SAFETY EVALUATION REPORT BY OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT TR-0516-49416-P, REVISION 2

“NON-LOSS-OF-COOLANT ACCIDENT ANALYSIS METHODOLOGY”

NUSCALE POWER, LLC

PROJECT NO. PROJ0769

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1. INTRODUCTION

1.1 Background

By letter dated January 10, 2017, NuScale Power, LLC (NuScale), hereinafter referred to as “the applicant,” submitted Topical Report (TR) TR-0516-49416-P, Revision 0, “Non-Loss-of-Coolant-Accident Analysis Methodology,” (non-LOCA EM) to the U.S. Nuclear Regulatory Commission (NRC) for review and approval (subsequently reissued as Revison 1 on August 10, 2017 (Agencywide Documents and Access Management System (ADAMS) Accession No. ML17222A827), to redact unmarked proprietary information and to submit a publicly available version). The applicant supplemented its submittal by letter dated March 7, 2017 (ML17066A463). The NRC accepted the TR for review on April 27, 2017 (ML17116A063). The applicant submitted Revision 2 of the TR on November 26, 2019 (ML19331A516).

The applicant submitted the TR in support of the design certification application (DCA) for the NuScale Power Small Modular Reactor. 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 the 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) that were not addressed as part of TR-0516-49422-P, “Loss-of-Coolant Accident Evaluation Model” (LOCA EM) (ML19331B585).

The applicant requested approval to use the non-LOCA EM 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.
- Non-LOCA initiating events, including their classification.
- The applicable acceptance criteria for non-LOCA events.
- Interfaces with other analyses (i.e., 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) 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.
- Results of sensitivity studies.
- Representative results of NRELAP5 calculations.
- A brief description of the quality assurance (QA) procedures.

The 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. The PIRT follows the short-term non-LOCA event progression, which is divided into three phases: pre-trip transient, post-trip transient, and stable natural circulation. The applicant defined figures of merit (FOMs) for each phase that reflect non-LOCA acceptance criteria and important factors relative to the NPM 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 results of sensitivity studies and representative analyses.

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 NuScale DCA Part 2 Tier 2, Section 15.0.5, "Long-Term Decay Heat and Residual Heat Removal," and the potential longer-term progression of non-LOCA events including one control rod stuck out of the core is addressed in NuScale DCA Part 2 Tier 2, Section 15.0.6, "Evaluation of a Return to Power."

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). Furthermore, the TR does not consider the evaluation of the accident radiological source term and dose since these aspects are covered in "Accident Source Term Methodology," TR-0915-17565, Revision 3, NuScale Power, dated April 2019 (ML19112A172). However, according to the applicant, the interface of the non-LOCA system transient analysis with the downstream subchannel and radiological analyses is considered part of the non-LOCA EM.

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, return to power assuming the worst-case stuck control rod, and analysis of peak containment pressure and temperature response.

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 Table 4-1, "Design basis events for which the non-LOCA system transient analysis is performed, event category, and event classification," of the TR. It 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 Criteria," of this safety evaluation report (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 specfic 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 NuScale DCA.

This SER describes the NRC staff's review of the methodology as documented in the TR and its related documents. Section 2, "Regulatory Criteria," 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 CRITERIA

2.1 Regulatory Requirements

Regulations under 10 CFR 52.47, "Contents of applications; technical information," and 10 CFR 52.79, "Contents of applications; technical information in final safety analysis report," require an applicant to provide a final safety analysis report to the NRC that, in part, presents a safety analysis of the structures, systems, and components (SSCs) provided for the prevention or mitigation of potential accidents and of the facility as a whole. An applicant used approved transient and accident analysis methodologies (e.g., the non-LOCA EM) 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 DCA.

- 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. The applicant has requested an exemption from GDC 27 in the NuScale DCA and has proposed NuScale-specific Principal Design Criterion 27.

- 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 DCA 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 Part 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) 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, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” 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
- EM
- 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 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 Criteria,” 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 higher-level guidance in the SRP and DSRS as well as regulatory requirements to ensure that they are either addressed in following the EMDAP or are addressed in the EM documentation.

In conducting its review, the NRC staff issued requests for additional information (RAIs) when additional information was needed to assess compliance with regulatory requirements. In addition, 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. The details of those non-LOCA EM audits are available in audit reports ML19039A090 and ML20036C849, which provide summaries of the audits and information examined during them.

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 structures, systems and components (SSCs) that are not classified as “safety-related SSCs” as described in Title 10 of the *Code of Federal Regulation* (CFR), Section 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 General Design Criteria (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 and the LOCA EM, and the TR references the LOCA EM TR for those 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 1 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 analysis methodology, control rod ejection analysis methodology, and return to power analysis.

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 includes periods in which the [[

]], a phenomenon that is further discussed in Section 3.5.1 of this SER. The non-LOCA EM is no longer applicable when 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 could occur "well after reactor trip and engineered safety features have responded to the initiating event."

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 acceptable for the purposes of assessing non-LOCA EM adequacy.

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, which is a derivative of the RELAP5-3D computer code.

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 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 in RG 1.203.

3.2.2 Regulatory Requirements

TR Section 2.2, "Regulatory Requirements," identifies regulatory requirements and guidance relevant to the non-LOCA transient analyses, including several GDC, RG 1.203, and specific DSRS and SRP sections. The NRC staff concludes that the applicant has specified the appropriate regulatory requirements and regulatory guidance discussed in SER Section 2, “Regulatory Criteria.”

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

TR Section 3.1, “Description of NuScale Plant,” briefly describes the configuration and unique features of the NPM. The NPM is a small integral pressurized water reactor (PWR), with the reactor core, the two helical coil steam generators (SGs), and the pressurizer contained within the RPV. The reactor core is much smaller than that of operating large light-water PWRs, and the NPM operates on natural circulation without the need for reactor coolant pumps. The NPM safety systems are passive and do not rely on ECCS pumps, accumulators, tanks, or piping. The RPV is housed within a steel containment vessel (CNV), which is partially immersed in the reactor pool (or ultimate heat sink (UHS)) for cooling and decay heat removal purposes. One or more NPMs form a NuScale Power Plant. Each NPM has its own chemical and volume control system (CVCS), ECCS, and DHRS.

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 pressurizer sprays and heaters. The SGs normally 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 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.

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 time frame covered by the non-LOCA methodology. In addition, isolation of the CVCS, demineralized water system, and pressurizer heaters are 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 is capable of removing 100 percent of the decay heat load via a passive condenser in the reactor pool. 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 includes three reactor vent valves (RVVs) on top of the RPV and two reactor recirculation valves (RRVs) on the side of the RPV in the downcomer region. 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 ECCS valves open when the differential pressure of the spring-loaded arming valves, which function as an inadvertent actuation block (IAB), decrease below the release pressure. If the IAB threshold is reached, the ECCS 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. 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 following the DHRS actuation is addressed in TR-0916-51299-NP, Revision 1, "Long-Term Cooling Methodology," dated August 2019 (ML19218A147).

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.

In its description of the RCS, the TR states that the pressurizer includes a baffle plate, integral to the pressurizer and SG tube sheets, which acts as a thermal barrier and allows for surge flow between the pressurizer and the RCS. The NRC staff confirmed by audit, as documented in an audit summary (ML19039A090), that the design configuration, function, operation and performance of the integral pressurizer baffle plate are not critical to the analysis, as the nominal flow rate through the baffle plate has an insignificant potential for thermal stratification and little effect on natural circulation. The analysis of the limiting heat-up event indicates that the reactor safety valves (RSVs) limits pressurization, making the RCS pressure relatively insensitive to the in-surge rates. Therefore, the NRC staff concludes that the docketed information regarding pressurizer modeling is appropriate.

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 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 for the NPM. The categories according to the frequency of occurrence are: AOOs, IEs, and PAs. The categories according to the event type are: increase in heat removal from the RCS, decrease in heat removal by the secondary system, reactivity and power distribution anomalies, increase in the reactor coolant inventory, and 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 DSRS 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. 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, are as follows:

<u>AOOs</u>

- 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*.
- Maximum fuel centerline temperature ≤ melting temperature (adjusted for burnup effects)*.
- 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*.

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

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. However, when margins to acceptance criteria are determined not to be challenged, representative results from the sensitivity calculations are used to demonstrate margin without extensive sensitivity studies to minimize the margin to those unchallenged acceptance criteria. The applicant further clarified (ML18270A469) 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 Transient Analysis," and bias direction sensitivity studies

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* These acceptance criteria are evaluated by the downstream subchannel analysis and are outside the scope of the TR. However, as discussed in Section 3.4.3 of this report, a pre-screening process using NRELAP5 helps to determine the cases evaluated in the subchannel analysis.

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, consistent with SRP Section 15.0.2.

3.4.3 Non-LOCA Transient Analysis Process

TR Section 4.3, “Non-LOCA Transient Analysis Process,” describes the six steps of the NuScale non-LOCA transient analysis process, which are evaluated in the corresponding 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. 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 4.3.1.1.2, “Axial Power Shape,” states that sensitivity studies on the axial power shape, confirm that the primary and secondary system pressure, flow, and fluid temperature responses are not significantly affected by the axial power shape; therefore, the NRELAP5 non-LOCA system transient analyses use a nominal center-peaked average axial power shape. The NRC staff notes that a top-peaked axial power shape is typically limiting. Therefore, the NRC staff audited calculations to confirm that RCS and secondary pressures are insensitive to power shape, as discussed in the associated audit report (ML19039A090). The NRC staff concludes that the audited sensitivity studies adequately support the docketed information and finds the use of a nominal center-peaked average axial power shape for the NRELAP5 non-LOCA system transient analyses acceptable. The NRC staff further notes that the applicant applied the most limiting axial power shape when evaluating SAFDLs in the downstream subchannel analysis.

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 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 audited a sensitivity calculation that examined the effect of reducing the energy deposition factor through direct moderator heating and observed only small changes in the results, as discussed in the associated audit report (ML19039A090). The NRC staff concludes that the audited sensitivity studies adequately support the docketed information and finds the use of a bounding-high energy deposition factor for the NRELAP5 non-LOCA system transient analyses 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, as documented in the associated audit report (ML19039A090). The applicant clarified that it assumed burnup corresponds to an average value for a typical UO_2 core ranging from about 12 gigawatt-days per metric ton (GWd/MT) at beginning of cycle (BOC) to about 24 GWd/MT at end of cycle (EOC). The NRC staff confirmed that these values are consistent with those in the NuScale DCA but recognizes that they may change if the fuel design or operation strategy changes. 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.

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 Evaluate 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 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.

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. 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 MCHFR-minimizing biases for the boundary input conditions. These conditions include maximum reactor power, maximum core exit pressure, maximum core inlet temperature, and minimum system flow rate. The cases evaluated for MCHFR are run at the minimum flow, with other initial conditions set to the limiting initialization for a given transient progression such that power, RCS pressure, and core inlet temperature are simultaneously maximized prior to a reactor trip.

The NRC staff notes that the directions of conservatism for these parameters in the NPM are logical and consistent with those for typical large PWRs except for the maximum core exit pressure. For large PWRs, a lower pressure typically results in lower margins to the departure from nucleate boiling. The TR further explains that the effect of subcooling plays an important role in the CHF versus pressure trend, noting that the subcooling effect is dominant for high reactor coolant mass flux (as would be observed in a traditional PWR) due to decreasing enthalpy rise. This allows for greater power capacity with increasing pressure. **[[**

]].

The NRC staff confirmed that this described the behavior of the CHF versus the pressure in Figure 5-1, "CHF vs. pressure for Stern preliminary prototypic and KATHY K8500 HMP™," of TR-0116-21012, Revision 1, "NuScale Power Critical Heat Flux Correlations," dated December 2018 (ML18360A632). This figure shows **[[**

]]. Furthermore, the pressure bias (as well as the power, core inlet temperature, and RCS flow biases) is consistent with the biases described in the subchannel TR.

However, it is not always clear which combination of initial conditions and transient response will define the limiting case for the subchannel analysis. For this reason, the TR specifies that a spectrum of cases may be analyzed from the limiting initialization. **[[**

]]

As described in an audit report ML19039A090)), the applicant explained how the NRELAP5 pre-screening process ensures that the appropriate cases for downstream subchannel analysis are identified and that the limiting MCHFR value is determined. The applicant stated that TR Figure 4-1, **[[**

]]

The NRC staff finds that the limiting MCHFR cases for downstream subchannel analysis will be appropriately identified using the non-LOCA EM because the MCHFR predicted using NRELAP5 follows the same trend as VIPRE-01. Further, when in doubt, an analyst will pass several potentially limiting cases to VIPRE-01.

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)). One or more transient analysis cases are identified as limiting with respect to accident radiological analysis. The conservative bias directions for these cases 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.

Various interface information is provided 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.

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. Therefore, 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.

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 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 all non-LOCA event types 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 riser, natural circulation is interrupted, and it is the end of Phase 3) • Subcriticality (limits heat source to decay heat levels)

The TR notes 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. Based on the evaluation of the information provided by the applicant, the NRC staff agrees that this scenario is not encountered in the short term following a non-LOCA event. NuScale DCA Part 2 Tier 2, Section 15.0.5, “Long-Term Decay Heat and Residual Heat Removal,” and Section 15.0.6, “Evaluation of a Return to Power,” address the time after which mixture level has dropped below the top of the riser.

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. As described in an audit report (ML19039A090), the NRC staff audited the applicant’s engineering report documenting the PIRT, which includes these phenomena and the rationale for all rankings.

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. During the audit discussions, the applicant clarified that the non-LOCA PIRT was developed before the specific EMs and looked holistically at all phenomena that may be important when evaluating all aspects of a non-LOCA event. Parameters 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, so PIRT updates were not necessary.

In the NPM, phenomena such as **[[ ]]** are of particular interest, among others. These and other highly ranked phenomena were the subject of extensive audit discussions that clarified how the phenomena were appropriately considered, ranked, and addressed. These discussions are summarized in an audit report (ML19039A090).

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 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

Section 5.3, “NRELAP5 Validation and Assessments for Non-LOCA,” of the TR discusses the SETs, IETs, and code-to-code assessment performed to validate the NuScale non-LOCA EM beyond what was done as part of the LOCA EM development (though brief discussion of LOCA EM assessments that examine heat transfer from the RCS to the SGs or DHRS is included in the non-LOCA TR). 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.

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. This assessment was part of the LOCA EM development; as such, the applicant included greater detail on the assessment in the LOCA TR. However, since the assessment was relative to the behavior of DHRS, which is not credited in the LOCA analysis, the NRC staff’s review of the assessment is documented in this SER.

Tables 5-4, “Comparison between NuScale Power Module decay heat removal system and KAIST test section dimensions,” through 5-6, “Comparison between NuScale Power Module decay heat removal system and KAIST NRELAP5 model nodalization,” of the non-LOCA TR, compare the geometry, operating parameters, and NRELAP5 model nodalization of the KAIST experiments and the NPM DHRS. Although the geometries and operating ranges are not identical between the KAIST facility and the NPM, the NRC staff finds that there is sufficient similarity in the geometry **[[ ]]** and operating conditions (i.e., pressure, steam flow, steam temperature, and Reynolds number (Re)) which overlap for the purposes of validation.

RG 1.203 recommends that the nodalization and option selections be consistent between the experimental facility and similar components in the nuclear power plant. The applicant provided a nodalization diagram (ML18270A469) for the NRELAP5 simulation of KAIST tests. Considering the differences in geometric dimensions, the NRC staff finds that the KAIST facility nodalization diagram shows reasonable similarity to that used for the NPM non-LOCA application.

The NRELAP5 DHRS condensation heat transfer model applies **[[**

]] to compute condensation heat transfer inside the DHRS tubes. However, the NRC staff noted that **[[ ]]** are determined from forced convection experimental data for inertially driven flows through horizontal- and downward-flowing tubes. In contrast, the NPM DHRS operates with no pumps or compressors, but rather through buoyancy and/or phase change-induced flow generated by natural circulation.

The applicant stated (ML18263A311) that even though the NPM DHRS flow is passively driven, the system flow is not natural circulation in the strict sense because it is not driven by buoyancy or temperature gradients. Rather, the applicant stated that the pressure differential resulting from boiling in the SG and condensation in the DHRS condenser, in addition to the relative elevation of these components, drives DHRS flow.

The NRC staff, however, considers the DHRS loop to be a two-phase natural circulation loop, but does agree that temperature gradients and buoyancy are not likely to be the principal drivers of the DHRS flow because phase change appears to be the primary flow driver. The NRC staff considers that the large density changes resulting from the boiling across the SG and condensation across the DHRS condenser establish the pressure differential noted by the applicant and drive the flow.

In addition, the applicant stated that **[[** **]]** is based on a wide range of test data and has broad applicability. The NRC staff notes that **[[** **]]** capture phase change, and the applicant showed that the range of the NPM DHRS flow conditions is covered by **[[** **]]**. However, the test data and the applications are predominantly **[[** **]]**. With little exception in the heat transfer literature (e.g., References 1, 2, and 3), forced convection heat transfer models are developed as functions of a Re, while free or natural convection models are developed as functions of a Grashof number (Gr) or Rayleigh number (Ra).

Alternative formulations to **[[** **]]**. The applicant did not consider any alternative formulations to **[[** **]]**. Therefore, the NRC staff performed an independent literature review and found alternative formulations for two-phase natural circulation heat transfer that may more accurately represent the flow physics. However, the application of these alternative formulations is relatively recent, and their availability is limited. Furthermore, **[[** **]]** that more accurately reflects the flow physics. Consequently, the NRC staff concludes that **[[** **]]** can be applied to the DHRS condenser provided that they are validated with appropriate experimental data.

The applicant validated its use of **[[** **]]** based on the comparison with KAIST tests. Section 5.3.1.3, "Assessment Results," of the TR concludes that the use of **[[** **]]** provides a reasonable-to-excellent agreement between the KAIST experimental data and the NRELAP5 simulations. The NRC staff confirmed through an audit, as described in an audit report (ML19039A090), that the **[[** **]]** are because the location of maximum heat transfer depends on pressure, and the spectrum of KAIST tests covered several different pressures. In addition, a laminar-to-turbulent heat transfer regime transition occurs at different elevations depending on the heat and mass flow rate in the test.

The NRC staff notes that the KAIST tests, in combination with the NIST-1 HP-03 SETs, adequately cover the expected ranges of DHRS operation. The NRC staff agrees 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) shows reasonable to excellent agreement (generally within five percent). Therefore, the NRC staff agrees 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 NIST-1 facility is a scaled version of the NPM including an RPV, a helical coil SG with a DHRS, a CNV, and a cooling pool vessel (CPV). RG 1.203 indicates that an applicant should perform scaling analyses to ensure that the data and associated models will be applicable to the full-scale analysis of the plant transient. The TR states that the applicant performed NIST-1 scaling analyses as part of the LOCA EM development. Section 8.3.2, "NuScale Facility Scaling," of the LOCA TR summarizes the scaling analysis, which is evaluated in the NRC staff's SER for the LOCA EM (ML20044E199).

TR Section 5.3.2.2, "Decay Heat Removal System Separate Effects Tast Matrix," briefly describes the NIST-1 SETs, and Section 5.3.2.3, "NRELAP5 Model Description," describes the separate effects NRELAP5 model of the NIST-1 facility. In part, Section 5.3.2.3 states that **[[**

]] but preserving fluid and structural time constants is also important for the simulation to ensure fluid flow is characterized similarly between the NPM and the test facility. The applicant provided additional information to justify the nodalization scheme of the NIST-1 facility compared to the NPM (ML18269A360 and ML19221B483).

The applicant stated that **[[**

]].

The applicant also performed several calculations to assess sensitivity to the degree of nodalization in the DHRS, DHRS heat exchanger tube thickness, and DHRS performance (considering DHRS heat transfer bias and one-train DHRS operation). The nodalization sensitivity calculations for the NPM (using a representative loss of AC power transient) and HP-04 show that varying the axial nodalization **[[ ]]** does not significantly change the DHRS performance. Varying the DHRS heat exchanger tube thickness **[[ ]]** for the representative loss of AC power calculation resulted in a slight difference in the plant conditions at the transient endpoint but there was no impact on the margin to the non-LOCA acceptance criteria. Finally, the applicant assessed the impact of a ±30 percent DHRS heat transfer bias on specific transients: loss of AC power, increase in feedwater flow (SG overfill scenario), and feedwater line break inside the containment. The applicant also examined a loss of AC power event in which only one DHRS train is assumed to operate. These DHRS performance sensitivity studies showed that system pressures and temperatures differed at the end of the transient calculation, with lower DHRS performance leading to a slower RCS cooldown, as expected. However, the applicant demonstrated that the overall impact of variations in the DHRS performance on the FOMs and event progression until the point of DHRS actuation is insignificant, and DHRS cooling is effective regardless of the variations.

The applicant stated that the insignificance of the DHRS performance, with respect to peak RCS pressure and MCHFR values and timing, is largely because the minimum margin to the respective acceptance criteria occurs before the DHRS cooling becomes effective. In addition, the RSVs limit peak RCS pressure to approximately the same value for each of the most challenging RCS pressure transients, and variations in DHRS heat transfer would not change that. Peak secondary pressure typically results from the DHRS actuation and is influenced by secondary inventory and primary side conditions at the time of the DHRS actuation. Therefore, the DHRS performance has negligible impacts on the margin to secondary pressure limits.

Based on the applicant's DHRS sensitivity calculations and the fundamental reasons underlying the calculation results, the NRC staff finds that the applicant has adequately addressed nodalization differences between the NIST-1 and NPM design (e.g., tube wall thickness). The NRC staff concludes that the axial nodalization preserves and/or conservatively bounds the fluid and structural characteristic time response of the NIST-1 and NPM DHRS models. Furthermore, the NRC staff finds that variations in the DHRS heat transfer have a negligible effect on the non-LOCA FOMs for the NPM design represented in the NPM model, Revision 2, which supports the lack of a DHRS heat transfer bias as part of the non-LOCA EM. However, if the NPM design changes significantly (e.g., MPS logic changes, reduced margin to acceptance criteria), additional justification would be needed to confirm that the application of a DHRS heat transfer bias is not necessary. Therefore, the NRC staff included a limitation in Section 4 of this SER, that an applicant or licensee seeking to apply this methodology to a design other than the design represented in the NPM model, Revision 2 (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.

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. For these tests, the heated primary system produced steam in the SGs, which was supplied to the full-height DHRS test section at a range of pressures, DHRS inlet mass flow rates, CPV temperatures, and steam inlet superheat values. The steam was condensed in the DHRS condenser tubes, and DHRS pressure was maintained by a control valve in the condensate line that controlled the rate of condensate discharge to the atmosphere. TR Table 5-9, “NIST-1 HP-03 test cases,” provides the specific test run parameters. As discussed above, the NRC staff finds that the NIST-1 HP-03 SETs, together with the KAIST tests adequately cover the expected ranges of the DHRS operation.

TR Section 5.3.2.5, “HP-03 Results,” discusses the assessment results for the NIST-1 HP-03 tests. For all HP-03 tests, the TR states that there is reasonable-to-excellent agreement for the predicted versus measured DHRS heat removal, and the NRC staff agrees; based on the presented comparisons. However, the NRC staff observed a potentially lower level of agreement for other parameters. The NRELAP5-predicted DHRS heat exchanger level for all HP-03 tests is generally outside the uncertainty bands of the associated data (TR Figures 5-12, “NIST-1 HP-03-01 decay heat removal system level code-to-data comparison,” 5-17, “NIST-1 HP-03-02c decay heat removal system level code-to-data comparison,” and 5-22, “NIST-1 HP-03-03-Part1 decay heat removal system level code-to-data comparison”), and the magnitude of fluctuations in the predicted level for tests HP-03-02c and HP-03-03 Part 1, differs from that of the measured level. The NRELAP5 prediction of DHRS internal fluid temperature also falls outside of the measurement uncertainty for test HP-03-01 (TR Figure 5-13, “NIST-1 HP-03-01 decay heat removal system internal fluid temperature code-to-data comparison”), and there is a disagreement between the average values of the predicted and measured DHRS internal fluid temperature for tests HP-03-02c (TR Figure 5-18, “NIST-1 HP-03-02c decay heat removal system internal fluid temperature code-to-data comparison”) and HP-03-03 Part 1 (TR Figure 5-23, “NIST-1 HP-03-03-Part1 decay heat removal system internal fluid temperature code-to-data comparison”). In addition, NRELAP5 did not predict thermal stratification in the CPV that was observed in the HP-03-02c test (TR Figure 5-19a, “NIST-1 HP-03-02c cooling pool vessel temperature code-to-data comparison (2 of 2)”).

The NRC staff audited underlying calculation notes for the HP-03 tests, as documented in audit reports (ML19039A090 and ML20036C849). In addition, the applicant submitted information (ML18270A472) that discussed the modeling of primary-to-secondary heat transfer and DHRS heat removal mechanisms and concluded that the discrepancies the NRC staff noted, are not due to compensating errors in NRELAP5 models or correlations. 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 applicant stated that the SIET tests show a reasonable-to-excellent agreement for the FOMs. The NRC staff finds that the applicant has implemented appropriate HCSG models in NRELAP5, and the NRELAP5 predictions of SIET tests show a reasonable-to-excellent agreement to the data. However, the assessments considering primary-to-secondary heat transfer were limited in scope, ultimately resulting in Condition 3 in Section 4, "Limitations and Conditions," of this SER. This Condition is meant to ensure that the application of no 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 one represented in NPM model Revision 2 or any NPM model update made pursuant to an NRC-approved change process for the NPM model. The NRC staff's evaluation of the NRELAP5 assessment against the SIET tests, including further discussion of Condition 3, is in Section 3.5.3.5, "Steam Generator Modeling," of this SER.

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

The applicant also showed that condensate temperature is relatively insignificant by **[[**

]].

Furthermore, the applicant referred to additional information it provided (ML18263A311 and ML18240A378), which demonstrated **[[**

]] based on the comparison of code predictions against the KAIST test data.

Finally, the applicant stated that uncertainties during testing, such as **[[**

]], may contribute to the prediction discrepancies.

Based on the additional tests performed to assess DHRS behavior, the **[[**

]], and the propensity for test facility uncertainties to affect prediction results, as confirmed during the NRC staff's audits and described in the associated audit reports (ML19039A090 and ML20036C849), the NRC staff finds that the applicant provided sufficient bases to justify the ability of NRELAP5 to reasonably predict DHRS phenomena of interest for the NPM non-LOCA analyses.

The NRC staff noted that some of the data from the medium- and high-pressure tests HP-03-02c and HP-03-03 show oscillations that are not predicted well by NRELAP5. However, the NRC staff confirmed during its audits, as described in the associated audit report (ML19039A090), that these oscillations were due to **[[**

]]. This information clarified the behavior shown in the non-LOCA TR and the basis for the applicant's conclusion of reasonable agreement for the oscillating parameters.

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 applicant concluded that NRELAP5 can accurately predict the overall heat transfer from the DHRS to the CPV fluid and that DHRS power has a reasonable-to-excellent or excellent agreement with the data, while the DHRS level and CPV level show a reasonable-or-better agreement. Based on its review of the HP-03 SETs, the NRC staff finds that NRELAP5 predicts the relevant phenomena reasonably when compared to the test data. The NRC staff also finds that the NRELAP5 predictions of the most important parameter – DHRS heat removal – are in a reasonable-to-excellent agreement with the data.

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.

TR Section 5.3.2.7.1, "HP-04-02 Run," describes the lower-pressure **[[** **]]** HP-04-02 run. TR Figure 5-26, "NIST-1 HP-04-02 decay heat removal system energy transfer rate," compares the measured HP-04-02 DHRS heat removal rate and the NRELAP5 simulation result. The NRC staff characterizes the overall agreement as reasonable. Early in the test **[[**

]].
The NRC staff reviewed these sensitivities provided by the applicant (ML19221B483) and agrees with the applicant's conclusions regarding **[[** **]]**.

TR Section 5.3.2.7.1, "HP-04-02 Run," states that NRELAP5 does not fully capture the experimentally observed DHRS condensate outlet temperature profiles for the HP-04-02 test. The NRC staff noted that the prediction (TR Figure 5-27, "NIST-1 HP-04-02 decay heat removal system condensate temperature") **[[**

]], the code captures the correct net energy transfer to CPV. The NRC staff considered the potential for compensating errors, resulting in a reasonable agreement in the DHRS heat removal rate for the HP-04 test series, as described in Section 3.5.3.2.1, "NIST HP-03 Separate Effects Tests," of this SER.

In addition, the NRC staff audited a sensitivity study performed by the applicant, as described in the associated audit report (ML19039A090), that **[[**

]]. This helps to confirm the applicant's docketed conclusions regarding the reasonable agreement of the DHRS heat removal rate despite the lack of agreement in the CPV fluid heatup profiles.

Furthermore, the NRC staff notes that the DHRS internal collapsed level and CPV level (Figures 5-28, "NIST-1 HP-04-02 decay heat removal system internal collapsed level," and 5-29, "NIST-1 HP-04-02 cooling pool vessel level," respectively) show a reasonable-or-better agreement between the predicted and measured values. The applicant indicated that these parameters have a strong influence on the heat transfer across the DHRS tubes, which also helps to explain the reasonable agreement observed for the DHRS heat removal rate.

TR Section 5.3.2.7.2, "HP-04-03 Run," describes the HP-04-03 test run, which was performed at higher DHRS pressures of **[[ ]]**. In general, the comparison of test data to NRELAP5 predictions is similar to that for the HP-04-02 test run. However, some parameters show a slightly better agreement for HP-04-03, including the DHRS heat removal rate.

In TR Section 5.3.2.7.3, "HP-04 Summary," the applicant concluded that NRELAP5 should be expected to adequately predict the DHRS heat removal rates for a large range of CPV liquid conditions. The NRC staff finds these conclusions to be acceptable based on the reasonable agreement between the NRELAP5 simulations and the test data for key parameters, the most important being the DHRS heat removal rate.

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 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 5-43, "NLT-02a transient pressurizer pressure comparison," 5-45, "NLT-02a transient pressurizer level comparison," and 5-46, "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 5-44, “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 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 or better, 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 cover the integral plant response from DHRS actuation to DHRS-driven cooling and depressurization. In this test, **[[**

]]. The NLT-02b test and NRELAP5 simulation were divided into four time-intervals, or phases, **[[**

]].

The applicant explained (ML18299A322) its modeling approaches related to some uncertain test facility conditions during all phases of the NLT-02b test, including secondary side inventory, RPV heat losses, and DHRS condensate line resistance. For secondary side inventory, the applicant noted that the test data indicated **[[**

]]. The NRC staff finds the above modeling approaches to be a reasonable way to address the related uncertainties.

TR Section 5.3.3.7, “NLT-2b Phase 1 Test Results,” compares NRELAP5 predictions to Phase 1 of the NLT-02b test. Phase 1 comprises the first **[[ ]]** of the test and consists of terminating feedwater flow and power to the core heater rods, DHRS actuation, and initial DHRS cooldown. In the first approximately **[[ ]]**, several model parameters deviate from the data by more than the measurement uncertainty. However, after approximately **[[ ]]** when the data appear to be quasi-steady, most key model parameters are in a reasonable-to-excellent agreement with the data.

Pressurizer level (TR Figure 5-55, “NLT-02b phase 1 transient pressurizer level comparison”) is not well predicted over most of Phase 1, which the applicant attributed to **[[**

]]. In addition, the NRC staff notes that the predicted DHRS power (TR Figure 5-61, “NLT-02b phase 1 transient decay heat removal system heat exchanger thermal power comparison”) is **[[**

]]. The NRC staff concluded that the **[[**

]] DHRS condensate temperature (TR Figure 5-65, “NLT-02b phase 1 transient decay heat removal system 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. SER Section 3.5.3.2.2, “NIST HP-04 Separate Effects Tests,” also describes the sensitivity studies that the applicant performed **[[**

]]. In addition, the applicant (ML18285A926) attributes **[[**

]].

TR Section 5.3.3.8, “NLT-2b Pase 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 decay heat removal system 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 5-82, “NLT-02b phase 2 transient decay heat removal system condensate temperature comparison,” respectively) **[[**

]]. The predicted DHRS condensate flow (TR Figure 5-83, “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.

TR Section 5.3.3.9, “NLT-2b Phase 3 Test Results,” describes NLT-02b Phase 3, which **[[**

]].

The level of agreement of Phase 3 key parameters is generally similar to that of parameters during Phases 1 and 2. The NRC staff agrees with the applicant that most predictions of key parameters show a reasonable-to-excellent agreement with the data for Phase 3, including the DHRS power. Again, the DHRS condensate temperature and the CPV temperature response are not well predicted by NRELAP5.

TR Section 5.3.3.10, “NLT-2b Phase 4 Test Results,” describes NLT-02b Phase 4. During this phase, **[[**

]]. A similar level of agreement in parameters is observed for Phase 4 as in the previous phases with a couple of exceptions. **[[**

]]. The predicted SG and DHRS power follow the trends of the data but [[

]]. Still, the agreement is reasonable, especially considering the relatively low magnitudes of these parameters.

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

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-15p2, transient RPV pressure short term"). [[

]].

Predicted values for pressurizer level (TR Figure 5-129, "NLT-15p2, transient pressurizer level") and RPV level (TR Figure 5-130, "NLT-15p2, 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 5-132, "NLT-15p2, transient core inlet temperature," through 5-134, "NLT-15p2, transient upper plenum temperature") are in reasonable to excellent agreement with the data.

The peak SG pressure (TR Figure 5-135, "NLT-15p2, transient secondary side pressure - 0 to 500 seconds") was [[

]]. The applicant stated that predicted SG (TR Figures 5-145, “NLT-15p2, transient steam generator tube coil level - long term,” and 5-146, “NLT-15p2, 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-15p2, 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-15p2, transient DHRS condensate line differential pressure”) and steam line (TR Figure 5-148, “NLT-15p2, transient DHRS steam line differential pressure”) to be **[[**

]]. The NRC staff agrees with the applicant’s assessment of these parameters because **[[**

]].

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

]]. The simulated SG power (TR Figure 5-149, “NLT-15p2, transient steam generator tube coil power removal”) and DHRS power (TR Figure 5-150, “NLT-15p2, 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.

For NIST-1 test NLT-02a, the NRC staff finds that there was reasonable to excellent agreement between the simulation and the data, with the riser mass flow rate showing the most deviation but still following the trend of the data. For test NLT-02b, the NRC staff finds that there was reasonable agreement with the data for pressurizer pressure, excellent agreement in the core inlet and outlet temperatures, and reasonable agreement in the SG pressure and power and DHRS power. The NRC staff finds that the applicant adequately justified instances in which the data and simulation were not in excellent or reasonable agreement, such as CPV temperature profiles.

TR Section 5.3.3.14,"NLT-15p2 Summary," summarizes the assessment of NRELAP5 against NIST-1 NLT-15p2 test data. Based on the model-to-data comparisons in the TR and the above evaluation, the NRC staff agrees that NRELAP5 predicted the important phenomena in the NIST-1 NLT-15p2 reasonably.

In summary, the NRC staff finds that the assessment of NRELAP5 against the NIST-1 IETs demonstrates that NRELAP5 can acceptably predict the plant thermal-hydraulic response to non-LOCA events.

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. RETRAN-3D is a general-purpose thermal-hydraulic code that the NRC staff has approved for generic use (ML010470342) and for plant-specific applications (e.g., ML18060A401). Therefore, the NRC staff agrees that the comparisons of NRELAP5 to RETRAN-3D results are useful as part of the NRELAP5 assessment.

The applicant based its RETRAN-3D model on the NRELAP5 model for the NPM, with differences such as **[[ ]]**. In addition, RETRAN-3D does not include specific models for HCSG heat transfer and wall friction. Therefore, the applicant introduced a modeling simplification **[[ ]]** such that the RETRAN-3D primary side heat transfer coefficients matched those calculated by NRELAP5 under steady-state conditions.

Differences between the code calculation results were attributed to differences in the nodalization and the codes calculation capabilities such as pressurizer performance.

The applicant compared simulation results for four reactivity events: uncontrolled rod withdrawal from full power, using a higher reactivity insertion rate; uncontrolled rod withdrawal from full power, using a lower reactivity insertion rate; power reduction from full power to 50 percent of rated power; and dropped control rod assembly from 50 percent rated power. The applicant clarified during the audits that the rod withdrawal scenarios are meant to simulate ranges of bank withdrawals representative of those in the DCA, as documented in the associated audit report (ML19039A090).

Prior to simulating the four transient reactivity events, the applicant performed RETRAN-3D and NRELAP5 simulations to obtain steady-state initial conditions. The NRC staff notes that the RETRAN-3D and NRELAP5 core flow and temperatures are not identical at the beginning of the transient because these parameters are not input by the user but are rather computed as part of the steady-state solution. Therefore, small differences between the RELAP5 and RETRAN-3D steady-state results are expected, and the NRC staff focused on the relative behavior of these parameters for the transient calculations.

As discussed in the associated audit report (ML19039A090), the applicant clarified that the appearance of zero initial reactivity insertion in TR Figures 5-159, “Total reactivity (fast uncontrolled rod withdrawal),” and 5-166, “Total reactivity (slow uncontrolled rod withdrawal),” is due to the ordinate scale. The reactivity addition due to reactor trip obscures the smaller reactivity addition due to rod withdrawal.

Overall, the NRC staff agrees with the applicant that the NRELAP5 predictions are in reasonable or better agreement with those from the RETRAN-3D code. NRELAP5 and RETRAN-3D predict similar power and reactivity responses, with the largest difference being that NRELAP5 predicts a slightly later trip for the uncontrolled rod withdrawal events. The applicant attributes the difference in timing to the difference in initial conditions calculated by the two codes. In general, the trends for RCS flow, core inlet temperature, and core outlet temperature are in good agreement considering the difference in initial conditions. The predictions of pressurizer pressure and level are in poorer agreement than the other parameters due to differences in pressurizer modeling between NRELAP5 and RETRAN-3D but still agree reasonably. In conclusion, the NRC staff finds that the comparisons between NRELAP5 and RETRAN-3D provide confidence in the ability of the NRELAP5 point kinetics model to acceptably predict reactivity feedback.

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 (TR-0516-49422), none are presented in non-LOCA TR. Furthermore, heat transfer from the HCSG to the DHRS is not credited in the LOCA EM, while it is credited in the non-LOCA EM. 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.

Section 6.7, “Helical Coil Steam Generator Component,” of the LOCA TR states that a new 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 [[

]]. Sections 7.4.2.4, "Special Analysis Techniques," and 7.4.2.5, "Assessment Results," of the LOCA TR note that [[

]], which the NRC staff evaluated in Section 3.6.1, "Thermal-Hydraulic Volumes and Heat Structures," of this SER.

The non-LOCA TR specifies that the NRELAP5 HLCOIL component is used to model the HCSG, and the NRELAP5 heat structure 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 [[

]].

The NRC staff agrees with the applicant 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.

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 appllicant'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 **[[**

]].

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 **[[**

]] over 40 percent MCHFR margin compared to the 95/95 limit for the limiting overcooling event in the NuScale DCA 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 RSV lifts to mitigate high reactor pressure, **[[**

]].

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

]].

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 **[[**

]].

Based on its review of the sensitivity study information, as confirmed in audits and described in the associated audit reports (ML20036C849), the NRC staff agrees that the FOMs for non-LOCA events are insensitive to reasonable variations in SG heat transfer for the NPM design described in the NuScale DCA.

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 DCA Part 2, Tier 2, Chapter 16, which are based on the values supported by the safety analysis, specifically DCA Part 2, Tier 2, 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 as described in the NuScale DCA. Because the applicant's sensitivity results were based on this particular design, the NRC staff requires additional justification 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 DCA. This is reflected in Condition 3 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.

3.5.4 <u>Conclusions of NRELAP5 Applicability for Non-LOCA</u>

TR Section 5.4, "Conclustions of NRELAP5 Applicability for Non-LOCA," summarizes the applicant's conclusions regarding the applicability of the NRELAP5 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 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.

3.6 NuScale NRELAP5 Plant Model

TR Section 6, "NuScale NRELAP5 Plant Model," describes how plant components and features are simulated by the NRELAP5 NPM non-LOCA transient model. 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 6.0, "NuScale NRELAP5 Plant Model," states that the non-LOCA model was developed following the NRELAP5 code manual user guidelines, supplemented by NuScale-specific modeling guidelines. Based on information reviewed as part of the NRC staff's audits and described in the associated audit report (ML19039A090), the NRC staff confirmed that the applicant did not maintain a separate, standalone modeling guidance document for non-LOCA analysis. However, only one plant design is used for the model, and any changes to the model options are documented as part of the design basis event calculations.

3.6.1 <u>Thermal-Hydraulic Volumes and Heat Structures</u>

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 figures showing nodalization.

Figure 6-2, "Typical primary and secondary side nodalization (heat structures and component cell details excluded)," presents a "typical" nodalization diagram that is meant to convey the overall structure of the model. **[[**

]] The NRC staff finds, consistent with the guidance in RG 1.203, that the nodalization described in the TR as detailed above provides an acceptable description of the nodalization used in the EM.

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

]].

During its audits, as documented in the associated audit report (ML19039A090), the NRC staff confirmed that **[[**

]].

The NRC staff agrees with the applicant’s assessment that **[[**

.**]]** The NRC staff noted that CHF is directly affected by natural circulation flow. Discussion of the 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 helical coil steam generator 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 past the SG, as long as the axial nodal resolution is sufficient to capture the thermal gradient along the flow path on both sides of the SG. The NRC staff confirmed that the SG model described in the TR provides sufficient axial nodal resolution to capture the thermal gradient along the flow path on both sides of the SG.

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 differential pressure for 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 **[[** **]]** were 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 DCA preserve a 5-degree F subcooling margin through the MPS high hot-leg and low pressurizer pressure trips, the NRC staff finds that the **[[** **]]** are 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 in an audit, 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).

Section 6.1.1, “Reactor Primary,” of the TR, states that **[[**

]]. As indicated above, several highly ranked phenomena identified by the applicant are related to multi-dimensional flows and complex flow behavior **[[** **]]**. The applicant described (ML18234A537) expected multi-dimensional flow and thermal behaviors in **[[**

]].

The results of the CFD calculations that the NRC staff audited, as documented in the associated audit report (ML20036C849), show that the coolant **[[**

]]. The applicant used these calculations, as compared to the NRELAP5 one-dimensional simulation, to conclude the acceptability of the NRELAP5 nodalization and analyses. Based on its review of the information, as confirmed by the audit, the NRC staff finds that **[[**

]] is adequately addressed by the non-LOCA EM.

According to TR Section 6.1.1, "Reactor Primary," **[[**

]]

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 a nominal average axial power shape with power distributed solely in the fuel pellet, 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, "Decay heat removal system division 1 nodalization," shows the nodalization for one of the DHRS trains. The TR states that **[[**

]]

The TR also states that **[[**

.**]]**

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 sensitivites concluded: **[[**(

]]. Based on its review of the applicant's sensitivity analyses, the NRC staff finds that the applicant's consideration of the effect of pool temperature and thermal stratification on the performance of the DHRS heat exchanger is acceptable.

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 **[[**

]].

In a sensitivity study for a representative loss of AC power event, 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 steam generator peak pressure **[[**

]].

The NRC staff reviewed the sensitivity studies 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 **[[**
]].

TR Section 6.1.6, “Emergency Core Cooling System,” describes the modeling of the ECCS in the NPM plant model. **[[**

.**]]** The NRC staff finds the ECCS modeling acceptable because these junction orientations maintain the fluid momentum associated with the downward physical orientation of the RRV into containment in the NPM design.

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 the NuScale design and is therefore acceptable.

TR Section 6.1.8, “Reactor Cooling Pool,” discusses the NRELAP5 representation of the reactor cooling pool. **[[**

]] The NRC staff finds this description adequately reflects the NuScale design, and the modeling of the reactor cooling pool is therefore acceptable.

The NRC staff finds that the description of the NuScale NRELAP5 Plant Model provided in TR Section 6.1, “Thermal-Hydraulic Volumes and Heat Structures,” accurately reflects or reasonably approximates the NuScale design and is therefore acceptable.

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. 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 current design.

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 pressurizer 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. TR Table 6-2, "NuScale Power Module safety logic with NRELAP5 signals in bold," is a representative list of MPS functions and signals for the NPM. The NRC staff notes that this list may not be fully consistent with a specific design (e.g., fewer MPS signals actuate the DHRS in the design described in NuScale DCA Revision 3 than are shown in TR Table 6-2); however, 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 DCA.

3.7 Non-LOCA Analysis Methodology

3.7.1 General Aspects of Non-LOCA Methodology

Section 7, "Non-LOCA Analysis Methodology," of the TR 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).

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 EDSS provides uninterrupted DC power for 72 hours to essential loads while shedding low importance or non-essential loads. EDSS batteries shed power to the ECCS valves after 24 hours such that these valves open when the RCS to CNV differential pressure is less than the IAB pressure. The applicant states that discharging to reactor coolant after 24 hours is not relevant to the short-term FOM addressed by this report (MCHFR and RCS and steam generator 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, TR-0516-49422, and discharge of reactor coolant after 24 hours is addressed by the Long-Term Cooling Methodology, TR-0916-51299.

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 highly reliable DC power system (EDSS)) 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 NuScale DCA. This is reflected in item 5 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, except for the IAB valve described below. 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-specifc basis as part of the application of this methodology to a specific design, such as the NuScale DCA Chapter 15 review.

The IAB valve is a first-of-a-kind, safety-significant, active component integral to the NuScale ECCS. In order to meet the requirements for the ECCS in 10 CFR Part 50, an applicant must show that it has addressed the single failure criterion (SFC). The SFC is defined in 10 CFR Part 50 Appendix K and derived from the definition of single failure in 10 CFR Part 50 Appendix A. During its review, the NRC staff noted that although the applicant assumed a single failure of a main ECCS valve to open, the applicant did not apply the SFC to the IAB valve in regard to the valve’s function to close. NuScale disagreed with the NRC staff’s application of the SFC to the IAB valve, which led the NRC staff to request Commission direction to resolve this issue, SECY-19-0036 “Application of the Single Failure Criterion to NuScale Power LLC’s Inadvertent Actuation Block Valves.”[†] 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,[‡] in which it informed the Commission how the NRC staff then generally applied the SFC, and, SECY-94-084,[§] in which the NRC staff requested Commission direction on 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 Commission direction on application of the SFC to the IAB valve’s function to close.

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

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

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

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

Based on the NRC staff's historic application of the SFC and Commission 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 single failure criterion. 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 addressed the use of the SFC and provided clarification on when to apply the SFC in other specific instances.

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 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 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. The initial condition biases are generally consistent with ranges expected during normal operation, accounting for steady-state fluctuations and calibration and instrument errors. 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, as discussed in the associated audit report (ML19039A090), the NRC staff confirmed that **[[**

]].

Table 1 summarizes the initial conditions and their example biased ranges that are extracted from TR Section 7.1.6.1, "Initial Conditions." The limiting biases are to be prescribed as part of the event-specific analysis methodology. The staff emphasizes that these are example values, and specific bias values applied in a licensing-basis calculation may vary depending on design-specific considerations.

Table 1 Initial Condition Biases

Parameter	Upper Range	Lower Range	Nominal Condition
Initial Core Power	+2%	-2%	100% Power**
Initial RCS Ave. Temperature	+10˚F	-10˚F	Nominal for power
Initial Pressurizer Pressure	+70 psi	-70 psi	Nominal for power

** *"The initial core power is biased high by an amount equal to the heat balance uncertainty."*

* *RTP – Rated Thermal Power*

Initial Pressurizer Level	+8%	-8%	Nominal for power
Initial Containment Pressure	2.0 psi	0.037 psi	High OR Low
Initial Steam Generator Pressure	+35 psi	-35 psi	Nominal for power
Initial Feedwater Temperature	+10˚F	-10˚F	Nominal for power
Initial RCS Flow Rate	690 kg/s	535 kg/s	590 kg/s for 100% RTP
Initial Core Ave. Temperature	1065˚F	960˚F	BOC Example
Initial Reactor Pool Temperature	200˚F	40˚F	High OR Low

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, "Examples of analytical limits and actuation delays (reactor trip system and engineered safety features actuation system)," 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 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. Therefore, the NRC staff finds the general description of the biasing of non-reactivity parameters appropriate and acceptable.

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:

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

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

3) 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 any operator action credited in non-LOCA transient analyses should be justified and consistent with operating procedures. However, operator action is not credited for any non-LOCA event for the NPM.

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 consisitent 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 4 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.

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 DCA. This is reflected in item 5 in Section 4, "Limitations and Conditions," of this evaluation.

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 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 DCA. This is reflected in item 5 in Section 4, "Limitations and Conditions," of this evaluation.

TR Section 7.2, "Event Specific Methodology," notes that initial RCS flow is biased low for most events since it is limiting for MCHFR. The applicant stated that **[[**

]].

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 **[[**

.**]]**

The NRC staff reviewed the results of the representative calculations in TR Section 8 with respect to the **[[**

]].

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 Termperature," 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. 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. The TR states that sensitivity studies are performed to identify the limiting feedwater cooldown rate, 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 the high power analytical limit is increased for overcooling events to account for the decalibration of the ex-core neutron detectors as downcomer density changes. The amount of increase is to be based on **[[**

]]. While the specific amount of increase is design-specific and not within the scope of the NRC staff's TR review, the NRC staff finds the overall approach for calculating it acceptable as long as the **[[**

]]. The NRC staff also finds the increase in high power analytical limit necessary to ensure a conservative calculation of overcooling transient response.

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 stated 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 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, initial RCS average temperature, and pressurizer pressure are conservative because these biases are consistent with known directions of conservatism for MCHFR (ML19067A256). In addition, the EOC moderator temperature coefficient 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.

The PCS function of automatic rod control is enabled, **[[**

]]. The NRC staff finds this assumption to be conservative because it will exacerbate the cooldown event.

The applicant presented sample results from sensitivity studies in TR Tables 7-8, "Representative fuel exposure study," through 7-10, "Representative feedwater temperature transient study," to demonstrate the effects of fuel exposure, initial fuel temperature, boundary condition type, and the single active failure of an MSIV to isolate. The results of the fuel exposure and initial fuel temperature sensitivity studies support the respective bias directions specified in TR Table 7-7, "Initial conditions, biases, and conservatisms – decrease in feedwater temperature."

The applicant also presented results from an example sensitivity study of feedwater temperature cooldown rate, which demonstrated that CHFR is minimized when the high-power trip is coincident with the riser high temperature trip. While faster feedwater cooldown leads to a high-power trip and slower cooldown leads to a high riser temperature trip, both result in slightly higher MCHFR than the case of both trips occurring simultaneously.

Based on information reviewed as part of the NRC staff's audits and discussed in the associated audit report (ML19039A090), 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 steam generator tube plugging is biased low for all sensitivity cases, which **[[**

]].

Based on the information submitted by the applicant, as confirmed by the NRC staff's review and audit, as documented in the associated audit report, 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, or high steam line pressure). Secondary system isolation ends the overcooling event, and DHRS provides decay heat removal.

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. 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. TR Tables 7-15, “Representative increase in feedwater flow study – high SG performance with maximum power and minimum RCS flow,” and 7-16, “Representative increase in feedwater flow study – low SG performance with maximum power and minimum RCS flow,” provide an example of representative sensitivity studies that might be performed to ascertain the limiting bias directions for an application of the increase in feedwater flow methodology. The representative sensitivity results indicate that **[[**

]].

Based on the information submitted by the applicant, as confirmed by the NRC staff’s review and audit, as documented in the associated audit report (ML19039A090), 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. Due to this event progression, the NRC staff agrees that MCHFR is the primary acceptance criterion of interest for this event.

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.

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 **[[**

]].

The representative 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. The results indicate that **[[**

]]. Based on information reviewed as part of the NRC staff’s audits and discussed in the associated audit report (ML19039A090), the NRC staff confirmed that differences in secondary side behavior result in a relatively small increase in steam flow being limiting for MCHFR for this event, compared to a relatively large increase in feedwater flow for the increase in feedwater flow event.

Based on the information submitted by the applicant, as confirmed by the NRC staff’s review and audit, as documented in the associated audit report, 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 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.

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, as documented in the associated audit report (ML19039A090), 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 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 Methdology," 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 DCA.

TR Table 7-24, "Initial conditions, biases, and conservatisms – steam line break," shows that most parameter initial conditions are 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 representative results for example sensitivity studies for the steam line break in terms of the integrated mass release and MCHFR. This example illustrates how a user of the methodology identify the parameter biases and assumptions that provide a bounding transient simulation.

Based on the information submitted by the applicant, as confirmed by the NRC staff's review and audit, as documented in the associated audit report, 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 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 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 stated 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. Some loss of containment vacuum/containment flooding cases result in reactor trip on high containment pressure, while others do not trip.

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.

TR Table 7-29, "Example sensitivity studies – containment flooding / loss of containment Vacuum," provides the NRELAP5 MCHFR estimates for example sensitivity studies for the containment loss of vacuum and the containment flooding events. The example studies suggested that the plant response for a containment flooding event bounds that of a loss of containment vacuum event. The NRC staff reviewed the applicant's supplementary information (ML18205A804) and determined that it adequately explained the results of the example sensitivity studies as discussed below.

The NRC staff agrees with the applicant that there is little variation in **[[**

]]. The applicant provided figures to justify that heat loss from the RCS resulting from the containment flooding event **[[**

]].

The applicant also noted that whether the event results in a reactor trip influences the effect of parameter variations. For example, there is very little difference in **[[**

]].

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 except that turbine trip initiates with turbine stop valve closure, while loss of external load initiates with turbine control valve closure. The NRC staff reviewed the applicant's methodology for these events 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.

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 due to the closure of the turbine control valves. 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 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 pressurizer 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. Based on information reviewed as part of the NRC staff's audits, as documented in the associated audit report (ML19039A090), the NRC staff confirmed some of the behavior and trends observed in the sensitivity studies. For example, **[[**

]]

Based on the information submitted by the applicant, as confirmed by the NRC staff's review and information discussed in the audit report, 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.

3.7.2.7 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. 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. A turbine trip and loss of feedwater would result in a sudden loss of the secondary cooling, heatup of the RCS, and pressurization of the secondary side. The applicant stated that a reactor trip and DHRS actuation would 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 7-34, "Acceptance criteria, single active failure, loss of power scenarios – loss of condenser vacuum," provides the acceptance criteria of interest for the LOCV event. Because the LOCV event results in rapid primary and secondary pressurization and heatup, the NRC staff agrees that primary and secondary pressures are the correct acceptance criteria of interest.

The NRC staff reviewed TR Table 7-36, "Initial conditions, biases, and conservatisms – loss of condenser vacuum," which provides the initial conditions, biases, and assumptions for the LOCV event. For the initial conditions whose bias directions are specified, the NRC staff confirmed that the bias directions are limiting for this event. The NRC staff noted that the contents of TR Table 7-36 are essentially identical to those in TR Table 7-32, “Initial conditions, biases, and conservatisms – turbine trip / loss of external load,” for the turbine trip/loss of external load events. Because the LOCV event is phenomenologically similar to the turbine trip/loss of external load events, the same reasoning discussed in Section 3.7.2.6, “Turbine Trip / Loss of External Load,” of this SER for Table 7-32 applies to TR Table 7-36.

TR Table 7-37, “Representative sensitivity studies – loss of condenser vacuum,” provides results of representative sensitivity studies that investigated the effects of varied initial conditions and assumptions on the RCS and SG pressure acceptance criteria. In reviewing TR Table 7-37, the NRC staff noted differences in the representative sensitivity study biasing between the MSIV closure event and the LOCV event. Based on the information provided as part of the audits and discussed in the associated audit report (ML19039A090), the NRC staff confirmed that an MSIV closure would result in faster steam flow isolation than an LOCV since the turbine and condenser are farther away from the module than the MSIV, which is at the top of the NPM. This accounts for some differences, such as the specification to vary initial pressurizer level and initial RCS flow for the MSIV closure event to identify the limiting bias directions. In addition, the applicant performed more extensive bias sensitivity cases for some heatup events than others as part of the non-LOCA EM. For example, the **[[**

]]. However, these are only representative sensitivity studies that will be repeated as part of licensing-basis evaluations. The NRC staff finds the sensitivity studies acceptable to demonstrate the general process to determine limiting biases and assumptions.

Based on the information submitted by the applicant, as confirmed by the NRC staff’s review and audit, as documented in the associated audit report, 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.

3.7.2.8 Main Steam Isolation Valve(s) Closure

TR Section 7.2.8, “Main Steam Line Isolation Valve(s) Closure,” discusses the main steam isolation valve(s) closure event-specific analysis methodology. The MSIV closure event may be initiated by a spurious closure signal, resulting in the 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. More of the parameters are varied for the MSIV closure event than 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. 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 representative sensitivity studies in Table 7-41, "Representative sensitivity studies – main steam isolation valve closure." The applicant used these results to substantiate the specification of a nominal initial feedwater temperature (e.g., no bias needs to be applied) for the MSIV closure event as well as the other heatup events except for loss of feedwater and feedwater line break. Based on insensitivity of the RCS and SG peak pressure to initial feedwater temperature, the NRC staff agrees that using a nominal feedwater temperature for the MSIV closure event, and events whose system response prior to reactor trip are similar, is acceptable.

As noted in the discussion of the LOCV event, the response to the MSIV closure event is similar to that of the LOCV event, but plant layout alters the limiting biases for some parameters. 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.

Based on the information submitted by the applicant, as confirmed by the NRC staff's review and audit, as documented in the associated audit report (ML19039A090), 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 nonemergency AC power event-specific analysis methodology. A loss of ac power results in a loss of feedwater and a turbine trip, increasing pressure in the RCS and SGs. For this reason, the NRC staff finds that the applicant correctly identified primary and secondary pressures as the acceptance criteria of interest for this event in Table 7-42, "Acceptance criteria, single active failure, loss of power scenarios – loss of normal AC power." The applicant stated that a reactor trip and DHRS actuation 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.

In the NPM, the loss of nonemergency AC power event is complex due to the interactive nature of the various electrical distribution systems and the lack of a Class 1E power supply. The TR examines three scenarios for the loss of normal AC power event:

1) Failure of the low voltage (480 V and 120 V) ac electrical distribution system (ELVS) upon the loss of nonemergency AC power, with EDNS and EDSS available.

2) Failure of the ELVS and EDNS upon the loss of nonemergency AC power, with EDSS available.

3) Failure of the ELVS and EDSS upon the loss of nonemergency AC power, with EDNS available.

The applicant concluded that the limiting pressure responses result from the loss of ELVS with EDNS and EDSS available (scenario 1 above) since scenario 2 results in immediate control rod drive mechanism (CRDM) drop and reactor trip, and scenario 3 results in immediate reactor trip, DHRS actuation, and containment isolation. In other words, scenarios 2 and 3 result in safety systems responding sooner and are therefore non-limiting for the loss of nonemergency AC power initiating event.

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 confirmed that the specified bias directions are limiting for this event. The NRC staff notes that the table is consistent with that of the LOCV event in TR Table 7-36, "Initial conditions, biases, and conservatisms – loss of condenser vacuum," which is expected due to the phenomenological similarities between the events, except for more control systems being enabled for the loss of nonemergency AC power event. However, these additional enabled control systems lose functionality as a result of the loss of AC power, so they have no influence on the event.

TR Table 7-45, "Representative sensitivity studies – loss of normal AC power," provides the results of representative sensitivity studies for the loss of nonemergency AC power. 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.

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 the applicant explained is due to increasing the initial SG inventory. The applicant stated that peak SG pressure is more sensitive to initial RCS average temperature than initial SG pressure because RCS temperature affects the saturation pressure in the DHRS loop. The

NRC staff finds these explanations acceptable to clarify the behavior in TR Table 7-45, "Representative sensitivity studies – loss of normal AC power." In addition, the NRC staff finds the representative sensitivity studies acceptable to demonstrate the general process to determine limiting biases and assumptions.

Based on the information submitted by the applicant, as confirmed by the NRC staff's review and audit, as documented in the associated audit report (ML19039A090), the NRC staff finds that the applicant's methodology for this event is consistent with DSRS Section 15.2.6 and 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 RCS temperature or high PZR pressure. Therefore, the NRC staff finds that the applicant correctly identified 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 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 representative results for sensitivity studies for a loss of normal feedwater flow. The NRC staff notes that these results are limited in scope, **[[**

]]. These representative 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 and discussed in the associated audit report (ML19039A090), the NRC staff confirmed the reason for the trend in peak SG pressure versus feedwater flow reduction.

The NRC staff finds that perfoming 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, as documented in the associated audit report, 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 as a whole 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 described three scenarios for consideration:

1) The inadvertent opening of a single valve at full power conditions. 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. This scenario is the most limiting for peak secondary system pressures.

2) A signal malfunction results in the unexpected actuation of one of the DHRS trains in which both the DHRS steam line valve and the condensate line valve open. The feedwater and steam systems associated with the affected DHR train is isolated. This rapid loss of heat removal from the RCS results in increases in the primary pressure.

3) A signal malfunction results in the unexpected actuation of both DHRS trains. Similar to the above scenario, feedwater and steam systems associated with both DHR trains are isolated. This rapid loss of heat removal from the RCS results in the most challenging scenario for primary pressure.

TR Section 7.2.11.1, “General Event Description,” states that the cooldown effect on the RCS in the first scenario is bounded by the more limiting case of an increase in feedwater flow, while scenario one is the most challenging case for the secondary side pressurization. The second scenario affects only one SG and is bounded by the third scenario, which represents the limiting condition of a complete loss of normal heat removal from the RCS and the delay associated with the DHRS becoming an effective heat sink. Scenario three is the most challenging case for the pressurization of the RCS.

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 steam generator pressure. 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 representative results for example sensitivity studies for inadvertent DHRS initiation in terms of maximum primary and secondary pressures. This example sensitivity study helps to illustrate the event-specific methodology, and the NRC staff finds that perfoming the sensitivity studies by varying parameters and assumptions as described in the TR 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.

3.7.2.12 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.”

A feedwater line break can occur inside or outside of containment since there are no feedwater line check valves inside containment. 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.

A feedwater line break outside containment causes a loss of feedwater flow to the SGs and a heatup of the RCS. The applicant stated that large breaks result in reactor trip on high pressurizer pressure, while smaller breaks result in reactor trips on either low steam pressure or high steam superheat. DHRS actuates in all cases such that the non-faulted steam generator loop provides cooling by removing heat from RCS to the reactor pool.

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 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 for primary and steam generator pressure. Many of the parameters are to be varied for each application of the methodology. For the parameters whose bias directions are specified, the NRC staff confirmed that the bias directions are appropriately conservative.

TR Table 7-57, "Representative sensitivity studies – feedwater line break," provides the results of representative sensitivity studies that 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 perfoming the sensitivity studies by varying the feedwater break size, single active failures, loss of power assumptions 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., 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 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 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 exceeds 300 degrees F). At lower RCS temperatures, 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."

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 representative sensitivity studies that 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. Based on the information evaluated during the audits, as dicussed in the associated audit report (ML19039A090), 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, **[[**

]].

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 15 percent rated thermal power 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 pressurizer pressure, or high riser temperature MPS signal. The limiting condition results for the reactivity insertion rate that causes the high core power, high pressurizer pressure, and high RCS riser temperature analytical limits to be reached almost simultaneously. Higher reactivity insertion rates cause an earlier reactor trip on high power rate. The NRC staff agrees with the general strategy of "synchronizing" the power, pressure, and hot leg temperature trips 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.

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 pressurizer pressure and level control are varied as part of the methodology and may be enabled if their operation worsens the consequences of the transient. The NRC staff finds pressurizer pressure and level control operation conservative if they delay a high-pressure trip such that it occurs nearly simultaneously with the high hot leg temperature and/or high-power trips.

TR Table 7-65, “Representative sensitivity studies – uncontrolled control rod bank withdrawal at power,” provides the results of representative sensitivity studies that 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. During its audits, as documented in the audit associated report (ML19039A090), the NRC staff confirmed that limiting fuel centerline temperature cases correspond to **[[**

]]. The applicant also clarified that the reactivity insertion rates examined as part of the sensitivity studies were chosen based on identifying the rate at which the reactor trip transitioned from occurring on high power, temperature, or pressure to high power rate. This helped the NRC staff to confirmed 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,”:

1) Withdrawing a single control rod assembly,

2) Dropping one or more control rod assemblies, or

3) 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 the reactivity insertion rate results in reaching the core power, pressurizer pressure, and riser temperature analytical limits simultaneously.

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 with initial reactor power less than or equal to 50 percent rated thermal power, a high power-rate trip does not occur, and the reactor eventually returns to the initial power level. The NRC staff notes that the applicant uses the terminology “return to power” to describe this behavior, which is different from the “return to power” scenario described in DCA Part 2, Tier 2, Section 15.0.6, in which the NPM may become recritical following a design-basis event with one rod stuck out. The limiting MCHFR results from rod drop cases initiated from hot full power conditions.

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.

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 agrees with 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 representative sensitivity studies for the single control rod assembly withdrawal event. Similar to the control rod bank withdrawal event, the NRC staff confirmed during its audits, as described in the associated audit report (ML19039A090), that the applicant specified an acceptable methodology to determine the limiting reactivity insertion rate.

TR Table 7-71, “Representative sensitivity studies – control rod misoperation, dropped control rod assemblies,” presents the results of the representative sensitivity studies to identify the limiting bias directions for the rod drop event. Based on the information provided as part of the NRC staff audits, as described in the associated audit report (ML19039A090), the NRC staff confirmed that the initial conditions are more important than the moderator feedback for rod drop scenarios that trip on high power rate caused by the rapid reactor trip. The NRC staff finds that the representative sensitivity studies for the control rod misoperation events adequately demonstrate the process that may be used to determine the limiting biases for a licensing-basis calculation.

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, as decribed in the associated audit report, 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 caused by failure of the blend system, either by controller or mechanical failure, or operator error. The event is terminated by isolating the diluted water source, which is accomplished by automatically closing the demineralized water system (DWS) isolation valves in Modes 1-3.

The inadvertent decrease in boron concentration event is evaluated for all the operational modes permitted in the plant Technical Specifications. For Mode 1 operation, three conditions are analyzed: hot full power (HFP), 25 percent rated thermal power, and hot zero power. 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. 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 (or lower) 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 mixing model to determine the remaining available shutdown margin and the time shutdown margin would be lost if the dilution source was not terminated.

The applicant determined that the inadvertent decrease in boron concentration in Mode 1 at 25 percent power is bounded by the hot full power and hot zero power cases. 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 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 mixing 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 1.7 ft^3/s (763 gpm). If the RCS flow rate is greater than or equal to 1.7 ft^3/s (763 gpm), 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 wave front model produces a conservatively high rate of reactivity insertion which results in a larger total reactivty 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. In Mode 5, Refueling, the Technical Specifications enforce limits on the pool boron concentration to provide adequate shutdown margin. The NRC staff notes that the large pool volume makes it highly unlikely that an inadvertent boron dilution would cause an unacceptable loss of shutdown margin.

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. The representative results for examples of such sensitivity studies are presented in TR Tables 7-75, “Representative results – inadvertent decrease in boron concentration in Mode 1 at hot full power with the Perfect Mixing Model,” 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 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 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 pressurizer level control system may result in the addition of makeup fluid, which will increase the pressurizer water level. Reactor trip on high pressurizer water level or high pressurizer pressure will 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 terminated by CVCS isolation (noting that the CVCS containment isolation valves are safety related) on high PZR level or low-low RCS flow.

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. Six parameters/control system assumptions are varied to maximize pressurization: initial RCS average temperature, initial RCS flow rate, initial pressurizer pressure and level, makeup temperature, and pressurizer 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 and demonstrates 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.

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, pressurizer spray lines, and high point vent (degassing) lines. Failure of a spray line or high point vent line is 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 a small line outside containment causes a decrease in pressurizer pressure and level and a reactor trip on low pressurizer pressure or low pressurizer level. CVCS isolation terminates the loss of fluid. 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. 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. 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, as documented in the associated audit report (ML19039A090), the NRC staff confirmed that a biased-low initial RCS flow rate and a biased-high initial RCS average temperature are **[[**

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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 representative results of the applicant’s example sensitivity studies. 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.

The NRC staff finds that perfoming the sensitivity studies by varying the 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 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, as documented in the associated audit report, 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 steam generator tube causes the pressurizer pressure and pressurizer level to decrease at a rate dependent upon the size and location of the fault. A reactor trip may be generated on low pressurizer pressure or low pressurizer 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.

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 confirmed that the bias directions that are specified are appropriately conservative with respect to effect on the acceptance criteria.

As discussed in the associated audit report (ML19039A090), the NRC staff also audited the example sensitivity studies for the SGTF event, which are shown in TR Table 7-92, "Representative break characteristics, initial conditions, loss of power, and single active failure sensitivity study - steam generator tube failure." In its audits, the NRC staff confirmed that the mass release is primarily driven by **[[**

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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 audits, as documented in the associated audit report, 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. Representative calculations are presented for each of the transient analysis categories:

1) Cooldown and/or Depressurization of the RCS (TR Section 8.1)

2) Heatup and/or Pressurization of the RCS (TR Section 8.2)

3) Reactivity Anomaly (TR Section 8.3)

4) Increase in RCS Inventory (TR Section 8.4)

5) Decrease in RCS Inventory (TR Section 8.5)

The information included for each scenario includes an event description, and results for key FOM and associated acceptance criterion resulting in the conclusions. These results are presented to demonstrate the application of the non-LOCA methodology to the NPM. The applicant stated that the fuel rod and core physics parameter inputs for the representative transients were developed using COPERNIC and SIMULATE5 computer codes, respectively, as described in the approved NuScale TRs TR-0116-20825-NP-A, “Appicability of AREVA Fuel Methodology for the NuScale Design,” (ML18040B306) and TR-0616-48793-NP-A, “Nuclear Analysis Codes and Methods Qualification,” (ML18348B036).

The NRC staff reviewed this section of the TR to understand how the methodology, as described in the other TR sections, would be expected to be implemented as part of a design-specific review. The NRC staff’s review of sample analyses indicated that the methodology was specified appropriately in the TR after observing that the sample calculations provided conservative and expected results.

3.8.1 Cooldown and/or Depressurization of the Reactor Coolant System.

TR Section 8.1, “Cooldown and/or Depressurization of the Reactor Coolant System,” presents representative calculations for transients that result in an increase in heat removal by the secondary system. These transients include decrease in feedwater temperature, increase in steam flow, and a break in the main steam line.

TR Section 8.1.1, “Decrease in Feedwater Termperature,” discusses the representative calculation of a decrease in feedwater temperature event. The assumptions for single failures, loss of power, and parameter biases defined in TR Section 7.2.1, “Decrease in Feedwater Temperature,” are utilized in the analysis. The analysis assumed a 1.18 °F/s linear rate of decrease in feedwater temperature is based on the limiting case from the representative sensitivity study on temperature decrease rate, which provided the limiting NRELAP5 MCHFR in TR Table 7-10, “Representative feedwater temperature transient study.” The sequence of events table (Table 8-1, “Decrease in feedwater temperature sequence of events”) confirms that

RTS is actuated by simultaneous high power and high RCS riser temperature signals. The moderator temperature feedback results in a core power increase from 163.2 MW(t) to 200.6 MW(t), an increase of approximately 23 percent core power.

TR Figure 8-2, "Power response for the representative decrease in feedwater temperature event," shows the power increase resulting from moderator temperature feedback for the decrease in feedwater temperature shown in TR Figure 8-1, "Temperature of feedwater during the representative decrease in feedwater temperature event." TR Figure 8-7, "Reactor coolant system flow rate for the representative decrease in feedwater temperature event," shows a dramatic decrease in RCS flow with subsequent flow oscillations. Oscillations in the core inlet temperature (Figure 8-8, "Core inlet temperature for the representative decrease in feedwater temperature event") and the net reactivity (TR Figure 8-9, "Net reactivity for the representative decrease in feedwater temperature event"), reflect the expected response following a reactor trip with concurrent DHRS actuation as described in TR Revision 2, Section 7.2, "Event Specific Methodology."

The NRC staff gained clarification during its audits, as documented in an audit report (ML19039A090), that the NPM physical behavior in Figure 8-8, "Core inlet temperature for the representative decrease in feedwater temperature event," between 500 and 750 seconds results from the characteristic oscillatory response of the NPM following DHRS actuation. Hot fluid is being generated in the core with a hot slug of water at the top of the riser and a cold slug of water at the core exit, resulting in a small RCS flow rate. From 600 to 700 seconds, the hot slug of water is sufficient to cause buoyant flow up the riser and a spike in RCS flow. Around 750 seconds, colder fluid from the downcomer flows into the core. Some radiant heat transfer carries heat away, which may contribute to an inflection point around this time.

The NRC staff also reviewed TR Section 8.1.2, "Increase in Steam Flow," which describes the representative calculation of the increase in steam flow event using the methodology discussed in TR Section 7.2.3, "Increase in Steam Flow." TR Section 8.1.2.1, "Event Description," states that the initial SG heat transfer and initial fuel temperature are biased low in the representative calculation. Although TR Table 7-21, "Representative steam flow study – steam generator heat transfer biased low," specifies that nominal values are to be used for these parameters, they are specified as such because of the insensitivity of MCHFR to biases of these parameters. Therefore, application of the low biases is acceptable for these parameters for the purposes of the representative calculation.

A step increase of 14.45 percent in steam flow initiates the representative increase in steam flow event. TR Figure 8-11, "Steam generator 2 pressure response for the representative increase in steam flow event," shows a secondary side pressure decrease of approximately 25 psi as a result of the step change in steam flow and an increase in the secondary heat removal. The core inlet temperature decrease shown early in TR Figure 8-13, "Core inlet temperature for the representative increase in steam flow event," results in the moderator temperature coefficient feedback increasing the core power to 199.97 MW(t) as shown in TR Figure 8-14, "Power response for the representative increase in steam flow event." The limiting MCHFR is reached at approximately 62 seconds, which occurs before the trip signal on high RCS riser temperature. The event is terminated after the RTS and DHRS actuate.

Based on information reviewed as part of the NRC staff's audits, as documented in the associated audit report (ML19039A090), the NRC staff confirmed that the SG heat transfer multiplier is applied **[[**

]]. The applicant adjusted **[[** **]]** to maintain the same RCS flow rate when the heat transfer was reduced with application of the SG heat transfer multiplier.

The NRC staff reviewed TR Section 8.1.3, "Main Steam Line Break," which describes the main steam line break representative calculation that was performed for a 3.3 percent split break outside containment. Even with such a small break, the increased steam flow causes in a decrease in the core inlet temperature resulting in a peak power of 202.8 MW(t) (24.3 percent) coincident with the NRELAP5-estimated MCHFR of 3.682. This results in a reactor trip signal on high reactor power at 49 seconds. The reactor power is shown in Figure 8-23, "Power response for the representative main steam line break event," of the TR.

The steam generator pressures, shown in TR Figure 8-24, "Steam generators 1 (unaffected) and 2 (affected) pressure response for the representative main steam line break event," initially decrease due to the small break, but then increase as the increasing core power causes the core outlet temperature to increase, resulting in increased primary to secondary heat transfer. The high steam generator pressure signal actuates the DHRS at approximately 59 seconds, isolating the SGs. After actuation of the DHRS, oscillations in various parameters occur similar to those observed in the decrease of feedwater temperature representative calculation and the increase in steam flow representative calculation.

Based on the information reviewed as part of the NRC staff's audits, as documented in the associated audit report (ML19039A090), the NRC staff observed that the target initial fuel temperatures are based on fuel performance code data that examine many different factors, including burnup and operating conditions. The applicant considers conservative fuel temperature ranges at BOC and EOC in the transient calculations. Further, the NRC staff observed that the difference in the transient response oscillations between the increase in steam flow event and the main steam line break event results from the failed SG blowdown in the main steam line break event, which causes a colder slug of water to build up. The resulting lower core inlet temperature causes an increase in the peak flow amplitudes in the oscillations.

Based on its review of the applicant's representative calculations for transients that result in an increase in heat removal by the secondary system described in this section, the NRC staff concluded that the sample analyses appropriately illustrate that implementation of the methodology as specified in the TR provide conservative and expected results. The NRC staff concludes that the methodology, when implemented, will provide conservative results appropriate for determining whether FOMs are met.

3.8.2 Heatup and/or Pressurization of the Reactor Coolant System

TR Section 8.2, "Heatup and/or Pressurization of the Reactor Coolant System," presents representative calculations for transients that result in a decrease in heat removal by the secondary system, including the loss of normal feedwater, the loss of nonemergency AC power, and a break in the feedwater line.

TR Section 8.2.1, “Loss of Normal Feedwater Flow,” examines two loss of normal feedwater cases: an RCS pressure limiting case and a secondary pressure limiting case.

TR Section 8.2.1.1, “Event Description – Reactor Coolant System Pressure Case,” discusses the loss of normal feedwater with assumptions that maximize the primary RCS pressure, including biasing the RCS temperature, feedwater temperature, pressurizer level and steam generator level high; biasing the RCS flow and pressurizer pressure low; increasing the SG primary and secondary side heat transfer coefficient by 30 percent; and assuming BOC reactivity coefficients.

The representative calculation for the RCS pressure case assumes a complete loss of normal feedwater at zero seconds. The loss of heat removal capability causes pressurization of the RCS and results in a reactor trip on high pressurizer pressure at 17.6 seconds, and the turbine trips shortly thereafter at 18.6 seconds. The RSV opens at a lift pressure of 2137.3 psia at 24.2 seconds, and the peak primary pressure of 2156.1 psia is reached at 24.6 seconds for a pressure overshoot of only 18.8 psi, as shown in TR Figure 8-29, “Reactor pressure vessel pressure response for the representative loss of normal feedwater flow event – reactor coolant system pressure case.” DHRS flow initiation is conservatively modeled to delay DHRS actuation valve opening until 49.3 seconds.

The NRC staff observed that after reactor trip and actuation of DHRS, the RCS flow rate in Figure 8-33 decreases rapidly. While large flow oscillations occur with DHRS initiation, flow reversal similar to that observed in the cases that result from overcooling events is not observed for this event. The core inlet and outlet temperatures (TR Figures 8-34, “Core inlet temperature for the representative loss of normal feedwater flow event – reactor coolant system pressure,” case and 8-35, “Core outlet temperature for the representative loss of normal feedwater flow event – reactor coolant system pressure case”) also show an oscillatory response with DHRS initiation. The oscillations are more frequent, better defined, and appear to dampen more slowly than for overcooling events, since the oscillatory behavior resulting from the DHRS cooling is mitigated to some degree in the heatup transients.

TR Section 8.2.1.4, “Event Description – Secondary Pressure Case,” discusses the loss of normal feedwater with assumptions that maximize the secondary pressure. The analysis assumptions are the same as described in TR Section 8.2.1.1, “Event Description – Reactor Coolant System Pressure Case,” but in this case, there is only a partial loss of feedwater flow. The feedwater flow is assumed to ramp down to 97.7 percent of normal over 0.1 seconds (e.g., a 2.3 percent reduction in FW flow). The RCS heatup (TR Figures 8-38, “Core inlet temperature for the representative loss of normal feedwater flow event – secondary pressure case,” and 8-39, “Core outlet temperature for the representative loss of normal feedwater flow event – secondary pressure case”) is much slower than the RCS pressure case and results in a reactor trip at 645.3 seconds on high riser temperature. In this case, the peak RCS pressure (TR Figure 8-40, “Reactor pressure vessel pressure response for the representative loss of normal feedwater flow event – secondary pressure case”) is lower than that in the RCS pressure limiting case, and the RSV does not lift. The small decrease in secondary heat removal capability does not appreciably change the secondary pressure (TR Figure 8-42, “Steam generator 2 pressure response for the representative loss of normal feedwater flow event – secondary pressure case”) until after reactor trip and DHRS actuation. Flow through the DHRS begins 30 seconds later, and the secondary pressure peaks at 1421.6 psia approximately 40 seconds after DHRS flow begins.

TR Section 8.2.2, “Loss of Normal AC Power,” describes the representative loss of normal AC power event for an RCS pressure limiting scenario. Although TR Table 7-44, “Initial conditions, biases, and conservatisms – loss of normal AC power,” does not specify a reactor pool temperature bias, TR Section 8.2.2.1, “Event Description,” states the reactor pool temperature is assumed to be at the maximum bounding value of 200 degrees F for events/cases that require heat removal using DHRS as well as the specific assumptions for this event. Based on the information reviewed as part of the NRC staff’s audits, as documented in the associated audit report ML19039A090), the NRC staff observed that the applicant did not perform sensitivity studies to examine the effect of biasing the pool water temperature high but agrees that a high bias is generally conservative for heatup events due to the reduced effectiveness of the heat sink.

The representative calculation assumes a loss of normal AC power at zero seconds with DC power systems available.

TR Figure 8-47, “Primary temperature response for the representative loss of AC power event,” shows that the core average temperature exceeds both the core outlet temperature and core inlet temperature for brief periods of time. Based on the information reviewed as part of the NRC staff’s audits, as discussed in the associated audit report (ML19039A090), the NRC staff observed that the applicant calculated the RCS average temperature as the average of the core outlet and core inlet temperatures, as expected. The 50-second delay in calculation of RCS average temperature results from the characteristic oscillatory response. After reactor trip, the reduction in RCS flow results in higher density water build-up in the riser, resulting in a stall in riser flow. While the core is continuing to heat fluid, the flow inlet to the core is also providing hotter fluid to the core as a result of the loss of the heat sink. This combination results in a peak in the core average temperature at an earlier time than the core exit temperature.

TR Figure 8-48, “System pressure response for the representative loss of AC power event,” shows the resulting RPV lower plenum and SG inlet pressure responses. The RCS peak pressure is 2,155 psia, and the secondary pressure is 1,250 psia.

TR Section 8.2.3, “Feedwater Line Break,” describes the representative calculation of an RCS pressure limiting case for the feedwater line break event. TR Table 7-56, “Initial conditions, biases, and conservatisms – feedwater line break,” provides the initial conditions, biases, and conservatisms for the feedwater line break event. Since Table 7-56 indicates that many of the parameters are varied to maximize pressurization, TR Section 8.2.3.1, “Event Description,” provides specific assumptions for varied parameters, including high biases for the initial RCS temperature, feedwater temperature, reactor pool temperature, pressurizer pressure and level, as well as BOC reactivity parameters. TR Section 8.2.3.1 states that the initial primary and initial feedwater temperatures are set to their maximum values to produce a higher energy system. The representative calculation for a feedwater line break outside of containment assumes a double-ended guillotine break just outside containment with a coincident loss of AC power and DC power systems available.

The high pressurizer pressure reactor trip setpoint is reached quickly, and the reactor trips at approximately 8 seconds. The peak primary pressure of 2,158 psia is reached at approximately 12 seconds, with approximately 20 psi of overshoot. TR Figure 8-56, “System pressure response for the representative feedwater line break event,” shows the primary and secondary pressure responses.

The primary temperature responses for the representative feedwater line break calculation are shown in TR Figure 8-55, “Primary temperature response for the representative feedwater line break event,” which displays the oscillations characteristic of post-trip and DHRS actuation conditions described in TR Section 7.2, “Event Specific Methodology.” Figure 8-61, “Reactor coolant system flow response for the representative feedwater line break event,” shows the RCS, feedwater, and steam flow rates. After reactor trip and DHRS actuation, the RCS flow rate decreases with strongly damped oscillations, and flow does not reverse.

Based on its review of the applicant’s representative calculations for transients that result in a decrease in heat removal by the secondary system described in this section, the NRC staff concluded that the sample analyses appropriately illustrate that implementation of the methodology as specified in the TR provided conservative and expected results, demonstrating that the methodology, when implemented, will provide conservative results appropriate for determining whether FOMs are met.

3.8.3 Reactivity Anomaly

TR Section 8.3, “Reactivity Anomaly,” presents representative calculations for reactivity and power distribution anomalies, including the uncontrolled control rod assembly withdrawal from subcritical or low power startup conditions and a control rod misoperation event.

TR Section 8.3.1, “Uncontrolled Control Rod Assembly Bank Withdrawal from Subcritical or Low Power Startup Conditions,” describes the representative calculation of the uncontrolled control rod assembly bank withdrawal from subcritical or low power startup conditions event. The representative calculation was performed at for an initial power of 15 percent. In the representative calculation, the high power-rate signal is not active, and core protection is provided by the high-power signal and the startup rate (intermediate range) signals. The high reactivity insertion rate results in a reactor trip signal on high startup rate to be generated at approximately 4 seconds, and the peak core power of 42 percent occurs at 7 seconds. Because little energy is added to the system before the scram, the event is terminated with only a small increase in pressurizer pressure as shown in TR Figure 8-62, “Pressurizer pressure response for the bank withdrawal from a low power startup condition,” and a relatively small increase in core outlet temperature before declining, as shown in Figure 8-67, “Core outlet temperature for the bank withdrawal from a low power startup condition.”

TR Section 8.3.2, “Control Rod Misoperation,” describes the representative calculation of the control rod misoperation event. The representative calculation examined a single rod withdrawal that initiated from 75 percent power, which appears to correspond to the limiting case from the example sensitivity studies in TR Table 7-69, “Representative sensitivity studies – control rod misoperation, single control rod assembly withdrawal.”

Positive reactivity insertion by the single rod withdrawal increases reactor power causing an increase in pressurizer pressure and level. A reactor trip signal is generated when the high riser temperature analytical limit is reached at approximately 139 seconds. The lowest-reading ex-core detector, in this case, does not reach the high power setpoint. The NRELAP5-estimated MCHFR (3.107) is reached at approximately 147 seconds, the same time that control rod insertion begins.

TR Section 8.3.2.2, "Single Rod Withdrawal MCHFR Case - Analysis Results," states that RCS flow decreases rapidly after reactor trip and DHRS actuation. Subsequently, flow observations are observed due to temperature and density differences between the riser and downcomer which is discussed in more detail in TR Section 7.2, "Event Specific Methodology."

To aid in understanding the general oscillatory behavior inherent in the NPM, the NRC staff confirmed in its audits, as documented in the associated audit report (ML19039A090), that during the period of oscillations, the flow is a function of the stable core heat-driven mass flow rate and the monometric balance of buoyant forces. Further, the NRC staff confirmed the periodicity in the decaying oscillations is a function not only of the facility dimensional characteristics but also a function of the event timing and DHRS response. The DHRS response affects the amplitude and timing of the periodicity due to the rate at which condensation becomes effective.

The NRC staff also confirmed, as described in the associated audit report (ML19039A090), that **[[ ]]** explains why it takes 187 seconds to increase from the minimum to maximum post trip core outlet temperature for the representative control rod misoperation event. In addition, the NRC staff observed that the difference between the core inlet specific volume and the approximate core exit and riser exit specific volume is approximately **[[ ]]**, which explains why the flow spike reaches a peak of approximately 40 percent of the initial steady state flow rate.

Based on its review of the applicant's representative calculations for for reactivity and power distribution anomalies described in this section, the NRC staff concluded that the sample analyses appropriately illustrate that implementation of the methodology as specified in the TR provided conservative and expected results, demonstrating that the methodology, when implemented, will provide conservative results appropriate for determining whether FOMs are met.

3.8.4 Increase in Reactor Coolant System Inventory

TR Section 8.4, "Increase in Reactor Coolant System Inventory," presents a representative calculation for the CVCS malfunction that increases RCS inventory event. The event is initiated at 102 percent power with a CVCS charging flow of 40 gpm at high temperature (150°F). The representative calculation assumed pressurizer spray was unavailable. The inventory increase results in increasing pressurizer pressure and level until a reactor trip and DHRS actuation is generated on high pressurizer pressure at approximately 513 seconds, as shown in TR Figure 8-84, "Pressure at the bottom of the pressurizer for increase in reactor coolant system inventory." CVCS isolation is not actuated until a high pressurizer level signal is generated at approximately 3,466 seconds. Prior to CVCS isolation, the primary pressure rises to the RSV setpoint, and a peak RCS pressure of 2,155 psia is reached at 3,371 seconds.

The representative calculation figures, especially those of RCS flow (TR Figure 8-87, "Reactor coolant system flow for increase in reactor coolant system inventory") and RCS temperatures (TR Figure 8-90, "Core inlet and exit coolant liquid temperature for increase in reactor coolant system inventory") show that system oscillations occur following DHRS actuation. The RCS flow rate displays the characteristic decline following reactor trip. In this case, the initial flow rate decline does not stagnate or reverse. The second decline appears to stagnate and may reverse, but this cannot be determined from the scale of the figure.

Based on its review of the applicant's representative calculation for the CVCS malfunction that increases RCS inventory described in this section, the NRC staff concluded that the sample analysis appropriately illustrated that implementation of the methodology as specified in the TR provided conservative and expected results, demonstrating that the methodology, when implemented, will provide conservative results appropriate for determining whether FOMs are met.

3.8.5 Decrease in Reactor Coolant System Inventory

TR Section 8.5, "Decrease in Reactor Cooland System Inventory," presents representative calculations for a break in small lines carrying primary coolant outside the containment and for a SGTF in the decrease in RCS inventory category.

TR Section 8.5.1, "Small Line Break Outside of Containment," presents the results of a representative calculation of a break in a small line outside of containment that assumes a double-ended guillotine break of the letdown line and a loss of AC power at the time of the break. The loss of RCS inventory causes the pressurizer pressure (TR Figure 8-94, "Reactor pressure vessel pressure response (0 to 350 sec) for the representative small break outside containment event") and level (TR Figure 8-93, "Pressurizer level response for the representative small break outside containment event") to briefly decrease before increasing to until after reactor trip. The loss of AC power and resulting turbine trip cause an increase in secondary side pressure (TR Figure 8-95, "Steam generator pressure responses for the representative small break outside containment event"), generating a reactor trip at 13.7 seconds due to high steam line pressure. The high steam line pressure signal also causes DHRS actuation and secondary side isolation. This calculation assumes that a double-ended guillotine CVCS makeup line break occurs concurrent with reactor trip. A low pressurizer pressure signal results in containment isolation, which isolates the CVCS and terminates the break with a maximum integrated release of 11,940 lbm. While break flow ceases at approximately 100 seconds, continued primary shrinkage continues, and the calculated pressurizer level drops to 0 at approximately 1,125 seconds. TR Figure 8-102, "Level above top of core response for the representative small break outside containment event," demonstrates that the level remains well above the top of the core for this representative calculation.

TR Section 8.5.2, "Steam Generator Tube Failure," presents the results of a representative calculation of a SGTF. The representative calculation assumes a 100 percent double-ended guillotine break with a single active failure of the primary MSIV to close, no loss of normal AC power, and biases consistent with those in TR Table 7-91, "Initial conditions, biases, and conservatisms – steam generator tube failure." Break flow out of the SGTF results in a decrease in pressurizer pressure and level. A reactor trip signal is generated on low pressurizer level at 146.0 seconds. A low pressurizer pressure signal results in DHRS actuation at 170.4 seconds, which also isolates the secondary side, except for the primary MSIV on the faulted steam generator. TR Table 8-12, "Sequence of events for steam generator tube failure," indicates that the secondary MSIV on the faulted SG is fully closed at 200.5 seconds, which isolates the environmental release. The maximum integrated break flow to the environment is 8,477 lbm.

TR Figure 8-104, "Reactor pressure vessel and steam generator pressure responses (0 to 500 sec) for the representative steam generator tube failure event (tube failure occurs in SG1)," shows the RCS and SG pressure responses for the representative calculation of a SGTF. After secondary side isolation, the steam generator pressures increase, with the faulted secondary side pressure increasing until equilibration with the RCS pressure at approximately 315 seconds. Secondary MSIV closure at 200.5 seconds stops the break flow to the environment but does not stop the influx of primary fluid into the faulted steam generator secondary. TR Figure 8-106, "Instantaneous break flow response for the representative steam generator tube failure event," shows that upon secondary side isolation at approximately 172 seconds, the break flow increases to a peak of approximately 47 lbs/sec as a result of the increased secondary side pressure in the faulted steam generator. The break flow then decreases until the RCS pressure and faulted steam generator pressures equilibrate.

Based on its review of the applicant's representative calculations for the break in small lines carrying primary coolant outside the containment and SGTF events described in this section, the NRC staff concluded that the sample analyses appropriately illustrate that implementation of the methodology as specified in the TR provided conservative and expected results, demonstrating that the methodology, when implemented, will provide conservative results appropriate for determining whether FOMs are met.

3.8.6 Summary of Representative Calculations.

The applicant performed the representative calculations following the event-specific non-LOCA analysis methodology specified in TR Section 7.2, "Event Specific Methodology," with specific inputs, biases, and assumptions appropriate to the determination of conservative parameters for comparison to the FOMs. Audit discussions, as documented in the associated audit reports (ML19039A090 and ML20036C849), and applicant responses to RAIs clarified information in Section 8, "Representative Calculations," of the TR. The representative analyses adequately demonstrate the application of the NuScale non-LOCA EM.

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 50 Appendix B and is implemented using the guidance of ASME NQA-1 2008 and NQA-1a-2009 Addenda (Reference 4).

The SRP requires that the EM be maintained under a QAP that meets the requirements of 10 CFR 50 Appendix B. The TR references the NuScale QAP which is indicated to comply with the NRC requirements. The QAP aspects are addressed in "NuScale Topical Report: Quality Assurance Program Description for the NuScale Power Plant," NP-TR-1010-859-NP-A. The NRC staff reviewed the QAP requirements and documented its approval in its SER (ML16347A405). Further, the NRC staff inspected NuScale's design control process and code development procedures. These inspections are documented in inspection reports dated October 7, 2017 (ML15268A186) and July 24, 2017 (ML17201J382). The TR states that the NRELAP5 computer code development has followed the NuScale QAP. In addition, it is also stated that the non-LOCA transient analysis is performed and documented in accordance to the NuScale QAP.

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. Any future changes or revisions to TR-0516-49422-P, "Loss-of-Coolant Accident Evaluation Model," November 2019, Revision 1 (ML19331B585) must be assessed by the applicant for their potential impact on the non-LOCA EM. Any subsequent changes to the non-LOCA methodology require NRC approval.

2. Use of the non-LOCA EM is limited to analysis of events described in 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, return to power assuming the worst-case stuck control rod, 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 long-term cooling evaluations and must be used in conjunction with separately approved EMs for those analyses.

3. An applicant or licensee seeking to apply this methodology to a design other than the design represented in NPM model Revision 2 (or any NPM model update made pursuant to a change process specifically approved by NRC for changes to the NPM model) must evaluate steam generator and DHRS heat transfer biases to determine if the elimination of the biases within this methodology remains justified based on margins to non-LOCA FOMs.

4. 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.

5. An applicant or licensee seeking to apply this methodology to a design must receive a separate approval through that design review for the event-specific electrical power assumptions (AC/DC), single failures, and the need for operator actions necessary to mitigate non-LOCA design basis events.

6. Use of the non-LOCA EM is limited to its use with NRELAP5 v1.4, in conjunction with NPM model Revision 2, unless changes are made pursuant to a change process specifically approved by NRC for changes to NRELAP5 and the NPM model.

5. CONCLUSION

The NRC staff reviewed TR-0516-49416-P, Revision 2, and the applicant's responses to staff RAIs and audited supporting documentation, as documented in the associated audit reports. 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. In addition, the NRC staff considers all RAI questions associated with the non-LOCA review closed and resolved.

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., DCAs) referencing the non-LOCA TR.

The NRC staff concludes that the non-LOCA methodology, as documented in TR Revision 2, is acceptable for analysis of the non-LOCA events in an NPM subject to the limitations and conditions stated in Section 4, "Limitations and Conditions," of this SER.

6. REFERENCES

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