UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 13, 2020

MEMORANDUM TO: Michael I. Dudek, Chief
New Reactor Licensing Branch
Division of New and Renewed Licenses
Office of Nuclear Reactor Regulation

FROM: Marieliz Vera, Project Manager ***/RA/ Omid Tabatabai for***
New Reactor Licensing Branch
Division of New and Renewed Licenses
Office of Nuclear Reactor Regulation

SUBJECT: U.S. NUCLEAR REGULATORY COMMISSION REPORT OF THE REGULATORY FOLLOW-UP AUDIT PERFORMED BETWEEN MAY 1, 2019, THROUGH SEPTEMBER 19, 2019, REGARDING THE NUSCALE STRESS AND FATIGUE ANALYSIS OF MAJOR DESIGN COMPONENTS

On January 6, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17013A229), NuScale Power, LLC (NuScale) submitted a design certification (DC) application via a transmittal letter dated December 31, 2016, for a Small Modular Reactor, to the U.S. Nuclear Regulatory Commission (NRC). The NRC staff started its detailed technical review of NuScale's DC application on March 15, 2017.

The NRC staff conducted an audit to verify that the component design in support of the NuScale Standard Plant DC application are being performed in accordance with the methodology and criteria described in the NuScale Final Safety Analysis Report. The audit was initiated on May 1, 2019, and ran through September 19, 2019, in accordance with the audit plan in ADAMS (ADAMS Accession No. ML19116A215).

The purpose of the audit was to: (1) gain a better understanding of the NuScale design; (2) verify information; (3) identify information that may require docketing to support the basis of the licensing or regulatory decision; and (4) review related documentation and non-docketed information to evaluate conformance with regulatory guidance and compliance with NRC regulations.

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The NRC staff conducted the audit via access to NuScale's electronic reading room. The audit was conducted in accordance with the NRC Office of New Reactors Office Instruction NRO-REG-108, "Regulatory Audits."

The publicly available version of the audit report is enclosed with this memorandum.

Document transmitted herewith contains sensitive unclassified information. When separated from the enclosure, this document is "DECONTROLLED."

Docket No. 52-048

Enclosure:
1. Audit Summary – (Non-Proprietary)
2. Audit Summary – (Proprietary)

cc: NuScale DC ListServ

SUBJECT: U.S. NUCLEAR REGULATORY COMMISSION REPORT OF THE REGULATORY FOLLOW-UP AUDIT PERFORMED BETWEEN MAY 1, 2019, THROUGH SEPTEMBER 19, 2019, REGARDING THE NUSCALE STRESS AND FATIGUE ANALYSIS OF MAJOR DESIGN COMPONENTS DATED: JANUARY 13, 2020

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ADAMS Accession No.: ML19319C232 ***via email** **NRR-106**

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U.S. NUCLEAR REGULATORY COMMISSION

NUSCALE POWER, LLC SUMMARY AUDIT REPORT OF STRESS AND FATIGUE ANALYSIS OF MAJOR DESIGN COMPONENTS

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1.0 BACKGROUND

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, Section 47, "Contents of applications; technical information," states that:

> *The application must contain a level of design information sufficient to enable the Commission to judge the applicant's proposed means of assuring that construction conforms to the design and to reach a final conclusion on all safety questions associated with the design before the certification is granted. The information submitted for a design certification must include performance requirements and design information sufficiently detailed to permit the preparation of acceptance and inspection requirements by the [U.S. Nuclear Regulatory Commission] NRC, and procurement specifications and construction and installation specifications by an applicant. The Commission will require, before design certification, that information normally contained in certain procurement specifications and construction and installation specifications be completed and available for audit if the information is necessary for the Commission to make its safety determination.*

On March 23, 2017, the U.S. Nuclear Regulatory Commission (NRC) accepted the Design Certification (DC) application for docketing for the NuScale Standard Plant Design Certification Application for a small module reactor (SMR) design submitted by NuScale Power, LLC, (NuScale), (Reference 1).

The NRC staff performed the design specification audit as a result of NuScale's response to the staff's Request for Additional Information (RAI) (RAI 9362, Question 03.08.02-16 and RAI 9459, Question 03.08.02-18) regarding the stress and fatigue analysis of the major components. As a result of the design specification audit (Reference 2 audit report), the staff identified in Table 1, audit items 4B, 103 and 104 are opened since the stress and fatigue evaluations were not available for review and will need to be reviewed in a next audit. To resolve these open audit items, the staff performed a subsequent audit of stress and fatigue analysis of the NuScale major components that were listed in the audit plan (Reference 3). This audit is to allow the staff to gain a better understanding of the supporting design calculations and to confirm a certain level of completeness of the NuScale design at the DC stage.

In the NuScale Letter LO-0219-64398 (Reference 4), dated February 4, 2019, NuScale provided the scope and schedule for the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) Section III stress analyses including fatigue to be completed for the NuScale Power Module (NPM) during the Design Certification Application (DCA) review. These stress analyses evaluate the loading combinations as required by the ASME design specifications for primary, secondary, and peak stresses, including the calculation of cumulative usage factors (fatigue). NuScale uploaded the stress and fatigue stress analyses documents in the electronic reading room (eRR) for the staff to review. In addition, NuScale provided additional documents for staff review during the onsite visit at the NuScale office in Corvallis, Oregon on September 10-13, 2019.

2.0 AUDIT RESULTS

Prior to this audit, the fatigue calculations were not available for review. NuScale evaluated the most critical locations within the NuScale Power Module, and provided the staff with a summary of the results showing the most critical locations for the reactor vessel, containment vessel, and reactor vessel internals, in the ASME Component Fatigue Screening Report, which was provided in the eRR.

The staff reviewed the stress and fatigue calculations that NuScale posted to the eRR. The calculations were in various stages of preparation. Some were letter revision levels and some were numeric revision levels. NuScale quality assurance procedure QP-0303-10267 "Design Control Process" Revision 13 defines letter and number revisions. A letter revision is a preliminary document, and a number revision is an approved/verified document. A letter revision document will ultimately be revised. If any of the assumptions in a document cannot be verified when it is made a numbered revision, Open Design Issues (ODIs) will be established for those unverified assumptions. The ODIs are used for assumptions that need to be verified. NuScale document EP-0303-310, "Open Design Item Management" Revision 7 provides instructions for the administration of ODIs.

During the stress and fatigue audit, the staff reviewed the updated primary stress analysis of the containment vessel which incorporated the new seismic loads and the updated design pressure of 1050 pounds per square inch absolute. The staff also reviewed updates to previously reviewed documents to ensure that the containment vessel with the updated design pressure and new seismic loads conformed to the requirements of the ASME BPV Code. These included the ASME Code reports for the containment vessel (CNV) FW Nozzle and CNV Main Steam nozzle and the ASME design specification for the Containment Vessel.

NuScale explained that the Primary Stress analysis for the CNV Nozzles was separated from the main Primary Stress analysis document for the CNV to facilitate updates. The staff noted in the updated Primary Stress analysis of the containment vessel that the controlling stress results in the vessel for design conditions was the general membrane stress, Pm, in the Upper CNV Shell. As the acceptance ratio was close to the ASME limit, the staff requested NuScale to detail any conservatisms in the stress calculation. NuScale stated that the Reactor Module Seismic Load Specification provides the updated seismic loads to qualify the membrane stress of the CNV shell. The seismic loads were applied at eight elevations of the CNV.

Conservatively, maximum forces and moments from all elevations were used to calculate membrane stresses. The staff reviewed the conservatisms detailed by NuScale and observed that the overall approach meets the ASME Code requirements for primary stress.

The staff noted in the Primary Stress analysis of the containment vessel nozzles that the Level C Allowable stress ratios for CNV nozzles 5-7, 11-14, 22 and 23 were close to the ASME limit and the staff requested NuScale to clarify what conservatisms were used in the stress calculation. NuScale stated that the development of the nozzle loads contained conservatism. NuScale explained that the nozzles are grouped by nominal dimensions of the connected piping and the maximum force and moment components from all grouped nozzles are used in the stress calculation. NuScale also provided the staff with an example showing the actual loads and the comparison to the bounding loads used. The staff reviewed the conservatisms detailed by NuScale and observed this meets the ASME Code requirements for primary stress.

The staff also reviewed the primary stress calculation for the CNV RPV support. The staff reviewed the Finite Element Analysis (FEA) performed on the CNV RPV support and observed that stresses are within Level D limits and meet the requirements of the ASME BPV Code, Section III, Subsection NF.

NuScale performed a fatigue evaluation to determine the most limiting locations. For the CNV, the refueling flange, the main steam nozzle, the feedwater nozzle, and the CVCS nozzles were identified as the most severe.

The staff reviewed the main steam and feedwater nozzle ASME Code calculation documents for the CNV. The staff noted that these were Revision A documents. NuScale had not revised the main steam and feedwater nozzle stress calculations to reflect the update to the primary stress analysis for the containment vessel, which incorporated the updated design pressure and updated seismic loads. NuScale stated that these Revision A documents are preliminary and that these calculation documents would be revised to include the latest design information as prescribed in the design specification for the CNV to complete the ASME design report for these components. The staff acknowledges that these calculation documents are preliminary and the staff will be able to verify the as-built design report as part of the Inspections, Tests, Analysis, and Acceptance Criteria (ITAAC) closure process.

The staff reviewed the CNV refueling flange stress analysis document, which also contained a fatigue evaluation. The stresses were within ASME Code requirements. The fatigue usage was less than 1.0 and meets ASME Code requirements.

The staff also reviewed the CNV CVC Nozzle/Weld Fatigue Analysis which detailed the fatigue analysis of welds connecting safe-ends of four CNV CVCS nozzles to their respective valve bodies and nozzle forgings. These were performed for CNV6 – RCS injection, CNV7 – PZR spray, CNV13 – RCS discharge, and CNV14 – RPV high point degasification. The bounding nozzle was CNV6 – RCS injection at the nozzle-to-top safe-end weld wet surface and lower transition zone of the nozzle. The fatigue usage was less than 0.4 for the welds and nozzle necks and less than 1.0 for the nozzle transitions through base metal. This meets ASME Code requirements.

The staff reviewed ER-A010-2529 "RXM Thermal Transient Load Definition Specification" Revision 0, and noticed that the temperature change denoted in Figure 4-2, RPV Head Gas

T&P – Reactor Cooldown from Hot Standby appeared to exceed the temperature limitations placed upon the pressurizer cooldown rate as denoted in FSAR Section 3.9.1.1.1, "Service Level A Conditions." FSAR Section 3.9.1.1.1, under Service Level A Transient 2 – Reactor Cooldown from Hot Shutdown states, "The steam and feedwater flow rates are controlled to keep the cooling rate below the maximum of 100 °F/hr (200 °F/hr in the pressurizer)." NuScale explained that the 200 °F/hr is applicable during the cooldown phase when the steam and feedwater systems are being used and the cooldown rate is controlled to be no greater than 200 °F/hr in the pressurizer. NuScale also explained that the near 200 degree temperature change depicted in Figure 4-2 of ER-A010-2529 is the result of opening the Emergency Core Cooling System valves, which results in the quick cooldown of the pressurizer and that the 200 °F/hr limit is not applicable. The staff suggests that NuScale clarify this section of the FSAR to note that other systems beyond the main steam and feedwater systems are engaged to cool the reactor down to safe shutdown conditions and that the 200 °F/hr limit is only applicable when the main steam and feedwater systems are in service and being used to cool down the reactor. The staff observed that the fatigue calculations for applicable parts of the NuScale Power Module used the appropriate cooldown rates to calculate the metal temperatures for use in the fatigue calculations.

The staff identified that there is inconsistency between the FSAR and ER-A010-2529 "RXM Thermal Transient Load Definition Specification" Revision 0. The FSAR states that the reactor cooldown from Hot Shutdown transient rate is 100 ºF/hr for the RCS and 200 ºF/hr for the pressurizer. Section 4.3.2 and Figure 4-54 of ER-A-010-2529 showed that the pressurizer temperature drops [redacted], which does not meet the 200 ºF/hr limit as stated in FSAR 3.9.1.1.1. In the NuScale letter LO-1019-67627 dated October 17, 2019 (Reference 5), the applicant issued the revisions to FSAR Section 3.9.1 to address the inconsistency.

The staff reviewed document EC-A011-7207 "ECCS Valve Flange Bolting Stress Calculation," Revision 0. The staff observed that NuScale has adequately demonstrated that the design of the bolted flange connections for the Emergency Core Cooling System Valves (i.e., reactor recirculation valves and the reactor vent valves) satisfy the pertinent bolting stress and fatigue design requirements of ASME Section III, NB-3200, as described in NuScale DCA Part 2, Tier 2, Section 3.6.2.7. The staff finds this acceptable.

The staff identified that the method used in determining environmentally assisted fatigue (EAF) correction factor may not be adequate. During its review of EC-A011-2707, "Integrated Steam Plenum ASME Code Stress Analysis" Revision A, the staff noted that NuScale used the average temperature of the transient to evaluate EAF correction factor (Fen). The staff noted that the NuScale's approach in defining average temperature did not properly account for the threshold temperature and in using its defined average temperature did not yield a comparable or conservative estimate of Fen when compared to the results of the detailed integrated approach, as noted in Revision 0 of the NUREG/CR-6909, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials." NuScale acknowledged the deficiency in determining Fen and performed a preliminary calculation using a different approach in determining Fen. NuScale also stated that average temperature method, without considering threshold temperature, had been applied to all fatigue analyses. A corrective action report (CR-0819-66550) was generated to update the fatigue calculations according to Revision 1 of NUREG/CR-6909. NuScale stated that all fatigue analyses will address this issue. On the basis that NuScale plans to revisit the fatigue analyses and the staff can verify these final analyses during the inspections of ITAAC as-built, the staff considered this approach acceptable.

The staff also observed that Table 2-6 of EC-A011-2707, “Nozzle external loads” listed the sum of the square (SRSS) force and SRSS moment. The staff noted that SRSS force/moment is not the traditional definition for the nozzle load. The staff asked NuScale to clarify how this loading was applied in the analysis model. NuScale responded that the analyst did not model the nozzle loads adequately in this preliminary analysis. NuScale also stated that the analysis will model the nozzle loads adequately by separating axial/shear forces and torsional/bending moments instead of using the SRSS force/moment. The staff can verify this during inspections of ITAAC, so the staff finds this acceptable.

The staff identified that the fatigue analysis did not consider the piping thermal load case during the review of EC-A013-2823 “CNV Feedwater Nozzle ASME Code Calculation” Revision A. The staff asked NuScale whether the thermal piping load was also neglected in other CNV nozzles’ analyses. NuScale explained that the design specification did not address piping thermal load. This deficiency is also applied to Main Steam Nozzle Report as shown in Table 2-6, “CNV Specification MS Nozzle External Loads” of EC-A010-2822 “CNV Main Steam Nozzle ASME Code Calculation” Revision A. The staff acknowledged that these calculations are preliminary, and the licensee will update the CNV design specification and design report. The staff will be able to verify the ITAAC as-built design report.

The staff also reviewed the fatigue pairing cycles and identified that the heatup to hot standby transient, with temperature increasing from 62 ºF to 422 ºF, and power ascent from hot standby transient, with temperature increasing from 422 ºF to 545 ºF, forms a fatigue pairing cycle. This is not the traditional fatigue pairing cycle. NuScale responded that the fatigue pairing used ANSYS computer code to select the peak stress and valley stress to form a fatigue pairing cycle. The staff reviewed the stress histograms and observed that these two transients stresses form a peak and valley and using ANSYS logic, which is adequate.

The staff also reviewed preliminary stress and fatigue calculations for the reactor internals, specifically the core support structures. Documents EC-A023-7294 “RVI Core Supports Fatigue Evaluation” Draft Rev 0, and EC-A023-6493 “RVI ASME Qualification for Level D Condition” Revision 1 were made available to the staff for review. Through reviewing of these reports and discussion with NuScale, NuScale showed that the stress and fatigue results for reactor internals are below the allowable limits set forth in the ASME Code Section III, Subsection NG, which is the primary design code of standard for reactor internals, with two exceptions that will result in a change in material selection:

1. The material selection for the four threaded structural fasteners that secure the upper core plate to the upper support blocks and the four threaded structural fasteners that secure the lower core plate to the core support blocks will be changed from the current material to a high strength material.

2. The weld selection for the welds that secure the upper support blocks to the core barrel will be changed to a full penetration weld.

NuScale identified that both of these changes are being made to increase the stress allowable for fatigue calculation.
Overall observation for the reactor internals is that although the stress and fatigue calculations are not completed at this point, NuScale showed that the preliminary calculations are acceptable, and the preliminary stress and fatigue results have adequate margins. NuScale also demonstrated that for the two material changes stated above, although the final selection

of the materials has not been made, the materials in consideration will have stress results that are acceptable in accordance with the stress allowable set forth in the ASME Code, Section III, Subsection NG.

3.0 DOCUMENTS REVIEWED

1. EC-A013-6689 “CNV CVC Nozzle/Weld Fatigue Analysis” (CNV6, CNV7, CNV13, CNV14) Revision1 dated 07/31/2019
2. EC-A013-5863 “CNV Refueling Flange Stress Analysis” Revision 0 dated 06/26/2019
3. ER-B020-6230 “Preliminary ASME Design Report for RRV” Revision 0 dated 02/20/2018
4. ER-B020-63-06 “Preliminary ASME Design Report for RVV” Revision 0 dated 03/09/2018
5. EC-A011-7207 “ECCS Valve Flange Bolting Stress Calculation” Revision 0 dated 08/30/2019
6. ER-A010-7203 “NuScale Methodology for Selecting DCA ASME Fatigue Evaluation Locations” Revision 1 dated 06/27/2019
7. EC-A011-2707 “Integrated Steam Plenum ASME Code Stress Analysis”. Revision A dated 05/22/2015
8. ER-A010-2529 “RXM Thermal Transient Load Definition Specification” Revision 0 dated 03/31/2016
9. EC-A023-7294 “RVI Core Supports Fatigue Evaluation” Draft Revision 0
10. EC-A023-6493 “RVI ASME Qualification for Level D Condition” Revision 1 dated 08/01/2019
11. ER-A010-1350 “Pressure and Thermal Transient for Analysis of NSSS Component,” Revision 2 dated 05/30/2019
12. EC-A011-7241 “Steam Generator Feedwater Plenum Stress Analysis,” Revision 0 dated 06/28/2019
13. NP12-01-A013-M-GA-2601-S01 through S09, “Lower CNV Section” Revision 1 dated 03/28/2016
14. NP12-01-A013-M-GA-2602-S01 through S20, “CNV Upper Section” Revision 3 dated 05/03/2017
15. EQ-A013-1826 “ASME Design Specification for Containment Vessel” Revision 2 dated 06/20/2019
16. ER-A010-3265 “RXM Nozzle Loading Specification” Revision A dated 02/23/2016
17. ECN-A013-6947 “CNV Design Pressure” Revision 0 dated 01/24/2019
18. EC-A013-3377 “Primary Stress Analysis of the Containment Vessel” Revision 1 dated 07/30/2019
19. EC-A013-7411 “Primary Stress Analysis of the Containment Vessel Nozzles” Revision 0 dated 07/30/2019
20. ER-A010-5511 “Reactor Module Seismic Load Specification” Revision 1 dated 02/28/2019
21. EC-A013-6690 “CNV CVC Nozzle/Weld Fatigue Analysis (CNV6, CNV7, CNV13, CNV14)” Revision 1 dated 07/31/2019
22. EC-A010-7271 “RPV Upper Supports Primary Stress Analysis” Revision 0 dated 07/25/2019
23. EC-A013-2823 “CNV Feedwater Nozzle ASME Code Calculation” Revision A dated 10/12/2015
24. EC-A013-2822 “CNV Main Steam Nozzle ASME Code Calculation” Revision A dated 10/13/2015
25. ER-A010-7203 “NuScale Methodology for Selecting DCA ASME Fatigue Evaluation Locations” Revision 0 dated 05/31/2019

26. EP-0303-310 “Open Design Item Management” Revision 7
27. QP-0303-10267 “Design Control Process” Revision 13

4.0 **REFERENCES**

1. NRC Letter, “NuScale Power, LLC – Acceptance of an Application for Standard Design Certification of a Small Modular Reactor,” dated March 23, 2017, ADAMS Accession No. ML17074A087,
2. U.S. Nuclear Regulatory Commission Staff Report of Regulatory Audit for NuScale Power, LLC; Follow-up Audit of Component Design Specifications, dated 02/11/2019 ADAMS Accession No. ML19018A140
3. Audit Plan for The Regulatory Audit of The NuScale Power, LLC, Component Design, dated 04/29/2019, ADAMS Accession No. ML19116A215,
4. NuScale Letter LO-0219-64398, “NuScale Power, LLC – ‘Submittal of Scope and Schedule for Component Stress Analysis.” dated February 4, 2019, ADAMS Accession No. ML19035A682
5. NuScale Power, LLC Submittal of Changes to Final Safety Analysis Report, Section 3.9.1, “Special Topics for Mechanical Components” dated October 17, 2019, ADAMS Accession No. ML19290F363