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SAFETY EVALUATION REPORT BY THE U.S. NUCLEAR REGULATORY COMMISSION

TOPICAL REPORT TR-0516-49422, REVISION 3

"LOSS-OF-COOLANT ACCIDENT EVALUATION MODEL"

NUSCALE POWER, LLC

Contents

1 INTRODUCTION AND BACKGROUND

Under the NRC Topical Report (TR) Program and by letter dated January 8, 2023, NuScale Power, LLC (NuScale or the applicant) submitted TR-0516-49422, “Loss-of-Coolant Accident Evaluation Model,” Revision 3, (hereafter referred to as the loss-of-coolant accident evaluation methodology topical report (LOCA EM TR) (Agencywide Documents Access and Management System (ADAMS) Accession No. ML23008A002), to the U.S. Nuclear Regulatory Commission (NRC) staff for review. By letter dated July 31, 2023 (ML23205A009), the NRC informed NuScale of its acceptance of TR-0516-49422-P, Revision 3, for a detailed technical review.

TR-0516-49422-P, Revision 3, presents NuScale’s LOCA evaluation model (LOCA EM) used to evaluate emergency core cooling systems’ (ECCS) performance in the NuScale Nuclear Power Module (NPM-20) for design-basis LOCAs and pipe breaks inside containment.

In addition to the ECCS performance model, it includes a description of a modified version of the LOCA EM that is used to evaluate the Inadvertent Opening of a Reactor Pressure Vessel (IORV) Valve event and the mass and energy release model for containment peak response evaluations.

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 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, Revision 3, and the NuScale EM used to evaluate the ECCS performance in the NuScale Power Module NPM-20. 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, for evaluating ECCS core cooling performance for LOCA and LOCA-type events. After Revision 2 of this TR was approved, NuScale submitted Revision 3 to TR-0516-49422 as a result of a new NPM design. The most significant change to the NPM-160 design was a power uprate from 160 megawatts thermal for the NPM-160 design to 250 megawatts thermal for the new NPM-20 design. Both designs use 37 fuel assemblies and the major dimensions of the two designs, including fuel height, are unchanged. For the NPM-20, the primary pressure was increased for the NPM-20 and the secondary side inlet temperature to the helical coil steam generators (HCSGs) was reduced for the NPM-20. The NPM-20 core inlet temperature was reduced and the total core flow increased to remove the additional core heat while minimizing the increase in core exit temperature.

The NuScale NPM-20 design is a small modular reactor designed to be deployed with several 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 cool down in the event of a LOCA. Each NPM has an independent nuclear steam supply system (NSSS) that includes a nuclear core, helical-coil 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. 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 with LOCA acceptance criteria such that there is no core uncovery or heatup for a design-basis LOCA inside containment.

The EM also evaluates the design-basis events (DBEs) resulting from an IORV, as well as containment pressure and temperature response during a LOCA and an IORV.

The TR-0516-49422, Revision 3, presents an updated NuScale LOCA EM and integrates the methodology to evaluate containment pressure and temperature response into the LOCA EM. The earlier version of the Containment Response Analysis Methodology (CRAM) for the NPM-160 approved design (NuScale US600 design) as described in a separate technical report, TR-0516-49084-P, “Containment Response Analysis Methodology Technical Report” (the CRAM TR), Revision 3 (ML20141L808 (nonproprietary) and ML20141L809 (proprietary)), that was incorporated by reference in the NuScale Design Certification Application (DCA), Final Safety Analysis Report (FSAR), Chapter 6. The CRAM has been updated for the design changes from NPM-160 to NPM-20 and is now a part of the LOCA EM TR, Revision 3. The CRAM methodology presented in the LOCA EM TR, Revision 3, is used in NuScale EC-A013-7725, “NPM-20 CNV Pressure and Temperature Response Analysis“ (ML23011A012), to perform the containment safety analyses for FSAR Section 6.2, “Containment Systems.” The CRAM is used to predict the containment pressure and temperature response and whether the CNV pressure would rapidly reduce and remain acceptably low.

NuScale made several changes to the ECCS valve designs and containment for the NPM-20, which include the following:

- Reduced the number of reactor vent valves (RVVs) from 3 to 2
- Removed Inadvertent actuation block valves (IABs) from the RVVs
- Two trip valves are used for each RVV and reactor recirculation valve (RRV) compared to single trip valves for the NPM-160
- Reduced the release pressure for the NPM-20 RRVs
- Added flow limiting venturis to RVVs and RRVs
- Changed the containment vessel steel material compositions

Other changes include ECCS actuation based on reactor pressure vessel (RPV) riser two-phase water level measured by thermal dispersion level sensors for the NPM-20 versus CNV level for the NPM-160. Two boron oxide dissolver baskets located on the inside surface of the CNV were added to the NPM-20 along with condensate rails to direct water during ECCS actuation to the dissolver baskets. The ultimate heat sink pool level was reduced for the NPM-20 and the maximum allowed CNV pressure was increased.

The increase in power noted above resulted in an increased coolant temperature change across the reactor core and density variation which resulted in an increased coolant natural circulation mass flow rate in the RCS. The increased coolant temperature variation required an increased pressurizer pressure. To accommodate the increased power, the feedwater temperature was decreased and the main steam/feedwater mass flow rate was increased in the secondary side SG. The main steam pressure was decreased to support improved operation of the secondary side considering SG superheat and turbine cycle efficiency. The reactor pool water level was decreased as mentioned above to mitigate the adverse impacts of the changed conditions on the RPV collapsed liquid level (CLL) during long-term cooling (LTC) with ECCS operation.

NuScale incorporated two changes to their currently approved EM used to analyze LOCA and IORV events. The first change was the incorporation of a new CHF correlation (NSPN-1) into NRELAP5 v1.7 which could be used during the events and provided a less conservative and more realistic estimate of CHF. The second change was allowing cross-flow to be considered during the events. This change would allow mixing between the hot and average assemblies, resulting in a less conservative and more realistic flow predictions in the limiting assembly.

This SER presents an assessment of the updated LOCA EM which is applicable to the new NPM-20. Section 5, "LIMITATIONS AND CONDITIONS," of this SER discusses the requirements should an applicant or licensee wish to apply the EM to either the NPM-160 design or a future design.

2 REGULATORY BASIS FOR LOCA EVALUATION MODEL TOPICAL REPORT REVIEW

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 TR-0516-49422, Revision 3, is acceptable for LOCA, IORV and containment analyses of the NuScale NPM-20 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).

The regulations in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, Criterion 38 for containment heat removal require the system safety function shall reduce rapidly the containment pressure and temperature following a LOCA and maintain them at acceptable low levels. Criterion 50, Containment Design Basis, requires the containment heat removal system can accommodate the calculated peak pressure and temperature conditions resulting from any LOCA without exceeding the design leakage rate and with sufficient margin.

The regulations in GDC, Criterion 10 for protection of fuel cladding integrity under normal operation require the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences, such as the inadvertent opening of an ECCS valve.

The various elements of the EMDAP are defined in RG 1.203, which provides a roadmap that relates the sections of this report to the elements and steps of the EMDAP. The EMDAP establishes the adequacy of a methodology for evaluating complex events.

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.

2.1.1.1 Emergency Core Cooling System 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, 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 LOCA. 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).

2.1.1.2 10 CFR Part 50 Appendix K

The regulation in 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 NPM-20 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 EM TR, Revision 3, 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 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 Part 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 NPM-20 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 methodology 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 limited to, post-CHF heat transfer models, fuel pin models that incorporates clad swelling, rupture, 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 EM and must be calculated for a number of postulated LOCAs of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated LOCAs are calculated."

2.1.1.3 Emergency Core Cooling System 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 Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants"

In TR-0516-49422, Revision 3, 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):

- 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.

- GDC 16, "Containment Design," which establishes that the reactor containment and associated systems shall be essentially leak-tight to prevent against the uncontrolled release of radioactive materials to the environment and to provide assurance that the containment design conditions important to safety are not exceeded according to the requirements of the postulated accident conditions.

- GDC 50, "Containment Design Basis," specifies that the reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be capable of withstanding the calculated pressure and temperature conditions as a result of any loss-of-coolant scenario without exceeding the design leakage rate and with sufficient margin. This margin reflects (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in SGs, metal-water, or other chemical reactors that may result from degradation but not total failure of emergency core cooling, (2) the limited experience and experimental data available for defining accidents and containment responses, and (3) the conservatism of the calculational model and input parameters.

NuScale Principal Design Criterion (PDC) 38, “Conformance or Exception,” states that a system to remove heat from the reactor containment shall be provided. The system safety function shall be to rapidly reduce the containment pressure and temperature following any loss-of-coolant scenario and maintain them at acceptably low levels while not impacting the functioning of other associated systems. Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to ensure that the safety system can function, assuming a single failure.

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 EM 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 V1.7 systems analysis computer code, which is developed from the Idaho National Laboratory (INL) RELAP5-3D computer code. 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.

2.2.2 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 20 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

2.2.3 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.

2.2.4 Regulatory Position 3, Documentation

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

2.2.5 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."

2.2.6 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."

The containment response analysis methodology is an extension of the NuScale LOCA, valve opening event and non-LOCA methodologies developed following the guidance of RG 1.203. This report identifies and justifies the differences in the containment response methodology inputs when compared to LOCA and reactor valve opening events.

2.3 NUREG-0800 Standard Review Plan

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.

The design specific review standard (DSRS) Section 6.2.1, "Containment Functional Design" (Reference 30), includes a high-level summary of an acceptable approach and content for a containment response analysis methodology, and references the lower-tier subsections with additional detail about the approach and contents. The comparison of the containment response analysis methodology to applicable content in DSRS Section 6.2.1 is provided in Table 2-3 of the LOCA EM TR, Revision 3.

The DSRS Section 6.2.1.1.A, "Containment" (Reference 30), includes, content related to containment design, including some elements that are associated with the capability to withstand M&E releases. The comparison of the containment response analysis methodology to applicable content in DSRS Section 6.2.1.1.A is provided in Table 2-4 of the LOCA EM TR, Revision 3. The DSRS Section 6.2.1.3, "Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs)" (Reference 30), includes the details of an acceptable approach and content for an M&E methodology for LOCAs. As noted, a comparison of NPM design reveals that some of the DSRS content is based on pressurized water reactor (PWR) large-break LOCA phenomena that are not applicable to the NuScale design. The comparison of the M&E methodology to applicable content in DSRS Section 6.2.1.3 is given in Table 2-5 of the LOCA EM TR, Revision 3.

The DSRS Section 6.2.1.4, "Mass and Energy Release Analysis for Postulated Secondary Pipe Ruptures" (Reference 30), includes the details of an acceptable approach and content for a M&E methodology for main steamline break (MSLBs) and FWLBs. The comparison of the M&E methodology to applicable content in DSRS Section 6.2.1.4 is provided in Table 2-6 of the LOCA EM TR, Revision 3.

3 NUSCALE LOCA EVALUATION METHODOLOGY SUMMARY

The NPM-20 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 CLL above the top of the active core through all potential LOCA events, as shown in NuScale LOCA EM 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 LOCA, for the NPM-20, the high-pressure water and steam leaving the RPV is contained in the CNV. The CNV has a design pressure of 1200 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 four ECCS valves, two RVVs and two RRVs, receive an ECCS actuation signal triggered by RPV riser two-phase water level sensors to open. Two RVVs will open immediately after the actuation of ECCS. However, two RRVs are blocked from opening by the IAB valve until the pressure difference between the RPV and CNV drops below the IAB threshold. Once these valves open, the RPV and CNV pressures equalize within a short period of time. After this pressure equalization, steam generated inside the RPV from decay heat and stored energy, either condenses inside the RPV through DHRS or 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-20 CNV design and the NuScale ECCS system, the NRC staff's review of the NuScale LOCA EM TR, Revision 3 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 DBE of a LOCA.

- The capability to predict Critical Heat Flux Ratio (CHFR) with credit taken for cross-flow so that the NuScale power module maintains CHF margin during the DBEs of a LOCA or IORV.

- The applicability of the NRELAP5 computer code to perform peak containment pressure and temperature analysis so that the CNV of NuScale power module is demonstrated to absorb 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 DBE of a LOCA

- Ensuring that the LOCA pipe break spectrum methodology included all susceptible RPV penetrations and break sizes inside containment.

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-2) facility and to confirm that the geometric dimensions and operating conditions of NIST-2, adequately represent the NPM-20 design.

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 Standard Design Application (SDA) that the CLL remains above the top of the core, is documented as part of the review of the NuScale SDA.

The NRELAP5 computer code, Version 1.6 (ML23011A012), 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, addition of the NSPN-1 CHF correlation to supplement Hench-Levy, and the addition of new containment condensation models to describe the unique design features of the ECCS operation of the NPM. Subsequently, NuScale submitted NRELAP5 Version 1.7 (ML24228A242) as the systems analysis computer code for the NuScale LOCA Evaluation Methodology, replacing NRELAP5 Version 1.6. In Version 1.7, NuScale made minor changes to **[[**

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In addition to the use of NRELAP5 for evaluating LOCAs, Section 9.8 of the LOCA EM TR applies the EM to analyze events described in SRP Section 15.6.1, "Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve," and DSRS 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 SRP 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 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, Revision 3, are discussed in detail below and summarized in Section 5 of this SER. The conclusions from the review are discussed in detail below and summarized in Section 6 of this SER.

In addition, the NRC staff conducted an audit of information provided by the applicant in support of the NRC staff's review of the TR. This audit is referenced throughout this SER. Unless otherwise noted, details of this audit are available in the audit report (ML24262A230).

4.1 Introduction and Scope

Section 1.1 of TR-0516-49422, Revision 3, states that the purpose of the NuScale EM is to evaluate ECCS performance in the NPM for design-basis LOCAs and ECCS valve opening events. In addition, the model is used to evaluate mass and energy release into the containment vessel and the subsequent peak containment pressure and temperature. Therefore, NuScale requests an approval to use the methodology to perform such analyses. NuScale stated that its LOCA EM follows the guidance provided in RG 1.203, "Transient and Accident Analysis Methods," and satisfies the applicable requirements of Appendix K to 10 CFR Part 50, "ECCS Evaluation Models." TR-0516-49422 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 does not provide any final licensing analyses and this review of TR-0516-49422 does not evaluate the acceptability of the NuScale NPM-20 design or provide any conclusions on the acceptability of the NuScale NPM-20 design.

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

1. Events as described in TR-0516-49416 Revision 5, “Non-Loss of Coolant Accident Analysis Methodology.”

2. LTC analysis and reactivity control methodology, “Extended Passive Cooling and Reactivity Control Methodology,” TR-124587, Revision 1.”

4.2 Background

Section 2 of TR-0516-49422, Revision 3, 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 MN-122626, “Quality Assurance Program Description,” (ML23361A122). The NRC staff reviewed the QA requirements and documented its approval in its SER (Refer to ML16347A405). Further, the NRC staff inspected NuScale’s design control process and code development procedures in February 2024 (ML24099A129); Limitation/Condition (L/C) #13 is placed on TR-0516-49422, Revision 3, approval, as described further in Section 4.6.1 of this SER.

NuScale indicated that the NPM-20 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 the requirements in 10 CFR Part 50, Appendix K, are not encountered in the design-basis NPM-20 event LOCAs, meaning that these phenomena have been eliminated by design. Thus, 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 EM TR, must demonstrate regulatory compliance with these Appendix K requirements, which could include seeking an exemption. The requirements are included in TR Table 2-2, “10 CFR Part 50, Appendix K required and acceptable features compliance.” This is reflected in Section 5.0, “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-0516-49422, Revision 3, 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 NSPN-1 CHF correlation used in NRELAP5 are contained in Section 4.11 of this SER.

4.3 NuScale Power Module Description and Operations

Section 3 of TR-0516-49422, Revision 3, provides a brief description of the NPM and a 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)
* Two integrated HCSGs and an PRV-integral pressurizer 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.
* A two-train safety-related two-phase natural circulation decay heat removal system (DHRS)
* Primary fluid in the SGs flow on the outside of the tube surfaces, and two-phase flow of the secondary fluid contained inside of the tubes
* A high-pressure steel CNV partially immersed in a water-filled pool that is integral to the ECCS capability to provide for emergency cooling

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

4.3.2 Plant Operation

In LOCA EM TR Section 3.2, NuScale provides a brief description of NPM operation, including systems modeled in the LOCA EM. The NRC staff reviewed the plant operation summary in the LOCA EM TR and finds that it provides a sufficient description to support the methodology description.

4.3.3 Safety-Related System Operation

In Section 3.3 of the TR, NuScale described operation of the safety-related systems and components, including the NuScale module protection system (MPS), ECCS, DHRS and the containment isolation valves. The NRC staff reviewed the safety-related system operation summary in TR-0516-49422, Revision 3, and finds that it provides a sufficient description to support the methodology description.

4.3.3.1 Emergency Core Cooling System

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. For the NPM-20, the ECCS is actuated on low and low-low RPV riser level setpoints signals, or 8 hours post scram, or at 24 hours after loss of alternating current (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 pounds per square inch differential (psid).

4.3.3.2 Decay Heat Removal System

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 each associated with one of the two NPM SGs. The DHRS is isolated during normal operation and is activated to remove decay heat for NPM long-term cooling for DBEs. The DHRS performance during LOCA conditions was benchmarked by NuScale using NIST-1 and NIST-2 integral tests and each train is designed with the goal 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, Revision 3, 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 EM TR. In 2022, the PIRT phenomena and ranking were re-assessed to address changes in analysis methodology and to accommodate design modifications. However, NuScale documented only the phenomena and processes of high importance in Section 4 of the LOCA EM TR, Revision 3. Therefore, the NRC staff reviewed both the LOCA EM TR and the 2015 NuScale LOCA PIRT as well as the 2022 LOCA PIRT applicability update. The 2015 NuScale PIRT and the 2022 LOCA PIRT applicability update provided the rankings for all four PIRT importance categories (high, medium, low, and inactive).

Previously, 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 ensure 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 panel. 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 ensure 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 panel members lists provided in Section 4.1 of the LOCA EM TR. Based on the NRC staff’s knowledge of the panel members listed, the NRC staff confirmed that all 2013 and 2015 panel members are highly regarded members of the nuclear community with extensive experience in the industry, a research institution, or nuclear academia. NuScale did not provide a member list for the 2022 PIRT phenomena and ranking re-assessment panel in the LOCA EM TR. The staff reviewed the 2022 re-assessment updated High-Ranked Phenomena for LOCA and IORV that listed in Tables 4-4 and 4-5 of Section 4.6, such as the two-phase choked flow in the RVVs, counter current flow limitation (CCFL) and water hold-up, flashing, void distribution in core, CHFR, and DHRS heat removal. The staff found the identified phenomena and ranking updates in the 2022 PIRT to be applicable to the NuScale NPM-20 during LOCA and IORV events.

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 break spectrum of break sizes and locations selected for the PIRT discussions inside the NPM-20 containment and determined that the only LOCAs that can occur inside containment 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 PWR designs that circulate reactor coolant through large pipes outside of the RPV, are not applicable for the NuScale NPM. The LOCA EM states that the breaks considered are those inside containment, therefore, consideration of any potential breaks outside containment is not included in this evaluation (see related L/C #9 in Section 5 of this SER).

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

]]. The NRC staff notes that for the NuScale design, breaks from the RVV nozzles and RRV flanges are not included in this TR as break locations and are the subject of L/C #9 in Section 5 of this SER.

Locations excluded from the LOCA break spectrum are not evaluated as part of the LOCA EM, and are considered through a licensing application, with appropriate exemptions, as specified in L/C #9 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.

4.4.1 Figures of Merit

In TR Section 4.3, NuScale discussed 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 an 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 stated 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 that for the NuScale LOCA EM FOMs (ML24326A329) 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 5, “Limitations and Conditions,” of this SER.

NuScale stated that to ensure ECCS performance, the 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. The limiting peak containment pressure and temperature are listed as other FOMs in LOCA EM TR.

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 L/C # 1 and L/C #2 in Section 5, “Limitations and Conditions,” of this SER.

4.4.2 Phenomenon Identification and Ranking Table Rankings

Sections 4.4 through 4.7 of the LOCA EM TR discuss the results of the NuScale LOCA 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 an NPM-20 LOCA, and the ranked relative importance of each, and assessed the knowledge level for 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.

Section 4.6 of the LOCA EM TR identified **[[**

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In Table 4-5 of the LOCA EM TR, phenomena ranked high for the IORV event are listed as follows:

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The staff notes the transport and potential accumulation of combustible, radiolytic non-condensable gases (NCGs), arising from water radiolysis in proximity to the nuclear, **[[ ]]** in the PIRT of the LOCA EM, which includes containment response analysis and in-vessel heat transfer evaluation. In letter dated XXX (MLXXXX), the applicant stated the US460 design relies on a passive autocatalytic recombiner (PAR) to maintain an inert containment following a design basis event or severe accident by preventing the accumulation of a combustible concentration of certain NCGs which, when combined and concentrated, form a combustible mixture, thus ensuring containment can perform its safety function as a fission product barrier.

During a regulatory audit (ML24211A09), the staff observed that the applicant specified that PAR is sized to mitigate the impact of radiolytic NCG on containment's function to transfer heat to the reactor pool by way of condensing steam generated in the RPV, as a part of the NPM-20 ECCS's operation. Therefore, the radiolytic gas transported to the containment is expected to have a very small concentration in the containment and does not alter the outcome of the existing PIRT. In addition, the staff assessed whether radiolytic gases could come out of solution and accumulate in the SG region during ECCS operation. In letter dated XXX (MLXXXX), the applicant determined that **[[**

]]. Because the low pressure region in the steam generator region and downward condensate droplets could impede this transport process, the staff performed a confirmatory analysis and concluded that the total amount of radiolytic gas reaching the downcomer region during ECCS operation will not be sufficient to exceed the combustible limit even assuming all the radiolytic gas is trapped in the downcomer region. Therefore, the staff finds that the existing PIRT for the in-vessel flow and heat transfer is not impacted by the generation and transport of the small amount of radiolytic gas.

NuScale has also performed and documented a PIRT for the non-LOCA events. The review of the non-LOCA PIRT is summarized in the SER for NuScale TR-0516-49416, "Non-Loss-of-Coolant Accident Analysis Methodology," Revision 4 (ML24334A049).

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 and the 2022 LOCA PIRT applicability update. 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 Sections 4.6.1 through 4.6.3 of the LOCA EM TR. The NRC staff finds 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 M&E release is a key factor but not the only key factor. The NRC staff found that the rate of M&E 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 **[[**

]]. The staff subsequently agreed that assessment of the rate of M&E release is implied in the assessment of choked and unchoked flow.

The IORV events are classified as AOOs, and hence the acceptance criteria and event progression are different from those of LOCAs. Table 4-5 of the LOCA EM TR identified several newly high-ranked phenomena relative to the LOCA PIRT. The staff evaluated two of these high-ranked IORV phenomena, **[[**

]] in detail during its review. Both phenomena may occur in Phase 0 of IORV and are classified as partially known with a large uncertainty. The **[[**

]] is classified as known with moderate or small uncertainties in the LOCA PIRT. However, **[[** **]]** is

classified as partially known with a large uncertainty due to **[[**
]] during IORV impacts the FOMs such as the CLL and the minimum CHF ratio (MCHFR). The **[[** **]]**, for both the hot and average fuel assemblies, are also affected by the RCS depressurization induced by the IORV. The staff agrees with the applicant regarding these newly identified high-ranked phenomena for IORV. Due to the higher power of the NPM-20 design, the CHF performance margin is reduced. One of the significant contributors to the CHF determination is the consideration of cross-flow. Therefore, the staff agrees with the applicant's ranking CHF phenomenon as a highly important phenomenon for IORV evaluations.

The NRC staff finds that the PIRT phenomena selections and knowledge level rankings are appropriate as a basis for the LOCA EM with its application to LOCA, IORV and containment performance evaluations.

4.4.3 Containment Response Analysis Methodology Phenomena Identification and Ranking Table

A PIRT was developed for the NuScale LOCA EM TR, Revision 3, by a panel of recognized industry and NuScale subject matter experts to provide an assessment of the relative importance of the phenomena and processes that may occur in the NPM-160 during a LOCA event. In 2022, the PIRT phenomena and rankings were re-evaluated to address changes in the NPM design and analysis methodology. The PIRT assessment performed in relation to specified FOMs is part of the EMDAP process prescribed by RG 1.203. The PIRT also established a knowledge ranking for each phenomenon identified. Using these FOMs, over two dozen phenomena were identified as important to be captured in the NuScale LOCA EM for the NPM-20, as shown in LOCA EM TR, Revision 3, Table 4-4. The applicability of the EM is based on the NuScale LOCA PIRT, which identifies and ranks those phenomena the EM must be qualified for in order to model a LOCA in the NPM. The EM does not have restrictions concerning operating setpoints or loss of offsite-power conditions as long as the phenomena that occur during the progression of a LOCA have been identified by the PIRT process.

The detailed review and evaluation of the latest PIRT for NPM-20 and the related high-ranked phenomena are presented in Section 4.4 of this SER. The qualification of the LOCA and non-LOCA methodologies, in particular the comparisons to separate effects tests (SETs) and integral effects tests (IETs) applicable to the containment response analysis methodology (CRAM), are presented in Section 4.7.5.8, "Assessment of NRELAP5 Prediction of Peak Containment Pressure," of this SER. The NRELAP5 simulation model used for the containment response analysis methodology (CRAM) is similar to the NRELAP5 simulation models used for LOCA, reactor valve opening events, and non-LOCA events. As "CNV pressure and temperature" is also a figure-of-merit in the LOCA PIRT, the LOCA scenario PIRT is also applicable to the primary and secondary system's M&E release and the resulting CNV pressure and temperature response. Therefore, the staff agrees that the PIRT developed and revised for the LOCA and non-LOCA methodologies for the NPM-20 is also applicable to the CRAM.

4.5 Evaluation Model Description

The NRC staff reviewed the NuScale LOCA EM description provided in Section 5 of the NuScale LOCA EM TR, Revision 3 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 NPM components and provides NuScale's rationale for these choices. Section 5 also provides NuScale's justification for the initial and boundary 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 is consistent with the SET and IET assessments used by NuScale to validate the NRELAP5 code (for its application to LOCA and non-LOCA analyses).

The staff reviewed Sections 5.5 and 5.6 of the LOCA EM TR, Revision 3. Section 5.5 describes the key modeling differences between the LOCA analysis and containment response analysis, while Section 5.6 provides detailed descriptions of the NRELAP5 containment response analysis M&E release model for the primary and secondary systems DBEs. Section 5.6 also documents the conservative initial and boundary conditions to ensure a bounding M&E release, as well as the CNV and reactor pool modeling assumptions used to minimize the CNV heat removal and maximize the CNV pressure. The staff finds the presented rationales to be appropriate and consistent with the submitted NRELAP5 containment model.

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

In LOCA EM TR, Revision 3, 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 the following:

- Choked and unchoked flow through the break and then ECCS flow via valves between the 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 FOMs. The NRC staff finds that NPM modeling developed adequately represents the key components and the key phenomena expected to occur during a LOCA.

4.5.1.1 NuScale Power Module NRELAP5 Model

The NuScale LOCA EM covers key components of the NPM participating in a LOCA. Revision 3 of the LOCA EM (the current revision) also covers the IORV AOO and the containment pressure and temperature response analysis methodology; it is the case that the components of an NPM involved in a LOCA are also involved in these methods. 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 between the riser and the downcomer

The NPM-20 nodalization diagram of these key components is shown in Figure 5-2, “Noding diagram of NRELAP5 loss-of-coolant accident input model for the NPM-20,” of the LOCA EM TR.

In LOCA EM TR, Revision 3, 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-20 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 EM TR, Revision 3, describes the modeling of each of the RCS components. The NRC staff’s findings on the modeling of those components are described below.

4.5.1.2 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 preserves the volume, and elevation changes in these three regions, and incorporates conservative loss factors to maximize core bypass flow, and the flow from the downcomer and to minimize the flow into the reactor core.

In the NPM-20 model, the riser nodal size has been changed in order to match the riser hole locations in the upper or lower riser sections. The NRC staff reviewed this model change and finds it is acceptable.

4.5.1.3 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 mostly parallel reactor core channels. The NRELAP5 model for NPM-20 connects the **[[**

]]. The core model includes form losses for top and bottom nozzles and grid spacers with appropriate hydrodynamic volumes based on fuel vendor data. In the NPM-20 model, the core nodal sizes have been changed per updated fuel vendor data. The NRC staff reviewed this model change and finds it is acceptable. NuScale modeled the core flow channels with individual NRELAP5 PIPE components. **[[**

]].

The NPM model sets the **[[**

]].

For LOCA analysis, NuScale assumed that actual core power is 102 percent of the rated 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 provides reasonable assurance 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 RPV and CNV metal components with 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 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). The NuScale model sets the moderator and Doppler coefficients for minimum negative worth-based beginning of cycle 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; the decay heat model selected is the 1973 revision of “Decay Energy Release Rate Following Shutdown of Uranium-Fueled Thermal Reactors” (Reference 52) standard which the NRC staff reviewed and determined to be consistent with the 1971 standard that complies with the regulatory requirements of 10 CFR 50.46, and Appendix K to 10 CFR Part 50. The NRC staff determined that this modeling approach meets the regulatory requirements for LOCA analysis.

4.5.1.4 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 liquid and vapor volumes, elevation changes and the energy stored in the metal components. It also models the connection of two RVVs, the pressurizer baffle plate, the pressurizer heaters, and the steam plenums interface.

4.5.1.5 Helical Coil Steam Generators (HCSG)

Section 5.1.3, "Helical Coil Steam Generators," of the NuScale TR, Revision 3 describes NuScale's steam generator design: two counter-spiraling "helical coil" sets that wrap around the upper half of the outer surface of the riser wall, in an annular region between the riser wall and the RPV wall called the steam generator region; the so-named SG region is the transition from the would-be hot-leg to the would-be cold leg in the NPM design. NRELAP5 has been programed with a variant of the "PIPE" hydraulic component which accounts for the flow physics differences for flow in a helical/spiraling pipe as compared to a straight pipe and NuScale uses this component to model their SGs. The NPM model includes noding required to calculate the heat transfer from the primary coolant to the secondary coolant in the HCSG tubes. In the NPM design and the models, subcooled secondary side coolant flows from the feedwater pumps to the feedwater headers, where the volume flow splits among the many tubes of the HCSGs. As the secondary side coolant flows upward in the helical coils, it heats, vaporizes, and becomes a superheated steam flow before exiting the helical coils into the steam headers; from the steam headers, the superheated steam flows enter the steam piping network, exiting containment and heading to the turbine. .

During a LOCA, the SGs are isolated from the rest of the secondary system and become part of the DHRS which receives steam from the generators and transfers heat to the pool, outside of the CNV, by condensation and returns the condensate to the SG. The DHRS provides an additional means of removing decay heat from the core and controlling the core conditions that was not in the previous versions of this LOCA EM, i.e., NPM-160. With the increase in power and other control system changes for the NPM-20, the DHRS is important for providing the capacity to remove decay heat during the initial blowdown period of a LOCA, prior to actuation of the ECCS, so NuScale credits actuation of the DHRS system in all LOCA cases. The staff evaluated the impact of DHRS heat transfer degradation related to two-phase instabilities for small break LOCA cases and determined that potential variability in DHRS loop condensate flow rate, during DHRS operation, does not significantly degrade DHRS heat transfer capability.

Although the energy transferred to or from the HCSGs is accounted for in the NRELAP5 LOCA evaluations, the initial operating temperature of the HCSGs (as part of the DHRS loops) is high (maximum temperature equaling primary side T_{hot}). The most significant impediment to the functioning of the HCSGs—as parts of DHRS loops—is resistance to flow during the early phases of a LOCA; this restriction dictates the heat removal rate capability from the RPV (i.e. the core) via HCSGs as parts of DHRS loops. In addition, the potential film boiling outside of the DHRS tube surface may also adversely impact the DHRS heat removal (e.g., instantaneous heat transfer rate, heat transfer rate oscillation). The LOCA EM references to the Non-LOCA EM for the description, validation, and uncertainty analysis of the DHRS modeling. However, to additionally address the concerns of potential high tube inlet heat transfer rates producing film boiling, the staff performed several sensitivity studies with NRELAP5 and TRACE and determined that if film boiling did occur, it would be limited to the tube inlet region and could not be sustained to the point where it would cause any significant degradation in DHRS design performance capability. Therefore, the staff finds that the DHRS modeling and coupled pool nodalization is sufficient to model the overall decay heat removal responses and heat transfer capability.

The NRC staff also finds that the NuScale approach to HCSG modelling is adequate for both initial steady state and LOCA transient analysis. The NRC staff agreed that since the LOCA analyses results are insensitive to the heat removal through HCSG/DHRS loop LOCA transients, the effects of tube-plugging and fouling would be negligible.

4.5.1.6 Containment Vessel and Reactor Pool

Section 5.1.4, "Containment Vessel and Reactor Pool," of the NuScale LOCA EM TR, Revision 3, describes the NPM-20 CNV **[[ ]]** initialized at a near-vacuum, maximum CNV operating pressure with dry NCG; the near-vacuum is effectively a form of thermal insulation for the RPV, useful during normal NPM operation.

In NRELAP5 models, the containment is represented with a PIPE hydraulic component, and it is thermally connected to the various hydraulic components that represent the RPV and the containment-penetrating piping components that penetrate the RPV or attach to the RPV. The containment-internal thermal connections are modeled using **[[ ]]** heat structures that link the containment to the components that are listed in the following paragraph.

In NRELAP5 models, the RPV is represented by the following hydraulically connected components: a BRANCH for the lower plenum, a PIPE for the downcomer (below the SG region), a PIPE for the SG region (above the downcomer), a BRANCH for the riser upper plenum, and a PIPE for the pressurizer. In NRELAP5 models, the portions of the feedwater and steam lines that are inside containment (for the two SGs) are represented with a total of four PIPE components; the portions of the CVCS injection and discharge lines that are inside containment are also represented with PIPE components (one PIPE for each).

The staff confirmed that the **[[**

]]. The staff reviewed the change and concluded that it did not have any safety-significance.

The LOCA EM TR nodalizes the containment volume between the RPV and CNV to accommodate the elevation of the RRV connection to the CNV in the middle of the RPV downcomer and the elevation of the RVV and reactor safety valve (RSV) connections at the top of the pressurizer. The NuScale model also provides CNV nodes at the elevations where breaks may occur (i.e., CVCS discharge, injection line, and high point vent line connections). The NRC staff finds that nodes connecting the RPV and CNV are provided at the proper elevations for LOCA breaks and ECCS valves.

The reactor pool is the ultimate heat sink in the NPM design, which is set to minimum level and maximum temperature. The heat transfer between the CNV and reactor pool is primarily driven by the temperature difference across the CNV wall. The reactor pool volume corresponding to an individual NPM is represented by a **[[**

]]. The staff is concerned **[[**
]] would not account for the impact of the pool's thermal stratification on the DHRS heat rejection. As documented in this section, the staff has issued Request for Additional Information (RAI)-10359 (ML24290A192 (nonprorietary) and ML24290A193 (proprietary)) about the outstanding staff concerns regarding the reactor pool modeling and nodalization around the DHRS and CNV and their potential impact on the containment thermal-hydraulic response. As documented later in this section, the staff concerns about the reactor pool modeling and nodalization around the DHRS and CNV and their potential impact on the containment thermal-hydraulic T/H response have been addressed

The NPM LOCA model **[[**

]] is adequate to conservatively model the behavior of the reactor pool with respect to the containment thermal-hydraulic response**.**

LOCA EM LTR, Revision 3, Section 9.6.1, presents the impact of three CNV-RPV nodalization schemes, summarized in Table 9-7, on the NPM-20 LOCA FOMs for the RCS injection line and high point vent line breaks, with no DHRS operation credited. No nodalization sensitivity results are presented for the limiting containment design-basis accident (DBA), i.e., RCS discharge line (DL) break. **[[**

]].

In the LOCA EM TR, Revision 3, Table 9-9 lists **[[** **]]** as the initial CNV wall temperature for the entire CNV wall above the pool water level. The staff conducted an audit of the CNV wall inner surface initial temperature distribution **[[**

]] that is the initial pool temperature documented in LOCA EM TR, Revision 3, Table 9-9. The staff reviewed the various assumptions for developing the initial CNV wall temperature distribution used in NRELAP5 and found them to be conservative.

The staff looked into the impact of **[[ ]]** in the upper part of the NPM-20 containment above the RPV (LOCA EM TR, Revision 3, Figure 5-1 versus Figure 5-2) that would have maximum thermal stratification and temperatures during the DBE transient due to the lowering of the pool level from **[[ ]]** in the US460 design and assuming an adiabatic boundary condition for the CNV heat structure outer surface above the pool water level. The staff observed that for the NPM-160 design, the peak containment pressure was not sensitive to containment nodalization. In addition, **[[ ]]** between NPM-160 and NPM-20 is kept the same, and the initial wall temperature for the upper containment nodes is uniform for both designs. This addresses the staff's safety significance concerns about the reduction in the number of nodes in the upper containment region above the RPV.

The NRC staff reviewed the information in the LOCA EM LTR, Revision 3, CRAM and determined that, to complete the review,additional information was required about the reactor cooling pool heat up sensitivity and natural convection heat transfer modeling uncertainty for the US460 design. The following is a summary of the staff concerns and their resolutions:

As opposed to US600, the US460 design relies on DHRS operation for the containment DBA mitigation, which could lead to an increasing level of CNV pressurization sensitivity to DHRS performance as the break size becomes smaller and the ECCS actuation is delayed. While no detailed pool nodalization study was performed, the reactor pool heat up and thermal stratification could degrade the DHRS capacity and reduce the CNV heat removal rate as the DBA progresses, which could potentially lead to a higher CNV pressurization under delayed ECCS actuation, especially for small break LOCA. The staff concerns were addressed by the evaluation of the full spectrums of the limiting RCS discharge line (DL) and high point vent line (HPV) break sizes down to 2.2 percent of the full-break area. **[[ ]]** the staff concerns about the degradation of the DHRS performance and CNV heat removal due to pool heat up and thermal stratification and their impact on the containment pressurization accompanied by delayed ECCS actuation, especially toward the smaller break end of the spectrum. The results for both the DL and HPV break spectrums showed that the peak CNV pressure and temperature were not sensitive to the conservative pool heat up assumption around the DHRS, and there is no resulting CNV response for both break spectrums that could potentially challenge the bounding peak containment pressure (PCP) and wall temperature calculated for the limiting 100% discharge line (DL) break for the NPM-20 containment design. It was also demonstrated that the CNV pressure is reduced to below 50% of the PCP within 24 hours for all DBEs and remains at an acceptably low level (as required by GDC 38) showing no rebound despite the reactor pool heat up to near-saturated pool temperature and DHRS performance degradation, a result that was also confirmed by the staff's NRELAP5 runs for the DL break event. In the revised RAI response (MLXXXX), the same trends were also observed in the long-term cooling analysis results with a high pool temperature boundary condition. The staff concludes that the NPM-20 containment thermal-hydraulic response is not sensitive to the reactor pool heat up and stratification and there is no containment safety significance associated with it.

The CRAM and LOCA EM have used **[[ ]]**

[[

]], the uncertainty in the convective heat transfer coefficient is not significant to demonstrating adequate and sustained CNV heat transfer. The staff agrees [[

]] throughout the transient and the pool conditions do not play a role in dictating the peak containment pressure. It was also determined that the NRELAP5 sensitivity assessment of NIST-1 testing for US600 DCA showed that thermal stratification in the reactor pool has a minimal impact on LOCA figures of merit, and the approach of modeling the reactor pool with one-dimensional pipe component is adequate. Therefore, the staff concluded that the uncertainty in natural convection heat transfer modeling from the CNV and DHRS to the pool due to thermal stratification would not be safety-significant with respect to the containment pressurization and DHRS capacity

4.5.1.7 Chemical and Volume Control System

The LOCA EM 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 (discharge). NuScale stated that [[

]].

The NRC staff finds that the NuScale break spectrum (see related L/C #9 in Section 5 of this SER) conservatively bounds any LOCA inside containment associated with the CVCS due to the consideration of LOCAs in the CVCS piping, the CVCS isolation function, and the fact that the model neglects water injected prior to isolation, and therefore, accepts this model for the CVCS.

4.5.1.8 Secondary System

The LOCA EM TR, Revision 3, describes the modeling of the secondary system in Section 5.1.6. NuScale stated that the SG secondary side is [[

]].

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 with adequate conservatism. The resolution of finding is discussed in Section 4.5.1.5 of this SER.

4.5.1.9 Decay Heat Removal System Model

The LOCA EM TR, Revision 3, 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 the ultimate heat sink pool.

The staff agrees that the function of the DHRS is essential to reduce the containment peak pressure and temperature after the LOCA initiation but before ECCS actuation, especially for small breaks. The LOCA EM references to the Non-LOCA EM for the description of the DHRS modeling, validation and uncertainty analysis used in the LOCA EM. The staff's resolution of DHRS design performance capacity and NRELAP5 modeling is discussed in Section 4.5.1.5 of this SER.

4.5.1.10 NRELAP5 Modeling Options

The NuScale NPM LOCA analysis for the NPM-20 is performed with Version 1.7 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 NCG 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, Revision 3. NuScale used **[[**

]].

The NRC staff finds that the NuScale selection of junction options as shown in Table 5-1, of the LOCA EM TR, is acceptable because the options 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, for modeling choke flow.

VOLUME OPTIONS

NuScale documented its selection of the volume options for its LOCA EM, in LOCA EM TR, Revision 3, 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, for modeling break flow phenomena and friction pressure drops.

HEAT STRUCTURE OPTIONS

NuScale discussed its selection of the heat structure options for its LOCA EM, in the LOCA EM TR, Revision 3, 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 are 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 the NuScale LOCA model uses Type 171 or Type 190 for fuel rod heat structure options and 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 the use of Option 170.

4.5.1.11 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 FOMs such as peak containment pressure and CLL in the RPV riser. The NRC staff previously 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, Revision 3 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 to 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, with the exception of an L/C on the ECCS actuation signal for RPV "level." The LOCA EM describes the NRELAP5 implementation of the riser level ECCS actuation signal (and associated setpoints) for a thermal dispersion switch type sensor. Such a sensor functions by distinguishing between the presence and absence of liquid water, where the absence of liquid water at operating temperature is an indication of a decrease in primary coolant inventory. In NRELAP5, the presence or absence of local liquid water is discerned from the computed, local, cell-homogenous void fraction at the elevation of the sensor, relative to the reference height for measuring RPV level. **[[ ]]**. The staff has reflected this requirement in L/C #11 in Section 5, "Limitations and Conditions," of this SER. Evaluation of the instrumentation is performed under the SDA review and is not a subject of this review.

4.5.3 Initial Plant Conditions

In Table 5-4, "Plant initial conditions," of the LOCA EM TR, Revision 3, 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), but not the initial mass of NCGs in containment and RPV, where some NCG in the RPV arises from radiolysis of the primary coolant that occurs when the nuclear core is emitting gamma radiation. NuScale stated 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 EM TR, Revision 3, lists the process parameters associated with the plant initial conditions, which serves as input to the LOCA EM. The NRC staff reviewed the LOCA EM TR 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 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 LOCA EM TR, Section 5.4, NuScale presents its break location, configuration and size, single failure, loss-of-power, and DHRS availability assumptions as part of its break spectrum definition. The break locations considered are a maximum of 2-inch piping.

4.5.4.1 Break Location, Configuration and Size

LOCA EM TR, Section 5.4 postulates break locations in the NPM RCS injection and discharge line, pressurizer spray supply line, and high point vent lines. The staff developed L/C #9 in Section 5, “Limitations and Conditions,” of this SER to ensure all potential break locations are considered in accordance with 10 CFR 50.46 and not just those locations listed in LOCA EM TR Section 5.4.

The NRC staff evaluated the LOCA break spectrum of break sizes and locations inside the NPM-20 containment. The break sites at the RRV and RVV flanges are not included within the break spectrum in this methodology and are subject to L/C #9 in Section 5, “Limitations and Conditions,” of this SER. The NRC finds the application of the LOCA EM to the identified break locations inside containment to be acceptable. Although the RRV and RVV flange breaks are not included within the spectrum described in the LOCA EM, the staff considers the LOCA EM method to be capable of producing conservative results for those break locations.

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

]]. The applicant did not consider losses of coolant in the CVCS piping system and associated welded connections between the CNV and the first isolation valve. The NRC staff reviewed the break spectrum flow area selection against the design and based on that review, agrees that this selection is appropriate, except for the CVCS piping system and associated welded connections outside the CNV, which are subject to L/C #9 in Section 5, “Limitations and Conditions,” of this SER. The NRC staff further noted that the break spectrum demonstrates that the **[[**

]]. The 100 percent CVCS DL break with the loss of direct current (DC) power produces the limiting MCHFR.

4.5.4.2 Single Failures

As noted by NuScale in LOCA EM TR, Section 5.4.3, “Single Failures,” 10 CFR Part 50, Appendix K requires that single failures be considered within the break spectrum. For single failures, NuScale considered failures of a single RVV or RRV valve to open, and failure of one division of ECCS valves to actuate.

LOCA EM TR Section 5.4.3 also notes that the methodology includes analyzing 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. The ECCS valves are held closed with non-safety-related DC. An inadvertent actuation of a division or inadvertent MPS signal when it should not be activated, causes the removal of DC power from that division resulting in the opening of one RVV and one RRV 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 four ECCS valves. The applicant stated that this scenario is non-limiting. **[[**

]].

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 of the NPM-160 design, 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. Because the NPM-20 design incorporates IAB valves, although only on the RRVs and with modified release thresholds, the staff determined that the following information regarding the decision on the application of the SFC to the IAB valves for the NPM-160 design also applies to the IABs present in the NPM-20 design.

For the NPM-160 design, 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

1 See SECY-19-0036, “Application of the Single Failure Criterion to NuScale Power LLC’s Inadvertent Actuation Block Valves,” (April 11, 2019) (ML19060A081).

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, in 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 SDA. This is reflected in L/C #5, in Section 5, "Limitations and Conditions," of this SER.

4.5.4.3 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 availability of dc power

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.

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2 See SECY-77-439, "Single Failure Criterion," (August 17, 1977) (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) (ML003708068), and associated SRM (June 30, 1994) (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) (ML19183A408).

Specific LOCA event limiting electric power assumptions are evaluated as part of a design-specific application of this methodology, such as the NuScale SDA for the US-460 design incorporating the NPM-20. This is reflected in L/C #5 in Section 5, “Limitations and Conditions,” of this SER.

4.5.4.4 Decay Heat Removal System

LOCA EM TR, Section 9.3, “Decay Heat Removal System,” states that the DHRS adds an additional heat sink capacity during the NPM. The LOCA break spectrum is not affected by the availability or performance of DHRS. The staff finds the assumption that the DHRS is available and operates as designed acceptable because the DHRS is a safety-related system.

4.5.5 Sensitivity Studies

Section 4.9 of this SER contains the NRC staff’s evaluation of NuScale’s sensitivity studies.

4.5.6 Initial and Boundary Conditions for the NPM-20 Input Model

The NuScale LOCA EM was developed using the EM development and assessment process (EMDAP) of RG 1.203, "Transient and Accident Analysis Methods," and satisfies the applicable requirements of "ECCS Evaluation Models," 10 CFR Part 50, Appendix K, and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors." Multiple layers of conservatism are incorporated in the LOCA EM to ensure a conservative bounding approach to analyzing LOCA transients. These conservatisms stem from the application of the modeling requirements of 10 CFR Part 50, Appendix K and through a series of conservative modeling features implemented through input modeling assumptions.

The staff reviewed LOCA EM TR, Revision 3, Sections 5.5 and 5.6. Section 5.5 describes the key modeling differences between the LOCA analysis and containment response analysis, while Section 5.6 provides detailed descriptions of the NRELAP5 containment response analysis M&E release model for the primary and secondary systems DBEs. Section 5.6 also documents the conservative initial and boundary conditions to ensure a bounding M&E release, as well as the CNV and reactor pool modeling assumptions used to minimize the CNV heat removal and maximize the CNV pressure.

The earlier version of the containment response analysis methodology (CRAM) for the NPM-160 approved design was described in a separate technical report, (TR-0516-49084-P, Revision 3, “Containment Response Analysis Methodology Technical Report” (the CRAM TR), May 2020) and was incorporated by reference in Chapter 6 of the NuScale DCA FSAR. The staff compared LOCA EM TR, Revision 3, Table 5-9 with CRAM TR Table 3-4 on the primary system initial conditions, and concluded that NuScale has documented the same conservatism rationale for the CRAM initial conditions in LOCA EM TR, Revision 3, as were documented in the CRAM TR. The staff also reviewed and established equivalence between LOCA EM TR, Revision 3, Table 5-10 and CRAM TR Table 3-5 for the CNV and reactor pool initial conditions.

LOCA EM TR, Revision 3, Table 5-11 and CRAM TR Table 3-6 document the assumptions and rationale for the boundary conditions used for the CNV peak pressure and temperature calculations for primary system release events. The staff concluded that the following NPM-20 design and analysis changes made in the US460 design are appropriately incorporated in Table 5-11:

a) Loss of normal ac and dc power may not occur at time zero or on the turbine trip,

b) RVVs do not have IAB anymore and open immediately,

c) RRVs have IABs and open at the IAB design criteria based on the differential release pressure,

d) DHRS is credited to the CNV design-basis scenarios.

LOCA EM TR, Revision 3, Table 5-12 and CRAM TR Table 3-7 document the assumptions and rationale for the secondary system initial conditions. The staff finds it appropriate that the NPM-20 design and analysis changes had no impact on the secondary system initial conditions.

LOCA EM TR, Revision 3, Table 5-13 and CRAM TR Table 3-8 document the assumptions and rationale for the boundary conditions used for the main steam line break CRAM. The staff concluded that the above-mentioned four NPM-20 design and analysis changes incorporated in Table 5-11 are appropriately reflected in LOCA EM TR, Revision 3, Table 5-13.

The staff compared LOCA EM TR, Revision 3, Table 9-9 with CRAM TR Table 5-1 on the initial conditions for the primary system's M&E release event analyses. The tables present the conservative parameters values for the CRAM initial conditions for NPM-20 and NPM-160, respectively. Many parameter values were identical. The staff conducted an audit of the NRELAP5 decks and ensured that the modified parameter values were used in the NRELAP5 containment decks for NPM-20. The staff also reviewed EC-A013-7725 (NPM-20 CNV Pressure and Temperature Response Analysis) (ML23011A012) which was submitted with the NRELAP5 decks to ensure that the containment analysis results presented in the LOCA EM TR, Revision 3 also reflected the NPM-20 design parameters. The following are the staff's specific observations about some key input CNV design parameters.

a) The LOCA EM TR, Revision 3, explains that **[[**

]] is more conservative with respect to the mass release upon an event initiation.

b) LOCA EM TR, Revision 3, Table 9-9 lists CNV free volume as **[[** **]]** and 6000 ft^3 used in the CRAM, while CRAM TR Table 5-1 documented it as **[[** **]]** CRAM TR Table 3-5 also mentions that the CNV free volume used is conservatively reduced to 6000 ft^3. The staff concludes that the CNV free volume value of 6000 ft^3 is documented in LOCA EM TR, Revision 3, and is consistently used in the NPM-20 CRAM demonstration. Containment free volume is typically verified by inspections, tests, analyses, and acceptance criteria (ITAAC) because it meets the various "key parameter" attributes underscored in several subsections of SRP Section 14.3, "Inspections, Tests, Analyses, and Acceptance Criteria." Therefore, NuScale has included a one-time ITAAC for the US460 design to verify the as-built CNV free volume for the first module ever built to conservatively bound the minimum value used in the NPM-20 CRAM.

c) An initial CNV atmospheric pressure of 3.0 psia is documented in both LOCA EM TR Table 9-9 and CRAM TR Table 5-1. The CNV atmosphere is maintained at a near-vacuum initial condition in the NPM design normal operation. The staff finds using a higher initial pressure (3.0 psia) in the CRAM to be conservative, as it bounds the CNV internal pressure as well as the initial NCG mass inside the CNV to maximize the adverse effect of NCGs on CNV wall condensation.

d) Comparing LOCA EM TR, Revision 3, Table 9-9 and CRAM TR Table 5-1 showed an increase in the initial reactor pool temperature from 110 ^{o}F to 140 ^{o}F from NPM-160 to NPM-20. The staff found the change to be conservative as it would reduce the heat transfer from the containment to the reactor pool, thereby, leading to a higher peak containment pressure and temperature.

e) Comparing LOCA EM TR Table 9-9 and CRAM TR Table 5-1 showed a decrease in the initial reactor pool water level above the pool floor from 65 ft for NPM-160 to 52 ft for NPM-20. The staff finds the change to be conservative as it would reduce the heat transfer from the containment to the reactor pool, thereby, leading to a higher peak containment pressure and temperature.

LOCA EM TR, Revision 3, Table 9-9 added some additional containment response analysis methodology parameters that were not included in CRAM TR Table 5-1. The following are the related staff observations.

a) LOCA EM TR Table 9-9 includes the CNV non-condensable gas (NCG) amounts used in the CRAM. **[[**

]].

b) LOCA EM TR Table 9-9 specifies **[[**

]] maximum initial CNV wall temperature above pool level is appropriate for the NPM-20.

c) LOCA EM TR Table 9-9 includes the modified RRV IAB release pressure range of 400-500 psid (biased low) for ECCS actuation, which is a change from the 900-1000 psid IAB release pressure range approved for the DCA with the CRAM TR. The staff finds that the new range is duly incorporated in the NRELAP5 containment analysis model.

d) T-ave is a parameter that is controlled at a lower nominal value of 540 ^{o}F for the NPM-20 than for the NPM-160. The staff investigated why, despite the 56 percent core power increase, the primary T-ave value of 545°F used for the NPM-20 is 10°F lower than 555°F used for the NPM-160. A cooler average initial RCS temperature appeared inconsistent with the FSAR Figure 4.4-2 that shows an increase in the analytical design operating limits in the FSAR with Thot increasing from 590°F to 605°F and the High [Thot] Temperature Analytical Limit increasing from 610°F to 620°F, when compared with DCA Figure 4.4-9. As Thot is a function of the core power level and the RCS flow rate, the higher Thot and higher Thot analytical limit are consistent with the design power increase. However, the staff found that the lower T-ave value for NPM-20 reflects its desired nominal operating condition maintained for the US460 design by utilizing other selected design parameters. In addition, the CRAM conservatively applies an additional 5 °F margin to the 540 °F T-ave value for using a 545 °F T-ave in the NPM-20 limiting M&E release and CNV response analyses that is consistent with the uncertainty and deadband used in the FSAR Chapter 15 analyses. Therefore, the staff concludes that a credible explanation exists for the apparent inverse trend between the RCS T-ave and the plant operating temperature envelope.

The staff investigated two additional conservatisms to ensure an NPM-20 model development with respect to a conservative M&E release into the containment:

a) LOCA EM TR, Revision 3, uses the 1973 ANS decay heat standard with a **[[ ]]** for the NuScale LOCA methodology for modeling the RPV FOMs to meet the Appendix K to 10 CFR Part 50 requirements. However, the containment response analysis methodology (CRAM), also documented in LOCA EM TR, Revision 3, is based on the American National Standard Decay Heat Power in Light Water Reactors, ANS-5.1-1979, decay heat standard and a two-sigma uncertainty assumption, which is less conservative. The staff determined that using the less conservative 1979 ANS decay heat standard in the CRAM instead of the 1973 standard is justified as SRP Section 6.2.1.3, “Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs),” allows the ANS-5.1-1979 decay heat model specified in SRP Section 9.2.5, “Ultimate Heat Sink,” as an acceptable means for calculating M&E release into the containment. A staff review of ANS-5.1-1979 standard showed that it defines the uncertainty statistically based on standard deviation in a normal distribution and suggests the 2-sigma level as an appropriate decay heat uncertainty for comparison with the 1973 ANS 5.2 standard and its uncertainties. The staff also noted that even though the CRAM allows for a 2-sigma uncertainty, the FSAR Chapter 6 containment safety analyses have used a bounding **[[ ]]** that was also approved for the US600 design for the DCA. The staff accepts the decay heat model and uncertainty used in the CRAM for the NPM-20.

b) The LOCA EM TR, Revision 3, documents that the effect of liquid droplet entrainment from the break/valve flow was evaluated **[[**

]]. Based on the consistency between the NPM-160 and NPM-20 design PIRTs with respect to PCP and temperature calculations and the similarity of the thermal-hydraulics of the liquid entrainment phenomenon inside the CNV, the staff accepts that a conservative modeling approach for liquid entrainment has been used.

The staff found the NPM-20 initial and boundary conditions, assumptions, and rationales presented in LOCA EM TR, Revision 3, Sections 5.5 and 5.6, to be conservative and consistent with the submitted NRELAP5 containment model.

4.6 NRELAP5 Computer Code

As stated in the LOCA EM TR, Revision 3, Section 6.0, NuScale used its proprietary NRELAP5 system thermal-hydraulics code for evaluating small break LOCA ECCS performance. This NuScale NRELAP5 code, Version 1.7, 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 major models and code changes implemented by NuScale to model unique design features and phenomena for an 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 most recent review of the NuScale NRELAP5 computer code for the NPM-20 focused on NuScale changes and additions after NRELAP5, Version 1.4, which is the NRELAP5 code version used for the NuScale NPM-160 in the US600 design (DCA). The general applicability of the NRELAP5 (Version 1.4) code changes—which is to say, what makes NRELAP different from RELAP5-3D—to a NuScale NPM were reviewed by the NRC staff during the DCA reviews. Since the completion of the DCA review, there have been NRELAP5 code version updates (now Version 1.7), changes to NPM design, and the scope of NuScale's LOCA LTR changed; considering both the code and NPM design changes, NRELAP5 Version 1.7 is generally applicable for the uses demonstrated in the expanded Revision 3 of the LOCA EM TR, subject to the limitations and conditions in Section 5 of this SER.

The NRC staff's review for the NPM-20 was initially based on NRELAP5 Version 1.6 (ML23011A012). The major code changes included the new NSPN-1 CHF options related to increase in module power with most changes being for convenience and/or minor improvements and corrections. NuScale subsequently submitted NRELAP5 Version 1.7 (ML24228A242) for the NuScale LOCA Evaluation Methodology, replacing NRELAP5 Version 1.6. In Version 1.7, NuScale made (1) minor changes to implementation of NSPN-1 correlation, (2) improvements to **[[** **]]**, and (3) adjustments to **[[** **]]**.

In letter dated xx, xx, xxxx (MLxxxx) the applicant provided an assessment of the code version change from Version 1.6 to Version 1.7, a listing of the NPM-20 basemodel geometry changes from Basemodel Revision 2 to Basemodel Revision 5, and the code documentation supporting the new version. For the version-to-version comparisons, the applicant provided evaluations of several LOCA and IORV events and separate effect benchmark tests to show that the results of the version change are as expected related to slight increase in choked flow mass flow rate. The staff reviewed these results and audited additional version-to-version benchmark results and determined that the code version update and model changes are acceptable and consistent with this NPM methodology.

4.6.1 Quality Assurance Requirements

Compliance with QA requirements is described in MN-122626-A, “NuScale Topical Report: Quality Assurance Program Description”. The NRC staff reviewed the Quality Assurance Program Description and documented its approval in its SER (ML23361A122). Further, the NRC staff inspected NuScale’s design control process and code development procedures, and these inspections are documented in the inspection report dated April 12, 2024 (ML24099A129), ; Based on the staff’s review of the categorization of some of the engineering documents and calculations underlying portions of the LOCA EM TR, L/C #13 is placed on this LOCA EM TR approval, requiring the entirety of the LOCA EM TR (which is considered to consist of the LOCA EM, IORV EM and CRAM) to be subject to design verification, and that an applicant or licensee referencing this report must ensure the engineering documents underlying the information and conclusions contained in this report were developed and are controlled consistent with 10 CFR Part 50, Appendix B, in accordance with the NuScale QAPD, Section 2.3.1, “Design Verification.”

4.6.2 NRELAP5 Hydrodynamic Model

In LOCA EM TR, Revision 3, 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 previous versions for LOCA analyses of the U.S. Evolutionary Power Reactor (EPR) design (ML110070113) and the U.S. Advanced Power Reactor (APR) 1400 design (ML18180A327), the NRC staff only reviewed the code changes and their applicability of the unique aspects to the NPM, which did not change for the NPM-20.

NuScale updated NRELAP5 code to Version 1.7. Part of the code upgrade altered the critical flow model. The staff’s evaluation of the critical flow model changes is in Sections 4.6 and 4.8.2.2 of this SER.

4.6.2.1 Field Equations

In Section 6.2.1, “Field Equations,” of the LOCA EM 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 Notice (IN) 92-02, "RELAP5/MOD3 Computer Code Error Associated with the Conservation of Energy Equation," dated January 3, 1992, and 92-02, Supplement 1, dated February 8, 1992. The RELAP5 code was modified by INL to resolve the issues raised by NRC IN 92-02 and 92-02, Supplement 1, prior to NRELAP5. 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 NuScale SwUM-0304-17023, "NRELAP5 Version 1.6 Theory Manual," dated January 6, 2023 (ML23011A012) (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). The staff also reviewed the NRELAP5 Theory Manual regarding code modifications from NRELAP5 Version 1.4 to the current Version of 1.7.

As discussed in the associated audit report (ML20010D112), 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 formulated and implemented in the code in a way that is acceptable for natural circulation nuclear reactor flows (that is, where there are not large sources and sinks of coolant momentum).

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).

4.6.2.2 State Relations

In the LOCA EM TR Section 6.2.2, NuScale discussed 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 previously reviewed and approved, these state equations are acceptable.

4.6.2.3 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 EM 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 EM 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 EM 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 EM TR.

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

4.6.2.4 Momentum Closure Relations

In LOCA EM 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 LOCA EM 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 LOCA EM 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 EM 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.

4.6.2.5 Heat Transfer

Section 6.2.5, "Heat Transfer," of the LOCA EM 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 NCGs 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 EM TR Figure 6-2, "NRELAP5 boiling and condensing curves," illustrates these three regions of the curve.

NRELAP5 default modeling uses the Chen boiling correlation for nucleate boiling up to the CHF point. NRELAP5 will issue a message and stop running if the CHFR reduces below one for core heat transfer. NuScale added this stop function to NRELAP5—for LOCA analyses only—when the core CHFR drops below one because maintaining core CHFR above one is a critical FOM for the NuScale LOCA EM design.

The NRC staff finds that the addition of the stop function to NRELAP5 is appropriate—for LOCA only—because acceptability of the NuScale LOCA EM depends on maintaining the water level above TAF. The detailed review of CHF correlations is documented in Section 4.11 of this SER and in the SER for Revision 2 of the LOCA EM (ML19331A516).

With respect to containment analysis, the staff compared LOCA EM TR, Revision 3, Table 5-7 with CRAM TR Table 3-1 for the NRELAP5 models used in the containment response analysis methodology (CRAM). The staff noted that the approved CRAM TR Table 3-1 had specified Cooper and Rohsenow boiling correlations to model the boiling heat transfer, while LOCA EM TR, Revision 3, omits the Cooper and Rohsenow correlations and rather uses Chen correlation for boiling. An audit showed that the Cooper and Rohsenow boiling correlations were only generically added to NRELAP5 but were not used in the containment response analyses. The staff concludes that the omission of the Cooper and Rohsenow boiling correlations from the LOCA EM TR, Revision 3 does not reflect a change in the CRAM, as the methodology only uses Chen correlation to model nucleate boiling. The staff also compared the CNV and reactor pool models as tabulated in LOCA EM TR, Revision 3, Table 5-8 and CRAM TR Table 3-2 and finds the information to be identical.

4.6.3 Heat Structure Models

As discussed in LOCA EM TR 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 (LWR) applications. Therefore, they are applicable to NuScale LOCA analyses. These boundary options can be used to specify unique heat transfer and CHF models and/or geometry configurations that invoke particular heat transfer schemes.

LOCA EM TR Section 6.3 also states 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 EM 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. These remain applicable to the NPM-20 design.

4.6.4 Point Reactor Kinetics Model

As described in LOCA EM 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 previously reviewed and approved. The NRC staff finds 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 previously reviewed and approved, these NRELAP5 trip and control system models are acceptable for NuScale applications.

4.6.6 Special Solution Techniques

As stated in LOCA EM TR 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.

4.6.6.1 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 requirements in 10 CFR Part 50, Appendix K. NRELAP5 includes options, as described in Section 6.6.1, "Choked Flow," of the LOCA EM TR, Revision 3, for switching between **[[**

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]]. The staff notes that this conservative option would require an applicant or licensee referencing this report to seek an exemption from the requirement of Appendix K I.C.1.b, which is specified in L/C #1 in Section 5 of this SER.

The LOCA EM TR, Revision 3, documents that containment response analysis methodology (CRAM) maximizes the CNV peak pressure and temperature by using a bounding mass and energy release from the RPV to CNV. For this purpose, the CRAM uses the [[

]], which is an important conservatism in modeling the critical flow for M&E release into the CNV consistent with Appendix K to 10 CFR Part 50. LOCA EM TR, Revision 3, Sections 8.2.2 and 8.2.3, demonstrate the adequacy of the EM for two-phase and single-phase choked and un-choked flows for predicting the M&E release based on the assessment of the NRELAP5 mass flow predictions with experimental data.

LOCA EM TR, Revision 3, which incorporates the CRAM, suggests a [[

]]. The staff concluded that the code changes did not make any significant changes in the M&E release or CNV T/H response that justifies the break-flow modeling approach and the fidelity of calculation of M&E release into the CNV.

4.6.6.2 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-contracts 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 previously approved for LOCA analyses, the NRC staff considers these models to be acceptable.

4.6.6.3 Counter Current Flow Limitation

NRELAP5 implements the general CCFL model in a form proposed by Bankoff which has the structure shown in LOCA EM 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 LOCA EM 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 Section 4.9.7, "Sensitivity Studies, of this SER." 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," of the LOCA EM TR,, Revision 3, "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. In its audit, the NRC staff also considered the potential distortion on core inlet temperature and SG steam temperature relative to the uncertainty for NIST-1 test results.

During its audit (ML24262A230), staff noted that Equation 2.6-112 in the NRELAP5 Theory Manual (Reference 20) appeared to **[[**

]], and is considered to be a more accurate formulation.

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 assumes no plugging or fouling in the HCSGs, and DHRS cooling does not generally have a significant role, the detailed SG model evaluation is found in the NRC staff's SER for the NuScale TR-0516-49416, Revision 4, "Non-Loss-of-Coolant Accident Analysis Methodology," (ML24334A049). L/C #12 in Section 5 of this SER is applied to this revision of LOCA EM, mandating that this edition of NuScale LOCA EM may only reference Revision 5 of the NuScale Non-LOCA EM. **[[**

]].

4.6.7.1 Helical Coil Tube Friction

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

]]. The staff notes that helical coil NRELAP5 component internal flow friction factor is modeled to be independent of internal tube wall roughness. The NRC staff considers the in-tube friction model acceptable for LOCA analyses.

4.6.7.2 Helical Coil Tube Heat Transfer

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

]] 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 data on separate effects heated tube wall and pressure drop data. Further detailed evaluation is documented in the NRC staff's SER of the NuScale TR-0516-49416, Revision 4 (ML24334A049).

4.6.8 Wall Condensation

In the event of a mass and energy (M&E) release into the containment vessel (CNV), CNV wall heat transfer plays a significant role in the NPM-20 reactor core cooling and containment heat removal. LOCA EM TR, Revision 3, Section 6.8, “Wall Heat Transfer and Condensation,” presents a detailed discussion of the wall heat transfer, condensation correlation, and the related NRELAP5 model. The process of condensation and retention of RCS within the CNV facilitates the passive transfer of decay heat energy to the reactor pool (UHS) and the RCS return to the RPV for the ECCS operation. The heat removal from the CNV is accomplished passively when the heat of steam condensing on the CNV inner wall is conducted across the CNV wall to the reactor pool in which the NPM is immersed. The resulting condensate film flows down the inner CNV wall and accumulates at the bottom of the CNV shell. Natural convection inside the condensate pool that collects in the lower portion of the CNV due to condensation also contributes to the CNV heat removal to the reactor pool. Following the conduction through the CNV wall, heat transfer on the outer surface of the CNV wall is primarily driven by natural convection inside the reactor pool (UHS), except when the outer surface temperature would exceed the local saturation temperature causing subcooled nucleate boiling heat transfer at higher CNV elevations.

The staff determined that NuScale had provided the same description, condensation correlation, and equations in the LOCA EM TR, Revision 3, that had been presented, analyzed, and approved as a part of LOCA EM TR, Revision 2. The PIRT ranking and knowledge level of the containment heat transfer phenomena that include condensation remain unchanged for NPM-20. The staff did not expect the applicability of the condensation correlation or the effect of non-condensable gas to be sensitive to the NPM-20 design as the containment free volume, PIRT, and the resulting containment pressure and temperature responses for NPM-20 are similar to that of NPM-160. Previously, during the LOCA EM TR, Revision 2, review, the staff had performed a detailed review of the condensation heat transfer modeling for the containment safety analyses presented in “Containment Response Analysis Methodology Technical Report”, TR-0516-49084-P, Revision 3 (CRAM TR), and had found the applicability and implementation of the condensation correlation in NRELAP5 to be conservative for calculating minimum CNV heat transfer and peak containment pressure.

As discussed in LOCA EM TR, Revision 3, Section 6.8, NRELAP5 uses **[[**

]].

Considering the safety significance of the condensation phenomenon, it is important to first summarize the most important aspects of the wall condensation modeling already established during the LOCA EM TR, Revision 2, review before finalizing the staff findings for LOCA EM TR, Revision 3. As condensation is a highly significant phenomenon for the ECCS performance and CNV thermal hydraulic response, the NRC staff reviewed the information provided by NuScale during the LOCA EM TR, Revision 2, review (ML17324B392 and ML19240C658) regarding the modeling of heat transfer from the CNV to the pool for a primary coolant release from the RPV, and audited additional calculations underlying the submitted information, as described in the LOCA EM TR, Revision 2, audit report (ML20010D112). The NRC staff had also reviewed and audited the NRELAP5 code, as discussed in the audit report, and noted that **[[**

]], the staff had also conducted an audit (ML19282C504) to determine if the use of this correlation was appropriate for the comparatively large diameter NuScale CNV.

NuScale states in LOCA EM TR, Revision 3, Section 6.8 that the **[[**

]]. The staff's LOCA EM TR, Revision 2, review had confirmed **[[**

]]. During the LOCA EM TR, Revision 3, review, the staff audited the NRELAP5 code changes, condensation convergence issue, and energy balance, as follows:

a) NuScale EC-A013-7725 (NPM-20 CNV Pressure and Temperature Response Analysis) (ML23011A012) describes **[[**

]] encountered with the NPM-20 wall condensation is not safety significant.

b) The staff audited the safety analysis graphic results for the limiting RCS discharge line break for the NPM-20 CNV design CNV liquid and vapor temperatures; CNV wall temperature; reactor pool temperature; and energy balance. The NPM-20 results were similar to the ones presented for NPM-160 in the CRAM Technical Report, TR-0516-49084-P, Revision 3, and no non-physical behavior or inconsistency was observed.

c) The staff conducted an audit of the NRELAP5 code changes implemented between the NPM-160 and NPM-20 designs that could impact the prediction of PCP and CNV wall temperature. The audited information covered the NRELAP5 applicability to NPM-20 analyses for the code changes from NRELAP5 Version 1.4 through Version 1.7. Based on the audited information, the staff concludes that the impact of the NRELAP5 code changes on the containment M&E release and calculated PCP is not significant.

The staff concludes that **[[** **]]**. The staff also recognizes that the condensation of steam in the NPM-20 occurs in a fairly comparable-to-slightly-lower pressure and temperature CNV environment than in the NPM-160. In addition, the PIRT ranking and the knowledge level of the condensation phenomenon have stayed the same and no new phenomenon is identified in the NPM-20 to affect the condensation characteristics inside the CNV. Therefore, the staff concludes that extending the applicability and usage of the **[[** **]]** to the NPM-20 design is appropriate. The conclusions drawn about the condensation modeling for the NPM-160 in the LOCA EM TR, Revision 2 are also applicable to the NPM-20 design and LOCA EM TR, Revision 3. The general condensation modeling approach is conservative for calculating CNV heat transfer, peak containment pressure, and collapsed water level above the core.

4.6.8.1 Effect of Non-condensable Gases on Condensation Heat Transfer

The LOCA EM TR, Revision 3, presents the formulation and assumptions for modeling the adverse effect of NCG on condensation heat transfer inside the containment that would lead to a higher CNV peak pressure as compared to pure steam. Air is assumed to be the only NCG species within the CNV during NPM-20 operation, for capturing the adverse effects of CNV NCGs on steam condensation on the CNV inner surface. Per LOCA EM TR, Revision 3, Table 9-9, air is initialized at a high initial pressure of 3.0 psia to maximize the NCG concentration inside the nearly evacuated CNV, at the beginning of the event. However, the staff found that the NRELAP5 containment biased decks, **[[** **]]** inside the CNV to be conservative.

NRELAP5 uses a two-fluid model for flow of a two-phase vapor/gas-liquid mixture that can contain NCG components in the vapor/gas phase as well. LOCA EM TR, Revision 3, Section 8.2.8.2 describes that **[[** **]]** can be used to model film condensation with and without the presence of NCGs. The presence of NCGs suppresses condensation due to the reduced saturation temperature corresponding to the partial pressure of vapors at the condensate film interface. With NCG present, NRELAP5 uses the Colburn-Hougen diffusion model (1934) to iteratively calculate the interface temperature between the bulk and pure NCG layer to satisfy the energy balance on the condensing wall. The staff reviewed the modeling details of non-condensable gas effects in LOCA EM TR, Revision 3, and found them identical to what was reviewed in LOCA EM, TR Revision 2.

The containment response analysis methodology (CRAM) as presented in LOCA EM TR, Revision 3 conservatively assumes that the maximum mass of NCGs that could exist within the

RPV during operation **[[** **]]** the effects of NCG on the CNV heat removal. LOCA EM TR, Revision 3, **[[** **]]**. The staff audited the NPM non-condensable gas mass calculations inside the NPM-20 under steady state operating conditions. The total NPM NCG source term is calculated **[[** **]]** case NCG analysis case was conservative based on the bounding initial CNV pressure/temperature, pressurizer level/temperature, and control rod position applicable to the containment design-basis analysis. The staff finds it reasonable to assume air as the only NCG species in the RPV and CNV as sensitivity studies of the NCG chemical composition during the US600 design certification had demonstrated that to be an appropriate assumption.

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)” of the LOCA EM TR, the interfacial drag model for large diameters is not considered reasonably accurate. However, this model can be used in the NuScale LOCA analyses, because there is a large margin in the minimum CLL in the inner vessel predicted for the spectrum of break sizes and locations in the LOCA analyses.

[[**]]**. Because the minimum CLL is above the TAF for the spectrum of break sizes, the uncertainties in the predicting level swell demonstrated in the GE level swell comparisons would not produce liquid levels below the TAF.

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 uranium-239 (239U) and plutonium-239 (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.

4.7 NRELAP5 Assessments

Section 7, "NRELAP5 Assessments," of the LOCA EM TR 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 SETs and IETs 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 SETs and IETs 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, containment peak pressure and temperature event analysis, and IORV occurrence.

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 EM 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 2008/1a - 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.

4.7.2.1 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 EM 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.

4.7.2.2 General Electric Level Swell Test – 1 foot

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 EM 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 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 LOCA EM TR, Section 4, for NuScale LOCA analyses is a CLL in the NPM riser, the NRC staff agrees with NuScale that the void fractions predicted by NRELAP5 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.

4.7.2.3 General Electric Level Swell Test – 4 feet

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 EM 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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4.7.2.4 KAIST

NuScale assessed the DHRS condensation modeling of its NRELAP5 code against experimental data from KAIST. The review of results of the assessment analysis is found in the NRC staff's SER of the NuScale TR-0516-49416, "Non-Loss-of-Coolant Accident Analysis Methodology," Revision 4 (ML24334A049); the outcome is deemed acceptable for the purpose of the LOCA SER for the NPM-20.

4.7.2.5 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 EM 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 EM 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.

4.7.2.6 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 EM TR. The specifics of the test configuration and instrumentation are described by NuScale in its LOCA EM 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 EM 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 EM 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 EM 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 EM 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 considers 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.

4.7.2.7 Semi Scale (S-NC-02 and S-NC-10)

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

NuScale modeled the Semi Scale test facility and compared the loop mass flowrate as a function of system inventory to the measured data from the Semi Scale 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 killowatts (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 Semi Scale tests is acceptable.

4.7.2.8 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 LOCA EM TR, Section 7.2.8, "Wilson Bubble Rise." The specifics of the test configuration and instrumentation are described by NuScale in its LOCA EM 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. In an NRELAP5 model of the experiment, using pressure close to the NPM-20 primary side normal full power condition, the code overpredicts void fraction. For a selection of modeled pressures, the code generally overpredicts void fractions but the error is usually within 25 percent. The NRC staff finds this part of the assessment acceptable.

4.7.2.9 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 EM TR. The specifics of the test configuration and instrumentation are described by NuScale in its LOCA EM 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.

4.7.2.10 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 EM TR. The specifics of the test configuration and instrumentation are described by NuScale in its LOCA EM TR, and the NRC staff reviewed these descriptions.

NuScale modeled the Bankoff Perforated Plate test facility and compared computed ratio of vapor superficial velocity to liquid superficial velocity to the same quantity derived from experimental data. 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.

4.7.2.11 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 EM TR. The specifics of the test configuration and instrumentation are described by NuScale in its LOCA EM 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 24," of the LOCA EM 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 4.11 of this SER and in the staff's SER for Revision 2 of the LOCA EM TR.

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. 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 Test Assessment Cases

The NuScale Power Module is dramatically different from other typical LWRs. As discussed in Section 7.5, "NuScale NIST Test Assessment Cases," of the LOCA EM TR, NuScale built the NIST test facility at Oregon State University to obtain test data relevant to its unique NPM design and approach to LOCA evaluations. NIST was designed to model the major components of the NPM: the NPM at **[[** **]]** scale. NuScale performed a number of NIST tests to assist in validation of the NRELAP5 system thermal-hydraulic code, and the NRC staff reviewed the summarized test information in the LOCA EM TR. Further, the NRC staff performed a detailed audit (ML24262A230) of the scaling analysis for NIST tests, as well as LOCA test assessments.

4.7.5.1 Test Facility

NuScale built the NIST 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 EM TR provides a schematic of the NIST facility. The original NIST facility was modified and upgraded several times to bring the facility in-line with the NPM-20 design configuration. The latest primary update was to increase the maximum allowable working pressure to better represent the NPM-20 configuration. Even though NuScale attempted to minimize the distortions between the NIST scaled test facility and the NPM, the NRC staff notes that distortions cannot be eliminated. Therefore, the NRC staff evaluated the NIST 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 staff noted that the NIST NRELAP5 model **[[** **]]** a riser hole sensitivity assessment provided by the applicant (ML24353A252 (nonproprietary) and ML24353A253 (proprietary)).

The purpose of the sensitivity study was to validate the effect of riser hole modeling used in NPM methodology analyses and NIST integral test assessments to confirm that NRELAP5 modeling shows expected phenomenal results from the presence of riser holes. The riser holes provide a flow path for liquid and boron transport to mitigate redistribution inside the RPV for (1) extended DHRS cooldown and (2) extended ECCS cooldown, respectively. The riser holes also affect RCS flow distribution during normal operations and accident conditions. The upper riser holes generally flow from the riser into various midpoints within the steam generator primary and account for approximately **[[ ]]** of the RCS flow. The lower riser holes generally flow from the downcomer back into the upper riser region, bypassing the core, and accounting for approximately **[[ ]]** of the RCS flow. The applicant indicated **[[ ]]**.

The applicant provided an assessment of selected LOCA and non-LOCA accidents and NIST benchmarks to show the sensitivity in model results in response to variations in riser hole size. Increasing the riser hole sizes increases mass flow to the steam generator midpoints but also increases the flow that bypasses core heating through the lower holes. During an NPM-20 transient, increases in riser hole size result in minor increases in steam generator primary and downcomer temperatures that cause corresponding minor effect differences in steam generator secondary and DHRS flow and pressures. For the NIST tests, **[[ ]]** for the sensitivity study. Based on the results of the various assessment sensitivity studies, the staff agreed with the applicant's conclusion that the NRELAP5 model responses are consistent with physics-based expected results and that there are very negligible effects on the event FOMs.

The NRC staff finds that one of the significant distortions of NIST relative to the NPM is that the NIST representation of the RPV, is not contained within the NIST representation of the CNV. In addition, the NIST representation of the CNV is not immersed in the NIST representation of the cooling pool vessel (CPV). The NIST models of the RPV and CNV are separate vessels connected by piping. LOCA EM TR, Figure 7-75, shows how the NIST valves that represent the RRVs and RVVs and potential pipe breaks enables the NIST facility to measure flow through this piping. The NIST representation of the CNV is connected to the NIST representation of the CPV through a heat transfer plate (HTP). The size of the NIST 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 test facility represents the NPM nuclear fuel with electrically heated rods. These NIST electrically heated rods establish a natural circulation flow up through the riser to the NIST SG and then back to the core like the NPM design. The NIST 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 Test Assessment Cases," of the LOCA EM TR, data from the NIST facility are used for both integral and separate effects validation of various phenomena. The NRC staff reviewed the NIST facility design and determined that the areas of potential distortions were appropriately identified and their impact on the degree of agreement of code-to data assessment. One area that the NRC staff focused its review on, was the reduction of the CNV distortion. For certain NIST tests, the NIST CNV is preheated to reduce the distortion between the NIST CNV arrangement and the actual NPM design. In NIST, **[[ ]]**. Other areas of scaling distortions that exist in RPV, SG, or CPV, are discussed in Section 4.8.3 of this SER.

4.7.5.2 *Integral Effects LOCA Test Procedure*

As described in Section 7.5.1.6, “Integral Effects LOCA Test Procedure,” of the LOCA EM TR, a valve and switch lineup is performed to configure the NIST facility for each test. The NIST 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 facility operates at a lower pressure than the NPM, and the fluid masses are scaled. The NRC staff audited the facility and test procedures for NIST-1 and found them to be in compliance with QA Test Control requirements.

The NRC staff understands the limitations of the NIST facility, which NuScale has accounted for in its test procedures. However, the NRC staff notes that these differences between NIST and the NPM mean that a limited direct comparison can be made between NIST tests to NPM LOCA results. The NRC staff notes that NIST is a test facility for LOCA code development and that NIST is the only facility that closely represents an NPM to simulate a LOCA.

4.7.5.3 *Facility NRELAP5 Model*

The NRC staff reviewed and audited details about NuScale’s NRELAP5 nodalization model, as described in the associated audit report (ML24262A230), which is similar to the model used for the NPM and is described in Section 7.5.2, “Facility NRELAP5 Model,” of the LOCA EM TR. The NRELAP5 model is a complete one-dimensional representation of the NIST test facility. The NRC staff finds that the NuScale NIST NRELAP5 model provides an acceptable representation of the NIST test facility to evaluate the capability of NRELAP5 to model NIST tests.

4.7.5.4 *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 EM 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.

4.7.5.5 *Separate Effect High Pressure Condensation Tests (NIST-1)*

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 LOCA EM TR, Section 7.5.4, “Separate Effect High Pressure Condensation Tests (NIST-1).”

NuScale reported 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 peak CNV pressure is over-predicted by NRELAP5. 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 previous associated NPM-160 EM 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 apply 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 determined as part of its review of the LOCA EM TR, Revision 2, that the NRELAP5 computation of pressure in HP-02 demonstrates a conservative computation of pressure.

During its review of Revision 2 of the LOCA EM, 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 noted that while there was a general agreement between the temperature data and NRELAP5 computations, the CNV fluid temperatures at the lower elevation were 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 1,800, 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) for the review of Revision 2 of the LOCA EM TR, 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 **[[**

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4.7.5.6 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 EM TR and the NRC staff reviewed these descriptions.

NuScale calculated form losses for a base run using Idelchik as referenced in the LOCA EM 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 EM 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) as part of its review of Revision 2 of the LOCA EM TR, 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.

4.7.5.7 Reactor Coolant System Discharge Line Loss-of-Coolant Accident Loss-of-Coolant Accident Integral Effects Tests (NIST-1)

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 EM 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 previous LOCA EM, Revision 2 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 previous LOCA EM TR, Revision 2, 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 previous LOCA EM TR, Revision 2, 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 finds that this information provides sufficient justification for applying the NRELAP5 pool condensation model.

4.7.5.8 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--106, “Comparison of NIST-1 HP-06 and HP-06b containment vessel pressure,” of the LOCA EM TR, showed that NRELAP5 slightly overestimated the measured NIST peak containment pressure with a negligible deviation. As discussed in the associated audit report (ML20034D464), as part of its review of Revision 2 of the LOCA EM, 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 completed tests performed to support Revision 2 of the LOCA EM, 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 for the NPM-160 DCA review, 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 were 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.

As part of its review of Revision 2 of the LOCA EM, 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 showed a conservative over-prediction of containment pressure by approximately 10 to 12 psi, therefore, there was sufficient margin to conclude that NRELAP5 adequately predicted 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 finds the assessment results to be acceptable to justify the use of the NRELAP5 code to perform peak containment pressure analysis. No additional testing was needed to support the CRAM, as presented in LOCA EM TR, Revision 3, and demonstrated for the NPM-20 design in US460.

4.7.5.9 NIST-1 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.

4.7.5.10 Spurious Reactor Vent Valve Opening Tests

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 the 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 EM TR, which include core power, RVV mass flow rate, RCS pressure and level, and 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.

4.7.5.11 Spurious Reactor Recirculation Valve Opening Integral Effects Test

NuScale performed the HP-49 test to assess the capability of NRELAP5 to predict the integral response of the NIST-1 facility to an inadvertent opening of an RRV without the DHRS. The entire NIST-1 facility except the CVCS, PZR spray, and DHRS was used in this IET. The SG was active, the HTP was preheated, the CNV was pre-pressurized, and the CPV was filled. The NRC staff reviewed the comparisons provided in the LOCA EM TR, which include valve mass flow rate, RPV and CNV pressure and level. The NRC staff agrees that the comparison between the NRELAP5 calculated, and test measured results are in a reasonable agreement and the assessment is, therefore, acceptable.

4.7.5.12 NIST-2 LOCA Integral Effects Test Series

NuScale upgraded the NIST-1 facility to the NIST-2 facility in 2018 primarily to increase the maximum allowable working pressures for the main SG, and the DHRS in order to perform integral effect tests for NPM-20. The NIST-2 LOCA integral effects tests are intended to investigate the Phase 1a and 1b LOCA and IORV phenomena. Seven NIST-2 LOCA IETs for various line break configuration, including both LOCA and IORV cases, are documented in Section 7.5.10 of LOCA EM TR. The NIST-2 IET test procedures are similar to the corresponding NIST-1 tests and the ECCS actuation varied from test-to-test. The staff focused their review on the high importance ranked LOCA and IORV phenomena that are listed in Section 4.6 of the LOCA EM TR. The applicant explained that the phenomena distortions observed in various NIST-2 tests are mainly due to facility limitations and different event timings, and are not significant compared to other dominant phenomena. The staff reviewed the response and accepted the distortions. (ML24326A347 (nonproprietary) and ML24326A348 (proprietary)) In general, the staff observed the NIST-2 test events and accident progression matched reasonably well to their corresponding NRELAP5 simulations. The staff noticed some deviations exist in RPV and CNV levels between NIST-2 Run 1 and Run 6 and their corresponding NRELAP5 simulations. The staff further audited (ML24262A230) the LOCA test report and agreed these discrepancies are mainly due to the boundary conditions and heat transfer modeling differences between NRELAP5 and NIST-2 tests. **[[**

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[[ ]]. Since these NIST-2 LOCA IETs are intended to investigate the phase 0 and phase 1 of the accident/event, the test-to-NRELAP5 comparison shows these IETs captured most transport processes in the RPV and CNV during the early phases of the accident, and hence, is acceptable.

4.7.5.13 NIST-2 IORV Integral Effects Test Series

The NIST-2 IORV IETs and the corresponding NRELAP5 simulations documented here are intended to demonstrate a better understanding of phenomena related to Phase 0 of an IORV event and to assess the NRELAP5 code's capability for the rapid depressurization process. The two NIST-2 IORV IETs, IORV Run 3 and Run 6 are performed with different valve opening configurations. The IORV cases and the corresponding NRELAP5 simulations are documented in Section 7.5.11 of LOCA EM TR. The test procedures are similar to those previously discussed in Section 4.7.5.11. Because the most important phenomena for IORV is CHF, for which the MCHFR occurred in Phase 0, the staff focused their review on the mass and energy exchange during the early blow down of the rapid depressurization phase. The PZR pressure, and M&E flows were evaluated. Based on the comparison between the measured core differential pressure (dp), PZR pressure and the calculated results, the staff agrees that the pressure trend of test data and NRELAP5 results match well which indicates these IETs captured the majority of the accident progression for the IORV depressurization process. The mass and energy release rate comparisons for Run 3 (all ECCS valves open case) shows good comparison and for Run 6 they are of minimal agreement. The staff considers this acceptable as it has little impacts on the FOM of IORV evaluations.

4.7.6 Containment Response Analysis Methodology Assessment – NPM-20 Only

The EMDAP requires an applicability demonstration of the computer code and tests. Extensive NRELAP5 code validation was performed during the review of LOCA EM TR, Revision 2 and the DCA, to ensure that the LOCA EM is applicable for all important phenomena and processes over the range encountered in the NPM LOCA. The NRELAP5 validation suite includes many legacy separate effects tests (SETs) and integral effects tests (IETs), as well as many SETs and IETs developed and run specifically for the NPM application. The IETs were performed at the Oregon State University NuScale Integral System Test (NIST) facility, which is a scaled representation of the complete NPM primary and secondary systems, as well as the reactor pool.

The NRC staff had conducted a detailed review of the assessment of the NRELAP5 code against NIST-1 test facility experimental data during the review of the LOCA EM TR, Revision 2. A unique aspect of the demonstration provided for the NPM is the comparison of NRELAP5 simulations of LOCA events to NIST test data and NRELAP5 simulation of the same LOCA event in an NPM. In those comparisons, the NPM results were scaled down to the NIST size using the scaling ratios used to design the NIST test facility. A reasonable-to-excellent agreement was obtained while comparing the NRELAP5 predictions with HP-49, HP-06, HP-06b, HP-07, HP-09 and HP-43 NIST test data, which established the applicability of NRELAP5 to accurately predict LOCA phenomena at both the NIST and NPM scales. The results showed a conservative prediction of containment pressure response to conclude that NRELAP5 adequately predicted peak CNV pressure for the NIST-1 facility tests. As no new PIRT phenomena were identified to impact the containment pressure and temperature response for the NPM-20, the staff found it reasonable that no additional experimental validation was performed for benchmarking the CRAM for the NPM-20 as presented in LOCA EM TR, Revision

3. The NRC staff finds the LOCA EM TR, Revision 2, assessment results to be acceptable to justify the use of the NRELAP5 code to perform peak containment pressure analysis for the NPM-20.

4.8 Assessment of Evaluation Model Adequacy

In Section 8, “Assessment of Evaluation Model Adequacy,” of the LOCA EM 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-160 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 EM 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 EM 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 EM 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.

4.8.2.1 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 the 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 cross-flow models. The result is a larger CHFR when cross-flow 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 for LOCA analysis.

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 EM 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 EM 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.

4.8.2.2 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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4.8.2.3 ***[[*** ***]]***

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

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

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

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

4.8.2.4 ***[[*** ***]]***

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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 EM TR, which indicates that **[[**

]].

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. **[[**

]]. 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. The paper notes with importance 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 **[[ ]]** to be acceptable.

4.8.2.5 ***[[***

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4.8.2.6 ***[[***

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4.8.2.7 ***[[***

]]. The staff has asked for a justification **[[**
]] from the CRAM. By letter dated XX, XX, XXXX (MLXXXX), NuScale revised both the LOCA and CRAM methodologies in LOCA EM TR, Revision 3 to use **[[**

]]. The staff agrees that heat transfer from the lower head to the reactor pool has a minor impact on the CNV pressure response. Therefore, using the **[[**

]] for modeling heat transfer from the lower hemispherical CNV head does not have any safety-significance with respect to the CNV T/H response

In general, the staff agrees with the LOCA EM TR, Revision 3, **[[**

]] as an overarching verification that the changes made in the NRELAP5 code have not altered the formulations and the numerical solutions of the fundamental balances and constituent models as coded in RELAP5-3D.

4.8.2.8 ***[[***

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4.8.2.9 ***[[***

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4.8.2.10 *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. The NRC staff judges NRELAP5 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 SET, where the model tends to over-predict two-phase levels and void distribution behavior in the axial direction.

4.8.2.11 ***[[***

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4.8.2.12 ***[[***

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4.8.2.13 ***[[***

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4.8.2.14 ***[[***

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4.8.2.15 ***[[***

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4.8.2.16 ***[[***

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4.8.2.17 ***[[***

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4.8.2.18 ***[[***

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4.8.2.19 ***[[***

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4.8.2.20 ***[[***

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4.8.2.21 ***[[***

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4.8.2.22 ***[[***

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4.8.3 Evaluation of Integral Performance (Top-Down Assessment)

4.8.3.1 Review of Code Governing Equations and Numerics

The NRELAP5 code governing equations and numerics are described in the LOCA EM 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 are discussed in Sections 4.6.2 of this SER. After reviewing the NIST-2 and NPM20 designs, the same conclusions in the LOCA EM TR Revision 2, SER (ML20181A269) apply.

4.8.3.2 NuScale Facility Scaling

The NIST facility is designed to simulate the integral system behaviors of a single NPM (NPM-160 or NPM-20) immersed in a reactor building pool. The NRC staff audited the applicant's scaling analysis, as discussed in the associated audit report (ML20034D464 for NIST-1 and ML24262A230 for NIST-2), to review the NIST 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, DHRS operation 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 to capture the high-ranked phenomena identified in the LOCA PIRT. The figure of merits (FOMs) in the NuScale design are the minimum CLL) in the core, the peak CNV pressure and CHF. The CLL is a surrogate for PCT.

NIST-1 and NIST-2 facilities have inherent distortions due to their small size and different component layouts compared to the NPM. Distortion also arises due to the difference in operating conditions for specific transients. The distortions identified in NIST-1 apply to NIST-2 since the major components are not changed. The scaling analysis covers the CVCS LOCA transient (HP-06) and additional LOCA scenarios: the high-point vent line break (HP-07) and inadvertent opening of RVV (HP-09) and RRV (HP-49). These break locations cover both reactor coolant vapor space and liquid space. The NRC staff audited (ML24262A230) the applicant's NRELAP5 analyses for NIST-1, NPM 160, NIST-2 and NPM-20, including evaluation of scaling distortions and their impacts on FOMs.

4.8.3.2.1 Scaling Methodology

Hierarchical Two-Tiered Scaling (H2TS) 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 EM TR, supplementary information provided by the applicant (ML19058A867) and audited the details of the implementation of this methodology in the NuScale scaling analysis. The NRC staff concludes the methodology is appropriately used in this scaling analysis.

The NRC staff audited both stages of the applicant's scaling analysis. 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 and the dimensions remain the same in NIST-2. 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 PI groups.

The NRC staff reviewed the four groups of transient phenomena analyzed by NuScale, including: vessel depressurization and containment pressurization during the blowdown and venting phases, the long-term recirculation phase and the reactor building pool heat up. The NRC staff found that NuScale's scaling analyses correctly identified the control volumes of interest, and the interactions between components and phases of event progression. In the NPM-20, the geometry of the entire RPV and SG remains the same. Similarly, the RPV and SG remains the same except for the secondary piping and DHRS. However, the thermal power has increased significantly from the NPM-160 to the NPM-20, namely the core power increases from 160 megawatts (MW) to 250 MW. Due to the facility limitations, the heater rods operate at a maximum level of 390 KW, lower than the scaled NPM power. The staff focused on auditing the LOCA decay power curve to ensure the decay curve is scaled from the NPM-20 according to the volume ratio. Some limitations of NIST-2 core heater were reported, which caused distortions in different phases of LOCA transients. The staff audited (ML24262A230) information made available by the applicant and found the distortion did not adversely impact to the similarity between NIST-2 and NPM-20 in the interested FOMs.

4.8.3.2.2 *Reactor Coolant System Natural Circulation*

The facility was designed to preserve event time and power-to-volume ratio. The NRC staff's audit focused on four RPV 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 facility was designed to have a much higher loop resistance than the NPM-20. The NIST SG scaling and the derivations of nondimensional PI 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 the scaling ratios of the NIST-1 and NPM-160 dimensions; they remain the same for NIST-2 and the NPM-20. Therefore, the conclusion drawn from NIST-1 applies to NIST-2.

The NRC staff reviewed single-phase natural circulation analyses for the NPM-160 at different powers (100 percent and 50 percent) and different pressures and confirmed that there is not much effect from the pressure on the NIST natural circulation flow. Therefore, the distortion due to a lower scaled power in 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-160 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 event starts, the phenomena are the same if the decay power is scaled from 100 percent NPM decay power. In NIST-2, the 100 percent power of the NPM-20 is higher than for the NPM-160. As in NIST-1, the steady state scaled flow ratio and temperature distribution in NIST-2 may deviate from the NPM-20, but the LOCA phenomena remain unaffected if the decay heat is correctly scaled. The staff has confirmed the decay heat scaling.

The NRC staff also performed NRELAP5 confirmatory calculations, which found that the appropriate NIST-1 loop resistance was established to confirm the one-to-one time ratio requirement. However, the staff needed to establish the scalability of NIST-2 through comparison of non-dimensional groups of buoyancy and resistance, especially in long-term cooling phase when the natural circulation will be in two-phase. The applicant provided justifications (ML24326A334) that NPM-20 operating conditions are within NPM-160 operating range which was validated in NIST-1 scaling. NuScale also explained that NIST-2 has inherent limitation in heater power and no new phenomena were expected in the power ramping process, thus there was no need to conduct HP-05 in the NIST-2 campaign. In addition, with **[[** **]]**, HP-05 assessment has improved agreement. The staff considers that the natural circulation flow in this test is single phase, and the physics involved is well-known. Based on this information, the NRC staff concludes that the scaling analyses for natural circulation is acceptable**.**

4.8.3.2.3 *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 EM 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 and concluded that it is correct in terms of identifying control volumes and phases of LOCA progression. Since the NIST-1 scaling analysis was reviewed and accepted (ML20181A269), the focus is in the NIST-2 scaling analysis. Before a scenario is simulated in NIST2, it is important to have properly scaled initial and boundary conditions. As NIST facility is limited in power and pressure, it can simulate LOCA transients at a point when the NPM pressure is 1650 psia. This will happen at different times for different LOCA transients. Sections 4.2 and 4.3 of the NIST-2 LOCA scaling report describe ideal scaled values and practical values used for initial conditions for four LOCA transients. As the transients begin at different times due to facility pressure limitations, the stored energy is different and creates some distortion. The staff reviewed **[[**

]]. The other important boundary condition is the core power since it is the driver and high-ranking phenomenon. As the tests were performed in the conditions when the NPM was in the decay heat mode, the decay heat in the facility needs to be scaled correctly from the NPM. The scaled core heater power is a curve fit shown in the NIST-2 LOCA scaling report. It shows that the actual decay powers used in the tests are slightly different than the ideally scaled decay power. The staff audited the curves used in the NIST-2 tests; they are appropriate.

4.8.3.2.4 Reactor Coolant System Depressurization Scaling

NuScale’s scaling formulation includes vessel M&E balance equations. **[[**
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]]. Therefore, the NRC staff finds that this phenomenon does not significantly affect the minimum collapsed water level during this period. However, the staff noted [[

]] and is therefore more conservative. Therefore, the conclusion in the distortion analysis is acceptable. [[

]]. Therefore, the NRC staff determined that NuScale's scaling approach for this phase is acceptable. Since the NPM-20 credits DHRS operation in LOCA events, the non-dimensional groups associated with secondary side (SG/DHRS) should have been included in top-down scaling. However, DHRS was not actuated in the NIST-2 LOCA and IORV experiments. The applicant stated that DHRS operation before ECCS actuation is similar to the non-LOCA loss of feedwater transient, and the justification of extending the scalability of NIST-2 DHRS to LOCA scenarios is provided in the non-LOCA EM TR Section 5.4, "Conclusions of NRELAP5 Applicability for Non-LOCA." After reviewing the applicant's rationales (MLXXXXX), the staff concluded that some uncertainties still exist but the impact to the LOCA FoMs is insignificant in terms of heat removal capacity relative to ECCS.

4.8.3.2.5 Containment Pressurization Scaling

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]] comparison between NIST-2 and the NPM; the distortion is similar, and the staff finds the approach NuScale takes to resolve this distortion to be acceptable.

4.8.3.2.6 Long-Term-Cooling Phase Scaling

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, with flow back into the vessel at the RRV, and heat loss to the pool. During this phase, the pressure drops between the RPV and CNV is determined by flow resistance and the flow rate. As the actual NIST hydro-static driving head was scaled to less than that of the NPM, the flow resistance in the NIST was evaluated to confirm the resistance ratio.

For the CNV inventory balance, the two important phenomena are RRV flow and phase change at the CNV wall. The RRV flow shows significant distortion but is in the acceptable range. For the energy balance, the most important phenomenon is the CNV wall heat transfer, and the next important one is RVV energy flow. The distortions in both phenomena are considered insignificant by the NRC staff.

4.8.3.2.7 Reactor Pool Heat Up Scaling

The NRC staff audited (ML24262A230) NuScale's scaling related to the ultimate heat sink. In NIST-1 and NIST-2, an HTP 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 and 2 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. This conclusion applies to NIST-2 and NPM-20.

4.8.3.2.8 *As-Built NuScale Facility Scaling Summary*

NuScale presents the NIST-1 scaling and distortions summary as a base for NIST-2 scaling design. Some significant NIST-1 distortions were identified during the DCA. These distortions are related to layout difference of RPV, CNV and pool compared to the plant configuration, and the initial and boundary conditions. The NRC staff concludes that the summary in the LOCA EM TR accurately described the distortions, which did not invalidate the scaling analysis results. Since NIST-2 is based on NIST-1, most primary systems are the same. Most distortions in NIST-1 are carried over to NIST-2 and the approaches NuScale took to mitigate these distortions are still applicable to NIST-2. Due to the power-to-volume differences from the NPM160 to the NPM-20, staff's review focuses mainly on the distortions created by this factor. Beyond the scaling distortions, NIST-1, and NIST-2 data are useful for determining code applicability.

4.8.3.3 Assessment of NuScale Facility Integral Effect Test Data

The NRC staff reviewed NuScale's NIST-1 and NIST-2 IET tests that support NPM calculations and NRELAP assessment of data in HP-05, HP-06, HP-07, HP-09, HP-43 and HP-49. NuScale performed sensitivity studies for these tests to evaluate the code uncertainties. The level of agreement between the prediction and test data is used to demonstrate the applicability of NRELAP5 in modeling high-ranked phenomena. The staff's review conclusions on the NIST-1 test data also apply to NIST-2. NuScale also conducted a series of NIST-2 LOCA and IORV tests. These tests are similar to NIST-1 test series. A few variations of these tests were added to examine the effects of NPM design changes, e.g. delayed ECCS response and venturi effects of the ECCS valves. The staff concludes that these tests do identify the new phenomena and the impact of NPM-20 design changes. However, these NIST-2 LOCA and IORV tests did not invoke DHRS operation. The test demonstrated only LOCA behaviors without DHRS heat removal. NuScale tested its DHRS with non-LOCA transients. Since the boundary conditions of the DHRS loop are different in LOCA and non-LOCA events, the staff reviewed the justifications for extending the DHRS applicability to LOCA scenarios in Section 4.8.3.2.4, which provides the staff's conclusion.

NuScale relied on validation with NIST1 data for NPM 160 in the DCA. The volume of NPM-20 remains the same as that of NPM-160, and yet the power increases by 58 percent. To clarify the influence of power increase as boundary condition during LOCA events, NuScale submitted the requested NIST-2 decay heat curves and their comparison to scaled NPM-20 decay power history in various tests (ML24326A331). The information was reviewed and deemed correct by the staff. The pre-break steady state power distortion affects the stored energy, which is important for the long term transient. The NRC staff evaluated the PI group representing the phenomenon and the distortion is acceptable. It is important that the scaling of the steady state scenarios before the LOCA initiates is correct to ensure a well-scaled initial condition. Since the NPM-20 has a higher power level than the NPM-160 at different steady state conditions, scaling analyses at different power levels is important. NuScale provided explanations (ML24326A334) to alleviate staff's concern via the HP-05 test discussed in 4.8.3.2.2. In reviewing the scaling analysis, the staff noted that the scaling analysis results were pre-test analysis estimates only used to develop NIST-2 test matrix and conditions. Therefore, the applicant performed (MLXXXX) an updated scaling and distortion analysis on representative tests based on the as-run conditions, i.e., HP-06 and HP-07. These two tests were chosen because larger scaling distortions were shown in the pre-test analysis and they cover both liquid space and steam space LOCAs. As to the as -run IORV scaling analysis update, the staff does not expect much difference after reviewing the updated HP-06 and HP-07 scaling results since the IORV test was performed using a similar procedure and the same facility. It is sufficient for the staff to review and evaluate the distortion based on the pre-test analysis. The overall as-run HP-06 and HP-07 results show slightly higher distortion compared to the pre-test PI group numbers in some phenomena. The applicant explained that these additional distortions are because **[[**

]]. These additional distortions do not change the phenomena and ranking, and the distortions are known from DCA and still within the tolerance range. The applicant performed extensive assessments in the LTR with these NIST-2 tests, and the code-to-data agreement is excellent for the figures of merit. Therefore, the staff concludes that the scaling analysis procedure outlined in the topical report has been fulfilled**.**

In the pre-test scaling analysis, the formulation of scaling and assessment of NIST-2 data are included. The code-to-data distortions were identified and explained. Because of compensating errors in the IET tests, the comparison of non-dimensional groups is essential to identify the ranking of important phenomena and the distortions. The analysis provides related PI groups to the figure of merits for four LOCA transients, HP-06, HP-07, HP-09 and HP-49. Some distortions were identified**.** In the analysis, NuScale provides **[[**

]].

These comparisons are helpful to see the overall design differences between the NPM-160 and NPM-20, but it is not sufficient as it does not directly relate the important phenomena to FOMs as the non-dimensional groups do. To demonstrates the phenomenon differences between the NPM-160 and NPM-20, NuScale further provided **[[**

]] all cases because of the higher values for the NPM-20 compared to NIST-2. The staff concludes that NIST-1 scalability for the NPM-160 is not completely extendable to NIST-2 for the NPM-20. Therefore, the similarity has to be established directly by comparing the high rank non-dimensional groups based on the goal of the specific test. In other words, some distortions can be allowed as long as the high ranked phenomena are reproduced in the test facility. In the scaling analysis, a direct comparison of PI groups between NIST-2 and NPM-20 were presented **[[**

]]. The high-ranking phenomena were of the same order of magnitude and so the scaling for NPM-20 and NIST 2 are acceptable for code validation.

4.8.3.4 Evaluation of NuScale Integral Effects Tests Distortions and NRELAP5 Scalability

The NRELAP5 code needs to be validated with Integral Effects and Separate Effects tests. These tests scale plant prototype into test facility model for intended phenomena and transients. However, in integral tests like NIST-1 and NIST-2, the data and code predictions may have good agreement due to compensating differences. Scaling analysis decomposes the integral effects into effects from individual phenomenon. and quantifies the distortion in these individual phenomena. Code predictions for IET data is sufficient to build confidence in the ability of the code to predict NPM performance only if the distortions in IET are within acceptable limits. The LTR refers to NIST-1 and NPM 160 scaling analyses and code calculations (ML20181A269, ML20010D112) because the test facilities have major components of interest in common. The analysis and calculation are referred in the TR for code applicability as well. The staff reviewed LOCA EM TR Section "7.5.10 NIST-2 LOCA Integral Effects Test Series" and Section 7.5.11 "NIST-2 IORV Integral Effects Test Series," to ensure consistency between the scaling findings in Section 8.3.2.4 "As-Built NuScale Facility Scaling Summary" and test data assessment. Some deviations identified were not directly related to scaling parameters but were because of the uncertainties in the test facility itself and the NRELAP models. The staff also concluded that some deviations could not be identified in the NRELAP assessment but were identified in the deviation of nondimensional scaling groups in Section 8.3. Some distortions identified are described below.

[[

]]. However, it was minor compared to the break energy phenomena from the scaling analysis, and therefore, the explanation provided by the applicant (ML24326A338) is acceptable to the staff.

The staff also noted **[[**

]]. The staff accepts this approach. In the real NPM operation, a higher pool temperature is also in the conservative direction for DHRS operation. Therefore, the staff accepts the proposed resolution to modeling the pool at a higher temperature than the expected operational limit.

While reviewing the NIST-2 HP-06 test, the staff identified a time scaling issue**.** The timing of the sequence of events distorted significantly. One obvious example is the lowest collapsed RPV level. In the NPM, with the help of DHRS, the staff would expect the NIST-2 RPV to reach the lowest level faster. According to the applicant's scaling methodology, NIST-2 was scaled with no time distortion, i.e., synchronicity. During the audit (ML24262A230), NuScale explained **[[**

]]. Therefore, the RPV CLL stays flat after it reaches 10 ft above TAF. The NRC staff agrees this distortion is partially contributed to by a code limitation and by multiple biases being applied in FSAR Chapter 15 safety analyses. To demonstrate the synchronicity scaling assumption through representative NIST-2 examples with appropriate initial and boundary conditions, in response to the staff's concern, the applicant demonstrated (MLXXXX) that **[[**

]]. With this additional analysis and justification, the staff concludes that NIST-2 chronology scaling is maintained as it was in the NIST-1 scaling.

In reviewing the NIST-2 LOCA RUN 6 test, the NRC staff noticed the RPV and CNV level behave differently than the base case. The deviation indicates some phenomena were not predicted in the model. NuScale explained (ML24326A336) that the deviation was due to the **[[**

]]. These differences cause the trends of RPV and CNV level to behave differently in timing. Also, NRELAP5 predictions in RUN 6 agree well with the data, which shows the NRELAP code applicability. Therefore, the staff accepts the explanations.

NIST-2 core heater power in the LOCA test is simulated by a fitted curve. In the pre-test IORV scaling reports, the staff audited (ML24262A230) the fitted curve was presented but no deviation was evaluated compared to the scaled NPM decay heat. **[[**

]]. The deviation affects the RPV depressurization (ML24326A340**)**. Evaluation of this distortion and its impact to the FOM is included in the staff's SER (ML24355A062) on the NuScale TR-124587, "Extended Passive Cooling and Reactivity Control Methodology."

As in the LOCA tests, the NIST-2 core heater power was scaled from the NPM decay power in the IORV tests. However, due to the limitation in the NIST-2 DACS system, the scaled NPM decay power could not be accurately prescribed. Some approximations were made in the tests, but the actual core power used was significantly lower in Phase 0 of the transient. This is presented in the IORV scaling analysis report. The distortion aggravates as the NPM power increases. **[[**

]], therefore the MCHFR thermal hydraulic condition and behaviors in Phase 0 is reproduced well in the fast depressurization transient. The staff accepts the explanations, and finds the test distortion is acceptable.

The NRC staff concludes that, because of the size of the NuScale test facilities (NIST-1 and NIST-2) and some inherent limitations in pressure and power, there are unavoidable scaling distortions. These distortions have been identified by the applicant, and their impact on the FOM have been fully assessed and mitigated. Therefore, the test results from NIST-1 and NIST-2 are acceptable for the NRELAP5 code assessment for NPM-20 LOCA and IORV events.

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-20 and focused on the 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 EM 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 EM TR identifies key models and correlations which are important for predicting the NPM LOCA ECCS performance following a LOCA. These are listed in LOCA EM TR, Section 8.2, “Evaluation of Models and Correlations (Bottom-Up Assessment),”

The NRC staff noted that the list does not contain Baker-Just correlation 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 the requirements of Appendix K to 10 CFR Part 50, as well as the decay heat model, which uses the 1973 ANS standard with the 1.2 multiplier and inclusion of actinide decay. In light of the higher power in the NPM-20, the model is considered conservative.

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 the NPM-20 due to the design and the quick cool down of the reactor vessel wall, the NRC staff agrees with NuScale that the downcomer boiling in the 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-20. 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 SET 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 (SETs) are dedicated to the study of a single particular phenomenological characteristic, so the measurement instrumentation can also be chosen more appropriately, and SETs can also be at larger scale with better control of initial and boundary conditions.

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 EM TR, were evaluated by the NRC staff, 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 SETs. However, the NRC staff concluded that these deficiencies would not have a significant effect on the 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 worst-case level 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 lack of sufficient number of integral tests for NRELAP5 validation. And, further, there are only the NIST-1 and NIST-2 facilities that apply directly to the NPM design, which NuScale successfully compared and benchmarked the NRELAP5 code to. It is evident that the NRELAP5 is capable of reproducing the NIST-1 and NIST-2 LOCA results. It was also recognized that there were distortions in NIST facilities due to size and configuration. However, high ranking phenomena were present in the NIST facility, making them adequate for validation. From this, pending closure of the above noted open and confirmatory items, it is the NRC staff's judgement that it is not unreasonable to expect that NRELAP5 is capable of producing the NPM-20 LOCA results.

In addition, NuScale successfully applied the NRELAP5 code to two Semi Scale 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-20.

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 EM 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 and NIST-2 data assessment. Based on the staff's evaluation of the applicant's applicability assessment, the NRC staff considers this approach to be acceptable, pending closure of the above noted open and confirmatory items.

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 EM 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., the CCFL model) or used conservative assumptions for scale-dependent phenomena (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 considers the approach acceptable, pending closure of the above noted open and confirmatory items.

Section 4.7.5.8 of this SER also presents an evaluation of the NRELAP5 applicability to the peak containment pressure analysis performed during the LOCA EM TR, Revision 2, review. The results showed a conservative prediction of containment pressure by NRELAP5 as validated by several NIST-1 facility tests. As the PIRT and CNV free volume have remained unchanged for LOCA EM TR, Revision 3, the staff finds the use of the NRELAP5 code to perform peak containment pressure analysis of the NPM-20 to be 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 example calculations presented by NuScale are for the NPM-20 design. The applicant stated that the LOCA EM remains applicable to the approved NPM-160 design as well; however the staff is imposing L/C #10 in Section 5 of this SER.

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 EM TR, and audited the underlying calculations, as described in the associated audit report (ML24262A230). 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 and no single failure. 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 FSAR Chapter 15 of an 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.

LOCA analyses for the NPM-20 are discussed in the following sections. This SER focuses on the design changes in the NPM-20 design. NPM-160 results are presented by the applicant in the LOCA EM TR, Revision 3, to provide a comparison to the NPM-20 examples. In some places, this SER retains certain aspects of the SER for NPM-160 that are applicable to the NPM-20 and did not need to be reanalyzed but do need to be discussed in order to present a comprehensive review of the NPM-20.

4.9.2 Phase 1 LOCA Results and Conclusions – NPM-20

For a 100 percent break of the RCS injection line (this is liquid space), as with the applicant's analysis of the NPM-160 design, the NuScale analysis of the NPM-20 design shows that the core remains covered with sufficient liquid and that the MCHFR is not violated. This demonstrates that the methodology given in the TR returns a conservative estimate of LOCA CLL and MCHFR behavior for liquid space break locations.

For all electrical power availability scenarios, the high point vent breaks (which are vapor space) bound the results for the pressurizer spray line break. With AC power lost, the high point vent line breaks are more limiting than the liquid space breaks due to the delay in RSV lift. For the most limiting EDAS power scenario, the discharge line break bounds all vapor space breaks.

For the Phase 1 NPM-20 results, all cases have a post scram MCHFR above the steady state CHFR value. The liquid space breaks give the highest CNV peak pressure. For all the break locations and sizes (see Section 9.2 of the LOCA EM TR), the CLL converges to approximately 8.5 feet above the TAF. The 100 percent injection line break (liquid space) with DHRS operation, no loss of power and failure of one ECCS division is the most limiting case for the minimum CLL above the TAF

It is important to note that, for the NPM-20 design, removal of the IABs on the RVVs allows the RVVs to begin to open immediately upon a loss of all electrical power (that is, both ac and dc), or in cases where at least dc power is available, after the MPS acts on the low riser level signal, which has a **[[ ]]**. This opening of the RVVs occurs much sooner than the time it took for the NPM-160 design to reach the RVVs IAB release pressure during small break size events. Elimination of the significant delay between ECCS actuation and RVV opening (due to the IAB lockout) also eliminates the very low minimum collapsed levels for smaller liquid space breaks (less than 35 percent pipe flow area) observed for the NPM-160. The RVV IAB lockouts for the NPM-160 led to lower collapsed liquid levels in the NPM-160 because of significant flashing of the RCS inventory and continuing inventory loss through the break during the IAB lockout period, as well as the fact that the prior revision of the LOCA EM did not credit DHRS for heat removal during that time.

Based on its review of the applicant's representative calculations for liquid and vapor space breaks 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 predictable results.

4.9.3 Break Size

The NRC staff reviewed NuScale's spectrum of break areas for different break locations given in the LOCA EM TR. The break areas range from 100 percent (2.23 in^2) down to 2.2 percent (0.05 in^2). The break locations include the RCS injection line, the RCS discharge line, the high point vent line, and the pressurizer spray line. A 1-in pipe diameter narrowing (intended to hold an orifice) limits the pressurizer spray line break to a 35 percent relative break area.

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 those for the maximum break size.

The area ratios between break area and maximum break area presented in Table 5-5, “Summary of analyzed break sizes,” of the LOCA EM TR and 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 LOCA EM TR, Section 9.2, “Break Size” and Section 9.3, “Decay Heat Removal System” of the LOCA EM TR.

NuScale presented the results of a spectrum of breaks in the discharge line, injection line, and high point vent, and pressurizer spray supply. The NRC staff observes that the limiting break was the 100 percent RCS injection line break for the minimum CHF ratio (MCHFR). The CLL above TAF in the RPV riser is a function of break size for both RCS injection and high point vent line breaks. The final equilibrium levels established after the pressure equalization are independent of break size and location. The minimum collapsed level is typically the equilibrium level of approximately 8.5 ft for all break sizes.

Besides the observations, NRC staff evaluated the LOCA break spectrum of break sizes and locations inside the NPM-20 containment.

The NRC staff finds that these analyses and sensitivity studies contain sufficient parameter and break size variations to properly identify the limiting break that produces the minimum liquid level above the TAF. The NRC staff finds, subject to L/C #9 in Section 5 of this SER, that the limiting small break injection line LOCA is correctly identified, based on the minimum liquid level above the TAF.

4.9.4 Decay Heat Removal System Availability (LOCA-Specific Perspective)

The applicant credited DHRS operation in all LOCA cases for this revision of the LOCA EM. The DHRS adds an additional heat sink capacity during the LOCA. When DHRS operation is taken into account, all break sizes behave similarly and minimum CLLs—a LOCA FOM—are maintained very close to the final equilibrium level for the various break sizes. The NRC staff finds that the DHRS is important to containment response by providing capability to remove thermal energy stored in the primary system, and decay heat, during the initial blowdown period of a LOCA but prior to actuation of ECCS. The staff has evaluated the impact of DHRS heat transfer degradation in small break LOCA cases if the SG is subjected to density wave oscillation (DWO) or if a whole DHRS loop is subject to various two-phase flow instabilities and found such instabilities to have insignificant impacts on the DHRS total integral heat removal from the reactor primary side.

4.9.5 Power Availability

The discussion in the previous sections assumes that both ac and dc power are available during the NPM-20 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. Because the IABs were removed from the RVVs in the NPM-20 design, the RVV valve opening is immediate (when this is a response to a loss of dc power, not a signal from the MPS) and not governed by the IAB release pressure as it is in the NPM-160 design. The immediate RVV opening caused by the loss of ac and EDAS (dc) power results in higher CNV peak pressures across the break sizes for the HPV break compared to the all-power available scenario. However, the RRVs are not opened until the pressure difference between the RPV and CNV reach the IAB release threshold.

The NRC staff reviewed NuScale's analysis results and as described previously, the minimum CLL is typically the equilibrium level which is approximately 9 feet above TAF for all break sizes and power scenarios. The loss of both ac and dc power has a significant impact on the steam space breaks, down to a 2 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 electrical power available. Based on its review of the applicant's representative calculations for the different assumptions on power availability described in this section, the staff concludes that the sample analyses appropriately illustrate that implementation of the methodology as specified in the LOCA EM TR will provide conservative and expected results.

4.9.6 Single Failure

The NRC staff reviewed NuScale's results of analyses assuming the single failures described in LOCA EM TR, Section 9.5, "Single Failure" of the LOCA EM 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 US460 SDAA. This is reflected in L/C #5 is Section 5 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.

4.9.7 Core Collapsed Liquid Level Calculation

The NRC staff previously reviewed NuScale's calculation of the core CLL and audited the calculations underlying it as described in the previous NPM-160 audit report (ML20034D464). The NRC staff noted 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 staff noted that the volume-based approach for CLL more appropriately represents "riser volume averaged liquid level" rather than a true riser CLL. This method could reduce the 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 EM TR acceptance criteria employs the CLL and CHF, both of which must be met, and because the NRC staff found 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 accepted the applicant's use of a volume-based CLL to determine minimum core inventory levels. The NRC staff noted that the NPM-20 uses of the same volume-based CLL which is also acceptable based on the reasons cited above.

4.9.8 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 EM TR and audited the underlying calculations. These parameters included nodalization, time-step size, counter current flow at the pressurizer baffle plate, and ECCS valve parameters. The NRC staff also reviewed NuScale's evaluation of the sensitivity of the LOCA results on the core power distribution, including axial power shape, hot fuel assembly radial peaking, and the initial reactor cooling pool temperature. Some of the sensitivity studies in Sections 9.6 of the LOCA

EM TR were performed for the NPM-160, but the generic nature of the evaluation makes them applicable to the NPM-20 as well.

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 EM 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.

4.9.8.1 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 9-7, "Number of volumes in RPV and containment vessel nodalization" of the LOCA EM 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 coarse NRELAP5 nodalization in the containment generates a slightly earlier ECCS actuation signal compared to the coarse and fine nodalization. NRELAP5 calculates a similar LOCA response in the 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 calculating 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.

Based on the discussion above, the NRC staff finds that the NRELAP5 nodalization used by NuScale is appropriate to conservatively predict the CLL and CHF margin. The evaluation of the NPM containment vessel and reactor pool nodalization is described as part of Section 4.5.1.6 within this SER that evaluates the NPM-20 containment vessel and reactor pool models that are discussed in the LOCA EM TR, Revision 3, Section 5.1.4.

4.9.8.2 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 9-95 through Figures 9-98 in the LOCA EM 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 EM TR, the NRC staff finds the NRELAP5 time step selection process to be acceptable.

4.9.8.3 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 9-99, "Effect of counter current flow limitation line slope on levels for 100 percent high point vent line break" of the LOCA EM TR 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.

4.9.8.4 Emergency Core Cooling System Valve Parameters – NPM-160

The content in this subsection is retained from the SER for the LOCA EM TR Revision 2 (for the NPM-160), because design changes for the NPM-20 did not require this portion of the staff's review to be repeated for this revision of the LOCA EM; changes for the NPM-20 that are not consistent with the contents of this section are discussed in the following subsection.

During its review of the LOCA EM Revision 2, 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. **[[**

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The NRC staff reviewed NuScale's sensitivity results, shown in LOCA EM TR, Figures 9-101 and 9-102, which illustrate the effect of RRV and RVV valve sizes on the peak CNV pressure and the minimum CLL as functions of break size. **[[**

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Additionally, the NRC staff performed sensitivity studies 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 the ECCS on pressure, particularly for steam space breaks and those with the DHRS active. Small liquid space breaks and those without use of the DHRS may continue to actuate on a high CNV level.

4.9.8.5 Emergency Core Cooling System Valve Parameters – NPM-20

The NuScale modeling of the NPM-20 ECCS valves is essentially the same as for the NPM-160 design; however, for the NPM-20: the number of RVVs is reduced, the NPM-20 ECCS valves include venturi nozzles instead of orifices (where venturi nozzles have much lower form losses once flow choking has ended), RVVs no longer have IABs, the number of trip valves on the RVVs and RRVs were doubled, and the release pressure for the RRVs was reduced. While the ECCS valves changed between the NPM-160 and the NPM-20, there are no changes to the methodology, specifically related to the valves, presented by the previous revision of the LOCA LTR (Revision 2); therefore, the staff maintains that the previous staff review and conclusions are suitably applied to the NPM-20. Note, there is discussion in Section 1 and Subsection 4.8.3.3 of this SER supporting or otherwise repeating the statements about the NPM-20 versus the NPM-160 ECCS valves in this subsection. While the ECCS actuation signal also changed from the NPM-160 to the NPM-20, the portion of the LOCA evaluation methodology specific to the ECCS valves is unaffected by that particular NPM design change.

4.9.8.6 Initial Reactor Pool Temperature

The NRC staff reviewed NuScale's sensitivity studies covering the range of initial pool temperatures from 40 ºF (4 ºC) to 140 ºF (60 ºC) to investigate the impact of the pool temperature on NRELAP5 calculated LOCA EM FOMs. NuScale evaluated the reactor pool temperatures, ranging from **[[**
]] 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. **[[**

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]]. 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.

4.9.8.7 Core Power Distribution – NPM-20

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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Based on the discussion above, as long as NuScale LOCA analyses continue to show that the minimum riser CLL remains above the TAF, the NRC staff finds that the core power distribution used in the NuScale LOCA EM is acceptable.

4.9.9 Loss-of-Coolant Accident Calculation Summary – NPM-20

The NRC staff reviewed NuScale's summary conclusions based on its break spectrum calculations as listed in the LOCA EM TR Section 9.1, "Loss-of-Coolant Accident Progression in the NuScale Power Module" and the break size sensitivity studies as listed in the LOCA EM TR Section 9.2, "Break Size". NuScale LOCA EM TR Sections 9.1 and 9.2 present 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 analyses for the NPM-20 design.

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 used the NPM-20 design changes and the applicant's NRELAP5 NPM-20 base model to develop the NPM-20 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 Analysis

Using the developed TRACE NPM model, the NRC staff performed a spectrum of LOCA analyses of different LOCA break locations and sizes. The staff performed sensitivity calculations to confirm LOCA progression trends and to evaluate the impact of different single failure assumptions and investigate margins to the key FOMs.

c. NRELAP NPM LOCA, IORV and Containment Analysis

The NRC staff audited NuScale's NPM-20 modeling using documents cited for reference or input to the submitted calculation documents that describe the various NRELAP5 input files used for analyses such as those on LOCA, IORV, and containment peak pressure and temperature. The staff's audit is described in the associated audit report (ML24262A230); the audited content has some overlap with what was audited for Revision 2 of the LOCA EM TR Review (associated audit report ML20010D112) in cases where the NRELAP5 input files were materially changed for the NPM-20 from the NPM-160. The staff focused its review on ensuring the code and scenarios evaluation methodology could still satisfactorily simulate the design given changes in the calculation and revisions in several areas where the design changed such as the nuclear power increase, the change in metal materials, the lowering of the reactor pool water level, the change in ECCS actuation signal, the changes in the ECCS valve, and the changes to input modeling style (e.g., NRELAP5 component selections and nodalizations).

The staff imported the NRELAP5 codes (v1.6 and then v1.7) and input models into the NRC's Symbolic Nuclear Analysis Package user interface processor for performing engineering analysis. Using these input models, the NRC staff performed LOCA, IORV, and containment peak pressure and temperature analyses for the limiting break locations and the opening of different ECCS valves to better understand key characteristics of the NPM LOCA evaluation methodology and the NPM-20 design.

d. TRACE NIST-2 Benchmark Analysis (LOCA RUN 1, LTC-01)

Similar to studies for NPM, the NRC staff used NIST-2 design information and the applicant's NRELAP5 models to develop the NIST-2 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-2 data, the NRC staff performed independent assessments for the key NIST-2 LOCA and LTC tests to confirm the LOCA progression trend and the applicability of TRACE.

e. NRELAP NIST-2 Benchmark Analysis (NIST-2 LOCA model)

The NRC staff also performed NIST-2 LOCA model benchmark confirmatory cases with NRELAP5 to better understand the NIST-2 facility's unique behavior and characteristics as well as the NIST-2 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).

4.9.10 Primary and Secondary System Release Scenario Containment Response Analysis – NPM-20 Only

The NPM-20 is an advanced, light-water, integral pressurized water reactor (PWR) that uses a compact, high-pressure steel containment vessel (CNV) partially immersed in a reactor pool coupled with passive safety-related systems. The secondary system includes a traditional steam-power conversion system including a steam turbine generator, condenser, and feedwater system. The piping in an NPM-20 containment that can potentially break is limited to the 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. Pipe breaks inside the CNV are evaluated as LOCA, while the ones outside the CNV are not defined as LOCA. Inadvertent opening of RVVs and RRVs leading to a decrease in RCS inventory inside the RPV are not included in the LOCA definition. However, the staff found it appropriate that the LOCA EM has been extended to model inadvertent RRV and RVV opening transients, as the thermal-hydraulic phenomena and the PIRT remain the same for LOCA and non-LOCA events.

LOCA EM TR, Revision 3, is not intended to provide the final NPM-20 design results. Rather, it provides example NRELAP5 analysis results to demonstrate the LOCA EM application to several design-basis transients for the NPM-20. The LOCA EM TR, Revision 3 presents these analyses for several design-basis break spectra that cover a range of break/opening locations. Each spectrum covers AC and EDAS power loss, ECCS actuation level, single failure, and corresponding initial and boundary conditions. The LOCA EM methodology is used to evaluate the mass and energy (M&E) release from the spectrum of primary and secondary systems design-basis transients and accidents, and the resulting CNV pressure and temperature response. The containment response analysis methodology (CRAM) documented in the LOCA EM TR, Revision 3, also supports analyses for non-LOCA events, as well as the extended passive cooling evaluation.

LOCA EM TR, Revision 3, Section 9.7, summarizes the containment response analysis methodology pressure and temperature results for a total of nine mass and energy (M&E) release scenarios that include six primary system, two secondary system (MSLB/FWLB), and one IORV M&E release event. The LOCA EM TR, Revision 3, also states that the input parameters and assumptions are chosen to maximize peak CNV temperature and pressure. LOCA EM TR Table 9-9 summarizes the common initial conditions for the M&E release analysis. The LOCA EM TR, Revision 3, presents the descriptions and results for the following nine initiating events of the M&E release into the CNV:

1. LOCA caused by RCS (CVCS) discharge line break from the downcomer (limiting CRAM event) (DL)

2. LOCA caused by RCS (CVCS) charging or injection line break from riser (limiting LOCA event) (CL)

3. LOCA caused by RPV high point vent line break near the top of the vessel (HPV)

4. AOO due to inadvertent opening of an RVV caused by a mechanical failure (RVV)

5. AOO due to inadvertent opening of an RRV caused by a mechanical failure (RRV)

6. AOO due to inadvertent opening of both RVVs caused by an inadvertent ECCS actuation signal (ECC)

7. MSLB

8. FWLB

9. IORV

The staff reviewed the results of each of the six primary containment M&E response analysis release cases as summarized in the LOCA EM TR, Revision 3, Sections 9.7.3.1 through 9.7.3.6. Each of the six sections contains a table of seven DBEs analyzed for the spectrum based on the respective assumptions for loss of AC power, loss of the EDAS, ECCS actuation level, and single failure (RVV, RRV, or RRV-RVV train). The table also includes the corresponding results for the reactor trip signal time (seconds), ECCS valves opening time (seconds), and peak CNV pressure occurrence (seconds); as well as the peak CNV pressure and peak CNV temperature values. Each section has the transient responses of all seven spectrum DBEs consolidated on the same graph, for each of the following quantities.

a) Pressurizer pressure (psia), to represent the RPV pressure response
b) Containment lower plenum pressure (psia), to represent the CNV pressure response
c) RPV level (ft), to represent the RPV level above the TAF
d) CNV level (ft), to represent the CNV level above the vessel bottom
e) Peak containment wall temperature (^{o}F) to represent the CNV wall temperature response

The staff compared the consolidated spectrum transients for the above quantities with the sequence of events for the limiting DBE laid out in LOCA EM TR, Revision 3, Table 9-10 and found that the various temperature, pressure, and level trends and abrupt changes physically correlated well with the resulting RVV and RRV opening. The staff also conducted some sensitivity studies using the submitted NRELAP5 decks for the limiting containment DBE and the M&E release and found the peak CNV pressure and temperature results to be almost the same as the ones documented in LOCA EM TR, Revision 4 as well as EC-A013-7725 (ML23011A012). The spectrum tables in the six primary system M&E release sections summarize the input assumptions and the containment pressure and temperature results for 42 DBEs. Of the 42 DBEs analyzed, the LOCA caused by RCS (CVCS) discharge line break from the downcomer (DL-2) with the loss of normal AC power (but no loss of EDAS) and no single failure gave the peak CNV pressure of 937.3 psia and the maximum CNV wall temperature of 533.1 ^{o}F, bounding all remaining 41 primary DBEs analyzed. More detailed results and discussion were provided for the limiting CNV peak pressure and temperature scenario. The staff finds the formulation of the seven DBEs for each spectrum to be appropriate for meeting the regulatory guidance for the sensitivity of the peak containment pressure and temperature to loss of AC power, loss of EDAS, ECCS actuation level, and single failure.

The staff also reviewed the results of the two secondary system M&E release inside the CNV, i.e. MSLB and main feedwater line break (FWLB), as summarized in the LOCA EM TR, Revision 3, Sections 9.7.4 and 9.7.5. Both the secondary system analyses used the same input parameters that are used in the primary system release analysis. Both MSLB and FWLB M&E release scenarios were analyzed for a spectrum of nine DBEs each. Sections 9.7.4 and 9.7.5 provide the MSLB and FWLB spectrum tables with consolidated input sensitivity assumptions and the peak CNV pressure and temperature results for a total of 18 DBEs. Both sections presented the consolidated transients of the nine spectrum DBEs for each of the aforementioned quantities of interest. The staff finds the formulation of the nine DBEs for both the secondary release events to be appropriate for meeting the regulatory guidance for the sensitivity of the peak containment pressure and temperature to the loss of AC power, loss of EDAS, ECCS actuation level, and single failure. The staff finds that the addition of the main steam isolation valve (MSIV) and feedwater isolation valve (FWIV) single failures to both MSLB and FWLB M&E release scenarios in addition to the single failures of RVV, RRV, or RRV-RVV train, completed the required sensitivity study for the secondary system release events. The staff finds it appropriate that the limiting events (SLB-3 and FLB-3) for both the MSLB and FWLB are mainly driven by an immediate loss of ac and EDAS power with no single failure that leads to an immediate opening of both RVVs and actuates both RRVs at the very start of the transient. The staff notes that the immediate opening of both RVVs due to the EDAS power loss coupled with the AC power loss corresponds to a somewhat delayed RRV opening and, thus, a delayed peak pressure timing, leading to a higher peak containment pressure.

LOCA EM TR, Revision 3, Section 9.8 also presents representative LOCA EM methodology and results for an IORV to meet the SRP Section 15.6.1 event guidance for the PWR Pressurizer Pressure Relief Valve event, and NuScale DSRS Section 15.6.6 (Inadvertent Operation of the Emergency Core Cooling System (ECCS)) event guidance for the inadvertent operation of the ECCS. With respect to the peak containment pressure and temperature, the most limiting of the three valve opening IORV events (851.0 psia, 492.0 ^{o}F) as documented in LOCA EM TR, Revision 3, Table 9-23, is less limiting than the limiting MSLB (900.0 psia, 529.6 ^{o}F) and FWLB (885.9 psia, 526.0 ^{o}F) events discussed above. Furthermore, the IORV results are not needed by the DSRS sections guidance applicable to NuScale FSAR Chapter 6.

4.9.11 Decay Heat Removal System Availability (Containment Response Specific Perspective)

The DHRS is a passive safety-related system that relies on a closed-loop, two-phase natural circulation to remove heat from the RCS through the SG and rejects it to the reactor pool through the DHRS condenser. It provides additional capacity to remove decay heat for both primary and secondary systems release, and is more effective during the initial blowdown period before the ECCS actuation. The DHRS was not credited in the containment response analysis methodology (CRAM) for the NPM-160, but it is credited to NPM-20 CNV analyses for both primary and secondary systems M&E release events.

As shown in LOCA EM TR, Revision 3, Figure 3-1, one DHRS train is available for each of the two SGs, capable of independently removing 100 percent of the decay heat load to cool the reactor’s primary-side inventory. With the individual passive DHRS condenser submerged in the reactor pool, the DHRS piping connects to the main steam and feedwater lines specific to the associated SG. During normal operation, the DHRS condensers are maintained with sufficient water inventory, but are isolated by valves on the steam side of the SG, and do not remove heat. After an event initiates, the M&E release into the CNV through the break or opening causes CNV pressurization and RPV depressurization. Depending upon the evaluated scenario, the high containment pressure signal or loss of EDAS power causes the reactor trip, containment isolation, and DHRS actuation. Upon the actuation of DHRS, the feedwater/steam lines isolation valves close and the DHRS actuation valves open, creating a closed loop between the SG and DHRS condenser, which establishes long-term decay heat removal using the unaffected SG and the associated DHRS. In the event of a postulated MSLB or FWLB inside the CNV, the M&E release into the CNV also includes the inventory present in the DHRS train associated with the SG loop with the break.

With DHRS operation, the primary system begins a gradual cooldown and depressurization. The DHRS provides secondary-side reactor cooling when normal feedwater is not available. A staff audit of the submitted NRELAP5 CRAM decks showed an appropriately modeled passive safety-related DHRS and ECCS systems, and both DHRS trains are included. The model considered the water in the affected and unaffected helical coil SGs and feedwater lines, feedwater transfer to the affected/unaffected helical coil SG before the closure of the isolation valves in the feedwater lines and upon flooding with the DHRS heat exchanger inventory in the affected loop, and steam in the helical coil SG. Because the DHRS is a closed system, the total water mass in the unaffected SG loop remains constant in NRELAP5 calculations for the DHRS system operation.

Starting from the base case, the staff conducted some sensitivity studies using the submitted NRELAP5 decks for the NPM-20 for the limiting containment DBE and M&E release. The study reproduced the graphical results as well as the peak CNV pressure and temperature for the limiting 100 percent RCS discharge line break LOCA. The staff performed a sensitivity study of the limiting 100 percent RCS discharge line break LOCA as well as a 5 percent RCS discharge line break small break Loss of Coolant Accident (SBLOCA), to study the impact of DHRS performance degradation by employing a fouling factor, on the CNV pressure and temperature transients. The study concluded that a deterioration in the DHRS performance does lead to a rise in peak CNV pressure and temperature. Even though the uncertainty in the DHRS performance did not challenge the ability to meeting the containment safety design acceptance criteria for the limiting RCS 100 percent discharge line break LOCA, the same could not be concluded for SBLOCA. The staff considers that the demonstration results summarized in Section 9.7 of the LOCA EM TR, Revision 3, do not appropriately address the uncertainties in the DHRS performance, especially in the SBLOCA scenario and the long-term, where the role of DHRS is likely to be more significant. The LOCA EM TR, Revision 3 did not present any containment results for SBLOCA that would be sensitive to DHRS performance degradation and could lead to re-pressurization of the RPV. As discussed in Section 4.5.1.6, NuScale provided additional containment pressure and temperature results for the containment SBLOCA analyses that adequately addressed the staff concerns.

The staff needed to establish the applicability of the CRAM to the NPM-20 CNV design over the applicable range of break spectrum and DBA conditions. As opposed to those for the NPM-160, NPM-20 containment safety analyses are credited with DHRS operation that reduces the M&E release into the CNV during the DBA. The staff anticipated an increasing level of sensitivity of the long-term CNV thermal-hydraulic (T/H) response to the DHRS heat transfer and ECCS actuation delay for smaller breaks in the LOCA break spectrum. The staff saw the lack of SBLOCA analysis and results for the NPM-20 as a gap in identifying the limiting NPM-20 CNV

DBE. The staff needed the CNV T/H response results along with the sensitivity studies that demonstrate the impact of bounding uncertainties associated with DHRS performance and ECCS actuation, for the LOCA break size spectrum. However, the staff could not find any description and results for the NPM-20 containment response and sensitivity studies for the break size spectrum, either in the LOCA EM TR, Revision 3, or EC-A013-7725 (NPM-20 CNV Pressure and Temperature Response Analysis). LOCA LTR, Revision 3, Table 5-5, does provide a summary of the analyzed break sizes and Section 9.6.7 provides break spectrum results evaluating effect of DHRS heat transfer capacity using LOCA model input but does not present the corresponding NPM-20 conservative containment analysis results. The staff needed the information to ensure that there is no SBLOCA CNV response excluded from the CRAM demonstration that could potentially challenge the limiting CNV design-basis RCS discharge-line LBLOCA.

NuScale had not provided the sensitivity studies to demonstrate the impact of bounding DHRS performance and ECCS actuation uncertainties on the long-term conservative containment thermal-hydraulic response for the RCS discharge line break spectrum analyzed by using the NRELAP5 CNV biased decks. Besides, NuScale has not covered the DHRS capacity below **[[ ]]** or demonstrated a maximum DHRS uncertainty to bound the CRAM results. By letter dated XX, XX, XXXX (ML####), NuScale provided supplemental information and as discussed in Section 4.5.1.6 of this SER, the staff reviewed new simulation results for SBLOCA and pool heat up analyses submitted to address the concerns about the sensitivity of the containment response to the DHRS capacity, and found them appropriate.

4.10 Evaluation Model for Inadvertent Opening of Reactor Valves

4.10.1 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. In Section 9.8 of the LOCA EM TR, a spurious opening of an ECCS valve equipped with an IAB is evaluated. In addition to mechanical failures, for ECCS valves that do not include the IAB, a spurious ECCS actuation or loss of power can also result in a valve opening. Thus, for the NPM-20, the spurious actuations of one or more RVVs and a single RRV are considered separately.

4.10.2 Accident Scenario Identification Process

The EM review criterion in SRP 15.0.2 [21] 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-20, NuScale performed a PIRT for IORV in 2021. The IORV PIRT applied the same LOCA ranking approach identified in Section 4.4 of the LOCA EM TR where the FOM was CHFR and is applicable to the initial portion of the LOCA event (Phase 0). A listing of the newly identified high-ranked phenomena relative to the Table 4-4 LOCA PIRT is given in Table 4-5. The high-ranked phenomena, such as flow through baffle plate, pressurizer non-equilibrium flashing, CHF, and interfacial drag/shear/void distribution, etc., are discussed further in Section 4.6.2 of the LOCA EM TR.

The NRC staff reviewed the IORV PIRT and did not identify any low to medium ranked phenomena or processes that should have been ranked high. Therefore, the NRC staff finds that the applicant's IORV PIRT table for high-ranking phenomena is acceptable.

4.10.3 IORV EM Evaluation

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 EM 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 those of LOCA pipe breaks. Therefore, the NRC staff's review focused on 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 the MCHFR since IORV is the limiting event for CHFR. The NRC staff reviewed the core modeling that used the cross-flow option. The initial conditions and biasing for the steady state portion of the transient are the same as for a LOCA.

The original EM for accidental IORV (i.e., RSV, RVV, or RRV) was previously reviewed and approved in Revision 2 of the LOCA EM (ML20189A644) [1]. Due to design changes in the NuScale reactor for the NPM-20 and the removal of the IAB valves from the RVVs, spurious opening of both RVVs must now be evaluated and is the limiting IORV event. While the IORV is an AOO, due to its similarities with a LOCA, it is simulated using the LOCA EM. However, due to the new consideration of the inadvertent ECCS event, certain changes to the LOCA EM are required. In order to evaluate these changes, the NRC staff evaluated the changes to the EM itself in this section and evaluated the CHF model and correlations in Section 4.11 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 low RPV riser 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 low RPV riser 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 LTC. Due to NPM-20 designs changes and the removal of the IAB from the RVVs, spurious opening of both RVVs is the limiting IORV event that produce the most bounding MCHFR. The NRC staff noted that the failure of an IAB to block (which would cause an additional 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.

While the IORV is an AOO, due to its similarities with a LOCA, it is simulated using the LOCA EM. However, due to the new consideration of the inadvertent ECCS from the lack of IABs on the RVVs, certain changes to the LOCA EM were required. The NRC staff followed the guidance of SRP Section 15.0.2 to assess the changes to the approved IORV methodology.

During the review of the changes, the NRC staff did note an inconsistency in the documentation of the approved methodology. As defined in Section 6.11.3 of the TR, NuScale provided their modification to the Hench-Levy correlation. The original Hench-Levy correlation is defined in 10 CFR Part 50, Appendix K as one of the approved correlations which are acceptable for use in LOCA analysis. The Hench-Levy correlation was developed at a single pressure (1,000 psia) and therefore a pressure correction multiplier was used for different pressures [22]. NuScale described the modification to the Hench-Levy correlation as an adjustment to the pressure correction portion. They provided equation 6-108 in the LOCA EM TR to define the pressure correction and provided Figure 6-7 to describe the behavior of the pressure correction. However, the equation and figure are inconsistent with each other and describe dramatically different behavior.

The behavior given in LOCA EM TR, Figure 6-7, is consistent with the description provided in the TR. The figure shows a modification to only the pressure corrected portion of the Hench-Levy correlation. In other words, both the original Hench-Levy and NuScale modified Hency-Levy generate the same result at 1000 psia. As the pressure increases, the prediction of CHF decreases. This decrease is captured in the original pressure correction term always returning a value below 1.0 for pressures above 1000 psia. As defined in the figure, NuScale's modified Hench-Levy correlation had similar performance, as the pressure correction term always returned a value below 1.0 for pressures above 1000 psia. The major difference between the original Hench-Levy pressure correction and NuScale's modification of it, was that increase in pressure resulted in a much more rapid decrease in CHF performance in the original pressure correction term compared to NuScale's pressure correction. To justify the higher CHF values predicted by the NuScale modification, NuScale provided measured data that supported the trend. Based on this justification, the NRC staff approved the modification to the Hench-Levy correlation [1].

However, during the current review, the NRC staff noticed that the equation that defines the new pressure correction term, equation 6-108, does not match the behavior of Figure 6-7. In general, the pressure correction term defined in equation 6-108 will result in a greater than 10 percent increase in the predicted CHF value compared to that of the pressure correction term defined in Figure 6-7.

Examining the NRC staff's previous conclusions on the Extended Hench-Levy correlation, it was not clear if the NRC staff approved the form of the correlation given in equation 6-108 or the form of the equation given in Figure 6-7. To resolve this issue, NuScale provided (ML24326A345) plots of the validation from the Extended Hench Levy CHF correlation, where the correlation was consistent with equation 6-108. The validation compared the correlations' predictions in NRELAP5 to experimental data from the KATHY test facility. Based on a review of this validation data, the NRC staff has determined that the previous staff's conclusions related to the Extended Hench-Levy Correlation [1] were consistent with this validation data, and therefore the form of the correlation which should be considered the approved form is that given in Equation 6-108 of the TR.

The NRC staff further evaluated the applicant's proposed cross-flow core modeling and best-estimated NSPN-1 CHF correlation, which are discussed in the following subsections.

An EM is the calculation framework for evaluating the behavior of the reactor coolant system during a postulated accident or transient. It includes one or more computer programs and other information necessary to apply the calculation framework to a specific transient or accident, such as mathematical models used, assumptions included in the programs, a procedure for treating the program input and output information, specification of those portions of the analysis not included in the computer programs, values of parameters, and other information necessary to specify the calculation procedure. EMs are sometimes referred to as a licensing methodology. The staff reviewed the IORV EM following SRP Section 15.0.2, Sub-section III.3.b guidelines, which contain review criteria for the Evaluation Model. The IORV EM review results are summarized below for each of the review criteria.

4.10.3.1 Previously Reviewed and Accepted Codes and Models

The majority of the IORV EM has been previously reviewed and approved in Revision 2 of TR-0516-49422-P-A [1]. The IORV NRELAP5 model of this TR is developed based on Revision 2 of NRELAP5 IORV EM with changes introduced for NPM-20. Therefore, the NRC staff finds it is acceptable that the previously approved codes and methods have been identified and used by NuScale.

4.10.3.2 Physical Modeling

While the physical models in NRELAP5 have been previously reviewed and approved by the NRC staff, NuScale's latest revision incorporates two changes that require additional NRC review. First, for both LOCA and IORV events, NuScale has developed a new CHF correlation, NSPN-1, to ensure that CHF does not occur. Second, for IORV events, **[[**

]]. Both of these changes were implemented to make the NRELAP5 simulation more of a best estimate simulation. The currently approved CHF correlations used in NRELAP5 include the extended Hench-Levy correlation and the modified Griffith-Zuber pool boiling correlation. While these correlations have been approved, they under-predict the CHF (i.e., are conservative) compared to NSPN-1. **[[**

]]. Hence, both changes were made to reduce the conservatism in the prediction of CHF.

NuScale followed the standard procedure when incorporating a new CHF correlation into a code by re-validating the correlation using the new code and ensuring that the new code and correlation could appropriately predict the experimental data used for validation. Additionally, cross-flow is a well understood and well-studied phenomenon which has been predicted by codes similar to NRELAP5 for decades. However, there are complexities in this case since NuScale uses CHF as its figure of merit, instead of peak cladding temperature, which is typical for lumped channel codes such as NRELAP5 (while subchannel codes are typically used for CHF FOM due to the necessary physics being captured in those codes). While the prior staff review and approval of the LOCA EM also used the CHF figure of merit, the correlations that were approved were conservative, not best estimate. Therefore, this was a focus area of the staff's review and audit (ML24262A230) of this new correlation.

Additionally, while validation of the new CHF correlation was performed, the NRC staff also focused its review on ensuring the applicability of that validation to future predictions using the correlation and ensuring that the error determined in validation would be a reasonable estimate for the predictive capability of the correlation when performing safety analysis. The applicability focus area is addressed in Section 4.12.1.9 Similarity Criteria and Scaling Rationale.

The NRC staff reviewed NRELAP5's ability to predict CHF in its safety evaluation as documented in the approved topical for NRELAP5 [1]. Much of the details are found in the staff's audit report focused on the CHF performance of NRELAP5 [24]. In its review of Revision 2 of this LTR, the NRC determined that there was adequate justification that the CHF correlations used in NRELAP5 provided reasonable assurance given specific CHF limits of 1.13 for IORV events using the high-flow CHF correlation, 1.37 for IORV events using the low-flow CHF correlation, and 1.29 for LOCA events using the high-flow and low-flow correlations.

Both the previously approved high-flow and low-flow CHF correlations **[[**

]]

[[

]].

To clarify this issue, NuScale submitted additional information [Ref for A-.LOCA.LTR-35] (ML24326A329). **[[**

]].

Based on the conservative predictive behavior of NRELAP-5 when using the IORV methodology compared to an approved methodology when simulating events similar to the IORV scenario, and the above considerations, the NRC staff finds that the new physical modelling approach crediting cross-flow and the incorporation of the NSPN-1 CHF correlation into NRELAP5 is acceptable.

4.10.3.3 Range of Validity of the Field and Closure Equations

Because this evaluation focuses only on changes to the approved LOCA EM based on the inclusion of the NSPN-1 CHF correlation and cross flow modelling capability, the NRC staff has determined that the range of validity of the field and closure equations remain unchanged with the exception of those discussed in Section 4.10.3.2, "Physical Modeling," and in Section 4.11 Critical Heat Flux Evaluation of this SE. Therefore, the validity of the new application ranges of the field and closure equations is confirmed.

4.10.3.4 Simplifying and Averaging Assumptions

Similar to the range of validity, the NRC staff has determined that no further assessment of the simplifying and averaging assumptions is required with the exception of those discussed in Section 4.10.3.2 and in Section 4.11 Critical Heat Flux Evaluation of this SE because the new IORV scenario modeling did not introduce new simplification and averaging assumptions.

4.10.3.5 Level of Detail in the Model

Because this evaluation focuses only on changes to the approved LOCA EM based on the inclusion of the NSPN-1 CHF correlation, the NRC staff has determined that no further assessment of the level of detail in the model is required with the exception of those discussed in Section 4.10.3.2 and in Section 4.11 Critical Heat Flux Evaluation of this SE.

4.10.3.6 Equations and Derivations

Because this evaluation focuses only on changes to the approved LOCA EM based on the inclusion of the NSPN-1 CHF correlation, the NRC staff has determined that no further assessment of the equations and derivations is required with the exception of those discussed in Section 4.10.3.2 and in Section 4.11 Critical Heat Flux Evaluation of this SE.

4.10.3.7 Validity of the Closure Relationships

Because this evaluation focuses only on changes to the approved LOCA EM based on the inclusion of the NSPN-1 CHF correlation, the NRC staff has determined that no further assessment of the validity of the closure relationships is required with the exception of those discussed in Section 4.10.3.2 and in Section 4.11 Critical Heat Flux Evaluation of this SE.

4.10.3.8 Similarity Criteria and Scaling Rationale

This evaluation focuses only on changes to the approved LOCA EM based on the inclusion of the NSPN-1 CHF correlation. Therefore, the NRC staff focused on the experimental data used to validate the NSPN-1 CHF correlation and if the error in the prediction of those experimental data would be consistent with the expected error in the prediction of real-world application of the NSPN-1 correlation to reactor fuel. In this review, the NRC staff noted that **[[**

]] This difference was considered in the assessment of the physical modeling of the NSPN-1 correlation and the determination that the physical model is considered adequate as documented in Section 4.10.3.2 of this SER.

Additionally, NuScale developed the NSPN-1 CHF correlations at steady state conditions and then is applying it to transient analysis. This assumes that the transient can be analyzed assuming quasi-steady state at each time step. This assumption is well known in CHF correlations, but becomes uncertain for the NSPN-1 correlation's application for LOCA and IORV cases which occur during rapid depressurization events. The uncertainty is further evaluated in SER Section 4.11.3.5.34.11.3.5.3 Transient Prediction. Therefore, the NRC staff has determined that no further assessment of the similarity criteria and scaling rationale is required with the exception of those discussed in Section 4.10.3.2 Physical Modeling and in

Section 4.11 Critical Heat Flux Evaluation (specifically, Section 4.11.3.5.3 Transient Prediction) of this SER.

4.10.3.9 Scaling Analysis

Because this evaluation focuses only on changes to the approved LOCA EM based on the inclusion of the NSPN-1 CHF correlation, the NRC staff has determined that no further assessment of the scaling analysis is required with the exception of those discussed in Section 4.10.3.2 Physical Modeling and in Section 4.11 Critical Heat Flux Evaluation of this SER.

4.10.4 Inadvertent Opening of a Reactor Pressure Vessel Valve Evaluation Model Conclusion

Based on the evidence evaluated in Section 4.10.3 of this SER, as well as the conclusions drawn related to the IORV EM, the limited review due to changes to the IORV EM, the conclusions on the CHF model in Section 4.11.4 of this SER, the conservative comparisons between the IORV EM and an NRC approved methodology, and consideration of the acceptable, yet higher level of uncertainty inherent in this first of a kind approach, the NRC staff concludes that the IORV EM will conservatively predict CHFR during the IORV or LOCA transient scenarios.

4.11 Critical Heat Flux Evaluation

The TR-0516-49422, " Loss-of-Coolant Accident Evaluation Model," describes how the NSPN-1 CHF correlation was developed from experimental data, behaves over its application domain, and how it will be applied in the future. The NRC staff's technical evaluation is focused on determining if the model is acceptable for use in reactor safety license calculations (i.e., that the model can be trusted). To perform this review, the NRC staff chose to use the review framework described in NUREG/KM-0013, "Credibility Assessment Framework for Critical Boiling Transition Models" [15].

As noted previously in Section 4.10 of this SER, one of the NRC staff's primary areas of focus in the review of the NSPN-1 correlation was whether the NSPN-1 correlation, which utilizes the same basic form as a modern subchannel CHF correlation, could predict the occurrence of CHF in the LOCA EM given the detailed subchannel physics is missing from the NuScale LOCA models.

One purpose of TR-0516-49422 is to provide the bases for NRC approval to use the NSPN-1 correlations in NRELAP5, within their range of applicability, along with their associated correlation limits for the NuScale NPM-20 and safety analysis of the NPM-20 with NuFuel-HTP2TM fuel. The NRC staff used the critical boiling transition model assessment framework [15].

4.11.1 Experimental Data

The experimental data for the NSPN-1 CHF correlation are from a series of tests (K9000, K9100, K9200, K9300) from the Framatome CHF facility in Karlstein Germany. These tests have been previously used to demonstrate the credibility of other CHF correlations that have been reviewed and accepted by the U.S. NRC. Given that the NRC has previously reviewed the test facility, measurement of the data, the ability of the test bundle to reproduce local conditions in the reactor in previous approvals, and that during this review the NRC staff has not discovered any reason why the data should not be trusted, the NRC staff has concluded that the experimental data which support the NSPN-1 model are appropriate.

4.11.2 Model Generation

4.11.2.1 Appropriate Mathematical Model

4.11.2.1.1 Model Parameters

In equation 6-114 of the TR, NuScale provides the parameters used for the NSPN-1 CHF correlation. These parameters are very similar to those parameters used in the NSP1 CHF correlation (equation 4-2 of Reference 26) and NSP4 (equation 7-1 of Reference 26). This form can be understood as an algebraic summation of key terms multiplied by constants. Those key terms may be multiplied by each other and may be multiplied by themselves. Those key parameters include:

- A ratio of the pressure divided by the critical pressure of 3204 psia.
- The local mass flux
- The local equilibrium quality
- **[[**
- **]]**.

These key parameters are identical to those used in the NSP4 CHF correlation and very similar to those used in the NSP1 CHF correlation. In general, using the same parameters in predicting CHF would seem to add to the credibility of the NSPN-1 CHF correlation. However, the NRC noted a very large difference in the NSP1, NSP4, and NPSN-1 CHF correlations. **[[**

]]. This concern is addressed in Section 4.10.3.2, of this SE. Because the model contains the parameters generally used to predict CHF, the NRC staff has concluded that this criterion has been met.

4.11.2.1.2 Model Form

In equation 6-114 of the LOCA EM TR, Revision 3, NuScale provides the model form used for the NSPN-1 CHF correlation. This form is very similar to the model form of the NSP1 CHF correlation (equation 4-2 of Reference 26) and the NSP4 CHF correlation (equation 7-1 of Reference 26). This form can be understood as an algebraic summation of key terms multiplied by constants. Those key terms may be multiplied by each other and may be multiplied by themselves. This form is very similar to the form used for the NSP4 CHF correlation. Because NuScale is using a form similar to previously approved CHF correlations, the NRC staff concludes that this criterion has been met.

4.11.2.2 Model Coefficient Generation

4.11.2.2.1 Training Data

In Section 6.11.5.1 of the LOCA EM TR, Revision 3, NuScale described the training process and identified the training data. Because NuScale has identified the data used to train the model, the NRC staff concludes that this goal has been met.

4.11.2.2.2 Coefficient Generation

To train the NSPN-1 CHF correlation, NuScale partitioned all of the data into three different sets, each of which was approximately the same size. They then trained the model on two of the three sets, reserving one of the three sets for pure validation purposes (i.e., 3 folds). This process was repeated three times in total, resulting in three different sets of coefficients for the NSPN-1 CHF correlation, each set corresponding to a set of coefficients developed from 2/3 of the total data set. **[[**
]]. By providing this explanation, the NRC staff concludes that this goal has been met.

The NRC staff notes that this, or any method, of coefficient generation should not be considered “accepted” by the NRC staff. In general, there is no one way to generate the coefficients of a data-driven model, and any method chosen has its advantages and disadvantages. Hence, this goal is not focused on demonstrating the acceptability of the method used to generate the coefficients, but only focused on describing such a method, which NuScale has done. The acceptability of the method will be demonstrated in the correlation’s prediction of the validation data.

4.11.2.2.3 BWR Specific Parameters

The NRC staff has concluded that this goal does not directly apply to the NSPN-1 correlation given that it is not being used in a BWR methodology. However, there is a related area the staff focused on regarding the fact that NSPN-1 cannot account for LOCA flow behavior or local radial power variations that is addressed in Section 4.10.3.2 of this SE.

4.11.3 Model Validation

4.11.3.1 Validation Error

The validation error used by NuScale is very similar to the validation error commonly used for CHF correlations. However, traditional CHF correlations are functions of local parameters (e.g., subchannel mass flux, subchannel quality) while the NPSN-1 correlation is a function of global parameters (e.g., total bundle mass flux, total bundle quality). The NRC staff focused its review on whether the error quantified in validation was applicable to the real-world system because the global behavior of the experimental test section will not be similar to the behavior of the real-world system. This issue is addressed in Section 4.10.3.2 of this SER. Because NuScale is using the validation error commonly used for CHF models, the NRC staff finds that this goal has been satisfied.

4.11.3.2 Data Distribution

4.11.3.2.1 Validation Data

NuScale identified the data used to validate the NSPN-1 as the same data which was used to train the model. These data were provided to the NRC (ML23214A211) [17]. Because NuScale has provided this information, the NRC staff concludes that this goal has been met.

4.11.3.2.2 Application Domain

In Table 6-8 of the LOCA EM TR, NuScale identified the application domain of the NSPN-1 correlation. The application domain is very large compared to other mixing vane application domains. For example, ORFEO-GAIA’s domain spans only 1,000 psia and limits the upper quality to 80 percent [18]. While WNG-1 also only spans 1000 psia and limits the upper quality to less than 43 percent [19]. In contrast, the NSPN-1 correlation spans a pressure range of almost 2,000 psia and a quality limits the upper quality at 100 percent. The mass flux range is much smaller than either ORFEO-GAIA and WNG-1, both of which have ranges approximately from 0.5-3.5 (Mlbm/hr-ft2), while the NuScale range is **[[ ]]**. In general, the application domain of the NPSN-1 CHF correlation is very different from the most recent CHF correlations that the NRC has reviewed and approved. It has a much larger span in pressure and quality and while it has a much smaller span in mass flux, it exists almost entirely in mass flux ranges that are well below the recently approved CHF correlations. The difference in application domains underscores the uniqueness of the NSPN-1 CHF correlation as discussed in Sections 4.10.3.2, 4.10.3.8, and 4.11.2.1.1 of this SER. Because NuScale provided this information, the NRC staff concludes that this goal has been met.

4.11.3.2.3 Expected Domain

The expected domain of the NPSN-1 CHF correlation has not been further defined from the application domain. Therefore, the entire application will be used as the expected domain as defined in LOCA EM TR, Table 6-8. Additionally, the details of the limiting IORV case could also be used as to define locations in the domain of special interest. Because the expected domain can be understood as the application domain, the NRC staff concludes that this goal has been met.

4.11.3.2.4 Data Density

In Section 6.11.5.5 of the LOCA EM TR, NuScale provided plots of the data over the application domain. Because these plots show data densities similar to previously approved CHF models, the NRC staff concludes that this goal has been met.

4.11.3.2.5 Sparse Regions

In section 6.11.5.5 of the LOCA EM TR, NuScale provided plots of the data over the application domain. Because these plots show data do not indicate regions with sparse data, the NRC staff concludes that this goal has been met.

4.11.3.2.6 Restricted Domain

It has already been determined that NuScale has appropriately addressed the restriction of other approved CHF correlations to their application domains, including the NPS4 correlation. Because NuScale has already demonstrated that it does restrict the correlation to its application domain, the NRC staff concludes that this goal has been previously met.

4.11.3.3 Consistent Model Error

4.11.3.3.1 Poolability

In Section 6.11.5.5 of the LOCA EM TR, NuScale provided an analysis of various subsets of the validation data and the maximum limit calculated from each subset. In that analysis, NuScale determined that considering subsets of pressure resulted in a limit that was slightly higher than the limit generated without considering subsets. Such small differences in limiting values are expected for such a large data set and are not a clear indication of non-poolable data. Additionally, NuScale has decided to use the limiting value from this analysis. Because NuScale's analysis demonstrates that the various subsets are similarly behaved, the NRC staff concludes that this goal has been met.

4.11.3.3.2 Non-Conservative Subregions

In Section 6.11.5.5 of the LOCA EM TR, NuScale provided an analysis of various subsets of the validation data and the maximum limit calculated from each subset. In that analysis, NuScale decided to use the limiting value from the analysis. This value is not a clear indication of a non-conservative subregion (i.e., the apparent difference could be due to random effects). However, by using that limit NuScale ensured that even if there is a small non-conservatism, it is accounted for in their safety limit. Because NuScale used a safety limit that bounds the regions in the application domain, and because upon further review, the NRC staff was also able to discover no evidence of a non-conversative subregion, the NRC staff concludes that this goal has been met.

4.11.3.3.3 Model Trends

In Section 6.11.5.5 of the LOCA EM TR, NuScale has provided the trends of the NSPN-1 Correlation's error versus each input. Because these plots show no trend with any input, the NRC staff concludes that this goal has been met.

4.11.3.4 Quantified Model Error

4.11.3.4.1 Error Data Base

In Section 6.11.5.5 of the LOCA EM TR, NuScale discussed the method for determining the error data base of the NPSN-1 CHF model. Instead of setting aside a certain percentage of data for validation purposes, NuScale chose to use a 3 fold technique of separating all of the data into three groups, training on two of the groups, and validating on the third group. This technique results in three separate 95/95 limits, each of which were within 0.01 of each other, and each of which was less than the final limit chosen of **[[ ]]**. The NRC staff has previously accepted such methods in lieu of reserving a portion of the data for validation.

Because NuScale has used a three-fold technique to determine the 95/95 limit, the NRC staff concludes that this goal has been met.

4.11.3.4.2 Statistical Method

In Section 6.11.5.5 of the LOCA EM TR, NuScale describes the parametric and non-parametric methods used to calculate the 95/95 limit. Because these methods are consistent with those methods previously used, the NRC staff concludes that this goal has been met.

4.11.3.4.3 Appropriate Bias for Model Uncertainty

In Section 6.11.5.5 of the LOCA EM TR, NuScale describes the method for choosing the 95/95 limit of 1.15. Consistent with previous approvals of CHF correlations, this limit is reflected in L/C #8 in Section 5 of this SER. Because the value chosen was the most limiting based on the sub-region analysis, the NRC staff concludes that this goal has been met.

4.11.3.5 Model Implementation

4.11.3.5.1 Same Computer Code

This goal has been reflected in L/C #4 stated in SER Section 5.0 of this SE.

4.11.3.5.2 Same Methodology

This goal has been reflected in L/C #4 stated in SER Section 5.0 of this SE.

4.11.3.5.3 Transient Prediction

Steady-state CHF testing, such as that used to develop NSPN-1, must be justified to be applicable to transient conditions. For many scenarios, this justification has been demonstrated through historical experiments as well experiments performed during a correlation's development. However, these experiments are typically limited to flow and power transients. As such, there have been very few analyses which have considered the application of steady state CHF data to depressurization transients. In the original VIPRE SE, this issue is briefly discussed. The reviewers concluded that the application of steady-state CHF data for depressurization transients, especially rapid depressurization transients (i.e., with depressurization rates of 100 to 350 psi per second – which is consistent with NuScale's depressurization rate) has shown "mixed results" where some correlations predicting the timing of CHF rather well while others fail to predict that CHF occurred as early as it did or that CHF occurred at all during the experiment. However, the reviewers do not resolve the issue as VIPRE was never intended to be applied to such rapid depressurization scenarios [25].

However, the same concern raised in the original VIPRE analysis applies to NSPN-1, and therefore the staff focused its review on evaluating whether the steady state CHF data measured at the KATHY facility accurately or conservatively reflects the CHF performance during a rapid depressurization.

To clarify this issue, NuScale submitted additional information (ML24326A329). **[[**

]], the NRC staff concludes that there is reasonable assurance that NRELAP5 model will result in an accurate or conservative prediction when applied in transient analysis. The NRC staff concludes that this goal has been met.

4.11.4 Critical Heat Flux Conclusion

Based on evidence provided in Section 4.11.1 of this SER, the NRC staff concludes that the experimental data supporting the NPSN-1 CHF correlation are appropriate. Based on the evidence in Section 4.11.2 of this SER, the NRC staff concludes that the NPSN-1 CHF correlation was generated in a logical fashion. Based on the evidence in Section 4.11.3 of this SER, the NRC staff concludes that the NPSN-1 CHF correlation has sufficient validation as demonstrated through appropriate quantification of its error. Therefore, based on the cumulative evidence, the NRC staff concludes that the NPSN-1 CHF correlation can be applied in IORV and LOCA analyses subject to the conditions and limitations listed in Section 5 of this SER.

5 LIMITATIONS AND CONDITIONS

This section provides a summary of the limitations and conditions based on the NRC staff's technical evaluation of the NuScale LOCA EM TR-0516-49422, "Loss-of-Coolant Accident Evaluation Methodology," Revision 3.

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 and the methodology to perform containment response analyses, can be used for the NuScale NPM design, subject to the following 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 in which: (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.

3. Types of Analyses Approved for the 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, or the containment pressure and temperature response methodology. 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, control rod ejection accidents, radiological consequences, non-LOCA events (other than an IORV and MSL/FW breaks inside containment), or evaluation of the long-term cooling (LTC) phase.

4. Limitations on NRELAP5 and NPM Model Approval.

   Unless changes are made pursuant to a change process specifically approved by the NRC staff for changes to NRELAP5 and the NPM model, use of NRELAP5 is limited to Version 1.7 in conjunction with NPM-20 basemodel Revision 5 (or a later NPM-20 basemodel revision if the revision is demonstrated to produce either essentially the same or conservative results, or if the revision is due to a change made to SSC physical/process input parameters only made via established change control processes (such as 10 CFR 50.59)).

   NRELAP5 Version1.7 and NPM model Revision 5 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 Version 1.7 and NPM model Revision 5, as described in this TR, are referenced in other EMs, those applications for use of NRELAP5 Version 1.7 and NPM model Revision 5 within another EM require separate approvals to ensure the models and assumptions are defined appropriately for the analyzed FOMs. Use of the NRELAP5 Version 1.7 and NPM model Revision 5 are therefore, not approved for standalone evaluation of the following events and must have separate EM approvals: thermal hydraulic instabilities in the secondary or primary system, control rod ejection accidents, non-LOCA events (other than IORV and MSL/FW breaks inside containment), radiological consequences and evaluation of the long-term cooling (LTC) phase.

5. 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 describe in its submittal the following analytical assumptions considered for the evaluation of design basis events described in this TR, including LOCA events, IORV events and containment pressure and temperature response events, and receive a separate approval for those assumptions: 1) single failures, 2) electrical power assumptions (AC/DC), or 3) operator actions relied on in the analysis (and therefore necessary to mitigate design basis events) within the 72 hours following event initiation to improve the results relative to the applicable figures of merit for a particular set of initial conditions, including actions taken to prevent accidents and transients from progressing to more severe events.

6. **[[** **]]**.

   [[

   ]]. Sections 5.5.2 and 5.5.3.2 of the SER for Revision 2 of the LOCA EM TR describes the basis for this limitation.

7. High- Flow CHF Correlation Range.

   Application of the high-flow **[[** **]]** CHF correlation is limited to its range of applicability as identified in SER Table 5.2-1. SER Sections 5.5.5.2.2.2 and 5.5.5.1 of the SER on Revision 2 of the LOCA EM TR (ML20181A270 (nonproprietary) and ML20181A269 (proprietary)) describe the basis for this limitation.

8. CHFR Minimum Value.

   The NSPN-1 CHFR limit of 1.20 is approved for analyses **[[** **]]**. The high-flow **[[** **]]** and low-flow **[[** **]]** CHFR limit of 1.29 is approved for analyses of LOCA or IORV events **[[** **]]** for high-flow and low-flow conditions. SER Sections 5.5.4.3 and 5.5.2.2.2 of SER Revision 2 for the LOCA EM TR describe the basis for this condition. The NSPN-1 CHF correlation is appro]]ved with a 95/95 critical heat flux ratio design limit of **[[** **]]** and is approved over the range defined in TR Table 6-8.

9. Regulatory Compliance with 10 CFR 50.46(a)(1) for LOCA Break Spectrum Excluded Locations.

   An applicant or licensee referencing this report will be required to address regulatory compliance with 10 CFR 50.46(a)(1), and thereby also satisfy General Design Criterion 35, which could include seeking an exemption from the required LOCA break locations, for those locations not analyzed within the break spectrum described in Sections 1.2 and 5.4 of the LOCA EM TR, including those discussed in Section 4.5.4.1 of this SER.

10. Limitation on NPM design LOCA EM is approved for

    Use of the LOCA EM TR, Revision 3, is limited to evaluations of the NPM-20 design. An applicant or licensee seeking approval to use the LOCA EM TR, Revision 3, for a design other than the NPM-20, such as the NPM-160, or another future NPM design, is required to demonstrate the applicability of the LOCA EM to the specific NPM design. The use of this methodology for a specific NPM design other than the NPM-20 requires NRC staff review and approval of the applicant's or licensee's determination of applicability.

11. ECCS RPV Riser Level Instrument Setpoint Modeling

    An applicant or licensee referencing this report for application to a specific design will be required to explain and justify the method for their approach to model ECCS actuation from the level sensor responses corresponding to the analytical limits (or range of analytical limits) specified for the setpoints, and to receive NRC staff review and approval of the methodology for such modeling, unless the method is followed that is described in section 5.2 of the LOCA EM TR that models the riser level instrument setpoint based on mixture level in the riser, using one of the approaches described in detail in section 5.2 (not including the application – specific alternate approach).

12. Link between LOCA EM and Non-LOCA EM

    The staff's approval is limited to the use of the LOCA EM with TR-0516-49416-P-A, Revision 4, "Non-Loss-of-Coolant Accident Analysis Methodology," and any future changes or revisions to TR-0516-49416-P-A, Revision 4, "Non-Loss-of-Coolant Accident Analysis Methodology," must be assessed by the applicant for their potential impact on the LOCA EM. Any subsequent changes to the LOCA EM require NRC approval.

13. LOCA EM Quality Assurance Requirements

    The entirety of the LOCA EM TR is considered to consist of the LOCA EM, IORV EM, and CRAM. As such, this report is considered to be subject to design verification, and an applicant or licensee referencing this report must ensure the engineering documents underlying the information and conclusions contained in this report were developed and are controlled consistent with 10 CFR Part 50, Appendix B, in accordance with Part II, "Quality Assurance Program Description Details," Section 2.3.1, "Design Verification," of TR MN-122626.

6 CONCLUSION

This SER documents the results of the technical evaluation of TR-0516-49422, "Loss-of-Coolant Accident Evaluation Model," Revision 3 applicable to the NPM-20 design. Subject to the closure of the open and confirmatory items noted in the above evaluation, 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 5.0 above. The NRC staff finds the NuScale LOCA EM appropriate for determining CHF and CLL conservatively, 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.

Subject to the closure of the open and confirmatory items noted in the above evaluation, the NRC staff finds the NuScale LOCA EM appropriate for analyzing the M&E release into the CNV for the spectrum of primary and secondary DBEs and determining the conservative CNV pressure and temperature response. Based on the demonstration of the containment response analysis methodology (CRAM) for NPM-20, the staff concludes that the methodology has appropriately incorporated the PIRT as it applies to modeling the NPM-20 containment response. NuScale has established the NRELAP5 applicability to the CRAM for modeling relevant phenomena including condensation heat transfer, non-condensable gas effect, decay heat, choked flow, DHRS/ECCS impact, and CNV heat removal to the reactor pool. The NRC staff finds the proposed CRAM methodology acceptable.

Subject to the closure of the open and confirmatory items noted in the above evaluation, the NRC staff concludes that the experimental data supporting the NPSN-1 CHF correlation is appropriate and the correlation was generated appropriately with sufficient validation. Therefore, the NPSN-1 CHF correlation can be applied in Phase 0 LOCA and IORV evaluation using NRELAP5 code subject to the conditions and limitations listed in Section 5. The NRC staff finds that the NRELAP5 IORV EM developed based on NPSN-1 CHF correlation and the cross-flow model conservatively predicts the minimum CHFR during the IORV scenario. Therefore, the NRELAP5 computer code and the NPM model are determined to be acceptable to evaluate the MCHFR for IORV and LOCA events.

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