U.S. NUCLEAR REGULATORY COMMISSION
AUDIT SUMMARY FOR THE REGULATORY AUDIT OF NUSCALE POWER, LLC DESIGN CERTIFICATION APPLICATION CHAPTER 4, "REACTOR"; CHAPTER 5, "REACTOR COOLANT AND CONNECTING SYSTEMS"; AND CHAPTER 9, "AUXILIARY SYSTEMS"

I. BACKGROUND

NuScale Power, LLC (NuScale) submitted by letter dated December 31, 2016, to the U.S. Nuclear Regulatory Commission (NRC), a Final Safety Analysis Report for its Design Certification (DC) application of the NuScale design (Reference 1). The NRC staff started its detailed technical review of NuScale's DC application on March 27, 2017.

The purpose of this regulatory audit of NuScale's subject line chapters was to: (1) gain a better understanding of the NuScale design; (2) verify information; (3) identify information that will require docketing to support the basis of the licensing or regulatory decision; and (4) review related documentation and non-docketed information to evaluated conformance with regulatory guidance. Additional background in available in the audit plan associated with this audit summary (Reference 2).

II. REGULATORY AUDIT BASES

Title 10 of the *Code of Federal Regulations* (CFR), Section 52.47(a)(3)(i) states:

> *A DC application must contain a final safety analysis report (FSAR) that includes a description of principle design criteria for the facility.*

This regulatory audit is based on the following regulations:

- 10 CFR 52.47, "Contents of applications; technical information in final safety analysis report."

- General Design Criteria 2 (GDC), "Design bases for protection against natural phenomena," of Appendix A to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," as it relates to structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

- GDC 4, “Environmental and dynamic effects design bases,” as it relates to the structures, systems, and components important to safety that shall be designed to accommodate the effects of and to compatible with the environmental conditions during normal plant operation as well as during postulated accidents.

- GDC 5, “Sharing of structures, systems, and components,” which requires that structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

- GDC 10, “Reactor Design,” which requires that reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

- GDC 11, “Reactor inherent protection,” which requires that the reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

- GDC 12, “Suppression of reactor power oscillations,” which requires that the reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

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

- GDC 14, “Reactor coolant pressure boundary,” which requires the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

- GDC 15, “Reactor coolant system design,” which requires that the reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

- GDC 19, “Control room,” which requires that control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal

conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

- GDC 23, "Protection system failure modes," which requires that the protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

- GDC 25, "Protection system requirements for reactivity control malfunctions," which requires that the protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

- GDC 26, "Reactivity control system redundancy and capability," which requires that two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

- GDC 27, "Combined reactivity control systems capability," which requires that 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.

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

- GDC 29, “Protection Against Anticipated Operational Occurrences,” as it relates to protecting system against anticipated operational occurrences such that the design of the protection and reactor control systems should assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

- GDC 31, “Fracture prevention of reactor coolant pressure boundary,” which requires that the reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

- GDC 33, “Reactor coolant makeup,” which requires that a system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

- GDC 34, “Residual heat removal,” which requires that a system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

- GDC 44, “Cooling Water,” which requires that a system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

- GDC 45, “Inspection of cooling water system,” which requires that the cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.

- GDC 46, “Testing of cooling water system,” which requires that the cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak tight integrity of its components, (2) the

operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

- GDC 54, "Piping systems penetrating containment," which requires that piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits. The cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.

- GDC 57, "Closed system isolation valves," which requires that each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

- GDC 60, "Control of releases of radioactive materials to the environment," which requires that the nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

- GDC 61," Fuel storage and handling and radioactivity control," which requires that the fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

- GDC 62, "Prevention of criticality in fuel storage and handling," ," which requires that criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

- 10 CFR 50.34(f)(2)(xxvi), with respect to the provisions for a leakage detection and

control program to minimize the leakage from those portions of the CVCS outside of the containment that contain or may contain radioactive material following an accident.

- 10 CFR 50.34(f)(2)(x) and 10 CFR 50.34(f)(2)(xi) require that reactor coolant system (RCS) SRVs meet Three Mile Island (TMI) Action Plan Items II.D.1 and II.D.3 of NUREG-0737.

- 10 CFR 52.47(a)(8) provides the requirement for design certification reviews to comply with the technically relevant portions of the TMI requirements in 10 CFR 50.34(f).

- 10 CFR 20.146, which requires the applicant to describe how facility design will minimize, to the extent practical, contamination of the facility and the environment, minimize generation of radioactive waste, and facilitate eventual decommissioning.

- 10 CFR 50.68, as it relates to preventing a criticality accident and to mitigating the radiological consequences of a criticality accident.

- 10 CFR 50.63, as it relates to the decay heat removal system (DHRS) providing adequate residual heat removal with the loss of all AC power.

- 10 CFR 50.46, 10 CFR 50.34, and 10 CFR 50.67, as they relate to the cooling performance analysis of the ECCS using an acceptable evaluation model and establishing acceptance criteria for light-water nuclear power reactor ECCSs.

- 10 CFR 52.47(b)(1), which requires that a DC application contain the proposed inspections, tests, analyses, and acceptance criteria (ITAAC) that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations.

III. AUDIT LOCATION AND DATES

The audit was conducted from the NRC headquarters via NuScale's electronic reading room.

Date: May 3, 2017 – November 29, 2017

Location: NRC Headquarters
Two White Flint North
11545 Rockville Pike
Rockville, MD 20852-2738

IV. AUDIT TEAM MEMBERS

Jeff Schmidt (NRO, Audit Lead)
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Jason Thompson (RES)
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John Budzynski (NRO)
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Bruce Bavol (NRO, Project Manager)

V. APPLICANT AND INDUSTRY STAFF PARTICIPANTS

NuScale Power, LLC

Darrell Gardner, Manager, Licensing

VI. DOCUMENTS AUDITED

FSAR Section 4.3:

- EC-A021-2404, Rev. 0, “Axial Offset and Axial Power Shape Analysis,” 11/23/2016.
- EC-0000-5323, Rev. 0, “Multi Module Flux Calculation,” 6/19/2017.
- EC-A021-4504, Rev. 1, “Shutdown Margin and Long Term Shutdown Capability,” 9/16/2016.

FSAR Section 5.4.3:

- EC-B030-2121, DHRS Thermal Hydraulic Analysis
- EQ-B030-2258, ASME Design Specification for DHRS Actuation Valves
- EQ-B030-3055, ASME Design Specification for DHRS Condenser

FSAR Section 9.1.1:

- EC-B160-3283, Rev. 0, “Spent Fuel Storage Rack Criticality Analysis,” 6/25/2015, Approved 2/4/2016.
- ECN-B160-4594, Rev. 0, “Update to remove proprietary markings on pitch dimension,” 9/23/2016.
- ECN-B160-4804, Rev. 0, “Reactivity Adjustment for 40 deg. F Water Temperature,” 11/3/2016.

- EC-B160-3283, Rev. 1, "Spent Fuel Storage Rack Criticality Analysis," 6/23/2017, Approved 6/26/2017.

- Criticality calculation output files:

  - sc-ns-bp-005
  - sc-ns-bp-005N
  - sc-ns-sens-001 through -003
  - sc-ns-sens-015
  - sc-ns-sens-020 through -023
  - sc-ns-sens-025
  - sc-ns-sens-029
  - sc-ns-sens-030
  - sc-ns-sens-049
  - sc-ns-sens-069
  - sc-ns-sens-070
  - sc-ns-sens-075
  - sc-ns-sens-077
  - sc-ns-sens-078
  - sc-ns-sens-080 through -085
  - sc-ns-sens-092
  - sc-ns-acc-002 through -005
  - sc-ns-acc-002N
  - sc-ns-acc-009
  - sc-ns-dm-1 through -10
  - sc-ns-bp-013A
  - sc-ns-bp-013barslah
  - sc-ns-bp-013Clah
  - sc-ns-bp-013Dlah
  - sc-ns-bp-013Elah
  - sc-ns-bp-013steelreflah
  - sc-ns-bp-013unboratedbarslah

VII. DESCRIPTION OF AUDIT ACTIVITIES AND SUMMARY OF OBSERVATIONS

<u>FSAR Section 4.3:</u>

Axial Offset and Power Dependent Insertion Limits

The NRC staff examined engineering calculation EC-A021-2404, Rev. 0, "Axial Offset and Axial Power Shape Analysis." During this examination NRC staff noted the purpose of the calculation and identified key inputs and their sources. In particular, NRC staff noted:

- The purpose of this calculation is to (1) confirm the suitability of an axial offset (AO) window and (2) identify a range of axial core-average distributions (shapes) that may occur at and within the AO windows. The axial shapes may then be used to determine bounding axial shapes for use in downstream analyses. The bounding axial shapes are determined by user-specific criteria and are not part of this calculation. This calculation also identifies the maximum F_{DH} and the axial shape corresponding to the highest relative nodal power (maximum F_Z).

- This calculation is a steady-state analysis of a representative equilibrium cycle and reflects best estimate reactor conditions, with consideration for [

  ]

- The software SIMULATE5 is used in this calculation.

- It is assumed that the neutron detectors in the in-core instrumentation (ICI) system will get a reliable signal at 25 percent of rated thermal power and above.

- Equilibrium cycle 8 is considered at beginning of cycle (BOC), middle of cycle (MOC), and end of cycle (EOC), represented by the first, middle and last burnup steps of the cycle at 0, 6, and 12.052 GWd/MT, respectively.

- A range of thermal-hydraulic conditions that include minimum, best-estimate, and maximum flow are considered at 1 percent, 5 percent, 25 percent, 50 percent, 75 percent, and 100 percent rated thermal power.
- The power dependent insertion limits (PDILs) for the control rods are [

  ]
- The axial offset (AO) window is taken from ER-0000-2486, Rev. 3, "Safety Analysis Analytical Limits Report."
- Calculations are performed to develop on the order of [ ] axial shapes representative of the entire axial offset window. Calculations are performed with the following perturbations:
  - 1, 5, 25, 50, 75, and 100 percent power.
  - BOC, MOC, and EOC.
  - [

    ]
  - Nominal and perturbed AO windows.
  - Minimum flow, best estimate (BE) flow, and maximum flow for thermal hydraulic conditions.
  - Average coolant temperature (associated with flow rate) and perturbed by[ ].
- Of the approximately [ ] axial shapes developed, [

  ] provided in from ER-0000-2486, Rev. 3, "Safety Analysis Analytical Limits Report."

EC-A021-2404, Rev. 0, "Axial Offset and Axial Power Shape Analysis," does not describe the process used to develop the AO window or PDILs. Accordingly, NRC staff requested that NuScale describe the process used to develop the AO window and PDILs. NuScale provided clarifying information which the NRC staff has summarized as follows. [

]

- [

]

- [

]

- The ability to maintain shutdown margin is more challenging for deeper PDILs because there is less available worth; however, [

]

NuScale summarized the process used to set the PDILs and AO window using the flowchart replicated by NRC staff in Figure 1.

[

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[

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[

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[

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[ ]

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[ ]

Listed Analyses:

- ER-0000-2486 (R4) Safety Analysis Analytical Limits Report
- EC-A021-2404 (R0) Axial Offset and Axial Power Shape Analysis
- EC-0000-2347 (R2) Steady-State Subchannel Analysis
- EC-0000-4888 (R0) NuScale LOCA Evaluation Model Supporting Calculations
- EC-0000-2749 (R0) Loss of Coolant Accident Resulting from a Spectrum of Postulated Piping Breaks

Shutdown Margin and Long Term Shutdown Capability

The NRC staff examined engineering calculation EC-A021-4504, Rev. 1, "Shutdown Margin and Long Term Shutdown Capability." During this examination, NRC staff noted the purpose of the calculation, identified key inputs and their sources, the calculation procedure, and key outputs from the calculation. In particular, NRC staff noted:

- The purpose of this calculation is to demonstrate the shutdown margin (SDM) and long term shutdown (LTSD) capability of the NuScale Reactor Core (RXC) under normal operation conditions.
- The calculation uses the PDIL limits provided in Table 1.

Table 1. Power Dependent Insertion Limits

Power (Percent Rated)	**PDIL (steps withdrawn)**		**Analyzed Rod Position (steps withdrawn)**	
	Reg. Bank 1	**Reg. Bank 2**	**Reg. Bank 1**	**Reg. Bank 2**
100	170.0	170.0	164.0	164.0

- The calculation uses best-estimate steady-state thermal-hydraulic conditions, provided in Table 2.

Table 2. Thermal-Hydraulic Conditions as a Function of Power

Power (Percent Rated)	Average Temperature (°F)	Primary Flow (Percent Rated)	Primary Flow (kg/s)
0.39	[ ]	[ ]	[ ]
0.58	[ ]	[ ]	[ ]
1.0	[ ]	[ ]	[ ]
5.0	[ ]	[ ]	[ ]
10.0	[ ]	[ ]	[ ]
15.0	543.3	47.7	280.2
100.0	543.3	100.0	587.0

- The redistribution worth calculation uses bounding axial offset limits of [ ], taken from ER-0000-2486, Rev. 2, "Safety Analyses Analytical Limits Report."

- The calculation accounts for 12 percent uncertainty in CRA bank worth.

- The minimum shutdown margin for hot full power (HFP) of 2041 pcm, corresponding to an eigenvalue of 0.98, is taken from ER-0000-2486, Rev. 2, "Safety Analyses Analytical Limits Report."

- The long term shutdown calculations assume an equilibrium temperature of [ ], defined by the cold shutdown criterion, which is taken from ER-0000-2486, Rev. 2, "Safety Analyses Analytical Limits Report."

- The SDM calculation is a balance between positive and negative reactivity contributions, where the balance must result in sufficient margin to satisfy the required SDM. Negative reactivity is provided by available worth from the CRAs. Positive reactivity considers:

    - The highest worth rod stuck out (WRSO) in the fully withdrawn position.

    - The effect of changes to moderator and fuel temperature that are collectively addressed in the power defect.

    - Worth penalties that account for axial flux redistribution.

    - CRA insertion to the PDIL.

- CRA worth uncertainty.

- The SDM calculation is performed using the following procedure:
  - [ ]
  - [ ]
  - [ ]
  - [ ]
  - [ ]
  - [ ]
  - [ ]
  - [ ]
  - [ ]

- The LTSD calculation is a balance between positive and negative reactivity contributions, where the balance must result in sufficient margin for the reactor to remain subcritical. Negative reactivity is provided by the available worth from CRAs. Positive reactivity considers:
  - The effect of changes to moderator and fuel temperature that are collectively addressed in the power defect.
  - Additional reduction in coolant temperature to long term equilibrium conditions.
  - Worth penalties that account for axial flux redistribution.
  - CRA insertion to the PDIL.
  - CRA worth uncertainty.

- Xenon decay.

- The general input setup and description of code options used to perform the LTSD calculation is described in the following bullets:
  - [ ]
  - [ ]
  - [ ]
  - [ ]
  - [ ]
  - [ ]
  - [ ]
  - [ ]
  - [ ]

- The SDM calculation utilized the statepoints provided in Table 3.

Table 3. SDM Configurations

#	Configuration	k_{eff}, BOC	k_{eff}, MOC	k_{eff}, EOC
1	[ ]	[ ]	[ ]	[ ]
2	[ ]	[ ]	[ ]	[ ]
3	[ ]	[ ]	[ ]	[ ]
4	[ ]	[ ]	[ ]	[ ]

5	[ ]	[ ]	[ ]	[ ]
6	[ ]	[ ]	[ ]	[ ]
7	[ ]	[ ]	[ ]	[ ]
8	[ ]	[ ]	[ ]	[ ]
9	[ ]	[ ]	[ ]	[ ]

- The LTSD calculation utilized the statepoint provided in Table 4.

Table 4. LTSD Configurations

#	Configuration	k_{eff}, BOC	k_{eff}, MOC	k_{eff}, EOC
1	[ ]	[ ]	[ ]	[ ]
2	[ ]	[ ]	[ ]	[ ]
3	[ ]	[ ]	[ ]	[ ]
4	[ ]	[ ]	[ ]	[ ]
5	[ ]	[ ]	[ ]	[ ]
6	[ ]	[ ]	[ ]	[ ]
7	[ ]	[ ]	[ ]	[ ]
8	[ ]	[ ]	[ ]	[ ]
9	[ ]	[ ]	[ ]	[ ]
10	[ ]	[ ]	[ ]	[ ]

- NRC staff requested clarification regarding the differences between configures 1, 3, and 7 in Table 3 (configurations 1, 3, and 8 in Table 4). NuScale provided the following clarification:
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- The SDM calculation produced net SDM margins of 4085 pcm, 3434 pcm, and 2696 pcm at BOC, MOC, and EOC, respectively. The SDM calculation also produced WRSO worths of 4766, 4922, and 5249 pcm at BOC, MOC, and EOC, respectively.

- The LTSD calculation produced net margins to critical of 5099, 3133, and 955 pcm at BOC, MOC, and EOC, respectively.

Multi-Module Considerations

The NRC staff examined engineering calculation EC-0000-5323, Rev. 0, “Multi Module Flux Calculation.” During this examination, NRC staff noted the purpose of the calculation and identified key inputs and their sources. In particular, NRC staff noted:

- The purpose of this calculation is to evaluate the multi module flux consequence to the NuScale Power module reactor.

- The analysis was performed using a conservative hand calculation. The applicant cited Section 14.5 of Martin (Reference 3) for the method, and several inputs, used in the hand calculation.

- The applicant took no credit for the presence of boron in the pool water. NRC staff notes that this is very conservative.

- The results of the applicant's analysis showed that the total flux at the containment nuclear vessel that has been attenuated from an adjacent nuclear power module (that is operating at 100 percent power) is negligible (i.e. the contribution to the flux observed by the ex-core detectors is several orders of magnitude less than the uncertainty associated with a startup neutron count rate).

Modeling of Axial Zoned Control Rod Assemblies

The NRC staff requested that NuScale clarify the approach used to model the axial zoned control rod assemblies (CRAs). NuScale provided the following explanation:

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Modeling Considerations for In-Core Instrumentation

The NRC staff requested that NuScale clarify whether the impact of in-core instrumentation on reactivity and power distributions was considered. NuScale provided the following explanation:

- In-core instruments can be modeled explicitly in CASMO5. However, detailed design of in-core instrumentation for the NuScale reactor is not yet finalized, so the CASMO5 and SIMULATE5 NuScale reactor core models used for benchmarks in TR-0616-48793 do not contain in-core instrumentation information and all instrumentation tubes are therefore modeled as water-filled. Because the in-core instruments reside in only 12 of the 37 assemblies in the reactor core, and the instrument tube is only 1 of 289 pin cells in an assembly, the impact to reactivity and power distributions of modeling in-core instrumentation is not considered significant. In addition, performing benchmarks of the NuScale reactor core with in-core instrumentation modeled has negligible impact on the ability of the codes to accurately predict reactivity and power distributions.

Modeling of Radial Reflector

The NRC staff requested the NuScale clarify the neutronic modeling of the radial reflector region. NuScale provided the following explanation:

- The modeling of the radial reflector in the NuScale RXC is split into multiple segments matching the square array and pitch of the fuel assemblies. The radial reflector segments are identified according to their position around the core and the relative material composition of the segment, as shown in Figure 3-5 of the TR-0616-48793, "Nuclear Analysis Codes and Methods Qualification Topical Report," (provided below as Figure 2 for completeness). [

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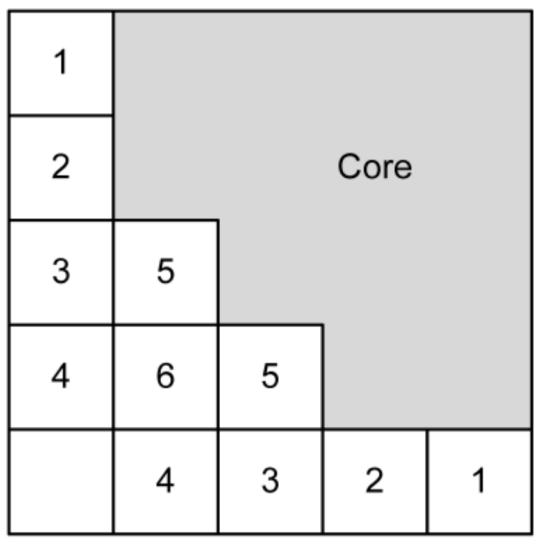

Figure 2. SIMULATE5 Radial Reflector Region Diagram

Modeling of Core Flow in Neutronic Analysis

The NRC staff inquired regarding the core flow used in the SIMULATE5 model. The applicant stated the following:

- The nominal total primary flow in the core design base model is [ ] (100 percent flow in SIMULATE5 inputs). Of the total primary flow, [

  ] This is any upward flow passing outside the assemblies that is not heated. This leaves [ ] percent of the total primary flow as

the active flow through the assemblies (or total core flow). Additional bypass flow through the guide tubes is specified [
] After accounting for both bypass flow paths, the remaining flow through heated channels is [ ] percent of the total primary flow.

- The flow types and rates used as input to and taken from output of the SIMULATE5 base model (Output Summary section) are provided in the Table 5 to aid in illustrating the discussion above.

Table 5. Flow Distribution in SIMULATE5 Analyses

Flow Type	**Percent of Primary Flow**	**Flow Rate (kg/s)**	**SIM5 Output (T/hr)**	**Percent of Primary Flow (Calc from SIM5 Output)**
Primary Flow (includes all bypass mechanisms)	[ ]	[ ]	[ ]	[ ]
Reflector Bypass	[ ]	[ ]	[ ]	[ ]
Active Assembly Flow	[ ]	[ ]	[ ]	[ ]
Guide Tube Bypass	[ ]	[ ]	[ ]	[ ]
Core Flow	[ ]	[ ]	[ ]	[ ]

Middle-of-Cycle in Core Analysis and Rod Ejection Analysis

The NRC staff identified that the value used to represent middle of cycle was not the same for all analyses (e.g., EC-A021-2404, Rev. 0, "Axial Offset and Axial Power Shape Analysis,") utilizes a burnup of 6.0 GWd/MTU and the CRA ejection analysis described in FSAR Section 15.4.8 utilizes a burnup of 4.0 GWd/MTU. NRC staff inquired regarding different burnup values used to represent middle of cycle in the core analysis calculations and rod ejection analysis calculations. The applicant stated the following:

- The nuclear analysis calculations are performed at beginning, middle and end of cycle statepoints corresponding to the initial, midpoint, and final burnups of the nominal cycle. The critical boron concentration letdown monotonically decreased over the duration of the cycle, so the initial and final burnups bound the core reactivity. Additionally, core parameters that are associated with analytical limits used in the safety analysis are the most bounding for initial or final burnups. In general, analytical limits used in the safety analysis are the most bounding for initial or final burnups. In general, this means that bounding parameters that are developed for use downstream in the safety analysis occur at the beginning or the end of the cycle. However, middle of cycle evaluations are also performed to verify that limiting values for core parameters are captured, and 6.0 GWd/MTU represents a nominal cycle midpoint.

- Although the above explanation generally holds true for all nuclear analysis calculations, the rod ejection transient analysis does not exhibit the same predictable linearity or trends as the steady state analyses do when key input parameters are perturbed. The radial peaking factor $F_{\Delta H}$ has a local and global maximum at 4.0 GWd/MTU for the equilibrium core. The rod ejection accident subchannel analysis uses 3D power distributions from the nuclear analysis and the potential impact that the radial peaking initial condition has on the progression of the transient is not known a priori. The radial peaking is a potential contributor to the bounding case, so the initial condition is evaluated at the statepoint where the peaking is limiting rather than the nominal middle of cycle statepoint. In contrast, the other subchannel analyses use a generic radial shape that assumes the radial peaking factor analytical limit as the initial condition and therefore a nominal middle of cycle statepoint is sufficient.

FSAR Section 5.4.3:

The NRC staff audited the design documentation associated with the thermal hydraulic DHRS system performance analysis. The primary tool used by the applicant to model the DHRS performance is NRELAP5 version 1.2. The applicability of the code for performing analyses of this type was not the subject of this audit, and was reviewed by the staff as part of the NRC staff safety evaluation for the non-LOCA topical report.

The applicant summarized the assumptions used in the analysis; in reviewing how these assumptions could impact the analysis, the NRC staff issued requests for additional information (RAI) 8745 and 9082, as detailed below in the RAIs section. Of particular concern to the NRC staff was that the design layout assumed in the calculation had no provision for verification in the docketed application.

The thermal hydraulic design document details the model used for the DHRS analysis, which is a simplified version of that used in base design model. This simplified model omits the RCS side modeling and instead includes a set forcing functions in order to more easily create constant primary side conditions for the steady state calculation. Additionally, superfluous secondary side components are also omitted for simplicity. Based on the NRC staff's review of the model, the simplifications made appear to be appropriate for the narrow purpose of evaluating the thermal-hydraulic model of the DHRS.

Various modifications were made to the model in order to run sensitivity studies and add conservatisms to the model for evaluating the limiting system performance. These modifications include additions to the secondary side to model failed isolation valves, additional boundary conditions for fouling and tube plugging, and the addition of non-condensable gases to the DHRS, among other changes. These modifications were used in some of the transient runs discussed below.

In order to setup the model, steady state cases were run for a suite of conditions, including nominal full power, high and low inventory, and lower reactor power conditions. These runs are then used as the restart file for the various transients examined in the analyses. Overarching inferences could be made based on the cases run by the applicant, such as the system performance as a function of inventory, RCS temperature, and pool temperature. In general, these DHRS performance analyses demonstrate the DHRS is capable of removing an acceptable amount of heat for most operating parameters, save for very low RCS temperatures and certain loop inventory conditions. The DHRS removing less heat at lower RCS

temperatures is largely a non-issue, as lower RCS temperatures are indicative that decay heat levels are low and/or that the DHRS has already performed its function. DHRS performance as a function of loop inventory is an important consideration in reviewing the limiting operation conditions, as the analyses showed that single phase heat transfer through the DHRS loop (whether because of too little liquid leading to steam or too much fluid such that no volume is available for phase change) could lead to conditions that yield unacceptable heat transfer performance.

The NRC staff reviewed the full suite of analyses, paying particular attention to three cases referenced in the FSAR: the nominal DHRS performance with two trains, and limiting performance under both high and low inventory conditions. Both the high and low inventory cases are specified to have only a single train of DHRS operating. These cases demonstrated DHRS performance for non-specific limiting transients, and showed that for a single operating train, achieving 420F in 36h is possible. The NRC staff reviewed the relevant inputs and determined them to be appropriately limiting for this demonstration, but notes that transient-specific issues could arise in Chapter 15 cases that would require evaluation of the case-specific effects.

FSAR Section 9.1.1:

The NRC staff examined engineering calculation EC-B160-3283, Rev. 0, “Spent Fuel Storage Rack Criticality Analysis,” two related engineering changes, EC-B160-3283, Rev. 1, and several spent fuel pool (SFP) criticality analysis input files to ensure consistency with the docketed technical report TR-0816-49833-P, Rev. 0, “Fuel Storage Rack Analysis.”

In its audit of EC-B160-3283, Rev. 0, and related engineering changes ECN-B160-4594, Rev. 0, and ECN-B160-4804, Rev. 0, the NRC staff noted that the information in those documents is largely consistent with, and provides more detailed bases for, docketed information. In particular, the NRC staff noted the following:

- The tolerance in poison plate movement is consistent with the actual movement allowance in the design.
- The applicant specified materials in the criticality analysis input files in terms of atom density and used a standard formula to calculate atom density.
- The stainless steel material properties (weight percent and atom density of some components) were incorrect in EC-B160-3283, Rev. 0; however, the values used in the analysis input files were consistent with the outlined methodology.
- Though EC-B160-3283, Rev. 0, lists the density of aluminum as 2.702 g/cm², the calculations used to determine atom density appeared to use a density of 2.705 g/cm². The NRC staff performed a sensitivity calculation and estimated the effect of using the higher density to be small and conservative.
- EC-B160-3283, Rev. 0, contained several typographical errors in the atom density calculations for the neutron absorber plates, but the values used in the analysis input files are consistent with the outlined methodology.
- The applicant determined the credited neutron absorber plate boron-10 (B-10) areal density by examining the change in the effective neutron multiplication factor (k_{eff}) with

varying B-10 areal density. The applicant performed a similar determination of the credited SFP boron concentration.

- The original criticality analyses assumed a water temperature of 67°F rather than the limiting temperature of 40°F. To account for this, the applicant calculated the bias due to using the higher temperature and applied it to all of its criticality calculations.

- The original criticality analyses were performed assuming a rack-to-rack spacing that was too large. The applicant performed a sensitivity study on two cases to assess the impact of using the correct versus incorrect spacing and found that using the incorrect spacing is non-conservative by about 20 percent millirho (pcm) for the single-rack model and by about 9 pcm for the whole-pool model. Because these differences are small, and per NEI 12-16, "Guidance for Performing Criticality Analysis of Fuel Storage at Light Water Reactor Power Plants," Rev. 1, the applicant chose to ignore the error. However, the staff felt that the small impact of the error, in combination with errors in the input files discussed below, could grow to be significant. Therefore, the staff requested additional information in RAI 9157.

The applicant subsequently provided EC-B160-3283, Rev. 1, for audit, so the staff focused on the changes since Rev. 0. The major changes, aside from editorial changes, are described below:

- As a result of RAI 8760 (ML17116A014), which was issued prior to the audit, the applicant included a thorough discussion of rack components, fuel assembly components, and structural materials that were omitted in the model in EC-B160-3283, Rev. 1. The applicant performed additional criticality analyses to assess the effects of modeling the components and structural material it determined may have non-negligible effects. The models considered the presence of end fittings on the fuel assembly and steel plates and bars in the flux trap region of the storage rack fuel cell. The most limiting case reflective of the actual rack design includes steel bars in the flux trap region, which is logical since displacing moderator in the flux trap reduces the effectiveness of the neutron absorber panels. Based on the limiting case, the applicant determined that an eigenvalue bias due to structural material of 0.00222 ± 0.00040 should be applied to borated moderator calculations, and an eigenvalue bias of 0.00223 ± 0.00038 should be applied to unborated moderator calculations. This conclusion is consistent with the response to RAI 8760.

- Also as a result of RAI 8760, the applicant performed 21 additional analyses for the accident scenario of a dropped fuel assembly, each examining a different drop location in the fuel assembly elevator area. The applicant corrected the spacing error described above for these dropped fuel assembly analyses since it affected the proximity of the dropped fuel assembly to the storage rack and clarified that the dimension describing spacing to the pool wall is relative to the storage cells (not the rack baseplate, as suggested by Figure 1-2 in TR-0816-49833-P, Revision 0). EC-B160-3283, Rev. 1, lists the results for each of the dropped fuel assembly cases, and the applicant provided the limiting drop results in response to RAI 8760.

The staff noted that most aspects of the criticality analysis input files it audited are consistent with the methodology described in TR-0816-49833-P, including materials, geometry, tolerances, operating conditions, and boundary conditions. In addition, the input files beginning with "sc-ns-

bp-013" that examined the effect of structural material are consistent with the configurations described in EC-B160-3283, Rev. 1. However, the staff made the following observations:

- The original whole-pool input files for the nominal SFP (sc-ns-sens-084) and storage of damaged fuel assemblies in the SFP (sc-ns-sens-084, -085, and -092) were missing fuel assemblies from the periphery cells of each rack except for the corners. However, the updated versions of those files (sc-ns-dm-1 through -10) corrected the error.

- The original (sc-ns-sens-85) and revised (sc-ns-dm-3 and -8) input files intending to analyze five damaged fuel assemblies stored in the corner locations of the racks actually contain only four damaged fuel assemblies, which the staff noted in RAI 9157.

- Contrary to TR-0816-49833-P, the analysis for all damaged fuel assemblies (sc-ns-sens-92) was originally run with the single rack model, not the whole pool model; however, the updated analysis (sc-ns-dm-5 and -10) used the whole pool model.

- Some damaged fuel input files (original files sc-ns-sens-084 and sc-ns-sens-085 and updated files sc-ns-dm-1 through -4 and sc-ns-dm-6 through -9) had one non-damaged rod in damaged assemblies, contrary to the statement in TR-0816-49833-P that a damaged assembly is assumed to have 100 percent cladding failure. The NRC staff included this issue in RAI 9157.

- Several analysis input files (original files sc-ns-sens-81 through -87 and sc-ns-sens-92 and updated files sc-ns-dm-1 through -10) appeared to use the smallest storage tube thickness allowed by tolerances, not the nominal value. Because the applicant's tolerance analysis showed this has a small non-conservative effect, the staff included this issue in RAI 9157.

- The input files for all damaged fuel (original file sc-ns-sens-92 and updated files sc-ns-dm-5 and sc-ns-dm-10) use a cross-section processing card that assume the gap is still a void, even though it is flooded with water. This issue was included in RAI 9157.

- The input files for the five damaged fuel assemblies at the center of the rack (original file sc-ns-sens-084 and updated files sc-ns-dm-2 and -7) included an extra damaged fuel assembly in a corner location, which would affect reactivity in a conservative manner.

VIII. EXIT BRIEFING

The NRC staff conducted an audit closeout meeting on November 29, 2017. At the exit briefing the NRC staff reiterated the purpose of the audit and discussed their activities. The NRC staff stated that they had identified areas where additional information is being requested to support the review, and briefly discussed the scope of these information requests. References to the detailed questions are provided in Section IX of this audit summary.

IX. REQUESTS FOR ADDITIONAL INFORMATION RESULTING FROM AUDIT

The NRC staff issued 3 RAIs based on information observations made during the audit. These RAIs are available in Agencywide Documents Access and Management System (ADAMS). ADAMS Accession Nos. are provided in Table 6.

Table 6. RAIs Resulting from Audit

RAI Number	Reference
9157	ML17335A115
8745	ML17123A285
9082	ML17318A580

X. OPEN ITEMS AND PROPOSED CLOSURE PATHS

Not applicable.

XI. DEVIATIONS FROM THE AUDIT PLAN

The audit was originally scheduled to exit on October 31, 2017, but was extended to November 29, 2017, to accommodate the evaluation and discussion of additional documentation requested by NRC staff.

XII. REFERENCES

1. NuScale Power, LLC Submittal of the NuScale Standard Plant Design Certification Application (NRC Project No. 0769), December 31, 2016. (ADAMS Accession No. ML17013A229).

2. Supplement 1 to the Audit Plan for the Regulatory Audit of NuScale Power, LLC Design Certification Application Chapter 4, “Reactor”; Chapter 5, “Reactor Coolant and Connecting Systems”; and Chapter 9, “Auxiliary Systems,” May 9, 2017. (ADAMS Accession No. ML17124A339).

3. Martin, James, “Physics for Radiation Protection – A Handbook,” Second Edition, Wiley-VCH Verlag GmbH & Co. KGaA, Weinhiem, 2006.