U.S. NUCLEAR REGULATORY COMMISSION

SAFETY EVALUATION REPORT FOR TOPICAL REPORT TR-0516-49422, REVISION 2, May 2020

"LOSS-OF-COOLANT ACCIDENT EVALUATION MODEL"

Enclosure 1

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1.0 INTRODUCTION AND BACKGROUND

Under the NRC Topical Report (TR) Program and by letter dated December 30, 2016, the applicant, NuScale Power, LLC (NuScale) submitted TR 0516 49422-P, “Loss-of-Coolant Accident Evaluation Model,” Revision 0, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17004A202), to the U.S. Nuclear Regulatory Commission (NRC) staff for review. The applicant supplemented its submittal by letter dated March 7, 2017 (ML17066A463). By letter dated April 27, 2017 (ML17116A063), the NRC informed NuScale of its acceptance of TR 0516 49422-P, Revision 0, for a detailed technical review. On November 27, 2019, (ML19331B585), NuScale submitted Revision 1 to TR 0516 49422-P and on May 27, 2020, (ML20148T471), NuScale Submitted Revision 2 to TR-0516-49422-P (hereafter referred to as the loss-of-coolant accident [LOCA] TR). By letter dated June 19, 2020, NuScale supplemented TR-0516-49422, Revision 2, (ML20175A345).

TR 0516 49422-P, “Loss-of-Coolant Accident Evaluation Model,” Revision 2, presents the NuScale’s evaluation model (EM) used to evaluate emergency core cooling systems’ (ECCS) performance in the NuScale Nuclear Power Module (NPM) for design basis LOCAs.

Additionally, Appendix B of TR-0516-49422, provides a description of a modified version of the LOCA EM that is used to evaluate the Inadvertent Opening of a Reactor Pressure Vessel (RPV) Valve (IORV) event. Section 5 includes details regarding the thermal limits evaluation during low flow, stagnation, and reverse flow, which occur during LOCAs and LOCA-like events. Specifically, it provides the bases to support: (1) the applicability of the **[[** **]]** critical heat flux (CHF) correlations (hereafter referred to as the high-flow and low-flow CHF correlations, respectively) for the analysis of the NPM, (2) the minimum CHF ratio (MCHFR) limit for each event, and (3) the range of applicability of these correlations. The TR also discusses the interfaces to the other analyses that assess the acceptance criteria not evaluated by the LOCA EM.

After the LOCA EM was developed, there was a design change to the NPM to ensure acceptable boron distribution during passive ECCS and decay heat removal system (DHRS) cooling modes. This included the addition of holes to the RPV riser and the addition of low RCS pressure emergency core cooling system actuation. Additionally, the TR was updated to describe the ECCS valves opening on low differential pressure between the RPV and CNV and the removal of the RPV level ECCS actuation. These changes were addressed in Revision 2 to TR-0516-49422-P.

NuScale stated that the LOCA EM was developed following the guidelines in the EM development and assessment process (EMDAP) of “Transient and Accident Analysis Methods,” Regulatory Guide (RG) 1.203 and that this model adheres to the applicable requirements under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix K, “ECCS Evaluation Models,” and 10 CFR Section 50.46, “Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors.” NuScale stated that multiple layers of conservatism are incorporated in the NuScale LOCA EM to ensure that a conservative analysis result is obtained. These conservatisms stem from an application of the modeling requirements of 10 CFR Part 50, Appendix K, and through a series of conservative modeling assumptions and modeling inputs.

This Safety Evaluation Report (SER) documents the results of the NRC staff’s in-depth technical evaluation of TR-0516-49422-P, “Loss-of-Coolant Accident Evaluation Model,” Revision 2, and the NuScale EM used to evaluate the ECCS performance in the NuScale NPM. The NRC staff performed a review to determine the technical applicability of the thermal hydraulic methods and modeling techniques as described in TR-0516-49422-P, for evaluating ECCS core cooling performance for LOCA and LOCA-like events.

The applicant developed the NuScale LOCA Evaluation Methodology to evaluate ECCS performance for the NuScale NPM. The NuScale design is a small modular reactor designed to be deployed with up to 12 NPMs at a specific site. Each NPM is a light-water, integral pressurized water reactor (PWR) that is enclosed by a high-pressure containment vessel (CNV) immersed in a reactor pool coupled with passive safety-related ECCS. The NPM is designed to shut down and cooldown in the event of a LOCA. Each NPM has an independent nuclear steam supply system (NSSS) that includes a nuclear core, helical-coil steam generator (SG), integral pressurizer, strategically placed ECCS valves, and a compact, high-pressure steel CNV that contains the NSSS. Each NPM has a secondary system that includes a traditional steam-power conversion system including a steam turbine generator, condenser, and feedwater system. The integral small PWR design does not have large reactor system piping found in conventional PWRs, therefore the number and size of pipe ruptures that would result in a LOCA are significantly reduced. The NuScale LOCA EM evaluates potential breaks in the reactor coolant system (RCS) injection line, RCS discharge line, pressurizer spray supply line, and pressurizer high point vent line. The RCS injection line is supplied by the chemical and volume control system (CVCS) and the discharge line returns to the CVCS. In addition, the applicant extended the EM to evaluate the design basis events resulting from an IORV. During normal operation, flow through the reactor is driven by natural circulation resulting from the thermal driving head produced by the temperature difference between the core and the heat sink afforded by the SGs. Natural circulation flow increases reliability that ECCS will successfully initiate recirculation flow by eliminating primary coolant pumps that can fail or lock up. NuScale designed the NPM so that there is no core uncovery or heatup for a design-basis LOCA.

2.0 REGULATORY BASIS FOR LOCA EM TOPICAL REVIEW

The NRC staff has reviewed the LOCA EM described in TR-0516-49422-P, Revision 2, entitled "Loss-of-Coolant Accident Evaluation Model," to determine whether this methodology is acceptable for performing LOCA analyses and IORV analyses. The NRC staff also reviewed the basis for applying the NRELAP5 code to predict certain highly-ranked phenomena that govern peak containment pressure analyses. The methodology for calculating the peak containment pressure and temperature performance is contained in the Containment Response Analysis Methodology (CRAM), TR-0516-49084, Revision 2 (ML19330F387), which is an extension of the LOCA TR methodology. This section of the SER describes the regulatory basis and supporting regulatory and guidance documents that the NRC staff uses to determine whether the methodology described in LOCA EM TR-0516-49422-P, Revision 2, is acceptable for LOCA and IORV analyses of the NuScale NPM design.

2.1 Regulatory Requirements

The requirements under 10 CFR 50.46 and 10 CFR Part 50, Appendix K, present the acceptance criteria for ECCS for light water nuclear power reactors and the required and acceptable features of the EMs employed. The NuScale LOCA EM is based on meeting the conservative 10 CFR part 50, Appendix K rule per 10 CFR 50.46(a)(1)(ii).

2.1.1 10 CFR 50.46 ECCS and Appendix K to 10 CFR Part 50 Requirements

The ECCS Rule at 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," requires in 10 CFR 50.46(a)(1)(i) that each PWR fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding must be equipped with an ECCS and that ECCS performance must be evaluated for the most severe postulated accident.

ECCS Analysis Method

The regulations at 10 CFR 50.46(a)(1)(i) requires that "ECCS cooling performance must be calculated in accordance with an acceptable evaluation model," and provides for two options, as mentioned above, for acceptable EM analytical techniques and methods: realistic or conservative.

Accordingly, 10 CFR 50.46(a)(1)(ii) describes an EM of the second category as a method that conservatively describes the behavior of the reactor system during a loss-of-coolant accident. and that such an EM "may be developed in conformance with the required and acceptable features" of Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50.

Furthermore, 10 CFR 50.46(c)(2) defines an EM as the calculational framework for evaluating the behavior of the reactor system during a postulated LOCA. An EM includes one or more computer programs and all other information necessary for applying the calculational framework to a specific LOCA (the mathematical models used, the assumptions included in the programs, the procedure for treating the program input and output information, the parts of the analysis not included in the computer programs, values of parameters, and all other information necessary to specify the calculational procedure).

10 CFR Part 50 Appendix K

The regulation at 10 CFR 50.46(a)(1)(ii) provides that an EM may be developed in conformance with the required and acceptable features that include specific conditions to be met as defined by Appendix K to 10 CFR Part 50.

Since the NuScale EM does not present any LOCA predictions that produce core uncovery and exposure of the active fuel region to steam cooling, neither fuel cladding damage or cladding oxidation is calculated, eliminating the potential of several 50.46 criteria from being exceeded. As such, the NuScale LOCA TR includes the provision that, "A feature "excluded" from the EM means that 10 CFR 50, Appendix K, directly requires the feature, without condition on the presence of a process or phenomena, but that the feature is not relevant to the NuScale LOCA EM. Table 2-2 technically justifies the exclusion of such feature from the model. However, an applicant or licensee referencing this report will be required to address regulatory compliance with 10 CFR 50.46 and 10 CFR 50, Appendix K (e.g., by seeking an exemption from that required feature)."

As stated in the applicability section of the TR, the review of the NuScale EM does not apply to conditions where the liquid level recedes below the top elevation of the core active fuel region. For this reason, the NuScale EM does not contain post-CHF clad damage models and metal water reaction methodologies that would normally accompany a PWR LOCA TR submitted for the NRC staff's review. However, should the predicted liquid level calculated using the NuScale

EM ever recede below the top elevation of the core, those conditions are beyond the capability of this methodology and modification to this EM would need to be submitted for the NRC staff's review and approval.

These modifications would include, but are not be limited to, post-CHF heat transfer models, fuel pin models that incorporates clad swelling, rupture and, oxidation, and calculation of the metal-water reaction rate using the Baker-Just Correlation.

Part II, "Required Documentation," of Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50, sets forth the EM documentation requirements for the required analyses as well as the need for additional sensitivity studies and comparisons of the EM to experimental data.

The regulation at 10 CFR 50.46(a)(1)(i), requires, in part, that "ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated."

ECCS Performance Criteria

The regulation at 10 CFR 50.46(a)(1)(i), requires, in part, that the ECCS calculated cooling performance following postulated LOCAs conforms to the criteria set forth in 10 CFR 50.46(b). This regulation defines the criteria for the calculated ECCS cooling performance during postulated LOCAs in 10 CFR 50.46(b)(1) through 10 CFR 50.46(b)(5) as follows:

(1) Peak Cladding Temperature.

The calculated maximum fuel element cladding temperature shall not exceed 2,200 degrees Fahrenheit (ºF) (1,477.59 K or 1,204.44 degrees Celsius (ºC)).

(2) Maximum Cladding Oxidation.

The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. This is based on the Baker-Just equation.

(3) Maximum Hydrogen Generation.

The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(4) Coolable Geometry.

Calculated changes in core geometry shall be such that the core remains amenable to cooling.

(5) Long-Term Cooling.

After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

2.1.2 10 CFR 50, Appendix A, “General Design Criteria for Nuclear Power Plants”

In TR-0516-49422, the applicant requests approval for two CHF models to be used in accordance with the analysis methodologies described in the TR. The CHF correlations, and their respective limits, are used to evaluate whether fuel cladding integrity is maintained during LOCA and LOCA-like events. Thus, approved CHF correlations and associated methodologies are used to establish a partial basis for compliance with the following general design criteria (GDC) in 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants.”

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

- GDC 19, “Control Room,” and 10 CFR 52.47(a)(2)(iv) as they relate to the evaluation and analysis of the radiological consequences from postulated accidents.

- GDC 35, “Emergency Core Cooling,” as it relates to demonstrating that the ECCS would provide abundant emergency core cooling to satisfy the ECCS safety function of transferring heat from the reactor core following any loss of reactor coolant at a rate that: (1) fuel and clad damage that could interfere with continued effective core cooling would be prevented, and (2) clad metal-water reaction would be limited to negligible amounts.

2.2 Regulatory Guide 1.203

RG 1.203, “Transient and Accident Analysis Methods,” provides guidance for developing and evaluating EMs for accident and transient analyses. Section D, “Implementation,” states that the guide is approved for use as an acceptable means of complying with the NRC regulations and for evaluating submittals of “new or modified EMs proposed by vendors or operating reactor licensees that, in accordance with 10 CFR 50.59, require NRC staffs review and approval.”

The LOCA EM is a deterministic analysis approach NuScale developed considering the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K. The LOCA TR states that the approach to the development of the model follows RG 1.203, “Transient and Accident Analysis Methods.” Within RG 1.203, the Phenomena Identification and Ranking Table (PIRT), is identified as a key requirement for EM development. Section 4 of the NuScale LOCA EM TR documents the PIRT that NuScale developed for the NPM. Section 4.4 of this SER provides the NRC staff’s review of this PIRT.

2.2.1 Evaluation Model Concept

In accordance with 10 CFR 50.46(c)(2), RG 1.203 states that the EM constitutes the calculational framework for evaluating the behavior of the reactor system during a postulated transient or a design-basis accident. As such, the EM may include one or more computer programs, special models, and all other information needed to apply the calculational framework to a specific event, such as procedures for treating the input and output information, specification of those portions of the analysis not included in the computer programs for which alternative approaches are used, or all other information needed to specify the calculational procedure. It is the entirety of an EM that ultimately determines whether the results comply with applicable regulations and therefore the development, assessment, and review processes must consider the entire EM. Most EMs used to analyze the events in SRP Chapter 15, “Transient and Accident Analysis,” rely on a systems code that describes the transport of fluid mass, momentum, and energy throughout the RCSs. The LOCA EM uses the NuScale NRELAP5 systems analysis computer code, which is developed from the Idaho National Laboratory (INL) RELAP5-3D computer code.

2.2.2 Evaluation Model Development and Assessment Principles

RG 1.203 defines the following six basic principles as important to follow in the EMDAP:

(1) Determine requirements for the EM.
(2) Develop an assessment base consistent with the determined requirements.
(3) Develop the EM.
(4) Assess the adequacy of the EM.
(5) Follow an appropriate quality assurance (QA) protocol during the EMDAP.
(6) Provide comprehensive, accurate, up-to-date documentation.

RG 1.203 discusses the NRC staff’s regulatory position, which provides guidance concerning methods for calculating transient and accident behavior. Part C of RG 1.203, provides guidance on aspects of an EMDAP that are related to the basic principles identified above and offers additional guidance.

Regulatory Position 1, EM Development and Assessment Process (EMDAP)

RG 1.203 identifies four basic elements developed to describe an EMDAP. The elements correspond to the first four EMDAP basic principles and provide guidance in twenty individual steps. In addition, Regulatory Position 1 includes requirements for reaching an adequacy decision. The basic elements of Regulatory Position 1 are identified below.

Element 1: Establish Requirements for EM Capability
Element 2: Develop Assessment Base
Element 3: Develop EM
Element 4: Assess EM Adequacy Decision

Regulatory Position 2, Quality Assurance

RG 1.203 discusses QA during development, assessment, and application of an EM and the requirements of Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” to 10 CFR Part 50.

Regulatory Position 3, Documentation

RG 1.203 provides guidance on the requirements to document the development of LOCA EMs.

Regulatory Position 4, General Purpose Computer Programs

RG 1.203 provides guidance on development of general-purpose transient analysis computer programs designed to analyze a number of different events for a wide variety of plants. Specifically, Regulatory Position 4 states that "application of the EMDAP should be considered as a prerequisite before submitting a general-purpose transient analysis computer program for review as the basis for EMs that may be used for a variety of plant and accident types."

Regulatory Position 5, Graded Approach to Applying the EMDAP Process

RG 1.203 provides guidance on the extent to which the full EMDAP should be applied for a specific application based on the following four EM attributes:

(1) Novelty of the revised EM compared to currently accepted models.
(2) Complexity of the event being analyzed.
(3) Degree of conservatism in the EM.
(4) Extent of any plant design or operational changes that would require reanalysis.

Appendix A of RG 1.203, "Additional Considerations in the Use of this RG for ECCS Analysis," describes uncertainty determination and provides guidance for best-estimate LOCA analyses. Appendix A of RG 1.203 refers to NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (SRP), Sections 15.6.5, "Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary," and 15.0.2, "Review of Transient and Accident Analysis Method."

2.3 NUREG-0800 Standard Review Plan

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (SRP), Section 15.0.2, "Review of Transient and Accident Analysis Methods," is the companion SRP section for RG 1.203.

SRP Section 15.6.5, "Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary," Revision 3, describes the review scope, acceptance criteria, review procedures, and findings relevant to ECCS analyses.

3.0 NUSCALE LOCA EVALUATION METHODOLOGY SUMMARY

The NuScale Power Module has several unique features that required the NRC staff to perform detailed reviews of the NuScale LOCA EM to determine whether this methodology is adequate. The NuScale design is a small modular PWR that relies on natural circulation during normal plant operation and uses a unique high-pressure containment as an integral part of the ECCS to keep the reactor core covered with the collapsed liquid level (CLL) above the top of the active core through all potential LOCA events, as shown in NuScale LOCA TR Figures 3-1, "A singular NuScale Power Module during normal operation," and 3-2, "Schematic of NuScale Power Module decay heat removal system and emergency core cooling system during operation."

During a NuScale NPM LOCA, the high-pressure water and steam leaving the RPV is contained in the CNV. The CNV has a design pressure of 1050 pounds per square inch absolute (psia) and is designed to enable the ECCS system to return cooled RCS liquid to the downcomer to prevent core uncovery during design basis LOCAs. During a LOCA, the five ECCS valves, three Reactor Vent valves (RVVs) and two Reactor Recirculation Valves (RRVs), receive a signal to open. However, the valves are blocked from opening by the Inadvertent Actuation Block (IAB) Valve until the pressure differential between the RPV and CNV drops below the IAB threshold. Once these valves open, the RPV and CNV pressures equalize within about 30 seconds. After this pressure equalization, steam generated inside the RPV from decay heat and stored energy, exits the RPV through the RVV, condenses on the inside of the CNV wall, and is returned from the CNV to the RPV through the RRVs.

Because of the unique features of the NuScale NPM containment (CNV) design and the NuScale ECCS system, the NRC staff's review of the NuScale LOCA EM TR focused particular attention on the ability of the NuScale LOCA EM to assess the following design issues and phenomena:

- The capability to predict the CLL in the RPV so that the NuScale power module maintains the CLL in the RPV above the reactor core during the design basis event of a LOCA.
- The capability to predict Critical Heat Flux Ratio (CHFR) so that the NuScale power module maintains the CHF margin during the design basis event of a LOCA.
- The applicability of NRELAP5 computer code to perform peak containment pressure so that the CNV of NuScale power module absorbs heat energy at a rate sufficient to maintain CNV pressure within design limits and to transfer heat energy from the RPV to the water pool outside the CNV during the design basis event of a LOCA
- Ensuring that the LOCA pipe break spectrum methodology included all susceptible RPV penetrations.

In addition, the NRC staff's review of the EM TR focused particular attention on the capability of the NuScale NRELAP5 computer code to accurately model the tests performed at the NPM scaled model NuScale Integral System Test Facility (NIST-1) facility and to confirm that the geometric dimensions and operating conditions of NIST-1, adequately represent the NPM full plant.

Because the NuScale design relies on maintaining a CLL above the top of the reactor core, the NRC staff's evaluation of the NuScale LOCA Evaluation Methodology is limited to the consideration of the conservative assumptions and modelling assumptions to determine that this design objective is adequately modeled. The determination to support the Design Certification that the CLL remains above the top of the core, is documented as part of the review of the NuScale Design Certification Application.

The NRELAP5 computer code, Version 1.4 (ML17066A463 and ML19162A086), was submitted as the systems analysis computer code for the NuScale LOCA Evaluation Methodology. NuScale's primary changes to the INL RELAP5-3D version included implementation of a new helical-coil SG (HCSG) component and the addition of new containment condensation models to describe the unique design features of the ECCS operation of the NPM.

In addition to the use of NRELAP5 for evaluating LOCAs, Appendix B of the LOCA TR extends the EM and methodology to analyze events described in SRP Section 15.6.1, "Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve," and SRP Section 15.6.6, "Inadvertent Operation of the Emergency Core Cooling System (ECCS) event." These inadvertent RPV valve events are classified as AOOs so the acceptance criteria are also slightly different than the Section 15.6.5 LOCA acceptance criteria. The event progression is essentially that of a LOCA which results in blowdown of the RCS inventory into the CNV and can be a steam region release or liquid space discharge.

4.0 TECHNICAL EVALUATION

This section of the SER summarizes and evaluates the information in each section of the TR against the regulatory requirements for that section. The Limitations and Conditions on the review of TR-0516-49422-P, are discussed in detail below and summarized in Section 6. The conclusions from the review are discussed in detail below and summarized in Section 7 of this SER.

In addition, the NRC staff conducted audits of information provided by the applicant in support of the NRC staff's review of the TR. These audits are referenced throughout this SER. Unless otherwise noted, details of these audits are available in audit reports at ADAMS Accession Nos. ML20010D112, ML20034D464, and ML20160A250, which provide summaries of the audits and the information examined during them.

4.1 Introduction and Scope

Section 1.1 of TR-0516-49422-P, states that the purpose of the NuScale EM is to evaluate ECCS performance in the NPM for design-basis LOCAs and requests approval to use the methodology to perform such analyses. NuScale stated that its LOCA EM follows the guidance provided in "Transient and Accident Analysis Methods," RG 1.203 and satisfies the applicable requirements of "ECCS Evaluation Models," 10 CFR Part 50, Appendix K. TR-0516-49422-P provides a description of the methodology used by NuScale for LOCA analyses and this methodology is reviewed in this SER for compliance with applicable regulatory criteria. However, TR-0516-49422-P does not provide any final licensing analyses and this review of TR-0516-49422-P does not evaluate the acceptability of the NuScale NPM or provide any conclusions on the acceptability of the NuScale design.

Further, the LOCA TR provides support for other analyses including:

1. events as described in TR-0516-49416-P, "Non-Loss of Coolant Accident Methodology";

2. containment peak pressure analysis as described in Technical Report TR-0516-49084-P, "Containment Response Analysis Methodology";

3. long term cooling as described in Technical Report, TR-0919-51299-P, "Long-Term Cooling Methodology"; and

4. IORV valves, including ECCS valves as described in Appendix B of the LOCA TR, "Evaluation Model for Inadvertent Opening of RPV Valves."

4.2 Background

Section 2 of the NuScale TR provides a description of how the NuScale LOCA EM conforms with the EMDAP guidance in RG 1.203. Additionally, NuScale stated that other provisions of

RG 1.203 related to establishing an appropriate QA program (QAP) and providing comprehensive, accurate, up-to-date documentation are described outside of the LOCA EM TR. QA requirements are included in “NuScale Topical Report: Quality Assurance Program Description for the NuScale Power Plant,” NP-TR-1010-859-NP-A. The NRC staff reviewed the QA requirements and documented its approval in its SER (ML16347A405). Further, the NRC staff inspected NuScale’s design control process and code development procedures. These inspections are documented in inspection reports dated October 7, 2017 (ML15268A186) and July 24, 2017 (ML17201J382).

NuScale indicated that the NPM is designed to reduce the consequences of design-basis LOCAs compared to existing PWRs for which 10 CFR Part 50, Appendix K was developed. Consequently, many of the phenomena that are the subject of 10 CFR Part 50, Appendix K requirements are not encountered in design-basis NPM event LOCAs, meaning that these phenomena have been eliminated by design, therefore, a number of the Appendix K requirements are satisfied by design rather than by analysis. NuScale has therefore limited the methodology to only pre-CHF heat transfer regimes and specified that an applicant or licensee seeking to reference the LOCA TR, must demonstrate regulatory compliance for these Appendix K requirements, which could include seeking an exemption. The requirements are included in Table 2-2, “10 CFR 50 Appendix K required and acceptable features compliance,” of the TR. This is reflected in Section 6.0, Limitations and Conditions, of this report. The staff’s review in this SER is therefore limited to pre-CHF heat transfer regimes.

The NuScale LOCA EM TR provides a summary of the NRELAP5 code modifications and modeling features added by NuScale to address the unique features and phenomena of the NPM design and states that the EM is consistent with the applicable requirements of 10 CFR Part 50, Appendix K and the Three Mile Island (TMI) Action Items applicable to the NuScale NPM as described in the Design-Specific Review Standard for NuScale, Section 15.6.5. The NRC staff’s review of the CHF correlations used in NRELAP5 are contained in Section 4.6.

4.3 NuScale Power Module Description and Operations

Section 3 of the NuScale LOCA EM TR provides a brief description of the NPM and a brief summary of NPM operation.

4.3.1 General Plant Design

Features of the NuScale plant design that are unique compared with existing operating PWR plants include:

* Reduced reactor core size.
* Natural circulation reactor coolant flow (i.e., no reactor coolant pumps).
* An integrated HCSG and a pressurizer inside the RPV that eliminates piping to connect the SG or pressurizer with the reactor.
* A safety-related ECCS system that does not require electrical power and does not use ECCS pumps.

* Primary fluid in the SGs flow on the outside of the tube surface, and two-phase flow of the secondary flow contained inside of the tubes.

* A high-pressure steel CNV immersed in a water-filled pool that is integral to the ECCS capability to provide for emergency cooling.

In TR Section 3.1, “General Plant Design,” NuScale describes how the NPM uses natural circulation to provide reactor core coolant flow without electrically power reactor coolant pumps. This section of the LOCA TR also describes the design of the NPM HCSG. As discussed in the LOCA TR, each NPM has a dedicated ECCS, CVCS, and DHRS. The NRC staff reviewed the summary of its general plant design in TR-0516-49422-P and found that it provided sufficient description of the design to support the methodology description.

4.3.2 Plant Operation

In TR Section 3.2, NuScale provides a brief description of NPM operation including systems modeled in the LOCA evaluation methodology. The NRC staff reviewed the plant operation summary in TR-0516-49422-P and found that it provided a sufficient description to support the methodology description.

4.3.3 Safety-Related System Operation

In Section 3.3 of the TR, NuScale describes operation of the safety-related systems and components, including the ECCS, the NuScale Module Protection System, DHRS and the containment isolation valves. The NRC staff reviewed the safety-related system operation summary in TR-0516-49422-P and found that it provided a sufficient description to support the methodology description.

The ECCS is a two-phase natural circulation system that is designed to maintain a water supply to the core during its operation in a LOCA scenario. The RPV and CNV geometry is designed such that ECCS actuation results in a CLL in the RPV that is generally significantly above the top of the core. The ECCS is actuated on high CNV level or low RCS pressure interlocked with RCS hot temperature and CNV pressure or at 24 hours with loss of AC power. If the ECCS is not already open by previous mechanisms, there is also a low differential RPV to CNV pressure feature due to the valve spring that can open the valves at about 15 psid.

The DHRS is a passive safety-related system that uses boiling condensation loop flow to remove heat from the RCS through the SG and reject heat to the reactor pool through the DHRS condensers. The DHRS is composed of two DHRS trains associated with the NPM SG and each train is designed with the capability to independently remove 100 percent of decay heat.

4.4 Phenomena Identification and Ranking

As discussed in Section 4 of the NuScale LOCA EM TR, NuScale developed the NPM PIRT in stages. NuScale developed its original PIRT in 2008 and updated this PIRT in 2013 and 2015. NuScale used the 2015 final PIRT as the basis for the presentation given in Chapter 4 of the LOCA TR. However, NuScale only documented phenomena and processes of high importance in Section 4 of the NuScale LOCA EM TR. Therefore, the NRC staff reviewed both the LOCA

TR and the NuScale 2015 PIRT. The 2015 NuScale PIRT provided the rankings for all four PIRT importance categories (high, medium, low, and inactive).

NuScale's first step in the PIRT development was to select the panel to support the PIRT review and to examine the qualifications of the PIRT board members to assure that they were qualified. NuScale's second step of PIRT development was to obtain an agreement between the NuScale staff and the PIRT panel on the accident scenarios and figures of merit (FOMs) identified by NuScale and the PIRT panels. NuScale's third step was to review of each of the phenomena and processes identified and ranked in the PIRT to determine the approximate fidelity of the rating assigned. In this process, NuScale did not require unanimous agreement on the reasons given for the PIRT ranking. However, NuScale did use a process to assure that no phenomena or process of high importance was missing and that the rankings of medium and low importance were reasonable.

The NRC staff reviewed the PIRT panel membership and the qualifications of the NuScale 2008, 2013 and 2015 PIRT panels as provided in Section 4 of the LOCA EM TR and the 2015 PIRT report that the staff audited (ML20010D112). NuScale provided the list of PIRT panel members in the LOCA TR. However, NuScale only provided panel member qualifications for the 2015 PIRT panel in the 2015 report. Based both on the NRC staff's knowledge of the panel members listed and the NRC staff's examination of the qualifications provided in the 2015 report, the NRC staff confirmed that all panel members are highly regarded members of the nuclear community with extensive experience in the industry, a research institution, or nuclear academia.

4.4.1 NuScale Loss-of-Coolant Accident Scenarios

NuScale discussed LOCA accident scenarios in LOCA EM TR Section 4.2, which it states are consistent with 10 CFR 50.46(c)(1). NuScale stated that it considered breaks of various sizes, types, and orientations in piping connected to the RPV. Because the NuScale NPM design eliminates most primary coolant piping, breaks are limited to the RCS injection and discharge lines, the pressurizer spray supply line, and the pressurizer high point vent line. **[[**

]] The TR provides a description of the progression of each LOCA scenario and divides the scenarios into two phases: LOCA blowdown (1a) and ECCS actuation to the time when stable long-term recirculation flow is established (1b).

The NRC staff evaluated the LOCA scenarios selected for the PIRT discussions and determined that the only LOCAs that can occur are those for penetrations of the RPV that pass into or through the CNV and originate within the RCS. These penetrations are few and small in cross-sectional area. Therefore, the NRC staff agreed that the large break LOCA scenarios for conventional pressurized reactor designs that circulate reactor coolant through large pipes outside of the RPV, are not applicable for the NuScale NPM.

The NRC staff found that NuScale's PIRT phenomenon selection of steam breaks from **[[**

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The NRC staff notes that for the NuScale design, breaks from the three RVV nozzles and two RRV flanges are excluded in this TR as break locations and are the subject of a separate evaluation in the DCD, Chapter 3, "Design of Structures, Systems, Components and Equipment," regarding their exclusion from consideration as break locations. Additionally, NuScale has evaluated inadvertent openings of an RRV or RVV as AOOs. The NRC staff's review of the NRELAP5 code for evaluation of an inadvertent opening of an RRV or RVV is discussed in Section 5 of this SER.

The NRC staff determined that the NuScale LOCA accident scenarios selected as the basis for their PIRT process are acceptable for establishing the ranking phenomena that must be considered in the LOCA EM because they consider the applicable features of the design.

4.4.2 Figures of Merit

In TR Section 4.3, NuScale discusses the FOMs selected for its LOCA EM, which are primarily CHFR and CLL above the top of active fuel (TAF). According to NuScale, the NPM retains sufficient water in the RPV such that the core will not be uncovered during any LOCA scenario. Therefore, peak clad temperature is not a FOM for the NuScale PIRT process. Instead, the CHFR is an important FOM used to demonstrate that the fuel clad does not reach the point of CHF where significant heat up of the cladding could occur. NuScale also states that maintaining the CLL above the core is an additional LOCA analysis FOM as it demonstrates that there is an adequate supply of liquid water available to preclude CHF in the core. As a result of the scenarios selected, the FOMs for the PIRT considerations are the fuel rod CHF value and the requirement to maintain CLL above the top of the active core fuel.

The NRC staff finds for the NuScale LOCA EM FOMs that: (1) the core fuel rods do not experience CHF and that (2) the CLL in the RPV remains above the core at all times during all LOCA scenarios, shows conservatism, and are acceptable FOMs for the NuScale LOCA EM, particularly in light of the fact that the EM is limited to pre-CHF heat transfer because these FOMs are more restrictive than that required by 10 CFR 50.46(b).

Because the NuScale LOCA EM includes only pre-CHF phenomena, credible LOCA break scenarios must not produce core uncovery or thermal-hydraulic conditions which result in exceeding the CHF limits. The NRC staff has determined that the validity of the NuScale LOCA EM model is limited to LOCA analyses that do not reach core uncovery and where the heat flux for fuel cladding remains below the CHFR limit. This Limitation is reflected in Section 6.0, “Limitations and Conditions,” of this SER.

NuScale stated that to ensure ECCS performance, CNV must be maintained and intact during all postulated accident scenarios. Therefore, the CNV must be kept below CNV design pressure and temperature design limits to ensure compliance with 10 CFR 50.46 criteria. However, NuScale stated that the limiting peak containment pressure and temperature are calculated with a different methodology from that described in the LOCA EM TR. NuScale developed methodology for ensuring that containment integrity is presented in TR-0516-49084-P, “Containment Response Analysis Methodology Technical Report” (ML19330F387).

Since the NuScale LOCA EM depends on showing that no credible LOCA scenario or break can result in a loss of containment integrity, the NRC staff has determined that the validity of the NuScale LOCA EM is limited to LOCA analyses that result in CNV temperature and pressure that remain below respective design limits for all LOCA events as required for the ECCS system to maintain sufficient liquid water in the CNV to ensure that the CLL remains above the top of the reactor core. This is reflected in Section 6.0, “Limitations and Conditions,” of this SER.

4.4.3 PIRT Rankings

Sections 4.4 through 4.7, of the LOCA EM TR discuss the results of the NuScale LOCA EM PIRT process and provide a list of High-Ranked Phenomena and the Phenomena Identification and Ranking Summary Table.

The NuScale PIRT panel identified phenomena and processes that could occur during a NPM LOCA, ranked relative importance of each, and assessed the knowledge level on each. Relative importance was ranked as High, Medium, Low, or inactive (not present or negligible). Knowledge level was divided into well known, known, partially known, or very limited. Finally, the portion of the NPM for which the phenomena or process was ranked, was identified.

In LOCA EM Table 4-4 "High-Ranked phenomena," NuScale provided the listing of the findings of its final PIRT for phenomena ranked of high importance as to position within the system where the phenomena are important, timing of importance during the accident, and knowledge level. NuScale assessed only the high ranked phenomena and processes in its LOCA EM TR. To assess medium to low ranked phenomena, the NRC staff reviewed the NuScale 2015 PIRT report. The NRC staff's review did not identify any low to medium ranked phenomena or processes that should have been ranked high. Therefore, the NRC staff finds that the NuScale LOCA EM table for high ranking phenomena, is acceptable.

The NRC staff also reviewed TR Section 4.6.1 of the LOCA TR, "Discussion of Phenomena Ranked High Importance." The NRC staff found that the rationale for the rankings in Table 4-4 is reasonable and appropriate but not always comprehensive. For example, the first entry in Section 4.6.1 **[[**

]] The NRC staff agrees that mass and energy release are a key factor but not the only key factor. The NRC staff found that the rate of mass and energy release is more important than the total amount of release in the determination of the CNV pressure and the flow changes in the RCS. However, several key important highly ranked phenomena included **[[**

]]. Staff subsequently agreed that assessment of rate is implied in the assessment of choke and unchoked flow. Therefore, the NRC staff finds that the conclusions and the PIRT phenomena selections and knowledge level rankings are appropriate as a basis for the NuScale LOCA EM.

4.5 Evaluation Model Description

The NRC staff reviewed the NuScale LOCA EM description provided in Section 5 of the NuScale LOCA EM TR to determine whether the analysis model described in Section 5 is suitable for performing LOCA safety analysis with NRELAP5. Section 5 describes the NPM nodalization and modeling input options selected by NuScale for each NPM component and provides NuScale's rationale for each choice. Section 5 also provides NuScale's justification for the boundary and initial conditions selected by NuScale, for the model. In addition, NuScale described the LOCA break spectrum selected by NuScale. NuScale stated that its NPM LOCA modeling used, is consistent with the Separate Effects Tests (SET) and Integral Effects Tests

(IET) assessments used by NuScale to validate the NRELAP5 code for its application to LOCA and Non-LOCA analysis.

4.5.1 NRELAP5 Loss-of-Coolant Accident Model for the NuScale Power Module

In LOCA EM TR Section 5.1, NuScale stated that its unique design features of the NPM allowed NuScale to use a simplified modeling approach to predict and evaluate consequences of postulated LOCAs.

The NRC staff's review of the NuScale LOCA EM is based on these key NuScale design assumptions. In Section 5, NuScale stated that in the event of a LOCA, these unique NPM design features result in a simple, predictable transient progression, that can be explained by a standard mass and energy balance over the RPV and CNV considering:

- Choked and unchoked flow through the break and then ECCS flow via valves between RPV and CNV,
- Core decay heat generation and RCS stored energy release, and
- Heat transfer between the CNV and the reactor pool that is characterized by steam condensation at the CNV inside surface and free convection at the CNV outside surface to the reactor pool.

The NRC staff reviewed the adequacy of the NRELAP5 modeling of these design features and determined that the modeling approach is adequate to evaluate the FOM. The NRC staff found that NPM modeling developed adequately represent the key components and the key phenomena expected to occur during a LOCA.

NuScale Power Module NRELAP5 Model

The NuScale LOCA EM covers key components of the NPM participating in a LOCA. These key components include the following:

1. RPV with internals:
    a. Lower plenum
    b. Reactor core
    c. Riser including the riser upper plenum
    d. Upper and lower downcomer
    e. Pressurizer
2. Containment (CNV)
3. SG secondary side
4. Reactor pool
5. ECCS valves
6. Postulated break locations

7. RPV internal heat structures and heat structures between components (i.e., RPV to the CNV to the reactor pool).

8. Riser holes located at approximately the midpoint of the SG (note that these holes were not incorporated into the models used in producing the example results described in the TR, but is in the model the staff is approving for use in this evaluation through limitation and condition 5 in Section 6 of this report. This is addressed later in this section of the report)

The nodalization diagram of these key components is shown in Figure 5-1, "Noding diagram of NRELAP5 loss-of-coolant accident input model for NuScale Power Module," of the LOCA EM TR.

In LOCA EM TR Section 5.1.1, "General Model Nodalization," NuScale stated that the NRELAP5 RCS noding was developed to provide appropriate resolution of fluid volumes as a function of elevation to account for natural circulation flow during NPM operations and to calculate the draining of fluids into the lower RCS volumes and circulation between the containment, the downcomer, and the core when the ECCS system is activated later in the LOCA.

The NRC staff finds that the LOCA model adequately represents the important components and phenomena required for evaluating LOCA scenarios for the NPM.

Section 5.1.2, "Reactor Coolant System," of the LOCA TR describes the modeling of each of the RCS components. The NRC staff's findings relative to the modeling of those components is described below.

Downcomer, Lower Plenum and Riser

The NRC staff reviewed the description of the modeling of these three regions and finds that the model is acceptable because the model correctly preserved the volume, elevation changes in these three regions and incorporates conservative loss factors to maximize core bypass flow, the flow from the downcomer and into the reactor core.

After the TR was initially developed, a design change added holes to the RPV riser. This design change was not incorporated into the model used in the examples in this TR and is not included in the noding diagram in Figure 5-1. However, the staff determined that these holes would not significantly impact the example calculations used to demonstrate an acceptable EM. Additionally, limitation and condition 5 in Section 6 of this report requires the use of NPM model Revision 3, which includes the riser holes.

Reactor Core Model

The NuScale RELAP5 model uses three axial channels in the reactor core to calculate reactor coolant flow through the core, including hot, average and bypass channels.

During development of the LOCA EM, NuScale recognized that flow reversal may occur within the core bypass at low or stagnant flow conditions. Therefore, NuScale selected sufficient axial nodes to account for the hydrostatic head in these three parallel reactor core channels. NuScale modeled the unheated sections of the hot and average core channels with single nodes below and above the core heated length. The core model includes form losses for top and bottom nozzles and grid spacers with appropriate hydrodynamic volumes based on fuel vendor data. NuScale modeled the core flow channels with individual NRELAP5 PIPE components. The crossflow between hot and average fuel channels was not credited to conservatively prevent hot and average channel fluid streams from mixing, which the NRC staff considers conservative.

The NPM model sets the **[[**

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For LOCA analysis, NuScale assumed that actual core power is 102 percent of rated the core power to account for uncertainty in measured power. The NPM core model assumes that the axial and radial power distribution is at the maximum limits set for core operation.

The NRC staff agrees that the assumption of 102 percent core power along with a power distribution set to the maximum allowed for plant operation reasonably assures that the maximum core operating power prior to a LOCA is conservatively modeled, which complies with the requirements of 10 CFR Part 50, Appendix K.

NuScale models the reactor pressure and CNV metal components as passive heat structures with an appropriate distribution of mass and heat transfer surfaces.

The NRC staff reviewed this basic modelling technique and finds it is acceptable because both RCS pressure vessel metal mass is conservatively modeled, and the conduction heat transfer is appropriately captured.

The NPM reactor model accounts for fission power due to prompt and delayed neutrons using a point kinetics approach. The model simulates reactivity changes due to reactor trip, fuel temperature changes (Doppler coefficient), and moderation changes (moderator density and temperature). The NuScale model sets the moderator and Doppler coefficients for minimum negative worth-based beginning of cycle (BOC) burnup and the reactivity feedback conditions that evolve during the LOCA. The scram rod worth is appropriately delayed for trip and insertion times and accounts for the most reactive rod stuck outside the core.

The point kinetics core model divides delayed neutrons into six precursors groups, and calculates core decay heat in accordance with the 1973 American Nuclear Society (ANS) standard which the NRC staff reviewed and determined was consistent with the 1971 standard that complies with the regulatory requirements of 10 CFR 50.46, Appendix K. The NRC staff determined that this modeling approach meets the regulatory requirements for LOCA analysis.

Pressurizer

The NRC staff reviewed the description of the modeling of the Pressurizer and finds that the Pressurizer model is acceptable because the model properly accounts for the fluid volume, elevation changes and the metal stored energy. It also models the connection of three RVVS, the pressurizer baffle plate, the pressurizer heaters, and the steam plenums interface.

Helical Coil Steam Generators (HCSG)

Section 5.1.3, “Helical Coil Steam Generators,” of the NuScale TR describes how NuScale models the two HCSGs within the NPM that is wrapped within the entire cylindrical cross-section of the upper cold side of the RCS loop using a newly developed NRELAP5 SG component model to capture the performance of this unique SG design. The NPM model includes noding required to calculate the heat transfer from the reactor coolant into the HCSG tubes. The HCSG model includes equivalent noding for the secondary coolant inside the HCSG tubes. The flow of the two-phase secondary coolant travels from the lower feedwater supply, into tubes to boil, and then up to the steam headers where it exits as superheated steam, is modeled.

During a LOCA, the SGs are isolated after reactor trip from the remainder of the secondary system but remain active via the DHRS which receives steam from the generators and transfers heat to the pool region, outside of the CNV, by condensation and returns the condensate to the SG. This would provide an additional means of removing decay heating from the core and controlling the core conditions. However, NuScale has chosen not to take credit for the DHRS in the LOCA EM. Thus, for the analysis of LOCAs and the inadvertent opening of ECCS valves, the SGs are isolated with stagnant tubes where heat transfer is then limited to the downcomer region only.

Although the energy transferred to or from the SGs is accounted for in the NRELAP5 LOCA evaluations, the initial operating temperature of the SGs is still high. The most significant factor for the HCSG, is the resistance to flow during early portion of the LOCA, as this influences the heat removal capability in the core. The HCSG secondary side impact on a LOCA, is the heat transfer capability. NuScale has tested both of these factors in prototype testing and incorporated the results of the testing into the new NRELAP5 SG component. The NRC staff notes that the primary impact of the HCSG is in the early phase of a LOCA. The detailed review of the NuScale Società Informazioni Esperienze Termoidrauliche (SIET) TF1 test validation is documented in the NRC staff’s SER of the Non-LOCA EM TR (ML20042E039).

The NRC staff finds that the NuScale approach to HCSG modelling is conservative for both initial steady state and transient analysis. The NRC staff agreed that since the role of the HCSGs are minor in the LOCA transients, the effects of tube-plugging and fouling are also negligible.

Containment Vessel and Reactor Pool

Section 5.1.4, “Containment Vessel and Reactor Pool,” of the NuScale TR describes how the NuScale LOCA model represents the CNV with **[[**

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Chemical and Volume Control System

The LOCA TR describes the modeling of the CVCS in Section 5.1.5, "Chemical and Volume Control System." Within the CNV, the CVCS system is comprised of small pipes connected to the RPV riser section for supply and the RPV downcomer section for letdown. NuScale stated that **[[**

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The NRC staff finds that the NuScale break spectrum conservatively bounds any LOCA associated with the CVCS due to the consideration of LOCAs in the CVCS piping, the CVCS isolation function, and the fact the model neglects water injected prior to isolation, and therefore, accepts this model for the CVCS.

Secondary System

The LOCA TR describes the modeling of the secondary system in Section 5.1.6. NuScale stated that the SG secondary side is **[[**

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The NRC staff finds that the NuScale LOCA model treatment of the secondary system is acceptable for LOCA evaluations because it treated secondary side energy contribution conservatively.

Decay Heat Removal System

The LOCA TR describes the modeling of the DHRS in Section 5.1.7. The NuScale DHRS design includes Isolation valves, closed during normal operation, that open upon activation of the DHRS. When these DHRS isolation valves open, steam generated in the HCSG tubes by heat transfer from the RCS is condensed in the DHRS heat exchanger by condensation on tubes cooled by ultimate heat sink pool.

For its LOCA EM, NuScale conservatively takes no credit for the DHRS. To justify the NuScale assumption that operation of the DHRS would not have an adverse impact on LOCA results, NuScale performed a break spectrum analysis for which the DHRS was active. The NuScale analysis of the active and inactive DHRS is presented in Section 9.3, "Decay Heat Removal System Availability," of the LOCA EM TR. The results show that the collapsed water level is not significantly impacted by assuming that the DHRS is active for larger breaks. For smaller breaks, operation of the DHRS prevents re-pressurization of the NPM. The NRC staff finds that the NuScale assumption that the DHRS is inactive, is a conservative assumption for the LOCA EM and is therefore, acceptable.

NRELAP5 Modeling Options

The NuScale NPM LOCA analysis is performed with Version 1.4 of NRELAP5. NRELAP5 uses the 'h2o95' water property table for all systems of the LOCA model. The NRELAP5 code has a default feature of termination of the transient if a system mass error exceeds one percent. NRELAP5 uses air as the only non-condensable gas for the partially evacuated CNV and NRELAP5 uses air at normal air pressure above the reactor pool water surface. Because these water property tables and air property are part of previously approved RELAP5 code features, the NRC staff finds these assumptions to be acceptable.

JUNCTION OPTIONS

NuScale provided a list of the junction options selected for its LOCA EM in Table 5-1, "Default junction options for the NRELAP5 loss-of-coolant accident model," of the LOCA EM TR. NuScale used **[[**

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The NRC staff finds that the NuScale junction options selection options as shown in Table 5-1, are acceptable because they properly model the fluid flow area and local loss coefficients and have added Moody critical flow models, which comply with the regulatory requirements in 10 CFR Part 50, Appendix K to model choke flow.

VOLUME OPTIONS

NuScale documented its selection of the volume options for its LOCA EM, in LOCA EM TR Section 5.1.8.2, "Volume Options." The NRC staff finds that these modeling options are acceptable for this application because they properly model the fluid interphase friction and wall friction consistent with the applicable regulatory requirements in 10 CFR Part 50, Appendix K to model break flow phenomena and friction pressure drops.

HEAT STRUCTURE OPTIONS

NuScale discussed its selection of the heat structure options for its LOCA EM, in LOCA EM TR Section 5.1.8.3, "Heat Structure Options."

The NRC staff notes that NuScale has added several boundary condition types to model unique aspects of the NPM and the NRC staff finds that this heat structure treatment is acceptable for the NuScale LOCA EM because it appropriately identifies that the options related to the FOM being CHF and CLL. In addition, the heat structure components modeled conservatively, accounts for the metal mass, the sensible heat and heat conduction during a LOCA transient.

The NRC staff notes that NRELAP5 v1.4 includes Option 170 with use of the **[[ ]]** for fuel rod CHF. However, the LOCA EM specifies use of Option 171 with use of the **[[ ]]**. As such, the NRC staff did not review and does not approve of use of Option 170.

Time Step Size Control

NuScale discussed NRELAP5 time step control in LOCA EM TR Section 5.1.9, "Time Step Size Control." NuScale stated that a sensitivity study was performed to demonstrate that the selected maximum time-step size has no significant impact on the LOCA FOM such as peak containment pressure and CLL in the RPV riser. The NRC staff audited this sensitivity study, as described in the associated audit report (ML20010D112), and finds that the analysis results are not sensitive to the time step size range. Therefore, the NRELAP5 time step selection process using **[[ ]]** is reasonable.

4.5.2 Analysis Setpoints and Trips

Section 5.2, "Analysis Setpoints and Trips," of the LOCA EM TR discusses and lists the system trips that NuScale incorporated in the LOCA EM. NuScale stated that signals that are not credited either do not play a role in a LOCA or provide conservatism by delaying actuation of safety-related systems that only reduce the consequences of a LOCA.

The NRC staff finds that the NuScale approach for selecting and modeling analysis setpoints and trips in their LOCA EM is acceptable based on the NRC staff's review of the trip set points included in the TR, which are the ones that are necessary for appropriately modeling the LOCA events in a conservative manner. Evaluation of the instrumentation is performed under the Design Certification review and is not a subject of this SER.

After the TR was developed, a design change added a low RCS pressure ECCS actuation and removed an RPV level actuation. These changes were not incorporated into the model used in the examples in this TR. However, the staff determined that these changes to the ECCS setpoints would not significantly impact the example calculations used to demonstrate an acceptable EM. Additionally, limitation and condition 5 in Section 6 of this report requires the use of NPM model Revision 3 which includes the updated ECCS setpoint logic .

4.5.3 Initial Plant Conditions

In Table 5-6, "Plant initial conditions," of the LOCA TR, NuScale listed initial plant conditions, including core power, RCS temperature and pressure, pressurizer level, CNV pressure, secondary system pressures and temperatures, and the initial level and temperature for the reactor pool (ultimate heat sink). NuScale states that these initial plant conditions are conservatively biased for LOCA analysis and that the plant conditions are selected to account for both the normal control system deadband and the system/sensor measurement uncertainty.

Section 5.3, "Initial Plant Conditions," of the LOCA TR lists the process parameters associated with the plant initial conditions, which serves as input the LOCA EM. The NRC staff reviewed NuScale's LOCA calculations (ML17066A463) to determine whether these parameters were chosen conservatively. Further, the NRC staff performed its own sensitivity studies using NuScale's NRELAP5 code and input model and varied parameters such as initial pool temperature, RCS temperature and pressure, and pressurizer level. The NRC staff confirmed through its sensitivity studies that the initial plant conditions listed in Table 5-6 are chosen conservatively. For example, **[[**

]] is conservatively used for the LOCA analysis.

The NRC staff further finds that the NuScale has provided sufficient detail to ensure that the appropriate bounding plant conditions have been selected for each LOCA analysis and that the system and measurement uncertainties established by NuScale in the DCA have been conservatively included in the LOCA analyses.

4.5.4 Loss-of-Coolant Accident Break Spectrum

In Section 5.4, NuScale presents its break location, configuration and size, single failure, loss-of-power, and Decay Heat Removal System availability assumptions as part of its break spectrum definition. The considered break locations are a maximum of 2-inch piping.

Break Location, Configuration and Size

NuScale postulates break locations in the NPM RCS injection and discharge line, pressurizer spray supply line, and high point vent lines. NuScale does not consider the RRV and RVV valve flange connections as break locations. The NRC staff reviewed the locations selected and compared it to the potential break locations and finds this identification of break locations acceptable due to break exclusion. The NRC staff notes that the RRV and RVV break exclusion zone review is performed as part of the Design Certification review in Chapter 3, "Design of Structures, Systems, Components and Equipment."

The NuScale break spectrum is based on piping that penetrates the RPV wall, connects to the CNV or passes through the CNV. There are four such entities; **[[**

]] The NRC staff reviewed the break spectrum flow area selection against the design and based on that review, agrees that this selection is appropriate. The NRC staff further noted that the break spectrum demonstrates that the 5 percent injection line break without DHRS operation, loss of alternating current (ac) power, and failure of one ECCS division is the most limiting break case that produces the minimum level. The 100 percent CVCS line break produces the MCHFR.

Single Failures

As noted by NuScale in TR Section 5.4.3, “Single Failures,” 10 CFR Part 50, Appendix K requires that single failures be considered within the break spectrum. 10 CFR Part 50, Appendix K requires analyzing single failures of a system or component classified as non-safety related if the inclusion of that system or component would introduce a more limiting condition for LOCA analysis. For single failures, NuScale considered failures of a single RVV or RRV valve to open, or failure of one division of ECCS valves to either actuate or inadvertently actuate.

A single failure inadvertent actuation of a division when it should not be activated, means that direct current (dc) power is removed from that division and two RVVs and one RRV will be available to open, and will do so once the differential pressure (dp) between the RPV and CNV drops below the IAB release pressure. If dc power remains available to the other division, that division’s valves will reposition on a valid actuation signal. If that actuation signal is received later than the IAB release pressure being achieved, then this creates a staggered release of the five ECCS valves (three earlier at the IAB release pressure and two later on the level actuation signal). The applicant stated that this scenario is non-limiting. **[[**

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The IAB valve is a first-of-a-kind, safety-significant, active component integral to the NuScale ECCS. To meet the requirements for the ECCS in 10 CFR Part 50, an applicant must show that it has evaluated the single failure criterion (SFC). The SFC is defined in 10 CFR Part 50, Appendix K and derived from the definition of single failure in 10 CFR Part 50, Appendix A. During its review, the NRC staff noted that although the applicant assumed a single failure of a main ECCS valve to open, the applicant did not apply the SFC to the IAB valve regarding the valve’s function to close. NuScale disagreed with the NRC staff’s application of the SFC to the IAB valve, which led the NRC staff to request the Commission’s direction to resolve this issue, SECY-19-0036, “Application of the Single Failure Criterion to NuScale Power LLC’s Inadvertent

Actuation Block Valves."[1] In SECY-19-0036, the NRC staff summarized the NRC's historical practice for applying the SFC. Specifically, the NRC staff summarized SECY-77-439,[2] in which it informed the Commission of how the NRC staff then generally applied the SFC, and, SECY-94-084,[3] in which the NRC staff requested the Commission's direction on the application of the SFC in specified fact- or application-specific circumstances. In view of this historical practice, the NRC staff in SECY-19-0036, requested the Commission's direction on the application of the SFC to the IAB valve's function to close.

In response to the paper, the Commission directed the NRC staff in SRM-SECY-19-0036, "Staff Requirements - SECY-19-0036 - Application of the Single Failure Criterion to NuScale Power LLC's Inadvertent Actuation Block Valves,"[4] to "review Chapter 15 of the NuScale Design Certification Application without assuming a single active failure of the inadvertent actuation block valve to close." The Commission further stated that "[t]his approach is consistent with the Commission's safety goal policy and associated core damage and large release frequency goals and existing Commission direction on the use of risk-informed decision-making, as articulated in the 1995 Policy Statement on the Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities and the White Paper on Risk-Informed and Performance-Based Regulation (in SRM SECY- 98-0144 and Yellow Announcement 99-019)."

Based on the NRC staff's historic application of the SFC and the Commission's direction on the subject, as described in SECY-77-439, SRM-SECY-94-084, and SRM-SECY-19-0036, the NRC has retained some discretion, fact- or application-specific circumstances, to decide when to apply the SFC. The Commission's decision in SRM-SECY-19-0036, provides direction regarding the appropriate application and interpretation of the regulatory requirements in 10 CFR Part 50, to the NuScale IAB valve's function to close. This decision is similar to those documented in previous Commission documents that evaluated the use of the SFC and provided clarification on when to apply the SFC in other specific instances.

Specific LOCA event limiting single failures are evaluated as part of a design-specific application of this methodology, such as the NuScale DCA. This is reflected in item 6, in Section 6.0 of this SER.

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1 See SECY-19-0036, "Application of the Single Failure Criterion to NuScale Power LLC's Inadvertent Actuation Block Valves," (April 11, 2019) (ADAMS Accession No. ML19060A081).

2 See SECY-77-439, "Single Failure Criterion," (August 17, 1977) (ADAMS Accession No. ML060260236).

3 SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs (March 28, 1994) (ADAMS Accession No. ML003708068), and associated SRM (June 30, 1994) (ADAMS Accession No. ML003708098).

4 See SRM-SECY-19-0036, "SECY-19-0036 Application of the Single Failure Criterion to NuScale Power LLC's Inadvertent Actuation Block Valves," (July 2, 2019) (ADAMS Accession No. ML19183A408).

Loss-of-Power

The NuScale LOCA evaluation methodology considers two scenarios for loss-of-power coincident with a postulated LOCA:

- Complete loss of normal ac and dc power
- Complete loss of only ac power with dc power availability

The NRC staff finds that the NuScale LOCA EM appropriately models the impacts of loss of ac and/or dc power coincident with a LOCA because it considers both scenarios.

Specific LOCA event limiting electric power assumptions are evaluated as part of a design-specific application of this methodology, such as the NuScale DCA. This is reflected in item 6 in Section 6.0 of this SER.

Decay Heat Removal System Availability

The NuScale LOCA EM does not credit the function of the DHRS. However, in Section 9.3, "Decay Heat Removal System Availability," of the EM TR, NuScale documented a sensitivity study that shows the beneficial outcome of the DHRS operation that the NRC staff reviewed. Because the DHRS operation is beneficial during a LOCA, the NRC staff finds that the NuScale assumption that the DHRS does not function, is conservative for the NuScale LOCA EM because it does not include this potential benefit from DHRS operation.

4.5.5 Sensitivity Studies

For the NRC staff's evaluation of NuScale's sensitivity studies, please see Section 4.9 of this SER.

4.5.6 Review Focus of TR Section 5

NRC regulation 10 CFR 50.46 requires applicants to show that they have analyzed the bounding break within the break spectrum relative to the FOMs for LOCA evaluations. The most critical FOMs for the NuScale LOCA EM are to show that minimum CLLs in the NPM riser remain above the active core and that the maximum fuel heat flux remains below the CHF. The NRC staff finds that the NuScale LOCA EM is capable of calculating these FOMs. The representative LOCA evaluations included in the NuScale LOCA EM TR, are meant to be examples to demonstrate the EM's capability but do not necessarily identify the most limiting LOCA, as this will need to be done through application of the methodology to a design. For example, as described in Section 4.2 of the TR, the example calculations were completed prior to the NPM design change to add an ECCS actuation on low RCS pressure. The staff determined that the example calculations presented in the TR reasonably represent the capability of the methodology.

4.5.7 Overall Conclusions of the review of TR Section 5

Subject to the limitations discussed above, the NRC staff finds that the NuScale LOCA EM, as described in LOCA EM TR-0516-49422-P, Section 5, is acceptable for referencing in the NuScale DCA because the modeling developed for the EM is deemed adequate to represent the key phenomena and features of the NPM.

4.6 NRELAP5 Computer Code

As stated in the LOCA EM TR, Section 6.0, NuScale used its proprietary NRELAP5 system thermal-hydraulics code for evaluating small break LOCA ECCS performance. This NuScale NRELAP5 code, developed from RELAP5-3D©, Version 4.1.3, includes hydrodynamics, heat transfer between structures and fluids, modeling of fuel, reactor kinetics models, and control systems models. Like previous versions of RELAP5 codes, the NuScale NRELAP5 code used a two-fluid, non-equilibrium, non-homogenous model to simulate system thermal-hydraulic response. In Section 6.0, NuScale provided a general overview of the NRELAP5 computer code structure, models, and correlations and a description of the LOCA code models and code changes implemented by NuScale to model unique design features and phenomena for the NPM.

NuScale added or revised the following models to NRELAP5, following the requirements of the NuScale QAP:

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The NRC staff's review of the NuScale NRELAP5 computer code focused only on NuScale changes and additions to the RELAP5-3D© code, and, the applicability of the NRELAP5 code to the NPM. The NRC staff's review was based on NRELAP5 Version 1.4. This is reflected in item 5 in Section 6.0 of this SER.

4.6.1 Quality Assurance Requirements

Compliance with QA requirements is described in "NuScale Topical Report: Quality Assurance Program Description for the NuScale Power Plant," NP-TR-1010-859-NP-A. The NRC staff reviewed the Quality Assurance Program Description and documented its approval in its SER (ML16347A405). Further, the NRC staff inspected NuScale's design control process and code development procedures. These inspections are documented in inspection reports dated October 7, 2017 (ML15268A186) and July 24, 2017 (ML17201J382).

4.6.2 NRELAP5 Hydrodynamic Model

In TR Section 6.2, "NRELAP5 Hydrodynamic Model," NuScale stated that the NRELAP5 hydrodynamic model is a transient, two-fluid model for flow of a two-phase vapor/gas-liquid mixture that can contain non-condensable components in the vapor/gas phase as well as a soluble component (i.e., boron) in the liquid phase. The NRELAP5 two-fluid equations of motion are formulated in terms of volume and time-averaged parameters of the flow. For most analyses, NRELAP5 uses empirical formulas to calculate bulk properties, such as friction and heat transfer.

Because the NRELAP5 hydrodynamic model framework is essentially identical to the RELAP5-3D© code and the NRC staff has approved its application for LOCA analyses of the US EPR design (ML110070113) and the US APR1400 design (ML18180A327), the NRC staff only reviewed the code changes and their applicability of the unique aspects to the NPM.

Field Equations

In Section 6.2.1, "Field Equations," of the LOCA TR, NuScale discusses the NRELAP5 thermal-hydraulic model. Because NRELAP5 basic field equations 6-1 to 6-10 are the same as the field equations in the RELAP5-3D© code (and the same equations as contained in the RELAP5/Mod3 code which has been thoroughly reviewed in the past by the NRC staff, as submitted by other vendors for review and approval), the NRC staff did not perform an in-depth review of these field equations and numerical solution techniques. The NRC staff confirmed that NRELAP5 correctly addressed NRC Information Notices 92-02 and 92-02, Supplement 1. Specifically, NuScale demonstrated that when the RELAP5 code is applied to situations in which the pressure drops significantly between cells, the energy in the downstream volume is not significantly underestimated), as INL modified RELAP5 to successfully resolve this issue. The modification preserves the total enthalpy across a junction where a large pressure gradient exists, as in a blow down event into a CNV, for example, where dissipation terms become more important. This modification also resolved the issues raised by NRC Information Notices 92-02 and 92-02, Supplement 1, as the modification is shown to conserve energy transfer across a junction where a large differential pressure exists. The NRC staff reviewed the break and/or orifice input model junction flags to ensure that they are appropriately set to involve the energy correction modeling (e=1).

The NRC staff audited the NRELAP5 Theory manual regarding code modifications made to the NRELAP5 code for application to the EM with focus on the changes from the INL RELAP5-3D code Version v.4.1.3, as described in the associated audit reports (ML20010D112 and ML20034D464).

As discussed in the associated audit report, the NRC staff audited NuScale's comparison of the NRELAP5 code prediction with known solutions for a simple oscillating manomenter, which showed that the NRELAP5 code properly predicts level and flow behavior for this benchmark exercise. This confirms that the treatment of the inertia term does not introduce significant error into the ability of the code to capture the correct amplitude and period of the oscillations. Adequate prediction of a simple manometer provides validation that NRELAP5 properly models hydrostatic forces that govern small break LOCA behavior as well as to assure the NRC staff that the momentum equation, with and without friction, is properly formulated and implemented in the code.

To confirm that the NRELAP5 code calculations do not show non-physical flow anomalies that would impact analysis of small break LOCAs, the NRC staff audited NuScale's results of the prediction of fluid flow behavior in a simple system containing parallel pipe components, as described in the associated audit report (ML20010D112) The NRC staff noted that flow anomalies that had been present in earlier versions of RELAP5, were not present in the NRELAP5 version of RELAP5-3D. The core nodalization consists of the application of the 1-D modeling technique in NRELAP5. The NRC staff further recognizes that 3-D modeling of a core is not necessary to accurately predict two-phase level swell following a small break LOCA. The 1-D and 3-D predictions of small break LOCA two-phase level swell have been shown to be in very close agreement, as the multi-dimensional flow capability does not cause the two-phase level to vary significantly across the radius of the nuclear core. NuScale further proved that the different nodalization included in NRELAP5 resolves the anomalous fluid behavior as exemplified by comparison to the dual and triple parallel pipe problems. Therefore, the NRC staff concluded that the current NRELAP5 core modeling would avoid the potential flow anomalies and that the 1-D channel model is reasonably accurate based on additional information provided by NuScale (ML18031B319).

State Relations

In the LOCA TR Section 6.2.2, NuScale discusses the NRELAP5 six-equation model based on five independent state variables with an additional equation for the non-condensable gas component. Because these state equations are the same as the field equations in the RELAP5-3D© code and its predecessors, which were reviewed and approved before, these state equations are acceptable.

Flow Regime Maps

In Section 6.2.3, "Flow Regime Maps," NuScale stated that one-dimensional field equations for the two-fluid model used in NRELAP5 precludes direct calculation of physical parameters, such as velocity or energy, that depend upon transverse gradients. Therefore, NRELAP5 adds algebraic terms to the conservation equations for a specific flow regime to provide closure to the two-fluid equations. NRELAP5 flow regime maps are based on the work of Taitel and Dukier and Ishii, as referenced in Section 6.2.3 of the LOCA TR, but further simplified by NuScale to efficiently apply these criteria in NRELAP5. A schematic of the vertical flow regime map, as coded in NRELAP5, is shown in LOCA TR Figure 6-1, "Schematic of Vertical flow-regime map indicating transitions," to illustrate flow-regime transitions as functions of void fraction, average mixture velocity and boiling. The NRELAP5 junction map is shown in LOCA TR Section 6.2.3.2, "Junction Flow Regime Maps." The NRELAP5 flow regime maps used for junctions are the same as used for the volumes and are based on the work of Taitel and Dukler, Ishii and Tandon, as referenced in Section 6.2.3 of the LOCA TR.

Because the NRELAP5 flow regime maps are the same as those in the RELAP5-3D© code and its predecessors, which were reviewed and approved before, the NRC staff considers these flow regime maps to be applicable to the NuScale application.

Momentum Closure Relations

In LOCA TR Section 6.2.4, "Momentum Closure Relations," NuScale states that NRELAP5 uses two different models for the phasic interfacial friction force computation: the drift flux method and the drag coefficient method. These are same models used in the base version RELAP5-3D© except for the revisions NuScale made to implement the new HCSG component.

NRELAP5 uses the drift flux approach only for bubbly and slug-flow regimes for vertical flow. The NRELAP5 drift flux equations are shown in TR Section 6.2.4. NRELAP5 uses the drag coefficient approach in all flow regimes other than vertical bubbly and slug-flow, as described in the equations in Section 6.2.4 of the TR. NRELAP5 determines wall friction based on the volume flow regime map. Because the NRELAP5 momentum equations in Section 6.2.4 of the LOCA TR are the same as the equations in the RELAP5-3D© code and its predecessors, which were reviewed and approved before, these flow regime maps are applicable to the NuScale application.

Heat Transfer

Section 6.2.5, “Heat Transfer,” of the LOCA TR describes the heat transfer equations. NRELAP5 solves for liquid and vapor/gas energy including energy added or removed by the heat flux to or from wall heat structures. NRELAP5 uses boiling heat transfer correlations when the wall surface temperature is above the saturation temperature. When a hydraulic volume is voided and the adjacent surface temperature is subcooled, vapor condensation on the surface is predicted. If non-condensable gases are present, the phenomena are more complex because condensation is based on the partial pressure of vapors present in the region. When the wall temperature is less than the saturation temperature based on total pressure, but greater than the saturation temperature based on vapor partial pressure, a convection condition exists. LOCA TR Figure 6-2, “NRELAP5 boiling and condensing curves,” illustrates these three regions of the curve.

NRELAP5 uses the Chen boiling correlation up to the CHF point. NRELAP5 will issue a message and stop running if CHFR reduces below one for core heat transfer. If the CHFR is below one on other structures outside of the core, NRELAP5 will calculate Post-CHF heat transfer on these surfaces outside the core. NuScale added this stop function to NRELAP5 when the core CHFR drops below one because maintaining core CHFR above one is a critical FOM for the NuScale LOCA EM.

The NRC staff finds that the addition of the stop function to NRELAP5 is appropriate because acceptability of the NuScale LOCA EM depends on maintaining the heat flux from the reactor fuel to the RCS below the CHF point. The detailed review of CHF correlations is documented in Section 5 of this SER.

4.6.3 Heat Structure Models

As discussed in Section 6.3, Heat Structure Models,” NRELAP5 calculates heat transfer from hydrodynamic volumes to adjacent solid heat structures. NRELAP5 has the capability to model various heat structures, allows the user to use standard thermal conductivities and heat capacities or input tables or functions and solves the one-dimensional heat equation with a finite difference method. NRELAP5 allows the user to specify spacing, internal heat source and material composition for each mesh. For nuclear fuel, NRELAP5 calculates the heat source with a reactor kinetics model, or tables of power versus time, or a control system variable. NRELAP5 also includes options for boundary conditions. These modeling features are typical for light water reactor applications. Therefore, they are applicable to NuScale LOCA analyses.

LOCA TR Section 6.3 also describes that the NRELAP5 has heat transfer correlations and a gap conduction model. NRELAP5 solves the heat conduction equation using the Crank-Nicolson method referenced in Section 6.3 of the LOCA TR.

Because the NRELAP5 heat structure and heat transfer models and equations discussed above are the same as those in the RELAP5-3D© code, the NRC staff did not perform an in-depth review of these heat structure and heat transfer models. However, as discussed below, the NRC staff did perform in-depth reviews of the specific heat structure modeling added to NRELAP5, including modeling of steam condensation on the inside wall of the CNV and heat transfer for the HCSG.

4.6.4 Point Reactor Kinetics Model

As described in LOCA TR Section 6.4, “Point Reactor Kinetics Model,” NRELAP5 calculates the total reactor core power from a user specified table or with a point-reactor kinetics model with reactivity feedback. The model uses the ANS 1973 decay heat standard to calculate reactor core power from decay of fission products. The NRC determined that this is similar to ANS 1971, but with higher accuracy, and in compliance with 10 CFR Part 50, Appendix K. Therefore, the NRC staff considers this modeling acceptable.

Furthermore, the selection of the delayed neutron fraction for the kinetics calculation can be justified as conservative for the core in the as-used state. This is typically done as part of the neutronics analysis of the core for a specific cycle design.

The staff confirmed that the NRELAP5 reactor core power and fission power models and equations discussed above are the same as those in the RELAP5-3D© code and its predecessors, which have been reviewed and approved before. The NRC staff found that the point kinetics modeling used is adequate to conservatively calculate the fission power and the decay heat power during a LOCA transient.

4.6.5 Trips and Control System Models

The NRELAP5 modelling of trip and control systems is described in Section 6.5, “Trips and Control System Models.” NRELAP5 provides several types of control variables based on NRELAP5 calculated parameters for each hydrodynamic volume, junction, pump, valve, heat structure, and reactor kinetics. Because the NRELAP5 trip and control system models are the same as those in the RELAP5-3D© code and its predecessors, which were reviewed and approved before, these NRELAP5 trip and control system models are acceptable for NuScale applications.

4.6.6 Special Solution Techniques

As stated in Section 6.6, “Special Solution Techniques,” NRELAP5 uses empirical models for certain processes that are too complex for the general solutions provided in NRELAP5. The NRC staff’s evaluation of these special NRELAP5 empirical models are discussed below.

Choked Flow

MOODY CRITICAL FLOW MODEL

NRELAP5 uses the Moody critical flow model, when the break flow is calculated to be two-phase, to comply with the 10 CFR Part 50, Appendix K requirements. NRELAP5 includes options, as described in Section 6.6.1, “Choked Flow,” for switching between **[[**

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Abrupt Area Change

As discussed by NuScale in Section 6.6.2, "Abrupt Area Change," the NRELAP5 hydrodynamic model provides analytical models for sudden area changes and orifices. NRELAP5 models abrupt area changes with the Borda-Carnot formulation for a sudden enlargement and the vena-contracta effect for a sudden contraction or sharp-edge orifice or both. This formulation does not include models for rounded or beveled enlargements, contractions, or orifices.

Because the NRELAP5 abrupt area change models are the same as those in the RELAP5-3D© code and its predecessor codes, which were approved for LOCA analyses before, the NRC staff considers these models to be acceptable.

Counter Current Flow Limitation

NRELAP5 implements the general CCFL model in a form proposed by Bankoff which has the structure shown in TR Section 6.6.3, "Counter Current Flow Limitation." NuScale provided an assessment of the NRELAP5 CCFL model against the Bankoff perforated plate test data in LOCA EM TR Section 7.2.10, "Bankoff Perforated Plate," and NuScale presented a sensitivity study of the effects of the CCFL as it applies to the NPM pressurizer baffle plate in TR Section 9.6.3 "Counter Current Flow Limitation Behaviour on Pressurizer Baffle Plate."

Because the NRELAP5 uses essentially the same CCFL model as the RELAP5-3D© code, the NRC staff did not perform an in-depth review of these NRELAP5 code implementation of the Bankoff CCFL model in NRELAP5. However, the NRC staff reviewed the assessment of the Bankoff model versus test data as shown in SER Section 4.7.2, "Legacy Test Data," and the NRC staff evaluated the NuScale Bankoff sensitivity study as shown in SER Section 4.9.7, "Sensitivity Studies." Since the counter current flow in the NuScale reactor is not a dominant physical phenomenon when the water level reaches the minimum value, this model is acceptable.

4.6.7 Helical Coil Steam Generator Component

As described in Section 6.7, "Helical Coil Steam Generator Component," NuScale added a new hydrodynamic component and heat transfer package to the NRELAP5 code to model flow and heat transfer inside and outside the HCSG tubes. These added models are specific to helical coil geometry heat transfer and wall friction correlations and were added because the models in the baseline RELAP5-3D© code did not provide adequate agreement with pressure drop and heat transfer performance against prototypic HCSG testing performed at SIET.

The adequacy of the added NRELAP5 HCSG were demonstrated by NuScale through prototypic assessments of the NuScale HCSG using SIET test data. These tests assessed heat transfer and pressure drop on both the secondary side (within tubes) and primary side (external to tubes) of the HCSG that showed good agreement with HCSG tube axial wall and secondary fluid temperature data.

The analysis of a LOCA depends on the initial stored energy in the primary coolant and the performance of the NPM HCSG can influence the temperatures and flow rates in the RPV. As described in the associated audit report (ML20010D112), the NRC staff audited a sensitivity study for a variation in the heat transfer performance of the HCSGs, above and below the nominal expected performance. The NRC staff also considered in its audit, the potential distortion on core inlet temperature and SG steam temperature relative to the uncertainty for NIST-1 test results.

The analyses audited included the effect of SG degradation on the initial conditions as well as a typical LOCA progression. Because the NRELAP5 LOCA model conservatively assumes no plugging or fouling in the HCSGs and no credit for DHRS cooling, the SG model was reviewed in detail in the NRC staff's SER for the NuScale Non-LOCA EM TR (ML20042E039). **[[**

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Helical Coil Tube Friction

NuScale implemented SG tube friction models into NRELAP5 for single phase and two-phase flow conditions. **[[**

]] Therefore, the NRC staff considers the in-tube friction model acceptable for LOCA analyses.

Helical Coil Tube Heat Transfer

A new heat transfer package has also been added to NRELAP5 and differs from that of the standard RELAP5 pipe geometry in **[[**

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The transition to **[[**

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The laminar heat transfer correlation developed by **[[**

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The NRC staff finds the friction and heat transfer models to be acceptable for the LOCA evaluations based on the good agreement with the SIET separate effects heated tube wall and pressure drop data. Further detailed evaluation is documented in the NRC staff's SER of the Non-LOCA EM TR (ML20042E039).

4.6.8 Wall Condensation

As discussed in Section 6.8, "Wall Heat Transfer and Condensation," the RELAP5-3D© code includes **[[**

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The NRC staff reviewed and audited the NRELAP5 code, as discussed in the associated audit report (ML20010D112), and noted that **[[**
]] Since condensation is of major significance to the prediction of ECCS performance, the NRC staff reviewed the information provided by NuScale (ML17324B392 and ML19240C658) regarding the modeling of heat transfer from the CNV to the pool for a loss of primary coolant from the RPV and audited additional calculations underlying the submitted information, as described in the associated audit report (ML20010D112). The NRC staff's review of use of the extended Shah correlation to model heat transfer in the DHRS was not reviewed as part of this TR, since DHRS heat transfer is not credited. The NRC staff's review of its use to model heat transfer in the DHRS is documented in the NRC staff's SER of the Non-LOCA EM TR (ML20042E039).

The information that the NRC staff reviewed, indicated that during **[[**

]] the NRC staff conducted an audit, as described in the associated audit report (ML19282C504), to determine **[[**

.]] As described in the associated audit report (ML19282C504), the NRC staff also audited [[

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The NRC staff finds that these modeling approaches are conservative for calculating minimum CNV heat transfer for maximum peak containment pressure and minimum collapsed water level above the core.

4.6.9 Interfacial Drag in Large Diameter Pipes

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As the NRELAP5 code assessment against General Electric (GE) level swell test showed reasonable agreement between the measured and the calculated as shown in Section 7.2.2, "GE Level Swell (1 ft)," the interfacial drag model for large diameters is reasonably accurate. Thus, this is applicable to the NuScale LOCA analyses.

4.6.10 Fission Decay Heat and Actinide Models

The NRELAP5 implementation of the ANS 1973 standard applies the Shure curve. Comparison of the ANS 1973 standard to the as implemented curve in NRELAP5 shows that the implemented curve reproduces the 1973 standard decay heat data.

The implemented model yields the result quoted in the 1979 Standard, the 1994 Standard, and the 2005 Standard. The 1973 actinide equations are identical to those in the 1979 standard. Comparison of the NRELAP5 model with this standard shows identical results.

Furthermore, infinite operation is assumed and a decay heat multiplier of 1.2 is employed as required by 10 CFR Part 50, Appendix K. The NRC staff notes that previous studies of the various decay heat standards identified the need to include the contributions from additional actinides (other than 239U and 239Pu), since actinide contribution grows significantly with shutdown time. Because the decay heat model used meets the applicable requirements of 10 CFR Part 50, Appendix K by assuming infinite full power operation and the approved ANS 1973 decay heat curve, the NRC staff considers this model to be acceptable.

4.6.11 Critical Heat Flux Models

The NRC staff's review of the proposed CHF correlations is documented in Section 5.4 of this SER.

4.7 NRELAP5 Assessments

Section 7, "NRELAP5 Assessments," provides a summary of the NuScale assessments of the SET and IET that NuScale performed. NuScale discussed the comparison of the NRELAP5 analysis of these separate and integral effects tests versus experimental data in Section 8.0, "Assessment of Evaluation Model Adequacy," and presents its justification of the adequacy for modeling of the high-ranked phenomena in the NuScale LOCA PIRT.

The NRC staff reviewed the separate and integral effects tests and focused on determining the acceptability of the NuScale LOCA evaluation methodology for performing design basis LOCA analyses. This NRC staff review was limited to the applicability of NuScale methodology and use of the NRELAP5 computer code to perform LOCA analysis for the break spectrum as defined by NuScale.

4.7.1 Assessment Methodology

NuScale used various special and integral experimental tests, and analytic problems to assess the performance of NRELAP5 using the process identified in Element 2 of RG 1.203. NuScale chose the tests and analytical problems to assess the adequacy of the NRELAP5 code to model the high-ranked phenomena shown in the NuScale LOCA PIRT as discussed in Section 4 of the LOCA TR. The NRC staff concludes that this process is consistent with that of RG 1.203 and is therefore, acceptable.

4.7.2 Legacy Test Data

Tests that NuScale evaluated in Section 7 were performed by others and were not done in compliance with the NuScale QAP. With the exception of Marviken JIT-11 data, NuScale qualified these tests by applying non-mandatory guidance provided by NQA-1 2008/2009 Addendum. NuScale used Marviken JIT-11 data based on published literature data. Because these legacy test results have been reviewed by the NRC staff previously for several RELAP5 code-based LOCA EM methods, the use of these data by NuScale is acceptable.

Ferrell-McGee

The Ferrell-McGee tests were performed in vertical pipes over a wide range of single phase and two-phase flow conditions with uniform, contraction, and expansion flow areas. NuScale performed analysis of these tests with NRELAP5 and compared the calculations with the experimental data to assess the ability of NRELAP5 code to calculate single- and two-phase pressure drop and void fraction under different pressures, flow rates, and inlet quality.

NuScale's NRELAP5 calculations and comparison to the Ferrell-McGee, are summarized in the LOCA EM TR. The NRC staff audited the calculations underlying the summary in the LOCA TR, as described in the associated audit report (ML20010D112), and found that the NuScale NRELAP5 code calculations show excellent agreement with test data for pressure drop in the bubbly to slug flow regime and satisfactory agreement with the test data in the annular-mist regime. Ferrell-McGee tests at void fractions approaching 1.0 where not usable for comparison to NRELAP5 analysis because the void fractions near 1.0 cannot be measured with sufficient accuracy and because pressure drop is strongly dependent on void fraction. The NRC staff found that NRELAP5 was able to adequately calculate void distribution for all of the Ferrell-McGee test cases based on the observed agreement between the measured and the calculated void fraction distribution. The difference between NRELAP5 calculations and measured pressure drop decreases with increased flow rate, increased pressure and increased hydraulic diameter.

GE Level Swell Test – 1 ft

NuScale assessed NRELAP5's ability to predict void distribution and level swell phenomena for depressurization transients by assessing it against the GE Level Swell Test referenced in Section 7.2.2, "GE Level Swell (1 ft)." The specifics of the test configuration and instrumentation are described by NuScale in its LOCA TR and the NRC staff reviewed these descriptions.

The GE Level Swell Test 1004-3 is a small-break blowdown of a vertical vessel for which GE measured the axial void fraction distribution. NuScale modeled the GE test facility and compared the two-fluid interphase level calculated by NRELAP5 to the measured void fraction distributions from the GE test. NuScale used these comparisons to assess whether NRELAP5 correctly predicts single phase and two-phase choked flow, liquid level, flashing, level swell, mixture level and phase slip and flow. NuScale used the Henry Fauske critical flow correlation with four contraction coefficients (1.0, 0.9, 0.7 and 0.6) to calculate the break flow. NuScale provided sensitivity analyses for blowdown line orientation and vessel nodalization.

NuScale determined that the selection of the Henry Fauske critical flow correlation with a 0.9 discharge coefficient provides the best comparison of NRELAP5 calculated vessel pressure to the GE test data. NuScale also determined that NRELAP5 analysis results are not sensitive to the other modeling options. The NuScale NRELAP5 model of the GE 1-foot (ft) (0.3 meters (m)) vessel generally over predicts void fraction. NRELAP5 only under predicts void fraction at the 12-feet (3.7m) elevation for times of 10 and 40 seconds.

Because the primary FOM as shown in TR Section 4 for NuScale LOCA analyses is a CLL in the NPM riser, the NRC staff agrees with NuScale that the NRELAP5 predicted void fractions are in reasonable agreement with the measured data and the Henry-Fauske critical flow correlations should be used for break flow in the subcooled region and Kataoka-Ishii and Zuber-Findlay for the interfacial drag model in pipes as discussed in Section 6.9.

GE Level Swell Test – 4 ft

NuScale assessed NRELAP5's ability to predict void distribution and level swell phenomena for depressurization transients by assessing it against the GE Level Swell Test referenced in Section 7.2.3, "GE Level Swell (4 ft)." The specifics of the test configuration and instrumentation are described by NuScale in its LOCA TR and the NRC staff reviewed these descriptions.

The GE 4-ft (1.2m) tank level swell tests measured time-dependent pressures and void fraction profiles in a large tank which was depressurized via a blowdown line. NuScale modeled the GE test facility and compared the two-fluid interphase level calculated by NRELAP5 to the measured void fraction distributions from the GE test. NuScale used these comparisons to assess whether NRELAP5 correctly predicts single phase and two-phase choked flow, liquid level, flashing, level swell, mixture level and phase slip and flow.

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Because the primary FOM as shown in TR Section 4 for NuScale LOCA analyses is a CLL in the NPM riser, the NRC staff found that the assessment results are in support of using the Henry-Fauske correlation for the critical break flow calculation in the subcooled region and Kataoka-Ishii and Zuber-Findlay for the interfacial drag model in the riser.

KAIST

NuScale assessed the DHRS condensation modeling of its NRELAP5 code against experimental data from KAIST. Since the assessment was relative to the behavior of DHRS, which is not credited in the LOCA analysis, the review results of the assessment analysis are documented in the NRC staff's SER of the Non-LOCA EM TR (ML20042E039).

FRIGG

NuScale assessed NRELAP5's ability to model interphase drag and heat transfer models under two phase flow conditions by assessing it against the FRIGG Tests referenced in Section 7.2.5, "FRIGG." The specifics of the test configuration and instrumentation are described by NuScale in its LOCA TR and the NRC staff reviewed these descriptions.

NuScale modeled the FRIGG test facility and compared the void distribution data to the measured void distributions from the FRIGG tests. NuScale used these comparisons to assess whether NRELAP5 correctly predicts the void distribution data in a rod bundle geometry as a function of mass flow, inlet subcooling, system pressure and thermal power.

Four tests were chosen for assessment. The NRC staff agrees with NuScale's conclusion that Figures 7-23 to 7-26 of the LOCA TR show that NRELAP5 predicted the experimental void fraction data with reasonable agreement, justifying use of one dimensional nodalization to obtain reasonable predictions of the axial void profile. These results validate the NRELAP5 interphase drag and heat transfer models for applications having similar core geometries.

FLECHT-SEASET

NuScale assessed NRELAP5's ability to model bundle boil-off by assessing it against the FLECHT-SEASET tests referenced in Section 7.2.6 of the LOCA TR. The specifics of the test configuration and instrumentation are described by NuScale in its LOCA TR and the NRC staff reviewed these descriptions.

NuScale modeled the FLECHT-SEASET test facility and compared the void distribution data to the measured void distributions from the FLECHT-SEASET tests. NuScale used these comparisons to assess whether NRELAP5 correctly predicts the void distribution data for a core boiloff configuration. The NRC staff audited NuScale's comparison calculations, as described in the associated audit report (ML20010D112), and noted a significant difference in the early part of the transient. As shown in Figure 7-28, "FLECHT-SEASET level 1 void fraction versus time – Test 35557," of the LOCA TR, voids appeared almost immediately after the initiation of the transient in the calculations, whereas there was a delay in the void generation in the experimental data. It appears to the NRC staff that the difference between the calculations and the data may be a time delay in the heatup of the rods. The LOCA TR shows the calculated void fraction history at various levels in the test section compared to data for one of the three tests. The calculations represented the trend of the data reasonably well. Early in time and at the lower levels, it appears the calculated entrainment rate is too high and thus the void fraction is over-calculated. The entrained liquid is carried up and out of the test section as evidenced by the lower calculated void fraction at elevations above the bottom cell during that time. This behavior also persists at later times as observed in the figures in LOCA TR Section 7.2.6. Simulation of the boiloff test seems to indicate that the interphase drag calculated by the code is too large. The rate of coolant lost out the bundle top in the calculation is greater than shown by the data. Figure 7-1, "Schematic of the Ferrell-McGee test section," of the LOCA TR indicates that these tests partially evaluated subcooled boiling at the spacers. Although there is some level of deviation between the measured void fraction and the calculated void fraction, the NRC staff considered that the NPM modeling derived from the FLECHT-SEASET boiloff test is reasonable because there is a large amount of collapsed water level above the core for the NuScale design.

SemiScale (S-NC-02 and S-NC-10)

NuScale assessed NRELAP5's ability to model **[[**
]] by assessing it against the SemiScale tests referenced in Section 7.2.7, "SemiScale (S-NC-02 and S-NC-10)," of the LOCA TR. The specifics of the test configuration and instrumentation are described by NuScale in its LOCA TR and the NRC staff reviewed these descriptions.

NuScale modeled the SemiScale test facility and compared the loop mass flowrate as a function of system inventory to the measured data from the SemiScale tests. NuScale used these comparisons to assess whether NRELAP5 correctly predicts natural circulation flow. For the test S-NC-2, the calculated results of flow versus inventory are under predicted at higher inventories and are adequately predicted at lower inventories for both power levels. In the 75-80 percent inventory for test S-NC-2, the code exhibits an oscillatory behavior. As described in the associated audit report (ML20010D112), the NRC staff audited NuScale's assessment, which attributed this to flow regime flip-flopping in the lowest core node. For test S-NC-10 at 100 kW, the calculated results compare well with the data in the 97 percent to 100 percent mass inventory range. In the lower inventory range, the flow rates were over predicted by as much as 40 percent. The over prediction is attributed by the applicant to the lowest node having a bubbly flow regime, resulting in more interfacial area and thus more drag compared to a slug regime, which would result in lesser drag force. However, since the assessment focused on the natural circulation and the NRELAP5 prediction matched the measured flow rate well, the code assessment against SemiScale tests are acceptable.

Wilson Bubble Rise

NuScale assessed NRELAP5's ability to model the void fraction distribution in the hot leg riser by assessing it against the Wilson Bubble Rise test referenced in Section 7.2.8, "Wilson Bubble Rise," of the LOCA TR. The specifics of the test configuration and instrumentation are described by NuScale in its LOCA TR and the NRC staff reviewed these descriptions.

NuScale modeled the Wilson Bubble Rise test facility and compared the void fraction at different pressures to the measured data from the Wilson Bubble Rise test. NuScale used these comparisons to assess whether NRELAP5 correctly predicts void fraction distribution in the hot leg riser. NuScale's assessment results show the void fraction is over predicted at lower flow rates and under predicted at higher flow rates. However, the overall comparison is within a normal range of error band. Therefore, the NRC staff finds this part of assessment acceptable.

Marviken Jet Impingement Test

NuScale assessed NRELAP5's single phase choked flow model by assessing it against the Marviken Jet Impingement test referenced in Section 7.2.9, "Marviken Jet Impingement Test (JIT) 11," of the LOCA TR. The specifics of the test configuration and instrumentation are described by NuScale in its LOCA TR and the NRC staff reviewed these descriptions.

NuScale modeled the Marviken Jet Impingement test facility and compared simulated mass flow rate and density for various values of the discharge coefficient to the measured data from the Marviken Jet Impingement test.

NuScale applied the **[[**

]]. Therefore, the NRC staff considers the code assessment against the Marviken test acceptable.

Bankoff Perforated Plate

NuScale assessed NRELAP5's ability to model countercurrent flow by assessing it against the Bankoff Perforated Plate test referenced in Section 7.2.10, "Bankoff Perforated Plate," of the LOCA TR. The specifics of the test configuration and instrumentation are described by NuScale in its LOCA TR and the NRC staff reviewed these descriptions.

NuScale modeled the Bankoff Perforated Plate test facility and compared the vapor superficial velocity to the measured data from the Bankoff Perforated Plate test. NuScale used these comparisons to assess whether NRELAP5 correctly predicts countercurrent flow. The NRC staff reviewed this comparison, which shows that NRELAP5 predictions are in excellent agreement, thus demonstrating that the correlation is correctly implemented in NRELAP5 and that the code can accurately model the countercurrent flow phenomena that occurs in the Bankoff tests.

Marviken Critical Flow Tests 22 and 24

NuScale assessed NRELAP5's ability to model blowdown conditions where discharge flow is limited by choked conditions by assessing it against the Marviken Critical Flow tests referenced in Section 7.2.11 of the LOCA TR. The specifics of the test configuration and instrumentation are described by NuScale in its LOCA TR and the NRC staff reviewed these descriptions.

NuScale modeled the Marviken Critical Flow test facility and compared the mixture density to the measured data from the Marviken Critical Flow tests. NuScale used these comparisons to assess whether NRELAP5 correctly predicts critical flow in piping breaks.

The NRC staff reviewed NuScale's sensitivity study, including sensitivity to time step, nodalization and critical flow at the break and the results for the four different critical flow correlations tested, as described in Section 7.2.11, "Marviken Critical Flow Test 22 and 21," of the LOCA TR. The NRC staff concludes that NuScale's analysis shows that NRELAP5 has the capability to perform critical flow calculations with reasonable agreement to test data.

4.7.3 NuScale Critical Heat Flux Tests

The NRC staff's review of the proposed CHF correlations, including the test data used in the development and validation of the correlation, is documented in Section 5.4 of this SER.

4.7.4 NuScale SIET Steam Generator Tests

NuScale conducted HCSG experiments at SIET laboratories, in Piacenza, Italy. The experiments were done to evaluate the heat transfer capability of the NuScale HCSG and develop the NuScale specific model. However, the LOCA analysis does not credit heat removal from the SG. Therefore, the detailed review of the SIET test and relevant assessments is documented in the NRC staff's SER of the Non-LOCA EM TR (ML20042E039).

4.7.5 NuScale NIST-1 Test Assessment Cases

The NuScale Power Module is dramatically different from other typical light water reactors. As discussed in Section 7.5, "NuScale NIST-1 Test Assessment Cases," of the LOCA TR, NuScale built the NIST-1 test facility at Oregon State University to obtain test data relevant to its unique NPM design and approach to LOCA evaluations. NIST-1 was designed to model the major components of the NPM: the NPM at 1:3.3 length scale, 1:227.5 volume scale, and 1:1 time scale. NuScale performed a number of NIST-1 tests to assist in validation of the NRELAP5 system thermal-hydraulic code, and the NRC staff reviewed the summarized test information in the LOCA TR. Further, the NRC staff performed a detailed audit of the test assessments, as described in the associated audit report (ML20010D112).

Test Facility

NuScale built the NIST-1 test facility to model a scaled representation of the NPM major components with minimum distortions relative to the actual NPM in order and provide the measurements necessary for validation of the NRELAP5 model used for LOCA safety analysis. Figure 7-75, "Schematic of NuScale integral test facility and NRELAP5 nodalization," of the LOCA TR provides a schematic of the NIST-1 facility. Even though NuScale attempted to minimize the distortions between the NIST-1 scaled test facility and the NPM, the NRC staff notes that distortions cannot be eliminated. Therefore, the NRC staff evaluates the NIST-1 facility design and tests for NRELAP5 code evaluation against important LOCA phenomena and not as testing to directly evaluate the safety or acceptability of the NPM design.

The NRC staff finds that one of the significant distortions of NIST-1 relative to the NPM is that the NIST-1 representation of the RPV, is not contained within the NIST-1 representation of the CNV. In addition, the NIST-1 representation of the CNV is not immersed in the NIST-1 representation of the cooling pool vessel (CPV). The NIST-1 models of the RPV and CNV are separate vessels connected by piping. Figure 7-75 shows how the NIST-1 valves that represent the RRVs and RVVs and potential pipe breaks enables the NIST-1 facility to measure flow through this piping. The NIST-1 representation of the CNV is connected to the NIST-1 representation of the CPV through a heat transfer plate (HTP). The size of the NIST-1 HTP is scaled to represent energy transfer from the entire NPM CNV inside surface to the pool.

To approximate the NPM natural circulation flow, the NIST-1 test facility represents the NPM nuclear fuel with electrically heated rods. These NIST-1 electrically heated rods establish a natural circulation flow up through the riser to the NIST-1 SG and then back to the core like the NPM design. The NIST-1 system pressure is controlled by the pressurizer component which contains heater rods to bring the pressurizer fluid up to saturation temperature at the design system pressure.

As described in Section 7.5, "NuScale NIST-1 Test Assessment Cases," of the LOCA TR, data from the NIST-1 facility is used for both integral and separate effects validation of various phenomena. The NRC staff reviewed the NIST-1 facility design and determined that the areas of potential distortions were appropriately identified and the impact on the assessment results. One area that the NRC staff focused its review on, was the reduction of the CNV distortion. For certain NIST-1 tests, the NIST-1 CNV is preheated to reduce the distortion between the NIST-1 CNV arrangement and the actual NPM design. In NIST-1, **[[**

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As described in the associated audit report (ML19282C504), the NRC staff audited NuScale's NRELAP5 sensitivity studies used to establish the temperature criterion for preheating. NRELAP5 computations compare the impact of **[[**

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The NRC staff concludes that CNV preheating provides an acceptable approach to minimizing the distortion introduced by the separated CNV vessel in NIST-1 for the reasons specified above.

NIST-1 Integral Effects LOCA Test Procedure

As described in Section 7.5.1.6, "Steam Generators," of the LOCA TR, a valve and switch lineup is performed to configure the NIST-1 facility for each test. The NIST-1 line modeling the LOCA break location specified for the test, is connected between the RPV and its associated CNV penetration. Orifices with the specified diameters are installed in the RVV and RRV lines to model the number of valves that are to open when ECCS actuates. The NIST-1 facility operates at a lower pressure than the NPM, and the fluid masses are scaled. The NRC staff audited the facility and test procedures, as described in the associated audit report, and conducted QA inspections. These inspections are documented in inspection reports dated October 7, 2017 (ML15268A186) and July 24, 2017 (ML17201J382).

The NRC staff understands the limitations of the NIST-1 facility, which NuScale has accounted for in its test procedures. However, the NRC staff notes that these differences between NIST-1 and the NPM mean that a limited direct comparison can be made between NIST-1 tests to NPM

LOCA results. The NRC staff notes that NIST-1 is a test facility for LOCA code development and that NIST-1 is the only facility that closely represents an NPM to simulate a LOCA.

NIST Facility NRELAP5 Model

The NRC staff reviewed and audited details about NuScale's NRELAP5 nodalization model, as described in the associated audit report (ML20010D112), which is similar to the model used for the NPM and is described in Section 7.5.2, "Facility NRELAP5 Model," of the TR. The NRELAP5 model is a complete one-dimensional representation of the NIST-1 test facility. The NRC staff finds that the NuScale NIST-1 NRELAP5 model provides an acceptable representation of the NIST-1 test facility to evaluate the capability of NRELAP5 to model NIST-1 tests.

NIST Facility Test Matrix

The NRC staff reviewed NuScale's test matrix, given in Table 7-6, "Facility high priority tests for NRELAP5 code validation," of the LOCA TR and finds the suite of tests are sufficient to benchmark the NRELAP5 computer code and justify its use for LOCA analyses. Each of this series of tests is evaluated below regarding its applicability to NuScale.

Separate Effect High Pressure Condensation Tests

NuScale performed Test HP-02 to assess the capability of NRELAP5 to predict condensation rates at high pressure test conditions. While HP-02 was a quasi-steady test, a transient was performed to achieve the desired steady-state test conditions. The HP-02 test included direct measurements of the CNV pressure, CNV level, CNV temperature, and CPV temperature response. The CNV was a closed vessel, so, condensed steam (water) accumulated to produce a rising liquid level. Details of the test procedures are presented in the LTR, Section 7.5.4, "Separate Effect High Pressure Condensation Tests."

NuScale reports generally good agreement between NRELAP5 and test data for CLL, upper containment wall temperature and upper containment wall temperature at the cooling pool. The reported pressure is over-predicted by NRELAP5 as the absolute pressure increases. This is an important test because it is the only "larger" scale test available to validate the Extended Shah condensation modeling in NRELAP5. As described in the associated audit report (ML19282C504), the NRC staff audited NuScale's test assessments and reviewed information regarding the pressure overprediction (ML18256A361) and noted that the primary cause of over-predicting the HP-02 peak pressure conditions is due to how NRELAP5 calculates the condensate film thickness when two heat structures connected to a single volume are both acting as condensing surfaces. The HP-02 prediction of peak pressure is affected by this code limitation during the initial containment pressurization because both the shell wall heat structure and HTP heat structure are initially cold.

NuScale performed sensitivity calculations and the user input heated hydraulic diameter was modified based on the CNV shell and HTP geometry to account for the code treatment of liquid in a condensing volume when calculating film thickness. The sensitivity calculations result in a significantly improved prediction of the pressure rate of increase and peak pressure. The results of these sensitivity calculations show that within the pseudo-steady state period, NRELAP5 can predict well the condensation and heat transfer rates when a single surface is the dominant condensing surface. The limitation arises when there is more than one surface with significant condensation connected to a hydraulic cell. Once the CNV shell has ceased participation in the condensation process, the assumption that only one dominating condensing surface exists becomes valid.

The NRC staff's review recognizes that the two condensing surfaces can distort the condensation processes for HP-02; however, the adjustments made to code input only applies to the initial heatup. Therefore, the use of the laminar film condensation correlation would result in an under-predicted film condensation coefficient and over-predicted pressure. The NRC staff has determined that the current NRELAP5 computation of pressure in HP-02 demonstrates a conservative computation of pressure.

The NRC staff also reviewed data and NRELAP5 overlay boundary condition plots for inlet steam flow, pressure, and inlet steam temperature (superheated), as well as plots for CLL and condensation rate (ML18256A361). The NRC staff notes that while there is a general agreement between the temperature data and NRELAP5 computations, the CNV fluid temperatures at the lower elevation are noticeably over-predicted.

NuScale noted that the overall comparisons between the NRELAP5 results and data indicate a reasonable to excellent agreement. Considering instrumentation and other measurement uncertainties, the NRC staff considers the results to be reasonable. The cause of the containment pressure over-prediction in the HP-02 test is the NRELAP5 treatment of film thickness when two heat structures connected to a single volume provide condensation surfaces.

The NRC staff noticed that most of the Reynolds numbers are reported near 1000. The laminar film regime ends at a Reynolds number of about 30 and enters the wavy-laminar regime out to a Reynolds number of about 1800, and that regime has higher condensation heat transfer coefficients. Thus, under-predicted film condensation coefficients could contribute to the over-predicted pressure in HP-02 Run 3.

The applicant reported (ML18256A361) that the HTP thermal conductivity was estimated from **[[**

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The NRC staff reviewed the results of NRELAP5 using radial nodes of 23, 29 (base), and 36, provided by the applicant (ML18256A361). The results demonstrate that there is no dependence between the CNV pressure response and the three nodalization schemes investigated. The temperature profile through the HTP from the CNV to CPV shows minimal changes between the sensitivities, and, there is no discernible difference in the integrated condensation rates.

The NRC staff reviewed the capability of the NRELAP5 code to adequately represent thermal stratification in the NIST-1 CNV and the applicant's justification that the validation of condensation is accurate. The NRC staff reviewed the impact of node size on interfacial condensation at the steam-water interface in HP-02 and agrees with the applicant that node size had a small impact on the computed interfacial condensation and pressure.

As described in the associated audit report (ML19282C504), the NRC staff audited NuScale's assessment calculations that applied an adiabatic boundary condition to assess the impact of including shell wall heat losses on the containment pressure response in HP-02. The NRC staff noted that the applicant's sensitivity calculation results show **[[**

]]." The NRC staff also audited (ML19282C504) the NIST-1 HP-49 test nodalization sensitivity analyses of lower CNV 49 noding for NIST-1 test HP-49 that considered an inadvertent RRV opening. Three node size configurations were considered – coarse, base and fine. The NRC staff confirmed that the NRELAP5 model sensitivity calculation results for a NIST-1 inadvertent RRV transient event show that further refined or coarser CNV nodalization has an insignificant effect on the containment pressurization response compared to the base nodalization.

Natural Circulation Test at Power

NuScale performed NIST-1 test HP-05, to assess the capability of NRELAP5 to predict natural circulation flow at various core powers and test conditions by comparing experimental data and NRELAP5 predictions. The specifics of the test configuration are described by NuScale in its LOCA TR and the NRC staff reviewed these descriptions.

NuScale calculated form losses for a base run using Idelchik as referenced in the LOCA TR for the various geometric configurations around the loop. NuScale modified the losses based on the individual experimental differential pressures measured around the flow loop and then confirmed the global response by comparing the experimental loop flow rate to that predicted by NRELAP5.

The NRC staff reviewed the comparisons illustrated in the LOCA TR and agrees that NRELAP5 is capable of predicting primary flow rate, core inlet temperature, and core outlet temperature with a reasonable-to-excellent agreement for natural circulation flow conditions. The NRC staff audited the supporting testing reports and the measured data from Test HP-05, as described in the associated audit report (ML20010D112), and observed that it was a calibration test to refine hydraulic loss coefficients to improve the NRELAP5 computation of natural circulation flow rates observed in NIST-1. The NRC staff reviewed the LOCA base deck and noted that the RCS form losses, were only based on theoretical formulations from Idelchik for flow regions outside of the core and SG.

NuScale tests showed that major pressure losses are in the reactor fuel and SG and that they are well characterized by fuel vendor testing and large scale HCSG testing at SIET Laboratories, in Piacenza, Italy. As the formulas from Idelchik are widely accepted for single phase flow and the dominant loss through the NPM RCS is from the core and SG, which were well characterized by fuel bundle and SG head loss testing, the NRC staff found that the HP-05 test supported the conclusion that the NRELAP5 code is capable of predicting the natural circulation of an NPM.

Chemical and Volume Control System Loss-of-Coolant Accident Integral Effects Tests

NuScale performed Test HP-06 and HP-06b, to assess the capability of NRELAP5 to predict the integral response and multiple phenomena of the NIST-1 facility for a single-ended discharge line break inside containment. The specific test conditions and configuration are detailed in Section 7.6.5 of the LOCA TR, which the NRC staff reviewed. NuScale compared several parameters to assess an agreement with NRELAP5, including: direct measurements of the CNV pressure, RPV pressure, CNV level, RPV level, primary flowrate, break orifice differential pressure, pressurizer level, CPV temperature, CNV temperature, and HTP temperature.

For both tests, NuScale reports that the comparison between the calculated and measured results are in a reasonable-to-excellent agreement. The NRC staff audited the supporting test reports, as well as an evaluation of the NIST-1 HP-06 test results and the impact of preheating of the NIST-1 containment, as described in the associated audit report (ML20010D112), which provided a justification to show how the NRELAP5 code correctly calculates the temperature, enthalpy and mass fraction of vapor and liquid as the containment pressure increases with time.

The experimental data plot that the NRC staff audited, as documented in the associated audit report (ML20010D112), showed the measured liquid temperatures versus time at four elevations. The plot clearly shows the accumulation of thermally stratified subcooled water in the presence of walls that have been preheated. The temperature response shown by the CNV thermocouples matches the expected thermal stratification trends. No adverse effects due to preheating are observed. Thus, the NRC staff agrees with the applicant that the test is judged to be adequate to assess the ability of NRELAP5 to model the thermal stratification phenomenon.

As discussed in the associated audit report (ML20010D112), the NRC staff also audited an analysis of condensation at the steam-water interface in the CNV (pool condensation). Physically, the thermal stratification of the CNV pool has the effect of limiting surface condensation, particularly during the phase when the containment pressure is increasing. The analysis indicates that an upper bound on the pressure error due to over-prediction of pool condensation is less than one percent for the HP-06b test. The applicant stated that given the small impact of pool condensation on pressure results, the nodalization is appropriate for purposes of modeling pool condensation. The NRC staff found that this information provides sufficient justification for applying the NRELAP5 pool condensation model.

Assessment of NRELAP5 Prediction of Peak Containment Pressure

The NRC staff notes that NuScale relied on the NRELAP5 LOCA methods to perform peak containment pressure analysis. Figure 7-102, “NIST-1 HP-06b containment vessel pressure comparison,” of the LOCA TR, showed that NRELAP5 slightly overestimated the measured NIST peak containment pressure with a negligible deviation. As discussed in the associated audit report (ML20034D464), the NRC staff audited information that indicated that there were uncertainties in the NIST containment pressure measurement instrumentation and core heater rod center line thermocouple readings. In addition, NuScale identified the uncertainties associated with the NRELAP5 NIST-1 model initial and boundary conditions. The applicant used this revised NIST-1 modeling in its assessment report (ML18268A365) regarding HP-49 RRV opening test results.

NuScale evaluated the heater rod model uncertainties using three recently completed tests, which included the HP-43, the NLT-15p2, and the HP-49 tests. The NIST-1 RPV core is made up of electrically heated rods, some of which are fixed with internal thermocouples. Each heater rod contains a heater element that is inserted into a thermowell where a nominal 0.005-inch gap, between the heater element and the thermowell, is completely filled with boron nitride to maintain sufficient heat transfer within the heater element to moderate heater element temperatures. The heater rods then are seal welded at the top but remained open at the bottom. NuScale explained that over time, the gap of some of those heater rods may have lost some of the boron nitride resulting in higher element temperatures and higher initial stored energy than when they were newly installed. The applicant's initial base NIST-1 NRELAP5 model included a uniformly applied rod model with no fixed air gap in the rods for all NIST-1 tests. This approach resulted in a potential underestimation of initial rod stored energy, and according to NuScale, accounted for under prediction of CNV pressure in the early test assessments. In its examination of the HP-43 and HP-49 test data, during the Containment audit, as discussed in the associated audit report (ML19282C504), and QA inspection (ML19093A669), the NRC staff found that NuScale used the maximum of measured temperature data rather than averaged values. Although NRELAP5 inputs should be based on average temperature where data is available and that conservative input should be used for older previous tests where rod temperature data was not collected, the NRC staff also noted that it is likely that the boron nitride layer eroded with time as more tests were completed, suggesting that lower heater element temperatures would be more realistic, especially for the earliest tests.

The NRC staff performed sensitivity studies and determined that differences in the results with the lower realistic initial temperature were minimal and not large enough to affect the overall conclusions of the assessment. Therefore, the NRC staff found the applicant's analysis and rod modeling to be acceptable. The results show a conservative over-prediction of containment pressure by approximately 10 to 12 psi, therefore, there is sufficient margin to conclude that NRELAP5 adequately predicts peak CNV pressure for NIST-1 facility tests. Therefore, based on the review of HP-49 test results and the re-analyses of HP-06, HP-06b, HP-07, HP-09 and HP-43 using the revised NRELAP5 NIST-1 model, the NRC staff found the assessment results to be acceptable to justify the use of the NRELAP5 code to perform peak containment pressure analysis.

Pressurizer Spray Supply Line Loss-of-Coolant Accident Integral Effects Test

NuScale performed the HP-07 test benchmark to assess the capability of NRELAP5 to predict the integral response of the NIST-1 facility modeling a single-ended pressurizer spray supply line break inside containment. The phenomena evaluated in the HP-07 test were the same as those in the HP-06 test. The NRC staff reviewed the applicant's comparisons, which showed the comparison between that the calculated and measured results are in a reasonable-to-excellent agreement. The NRC staff agrees that they are in a good agreement and the assessment is therefore, acceptable.

Spurious Reactor Vent Valve Opening Test

NuScale performed the HP-09 test to assess the capability of NRELAP5 to predict the integral response of the NIST-1 facility modeling to inadvertent depressurization of the RPV initiated by a spurious opening of an RVV without DHRS. Furthermore, this test also provided a bounding depressurization rate for a LOCA initiated by break from pressurizer gas space.

The phenomena evaluated in the HP-09 test were the same as those in the HP-06 test. The NRC staff reviewed the comparisons provided in the LOCA TR, which include core power, RVV mass flow rate, RCS pressure and level, containment pressure and level. The NRC staff agrees that the comparison between the calculated and measured results are in a reasonable-to-excellent agreement and the assessment is therefore, acceptable.

4.8 Assessment of Evaluation Model Adequacy

In Section 8, "Assessment of Evaluation Model Adequacy," of the LOCA TR, NuScale presented its assessment of the adequacy of its LOCA EM based on the NRELAP5 computer code Version 1.4 and Revision 2 of the NPM plant base model for analysis of design-basis LOCAs. NuScale demonstrated LOCA EM adequacy by closure model and correlation reviews, and assessments against relevant experimental data. The NRC staff focused its review on being consistent with the EMDAP (RG 1.203).

4.8.1 Adequacy Demonstration Overview

Section 8.1, "Adequacy Demonstration Overview," of the LOCA TR provides a summary of the NuScale process for demonstrating model adequacy. NuScale used the results of its PIRT process discussed in Section 4 of the LOCA TR, to select the important phenomena for demonstrating LOCA model adequacy. The NRC staff's findings on the NuScale LOCA EM are provided below for each of NuScale's adequacy determinations.

4.8.2 Evaluation of Models and Correlations (Bottom-Up Assessment)

As discussed in Section 8.2, "Evaluation of Models and Correlations (Bottom-Up Assessment)," of the LOCA TR, NuScale evaluated the adequacy of NRELAP5 for modeling the PIRT high ranked phenomena by comparing NRELAP5 analyses against appropriate fundamental and special effects data. As discussed further below, the NRC staff reviewed NuScale's process for selecting fundamental and special effects test data to evaluate its LOCA EM for highly ranked PIRT phenomena and finds it to be acceptable because it conforms to the process described in RG 1.203.

Important Models and Correlations

The NRC staff reviewed NuScale's identified high ranked PIRT phenomena and the dominant NRELAP5 models and correlations required to assess these phenomena, as well as the key parameters, special situations associated with the phenomena and NRELAP5 assessments with NuScale and legacy test data used. As part of its review, the NRC staff reviewed information (ML17310B505) provided by NuScale explaining the validation of the modeling of flow through the ECCS valves. The NRC staff observed that **[[**

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The NRC staff assessed the adequacy of NRELAP5 for modeling interruption of natural circulation. The NRC staff reviewed information provided by NuScale (ML18031B319) to demonstrate the adequacy of modeling the core as two parallel channels without crossflow, including the applicant's computational assessment of crossflow modeling and CHF computations using VIPRE-01. The NRC staff agrees with the applicant's conclusions that the VIPRE-01 computations show that allowing full crossflow (base case), produces a higher flow rate and an associated lower void fraction in the hot assembly than for the other two restrictive crossflow models. The result is a larger CHFR when crossflow is allowed. The reported results support the applicant's conclusion that the closed channel model in NRELAP5 produces a more conservative CHF margin than an open channel model. Therefore, based on the above discussion, the NRC staff concludes that the modeling in NRELAP5 for the interruption of natural circulation is sufficient.

The NRC staff reviewed the applicant's estimated range of key NPM steady-state and design basis LOCA parameters that NuScale used to evaluate the adequacy of its LOCA EM TR models and correlations. NuScale stated that these parameter ranges shown in Table 8.2, "NuScale Power Module range of process parameters," of the LOCA TR identify the minimum range for demonstrating NRELAP5 adequacy, but that the applicability of models and correlations are not restricted to these ranges. NuScale determined that these parameter ranges from several sources including design values, proposed technical specification limits, and limiting initial and boundary conditions. NuScale obtained the ranges for some parameters from the NRELAP5 LOCA break spectrum calculations described in Section 9.0, "Loss-of-Coolant Accident Calculations," of the LOCA TR. The NRC staff finds that the NuScale process used for determining parameter ranges is acceptable because the values are based on the design, or are conservative, or are limited by technical specifications.

Two-Phase and Single-Phase Choked Flow (Mass and Energy Release)

As discussed in Section 4.6.6.1 of this SER, NRELAP5 employs a critical flow model that uses the **[[**

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The NRC staff reviewed NuScale's comparison of this model to **[[**

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The NRC staff agrees with NuScale's conclusions that the NRELAP5 comparisons to these two tests demonstrated a good agreement with the data during the subcooled portion of the tests, while over-predicting the break flow for saturated conditions, thereby displaying a conservative prediction of break mass flow rate. The NRC staff finds that these critical flow tests comparisons are sufficient to demonstrate an acceptable performance supported by the finding that the **[[**

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The NRC staff reviewed the applicant's comparison of the NRELAP5 model for **[[**

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The NRELAP5 code predicted-versus measured-pressure drop, of Figure 7-2, "Predicted versus measured pressure drop for selected contraction tests," of the LOCA TR, **[[**

]] showed an overall acceptable agreement. **[[**

]] The NRC staff reviewed the comparison results and agrees that the values are conservative. Therefore, the NRC staff finds this modeling approach to be acceptable.

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The NRC staff reviewed Table 8-6, "Dimensions of NuScale Power Module, NIST-1 and Bankoff pressurizer plate," in the LOCA TR, which indicates that **[[**

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The NRC staff noted that the verification of the correlation is based on the fact that the pressurizer drainage is well predicted in the test simulation. Moreover, the NRC LOCA verification runs with and without CCFL indicate that the results are not sensitive to CCFL at the pressurizer baffle plate or the core upper plate. Furthermore, regarding the limiting small break LOCA, all potential CCFL effects will have subsided due to the very low steaming rate at the time that the minimum liquid level is reached late in the event, where the liquid level and hence two-phase swelled level, in the vessel remains well above the top elevation of the core.
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]] Review of a CCFL paper by Stephen and Mayinger, "Experimental and Analytical study of Countercurrent Flow Limitations in Vertical Gas Liquid Flows," Chem. Eng. Tech. 15 (1992) pp 51-62, shows comparisons of the Wallis and Kutateladze forms of which Bankoff is intermediate, to a range of pressure flooding conditions up to about 200 psia. They importantly noted that the reducing effect of high gas-phase densities on gas velocities during flooding was satisfactorily predicted by these correlations.

Based on the NRC staff's assessment, as discussed above, the NRC staff finds **[[**
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Flashing

The NRC staff imposes no special findings or limitations on the modeling of NPM due to the flashing models implemented in the NRELAP5 code. NRELAP5 is judged by the NRC staff to be able to predict flashing during a depressurization event. This is irrespective of the non-conservative behavior shown by the interfacial drag model to accurately predict void and level swell separate effects test, where the model tends to over-predict two-phase levels and void distribution behavior in the axial direction.

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4.8.3 Evaluation of Integral Performance (Top-Down Assessment)

The NRC staff reviewed NuScale's primary areas described in LOCA TR Section 8.3, "Evaluation of Integral Performance (Top-Down Assessment)," of its top-down assessment, including the code governing equations, numerical solution and underlying assumptions; the integrated performance of the code against the IETs conducted in the NIST-1 facility; and calculations to evaluate differences and distortions between the NIST-1 facility and the NPM design.

The NRC staff reviewed the NRELAPS NIST-1 and NPM input models and found that these models were developed using generally consistent nodalization and option selection and that the IETs at the NIST-1 facility are appropriate for evaluating the key phenomena for NPM LOCA analyses.

Review of Code Governing Equations and Numerics

The NRELAP5 code governing equations and numerics are described in the LOCA TR, Section 8.3.1, "Review of Code Governing Equations and Numerics," and are the same as that of the original RELAP5-3D code. Therefore, the NRC staff determined a further in-depth review was not necessary. The NRC staff's review of the hydrodynamic model and field equations were discussed earlier in Sections 4.6.2 of this SER.

NuScale Facility Scaling

The NIST-1 facility is designed to simulate the integral system behaviors of a single NPM immersed in a reactor building pool. The NRC staff audited the applicant's scaling analysis, as discussed in the associated audit report (ML20034D464), to determine the NIST-1 dimensions and operating conditions. Distortion between the test facility and prototype is also analyzed in the scaling analysis. The hierarchical two-tiered scaling (H2TS) methodology was adopted by NuScale to scale the phenomena including RCS natural circulation, LOCA progression and ECCS operations. The NRC staff focused its review on confirming that non-dimensional numbers (π groups) representing phenomena are preserved for both the NPM and NIST-1 to capture the high-ranked phenomena identified in the LOCA PIRT. The FOMs in the NuScale design are the minimum CLL in the core, the peak CNV pressure and CHF.

NIST-1 has inherent distortions due to its small size and different component layout compared to NPM. Distortion also arises due to the difference of operating conditions in specific transients. The scaling analysis covered CVCS LOCA transient (HP-06) and additional LOCA scenarios: the high-point vent line break (HP-07) and inadvertent opening of RVV (HP-09). These break locations cover reactor coolant liquid space and vapor space. The NRC staff audited the applicant's NRELAP5 analyses for NIST-1 and the NPM, including its evaluation of scaling distortions and their impacts on FOMs.

Hierarchical Two-Tiered Scaling is a proven methodology developed by the NRC and has been used in several reactor designs. The NRC staff reviewed the scaling summary in the LOCA TR, supplementary information provided by the applicant (ML19058A867) and audited the details of the implementation of this methodology in the NuScale scaling analysis to determine whether it is appropriately used.

The NRC staff audited both stages of the applicant's scaling analyses. The first stage was steady state single-phase natural circulation in the RPV. In NIST-1, the maximum power level is scaled **[[ ]]** The NIST-1 vessel dimensions were determined in this stage. In the second stage, the applicant performed scaling on LOCA phenomena at different phases. Potential distortions were analyzed and identified through the difference in non-dimensional π groups.

The NRC staff reviewed the four groups of transient phenomena NuScale analyzed, including: vessel depressurization and containment pressurization during the blowdown and venting phases, the long-term recirculation phase and reactor building pool heat up. The NRC staff found that NuScale's scaling analyses correctly identified the control volumes of interest, the interactions between components and phases of event progression.

STEADY STATE NATURAL CIRCULATION OPERATION SCALING

The NRC staff reviewed the scaling ratios of the NIST-1 and NPM dimensions. The facility was designed to preserve event time and power-to-volume ratio. The NRC staff's audit focused on four areas of interest: the downcomer to lower plenum flow path, the central core region, the flow path between the upper riser and annulus, and the SG external flow. Among these, the SG frictional pressure losses dominate. The NRC staff found that NuScale applied appropriate scaling factors and initial steady state conditions with buoyancy forces balancing frictional losses, resulting in the correct flow rate comparisons. The one-to-one time ratio (isochronicity) requirement was met. Based on the analysis, the NIST-1 facility was designed to have a much higher loop resistance than the NPM. The NIST-1 SG scaling and the derivations of non-dimensional π groups for steady state natural circulation were confirmed by NuScale using additional NRELAP5 analysis with an excellent agreement of flow predictions and data.

The NRC staff reviewed single-phase natural circulation analyses for the NPM at different powers (100 percent and 50 percent) and for NIST-1 at different pressures and confirmed that there is not much effect from the pressure on the NIST-1 natural circulation flow. Therefore, the distortion due to lower pressure in the NIST-1, does not impact the results of natural circulation scaling. The matching of the natural circulation number and loop energy ratio lead to correctly scaled flow rates in NIST-1 compared to the NPM at 50 percent rated power. In the 100 percent NPM power condition, the scaled flow ratio and core temperature's distribution are slightly different than those in the 50 percent power condition. However, as the LOCA starts, the phenomena are the same if the decay power is scaled from 100 percent NPM decay power.

The NRC staff also performed TRACE and NRELAP5 confirmatory calculations and confirmed that the appropriate NIST-1 loop resistance was established to confirm the one-to-one time ratio requirement. Therefore, the NRC staff concludes that the scaling analyses for natural circulation are acceptable.

LOSS OF COOLANT ACCIDENT AND EMERGENCY CORE COOLING SYSTEM SCALING

Section 8.3.2.3, “Loss-of-Coolant Accident and Emergency Core Cooling System Scaling,” of the LOCA TR, summarizes the scaling of vessel depressurization, containment pressurization, long-term recirculation and building pool heat up phenomena for the CVCS line break event. The NRC staff audited the scaling approach detailed in NuScale’s scaling reports, as discussed in the associated audit report (ML20034D464), and concluded that it is correct in terms of identifying control volumes and phases of LOCA progression.

Vessel Depressurization

NuScale’s scaling formulation includes vessel mass and energy balance equations. **[[**

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Containment pressurization

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Long Term Recirculation Cooling Phase

The NRC staff reviewed NuScale’s top-down and bottom-up scaling flow path and notes that it correctly identified important phenomena that controls the steam flow and the return of condensate. The NRC staff also reviewed NuScale’s containment pressure equation, which is formulated with flow out of the vessel through the break and RVV, flow back into the vessel at the RRV, and heat loss to the pool. During this phase, the pressure drop between the RPV and CNV is determined by flow resistance and the flow rate. As the actual NIST-1 hydro-static driving head was scaled less than that of the NPM, the flow resistance in NIST-1 was evaluated to confirm the resistance ratio. For the CNV inventory balance, the two important phenomena are RRV flow and CNV wall phase change. RRV flow shows a large distortion, but in the acceptable range. For the energy balance, the most important phenomenon is the CNV wall heat transfer and the next important phenomenon is RVV energy flow. The distortions in both phenomena are less than 15 percent and are considered to be insignificant by the NRC staff. The NRC staff audited NuScale's calculations, as discussed in the associated audit reports (ML20034D464), and concluded that the scaling formulations and approaches for this phase are appropriate because there is less than 15 percent distortion.

Building Pool Heat up

The NRC staff audited NuScale's scaling related to the ultimate heat sink. NIST-1 has a HTP that connects the CNV with the pool, and the pool is a separate tank with its volume scaled as the power of the reactor for only one bay of the common pool. Therefore, the natural circulation pattern in NIST-1 is different than the multi-module pool in the NPM. Less horizontal thermal diffusion is expected. NuScale recognized the complexity of mixing behaviors in the stratified layer near the CNV wall but did not include the scaling of diffusion flows. The approach of neglecting the thermal diffusion in scaling is conservative since the diffusion helps cooling, and the diffusion flow will eventually reduce as the pool warms up after the initial period. Because the approach is conservative, the NRC staff finds it acceptable.

HIGH VENT LINE BREAK (HP07) AND INADVERTENT OPENING OF ONE RVV (HP09)

In addition to the CVCS break LOCA, the applicant performed scaling analyses for two other events: High Vent Line break (HP07) and inadvertent opening of one RVV (HP09).

The NRC staff reviewed the values of scaling groups at four snapshots in time during the high point vent line break (HP-07). The snapshots are: 1) when RPV pressure reaches 1500 psia, 2) near peak pressure, 3) right before ECCS actuation, and 4) long term cooling with RRV flow reversal. The NRC staff noted that the inventory is controlled by flashing and there is negligible distortion in flashing phenomenon for the RPV. The energy balance involves three important phenomena, break energy flow, core heat transfer and SG heat loss. The largest distortion is in SG heat transfer, but it is considered by NuScale to be small compared to break energy. In CNV scaling, the NRC staff noted that the scaling groups are small and the transient is mild, as compared to HP-06. The NRC staff concluded that in general, the dominant phenomena are well represented by NIST-1, as distortions are small in this transient.

HP09 is a fast transient, and the NRC staff reviewed the applicant's estimates for three snapshots in time: initial pressure, peak CNV pressure and long-term cooling.

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The NRC staff notes that even acceptable scaling distortions can have a cumulative effect on FOMs. The NRC staff observed this during its audit of NuScale's distortion report, as described in the associated audit report (ML20034D464). However, as the scaling groups for important phenomena were in the same order of magnitude for both NIST and the NPM, the NRC staff finds that the data are appropriate for code validation. The impact of scaling distortions is discussed below.

Assessment of NuScale Facility Integral Effect Test Data

The NRC staff reviewed NuScale's summary of NIST-1 IET tests that support NPM calculations and its assessment of data in these tests: HP-05, HP-06, HP-07, HP-09, HP-43 and HP-49, with NRELAP5. NuScale performed a sensitivity study for these tests to evaluate the potential uncertainties. NuScale used the agreement between the prediction and test data for these tests to demonstrate the applicability of NRELAP5 in modeling high-ranked phenomena in the NuScale design.

Evaluation of NuScale Integral Effects Tests Distortions and NRELAP5 Scalability

The NRC staff reviewed the applicant's summary of distortions concerning biases of initial/boundary conditions (IC/BC) and audited its actual operating procedure for the as-built

NIST-1 facility. The NRC staff audited the applicant's NRELAP predictions for tests HP-05, HP-06, HP-07 and HP-09. In addition, the NRC staff audited the applicant's three NPM sensitivity calculations for each transient (base case for proposed design without Appendix K assumptions, IC/BC case with the same initial and boundary conditions as in NIST-1, and a distortion case including scaling distortions). The NPM results were adjusted using scaling factors before comparing to NIST-1 data. The NRC staff audited the applicant's NRELAP5 prediction of NIST-1 tests with NIST-1 data, as described in the associated audit report (ML20010D112). The NRC staff concludes that the summary in the LOCA TR accurately described the distortions, which did not invalidate the scaling analysis results.

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The NRC staff concludes that, because of the size of the NIST-1 facility and its development history, there are scaling distortions. These distortions have been identified by the applicant and its impact on the FOM have been fully assessed and understood. The test results from NIST-1 were not only used to assess NRELAP-5 code, but also can be representative for NPM during LOCA and IORVs.

4.8.4 Summary of Adequacy Findings

The NRC staff reviewed the adequacy of the NRELAP5 code for analysis of design-basis LOCAs in the NPM and focused on the NuScale identification of key phenomenological models in a PIRT that are needed to successfully predict ECCS performance following a LOCA. This is demonstrated by choosing the proper closure models and correlations, and then assigning the many assessments against relevant separate effects tests and integral experiments to validate the important models listed in the PIRT. The NRC staff considers this a key step in establishing the adequacy of the NRELAP5 code as an acceptable component of the NuScale LOCA methodology as part of the EMDAP given in RG 1.203. The NRC staff also reviewed the subsequent steps to this objective, including documentation of the bottom-up assessment of the NRELAP5 models and correlations to determine their adequacy to predict the high (H) ranked phenomena in the PIRT, as well as a top-down assessment of the EM, including a review of EM governing equations and numerics to determine their applicability to NPM LOCA analysis, and evaluation of the integral code performance based on the assessments of the EM against relevant IETs. The NRC staff reviewed the applicant's summary in the LOCA TR of the adequacy findings, which showed how each PIRT high (H) ranked phenomenon is covered by the LOCA methodology models and correlations. The NRC staff also reviewed the applicant's identification of models that are marginally adequate, or ranges in which PIRT phenomena are not covered, and the manner of compensating for code limitations.

The LOCA TR identifies key models and correlations which are important to predicting the NPM LOCA ECCS performance following a LOCA. These are listed in Section 8.2, "Evaluation of Models and Correlations (Bottom-Up Assessment)," of the LOCA TR. The NRC staff noted that the list does not contain Baker Just for oxidation nor a rod swelling and rupture model since the acceptance criteria for acceptable ECCS performance does not include core uncovery. The Henry/Fauske Moody critical flow model meets Appendix K requirements, as well as the decay heat model, which uses the 1973 ANS standard with the 1.2 multiplier and inclusion of actinide decay.

The NRC staff notes that for current generation plants, downcomer boiling is an important phenomenon affecting the event progression following large break LOCAs. Because large breaks are not possible for NPM due to the design and the quick cool down of reactor vessel wall, the NRC staff agrees with NuScale that the downcomer boiling in NPM is not significantly large enough to produce a lower long-term liquid level in the core and riser region.

The NRC staff also agrees that these are relevant phenomenological models for simulating small break LOCAs in the NuScale NPM. Further, the NRC staff believes that the code predictions of the basic phenomena, such as these, with behavior observed in single situations created in individual separate effects test facilities allow a more focused and better assessment of the accuracy of the specific models in the code to be made than is possible using integral experimental data. This is because separate effects tests are dedicated to the study of a single particular phenomenological characteristic, so the measurement instrumentation can also be chosen more appropriately.

The 21 dominant NRELAP5 models and correlations for LOCA modeling, listed by the applicant in Section 8.2.1, "Important Models and Correlations," of the LOCA TR, were evaluated by the NRC staff in Sections 4.6.2 through 4.6.11 of this SER, and again summarized in Sections 4.8.2.1 through 4.8.2.22 of this SER. In brief, the NRC staff noted that the interfacial drag model was not considered accurate enough for determining the potential for core uncovery since the model over predicted the level swell and the axial void profile in many of the separate effect tests. However, the NRC staff concluded that these deficiencies would not have a significant effect on FOMs. This was noted in Section 4.6.8 of this SER as this modeling was found to be reasonably conservative in nature for the phenomenon of interest. The NRC staff concluded that the CNV condensation modeling is adequate to determine that the worst small break LOCA has been identified which displays the minimum liquid level in the core. Further, the minimum level worst case can be demonstrated to determine the liquid level above the top of the core, and the NRC staff believes the methodology is sufficient to predict the potential for two-phase uncovery of the core.

The NRC staff recognizes that there is a deficiency of integral test data against which the NRELAP5 code was validated against. And, further, there is only the NIST-1 facility that applies directly to the NPM design, which NuScale successfully compared and benchmarked the NRELAP5 code to. It is evident that the NRELAP5 modeling is capable of reproducing the NIST-1 LOCA results. From this, it is the NRC staff's judgement that it is not unreasonable to expect that NRELAP5 is capable of producing the NPM LOCA results.

In addition, NuScale successfully applied the NRELAP5 code to two Semiscale natural circulation tests. It is also noted that this facility is a much smaller scale, and there were no specific requirements for nodalization to successfully model natural circulation. As such, given that the condensation modeling was determined to use a conservative approach, the NRC staff found that the SET and IET code qualification effort supports the acceptance of the NRELAP5 code for evaluating ECCS performance following a small break LOCA in the NuScale NPM.

The NRC staff noted that distortions may compensate and result in seemingly conservative predictions for the tests. To determine the conservativeness of an EM, the applicability and scalability of the code need to be evaluated. An assessment of the applicability of NRELAP5 based on model correlations and bottom-up phenomenon is conducted and summarized in the LOCA TR, in Table 8-18, "Summary of bottom-up evaluation of NRELAP5 models and correlations," and Table 8-19, "Applicability summary for high-ranked phenomena," lists 21 high ranking phenomena. The applicant justified the applicability of the NRELAP5 code based on the agreement of NIST-1 data assessment. Based on the staff's evaluation of the applicant's applicability assessment the NRC staff considers this approach to be acceptable.

Due to the scale-dependent correlations used in the code, a scalability evaluation of NRELAP5 models was conducted. The applicant assessed the scalability issue in Table 8-18 of the LOCA TR by examining the scale dependency of important phenomena. These scale dependent models include choked flow, CCFL model, wall film condensation, riser flow regime and 3-D core flow distribution. The applicant either performed sensitivity studies on the coefficients of the correlations (e.g. CCFL model) or used conservative assumption for scale-dependent phenomenon (e.g. laminar regime film condensation heat transfer coefficient for turbulent regime) to ensure that the FOMs are not compromised. Based on the staff's evaluation of the applicant's scalability evaluation, the NRC staff found the approach acceptable.

4.9 Loss-of-Coolant Accident Calculations

NuScale stated that "the primary purpose of the break spectrum calculations and sensitivity studies is to support the development of the LOCA EM and to demonstrate its application for the evaluation of the NPM ECCS performance during postulated LOCAs."

The initial/boundary conditions and inputs for key LOCA EM parameters used for this analysis, are summarized in Appendix A of the LOCA TR.

4.9.1 Loss-of-Coolant Accident Progression in the NuScale Power Module

The NRC staff reviewed the applicant's LOCA analyses summarized in Section 9.1, "Loss-of-Coolant Accident Progression in the NuScale Power Module," of the LOCA TR, and audited the underlying calculations, as described in the associated audit report (ML20010D112). These calculations are for a representative liquid space break (100 percent break of the RCS injection line) and a representative steam space break (100 percent break of the high point vent line).

NuScale described representative LOCA scenarios that assume full-break area, no loss of ac or dc power, no single failure, and do not credit either decay heat removal train. The NRC staff agrees with NuScale that the representative LOCAs are appropriate to show a typical application of the NuScale 10 CFR Part 50, Appendix K LOCA EM. The NRC staff also noted that these may not be the limiting LOCA cases regarding the LOCA evaluations in Chapter 15 of the NuScale DCA, or any other application of this methodology to a specific design. Specific analytic results for LOCAs are evaluated as part of a design specific application of this methodology, such as the NuScale DCA.

Liquid Space Break

NRELAP5 calculates immediate choked flow for a 100 percent break of the RCS injection line. The mass and energy releases into the CNV through the break results in rapid pressurization of the CNV and depressurization of the RPV. The applicant described the sequence of events in Section 9.1.1, "Liquid Space Break," of the LOCA TR. The NRC staff reviewed the applicant's NRELAP5 calculated RPV and CNV pressure responses and noted that the NRELAP5 calculated energy release to the CNV through the break and ECCS valves is significantly larger than the energy release to the RPV by core heat transfer. The NRELAP5 calculated the heat transfer into the CNV wall results in continuous depressurization of both the RPV and CNV, after the initial pressurization of the CNV.

The NuScale analysis shows that the core remains covered and that MCHFR is not violated; thus, the peak cladding and fuel centerline temperatures are the values at a steady state before the LOCA. Based on its review of the applicant's representative calculations for liquid space breaks described in this section, the NRC staff concluded that the sample analyses appropriately illustrate that implementation of the methodology as specified in the TR will provide conservative and expected results.

Steam Space Break

The NRC staff also reviewed NuScale's analysis of a steam space break that occurs on the RCS high point vent line. Similar to the RCS injection line break discussed in TR Section 9.1, "Loss-of-Coolant Accident Progression in the NuScale Power Module," the 100 percent break on the high point vent line causes a reactor trip signal based on the high containment pressure followed by the reactor trip and containment and secondary isolation. NRELAP5 calculates reduced recirculation flow after the reactor trip; and NRELAP5 calculates a brief period of flow reversal in the average core channel with upward flow re-established on both average and hot fuel assemblies during the remainder of the transient.

The NuScale analysis shows that the MCHFR in the transient is at a steady-state and the MCHFR margin increases with time, due to the power and flow mismatch. Based on its review of the applicant's representative calculations for steam space breaks described in this section, the NRC staff concluded that the sample analyses appropriately illustrate that implementation of the methodology as specified in the TR will provide conservative and expected results.

4.9.2 Break Size

The NRC staff reviewed NuScale's spectrum of break areas for different break locations given in the LOCA TR. The break areas range from **[[**

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For all breaks, the break size impacts the timing of events because the break flow rate is proportional to the area for similar upstream conditions. The smaller break sizes produce slower depressurization and lower mass/energy loss rates. For example, events for the 10 percent break size take about 10 times as long as compared to the maximum break size.

The area ratios between break area and maximum break area presented in Table 5-7, "Summary of analyzed break sizes," of the LOCA TR are used to define "scaled time." This allows the presentation of computed results for different break areas to be presented on the same plot using "scaled time." This is important to interpret many of the figures presented in Sections 9.2, "Break Size," and 9.3. "Decay Heat Removal System Availability," of the LOCA TR.

NuScale presented the results of a spectrum of breaks in the discharge line, injection line, and high point vent, and pressurizer spray supply. In the calculations audited by the NRC staff, as described in the associated audit report (ML20010D112), NuScale assumes a 10 percent tube plugging as well as a fouling equivalent to 10^{-4} ft crud thickness on the inner surface, in sensitivity studies.

The NRC staff observes that the limiting break was the 5 percent break with the minimum liquid level above the TAF. This behavior was stated as a "direct" result of the ECCS actuation determined by the IAB release pressures where the ECCS valves do not open until the core collapsed levels are below the final equilibrium value. The NRC staff audited the sensitivity studies applied to this break spectrum, as described in the associated audit report (ML2010D112), including nodalization, time step size, CCFL at the pressurizer baffle plate, ECCS valve parameters (IAB release pressure differential threshold, size/capacity), core power distribution radial peaking assigned to the hot assembly, and CNV initial pool temperatures covering the range 140 ºF (60 ºC) down to and including 40 ºF (4 ºC). The most significant impact from the sensitivity studies were the DHRS unavailability (not credited in the LOCA analyses), IAB release pressure (low is most limiting for minimum liquid level), and proposed variation in ECCS valve sizes.

The NRC staff finds that these analyses and sensitivity studies contain sufficient parameter and break size variations to properly identify the limiting 5 percent injection line break that produces the minimum liquid level above the TAF. The NRC staff finds that the limiting small break injection line LOCA is correctly identified, based on the minimum liquid level above the TAF.

The NRC staff further observes that for the very small breaks where depressurization is very slow, the ECCS valves could remain closed for long periods of time, resulting in large losses of primary liquid. Most important however, is the heat transfer from the RPV to the liquid in the CNV which accumulates the lost primary liquid. It is the conduction of heat from the RPV into the CNV, in addition to the break that depressurizes the RPV to conditions that allow the ECCS valves, via the IAB release set point, to eventually open and thereby assure long term cooling. Since the operation of the DHRS is not credited in the LOCA analyses, the NRC staff considers LOCA events, particularly the smaller breaks in the spectrum, to be conservatively treated since this additional heat removal and depressurization mechanism provided by the DHRS is not credited. As such, the NRC staff notes that the IAB release set point, heat removal from the RPV into the CNV due to the lower temperature liquid in this region, and the DHRS all can effectively contribute to depressurizing the RPV and provide a means of assuring post-LOCA long term cooling for the NuScale NPM.

Based on its review of the applicant's representative calculations for different break sizes described in this section, the NRC staff concludes that the sample analyses appropriately illustrate that implementation of the methodology as specified in the TR will provide conservative and expected results.

4.9.3 Decay Heat Removal System Availability

The discussion in the previous sections indicates that the applicant took no credit for the DHRS operation. The DHRS adds an additional heat sink capacity during the NPM LOCA that is of benefit to the collapsed level above the TAF, primarily for the smaller break sizes. When the DHRS operation is taken into account, all break sizes behave similarly and minimum CLLs that are maintained very close to the final equilibrium level for most all of the break sizes. The NuScale reported analysis demonstrated that more adverse conditions are not created when crediting the DHRS. The NRC staff finds that not crediting DHRS for LOCA analyses is a conservative approach.

4.9.4 Power Availability

The discussion in the previous sections assumes that both ac and dc power are available during the NPM LOCA. Loss-of-power is defined as either loss of only ac power or loss of both ac and dc power. The loss of all power causes an immediate reactor trip and de-energizes the ECCS valves. However, the ECCS valves are not opened until the differential pressure between the RPV and CNV reach the IAB release set-point.

The NRC staff reviewed NuScale's analysis results showing that the loss of both ac and dc power has a significant impact on the steam space breaks, down to a two percent break size, but has a minimal impact on the liquid space breaks. The resulting peak containment pressures are not significantly higher than with all power.

Based on its review of the applicant's representative calculations for different assumptions on power availability described in this section, the NRC staff concludes that the sample analyses appropriately illustrate that implementation of the methodology as specified in the TR will provide conservative and expected results.

4.9.5 Single Failure

The NRC staff reviewed NuScale's results of analyses assuming the single failures described in Section 9.5, "Single Failure," of the LOCA TR. As described in Section 4.5.4 of this SER, specific LOCA event limiting single failures are evaluated as part of a design-specific application of this methodology, such as the NuScale DCA. This is reflected in item 6 under Section 6.0 of this SER. Based on its review of the applicant's representative calculations for different single failure assumptions, the NRC staff concludes that the sample analyses appropriately illustrate that implementation of the methodology as specified in the TR will provide conservative and expected results.

4.9.6 Core Collapsed Liquid Level Calculation

The NRC staff reviewed NuScale's summary of its calculation of the core CLL, and audited the calculations underlying it as described in the associated audit report (ML20034D464). The NRC staff notes that the minimum core CLL calculation is not the traditional axial formulation but is instead volume based. The axially based approach has some inherent conservativisms since it uses only stacked node height times the computed node liquid fraction, but it is the only method that can capture true "minimum CLL." The volume-based approach provides a distorted estimate of approximately an additional 2 ft (0.6m) of core level since it credits liquid that is in the riser that is not necessarily available to the core. The volume-based CLL term more appropriately represents "riser volume averaged liquid level" rather than a riser minimum CLL. This method could erode true core CLL margin such that MCHFR may be encountered in the hot channel before the volume-based CLL term reaches the TAF.

Because the LOCA TR acceptance criteria employs the CLL and CHF, both of which must be met, and because the NRC staff finds that the Hench-Levy/Griffith-Zuber correlation employed by the applicant is sufficiently conservative and reasonably bounding in terms of predicting CHF, the NRC staff accepts the applicant's use of a volume-based CLL. The NRC staff notes that the use of a volume-based CLL in the absence of a CHF criteria would not be acceptable, for the reasons cited above.

4.9.7 Sensitivity Studies

The NRC staff reviewed NuScale's evaluation of the sensitivity of the LOCA EM results to the changes in modeling parameters summarized in Section 9.6, "Sensitivity Studies," of the LOCA TR and audited the underlying calculations, as described in the associated audit report (ML20010D112). These parameters included nodalization, time-step size, counter current flow at the pressurizer baffle plate, and ECCS valve parameters (IAB release pressure, differential threshold, valve size/capacity, and valve stroke time). The NRC staff also reviewed NuScale's evaluation of the sensitivity of the LOCA results to the core power distribution, including axial power shape and hot fuel assembly radial peaking, and initial reactor cooling pool temperature.

The NRC staff also reviewed how NuScale used the sensitivity studies shown in Section 9.6 to support its selection of input values and modeling assumptions for its LOCA EM. These NuScale studies are limited to the range of breaks (2.23 in^2 to .05 in^2) as defined by the NuScale break spectrum given in Section 5.4, "Loss-of-Coolant Accident Break Spectrum, "of the LOCA TR. Based on its review of the applicant's representative sensitivity studies for varying the modeling parameters described in this section, the NRC staff concludes that the sample analyses appropriately illustrate that implementation of the methodology as specified in the TR will provide conservative and expected results.

Model Nodalization

The NRC staff reviewed NuScale's nodalization sensitivity study to determine the impact of nodalization on the key LOCA FOMs including containment pressure and collapsed RPV riser liquid level. To assess the impact of nodalization on the NPM LOCA behavior, NuScale evaluated the three nodalization schemes shown in Table 9-3, "Number of volumes in reactor pressure vessel and containment vessel nodalization," of the LOCA TR for the full range of break sizes for both the RCS injection line and the high point vent line breaks.

NuScale evaluated both break locations without DHRS operation, loss-of-power, and with no single failure. The NRELAP analyses results for the 100 percent high point vent line break are similar for the different NRELAP5 nodalization schemes. The coarser NRELAP5 nodalization in the containment generates slightly early ECCS actuation signal compared to the coarse and fine nodalization. NRELAP5 calculates similar LOCA response in RPV and CNV pressures and collapsed levels for the high point vent line break scenario.

The NRC staff did not identify any issues with the NuScale selected NRELAP5 nodalization for the calculation of the riser CLL. The NRC staff noted that the accumulation of subcooled water in the CNV resulting from a LOCA blowdown, is important to the computation of the maximum containment pressure. The nodal solution of NRELAP5 must be able to capture the stratification of subcooled water during blowdown for the determination of the maximum CNV pressure.

The NRC staff reviewed information relative to NuScale's CNV response analysis (ML18298A360 and ML19073A241) that evaluated the effects of using a set of coarser and finer axial nodalizations for the CNV volume, a finer reactor pool nodalization, and a finer CNV heat structure radial nodalization to determine the most limiting nodal representation with respect to CNV peak pressure and temperature. The information that the NRC staff reviewed, showed the expected thermal stratification with minimal temperature differences between the three nodal selections. The NRC staff also reviewed the applicant's information (ML19151A837) providing similar NRELAP5 containment model nodalization studies for the NIST-1 facility containment for the HP-49 test, which involved the largest RPV liquid discharge into the containment with the lowest elevation. The NRC staff concluded that the NIST-1 nodalization study results and convergence trends are very similar to that of the NPM nodalization studies, and the sensitivity of peak containment pressure to nodalization is not significant. Thus, the NRC staff concluded that the nodal selection has a small impact on the maximum CNV pressure.

Based on the discussion above, the NRC staff found that the NRELAP5 nodalization, used by NuScale, is appropriate to conservatively predict the CLL and CHF margin.

Time-Step Size Selection

NRELAP5 restricts time-step size by the courant time-step size and the accumulation of the mass-error during the time integration. NRELAP5 LOCA simulations set the courant time-step size to evaluate the effect of time-step size selection on the key NPM LOCA FOMs. The NRC staff reviewed Figures 9-26 through Figures 9-29 in the LOCA TR for the full-size injection line and high point vent line breaks, which illustrate that the maximum time-step size allowed for the NRELAP5 calculations is mainly determined by the mass-error management. These figures show that the containment and RPV pressures, minimum collapsed level above the TAF in the RPV riser, hot channel mass flux, and hot channel MCHFR, are all independent of the time-step sizes selected for the simulation.

Based on the information that the NRC staff reviewed above, and provided that all NRELAP5 calculations continue to show that the collapsed liquid riser level remains above the TAF, as is specified in the LOCA TR, the NRC staff finds the NRELAP5 time step selection process to be acceptable.

Counter Current Flow Limitation Behavior on Pressurizer Baffle Plate

The NRC staff reviewed NuScale's use of the Bankoff CCFL correlation at the pressurizer baffle plate with a slope of 1.0. A few of the break spectrum cases activated the CCFL flag at the pressurizer baffle plate, which did not allow liquid to readily drain from the pressurizer to the downcomer in the presence of upward steam flow. These break cases were limited to the larger pressurizer spray and vent line breaks. The NRC staff reviewed NuScale's analysis of liquid and steam breaks to assess the effects of increasing the Bankoff CCFL model slope between 1.0 and 2.0.

The NRC staff noted that the change in the CCFL slope had a significant effect on the immediate NRELAP5 calculated pressurizer level and this change also affected the instantaneous collapsed riser liquid level as shown in Figure 9-30, "Effect of counter current flow limitation line slope on levels for 100 percent high point vent line break," for the high point vent line break. A higher CCFL slope causes a lower CLL above the TAF because the water is held up in the pressurizer for longer periods of time. However, this change in the Bankoff slope did not impact the riser level for the entire transient because the pressurizer eventually empties and the responses merge before reaching the minimum CLL above the TAF. Hence, the slope input for the CCFL correlation has no impact on the FOM for the minimum collapsed riser level.

Because all NRELAP5 LOCA analyses continue to show that the pressurizer has completely emptied well in advance of the calculated riser CLL reaching the minimum value above the TAF, the NRC staff accepts the Bankoff slope used by NuScale.

Emergency Core Cooling System Valve Parameters

The staff reviewed NuScale's modeling of the ECCS valve characteristics. The NuScale DCA provides minimum and maximum valve sizes and a range of differential pressures at which the IAB arming valve closes (locks) and opens (releases). The staff reviewed NuScale's evaluation of liquid and steam breaks, which evaluate separate and combined effects of the range of these valve characteristics on the NRELAP5 calculated LOCA FOMs. The NRELAP5 calculated minimum riser collapsed liquid level shows no dependence on IAB release pressure for the larger breaks. However, for the smaller breaks (less than 35 percent), the staff noted that the minimum collapsed liquid level decreases with decreases in IAB release pressure because the ECCS actuation is determined by the IAB lockout and release pressures where the valves do not open until collapsed levels are well below the stabilized level entering the long-term cooling phase.

The staff also noted that large breaks (greater than 35%) cause relatively rapid RCS depressurization, and CNV pressure is not affected by the IAB release pressure. For the smaller breaks (less than 35 percent area), the peak CNV pressure is higher at a higher IAB release pressure because ECCS activation is determined by the IAB release pressure. However, the peak CNV pressures for these smaller breaks are still lower than the peak CNV pressure for larger breaks.

The NRC staff reviewed NuScale's sensitivity results, shown in LOCA TR Figures 9-32 and 9-33, which illustrate the effect of RRV and RVV valve sizes on the peak CNV pressure and the minimum CLL as functions of break size. Overall, the impact of the ECCS valve size on the peak CNV pressure and the collapsed riser liquid level is small. NuScale concludes that a larger RRV size and the lower IAB release pressure set points, generate lower minimum riser collapsed levels above the TAF. The NRC staff agrees with this conclusion as it is expected since larger RRV and delayed IAB release would increase RCS inventory lost before ECCS is actuated.

Additionally, NRC staff performed sensitivity of ECCS results with riser flow holes added and the expanded ECCS actuation signal based on RCS pressure that is interlocked with RCS hot temperature and CNV pressure. This feature results in earlier actuation of ECCS on pressure, particularly for steam space breaks and those with DHRS active. Small liquid space breaks and those without use of DHRS may continue to actuate on high CNV level.

Initial Reactor Pool Temperature

The NRC staff reviewed NuScale's sensitivity studies covering the range of initial pool temperatures to investigate the impact of the pool temperature on NRELAP5 calculated LOCA EM FOMs. NuScale evaluated the reactor pool temperatures, ranging from 40 ºF (4 ºC) to 140 ºF (60 ºC). NuScale evaluated that the RCS injection line breaks down to 5 percent of the full-break size break area, analyzed. The NRC staff noted that the effect of the initial pool temperature on the peak CNV pressure is more pronounced for smaller breaks. For a 100 percent injection line break, the break energy and energy release by the ECCS valve flows are larger than that of the CNV to the reactor pool energy transfer. The energy transfer from the CNV wall to the reactor cooling pool becomes comparable to the break energy for the smaller breaks; this results in significantly lower peak CNV pressures for smaller breaks. However, the maximum peak CNV pressure for all the break sizes considered, occurs at larger breaks. The applicant therefore concluded that the initial pool temperature has negligible impact on the maximum peak CNV pressure as NuScale's LOCA EM FOM. The slight increase in the peak CNV pressures at larger break sizes is conservatively considered by biasing the initial pool temperature to its maximum value. The applicant also reached a similar conclusion for the minimum CLL above TAF, as the maximum pool temperature produces lower momentary minimum levels at smaller break sizes. For all the initial pool temperatures investigated in the sensitivity calculation, the NRC staff notes that no CHF violation is observed; therefore, the minimum MCHFR is defined by the steady state value. Based on its review of NuScale's sensitivity studies, the NRC staff agrees that the maximum pool temperature is a conservative assumption for the NuScale maximum CNV pressure analysis.

Core Power Distribution

The NRC staff reviewed NuScale's sensitivity study of the impact of core power distribution for the full-range of the RCS injection line break sizes including axial power shapes and radial hot fuel assembly power peaking to investigate the effect of core peaking on NRELAP5 calculated LOCA FOMs. NuScale choose axial power shapes to represent **[[**

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[[
,**]]** the NRC staff finds that the core power distribution used in the NuScale LOCA EM, is acceptable.

4.9.8 Loss-of-Coolant Accident Calculation Summary

The NRC staff reviewed NuScale's summary conclusions based on its break spectrum calculations and sensitivity studies as listed in the LOCA EM TR Section 9.7, "Loss-of-Coolant Accident Calculation Summary." NuScale LOCA EM TR Section 9.7 presents a range of representative LOCA analyses used by NuScale to demonstrate that NRELAP5 is capable of evaluating NPM LOCAs against the primary FOMs (i.e. maintaining the CLL in the riser above the TAF and maintaining the CHF ratio above the MCHFR).

The NRC staff finds that the accidents evaluated appropriately cover the range of applicability to adequately show that NRELAP5 is capable of performing LOCA analysis to support the NuScale DCA.

In addition, the NRC staff performed extensive confirmatory analyses independently using both the TRACE and NRELAP5 computer codes. The scope of the confirmatory analyses includes the following categories of calculations:

a. TRACE NuScale NPM model development

   The NRC staff noted in audit reviews, as documented in the associated audit reports (ML20010D112), that NuScale used an ANSYS solids model to develop all the NPM geometry and mass information based on "cold" RCS dimensions from released drawings. Using the guidelines developed for NIST-1 modeling, NuScale used this NPM design information to develop the base NRELAP5 NPM input model. The NRC staff then used the NPM solids model design information along with the applicant's NRELAP5 base model to develop the NPM TRACE model, which models the core, the SG, the RPV and the containment with 3-D VESSEL components. TRACE VALVE components, control blocks and signal variables were used to model the ECCS valves and the RCS protection systems.

b. TRACE NPM Best Estimate LOCA and IORV Analysis

   Using the developed TRACE NPM model, the NRC staff performed a spectrum of LOCA analyses of different LOCA break locations and sizes. Sensitivity calculations were performed to confirm LOCA progression trends and to evaluate the impact of different single failure assumptions, IAB block and release pressure set points, and investigate margins to the key FOMs.

c. NRELAP NPM LOCA, IORV and Containment Pressure Analysis

The NRC staff audited NuScale NPM computer code and modeling, as described in the associated audit report (ML20010D112), and incorporated the NRELAP5 code and input models into the NRC's Symbolic Nuclear Analysis Package (SNAP) user interface processor for performing engineering analysis. Using these input models, the NRC staff performed LOCA and containment peak pressure analysis for the limiting break locations and the opening of different ECCS valves to better understand key characteristics of the NPM design.

d. TRACE NIST-1 Benchmark Analysis (HP-02, HP-05, HP-06b, HP-43, and HP-49)

Similar to studies for NPM, the NRC staff used NIST-1 design information and the applicant's NRELAP5 models to develop the NIST-1 TRACE model which modeled the core, the RPV and the containment with 3-D VESSEL components. Using the developed TRACE models and the audited NIST-1 data, the NRC staff performed independent assessments for the key NIST-1 tests.

e. NRELAP NIST-1 Benchmark Analysis (HP-02, HP-06b, and HP-49)

The NRC staff also performed sensitivity assessments with NRELAP5 and the applicant's modeling to better understand unique behavior and characteristics and results of the NIST-1 modeling and test benchmark results.

All these benchmarks and calculations confirmed to the NRC staff that the LOCA EM has adequate basis and validation to sufficiently predict key FOMs for the NPM design (i.e., the collapsed water level remains above the TAF, the MCHFR is not violated, and the peak containment pressure is much lower than the design limit pressure).

5.0 EVALUATION MODEL FOR INADVERTANT OPENING OF RPV VALVES

5.1 Event Description and Classification

An accidental IORV (i.e., RSV, RVV, or RRV) results in reactor vessel depressurization and a decrease of reactor vessel coolant inventory that could be caused by a spurious electrical signal, hardware malfunction, or operator error. The EM and methodology applied to analyze SRP Section 15.6.1, "Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve," and SRP Section 15.6.6, "Inadvertent Operation of the Emergency Core Cooling System (ECCS)," events are developed by extending the LOCA Methodology. These inadvertent RPV valve events are classified by NuScale as AOOs.

The applicant's description confirms that each ECCS valve system includes an IAB valve to block the main ECCS valve from opening based on a set release pressure difference threshold between the CNV and RPV. The IAB effectively reduces the frequency of inadvertent openings of the valves during power operation. The limiting event analyzed is the mechanical failure of an ECCS valve that depressurizes the control chamber at operating pressure.

5.2 Evaluation Model

The NRELAP5 model utilized for IORV analysis is developed from a base plant model that is modified for the important aspects for the IORV, with similar modifications that are applied to the LOCA EM as described in Section 5.1, “NRELAP5 Loss-of-Coolant Accident Model for the NuScale Power Module,” of the LOCA TR. The NRC staff reviewed the primary differences in the modeling, which are related to those necessary to better align the analysis with AOO acceptance criteria instead of postulated accident criteria.

The NRC staff reviewed the overall EM objectives for mitigation, which are the same as for LOCAs in that: (1) the CNV must contain the loss of inventory from the RCS, (2) the remaining ECCS valves must actuate to depressurize the RPV into the CNV until pressure equalization, which allows the return of discharged fluid back into the RPV to cool the core, and (3) stable natural circulation flow must be maintained via ECCS steam condensation cooling to the reactor pool. Initial plant conditions are conservatively biased similar to the initial conditions used for LOCA analyses.

The applicant used NRELAP5 LOCA modeling methods for its analysis of IORV events because the transient progression of the event and the PIRT phenomena are similar to that of LOCA pipe breaks. Therefore, the NRC staff’s review focused in the areas of the differences from the LOCA modeling. The method specifies selection of input parameters and initial conditions to provide a conservative calculation relative to MCHFR since IORV is the limiting event for CHFR. The NRC staff reviewed and audited the core modeling, as described in the associated audit report (ML20034D464), that is based on heat transfer options that specify the extended Hench-Levy CHF correlation for high flow conditions, and the Griffith-Zuber CHF correlation for low flow conditions. The initial conditions and biasing for the steady state portion of the transient are the same as for a LOCA.

The NRC staff reviewed the information on those differences (ML18264A338), which are:

1. The ECCS valve opening stroke is reduced to 0.1 seconds for faster opening, which produces higher flow rates and faster depressurization.

2. The ECCS valve inadvertent opening choked flow is modeled as a break with Moody/Henry-Fauske (c=3) to maximize two-phase flow which is consistent with LOCA break methodology.

3. An Additional 2 second scram delay conservatism used in a LOCA is removed (not used for AOO calculations).

4. The 15 percent bias on fuel thermal conductivity and heat capacity to increase stored thermal energy for a LOCA is removed (not used for AOO calculations).

5. A fuel gap conductance is varied via sensitivity analyses instead of the bounding value used in a LOCA.

6. Although the same CHF correlation methods are used, less limiting 95/95 tolerance limit of 1.13 is applied in AOO calculations.

The NRC staff further reviewed the following modifications and audited the associated underlying calculations and sensitivity analyses, as described in the associated audit reports (ML20034D464). The initial conditions and biasing included sensitivity cases for: (1) all electric power available, (2) loss of normal ac power, and (3) loss of normal ac and dc power. The loss of dc power will impact the LOCA progression by immediately triggering the ECCS valves to go to their fail-safe position, where each valve is held closed by its IAB.

The single failure assumptions included: (1) no single failure, (2) failure of a single RVV to open, (3) failure of a single RRV to open, and (4) failure of one ECCS division (i.e., one RVV and one RRV). The NRC staff noted that the failure of an IAB to block (which would cause a second ECCS valve to prematurely open above the IAB threshold release pressure) was not considered. The treatment of the IAB valve's function to close relative to single failure is discussed further in Section 4.5.4 of this SER. Therefore, for analysis of these events, based on the applicant's submittal and the discussion in Section 4.5.4 of this SER, the inadvertent operation of ECCS consists of the opening of one RVV or one RRV.

The NRC staff also reviewed and audited sensitivity cases on model parameters for fuel rod gap conductance, axial power shape, ECCS valve sizing, ECCS valve opening rate, and DHRS availability. The IORV event is modeled as a mechanical failure resulting in opening of a single ECCS valve. The valve opening is the initiating event and partial valve opening is not considered as a single failure.

Specific limiting single failures, electrical power assumptions, and whether any operator actions are needed for this event are evaluated as part of a design-specific application of this methodology, such as the NuScale DCA. This is reflected in item 6 under Section 6.0 of this SER.

For the typical inadvertent RVV or RRV opening with or without the loss of ac power, the MCHFR occurs very early in the transient before the rods are fully inserted from the reactor trip on high containment pressure. The remaining ECCS valves open much later, when the ECCS system actuates on the high containment level. Minimum water level above the core occurs as the RPV and containment water levels equalize. The overall RRV transient is like the RVV. However, the liquid-space discharge results in a slower depressurization, accompanied by a greater decrease in core inlet flow as coolant discharges from the downcomer region into the containment. The liquid-space discharge generates an ECCS actuation signal on the high CNV level that occurs earlier than for the RVV transient. After the remaining ECCS valves open, the RRV scenario and the RVV scenario follow similar trends for fluid conditions and heat transfer for long term cooling.

The NRC staff reviewed NuScale's sensitivity analyses for key model parameters including: (1) Fuel Rod Gap Conductance, (2) Axial Power Shape, (3) ECCS Valve Sizing, (4) ECCS Valve Opening Stroke Time, (5) DHRS Operation, (6) Single Active Failures, and (7) Electric Power Availability. The NRC staff also reviewed the limiting MCHFR case for RVV inadvertent opening with power available, high RCS average temperature, low RCS flow, high RCS pressure, maximum gap conductance, and high PZR level, however the NRC staff's review of the final limiting case will be documented in the NRC staff's evaluation of a NPM application (e.g. the staff's SER for the NuScale DCA). The NRC staff's review was intended to confirm that the

LOCA/IORV methodology is capable of predicting the limiting cases, and the expected key event sequences and parameter trends.

The applicant's assessment methodology for the IORV events is based on LOCA experimental test benchmarks, inclusive of SETs for CHF and IETs for phenomenological system responses (per Table 7-1, "NRELAP5 loss-of-coolant accident assessment matrix," of the LOCA TR). The high-ranked phenomena derived in the NuScale LOCA PIRT are indicated by the applicant as identical to the IORV event, since the progression of the transient is very similar. The NRC staff noted that although the progression is similar, the effective break flow areas of IORV events are much larger. The LOCA break sizes were limited to 2.0-inch pipes in CVCS or PZR spray, whereas limiting flow areas for ECCS valves are much larger, 2.5-inch for the RRVs and 4.75-inch for the RVVs. Additionally, the NRC staff observed that the RRVs are located at a lower elevation relative to the TAF in comparison to the CVCS charging and discharge lines. The LOCA PIRT high-ranked phenomena (per Table 4-4, "High-ranked phenomena," of the LOCA TR) included and evaluated the RRVs and RVVs but only as related to ECCS actuation, i.e., phase 1b, and not as an initiating event.

The NRC staff concludes that since the RRV flow area and location within the RPV are relatively similar, two-phase critical flow phenomena (knowledge level 2) can be extended from the CVCS break assessment. For the RVVs, the flow area is significantly larger but the critical break flow phenomena is primarily single phase (knowledge level 4).

The applicant additionally uses the NIST-1 Integral Effects Tests, HP-06b and HP-43, to provide assessments for NRELAP5 predictions of key critical flow phenomena. The HP-43 test is an updated version of the HP-09 test where the depressurization of the RPV is initiated by a spuriously opened RVV sized to represent one of three RVVs, and the remaining RVV, which opens on ECCS actuation, is sized to represent flow from two RVVs. The NRC staff audited this test and the test assessments, as described in the associated audit report (ML20034D464), and concludes that the code predictions adequately matched the data trends and key FOMs. The NRC staff also reviewed supplementary information (ML18268A365) and audited the applicant's test results and assessment reports for NIST-1 HP-49, as described in the associated audit report (ML20034D464), which provide data for a spurious RRV opening event. The NRC staff similarly concluded that these code predictions also adequately matched the FOMs.

Based on its review of the applicant's representative calculations for the different IORV events with single failure assumptions described in this section, the NRC staff concludes that the sample analyses appropriately illustrate that implementation of the IORV methodology as specified in Appendix B of the TR (1) is consistent with LOCA EM methodology such that the validations for LOCA are applicable to the IORV event and (2) will provide conservative and expected results.

The NRC staff also performed sensitivity calculations of the IORV results with riser flow holes added and the expanded ECCS actuation signal based on RCS pressure. The results indicated that the limiting CHF are not affected by these design changes.

5.3 Accident Scenario Identification Process

The EM review criterion per RG 1.203, recommends that applicants follow a structured process for the identification and ranking of physical phenomena relevant to the accident scenarios to which the EM will be applied. For the NPM, the applicant has indicated there are no significant differences in the physics of the fundamental phenomena between the LOCA and IORV events, i.e., the initiating locations and effective break flow areas are different but the governing thermal hydraulic code processes would be very similar. NuScale indicated that the high-ranked phenomena from the LOCA PIRT shown in LOCA TR Table 4-4 also apply to the IORV event scenarios, and that the event phases of initial blowdown (1a) and ECCS actuation (1b) are also identical for LOCA and IORV.

5.4 CHF Evaluation

The NRC staff's evaluation of the high-flow and low-flow CHF models uses the critical boiling transition model assessment framework developed in Appendix A of the safety evaluation (SE) for TR-0116-21012-A, "NuScale Power Critical Heat Flux Correlations," (ADAMS Accession No. ML18360A632). This framework assesses the CHF correlation through a top-down approach whereby the high level finding that, "the CHF correlation is acceptable," is broken down into lower level goals (G). The lowest level goals are directly supported by evidence.

In addition to the material submitted by NuScale, the NRC staff considered the historical and publicly available data from NUREG/CR-1559, "Transient Critical Heat Flux and Blowdown Heat-Transfer Studies," (Office of Science and Technical Information (OSTI) Identifier 5824873) during their review.

The NRC staff notes that information pertaining to the high-flow and low-flow CHF models is also presented in Section 5.1.8.3, Section 6.11, and Section 7.3 of the TR. The NRC staff's review of the CHF information in TR-0516-49422 is provided in Section 5.5 below.

5.5 Experiential Data

Section B.5.1 of the TR describes the testing used to validate the high-flow CHF correlation, which was obtained at the KArlstein Thermal HYdraulic test loop (KATHY) in Karlstein, Germany. Additionally, Section 6.11, Section 7.3, and Section 5.3 of the TR clarify that Stern CHF data is also used in the development of the high-flow and low-flow CHF correlations as implemented into NRELAP5. The NRC staff previously reviewed the experimental data supporting the CHF model development as part of the review of TR-0116-21012-A, "NuScale Power Critical Heat Flux Correlations," (ADAMS Accession No. ML18360A632). Accordingly, the NRC staff's findings for all of the items under G1 of the assessment framework, documented in the SE for TR-0116-21012-A, are applicable to the evaluation of the high-flow and low-flow CHF correlations with the exception of G1.3.1 and G1.3.2. G1.3.1 and G1.3.2, which are evaluated for the high-flow and low-flow CHF correlations below.

Equivalent Geometries

The test bundle used in the experiment should have equivalent geometric dimensions to that of the fuel bundle used in the reactor for all major components.

G1.3.1, Review Framework for Critical Boiling Transition Models

The NRC staff's SE for TR-0116-21012-A, "NuScale Power Critical Heat Flux Correlations," (ADAMS Accession No. ML18360A632) established the finding that the test bundles used to obtain validation data for the NSP2 and NSP4 CHF correlations have equivalent dimensions to that of the fuel bundle in the reactor for all major components because: (1) the data collected from the KATHY test loop used test bundles that are representative of prototypical NuFuel-HTP2TM fuel, and (2) the KATHY test loop data was used for the validation of the NSP2 and NSP4 CHF correlations. This finding is applicable to the high-flow CHF correlation because the KATHY data is used to validate this correlation. However, the Stern data is used to validate the low-flow CHF correlation, and the test bundle used at Stern was a preliminary prototype fuel that is not prototypical of NuFuel-HTP2TM fuel.

Section B.5.3 of the TR describes that no KATHY test data exists in the range where the low-flow CHF correlation is used. Therefore, the applicant proposed to use the subset of Stern data that was collected for lower flow rates. In Section 6.1, "Comparison of Stern Preliminary Prototypic to KATHY NuFuel-HTP2™ Test Data," of TR-0116-21012-A, "NuScale Power Critical Heat Flux Correlations," (ADAMS Accession No. ML18360A632), the applicant compared the results of CHF testing from Stern to the results from the KATHY test loop. This comparison demonstrated that **[[**

]] Based on the information provided in Section 6.1 of TR-0115-21012-A, the NRC staff finds the use of Stern data, to validate the low-flow CHF correlation, to be acceptable because the applicant demonstrated that the grid spacers used in the NuFuel-HTP2TM fuel design do not adversely impact CHF performance.

Equivalent Grid Spacers

The grid spacers used in the test bundle should be prototypical of the grid spacers used in the reactor assembly.

G1.3.2, Review Framework for Critical Boiling Transition Models

The same argument that was applied to G1.3.1 also applies to G1.3.2. Specifically: (1) the KATHY test loop data that is used to validate the high-flow CHF correlation used grid spacers that are prototypical of the grid spacers used in the reactor assembly, and (2) the Stern test data that is used to validate the low-flow CHF correlation does not use a prototypical grid spacer but is acceptable because the applicant demonstrated that the grid spacers used in the NuFuel-HTP2TM fuel design do not adversely impact CHF performance.

5.5.1 Model Generation

5.5.1.1 Mathematical Form

Sections 6.11.3, **[[ ]]** and 6.11.4, **[[ ]]** of the TR describe the high-flow and low-flow CHF correlations, respectively. NuScale selected historical CHF models as the basis for CHF modeling in the TR.

The NRC staff compared the high-flow CHF correlation with the **[[ ]]** presented in Todreas and Kazimi[5] and finds them to be consistent. However, NuScale **[[ ]]** (discussed further in Section 5.5.1.2 of this SER). The NRC staff compared the low-flow CHF correlation with the **[[ ]]** presented in Appendix G of NUREG/CR-1559 and finds that the parameters and forms are consistent. The NRC staff observed that the coefficient for the low-flow CHF model, given by Equation 6-105 of the TR, differs from the value provided in NUREG/CR-1559 but it is acceptable because the coefficient used in the TR is smaller, resulting in a conservative calculated value for CHF. Additionally, the low-flow CHF correlation uses a CHF multiplier which, as discussed further in Section 5.5.2 and 5.5.3.2 of this SER, resulted in the NRC staff creating Limitation 7 on the application of the low-flow CHF correlation. Based on the information provided in Sections 6.11.3 and 6.11.4 of the TR, and subject to Limitations 7 and 8 for correlation applicability ranges, the NRC staff finds that the high-flow and low-flow CHF correlations contain the appropriate parameters (G2.1.1) and have an acceptable model form (G2.1.2) because these models are consistent with historical models that have been tested and validated.

5.5.1.2 Model Coefficient Generation

5.5.1.2.1 Training Data

> **Training Data**
>
> *The training data (i.e., the data used to generate the coefficients of the model) should be identified.*
>
> G2.2.1, Review Framework for Critical Boiling Transition Models

Sections 6.11.3 and 6.11.4 of the TR describe the high-flow and low-flow CHF correlations as historical correlations with a modification to the high-flow CHF correlation. Section 6.11.3 of the TR describes the modification to the high-flow CHF correlation as a **[[ ]]**. Specifically, the discussion clarifies that **[[ ]]** using CHF data collected at Stern Laboratories because the data collected at Stern Laboratories **[[ ]]**. Section B.5.3 of the TR discusses the development of the CHFR limit (i.e., validation) for the high-flow CHF correlation as using data collected at the KATHY test loop for prototypical Nufuel-HTP2TM fuel, which is independent of the data collected at Stern Laboratories. Section 6.11.4 of the TR clarifies that no data was collected to train the low-flow CHF correlation (i.e., the collected data was used only for validation purposes). Based on the information provided in Section 6.11.3, 6.11.4, and B.5.3 of the TR, the NRC staff finds the selected training data to be acceptable because it is independent of the validation data.

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5 N. E. Todreas and M.S. Kazimi (1990), *Nuclear Systems I – Thermal Hydraulic Fundamentals*, Taylor & Francis

5.5.1.2.2 Coefficient Generation

> **Coefficient Generation**
>
> *The method for calculating the model's coefficients should be described.*
>
> G2.2.2, Review Framework for Critical Boiling Transition Models

Section 6.11.3 of the TR describes the curve fitting for the **[[** **]]** of the high-flow CHF correlation. The resulting coefficients and **[[** **]]** are provided in Table 6-5 and Figure 6-6 of the TR, respectively. Based on the description provided in Section 6.11.3 of the TR, and the results shown in Figure 6-6 of the TR, the NRC staff finds the coefficient generation to be acceptable because the resulting **[[** **]]** trends are consistent with the data from Stern Laboratories.

5.5.2 Model Validation

5.5.2.1 Validation Error

> **Validation Error**
>
> *The validation error has been correctly calculated.*
>
> G3.1, Review Framework for Critical Boiling Transition Models

NuScale uses a database to assess and validate the high-flow and low-flow CHF correlations, which are based on historical CHF correlations. The validation data for the high-flow and low-flow CHF correlations are obtained from testing the NuFuel-HTP2TM simulated fuel. The high-flow correlation applies the **[[**

]]

Section B.5.3 of the TR explains that a comparison between the NRELAP5 predicted CHF values and measured KATHY test values **[[**

.**]]** The NRC staff finds this approach to be acceptable because it results in a conservative CHFR limit.

Section B.5.3 of the TR explains that there is no KATHY CHF test data for the low-flow CHF correlation conditions. Therefore, the CHF limit derived from the Stern CHF data described in TR Section 7.3.6, "Assessment Results," is used. **[[**

.]] Therefore, the NRC staff finds this approach to be acceptable because it results in a reasonable CHFR limit and the CHFR limit is typically challenged within the range of the high-flow correlation.

Section 7.3.5 and B.5.2 of the TR describes the NRELAP5 model for the CHF tests. The TR states that the **[[**

]]. The NRC staff reviewed the NRELAP5 test models and comparisons and confirmed that an acceptable measured value is used for the comparison.

Based on the acceptable comparisons of measured (test data) vs predicted (calculated) power, the NRC staff finds that the calculated validation error (CHFR Limits) is acceptable and is calculated using an acceptable measured value.

5.5.2.2 Data Distribution

5.5.2.2.1 Validation Data

> **Validation Data**
>
> *The validation data (i.e., the data used to quantify the model's error) should be identified.*
>
> G3.2.1, Review Framework for Critical Boiling Transition Models

TR Sections 7.3.6 and B.5 describes the validation process used to establish the CHF correlation limits for the high-flow and low-flow CHF correlations. The methodology for determining the correlation limit is identical to that used in the development of the NSP2 and NSP4 CHF correlations described in TR-0116-21012, "NuScale Power Critical Heat Flux Correlations," Revision 1, issued November 2017 (ADAMS Accession No. ML17335A089), which the NRC staff has reviewed and approved (ADAMS Accession No. ML18214A480). The referenced TR explains that **[[**

]]. The limits for these correlations were established by comparison to test data which is completely independent of the historical correlation development. Based on the use of historical correlations for comparison to test data and the **[[** **]]**, the NRC staff finds the use of the data to perform validation and to determine the CHFR limit to be acceptable.

5.5.2.2.2 Application Domain

> **Application Domain**
>
> *The application domain of the model should be mathematically defined.*
>
> G3.2.2, Review Framework for Critical Boiling Transition Models

Table 6-6 of the TR provides the application domain for the high-flow CHF correlation. The low-flow correlation application domain is characterized by **[[** **]]** in TR Section 6.11.4. **[[**

]] Therefore, the NRC staff finds the **[[** **]]** to be acceptable.

The NRC staff confirmed that the test data used to validate the high-flow correlation covers the high-flow correlation application range. The NRC staff observed that the test data used to validate the low-flow correlation **[[**

]].

The NRC staff finds the definition of the application domain to be acceptable because: (1) the high-flow correlation application domain is supported by test data (2) the low-flow correlation application domain is **[[**

]].

5.5.2.2.3 Expected Domain

> **Expected Domain**
>
> *The expected domain of the model should be understood.*
>
> G3.2.3, Review Framework for Critical Boiling Transition Models

In TR-1113-5374-NP, "Critical Heat Flux Test Program Technical Report," January 2014 (ADAMS Accession No. ML14024A455), Section 2.5, the applicant performed preliminary analyses of the NPM to develop conditions covering normal operation, AOOs and postulated accidents. The NRC staff compared the range of test conditions, identified in Table 2-5, "Range of test conditions," of TR-1113-5374-NP, with the range of applicability of the high-flow and low-flow CHF correlations, provided in Table 6-6 and Section 6.11.4 of the TR, and determined that the range of test conditions encompasses the range of applicability for the high-flow correlation and is reasonable for the low-flow correlation. Based on the performance studies of the NPM presented in TR-1113-5374-NP, the NRC staff finds the expected range to be acceptable.

5.5.2.3 Data Density

Data Density

There should be an appropriate data density throughout the expected domain.

G3.2.4, Review Framework for Critical Boiling Transition Models

Table 6-6 and Section 6.11.4 of the TR mathematically define the application domains for the high-flow and low-flow CHF correlations, respectively. Based on prior experience with CHF correlation reviews, the NRC staff recognizes that the defined application domain of a CHF correlation contains regions where there are no or sparse underlying experimental data and where the correlation may not be used. The NRC staff conducted an audit as part of the review, which included the data density throughout the expected domain, as described in the associated audit report (ADAMS Accession No. ML20034D464). During the audit, the NRC staff observed several plots that show the data collection within the expected domain. These plots show that the **[[**

]]. Based on the information observed during the audit, the NRC finds that there is appropriate data density throughout the expected domain.

5.5.2.4 Sparse Regions

Sparse Regions

Sparse regions (i.e., regions of low data density) in the expected domain should be identified and justified to be appropriate.

G3.2.5, Review Framework for Critical Boiling Transition Models

As described in Section 5.5.2.3 of this SER, the expected mass flux and pressure regions have appropriate data density throughout the expected domain. Additionally, Section 5.5.2.3 states that the plot comparisons to data show **[[**

]]. Accordingly, the NRC staff finds that the sparse regions in the expected domain are identified and appropriately justified.

5.5.2.5 Restricted Domain

Restricted Domain
The model should be restricted to its application domain.
G3.2.6, Review Framework for Critical Boiling Transition Models

Table 6-6 and Section 6.11.4 of the TR mathematically define the application domains for the high-flow and low-flow CHF correlations, respectively. Additionally, Section 1.1 of the TR states that approval of the EM, which includes the CHF correlations range of applicability, for design basis LOCA events is requested. Section 1.1 of the TR extends the purpose of the EM to include IORV. The NRC staff finds that this is consistent with the established precedent and is sufficient for restricting the domain of applicability in conjunction with the limitations and conditions described in this SER.

5.5.3 Consistent Model Error

5.5.3.1 Poolability

Poolability
The validation error should be investigated to ensure that it does not contain any subgroups that are obviously not from the same population (i.e., non-poolable).
G3.3.1, Review Framework for Critical Boiling Transition Models

Section B.5.3 of the TR provides the process for determining the CHF correlation limits. The NRC staff observed that the NuScale process for developing the CHFR limit involved several statistical tests to determine: (1) whether subregions can be combined (i.e., pooled), and (2) whether the data can be treated as normally distributed. Figure B-4, "CHF Statistical Methods Flow Chart," of the TR provides the NuScale process for developing the CHFR limit. The statistical tests used by NuScale, and as shown in Figure B-4, are consistent with those described in NUREG-1475, Revision 1, "Applying Statistics," March 2011 (ML11102A076). Based on the description of these methods in NUREG-1475, the NRC staff finds that these statistical tests are consistent with established guidance and are therefore, acceptable. Additionally, based on NuScale selecting a bounding CHFR limit the NRC staff finds the process used to select the CHFR limit acceptable.

5.5.3.2 Nonconservative Subregions

> **Nonconservative Subregions**
>
> *The expected domain should be investigated to determine if it contains any non-conservative subregions which would impact the predictive capability of the model.*
>
> G3.3.2, Review Framework for Critical Boiling Transition Models

The NRC staff analyzed the measured-to-predicted performance of the high-flow CHF correlation. The NRC staff audited the subregions used to determine the correlation CHF limit, as described in the associated audit report (ADAMS Accession No. ML20034D464). During the audit, the NRC staff confirmed that the **[[**

]]. The NRC staff finds that the high-flow CHFR limit is suitably conservative over the application domain.

The NRC staff analyzed the measured-to-predicted performance of the low-flow CHF correlation. The NRC audited the subregions used to determine the correlation CHF limit, as described in the associated audit report (ADAMS Accession No. ML20034D464). The NRC staff observed that there is a **[[**

]] discussed in Section 6.11.4 of the TR and the information reviewed during the audit, the NRC staff finds that the low-flow CHFR limit is suitably conservative over the application domain established in the TR.

5.5.3.3 Model Trends

> **Model Trends**
>
> *The model is trending as expected in each of the various model parameters.*
>
> G3.3.3, Review Framework for Critical Boiling Transition Models

The NRC staff analyzed the measured-to-predicted performance of the high-flow CHF correlation. The NRC staff audited the model trends for measured-to-predicted performance with respect to mass flux, pressure, subcooling and critical quality, as described in the associated audit report (ADAMS Accession No. ML20034D464). During the audit, the NRC staff confirmed that **[[**

]], the NRC staff finds that the high-flow correlation trends are as expected in each of the various model parameters.

The NRC staff analyzed the measured-to-predicted performance of the low-flow CHF correlation. The NRC audited the model trends for measured-to-predicted performance with respect to mass flux, pressure, subcooling and critical quality, as discussed in the associated audit report (ADAMS Accession No. ML20034D464). During the audit, the NRC staff observed that **[[**

]], the NRC staff finds that the low-flow correlation trends are as expected in each of the various model parameters.

5.5.4 Quantified Model Error

5.5.4.1 Error Data Base

> **Error Data Base**
>
> *The model's error should be calculated from an appropriate data base.*
>
> G3.4.1, Review Framework for Critical Boiling Transition Models

As described in Section 5.5.2.2.1 of this SER, NuScale **[[** **]]** to establish the CHFR limit for the high-flow and low-flow correlations. Based on the use of **[[** **]]** to set the CHFR limits, the NRC staff finds that the error database used to determine the CHFR limits for the high-flow and low-flow CHF correlations are acceptable.

5.5.4.2 Statistical Method

> **Statistical Method**
>
> *The model's error should be calculated using an appropriate statistical method.*
>
> G3.4.2, Review Framework for Critical Boiling Transition Models

Section B.5.3 of the TR describes the statistics used to develop the CHF correlation limits. As evaluated in Section 5.5.3.1 of this SER, the statistical tests for poolability and normality are consistent with established guidance and are therefore acceptable. If normality testing determines that the data are normally distributed, NuScale determines the CHFR limit by adding the standard deviation times an appropriate tolerance factor (i.e., sufficient to establish a 95/95 CHFR limit) to the predicted-to-mean distribution average. If normality testing determines that the data are not normally distributed, then NuScale uses nonparametric statistics to establish a

95/95 CHFR limit. The NRC staff finds NuScale's statistical methodology acceptable because it is consistent with established guidance.

5.5.4.3 Appropriate Bias for Model Uncertainty

Appropriate Bias

The model's error should be appropriately biased in generating the model uncertainty.

G3.4.3, Review Framework for Critical Boiling Transition Models

Section 7.3.6 and B.5.3 of the TR develop the CHFR limits for the high-flow and low-flow CHF correlations. The CHFR limit of 1.05 for the high-flow correlation for IORV events, presented in Section B.5.3 of the TR, **[[**
]]. NuScale selected a CHFR limit of 1.29 for the low-flow correlation for IORV events, **[[**
]]. Additionally, a 3 percent engineering uncertainty factor and a 3 percent fuel rod bowing factor is applied to the CHFR limits for IORV events resulting in final CHFR limits that are rounded up to 1.13 and 1.37 respectively. NuScale selected a CHFR limit of 1.29 for the high-flow and low-flow correlations for LOCA events, **[[**
]]. Based on the use of bounding values for the CHFR limit, the NRC staff finds that the CHFR limits for the high-flow and low-flow correlations are appropriately biased. To ensure that the high-flow and low-flow CHFR limits are used in a manner consistent with their approved biases, the NRC staff established Condition 9 on the application of the CHFR limits for IORV and LOCA events.

5.5.5 Model Implementation

5.5.5.1 Same Computer Code

Same Computer Code

The model has been implemented in the same computer code that was used to generate the validation data.

G3.5.1, Review Framework for Critical Boiling Transition Models

Section 5.0 and B.6 of the TR provide the description of the EM which includes the use of the NRELAP5 computer code. Section 7.3 and B.5.3 describes the validation of the computer code to the test data. To ensure that the high-flow and low-flow CHF correlations are used in a manner consistent with their validation, the NRC staff established Limitations 7 and 8 on the use of NRELAP5 calculations using the high-flow and low-flow CHF correlations. Based on the description in Sections 5.0, B.6, 7.3 and B.5.3 of the TR, and pursuant to Limitations 7 and 8, the NRC staff finds that the high-flow and low-flow CHF correlations are implemented using the same computer code used to generate validation data, and are therefore acceptable.

5.5.5.2 Same Methodology

Same Methodology
The model's prediction of critical boiling transition is being applied in the same manner as it was when predicting the validation data set.
G3.5.2, Review Framework for Critical Boiling Transition Models

As described in Section 5.5.5.1, "Same Computer Code," of this SER, the NRC staff established Limitations 7 and 8 to ensure that the high-flow and low-flow CHF correlations are used in a manner that is consistent with their validation. Based on the description in Sections 5.0, B.6, 7.3 and B.5.3 of the TR, and pursuant to Limitations 7 and 8, the NRC staff finds that the high-flow and low-flow CHF correlations are being applied in the same manner as when predicting the validation data set, and are therefore acceptable.

5.5.5.3 Transient Behavior

Prediction of Transient Behavior
The model results in an accurate or conservative prediction when it is used to predict transient behavior.
G3.5.3, Review Framework for Critical Boiling Transition Models

During an audit with NuScale during June 13 – 15, 2017, as described in the associated audit report (ML17278A168), the NRC staff observed that NuScale performed **[[**

]] Based on the results of these tests, the NRC staff finds that the high-flow and low-flow CHF correlations provide suitably conservative predictions for CHF when used to predict transient behavior.

5.6 Summary

The NRC staff approves the CHF modeling described in TR-0516-49422, Revision 2, subject to the limitations and conditions identified in Section 6.0 of this SER. In particular, the NRC staff finds that: (1) the high-flow CHF correlation is acceptable for use in performing safety analyses of the NPM with NuFuelHTP2™ fuel, with a CHFR limit of 1.13 for IORV events and 1.29 for LOCA events, over the range of applicability provided in Table 5.2-1, "High-Flow CHF Correlation Range of Applicability," of this SER, and (2) the low-flow CHF correlation is acceptable for use in performing safety analyses of the NPM with NuFuelHTP2™ fuel, with a CHFR limit of 1.29 for LOCA events and 1.37 for IORV events. These conclusions are based on the following three findings:

1. The experimental data supporting the high-flow and low-flow CHF correlations are appropriate as evidenced by meeting all the supporting goals discussed in Section 5.5 of this SER.

2. The high-flow and low-flow CHF correlations were generated in a logical fashion as evidenced by meeting all the supporting goals discussed in Section 5.5.1 of this SER.

3. The high-flow and low-flow CHF correlations have sufficient validation, demonstrated through appropriate quantification of their error, as evidenced by meeting all the supporting goals discussed in Sections 5.5.2 through 5.5.5 of this SER.

6.0 LIMITATIONS AND CONDITIONS

This section provides a summary of the limitations and conditions based on the technical evaluation of the NuScale TR 0516-49422, “Loss-of-Coolant Accident Evaluation Methodology”, Revision 2. As a result of its in-depth technical evaluation, the NRC staff determined that the NuScale LOCA EM, including the methodology to analyze events initiating from the IORV specified in Appendix B, can be used for the NuScale NPM design, subject to the limitations and specific restrictions on the use of this model as listed below.

1. Regulatory compliance with 10 CFR Part 50, Appendix K for application of the LOCA EM for features not evaluated in the TR.

   An applicant or licensee referencing this report will be required to address regulatory compliance with 10 CFR 50.46 and 10 CFR Part 50, Appendix K, which could include seeking an exemption from the required features not addressed by this EM as described in Table 2-2 of this TR, including: those related to post-CHF heat transfer models; fuel pin models that incorporate clad swelling, rupture and, oxidation; calculation of the metal-water reaction rate using the Baker-Just Correlation and radiation heat transfer.

2. CLL, CHF and Peak Containment Pressure and Temperature Requirements.

   The NuScale LOCA EM is limited to the evaluation of LOCAs where: (1) the CHF is not exceeded; (2) the CLL remains above the top elevation of the core active fuel region for the full spectrum of break sizes and locations and (3) the containment peak temperature and pressure remain below the design limits.

   NRELAP5 does not apply to LOCA conditions where CHF is achieved and core uncovery is predicted to occur since the NuScale LOCA EM has not been demonstrated as adequate to evaluate peak cladding temperature, core wide oxidation, rod swelling and rupture behavior that could occur if the CLL drops below the top of the reactor core sufficiently to cause the active fuel to be exposed to steam cooling. Further, the NuScale design is subject to a potential return to power subsequent to a LOCA with one rod stuck full out of the core. The LOCA EM does not include a method for calculating the number of failed pins from either an initial LOCA or a subsequent return to power with pre-existing fuel pin failures from the LOCA event.

3. No Credit for DHRS Heat Removal in LOCA EM without Further NRC Review and Approval.

   The NRC staff did not review the DHRS heat removal modeling as part of this overall methodology. Any future credit for DHRS requires review and approval by the NRC.

4. Types of Analyses Approved for LOCA EM.

   Use of the LOCA EM is limited to evaluations of the analyses for the FOMs described in the TR: the short term LOCA or an IORV event. The LOCA EM is not approved for use in evaluations for thermal hydraulic analyses not described in the methodology presented in the TR. Use of the LOCA EM is not approved for use in analysis of thermal hydraulic instabilities in the secondary or primary system, peak containment pressure (such as following an IORV), control rod ejection accidents, radiological consequences, non-LOCA events (other than an IORV), return to power analysis assuming the worst-case stuck control rod, and evaluation of the long-term cooling phase.

5. Limitations on NRELAP5 and NPM Model Approval.

   Use of NRELAP5 is limited to v1.4, in conjunction with NPM model Revision 3, unless changes are made pursuant to a change process specifically approved by the NRC staff for changes to NRELAP5 and the NPM model. NRELAP5 v1.4 and NPM model Revision 3 are approved for use in this TR as part of the LOCA EM. NRELAP5 is not approved for analysis of thermal hydraulic instabilities in the secondary or primary system. When NRELAP5 v1.4 and NPM model Revision 3, as described in this TR, are referenced in other EMs, those applications for use of NRELAP5 v1.4 and NPM model Revision 3 within another EM; require separate approvals to ensure the models and assumptions are defined appropriately for the analyzed FOMs. Use of the NRELAP5 v1.4 and NPM model Revision 3 are therefore not approved for standalone evaluation of the following events and must have separate EM approvals: peak containment pressure (such as following an IORV), control rod ejection accidents, non-LOCA events (other than IORV), return to power analysis assuming the worst-case stuck control rod, and evaluation of the long-term cooling phase.

6. Single Failures, Electrical Power Assumptions (ac/dc) and Need for Operator Actions Not Approved Within this Methodology.

   An applicant or licensee seeking to apply this methodology to a design must receive a separate approval through that design review for the single failures, electrical power assumptions (ac/dc) or the need for operator actions necessary to mitigate design basis events necessary to consider for the evaluation of LOCA events, or the IORV events.

7. **[[** **]]**.

   [[

   ]]. Sections 5.5.2 and 5.5.3.2 of this SER describes the basis for this limitation.

8. High Flow CHF Correlation Range.

Application of the high-flow CHF correlation is limited to its range of applicability as identified in Table 5.2-1. Sections 5.5.5.2.2.2 and 5.5.5.1 of this SER describe the basis for this limitation.

Table 5.2-1. High-Flow CHF Correlation Range of Applicability

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

9. CHFR Minimum Value.

The high-flow CHFR limit of 1.13 is required for analyses of IORV events for high-flow conditions. The low-flow CHFR limit of 1.37 is required for analyses of IORV events for low-flow conditions. The high-flow and low-flow CHFR limit of 1.29 is required for analyses of LOCA events for high-flow and low-flow conditions. Sections 5.5.4.3 and 5.5.2.2.2 of this SER describe the basis for this condition.

7.0 CONCLUSION

This SER documents the results of the technical evaluation of TR-0516-49422-P, “Loss-of-Coolant Accident Evaluation Model,” Revision 2. The NRC staff finds that the proposed methodology is acceptable for meeting the requirements of 10 CFR 50.46 and Appendix K evaluated in this TR, for evaluation of the ECCS performance in the NuScale NPM for design basis LOCAs, subject to the limitations, conditions, and restrictions identified in Section 6.0 above. The NRC staff finds the NuScale LOCA EM appropriate for determining CHF and CLL results, excluding peak cladding temperature, clad oxidation and core wide clad oxidation, but requires that this information, along with the worst break minimum liquid level in the vessel above the top of the active fuel be reported on a plant specific application, which uses this version of the NuScale LOCA EM. The NRELAP5 computer code is also determined to be applicable to predict peak containment pressure and temperature when referenced in a separately approved EM, subject to specific modeling requirements necessary for prediction of these FOMs. The NRC staff finds the CHF modeling described in TR-0516-49422, Revision 2, acceptable subject to the limitations and conditions identified in Section 6.0 of this SER. Therefore, the NRELAP5 computer code and the NPM model are determined to be acceptable to evaluate the MCHFR for IORV and LOCA events.

For uses other than that intended and approved as part of the NuScale LOCA methodology, the process and all of its elements, including a description of the intended use and justification, must be submitted to the NRC for review and approval. The NRC staff emphasizes that the criterion for acceptable ECCS performance following all LOCA break sizes is that the CLL in the RPV remain above the top of the active fuel and the CHF limit be met. TR-0516-49422-P, Revision 2, constitutes a separate and unique methodology, and as such, any other version derived from this TR, such as an update designated by a new revision number, amendment number, addendum number or equivalent designation would constitute a definition of a new methodology requiring the NRC staff's review and acceptance prior to a generic application and prior to any specific plant licensing application of a new methodology derived from this TR.