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

February 28, 2020

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

FROM: Getachew Tesfaye, Senior Project Manager ***/RA/***
New Reactor Licensing Branch
Division of New and Renewed Licenses
Office of Nuclear Reactor Regulation

SUBJECT: AUDIT REPORT FOR THE REGULATORY AUDIT OF NUSCALE POWER, LLC DESIGN CERTIFICATION APPLICATION, CHAPTER 15, “TRANSIENT AND ACCIDENT ANALYSES,” AND CHAPTER 6, “ENGINEERED SAFETY FEATURES”

By letter dated December 31, 2016, NuScale Power, LLC (NuScale) submitted to the U.S. Nuclear Regulatory Commission (NRC) a Final Safety Analysis Report (FSAR) for its Design Certification Application (DCA) of the NuScale design (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17013A229). By letter dated August 26, 2019 (ADAMS Accession No. ML19238A307), the NRC was notified that the inadvertent actuation block (IAB) valve, which is a component of the emergency core cooling system (ECCS), pressure differential operating range had changed from that assumed in the DCA, Revision 2.

This design change affects, at a minimum, the Chapter 15 analyses associated with the loss-of-coolant accident (LOCA) analysis (Section 15.6.5), inadvertent operation of the ECCS (Section 15.6.6), and the containment response in DCA Section 6.2. The purpose of this regulatory audit is to understand the impact of the revised IAB threshold operating range on figures and components of merit used to make regulatory findings.

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The audit was conducted from the NRC headquarters via NuScale's electronic reading room, the NuScale office, and via telephone conferences.

Docket No. 52-048

Enclosure:
As stated

cc w/encl.: DC NuScale Power LLC Listserv

SUBJECT: AUDIT REPORT FOR THE REGULATORY AUDIT OF NUSCALE POWER, LLC DESIGN CERTIFICATION APPLICATION, CHAPTER 15, "TRANSIENT AND ACCIDENT ANALYSES," AND CHAPTER 6, "ENGINEERED SAFETY FEATURES" DATED: FEBRUARY 28, 2020

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ADAMS Accession No.: ML20054A060 ***via email** **NRR-106**

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AUDIT REPORT FOR THE REGULATORY AUDIT OF NUSCALE POWER, LLC DESIGN CERTIFICATION APPLICATION, CHAPTER 15, "TRANSIENT AND ACCIDENT ANALYSES" AND CHAPTER 6, "ENGINEERED SAFETY FEATURES"

A. BACKGROUND

By letter dated December 31, 2016, NuScale Power, LLC (NuScale) submitted to the U.S. Nuclear Regulatory Commission (NRC) a Final Safety Analysis Report (FSAR) for its Design Certification Application (DCA) of the NuScale design (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17013A229). By letter dated August 26, 2019 (ADAMS Accession No. ML19238A307), the NRC was notified that the inadvertent actuation block (IAB) valve, which is a component of the emergency core cooling system (ECCS), pressure differential operating range had changed from that assumed in the DCA, Revision 2.

This design change affects, at a minimum, the Chapter 15 analyses associated with the loss-of-coolant accident (LOCA) analysis (Section 15.6.5), inadvertent operation of the ECCS (Section 15.6.6), and the containment response in DCA Section 6.2. The purpose of this regulatory audit is to understand the impact of the revised IAB threshold operating range on figures of merit used to make regulatory findings.

B. REGULATORY AUDIT BASIS

This audit is based on the following regulatory requirements:

- 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," which requires in part that the applicants take provisions to accommodate and appropriately protect structures, systems and components important to safety against the environmental conditions, including dynamic effects, that may result from normal operation, maintenance, testing, equipment failures and postulated accidents,

- GDC 10, "Reactor Design," as it pertains to ensuring the specified acceptable fuel design limits are not exceeded during normal operation, including the effects of anticipated operational occurrences (AOO),

- GDC 16, "Containment design," which requires in part that a reactor containment and associated systems be provided to establish an essentially leak-tight barrier and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require,

- GDC 35, "Emergency core cooling system," as it pertains to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts,

- GDC 38, “Containment heat removal,” which requires that:

    - The containment heat removal system be capable of rapidly reducing the containment pressure and temperature following a LOCA and to maintain these parameters at acceptably low levels,

    - The containment heat removal system performs in a manner consistent with the function of other systems,

    - The safety-grade design of the containment heat removal system provides suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capability to ensure 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 in the event of a single failure,

- GDC 50, “Containment design basis,” which requires in part that the reactor containment structure and its internal compartments to accommodate the calculated pressure and temperature conditions resulting from any LOCA,

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

- 10 CFR 50.43(e), concerning testing to qualify and assess the capability of submitted designs to meet safety criteria,

- 10 CFR 50.46, “Acceptance criteria for ECCS for Light-Water Nuclear Power Reactors.”

C. AUDIT LOCATION AND DATES

The audit was conducted from the NRC headquarters via NuScale’s electronic reading room, the NuScale Office, and via telephone conferences.

Dates: September 17, 2019, through December 18, 2019

Locations: NRC Headquarters
Two White Flint North
11545 Rockville Pike
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D. AUDIT TEAM MEMBERS

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E. APPLICANT PARTICIPANTS

Mike Melton
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Ben Bristol
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Austin Thody
Adam Brigantic
Scott Barnes
Taylor Coddington
Yeon Jong Yoo
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F. AUDIT DOCUMENTS

The staff audited the following documents provided by NuScale:

Electronic Reading Room Document Lists	
IAB Closure Plan Audit Folder	
Document Number	Document Title
FSAR, Draft Revision 4	Chapter 6 IAB Changes CP-1967
TR-0516-49084, Draft Revision 2	CNV Technical Report IAB Changes CP-1968
TR-0516-49084, Draft Revision 2	Containment Response Analysis Methodology
FSAR Section 6.2, Draft Revision 4	DRAFT Chapter 6 IAB Changes Post-Audit
TR-0516-49084, Draft Revision 2	DRAFT TR-0516-49084 IAB Changes Post-Audit
EC-0000-2749	Draft Compare R1 to R2
ECN-A013-7650, Revision 0	Documentation of Limiting CNV Peak Pressure Cas Identified in ECN-A013-7531
ECN-A013-7531, Revision 0	IAB Range Effect on Containment Accident Pressure
FSAR, Draft Revision 4	FSAR 15.6.5 LOCA CP-1931 Final
FSAR, Draft Revision 4	FSAR 15.6.6 CP-1971 - Final Approved Change Package
FSAR, Draft Revision 4	FSAR 15.6.6 CP-1971 10-22-2019
FSAR, Draft Revision 4	FSAR Chapter 6 IAB Range Impact CP-1919

FSAR, Draft Revision 4	FSAR Section 15.6.6 IORV IAB Update CP-1924
N/A	IAB Range Calculation Impact Evaluation
EC-0000-4684, Revision 2	Inadvertent Opening of an RPV Valve
EC-0000-2749, Revision 2	Loss of Coolant Accident Resulting from a Spectrum of Postulated Piping Breaks
FSAR, Generic Technical Specifications Volume 2: Bases, Draft Revision 4	Tech Spec CNV Peak Pressure CP-1963
TR-0516-49084, Revision 2	Containment Response Analysis Methodology Technical Report

G. DESCRIPTION OF AUDIT ACTIVITIES AND SUMMARY OF OBSERVATIONS

The audit team reviewed supporting calculations and documentation associated with the DCA, Tier 2, Section 6.2, "Containment Response Analyses," and Chapter 15, "Transient and Accident Analyses," which may have been impacted by the revised IAB threshold operating range.

<u>DCA Section 6.2</u>

The main objective of this audit for DCA Section 6.2 was to ensure that NuScale DCA Section 6.2.1.1 and Containment Response Analysis Methodology (CRAM) technical report (TeR) are appropriately revised and are mutually consistent in handling the peak containment pressure (PCP) analysis of record for the limiting design basis event as well as the containment vessel (CNV) design margin. The staff audit of "Chapter 6 IAB Changes CP-1967" and "CNV Technical Report IAB Changes CP-1968" ascertained that the applicant had updated DCA Section 6.2.1.1 to document the 994-psia limiting peak CNV pressure design basis event and assumptions of the staggered inadvertent opening of both reactor recirculation valves (RRVs) at 1,000 psid and three reactor vent valves (RVVs) at 900 psid, reflecting their design tolerance. While the CRAM TeR captured the 986-psia, single-failure PCP variant of the AOO, its audited revisions documented that the limiting 994-psia PCP, staggered RRV-RVV opening variant is described in the DCA Section 6.2.1.1. The staff noted the focus of CRAM TeR Case 5 is on its single-failure, 986-psia variant with inadvertent opening of single RRV, with minimal reference to the 994-psia variant. This approach does capture the minimum details needed to describe the difference between the two variants of the same AOO.

The audit ensured that NuScale had added the Sequence of Events Table (6.2-7) for the limiting peak CNV pressure case of the staggered RRV-RVV opening to address the staff concerns about properly documenting the limiting PCP case in the DCA. NuScale also included the Sequence of Events Table (6.2-8) for the limiting peak CNV wall temperature case in the DCA. The staff ensured that the applicant had included all relevant graphs in the DCA for the internal pressure, wall temperature, break/ECCS mass and energy releases, for the limiting peak CNV pressure and limiting peak CNV wall temperature transients (RCS injection line break LOCA). It was also found that the applicant had analyzed the staggered opening impact on the peak CNV temperature case and found it to be non-limiting. The staff found all the trends of the modified limiting case, e.g. faster initial depressurization, a lower total mass release rate in the PCP limiting case, were reasonable. Later, the staff also independently verified the 994-psia peak containment pressure RELAP5 prediction.

The staff also looked into the 14.7 psia atmospheric pressure that had already been credited to the CNV structure ASME stress analysis that resulted in 1,050 psia CNV design pressure. Discussions with the applicant clarified that crediting the liquid pool hydrostatic head in the containment stress (or dP) analysis as the external pressure boundary condition essentially shifts the point of peak CNV dP from the bottom of the containment to the top. This gives about 7 psi additional margin which is essentially the containment liquid hydrostatic pressure head. This reduced 22 psi additional potential margin documented in the CRAM Revision 1 to 7.3 psi. Auditing EC-A013-2341, Table 5, that was provided to the staff, the staff was able to confirm the 7 psi additional margin to the CNV structure design due to the containment water level hydrostatic pressure head, which NuScale currently does not credit. As a result, NuScale planned to initiate a CR to revise the CRAM references to "22" psi additional available margin and "0" psia CNV external pressure boundary condition for the CNV structure analysis. CNV Technical Report IAB Changes CP-1968 was appropriately updated to reflect the modified hydrostatic head information during the audit.

<u>DCA Section 15.6.5</u>

The NRC staff examined engineering calculation EC-0000-2749, Revisions 1 and 2, to check that the applicant adequately addressed changes in: (1) ECCS CNV level activation setting, (2) IAB block and release parameters (950 +/-50 psi), and (3) potential of staggered release of RRV and RVVs. The staff also questioned the revised methodology to compute core collapsed liquid level. The staff also noticed significant difference in ECCS activation times for cases with all power available between the two revisions. The applicant explained that the difference is due to the increased CNV level setpoint. For cases with loss of all power (alternating current and direct current between the two revisions, staff noticed a slight delay in ECCS activation times for Revision 2, this was due to the decrease in IAB release pressure from 1,000 to 900 psid. The staff observed that consideration of staggered release of ECCS valves did not produce more limiting results for LOCA cases.

The NRC staff reviewed overall conservatism of inputs, and ECCS setpoint and valve input changes. The staff agreed that the inputs appeared appropriate. The staff asked NuScale to provide clarifications regarding the revised methods to compute core collapsed liquid level in the LOCA TR (TR-0516-49422). The staff notes that the applicant's response to questions as to why sensitivities shown in Table 4-11 (RVV1000, RRV 900) and Table 4-12 (RVV900, RRV1000) of Revision 2 contained the same results was unclear. The staff believes any differences would be relatively small but would expect a minor difference in results; therefore, it is acceptable.

The staff also reviewed overall results and trends related to ECCS IAB changes for LOCA Section 15.6.5 and did not find any significant issues.

<u>DCD Section 15.6.6</u>

The NRC staff examined the updated engineering calculation EC-0000-4684, Revision 2. This update resulted in changes to the limiting case (RVV became more limiting than RRV) and limiting minimum critical heat flux ratio (MCHFR) (from 1.41 to 1.32). The staff observed that these changes are not necessarily as a result of changes to the IAB block and release parameters noted above; rather as a result of changes to bounding kinetics and feedback coefficients, RCS flow assumptions, and other changes to the base model. The staff notes that the calculation included appropriate sensitivity studies on gap conductance, axial power shape, DHRS operation, ECCS valve capacity, electrical power availability, valve stroke time, and the potential for staggered ECCS valve release.

The staff also reviewed overall results and trends related to ECCS IAB changes document DCA Section 15.6.6 draft Revision 4 and did not find any issues.

H. EXIT BRIEFING

The staff conducted an audit closing meeting via teleconference on December 18, 2019. During the meeting, the staff reviewed the purpose of the audit, discussed the audit activities, and reviewed major accomplishments. The staff thanked NuScale personnel and indicated that the information was sufficient to address all staff questions or concerns.

I. REQUESTS FOR ADDITIONAL INFORMATION RESULTING FROM AUDIT

No requests for additional information were issued as a result of this audit.

J. OPEN ITEMS AND PROPOSED CLOSURE PATHS

No open items were identified as a result of this audit.

K. REFERENCES

1. NRO-REG-108, “Regulatory Audits,” April 2, 2009 (ADAMS Accession No. ML081910260).

2. NRC Audit Plan for NuScale DCA Chapter 6 and Chapter 15, and Containment Response Analysis Changes Due to Change in the Inadvertent Actuation Block Valve Operating Range (ADAMS Accession No. ML19255F022).