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

TOPICAL REPORT TR-0516-49416, REVISION 4

"NON-LOSS-OF-COOLANT ACCIDENT ANALYSIS METHODOLGY"

NUSCALE POWER, LLC

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1 INTRODUCTION

1.1 Background

On January 5, 2023, NuScale Power, LLC (NuScale), hereinafter referred to as "the applicant," submitted Topical Report (TR) TR-0516-49416-P, Revision 4, "Non-Loss-of- Coolant-Accident Analysis Methodology," to the U.S. Nuclear Regulatory Commission (NRC) for review and approval in support of the NuScale US460 Standard Design Approval (SDA). (ADAMS Accession Number (ML23005A305). The NRC accepted the TR for review on 7/31/2023 (ML23206A107). The completeness determination was updated to include a resource estimate on 9/22/2023 (ML23265A154). The completeness determination was revised on 6/27/2024 (ML24178A422) to update the schedule and resource estimates, to account for issuing requests for additional information (RAIs). The applicant submitted Revision 5 of the TR on xxxxx (MLxxxxx), which addressed NRC staff RAIs and issues developed during the staff's regulatory audit (ML24262A257). The conclusions in the staff's SER are based on markups to Rev. 4 of the TR provided by NuScale in advance of Rev. 5 of the TR.

The TR seeks approval for the application of the proposed evaluation model (EM) for the analysis of system transient response to non-loss-of-coolant accident (non-LOCA) initiating events for a NuScale Power Module (NPM). The non-LOCA EM is limited to a short time frame following a design-basis non-LOCA event (e.g., a steam line break) in which the coolant mixture level remains above the top of the riser and primary side natural circulation is maintained

> The EM uses a modified version of the RELAP5 computer code, referred to as NRELAP5, and follows a graded approach outlined in Regulatory Guide (RG) 1.203, "Transient and Accident Analysis Methods," dated December 2005 (ML053500170). The TR addresses the high-ranked phenomena identified by the non-LOCA phenomena identification and ranking table (PIRT). TR-0516-49422-P-A, "Loss-of-Coolant Accident Evaluation Model" (LOCA TR), Revision X (MLXXX) (Reference 8) addresses the high-ranked phenomena that are related to the performance of the NRELAP5 computer code but are not addressed in the non-LOCA TR.

The applicant requested approval of the non-LOCA EM to use for analyses of NPM design basis non-LOCA events that require system analysis, including anticipated operational occurrences (AOOs), infrequent events (IEs), and postulated accidents (PAs). The applicant stated that the representative analysis results presented in Section 8 of the TR, "Representative Calculations," are illustrative of the non-LOCA methodology and are not necessarily representative of the applicant's final design. Therefore, the applicant is not seeking approval of the calculational results described in Section 8 of the non-LOCA TR.

The scope of the TR includes the applicability and acceptability of the proposed methodology to evaluate the primary and secondary system pressure acceptance criteria found in Section 15.0, "Introduction – Transient and Accident Analyses," of the NuScale Design Specific Review Standard (DSRS), dated June 2016 (ML15355A295). The TR also discusses the interfaces to the other analyses that assess the acceptance criteria not evaluated by the non-LOCA EM.

1.2 Scope of the Submittal

The TR includes information on the following topics:

- The EM roadmap and relevant regulatory requirements.

- Key NPM design characteristics.
- Credited plant control, protection, and instrumentation.
- Non-LOCA initiating events, including their classification.
- The applicable acceptance criteria for non-LOCA events.
- Interfaces with other analyses (i.e., nuclear, subchannel, and radiological analyses).
- A summary of the PIRT for non-LOCA transient analysis.
- Discussion of NRELAP5 applicability to the NPM.
- Assessment of NRELAP5 results against recent data from NuScale Integral Test Facility (NIST), NIST-1 and NIST-2, and other experiments.
- A description of the NRELAP5 plant model.
- Selection of input parameter and initial conditions.
- Identification of the limiting single failure and limiting loss of power scenarios.
- Methods for sensitivity studies and results.
- Representative results of NRELAP5 calculations.
- A brief description of the quality assurance (QA) procedures.

The licensing topical report (TR) cites several General Design Criteria (GDC) in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, and the guidance in RG 1.203, several DSRS sections, and several NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” (Standard Review Plan (SRP)) (ML070660036) sections as relevant to non-LOCA transient system analysis EM development and application.

The TR also presents a summary of the PIRT process and a list of highly ranked phenomena applicable to non-LOCA events relevant to the NPM-20 reactor design. The PIRT follows the short-term non-LOCA event progression, which is divided into three phases: pre-trip transient, post-trip transient, and initiating event is mitigated and stable cooling is established.

The applicant defined figures of merit (FOMs) for each phase that reflect non-LOCA acceptance criteria and important factors relative to the NPM-20 design. The applicant assigned each identified phenomenon an importance ranking according to its influence on an FOM (i.e., high (significant influence), medium (moderate influence), low (small influence), and inactive (not present or negligible)). The summary of the highly ranked phenomena provides rationale for the ranking of each phenomenon.

Furthermore, the TR discusses the applicability of NRELAP5 for non-LOCA analyses, including experimental assessment bases of the NRELAP5 models based on separate effects test (SET) and integral effects test (IET) data, details of the NRELAP5 model nodalization for NPM, and required sensitivity studies. Representative analyses are shown to illustrate how the EMs are used to analyze the non-LOCA transients.

The TR is focused on the short-term non-LOCA transient progression, defined as the time frame during which the mixture level remains above the top of the riser and primary side natural circulation is maintained. The applicant's long-term cooling analysis methodology, including events that transition from decay heat removal system (DHRS) cooling to emergency core cooling system (ECCS) heat removal, is addressed in TR-14587-P-A, Revision X, "Extended Passive Cooling and Reactivity Control Methodology," (MLXXXX) (Reference 7).

The TR does not address the evaluation of specified acceptable fuel design limits (SAFDLs), which are evaluated in "Subchannel Analysis Methodology," TR-0915-17564-NP-A, Revision 2, NuScale Power, dated February 2019 (Subchannel TR) (ML19067A256) and its supplemental Topical Report TR-108601-P-A, "Statistical Subchannel Analysis Methodology" Revision 4, (ML24106A160). Furthermore, the TR does not include the evaluation of the accident radiological source term and dose since these aspects are covered in "Accident Source Term Methodology," TR-0915-17565-NP-A, Revision 4, NuScale Power, dated February 2020 (ML20057G132). However, the non-LOCA EM does provide input to the subchannel and statistical subchannel analysis models for evaluation of SAFDLs. It also provides input to the accident source term model for evaluation of accident radiological source term and dose.

Other events that are covered by separate methodologies and are therefore excluded from the scope of the TR include control rod ejection accidents, inadvertent opening of an ECCS valve, loss of coolant accidents, thermal-hydraulic instabilities, and analysis of peak containment pressure and temperature response.

There are holes in the riser of the NuScale reactor designs. The purpose of these holes is to mitigate potential boron dilution in the downcomer during extended periods of DHRS operation in which the riser may uncover and steam condenses on the steam generator (SG) tubes. The riser holes were not designed to influence the short term DHRS cooldown during the non-LOCA phase. The TR states that the plant design overview description in TR Section 3, PIRT and test assessment discussion in TR Section 5, input and biasing discussion in TR Section 7, and NRELAP5 example calculations in TR Section 8 do not incorporate the small holes in the riser. However, the applicant's NRELAP5 models for non-LOCA event analyses include the riser holes and the model description in Section 6 of the TR reflects the riser hole flow by using a junction to account for the effects of riser hole flow on the total primary flow and temperature distribution. The impact of riser holes on the PIRT and test assessment are discussed in Section 3.5 of this safety evaluation report (SER), and the NRC staff's review of the sensitivity studies and example calculations are in Sections 3.7 and 3.8 of this SER, respectively. Broadly, neither the event-specific sensitivity studies nor the example calculations are representative of the NPM-20 design, including the presence of riser holes. Although the riser holes were not discussed in these sections of the TR, NRELAP5 models implementing the TR methodology must include riser holes. The applicant further states that during the short-term time frame considered in the non-LOCA EM, the mixture level remains above the top of the riser and primary side natural circulation is maintained. During its audit review (ML24262A257), the staff confirmed that the applicant's evaluation demonstrated that the riser holes remain covered by water during the non-LOCA events and the holes have an insignificant effect on steady-state parameters as well as short-term non-LOCA transient progressions and FOMs. The staff's sensitivity analyses support this conclusion.

1.3 Scope of the Review

This review focused on the acceptability and applicability of the methodology described in the TR to non-LOCA event analysis for the events listed in the TR, Table 4-1, "Design basis events for which the non-LOCA system transient analysis is performed, event category, and event classification," of the TR.

The review considered the application of the graded approach to the EM development and assessment process (EMDAP) described in RG 1.203. The NRC staff evaluated the EM against the NRC's regulatory requirements and guidance listed in Section 2, "Regulatory Basis for Non-LOCA EM Review," of this SER. The NRC staff's review covered all topics in the bulleted list in Section 1.2, "Scope of the Submittal," of this SER except for the NPM design; the event-specific limiting single failures, electric power assumptions, and the necessity for operator actions to mitigate specific non-LOCA events; results of representative calculations; and QA. These topics are evaluated as part of the review of a design-specific application of the methodology, such as the review performed for the US460 NuScale SDA.

This SER describes the NRC staff's review of the methodology as documented in the TR and its related documents. Section 2, "Regulatory Basis for Non-LOCA EM Review," discusses the regulatory criteria used to guide the review. Section 3, "Technical Evaluation," contains the NRC staff's technical evaluation. Section 4, "Limitations and Conditions," lists the applicable conditions and limitations, and Section 5, "Conclusion," presents the conclusions of the NRC staff's review.

2 REGULATORY BASIS FOR NON-LOCA EM REVIEW

2.1 Regulatory Requirements

Regulations under 10 CFR 52.47, "Contents of applications; technical information," 10 CFR 52.79, "Contents of applications; technical information in final safety analysis report," and 10 CFR 52.137, "Contents of applications; technical information," require an applicant to provide a final safety analysis report (FSAR) to the NRC that, in part, presents a safety analysis of the structure, system, and components (SSCs) provided for the prevention or mitigation of potential accidents of the facility as a whole (in the case of 10 CFR 52.47 and 10 CFR 52.79) or a major portion thereof in 10 CFR 52.137. The applicant presents accident analysis methodologies for the non-LOCA events in this TR to perform the required safety analyses. The results of the transient and accident analyses form a partial basis for compliance with the following GDC applicable to non-LOCA events:

- GDC 5, "Sharing of structures, systems and components," as it relates to the requirement that any sharing among nuclear power units of SSCs important to safety will not significantly impair their safety function.

- GDC 10, "Reactor design," as it relates to the reactor coolant system (RCS) being designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations, including AOOs.

- GDC 13, “Instrumentation and control,” as it relates to instrumentation and controls provided to monitor variables over anticipated ranges for normal operations, for AOOs, and for accident conditions.

- GDC 15, “Reactor coolant system design,” as it relates to the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the pressure boundary will not be breached during normal operations, including AOOs.

- GDC 17, “Electric power systems,” as it relates to the requirement that an onsite and offsite electric power system be provided to permit the functioning of SSCs important to safety. The safety function for each system (assuming the other system is not working) shall be to provide sufficient capacity and capability to ensure that the acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary (RCPB) are not exceeded during an AOO and that core cooling, containment integrity, and other vital functions are maintained in the event of an accident. The applicant has requested an exemption from GDC 17 in the NuScale US460 SDA.

- GDC 20, “Protection system functions,” as it relates to the reactor protection system being designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that the plant does not exceed SAFDLs during any condition of normal operation, including AOOs.

- GDC 25, “Protection system requirements for reactivity control malfunctions,” as it relates to the requirement that the reactor protection system be designed to ensure that SAFDLs are not exceeded for any single malfunction of the reactivity control system, such as accidental withdrawal of control rods.

- GDC 26, “Reactivity control system redundancy and capability,” as it relates to the reliable control of reactivity changes to ensure that SAFDLs are not exceeded even during AOOs. This is accomplished by ensuring that the applicant has allowed an appropriate margin for malfunctions such as stuck rods.

- GDC 27, “Combined reactivity control systems capability,” as it relates to controlling the rate of reactivity changes to ensure that, under PA conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained.

- GDC 28, “Reactivity limits,” as it relates to limits on the potential amount and rate of reactivity increase to ensure that the effects of postulated reactivity accidents can neither: (1) result in damage to the RCPB greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other reactor pressure vessel (RPV) internals to impair significantly the capability to cool the core.

- GDC 31, “Fracture prevention of reactor coolant pressure boundary,” as it relates to the RCS being designed with sufficient margin to ensure that the boundary behaves in a nonbrittle manner, and that the probability of propagating fracture is minimized.

- GDC 34, “Residual heat removal,” as it relates to the capability to transfer decay heat and other residual heat from the reactor so that fuel and pressure boundary design limits are not exceeded. The applicant has requested an exemption from GDC 34 in the NuScale US460 SDA and has proposed NuScale-specific Principal Design Criterion 34.

2.2 Regulatory Guidance

The SRP provides guidance for reviewing safety analysis reports, and the NuScale DSRS provides guidance for areas where existing SRP sections do not address the unique features of the NuScale design. DSRS Section 15.0, "Introduction – Transient and Accident Analyses," provides guidance for the review of transient and accident analyses, including event categorization and acceptance criteria as well as a discussion of the safety analysis EMs.

The acceptance criteria for AOOs, as listed in DSRS Section 15.0, are:

- Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

- Fuel cladding integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit.

- An AOO should not generate a PA without other faults occurring independently or result in a consequential loss of function of the RCS or reactor containment barriers.

The acceptance criteria for IEs and PAs, as listed in DSRS Section 15.0, are:

- Pressure in the RCS and main steam system should be maintained below acceptable design limits, considering potential brittle as well as ductile failures.

- Fuel cladding integrity will be maintained if the minimum DNBR remains above the 95/95 DNBR limit. If the minimum DNBR does not meet these limits, then the fuel is assumed to have failed.

- The release of radioactive material shall not result in offsite doses in excess of the guidelines of 10 CFR 52.47(a)(2)(iv) and 10 CFR Part 100. The acceptance criterion for IEs is a small fraction (10 percent) of 10 CFR 52.47(a)(2)(iv) and 10 CFR Part 100.

- A PA, including an IE, shall not, by itself, cause a consequential loss of required functions of systems needed to cope with the fault, including those of the RCS and the reactor containment system.

Event-specific SRP and DSRS sections provide additional acceptance criteria for AOOs, such as fuel centerline temperatures not exceeding the melting point for reactivity-initiated events.

NUREG-0800, Section 15.0.2, "Review of Transient and Accident Analysis Methods," Revision 0, dated March 2007 (ML070820123) provides guidance for the review of the methods used in transient and accident analyses, including the EM, and specifies recommended features of the EM.

In addition, RG 1.203 provides guidance for the development and assessment of transient and accident analysis EMs. It describes the EMDAP, a framework for developing and determining the adequacy of EMs, and fundamental elements of the EM documentation.

Chapter 15 of the DSRS and SRP recommend that an applicant use approved EMs or computer codes to analyze most events. Furthermore, SRP Section 15.0.2 and RG 1.203 identify six individual areas of review for transient and accident analysis methods:

- Documentation
- Evaluation models (EMs)
- Accident Scenario Identification Process
- Code Assessment
- Uncertainty Analysis
- Quality Assurance Plan

Each of these areas is discussed below.

2.2.1 Documentation

SRP Section 15.0.2 states that the EM documentation must be scrutable, complete, unambiguous, accurate, and reasonably self-contained. It must also be sufficiently detailed such that a qualified engineer can understand the documentation without recourse to the originator as required of any design calculation that meets the design control requirements of Appendix B to 10 CFR Part 50.

2.2.2 Evaluation Model

SRP Section 15.0.2 states that the EM should include all computational and non-computational elements, including field equations, constitutive and closure relations, and simplifying assumptions used to perform transient and accident analyses, and the NRC staff should review these elements to determine their applicability and adequacy.

2.2.3 Accident Scenario Identification Process

SRP Section 15.0.2 recommends that an applicant supply a complete description of the accident scenarios, including plant initial conditions; the initiating event and all subsequent events and phases of the accident; and the important physical phenomena and systems and/or component interactions that influence the outcome of the accident. This review criterion also recommends that the applicant use a structured process to identify and rank phenomena relevant to accident scenarios to which the analysis methodology will be applied, to determine the importance of the phenomena and their impact on the selected FOM. The predictive fidelity of the models in the EM should be commensurate with the importance of the associated phenomena.

2.2.4 Code Assessment

SRP Section 15.0.2 states that all code models, or changes to such models, that will be used in the EM should be assessed against SETs and IETs, including consideration of scaling and distortions.

2.2.5 Uncertainty Analysis

SRP Section 15.0.2 states that transient and accident methods should either estimate the uncertainty associated with the calculations, as is performed for best estimate analyses, or should provide a demonstrably conservative evaluation. If bounding analyses rather than uncertainty analyses are to be performed, bounding values for input parameters similar to those described in the SRP sections or RGs can be used for plant operating conditions such as accident initial conditions, setpoint values, and boundary conditions.

SRP Section 15.0.2 states that uncertainty analyses should address all important sources of code uncertainty, including the mathematical models in the code, and the user-selected inputs such as model nodalization. The major sources of uncertainty should be assessed in a manner consistent with the results of the accident scenario identification process. SETs should be used to determine the uncertainty bounds of individual physical models. IETs should be performed to demonstrate that the interactions between different physical phenomena and RCS components and subsystems are identified and predicted correctly.

2.2.6 Quality Assurance Plan

The SRP states that the EM should be maintained under a QA program (QAP) that meets the requirements of 10 CFR Part 50, Appendix B.

3 TECHNICAL EVALUATION

The technical evaluation of the TR is guided by the regulatory requirements and regulatory guidance described in Section 2, “Regulatory Basis for Non-LOCA EM Review,” of this SER. The evaluation starts with the principles of the EMDAP since the EMDAP guides the development of the EM. The technical evaluation also includes the aspects of RG 1.203 that are not specifically included in the EMDAP and considers the regulatory requirements as well as the higher-level guidance provided in the SRP and DSRS to ensure that they are either addressed in following the EMDAP or are addressed in the EM documentation.

The NRC staff performed audits of information provided by the applicant in support of the NRC staff’s review of the TR that are referred to throughout this SER. Information regarding this non-LOCA EM audit review is available in audit report (ML24262A257) which provides a summary of the information examined. The NRC staff requested audit responses that support assessment of the methodology with regulatory requirements to be appended to the TR. In addition, the NRC staff issued RAIs when additional information was needed to assess compliance with regulatory requirements for issues that could not be resolved through the audit process.

For consistency with the applicant’s terminology in the TR, the NRC staff uses the term “non-safety-related” in this SER to refer to SSCs that are not classified as “safety-related SSCs” as described in 10 CFR 50.2, “Definition.” However, among the “non-safety-related” SSCs, there are those that are “important to safety” as that term is used in the GDC listed in Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50, and others that are not considered “important to safety.”

3.1 Introduction

3.1.1 Purpose

TR Section 1.1, “Purpose,” describes the purpose of the TR and states that the NuScale non-LOCA EM follows a graded approach to the EMDAP. Significant overlap exists between the non-LOCA EM (MLXXX), the XPC EM (MLXXX), and the EM documented in the LOCA TR (Reference 8). The non-LOCA TR references the EMs in these TRs for the overlapping areas to avoid duplication of information. The NRC staff notes that a graded approach to the EMDAP, as discussed in RG 1.203, may be acceptable, provided that the modifications that form the EM are based on a previously approved EM. Therefore, any future changes to the LOCA EM need to be assessed by the applicant for their potential impact on the non-LOCA EM. Any subsequent changes to the non-LOCA methodology will require NRC approval. This is listed as Condition 2 in Section 4, "Limitations and Conditions," of the SER.

3.1.2 Scope

TR Section 1.2, "Scope," describes the scope of the non-LOCA EM, including specification of the computer codes used, the events considered, the development approach, and the analysis methodology. It also describes items not included in the scope of the non-LOCA EM, including SAFDL evaluation, radiological source term and dose analysis, long-term cooling and return to power analysis, control rod ejection analysis methodology.

The non-LOCA EM is applicable for the short-term transient progression, during which the RCS primary mixture level remains above the top of the riser and primary side natural circulation is maintained. This short-term transient progression also includes periods in which the **{{**

}}, a phenomenon that is further discussed in Section 3.5.1 of this SER.

During its review, the staff notes that although the TR does not reference a specific NPM design, the methodologies presented in the TR for analyses of the non-LOCA events are based on the design features, trip signals, and setpoints of the NPM-20. In addition, all example and supporting analyses provided by the applicant during the audit review (ML24262A257) are based on the NPM-20 design. The applicant stated that the non-LOCA EM remains applicable to the approved NPM-160 design as well, however, the staff is imposing Limitation and Condition (L&C) No. 1, which discusses the requirements an applicant or licensee wishing to apply the EM to either the NPM-160 design or a future design needs to fulfill.

L&C No. 3 discusses that the non-LOCA EM is no longer applicable when the coolant in the RCS shrinks sufficiently to drop below the top of the riser, which TR Section 5.1.3, "Phenomena Identification and Ranking Table Figures-of-Merit and Phenomenon Ranking," clarifies that this phenomenon could occur "well after reactor trip and engineered safety features have responded to the initiating event." The XPC EM provides methodologies for analyses of the system performance when RCS coolant level shrinks sufficiently to go below the top of the riser.

The NRC staff finds that the applicant clearly stated the intended use of the non-LOCA EM, and the scope of information provided in the TR, and other supporting documentation is consistent with the intended use, and therefore is acceptable for the purposes of assessing the scope of the non-LOCA EM.

3.2 Background

TR Section 2, "Background," describes the basic principles identified in RG 1.203 that are important in the development and assessment of an EM. This section also specifies that the EM uses the NRELAP5 computer code Version 1.7 **[OPEN ITEM]**, which is a descendant of the Idaho National Laboratory (INL) RELAP5-3D© computer code. NuScale initially submitted Version 1.6 of NRELAP5 (ML23011A012) for the US460 Standard Design Approval Application (SDAA) as the systems analysis computer code for LOCA and non-LOCA Evaluation

Methodologies. With respect to relevance to non-LOCA transients, the significant difference between RELAP5-3D and NRELAP5 is the helical coil hydraulic component (HLCOIL), which is used to model an NPM's unique SGs; the NPM SGs are parts of the DHRS loops and the DHRS is of central importance to non-LOCA transients. Subsequently during the US460 review, NuScale submitted NRELAP5 Version 1.7 (ML24228A242) as the systems analysis computer code for the NuScale LOCA Evaluation Methodology, replacing NRELAP5 Version 1.6.

3.2.1 Non-LOCA Evaluation Model Roadmap

Section 2.1, "Non-LOCA Evaluation Model Roadmap," of the TR provides the roadmap to the non-LOCA EM and refers to the EMDAP in RG 1.203. TR Figure 2-1, "Evaluation model development and assessment process," shows the elements and steps in the EMDAP, and TR Table 2-1, "Evaluation model development and assessment process steps and associated application in the non-LOCA evaluation model," cross-references, where in the documentation, each step of the EMDAP is addressed.

As discussed in Section 3.1.1, "Purpose," of this SER, the NRC staff finds a graded approach to the EMDAP to be consistent with the guidance provided in RG 1.203 and therefore to be acceptable, given approval of the LOCA EM. The NRC staff concludes that the applicant has acceptably documented the use of the graded approach and that the non- LOCA EM roadmap is complete and consistent with the guidance provided in RG 1.203.

3.2.2 Regulatory Requirements and Regulatory Guidance

TR Section 2.2, "Regulatory Requirements and Regulatory Guidance," identifies regulatory requirements and guidance relevant to the non-LOCA transient analyses, including several GDCs, RG 1.203, and specific DSRS and SRP sections. The NRC staff reviewed the discussions and concluded that the applicant has specified the appropriate regulatory requirements and regulatory guidance discussed in SER Section 2, "Regulatory Basis for Non-LOCA EM Review."

3.3 Plant Design Overview

The NRC staff reviewed the plant design information in Section 3, "Plant Design Overview," of the TR only to identify aspects relevant to the non-LOCA EM. This review does not evaluate the plant design. The major details of the plant design relevant to the non-LOCA EM are described below.

3.3.1 Description of NuScale Plant

Non-LOCA TR Section 3.1, "Description of NuScale Plant," briefly describes the configuration and features of the NuScale plant design that are unique compared with existing operating pressurized-water reactor (PWR) plants. The NPM is a small integral PWR. One or more NPMs form a NuScale Power Plant. Each NPM has its own chemical and volume control system (CVCS), ECCS, and DHRS.

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 helical coil steam generators (HCSGs) and an RPV-internal pressurizer (PZR) that eliminates piping to connect the SG or PZR 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 DHRS

- Primary fluid in the SGs flows on the outside of the tube surfaces, and two-phase flow of the secondary fluid is contained inside the tubes

- A high-pressure steel CNV partially immersed in a water-filled pool (or ultimate heat sink (UHS)) for cooling and decay heat removal purposes.

- Reactor vent valves (RVVs) without inadvertent actuation block (IAB)

The NRC staff reviewed the general description of the plant design in TR-0516-49416-P, Revision 4 and finds it provides a sufficient description of the design to support the description of the methodology.

3.3.2 Plant Operation

TR Section 3.2, "Plant Operation," briefly describes the plant configuration during normal operation as well as control and protection systems for the individual power modules and overall plant. Control systems that are active during normal operation include the CVCS and PZR sprays and heaters. Normally, the steam generators (SGs) transfer heat to the feedwater and DHRS is isolated. In addition, the CNV is evacuated during normal operation, which reduces the convective heat load on the CNV shell.

The module control system (MCS) and plant control system (PCS) provide monitoring and control to non-safety-related plant systems, such as RCS pressure control, feedwater and turbine control, and rod control and position indication. The reactor trip system (RTS) and engineered safety features actuation system (ESFAS) comprise the module protection system (MPS), which provides automatic protection functions during off-normal conditions. Depending on the MPS signal, the protection functions may include a reactor trip; isolation of feedwater, main steam, CVCS, and/or containment; and actuation of the DHRS and/or the ECCS.

The makeup function of CVCS is not credited in non-LOCA event evaluations. TR Sections 6.3.1.2, 7.1.2, and tables in Section 7.2 show event-specific assumptions concerning non-safety system operation.

TR Section 3.2, "Plant Operation," states that the systems credited to mitigate non-LOCA events include the DHRS, ECCS, MPS, and RTS. As discussed in SER Section 3.3.4, "Emergency Core Cooling System," the ECCS does not actuate in the timeframe covered by the non-LOCA methodology. In addition, isolation of the CVCS, containment, demineralized water system, and PZR heaters is credited. The UHS is the only safety system shared among modules.

3.3.3 Decay Heat Removal System

TR Section 3.3, "Decay Heat Removal System," states that the DHRS is a closed-loop, two-phase natural circulation cooling system. The DHRS consists of two trains, one attached to each SG loop. Each train removes decay heat from the RPV and rejects it to the reactor pool via condensers submerged in the reactor pool. The staff notes that DHRS capability is assessed through the design review in a licensing application. The representation of decay heat for non-LOCA analysis is discussed in Section 3.6.1, "Thermal-Hydraulic Volumes and Heat Structures," of this SER.

3.3.4 Emergency Core Cooling System

TR Section 3.4, "Emergency Core Cooling System," briefly discusses the ECCS design. The ECCS for the NPM-20 includes two RVVs on top of the RPV and two reactor recirculation valves (RRVs) on the side of the RPV in the downcomer region. Each RRV includes an IAB to prevent the RRV from an inadvertent opening. The ECCS is actuated by the simultaneous opening of the RVVs and RRVs, which allows a natural circulation cooling path to be established. Vaporized water leaving the core exits the RVVs as steam, condenses and collects in containment, and flows back into the RPV through the RRVs.

The TR states that the IAB consists of a spring loaded arming valve in the vent port path from the main disc chamber to the vent line and the RRV main valves open when the differential pressure of the spring-loaded arming valve decreases below the release pressure. If the IAB threshold is reached, the RRV valves will fail open on a loss of power. Section 3.7.1, "General Aspects of Non-LOCA Methodology," of this SER further discusses loss of power scenarios and ECCS actuation.

The ECCS is not actuated in the short time frame of any of the non-LOCA events documented in the TR, presuming there is not a loss of EDAS power. Once, or if, the ECCS valves open, those events or event phases are analyzed according to the methodology in TR-0516-49422, "Loss-of-Coolant Evaluation Model," Revision X (MLXXXX). The consequences of the non-LOCA events described in this TR are mitigated by the actuation and the operation of the DHRS for the short-term transient period evaluated with this methodology. The ECCS actuation, which includes the ESB feature, following the DHRS actuation is addressed in the XPC TR (Reference 7).

3.3.5 Other Important Systems and Functions

TR Section 3.5, "Other Important Systems and Functions," provides a general discussion of the RCS, feedwater system, main steam system, CVCS, CNV, and reactor pool.

The staff reviewed the information provided in Section 3 of the TR and finds that it includes sufficient descriptions of the SSCs important to the safety analyses of the non-LOCA events as defined in Section 1.2 of the TR.

3.4 Transient and Accident Analysis Overview

TR Section 4, "Transient and Accident Analysis Overview," discusses event classifications, acceptance criteria, and the transient analysis process. The discussion also includes the interfaces of the non-LOCA EM with other methodologies.

3.4.1 Design-Basis Events and Event Classification

TR Section 4.1, “Design-Basis Events and Event Classification,” provides the event categories. The categories according to the frequency of occurrence are: AOOs, IEs, and PAs. The categories according to the event type are: (1) increase in heat removal from the RCS, (2) decrease in heat removal by the secondary system, (3) reactivity and power distribution anomalies, (4) increase in the reactor coolant inventory, and (5) decrease in the reactor coolant inventory. The NRC staff notes that the event categories are, in general, similar to those for traditional large PWRs and also are consistent with SRP Section 15.0. The exception is the lack of a decrease in the RCS flow rate category, which the NRC staff notes is acceptable in this case, as there is no forced cooling in the NPM design.

The TR states that event classification is based on historical precedent for initiating events similar to those in currently operating plants and certified designs. For events that are unique to the NPM design or where differences relative to operating and certified designs are known to exist, the TR states that event frequencies are based on results of the probabilistic risk assessment.

The one unique event for the NPM is the failure of small lines carrying primary coolant outside of the containment, which is classified as an IE for consistency with SRP/DSRS guidance for dose consequences.

TR Section 4.1, “Design-Basis Events and Event Classification,” notes that the non-LOCA EM analyses are performed for a single module. Some initiating events, such as a loss of AC power, may affect multiple modules. Since the only shared safety system among modules is the UHS, the applicant assumes a pool temperature that bounds possible interactions between modules (ML24305A000). Section 3.6.1, “Thermal-Hydraulic Volumes and Heat Structures,” of this SER further discusses the pool temperature assumption.

3.4.2 Design Basis Event Acceptance Criteria

TR Section 4.2, “Design Basis Event Acceptance Criteria,” discusses acceptance criteria for AOOs, IEs, and PAs. The acceptance criteria relevant to non-LOCA system transient analyses, excluding containment and radiological acceptance criteria (as margin to these acceptance criteria are determined with other analysis methodologies), are as follows:

AOOs

- Maximum RCS primary system pressure ≤ 110 percent of design pressure.
- Maximum main steam secondary system pressure ≤ 110 percent of design pressure.
- Minimum critical heat flux ratio (MCHFR) > 95/95 critical heat flux ratio (CHFR) limit.[1]
- Maximum fuel centerline temperature ≤ melting temperature (adjusted for

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burnup effects).[1]

- An AOO should not result in a significant loss of reactor containment barrier.[2]

- An AOO should not generate a postulated accident without other faults occurring independently.

<u>IEs and PAs</u>

- Maximum RCS primary system pressure ≤ 120 percent of design pressure.

- Maximum main steam secondary system pressure ≤ 120 percent of design pressure.

- Fuel cladding integrity: If MCHFR ≤ 95/95 CHFR limit, or if maximum fuel centerline temperature > melting temperature, fuel rod is assumed to be failed[1].

- Containment integrity: Margins to containment pressure and temperature limits are maintained.[2]

The NRC staff finds these acceptance criteria acceptable because they are consistent with those listed in SRP and DSRS Sections 15.0. DSRS Section 15.0 is listed in Section 2.2, "Regulatory Guidance," of this SER.

The applicant also includes in Table 4-2, Table 4-3, and Table 4-4 of the TR a few acceptance criteria that are outside the scope of the Non-LOCA TR (i.e. containment integrity and release of radioactive material). For containment integrity the applicant does not provide any evaluation methodologies in this TR and this SER does not include discussions of these aspects.

SRP Section 15.0.2 states that a complete uncertainty analysis is not needed if suitably conservative input parameters are used. TR Section 4.2, "Design Basis Event Acceptance Criteria," states that the methodology includes performing sensitivity calculations to determine that suitably conservative inputs that result in the minimum margins to acceptance criteria are chosen. Section 7.2 of the TR specifies a minimum set of sensitivity studies that must be performed when the TR is implemented in order to assess margin to acceptance criteria. However, when these results show that margins to acceptance criteria are challenged, additional sensitivity studies on additional parameters are performed in order to ensure that acceptance criteria are met. In this way, the methodology may demonstrate margin without extensive sensitivity studies to minimize the margin to those unchallenged acceptance criteria. The applicant further clarified (ML18270A469) during the review of the previous revision of this topical report that it developed the non-LOCA methodology to ensure that combinations of models and inputs at extremes do not result in non-conservative predicted results. The applicant stated that it ensured consistent behavior based on bias directions (e.g., biasing initial pressure high always results in a higher peak pressure) and consistent input importance. The applicant referenced validation studies in TR Section 5, "NRELAP5 Applicability for Non-LOCA

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Transient Analysis," and bias direction sensitivity studies in TR Section 7, "Non-LOCA Analysis Methodology," which demonstrate the consistent behavior.

Based on its review of the validation studies and event-specific sensitivity study results, discussed in Sections 3.5, "NRELAP5 Applicability for Non-LOCA Transient Analysis," and 3.7, "Non-LOCA Analysis Methodology," of this SER, the NRC staff finds that the applicant's approach provides for use of suitably conservative input parameters and is consistent with SRP Section 15.0.

3.4.3 Non-LOCA Transient Analysis Process

TR Section 4.3, "Non-LOCA Transient Analysis Process," describes the six steps in the non-LOCA transient analysis process. The staff's evaluations of these steps are documented in the subsections below.

3.4.3.1 Develop Plant Base Model NRELAP5 Input

The NRELAP5 computer code, which is based on modifications to the RELAP5-3D (Version 4.1.3) computer code developed by Idaho National Laboratory, is the system thermal-hydraulics code that the applicant uses for its non-LOCA system transient analyses. The NRELAP5 code is being maintained within NuScale's QAP. The NRELAP5 computer code, Version 1.6 (ML23011A012), was submitted for the US600 SDA as the systems analysis computer code. 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. Of note in Version 1.7, NuScale modified some algorithms used to blend between process models, particularly the choking models, to return smoother physics transitions; also, a parametric card option was added to allow users to specify the magnitude of a numerical stability modifier applied at cell junctions. Generally, topics covered by the non-LOCA TR are unaffected by the difference(s) between Versions 1.6 and 1.7 of NRELAP5 (ML24305A000). In addition to the FOMs for Non-LOCA system transient analyses listed in Section 3.4.2, "Design Basis Event Acceptance Criteria," of this SER (i.e., maximum RCS pressure and maximum secondary pressure), TR Section 4.3.1, "Develop Plant Base Model NRELAP5 Input," states that the RCS water level response is also evaluated for non-LOCA events that result in an RCS inventory decrease.

TR Section 4.3.1.1, "Interface with Core Design (Input to the Transient Analysis)," discusses the inputs to the non-LOCA EM that result from the interface with core design, including the reactor kinetics parameters, the moderator temperature and Doppler temperature coefficients, and the expected axial power distributions from which an appropriate axial power distribution should be selected for the transient analysis.

TR Section 4.3.1.1.1, "Reactor Kinetics Model," provides a discussion of the point kinetic model used in the NRELAP5 computer code. The TR states that the total core power during a non-LOCA transient is the combination of the fission power and the decay heat, with the fission power response modeled using the separable point reactor kinetics model in NRELAP5. The TR also provides a list of the parameters that are used in calculations of the total core power during a non-LOCA transient (ML24348A052).

The staff reviewed the method for calculating the total core power during a non-LOCA transient in the TR and finds the method to be consistent with the basic reactor physics point kinetic equation and decay heat generation during a non-LOCA transient. On this basis, the staff determined that the methodology implemented in the NRELAP5 to be acceptable.

TR Section 4.3.1.1.2, “Axial Power Shape,” states that for the system transient analysis, a single channel core model is used. The single channel model is described in Section 6.0 of the TR. A nominal center-peaked average axial power shape is used as input. This is consistent with the single channel core for reactivity feedback coefficients determined in the core design analyses. Uncertainties associated with the axial power shape and radial power peaking factors that can affect the MCHFR and peak centerline fuel temperature are accounted for in the downstream subchannel analyses as described in, TR-0915-17564-P-A, “Subchannel Analysis Methodology,” Revision 2” (ML19067A257), which is supplemented by, TR-108601-P (Supplement 1 to TR-0915-17564-P-A, “Statistical Subchannel Analysis Methodology,”), Revision 4. (ML24348A042 & ML24348A084).

During its review, the staff audited the applicant’s nuclear design analyses and conformed that the NPM-20 core does have a center peaked axial power shape under full power. On this basis, the staff finds it acceptable to assume nominal center-peaked average axial power shape in the single core channel model.

The applicant provides methodologies in the TR that must be used for performing sensitivity studies on the impacts of the axial power shape on the primary and secondary system pressure, flow and fluid temperature responses, to include the parameters and acceptance criteria. The staff reviewed the methodologies presented in the TR and finds that there is reasonable assurance that the methodologies will be able to capture the impacts of the axial power shape on primary and secondary system pressure, flow rate and fluid temperature responses.

TR Section 4.3.1.1.3, “Energy Deposition Factor,” states that a bounding-high energy deposition factor, i.e., the portion of the energy generated in the core that is directly deposited in the fuel, is assumed for non-LOCA calculations and further states that sensitivity studies using the non-LOCA EM demonstrate that margins to acceptance criteria are insensitive to changes in the energy deposition factor.

The NRC staff reviewed the methodology for determining the bounding energy deposition factor. The staff finds that the methodology and acceptance criterion provide reasonable assurance that the bounding factor will be identified by the sensitivity studies that must be performed. TR Section 4.3.1.1.3 states that a bounding high energy deposition factor is used such that all energy is assumed to be deposited in the fuel. In response to the staff audit question, the applicant clarified (ML24348A048 (NP) & ML24348A049 (P)) that the energy deposition factor is the same as used in Revision 3 of this TR (ML20148M391 (Publicly Available) & ML20148M392 (Non-Publicly Available) which uses the average burnup to determine the energy deposition factor for each loading batch. The staff finds this approach is consistent with the loading pattern of the NPM-20 design and therefore acceptable. The applicant performed sensitivity studies on varying the energy deposition factor. The staff audited the sensitivity study (ML24262A257) performed by the applicant and concludes that that methodology presented in the TR for performance of the sensitivity study is acceptable because it includes examination of the effect of reducing the energy deposition factor through direct moderator heating. The NRC staff concludes that the EM requires use of bounding-high energy deposition factor for the NRELAP5 non-LOCA system transient analyses and finds this acceptable.

TR Section 4.3.1.2, “Interface with Fuel Rod Performance Design (Input to Transient Analysis),” discusses the inputs to the non-LOCA EM that result from the interface with fuel rod design, including fuel geometry, fuel thermo-mechanical properties, and fuel performance data. Section 4.3.1.2.2, “Fuel Rod Material Properties,” of the TR states that fuel thermal conductivity is calculated based on a representative time-in-cycle core average burnup. However, the TR does not provide details on the representative core average burnup and the dependence of fuel conductivity on burnup.

The NRC staff discussed these topics with the applicant during its audits (ML24262A257). The applicant clarified that it assumed burnup corresponds to an average value for a typical UO_2 core ranging from 0 gigawatt-days per metric ton of uranium (GWd/MTU) at beginning of cycle (BOC) to about **{{ }}** at end of cycle (EOC) (ML24348A054 (NP) & ML24348A055 (P)). The NRC staff confirmed that these values are consistent with those in the NuScale NPM-20 but recognizes that they may change if the fuel design or operation strategy changes. As TR Section 4.3.1.2.2 requires use of representative time-in-cycle core average burnup, it requires users applying the non-LOCA TR to examine these cycle-average fuel burnup values to assure they are within the assumed range because burnup has a significant impact on fuel thermal conductivity. NuScale described through response to audit questions (ML24348A054 (NP) & ML24348A055 (P)) that maximum core-average fuel exposure is calculated by taking an average of the highest end-of-cycle exposures for each batch of fuel. The NRC staff audited the NPM-20 NRELAP5 non-LOCA model calculation package and confirmed that the maximum core-average fuel exposure was calculated in this way. The applicant also stated that the fuel thermal conductivity is consistent with the burnup-dependent value calculated by the fuel performance code. This additional information adequately clarified the docketed material, and the NRC staff concludes that the interface with the fuel design analysis, as described in the TR, is acceptable.

In Section 4.3.1.2.3, the TR discusses the fuel performance data that are used in the NRELAP5 models for analyzing the non-LOCA transients. The TR states that the fuel rod gap conductance, specific heat, and density in the NRELAP5 fuel rod heat structure are used to set the initial core average fuel temperature.

Bounding values for fuel rod gap conductance are selected to provide conservatively high or low core average fuel temperature for the time-in-life of interest for the calculation. TR Section 4.3.1.2.3 states that the conservative core average fuel temperatures are confirmed on a cycle-specific basis.

The staff reviewed the selection of the fuel performance data and finds it to be acceptable because the methodology requires use of bounding parameters, and the values must be confirmed on a cycle-specific basis. The staff also finds that confirmation of the bounding parameters for each cycle provides further assurance that accurate fuel performance data will be used because the thermal conductivity will degrade as burnup increases.

3.4.3.2 Adapt Plant Base Model NRELAP5 Input for Event-Specific Transient Analysis

TR Section 4.3.2, "Adapt Plant Base Model NRELAP5 Input for Event-Specific Transient Analysis," states that the NRELAP5 plant base model is adapted for the event-specific analyses, including biasing of initial and boundary conditions, single failures, and loss of power scenarios. These adaptations are described in TR Section 7, "Non-LOCA Analysis Methodology," and are evaluated in Section 3.7, "Non-LOCA Analysis Methodology," of this SER.

3.4.3.3 Perform NRELAP5 Steady-State and Transient System Analysis Calculations

TR Section 4.3.3, "Perform NRELAP5 Steady State and Transient System Analysis Calculations," states that at least one steady-state initialization calculation is performed for each transient analysis and transient calculations are performed after confirming that acceptable steady-state conditions have been reached. TR Section 7.1, "General," further discusses this process and is evaluated in Section 3.7.1, "General Aspects of Non-LOCA Methodology," of this SER.

3.4.3.4 Evaluation of the Results of Transient Analysis Calculations

TR Section 4.3.4, Evaluate Results of Transient Analysis Calculations," describes how the transient analysis results are evaluated for acceptability, including evaluation against RCS and SG pressure acceptance criteria. Section 4.3.4 also describes conditions that are to be demonstrated for typically a few hundred seconds following the last expected safety system actuation in the short-term transient progression:

- MPS actuations expected in direct response to the initiating event have occurred
- If reactor trip occurs, power is reduced to decay heat levels and decreases with time
- Core average temperature is stable or decreasing following reactor trip
- RCS pressure is stable or decreasing
- RCS fluid inventory is stable
- Containment pressure is stable or decreasing

The NRC staff reviewed these criteria and concludes that meeting these conditions is sufficient to demonstrate that the minimum margin to acceptance criteria has occurred and that adequate core cooling has been established.

The NRC staff notes that, in some circumstances, the criterion that AOOs should not generate a postulated accident without other faults occurring independently or result in consequential loss of function of the RCS or containment barriers (the non-escalation criterion) may be challenged if SSCs are subjected to circumstances for which they are not qualified. For example, pressure relief valves may not be qualified to relieve water, and failure of such a valve to reseat could result in a design-basis accident (small-break LOCA). However, the staff notes that component qualification is typically reviewed during licensing applications for specific plant designs. As such, in addition to the plant-response evaluated by the non-LOCA evaluation model, plant-specific applications implementing this methodology will need to address the range of component qualification when establishing that the non-escalation criterion is satisfied.

3.4.3.5 Identification of Cases for Subchannel Analysis and Extraction of Boundary Condition Data

TR Section 4.3.5, "Identification of Cases for Subchannel Analysis and Extraction of Boundary Condition Data," states that the VIPRE-01 computer code is used to determine the MCHFR and maximum fuel centerline temperature. NRELAP5 results provide initial and boundary conditions for the VIPRE-01 calculations. The following NRELAP5 time-dependent results are provided as input to the VIPRE-01 calculation: reactor power, core exit pressure, core inlet temperature, and total RCS flow rate. The TR states that the cases selected for the downstream subchannel analysis are those with conservative bias directions for these boundary conditions, including maximum reactor power, maximum core inlet temperature, and minimum system flow rate. The NRC staff notes that the directions of conservatism for these parameters in the NPM are logical and consistent with the conservative bias directions for typical large PWRs.

TR Section 4.3.5 states that the impact of pressure on critical heat flux (CHF) is evaluated on a case-by-case basis by varying initial PZR pressure. The NRC staff notes that variation of other parameters, such as operation of pressure control systems or evaluating a spectrum of initiating events, also aid in variation of RCS pressure. The staff reviewed methodology provided in the TR for determining the effect of pressure on CHF and audited the applicant's calculations examining the sensitivity of CHF to RCS pressure for NPM-20 operating conditions and finds that there is a reasonable assurance that the methodology will be able to identify and capture the impact of pressure on the CHF.

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Based on the description of the process to identify limiting cases as described in the TR, and confirmed during the staff's audit, the staff finds that there is a reasonable assurance that the NRELAP5 calculations will identify the potential challenging cases necessitating subchannel analyses.

The TR further requires that the system transient parameters are provided for subchannel analysis for a sufficient time for the subchannel analyses to demonstrate that the MCHFR has occurred, typically at least 10-15 seconds following reactor trip. While the typical simulation time given in the TR is just an example, the staff understands that the methodology requires the user of the TR to ensure that the simulation time is sufficient to calculate the minimum CHFR for each event. The methodology also requires user to pass on the appropriate portion of the transient to the subchannel analysis such that minimum CHFR is identified. The staff finds these requirements to be appropriate and acceptable because they provide reasonable assurance that the MCHFR is captured.

3.4.3.6 Identification of Cases for Accident Radiological Analysis

As discussed in TR Section 4.3.6, "Identification of Cases for Accident Radiological Analysis," NRELAP5 transient analysis results are provided as input to accident radiological analyses for events that result in reactor coolant loss outside of the containment (e.g., failure of small lines carrying primary coolant outside containment and steam generator tube failure (SGTF)). The conservative bias directions for accident radiological analyses are:

- Maximum integrated mass release outside of containment prior to isolation of the RCS mass release.

- Maximum integrated mass release between time of reactor trip and time of isolation of the RCS mass release.

The applicant provided a detailed discussion of the bases for identifying the above-mentioned two cases as bounding scenarios for a particular initiating event. The staff reviewed the discussions and finds that the applicant established bases for its conclusions that these two scenarios are bounding and agrees that these two scenarios are likely to result in bounding radiological consequences for radiological analyses of specific non-LOCA transients.

The TR provides various interface information to the radiological consequence analysis, including time of the reactor trip, time of the reactor coolant release isolation, time-dependent mass release, and other time-dependent system parameters.

The staff notes that, this is similar to traditional large PWRs, accident radiological consequences for the NPM tend to increase with increasing integrated mass release outside of containment prior to isolation of the source, with iodine spiking and the timing of events potentially affecting the radiological consequences. On this basis, the NRC staff finds that the approach for identification of cases for accident radiological analysis is appropriate for the NPM design and it is therefore, acceptable.

As an alternative to using values obtained from transient analysis, the TR allows for using bounding values for both mass release and isolation times for accident radiological analysis. Additional information on this approach was provided by the applicant (ML24262A257). The applicant provides a generic description and an example for this method in which mass release is calculated as the mass required to reduce PZR level from a high PZR level trip setpoint to a low PZR level trip setpoint that results in RCS isolation, accounting for sensing and actuation delays and PZR level uncertainty. System transient analysis is still required to confirm that these assumptions are bounding. While NRC staff finds this general approach to be acceptable TR does not specify a particular methodology for determining these bounding values (ML24348A069 (NP)). Therefore, values of and justification for inputs to the radiological consequence methodology developed using this approach are reviewed as part of a design-specific application of this methodology, such as the NPM-20 SDA. This is reflected in item 9 in Section 4, “Limitations and Conditions,” of this evaluation.

3.5 NRELAP5 Applicability for Non-LOCA Transient Analysis

3.5.1 Non-LOCA Phenomena Identification and Ranking Table

A panel of experts developed the non-LOCA PIRT described in TR Section 5.1, “Non-LOCA Phenomena Identification and Ranking Table and Evaluation of High-Ranked Phenomena,” based on the state-of-knowledge at the time of the PIRT development. The PIRT was originally developed based on the earlier NPM-160 design and applied to the NPM-20 design. The applicant compared these two designs and identified that the major differences between the two designs are: (1) NPM-20 has a higher power output (250 MWth), (2) increased reactor temperature, and (3) increased reactor pressure, and some geometric and dimensional changes such as minor reflector bypass flow area, additional riser holes. However, the only significant geometry changes affecting the NPM-20 non-LOCA system transient behavior are those associated with the DHRS loop. In addition, a new ECCS supplemental boron system is added to address potential re-criticality concern under long term core cooling operation. The methodologies for analyzing long term cooling events are discussed in the XPC TR (Reference 7).

The applicant assessed the impact of the design changes to the PIRT based on a comparison of the NPM-20 design with the NPM-160 design. The comparison includes the normal operating conditions at full power, design limits, and the geometric parameters. The results of the assessment identified no new phenomena that are applicable during the non-LOCA event phases although the time dependent event progression and setpoints associated with the NPM-20 design are different. The assessment concluded that the non-LOCA PIRT remains applicable to the NPM-20 design, consistent with the description in Section 3.0 of the TR. TR Table 5-3 provides the PIRT developed for up to phase 3 of the NPM-20 design.

The staff reviewed the discussion of the PIRT process used by the applicant. Based on a review of the applicant's responses to audit questions (ML24262A257), the staff finds as acceptable the conclusion made by the applicant that PIRT remains applicable to the NPM-20 design because the power density, RCS pressure and temperature, and additional riser holes in the NPM-20 will not changes the events, system responses, and FOMs. The change in the reflector hole design has insignificant impact on the non-LOCA events because the change in the reflector flow area is very small.

The non-LOCA PIRT identifies key phenomena that may occur in the NPM during a non-LOCA event, ranks their relative importance with respect to FOM, and ranks the knowledge level of each phenomenon. The PIRT panel considered non-LOCA event types considered in the TR by dividing the events into five different categories and evaluating one representative design-basis event from each category:

- Cooldown/depressurization events: main steam line break inside the containment
- Heatup/pressurization events: feedwater line break inside the containment
- Reactivity-initiated events: control rod assembly (CRA) withdrawal
- Events that result in an increase in RCS inventory: CVCS malfunction
- Events that result in a decrease in RCS inventory: SGTF

The NRC staff notes that these representative events are the most challenging non-LOCA events with respect to FOMs in each of the respective event categories and are therefore appropriate for evaluation.

The PIRT panel divided the non-LOCA event progression into three distinct phases and defined the FOM that is important for each phase, as shown in the table below:

Phase	Phase Description	FOM

1 – Pre-trip transient	Begins with the event initiation and ends with the actuation of the MPS.	• CHF (may be challenged by cooldown and reactivity-initiated events) • Primary pressure (may be challenged by heatup and RCS inventory increase events)
2 – Post-trip transition	Begins with MPS actuation (and often DHRS actuation). Reactor power and RCS flow rates transition towards decay heat levels.	• CHF • Primary pressure • Secondary pressure (maximum secondary pressure may occur due to DHRS actuation) • Containment pressure (indicates containment integrity; non-LOCAs may release mass and energy into containment)
3 – Stable natural circulation	Stable primary and DHRS (if applicable) natural circulation conditions are established. Primary temperature and pressure, and secondary side flow rate and pressure, decrease.	• CHF • Coolant mixture level (indicates whether primary side natural circulation is maintained; if DHRS heat removal is sufficient to drop the RCS water level below the top of the riser, natural circulation is interrupted, and it is the end of Phase 3) • Subcriticality (limits heat source to decay heat levels)

The TR states that if the coolant mixture level is not maintained above the top of the riser, natural circulation may be interrupted, ending Phase 3, and that this is well after reactor trip and engineered safety features have responded to the initiating event. The EM is applicable for the short-term non-LOCA transient progression, and during this time frame, the mixture level remains above the top of the riser and primary side natural circulation is maintained. The reactivity control and extended passive cooling analysis methodology in the long-term, including events that transition from DHRS cooling to ECCS cooling, is addressed in the XPC TR (Reference 7).

The staff performed confirmatory analyses on the liquid level during non-LOCA transients and confirmed that the mixture level remains above the top of the riser and that primary side natural circulation is maintained during the non-LOCA transient time frame. Based on the evaluation of the information provided by the applicant, the NRC staff agrees that the riser uncovery and interrupted natural circulation scenario will not be encountered in the short- term following those non-LOCA events within the scope of the TR. The XPC EM TR (Reference 7) addresses the time after which mixture level has dropped below the top of the riser.

With respect to subcriticality, the TR notes that the boron in the primary system during Phase 3 is limited to the soluble boron at the RCS critical boron concentration from normal operating conditions and not the addition of supplemental boron by ECCS. The addition of any supplemental boron by ECCS is outside the non-LOCA Phase 3 scope and is assessed separately in the XPC TR.

Each PIRT phenomenon was assigned an importance ranking and knowledge level considering all five representative non-LOCA events. The importance rankings are defined as:

- High (H) - Significant influence on the FOMs
- Medium (M) - Moderate influence on the FOMs
- Low (L) - Small influence on the FOMs
- Inactive (I) - Phenomenon is not present or negligible.

The knowledge level rankings are defined as:

- 4 - Well-known/small uncertainty
- 3 - Known/moderate uncertainty
- 2 - Partially known/large uncertainty
- 1 - Very limited knowledge/uncertainty cannot be characterized

TR Section 5.1.4, “Highly Ranked Phenomena,” lists the highly ranked phenomena identified for the non-LOCA PIRT, including the knowledge level, the systems and components in which the phenomenon was highly ranked, the basis for the ranking, and how the phenomenon is addressed (e.g., by the downstream subchannel analysis, specifying appropriately conservative input, or NRELAP5 assessment studies). The TR does not list or discuss phenomena of moderate or small influence on the FOM.

The applicant provides detailed discussions on the highly ranked phenomena in TR Subsections 5.1.4.1 to 5.1.4.1.57. The discussions include identification of the causes, progression, safety impacts (ranking) of each phenomenon and the knowledge on the phenomenon. These subsections also discuss the methodologies for analyzing the phenomena.

The staff reviewed the discussions presented in TR Subsections 5.1.4.1 to 5.1.4.1.57 and finds that the applicant has correctly identified all phenomena that have significant impacts on the safety of the NPM design, the causes, progressions, safety significance, and the corresponding ranking. The staff also finds that the applicant followed the guidance and industry practice in identifying and ranking the phenomena and knowledge gaps. On these bases, the staff finds with reasonable assurance that the applicant has correctly identified and ranked the highly safety significant phenomena and the knowledge gaps.

TR Section 5.1.4, “Highly Ranked Phenomena,” also details how certain highly ranked phenomena, such as **{{ }}**, are addressed in the subchannel analysis rather than in the non-LOCA EM, and how they relate back to the non-LOCA EM.

The applicant states that some parameters, such as **{{ }}**, that are important for the subchannel analysis may not be important to the non-LOCA transient response and were therefore not included in the non-LOCA EM. The applicant also noted that the code and plant design changes since the original PIRT was developed, have insignificant effects on the PIRT for the NPM-20 design, therefore, PIRT updates were not necessary.

In the NPM, phenomena such as **{{**

}} are of particular interest, among others. These and other highly ranked phenomena were discussed extensively in the audit (ML19039A090) conducted in the review of the previous version of this TR. The discussions clarified how the phenomena were appropriately considered, ranked, and addressed. For review of the applicability of the PIRT to the NPM-20 design, the staff audited (ML24262A257) relevant documentation.

The NRC staff finds that the applicant adequately identified highly ranked phenomena and provided the corresponding knowledge levels, systems/components in which the phenomena are applicable, bases for the rankings, and explained how the phenomena are addressed. This conclusion is based on the NRC staff's knowledge and understanding of the NuScale design, the NRC approved PIRT for NPM-160 design, and information from other LWR PIRTs that have been previously developed and/or approved by the NRC staff.

3.5.2 Evaluation of Non-LOCA Phenomena Identification and Ranking Table High-Ranked Phenomena

TR Section 5.1.4, "Highly Ranked Phenomena," discusses the evaluation of highly ranked phenomena. Therefore, TR Section 5.2, "Evaluation of Non-LOCA Phenomena Identification and Ranking Table High-Ranked Phenomena," simply points to TR Section 5.1.4, which is evaluated in Section 3.5.1, "Non-LOCA Phenomena Identification and Ranking Table," of this SER.

3.5.3 NRELAP5 Validation and Assessments for Non-LOCA

NRELAP5 is the system thermal-hydraulics code used to simulate an NPM system response during non-LOCA short-term transient event progression. The NRELAP5 assessments performed as part of the development of the LOCA TR (Reference 8), demonstrate the capability of the code to simulate an NPM-20 response to LOCA events. TR Section 5.3, "NRELAP5 Validation and Assessments for Non-LOCA," of the TR discusses the SETs, IETs, computational fluid dynamics (CFD), and code-to-code comparison performed to assess (1) the applicability of the NRELAP5 computer code for predicting the system responses to non-LOCA transients, and (2) the NuScale non-LOCA EM phenomena beyond what was done as part of the LOCA EM development (concentrating on non-LOCA EM assessments that examine heat transfer from the RCS to the DHRS and reactor pool via the SGs). The TR states that the agreement between NRELAP5 predictions and data or the code-to-code comparison is assessed in accordance with RG 1.203 definitions of excellent, reasonable, minimal, or insufficient agreement. The following subsections of this SER document the staff's evaluations of the assessment performed by the applicant of the NRELAP5 computer code for performing transient analyses for the NPM-20 design.

3.5.3.1 KAIST

As discussed in TR Section 5.3.1, "KAIST," the applicant used high-pressure condensation data from experiments performed at the KAIST facility to assess NRELAP5 predictions of condensation inside, and heat transfer across, DHRS tubes. The staff has reviewed the applicability of the KAIST experiment data for validation of the NRELAP5 computer code for the NPM-160 design. The staff's conclusions are documented in Non-LOCA TR-0516-49416-P-A, Revision 3 (MLXXX). The staff reviewed the conclusions and finds them to be applicable to the

NPM-20 design. Based on its review, the staff finds that the KAIST tests verify the EM against the various phenomena identified in the PIRT. Since the PIRT for the NPM-160 was confirmed to be applicable to the NPM-20, the KAIST tests are applicable to the NPM-20. For this reason, the staff decided not include a detailed discussion of the assessment in this SER version.

In summary, the NRC staff noted that the KAIST tests, in combination with the NIST-1 HP-03 SETs discussed further in section 3.5.3.2.1 of this evaluation, adequately covered the expected ranges of DHRS operation. The NRC staff agreed with the applicant that the predicted heat transfer coefficients, wall temperatures, and condensed liquid flow rates for the KAIST experiments provided in the LOCA TR show reasonable to excellent agreement with the test data. In addition, the more holistic measure of total heat transfer as a function of pressure provided by the applicant (ML18240A378) showed reasonable to excellent agreement (generally within five percent). Therefore, the NRC staff agreed with the applicant's conclusion that NRELAP5 predicts **{{** **}}** with reasonable to excellent agreement.

3.5.3.2 NIST-1 Decay Heat Removal System Separate Effects Tests

As part of the non-LOCA EM validation, the applicant performed SETs at the NuScale Integral System Test-1 (NIST-1) test facility, which is described in Section 5.3.2.1, "NIST-1 Facility," of the TR. The staff previously reviewed the applicability of the NIST-1 experiment data for validation of the NRELAP5 computer code for the NPM-160 design. The staff's conclusions are documented in non-LOCA TR-P-A, Revision 3 (ML20148M391). The staff reviewed the conclusions and finds them to be applicable to the NPM-20 design. For this reason, the staff decided not to include detailed discussion of the assessment in this SER. The staff reviewed these conclusions, and the information provided in the response to the RAI (ML19221B483), and finds they remain appropriate for the NPM-20 design because determining the time step size based on the material Courant limit is derived from the basic physical principle and there is no change in the NPM-20 dimensions compared to NPM-160.

The L&C specified in the SER for Revision 3 of the TR (ML20148M391) states that if the NPM design changes significantly from what the staff has reviewed (e.g., MPS logic changes that impact non-LOCA transient progressions, reduced margin to acceptance criteria), additional justification would be needed to confirm that the application of a DHRS heat transfer bias is not necessary. The staff notes that this L&C remains valid because the TR remains the same as in Revision 3 with respect to relevant integral and separate effects tests for NRELAP5 code validation. Therefore, the NRC staff retained a modified L&C in Section 4 of this SER, "Limitations and Conditions," that an applicant or licensee seeking to apply this methodology to a design other than the design represented in the NPM model, Revision 4 (or any NPM model update made pursuant to a change process specifically approved by the NRC for changes to the NPM model) must evaluate SG and DHRS heat transfer biases to determine if the elimination of the biases within this methodology remains justified based on the margins to non-LOCA FOMs.

Sections 5.3.6 and 5.3.7 of TR Rev 4 are meant to provide additional benchmark comparisons to improve the validation basis **{{** **}}** as described in SER Sections 3.5.3.6 and 3.5.3.7.

The NRC staff finds that the NIST-1 SET model is consistent with the descriptions of the NIST-1 SETs and uses nodalization sufficiently similar to that used for the NPM non-LOCA application, in accordance with the guidance in RG 1.203.

3.5.3.2.1 NIST HP-03 Separate Effects Tests

TR Section 5.3.2.4, “HP-03 Test Description,” describes the NIST-1 HP-03 tests, which used a full-height DHRS heat exchanger to assess the ability of NRELAP5 to predict condensation within, and heat transfer across, the DHRS tubes. As discussed in the SER for non-LOCA TR-P-A, Revision 3 (ML20148M391), the NRC staff finds that the NIST-1 HP-03 SETs, together with the KAIST tests adequately cover part of the expected ranges of the DHRS operation because the NIST-1 HP-03 tests use DHRS pressure and temperature that are lower than that of the NPM-20 design. Nevertheless, the staff finds that the tests are still valid for the lower ranges of the NPM-20 DHRS pressure and temperature during DHRS operation; recognizing that the DHRS temperature and pressure may be higher than the NIST HP-03 tests at actuation and then go down as the system cools.

For primary to secondary heat transfer, the applicant referenced the assessment of NRELAP5 predictions of the SIET TF-1 (for secondary side heat transfer) and SIET TF-2 (for primary to secondary side heat transfer) tests. The NRC staff’s evaluation of the NRELAP5 assessment against the SIET tests is documented in Section 3.5.3.5, “Steam Generator Modeling,” Section 3.5.3.6, “Heat Transfer Correlation Comparison” and Section 3.5.3.7, “NIST-2 Steam Generator – Decay Heat Removal Systems Integral Effects Tests,” of this SER. However, the assessments considering primary-to-secondary heat transfer were limited in scope, ultimately resulting in Condition 4 in Section 4, “Limitations and Conditions,” of this SER. This Condition is meant to ensure that the exclusion of the SG heat transfer bias as part of the non-LOCA EM remains justified if an applicant or licensee applies the non-LOCA EM to a design other than the NPM NRELAP5 non-LOCA model Revision 4 or any NPM model update made pursuant to an NRC-approved change process for the NPM model. Further discussion of Condition 4 is also documented in SER Sections 3.5.3.5, 3.5.3.6, and 3.5.3.7.

With respect to heat removal mechanisms in the DHRS, the applicant stated that the one-dimensional cooling pool model **{{**

}} by bounding the pool temperature boundary condition in the non-LOCA plant analyses. **{{**

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As discussed in Section 3.6.1, “Thermal-Hydraulic Volumes and Heat Structures,” of this SER, the NRC staff finds this treatment acceptable.

TR Section 5.3.2.5.4, “HP-03 Summary,” summarizes the comparison of the NRELAP5 code predictions to the HP-03 test series data. The staff’s evaluation of the HP-03 setup and testing results is documented in the approved version of Revision 3 for this TR. The staff reviewed the test facility, compared with the reactor design, and finds that the test results remain appropriate for application to the NPM-20 design.

3.5.3.2.2 NIST HP-04 Separate Effects Tests

TR Section 5.3.2.6, “HP-04 Test Description,” describes the NIST-1 HP-04 test series performed to assess the ability of NRELAP5 to predict **{{**

}}. Like the HP-03 test setup, steam produced in the SG was routed to the simulated full-height DHRS, and the condensate line discharged to the environment. The HP- 04 test series consists of two runs at different DHRS pressures, as shown in TR Table 5-10, “NIST-1 HP04 test ranges.”

TR Section 5.3.2.7, “HP-04 Test Results,” discusses the HP-04 test series results at a high level. The applicant concluded that the NRELAP5 test simulations predicted the data with a reasonable-to-excellent agreement, acknowledging that NRELAP5 does not fully capture the CPV heat-up response. Despite this, the applicant stated that NRELAP5 can accurately predict the energy transfer from the DHRS to the CPV fluid. The staff’s review and conclusions on the NIST HP-04 Separate Effects Tests are documented in the SER for Revision 3 of the approved Non-LOCA EM (ML20148M391).

Section 3.5.3.2.2, “NIST HP-04 Separate Effects Tests,” for Revision 3 of the approved non-LOCA EM (ML20148M391) also describes the sensitivity studies that the applicant performed **{{**

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The staff reviewed the NIST HP-04 test facility and results and finds it to be sufficiently similar to the NPM-20 design and the test results are appropriate for the NRELAP5 code validation for analyses of **{{** **}}** phenomena. The SER for

Revision 3 of the non-LOCA EM (ML20148M391) provides the detailed staff evaluation of the descriptions and test results of the NIST HP-04 test facility.

3.5.3.3 NIST-1 Non-LOCA Integral Effects Tests

TR Section 5.3.3, "NIST-1 Non-LOCA Integral Test," discusses the NIST-1 facility non-LOCA IETs, which include NLT-02a, NLT-02b, and NLT-15p2. The objectives of these tests were, respectively: to measure the integral response to a loss of feedwater transient to the point of a reactor trip; to examine DHRS-driven cooling following the initial DHRS actuation; and to measure the integral response to a loss of feedwater transient and subsequent DHRS cooling.

TR Section 5.3.3.3, "NRELAP5 Model Description," describes the NRELAP5 model and provides NIST-1 nodalization schematics for the primary and secondary sides. The applicant compared the nodalization and **{{ }}** for NIST-1 and the NPM; and provided justification for the differences (ML18270A469). Based on this information, the NRC staff was able to confirm that the nodalization for the NIST-1 IET models is sufficiently similar to that of the NPM-20 model.

3.5.3.3.1 NIST-1 NLT-02a Test

TR Section 5.3.3.4, "NLT-2a Test Description," provides selected initial conditions and the sequence of events for the NLT-02a loss of feedwater test, and the test results are presented in TR Section 5.3.3.5, "NLT-2a Test Results." **{{**
}}. TR Section 5.3.3.5 compares the NRELAP5-calculated values for primary and secondary parameters against the test data for the first 150 seconds after feedwater flow interruption. Feedwater flow (TR Figure 5-41, "NLT-02a transient feedwater flow comparison"), core heater rod power (TR Figure 5-42, "NLT-02a transient core heater rod power comparison"), and steam line pressure (TR Figure 5-51) were boundary conditions for the NRELAP5 simulation. Primary pressure and core inlet temperature simulation results (TR Figures 5-43a, "NLT-02a transient pressurizer pressure comparison," 5-45a, "NLT-02a transient pressurizer level comparison," and 5-46a, "NLT-02a transient core inlet temperature") are within the data uncertainty bands and follow the trend of the data well, and therefore, the NRC staff agrees with the applicant that these parameters show a reasonable-to-excellent or excellent agreement with the test data. The applicant concluded that all other calculated parameters demonstrate reasonable agreement and, based on its review of the parameters, the NRC staff agrees.

The NRC staff noted that the riser mass flow rate (TR Figure 5-44a, "NLT-02a transient riser mass flow rate comparison") generally showed the least agreement of the parameters, as the prediction was outside the measurement uncertainty for the duration of the test. The NRC staff previously audited sensitivity studies, as described in the associated audit report (ML20036C849), that the applicant performed to assess **{{**

}}. The applicant also described (ML18270A466) modeling approaches associated with some level of uncertainty that could contribute to the overpredicted riser mass flow rate, such as **{{**
}}.

The NRC staff finds that NRELAP5 predicted the behavior of major parameters from the NLT-02a test reasonably well, which, in combination with the other IETs, demonstrates the ability of NRELAP5 to provide acceptable predictions of non-LOCA events.

3.5.3.3.2 NIST-1 NLT-02b Test

TR Section 5.3.3.6, "NLT-2b Test Description," describes the NLT-02b test, which was intended to investigate the integral plant response from DHRS actuation to DHRS-driven cooling and depressurization.

PZR level presented in TR Figure 5-55a, "NLT-02b phase 1 transient pressurizer level comparison") was calculated using Version 1.6 of the NRELAP5 code **{{**

}}. In addition, the NRC staff notes that similar trends are observed on the predicted DHRS power using NRELAP5 Version 1.7 **[OPEN ITEM]** as shown in Figure 5-65a and Figure 5-66a comparing to TR Figure 5-61, "NLT-02b phase 1 transient DHRS heat exchanger thermal power comparison," is **{{**

}}. Based on the comparisons between the calculated and test results, the NRC staff concluded that the **{{** **}}** DHRS condensate temperature (TR Figure 5-65a, "NLT-02b phase 1 transient DHRS condensate temperature comparison") and **{{**

}} were sufficiently addressed by the applicant, as described in Section 3.5.3.2.1, "NIST HP-03 Separate Effects Tests," of this SER. On these bases, pending closure of the above noted open item, the staff determined that this conclusion remains valid for the NPM-20 design.

TR Section 5.3.3.8, "NLT-2b Phase 2 Test Results," compares NLT-02b Phase 2 test results to NRELAP5 predictions. Phase 2 spans the period of **{{**

}}. The NRC staff agrees with the applicant that most key parameters show a reasonable-or-better agreement between the predictions and data for Phase 2. Like Phase 1, the Phase 2 CPV and condensate temperatures (TR Figures 5-87a, "NLT-02b Phase 2 transient cooling pool vessel region 5 temperature comparison (near bottom of DHRS heat exchanger)," to 5-88, "NLT-02b phase 2 transient cooling pool vessel Region 7 temperature comparison (just above the decay heat removal system heat exchanger tube region)," and 5-82a, "NLT-02b Phase 2 transient decay heat removal system condensate temperature comparison," respectively) **{{**

}}. The predicted DHRS condensate flow (TR Figure 5-83a, "NLT-02b phase 2 transient decay heat removal system condensate flow comparison"), and consequently, the DHRS power (TR Figure 5-79, "NLT-02b Phase 2 transient decay heat removal system heat exchanger thermal power comparison"), **{{** **}}** but still exhibit a reasonable agreement with the data.

The overall trends observed between NRELAP5 v1.6 and 1.7 remain similar. On these bases, pending closure of the noted open item, the staff determined that this conclusion remains valid for the NPM-20 design. **[OPEN ITEM]**

TR Section 5.3.3.9, "NLT-2b Phase 3 Test Results," describes NLT-02b Phase 3. The staff's evaluation of NLT-2b Phase 3 Test is documented in the SER for non-LOCA-P-A, Revision 3 (ML20148M391). The staff reviewed the SER and finds that the conclusions made in the review of Revision 3 of this TR remain applicable to the NPM-20 design because between the minor geometric changes to the DHRS loop and despite the substantial core thermal power increase for NPM-20 over NPM-160, the DHRS is a natural circulation flow driven by the heat input into it. Within reasonable limits, the condensate level in the condenser tubes adjusts to the applied steam pressure such that the DHRS will transfer heat to the reactor pool at a rate in proportion to the rate heat is supplied to it.

3.5.3.3.3 NIST-1 NLT-15p2 Test

TR Section 5.3.3.12, "NLT-15-p2 Test Description," describes the NIST NLT-15p2 integral test of a loss of feedwater event leading to actuation of the DHRS. During this test, **{{**

}}. TR Section 5.3.3.13, "NLT-15 p2 Test Results," provides the test results and NRELAP5 predictions.

The applicant stated that predicted primary pressure is in reasonable agreement with the data near the beginning of the event when peak pressures occur (TR Figure 5-127, "NLT-15-p2, transient RPV pressure short term"). **{{**

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The results show that the predicted values for PZR level (TR Figure 5-129a, "NLT-15-p2, transient pressurizer level") and RPV level (TR Figure 5-130a, "NLT-15-p2, transient RPV level") are in excellent agreement with the data. The applicant deems the agreement in riser flow (TR Figure 5-131, "NLT-15p2, transient riser mass flow rate") as reasonable **{{**

}}. Predicted RPV loop temperatures (TR Figures 5-133a, "NLT-15-p2, transient core inlet temperature," through 5-134a, "NLT-15-p2, transient upper plenum temperature") are in reasonable to excellent agreement with the data.

The results also show that the peak SG pressure (TR Figure 5-135, "NLT-15-p2, transient secondary side pressure - 0 to 500 seconds") was **{{**

}}. The applicant stated that predicted SG (TR Figures 5-145, "NLT-15-p2, transient steam generator tube coil level - long term," and 5-146, "NLT-15-p2, transient steam generator tube coil level - short term") and DHRS (TR Figures 5-138, "NLT-15p2, transient DHRS HX level - 0 to 500 seconds," and 5-144, "NLT-15-p2, transient DHRS HX level,") levels showed reasonable agreement with the data. The NRC staff notes that **{{**

}}. The applicant judged the NRELAP5 predictions for differential pressures across the DHRS condensate line (TR Figure 5-147, "NLT-15-p2, transient DHRS condensate line differential pressure") and steam line (TR Figure 5-148, "NLT-15-p2, transient DHRS steam line differential pressure") to be **{{**
}}. The NRC staff agrees with the applicant's assessment of these parameters because **{{**

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The simulated DHRS loop mass flow rate (TR Figures 5-142, "NLT-15-p2, transient DHRS loop flow - short term," and 5-143a, "NLT-15-p2, transient DHRS loop flow rate - long term") **{{**

}}. The simulated SG power (TR Figure 5-149a, "NLT-15-p2, transient steam generator tube coil power removal") and DHRS power (TR Figure 5-150a, "NLT-15-p2, transient DHRS power removal") **{{**

}}. Therefore, the NRC staff agrees with the applicant that the predicted DHRS mass flow, SG power, and DHRS power show reasonable agreement with the data. As discussed previously for the NLT-02a tests and NLT-02b tests, NRELAP5 did not capture the CPV temperature profile, but this does not affect prediction of DHRS heat removal.

3.5.3.3.4 NIST-1 Integral Effects Tests Summary

TR Section 5.3.3.11, "NLT-2 Summary," summarizes the results of NRELAP5 assessments against the NIST-1 NLT-02 tests. The applicant concluded that NRELAP5 can reasonably predict primary heatup and pressurization resulting from a loss of feedwater, as supported by comparisons against NLT-02a. The applicant also concluded, based on comparisons to NLT-02b, that the code can predict the heat transfer from the primary side to the SG and from the DHRS to the CPV with reasonable to excellent agreement. The applicant described parameter predictions that were not in good agreement with the data but concluded that the important parameters could be reasonably calculated within the limitations of the NRELAP5 computer code. The staff's evaluation of the NRELAP5 computer code for predicting primary side heatup and pressurization resulting from a loss of feedwater is documented in the SER for Revision 3 of the non-LOCA EM (ML20148M391). The staff reviewed the SER and finds that the conclusions made remain applicable to the NPM-20 design. The NIST-1 tests verify the EM against the various phenomena identified in the PIRT. Since the PIRT for the NPM-160 was confirmed to be applicable to the NPM-20, the NIST-1 tests are applicable to the NPM-20.

Additionally, DHRS flow is driven by the heat input to it and any geometric changes to DHRS between NPM-160 and NPM-20 seem to have had insignificant inpact.

3.5.3.4 Code-to-Code Benchmark for Integral Assessment of Reactivity Event Response

TR Section 5.3.4, "Code-to-Code Benchmark for Integral Assessment of Reactivity Event Response," describes the code-to-code benchmark against the RETRAN-3D code that the applicant performed primarily to assess the performance of the NRELAP5 point kinetics model and to supplement the assessment of NRELAP5 primary side thermal-hydraulic response for reactivity transient events.

RG 1.203 recognizes that code-to-code comparisons can be useful in code assessment with key limitations as follows:

"For some plants and transients, code-to-code comparisons can be very helpful. In particular, if a new code or device is intended to have a limited application, the results may be compared to calculations using a previous code. However, the previous code should be well-assessed to integral or plant data for the plant type and transient being considered for the new device. Differences in key input (such as system nodalization) should be explained so that favorable comparisons provide the right answers for the right reasons. Such benchmark calculations would not replace assessment of the new code."

The staff agrees with the applicant that the previous assessment on the adequacy of the point kinetic model for modeling NPM-160 is acceptable for the NPM-20 design because the power increase and possible larger flux gradients do not impact the system's response to reactivity change, which is a global parameter. The SER for non-LOCA TR, Revision 3 (ML20191A285) provides more detailed discussions on validation of the NRELAP5 computer code that the staff finds to be valid for the NPM-20 design.

3.5.3.5 Steam Generator Modeling

The NRELAP5 code validation for the HCSG was accomplished as part of the LOCA EM, with testing performed at the SIET facility and other legacy experiments. Section 5.3.5, "Steam Generator Modeling," of the TR describes the applicant's assessment of the NRELAP5 HCSG model for performing NPM non-LOCA analyses. This assessment is an extension of that performed in the LOCA TR against the SIET TF-1 and TF-2 tests. Since the description of the SIET facility, tests, test data, and model-to-data comparisons are provided in Section 7.4, "NuScale SIET Steam Generator Tests," of the LOCA TR (Reference 8), none are presented in non-LOCA TR. Furthermore, the applicant showed that the decay heat transfer from the HCSG to the DHRS is generally dominated by opening of the ECCS in the LOCA EM applications. However, this is the only mode of decay heat removal in the non-LOCA EM applications. Therefore, the NRC staff's review of the NRELAP5 HCSG heat transfer model validation and its applicability to the non- LOCA methodology includes review of Sections 6.7, "Helical Coil Steam Generator Component," and 7.4 of the LOCA TR, as applicable to the non-LOCA methodology.

The NIST-2 testing primarily focused on SG-DHRS performance to enhance NRELAP5 code validation for DHRS flow and heat transfer. The applicant used its previous evaluation of the SG on the NPM design (i.e. NIST-1 DHRS separate effects testing) to justify modeling the NPM-20 SG with NRELAP5. The applicant affirms the applicability of the NRELAP5 key physical models for SG and DHRS heat removal during non-LOCA events for the NPM-20 because the PIRT was unchanged following the NIST-2 DHRS integral effects tests, despite initial concerns the applicant had about certain changes to the design of DHRS (e.g., relative geometries such as lengths of pipes) potentially affecting DHRS-relevant PIRT. Based on the staff's previous review of DHRS testing (NIST-1) and the review of the applicant's documentation and because NPM-160 and NPM-20 DHRS are mostly similar, the staff agrees that the previous testing as described below is applicable to the NPM-20 design. Additionally, some NIST-1 NRELAP5 simulations were updated to use Version 1.6 of NRELAP5 for the revision of the non-LOCA TR this version of the SE focuses on, adding credibility to the use of the code for NPM-20 designs based on previously conducted experiments.

Section 6.7, “Helical Coil Steam Generator Component,” of the LOCA TR states that a hydrodynamic component (designated as “HLCOIL”) and heat transfer package were added to NRELAP5 for modeling pressure drop and heat transfer on the secondary side of the SG. The HLCOIL component applies helical coil friction factor models that are summarized in Section 6.7.1, “Helical Coil Tube Friction,” of the LOCA TR. The helical coil single and two-phase friction factor correlations applied inside the SG tubes (corresponding to boundary condition **{{**

}}. Secondary side laminar and turbulent heat transfer correlations for single-phase flow discussed in Section 6.7.2.1, “Helical Coil Single-Phase Heat Transfer,” of the LOCA TR **{{**

}}. As described in Section 6.7.2.2, “Helical Coil Two-Phase Subcooled and Saturated Flow Boiling Heat Transfer,” of the LOCA TR, two-phase subcooled and saturated boiling heat transfer are **{{**

}}. The applicant stated (ML18002A610) that the primary side heat transfer correlation (corresponding to boundary condition **{{**

}}) cannot not accurately represent the physical phenomenon of the heat transfer on the primary side of the HCSG and therefore shall not be used in safety analyses for NPM modules. This condition is reflected in L&C No. 7.

The non-LOCA TR specifies that the NRELAP5 HLCOIL component is used to model the HCSG, and the NRELAP5 heat structure geometry options **{{** **}}** are used for the primary and secondary, respectively. The applicant assessed these models and correlations against experimental data, as described in Section 7.4, “NuScale SIET Steam Generator Tests,” of the LOCA TR.

Sections 5.3.5.1, “Background,” and 5.3.5.2, “Helical Coil Steam Generator Modeling,” of the non-LOCA TR reference Section 7.4, “NuScale SIET Steam Generator Tests,” of the LOCA TR. TR Section 5.3.5.3, “Helical Coil Steam Generator Operating Ranges vs. Validated Ranges,” compares the operating ranges for some key SG parameters to the validated ranges in NRELAP5 and notes that **{{**

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The NRC staff finds that, **{{**

}}. Therefore, the NRC staff accepts application of the SIET results reported in the LOCA TR for evaluating the NRELAP5 SG model for the non-LOCA transient analysis.

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Section 7.4.1, "SIET Tests," of the LOCA TR discusses the SIET TF-1 tests and assessment of the NRELAP5 predictions against test data. The TF-1 tests included adiabatic and diabatic tests to assess flow inside the SG tubes. During the diabatic tests, the coils were electrically heated, with three separate heating zones in the axial direction. The applicant concluded that the predicted pressure drops, wall temperatures, and fluid temperatures along the tube are in reasonable to excellent agreement with the TF-1 test data. Based on its review of the figures provided in Section 7.4.1 of the LOCA TR, the NRC staff agrees with the applicant's assessment. In addition, the NRC staff's confirmatory calculations using the TRACE code showed reasonable to excellent agreement with the TF-1 data and support the applicant's TF-1 conclusion noted above.

Section 7.4.2, "SIET Fluid-Heated Test," of the LOCA TR discusses the SIET TF-2 tests and their use to assess the NRELAP5 SG model. The SIET TF-2 tests were performed to validate NRELAP5 primary-to-secondary side SG heat transfer and primary side SG loss coefficients. Based on staff concerns regarding test primary side flow rates, the applicant **{{ }}** (ML18194A749, ML18228A817). The applicant justified the validity of the **{{ }}** TF-2 validation tests by performing a primary-to-secondary side heat balance assessment demonstrating the tests' acceptability.

While compensating errors during calculation of primary and secondary side heat balances that might mask errors in primary flow are possible, the NRC staff observed no such errors during its review of the revised TF-2 assessment data.

While the TF-2 facility consisted of five tube banks representing the **{{**

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The NRC staff agrees that the TF-2 test data-to-model comparisons presented in the LOCA TR are in reasonable to excellent agreement. However, due to the concerns and potential limitations noted above, the NRC staff could not confirm that the TF-2 tests fully represent NPM steady-state and non-LOCA transient conditions or that the SG heat transfer coefficient biases were appropriately conservative for non-LOCA events. To address the question of SG heat transfer biasing, the applicant performed a series of SG heat transfer sensitivity analyses and evaluated the resultant changes relative to the FOMs for the five non-LOCA transient classes (increase in heat removal from the secondary, decrease in heat removal from the secondary, reactivity and power distribution anomalies, increase in reactor coolant inventory, and decrease in reactor coolant inventory) (ML19212A796).

For the five increase in heat removal from secondary events considered, the key FOM is MCHFR. The applicant calculated a **{{**

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{{

}} over 40 percent MCHFR margin compared to the 95/95 limit for the limiting overcooling event in the NuScale SDA when using the subchannel methodology.

For the six decrease in heat removal from the secondary events considered, the key FOMs are primary and secondary pressures. Because the reactor safety valves (RSV) lifts to mitigate high reactor pressure, **{{**

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For the reactivity transients, the applicant concluded that **{{**

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For the increase in RCS inventory event due to CVCS malfunction, primary and secondary pressures are the FOMs of interest. The applicant's calculations resulted in conclusions similar to those for the decrease in heat removal events. For the two decrease in RCS inventory events, the limiting FOM is dose. The applicant found **{{**

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Based on its review of the sensitivity study information, as confirmed in audits and described in the associated audit report (ML20036C849), the NRC staff agrees that the FOMs for non-LOCA events are insensitive to reasonable variations in SG heat transfer for the design reflected in NPM model Revision 2, which was the latest model revision at the time those calculations were performed. However, the staff has limited the application of this methodology to NPM-20 model Revision 4 (or updates made pursuant to an approved change process), as further discussed below. For the same reasons discussed in Section 3.5.3.2 of this SER, "NIST-1 Decay Heat Removal System Separate Effects Tests," with regard to DHRS heat transfer, and because of the minimal effect of the riser holes on steady-state parameters, the staff concluded that the NPM model changes between Revisions 2 and 3 do not change the SG heat transfer conclusions. Therefore, the staff finds that the applicant's sensitivity studies using NPM model Revision 2 are sufficient to support the lack of a SG heat transfer bias as part of the non-LOCA EM.

For post-trip heat removal, the effect of the SG heat transfer uncertainty is minimal since the DHRS heat exchanger capacity is the limiting factor. The heat transfer surface area of the DHRS is **{{** **}}**, so the heat transport capability of the DHRS is much less than that of the SG, consistent with the requirements to remove decay power versus full power.

The NRC staff notes that the effect of SG heat transfer on normal operations (steady-state initial conditions) is addressed by the applicant's technical specifications (TS) in the NuScale SDAA

FSAR, Chapter 16, which are based on the values supported by the safety analysis, specifically SDAA FSAR Table 15.0-6, “Module Initial Conditions Ranges for Design Basis Event Evaluation.”

Based on the relative lack of sensitivity of the non-LOCA transient analyses FOMs to variations in SG heat transfer, the post-trip DHRS heat removal capability, and TS providing the permissible range of primary temperatures for steady state operation, the NRC staff finds the application of a NRELAP5 SG heat transfer coefficient uncertainty unnecessary for the NPM design reflected by NRELAP5 non-LOCA model Revision 4 (noting that input file version numbers were reset to 1 for the US460 SDAA NPM design). NRC will require additional justification be provided to ignore SG heat transfer biases if the NPM design is updated (including, but not limited to, design or MPS logic changes) such that margins to non-LOCA FOMs decrease relative to those in the NuScale SDA. This is reflected in Condition 4 in Section 4, “Limitations and Conditions,” of this SER.

In summary, the NRC staff finds that the applicant has implemented appropriate HCSG models in NRELAP5, and the NRELAP5 predictions of SIET tests show reasonable to excellent agreement to the data. However, the assessments considering primary-to-secondary heat transfer were limited in scope, ultimately resulting in the condition described above.

3.5.3.6 Heat Transfer Correlation Comparison

For the NPM-160, the staff previously audited (ML19039A090) the CFD calculations and found that the applicant analyzed several cases for comparison to literature heat transfer correlations (i.e., **{{ }}**) as well as for parameter and mesh size sensitivity evaluation.

In TR Section 5.3.6.3 the applicant states that NRELAP5 Version 1.7 provides the option of using either the ESDU correlation in stand-alone form or in RSS combination with the Dittus-Boelter correlation. The staff discussed the ESDU correlation in combination with the Dittus-Boelter correlation with the applicant during its audit review and both agreed that this option may not accurately represent the actual physical heat transfer phenomena in the helical coil SG and therefore should not be used.

Based on the above evaluations, the staff finds that the ESDU correlation provided in NRELAP5 Versions 1.6+ is acceptable, when applied to the exterior of the helical coils, for performing analyses for non-LOCA events for the NMP-20 design.

3.5.3.7 NIST-2 Steam Generator – Decay Heat Removal System Integral Effects Tests

The applicant performed tests for the performance of the DHRS under Non-LOCA events. To assess and validate the NRELAP5 code in predicting SG-DHRS heat transfer behavior, the applicant developed the NIST-2 testing facility. The NIST-2 facility is an upgrade to the NIST-1 facility, as discussed in Section 3.5.3.3 of this SER. The key upgrades were to increase the maximum allowable pressure for the main steam and DHRS piping systems to perform scaled separate and integral effects tests for the NRELAP5 code that is used for NPM-20 safety analyses. When compared to NIST-1, the NIST-2 facility test conditions are closer to the NPM-20 operational conditions. Per the Non-LOCA PIRT discussion in Section 3.5.1 of this SER, several high ranked phenomena are identified to have particular relevance to DHRS operation in non-LOCA Phase 3 (stable natural circulation phase). The staff reviewed the description of the NIST-2 tests (ML24215A251) to evaluate the DHRS performance during LOCA and non-LOCA transients. The NIST-2 test assessment for LOCA events are primarily discussed in Section 4.7.5 of the LOCA TR SER (see Reference 8)). In this section, the NRC staff focused on the NIST-2 non-LOCA SG-DHRS performance during a limiting loss of feedwater (LOFW) transient because this event is related to the performance of the secondary side of the SG which also serves as the heat removal function of the DHRS. Besides Section 5.3.7 "NIST-2 SG-DHRS Integral Effects Tests" of the non-LOCA TR, staff also audited (ML24215A240) the engineering calculation packages to assess the applicability, including scalability and distortion, of the test facility and the testing **{{**

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The staff audited (ML24262A257) the IET data evaluation report on the comparison NRELAP prediction of DHRS performance with data for a few transient runs in NIST-2. Some distortions were observed. For example, **{{**

}}. Several studies were provided to address the staff's concern on the similarity between NIST-2 and NPM. The staff discovered that in the applicant's NIST-2 model, {{

}} (ML24215A242). The staff issued an {{

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{{

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3.5.4 Conclusions of NRELAP5 Applicability for Non-LOCA

TR Section 5.4, "Conclusions of NRELAP5 Applicability for Non-LOCA," summarizes the applicant's conclusions regarding the applicability of the NRELAP5 Version 1.7 computer code to the non-LOCA transient analyses. The applicant concluded that, based on the highly ranked non-LOCA phenomena and the various methods used to address them, NRELAP5 is applicable to the non-LOCA analysis. Based on the evaluations in the preceding subsections, the NRC staff finds, pending resolution of the open item, **[OPEN ITEM]** that the applicant has adequately addressed the phenomena important to non-LOCA events and has demonstrated that NRELAP5 is an acceptable tool for non-LOCA event analysis, subject to the limitations and conditions in Section 4 of this evaluation.

3.6 NuScale NRELAP5 Plant Model

TR Section 6, "NuScale NRELAP5 Plant Model," describes how NPM plant components and processes are modeled with NRELAP5 for non-LOCA transients. The descriptions cover modeling of the reactor primary and secondary (SG) systems, fuel, ECCS, DHRS, CNV, reactor pool, and protection and control systems. TR Section 6 also provides modeling of the reflector bypass region and modeling of the lower and upper riser holes. The TR further states that an equivalent modeling approach for the safety valve may be used provided that the valve capacity is conservatively modeled in the analysis.

NRC staff noted in multiple instances that plant descriptions are written to apply to the NPM-20 or NPM-160 design. Additionally, several figures showing model nodalization were not updated from Revision 3 to Revision 4 of the topical report, and still show features that are specific to the NPM-160 design, such as inclusion of three RVVs. However, during the review NRC staff noted several changes to the topical report methodology that are based on the NPM-20 design and raised questions regarding the applicability of the methodology to the NPM-160 design. Although the approved version of the topical report includes these figures representing the NPM-160 design and generic description in the topical report, Revision 4 of this methodology was reviewed and approved for application to the NPM-20 design, as specified in L/C # 1.

3.6.1 Thermal-Hydraulic Volumes and Heat Structures

TR Section 6.1, “Thermal-Hydraulic Volumes and Heat Structures,” describes the thermal-hydraulic components, heat structures, and junctions in the NRELAP5 plant model. It also provides multiple NRELAP5 component diagrams.

Figure 6-2, “Typical primary and secondary side model (heat structures and component cell details excluded),” presents a “typical” component system diagram that is meant to convey the overall structure of the non-LOCA NRELAP5 model. **{{**

}} The NRC staff finds, consistent with the guidance in RG 1.203, that the component systems, and nodalizations described in the TR provides an acceptable description of the model used in the EM.

Section 6.1, “Thermal-Hydraulic Volumes and Heat Structures,” of the TR states that **{{**

}}. The NRC staff agrees with the applicant’s assessment that **{{**

}}. The staff audited the NPM-20-updated CFD calculation for flow and heat transfer in the primary; the conclusion about safely neglecting some of the riser-internals masses in the NRELAP5 model remains valid because these NPM aspects were effectively unchanged between NPM-160 and NPM-20.

The NRC staff noted that CHF is directly affected by natural circulation flow. Discussion of the high importance phenomena **{{**

}} listed in TR Table 5-3, “High-ranked phenomena for non-LOCA events,” states that **{{**

}}. The NRC staff finds that the effect on CHF margin is conservative **{{**

}}.

TR Section 6.1.1, “Reactor Primary,” describes the NRELAP5 representation of the primary fluid volumes and heat structures. The HCSG is unique to the NuScale reactor design and differs from those of conventional PWRs. **{{** **}}**. The NRC staff finds that **{{** **}}** are adequate to represent primary flow and heat transfer through the SG shell, as long as the axial nodal resolution is sufficient to capture the temperature change through the SG; the NRC staff confirmed that the SG model described in the TR is adequate for this purpose.

Section 7.4.2.4, “Special Analysis Techniques,” of the LOCA TR specifies that **{{** **}}**, which the applicant benchmarked against the adiabatic TF-2 test data and concluded that provided a good prediction of the pressure change due to flow across the tube bundles on the primary side. Based on the agreement between the TF-2 results and the NRELAP5 predictions, the NRC staff agrees that the **{{** **}}** are acceptable for simulating similar NPM SG form losses. The NRC staff notes that the **{{** **}}** was developed with data from liquid flows and should not be applied to gas or two-phase flow conditions across the primary side of the SG. Because the primary side analytical limits specified by the applicant in the SDA preserve a 5 delta-degree F subcooling margin through the MPS high hot-leg and low PZR pressure trips, the NRC staff finds that the **{{** **}}** is applicable under normal operation and non-LOCA events.

{{ **}}**.

As discussed in Section 3.5.1, “Non-LOCA Phenomena Identification and Ranking Table,” of this SER, several non-LOCA highly ranked phenomena that are identified in Section 5.1.4, “Highly Ranked Phenomena,” of the non-LOCA TR, including **{{** **}}**, are not reflected in the NRELAP5 non-LOCA EM representation of the NPM**.** Table 5-3, “High-ranked phenomena for non-LOCA events,” of the TR states that the phenomena are addressed by the subchannel analysis except for **{{** **}}**. The NRC staff confirmed through audit for the US600 DCA, as documented in the associated audit report ML19039A090, that those highly ranked phenomena are not relevant to the non-LOCA TR except for **{{** **}}** and are instead applicable to other portions of the non-LOCA EM (e.g., the subchannel analysis), and, the staff position remains the same for the NPM-20.

Section 6.1.1, “Reactor Primary,” of Revision 3 of the TR, stated that **{{** **}}.** The NRELAP5 modeling for US460 design, along with Revision 4 of the TR, have been modified to combine the lower and upper riser sections that were named at the start of this paragraph. As indicated above, several highly ranked phenomena identified by the applicant are related to multi-dimensional flows and complex flow behavior **{{**

}}.

For the US600 design certification application (DCA), the applicant described (ML18234A537) expected multi-dimensional flow and thermal behavior in **{{**

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Based on its review of the information and the audit for the NPM-160, the NRC staff found that **{{** **}}** was adequately addressed by the non-LOCA EM for the NPM-160 design.

While a 1-D model such as that used in NRELAP5 simulations of an NPM could not simply be expected to accurately model a 3-D flow, the 1-D NRELAP5 models have been adjusted using loss coefficients such that the steady state primary loop mass flow rate computed comports with what has been observed in the various physical experiments. In this way, the 1-D NRELAP5 models implicitly account for the 3-D reality of primary coolant flow in an NPM. The Staff reviewed updated CFD calculation results documentation for NPM-20, which are much more detailed than for the DCA audit response, and finds that **{{**

}}; the natural circulation mechanism automatically adapts flow speed to prevent a net change in stored thermal energy in either the core region or the SG region**.**

According to TR Section 6.1.1, “Reactor Primary,” **{{**

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TR Section 6.1.2, “Core kinetics,” discusses the core kinetics in the NRELAP5 plant model of the NPM. The TR states that the non-LOCA decay heat model is in accordance with the 1973 American Nuclear Society (ANS) standard. As discussed in Section 3.7.1, “General Aspects of Non-LOCA Methodology,” of this SER, the NRC staff finds use of the 1973 ANS decay heat standard, in conjunction with bounding decay heat multipliers and appropriate actinide contribution, to be acceptable for use in non-LOCA analyses.

TR Section 6.1.3, “Fuel rod design input,” discusses the fuel rod design input used in the NRELAP5 plant model of the NPM. The fuel rods are modeled similar to those in typical large PWRs and use interface data from fuel performance codes. The core power distribution to be used for the non-LOCA transient analysis is based on peaking factors selected to trend reasonably with the downstream subchannel analysis (ML24348A048) with power distributed solely in the fuel pellet as specified by the applicant (ML24348A048 Public, ML24348A049 Non-Public), which the NRC staff finds to be acceptable, as discussed in Section 3.4.3.1, “Develop Plant Base Model NRELAP5 Input,” of this SER.

TR Section 6.1.4.1, “Feedwater System,” discusses the NRELAP5 representation of the feedwater system. The NRC staff finds that this description adequately represents the NuScale design, and the modeling of the feedwater system is therefore, acceptable.

TR Section 6.1.4.2, “Steam Generator Secondary,” discusses the NRELAP5 representation of the SG secondary side. The NRC staff finds that this description adequately reflects the NuScale design and the modeling of the SG secondary system side, and is therefore, acceptable.

TR Section 6.1.4.3, “Main Steam System,” describes the NRELAP5 model of the main steam system in the NPM plant model. **{{**

}}. The NRC staff finds that the description adequately reflects the NuScale design and is therefore, acceptable.

Section 6.1.5, “Decay Heat Removal System,” of the TR describes the NRELAP5 DHRS model in the NPM plant model. Figure 6-13, “Typical decay heat removal system division 1 model,” shows the NRELAP5 component diagram for DHRS division 1. The TR states that **{{**

}}.

The TR also states that **{{**

}}. The staff finds this acceptable.

During the review of US600 DCA, the NRC staff reviewed the results of the applicant’s sensitivity studies (ML18234A521), which used a simplified DHRS model in steady-state mode and a representative loss of ac power transient to assess the impacts of pool heat sink boundary condition modeling. The simplified DHRS model sensitivities concluded:**{{**

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{{

}} applicant's consideration of the effect of pool temperature and thermal stratification on the performance of the DHRS is acceptable; considering the similarities of DHRS design between NPM-160 and NPM-20, staff does not find a reason to re-evaluate the magnitude of scalar heat flux argument that was made for the NPM-160 finding on the topic of pool thermal stratification.

The applicant uses the **{{** **}}** to calculate the pool boiling heat transfer coefficient external to the DHRS in the cooling pool. The applicant submitted a justification (ML18299A296) for the use of **{{** **}}** under the condition of pool boiling, since the **{{** **}}** was not developed for pool boiling applications. The applicant described the components of the NRELAP5 implementation of **{{**

}}
{{ **}}**. For the US460 SDAA, staff noted NuScale's existing response to "eRAI No. 9466," (ML18128A341) which was submitted during the US600 DCA review. Considering the NRELAP5 non-LOCA model for NPM-20 still uses the same correlations, and the code is the same (in terms of relevant algorithm(s) for picking correlations) as it was for US600 DC review, staff has no reason to re-evaluate these findings that were made for the US600 DC.

In a sensitivity study for a representative loss of ac power event (US600 DCA review; "eRAI No. 9466," ML18128A341), the applicant compared the results using the **{{** **}}**.
There was no difference in the peak RPV pressure, and only a very small variation in the peak SG peak pressure **{{**

}}.

The NRC staff reviewed the sensitivity studies (US600 DCA review; "eRAI No. 9466" ML18128A341) and finds that the use of the **{{** **}}** is acceptable, since the **{{** **}}** incorporates the **{{** **}}** and sensitivities for a representative non-LOCA event demonstrate that there is little difference in the peak primary pressure, peak secondary pressure, and transient progression when using the **{{**

}}. While this decision was made with respect to the US600 design, the staff has no reason to re-evaluate this finding for the US460.

TR Section 6.1.6, "Emergency Core Cooling System," describes the modeling of the ECCS in the NPM plant model. **{{**

}}. The applicant did not describe certain NPM-20 specific features that do not affect the progression of non-LOCA events prior to establishment of long-term cooling, such as the ESB feature of the ECCS (ML24348A048 Public, ML24348A049 Non-Public). The NRC staff finds the ECCS modeling acceptable.

TR Section 6.1.7, "Containment Vessel," discusses the NRELAP5 model for the containment in the NPM plant model. NRELAP5 **{{**

}}. The NRC staff finds this description adequately reflects both the NPM-20 design and the NRELAP5 model of NPM-20 and is therefore acceptable.

TR Section 6.1.8, "Reactor Cooling Pool," discusses the NRELAP5 representation of the reactor cooling pool. **{{**

}}. Since the sensitivity of transients to changing pool temperature can be accounted for by running multiple cases, of a given transient, with different reactor pool constant temperature, the NRC staff finds this simplified pool heat up model acceptable.

The NRC staff finds that the description of the NuScale NRELAP5 Plant Model provided in Revision 4 of TR Section 6.1, "Thermal-Hydraulic Volumes and Heat Structures," is a sufficient description of the model.

3.6.2 Material Properties

TR Section 6.2, "Material Properties," discusses the thermal conductivity and volumetric heat capacity associated several materials used in the heat structures. It states that the material properties will be amended as the NPM design evolves; staff note that, at the time of this writing, the non-LOCA basemodel (r1) contains one additional material (316 steel) than listed in the current Revision (4) of non-LOCA EM. The NRC staff finds this acceptable as the method specifies that the material properties used in the model will reflect the operating plant, and the details in this section also reflect the NPM-20 design. **{{**

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3.6.3 Control and Protection Systems

Section 6.3, “Control Systems,” describes the NPM control and protection systems that are modeled in the NRELAP5 non-LOCA EM. In general, control and protection functions are accomplished through trips, control functions, and user-specified tables. The non-safety-related MCS consists of the PZR pressure control (i.e., heaters and spray), CVCS control, RCS temperature control, steam pressure control, feedwater and turbine load control, and containment pressure control functions, and is briefly described in Section 6.3.1, “Module Control System (Nonsafety-related),” of the TR. Section 6.3.2, “Module Protection System (Safety-related),” of the TR describes the safety-related MPS, including the use of analytical limits and fixed delay times. The TR lists a representative list of MPS functions and signals for the NPM. The list helps to illustrate how the MPS logic is implemented within the non-LOCA EM. The NRC staff review of the acceptability of MPS signals, the associated analytical limits, and time delays is performed as part of a design-specific application of the non- LOCA EM, such as the NuScale SDA.

3.7 Non-LOCA Analysis Methodology

3.7.1 General Aspects of Non-LOCA Methodology

TR Section 7, “Non-LOCA Analysis Methodology,” describes the NuScale non-LOCA analysis methodology. Section 7.1, “General,” provides the general non-LOCA analysis methodology, including the list of typical initial conditions; the typical initialization process; the general process for treating plant controls, loss of power, and single failures; the process for treating reactivity parameters; the biasing of other analysis parameters; and typical MPS signals and associated analytical limits and time delays.

TR Section 7.1.1.2, “Identification of Relevant Parameters,” discusses the list of initial conditions developed for the non-LOCA transient analyses. TR Table 7-1, “Typical list of initial conditions considered,” provides a typical list of initial conditions that are considered for the non-LOCA transient analysis, including parameters directly input to NRELAP5 and calculated parameters that are “target” parameters established during code initialization. TR Section 7.1.1.3, “Prioritization of Initial Conditions,” describes the prioritization of the initial conditions. As part of the steady state initialization, the important parameters are to be checked to confirm that they are within the allowable target value range or that the parameter conservatively bounds the target, and that the parameters are within the acceptable tolerances. A parameter that is not important may or may not be checked during the steady state initialization. TR Section 7.1.1.4, “Typical Initialization Process,” provides a list of the critical parameters necessary to establish the desired steady-state condition and describes the conditions for achieving a steady-state.

After a successful steady state simulation, a “null transient” is performed, which corresponds to a restart of the steady state with biased initial conditions. **{{**

}}. The NRC staff finds the null transient process used to establish biased NRELAP5 stable, steady-state, initial conditions for non-LOCA transient analyses reasonable and acceptable based on standard industry practice and as confirmed by audit discussions on the bias application methodology (ML19039A090). The staff finds that the conclusions remain valid for Revision 4 of this TR because the methodology for identifying the relevant parameters of the initial conditions remains the same for the NPM-20 design.

TR Section 7.1.2, “Treatment of Plant Controls,” discusses the treatment of normal, non-safety related PCSs in the NRELAP5 non-LOCA analyses based on their impact on the calculated consequences relative to the acceptance criteria. The applicant states that PCS operation is disabled if it would lead to a less severe transient response, while PCS operation is enabled if it leads to more severe consequences. The NRC staff finds this to be a conservative, and therefore acceptable, approach. The NRC staff confirmed that the PCS functions considered for non-LOCA transient analyses are consistent with the PCSs that are part of the current NPM design. The column entitled “Basis” in the event-specific tables entitled “Initial conditions, biases, and conservatisms” in Section 7.2, “Event Specific Methodology,” provide the operational assumptions for the PCS. Assessment of the event-specific PCS performance conditions is performed as part of the event-specific methodology evaluations in Section 3.7.2, “Event Specific Methodology,” of this SER.

TR Section 7.1.3, “Loss of Power Conditions,” discusses the loss of AC and DC power. The applicant states that the natural circulation flow in the NPM makes the loss of power less important in the NPM design compared to a conventional PWR. The NPM design thereby eliminates the need to consider loss of forced RPV flow events (e.g., reactor coolant pump trip or pump rotor seizure). The NRC staff agrees failure of forced coolant flow is not applicable to the NPM due to the lack of reactor coolant pumps.

TR Sections 7.1.3.1, “Background,” through 7.1.3.3, “Electrical Systems with Important Loads,” discuss the electric power requirements and supply duration. The applicant states that EDAS provides uninterrupted DC power for 72 hours to essential loads while shedding low importance or non-essential loads. Following loss of normal AC power, EDAS batteries power the ECCS valves for 24 hours, then these valves open: RVVs first, then, RRVs when RPV-to-CNT pressure difference matches the IAB release threshold. ECCS actuation due to load shedding may be pre-empted by a signal that actuates ECCS 8 hours after any reactor trip in the NPM-20 design. The applicant states that actuating ECCS after 8 or 24 hours is not relevant to the short-term FOM(s) addressed by this report (MCHFR and RCS and SG maximum pressures). The NRC staff agrees and notes that the EM for an inadvertent opening of an ECCS valve is addressed in the LOCA TR, (Reference 8) and discharge of reactor coolant after 24 hours is addressed in the XPC TR (Reference 7).

TR Section 7.1.3.4, “Timing of Loss of Power,” discusses the timing for the loss-of-power. The loss of normal AC power is assumed to occur either coincident with the initiation of the event or coincident with turbine trip. The basis for selecting these two times is that the loss of AC power could be the event initiator or be caused as a result of the event. The applicant also notes that the random loss of non-safety related electrical systems are not assumed for the NuScale non-LOCA EM, but the failure of the DC power (normal DC power system (EDNS) and augmented DC power system (EDAS)) are related to the loss of AC power or at the time of the initiating event. The specific electric power assumptions are reviewed as part of a design-specific application of this methodology, such as the NPM-20 SDA. This is reflected in item 6 in Section 4, “Limitations and Conditions,” of this evaluation.

TR Section 7.1.4, “Single Failures,” discusses the single failure assumptions for the NuScale non-LOCA EM. The applicant notes in TR Section 7.1.4.3, “Consideration of Passive Single Failures,” that passive failures of fluid systems, components that do not have to change position or state (e.g., piping or heat exchanger) are not considered for the non-LOCA transient analyses during the short term (up to 24 hours). This is consistent with the SECY-94-084 SRM and past precedence and therefore is acceptable. Components that change state or position in a fluid system are considered active components and are subject to the single failure criteria. The staff notes that a failure of the MPS which causes the opening of ECCS valves is not considered within the scope of the non-LOCA methodology and instead is analyzed using the methodology provided in the LOCA TR (Reference 8). TR Section 3.4 provides a description of the IAB design.TR Section 7.1.4.2, “Consideration of Single Failures,” describes the various means used to identify the potential active single failures. Passive electrical failures are also considered, consistent with the SECY-94-084 SRM. TR Section 7.1.4.4, “Single Failures to Evaluate,” identifies the single active failures considered for the NuScale non-LOCA EM analyses and identifies a passive electrical single failure in the MPS as the failure to signal one ECCS RRV and one RVV to open upon demand. The evaluation of the appropriate active and passive single failures is performed on an event-specific basis as part of the application of this methodology to a specific design, such as the NuScale SDA Chapter 15 review.

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 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.” [3] In SECY-19-0036, the NRC staff summarized the NRC’s historical practice for applying the SFC. Specifically, the NRC staff summarized SECY-77-439, [4] in which it informed the Commission of how the NRC staff then generally applied the SFC, and, SECY-94-084, [5] 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,” [6] 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

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[3] See SECY-19-0036, “Application of the Single Failure Criterion to NuScale Power LLC’s Inadvertent Actuation Block Valves,” (April 11, 2019) (Agencywide Documents Access and Management System (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19060A081).

[4] See SECY-77-439, "Single Failure Criterion," (August 17, 1977) (ML060260236).

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

[6] 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).

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 non-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 item 1, in Section 4.0 of this SER.

TR Section 7.1.5, "Bounding Reactivity Parameters," discusses the use of bounding reactivity parameters in non-LOCA analyses. Section 7.1.5.1, "Moderator Temperature Coefficient," discusses the moderator temperature coefficient (MTC) and provides example values. Section 7.1.5.2, "Doppler Temperature Coefficient," discusses the Doppler temperature coefficient and provides example values. The use of low and high multipliers on the decay heat contribution and inclusion or exclusion of the actinide contribution is discussed in TR Section 7.1.5.3, "Decay Heat Contribution." TR Figure 7-1, "Example of decay heat comparisons," shows that use of the multipliers and inclusion or exclusion of actinide contribution conservatively bounds the best- estimate decay heat calculated using the ORIGEN code for a generic equilibrium cycle. TR Section 7.1.5.3 states that a review of the applicable core physics parameters will be performed for each cycle to confirm that the multipliers remain bounding. The NRC staff finds the applicant's use of the 1973 ANS decay heat standard with appropriate multipliers and actinide contribution confirmed on a cycle-by-cycle basis acceptable because it ensures that the values used in the analyses remain bounding.

The scram worth is defined in TR Section 7.1.5.4, "Scram Worth," while Table 7-2, "Example of normalized trip worth vs. time after trip," provides an example table of normalized trip worth as a function of time after reactor trip.

The staff finds that use of bounding reactivity parameters is conservative and consistent with SRP/DSRS Chapter 15 guidance and is therefore acceptable.

TR Section 7.1.6, "Biasing of Other Parameters," provides a brief description of biasing non-reactivity parameters in the NuScale non-LOCA EM, including initial conditions, valve characteristics, and analytical limits and associated response times. The TR does not contain any methodologies for uncertainty analysis. Instead, reliance is placed on defining biases, conservatisms and use of sensitivity calculations to demonstrate compliance with relevant acceptance criteria applicable to non-LOCA transients.

TR Section 7.1.6.1, "Initial Conditions," discusses how the initial conditions are chosen for non-LOCA analyses. The applicant states that the most challenging initial conditions for the event and acceptance criterion of interest are applied to the analyses. While example initial condition biases are presented in this section, in implementation biased initial conditions will be consistent with plant-specific ranges expected during normal operation, accounting for steady-state fluctuations and calibration and instrument errors that are consistent with the plant design. However, nominal conditions may be used if the event is insensitive to the parameter.

The NRC staff notes that several parameters identified in Section 7.1.6.1, “Initial Conditions,” of the TR are not truly independent initial conditions but must be determined through initial steady-state calculations. For example, initial RCS flow rate for a natural circulation NPM is related to the power input by the reactor, the heat removal by SGs, and the hydraulic characteristics of the circuit. Therefore, **{{**

}}, it would not be possible to arbitrarily specify the initial flow without violating the conservation of mass, energy, and momentum. Based on the information reviewed as part of the NRC staff’s audits (ML24262A257), the NRC staff confirmed that **{{**

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TR Section 7.1.6.2, “Valve Characteristics,” discusses the valve characteristics for the pressure relief valves, isolation valves, DHRS valves, nonsafety-related feedwater check valves, and turbine stop valves. The valve characteristics are basic design information necessary to represent, in part, the plant design and operation of a system, structure, or component. While the stroke times provided in the TR are examples, the staff finds the overall strategy of providing the most conservative characteristics for the acceptance criterion of interest acceptable.

Section 7.1.6.3, “Analytical Limits and Response Times,” of the TR discusses analytical limits and response times modeled in the NuScale non-LOCA transient analyses. Table 7-3 of the TR provides examples of analytical limits and actuation delays. While many of these functions are comparable to protection system actuation functions in traditional large PWRs, some functions, such as high or low steam superheat, are specific to the NPM design. The staff reviewed the parameters and finds that the information is sufficient from the perspective of an evaluation of the methodology. However, the staff notes that these are examples and the actual values of these parameters are directly associated with a specific NPM design. The methodology requires use of actuation setpoints and delays that are associated with the specific NPM design when the methodology is applied. Changes in the signals, setpoints, and delays used for both reactor trip and ESFAS actuations may alter event-specific transient progressions and limiting bias directions compared to what is shown in the report. This requirement is reflected in Limitation and Condition No. 8.

The NRC staff also notes that this topical report was submitted concurrently with SDAA for the NPM-20 design. Calculations implementing the non-LOCA methodology were audited during this review, as discussed in the following sections. During its review, the staff noted and the applicant confirmed (ML24348A033 (NP)) that these values are not comparable with the designed operating conditions of the NPM-20. Therefore, deviations between the signals, setpoints, and delays used for the NPM-20 design should be considered when applying this methodology to designs with different RTS and ESFAS configurations.

The NRC staff finds the biasing of non-reactivity parameters is dependent upon the specific non-LOCA event. The NRC staff finds that the input range determination is consistent with DSRS Section 15.0. The examples provided for valve operational timing are consistent with stroke times in typical non-LOCA EMs. The example of analytical limits (setpoints used in the non-LOCA analyses) and actuation delays are consistent with typical non-LOCA analyses.

Section 7.1.7, “Credit for Nonsafety-related Components or Operator Actions,” of the TR describes the non-safety-related components and operator actions for which credit is taken in the NuScale non-LOCA safety analyses. The applicant indicates that the following non-safety-related equipment or components are credited for event mitigation as part of the non-LOCA transient analyses:

- Non-safety-related secondary MSIV as the backup isolation device for main steam system piping penetrating containment.

- Non-safety-related feedwater regulating valves as backup isolation of the feedwater system piping penetrating containment.

- Non-safety-related feedwater check valve as backup isolation of the DHRS when reverse flow is experienced during a break in the feedwater piping system.

Section 7.1.7 of the TR also indicates that operator action is not credited in the non-LOCA evaluation model. The determination for the need of operator actions to mitigate specific non-LOCA events is to be evaluated as part of a design-specific application of this methodology, such as the NuScale SDA. This is reflected in item 6 in Section 4, “Limitations and Conditions,” of this evaluation (ML24305A000).

The NRC staff finds use of the non-safety-related feedwater regulating and check valves acceptable as a backup to safety-related components as it is consistent with NUREG-0138, Issue 1. The use of the secondary MSIV is an extension of NUREG-0138, Issue 1 as it deals with maintaining primary side inventory. The NRC staff finds this acceptable subject to item 5 in Section 4, “Limitations and Conditions,” of this evaluation, which requires an applicant or licensee using this methodology and seeking to credit the non-safety-related MSIV in the analysis of a SGTF event to receive specific approval to credit the non-safety-related MSIV through the design review.

3.7.2 Event-Specific Methodology

Section 7.2, “Event Specific Methodology,” of the TR describes the NuScale non-LOCA analysis methodology specific to each event and states that the non-LOCA event simulations are performed using conservative methodologies. TR Table 7-4, “Regulatory Acceptance Criteria,” provides the regulatory acceptance criteria. The table notes that other methodologies are used for most of the acceptance criteria (CHF, fuel centerline temperature, peak containment pressure, and dose). The criteria for RCS and SG pressure are considered within the non-LOCA EM. The NRC staff reviewed TR Section 7.2 and audited the supporting transient analysis methodology report (ML24215A253) to confirm that the applicant’s methodology for each event specifies appropriate assumptions and biases for the applicable parameters, that the necessary acceptance criteria will be checked, and that the methodology will ensure conservative results when implemented**.** Event-specific single failures, electrical power assumptions (AC and DC), and the potential need for operator actions to mitigate non-LOCA events are not evaluated as part of this review. The determination of event-specific single failures, electrical power assumptions, and potential operator actions are to be evaluated as part of a design-specific application of this methodology, such as the NuScale SDA. This is reflected in item 6 in Section 4, “Limitations and Conditions,” of this evaluation.

TR Section 7.2, “Event Specific Methodology,” states that initial RCS flow is biased low for most events since it is limiting for MCHFR. The applicant stated that **{{**

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The NRC staff agrees that biasing the initial RCS flow low tends to conservatively reduce the MCHFR due to the lower heat transfer at the lower mass flux. **{{**

}}. For these reasons, the NRC staff finds that minimizing the initial RCS flow is acceptable.

Section 7.2, “Event Specific Methodology,” of the TR also explains the **{{**

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The applicant provided sample sensitivity studies for each event as discussed below. The NRC staff notes that these sensitivity studies were not performed based on the NPM-20 design, rather they were provided to illustrate how future TR users would perform sensitivity studies when the methodology is implemented. As such, the staff did not audit the calculations themselves or verify the results and did not consider the results when evaluating the methodology for application to the NPM-20.

Transition to a safe and stable condition is dependent on DHRS performance, which can be challenged in certain ranges of SG-DHRS loop inventory. **{{**

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The NRC staff finds the applicant's treatment of initial conditions and parameters which are varied to be acceptable because sensitivity studies will be performed as part of the event-specific methodologies, to identify the limiting bias direction for licensing basis calculations.

3.7.2.1 Decrease in Feedwater Temperature

TR Section 7.2.1, "Decrease in Feedwater Temperature," describes the decrease in feedwater temperature event-specific methodology. The NRC staff reviewed the applicant's methodology for this event to determine whether it specified appropriate biases and conservatisms for the applicable parameters, whether the necessary acceptance criteria would be checked, and whether the methodology as a whole would ensure conservative results when implemented.

The event is caused by an unspecified feedwater system malfunction. The decrease in feedwater temperature results in decreased primary coolant temperature, increased reactivity, and increased core power. A reactor trip may result from high reactor power or high coolant temperature in the riser. Feedwater isolation valve (FWIV) closure resulting from DHRS actuation stops the RCS over- cooling.

The methodology assumes that the initial feedwater temperature starts at the full-power feedwater temperature and decreases to the coldest temperature in the secondary side. The TR states that sensitivity studies are performed to identify the limiting condition which occurs for concurrent high core power and high riser temperature reactor trips. The NRC staff finds this strategy acceptable because it identifies the limiting cooldown rate from a bounding spectrum of cooldown rates.

The TR also states that **{{**

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Table 7-6, "Acceptance criteria – decrease in feedwater temperature," discusses all non-LOCA acceptance criteria in the context of the decrease in feedwater temperature event. The applicant states that peak primary and secondary pressures are bounded by undercooling events discussed in other parts of the TR, and therefore, sensitivities to maximize these parameters are not analyzed as part of the decrease in feedwater temperature event. The NRC staff agrees that primary and secondary peak pressures are bounded by other events and that MCHFR is the principal FOM for the decrease in feedwater temperature event.

The NRC staff reviewed the initial conditions, biases, and conservatisms in TR Table 7-7, "Initial conditions, biases, and conservatisms – decrease in feedwater temperature," including the PCS operating function assumptions. For the parameters that are not varied as part of each application of the methodology (i.e., parameters whose bias directions are specified in the TR), the NRC staff confirmed that the bias directions are appropriately conservative or otherwise appropriate. For example, the initial bias directions for reactor power and initial RCS average temperature are conservative because these biases are consistent with known directions of conservatism for MCHFR (ML19067A256). In addition, the EOC MTC bias provides the largest reactivity change during cooling and minimizes MCHFR. Some parameters **{{**

}} are set to a nominal initial value, which is acceptable because a conservative bias direction does not exist for these parameters.

TR Table 7-7 indicates that the initial PZR water level is biased to high based on the results of the applicant's sensitivity analysis which shows that MCHFR does not change appreciably as a function of PZR level. The staff considers this conclusion reasonable because the range of possible PZR water levels hardly changes the thermophysical properties of water in the core region (via hydrostatic pressure) (e.g., viscosity, thermal conductivity, specific heats); similarly, the PZR-water level essentially does not influence the primary loop flow speed because the PZR baffle plate essentially decouples the liquid water in the PZR from the liquid water that is circulating as the primary loop flow. Additionally, RCS pressure changes due to variation in initial PZR water level and the accompanying change in hydrostatic head are small in comparison to nominal variations in RCS pressure. The TR methodology requires variation of the initial RCS pressure and operation of the PZR spray in order to identify the limiting MCHFR considering variations in RCS pressure.

In TR Table 7-7, the applicant states that decay heat is biased high to maximize the impact on DHRS. The staff finds this approach to be acceptable because higher decay heat requires higher DHRS capacity.

The PCS function of automatic rod control is **{{**
}}. The staff finds this approach to be acceptable because it is designed to get the minimum MCHFR.

The TR Table 7-7 states that initial PZR pressure and operation of PZR spray are varied in order to identify the limiting MCHFR (ML24348A040). These parameter treatments deviate from the previous revision of this TR (i.e. TR-0516-49416-P-A, Revision 3). Because the effect of pressure on MCHFR is evaluated each time the methodology is applied in order to identify the limiting MCHFR, the NRC staff finds this acceptable.

The applicant presented sample results from sensitivity studies in TR Tables 7-8, “Representative fuel exposure study,” through 7-11, “Representative boundary condition type / single active failure studies,” to demonstrate the effects of fuel exposure, initial fuel temperature, boundary condition type, and the single active failure of an MSIV to isolate. While the results are not applicable to the NPM-20 design, sensitivity studies on boundary condition type and single active failure illustrate how parameters will be varied to perform the sensitivity studies specified in TR Table 7-7, “Initial conditions, biases, and conservatisms.” (ML24305A000).

The applicant also presented results from an example sensitivity study of feedwater temperature cooldown rate in Table 7-10 which demonstrated that CHFR is minimized when **{{**

}}. While these results are not applicable to the NPM-20 design, they illustrate how the methodology requires variation of the rate of feedwater temperature change in order to identify transitions between trip signals, as limiting MCHFR may occur at these points (ML24305A000).

Based on information reviewed as part of the NRC staff’s audits (ML24262A257), the NRC staff confirmed that the parameters and initial condition biases applied to the event would result in a conservative bounding value of MCHFR. The NRC staff further confirmed that **{{**
}} means that the SG tube plugging is determined by sensitivity studies to maximize the effect on MCHFR.

Based on the information submitted by the applicant, as confirmed by the NRC staff’s review and audit, the NRC staff finds that the applicant’s methodology for this event will ensure conservative results when implemented.

3.7.2.2 Increase in Feedwater Flow

TR Section 7.2.2, “Increase in Feedwater Flow,” describes the increase in feedwater flow event-specific methodology. The NRC staff reviewed the applicant’s methodology for this event to determine whether it specified appropriate biases and conservatisms for the applicable parameters, whether the necessary acceptance criteria would be checked, and whether the methodology as a whole would ensure conservative results when implemented.

The event is initiated by a malfunction that increases the feedwater flow rate. Like the decrease in feedwater temperature event, the overcooling of the RCS decreases the core inlet temperature and increases the core power. Reactor trip may result from various MPS signals (e.g., high power, low steam line superheat, high PZR pressure, or high steam line pressure). Secondary system isolation ends the overcooling event, and DHRS provides decay heat removal. Additionally, increase in feedwater flow transients increase secondary side inventory. After DHRS actuation, a larger secondary side inventory may challenge DHRS performance because a higher collapsed liquid level in the DHRS condensers reduces the surface area available for steam condensation.

TR Table 7-13, “Acceptance criteria – increase in feedwater flow,” discusses all the acceptance criteria for the increase in feedwater flow event. Like the decrease in feedwater temperature event, the NRC staff agrees that primary and secondary peak pressures are bounded by other events and that MCHFR is the principal FOM for increase in feedwater flow event.

The applicant stated that the limiting MCHFR results when **{{**

}}. The methodology specifies that sensitivity studies to determine **{{** **}}**, as well as limiting bias directions for certain parameters, should be performed.

TR Table 7-14, “Initial conditions, biases, and conservatisms – increase in feedwater flow,” provides the initial conditions, biases and conservatisms for the increase in feedwater flow event. Some of the biases use bounding MCHFR and others (e.g., Initial PZR pressure, PZR spray operation), are determined by sensitivities to get the most challenging conditions (ML24262A257). The initial condition biases for the increase in feedwater flow event are largely the same, and based on similar rationale, as those applied to the decrease in feedwater temperature event described in Section 7.2.1, “Decrease in Feedwater Temperature,” of the TR. Because the RCS response is similar between the decrease in feedwater temperature and increase in feedwater flow events, the NRC staff finds that appropriate bias directions were also applied for the increase in feedwater flow event.

The **{{**

}} described in Section 7.2.1, “Decrease in Feedwater Temperature,” of the TR. While the effect of downcomer coolant temperature changes on the ex-core detector response is not event-specific, modifications to these setpoints may only be valid for certain event progressions. However, because the event timing and trip signals are similar between the increase in feedwater flow and decrease in feedwater temperature events, the NRC staff finds that application of these trip modifications to the increase in feedwater flow event is also acceptable.

TR Table 7-14 provides initial conditions, biases, and conservatisms used in cases that challenge DHRS performance for increase in feedwater temperature. The NRC staff reviewed the initial condition biases and the rationale provided that these conditions will maximize secondary side inventory prior to DHRS actuation. Additionally, NRC staff audited “Increase in feedwater flow” calculations provided with the US460 SDAA, which was submitted concurrently with this topical report. **{{**

}}. Therefore, the NRC staff finds the initial conditions, biases, and conservatisms for these cases acceptable**.**

TR Table 7-16, “Representative increase in feedwater flow study – high and low SG performance with maximum power and minimum RCS flow,” provides an example of sensitivity studies that might be performed to ascertain the limiting bias directions for an application of the increase in feedwater flow methodology. While the sensitivity study results do not apply to the NPM-20 design, they would indicate that **{{** **}}** **{{**

}} (ML24305A000). These sensitivity studies will be repeated when the methodology is implemented, as described in TR Table 7-14.

Based on the information submitted by the applicant, as confirmed by the NRC staff’s review and audit (ML24262A257), the NRC staff finds that the applicant’s methodology for this event will ensure conservative results when implemented.

3.7.2.3 Increase in Steam Flow

TR Section 7.2.3, “Increase in Steam Flow,” describes the increase in steam flow event-specific analysis methods. The NRC staff reviewed the applicant’s methodology for this event to determine whether it specified appropriate biases and conservatisms for the applicable parameters, whether the necessary acceptance criteria would be checked, and whether the methodology as a whole would ensure conservative results when implemented.

An increase in main steam flow causes an increase in heat transfer from the primary to the secondary, a decrease in RCS temperature, an increase in core power and heat flux, and a decrease in RCS and SG pressures. The decreasing RCS temperature causes the rod control system to withdraw the regulating bank. Reactor trip may occur on high power, high riser temperature, or low steam pressure signals. **{{**

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The TR states that the limiting MCHFR occurs when the event is initiated from full power conditions, and the power increase resulting from the increased steam flow remains just below the high-power analytical limit such that the reactor trip occurs due to high RCS riser temperature or low steam pressure.

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The NRC staff reviewed the initial condition biases and assumptions for the increase in steam flow in TR Table 7-19, “Initial conditions, biases, and conservatisms – increase in steam flow.” The NRC staff notes that they are very similar to those applied to the increase in feedwater flow event described in Section 7.2.2, “Increase in Feedwater Flow,” of the TR. This is appropriate given the similarity of RCS behavior between the two events. One notable difference is that the initial SG pressure is biased high for the increase in steam flow event **{{**
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The sensitivity study results presented in Tables 7-20, “Representative steam flow study – nominal steam generator heat transfer,” and 7-21, “Representative steam flow study – steam generator heat transfer biased low” of the TR illustrate how a user of the methodology could identify the limiting steam flow increase. While the results are not applicable to the NPM-20 design, they would indicate that **{{**
}} (ML24305A000). These sensitivity studies will be repeated when the methodology is implemented per TR Table 7-19.

Based on the information submitted by the applicant, as confirmed by the NRC staff’s review and audit, the NRC staff finds that the applicant’s methodology for this event will ensure conservative results when implemented.

3.7.2.4 Steam System Piping Failure Inside or Outside of Containment

TR Section 7.2.4, “Steam System Piping Failure Inside or Outside of Containment,” describes the steam system piping failure inside or outside of containment event-specific analysis methodology. A steam line break is defined as a pipe break in the main steam system, which results in excessive RCS cooldown and causes the core reactivity to increase. The methodology considers a range of sizes for steam line breaks inside or outside of containment in the NPM. For a break inside containment, even a very small steam line break would lead to a reactor trip on high containment pressure since the containment operates at sub-atmospheric conditions or near vacuum. For breaks outside of containment, larger breaks will result in a reactor trip on low steam pressure or high core power and flow out of the break is terminated by the closure of the MSIV. For smaller breaks outside of containment, reactor trip will eventually occur on high core power.

The methodology considers split breaks to be of a higher frequency of occurrence and thus evaluated against the more restrictive AOO acceptance criteria for MCHFR, system pressures, and fuel centerline melt; the larger double-end guillotine breaks are evaluated against the applicable radiological dose criteria which is assessed in the downstream radiological dose analysis.

The NRC staff reviewed the applicant’s methodology for this event to determine whether it specified appropriate biases and conservatisms for the applicable parameters, whether the necessary acceptance criteria would be checked, and whether the methodology as a whole would ensure conservative results when implemented.

The initial conditions, biases and control system responses, including the sensitivity studies to be considered are expected to ensure conservative results for the individual plant licensing applications. Decalibration of the excore neutron detectors due to the increase in downcomer density from the cooldown is accounted for by **{{**

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Flow through the break is modeled using the **{{**

}}, as noted in the TR, has some dependence upon the configuration near the break. Essentially, the double-ended rupture of one of the steam lines would have different critical flow behavior than the equivalent size split rupture in the merged piping. The NRC staff confirmed during its audits of supporting documentation (ML24262A257**),** that the NRELAP5 model appropriately reflects the design of the main steam line upstream of the MSIVs relative to how a circumferential break of one steam line inside containment affects fluid and steam flow in the SGs. The applicant models **{{**

}}, which the NRC staff agrees is a conservative approach.

TR Table 7-22, “Acceptance criteria, single active failure, loss of power scenarios – steam line break,” states that for the steam line break, the limiting MCHFR is not adversely affected by a single failure or the loss of power. However, limiting mass and energy release for radiological consequences results are considered when there is a single failure of one MSIV to close on the piping with the break outside containment and limiting mass and energy release for radiological consequences results when there is a single failure of one FWIV to close on the piping with the break inside containment. As noted in SER Section 3.7.2, “Event Specific Methodology,” event-specific single failures, electrical power assumptions (AC and DC), and the potential need for operator actions to mitigate non-LOCA events are not evaluated as part of this review. The determination of event-specific single failures, electrical power assumptions, and potential operator actions are evaluated as part of a design-specific application of this methodology, such as the NuScale SDA Application.

TR Table 7-24, “Initial conditions, biases, and conservatisms – steam line break,” shows that most parameter initial conditions are biased to conservative conditions or varied for each application of the methodology to identify the limiting bias directions or assumptions. As discussed in Section 3.7.2, "Event Specific Methodology," of this SER, this is acceptable to the NRC staff.

TR Table 7-25, "Steam line break study," presents results for example sensitivity studies for the steam line break in terms of MCHFR. While the results are not applicable to the NPM-20 design, the example studies illustrate how a user of the methodology should identify the parameter biases and assumptions that provide a bounding transient simulation (ML24305A000).

Based on the information submitted by the applicant, as confirmed by the NRC staff's review, and audit (ML24262A257), the NRC staff finds that the applicant's methodology for this event will ensure conservative results when implemented.

3.7.2.5 Containment Flooding or Loss of Containment Vacuum

TR Section 7.2.5, "Containment Flooding / Loss of External Load," discusses the containment flooding or loss of containment vacuum event, which is unique to the NPM.

The TR defines a loss of containment vacuum as the ingress of vapor, air, or minimal amounts of water into the CNV that does not cause water buildup in the CNV. Containment flooding does result in liquid buildup in the CNV. The applicant states that the containment flooding event is considered only for a break in the reactor component cooling water (RCCW) line since breaks in other lines that could result in liquid buildup in the CNV are evaluated as separate initiating events. As a consequence of, the overcooling effect of the event, reactor power level may increase, resulting in a decrease in MCHFR. Sensitivity analyses are performed for licensing applications to determine the containment flooding / loss of vacuum cases with lowest MCHFR for this event as delineated in TR Table 7-28.

The applicant further states that the loss of CNV vacuum or CNV flooding from the RCCW results in increased conductive and convective heat removal from the RCS making the event less limiting than other AOOs such as overheating events. The staff finds this conclusion to be acceptable because the event does not introduce more challenging conditions for RCS pressure and secondary side pressure.

The NRC staff reviewed TR Table 7-28, "Initial conditions, biases, and conservatisms – containment flooding / loss of containment vacuum," which presents the initial conditions, biases and conservatisms for the containment flooding or loss of containment vacuum events and confirmed that the specified biases and control system assumptions are appropriately conservative or otherwise acceptable.

The TR describes modifications to the high power trip setpoints and high power rate trip setpoint in response to changes in downcomer temperature. NRC staff audited sample calculations for this event provided during review of the NPM-20 SDAA and noted that even without modifications to trip setpoints, none of the analyzed transients result in a reactor trip. Additionally, the NRC staff understands that flooding the containment vessel (CNV) would result in a more significant decalibration of excore detectors. The non-LOCA model does not consider decalibration resulting from containment flooding. NRC staff expects that modeling this effect could result in earlier power rate trips for some containment flooding cases. Because neglecting a potential reactor trip is conservative for non-LOCA acceptance criteria, and because the NPM-20 design does not feature control systems that accept ex-core detector power as an input, the NRC staff finds this approach acceptable for this event.

Based on the information submitted by the applicant, as confirmed by the NRC staff's review, the NRC staff finds that the applicant's methodology for this event is consistent with DSRS Section 15.1.6 and will ensure conservative results when implemented.

3.7.2.6 Turbine Trip / Loss of External Load

TR Section 7.2.6, "Turbine Trip / Loss of External Load," describes turbine trip/loss of external load event-specific methodology. The applicant grouped these events together because they are essentially identical with respect to the system response, except that turbine trip initiates with turbine stop valve closure, while loss of external load initiates with turbine control valve closure.

A loss of external load event is caused by the disconnection of the turbine generator from the electrical distribution grid. A loss of external load generates a turbine trip, which results in a reduction in steam flow from the SGs to the turbine. A turbine trip may also occur independently, resulting in closure of the turbine stop valves. In the NPM, turbine bypass valves would normally open to allow the reactor to remain in operation in the event of a turbine trip. However, the applicant does not credit the turbine bypass valves for event mitigation.

The reduction in heat removal because of reduced steam flow to the turbine results in pressurization of the RCS. The closure of the turbine stop valve or the turbine control valve results in pressurization of the secondary. Because of the rapid pressurization of the primary and secondary systems, the NRC staff finds that the applicant has appropriately identified the primary and secondary pressures as the acceptance criteria of interest for this event. The applicant stated that a reactor trip and DHRS actuation would transition the NPM to a safe, stable condition.

The NRC staff reviewed the initial conditions, biases, and conservatisms for the NPM-20 turbine trip/loss of external load events in TR Table 7-32, "Initial conditions, biases, and conservatisms – turbine trip / loss of external load." For the initial conditions whose bias directions are specified, the NRC staff confirmed that the bias directions are limiting for these events. For example, initial reactor power is biased high, which is consistent with guidance in DSRS Section 15.2.1-15.2.5. The NRC staff finds the assumption of BOC reactivity feedback and kinetics conservative for these events because the least-negative reactivity coefficients minimize negative reactivity feedback resulting from temperature increases. In addition, biased-high decay heat is generally limiting for overheating events since it presents the greatest challenge to heat removal.

The NRC staff also finds the assumptions regarding the control systems in TR Table 7-32, "Initial conditions, biases, and conservatisms – turbine trip / loss of external load," such as disabling PZR spray and RCS letdown, appropriate because they present the greatest challenge to the primary and secondary pressure acceptance criteria.

TR Table 7-33, "Representative Sensitivity Studies – Turbine Trip / Loss of External Load," provides an example of sensitivity studies that might be performed to ascertain the limiting bias directions for an application of the turbine trip / loss of external load methodology. While the results are not applicable to the NPM-20 design, they illustrate how sensitivity studies will be performed to identify limiting conditions for this event. For the turbine trip/loss of external load event, the TR includes a requirement for a sensitivity study on the initial primary temperature and primary/secondary pressures (as indicated in TR Table 7-32) to identify the conditions that maximize peak primary and secondary pressures. The TR requires additional sensitivity studies on other parameters when margins to acceptance criteria fall below limits or when the capacity of the RSVs is challenged. Circumstances in which additional sensitivity studies are required are specified in TR Section 4.2. These additional sensitivity studies are performed to ensure that case(s) with the potentially limiting peak primary and secondary pressures are identified. The additional sensitivity studies would be performed on parameters beyond those listed as "varied" in Table 7-32, "Initial conditions, biases, and conservatisms - turbine trip / loss of external load," although the studies may involve simultaneous variation of additional parameters and those listed as "varied" (ML24305A000).

The staff reviewed the applicant's methodology for these events to determine whether it selected the appropriate parameters and specified appropriate biases and conservatisms for the applicable parameters, whether the necessary acceptance criteria would be checked, and whether the methodology as a whole would ensure conservative results when implemented.

Based on the information submitted by the applicant, as confirmed by the NRC staff's review and audit (ML24262A257), the NRC staff finds that the applicant's methodology for this event is consistent with DSRS Sections 15.2.1 and -15.2.25 and will ensure conservative results when implemented. On these bases, the staff finds methodology acceptable because the methodology and the NRELAP5 code are adequate for analyzing the system's response to the turbine trip/loss of external load accident and produce reliable results.

3.7.2.7 Loss of Condenser Vacuum

TR Section 7.2.7, "Loss of Condenser Vacuum," describes the loss of condenser vacuum (LOCV) event-specific methodology. The loss of condenser vacuum results in turbine stop valve closure and a loss of feedwater flow. By design, the turbine bypass valves would normally open to allow the reactor to remain in operation in the event of a turbine trip. However, the applicant does not credit the turbine bypass valves for event mitigation. A turbine trip and loss of feedwater would result in a sudden loss of the secondary side heat removal, heatup of the RCS, and pressurization of the secondary side. Rising system pressures typically result in a rapid RTS actuation on either high PZR or steam pressure. The applicant stated that a reactor trip and DHRS actuation would terminate the event and transition the NPM to a safe, stable condition.

TR Section 7.2.7.1, "General event description," states that that the loss of condenser vacuum event methodology is essentially equivalent to the methodology used for the turbine trip / loss of external load events. The main difference is that the loss of condenser vacuum event includes a loss of feedwater flow at event initiation. However, because the turbine trip / loss of external load events considers a loss of normal AC power at event initiation, those events also model a loss of feedwater flow at event initiation. As a result, the scenarios analyzed as part of Section 7.2.6 address the loss of condenser vacuum event. TR Section 7.2.7.1 concludes that the relevant acceptance criteria, SAF, and LOP scenarios from Table 7-30 are also applicable to the loss of condenser vacuum event.

The NRC staff audited the applicant's analyses for the turbine trip / loss of external load events and the loss of condenser vacuum event that were submitted with the US460 SDAA, as these calculations were performed using the TR EM and were submitted concurrently with this TR.

The NRC staff considered the initial conditions regarding assumed loss of feedwater flow at event initiation. The loss of condenser vacuum event results in a loss of feedwater flow at event initiation, but the turbine trip / loss of external load does not. However, the applicant also assumes that normal AC power is lost at event initiation for the turbine trip / loss of external load events, which results in the same initiation of loss of feedwater flow occurring at event initiation. For these reasons, the NRC staff finds that the scenarios analyzed as part of Section 7.2.6 address the loss of condenser vacuum event. In addition, the staff agrees that the relevant acceptance criteria, SAF, and LOP scenarios from Table 7-30 are also applicable to the loss of condenser vacuum event.

TR Section 7.2.7.2, "Acceptance Criteria," states that the evaluation of the most challenging case(s) relative to the acceptance criteria for the turbine trip / loss of external load events presented in Table 7-31, "Acceptance criteria – turbine trip / loss of external load" is applicable to the loss of condenser vacuum event. Because NRC staff finds that scenarios analyzed as part of Section 7.2.6 address the loss of condenser vacuum event as discussed above, the NRC staff finds this acceptable.

TR Section 7.2.7.3, "Biases, conservatisms, and sensitivity studies," states that the biases and conservatisms presented in Table 7-32 for the turbine trip / loss of external load events are applicable to the loss of condenser vacuum event. Revision 4 of the TR does not include a separate table for the initial conditions, biases, and conservatisms for the LOCV transient. The staff finds this approach to be acceptable because the LOCV event initiation, progression, and consequences are similar to the turbine trip / loss of external load event as discussed above.

L&C No. 6, described in Section 4 of this SER, requires an applicant or licensee seeking to apply this methodology to a design to receive separate approval through that design review for the event-specific electrical power assumptions. This review should confirm that NRC staff findings concerning the similarity of the turbine trip / loss of external load event and the loss of condenser vacuum event remain valid.

Based on the information submitted by the applicant, as confirmed by the NRC staff's review and audit (ML24262A257), the NRC staff finds that the applicant's methodology for this event is consistent with DSRS Section 15.2.1-15.2.5 and there is reasonable assurance that it will produce conservative results when implemented. Therefore, the NRC staff finds the methodology acceptable because the methodology and the NRELAP5 code are adequate for analyzing the system's response to the loss of condenser vacuum event.

3.7.2.8 Main Steam Isolation Valve(s) Closure

TR Section 7.2.8, "Main Steam Line Isolation Valve(s) Closure," discusses the main steam isolation valve (MSIVs) closure event-specific analysis methodology. The MSIV closure event may be initiated by a spurious closure signal, resulting in the inadvertent closure of one or both MSIVs and subsequent pressurization of the secondary system and overheating and pressurization of the RCS. Table 7-38, "Acceptance criteria, single active failure, loss of power scenarios – main steam isolation valve closure," identifies primary and secondary pressures as the FOMs of interest for the MSIV closure event, and the NRC staff agrees based on the rapid pressurization effect of this event.

The NRC staff reviewed the applicant's methodology for this event to determine whether it specified appropriate biases for the applicable parameters and whether the methodology would ensure conservative results when implemented. Table 7-40, “Initial conditions, biases, and conservatisms – main steam isolation valve closure,” lists the initial conditions, biases and conservatisms. Initial RCS flow rate and PZR level are additional parameters varied for the MSIV closure event over those for the turbine trip/loss of external load and LOCV events, which is appropriate given the difference in proximity of the MSIVs to the NPM compared to the turbine stop/closure valves. For the initial conditions whose bias directions are specified, the NRC staff confirmed that the bias directions are limiting for this event (ML24305A000). The NRC staff notes that some assumptions regarding the control systems in TR Table 7-40 differ from the turbine trip/loss of external load and LOCV events, particularly, enabling of turbine throttle valves and feedwater pump speed for the MSIV closure event. However, the NRC staff finds that these functions are inconsequential for the MSIV closure event, and the related assumptions are therefore acceptable.

The applicant provided results of example sensitivity studies in Table 7-41, “Representative sensitivity studies – main steam isolation valve closure.” The results are not applicable to the NPM-20 design, but the example sensitivity studies illustrate how a user of the methodology could vary parameters in order to identify the appropriate limiting initial conditions (ML24348A072 (NP)). Although Table 7-41, “Representative sensitivity studies – main steam isolation valve closure,” only shows results for closure of both MSIVs, the TR text specifies that sensitivity studies on the number of MSIVs closing is performed as part of the methodology. The NRC staff finds the representative sensitivity studies acceptable to demonstrate the general process to determine limiting biases and assumptions (ML24305A000).

Based on the information submitted by the applicant, as confirmed by the NRC staff’s review and audit (ML24262A257), the NRC staff finds, pending closure of the below open item, that the applicant’s methodology for this event to be acceptable because the code validation [**OPEN ITEM**] and review of the event initial conditions and progression and sensitivity analyses have demonstrated that the methodology and the NRELAP5 code are appropriate for analyzing the system’s response to the turbine trip/loss of external load accident and produce reliable results.

Based on the information submitted by the applicant, as confirmed by the NRC staff’s review audit, (ML24262A257) the NRC staff finds that the applicant’s methodology for this event is consistent with DSRS Section 15.2.1-15.2.5 and will ensure conservative results when implemented.

3.7.2.9 Loss of Nonemergency AC Power

TR Section 7.2.9, “Loss of Nonemergency AC Power,” describes the loss of normal nonemergency AC power event-specific analysis methodology. As described in TR Section 7.1.3.1, the normal source of AC electrical power for an operating NPM is from an operating NPM turbine-generator, not the offsite power grid. In addition, there are no onsite safety-related power sources, however the NPM-20 plant design includes two different DC power systems, both of which are equipped with battery backup. The EDNS is a plant common DC power source that does not serve any safety-related loads during either NPM startup, normal operations, shutdown, or abnormal plant operation; and the EDAS supplies plant common loads as well as module-specific loads up to a 72 hour duty, including module-specific power to the MPS.

The loss of normal nonemergency AC power means a loss of power from either the high voltage (EHVS), medium voltage (EMVS) or low voltage (ELVS) AC electrical distribution system. A loss of AC power results in the turbine generator tripping and a loss of pumps on the secondary side, causing an increase in RCS and SG pressure. The NRC staff finds that the applicant has correctly identified primary and secondary pressures as the acceptance criteria of interest for this event as provided in Table 7-42, “Acceptance criteria, single active failure, loss of power scenarios – loss of normal AC power.” The applicant states that a reactor trip and DHRS actuation (which result from the MPS response to loss of AC power to the battery chargers or other trip signals, depending on the scenario and relevant actuation delays) end the transient and transition the NPM to a safe, stable condition.

The NRC staff reviewed the applicant’s methodology for this event to determine whether it specified appropriate biases and conservatisms for the applicable parameters, whether the necessary acceptance criteria would be checked, and whether the methodology as a whole would ensure conservative results when implemented.

The TR states that of the various loss of normal power scenarios, the limiting scenario typically does not result in immediate reactor trip or full CRA insertion at event initiation. The NRC staff finds that scenarios where reactor trip on loss of AC power does not occur or is not credited will be limiting as this will result in more rapid heating and pressurization of the primary side.

The topical report also states that the typical limiting scenario is loss of ELVS at event initiation with EDNS and EDAS/EDSS available. Based on the level of information provided in the topical report regarding the consequences of loss of these various systems, NRC staff cannot confirm that this scenario is limiting and loss of DC power scenarios should be considered on application of this topical report, consistent with the general loss of power treatment discussed in TR Section 7.1.3 and statements in TR Section 7.2.9.1 that a review of the plant-specific electrical system is performed for each licensing application to determine the impact on plant equipment from the loss of power to ensure the limiting scenario is identified.

The NRC staff reviewed the initial conditions, biases and conservatisms for the event in TR Table 7-44, “Initial conditions, biases, and conservatisms – loss of normal AC power,” and finds that the specified bias directions are typically limiting for this event. The staff notes that the initial RCS flow rate will be **{{ }}**, which is a change from the previous Revision 3 of the topical report wherein it is **{{ }}**. Sensitivity analyses are required for licensing applications to identify the most limiting cases.

TR Table 7-45, “Representative sensitivity studies – loss of normal AC power,” provides the results of sensitivity studies for the loss of nonemergency AC power performed on the NPM-160 design (ML24305A000). During review of Revision 3 of TR-0516-49416-P, the NRC staff reviewed supplementary information provided by the applicant (ML18184A589), including the times to reach analytical limits and actuate RTS, DHRS, and CNV isolation, to understand the trends and behavior in TR Table 7-45 and their implications on the biases for the parameters in TR Table 7-44, “Initial conditions, biases, and conservatisms – loss of normal AC power.” Due to the modeled relief capacity of the RSV, the peak primary pressure is nearly invariant for a wide range of differing bias conditions. The one case in which the primary pressure is lower than the rest of the cases results when the combined effect of the initial condition biases delays the RCS pressure rise such that reactor trip occurs before the RSV lift setpoint is reached. Although not shown in TR Table 7-45, nominal biasing of parameters also does not result in reaching the RSV lift setpoint. During review of Chapter 15 calculations supporting the NPM-20 design, the NRC staff also observed that peak RCS pressure was nearly invariant for a wide range of differing initial conditions that approach the RCS pressure limit and result in RSV lift.

Biasing the initial RCS average temperature high tends to result in higher peak SG pressures. Furthermore, a higher initial SG pressure also tends to result in a higher peak SG pressure, which is due to increasing the initial SG inventory.

Based on the information submitted by the applicant, as confirmed by the NRC staff's review, the NRC staff finds that the applicant's methodology for this event is consistent with DSRS Section 15.2.6 and there is reasonable assurance that it will ensure conservative results when implemented.

3.7.2.10 Loss of Normal Feedwater

TR Section 7.2.10, "Loss of Normal Feedwater," discusses the loss of normal feedwater event-specific analysis methodology. A partial or complete loss of feedwater flow results in a boil-off of the water in the SGs, resulting in a loss of the SGs as a heat sink. This causes an increase in the RCS temperature and pressure until the reactor trips due to high PZR pressure. Therefore, the NRC staff finds that the applicant correctly identified (ML24348A058 (NP)) primary and secondary pressures as the acceptance criteria of interest for this event in Table 7-46, "Acceptance criteria, single active failure, loss of power scenarios – loss of normal feedwater flow." The applicant stated that the reactor trip and DHRS actuation terminate the transient and transition the NPM to a safe, stable condition.

The NRC staff reviewed the applicant's methodology for this event to determine whether it specified appropriate biases and conservatisms for the applicable parameters, whether the necessary acceptance criteria would be checked, and whether the methodology as a whole would ensure conservative results when implemented**.**

TR Table 7-48, "Initial conditions, biases, and conservatisms – loss of normal feedwater flow," presents the initial conditions, biases, and conservatisms considered in the methodology to identify a bounding transient simulation for primary and SG pressure. Additionally, TR Section 7.2.10.3, "Biases, Conservatisms, and Sensitivity Studies," states that sensitivity studies are performed as needed, varying the appropriate parameters in Table 7-48 to identify the limiting loss of normal feedwater scenario(s) with regard to primary and secondary pressures.

The NRC staff confirmed that the bias directions that are specified in TR Table 7-48, as well as control system assumptions, are limiting or otherwise appropriate for this event. They are nearly identical to those for the MSIV closure event, with the most notable difference being that initial feedwater temperature is varied for the loss of normal feedwater event/ The NRC staff finds this difference appropriate given that initial feedwater temperature can have a compounding effect with the feedwater flow reduction.

TR Table 7-49, "Sensitivity studies – loss of normal feedwater flow," presents results of the sensitivity studies for a loss of normal feedwater flow that were performed in the NPM-160 design (ML24305A000). The NRC staff notes that these results are limited in scope, **{{**

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These results show that limiting RCS pressure case results from a complete loss of feedwater, while the limiting SG pressure case results from a partial loss of feedwater flow. Based on information provided as part of the audits (ML24262A257), the NRC staff confirmed the reason for the trend in peak SG pressure versus feedwater flow reduction.

The NRC staff gathered additional information from the DCA and SDA Chapter 15.2.7

presented results, which are the relevant sections for the loss of feedwater flow event, The staff reviewed the previous TR sensitivity analyses to ascertain the limiting bias directions for an application of the loss of feedwater flow event methodology. in addition, the NRC staff reviewed information of the NRC staff's previous audits, as documented in the associated audit report (ML19039A090). Based on the review of the FSAR Section 15.2.7 presented results, the sensitivity results in Revision 4 of the TR, and the previous audit report, the NRC staff confirmed some of the behavior and trends observed in the sensitivity studies supporting the initial conditions, biases, and conservatisms listed in TR Table 7-48.

Based on its review of the sensitivity analyses, the NRC staff agrees that that changes to the heat transfer performance of the SG and possible feedwater flow oscillations, caused by potential instabilities, do not significantly reduce margin to SAFDLs or challenge primary pressure acceptance criteria.

Based on the information submitted by the applicant, as confirmed by the NRC staff's review and audit (ML24262A257), the NRC staff finds that performing sensitivity studies by varying the parameters identified in Table 7-48, "Initial conditions, biases, and conservatisms – loss of normal feedwater flow," and considering possible single active failures and loss of power assumptions provides a bounding transient simulation to identify the limiting response(s) for primary and secondary pressure. Based on the information submitted by the applicant, as confirmed by the NRC staff's review and audit (ML24262A257), the NRC staff finds that the applicant's methodology for this event will ensure conservative results when implemented.

3.7.2.11 Inadvertent Decay Heat Removal System Actuation

TR Section 7.2.11, "Inadvertent Decay Heat Removal System Actuation," describes the inadvertent DHRS actuation event-specific methodology. The NRC staff reviewed the applicant's methodology for this event to determine whether it specified appropriate biases and conservatisms for the applicable parameters, whether the necessary acceptance criteria would be checked, and whether the methodology would ensure conservative results when implemented.

The inadvertent DHRS actuation event is unique to plants that incorporate a passive decay heat removal design. In the NPM, the inadvertent actuation of the DHRS may result from an unexpected DHRS valve actuation or a spurious DHRS actuation signal. The applicant evaluates scenarios for the inadvertent operation of DHRS consistent with its design and multiple configurations for operation. The applicant described the five scenarios for consideration:

The inadvertent opening of a single valve at full power conditions or reduced power conditions.

Scenario 1: In this event scenario, the plant does not trip on low turbine inlet temperature or low steam superheat since the safety-related main steam line temperatures are measured just upstream of the junctions between the main steam lines and the DHRS steam lines. However, some of the feedwater is diverted through the DHRS, and a gradual heatup of the RCS occurs until it reaches the maximum analytical temperature limit and signals the MPS resulting in a reactor trip.

Scenario 2: Inadvertent actuation signal isolates one SG and initiates one DHRS train. This scenario is bounded by Scenario 3.

Scenario 3: Inadvertent actuation signal isolates both SGs, and initiates both DHRS trains. This scenario results in a total loss of normal heat removal from the RCS. The applicant biases the initial conditions to maximize system pressures, primarily RCS.

Scenario 4: Inadvertent isolation of one SG and associated DHRS train not actuated. In this scenario the isolation of one SG causes a heatup. This then causes an increase in primary pressure, a reactor trip, and RSV opening. DHRS actuation occurs later in the transient after trip signals are reached and secondary pressure peaks following DHRS actuation.

Scenario 5: Inadvertent isolation of both SGs and no DHRS trains are actuated. The system response is like Scenario 4, except the increase in primary pressure is more rapid and results in earlier reactor trip and RSV opening. Likewise, DHRS actuation occurs later in the transient after trip signals are reached and secondary pressure peaks following DHRS actuation.

TR Section 7.2.11.1 further states that the methodology considers each scenario to determine the limiting cases for the acceptance criteria and sensitivity studies are performed to identify the conditions that maximize peak system pressure.

TR Table 7-52, "Initial conditions, biases, and conservatisms – inadvertent decay heat removal system actuation," presents the initial conditions, biases, and conservatisms that are considered in the methodology to identify a bounding transient simulation for primary and SG pressure .TR Table 7-52 provides the parameters that are either bounding or determined by sensitivity studies. Most of the parameters are to be varied in licensing-basis calculations to identify the limiting bias directions, which is acceptable, as discussed in SER Section 3.7.2, "Event Specific Methodology."

The NRC staff confirmed that the bias directions that are specified in TR Table 7-52, as well as control system assumptions, are appropriately conservative.

TR Table 7-53, "Representative sensitivity studies – inadvertent decay heat removal system actuation," presents results for example sensitivity studies for inadvertent DHRS initiation in terms of maximum primary and secondary pressures. While the results are not applicable to the NPM-20 design, the example sensitivity study helps to illustrate how parameters will be varied to identify the limiting condition in the event-specific methodology. Although TR Table 7-53 only shows results for Scenarios 1, 2, and 3, NRC staff confirmed through audit of calculations implementing the TR-0516-49416-P, Revision 4 methodology that the methodology requires perform similar sensitivity studies for Scenarios 4 and 5. Based on the information submitted by the applicant, as confirmed by the NRC staff's review and audit (ML24262A257), NRC staff finds that performing the sensitivity studies by varying parameters and assumptions as described in the TR with considerations of the above five scenarios provides a bounding transient simulation. The NRC staff finds that the applicant's methodology for this event will ensure conservative results when implemented.

3.7.2.12 Feedwater System Pipe Break Inside or Outside of Containment

The event-specific analysis methods for feedwater system pipe break inside or outside of containment is discussed in TR Section 7.2.12, "Feedwater System Pipe Break Inside of Outside of Containment." The applicant states that both split breaks and double-ended guillotine breaks are analyzed and the more restrictive AOO criteria for system pressures, CHFR, and fuel centerline melt applicable to breaks with higher event frequency are used in the evaluation.

A feedwater line break can occur inside or outside of containment There are no feedwater line isolation valves since there are no feedwater line check valves inside the CNV. A feedwater line break inside containment results in a loss of containment vacuum and a high containment pressure signal that actuates a reactor trip, isolates the secondary system and CVCS, and opens the DHRS valves. The SG, DHRS piping, and DHRS condenser for the faulted SG drain through the break into the containment. The non-faulted SG and DHRS loop provide cooling to the RCS via heat transfer to the reactor pool. The response of smaller feedwater line breaks inside containment is similar to the feedwater line break except that other MPS setpoints, such as high PZR pressure, may be reached before high containment pressure.

A feedwater line break outside containment causes a loss of feedwater flow to the SGs and a heatup of the RCS. The applicant states that large breaks result in reactor trip on high PZR pressure, while smaller breaks result in a more gradual heatup of the RCS and a reactor trip on other MPS signals. DHRS actuates in all cases such that the non-faulted SG loop provides cooling by removing heat from RCS to the reactor pool. Reactor trip and transition to stable DHRS flow terminates the transient with the NPM in a safe,
stable condition.

The NRELAP5 model of the feedwater line break **{{**

}} is briefly discussed as part of the technical evaluation of the Steam System Piping Failure Inside or Outside of Containment event. The applicant further states that sensitivity studies on primary and secondary conditions, and break size/location are performed to identify the conditions that maximize peak primary and secondary pressures.

The NRC staff reviewed the applicant's methodology for this event to determine whether it specified appropriate biases and conservatisms for the applicable parameters, whether the necessary acceptance criteria would be checked, and whether the methodology as a whole would ensure conservative results when implemented.

TR Table 7-56, "Initial conditions, biases, and conservatisms – feedwater line break," presents the initial conditions, biases, and conservatisms that are considered in the methodology to identify a bounding transient simulation for primary and SG pressure. Many of the parameters are to be varied for each licensing application of the methodology. For the parameters whose bias directions are specified, the NRC staff concurs that the bias directions are appropriately conservative. The staff notes that the initial RCS average temperature assumption is biased to high and find this approach to be acceptable because it maximizes the initial system energy and is conservative.

The applicant states that sensitivity studies are performed for licensing applications to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event (i.e., system pressures for overheating events, MCHFR for overcooling events) (ML24305A000).

TR Table 7-57, "Representative sensitivity studies – feedwater line break," provides the results of example sensitivity studies. While the results are not applicable to the NPM-20 design, the studies help to illustrate how the non-LOCA EM could be applied to identify the limiting conditions for the feedwater line break event. The NRC staff finds that performing the sensitivity studies by varying the feedwater break size, location, single active failures (per TR Section

7.1.4.1), loss of power assumptions (per TR Section 7.1.3.1), and parameters identified in Table 7-56, “Initial conditions, biases, and conservatisms – feedwater line break,” to identify the limiting response(s) for the acceptance criteria challenged by the event provides a bounding transient simulation. Based on the information submitted by the applicant, as confirmed by the NRC staff’s review, the NRC staff finds that the applicant’s methodology for this event will ensure conservative results when implemented.

3.7.2.13 Uncontrolled Control Rod Assembly Bank Withdrawal from Subcritical or Low Power Startup Conditions

TR Section 7.2.13, “Uncontrolled Control Rod Assembly Bank Withdrawal from Subcritical or Low Power Startup Conditions,” discusses the uncontrolled control rod assembly bank withdrawal from subcritical or low power startup conditions (i.e., below the hold point at which the high power trip setpoint is changed from the low to the high level and at power levels up to 15 percent rated thermal power) event-specific analysis methods.

In the NPM design, source range count-rate and source and intermediate range flux rate signals provide protection during low-power conditions. Therefore, the applicant examined two scenarios.

In Scenario 1, power is low enough that the intermediate range channel does not have an established signal, and high count-rate and startup rate (source range) signals provide protection. The applicant determined that the limiting case in scenario 1 typically results when **{{**

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In Scenario 2, power is high enough for the intermediate range channel to have an established signal. Therefore, the high count-rate signal is not available, and the high power-rate signal is also not active below 15 percent thermal power. Protection is provided by the high power (low setting) and startup rate (intermediate range) signals. The applicant stated that the highest core power typically occurs when the high power (low setting) and the startup rate (intermediate range) setpoints are reached simultaneously. This establishes the highest initial core power while also allowing for the largest reactivity insertion rate.

Further, TR Section 7.2.13.1, “General Event Description and Methodology,” states that the SGs may provide decay heat removal following the uncontrolled CRA bank withdrawal from subcritical or low-power conditions with at least one feedwater pump operating (which would be the case when RCS temperature is heated to temperatures that permit an approach to criticality). When normal feedwater flow is not available, either the flooded containment or DHRS provides decay heat removal. The maximum power and minimum CHFR occur just after reactor trip, and the peak power and power spike duration do not cause a significant temperature or pressure increase to challenge the RCS or SG pressure acceptance criteria. Therefore, the NRC staff agrees with the applicant’s identification of MCHFR and maximum fuel centerline temperature as the acceptance criteria of interest for this event in TR Table 7-58, “Acceptance criteria, single active failure, loss of power scenarios – uncontrolled control rod bank withdrawal from subcritical or low power startup conditions.”

While TR Table 7-58 includes high-level statements about loss of power scenarios, the specific electric power assumptions are reviewed as part of design-specific applications of this methodology, such as the NPM-20 SDA. This is reflected in item 6 of Section 4, “Limitations and

Conditions," of this evaluation.

The NRC staff reviewed the applicant's methodology for this event to determine whether it specified appropriate biases and conservatisms for the applicable parameters, whether the necessary acceptance criteria would be checked, and whether the methodology as a whole would ensure conservative results when implemented.

TR Table 7-60, "Initial conditions, biases, and conservatisms – uncontrolled control rod bank withdrawal from subcritical or low power startup conditions," lists the initial conditions, biases, and conservatisms for the uncontrolled control rod bank withdrawal from subcritical or low power startup conditions. The NRC staff ensured that the bias directions that are specified, as well as control system assumptions, are appropriately conservative or otherwise acceptable.

The major parameters varied for this event are the initial power level and the reactivity insertion rate. Several parameters are set to nominal values, which is acceptable given that the parameters typically vary as a function of power below a certain power level. In addition, the NRC staff does not expect the parameters set to nominal values to significantly impact MCHFR or fuel centerline temperature due to the low initial power level. The NRC staff also notes that BOC conditions, including the most positive MTC, are appropriate for this event because they minimize negative reactivity feedback as moderator temperatures increase.

TR Table 7-61, "Representative sensitivity studies – uncontrolled control rod bank withdrawal from subcritical or low power startup conditions," provides the results of example sensitivity studies. While the results are not applicable to the NPM-20 design, the example studies help to illustrate how the non-LOCA EM could be applied to identify the limiting conditions for the subcritical or low power control rod withdrawal cases (ML24305A000).

Based on the information evaluated during audit of the NPM-20 SDAA (ML24262A257), the NRC staff confirmed reasons for some of the trends and behavior observed in the sensitivity studies for cases that fall under Scenario 1. The power for these cases is very low, so the reactivity feedback effects are small. For a set reactivity insertion rate, **{{**

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Based on the information submitted by the applicant, as confirmed by the NRC staff's review and audit, the NRC staff finds that the applicant's methodology for this event is consistent with SRP Section 15.4.1 and will ensure conservative results when implemented.

3.7.2.14 Uncontrolled Control Rod Assembly Bank Withdrawal at Power

TR Section 7.2.14, "Uncontrolled Control Rod Assembly Bank Withdrawal at Power," discusses the uncontrolled control rod assembly bank withdrawal at power event-specific analysis methodology, which applies for initial power levels ranging from the low setting level (i.e., the power at which the high power trip is changed from the "low" to the high" setting) to hot full power. The withdrawal of the control rod assembly bank inserts positive reactivity, increasing core power as well as RCS temperature and pressure. The applicant stated that reactor trip may result from the high power, high power rate, high PZR pressure, or high RCS temperatures MPS signals. The limiting condition typically results for the reactivity insertion rate that causes the high core power, high PZR pressure, and high RCS riser temperature signals to be sent almost simultaneously. Higher reactivity insertion rates cause an earlier reactor trip on high power rate. The NRC staff agrees with the general strategy of ensuring the spectrum of reactivity insertion rates **{{** **}}**. In particular, **{{** **}}** should be evaluated because it maximizes the RCS conditions that are known to contribute to the lowest MCHFR.

The NRC staff reviewed the applicant's methodology for this event to determine whether it specified appropriate biases and conservatisms for the applicable parameters, that the necessary acceptance criteria would be checked, and that the methodology as a whole would ensure conservative results when implemented.

TR Table 7-62, "Acceptance criteria, single active failure, loss of power scenarios – uncontrolled control rod bank withdrawal at power," identifies MCHFR and maximum fuel centerline temperature as the primary acceptance criteria of interest for this event, which is consistent with SRP Section 15.4.2 and therefore acceptable. Although primary and secondary pressures increase during this type of event, the decrease in heat removal by the secondary system events are bounding due to the more rapid pressurization rates. While TR Table 7-62 includes high-level statements about loss of power scenarios, the specific electric power assumptions are reviewed as part of design-specific applications of this methodology, such as the NPM-20 SDA. This is reflected in item 6 of Section 4, "Limitations and Conditions," of this evaluation.

TR Table 7-64, "Initial conditions, biases, and conservatisms – uncontrolled control rod bank withdrawal at power," provides the initial conditions, biases, and conservatisms for the uncontrolled control rod bank withdrawal at power event. Most of the RCS conditions are varied as part of each analysis. For the bias directions that are specified, the NRC staff confirmed that the biases are appropriately conservative or otherwise acceptable. For example, **{{**

}}. The NRC staff notes that **{{**

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TR Table 7-65, "Representative sensitivity studies – uncontrolled control rod bank withdrawal at power," provides the results of example sensitivity studies. While the results are not applicable to the NPM-20 design, they help to illustrate how the non-LOCA EM could be applied to identify the limiting conditions for the subcritical or low power control rod withdrawal cases (ML24305A000). During its audits, as documented in the audit associated report (ML24262A257), the NRC staff examined calculation documents implementing the topical report methodology in support of the NPM-20 SDAA. The NRC staff confirmed that limiting fuel centerline temperature cases correspond to **{{**

}}. The TR states that the reactivity insertion rates are examined **{{**

}} (ML24305A000). This helped the NRC staff to confirm that the applicant had defined an appropriate method for identifying the limiting reactivity insertion rate.

Based on the information submitted by the applicant, as confirmed by the NRC staff's review and information provided as part of the audits, as documented in the audit associated report, the NRC staff finds that the applicant's methodology for this event is consistent with SRP Section 15.4.2 and will ensure conservative results when implemented.

3.7.2.15 Control Rod Misoperation

TR Section 7.2.15, "Control Rod Misoperation," describes the control rod misoperation event-specific analysis methodology. For the NPM, three different scenarios are postulated, as defined in TR Section 7.2.15.1, "General Event Description and Methodology,":

- Withdrawing a single control rod assembly,
- Dropping one or more control rod assemblies, or
- Leaving one or more control rod assemblies behind when inserting or withdrawing a control bank.

Withdrawing a single control rod assembly inserts positive reactivity, and the transient is similar to the uncontrolled control rod assembly bank withdrawal event described in Section 3.7.2.14, "Uncontrolled Control Rod Assembly Bank Withdrawal at Power," of this SER except for the power asymmetry and lower reactivity insertion rate associated with the single rod withdrawal. Like the bank withdrawal event, the applicant stated that the limiting single rod withdrawal results when **{{**

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Dropping one control rod assembly adds negative reactivity, reducing the core power. The rod control system would normally attempt to restore the power level but cannot react quickly enough to preclude a reactor trip on high power rate. For some cases a high power-rate trip does not occur, and the reactor eventually returns to the initial power level. The phase of the transient in which the reactor returns to the initial power level is effectively a CRA bank withdrawal with a single CRA fully inserted, which may increase power peaking. However, the total reactivity insertion will be on the order of the negative reactivity insertion of the dropped CRA, and the combination of initial power level and total reactivity insertion are reduced compared to the transients discussed in SER Section 3.7.14 as high dropped CRA worth and high initial power cases tend to result in high power-rate trips. Based on the limited total reactivity insertion and potentially higher power peaking, the applicant evaluates rod drops that do not result in immediate rate trip to determine whether they are bounded by single CRA withdrawals.

TR Section 7.2.15.1, "General Event Description and Methodology," also notes that the high power-rate signal is based on the most limiting ex-core detector reading considering the asymmetry due to the single rod withdrawal or rod drop. The methodology specifies use the lowest- (for single rod withdrawal) or highest- (for single rod drop) reading ex-core detector and multiplies the core average power by the minimum (for single rod withdrawal) or maximum (for single rod drop) post-event to pre-event ratio of the radial peaking factors for the outer row of fuel assemblies.

For the condition in which one or more control rod assemblies do not move for a control rod bank demand, referred to as a control rod assembly misalignment, the applicant does uses the subchannel methodology rather than the non-LOCA methodology because it is a static event.

The NRC staff reviewed the applicant's methodology for this event to determine whether it specified appropriate biases and conservatisms for the applicable parameters, that the necessary acceptance criteria would be checked, and that the methodology as a whole would ensure conservative results when implemented.

TR Table 7-66, "Acceptance criteria, single active failure, loss of power scenarios – control rod misoperation," identifies MCHFR and maximum fuel centerline temperature as the acceptance criteria of interest for the control rod misoperation events, which is consistent with SRP Section 15.4.3 and therefore acceptable. While TR Table 7-66 includes high-level statements about loss of power scenarios, the specific electric power assumptions are reviewed as part of design-specific applications of this methodology, such as the NPM-20 SDA. This is reflected in item 6 of Section 4, "Limitations and Conditions," of this evaluation.

TR Tables 7-68, "Initial conditions, biases, and conservatisms – control rod misoperation, single control rod assembly withdrawal," and 7-70, "Initial conditions, biases, and conservatisms – control rod misoperation, dropped control rod assemblies," describe the initial conditions, biases, and conservatisms used in the evaluation of single control rod assembly withdrawal and rod drop events, respectively. The NRC staff reviewed the list of biased parameters and finds that the applicant's choice of parameters and bias directions to yield a conservative MCHFR and maximum fuel centerline temperature.

TR Table 7-69, "Representative sensitivity studies – control rod misoperation, single control rod assembly withdrawal," shows the results of sensitivity studies for the single control rod assembly withdrawal event (ML24305A000). While the results of these sensitivity studies are not applicable to the NPM-20 design, they help to illustrate how parameters will be varied to identify a set of potentially limiting single CRA withdrawal cases for subchannel analysis.

TR Table 7-71, "Representative sensitivity studies – control rod misoperation, dropped control rod assemblies," presents the results of example sensitivity studies. While these results are not applicable to the NPM-20 design, they illustrate how a user of the methodology would identify the limiting bias directions for the rod drop event. These sensitivity studies would be performed for rod drops that are not screened from transient analysis (ML24305A000). Based on the information provided as part of the NRC staff audit ML24262A257, the NRC staff confirmed that the methodology ensures CRA drops that do not result in an immediate rate trip are bounded by single CRA withdrawal with a method that conservatively accounts for power overshoot and variations in the reactivity insertion rate for bank withdrawal due to automatic control system response compared to the reactivity insertion rates modeled in the single CRA withdrawal analysis. The NRC staff finds that the sensitivity studies provided for audit with the SDA review for the NPM-20 design for the control rod misoperation events adequately demonstrate the process that may be used to screen non-limiting cases from transient evaluation to ensure that CRA drop consequences are bounded by the single CRA withdrawal transient.

Based on the sensitivity studies for a significantly large number of cases as shown in Figure 7-3 of the TR, the applicant categorized the drop cases into two groups: (1) cause reactor trip within a short period on the high power rate trip signal due to the negative reactivity inserted by the dropped CRA and (2) no immediate reactor trip. The TR provides an alternative approach for analyses of the rod misoperation events. The alternative approach screens rod drop scenarios from MCHFR or fuel temperature evaluations based on qualitative arguments that certain scenarios are bounded by either steady state operation or single CRA withdrawal scenarios. For the first group, the applicant states **{{**

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}}, then system transient analysis is performed, and subchannel analysis is performed to ensure that acceptance criteria are met.

Example results provided by the applicant demonstrate that this screening methodology is applicable to plants with a CRA drop time, rod power dependent insertion limit setting, and high power rate trip setting such that most CRA drops (particularly those from high initial power levels and with high dropped CRA worths) result in a reactor trip before reactor power begins to recover. The applicant justifies the aforementioned CRA reactivity screening criterion based because the CRA control system will only insert positive reactivity to compensate for the negative reactivity of the dropped CRA, resulting in minimal power overshoot. NRC staff finds that for plant designs in which the control system is tuned to ensure minimal power overshoot in the event of a CRA drop, the screening criteria are sufficient to provide reasonable assurance that MCHFR and peak LHGR occurring during plant response to a dropped CRA that does not promptly result in a reactor trip will be bounded by MCHFR and peak LHGR realized during single CRA withdrawal analysis.

The staff reviewed the alternative approach and finds that it provides reasonable assurance that rod drops are either bounded by other control rod misoperation scenarios or captures the fundamental event progression and consequences through system transient and subchannel analysis and therefore is acceptable.

Based on the information submitted by the applicant, as confirmed by the NRC staff's review and information provided as part of the NRC staff's audit, the NRC staff finds that the applicant's methodology for this event is consistent with SRP Section 15.4.3 and will ensure conservative results when implemented.

3.7.2.16 Inadvertent Decrease in Boron Concentration

TR Section 7.2.16, "Inadvertent Decrease in Boron Concentration," describes the inadvertent decrease in boron concentration event-specific methodology. The NRC staff reviewed the applicant's methodology for this event to determine whether it specified appropriate biases and conservatisms for the applicable parameters, whether the necessary acceptance criteria would be checked, and whether the methodology as a whole would ensure conservative results when implemented.

An inadvertent decrease in boron concentration is typically caused by failure of the blend system, either by controller or mechanical failure, or operator error. TR Section 7.2.16.1 states the event is terminated by isolating the diluted water source, which is accomplished by automatically closing the demineralized water system (DWS) isolation valves.

The inadvertent decrease in boron concentration event is evaluated for all the operational modes permitted in the plant Technical Specifications. For Mode 1 operation, a range of initial power levels between hot full power and 25 percent rated thermal power, as well as hot zero power are analyzed. The methodology specifies use of the perfect mixing and wave front models to determine the reactivity insertion rate in Mode 1. The perfect mixing model assumes instantaneous mixing and calculates a lower reactivity insertion rate that could delay detection, while the wave front model assumes mixing only at the CVCS injection point and calculates the maximum reactivity insertion rate as the concentration wave front sweeps the core. The NRC staff finds the use of these two models for Mode 1 acceptable because they show the two extremes of the reactivity insertion rates.

In Mode 1 at hot full power, the methodology states that the uncontrolled control rod bank withdrawal at power event results for the case with the same initial power, the same reactivity insertion rate, and the longest time to reactor trip is used to determine the time of reactor trip and isolation of the dilution source via closure of the DWS isolation valves. Calculations are performed with the perfect mixing model to determine the remaining available shutdown margin and the time shutdown margin would be lost if the dilution source was not terminated.

In the Mode 1 hot zero power case, the applicant stated that results from the uncontrolled control rod assembly withdrawal at low power startup conditions event corresponding to the reactivity insertion rates from both the perfect mixing model and the wave front model are used for the time of reactor trip and isolation of the dilution source via closure of DWS isolation valves. Calculations are performed using the wave front model to determine the remaining available shutdown margin and the time shutdown margin would be lost if the dilution source was not terminated.

During Mode 2 (Hot Shutdown) and Mode 3 (Safe Shutdown), the inadvertent decrease in boron concentration case in the NPM depends upon the RCS flow rate. The low RCS flow rate MPS signal is credited to isolate DWS if the RCS flow rate is less than a particular setpoint such as 1.7 ft^3/s (763 gpm). If the RCS flow rate is greater than or equal to that setpoint, the high count-rate signal is credited to isolate the DWS. Calculations are performed using the wave front model to determine the remaining available shutdown margin and the time shutdown margin would be lost if the dilution source was not terminated. The NRC staff finds the use of the wave front model for Modes 2 and 3 acceptable because the delays associated with the wave front model combined with reliance on a count rate trip produce a conservatively large total reactivity insertion, and shutdown margin degradation at the time the DWS isolation valves close.

Mode 4 is defined as Transition, and all CVCS connections to the NPM are disconnected, isolated, or locked out. This prevents an inadvertent decrease in boron concentration. **{{**

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The NRC staff reviewed the initial conditions, biases, and conservatisms for the inadvertent decrease in boron concentration event in TR Table 7-74, “Initial conditions, biases, and conservatisms – inadvertent decrease in boron concentration.” While many of the listed parameters are irrelevant due to not being part of the mixing model, the NRC staff confirmed that for the initial conditions whose bias directions are specified in TR Table 7-74, the bias directions are limiting for this event. The TR also states that studies are performed as needed to demonstrate the source of dilution is isolated before shutdown margin is lost. NRC staff audited calculations (ML24215A229) performed using this methodology that were submitted with the NPM-20 SDAA concurrent with this topical report. During the audit NRC staff observed that **{{** **}}** to terminate the event. As identified in L&C No. 6, review and approval of operator actions will be performed during the design-specific licensing review and is not within the scope of the staff’s evaluation of this TR. The TR also includes results for examples of such sensitivity studies performed for the NPM-160 design in TR Tables 7-75, “Representative results – inadvertent decrease in boron concentration in Mode 1 at hot full power,” through 7-79, “Representative results – inadvertent decrease in boron concentration in Mode 3.”

Based on the information submitted by the applicant, as confirmed by the NRC staff’s review, the NRC staff finds that the applicant’s methodology for this event will ensure conservative results when implemented.

3.7.2.17 Chemical and Volume Control System Malfunction that Increases Reactor Coolant System Inventory

TR Section 7.2.17, “Chemical and Volume Control System Malfunction that Increases Reactor Coolant System Inventory,” describes the event-specific analysis methods for the CVCS malfunction that increases RCS coolant inventory. The NRC staff reviewed the applicant’s methodology for this event to determine whether it specified appropriate biases and conservatisms for the applicable parameters, whether the necessary acceptance criteria would be checked, and whether the methodology as a whole would ensure conservative results when implemented.

Malfunctions in the charging (makeup) system or PZR level control system may result in the addition of makeup fluid, which will increase the PZR-water level. Reactor trip on high PZR-water level or high PZR pressure will typically result. For this event, the transient analysis conservatively assumes that the malfunction isolates letdown and actuates both makeup pumps at maximum capacity, which provides a bounding increase in RCS inventory.

TR Section 7.2.17.1, “General Event Description,” states that the full power initial condition is limiting, and the event is typically terminated by CVCS isolation (noting that the CVCS containment isolation valves are safety related) on high PZR level.

TR Table 7-81, “Acceptance criteria – reactor coolant system inventory increase,” assesses each of the non-LOCA FOMs relative to this event, and TR Table 7-80, “Acceptance criteria, single active failure, loss of power scenarios – reactor coolant system inventory increase,” identifies primary and secondary pressures as the acceptance criteria of interest. The NRC staff agrees that the pressures are challenged due to the postulated RCS inventory addition.

The NRC staff reviewed the initial conditions, biases, and conservatisms in TR Table 7-82, "Initial conditions, biases, and conservatisms – reactor coolant system inventory increase," for the CVCS malfunction that increases inventory. Five parameters/control system assumptions are varied **{{ }}**: initial RCS average temperature, initial RCS flow rate, initial PZR pressure and level, makeup temperature, and PZR spray operation. For the initial conditions whose bias directions are specified, the NRC staff confirmed that the bias directions are limiting for these events. For example, initial fuel temperature and reactivity and kinetics parameters are biased such that they would **{{ }}** resulting from addition of colder water to the RCS.

TR Table 7-83, "Representative sensitivity studies – reactor coolant system inventory increase," provides the results of the example sensitivity studies for the CVCS malfunction that increases inventory. While the results are not applicable to the NPM-20 design, they demonstrate the type of methodology that would be followed to identify the limiting biases for a licensing-basis calculation. The example sensitivity studies indicate that the interplay of the parameter biases may be an important consideration for analyses of this event (ML24305A000). Based on the information submitted by the applicant, as confirmed by the NRC staff's review, the NRC staff finds that the applicant's methodology for this event will ensure conservative results when implemented.

3.7.2.18 Failure of Small Lines Outside Containment

TR Section 7.2.18, "Failure of Small Lines Outside Containment," discusses the failure of small lines outside containment event-specific analysis methodology. The NRC staff reviewed the applicant's methodology for this event to determine whether it specified appropriate biases and conservatisms for the applicable parameters, whether the necessary acceptance criteria would be checked, and whether the methodology as a whole would ensure conservative results when implemented.

The failure of small lines outside of containment is assumed to occur in the CVCS since it is the only system in which primary coolant is carried outside of containment. These lines include makeup lines, letdown lines, PZR spray lines, and high point vent (degassing) lines. The non-LOCA methodology only considers failures in these lines outboard of containment isolation valves. Failures within the CVCS between the CNV and containment isolation valves are not addressed within this methodology or in the methodology for analysis of LOCAs (Reference 8); however, a Limitation and Condition in the corresponding safety evaluation for the LOCA methodology requires these failures in the CVCS lines to be addressed via regulatory compliance with 10 CFR 50.46(a)(1).

Failure of a spray line or high point vent line outboard of the containment isolation valves is typically less limiting in terms of mass and energy release than a break in a makeup line or letdown line so the non-LOCA EM excludes evaluation of spray or high point vent line breaks.

The release of reactor coolant resulting from the failure of either a CVCS makeup or letdown line outside containment causes a decrease in PZR pressure and level and a reactor trip on low PZR pressure or low PZR level. CVCS isolation (typically, low PZR level) terminates the fluid mass release from the reactor vessel. After the reactor trip, during the period up to CVCS isolation, the applicant states that mass and energy release is maximized by increasing the break area to include both lines because the mismatch between reactor heat generation and SG heat removal causes a level and pressure decrease nearly independent of break flow. Conversely, iodine spiking is maximized when the break is in a single location.

TR Table 7-85, “Acceptance criteria – breaks in small lines carrying primary coolant outside containment,” discusses the non-LOCA FOMs relative to the failure of a small line outside containment event. The NRC staff agrees with the applicant’s identification of radiological consequences as the acceptance criterion of interest in TR Table 7-84, “Acceptance criteria, single active failure, loss of power scenarios – breaks in small lines carrying primary coolant outside containment,” because the event postulates that RCS inventory is lost outside containment, and the event does not challenge other non-LOCA acceptance criteria.

The NRC staff reviewed the initial conditions, biases, and conservatisms in TR Table 7-86 that are considered in the methodology to identify a bounding transient simulation. The NRC staff finds that some of the initial conditions, biases, and conservatisms listed in Table 7-86 require sensitivity studies to find the limiting case for mass release and iodine spiking; and the initial fuel temperature will be varied **{{** **}}**, both of which provide a more rigorous evaluation. On these bases, the staff finds these acceptable. The reactor point kinetics parameters, however, are be set at nominal. The staff reviewed these parameters and finds setting them as normal is acceptable because they do not have significant influence on either mass release or iodine spiking. For the initial conditions whose bias directions are specified, the NRC staff confirmed that the bias directions are limiting for these events. Based on the information provided as part of the NRC staff’s audits (ML24262A257), the NRC staff confirmed that a biased-low initial RCS flow rate and a biased-high initial RCS average temperature are **{{** **}}**.

TR Section 7.2.18.3, “Biases, Conservatisms, and Sensitivity Studies,” states that sensitivity studies are performed as needed, varying break size and location, single active failures, loss of power assumptions, and parameters identified in Table 7-86, “Initial conditions, biases, and conservatisms – breaks in small lines carrying primary coolant outside containment,” to identify the limiting mass release and iodine spiking scenarios.

TR Table 7-87, “Representative break, time in life, power, flow, and temperature sensitivity study for mass release - breaks in small lines carrying primary coolant outside containment,” provides results of the applicant’s example sensitivity studies. While the results are not applicable to the NPM-20 design, the studies illustrate how sensitivity studies will be performed to identify the limiting conditions when the methodology is applied. In this example, the largest integrated mass release occurs for the 100 percent break of the letdown line full power plus the heat balance uncertainty with biased-high RCS average temperature and assuming a 100-percent break in the makeup line at the time of reactor trip. The maximum iodine spiking time case also assumes biased-high initial RCS average temperature (ML24305A000).

The NRC staff finds that performing the sensitivity studies by varying the break size and location, single active failures (per TR Section 7.1.4.1), loss of power assumptions (per TR Section 7.1.3.1) , and parameters identified in Table 7-86, “Initial conditions, biases, and conservatisms – breaks in small lines carrying primary coolant outside containment,” to identify the limiting response(s) for the acceptance criteria parameter(s) challenged by the event provides a bounding transient simulation.

Based on the information submitted by the applicant, as confirmed by the NRC staff's review and the information provided as part of the NRC staff's audit (ML24262A257), the NRC staff finds that the applicant's methodology for this event will ensure conservative results when implemented.

3.7.2.19 Steam Generator Tube Failure

TR Section 7.2.19, "Steam Generator Tube Failure," discusses the SGTF event-specific analysis methodology. The NRC staff reviewed the applicant's methodology for this event to determine whether it specified appropriate biases and conservatisms for the applicable parameters, whether the necessary acceptance criteria would be checked, and whether the methodology as a whole would ensure conservative results when implemented.

The failure of a SG tube causes the PZR pressure and PZR level to decrease at a rate dependent upon the size and location of the fault. A reactor trip may be generated on low PZR pressure or low PZR level assuming no loss of ac power at event initiation. The DHRS is eventually actuated, and the closure of the MSIVs and FWIVs terminates the release of mass and energy to the environment.

The SGTF size and location and the timing of the secondary side isolation determine the amount of radiological material potentially released to the environment. The methodology specifies performing sensitivity analyses for a range of break sizes and locations to determine the limiting cases and indicates that a break location at the top of the SG typically provides the greatest total mass release. The mass release from the primary system due to the SGTF is provided as an input to the downstream radiological dose calculation. The applicant states further that, alternatively, the radiological dose calculation may be evaluated using bounding assumptions as described in Section 4.3.6 of the topical report.

TR Table 7-91, "Initial conditions, biases, and conservatisms – steam generator tube failure," provides the initial conditions, biases, and conservatisms for the SGTF event. Several parameters are varied **{{** **}}**. The NRC staff concurs that the bias directions that are specified are appropriately conservative with respect to effect on the acceptance criteria. Sensitivity studies are also performed to identify the limiting scenarios for each of the acceptance criteria challenged by the SGTF event.

TR Table 7-92, "Representative break characteristics, initial conditions, loss of power, and single active failure sensitivity study - steam generator tube failure" shows example sensitivity studies performed for the SGTF event. While the results are not applicable to the NPM-20 design, these sensitivity studies illustrate the methodology for identifying conditions that result in limiting integrated mass release and iodine spiking time (ML24305A000).

Based on the information submitted by the applicant, as confirmed by the NRC staff's review and the information provided as part of the NRC staff's audit (ML24262A257), the NRC staff finds that the applicant's methodology for this event will ensure conservative results when implemented.

3.8 Representative Calculations

The TR Section 8.0, “Representative Calculations,” includes the results of several representative transient calculations as a demonstration of the analysis methodology based upon the logical framework established in earlier sections. Furthermore, the TR states that the results are limited to demonstrating the application of the non-LOCA methodology to the NPM and the applicant is not seeking approval of these demonstrative calculations. Therefore, the staff did not review Section 8.0 of the TR since these representative calculations are for illustration purpose and based on design features of NPM-160. In addition, since the NPM-20 SDAA was under staff review in parallel with this new revision, and since per L&C No. 1 the staff is limiting the application of this method to the NPM-20, the staff leveraged the review of those calcs performed for Chapter 15 of the SDAA in determining the that the method can be applied and produces adequate and expected results. As such, this SER does not include evaluation of the sample calculations.

3.9 Quality Assurance

In TR Section 9, “Quality Assurance,” the applicant describes how the NuScale QA TR and their implementing QAP are used to control the activities supporting this TR. They state that their QAP complies with the requirements of 10 CFR Part 50, Appendix B and is implemented using the guidance of ASME NQA-1 2008 and NQA-1a-2009 Addenda (Reference 6).

The SRP requires that the EM be maintained under a QAP that meets the requirements of 10 CFR Part 50, Appendix B. The TR references the NuScale QAP which is indicated to comply with the NRC requirements. Compliance with QA requirements is described in “NuScale Topical Report: Quality Assurance Program Description,” MN-122626-A. The NRC staff reviewed the Quality Assurance Program Description and documented its approval in its SER (ML16347A405). Further, the NRC staff inspected NuScale’s design control process and code development procedures, and these inspections are documented in the inspection report dated April 12, 2024 (ML24099A129).

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4 LIMITATIONS AND CONDITIONS

The TR provides a reasonable methodological framework for use in licensing applications in conjunction with the following limitations and conditions.

1. Use of the non-LOCA EM, Revision TR, Rev. X, is limited to evaluations of the NPM-20 design. An applicant or licensee seeking approval to use the non-LOCA EM TR, Rev. X 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 non-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.

2. The staff's approval is limited to the use of the non-LOCA EM with TR-0516-494-P-A, "Loss-of-Coolant Accident Evaluation Model," Revision X, XXX (MLXXXX), and any future changes or revisions to TR-0516-494-P-A, "Loss-of-Coolant Accident Evaluation Model," Revision X (MLXXX), must be assessed by the applicant for their potential impact on the non-LOCA EM. Any subsequent changes to the non-LOCA EM require NRC approval.

3. Use of the non-LOCA EM is limited to analyses of events described in non-LOCA EM TR Table 4-1, "Design basis events for which the non-LOCA system transient analysis is performed, event category, and event classification," up until the time when riser level uncovers due to RCS shrinkage, for the determination of primary and secondary pressures, and the potential for consequential loss of system functionality, as defined in the non-LOCA TR. The non-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 non-LOCA EM is not approved for use in evaluations for: inadvertent opening of an RPV valve, analysis of peak containment pressure and temperature response and thermal hydraulic instabilities in the secondary or primary system. It is also not approved for standalone evaluation of margin to SAFDLs, analysis of radiological consequences, control rod ejection accidents and evaluation of the long-term cooling phase and must be used in conjunction with separately approved EMs for those analyses.

4. The uncertainty in the model of the DHRS heat transfer has not been quantified and is not approved. An applicant or licensee seeking to apply this methodology to the NPM-20 must evaluate SG and DHRS heat transfer biases to determine if the treatment of uncertainty is justified based on margins to non-LOCA FOMs. Subsequent changes made to the NPM-20 also require this evaluation, except for changes to SSC physical/process input parameters only via established change control processes (such as 10 CFR 50.59) not otherwise requiring NRC approval. Future changes to the non-LOCA LTR methodology must include an evaluation of the uncertainty in DHRS heat transfer. An applicant or licensee seeking to apply this methodology to a design other than the NPM-20 must evaluate SG and DHRS heat transfer biases to determine if the treatment of uncertainty remains justified based on margins to non-LOCA FOMs.

5. An applicant or licensee seeking to apply this methodology to a design and take credit for the non-safety MSIVs must receive specific approval through that design review for crediting the non-safety MSIVs in analysis of a SGTF event, due to extension of NUREG-0138, Issue 1, to components protecting against primary side coolant loss.

6. 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 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.

7. 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 later NRELAP5 versions and/or new NPM-20 basemodel revisions if the revisions are demonstrated to produce either essentially the same or conservative results and are consistent with the approved methodology, or if the revision is to the basemodel and due to a change made to SSC physical/process input parameters only made via established change control processes (such as 10 CFR 50.59)).

   When NRELAP5 v1.7 and NPM model Revision 5, as described in the TR, are referenced in other EMs, those applications for use of NRELAP5 v1.7 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 v1.7 and NPM model Revision 5 are therefore approved only for the events listed in Table 4-1 of this TR.

   Option 134 of the NRELAP5 code, i.e. use of combination of ESDU and Dittus-Boelter correlations is not approved for use.

8. An applicant or licensee seeking to apply this methodology to a design must receive a separate approval through that design review for the values for the parameters and setpoints in Table 7-3 of this TR and must assess the potential impact on the event-specific bias directions. Any subsequent changes to the parameters and setpoints in Table 7-3 of this TR made by an applicant or licensee following the initial design approval must also be reassessed by the applicant or licensee for any potential impact on the event-specific bias directions. Any identified changes to the event-specific bias directions specified in the non-LOCA methodology require NRC approval.

9. An applicant or licensee seeking to apply this methodology to a design must receive a separate approval through that design review for inputs to the radiological consequence methodology that are not derived from transient analysis.

5 CONCLUSION

The NRC staff reviewed TR-0516-49416-P, revision, and the applicant's responses to staff RAIs and audited supporting documentation. As a result of this review, in accordance with the applicable NRC regulations documented in Section 2, "Regulatory Criteria," of this SER the NRC staff finds that the use of the NRELAP5 code with the non-LOCA analysis methodology described in the TR is appropriate for the non- LOCA safety analyses of the NuScale NPM design.

The non-LOCA TR uses many example values of input parameters to demonstrate the application of the non-LOCA EM to perform non-LOCA analyses. The TR includes analysis results for the sole purpose of enhancing the understanding of the analytical methods. Therefore, this SER does not approve the use of any specific example value input or result presented in the TR. In various subsections of this SER, the NRC staff documents the review of various input parameters and determines whether or not the related bias direction or assumptions are approved. The NRC staff would review and approve specific input values and ensuing results for the reactor design for the subsequent licensing submittals (e.g., SDAs) referencing the non-LOCA TR.

The NRC staff concludes that the non-LOCA methodology, as documented in TR Revision X, is acceptable for analysis of the non-LOCA events in the NPM-20 design subject to the limitations and conditions stated in Section 4, "Limitations and Conditions," of this SER.

6 REFERENCES

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4. Gartia, MR, Vijayan PK, and Pilkwal DS, "A Generalized Flow Correlation for Two-Phase Natural Circulation Loops," 18th National & 7th ISHMT-ASME Heat and Mass Transfer Conference, IIT Guwahati, India, January 2006.

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6. American Society of Mechanical Engineers, Quality Assurance Program Requirements for Nuclear Facility Applications, ASME NQA-1-2008, NQA-1 a-2009 Addenda.

7. NuScale Power, LLC, "Extended Passive Cooling and Reactivity Control Methodology," TR-124587-P-A, Revision X.

8. NuScale Power, LLC, "Loss-of-Coolant Accident Evaluation Model," TR-0516-49422-P-A, Revision X.