15.6.6

- EC-0000-4684, Revision 0, “Spurious Opening of an RPV Valve”
- EC-0000-3221, Revision 0, “Identification and Classification of Deterministic Design Basis Events for the NuScale Power SMR”
- EC-0000-4888, Revision 0, “NuScale LOCA Evaluation Model Supporting Calculations”
- EC-A010-1782, Revision 0, “NuScale NRELAP Module Basemodel”

Other

- EC-0000-2347, Revision 3, “Steady-State Subchannel Analysis”
- ER-0000-2486, Revision 5, “Safety Analysis Analytical Limits Report”
- ER-0000-3641, Revision 0, “Transient Analysis Interface with Fuel Performance, with Methodology for Bounding Values”
- ER-A010-2009, Revision 4, “NuScale Reactor Module Design Parameters”

7 DESCRIPTION OF AUDIT ACTIVITIES AND SUMMARY OF OBSERVATIONS

The NRC staff audited information related to multiple aspects of FSAR Chapter 15. In addition, the staff held multiple audit discussions with the applicant via telephone. Summaries relevant to the sections audited are provided below.

FSAR Section 15.0.6

The NRC staff audited engineering calculation (EC) EC-0000-4820, Revision 1, “Overcooling Return to Power Analysis,” and EC-0000-4848, Revision 0, “ECCS Overcooling Reactivity Coping Analysis.” EC-0000-4820 uses a version of the non-loss-of-coolant accident (non-LOCA) NRELAP5 model to first perform a decay heat removal system (DHRS) cooldown, which maximizes the return to power for a range of decay heat values. The staff examined the input conditions and compared the NRELAP5 model to that given in the Non-LOCA topical report (TR), TR-0516-49416, “Non-Loss-of-Coolant Accident Analysis Methodology.” The staff examined the reactor coolant system (RCS) parameters associated with the cooldown to confirm the expected system response. The second part of EC-0000-4820, uses NRELAP5 with the hot channel model from the LOCA NRELAP5 deck and actuates the emergency core cooling system (ECCS) at and around the maximum return to power. The staff examined results for the expected behavior, especially with regard to minimum critical heat flux ratio (MCHFR). Regarding the audit of EC-0000-4820, the staff noted the following:

- A conservative value of 40 degrees Fahrenheit (°F) was assumed for the reactor building pool temperature.
- A series of decay heat levels were assumed to determine if there was any impact on final maximum return to power.

- The heat removal capability of the two DHRS trains was increased by 30 percent to maximize the cooldown.

- The applicant ran the analysis at hot zero power (HZP) conditions so an initial RCS inventory was achieved such that the natural circulation is preserved throughout the cooldown. In Request for Additional Information (RAI) 9508, the NRC staff asked if initial conditions exist such that reactor pressure vessel (RPV) level could drop below the riser, thereby losing single phase natural circulation, and if under these conditions, whether fuel cladding temperatures remain acceptably low such that the specified acceptable fuel design limits are preserved consistent with NuScale Principal Design Criterion 34.

- Since the analysis was performed at HZP, the applicant used a moderator temperature coefficient (MTC) corresponding to HZP. In RAI 9488, the NRC staff asked if using the HZP MTC was conservative, as an at-power MTC would be more negative assuming natural circulation could still be preserved.

- The ECCS actuation cases uses the hot channel model from the NRELAP5 LOCA evaluation model (EM). The NRC staff noted that axial power distribution is bottom-peaked, which is not consistent with a stuck rod axial power shape. In RAI 9487, the staff asked the basis for using a bottom-peaked axial power shape in the hot channel.

- Consistent with the NRELAP5 LOCA model, the return to power model uses the Hench-Levy critical heat flux (CHF) correlation, which was assessed against the KATHY NuFuel HTP2 CHF database for high flow and the Stern CHF database for low flow conditions to determine the 95/95 design limit. In RAI 9536, the staff requested the applicant provide additional information as to how the Hench-Levy 95/95 correlation limit was determined and its applicability to the limiting return to power MCHFR conditions.

The NRC staff also generated other more minor RAIs. The staff has requested the applicant to document the basis for the Hench-Levy CHF correlation 95/95 correlation limit in a TR consistent with other CHF correlation limit determinations. The review and potential approval of the correlation limit will be documented in the TR safety evaluation and referenced in the return to power analyses.

FSAR Section 15.1.1

The NRC staff audited EC-0000-2017, Revision 0, “Decrease in Feedwater Temperature Analysis,” and the attached engineering change notice (ECN) ECN-0000-4863, “Decrease in Feedwater Temperature Impact Analysis,” to examine key inputs and assumptions for the analysis and to confirm the FSAR presented the most limiting cases. The calculation note largely confirmed the information presented in the FSAR. In addition, the staff noted the following:

- The minimum feedwater temperature of 100 °F is based on the saturation temperature of the condenser (the coldest point in the secondary system) at vacuum conditions.

- The increase in the high-power analytical limit to 125 percent power is based on [[

]]. The staff issued RAI 9483, Question 15.01.01-2, requesting a docketed basis for the assumed 5 percent increase in the high-power analytical limit.

- To maximize the cooldown, the reactor pool is assumed to be the minimum temperature of 40 °F and is treated as nearly infinite by increasing the pipe component area, and DHRS heat transfer is increased by 30 percent. RAI 9483, Question 15.01.01-8, requested the applicant to add these assumptions to the FSAR.

- Pressurizer heater operation is allowed because it increases RCS pressure, which is limiting for MCHFR.

- The steam outlet boundary is specified as a constant pressure boundary, contrary to a statement in FSAR Section 15.1.1, Revision 1. The staff issued RAI 9483, Question 15.01.01-10, in part, for clarification of this inconsistency.

- Minimum decay heat (a multiplier of 0.8 and no actinide contribution) is assumed to minimize RCS heating.

- The minimum end-of-cycle (EOC) hot full power volume average fuel temperature of 740 °F cannot be achieved through increasing gap conductivity, so the temperature used is 820.5 °F. Sensitivity results show that MCHFR is relatively insensitive to the fuel temperature.

- To find the limiting decrease in feedwater temperature scenario, the applicant examined 10 initial condition cases, each with seven temperature decrease rates.

FSAR Section 15.1.2

The NRC staff audited EC-0000-2016, Revision 0, “Increase in Feedwater Flow Analysis,” and the attached ECN-0000-4864, “Increase in Feedwater Flow Impact Analysis,” to confirm the use of conservative assumptions and inputs in the analysis and to confirm that the FSAR presents the most limiting case. The documents largely confirmed the information presented in the FSAR. In addition, the staff noted the following:

- The reactor pool temperature is assumed to be 40 °F, and DHRS heat transfer is decreased by 30 percent for the limiting case. RAI 9483, Question 15.01.01-8, requested the applicant to add these assumptions to the FSAR.

- A minimum feedwater temperature of 290 °F is assumed to exacerbate the cooldown.

- Much of the other modeling is similar to that described in EC-0000-2017.

- The applicant examined several amounts of steam flow increase, up to 100 percent, over 0.1 seconds.

- The applicant investigated scenarios of potential steam generator (SG) overfill by examining two initializations: (1) a high initial SG level, achieved by biasing feedwater temperature low, SG pressure high, and SG heat transfer low; and (2) a low initial SG level, achieved by biasing feedwater temperature high, SG pressure low, and SG heat transfer high. A feedwater isolation valve (FWIV) is assumed as the single active failure.

> The maximum transient SG level is 82 percent, and the calculations showed that DHRS capability was not degraded. However, based on the applicant's sensitivity study results, the staff noted that the maximum transient SG level is not directly translatable to initial SG level. Furthermore, peak pressures for both SG trains are identical despite one train having a failed FWIV. RAI 9483, Question 15.01.01-7, asked the applicant to further justify that the SGs do not overfill and impede DHRS capability.

The NRC staff also audited EC-0000-3077, Revision 1, "Subchannel Analysis of an Increase in Feedwater Flow," to ensure the limiting NRELAP5 cases were passed to the subchannel analysis.

FSAR Section 15.1.5

The NRC staff audited calculation files associated with the applicant's main steam line break analysis to support the review of FSAR Tier 2, Section 15.1.5, "Steam Piping Failures Inside and Outside of Containment."

The NRC staff examined the initial conditions, boundary conditions, assumptions, and input to confirm the applicant's method of biasing input parameters in the conservative direction. The staff's audit included confirming the expected outcomes of certain parameter biasing to ensure the results were reasonable. The staff generated several RAIs to request the applicant to provide additional information regarding selected assumptions and justifications for using certain input parameters. The staff also examined the audit calculation files to confirm proper application of the single failure criterion presented in FSAR Tier 2, Section 15.1.5.

The NRC staff examined the audited documents to confirm that the applicant adequately followed the non-LOCA methodology referenced in this section of the FSAR. In addition, the staff confirmed that the subchannel analysis methodology was followed for calculating the MCHFR associated with this Chapter 15 event.

The NRC staff examined the audited documents to confirm the break size/location sensitivity calculations. The staff requested to meet with the applicant regarding an apparent issue with its break size sensitivity analysis. The applicant indicated in the FSAR that a 7.5 percent split break is limiting for the radiological release case; however, the staff noted that sensitivity calculations indicated that a 2 percent split break was even more limiting from the perspective of mass release. The staff intends to examine additional documentation during a follow-on audit and is tracking this issue via RAI 9478, Question 15.01.05-2.

FSAR Section 15.1.6

The NRC staff audited EC-0000-2878, Revision 0, to gain a better understanding of and to confirm various information in FSAR Section 15.1.6. The staff confirmed the statement in the FSAR that a loss of containment vacuum (without flooding) is bounded by a containment flooding event. The applicant's loss of containment vacuum event calculation showed that the RPV heat loss increases by a small amount, and the NuScale Power Module (NPM) remains at power. The RCS parameters and core power reach a quasi-steady state and differ little from pre-event conditions. This behavior is like that of a containment flooding event, except the containment flooding events are more limiting with respect to figures of merit.

In addition, the NRC staff audited the applicant's sensitivity studies for the containment flooding event and confirmed that the initial conditions maximize the consequences of the event. This event is different from most other Chapter 15 events since it results in the NPM approaching a new steady state without a reactor trip. The audited material supports the conclusions regarding the NPM meeting acceptance criteria for this event.

FSAR Sections 15.2.1-15.2.3

The NRC staff audited EC-0000-1997, Revision 0, "Loss of External Load, Turbine Trip, Loss of Condenser Vacuum," and attached ECN-0000-4523, "Additional summary tables for loss of external load, turbine trip, loss of condenser vacuum analysis"; ECN-0000-4862, "Loss of External Load, Turbine Trip, Loss of Condenser Vacuum – Impact Analysis"; and ECN-0000-4950, "Loss of External Load, Turbine Trip, Loss of Condenser Vacuum – Impact Analysis Timetable Results for Subchannel Analysis." The staff examined the assumptions, inputs, and results in these documents, which largely confirmed the information presented in the FSAR. The staff also noted the following:

- The turbine stops valve (TSV) closure time is assumed to be 0.001 seconds (s), more limiting than the typical value of 0.10 s in SRP Section 15.2.1-15.2.5, since the TSV does not have an associated design specification and is not safety-related.

- The turbine control valve closure time is assumed to be 0.15 s, consistent with SRP Section 15.2.1-15.2.5.

- The only difference between the turbine trip and loss of external load events is the valve closure time, which is faster for the turbine trip (as discussed above).

- The loss of condenser vacuum event is similar to the turbine trip event except the condensate pumps trip at transient initiation, resulting in a loss of feedwater. Loss of feedwater is the most challenging condition for RCS pressure, as nearly all events with a loss of feedwater result in lifting of the reactor safety valve (RSV).

- In general, control systems are only allowed to respond normally if they increase the severity of the transient. For example, the model maintains pressurizer heater output constant with no spray actuation since normal operation would reduce RCS pressure.

- ECN-0000-4862 indicated that the heat transfer option originally used to model heat transfer between the DHRS and the pool for these events was different from that used in the non-LOCA TR. The staff issued RAI 9407, Question 15.02.01-6, requesting clarification of what heat transfer option is used in the FSAR analysis and any differences relative to the TR methodology.

- The applicant performed sensitivity studies for the turbine trip, loss of external load, and loss of condenser vacuum events that varied the initial SG pressure, RCS average temperature (T_{avg}), pressurizer pressure, and SG heat transfer individually and in combination. The studies also investigated the effects of single failures and loss of alternating current (AC) power.

- The sensitivity studies did not investigate the effects of biased-low SG heat transfer, which the staff felt may be more of a challenge for RCS pressure or MCHFR, so the staff issued RAI 9407, Question 15.02.01-5, requesting justification.

- The loss of external load results was very similar to those for turbine trip, rendering the slower valve closure time almost negligible.

- The limiting event in terms of RCS pressure and MCHFR is a loss of condenser vacuum. For RCS pressure, an assumed failure of a FWIV to close with a loss of AC power is marginally more challenging. The limiting event for SG pressure is a turbine trip, and an assumed failure of a FWIV to close is slightly more challenging. The staff issued RAI 9407, Question 15.02.01-3, requesting the applicant to document these assumptions that produced the limiting results in the FSAR.

- The timing in the sequence of events tables in ECN-0000-4862 differed from what was presented in the FSAR, so the staff issued RAI 9407, Question 15.02.01-7.

- The RSV lifts for the loss of condenser vacuum limiting MCHFR case, which is not clear from the FSAR. The staff issued RAI 9407, Question 15.02.01-4, requesting clarification in the FSAR.

- The applicant examined a limiting RCS pressure transient assuming RPV pressure was not limited by means other than the transient response. The peak RCS pressure was under 2,200 pounds per square inch absolute (psia), which is well below the anticipated operational occurrence limit of 2,310 psia.

FSAR Section 15.2.4

The NRC staff audited EC-0000-2995, Revision 1, “Closure of Main Steam Isolation Valve Transient Analysis,” to examine the inputs and assumptions and to confirm the most limiting case was identified and presented in the FSAR. The calculation notes largely confirmed the information presented in the FSAR. In addition, the staff noted the following:

- Failure of a FWIV to close is not considered because the applicant stated it has little effect on the transient. After main steam isolation valve (MSIV) closure, the SG pressure is generally high enough to prevent feedwater inflow, and peak SG pressure is insensitive to FWIV closure since it occurs after the backup FWIV is closed.

- The document states that the downstream subchannel analyses consider both high and low initial pressurizer pressure because the bias direction dictates whether the high pressurizer trip signal or the high SG pressure trip signal occurs first.

- Based on the applicant’s sensitivity studies, for RCS pressure cases that do not result in the RSV lifting, RCS pressure increases with decreasing initial T_{avg}. For MCHFR cases, NRELAP5 MCHFR decreases with decreasing initial pressurizer pressure. Since these bias directions do not appear consistent with those in the FSAR, the staff issued RAI 9407, Question 15.02.01-11.

FSAR Section 15.2.6

The staff audited EC-0000-2908, Revision 1, “Loss of Non-Emergency AC Power to the Station Auxiliaries Analysis,” to examine key inputs and assumptions to the analysis and to confirm the FSAR presented the most limiting cases. The calculation notes largely confirmed the information presented in the FSAR. In addition, the staff noted the following:

- In reality, when AC power is lost, there will be a slight time delay before the feedwater pumps trip and a pump coast-down. However, for conservatism, the model assumes an immediate decrease of feedwater to zero flow coincident with the loss of AC power.
- Control rods are assumed to be in manual mode.
- The applicant performed sensitivity studies in which it varied the initial SG pressure, T_{avg}, pressurizer pressure, and SG heat transfer individually and in combination. The studies also investigated the effects of single failures.
- In general, the peak RCS pressure was relatively insensitive to initial condition biasing, varying up to 10 psia depending on the bias directions. This suggests that a loss of AC power is the dominant factor in peak RCS pressure, and any differences due to initial condition biasing are largely mitigated by RSV actuation.
- For the limiting RCS pressure case, the sensitivity studies showed that the initial T_{avg} was biased low, and the initial SG pressure was biased high, which was inconsistent with what was reported in the FSAR. The staff issued RAI 9416, Question 15.02.06-1, for clarification.
- The applicant used the limiting RCS pressure case to examine the effects of top- and bottom-skewed power shapes, direct moderator heating (2.5 percent and 5 percent), and end-of-cycle fuel temperature on the figures of merit for the limiting RCS pressure case. The effects on the NRELAP5-calculated RCS pressure, SG pressure, and MCHFR were negligible.

FSAR Section 15.2.7

The NRC staff audited calculation files associated with the applicant’s loss of normal feedwater flow analysis to support the review of FSAR Tier 2, Section 15.2.7, “Loss of Normal Feedwater Flow.”

The NRC staff examined the initial conditions, boundary conditions, assumptions, and input to confirm the applicant’s method of biasing input parameters in the conservative direction. The staff’s audit included confirming the expected outcomes of certain parameter biasing to ensure that the results were reasonable. The staff generated an RAI to request the applicant to provide additional justification in the FSAR regarding the biasing of various parameters. As a result, the applicant submitted a response with additional sensitivities to provide further justification for the input parameter biasing. The staff also examined the audit calculation files to confirm proper application of the single failure criterion presented in FSAR Tier 2, Section 15.2.7.

The NRC staff examined the audited documents to confirm that the applicant adequately followed the non-LOCA methodology referenced in this section of the FSAR. In addition, the staff confirmed that the applicant followed the subchannel analysis methodology for calculating the MCHFR associated with this Chapter 15 event.

FSAR Section 15.2.8

The NRC staff audited EC-0000-2250, Revision 0, “Feedwater Piping Failure Analysis,” and EC-0000-2357, Revision 1, “Subchannel Analysis of Feedwater System Pipe Breaks,” to support the review of FSAR Tier 2, Section 15.2.8, “Feedwater System Pipe Breaks Inside and Outside of Containment.”

The NRC staff examined the initial conditions, boundary conditions, assumptions, and input to confirm the applicant’s method of biasing input parameters in the conservative direction. The staff’s audit included confirming the expected outcomes of certain parameter biasing to ensure that the results were reasonable. The staff generated RAI No. 8744 to request the applicant to provide additional information in the FSAR regarding the biasing of the SG heat transfer uncertainty. As a result, the applicant submitted a response with further clarification for the chosen input parameter bias. The staff also examined the audit calculation files to confirm proper application of the single failure criterion presented in FSAR Tier 2, Section 15.2.8. As a result, the staff issued several RAIs to request the applicant to justify crediting valves that are not safety-related to mitigate a design basis accident.

The NRC staff examined the audited documents to confirm that the applicant adequately followed the non-LOCA methodology referenced in this section of the FSAR. In addition, the staff confirmed that the applicant followed the subchannel analysis methodology for calculating the MCHFR associated with this Chapter 15 event.

The NRC staff examined the audited documents to confirm the break size/location sensitivity calculations. In addition, the staff audited the calculation files to examine what occurs when ECCS actuates after the containment vessel has been filled with diluted water from the feedwater line break. The staff issued RAIs to request information regarding boron transport during recirculation after a feedwater line break.

FSAR Section 15.4.1

The NRC staff audited EC-0000-2910, Revision 1, “Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition,” and attached ECN-0000-5268, “Additional UCRWS Plots,” to examine the inputs and assumptions to the transient analysis for this event. The calculation notes largely confirmed the information presented in the FSAR. In addition, the staff made the following observations:

- An earlier, preliminary version of the document showed that S3K, a multi-dimensional neutronics code, produced results similar to NRELAP, with NRELAP5 predicting a higher power peak. The applicant concluded that multi-dimensional neutronic modeling is not needed for the transient analysis provided conservative differential rod worth and reactivity feedback are used.

- NRELAP5 does not calculate counts per second for the source range overpower analytical limit. Instead, the analysis assumes [[ ]]. Given this assumption, the staff issued RAI 9507, Question 15.04.01-4, requesting demonstration that the analytical limit will be adequately protected.

- The module heatup flow rate is [[ ]], and the initial RCS flow rate for the limiting MCHFR and fuel temperature cases is [[ ]]. The initial RCS flow rate for the limiting RCS pressure case is [[ ]].

- Feedwater temperature is assumed to be 100 °F, corresponding to hot shutdown.

- The analysis examined different initial power levels ranging from 1 Watt to 15 percent power. For cases initiating from the source range, the maximum power case occurred when the high source range count rate and the source range power rate analytical limits were reached simultaneously. Similarly, for cases initiating from the intermediate range, simultaneous trips on the high power and high intermediate range power rate are limiting for maximum power.

- A minimum initial reactor power is not limiting for the NPM because the power rate trip prevents a significant increase in power.

- The sensitivity studies examined the effect of initial core inlet temperature at either 420 °F (the minimum temperature for criticality) or [[ ]] and SG pressure at either [[ ]].

- A relatively low reactivity insertion rate is limiting for RCS pressure because it avoids the power-related trips; however, pressure barely exceeds the trip setpoint. For this reason, initial pressurizer pressure was not investigated as a sensitivity for the maximum RCS pressure case.

- The limiting MCHFR case in NRELAP assumed a relatively low reactivity insertion rate, but a higher reactivity insertion rate was limiting in the downstream subchannel analysis. Although the core inlet temperature is maximized with a low reactivity insertion rate, power ascension is slow, and RCS flow rises to prevent MCHFR from dropping significantly.

- Because core inlet flow increases with increasing power, using the initial RCS flow rate in VIPRE is conservative.

In addition, the NRC staff audited EC-0000-3080, Revision 1, “Subchannel Analysis of an Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power,” and attached ECN-0000-5726, “NSP4 Update – Subchannel Analysis of UCRW from Subcritical or Low Power,” to examine the inputs and assumptions to the subchannel analysis and to confirm the appropriate cases from the transient analysis underwent a subchannel analysis. The staff noted the following:

- The axial power shape used for the limiting event is a 25 percent power-derived shape. Since the event initiates at 1 megawatt, not 25 percent power, the staff issued RAI 9507, Question 15.04.01-6, requesting confirmation that the axial power shape used is conservative.

- In EC-0000-3080, Revision 1, three of 10 cases passed to the subchannel analysis resulted in an MCHFR of 10 (the maximum value) or less. Of them, the mass flow rate of one was outside the range of the NSP2 CHF correlation. For this case, the applicant

noted that the void fraction and quality was zero, and the maximum outer cladding temperature on a partial length of the hot rod was about 2 °F above saturation temperature at 1850 psia. The conclusion was that no phase change had occurred, and therefore this case did not represent MCHFR.

- In ECN-0000-5726, the applicant re-performed the subchannel analyses using the NSP4 CHF correlation. Each of the three cases above now showed an MCHFR of 10.

FSAR Section 15.4.2

The NRC staff audited EC-0000-1999, Revision 2, "Uncontrolled Control Rod Assembly Withdrawal at Power Transient Analysis," to examine the inputs and assumptions to the transient analysis for this event. The calculation notes largely confirmed the information presented in the FSAR. In addition, the staff made the following observations:

- The reactor pool temperature is assumed to be 100 °F (normal operating temperature).

- While higher power, RCS pressure, and RCS temperature are in general associated with lower MCHFR, biasing each of these parameters high may not be conservative since it could cause an earlier reactor trip. In general, the limiting MCHFR cases occur when the high reactor power, high hot leg temperature, and high pressurizer pressure trips occur nearly simultaneously. Based on this, it appears that the applicant's sensitivity studies adequately determined the limiting MCHFR cases.

- The most limiting NRELAP5 MCHFR case initiated from 102 percent power; however, several limiting NRELAP5 MCHFR cases were passed to the downstream subchannel analysis, which showed that the limiting MCHFR case initiated from 75 percent power.

- The reactor power for the limiting MCHFR case initiating from 75 percent power did not reach the high reactor power analytical limit of 120 percent power, contrary to statements in FSAR Chapter 15. Therefore, the staff requested that the applicant correct the statements in the FSAR in RAI 9509, Question 15.04.02-3.

In addition, the NRC staff audited EC-0000-2899, Revision 2, "Subchannel Analysis of Uncontrolled Rod Assembly Withdrawal at Power," and attached ECN-0000-5714, "NSP4 Update – Subchannel Analysis of Uncontrolled Control Rod Assembly Withdrawal at Power"; ECN-0000-5262, "Change of time scales in Figure 5-1"; and ECN-0000-5635, "Additional Subchannel Cases for Uncontrolled Control Bank Withdrawal at Power," to examine the inputs and assumptions to the subchannel analysis and to confirm the appropriate cases from the transient analysis underwent a subchannel analysis. The staff noted that the most limiting MCHFR cases from NRELAP5 were passed on to the subchannel analysis and that the most limiting VIPRE MCHFR is consistent with what is presented in FSAR Revision 1.

FSAR Section 15.4.3

The staff audited the neutronic analysis for the control rod assembly drop event in EC-A021-1977, Revision 1, "Control Rod Assembly Drop Analysis," and attached ECN-A021-5136, "Supplementary Single Rod Drop Summary Results," and noted the following:

- The applicant performed steady-state neutronic analyses of an equilibrium cycle using Studsvik Scandpower Core Management Software, Version 5 (CMS5) to identify the worst-case CRA drop. The worst drop is defined as the one that yields the largest change in the enthalpy rise hot channel factor (FΔH). The applicant considered various times in life (beginning, middle, and end of cycle [BOC, MOC, and EOC]), power levels (1, 25, 50, 75, and 100 percent), control rod insertion schemes (i.e., all rods out versus rods at the power-dependent insertion limits), dropped rods and dropped subgroups, initial axial power shapes (nominal, limiting positive axial offset, and limiting negative axial offset), and the effects of xenon redistribution.

- A single CRA drop was more limiting than a subgroup drop, and full insertion of the CRA is limiting.

- CRA S-08 is the limiting dropped CRA for all initial power levels.

- In general, the change in FΔH increased with decreasing initial power level.

- The largest post-drop FΔH occurs for MOC conditions for CRA S-08. The largest change in FΔH occurs for MOC for 100 percent power but EOC for lower power levels.

The NRC staff audited EC-0000-2139, Revision 1, "Control Rod Misoperation Transient Analysis," and attached ECN-0000-5067, "Linear Power Range Negative Rate Trip Definition," to examine the inputs and assumptions for these events. The calculation notes largely confirmed the information presented in the FSAR. In addition, the staff made the following observations:

- The single rod withdrawal (SRW) analysis accounts for core asymmetry by multiplying the reactor power by the ratio of the transient radial power factor to the pre-withdrawal radial power factor, thereby calculating the lowest power seen by excore detectors.

- The rod drops parameters (rod worths, FΔH, change in FΔH) are consistent with those in EC-A021-1977.

- The module control system assumptions for the SRW and rod drop events appear to be conservative, including disabling the rod control system for the SRW and allowing pressurizer heater operation during the rod drop since higher pressure is generally conservative for MCHFR.

- The applicant investigated both BOC and EOC conditions for the control rod drop analysis and found that EOC was limiting.

- The power-dependent MTC assumed for the SRW is as follows:

Power (%)	MTC (percent millirho (pcm) per °F)
[[	
	]]

- Similar to the uncontrolled bank withdrawal at power event, limiting conditions are achieved for the SRW by synchronizing the trips.

- The applicant examined two SRW cases initiating from [[ ]]. Both [[ ]. Therefore, it appeared that more limiting cases initiating from [[ ]] may be possible by biasing parameters to further delay the [[ ]] and by considering other reactivity insertion rates.

- Therefore, the staff requested additional justification that the limiting single CRA withdrawal case has been identified in RAI 9512, Question 15.04.03-5.

In addition, the staff audited EC-0000-2897, Revision 1, "Subchannel Analysis of Control Rod Misoperation," and attached ECN-0000-5715, "NSP4 Update – Subchannel Analysis of Control Rod Misoperation"; and ECN-0000-5263, "Change of time scales in Figure 5-1," to examine the inputs and assumptions to the subchannel analysis and to confirm the appropriate cases from the transient analysis underwent a subchannel analysis. The staff noted the following:

- The appropriate cases from NRELAP5 were passed on to the subchannel analysis.

- For SRW, the applicant examined several combinations of axial offset and power shapes, control bank position, xenon conditions, and time in life. The axial power shape used in the subchannel analysis corresponds to the conditions that produced the limiting augmentation factor.

- For the rod drop event, the power shape used is the core-average power shape from document EC-0000-2347.

- The staff confirmed that the limiting cases are consistent with those documented in the FSAR.

For the control rod misalignment event, the staff audited the nuclear design calculations in EC-A021-2405, Revision 1, "Control Rod Misalignment Analysis," and noted the following:

- ODI-16-1030 sought verification that the low power hold point is at 25 percent power because the subchannel analysis does not postulate a misalignment event that occurs below this hold point. The staff issued RAI 9512, Question 15.04.03-3, for clarification regarding the ODI.

- The largest augmentation factor occurs for the configuration with the highest $F\Delta H$, which results for a partial misalignment of a Group 1 rod assuming EOC conditions, a positive axial offset, and 25 percent initial power.

Furthermore, the NRC staff audited EC-0000-4309, Revision 1, "Subchannel Analysis of a Control Rod Misalignment," 12/8/2016, and attached ECN-0000-5716, "NSP4 Update – Subchannel Analysis of a Control Rod Misalignment." The staff noted the following:

- The applied system pressure bias was [[ ]], and the applied core inlet temperature bias was [[ ]], because these biases did not appear consistent with

those in FSAR Tier 2, Table 15.0-6, "Module Initial Conditions Ranges for Design Basis Event Evaluation," the staff issued RAI 9512, Question 15.04.03-2.

- The axial power shape and system boundary conditions for the control rod misalignment subchannel analysis correspond to [[ ]] power, while the limiting radial augmentation factor is for [[ ]]. Therefore, the staff requested justification for the axial power shape and boundary conditions chosen in RAI 9512, Question 15.04.03-3.

FSAR Section 15.4.7

The NRC staff audited EC-0000-2646, "Subchannel Analysis of Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position," and attached ECN-0000-5727, "NSP4 Update – Subchannel Analysis of Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position," to confirm that the analysis used conservative inputs to obtain the most limiting result and to confirm the consistency of the information with that presented in the FSAR. In particular, the staff noted that the applied system pressure bias was [[ ]], and the applied core inlet temperature bias was [[ ]], because these biases did not appear consistent with those in FSAR Tier 2, Table 15.0-6, the staff issued RAI 9504, Question 15.04.07-3.

Other

The NRC staff audited EC-0000-2347, Revision 3, "Steady-State Subchannel Analysis," and noted that the axial power shape analysis determines the limiting shape to use to calculate MCHFR and examined over 4,000 different power shapes from SIMULATE for power levels greater than or equal to five percent. Ultimately, the axial power shape for each power level that results in the lowest MCHFR is selected for use. The document stated that the limiting axial power shape for all power levels is middle-peaked because the magnitude of peaking is higher than for a top- or bottom-peaked shape, and it produces a limiting MCHFR.

The NRC staff also audited ER-0000-2486, Revision 5, "Safety Analysis Analytical Limits Report." The staff examined the ranges of initial conditions in this document, which are stated to account for both normal control system dead-bands and system/sensor measurement uncertainties, and noted they are consistent with FSAR Tier 2, Table 15.0-6. The staff also confirmed that the actuation analytical limits and safety-related valve analytical characteristics in this document are consistent with those in FSAR Chapter 15.

In the NRC staff's audit of ER-0000-3641, Revision 0, "Transient Analysis Interface with Fuel Performance, with Methodology for Bounding Values," the staff noted the following:

- The volume-weighted core average fuel temperature is not directly input. Rather, fuel rod gap conductance is tuned until the desired temperature is reached. From several calculation notes (e.g., EC-0000-2139), the low- and high-bounding gap conductance values are [[ ]], respectively.

- The decay heat standard and multipliers are consistent with the description in the FSAR and the non-LOCA TR. A figure in the document showed that the minimum and maximum decay heat conditions enveloped a best-estimate ORIGEN equilibrium cycle decay heat calculation.

FSAR Section 15.6.2

The staff audited EC-0000-2786 and confirmed that the calculation follows the methodology described in the non-LOCA TR. Furthermore, the staff notes that the calculation supports the conclusions about the acceptance criteria in the FSAR.

FSAR Section 15.6.3

The NRC staff audited NuScale calculation EC-0000-1735 to assess the bounding case results and to better understand the responses, in particular, related to the SG unique design. The staff issued RAI Nos. 8794 and 9368 to question the ability of the MSIVs to close under potentially water-logged conditions. Remaining issues include staff's review of modeling changes and assumptions related to the revised base NRELAP5 model and any changes related to code version change from version 1.3 to 1.4 (RAI No. 9325). Other issues include use of non-safety FWIVs and MSIVs (RAI No. 9237) related to NUREG-0138, "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum from Director, NRR to NRR Staff." The staff requested in RAI No. 9420 for the applicant to rerun the steam generator tube failure cases assuming these valves fail to close and confirm that radiological releases remain under guidelines of 10 CFR Part 100, "Reactor Site Criteria."

FSAR Section 15.6.5

The NRC staff is reviewing the long-term cooling (LTC) technical report, TR-0916-51299, "Long-Term Cooling Methodology," in conjunction with the LOCA TR, TR-0516-49422, "Loss-of-Coolant Accident Evaluation Model." The staff audited the LOCA EM calculation EC-0000-4888, as well as EC-0000-3921, EC-A010-4270, EC-0000-4576, EC-0000-2749, and EC-T080-4505. The LTC technical report appears to preclude the potential for boron dilution without providing adequate justification. The staff issued RAI Nos. 8744 and 8930 to address de-borated water from secondary pipe breaks or LOCAs during LTC, where steam condensate from reactor vent valves or feedwater line break flow could form slugs of fresh water that could potentially dilute the core and reduce shutdown margin. The staff issued RAI Nos. 9470, 9471, and 9479 to address aspects of the LTC model features and assumptions. The LTC cases were biased for either maximum or minimum cooldown rate. Staff also examined NIST-1 HP-19a and HP-19b assessment results.

The NRC staff had difficulty understanding differences in LOCA modeling versus that used for LTC modeling. The applicant simplified reduce code mass errors, and it did not appear that the changes would produce conservative results (RAI Nos. 9470 and 9516). LTC modeling changes and assumptions related to the revised base NRELAP5 model and any changes related to RAI No. 9325 also need to be understood.

FSAR Section 15.6.6

The NRC staff audited EC-0000-4684, Revision 0, "Spurious Opening of an RPV Valve" to support the review of FSAR Tier 2, Section 15.6.6, "Inadvertent Operation of Emergency Core Cooling System." The staff also audited supporting calculations EC-0000-3221, Revision 0, "Identification and Classification of Deterministic Design Basis Events for the NuScale Power SMR", EC-0000-4888, Revision 0, "NuScale LOCA Evaluation Model Supporting Calculations," and EC-A010-1782, Revision 0, "NuScale NRELAP Module Base model."

The NRC staff examined the initial conditions, boundary conditions, assumptions, and input to confirm the applicant's method of biasing input parameters in the conservative direction. The staff's audit included confirming the expected outcomes of certain parameter biasing to ensure that the results were reasonable. The staff also examined the audit calculation files to confirm proper application of the single failure criterion presented in FSAR Tier 2, Section 15.6.6. As a result, the staff issued several RAIs to request the applicant to justify the analysis not assuming the single failure of an inadvertent actuation block (IAB) valve. As a result of the responses and audit, the staff determined that the IAB is an active component. On December 17, 2018, NuScale submitted a letter to the Commission (ADAMS Accession No. ML18351A145) addressing the IAB single failure issue.

Due to inadequacies associated with the description of the methodology in this section of the FSAR, the NRC staff issued RAI No. 9373 to request the applicant to adequately define the methodology in the FSAR. The staff received the RAI response, which added an appendix to the LOCA methodology TR detailing the methodology used for this event. The staff is currently reviewing the methodology and will further audit supporting FSAR documents in a follow-on audit to confirm the methodology is followed for this analysis.

The NRC staff examined the audited documents to ensure that the applicant considered the worst limiting inadvertent opening of an RPV valve; however, part of this evaluation is dependent upon the single failure assumption of an IAB, which is being resolved as mentioned above.

8 EXIT BRIEFING

The NRC staff conducted an audit exit meeting at the NuScale Rockville office on March 20, 2018. During the exit meeting, the staff reiterated the purpose of the audit and discussed the audit activities and outcome. The staff also indicated that RAIs would be forthcoming to obtain information necessary for the NRC staff to complete its review of the NuScale DCA. The staff's RAIs are publicly available in ADAMS.