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

February 12, 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 PHASE 4 OF THE REGULATORY AUDIT OF NUSCALE POWER, LLC TOPICAL REPORT TR-0516-49422-P, "LOSS-OF-COOLANT ACCIDENT EVALUATION MODEL"

By letter dated December 30, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No ML17004A202), NuScale Power, LLC (NuScale) submitted Topical Report (TR) TR-0516-49422, "Loss-of-Coolant Accident Evaluation Model," Revision 0, (ADAMS Accession No. ML17004A202) to the U.S. Nuclear Regulatory Commission (NRC) for review and approval. The applicant submitted the TR in support of the design certification application (DCA) for the NuScale Power Small Modular Reactor, which the NRC accepted for review on March 23, 2017 (ADAMS Accession No. ML17074A087). On April 27, 2017, the NRC issued a letter accepting TR-0516-49422-P, Revision 0 for review (ADAMS Accession No. ML17116A063).

The first loss-of-coolant accident (LOCA) audit was started in May 2017, which included Phases 1, 2, and 3, and was completed in March 2018 (ADAMS Accession No. ML20010D112). As NuScale responded to staff's requests for additional information (RAIs), their submittals included draft changes to Revision 1 of TR-0516-49422-P and an expansion of the scope of the LOCA topical report to cover the evaluation model for inadvertent opening of reactor pressure vessel valves and other lesser changes (ADAMS Accession Nos. ML18271A166, ML18264A337, and ML19240C658). Consequently, staff opened a follow-up Phase 4 audit (ADAMS Accession No. ML19008A355) to review documents supporting these changes. The audit entrance meeting was held on January 10, 2019.

CONTACTS: Getachew Tesfaye, NRR/DNRL
301-415-8013

The purpose of Phase 4 was to examine the underlying calculations, analyses, and evaluations supporting several RAI responses as well as Appendix B of the draft revision 1 which was submitted (ADAMS Accession No. ML18264A338) as part of the response to RAI 9536 on September 1, 2018, and several other relevant changes being implemented. In conducting this audit, the staff gained a better understanding of new or changed calculations and information that support the revision of TR-0516-49422-P and review the underlying analyses and reports in support of Appendix B and many other changes, such as updates of the NRELAP5 code from Version 1.3 to 1.4, and changes to emergency core cooling system setpoints.

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

Docket No. 52-048

Enclosure:
As stated

cc w/encl.: DC NuScale Power LLC Listserv

SUBJECT: AUDIT REPORT FOR PHASE 4 OF THE REGULATORY AUDIT OF NUSCALE POWER, LLC TOPICAL REPORT TR-0516-49422-P, “LOSS-OF-COOLANT ACCIDENT EVALUATION MODEL”
DATED: FEBURARY 12, 2020

DISTRIBUTION:
PUBLIC
GTesfaye, NRR
MDudek, NRR
SGreen, NRR
RPatton, NRR
CThurston, NRR
RidsNrrDnrl
RidsNrrDnrlNrlb
RidsNrrDss
RidsEdoMailCenter
RidsAcrsMailCenter
RidsOgcMailCenter
RidsNrrMailCenter
RidsNrrLACSmithResource

ADAMS Accession Nos.:
PKG: ML20034D464
PROPRIETARY: ML20034D492
PUBLIC: ML20042E024 ***via email** **NRR-106**

OFFICE	DNRL/NRLB:PM	DNRL/NRLB:LA	DE/SNRB: BC
NAME	GTesfaye	SGreen*	BPatton*
DATE	01/31/2020	02/03/2020	01/31/2020
OFFICE	DNRL/NRLB: BC	DNRL/NRLB:PM	[redacted]
NAME	MDudek	GTesfaye	[redacted]
DATE	02/11/2020	02/12/2020	[redacted]

OFFICIAL RECORD COPY

AUDIT REPORT FOR PHASE 4 OF THE REGULATORY AUDIT OF
NUSCALE POWER, LLC TOPICAL REPORT TR-0516-49422-P,
"LOSS-OF-COOLANT ACCIDENT EVALUATION MODEL"

1 BACKGROUND

By letter dated December 30, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No ML17004A202), NuScale Power, LLC (NuScale) submitted Topical Report (TR) TR-0516-49422, "Loss-of-Coolant Accident Evaluation Model," Revision 0, (ML17004A202) to the U.S. Nuclear Regulatory Commission (NRC) for review and approval. The applicant submitted the TR in support of the design certification application (DCA) for the NuScale Power Small Modular Reactor, which the NRC accepted for review on March 23, 2017 (ADAMS Accession No. ML17074A087). On April 27, 2017, the NRC issued a letter accepting TR-0516-49422-P, Revision 0 for review (ADAMS Accession No. ML17116A063).

The first loss-of-coolant accident (LOCA) audit was started in May 2017, which included Phases 1, 2, and 3, and was completed in March 2018 (ADAMS Accession No. ML20010D112). As NuScale responded to staff's requests for additional information (RAIs), their submittals included draft changes to Revision 1 of TR-0516-49422-P and an expansion of the scope of the LOCA topical report to cover the evaluation model for inadvertent opening of reactor pressure vessel (RPV) valves and other lesser changes (ADAMS Accession Nos. ML18271A166, ML18264A337, and ML19240C658). Consequently, staff opened a follow-on audit including Phases 4 and 5 (ADAMS Accession No. ML19008A355) to review documents supporting these changes while completing the review. The audit entrance meeting was held on January 10, 2019.

The purpose of Phase 4 more specifically was to examine underlying calculations, analyses, and evaluations supporting several RAI responses as well as Appendix B of the draft Revision 1 which was submitted (ADAMS Accession No. ML18264A338) as part of the response to RAI 9536 on September 1, 2018, and several other relevant changes being implemented. In conducting this audit, the staff gained a better understanding of new or changed calculations and information that support the revision of TR-0516-49422-P and review the underlying analyses and reports in support of Appendix B and many other changes, such as updates of the NRELAP5 code from Version 1.3 to 1.4, and changes to emergency core cooling system (ECCS) setpoints.

During the audit, the NRC staff, assisted by NUMARK contractors, focused on examination of associated reference documents and analyses which supported updates and changes to the analyses, evaluations, and conclusions in the topical report documents, new RAI responses, and supplements to existing RAI responses.

2 REGULATORY AUDIT BASIS

Title 10 of the *Code of Federal Regulations* (CFR), Section 50.46 and Appendix K state the acceptance criteria for ECCS for light-water nuclear power reactors. Regulatory Guide (RG) 1.203, "Transient and Accident Analysis Methods," December 2005, provides guidance for preparing and reviewing computer codes and analysis methods for safety analyses to support licensing applications including DCAs.

10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," requires:

> *(a)(1)(i) that each PWR, fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding, must be equipped with an ECCS. The ECCS must be designed in compliance with certain requirements pertaining ECCS analysis method, ECCS analysis scope, and ECCS performance criteria.*

The following general design criteria (GDC) from 10 CFR Part 50, Appendix A are applicable to the LOCA analysis:

- GDC 5, "Sharing of structures, systems, and components" GDC 10, "Reactor design"
- GDC 13, "Instrumentation and control"
- GDC 15, "Reactor coolant system design"
- GDC 17, "Electric power systems" GDC 20, "Protection system functions"
- GDC 26, "Reactivity control system redundancy and capability"
- GDC 27, "Combined reactivity control systems capability"
- GDC 31, "Fracture prevention of reactor coolant pressure boundary"
- GDC 34, "Residual heat removal"

In addition, relevant regulatory guidance includes:

- NuScale Design-Specific Review Standard (DSRS) 15.6.5, "Loss of Coolant Accident" Revision 0, June 2016.
- Standard Review Plan (SRP) Section 15.0.2, "Review of Transient and Accident Analysis Method," Revision 0, March 2007.
- SRP Section 15.6.5, "Loss of Coolant Accident," Revision 3, March 2007.
- NuScale DSRS 15.0, "Introduction – Transient and Accident Analyses," Revision 0.
- DSRS 6.3, "Emergency Core Cooling System," Revision 0.
- RG 1.203, "Transient and Accident Analysis Methods," December 2005.

3 AUDIT LOCATION AND DATES

The audit was conducted from the NRC headquarters via NuScale's electronic reading room (eRR) and via telephone conferences.

Dates: Phase 4: January 10, 2019, through January 28, 2020

Locations: NRC Headquarters
Two White Flint North
11545 Rockville Pike

Rockville, MD 20852-2738
NuScale
1100 NE Circle Blvd., Suite 200
Corvallis, OR 97330

4 AUDIT TEAM MEMBERS

NRC Staff

Shanlai Lu	Sr. Nuclear Engineer, Lead	(NRR/DSS/SNRB)
Carl Thurston	Nuclear Engineer	(NRR/DSS/SNRB)
Peter Lien	Sr. Reactor Systems Engineer	(RES/DSA/CRAB)
Timothy Drzewiecki	Reactor Systems Engineer	(NRR/DANU/UART)
Antonio J Barrett	Reactor Systems Engineer	(NRR/DANU/UART)
Syed Haider	Nuclear Engineer	(NRR/DSS/SNSB)
Jeffrey Schmidt	Sr. Reactor Systems Engineer	(NRR/DANU/UART)
Rani Franovich	Sr. Project Manager	(NRR/DNRL/NRLB)

Brookhaven National Laboratory

Upendra Singh (Kumar) Rohatgi Senior Scientist

NUMARK Associates, Inc.

Marvin Smith - Project Manager
Bert Dunn - Consultant Technical Expert
Donald Rowe - Consultant Technical Expert
Leonard Ward - Consultant Technical Expert

5 APPLICANT PARTICIPANTS

Brian Wolfe - Code Development Supervisor
Meghan McCloskey - Nuclear Methods Safety Analyst
Eric Coryell - Transient Analysis Supervisor
Morris Byram - Licensing Engineer
Jeffrey Luitjens - Code Development
Zackary Rad - Director, Regulatory Affairs
Selim Kuran - Safety Analysis Engineer
Ben Bristol - Safety Analysis Engineer
Patrick Byfield - Code Development
Pravin Sawant - Nuclear Methods

6 AUDIT DOCUMENTS

The staff audited the following documents provided by NuScale:

LOCA Electronic Reading Room Document Lists	
LOCA Audit Folder – January 2019 (Folder)	
Document Number	Document Title
EC-0000-4909, Revision 1	NRELAP5 Comparison to KATHY NuFuel HTP2TM CHF Data
ECN-0000-6691, Revision 1	ECN: Impact of Condition Reports on the 'NRELAP5 Comparison to KATHY NuFuel HTP2TM CHF Data' (EC-0000-4909 R1)
ECN-T080-6785, Revision 0	ECN: NIST-1 Base Model Correlations, Heater Rod Model Updates, and Addition of Lumped Secondary Side Model
ECN-T080-6833, Revision 0	ECN: HP-06 Assessment Results Using Updated Input Model and NRELAPS, Version 1.4
ECN-T080-6857, Revision 0	ECN: HP-06b Assessment Results Using Updated Input Model and NRELAPS, Version 1.4
ECN-T080-6854, Revision 0	HP-07 Assessment Results Using Updated Input Model and NRELAPS, Version 1.4
ECN-T080-6858, Revision 0	HP-09 Assessment Results Using Updated Input Model and NRELAPS, Version 1.4
EC-T080-6943, Revision 0	Top-down Scaling of NIST-1 Integral Test Facility
EC-0000-3853, Revision 2	Calculations to Support NIST-1 distortion Analysis and Modeling of Containment and Pool Heat Transfer
ECN-0000-6991, Revision 0	ECN: Impact of Condition Reports on the "NRELAPS Comparison to KATHY NuFuel HTP2TM CHF Data
EC-0000-3216, Revision 1	NRELAP5 Comparison to Stern CHF Data
LOCA CHF Slide Presentation	CHF Correlations in NRELAP5 - Predicting Margins during LOCA, IORV
LOCA Slide Presentation	Technical Slides in Support of LOCA Call, May 15, 2019
LOCA Scaling Question Presentation	LOCA Scaling Questions - June 12, 2019, Reference
EC-0000-4487, Revision 1	Comparison of CHFR Calculated by NRELAP5 and VIPRE
EC-0000-4684, Revision 1	Inadvertent Opening of an RPV Valve - With ECN
EC-0000-6108, Revision 0	VIPRE-01 Comparison of Calculations with Crossflow and No Crossflow
09.09.2019.TR-0516-49422 LOCA EM living Revision 1 Draft	9.9.2019 - TR-0516-49422_LOCA EM_living_Rev1_P
EC-0000-4888, Revision 1	NuScale LOCA TR Evaluation Model Supporting Calculations
EC-A030-2359, Revision 5	RCS Flow Form Loss Calculation
EC-0000-7510, Revision 0	CHF Method Comparison
EC-A013-2341, Revision 3	Containment Pressure and Temperature Response to Design Basis Event
TSD-A010-57534, Revision 1	HP-49 ECCS RRV Spurious Opening Test Specification
NP12-00-B020-M-GA-5678, Revision 1	Inadvertent Actuation Block Drawing
EC-0000-5687, Revision 0	LOCA Base Model
ER-0000-3921, Revision 0	Long Term Cooling Methodology Report
EC-T080-4872, Revision 0	NRELAP5 Assessment Against NuScale Integral System CVCS Discharge Line Break Test HP-06b
EC-T080-3820, Revision 0	NRELAP5 Assessment Against NuScale Integral System Pressurizer Spray Line Break Test NIST HP-07-1002

EC-T080-3822, Revision 2	NRELAP5 Assessment Against NuScale Separate Effects high Pressure Condensation Test Series NIST-1 HP-02
EC- T080-4505, Revision 0	NRELAP5 Assessment of Spurious RVV Opening Test NIST-1 HP19a
EC- T080-4506, Revision 0	NRELAP5 Assessment of Spurious RVV Opening Test NIST-1 HP-19b
ER-0000-4738, Revision 1	NRELAP5 Version 1.3 Assessment Report
EC-T080-5045, Revision 0	NRELAP5 Assessment of Spurious RVV Opening Test with 3 RVVs NIST-1 HP-43
SDR-0615-15509, Revision 4	OSU NIST-1 Facility Description Report
EC-A030-2713, Revision 2	Primary and Secondary Steady State Parameters
NP12-01-A010-M-GA-1957-S01, Revision 1	Reactor Module Assembly
NP12-00-B020-M-GA-2650, Revision 1	Reactor Recirculation Valve Drawing
NP12-00-B020-M-GA-2617, Revision 1	Reactor Vent Valve Drawing
EC-A010-1507, Revision 3	System Transient Model Input Parameters Calculation
EC-A010-1782, Revision 1	NuScale NRELAP5 Module Base model
EC-T080-6943, Revision 0	Top-down Scaling of NIST-1 Integral Test Facility
Updated NIST Heater Rod Information_2019.02.28a (August 6, 2019, 6:35 PM CST)	Updated NIST_Heater_Rod_information_2019.02.28a
RRV NIST Testing (Folder)	
EE-T080-5608-Attachment1SS, Revision 0	Attachment 1: NIST-1 Heater Rod Characterization (HP-43 Steady-state)
EE-T080-5608-Attachment2T, Revision 0	Attachment 2: NIST-1 Heater Rod Characterization (HP-43 Transient)
EE-T080-5608-Attachment3-NLT2a, Revision 0	Attachment 3: NIST-1 Heater Rod Characterization (NLT-2a Transient)
ED-A026-4393, Revision 1	Control Rod Assembly Drawings
SDR-0815-16916, Revision 6	Data Acquisition and Control System Configuration Report
NIST-1-HP-49-R3 HP-49, Revision 3	HP-49 ECCS RRV Spurious Opening Test Procedure
TSD-A010-57534, Revision 2	HP-49 ECCS RRV Spurious Opening Test Specification
NCI-0317-53479_Part1	NCI-0317-53479 - Part 1 - Error fixes
NCI-0317-53479_Part2	NCI-0317-53479 - Part 2 - Error fixes
NCI-0317-53479_Part3	NCI-0317-53479 - Part 3 - Error fixes
NCI-0317-53479_Part4	NCI-0317-53479 - Part 4 - Error fixes
NCI-0317-53499	NCI-03174-53499 - Misc. fixes
NCI-0517-54308_Part1	NCI-0517-54308 - Part 1 - Chang CHF fixes
NCI-0517-54308_Part2	NCI-0517-54308 - Part 2 - Chang CHF fixes
NCI-0517-54308_Part3	NCI-0517-54308 - Part 3 - Chang CHF fixes
NCI-0517-54308_Part4	NCI-0517-54308 - Part 4 - Chang CHF fixes
NCI-0617-54727	NCI-0617-54727 - Misc. fixes
NCI-0917-56167_Part1	NCI-0917-56167 - Part 1 - Misc. fixes
NCI-0917-56167_Part2	NCI-0917-56167 - Part 2 - Misc. fixes
NCI-0917-56167_Part3	NCI-0917-56167 - Part 3 - Misc. fixes
NCI-0917-56167_Part4	NCI-0917-56167 - Part 4 - Misc. fixes
NCI-1017-56364_Part1	NCI-1017-56364 - Part 1 - Condensation fixes
NCI-1017-56364_Part2	NCI-1017-56364 - Part 2 - Condensation fixes
NCI-1017-56364_Part3	NCI-1017-56364 - Part 3 - Condensation fixes

NCI-1017-56364_Part4	NCI-1017-56364 - Part 4 - Condensation fixes
NCI-1017-56795	NCI-1017-56795 - Windows Executable
NCI-1017-56796	NCI-1017-56796 - Plotting fixes
NCI-1017-56796_Part1	NCI-1017-56796 - Part 1 - Plotting fixes
NCI-1017-56796_Part2	NCI-1017-56796 - Part 2 - Plotting fixes
NCI-1017-56796_Part3	NCI-1017-56796 - Part 3 - Plotting fixes
NCI-1017-56960	NCI-1017-56960 - Misc. fixes
NCI-1117-57430	NCI-1117-57430 - Misc. fixes
EC-B020-4834, Revision 0	NIST-1 ECCS Orifice Sizing
EE-T080-5608, Revision 0	NIST-1 Heater Rod Characterization
EC-T080-3468, Revision 3	NIST-1 NRELAP5 Base Input Model
EC-T080-6594, Revision 0	NRELAP5 Assessment Against NuScale Loss-of-Feedwater Flow with DHRS Test NIST-NLT-15 Phase 2
EC-T080-6557, Revision 0	NRELAP5 Assessment Against NuScale Loss of Feedwater Flow Test NiST-NLT-02a
EC-T080-4330, Revision 1	NRELAP5 Assessment Against NuScale Loss of Feedwater Flow with DHRS Test NIST-NLT-02b
EC-T080-6620, Revision 0	NRELAP5 Assessment of Spurious RRV Opening Test NIST-1 HP49
SwUM-0304-17111, Revision 5	NRELAP5 Developers Manual
SwTR-0304-17153, Revision 5	NRELAP5 Version 1.4 Acceptance Test Report
SwUM-0304-15495, Revision 5	NRELAP5 Version 1.4 Input Data Requirements
SwRN-0304-54421, Revision 0	NRELAP5 Version 1.4 Software Release Notes
SwUM-0304-17023, Revision 6	NRELAP5 Version 1.4 Theory Manual
EE-T080-13757, Revision 2	NuScale Integral System Test (NIST-1) Facility Scaling Analysis
EE-T080-5098, Revision 0	Pre-Test Prediction of Maximum PZR Spray Line Break Test NIST-1 HP-38
EE-T080-5993, Revision 0	Pre-Test Prediction of Nominal Spurious RRV Opening Test NIST-1 HP-49
NP12-01-A023-M-GA-2305, Revision 2	Reactor Vessel Internals - Core Support
NP12-01-A023-M-GA-2304, Revision 1	Reactor Vessel Internals - Lower Riser

7 DESCRIPTION OF AUDIT ACTIVITIES AND SUMMARY OF OBSERVATIONS

In Phase 4 of the audit, the staff audited information that supported RAI responses related to NuScale's LOCA analysis methodology. During the first three phases, the audit team became familiar with the initial set of information provided in the NuScale eRR (particularly LOCA calculation, EC-0000-4888, Revision 0) and visited NuScale's Integral System Test -1 (NIST-1) facility during a week-long on-site audit. The audit team continued its examination of documents provided in the eRR through completion of Phase 3 of the audit. In Phase 4, the staff focused on audit of:

- NRELAP-5 condensation modeling (RAI 8990)
- NRELAP5 code changes for Version 1.4
- NIST-1 scaling and distortion issues (RAI 9208)
- ECCS system changes (elimination of riser level ECCS trip input)
- NRELAP5 base model changes and their effect on the evaluation model (RAI 9325)

- NIST-1 RRV integral effects test
- HP-49
- NIST-1 NRELAP5 heater rod model changes
- Inadvertent opening of RPV additions to the LOCA TR to extend the EM to cover analysis for Final Safety Analysis Report Chapter 15.6.6
- Critical heat flux (CHF) models for LOCA and inadvertent opening of RPV analyses.

The staff also reviewed the assessment and evaluation of NIST-1 Test HP-49, which was added as Appendix C to the LOCA TR.

The specific observations are documented below.

7.1 NRELAP5 Condensation Modeling

The staff reviewed original and supplemental responses to RAI 8990 (ADAMS Accession Nos. ML17324B392 and ML19240C658) and observed that although Extended Shah is highlighted as the correlation for condensation in the TR, the actual modeling used is based on laminar Nusselt theory correlation, which has much wider application (e.g., encompasses flat plate configuration that is closer to actual NuScale CNV geometry as compared to small pipes). The staff reviewed the code modeling description in the NRELAP5 Version 1.4 Theory Manual "SwUM-0304-17023" to understand the code changes made by NuScale in implementing the Shah 2009 correlation (Reference 2). The staff noted that in regard to application to the CNV wall, this correlation was being used outside of its range of applicability for hydraulic diameter. The data to develop the correlation was based on flow inside small tubes. The staff considered condensation on the CNV walls to be more representative to that of flat plates. The staff then noted that implementation of Shah reverts to the Nusselt correlation in the laminar flow regime, which is Regime 3 (Section 2.6.2.5 of the theory manual). The Nusselt number correlation (developed in 1916) is included in the base version of RELAP5-3D and has a wide range of applicability including flat plates. The staff found this use of the Nusselt correlation to be consistent with calculation results shown in RAI 8990 responses.

Additionally, staff noted that film thickness was conservatively applied, which indicated to the staff that overall the code modeling was reasonable and acceptable. The supplemental response provided for a clearer explanation of how Nusselt correlation is applied in modeling the condensation process during blown and after ECCS actuation in the CNV.

7.2 NRELAP5 Code Version 1.4

NRELAP5 Version 1.3 was initially used to develop the TR and over the course of the staff's review, NuScale identified several modifications, corrections, and system improvements that were implemented into the code to update the version used for the TR to Version 1.4. The staff examined the significant NRELAP5 Change Implementations (NCIs), contained in the audit list, principally NCI-1017-56364, NCI-0317-53479, NCI-0317-56995, and NCI-0517-54308, and the assessment report, SwTR-0304-17153, the software release notes, SwUM-0304-15495, and the theory manual, SwUM-0304-17023. The staff determined that one of the more important changes included correcting the multiplier on the Nusselt heat transfer coefficient, which results in slightly lower peak CNV pressure predictions, NCI-1017-56364. Most other changes were related to improvements that do not affect code results like the 64-bit executable capability

(NCI-0317-56995), changes to CHF correlations not used in the LOCA EM (NCI-0517-54308), and fixes to plotting routines (NCI-1017-56796).

The staff determined that the implementation of code changes and the assessment results and regression validation testing (SwTR-0304-17153) were acceptable.

7.3 NIST-1 Scaling and Distortion Analysis

The staff issued several RAIs (9208, 9390, and 9494) related to NIST-1 scaling and distortion analysis. The staff noted that NuScale report EE-T080-13757, Revision 2, which the staff audited, identified significant distortions as being related to NIST-1 CNV scaling including: (a) flow geometry [[ ]], (b) 3-D core flow and heat transfer pattern in the NPM versus 1-D in NIST-1, (c) limitation on NIST-1 secondary pressure, and (d) [[ ]].

It appeared to staff that the applicant only partially addressed some distortions and did not address the inter-relationship among the significant distortions clearly. The staff was concerned that the existing distortion reports, which the staff audited, may not adequately capture all the high-ranking important phenomena in the PIRT, nor correctly quantify the distortions to determine the uncertainty in the figure of merit (FOM) predictions. The staff believed this could cause an over prediction of condensation in the NPM.

The staff reviewed the applicant's responses and determined that these distortions did not significantly affect the key figures of merit. The staff audited NuScale's modified scaling analysis EC-T080-6943 and noted that the methods and results were appropriate.

7.4 ECCS System Changes

7.4.1 Elimination of Riser Level ECCS Trip Input

LOCA LTR (TR-0516-49422-P), Table A.3 describes low RPV riser level range as one of two level trips to actuate ECCS. Based on Heat Removal System (DHRS) and other control system changes, NuScale determined that the RCS level trip was not needed and would be eliminated. NuScale submitted those design changes to NRC on April 15, 2019 (ADAMS Accession No. ML19105B292). Revision 1 of the TR now includes a footnote to acknowledge that the RCS level trip was not removed from the modeling used to generate the TR results (EC-0000-4888, Rev. 1), but it was not activated in any of the analyses and it was not used in the final analysis modeling in the Design Certification Application (DCA) Part 2, Tier 2 analysis results in Section 15.6.5.

7.4.2 IAB Actuation Changes

Based on valve testing, NuScale determined that IAB actuation setpoints for the ECCS valve IABs (ECN-B020-7507) should change. The design change specified that the IABs would block at delta (CNV- reactor coolant system (RCS)) pressures of 1300 psid or greater and release at delta pressures of 950 psid +/- 50 psi. Revision 1 of the TR has a footnote added to Table A.3 to acknowledge that old release trip setpoints were not removed from the modeling used to generate TR results (EC-0000-4888, Rev. 1), but the correct activation setpoints are used in the final analysis modeling in the DCA Part 2, Tier 2 analysis results in Section 15.6.5.

7.4.3 CNV Level Setpoint Changes

The LOCA LTR (TR-0516-49422-P), Table A.3 also describes high CNV level setpoint as 260-220 inches and based on DHRS and other control system changes, NuScale determined that the CNV level trip would increase to 282 +/-18 inches. NuScale submitted those design changes to NRC on April 15, 2019 (ADAMS Accession No. ML19105B292). Revision 1 of the TR has a footnote added to Table A.3 to acknowledge that old trip setpoints were not removed from the modeling used to generate TR results (EC-0000-4888, Rev. 1), but the correct activation setpoints were used in the final analysis modeling in the DCA Part 2, Tier 2 analysis results reported in Section 15.6.5.

7.5 <u>NRELAP5 Base Model Changes</u>

The staff noted that the NRELAP5 ANSYS solids model and parameter input document (EC-A010-1507, Revision 3) was out of sequence with the base model document (EC-A010-1782). The staff was subsequently informed that base model (EC-A010-1782) and the LOCA TR calculation document (EC-0000-4888) were under revision. Revision 1 of the NRELAP5 base model included a total re-nodalization of the model previously developed. The staff audited the calculation model revision to understand the effect of changes made and staff noted that NuScale replaced the modeling inputs in their entirety.

Overall staff observed that changes were modest with noted modifications in: (1) spillover [[ ]] at top of riser and changes in SG noding, (2) RCS form losses used, (3) feedwater piping size corrections, (4) HCSG tubesheet inlet flow restriction loss increase, (5) material of CNV lower head and cylindrical portion material changed to SA-965, (6) upper head, not modeled in Revision 0, was added and upper material type remained carbon steel clad by stainless, and (7) downcomer and riser collapse liquid calculation (CVARs 130 and 133) was changed to volume-based level calculations.

The staff audited details regarding NuScale's use of a volume-based collapse liquid level (CLL) calculation used in Revision 1 of the base model versus the axial-based CLL calculation used in Revision 0. The Revision 1 calculation appeared to show unrealistically higher levels in the core compared to the liquid level in the downcomer. The staff audited additional information made available by NuScale to clarify the issue. The staff examined this information and determined that the revised method departs from traditional conservative axial methods in that it credits liquid in the upper riser. NuScale subsequently described this method in Revision 1 of the LOCA TR. The staff noted that since the methodology specifies that both the CLL and MCHFR criteria must be met, and the MCHFR correlations were adequately conservative to protect against core heat up, that the methodology as a whole remains conservative even if the level

criteria, when considered in isolation, may not adequately protect against core heat-up due to the volume-based method NuScale uses to calculate CLL.

7.6 NIST-1 Heater Rod Model Changes

On Page 217 of TR Revision 0, NuScale stated that the heater rods are not in contact with water because they are placed in thermal-wells. During the Phase 2 Corvallis on-site audit, NuScale explained that the thermal-wells are metal cladding with high thermal conductivity material completely filling in the gap between the cladding and heater rod. The heater rods simulate nuclear fuel by providing steady state power to initiate transients and programmed decay heat power for post-trip core response.

However, staff noted in its audit that for the follow-on 2018 NIST-1 Test HP-49 assessments (ADAMS Accession Nos. ML18177A087, ML19308A058, and EC-T080-6620), NuScale revised the NRELAP5 heater rod modeling, which significantly increased the thermal resistance and thereby the stored energy in each rod. The NIST-1 RPV core is made up of electrically heated rods, some of which are fixed with internal thermocouples. The rods each contain a heater element that is inserted into a thermowell where the 0.005” gap is filled with boron nitride to maintain good transfer with the heater element and keep heater element temperatures low. Staff noted that in the base NRELAP5 NIST-1 modeling, the heater rod input neglected the boron nitride coating since its thermal properties were very similar to the stainless steel thermowell sleeve. With this approach (that is used for all previous test assessments), the NRELAP5 computed rod center temperatures were 600 to 650 °F, and the NIST-1 testing program had not acquired any center line temperature data to check these values.

Based on assessments for NIST-1 test NLT-2, the applicant suspected that rod center temperatures (i.e., rod stored energy) were higher than considered in the current NRELAP5 modeling, so for the test HP-43, NuScale collected rod center line data, which the staff examined. The staff noted that the data indicated a significantly higher centerline temperature, which implied that there were likely air gaps and movement (relocation) of the boron nitrate due to rod heat-up and cooldown cycles resulting in considerably higher thermal resistance in this gap layer.

The staff also examined the applicant’s rod characterization studies (EE-T080-5608) and noted the applicant found a rather wide variation of percentage air in the gap that would best match the rod centerline temperature data. The staff also examined the applicant’s revised and updated base NIST-1 NRELAP5 model (EC-T080-3468), which included a uniformly applied rod model with a set air gap for all NIST-1 tests that significantly increased initial rod centerline temperature and, consequently, core stored energy. The staff observed that the increase in stored energy caused a substantial increase in the peak pressure prediction, approximately 50-60 psi higher than computed with the original rod modeling.

In the staff’s audit of HP-43 (TD-1216-524740) and HP-49 test data (TD-0718-6600), the staff found an inconsistency where the applicant used the maximum of data rather than average temperature. The staff considers that NRELAP5 Rod Model inputs should be based on average temperature where data is available and that conservative (lower temperature) inputs should be used for previous earlier tests where data was not collected. Since the rod modeling has an overwhelmingly dominant effect on pressure test results, staff was concerned that the rod model changes could have been masking other phenomena in the applicant’s assessment. Additionally, since the applicant’s analysis indicated wide variability, that implied to the staff that

more conservative inputs should be used since it is likely that the boron nitride layer eroded progressively over time as more tests were completed. This suggested to the staff that lower temperatures would be more realistic, especially for the earliest tests.

Separate from its audit activities, the staff determined via its own sensitivity studies that differences in the results with a lower realistic initial temperature were minimal and not large enough to affect the conclusions of the assessment. The staff's conclusion and finding relative to this issue will be documented in the SER for the LOCA TR.

7.7 NIST-1 RRV Integral Effects Test Results, HP-49

The staff noted that the NIST-1 LOCA tests modeled several RVV opening events from full nominal power and pressure conditions, but did not address an RRV event. The staff subsequently issued RAIs 8985 and 9549 requesting the applicant to provide justification as to why high-ranked PIRT phenomena with high ranked uncertainty specific to an RRV opening were not validated by NIST-1 tests. The applicant added an RRV test, which was designated as NIST-1 HP-49. The test was performed in June 2018, and the NRC staff were present to witness the test (ADAMS Accession No. ML18177A087). As indicated above, with the revised heater rod modeling, the CNV peak pressure predictions improved significantly. The staff completed its review of the test data and test assessment results for HP-49 and found them acceptable.

The staff also reviewed NuScale's re-analyses of HP-06, HP-06b, HP-07, HP-09 and HP-43 test assessments with the revised rod modeling and found them to be acceptable.

7.8 LOCA EM Extension to Address Inadvertent Opening of RPVs

The staff reviewed the applicant's implementation of the Hench-Levy and modified Zuber CHF correlations (for heat transfer options 171). The staff also noted that NuScale made code updates for option 170 (Hench-Levy and Chang CHF correlations) for NRELAP5 version 1.4. The staff audited these changes in the review of NCI updates and in ER-0000-2413 "CHF Modeling from Stern's Data." The staff noted that option 170 is not used for LOCA or the inadvertent opening of an RPV valve (IORV) analysis, so it has no impact on the analysis results.

7.9 CHF Models

Critical Heat Flux (High-Flow)

The NRC staff examined engineering calculation EC-0000-4909, Revision 1, "NRELAP5 Comparison to KATHY NuFuelTM CHF Data." During this examination, the staff noted the purpose of the calculation, identified key inputs and their sources, and noted the key results. In particular, the staff noted:

- The purpose of this report is to compare the NuFuel HTP2™ CHF data from KATHY Laboratories in Karlstein, Germany, with the performance predicted by NRELAP5 using CHF correlations implemented in Version 1.3 of the code.

- The statistical analysis used the subregion approach described in TR-0116-21012, “NuScale Power Critical Heat Flux Correlations,” Revision 1.

- NRELAP5 analyses are performed using the heat structure right boundary type 171, which is described in TR-0516-49422.

- NRELAP5 analyses are performed as follows:

    - Analyses are performed [[

      ]].

    - The NRELAP5 model of the KATHY NuFuel HTP2™ test sections are modeled [[

      ]].

    - The NRELAP5 calculations [[

      ]].

    - The Experimental Power Ratio (EPR) is used to validate the CHF correlation and is defined as [[

      ]]

- The CHF data obtained from the KATHY test loop that correspond to the pressure, inlet subcooling, and flow ranges identified in Table 1 were used to validate the [[
  ]] (hereafter referred to as the high-flow CHF correlation).

Table 1. Range limits for KATHY CHF data used to Validate the High-Flow CHF Correlation

Parameter	Min	Max
Pressure	[[ ]]	[[ ]]
Inlet Subcooling	[[ ]]	[[ ]]
Mass Flux	[[ ]]	[[ ]]

- CHF correlation performance is investigated [[

]].

- A plot of predicted-to-measured CHF values shows that the NRELAP5 analyses tend to conservatively under-predict CHF values.

- The CHFR limit is established as follows:

  - The KATHY data was partitioned into subsets. These subsets are combined into the composite subsets identified in Table 2, using the statistical tests described in TR-0116-21012, “NuScale Power Critical Heat Flux Correlations,” Revision 1. The resulting limits for each subset are provided in Table 3.

Table 2. Composite Subsets of NuFuel-HTP2™ Data Used to set CHFR Limit

Subset	**Test ID**	**Pressure MPa**	**Mass Flux kg/s-m^2**	**Inlet Subcooling K**	**Exit Quality**
1	[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]], [[ ]], [[ ]]
2	[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]], [[ ]]
3	[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]], [[ ]]
4	[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]]

Table 3. CHFR Limit from Composite Subsets of NuFuel-HTP2™ Data

Subset Basis	**Composite Subset**	**Type**	**Limit**
Test ID	[[ ]] [[ ]]	[[ ]] [[ ]]	[[ ]] [[ ]]
Pressure	[[ ]] [[ ]] [[ ]] [[ ]]	[[ ]] [[ ]] [[ ]] [[ ]]	[[ ]] [[ ]] [[ ]] [[ ]]
Mass Flux	[[ ]]	[[ ]]	[[ ]]
Inlet Subcooling	[[ ]] [[ ]]	[[ ]] [[ ]]	[[ ]] [[ ]]
Exit Quality	[[ ]], [[ ]] [[ ]], [[ ]] [[ ]]	[[ ]] [[ ]] [[ ]]	[[ ]] [[ ]] [[ ]]
Overall Maximum			**1.06**

- An investigation into non-conservative subregions is performed by plotting all of the points on a three-dimensional plot and identifying the data points that exhibited the lowest margin to CHF (bottom 3 percent). The NRC staff notes that this is the same type of plot that was submitted to NRC in response to RAI 8931, Question 04.04-6 (ADAMS Accession No. ML17268A385), which was issued as part of the review of

TR-0116-21012. This plot showed that the points of lowest CHF margin were spread through the application domain and not concentrated in a particular region.

- The ranges of applicability for the High-Flow CHF correlation are established by the range of data used to establish the CHFR limit using the KATHY data and are provided in Table 4.

Table 4. Range of Applicability for the High Flow CHF Correlation using KATHY data

Parameter	Range of Applicability
Pressure	[[ ]]
Inlet Mass Flux	[[ ]]
Exit Quality	[[ ]]
Inlet Quality	[[ ]]

The NRC staff examined engineering change notice ECN-0000-6991, Revision 1, "Impact of Condition Reports on the 'NRELAP5 Comparison to KATHY NuFuel HTP2TM CHF Data' (EC-0000-4909 R1)." During this examination the staff noted the purpose of the document, identified key inputs and their sources, and verified the key results. In particular, the staff noted:

- The purpose of this document is to address document changes due to (1) errors discovered as part of the corrective action process, and (2) develop the necessary document changes associated with NRC requests for additional information. This document also contains changes to the high-flow CHFR limit.

- All KATHY cases supporting the high-flow CHF correlation are rerun using NRELAP5 v1.4 to address CR-0119-64068. The results of these calculations are:

    - CR-0119-64068 [[ ]]. There were no changes in the output calculation between NRELAP v1.3 and v1.4 for the [[ ]] series of tests.

    - CR-0119-64068 [[ ]]. The impact of this change is less than 1 CHFR point for the [[ ]] series of tests.

- The high-flow CHFR limit is reevaluated to address CR-0119-64241. This reevaluation resulted in the following changes to EC-0000-4909, Revision 1. The staff notes that there is no change to the range of applicability documented in Table 4.

    - The composite subsets from Table 2 are updated to Table 5. The resulting CHFR limits for each subset are updated from Table 3 to Table 6 for an overall maximum CHFR limit of 1.05.

- The three-dimensional plot used to investigate non-conservative subregions was recreated. The plot continues to show that the points of lowest CHF margin are spread through the application domain and not concentration in a particular region.

Table 5. Updated Composite Subsets of NuFuel-HTP2™ Data Used to set CHFR Limit

Subset	Test ID	Pressure MPa	Mass Flux $kg/s\text{-}m^2$	Inlet Subcooling K	Exit Quality
1	[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]], [[ ]]
2	[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]], [[ ]], [[ ]]
3	[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]], [[ ]]

Table 6. Updated CHFR Limit from Composite Subsets of NuFuel-HTP2™ Data

Subset Basis	Composite Subset	Type	Limit
Test ID	[[ ]] [[ ]]	[[ ]] [[ ]]	[[ ]] [[ ]]
Pressure	[[ ]] [[ ]]	[[ ]] [[ ]]	[[ ]] [[ ]]
Mass Flux	[[ ]]	[[ ]]	[[ ]]
Inlet Subcooling	[[ ]] [[ ]]	[[ ]] [[ ]]	[[ ]] [[ ]]
Exit Quality	[[ ]], [[ ]] [[ ]],[[ ]], [[ ]] [[ ]], [[ ]]	[[ ]] [[ ]] [[ ]]	[[ ]] [[ ]] [[ ]]
Overall Maximum			**1.05**

- ECN-0000-6991, Revision 1 includes an update to EC-0000-4909, Revision 1 to provide the database used to validate the high-flow CHF correlation.

Critical Heat Flux (Low-Flow)

The NRC staff examined engineering calculation EC-0000-3216, Revision 1, “NRELAP5 Comparison to Stern CHF Data.” During this examination the staff noted the purpose of the calculation, identified key inputs and their sources, and verified the key results. In particular, the staff noted:

- The purpose of this report is to compare the CHF data from Stern Laboratories with the performance predicted by NRELAP5 Version 1.3 of the code.

- NRELAP5 CHF analyses of the testing at Stern Laboratories is performed by:

- [[ ]].
- [[ ]].
- [[ ]].

- This calculation, EC-0000-3216, Revision 1, investigates the use of the several NRELAP heat structure options including:
  - [[ ]].
  - [[ ]].

- The modeling in NRELAP5 uses [[ ]].

- The CHF data obtained from Stern Laboratories that correspond to the pressure, inlet subcooling, and flow ranges identified in Table 1 are also used in EC-0000-3216, Revision 1 to validate the high-flow CHF correlation.

- Data obtained from Stern Laboratories at flow rates of [[ ]] are used to validate the low-flow CHF correlation.

- Sensitivity studies were performed to investigate the impact of [[ ]].
  - [[ ]].
  - [[ ]].

- [[

  ]].

- [[

  ]]. The results from the [[ ]], which corresponds to the NRELAP5 model described in Section 5.1.2.2.1 of TR-0516-49422, "Loss-of-Coolant Accident Evaluation Model," are provided in Table 7.

Table 7. Low-Flow CHFR Values Calculated from Stern Data

Test	No. of Points	Mean	Std.	Max.
[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]]
[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]]
[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]]

- CHF correlation performance is investigated [[
  ]].
  - [[

    ]].
  - [[

    ]].
  - [[
    ]].

8 EXIT BRIEFING

The staff conducted an audit closure meeting at NuScale's Rockville, MD office on January 28, 2020. 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 close and resolve the issues identified.

9 REQUESTS FOR ADDITIONAL INFORMATION RESULTING FROM AUDIT

Not applicable.

10 OPEN ITEMS AND PROPOSED CLOSURE PATHS

Not applicable.

11 DEVIATIONS FROM THE AUDIT PLAN

The audit was originally scheduled to exit following the original audit end date of July 31, 2019, but was extended to January 28, 2020, to accommodate the examination and discussion of additional documentation requested by the NRC staff.

12 REFERENCES

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

2. M. Mohammed Shah, "An Improved and Extended General Correlation for Heat Transfer During Condensation in Plain Tubes," HVAC&R Research, vol. 15, no. 5, September 2009.