U.S. NUCLEAR REGULATORY COMMISSION
AUDIT SUMMARY FOR THE REGULATORY AUDIT OF NUSCALE POWER, LLC DESIGN CERTIFICATION APPLICATION, DESIGN CONTROL DOCUMENT, TIER 2, CHAPTER 5, SECTION 5.4.1, "STEAM GENERATORS"

I. Background

By letter dated December 31, 2016, NuScale Power, LLC (NuScale) submitted to the U.S. Nuclear Regulatory Commission (NRC) a Design Control Document for its Design Certification (DC) application of the NuScale reactor design (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17013A229).

The purpose of the audit was to: (1) gain a better understanding of information underlying the application in the area of steam generator (SG) tube integrity, (2) identify information that will require docketing to support the basis of the licensing or regulatory decision; and (3) develop an understanding of the topics to support issuing clear requests for additional information (RAI). Specifically, the Audit Team reviewed information related to the SG degradation and tube integrity assessments, qualification of tube cleaning and inspection methods, and the determination of the tube plugging criterion. The "Audit Plan for the Regulatory Audit of NuScale Power, LLC, Design Certification Application, Design Control Document, Tier 2, Chapter 5, Section 5.4.1, 'Steam Generators'," is available in ADAMS under Accession No. ML17177A680.

II. Regulatory Audit Bases

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

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

An audit is required to examine detailed information related to the applicant's principle design criteria, and reach a safety conclusion on the NuScale application sections in the scope of this audit plan. The NRC staff must have sufficient information to ensure that acceptable risk and reasonable assurance of safety can be documented in the NRC staff's safety evaluation.

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 (GDC) 4, "Environmental and Dynamic Effects Design Bases," of Appendix A to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires that structures, systems, and components important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions during normal plant operation as well as during postulated accidents.

Enclosure 1

- GDC 14, “Reactor Coolant Pressure Boundary,” requires that the RCPB be designed with sufficient margin to assure that the design conditions are not exceeded during normal operation, including anticipated operational occurrences.

- GDC 32, “Inspection of Reactor Coolant Pressure Boundary,” requires that the reactor coolant pressure boundary be designed to permit periodic inspection and testing to assess structural and leakage integrity.

- 10 CFR 50.36, “Technical Specifications,” as it relates to the Steam Generator Program in the technical specifications.

- 10 CFR 50.55a(g) requires that inservice inspection programs meet the applicable inspection requirements in Section XI of the ASME Boiler and Pressure Vessel Code.

- 10 CFR 50.65, “Preservice and Inservice Inspection Requirements,” requires that licensees be able to monitor the condition of the steam generator tubes to provide reasonable assurance that the tubes are capable of fulfilling their intended functions.

- Appendix B to 10 CFR Part 50 applies to implementation of the Steam Generator Program.

III. Audit Location and Dates

Date: July 10, 2017 – October 11, 2017.

Location: Electronic Reading Room (eRR).

Status teleconferences with NuScale were held on August 15, September 12, and September 27, 2017.

IV. Audit Team Members

Gregory Makar (NRC, Audit Lead)
Leslie Terry (NRC)
Matthew Mitchell (NRC)
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Gery Wilkowski (Engineering Mechanics Corporation of Columbus (Emc^2))
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V. Applicant and Industry Staff Participants

<u>NuScale Power, LLC</u>
Marty Bryan
Darrell Gardner
Steven Mirsky
Jennie Wike
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Matt Mallet
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Karl Gross
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VI. Documents Audited

A. NP-EE-A014-2313, Rev. 0, "Steam Generator Program." (Note: This document is an engineering study and the observations in this summary made from this document are for information only.)

B. EC-A014-3224, Rev. 0, "SG Tube Structural Integrity Performance Criterion Calculation."

C. EQ-A011-1775, Rev. 1, "ASME Design Specification for Reactor Pressure Vessel."

D. ER-A010-2807, Rev. 0, "NuScale Reactor Module Degradation Mechanism and Inspectability Assessment."

E. ER-A010-4009, Rev. 0, "Supplement to NuScale Reactor Module Degradation Mechanism and Inspectability Assessment."

F. ER-A014-4350, Rev. 0, "Steam Generator Flow Restrictor Testing Final Technical Report."

G. ER-A014-2549, Rev. 1, "Flow Restrictor Test Development Plan."

H. ER-A014-2705, Rev. 0, "Steam Generator Tube Stabilizer Design Report."

I. ER-A014-2917, Rev. A, "Steam Generator Tube Corrosion Testing Needs Development."

J. ER-A014-3060, Rev. 0, "Steam Generator Degradation Assessment."

K. ER-A014-3093, Rev. A, "SG Sludge Deposition and Fouling Testing Needs Development."

L. ER-A014-3354, Rev. 0, "Steam Generator Tube Plugging Criterion and Structural Integrity Performance Criteria."

M. NP-ER-A014-2000, Rev. 1, "Steam Generator Tube Inspection Feasibility Report."

N. ED-A011-2658, Rev. 2, "Upper RPV Section Dimension Additions."

O. NP12-01-A011-M-SA-2689-S01 - Schematics.

P. PL-0702-48438, Rev. 0, “Steam Generator Flow Restrictor Testing Plan.”

Q. SD-A030-1929, Rev. 0, “Reactor Coolant System Design Description.”

R. SD-A014-4027, Rev. 0, “Steam Generator System Design Description.”

S. Steam Generator OE Document List (references used in the design related to operating experience).

T. Investigations on the Nuclear Steam Generator of NS Otto Hahn (German/English Translation).

U. TSD-T050-3424, Rev. 2, “Test Specification – Steam Generator Flow Restriction Device Test.”

VII. Summary of Observations

<u>Feasibility of SG Tube Inspection</u>

The NRC staff reviewed audit documents to gain a better understanding of the inspectability of the NuScale helical SG tubes and observed, in NP-ER-A014-2000, Rev. 1, NuScale performed an inspection feasibility demonstration that consisted of two mockup tubes, which were approximately [ ] long and representing the smallest [ ] and largest [ ] diameter helixes.

The NRC staff noted the following concerning the mockup tubes versus the NuScale design. The mock up tubes:

- had a smaller inner diameter (ID),
- were longer in length,
- contained welds near the transition bends; however the NuScale design tubes are seamless, and
- had less rigorous control for ovality.

NuScale asserts that actual tube conditions would be more favorable for inspection. Three different probe shafts [ ] and two different shaft tips were evaluated in the demonstration.

The following demonstration tests were performed:

- Hand-pushing each probe shaft into each mockup tube in the horizontal position.
- Hand-pushing the best performing probe shaft, [ ], with a tip into the small helix tube in the horizontal position. [ ]
- Inserting and withdrawing from the top the best performing probe shaft with a tip in each mockup tube in the vertical position. [

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- Pushing a dummy probe from the top with a probe drive system into each mockup tube in the vertical position and then retracting the dummy probe. [ ]

- Pushing a dummy probe from the bottom with a probe drive system into the large helix tube.

The NRC staff observed that most of the demonstration tests showed that an eddy current probe can transverse the entire length of the helical SG tube. The test consisting of pushing the dummy probe from the bottom with a probe drive system into the large helix tube was unsuccessful. The NRC staff notes that it would stand to reason the same test on the small helix tube would also be unsuccessful. [ ] While the NRC staff did not observe any demonstration tests, NuScale asserts it is feasible to develop [ ] so that inspections can be performed from the bottom. In addition, the dummy probe could not be retracted after being inserted past the lower transition region. NuScale asserts that the welds near the transition bends in the mockup tubes caused this retraction issue.

In NP-EE-A014-2313, Rev. 0, NuScale notes the signal to noise ratio, for bent tubes, is expected to be greater than for straight tubing and may result in defect detection and sizing limitations. The document goes on to note that iron deposits on the inside surface of the SG tubes may impact the ability to pass an eddy current probe through the tubes and may impact defect detection.

The NRC staff observed, from SD-A014-4027, Rev. 0, NuScale evaluated whether the SG tube support bars and cantilevers, steam and feed nozzles, piping, and steam and feed plenums are inspectable; and determined that all applicable elements of these components are inspectable.

<u>Flow Restrictors</u>

The NuScale SG design includes flow restrictors that extend into the tubes and are mounted on a plate in each feed plenum that is attached to the secondary-side face of the tubesheets. NuScale asserts that the flow restrictors will provide the necessary secondary-side pressure drop for flow stability and preclude density wave oscillations. The NRC staff reviewed ER-A014-4350, Rev. 0, and observed that NuScale conducted tests for nine flow restrictor configurations to determine the pressure drop as a function of flow rates for NuScale's full range of operating conditions, determine performance variation due to manufacturing and installation tolerances for the different configurations, and assess vibration behavior of the flow restrictors. The tests were performed on three different [ ], three different [ ], and three different [ ]. The [ ]. Each flow restrictor had a specific mounting plate design. Different insertion depths were evaluated for the [ ]. Different insertion depths were not evaluated for the [

] because its total length was [ ]. The [ ] also had different outside diameters (OD) [ ]. The tubing for the tests were Type 304 seamless stainless steel with the outer diameter matching the tubes in the NuScale design and the wall thickness slightly thinner than the tubes in the NuScale design. NuScale asserts that the thinner wall thickness falls within expected manufacturing tolerances for the ID. Each flow restrictor was tested at six selected flow/differential pressure conditions. Additional tests were conducted, including misalignment tests where the restrictor was flush with the inside of the tube. Based on the tests performed, NuScale concluded that the [ ] produced the highest loss coefficient. The vibration assessment performed by NuScale concluded that the impact of vibration on the restrictor design was likely negligible because the test conditions when vibration did occur were several times those of the planned NuScale conditions.

Given that the NRC staff observed little information regarding the potential degradation related to the flow restrictors and associated hardware, the NRC issued RAI 9231, Question 05.04.02.01-8 (Accession No. ML17347B444).

<u>Tube Supports</u>

NP12-01-A011-M-SA-2689-S01 notes that the NuScale SG support structure consists of eight sets of vertical bars that span the full height of the tube bundle between each column of tubes and are spaced at [ ] degree intervals. Each set of tube support bars is mounted to the reactor pressure vessel wall by attachment to the pressurizer baffle plate above the tube bundle and to a cantilever support below the tube bundle. The support bars contain short, [ ] punched tabs that provide vertical support. ER-A014-3354, Rev. 0, describes the details of how the support structures will contact the tubes. The punched tabs are curved, and may have up to 90 degrees of circumferential contact with the tubes. Contact between the tab and the top of a tube, and contact between the tubes and support plates, are axial line contacts. Document C notes that the tube support tabs for the eight inner column tube supports rest between a pair of backing strips.

NuScale notes in EQ-A011-1775, Rev. 1, the SG tube supports are considered “internal structures” because while the supports are within the pressure retaining boundary of the reactor pressure vessel, they are outside the pressure retaining boundary of the SG. In addition, EQ-A011-1775, Rev. 1, notes that the supports will be constructed in accordance with ASME BPV Code Section III, Subsection NG. The tube-to-support radial clearance is nominally [ ] which is similar to design clearances found in conventional nuclear SGs. The support material is Type 304/304L which is similar to newer conventional SGs. The SG tube supports are included in the NuScale SG program. Given that the NRC staff did not observe specific information regarding the performance of the support structures (i.e., providing adequate support and resisting degradation), the NRC issued RAI 9231, Question 05.04.02.01-6 (Accession No. ML17347B444).

<u>Degradation Assessment</u>

NuScale completed a preliminary degradation assessment and noted that it would be revised prior to the preservice inspection. In ER-A014-3060, Rev. 0, the NRC staff observed that NuScale’s preliminary degradation assessment consisted of evaluating how degradation mechanisms observed in the current operating fleet apply to the NuScale design, and determining whether there are degradation mechanisms unique to NuScale. NuScale determined that the following degradation mechanisms for the secondary side (inside SG tubes) are possible:

- Pitting formed under deposits on the tube inner diameter. The pitting could potentially be activated by exposure to oxygen during secondary side drains to allow inservice inspection.

- Potential intergranular attack (IGA) or intergranular stress corrosion cracking (IGSCC) under deposits in the boiling zone.

- Erosion-enhanced corrosion along the tube ID due to potential impingement of debris or corrosion products along the curved tubes due to centrifugal effects.

NuScale determined that the following degradation mechanisms for the primary side (outside SG tubes) are possible:

- Pitting under crud deposits potentially activated during drain downs for refueling activities.

- Wear between tubes and tube supports or between tubes and potential foreign material.

- IGA or IGSCC in the tube-to-tubesheet crevice.

NuScale notes that the potential for pitting could be greater than for conventional SGs. NuScale also notes that secondary-side foreign objects would most likely result in damage to tube end welds in the feed plenums. NuScale notes that radioactive crud can collect in the physical crevices between the tubes and the tube supports. Overall, NuScale determined that wear at tube supports is the most likely degradation mechanism. While high-cycle fatigue was not determined by NuScale to be a likely degradation mechanism, NuScale plans to further evaluate the potential as part of the detailed design because some fluctuation in the boiling interface is expected inside the tubes which may result in thermal stresses. NuScale determined that denting is unlikely due to the corrosion resistant material (Type 304) used for the tube support structures. Primary water stress corrosion cracking and primary-side IGA are considered unlikely degradation mechanisms by NuScale because the SG tubes are thermally-treated Alloy 690. In addition, NuScale determined that wastage/thinning is an unlikely degradation mechanism because of the use of an all-volatile treatment chemistry. NuScale notes that some sludge/scale deposition on the ID of the tubes is expected due to iron transports from the secondary systems and the inability to blowdown the secondary side of the SGs. NuScale determined degradation growth rates for the likely degradation mechanisms, including high-cycle fatigue, using the default growth rates in the Electric Power Research Institute (EPRI) SG Tube Integrity Assessment Guidelines. The EPRI default growth rates were used because there is not operating experience for NuScale. NuScale asserts in ER-A014-3060, Rev. 0, the growth of potential degradation is expected to be similar to or less then what is found in the currently operating fleet. Wear flaws are considered by NuScale to bound other volumetric degradation mechanisms. NuScale asserts that existing nondestructive examination techniques are available to detect the likely degradation mechanisms for the SGs.

As a result of the preliminary degradation assessment, NuScale concluded that SG tube degradation can be monitored and managed using inspection strategies and techniques similar to those in the currently operating fleet consistent with NEI 97-06.

Plugging Criteria

The NRC staff and Emc^2 staff reviewed audit documents to gain a better understanding of how the SG tube plugging criteria was determined and how Combined License (COL) applicants referencing the NuScale design certification will determine a site-specific SG tube plugging criteria. [ ]

The NRC staff has determined, based on the audit, that additional information is required regarding the SG tube plugging criteria and issued RAI 9273, Question 05.04.02.01-1 (Accession No. ML17340A632).

Tube Plugging and Stabilization

The NRC staff reviewed audit documents to gain a better understanding of NuScale's specific plans for tube plugging and stabilization hardware, procedures, and inspection requirements. While a tube plug design has not yet been selected, it is NuScale's intent to use an existing industry mechanically installed plug which may need to be adapted for the specific NuScale SG tube geometry. Because of this specific tube geometry, NuScale developed ER-A014-2705, Rev. 0, that defines a tube stabilizer design concept and basic tube stabilizer requirements. The document noted that the NuScale degradation assessment informed the stabilizer design concept. Based on the degradation assessment, NuScale concluded that OD stress corrosion cracking at the tube-to-tubesheet crevice in both the steam and feed tubesheet regions is the only location where tube stabilization would be required. For stabilization in this region, NuScale described a [ ]. NuScale asserts that the length of the stabilizer depends on tube plug design. NuScale also described the potential need for a tube retention device for stabilizers installed at the steam plenum tubesheet to mitigate a severed tube from displacing downward due to gravity. [ ] The NRC staff observed that NuScale recognized the need to be able to stabilize a tube at any location. [ ]. NuScale asserts that tube stabilizers [ ] would be acceptable because they would be compatible with the primary and secondary water chemistries, and that the tube stabilizers would not be susceptible to stress corrosion cracking because they would not experience tensile stress.

ER-A014-3060, Rev. 0, states a preservice inspection of SG tube plugs shall be performed after installation and before the initial cycle of power operation with the installed plugs. The document goes on to state that the preservice inspection verifies that the plugs were installed in the proper location and in accordance with pre-established acceptance criteria. ER-A014-3060, Rev. 0, also states any tube plugs that were installed prior to operation will be visually examined in accordance with “EPRI Steam Generator Management Program: Pressurized Water Reactor Steam Generator Examination Guidelines,” during the first inservice inspection. Subsequent inservice inspection requirements will be developed after specific designs are developed but are expected to be consistent with inspection plans used in the currently operating fleet.

Tube Cleaning

According to NP-EE-A014-2313, Rev. 0, the need for periodic secondary side SG cleaning is likely based on a preliminary evaluation of potential deposition of steam plant corrosion products inside the SG tubes. NP-EE-A014-2313, Rev. 0, also notes that NuScale investigated various commercial SG cleaning processes. NuScale notes in SD-A014-4027, Rev. 0, that cleaning methods will depend on the deposits that need to be removed. SD-A014-4027, Rev. 0, also states that required cleaning will be performed in the inspection bay during refueling outages and chemical cleaning of the secondary side of the SG system will be by [

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Monitoring of Primary-to-Secondary Leakage

The NRC staff inquired whether a gap regarding primary-to-secondary leakage that was identified in NP-EE-A014-2313, Rev. 0, had been closed or justified. Specifically, Section 4 of this document noted that primary-to-secondary leakage monitoring could not be performed in Modes 2, 3, and 4. NuScale staff noted that the gap is closed and that primary-to-secondary leakage monitoring is in Limiting Condition for Operation 3.4.5 of the Technical Specifications.

EPRI Alloy 690 Steam Generator Tubing Specification Sourcebook

The 2014 EPRI Alloy 690 Steam Generator Tubing Specification Sourcebook, which NuScale references, was superseded by the December 2016 version. The NRC staff inquired how NuScale was addressing the changes in the December 2016 version. NuScale staff noted that the only change from the 2014 version was the [

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Access to Primary Side (Outside SG Tubes)

The NRC staff inquired how the necessary tools for inspection and foreign object retrieval will access the primary side of the SG. The NuScale staff discussed how, overall, access to the primary side is improved by the design of the SG. Specifically, when the upper NuScale Power Module that consists of the upper containment vessel and upper reactor pressure vessel is in the inspection rack in the dry dock, there is 360 degree direct access to the primary side of the SG. NuScale asserted that inspection and foreign object removal tooling may be passed from the bottom of the SG to the top of the SG and between each tube column. In addition, NuScale noted that no special tooling for inspection or foreign object retrieval is required for the NuScale SG design.

Additional Observations

Consistent with EQ-A011-1775, Rev. 1, the NRC staff observed in ER-A014-3060, Rev. 0, that the SG tubes are welded to the cladding on the inner plenum surface and in ER-A014-2705, Rev. 0, that the SG tubes will be hydraulically expanded into the tubesheet holes to within [ ] of the primary face of the tubesheet.

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VIII. Status Briefing

The NRC staff conducted an audit status meeting on August 15, 2017, where they provided a status of the audit and development of RAIs, and discussed the basis for the audit extension. On September 12, 2017, and September 27, 2017, the NRC staff conducted audit status meetings to discuss questions arising from the audit. The main focus of the questions and discussions for both meetings was NuScale's SG tube plugging criteria.

IX. Exit Briefing

The NRC staff conducted an audit closeout meeting on October 11, 2017. At the exit briefing the NRC staff reiterated the purpose of the audit and discussed their activities. The NRC staff stated that the audit assisted them in obtaining a better understanding of the applicant's helical coil SG design, inspection feasibility, degradation assessment, plugging criteria, and SG program. Additionally, the NRC staff stated that they had modified or eliminated proposed questions for a request for additional information as a result of the audit, and that the audit resulted in one question focused on the SG tube plugging criteria (*see* Section X below).

X. Requests for Additional Information Resulting from Audit

The audit resulted in one question with multiple parts that will be included in the request for additional information on Section 5.4.1 in Tier 2 of the FSAR. RAI 9273, Question 05.04.02.01-1 focuses on the determination of the SG tube plugging criteria, and is available in ADAMS under Accession Number ML17340A632.

XI. Open Items and Proposed Closure Paths

The NRC staff inquired about four open design items (ODIs) identified during the audit. NuScale staff noted that ODI-15-0137 and ODI-15-0138 related to design and normal operating conditions were verified with no changes to the ODI assumptions. NuScale also noted that ODI-16-0139 related to overpressure protection provided by a single thermal relief valve on each feedwater line was verified with no changes to the ODI assumptions. On September 12, 2017, and September 28, 2017, NuScale noted that ODI-15-0139 related to seismic accelerations had not been verified. The NRC staff noted on September 27, 2017, that it would continue to monitor the closure of ODI-15-0139 through the appropriate NRC staff.

XII. Deviations from the Audit Plan

The duration of the audit was extended from August 15, 2017, to October 11, 2017, Emc^2 staff were added to the Audit Team, and NuScale added additional documents to the eRR upon request from the Audit Team.